15.7 Radioactive Release from Subsystems and Components

15.7.1 Radiological Consequences of a Radioactive Gas Waste System Leak or Failure

The offgas system charcoal beds are assumed to be bypassed for the accident analysis. The analysis documented below is valid for the offgas system design as described in Section 11.3 of this DCD.

15.7.1.1 Basis and Assumptions

The radiological consequences for an offgas system accident as specified in Standard Review Plan 11.3, Branch Technical Position ESTB 11-5 are presented. The branch technical position assumptions were used except as detailed below to evaluate this accident. The results are presented in Tables 15.7-1 through 15.7-3 and show the ABWR design to be compliant with the requirements of the position paper.

The ABWR offgas system is detailed in Subsection 11.3. The system is designed to be both detonation and seismic resistant and meets all criteria of Regulatory Guide 1.143. As such the failure of a single active component leading to a direct release of radioactive gases to the environment is highly unlikely. Therefore, inadvertent operator action with bypass of the delay charcoal beds is analyzed for compliance to ESTB 11-5. A top level diagram of the ABWR offgas system can be found in Figure 11.3-1 (see also Figure 15.7-2) which shows that the ABWR charcoal beds consists of five charcoal tanks.The first or guard tank contains 4,721 kg of charcoal followed by four tanks each containing 27,200 kg of charcoal. Bypass valves exists to direct flow around the (1) guard tank, (2) four series of follow-on tanks or (3) all tanks. To bypass either pathway (1) or (2) above requires the operator to enter a computer command with a required permissive. To bypass all tanks (pathway (3)) requires the operator to key in the command with two separate permissives. Since pathway (3) would require both inadvertent operation upon the operator (keying in the wrong command) plus getting two specific permissives for three incorrect decisions, it is not assumed that pathway (3) is likely to occur. Redundant upon human decision making and downstream of the charcoal beds and the post charcoal bed particle filter shown in Figure 11.3-1 are a series of two redundant radiation monitoring instruments and an air operated isolation valve which will alarm the control room and automatically shut off all flow from the offgas system for radioactivity levels in excess of environmental limits which are defined by 10CFR20 as not greater than $2x10^{-2}$m Sv/h at the site boundary. Therefore, bypass of the charcoal beds during periods with significant radioactive flow through the offgas system will be limited and/or automatically terminated by actuation of the downstream sensors.

To evaluate the potential radiological consequences of an inadvertent bypass of the charcoal beds, it was assumed that operator error or computer error has led to the bypass of the four follow-on beds in addition to the failure of the automated air operated downstream isolation valve. It is also assumed that during this period, the plant is running at and continues to run at

the maximum permissible offgas rate of 14.8 GBq/s (based upon the assumption of 0.0037 GBq/s/MWt as stipulated in Standard Review Plan 11.3) evaluated to a decay time of 30 minutes from the vessel exit nozzle. Even with the failure of the downstream isolation valve, it is not anticipated or assumed that the isolation instrumentation would fail but would instead alarm the control room with a high radiation alarm causing the operator to manually isolate the offgas system (i.e., close suction valves) within 30 minutes of the alarm. Therefore, this analysis differs from the branch technical position on the following points:

(1) Flow is through a single 4,721 kg charcoal tank with an evaluated hold up time given by NUREG-0016, equation 1.5.1.6 using K_d's for Kr and Xe from NUREG-0016.

(2) An isolation valve prevents flow through the follow-on charcoal tanks while in bypass and therefore no activity from these tanks is included in the final release calculations.

(3) With redundant instrumentation, it is expected that operator intervention to either shut off the bypass or isolate the offgas system is predicted to occur within 30 minutes and therefore the total flow from the system is evaluated for 30 minutes and not the 2 hour period stipulated in the branch technical position.

15.7.1.2 Design Basis Accident

The DBA evaluation assumptions are given in Table 15.7-1 with the isotopic flows and releases given in Table 15.7-2, and the meteorology and dose results given in Table 15.7-3.

