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. 15.00.03-02

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 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?

Response

a. The reactor coolant source term data in Table 11.1-2 represents the design basis fission and corrosion product specific activities calculated based on 1% fuel defect and continuous gas stripping operation. In order to maximize the RCS initial source term used for the dose analyses, values given in Table 15A-3 are calculated based on 0.25% fuel defect with no credit for gas stripping operation, which are provided in Table 12.2-5, and multiplied by a factor of 4 to account for the increase of fuel defect from 0.25% to 1%. For clarity, Table 15A-3 will be updated.

b. i The radiological consequence analyses in Chapter 15 are based on the thermal-hydraulic (T/H) analyses. Since the T/H analyses for different DBAs use different initial RCS mass to maximize the radiological consequences, the total reactor coolant activity values in Column 2 of Tables 15A-4 through 15A-7 are different. The RCS mass values applied in each DBA are given in Table 1. The total RCS activities in Columns 3 through 6 in Table 1 are calculated by multiplying the DE I-131 concentration (Bq/g) in Column 2 by the corresponding RCS mass value for each DBA.

Table 1. RCS Mass and Initial Iodine Total Activities Used in Dose Analyses

Nuclides	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131 Activity Concentration (Bq/g)	Total Activity in RCS (Bq)				
		Steam Line Break (SLB)		Steam Generator Tube Rupture (SGTR)	Failure of Small Lines Carrying Primary Coolant Outside Containment (LDLB)	Feedwater Line Break (FLB)
		SLBFPDLOOP	SLBZPLOPD			
RCS Mass (kg)	-	274,392	286,829	290,680	292,431	288,086
I-131	2.93E+04	8.05E+12	8.41E+12	8.53E+12	8.58E+12	8.45E+12
I-132	7.88E+03	2.16E+12	2.26E+12	2.29E+12	2.30E+12	2.27E+12
I-133	4.16E+04	1.14E+13	1.19E+13	1.21E+13	1.22E+13	1.20E+13
I-134	4.81E+03	1.32E+12	1.38E+12	1.40E+12	1.41E+12	1.39E+12
I-135	2.37E+04	6.49E+12	6.78E+12	6.88E+12	6.92E+12	6.82E+10

i. The iodine appearance rate are also calculated using an accident-specific mass as with the Responses #2.b.i above, and the methods to calculate letdown purification removal rate are provided in detail in DCD 15A.1.2.2.

c. The quantitative evaluation was performed to determine the dose contribution of alkali metals based on the initial concentration in the reactor coolant corresponding to a 1.0% fuel defect and, for the events that experience fuel cladding damage, the additional concentration activities resulted from the failed fuel. For this re-analysis of the radiological consequences, the updated on-site χ/Qs were used, which was submitted as the responses to RAI 20-7912 -Question 02.03.04-1 and RAI 174-8211-Question 02.03.04-05. The evaluation results have shown that their impact was not insignificant compared to the total doses. The transport of alkali metals such as cesium (Cs) and rubidium (Rb) in the reactor coolant to the environment should be therefore taken into account.

   The related descriptions in DCD Chapter 15 and tables indicating results for the radiological consequences will be revised to integrate consideration of the alkali metals.

d. According to RG 1.183, Appendix F, Section 5.1, the primary-to-secondary leak rate in the SGs should be assumed to be the leak rate limiting condition for operation specified in the Technical Specifications (TS). The RCS operational leakage in technical specifications 3.4.12 is intended to limit to 150 gpd (0.39 L/min) per any one SG. In the DBA analysis for APR1400, however, the primary-to-secondary leakage of 0.6 gpm (2.27 L/min) for both SGs (total SGs) was used, which is higher than this TS limit. Section 3.4.12 of the TS Bases indicates that:

   - The safety analysis for an event resulting in steam discharge to the atmosphere conservatively assumes a 1.13 L/min (0.3 gpm) primary to secondary leakage as the initial condition.

In addition, the primary to secondary leakage of 0.3 gpm for any one SG applied to the dose analysis corresponds to the maximum accident-induced leakage limit specified in the technical specification 5.5.9 "Steam Generator (SG) Program", which is determined based on design basis accident considerations. Therefore, although the primary-to-secondary leakage assumed in the radiological dose assessment is not consistent with the guidance specified in RG 1.183, Appendix F, Section 5.1, this leakage is conservatively used in the dose analyses to maximize the offsite doses.

Impact on DCD

DCD Table 15A-3 will be revised as indicated in the Attachment 1, and DCD Sections 15.1. and Table 15. Will be revised as indicated in the Attachment 2.

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-3

Reactor Coolant Iodine Concentrations for Various Conditions

Nuclides	1.0 % Failed Fuel RCS Iodine Activity Concentration ~~(Bq/g)~~ (Bq/g)(1)	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131 Activity Concentration (Bq/g)	2.2×10^6 Bq/g (60 μCi/g) DE I-131 Activity Concentration (Bq/g)
I-131	9.92×10^4	2.93×10^4	1.76×10^6
I-132	2.66×10^4	7.88×10^3	4.72×10^5
I-133	1.41×10^5	4.16×10^4	2.50×10^6
I-134	1.63×10^4	4.81×10^3	2.89×10^4
I-135	7.99×10^4	2.37×10^4	1.42×10^6

(1) These values are calculated by multiplying the values based on 0.25% failed fuel by a factor of 4 to be based on 1% failed fuel.

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Table 6.4-2

MCR and TSC Doses from Design Basis Accidents

Design Basis Accident		TEDE (mSv)(1)
Steam system piping failure	1 % Fuel Failure	~~3.26E+01~~
	Pre-accident spike	~~7.82E+00~~
	Event-generated spike	~~1.29E+01~~
Feedwater system pipe break		~~5.91E+00~~
RCP rotor seizure		~~1.93E+01~~
Control element assembly ejection	Containment leakage	~~1.84E+01~~
	Steam system release case	~~3.41E+01~~
Failure of small lines carrying primary coolant outside containment		~~6.41E+00~~
Steam generator tube rupture	Pre-accident spike	~~1.00E+01~~
	Event-generated spike	~~7.94E+00~~
Loss of coolant accident		~~3.88E+01~~
Fuel handing accident		~~1.19E+01~~

(1) TEDE: Total effective dose equivalent

~~3.78E+01~~
~~1.30E+01~~
~~1.81E+01~~
~~1.11E+01~~
~~2.45E+01~~
~~2.36E+01~~
~~3.94E+01~~
~~1.16E+01~~
~~1.52E+01~~
~~1.32E+01~~
~~4.40E+01~~
~~6.25E+00~~

3.58E+01
2.10E+01
2.22E+01
1.98E+01
2.28E+01
3.54E+01
2.91E+01
1.95E+01
2.02E+01
1.96E+01
4.69E+01
8.55E+00

For the alkali metal, the initial concentrations in the primary side are conservatively assumed to be based on one percent fuel defect.

3) A failed fuel condition in the core

b. GIS calculations employ an iodine spiking factor of 335 for a steam generator tube rupture and 500 for other non-LOCAs.

c. For the PIS case, the iodine concentration is increased to 60 times the iodine concentration in the Technical Specifications.

d. The LCO concentrations in the Technical Specifications are employed for the initial iodine activity concentrations for the primary and secondary systems, which are 3.7×10^4 Bq/g (1.0 μCi/g) DE I-131 and 3.7×10^3 Bq/g (0.1 μCi/g) DE I-131, respectively. The initial noble gas concentrations in the primary side are conservatively assumed to be at 2.15×10^7 Bq/g (580 μCi/g) DE Xe-133, even though the LCO concentration in the Technical Specifications limits noble gas concentrations below 1.11×10^7 Bq/g (300 μCi/g) DE Xe-133.

e. The timing of an operator action may vary for each event. It is generally assumed that an operator action is not credited in the analysis before 30 minutes after event initiation unless an earlier operator action results in more adverse consequences.

f. An overall steam generator tube leakage of 2.27 L/min (0.6 gpm) is assumed for the duration of the transient.

g. Conservative partition coefficients for iodine and alkali metals are assumed for various occurrences as follows:

1) For primary coolant release via steam generator:

	With SG Dried-Out	Without SG Dried-Out
Flashed Portion	1	1
Unflashed Portion	1	~~100~~ 100 for iodine, 200 for alkali metal

Table 15.0-10

Results of Radiological Consequences of APR1400 Design Basis Accidents

Design Basis Accidents		TEDE (mSv) EAB	TEDE (mSv) LPZ	Dose Criteria (mSv TEDE)
Steam system piping failure	1.0 % Fuel Failure	~~3.33E+01~~	~~2.08E+01~~	2.50E+02
	Pre-accident spike	~~2.66E+00~~	~~1.57E+00~~	2.50E+02
	Event-generated spike	~~1.15E+01~~	~~6.20E+00~~	2.50E+01
Feedwater system pipe break		~~4.22E-01~~	~~2.22E-01~~	2.50E+01
RCP rotor seizure		~~2.09E+01~~	~~1.07E+01~~	2.50E+01
Control element assembly ejection	Containment	~~3.99E+01~~	~~3.77E+01~~	6.30E+01
	Steam generator	~~3.99E+01~~	~~2.23E+01~~	6.30E+01
Failure of small lines carrying primary coolant outside containment		~~1.36E+01~~	~~3.18E+00~~	2.50E+01
Steam generator tube rupture	Pre-accident spike	~~1.10E+01~~	~~2.53E+00~~	2.50E+02
	Event-generated spike	~~6.32E+00~~	~~1.57E+00~~	2.50E+01
Loss-of-coolant accidents		~~2.04E+02~~	~~1.08E+02~~	2.50E+02
Fuel handling accident		~~3.89E+00~~	~~8.56E+00~~	6.30E+01

4.91E+01
3.54E+00
1.04E+01
4.72E+00
1.56E+01
5.90E+01
4.00E+01
4.17E+00
8.07E+00
4.79E+00
2.00E+01
3.89E+01

3.80E+01
1.53E+00
5.08E+00
2.10E-01
8.30E+00
5.58E+01
2.18E+01
1.07E+00
1.77E+00
1.05E+00
1.07E+01
8.56E+00

Release via the Affected Steam Generator

the radionuclides

and alkali metal

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~~ are 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.

are

the partition coefficients for the radioiodine and alkali metal

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 assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient.

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.

Per the accident analyses performed for the APR1400, the radiological consequence analysis for the MSLB are performed for the two (2) cases out of six (6) cases: (1) SLBFPDLOOP and (2) SLBZPLOOPD, which are the most liming cases.

Consistent with NRC RG 1.183, Appendix E, Section 2, the activity assumed in the analysis is based on the activity associated with the projected fuel damage or the maximum Technical Specification values, whichever maximizes the radiological consequences. For SLBZPLOOPD case (Case 6), since fuel damage is not postulated, the maximum Technical Specification values are used. For the SLBFPDLOOP case (Case 5), because 1 percent fuel damage is postulated, the activity due to the projected fuel damage (i.e., 1.0 percent fuel damage) is used. Therefore, two iodine spiking cases (pre-accident iodine spike and concurrent iodine spike) are analyzed for SLBZPLOOPD and the fuel damage case is analyzed for SLBFPDLOOP.

a. It is assumed for the pre-accident iodine spike that a reactor transient has occurred prior to the postulated MSLB and has raised the primary coolant iodine concentration to the maximum value of 2.22×10^6 Bq/g (60 μCi/g) DE I-131.

b. For the event-generated iodine spike that the primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the equilibrium primary coolant iodine concentration of 3.7×10^4 Bq/g (1.0 μCi/g) DE I-131. The assumed iodine spike duration is 8 hours. It is assumed that the iodine activity released from the fuel to RCS is mixed instantaneously and homogeneously with the primary coolant. The event-generated iodine spike isotopic iodine activity appearance rates and resulting iodine activities in the RCS are presented in Table 15A-4.

c. The maximum RCS noble gas concentration for the APR1400 is 2.15×10^7 Bq/g (580 μCi/g) DE Xe-133.

For the alkali metal, the initial concentrations in the primary side are conservatively assumed to be based on one percent fuel defect.

The RCS is assumed to leak into the unaffected and affected SGs at 2.27 L/min (0.6 gpm). From 0 to 0.5 hours, one-half of the P-T-S leakage is into the affected SG, and one-half of the P-T-S leakage is into the unaffected SG. From 0.5 to 8 hours, the all of 2.27 L/min

(0.6 gpm) P-T-S leakage is into the affected SG. It is assumed that the P-T-S leakage continues until the primary system pressure is less than the secondary system pressure or until the temperature of the leakage is less than 100 °C and shutdown cooling is in operation.

The chemical forms of iodine released from the steam generators to the environment are assumed to be 97 percent elemental and 3 percent organic.

All noble gas radionuclides released from the primary system via the P-T-S leak are released to environment without reduction or mitigation.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and are given in Tables ~~2.3-2~~ 2.3-1 through 2.3-12; the breathing rates are given in Table ~~15A-11~~ 15A-12.

15.1.5.5.3 Results

The radiological consequences due to an MSLB accident are presented in Table 15.1.5-13. The results of the MSLB accident analyses indicate that the EAB and LPZ doses due to an MSLB accident with a pre-accident iodine spike, an event-generated iodine spike, and 1 percent fuel failure are within their allowable dose criteria limits, which are 100 percent, 10 percent, and 100 percent of the 10 CFR 50.34(a)(1) value, respectively.

The MCR and TSC doses for all cases are also within the dose limits in GDC 19.

15.1.5.6 Conclusions

A post-trip RTP does not occur in all cases so that the fuel integrity is not challenged by this event (an increase in heat removal by the secondary system). Consequently, the core remains in place and is unaffected with no loss of core cooling capability.

Also, the RCS pressures remain below 193.34 kg/cm^{2}A (2,750 psia), and the steam generator pressures remain below 92.83 kg/cm^{2}A (1,320 psia).

For pre-trip fuel degradation, the maximum potential for radiological releases due to fuel failure occurs in SLBs outside the containment with a LOOP concurrent with

Table 15.1.5-12 (1 of 3)

Parameters Used in Evaluating the Radiological Consequences of the Steam Line Break Outside Containment

Parameter	Value
Source Terms	
Reactor Core Power Level	4,062.66 MWt
Percent of Fuel Assumed to Undergo Departure from Nucleate Boiling (DNB)	1 % For SLBFPDLOOP 0 % For SLBZPLOOPD
Percent of Fuel Assumed to Melt	0 % For SLBFPDLOOP 0 % For SLBZPLOOPD
Radial Peaking Factor	1.80
Initial RCS Mass	274,392 kg (604,930 lbm) for SLBFPDLOOP 286,829 kg (632,340 lbm) for SLBZPLOOPD
Initial Steam Generator Liquid Mass per SG	54,592 kg (120,353 lbm) for SLBFPDLOOP 187,658 kg (413,709 lbm) for SLBZPLOOPD
Initial RCS Iodine Specific Activity	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131
Initial Secondary Liquid Iodine Specific Activity	3.7×10^3 Bq/g (0.1 μCi/g) DE I-131
Initial RCS Noble Gas Specific Activity	2.15×10^7 Bq/g (580 μCi/g) DE Xe-133
RSC Iodine Specific Activity Used for Pre-accident Iodine Spike Case	2.22×10^6 Bq/g (60 μCi/g) DE I-131
Event-generated Iodine Spiking Factor	500
Duration of Event-generated Iodine Spike	8 hr
Chemical Forms of Iodine Released from the SG to the Environment	97 % elemental and 3 % organic

Initial RCS Alkali Metal Specific Activity RCS concentrations with 1% fuel defect

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) 39,010 kg (86,000 lbm) 0.0 kg (0.0 lbm)
Termination of Release from Affected SG	30 min
Unaffected SG P-T-S Leak Duration, and Termination of Release from Unaffected SG	8 hr
SG Liquid Iodine Partition Coefficient	100
Letdown System Flow Rate	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

SG Liquid Alkali Metal Partition Coefficient 200

Table 15.1.5-12 (3 of 3)

Parameter	Value
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodine removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m^3/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors 0 ~ 24 hr 24 ~ 96 hr 96 ~ 720 hr	 100 % 60 % 40 %
Onsite χ/Qs	See Tables 2.3.2 ~ 2.3.12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

Table 15.1.5-13 (1 of 2)

Radiological Consequences of Steam Line Breaks Outside Containment

Pre-accident Iodine Spike Case (SLBZPLOOPD)

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	1.70E+00	2.18E+00	1.46E+00
P-T-S Noble Gas Release	1.38E-02	1.20E-02	8.55E-03
Secondary Liquid Iodine Release	4.27E-01	4.64E-01	1.04E-01
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	1.30E+01	2.66E+00	1.57E+00
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

Replace this with A

Table 15.1.5-13 (2 of 2)

Event-generated Iodine Spike Case (SLBZPLOOPD)

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	6.77E+00	1.11E+01	6.09E+00
P-T-S Noble Gas Release	1.38E-02	1.20E-02	8.55E-03
Secondary Liquid Iodine Release	4.27E-01	4.64E-01	1.04E-01
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	1.81E+01	1.15E+01	6.20E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

1 % Fuel Failure Case (SLBFPDLOOP)

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	2.26E+01	2.89E+01	1.96E+01
P-T-S Noble Gas Release	7.87E-01	1.14E+00	4.96E-01
Secondary Liquid Iodine Release	3.52E+00	3.18E+00	7.04E-01
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	3.78E+01	3.33E+01	2.08E+01
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

Replace this with A

Table 15.1.5-13 (1 of 2)

Radiological Consequences of Steam Line Breaks Outside Containment

Pre-accident Iodine Spike Case (SLBZPLOOPD)

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	4.08E-01	1.62E+00	1.08E+00
P-T-S Noble Gas Release	7.84E-03	8.90E-03	6.36E-03
P-T-S Alkali Metal Release	6.85E-03	2.75E-02	1.78E-02
Secondary Liquid Iodine Release	1.43E+00	1.88E+00	4.21E-01
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	2.10E+01	3.54E+00	1.53E+00
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

Event-generated Iodine Spike Case (SLBZPLOOPD)

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	1.64E+00	8.44E+00	4.64E+00
P-T-S Noble Gas Release	7.84E-03	8.90E-03	6.36E-03
P-T-S Alkali Metal Release	6.85E-03	2.75E-02	1.78E-02
Secondary Liquid Iodine Release	1.43E+00	1.88E+00	4.21E-01
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	2.22E+01	1.04E+01	5.08E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

A

Table 15.1.5-13 (2 of 2)

1 % Fuel Failure Case (SLBFPDLOOP)

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	7.67E+00	2.43E+01	1.95E+01
P-T-S Noble Gas Release	4.38E-01	8.42E-01	3.67E-01
P-T-S Alkali Metal Release	7.14E+00	2.27E+01	1.78E+01
Secondary Liquid Iodine Release	1.44E+00	1.19E+00	2.62E-01
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	3.58E+01	4.91E+01	3.80E+01
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

A

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.

the radioiodine and alkali metal are

Release via Affected Steam Generator

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

and alkali metal

are

the affected SG ~~is~~ assumed to flash to vapor and ~~is~~ leaked into the containment through the break in the feedwater line without crediting any holdup in the affected SG.

Release via the Unaffected Steam Generator

During the first 30 minutes of the FLB event, the iodine and alkali metal activity in the P-T-S leakage is mixed with the SG liquid and assumed to become vapor at a rate that is a function of the steaming rate and a ~~an iodine~~ partition coefficient, which is released through the MSSVs of the unaffected SG. At 30 minutes, operator action is taken to open the unaffected SG ADV to cool down the RCS. The radioactivity in the bulk water is assumed to become vapor at a rate that is ~~the~~ function of the steaming rate and ~~the~~ partition coefficient. The steam release from the unaffected SG continues for 8 hours until the shutdown cooling system is aligned to dissipate heat.

a

Release via the Condenser

Prior to the LOOP, the contaminated secondary steam in the unaffected and affected SGs is released to the condenser. The steam release to the condenser is not considered in the post-FLB activity release to the environment due to the tortuous path to the condenser through the turbines and moisture separators, and condenser hold-up time.

Figure 15A-2 in Appendix 15A shows the leakage paths and transport of the activity released to environment, MCR, and TSC during an FLB event.

15.2.8.5.2 Input Parameters and Initial Conditions

The design basis FLB accident is analyzed using a conservative set of assumptions based on NRC RG 1.183 and the APR1400 design inputs. Input parameter values used for an FLB radiological consequence evaluation are presented in Table 15.2.8-3.

No fuel damage is postulated for the FLB accident. The iodine activity in the RCS is assumed to be the maximum coolant activity allowed by the Technical Specifications including effect of event-generated iodine spike that increases the equilibrium fission product activity release rate from fuel by a factor of 500. The event-generated iodine spike case is considered to evaluate the resulting EAB, LPZ, MCR, and TSC doses because:

a. The iodine release due to the event-generated iodine spike by a factor of 500 is higher than the pre-accident iodine spike of 2.22×10^6 Bq/g (60 μCi/g).

b. The offsite allowable dose limits for the event-generated iodine spike are limiting in comparison to the offsite dose limits for the pre-accident iodine spike per SRP 15.0.3, Table 1.

It is assumed that the primary system transient associated with the FLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the equilibrium primary coolant iodine concentration of 3.7×10^4 Bq/g (1.0 μCi/g) DE I-131. The assumed iodine spike duration is 8 hours. The iodine activity released from the fuel to RCS is assumed to be mixed instantaneously and homogeneously with the primary coolant. The event-generated iodine spike isotopic iodine activity appearance rates in the RCS are calculated in Table 15A-7.

The maximum RCS noble gas concentration is 2.15×10^7 Bq/g (580 μCi/g) DE Xe-133.

The RCS is assumed to leak into both the affected and unaffected SGs at a total P-T-S leak rate of 2.27 L/min (0.6 gpm). It is assumed that the P-T-S leakage into the SGs continues until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F) and shutdown cooling is in operation. The RCS is assumed to leak into the SGs for 8 hours until the shutdown cooling system is initialized. For the alkali metal, the initial concentrations in the primary side are conservatively assumed to be based on one percent fuel defect.

The chemical forms of iodine released from the SGs to the environment are assumed to be 97 percent elemental and 3 percent organic.

All noble gas radionuclides released from the primary system via the P-T-S leak are released to environment without reduction or mitigation.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and are provided in Tables ~~2.3-2~~ 2.3-1 through 2.3-12; the breathing rates are given in Table ~~15A-11~~ 15A-12.