15.7.1.3 Results

The dose results are given in Table 15.7-3 and are within the limiting 2.5×10^{-2}Sv whole body dose for an infrequent event or the 0.5×10^{-2}Sv frequent event limitation of the Branch Technical Position ESTB-11-5.

15.7.2 Liquid Radioactive System Failure

This section of the Standard Review Plan has been deleted.

15.7.3 Postulated Radioactive Release Due to Liquid Radwaste Tank Failure

The iodine inventory in the liquid radwaste tanks as discussed in this section is greater than the inventory in the design described in Section 11.2. Thus the analysis documented in this section is bounding.

15.7.3.1 Identification of Cause and Frequency Classification

An unspecified event causes the complete release of the radioactive inventory in all tanks containing radionuclides in the Liquid Radwaste System. Postulated events that could cause a release of the inventory of a tank are sudden unmonitored cracks in the vessel or operator error.

Small cracks and consequent low-level releases are bounded by this analysis and should be contained without any significant release.

The ABWR Radwaste Building is designed in accordance with the requirements of Regulatory Guide 1.143. In addition, all compartments containing liquid radwastes are steel-lined up to a height capable of containing the release of all the liquid radwastes into the compartment. Because of these design capabilities, it is considered remote that any major accident involving the release of liquid radwastes into these volumes would result in the release of these liquids to the environment via the liquid pathway. Releases as a result of major cracks would instead result in the release of the liquid radwastes to the compartment and then to the building sump system for containment in other tanks or emergency tanks. A complete description of the Liquid Radwaste System is found in Section 11.2, except for the tank inventories, which are found in Section 12.2.

A liquid radwaste release caused by operator error is also considered a remote possibility. Operating techniques and administrative procedures emphasize detailed system and equipment operating instruction. A positive action interlock system is also provided to prevent inadvertent opening of a drain valve. Should a release of wastes occur, the steel lining would contain the release until the floor drain sump pumps in the building capture and contain such spills.

The probability of a complete tank release is considered low enough to warrant this event as a limiting fault.

15.7.3.2 Sequence of Events and Systems Operation

Following a failure, the area radiation alarms would be expected to alarm at one minute with operator intervention following at approximately five minutes after release. However, the rupture of a waste tank would leave little recourse for the operator. Gaseous wastes would be trapped following alarm initiation, since isolation would occur upon alarm initiation; however, no credit for isolation is taken for this aspect and gaseous releases are expected to be purged to the environment.

Liquid release would be contained within the steel liner and would present no immediate threat to the environment leaving the operator sufficient time (on the order of hours) in which to recover systems to pump the release into holding tanks or emergency tanks.

15.7.3.3 Design Basis Accident

Based upon the above discussion, a single pathway is considered for release of fission products to the environment via airborne releases. The liquid pathway is not considered due to the mitigative capabilities of the Radwaste Building.

For the airborne pathway, volatile iodine species in the tanks using conservative inventories are considered. Though isolation is expected within minutes of this occurrence, release of 10% of

the iodine inventory is conservatively assumed over a two hour period. Specific values for this analysis are found in Tables 15.7-5 and 15.7-6.

15.7.3.4 Results

No liquid or significant (from airborne species) ground contamination is expected. Airborne doses are given in Table 15.7-7 and are a fraction of 10CFR100 criteria. COL applicants need to update the dose calculations to conform to the as-designed plant and site-specific parameters (see Subsection 15.7.6.1 for COL license information.).

15.7.4 Fuel-Handling Accident

The radiological results documented in this section are based on an 8x8 BWR fuel design. The SVEA-96 Optima2 fuel design as described in this DCD weighs approximately 15 kg less than this 8x8 fuel, and has essentially the same energy absorption to 1% plastic strain. Applying the methodology described in this section to the Optima2 fuel, less than 145 Optima2 fuel rods are calculated to fail, and these rods contain less fission product inventory than the 115 failed 8x8 BWR fuel design rods. Since the fuel channels are not accounted for in the drop analysis, channel differences are irrelevant. Therefore the radiological results provided in this section are bounding for the SVEA-96 Optima2 fuel.