Table 15.2.8-3 (1 of 3)

Parameters Used in Evaluating the Radiological Consequences of a Feedwater Line Break

Parameter	Value
Source Terms	
Reactor Core Power Level	4,062.66 MWt
Percent of Fuel Assumed to Undergo Departure from Nucleate Boiling (DNB)	0 %
Percent of Fuel Assumed to Melt	0 %
Initial RCS Mass	288,086 kg (635,120 lbm)
Initial Steam Generator Liquid Mass	89,721 kg/SG (197,802 lbm/SG)
Initial RCS Iodine Specific Activity	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131
Initial Secondary Liquid Iodine Specific Activity	3.7×10^3 Bq/g (0.1 μCi/g) DE I-131
Initial RCS Noble Gas Specific Activity ~~Limit~~	2.15×10^7 Bq/g (580 μCi/g) DE Xe-133
Event-generated Iodine Spiking Factor	500
Duration of Event-generated Iodine Spike	8 hr
Chemical Forms of Iodine Released from the SG to the Environment	97 % elemental and 3 % organic

Initial RCS Alkali Metal Specific Activity RCS concentrations with 1% fuel defect

Table 15.2.8-3 (2 of 3)

Parameter	Value
Secondary System Activity Transport Model	
Integrated P-T-S leakage 0 – 2 hr 0 – 8 hr	 272 kg (601 lbm) 818 kg (1,803 lbm)
Release from affected SG through MSSV Total liquid mass Duration of release	 2,810 kg (6,200 lbm) 20 sec
Total mass released from affected SG to containment through the break in the feedwater line (0 – 0.5 hr)	1,080 kg (239,000 lbm)
Total steam mass release from intact SG 0 – 0.5 hr 0.5 – 2 hr via ADV 2 – 8 hr via ADV	 69,800 kg (154,000 lbm) 481,000 kg (1,060,000 lbm) 490,000 kg (1,080,000 lbm)
FLB isolation time	30 minutes
Primary-to-secondary Leakage Rate through SGs	2.27 L/min (0.6 gpm)
Termination of intact SG P-T-S leak	8 hr
SG liquid iodine partition coefficient	100
Letdown system flow rate	18,100 kg/hr 10^4 (39,842 lbm/hr)
RCS Fluid Released to IRWST	1,660 kg (3,651 lbm)
Duration of RCS Fluid Released to IRWST	1 min

SG Liquid Alkali Metal Partition Coefficient 200

Table 15.2.8-3 (3 of 3)

Parameter	Value
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodine removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m³/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors	
0 – 24 hr	100 %
24 – 96 hr	60 %
96 – 720 hr	40 %
Onsite χ/Qs	See Tables 2.3.2 ~ 2.3.12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

Table 15.2.8-4

Radiological Consequences of Feedwater Line Break

Post-FLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
Containment Leakage	6.78E-03	7.17E-03	3.03E-02
P-T-S Iodine Release	1.48E-01	2.80E-01	1.37E-01
P-T-S Noble Gas Release (Unaffected SG)	6.63E-03	5.98E-03	4.28E-03
P-T-S Noble Gas Release (Containment)	9.22E-08	1.20E-07	4.09E-07
Secondary Liquid Iodine Release	6.95E-02	1.29E-01	5.08E-02
External Cloud	6.88E-01	0.00E+00	0.00E+00
Containment Shine	0.00E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	1.11E+01	4.22E-01	2.22E-01
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

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Table 15.2.8-4

Radiological Consequences of Feedwater Line Break

Concurrent-accident Iodine Spike Case (FLB)

Post-FLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	3.67E-01	3.01E+00	1.63E+00
P-T-S Noble Gas Release	4.31E-03	7.86E-03	5.60E-03
P-T-S Alkali Metal Release	1.18E-03	9.69E-03	5.06E-03
Containment Leakage (Iodine & Alkali Metal)	3.27E-03	5.26E-03	2.13E-02
Containment Leakage (Noble Gas)	9.56E-08	8.19E-08	2.79E-07
Secondary Liquid Iodine Release	2.64E-01	1.69E+00	4.35E-01
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	1.98E+01	4.72E+00	2.10E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

15.3.3.5.1 Evaluation Model

Figure 15A-3 in Appendix 15A shows the leakage paths and transport of the activity released to environment and control room during RCP rotor seizure event.

Release via the Steam Generators

are The iodine and alkali metal

In this analysis, the affected steam generator (SG) is the SG associated with the RCS loop with the RCP rotor seizure. Both the affected and unaffected SG tubes are expected to be uncovered during the first 30 minutes of the RCP rotor seizure. During this period, the P-T-S leakage flashing fraction averages 15.0 percent. After 30 minutes, the tubes in both SGs are assumed to remain covered during the RCP rotor seizure event. ~~Per NRC RG 1.183, the iodine~~ activity in the P-T-S leakage ~~is~~ assumed to mix with SG coolant without flashing during the period of submergence, and ~~an iodine partition coefficient of 100 is~~ assumed. The RCS ~~iodine~~ release continues from 0 to 2 hours in the affected SG and from 0 to 8 hours for the unaffected SG until the shutdown cooling system is initialized at 8 hours.

the partition coefficients of 100 for iodine and 200 for alkali metal are

Release via the Condenser

The release of fission products from the secondary system is evaluated with the assumption of a coincident LOOP. The offsite power is assumed to be lost so that the main steam condenser is not available for removal of the decay heat.

15.3.3.5.2 Input Parameters and Initial Conditions

The design basis RCP rotor seizure accident is analyzed using a conservative set of assumptions and the APR1400 design inputs. The analysis is performed using the guidance in NRC RG 1.183, Appendix G.

The APR1400 RCP rotor seizure analysis predicts that no more than 7 percent of the fuel in the core will fail due to DNBR, and no fuel will melt. The assumed inventory of fission products in the reactor core and available for release to the reactor coolant is based on the maximum power level of 4,062.66 MWt corresponding to current fuel enrichment and fuel burnup, which is 1.02 times the APR1400 licensed thermal power of 3,983 MWt. The

activity released from the fuel is assumed to be released instantaneously and homogeneously through the primary coolant.

The chemical forms of iodine released from the steam generators to the environment are 97 percent elemental and 3 percent organic. These iodine chemical forms are applied to iodine releases from the P-T-S leakage and from the secondary liquid.

The secondary coolant iodine concentration is limited to 3.7 × 10^3 Bq/g (0.1 μCi/g) DE I-131. The RCS DE I-131 isotopic concentration profile is multiplied by a factor of 0.1 representing the ratio of the secondary to primary limits to determine the secondary coolant iodine concentration. The secondary coolant iodine concentration is multiplied by the total coolant mass in both steam generators to calculate the total secondary coolant iodine inventory.

This analysis models the RCS leakage limit of 1.14 L/min (0.3 gpm) gpm of P-T-S leakage through each SG. The primary coolant density used in converting the volumetric P-T-S leak rates to mass leak rates is 1.0 g/cm^3 (62.4 lbm/ft^3). The P-T-S leakage continues until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F), and the release of radioactivity is assumed to continue until shutdown cooling is in operation and releases from the steam generator have terminated. In this analysis, the RCS is conservatively assumed to leak into the affected SG for 0 to 2 hours and into unaffected SG for 0 to 8 hours until the shutdown cooling system is initialized. All noble gas radionuclides released from the primary system (through the P-T-S leak) are released to the environment without reduction or mitigation. Input parameter values used for RCP rotor seizure radiological consequence evaluation are presented in Table 15.3.3-3.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and are in Tables 2.3-1 through 2.3-12; breathing rates are given in Table ~~15A-11~~.

15A-12

15.3.3.5.3 Results

The radiological consequences due to RCP rotor seizure accident are presented in Table 15.3.3-4. The results of the RCP rotor seizure accident analyses indicate that the EAB and LPZ doses are within their allowable dose criteria limit, which is 10 percent of the 10 CFR

Initial RCS Alkali Metal Specific Activity RCS concentrations with 1% fuel defect

Table 15.3.3-3 (1 of 2)

Parameters Used in Evaluating the Radiological Consequences of the Reactor Coolant Pump Rotor Seizure

Parameter	Value
Source Terms	
Reactor Core Power Level	4,062.66 MWt
Percent of Fuel Assumed to Undergo Departure from Nucleate Boiling (DNB)	7.0 %
Percent of Fuel Assumed to Melt	0 %
Radial Peaking Factor	1.80
RCS Mass	272,000 kg (600,000 lbm)
Initial RCS Iodine Specific Activity	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131
Initial RCS Noble Gases Specific Activity	2.15×10^7 Bq/g (580 μCi/g) DE X-133
Initial Secondary Liquid Iodine Specific Activity	3.7×10^3 Bq/g (0.1 μCi/g) DE I-131
Secondary System Activity Transport Model	
Primary-to-Secondary Leak Rate through SGs	2.27 L/min (0.6 gpm) for two SGs
Density of P-T-S Leakage Measurement	1 g/cm^3 (62.4 lbm/ft^3)
Steam Mass Release from Affected SG 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 56,800 kg (125,000 lbm) 193,000 kg (425,000 lbm) 0.0 kg (0.0 lbm)
Steam Mass Release from Unaffected SG 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 53,900 kg (119,000 lbm) 199,000 kg (439,000 lbm) 609,000 kg (1,340,000 lbm)
Initial SG Liquid Mass	84,200 kg (186,000 lbm)
Primary-To-Secondary Leak Flashing Fraction	15.0 % average for 30 min after onset of accident
SG Liquid Iodine Partition Coefficient ~~for SG Liquid Iodine Release~~	100

SG Liquid Alkali Metal Partition Coefficient 200

Table 15.3.3-3 (2 of 2)

Parameter	Value
Chemical Form of Iodine Released from SG to Environment	
Elemental	97 %
Organic	3 %
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodine removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m³/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors	
0 ~ 24 hr	100 %
24 ~ 96 hr	60 %
96 ~ 720 hr	40 %
Onsite χ/Qs	See Tables 2.3.2 ~ 2.3.12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

Table 15.3.3-4

Radiological Consequences of the Reactor Coolant Pump Rotor Seizure

Post-RCP Rotor Seizure Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	9.19E+00	1.29E+01	8.10E+00
P-T-S Noble Gas Release	3.94E+00	7.89E+00	2.57E+00
Secondary Liquid Iodine Release	5.03E-02	7.51E-02	3.70E-02
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	2.45E+01	2.09E+01	1.07E+01
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

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Table 15.3.3-4

Radiological Consequences of the Reactor Coolant Pump Rotor Seizure

Post-RCP Rotor Seizure Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	1.54E+00	6.84E+00	4.46E+00
P-T-S Noble Gas Release	1.45E+00	5.85E+00	1.91E+00
P-T-S Alkali Metal Release	7.06E-01	2.82E+00	1.91E+00
Secondary Liquid Iodine Release	1.38E-02	5.96E-02	2.77E-02
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	2.28E+01	1.56E+01	8.30E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

15.4.8.5.1 Evaluation Model

Figure 15A-4 in Appendix 15A shows the leakage paths and transport of the activity released to environment, MCR, and TSC during a CEA ejection event.

Containment Leakage Case

, alkali metals

For this case, the activity released from the failed fuel is assumed to be instantaneously and homogeneously distributed in the containment following a CEA ejection. This analysis also releases 100 percent of the iodine and noble gases initially present in the RCS. The activity in the containment is subject to be released by the design leak rate specified in the Technical Specifications.

Secondary System Release Case

~~The preferred means to release steam for the cooldown is by dumping steam to the condenser.~~ ~~To maximize the secondary system release doses, it is assumed~~ that offsite power is lost so that the main steam condenser is not available. Following the CEA ejection event, the reactor is shut down and the plant is cooled down by discharging secondary coolant through the two SGs using the combination of one or more ADVs and MSSVs.

The SG tubes are expected to be uncovered during the first 30 minutes of the CEA ejection event because the MSSVs are open to cool down the RCS. During this period, the P-T-S leakage flashing fraction averages 15.0 percent. After 30 minutes, the SG tubes remain covered during the CEA ejection. During the first 30 minutes, the unflashed P-T-S leakage mixes with the SG secondary coolant, and after 30 minutes all of the P-T-S leakage mixes with the SG secondary coolant. ~~An iodine partition coefficient of 100 is assumed for transporting the iodine from the SG secondary coolant.~~ ~~For the secondary liquid iodine release from the SG, an iodine partition coefficient of 100 is assumed, which is consistent with the value recommended in NRC RG 1.183.~~

The partition coefficient of 100 for iodine and 200 for alkali metal are assumed for the secondary liquid release from the SG.

15.4.8.5.2 <u>Input Parameters and Initial Conditions</u>

The design basis CEA ejection accident is analyzed using a conservative set of assumptions and APR1400 design inputs. The CEA ejection analysis is performed using the guidance in NRC RG 1.183, Appendix H.

and 12 percent of the core inventory of the alkali metals are

Per NRC RG 1.183, Appendix H, Section 1, for the CEA ejection accident, the release from the failed fuel is based on the number of fuel rods to undergo DNB and the assumption that 10 percent of the core inventory of the noble gases and iodines ~~is~~ in the fuel gap. The expected number of fuel rods in DNB is 10 percent of the core. The failed fuel is modeled with a radial peaking factor of 1.80. Fuel melt is not expected to occur during the CEA ejection. The CEA ejection releases more iodine and noble gases from fuel gap than the other non-LOCA events as specified in NRC RG 1.183, Table 3. The assumed inventory of fission products in the reactor core and available for release to the reactor coolant is based on the maximum power level of 4,062.66 MWt corresponding to current fuel enrichment and fuel burnup, which is 1.02 times the APR1400 licensed thermal power of 3,983 MWt with the cycle burnup of 56.4 GWD/MTU.

The secondary coolant iodine concentration is limited to 3.7×10^3 Bq/g (0.1 μCi/g) DE I-131. The RCS DE I-131 isotopic concentration profile is multiplied by a factor of 0.1, representing 10 percent of primary coolant concentration to determine the secondary coolant iodine concentration. The secondary coolant iodine concentration is multiplied by the total coolant mass in both steam generators to calculate the total secondary coolant iodine inventory.

The chemical forms of iodine released from the steam generators to the environment are 97 percent elemental and 3 percent organic. These iodine chemical forms apply to iodine releases from the P-T-S leakage and from the secondary liquid.

Input parameter values used for CEA ejection radiological consequence evaluation are presented in Table 15.4.8-4.

A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, or other engineered safety features can be taken into account. This analysis

credits aerosol removal by natural deposition. No credit is taken for elemental iodine or aerosol removal by containment sprays.

The primary containment leaks at a Technical Specification peak pressure leak rate of 0.1 percent by volume for the first 24 hours. This leak rate is reduced to 0.05 weight percent after 24 hours.

The P-T-S leakage is at the RCS operational leakage limit of 1.14 L/min (0.3 gpm) through any one SG as specified in the Technical Specification. The P-T-S leak exists until shutdown cooling is in operation and releases from the steam generators have been terminated at 8.0 hours. The primary coolant density used in converting the volumetric P-T-S leak rates to mass leak rates is 1.0 g/cm^3 (62.4 lbm/ft^3). All noble gas radionuclides released from the primary system via the P-T-S leaks are released to the environment without reduction or mitigation.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and are provided in Tables ~~2.3-2~~ 2.3-1 through 2.3-12; breathing rates are given in Table ~~15A-11~~ 15A-12.

15.4.8.5.3 <u>Results</u>

The radiological consequences due to a CEA ejection are presented in Table 15.4.8-5. The results of the CEA ejection analyses indicate that the EAB and LPZ doses are within their allowable dose limit, which is 25 percent of the 10 CFR 50.34(a)(1) value, as specified in SRP 15.0.3. The MCR and TSC doses are also within the dose limit in GDC 19.

15.4.8.6 <u>Conclusions</u>

For the spectrum of CEA ejection evaluated, none of the power excursions causes the fuel temperatures to reach the limiting fuel melting temperature or the fuel enthalpy limits. For the events that exceeded the DNBR limit, the number of fuel failures was less than the value allowed for the radiological release limit. The peak RCS pressure remains below 110 percent of the RCS design pressure. The stresses due to the primary pressure response during the transients do not exceed Service Limit C as defined in the ASME Code.

The doses at the EAB, LPZ, MCR, and TSC are within their allowable dose criteria.

Table 15.4.8-4 (1 of 3)

Parameters Used in Evaluating the Radiological Consequences of a CEA Ejection

Parameter	Value
Source Terms	
Reactor Core Power Level	4,062.66 MWt
Percent of Fuel Assumed to Undergo DNB	10 %
Initial RCS Iodine Specific Activity	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131
Initial RCS Noble Gases Specific Activity	2.15×10^7 Bq/g (580 μCi/g) DE Xe-133
Initial Secondary Liquid Iodine Specific Activity	3.7×10^3 Bq/g (0.1 μCi/g) DE I-131
Radial Peaking Factor	1.80
RCS Initial Mass	267,620 kg (590,000 lbm)
Containment Leakage Transport Model	
Containment Net Free Volume	8.86×10^4 m^3 (3.13×10^6 ft^3)
Reactor Coolant Mass Released to Containment	2.68×10^5 kg (5.90×10^5 lbm)
Credit for Radioactive Decay during Hold up in Containment In Transit to Dose Points	 Applicable Not Applicable
Iodine Chemical Form Aerosol (CsI) Elemental Organic	 95.0 % 4.85 % 0.15 %
Containment Aerosol Natural Deposition Removal	Powers model with a 10-percentile probability
Containment Leak Rate	0.1 %/day (0 ~ 24 hr) 0.05 %/day (24 ~ 720 hr)

Initial RCS Alkali Metal Specific Activity RCS concentrations with 1% fuel defect

Table 15.4.8-4 (2 of 3)

Parameter	Value
Secondary System Release Transport Model	
Primary-To-Secondary Leak Rate through SGs	2.27 L/min (0.6 gpm) through all SGs 1.14 L/min (0.3 gpm) through any one SG
Steam Generator Liquid Mass	104,326 kg (230,000 lbm)
Primary-To-Secondary Leak Duration	8 hr
Steam Mass Released from Both Intact SGs to Environment 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 118,660 kg (261,600 lbm) 650,737 kg (1,434,630 lbm) 624,538 kg (1,376,870 lbm)
Primary-To-Secondary Leak Flashing Fraction	15.0 % average for 1,800 sec after onset of accident
SG Liquid Iodine Partition Coefficient ~~for SG Liquid Iodine Release~~	100
~~Alkali Material (Cs, Rb) Partition coefficient~~ SG Liquid Alkali Metal Partition Coefficient	~~5.0×10^{-3}~~ 200
Chemical Form of Iodine Released from SG Elemental Organic Iodine	 97 % 3 %

Table 15.4.8-4 (3 of 3)

Parameter	Value
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodine removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m^3/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors 0 ~ 24 hr 24 ~ 96 hr 96 ~ 720 hr	 100 % 60 % 40 %
Onsite χ/Qs	See Tables 2.3.2 ~ 2.3.12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

Table 15.4.8-5

Radiological Consequences of CEA Ejection

Containment Leakage Case

Post-CEA Ejection Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
Containment Leakage	1.27E+01	3.99E+01	3.77E+01
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	2.36E+01	3.99E+01	3.77E+01
Allowable TEDE Limit	5.00E+01	6.30E+01	6.30E+01

Steam System Release Case

Post-CEA Ejection Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	1.54E+01	1.94E+01	1.35E+01
P-T-S Noble Gas Release	1.30E+01	2.04E+01	8.78E+00
Secondary Liquid Iodine Release	6.62E-02	1.22E-01	4.66E-02
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	3.94E+01	3.99E+01	2.23E+01
Allowable TEDE Limit	5.00E+01	6.30E+01	6.30E+01

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Table 15.4.8-5

Radiological Consequences of CEA Ejection

Containment Leakage Case

Post-CEA Ejection Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
Containment Leakage	1.63E+01	5.90E+01	5.58E+01
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	3.54E+01	5.90E+01	5.58E+01
Allowable TEDE Limit	5.00E+01	6.30E+01	6.30E+01

Steam System Release Case

Post-CEA Ejection Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	2.60E+00	1.46E+01	1.01E+01
P-T-S Noble Gas Release	6.55E+00	2.04E+01	8.79E+00
P-T-S Alkali Metal Release	8.48E-01	5.01E+00	2.95E+00
Secondary Liquid Iodine Release	1.83E-02	1.22E-01	4.65E-02
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	2.91E+01	4.00E+01	2.18E+01
Allowable TEDE Limit	5.00E+01	6.30E+01	6.30E+01

methodology guidance in NRC RG 1.183, the TEDE dose criteria, and the plant-specific bounding design information applicable to the APR1400.

15.6.2.5.1 Evaluation Model

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

Release via the Auxiliary Building

except for the alkali metal

RCS fluid is released into the Auxiliary Building (AB) through the letdown line break, and from there, is assumed to be directly released to the environment through the AB exhaust vent without mixing with the AB volume for conservatism. The fraction ~~of iodine~~ assumed to become airborne and available for release to the atmosphere through the AB exhaust vent, without credit for plateout, is equal to the fraction of the coolant flashing into steam in the depressurization process. In addition, a portion of the iodine in the unflashed leakage vaporizes with a partition coefficient. The LDLB iodine is released through the building ventilation system to the environment without any credit for filtration of radioactivity.

Release via the Steam Generators

In order to remove decay heat, the plant begins to release the secondary coolant to atmosphere. After 30 minutes, the SGs are used to cool down the RCS and the contaminated steam present in the SGs is released to the environment through the ADV. The RCS iodine activity entering the SGs via P-T-S leakage is assumed to mix with, and be diluted within, the bulk water in the SGs. The steam release from the SGs continues to 8 hours until the shutdown cooling system is aligned to dissipate heat.

Release via the Condenser

The contaminated secondary steam in the SGs is released to the condenser until operator actions are taken. But the steam release to the condenser is not considered in the post-LDLB activity release to the environment due to the path to the condenser through the turbines and moisture separators, and condenser hold-up time.

Figure 15A-5 in Appendix 15A shows the leakage paths and transport of the activity released to environment, MCR, and TSC during an LDLB event.

15.6.2.5.2 Input Parameters and Initial Conditions

The design basis LDLB accident is analyzed using a conservative set of assumptions and the APR1400 design inputs. Input parameter values used for LDLB radiological consequence evaluation are presented in Table 15.6.2-4.

No fuel damage is postulated for the LDLB accident. Consistent with SRP 15.6.2, it is assumed for the event-generated iodine spike that the primary system transient associated with the LDLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the equilibrium primary coolant iodine concentration of 3.7×10^4 Bq/g (1.0 μCi/g) DE I-131. The assumed iodine spike duration is 8 hours. It is assumed that the iodine activity released from the fuel to RCS is mixed instantaneously and homogeneously with the primary coolant. The event-generated iodine spike isotopic iodine activity appearance rates in the RCS are calculated in Table 15A-6.

The maximum RCS noble gas concentration for the APR1400 is 2.15×10^7 Bq/g (580 μCi/g) DE Xe-133 as shown in Table 15A-8.

The RCS flashing fraction is determined to be 0.259 based on the enthalpy difference under circumstance of primary coolant leak by assuming the leakage to be constant enthalpy process. Iodine partition coefficient of 10 is conservatively considered for the unflashed RCS fluid. For the alkali metal, the initial concentrations in the primary side are conservatively assumed to be based on one percent fuel defect.

The RCS is assumed to leak into the SGs at a P-T-S leak rate of 2.27 L/min (0.6 gpm). It is assumed that the P-T-S leakage into the SGs continues until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F) and shutdown cooling is in operation. The RCS is assumed to leak into the SGs for 8 hours until the shutdown cooling system is initialized.

The time required to isolate any break by operator action is conservatively assumed to be 30 minutes.

It is assumed that the chemical forms of iodine released from the steam generators to the environment are 97 percent elemental and 3 percent organic.

All noble gas radionuclides released from the primary system via the P-T-S leak are released to environment without reduction or mitigation. The iodine activity in the bulk water is assumed to become vapor at a rate that is a function of the steaming rate and iodine partition coefficient.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and listed in Tables ~~2.3-2~~ 2.3-1 through 2.3-12; breathing rates are given in Table ~~15A-11~~ 15A-12.

15.6.2.5.3 Results

The radiological consequences due to LDLB accident are presented in Table 15.6.2-5. The results of the LDLB accident analyses indicate that the EAB and LPZ doses due to a LDLB accident with an event-generated iodine spike are within their allowable dose criteria limits, which are 10 percent of the 10 CFR 50.34(a)(1) value as specified in SRP 15.0.3. The MCR and TSC doses are also within the limit in GDC 19.

15.6.2.6 Conclusions

The double-ended break of a letdown line outside the containment upstream of the letdown isolation valve results in a gradual depressurization of the RCS. The minimum DNBR remains above 1.29, thus providing reasonable assurance of fuel cladding integrity. The doses at the EAB, LPZ, MCR, and TSC are within the allowable criteria specified. Also, the RCS pressures remain below 193.34 kg/cm^{2}A (2,750 psia), and the steam generator pressures remain below 92.83 kg/cm^{2}A (1,320 psia).