15.7.4.1 Identification of Causes and Frequency Classification

15.7.4.1.1 Identification of Causes

The fuel-handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanism, resulting in dropping a raised fuel assembly onto the reactor core.

15.7.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

15.7.4.2 Sequence of Events and Systems Operation

15.7.4.2.1 Sequence of Events

Sequence of Events	Elapsed Time (min)
(1) Channeled fuel bundle is being handled by a crane over reactor core. Crane motion changes from horizontal and the fuel grapple releases, dropping the bundle.	0
(2) Some rods in both the dropped and struck bundles fail, releasing radioactive gases to the pool water.	0
(3) Gases pass from the water to the Reactor Building, fuel-handling area.	0
(4) The Reactor Building ventilation system high radiation alarm alerts plant personnel.	10
(5) Operator actions begin.	10

15.7.4.2.2 Identification of Operator Actions

The following actions are carried out:

(1) Initiate the evacuation of the Reactor Building fuel-handling area and the locking of the fuel storage building doors.

(2) The fuel-handling foreman gives instructions to go immediately to the radiation protection personnel decontamination area.

(3) The fuel-handling foreman makes the operations shift engineer aware of the accident.

(4) The shift engineer determines if the normal ventilation system has isolated and the SGTS is in operation.

(5) The shift engineer initiates action to determine the extent of potential radiation doses by measuring the radiation levels in the vicinity of or close to the Reactor Building.

(6) The plant superintendent or delegate determines if the SGTS is performing as designed.

(7) The duty shift engineer posts the appropriate radiological control signs at the entrance of the Reactor Building.

(8) Before entry to the refueling area is made, a careful study of conditions, radiation levels, etc., is performed.

15.7.4.2.3 System Operation

Normally, operating plant instrumentation and controls are assumed to function, although credit is taken only for the isolation of the normal ventilation system and the operation of the standby gas treatment system. Operation of other plant or reactor protection systems or ESF systems is not expected.

15.7.4.3 Core and System Performance

15.7.4.3.1 Mathematical Model

The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a reasonable, yet conservative assessment of the consequences.

To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

The fuel assembly is expected to impact on the core at a small angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods absorb little energy prior to failure as a result of bending. Actual bending tests with concentrated point-loads show that each fuel rod absorbs approximately 1.35 N•m prior to cladding failure. Each rod that fails as a result of gross compression distortion is expected to absorb approximately 0.339 kN•m before cladding failure (based on 1% uniform plastic deformation of the rods). The energy of the dropped assembly is conservatively assumed to be absorbed by only the cladding and other pool structures. Because an unchanneled fuel assembly consists of greater than 70% fuel by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations that follow.

15.7.4.3.2 Input Parameters and Initial Conditions

The assumptions used in the analysis of this accident are:

(1) The fuel assembly is dropped from the maximum height allowed by the fuel-handling equipment, about 13.4m.

(2) The entire amount of potential energy, referenced to the top of core, is available for application to the fuel assemblies involved in the accident. This assumption neglects the dissipation of some of the mechanical energy of the falling fuel assembly and requires the complete detachment of the assembly from the fuel hoisting equipment. This is only possible if the fuel assembly handle, fuel grapple or grapple cable breaks.

(3) None of the energy associated with the dropped fuel assembly is absorbed by the fuel material (uranium dioxide).

Regulatory Guide 1.25 provides assumptions acceptable to the NRC that may be used in evaluating the radiological consequences of a postulated fuel-handling accident resulting in damage to the fuel cladding and subsequent release of radioactive materials.

The key implementation assumptions used in the analyses are as follows:

(1) Two-hour site boundary meteorology calculated in accordance with USNRC RG 1.3 and 1.145 for ground level release.

(2) SGTS filter efficiency 99% for all iodine forms.

(3) All activity released to the environment is via the SGTS.

(4) 115 fuel rods are calculated to be damaged.

15.7.4.3.3 Results

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason, a simplified energy approach was taken and numerous conservative assumptions were made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods was determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The wet weight of the dropped bundle is 279.87 kg and the wet weight of the grapple component is 158.76 kg. The drop distance is 13.38m. The total energy to be dissipated by the first impact is:

$$E = (279.87 + 158.76)\,(13.38) = 57.56 \text{ kN}\cdot\text{m}$$

One half of the energy was considered to be absorbed by the falling assembly and one half by the impacted assemblies.

No energy was considered to be absorbed by the fuel pellets (i.e., the energy was absorbed entirely by the non-fuel components of the assemblies). The energy available for clad deformation was considered to be proportional to the mass ratio:

$$\frac{\text{mass of cladding}}{(\text{mass of assembly} - \text{mass of fuel pellets})}$$

and is equal to a maximum of 0.519 for the fuel designs considered here.

The energy absorbed by the cladding of the four impacted assemblies is

$$(28.78 \text{ kN•m})\ (0.519) = 14.93 \text{ kN•m}$$

Each rod that fails is expected to absorb approximately 0.339 kN•m before cladding failure, based on uniform 1% plastic deformation of the cladding.

The number of rods failed in the impacted assemblies is:

$$N_F = \frac{(14.93 \text{ kN•m})}{(0.339 \text{ kN•m})} = 44 \text{ rods}$$

The dropped assembly was considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason, it was assumed that all the rods in the dropped assembly fail. The total number of failed rods on initial impact was $62 + 44 = 106$.

The assembly was assumed to tip over and impact horizontally on the top of the core. The remaining available energy was used to predict the number of additional rod failures. The available energy was calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

$$E = W_G H_G + \int_0^{H_B} \frac{W_B}{H_B} y \text{ dy}$$

$$= W_G H_G + \frac{1}{2} W_B H_B$$

$$= (158.76 \text{ kg})(4.06m) + \frac{1}{2}(279.87)(4.06m)$$

$$= 11.9(\text{ kN•m})$$

As before, the energy was considered to be absorbed equally by the falling assembly and the impacted assemblies and the fraction available for clad deformation was 0.519. The energy available to deform clad in the impacted assemblies was:

$$E_c = (0.5)(11.9 \text{ kN•m})(0.519)$$

$$= 3.09 \text{ kN·m}$$

and the number of failures in the impacted assemblies was

$$N_F = \frac{(3.09 \text{ kN•m})}{(0.339 \text{ kN•m})} = 9 \text{ rods}$$

Since the rods in the dropped assembly were considered to have failed in the initial impact, the total failed rods in both impacts is $106 + 9 = 115$.

15.7.4.4 Not Used

15.7.4.5 Radiological Consequences

Radiological analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines. The analysis is referred to as the "Design Basis Analysis".

The fission product inventory in the fuel rods assumed to be damaged is based on 1000 days of continuous operation at 4005 MWt. A 24 h period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 h following initiation of reactor shutdown. Figure 15.7-1 shows the leakage flow path for this accident.

15.7.4.5.1 Design Basis Analysis

The Design Basis Analysis is based on Regulatory Guide 1.25. The specific models, assumptions and the program used for computer evaluations are described in Reference 15.7-1. Specific values or parameters used in the evaluation are presented in Table 15.7-8.

15.7.4.5.1.1 Fission Product Release from Fuel

Per the conditions in Regulatory Guide 1.25, The following conditions are assumed applicable for this event:

(1) Power Level—4005 MWt for 3 years

(2) Plenum Activity—10% of the radioactivity for iodine and noble gases except Kr-85 and 30% for Kr-85

(3) Fission Product Peaking Factor—1.5 for those rods damaged

(4) Activity Released to Reactor Building—10% of the noble gas activity and 0.1% for the iodine activity

Based on the above conditions, the activity released to the Reactor Building is presented in Table 15.7-9.