15.6.3 Steam Generator Tube Failure

The steam generator tube rupture (SGTR) accident is a penetration of the barrier between the RCS and the main steam system and is the result of a double-ended guillotine break of a steam generator U-tube. An SGTR is classified as a PA and is not expected during the lifetime of the plant, but the event is postulated because the consequences include the potential release of significant amounts of radioactivity.

Release through the Affected Steam Generator

The post-SGTR thermal hydraulic condition in the affected SG is such that a large amount of RCS coolant released through the ruptured tube increases the secondary side coolant mass inventory, which eliminates the possibility of a steam generator dryout condition. The ~~iodine and noble gas~~ radioactivity in the P-T-S rupture flow is assumed to flash to vapor in the affected SG and be released to the environment without mitigation. The P-T-S rupture flow that does not flash is assumed to mix with the secondary liquid during the periods of SG tube submergence. The contaminated steam present in the affected SG is released to the environment through the MSSVs until operator action is taken to open the ADV aligned to the unaffected SG. The radioactivity in the bulk water is assumed to become vapor at a rate that is ~~the~~ a function of the steaming rate and ~~the~~ a partition coefficient.

Release via the Unaffected Steam Generator

With regard to the unaffected SG used for plant cooldown, the P-T-S leakage is assumed to mix with the secondary water without flashing during periods of total tube submergence and become vapor at a rate that is a function of the steaming rate and an assumed iodine partition coefficient. The steam release from the unaffected SG continues to 8 hours until the shutdown cooling system is aligned to dissipate heat.

Release via the Condenser

Prior to the LOOP, the contaminated secondary steam in the unaffected and affected SG is released to the condenser. But the steam release to the condenser is not considered in the post-SGTR activity release to the environment due to the path to the condenser through the turbines and moisture separators, and condenser hold-up time.

Figure 15A-6 in Appendix 15A shows the leakage paths and transport of the activity released to environment, MCR, and TSC during an SGTR event.

15.6.3.2.5.2 Input Parameters and Initial Conditions

The design basis SGTR accident is analyzed using a conservative set of assumptions based on NRC RG 1.183, Appendix F and the APR1400 design inputs. Input parameter values used for SGTR radiological consequence evaluation are presented in Table 15.6.3-5.

Since no fuel damage is postulated for the SGTR event, the activity released is the maximum RCS activity allowed by the Technical Specifications. Two cases of iodine spiking corresponding to a pre-accident iodine spike and an event-generated iodine spike are assumed.

It is assumed for the pre-accident iodine spike that a reactor transient has occurred prior to the postulated SGTR and has raised the primary coolant iodine concentration to the maximum value of 2.22×10^6 Bq/g (60 μCi/g) DE I-131.

For the event-generated iodine spike, the primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant, which is expressed in curies per unit time, increases to a value 335 times greater than the release rate corresponding to the equilibrium primary coolant iodine concentration of 3.7×10^4 Bq/g (1.0 μCi/g) DE I-131. The assumed iodine spike duration is 8 hours. It is assumed that the iodine activity released from the fuel to RCS is mixed instantaneously and homogeneously with the primary coolant. For the alkali metal, the initial concentrations in the primary side are conservatively assumed to be based on one percent fuel defect.

The maximum RCS noble gas concentration for the APR1400 is 2.15×10^7 Bq/g (580 μCi/g) DE Xe-133.

The total primary coolant break flow into the affected SG is 64,200 kg (141,448 lbm), which is calculated by thermal-hydraulic analysis.

The RCS is assumed to leak into the unaffected and affected SGs at a P-T-S leak rate of 2.27 L/min (0.6 gpm). The total primary coolant flow into the unaffected and affected SG via P-T-S is 1,090 kg (2,404 lbm). The P-T-S leakage into the SG continues until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100 °C (212 °F) and shutdown cooling is in operation.

It is assumed that all noble gas radionuclides released from the primary system are released to environment without reduction or mitigation. The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient.

The chemical forms of iodine released from the steam generators to the environment are assumed to be 97 percent elemental and 3 percent organic.

All noble gas radionuclides released from the primary system via the P-T-S leak are released to environment without reduction or mitigation.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and are in Tables ~~2.3-2~~ 2.3-1 through 2.3-12, and the breathing rates are given in Table ~~15A-11~~ 15A-12.

15.6.3.2.5.3 Results

The radiological consequences due to SGTR accident are presented in Table 15.6.3-6. The results of the SGTR accident with pre-accident and event-generated iodine spike analyses indicate that the EAB and LPZ doses due to an SGTR accident are within their allowable dose criteria limits, which are 100 percent and 10 percent of the 10 CFR 50.34(a)(1) value, respectively. The MCR and TSC doses are also within the dose limit in GDC 19.

15.6.3.2.6 Conclusions

The radiological consequences for the SGTR accident with a LOOP are within the allowable criteria. The RCS and secondary system pressures are below the 110 percent of the design pressure limits, thus providing reasonable assurance of the integrity of these systems. The minimum DNBR is above the DNBR SAFDL value of 1.29. The acceptance criterion regarding fuel performance is met.

For the limiting SGTR event with respect to SG overfill considerations (SGTR with a LOOP), the maximum liquid inventories do not result in an overfill and consequent introduction of liquid water into the steam lines, providing reasonable assurance of the integrity of these steam lines.

After 30 minutes, the operator uses the plant emergency procedure for the SGTR to cool down the plant to shutdown cooling entry conditions.

constant enthalpy process based on the maximum time-dependent temperature of the IRWST water circulating outside containment.

The post-LOCA sump water temperature for the APR1400 is higher than 107 °C (225 °F) between 11,000 seconds (≈ 3.0 hours) and 60,000 seconds (16.67 hours), and it reaches the maximum values of 113 °C (235.5 °F) at 27,500 seconds (7.64 hours). Assuming the IRWST water temperature is 107 °C (225 °F) yields an iodine FF of 1.35 percent. The iodine FF of 2.39 percent is calculated for the maximum IRWST water temperature at 113 °C (235.5 °F), and an average FF of 2 percent is calculated for the IRWST water temperature between 3.0 hours and 16.67 hours. For the remainder of the accident, the IRWST water is conservatively assumed to remain at less than 100 °C (212 °F), and the FF of 10 percent is used during this period to be consistent with NRC RG 1.183. The post-LOCA ESF leakage release rates based on the calculated FF and an assumed design basis ESF leakage of 18.9 L/hr (5 gal/hr) (doubled to 37.8 L/hr [10 gal/hr]) are used to calculate the resulting dose consequences.

15.6.5.5.1.3 Containment Low Volume Purge System Release

If the primary containment is routinely purged during power operations, releases through the purge system prior to containment isolation are analyzed, and the resulting doses are summed with the postulated doses from other release paths. The containment low volume purge system (CLVPS) release occurs following the large break LOCA and before containment isolation. The isolation valves of the CLVPS are closed by the containment isolation actuation signal (CIAS) after a LOCA with a LOOP within 5.0 seconds. The CLVPS release evaluation assumes that 100 percent of the radionuclide inventory in the RCS liquid is released to the containment at the initiation of the LOCA and homogeneously mixed in the containment atmosphere. A release of gap activity into the containment is not considered because the CLVPS release terminates within 5 seconds, which is before the onset of the gap release, which occurs at 30 seconds.

CLVPS Release Source Term

~~The RCS isotopic iodine concentrations are based on the Technical Specification for RCS equilibrium activity and the thyroid dose conversion factors specified in Federal Guidance Report 11 (Reference 55). The noble gas concentrations are based on one percent failed fuel. Consistent with NRC RG 1.183, iodine spiking is not considered.~~

Consistent with NRC RG 1.183, the radionuclide inventories in the RCS liquid are based on the Technical Specification for the RCS equilibrium activity, and the iodine spikes is not considered. Additionally, the alkali metal concentration is based on the RCS concentrations with 1% fuel defect.

conservatisms in the analysis. Many of the design input parameter values used in the analysis are those specified in the Technical Specifications.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and are listed in Tables 2.3-1 through 2.3-12; breathing rates are given in Table ~~15A-11~~ 15A-12

15.6.5.5.3 Results

The radiological consequences due to large break LOCA are presented in Table 15.6.5-14. The results of large break LOCA analyses indicate that the EAB and LPZ doses are within their allowable dose limits in 10 CFR 50.34(a)(1). The MCR and TSC doses are also within the limit in GDC 19.

15.6.6 Combined License Information

No COL information is required with regard to Section 15.6.

Initial RCS Alkali Metal Specific Activity RCS concentrations with 1% fuel defect

Table 15.6.2-4 (1 of 2)

Parameters Used in Evaluating the Radiological Consequences of a Double-Ended Break of the Letdown Line Outside Containment

Parameter	Value
Source Terms	
Reactor Core Power Level	4,062.66 MWt
Percent of Fuel Assumed to Undergo Departure from Nucleate Boiling (DNB)	0 %
Percent of Fuel Assumed to Melt	0 %
Initial RCS Mass	292,431 kg (644,700 lbm)
Initial Steam Generator Liquid Mass per SG	89,086 kg (196,400 lbm)
Initial RCS Iodine Specific Activity	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131
Initial Secondary Liquid Iodine Specific Activity	3.7×10^3 Bq/g (0.1 μCi/g) DE I-131
Initial Noble Gas Specific Activity	2.15×10^7 Bq/g (580 μCi/g) DE Xe-133
Event-generated Iodine Spiking Factor	500
Duration of Event-generated Iodine Spike	8 hr
Chemical Forms of Iodine Released from the SG to the Environment	97 % elemental and 3 % organic
Secondary System Activity Transport Model	
Primary-to-secondary Leak Rate through SG	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 Steam Mass Release from Both SGs 0 ~ 0.5 hr 0.5 ~ 2 hr via ADV 2 ~ 8 hr via ADV	 0.00 kg (0.0 lbm) 461,000 kg (1,016,000 lbm) 54,600,000 kg (1,203,000 lbm)
LDLB Isolation Time	30 min
Unaffected SG P-T-S Leak Duration, and Termination of Release from Unaffected SG	8 hr
SG Liquid Iodine Partition Coefficient	100
~~Alkali Material (Cs, Rb) Partition coefficient~~	~~5.0×10^{-3}~~
Letdown System Flow Rate	18,100 kg/hr (39,842 lbm/hr)

SG Liquid Alkali Metal Partition Coefficient 200

Table 15.6.2-4 (2 of 2)

Parameter	Value
Secondary System Activity Transport Model (cont.)	
RCS Mass Release Outside Containment	20,300 kg (44,700 lbm)
RCS Fluid Flashing Factor	0.259
Unflashed Letdown Line Break Fluid Iodine Partition Coefficient	10
Auxiliary Building Controlled Area Exhaust System Filter Efficiencies Elemental and Organic Iodine Particulate	 0 % 0 %
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodine removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m^3/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors 0 ~ 24 hr 24 ~ 96 hr 96 ~ 720 hr	 100 % 60 % 40 %
Onsite χ/Qs	See Tables 2.3.2 ~ 2.3.12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

Table 15.6.2-5

Radiological Consequences of a Double-Ended Break
of the Letdown Line Outside Containment

Post-LDLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
LDLB Iodine and Noble Gas Release	3.96E-01	1.31E+01	2.88E+00
P-T-S Iodine Release	2.95E-01	4.53E-01	2.73E-01
P-T-S Noble Gas Release	1.32E-02	1.19E-02	8.54E-03
Secondary Liquid Iodine Release	2.95E-02	5.99E-02	2.52E-02
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	1.16E+01	1.36E+01	3.18E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

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Table 15.6.2-5

Radiological Consequences of Double-Ended Break of the Letdown Line Outside Containment

Post-LDLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
LDLB Iodine & Alkali metals Release	2.29E-01	3.09E+00	6.80E-01
LDLB Noble Gas Release	3.39E-02	6.64E-01	1.46E-01
P-T-S Iodine Release	7.27E-02	3.45E-01	2.08E-01
P-T-S Noble Gas Release	7.60E-03	8.85E-03	6.33E-03
P-T-S Alkali Metal Release	1.14E-04	5.45E-04	3.12E-04
Secondary Liquid Iodine Release	9.52E-03	5.98E-02	2.51E-02
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	1.95E+01	4.17E+00	1.07E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

Initial RCS Alkali Metal Specific Activity RCS concentrations with 1% fuel defect

Table 15.6.3-5 (1 of 3)

Parameters Used in Evaluating the Radiological Consequences of the Steam Generator Tube Rupture with a Loss of Offsite Power

Parameter	Value
Source Terms	
Reactor Core Power Level	4,062.66 MWt
Percent of Fuel Assumed to Undergo Departure from Nucleate Boiling (DNB)	0 %
Percent of Fuel Assumed to Melt	0 %
Initial RCS Mass	290,680 kg (640,840 lbm)
Initial Steam Generator Liquid Mass	117,688 kg/SG (259,457 lbm/SG)
Initial RCS Iodine Specific Activity	3.7×10^4 Bq/g (1.0 μCi/g) DE I-131
Secondary Liquid Iodine Specific Activity	3.7×10^3 Bq/g (0.1 μCi/g) DE I-131
Initial RCS Noble Gas Specific Activity	2.15×10^7 Bq/g (580 μCi/g) DE Xe-133
Used for Pre-accident Iodine Spike Case RCS Iodine Specific Activity	2.22×10^6 Bq/g (60 μCi/g) DE I-131
Event-generated Iodine Spiking Factor	335
Duration of Event-generated Iodine Spike	8 hr
Chemical Forms of Iodine Released from the Steam Generators to the Environment	97 % elemental and 3 % organic

Table 15.6.3-5 (2 of 3)

Parameter	Value
Secondary System Activity Transport Model	
Primary-to-secondary Leak Rate through SG	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)
Integrated Break Flow into Affected SG 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 40,200 kg (88,640 lbm) 24,000 kg (52,808 lbm) 0 kg (0.0 lbm)
Integrated Flashed Break Flow into Affected SG 0 ~ 0.5 hr 0.5 ~ 2 hr 2 ~ 8 hr	 2,450 kg (5,400 lbm) 2,900 kg (6,400 lbm) 0 kg (0.0 lbm)
Steam Mass Release from Affected SG 0 ~ 0.5 hr via MSSV 0.5 ~ 2 hr 2 ~ 8 hr	 77,600 kg (171,000 lbm) 0 kg (0.0 lbm) 0 kg (0.0 lbm)
Steam Mass Release from Unaffected SG 0 ~ 0.5 hr via MSSV 0.5 ~ 2 hr via ADV 2 ~ 8 hr via ADV	 64,400 kg (142,000 lbm) 463,000 kg (1,021,000 lbm) 586,000 kg (1,293,000 lbm)
Termination of Release from Affected SG by Operator Action	30 min
Unaffected SG P-T-S Leak Duration, and Termination of Release from Unaffected SG	8 hr
SG Liquid Iodine Partition Coefficient	100
Letdown System Flow Rate	18,100 kg/hr (39,842 lbm/hr)
Total RCS Leak Rate	41.6 L/min (11 gpm)

SG Liquid Alkali Metal Partition Coefficient 200

Table 15.6.3-5 (3 of 3)

Parameter	Value
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodines removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m³/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors 0 ~ 24 hr 24 ~ 96 hr 96 ~ 720 hr	 100 % 60 % 40 %
Onsite χ/Qs	See Tables ~~2.3-6 and 2.3-7~~ 2.3-2 ~ 2.3-12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

Table 15.6.3-6

Radiological Consequences of the Steam Generator Tube Rupture with a Loss of Offsite Power

Pre-accident Iodine Spike Case

Post-SGTR Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	3.25E+00	8.38E+00	1.93E+00
P-T-S Noble Gas Release	9.84E-01	2.46E+00	5.45E-01
Secondary Liquid Iodine Release	8.79E-02	1.42E-01	5.87E-02
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	1.52E+01	1.10E+01	2.53E+00
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

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Event-generated Iodine Spike Case

Post-SGTR Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	1.19E+00	3.72E+00	9.70E-01
P-T-S Noble Gas Release	9.84E-01	2.46E+00	5.45E-01
Secondary Liquid Iodine Release	8.79E-02	1.42E-01	5.87E-02
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	1.32E+01	6.32E+00	1.57E+00
Allowable TEDE Limit	5.00E+00	2.50E+01	2.50E+01

Table 15.6.3-6

Radiological Consequences of the Steam Generator
Tube Rupture with a Loss of Offsite Power

Pre-accident Iodine Spike Case

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	8.44E-01	6.02E+00	1.32E+00
P-T-S Noble Gas Release	2.55E-01	1.83E+00	4.02E-01
P-T-S Alkali Metal Release	1.26E-02	8.81E-02	1.94E-02
Secondary Liquid Iodine Release	2.13E-02	1.42E-01	3.13E-02
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	2.03E+01	8.07E+00	1.77E+00
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

Event-generated Iodine Spike Case

Post-MSLB Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
P-T-S Iodine Release	2.16E-01	2.74E+00	6.02E-01
P-T-S Noble Gas Release	2.55E-01	1.83E+00	4.02E-01
P-T-S Alkali Metal Release	1.26E-02	8.81E-02	1.94E-02
Secondary Liquid Iodine Release	2.13E-02	1.42E-01	3.13E-02
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	1.96E+01	4.79E+00	1.05E+00
Allowable TEDE Limit	5.00E+01	2.50E+01	2.50E+01

Table 15.6.5-13 (3 of 3)

Parameter	Value
Chemical Form of Iodine in ESF Elemental Organic	 97 % 3 %
Fraction of Core Iodine in Sump Water	40 %
Auxiliary Building Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodines removal)	95%
MCR Parameters	
MCR Wall Thickness East West North South Ceiling	 0.91 m (3.0 ft) 0.91 m (3.0 ft) 0.91 m (3.0 ft) 0.91 m (3.0 ft) 0.46 m (1.5 ft)
Minimum MCR Envelope Concrete Shielding	0.46 m (1.5 ft)
Emergency Ventilation HVAC Filter Charcoal Density	0.45 g/cc (28.1 lb/ft^3)
Emergency Ventilation HVAC Filter Charcoal Tray Dimension	1.65 m (L) × 1.65 m (W) × 2.34 m (H) 1.65 m (L) × 1.65 m (W) × 1.65 m (H)
Other MCR Parameters	See Table 15.3.3-3
Containment Low Volume Purge System (CLVPS) Release Parameters	
CLVPS Valve Closure Time	5.0 sec
Volume Flow Rate of CLVPS Release	11 m^3/sec (2.34 × 10^4 cfm)
Reactor Coolant Mass	300,000 kg (661,000 lbm)
Reactor Coolant Specific Activity	≤ 3.7 × 10^4 Bq/g (1.0 μCi/g) DE I-131
Onsite χ/Qs	See Tables 2.3.2 ~ 2.3-12
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Minimum Concrete Density	2,240 kg/m^3 (140 lb/ft^3)

2.15 × 10^7 Bq/g (580 μCi/g) DE Xe-133
RCS concentrations with 1% fuel defect for Alkali Metal

Table 15.6.5-14

Radiological Consequences of a Large Break LOCA

Post-LOCA Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
Containment Leakage	3.26E+01	2.03E+02	1.01E+02
ESF Leakage	4.99E-01	1.20E+00	6.84E+00
CLVPS Release	2.81E-02	5.68E-03	1.25E-03
Containment Shine	0.00E+00	0.00E+00	0.00E+00
External Cloud	6.88E-01	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.02E+01	0.00E+00	0.00E+00
Total	4.40E+01	2.04E+02	1.08E+02
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

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Table 15.6.5-14

Radiological Consequences of a Large Break LOCA

Post-LOCA Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
Containment Leakage	2.73E+01	1.99E+01	9.94E+00
ESF Leakage	3.16E-01	1.18E-01	6.70E-01
CLVPS Release	1.54E-01	2.30E-03	5.05E-04
Containment Shine	0.00E+00	0.00E+00	0.00E+00
External Cloud	6.22E+00	0.00E+00	0.00E+00
Emergency Ventilation Filter Shine	1.29E+01	0.00E+00	0.00E+00
Total	4.69E+01	2.00E+01	1.06E+01
Allowable TEDE Limit	5.00E+01	2.50E+02	2.50E+02

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 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.

the iodines, alkali metals and noble gases

The radioactivity release from the fuel pool is assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. In this analysis, no dilution or mixing is credited in the fuel handling building.

The χ/Q values used in the analysis for EAB, LPZ, MCR, and TSC are described in Subsection 2.3.4 and listed in Table 15.7.4-1; breathing rates are given in Table ~~15A-11~~ 15A-12.

15.7.4.3 Results

The radiological consequences due to FHA are presented in Table 15.7.4-2. The results of the FHA analyses indicate that the EAB and LPZ doses are within their allowable dose limit, which is 25 percent of the 10 CFR 50.34(a)(1) value as specified in SRP 15.0.3. The MCR and TSC doses are also within the dose limit in GDC 19.

15.7.4.4 Conclusion

The potential radiological consequences of a postulated FHA have been conservatively analyzed, using the assumptions and models described in the preceding subsections. The calculated doses are within the criteria specified in SRP 15.0.3.

15.7.5 Spent Fuel Cask Drop Accident

A spent fuel cask handling accident is evaluated if the spent fuel cask can be dropped from a height exceeding 9.14 m (30 ft) onto a hard unyielding surface or if it can be dropped or tipped onto stored irradiated fuel.

The fuel handling system and plant layout of the APR1400 is designed to meet the following criteria:

a. All spent fuel cask lifts from the cask transporter to the cask laydown area are limited to less than 9.14 m (30 ft)

b. The spent fuel cask handling crane operating procedures establish the requirements for operator training, crane inspections, and approved cask handling procedures

Table 15.7.4-1 (2 of 2)

Parameter	Value
MCR and TSC Model Parameters	
Envelope Volume	5,663 m^3 (200,000 ft^3)
Normal Ventilation Flow Rate (unfiltered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Makeup Rate (filtered)	105 m^3/min (3,700 cfm)
Emergency Ventilation Recirculation Flow Rate (filtered)	122 m^3/min (4,300 cfm)
Emergency HVAC Delay Time	5 min
Emergency Ventilation Charcoal Filter Efficiency (elemental and organic iodine removal)	99 %
Emergency Ventilation HEPA Filter Efficiency (particulate removal)	99 %
Unfiltered Inleakage	~~8.50 m³/min (300 cfm)~~ 2.83 m^3/min (100 cfm)
Occupancy Factors 0 ~ 24 hr 24 ~ 96 hr 96 ~ 720 hr	 100 % 60 % 40 %
Onsite χ/Qs	See Tables 2.3-2 ~ 2.3-12
~~Limiting Onsite χ/Qs for MCR Air Intake (s/m³)~~ ~~0 ~ 2 hr~~ ~~2 ~ 8 hr~~ ~~8 ~ 24 hr~~ ~~24 ~ 96 hr~~ ~~96 ~ 720 hr~~	 ~~2.59E-04~~ ~~2.04E-04~~ ~~8.98E-05~~ ~~5.93E-05~~ ~~4.58E-05~~
~~Limiting Onsite χ/Qs for MCR Unfiltered Inleakage (s/m³)~~ ~~0 ~ 2 hr~~ ~~2 ~ 8 hr~~ ~~8 ~ 24 hr~~ ~~24 ~ 96 hr~~ ~~96 ~ 720 hr~~	 ~~1.04E-03~~ ~~8.18E-04~~ ~~3.59E-04~~ ~~2.37E-04~~ ~~1.83E-04~~
Offsite Model Parameters	
χ/Qs	See Table 2.3-1
Breathing Rate	See Table ~~15A-11~~ 15A-12
Dose Conversion Factors	See Table ~~15A-10~~ 15A-11

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Table 15.7.4-2

Radiological Consequences of Fuel Handling Accident

8.56E+00

~~8.56E-01~~

Activity Release Path	TEDE Dose (mSv)		
	MCR and TSC	EAB	LPZ
Fuel handling area vent release	~~6.25E+00~~	~~3.89E+00~~	~~8.56E+00~~
Allowable TEDE limit	5.00E+01	6.30E+01	6.30E+01

8.55E+00

3.89E+01

15A.1.2.3 Noble Gas Inventory

For pre-accident noble gas concentrations, the initial concentration is assumed to be at 2.15 $\times 10^7$ Bq/g (580 μCi/g) DE Xe-133 in the primary system, based on one percent fuel defect. This assumption is conservative because the Technical Specifications restrict the noble gas concentrations in the RCS to less than 1.11 x 10^7 Bq/g (300 μCi/g). The resulting activities are presented in Table 15A-8.