15.7.4.5.1.2 Fission Product Transport to the Environment

Also, per the conditions of Regulatory Guide 1.25, it is assumed that the airborne activity of the Reactor Building (Table 15.7-9) is released to the environment over a 2 hr period via a 99% iodine efficient SGTS. The total activity released to the environment is presented in Table 15.7-10.

15.7.4.5.1.3 Results

The calculated exposures for the Design Basis Analysis are presented in Table 15.7-11 and are within the guidelines of 10CFR100. COL applicants need to update these calculations to conform to the as-designed plant and site-specific parameters. See Subsection 15.7.6.1 for COL license information.

15.7.5 Spent Fuel Cask Drop Accident

The radiological results documented in this section are based on an 8x8 BWR fuel design. The SVEA-96 Optima2 fuel design as described in this DCD has essentially the same post-discharge isotopic inventory as this 8x8 fuel.

15.7.5.1 Identification of Cause

Due to the redundant nature of the crane, the cask drop accident is not believed to be a credible accident. However, the accident is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fall from the level of the refueling floor to ground level through the refueling floor maintenance hatch.

15.7.5.2 Radiological Analysis

The largest size BWR fuel cask is conservatively assumed to be dropped approximately 29 m from the refueling floor level to ground level on transport from the decontamination pit out of the reactor building.

It is conservatively assumed that all fuel rods are damaged and the fission gases in the fuel rod gap space are released to the reactor building and then to the environment over a two-hour period. Table 15.7-12 provides the assumptions for this analysis and Table 15.7-13 radiological consequences. As can be seen from Table 15.7-13, the radiological releases are within guidelines. COL applicants need to update these calculations to conform to the as-designed plant and site specific parameters. See Subsection 15.7.6.1 for COL license information.

15.7.6 COL License Information

15.7.6.1 Radiological Consequences of Non-Line Break Accidents

The COL applicant will evaluate the radiological consequences of the failure of the liquid radwaste tank, the fuel handling accident, and the fuel cask drop accident for the final plant design and site parameters (see Subsections 15.7.3.4, 15.7.4.5.1.3, and 15.7.5.2.).

15.7.7 References

15.7-1 H. Careway, V. Nguyen, et al., “Radiological Accident Evaluation—The CONAC03 Code”, December 1981 (NEDO-21143-1).

Table 15.7-1 Offgas System Failure Accident Parameters

I. Data and Assumptions Used to Estimate Source Terms		
A.	Power Level	4005MWt
B.	Offgas Release Rate	14.8 GBq/s (referenced to 30 min)
C.	Charcoal Mass	Guard Tank, 4,721 kg
D.	Charcoal Delay[1]	
	Kr	2.07 h
	Xe	36.9 h
E.	Duration of Release	30 min
F.	Design Basis Rate	3.7 GBq/s
G.	Maximum Technical Specification Rate	14.8 GBq/s
II. Dispersion and Dose Rate		
A.	Meteorology	Table 15.7-3
B.	Dose Methodology	Reference 15.7-1
C.	Dose Conversion Assumptions	Reference 15.7-1, RG 1.109
D.	Activity Releases	Table 15.7-2
E.	Dose Evaluations	Table 15.7-3

Note 1:Charcoal Delay calculated based upon charcoal mass using equation 1.5.1.6 of NUREG-0016 and K_d's taken from 1.5.2.19 and 1.5.2.20 of NUREG-0016.