15A.1.2.4 Alkali Metal Inventory

~~It is possible that alkali metal activities in the RCS are ignored because the dose contribution from alkali metals is insignificant compared to the dose contribution from noble gases that are released directly to the environment without holdup. The alkali metals have a low partition coefficient from the coolant to steam phase so the alkali metal activities released to the environment are significantly reduced and the dose contribution is also negligible.~~

For the pre-accident alkali metal, the initial concentration in the reactor coolant is assumed to be at the design basis values corresponding to a 1.0% fuel defect, although the Technical Specification does not specify the allowable maximum activity at equilibrium condition for the alkali metals in compliance with the RG 1.183 guidance. The resulting activities are presented in Table 15A-9.

15A.1.3 Secondary Coolant Source Term

The iodine activity of the secondary coolant system is assumed to be at the Technical Specification limit of 3.7 × 10^3 Bq/g (0.1 μCi/g) DE I-131. This is one-tenth of the maximum equilibrium reactor coolant activities and is given in Table ~~15A-9~~ 15A-10.

a

the primary and secondary coolant. The leakage that flashes to vapor is assumed to rise through the bulk water of the SG and enter the steam space without any credit for removal.

For the cases of unaffected SG, of which tubes are fully submerged by the secondary water, the P-T-S leakage is assumed to mix with the secondary water without flashing during period of total tube submergence. The radioactivity in the bulk water is assumed to become vapor at a rate that is ~~the~~ function of the steaming rate and ~~the~~ partition coefficient. A partition coefficient ~~for iodine of 100~~ is assumed.

of 100 for iodine and 200 for alkali metal

15A.3.3 Activity Transport from Spent Fuel Pool

The DFs for the elemental and organic iodine in the pools are 500 and 1, respectively, because the depth of water above the stored spent fuel is designed to be 7.0 m (23 ft) or greater. This DFs result in overall effective DF of 200, which means that 99.5 percent of the total iodine released from the failed fuel rods is retained in the pool water.

Noble gases are released out of the pool surface without scrubbing. Particulate radionuclides are assumed to be retained by the water in the fuel pool or refueling pool (i.e., infinite decontamination factor).

15A.3.4 Flashing Fraction

The flashing fraction, the portion of discharged fluids that flashes to vapor, is calculated based on the enthalpy difference under circumstance of coolant leakage by assuming the leakage to be constant enthalpy process and expressed as follows:

$$\text{Flashing fraction (FF)} = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

Where,

h_{f_1} = Enthalpy at system temperature and pressure before leaking from a component

h_{f_2} = Enthalpy at saturation condition after leaking from a component

h_{fg} = Enthalpy of steam at saturation condition after leaking from a component

15A.4 Event-specific Activity Transport Model

15A.4.1 Steam Line Break

The radioactivities ~~Radioiodines~~ initially contained in the primary coolant transfer to the SG through the SG tube leaks. The secondary coolant releases directly to the environment through the ruptured steam line. A portion of the iodine activity initially contained in the unaffected SG, and noble gas, alkali metal and iodine activities due to SG tube leakage is released to environment through ADVs or MSSVs. The primary coolant discharged to in-containment refueling water storage tank (IRWST) through POSRV during the accident is released to the environment due to containment leakage. The appropriate partitioning coefficient, flashing fraction, and iodine spiking effects are considered for dose calculation. More specific evaluation model, assumptions, and input data used for this event are discussed in Subsection 15.1.5.5. The activity transport paths from containment, the affected and unaffected SGs to the environment (or the main control room) for the event are illustrated in Figure 15A-1.

15A.4.2 Feedwater Line Break

For the affected SG, radioiodines contained in the primary leaking through SG tubes and those in the secondary coolant are released to the containment through the break in the feedwater line. Throughout the event, P-T-S leakage entering the affected SG is conservatively assumed to be directly released to the environment through the MSSVs. The unaffected SG releases steam when the intact SG MSSVs and ADVs open. The RCS fluid is released to the IRWST located inside containment through POSRV during the accident and from there is released to the environment due to the containment leakage. The appropriate partitioning coefficient, flashing fraction, and iodine spiking effects are considered for dose calculation. More specific evaluation models, assumptions, and input data used for this event are discussed in Subsection 15.2.8.5. The activity transport paths from the feedwater line break through the containment building, and the P-T-S leakage through the two SGs and to the environment (or the main control room) through the main steam safety valve (MSSV) and atmospheric dump valve (ADV) steaming are illustrated in Figure 15A-2.

15A.5 Dose Calculation Methodology

15A.5.1 Atmospheric Dispersion Factor

Accident atmospheric dispersion factors (χ/Q) for the exclusion area boundary and the low population zone are used to calculate the potential offsite doses. The representative χ/Q values are determined as described in Subsection 2.3.4 and are given in Table 2.3-1. Main control room χ/Q values are also addressed in Subsection 2.3.4 and given in Tables 2.3-2 through ~~2.3-13~~ 2.3-12.

15A.5.2 Dose Conversion Factor

15A.5.2.1 Immersion Dose Conversion Factor

Consistent with NRC RG 1.183, effective dose equivalent (EDE) is used in determining the contribution of external dose to the TEDE. This calculation models the EDE dose conversion factors in the column headed "effective" in Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil," (Reference 3). The dose conversion factors for calculation of EDE doses are shown in Table ~~15A-10~~ 15A-11.

15A.5.2.2 Inhalation Dose Conversion Factor

Consistent with NRC RG 1.183, the exposure-to-committed effective dose equivalent (CEDE) factors for inhalation of radioactive material are derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers." This calculation models the CEDE dose conversion factors in the column headed "effective" yield doses in Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," (Reference 4). The dose conversion factors for calculation of CEDE doses are shown in Table ~~15A-10~~ 15A-11

15A.5.3 Breathing Rate

Consistent with NRC RG 1.183, for the first 8 hours, the receptor offsite breathing rate is assumed to be 3.5×10^{-4} m^3/sec after the initiation of the event. From 8 to 24 hours after

the accident, the breathing rate is assumed to be 1.8×10^{-4} m^3/sec. After 24 hours and until the end of the accident, the rate is assumed to be 2.3×10^{-4} m^3/sec. For the control room, the breathing rate of the individual is assumed to be 3.5×10^{-4} m^3/sec during the entire period of the accident. These breathing rates are listed in Table ~~15A-11~~ 15A-12.

15A.5.4 Offsite Dose Calculation Method

15A.5.4.1 Assumptions

The following assumptions are used in modeling the external effective dose equivalent from immersion in a cloud of radioactivity and the internal effective dose equivalent from inhalation of radioactivity.

a. The dose contribution of direct radiation from sources other than the leakage cloud is negligible compared to the dose due to immersion in the leakage cloud

b. All radioactivity releases are treated as ground level releases regardless of the point of release

c. Radioactive decay from the point of release to the dose receptor is neglected

15A.5.4.2 Immersion Dose

The EDE is obtained by considering the dose receptor to be immersed in a radioactive cloud which is infinite in all directions above the ground plane (i.e., a semi-infinite cloud).

The concentration of radioactive material within this cloud is considered to be uniform and equal to the maximum centerline ground level concentration.

The external effective dose equivalent is a result of exposure to external gamma radiation. The EDE due to immersion in a semi-infinite cloud is given by the following equation:

$$D_E = \chi/Q \cdot \sum_i (Q_i \cdot DCF_{E,i})$$

Where:

D_E = external effective dose from immersion in a semi-infinite cloud for a given time period, Sv

Table 15A-1 (1 of 2)

Maximum Core Fission Product Inventories
(Core Power: 4,062.66 MWt, Burnup: 56.4 GWD/MTU)

Br-84 1.37×10^{18}

Rb-88 4.67×10^{18}
Rb-89 6.20×10^{18}

Nuclides	Core Inventory (Bq)	Nuclides	Core Inventory (Bq)
Co-58$^{(1)}$	-	Sb-127	4.32×10^{17}
Co-60$^{(1)}$	-	Sb-129	1.48×10^{18}
Kr-85	5.86×10^{16}	Te-127	4.28×10^{17}
Kr-85M	1.58×10^{18}	Te-127M	7.21×10^{16}
Kr-87	3.23×10^{18}	Te-129	1.41×10^{18}
Kr-88	4.57×10^{18}	Te-129M	2.88×10^{17}
Rb-86	1.29×10^{16}	Te-131M	9.42×10^{17}
Sr-89	5.71×10^{18}	Te-132	6.28×10^{18}
Sr-90	5.13×10^{17}	I-131	4.43×10^{18}
Sr-91	7.68×10^{18}	I-132	6.42×10^{18}
Sr-92	7.75×10^{18}	I-133	9.37×10^{18}
Y-90	5.41×10^{17}	I-134	1.07×10^{19}
Y-91	7.00×10^{18}	I-135	8.84×10^{18}
Y-92	7.82×10^{18}	Xe-133	9.33×10^{18}
Y-93	5.61×10^{18}	Xe-135	2.80×10^{18}
Zr-95	8.18×10^{18}	Cs-134	1.38×10^{18}
Zr-97	7.77×10^{18}	Cs-136	3.55×10^{17}
Nb-95	8.15×10^{18}	Cs-137	7.65×10^{17}
Mo-99	8.53×10^{18}	Ba-139	8.77×10^{18}
Tc-99M	7.51×10^{18}	Ba-140	8.68×10^{18}
Ru-103	5.89×10^{18}	La-140	8.70×10^{18}
Ru-105	3.48×10^{18}	La-141	8.00×10^{18}
Ru-106	5.27×10^{18}	La-142	7.81×10^{18}

Xe-133M 2.86×10^{17}

Xe-131M 6.52×10^{16}

Cs-135 2.97×10^{12}

Cs-138 9.11×10^{18}

Xe-135M 1.94×10^{18}
Xe-135 8.65×10^{18}

(1) Co-58 and Co-60 activities, 2.55×10^2 Ci/MW$_t$ and 1.95×10^2 Ci/MW$_t$, respectively, are conservatively assumed to be added into the LOCA radiological consequence analysis, which is obtained from the RADTRAD User's Manual, Table 1.4.3.2-2.