Table 15.7-2 Isotopic Source and Release to the Environment

Isotope	Isotope Flow Rates (Design Basis) T=0 (MBq/s)	T=2 min (MBq/s)	T=30 min (MBq/s)	Integrated Releases (at Max Tech Spec) T=30 min (MBq/s)	Charcoal Delay (MBq/s)	Total (MBq/s)
Kr-83m	1.27E+2	1.25E+2	1.05E+2	7.57E+5	4.07E+5	1.16E+6
Kr-85m	2.23E+2	2.22E+2	2.06E+2	1.48E+6	1.15E+6	2.63E+6
Kr-85	6.96E–1	6.96E–1	6.96E–1	5.18E+3	5.18E+3	9.99E+3
Kr-87	7.66E+2	7.51E+2	5.81E+2	4.19E+6	1.69E+6	5.88E+6
Kr-88	7.66E+2	7.59E+2	6.77E+2	4.87E+6	3.23E+6	8.10E+6
Kr-89	4.74E+3	3.05E+3	6.55E+0	4.74E+4	0.00	4.74E+4
Kr-90	1.04E+4	7.96E+2	1.75E–13	0.00	0.00	0.00
Kr-91	1.27E+4	8.92E–1	0	0.00	0.00	0.00
Kr-92	1.27E+4	2.96E–16	0	0.00	0.00	0.00
Kr-93	3.34E+3	1.18E–25	0	0.00	0.00	0.00
Kr-94	8.21E+2	0	0	0.00	0.00	0.00
Kr-95	7.66E+1	0	0	0.00	0.00	0.00
Kr-97	5.00E–1	0	0	0.00	0.00	0.00
Total	4.66E+4	5.71E+3	1.58E+3	1.14E+7	6.48E+6	1.78E+7
Xe-131m	5.44E–1	5.44E–1	5.44E–1	4.07E+3	3.70E+3	7.40E+3
Xe-133m	1.04E+1	1.04E+1	1.04E+1	7.47E+4	4.59E+4	1.21E+5
Xe-133	2.92E+2	2.92E+2	2.92E+2	2.10E+6	1.80E+6	3.90E+6
Xe-135m	9.73E+2	8.92E+2	2.50E+2`	1.80E+6	0.00	1.80E+6
Xe-135	8.36E+2	8.33E+2	8.03E+2	5.79E+6	3.39E+5	6.13E+6
Xe-137	5.44E+3	3.77E+3	2.42E+1	1.74E+5	0.00	1.74E+5
Xe-138	3.20E+3	2.90E+3	7.40E+2	5.33E+6	0.00	5.33E+6
Xe-139	1.04E+4	1.33E+3	4.03E–10	0.00E+0	0.00	0.00
Xe-140	1.11E+4	2.46E+1	0	0.00E+0	0.00	0.00
Xe-141	9.07E+3	9.03E–18	0	0.00E+0	0.00	0.00
Xe-132	2.65E+3	6.51E–27	0	0.00E+0	0.00	0.00
Xe-143	4.44E+2	0	0	0.00E+0	0.00	0.00
Xe-144	2.09E+1	0	0	0.00E+0	0.00	0.00
Total	4.45E+4	1.01E+4	2.12E+3	1.53E+7	2.19E+6	1.75E+7
Kr+XE	9.11E+4	1.58E+4	3.70E+3	2.66E+7	8.66E+6	3.53E+7

Table 15.7-3 Offgas System Failure Meteorology and Dose Results

Meteorology	Whole Body Dose
1.37E-3 s/m3	2.75 mSv

Table 15.7-4

Not Used

Table 15.7-5 Radwaste System Failure Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
	A. Power level	4005 MWt
	B. Source inventory	Table 12.2-13
	C. Fraction of iodine released	10%
	D. Duration of accident	2 h
II	**Dispersion and Dose Data**	
	A. Meteorology	Table15.7-7
	B. Dose commitment distance	800m
	C. Method of dose calculation	Reference 15.7-1
	D. Dose conversion assumptions	Reference 15.7-1, RG 1.109, ICRP 30.
	E. Activity released	Table 15.7-6
	F. Dose evaluations	Table 15.7-7

Table 15.7-6 Isotopic Release to Environment (megabecquerel)

Isotope	1-min	10-min	1-hour	2-hour
I-131	4.8E+4	4.4E+5	1.6E+6	1.9E+6
I-132	4.1E+3	3.5E+4	1.1E+5	1.3E+5
I-133	2.8E+4	2.6E+5	8.9E+5	1.1E+6
I-134	2.6E+3	2.2E+4	6.3E+4	7.0E+4
I-135	1.2E+4	1.1E+5	3.7E+5	4.4E+5
Total I	9.5E+4	8.7E+5	3.0E+6	3.6E+6

Table 15.7-7 Radwaste System Failure Accident Meteorology* and Dose Results

Meteorology (s/m^3)	Distance (m)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
1.42E–03	Max	3.0E–1	2.4E–4
1.37E-03	Chp 2	2.9E–1	2.3E–4
1.18E–03	300	2.5E–1	2.0E–4
2.19E–04	800	4.6E–2	3.8E–5

* Meteorology calculated using Regulatory Guide 1.145 for ground level 1.0 m/s, F stability release. “Max” = maximum meteorology to meet 10% of 10CFR100.

Table 15.7-8 Fuel-Handling Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
	A. Power level	4005 MWt
	B. Radial peaking factor	1.5
	C. Duration of accident	2 h
	D. No. rods damaged	115 rods
	E. Minimum time to accident	24 h
	F. Peak linear power density	44 kW/m
	G. Average burnup	32,000 MW•d/t
	H. Maximum fuel centerline temperature	1824°C
	I. Fraction of activity released	10% of all isotopes except 30% Kr-85
II	**Data and Assumptions Used to Estimate Activity Released**	
	A. Species fraction	
	(1) Organic iodine	0.25%
	(2) Inorganic iodine	99.75%
	(3) Noble gas	100%
	B. Pool Retention decontamination factor	
	(1) Organic iodine	1
	(2) Inorganic iodine	133
	(3) Noble gas	1
	C. SGTS filtration efficiency*	
	(1) Organic iodine	99%
	(2) Inorganic iodine	99%
	(3) Noble gas	0%
	D. Reactor Building Release Rate	300%/2 h
III	**Dispersion and Dose Data**	
	A. Meteorology	Table 15.7-11
	B. Boundary and LPZ distances	Table 15.7-11
	C. Method of dose calculation	Reference 15.7-1
	D. Dose conversion assumptions	Reference 15.7-1 RG 1.109, ICRP30
	E. Activity inventory/releases	Table 15.7-9, Table 15.7-10
	F. Dose evaluations	Table 15.7-11

* No SGTS filtration for first 20 minutes of accident.

Table 15.7-9 Fuel-Handling Accident Reactor Building Inventory (megabecquerel)

Isotope	1 minute	10minute	1 hour	2 hours
I-131	1.13E+07	8.99E+06	6.88E+06	6.73E+06
I-132	1.45E+07	1.11E+07	6.59E+06	4.77E+06
I-133	1.17E+07	9.25E+06	6.92E+06	6.55E+06
I-134	6.29E–01	4.44E–01	1.77E–01	7.88E–02
I-135	1.91E+06	1.50E+06	1.06E+06	9.32E+05
Total	3.93E+07	3.08E+07	2.14E+07	1.90E+07
Kr-83m	5.99E+05	4.51E+05	2.53E+05	1.69E+05
Kr-85m	7.66E+06	5.96E+06	4.03E+06	3.38E+06
Kr-85	4.18E+07	3.33E+07	2.56E+07	2.50E+07
Kr-87	1.18E+03	8.70E+02	4.22E+02	2.41E+02
Kr-88	2.21E+06	1.70E+06	1.07E+06	8.18E+05
Kr-89	2.32E–05	2.59E–06	3.55E–11	3.70E–16
Xe-131m	7.29E+06	5.81E+06	4.44E+06	4.37E+06
Xe-133m	9.66E+07	7.70E+07	5.85E+07	5.62E+07
Xe-133	2.46E+09	1.96E+09	1.50E+09	1.46E+09
Xe-135m	2.80E+07	1.51E+07	1.26E+06	8.66E+04
Xe-135	5.62E+08	4.44E+08	3.20E+08	2.90E+08
Xe-137	5.22E–05	8.14E–06	7.36E–10	1.04E–14
Xe-138	5.62E–05	2.90E–05	1.93E–06	1.00E–07
Total	3.21E+09	2.54E+09	1.91E+09	1.84E+09