Table 15A-2

Fraction of Fission Product Inventory in Gap

Nuclides		Fraction of Core Inventory		
			Non-LOCA	
		LOCA	Non-CEA Ejection	CEA Ejection
Noble gases	Kr-85	0.05	0.10	0.10
	Others	0.05	0.05	0.10
Halogens	I-131	0.05	0.08	0.10
	Others	0.05	0.05	~~0.0~~
Alkali metals	All	0.05	0.12	~~0.0~~

0.10 for iodine, 0.05 for others

0.12

Table 15A-4

Iodine Appearance Rates for Event-generated Iodine Spike (Steam Line Break)

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})	500 Times of Iodine Appearance Rate (Bq/sec)
I-131	8.05×10^{12}	9.98×10^{-7}	~~2.00×10^{-5}~~	~~8.43~~ $\times 10^{10}$
I-132	2.16×10^{12}	8.37×10^{-5}	~~2.00×10^{-5}~~	~~1.12~~ $\times 10^{11}$
I-133	1.14×10^{13}	9.26×10^{-6}	~~2.00×10^{-5}~~	~~1.67~~ $\times 10^{11}$
I-134	1.32×10^{12}	2.20×10^{-4}	~~2.00×10^{-5}~~	~~1.58~~ $\times 10^{11}$
I-135	6.49×10^{12}	2.91×10^{-5}	~~2.00×10^{-5}~~	~~1.59~~ $\times 10^{11}$

2.02×10^{-5}

8.52
1.12
1.68
1.58
1.60

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}~~ $8.53\mathrm{x}10^{12}$	9.98×10^{-7}	~~1.90×10^{-5}~~ $1.97\mathrm{x}10^{-5}$	~~5.91×10^{10}~~ $5.90\mathrm{x}10^{10}$
I-132	~~2.36×10^{12}~~ $2.29\mathrm{x}10^{12}$	8.37×10^{-5}	~~1.90×10^{-5}~~ $1.97\mathrm{x}10^{-5}$	~~8.14×10^{10}~~ $7.93\mathrm{x}10^{10}$
I-133	~~1.25×10^{13}~~ $1.21\mathrm{x}10^{13}$	9.26×10^{-6}	~~1.90×10^{-5}~~ $1.97\mathrm{x}10^{-5}$	~~1.18×10^{11}~~ $1.17\mathrm{x}10^{11}$
I-134	~~1.44×10^{12}~~ $1.40\mathrm{x}10^{12}$	2.20×10^{-4}	~~1.90×10^{-5}~~ $1.97\mathrm{x}10^{-5}$	~~1.15×10^{11}~~ $1.12\mathrm{x}10^{11}$
I-135	~~7.10×10^{12}~~ $6.87\mathrm{x}10^{12}$	2.91×10^{-5}	~~1.90×10^{-5}~~ $1.97\mathrm{x}10^{-5}$	~~1.15×10^{11}~~ $1.12\mathrm{x}10^{11}$

Table 15A-6

Iodine Appearance Rates for Event-generated Iodine Spike
(Failure of Small Lines Carrying Primary Coolant Outside Containment)

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})	500 Times of Iodine Appearance Rate (Bq/sec)
I-131	~~8.58×10^{12}~~ 8.58×10^{12}	9.98×10^{-7}	~~1.89×10^{-5}~~ 1.95×10^{-5}	~~8.55×10^{10}~~ 8.81×10^{10}
I-132	~~2.30×10^{12}~~ 2.30×10^{12}	8.37×10^{-5}	~~1.89×10^{-5}~~ 1.95×10^{-5}	~~1.18×10^{11}~~ 1.19×10^{11}
I-133	~~1.22×10^{13}~~ 1.22×10^{13}	9.26×10^{-6}	~~1.89×10^{-5}~~ 1.95×10^{-5}	~~1.72×10^{11}~~ 1.76×10^{11}
I-134	~~1.41×10^{12}~~ 1.41×10^{12}	2.20×10^{-4}	~~1.89×10^{-5}~~ 1.95×10^{-5}	~~1.68×10^{11}~~ 1.68×10^{11}
I-135	~~6.92×10^{12}~~ 6.91×10^{12}	2.91×10^{-5}	~~1.89×10^{-5}~~ 1.95×10^{-5}	~~1.66×10^{11}~~ 1.68×10^{11}

Table 15A-7

Iodine Appearance Rates for Event-generated Iodine Spike (Feedwater Line Break)

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})	500 Times of Iodine Appearance Rate (Bq/sec)
I-131	$\cancel{8.45 \times 10^{12}}$ 7.41×10^{12}	9.98×10^{-7}	$\cancel{1.92 \times 10^{-5}}$ 1.98×10^{-5}	$\cancel{8.54 \times 10^{10}}$ 7.71×10^{10}
I-132	$\cancel{2.27 \times 10^{12}}$ 1.99×10^{12}	8.37×10^{-5}	$\cancel{1.92 \times 10^{-5}}$ 1.98×10^{-5}	$\cancel{1.17 \times 10^{11}}$ 1.03×10^{11}
I-133	$\cancel{1.20 \times 10^{13}}$ 1.05×10^{13}	9.26×10^{-6}	$\cancel{1.92 \times 10^{-5}}$ 1.98×10^{-5}	$\cancel{1.71 \times 10^{11}}$ 1.53×10^{11}
I-134	$\cancel{1.39 \times 10^{12}}$ 1.22×10^{12}	2.20×10^{-4}	$\cancel{1.92 \times 10^{-5}}$ 1.98×10^{-5}	$\cancel{1.66 \times 10^{11}}$ 1.46×10^{11}
I-135	$\cancel{6.82 \times 10^{10}}$ 5.97×10^{12}	2.91×10^{-5}	$\cancel{1.92 \times 10^{-5}}$ 1.98×10^{-5}	$\cancel{1.65 \times 10^{11}}$ 1.46×10^{11}

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Table ~~15A-9~~

Secondary Coolant Iodine Concentrations

Nuclide	Secondary Coolant Activity of 3.7×10^3 Bq/g (0.1 μCi/g) (Bq /g)
I-131	2.93×10^3
I-132	7.88×10^2
I-133	4.16×10^3
I-134	4.81×10^2
I-135	2.37×10^3

Insert new Table 15A-9 in Next Page

Table 15A-9

Primary Coolant Alkali Metal Concentration

Nuclides	1.0 % Fuel Defect RCS Alkali Metal Concentration (Bq/g)[1]
Rb-88	9.16×10^4
Cs-134	1.41×10^4
Cs-136	1.92×10^3
Cs-137	1.63×10^4

[1] These values are calculated by multiplying the values based on 0.25% fuel defect by a factor of 4 to be based on 1% fuel defect.

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Table 15A-10 (1 of 4)

Dose Conversion Factors

Nuclide	EDE Dose Conversion Factor (Sv-m^3/Bq-sec)	CEDE Dose Conversion Factor (Sv/Bq)
Noble Gases		
Kr-85	1.19×10^{-16}	-
Kr-85 m	~~7.49~~ 7.48 $\times 10^{-15}$	-
Kr-87	~~4.11~~ 4.12 $\times 10^{-14}$	-
Kr-88	1.02×10^{-13}	-
Xe131m	3.89×10^{-16}	-
Xe133m	1.37×10^{-15}	-
Xe-133	1.56×10^{-15}	-
Xe135m	2.04×12^{-14}	-
Xe-135	1.19×10^{-14}	-
Xe-138	~~5.76~~ 5.77 $\times 10^{-14}$	-
Halogens		
Br-84	9.41×10^{-14}	2.61×10^{-11}
I-131	1.82×10^{-14}	8.89×10^{-9}
I-132	1.12×10^{-13}	1.03×10^{-10}
I-133	~~2.95~~ 2.94 $\times 10^{-14}$	1.58×10^{-9}
I-134	1.30×10^{-13}	3.55×10^{-11}
I-135	~~7.97~~ 7.98 $\times 10^{-14}$	3.32×10^{-10}
Alkali Metals		
Rb-86	4.81×10^{-15}	1.79×10^{-9}
Rb-88	3.36×10^{-14}	2.26×10^{-11}
Rb-89	1.06×10^{-13}	1.16×10^{-11}
Cs-134	7.57×10^{-14}	~~1.25×10^{-9}~~ 1.25×10^{-8}
Cs-135	5.65×10^{-19}	1.23×10^{-9}
Cs-136	1.06×10^{-13}	1.98×10^{-9}
Cs-137	7.74×10^{-18}	8.63×10^{-9}
Cs-138	1.21×10^{-13}	2.74×10^{-11}

Table ~~15A-10~~ 15A-11 (2 of 4)

Nuclide	EDE Dose Conversion Factor (Sv-m^3/Bq-sec)	CEDE Dose Conversion Factor (Sv/Bq)
Barium and Strontium		
Sr-89	7.73×10^{-17}	1.12×10^{-8}
Sr-90	7.53×10^{-18}	3.51×10^{-7}
Sr-91	3.45×10^{-14}	$\cancel{4.55 \times 10^{-10}}$ 4.49×10^{-10}
Sr-92	6.79×10^{-14}	2.18×10^{-10}
Ba-139	2.17×10^{-15}	4.64×10^{-11}
Ba-140	8.58×10^{-15}	$\cancel{1.10 \times 10^{-9}}$ 1.01×10^{-9}
Tellurium Group		
Sb-127	3.33×10^{-14}	1.63×10^{-9}
Sb-129	$\cancel{7.14 \times 10^{-18}}$ 7.14×10^{-14}	1.74×10^{-10}
Te-127	2.42×10^{-16}	$\cancel{8.60 \times 10^{-9}}$ 8.60×10^{-11}
Te-127m	1.47×10^{-16}	5.81×10^{-9}
Te-129	2.75×10^{-15}	2.09×10^{-11}
Te-129m	1.55×10^{-15}	$\cancel{6.48 \times 10^{-9}}$ 6.47×10^{-9}
Te-131m	7.01×10^{-14}	$\cancel{1.76 \times 10^{-9}}$ 1.73×10^{-9}
Te-132	1.03×10^{-14}	2.55×10^{-9}
Noble Metals		
Co-58	4.76×10^{-14}	2.94×10^{-9}
Co-60	1.26×10^{-13}	5.91×10^{-8}
Mo-99	7.28×10^{-15}	1.07×10^{-9}
Tc-99m	5.89×10^{-15}	8.80×10^{-12}
Ru-103	2.25×10^{-14}	2.42×10^{-9}

Table ~~15A-10~~ 15A-11 (3 of 4)

Nuclide	EDE Dose Conversion Factor (Sv-m^3/Bq-sec)	CEDE Dose Conversion Factor (Sv/Bq)
Ru-105	3.81×10^{-14}	1.23×10^{-10}
Ru-106	~~1.04×10^{-14}~~ $0.00 \times 10^{+0}$	1.29×10^{-7}
Rh-105	3.72×10^{-15}	~~2.58×10^{-9}~~ 2.58×10^{-10}
Lanthanides		
Y-90	~~1.91×10^{-16}~~ 1.90×10^{-16}	2.28×10^{-9}
Y-91	~~2.61×10^{-16}~~ 2.60×10^{-16}	1.32×10^{-8}
Y-92	1.30×10^{-14}	2.11×10^{-10}
Y-93	4.80×10^{-15}	5.82×10^{-10}
Zr-95	3.60×10^{-14}	6.39×10^{-9}
Zr-97	9.02×10^{-15}	1.17×10^{-9}
Nb-95	3.74×10^{-14}	1.57×10^{-9}
La-140	1.17×10^{-13}	1.31×10^{-9}
La-141	2.39×10^{-15}	~~1.57×10^{-9}~~ 1.57×10^{-10}
La-142	1.44×10^{-13}	6.84×10^{-11}
Pr-143	2.10×10^{-17}	2.19×10^{-9}
Nd-147	6.19×10^{-15}	1.85×10^{-9}
Am-241	8.18×10^{-16}	1.20×10^{-4}
Cm-242	5.69×10^{-18}	4.67×10^{-6}
Cm-244	4.91×10^{-18}	6.70×10^{-5}

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Table ~~15A-10~~ (4 of 4)

3.43 × 10^{-15}

Nuclide	EDE Dose Conversion Factor (Sv-m^3/Bq-sec)	CEDE Dose Conversion Factor (Sv/Bq)
Cerium Group		
Ce-141	~~3.43 × 10^{15}~~	2.42 × 10^{-9}
Ce-143	1.29 × 10^{-14}	9.16 × 10^{-10}
Ce-144	8.53 × 10^{-16}	1.01 × 10^{-7}
Np-239	7.69 × 10^{-15}	6.78 × 10^{-10}
Pu-238	4.88 × 10^{-18}	7.79 × 10^{-5}
Pu-239	4.24 × 10^{-18}	8.33 × 10^{-5}
Pu-240	4.75 × 10^{-18}	8.33 × 10^{-5}
Pu-241	7.25 × 10^{-20}	1.34 × 10^{-6}

15A-12

Table ~~15A-11~~

Breathing Rates

Time from Start of Accident	Breathing Rate (m^3/sec)
Offsite	
0 ~ 8 hrs	3.5×10^{-4}
8 ~ 24 hrs	1.8×10^{-4}
1 ~ 30 days	2.3×10^{-4}
Main Control Room (MCR)	
0 ~ 30 days	3.5×10^{-4}