Table 15.7-10 Fuel-Handling Accident Isotopic Release to Environment (megabecquerel)

Isotope	1 minute	10 minute	1 hour	2 hours
I-131	2.85E+05	2.56E+06	4.55E+06	4.55E+06
I-132	3.67E+05	3.22E+06	5.62E+06	5.62E+06
I-133	2.95E+05	2.64E+06	4.70E+06	4.70E+06
I-134	1.60E–02	1.36E–01	2.28E–01	2.28E–01
I-135	4.85E+04	4.29E+05	7.62E+05	7.62E+05
TOTAL	9.96E+05	8.85E+06	1.56E+07	1.56E+07
Kr-83m	1.52E+04	1.32E+05	2.33E+05	2.38E+05
Kr-85m	1.94E+05	1.72E+06	3.08E+06	3.16E+06
Kr-85	1.05E+06	9.47E+06	1.72E+07	1.77E+07
Kr-87	3.00E+01	2.59E+02	4.51E+02	4.55E+02
Kr-88	5.62E+04	4.92E+05	8.81E+05	8.99E+05
Kr-89	6.55E–07	2.77E–06	3.01E–06	3.01E–06
Xe-131m	1.84E+05	1.65E+06	3.00E+06	3.09E+06
Xe-133m	2.44E+06	2.18E+07	3.96E+07	4.07E+07
Xe-133	6.22E+07	5.59E+08	1.01E+09	1.04E+09
Xe-135m	7.25E+05	5.44E+06	8.18E+06	8.18E+06
Xe-135	1.42E+07	1.27E+08	2.29E+08	2.36E+08
Xe-137	1.45E–06	6.77E–06	7.66E–06	7.66E–06
Xe-138	1.46E–06	1.07E–05	1.59E–05	1.59E–05
TOTAL	8.11E+07	7.26E+08	1.32E+09	1.35E+09

Table 15.7-11 Fuel-Handling Accident Meteorological* Parameters And Radiological Effects

Meteorology (s/m^3)	Distance (m)	Thyroid Dose (Sv)	Whole Body Dose (Sv)
1.37E–03	max/Chp 2	7.5E–01	1.2E–02
1.18E–03	300	6.4E–01	1.1E–02
2.19E–04	800	1.2E–01	2.0E–03

* Meteorology calculated using Regulatory Guide 1.145 for a ground level 1.0 m/s, F stability release. "Max" = maximum meteorology to meet 25% of 10CFR100 limitation.

Table 15.7-12 Fuel Cask Drop Accident Parameters

I	**Data and Assumptions Used to Estimate Source Terms**	
	A. Power level of reactor while fuel was in core	4005 MWt
	B. Radial peaking factor while fuel was in core	1.5
	C. Fuel bundles in cask	18
	D. Fuel damaged	1116 rods
	E. Minimum time of fuel in storage prior to accident	120 days
	F. Peak linear power density	44 kW/m
	G. Average burnup of fuel	32,000 MW•d/t
	H. Maximum fuel centerline temperature	1824°C
	I. Fraction of activity released	10% of all isotopes except 30% Kr-85
	J. Time period for Reactor Building release	2 h
	K. Iodine filter efficiency	None
II	**Dispersion and Dose Data**	
	A. Meteorology	Table 15.7-13
	B. Boundary and LPZ distances	Table 15.7-13
	C. Method of dose calculation	Reference 15.7-1
	D. Dose conversion assumptions	Reference 15.7-1 and RG 1.109
	E. Activity inventory/releases	Table 15.7-13
	F. Dose evaluations	Table 15.7-13

Table 15.7-13 Cask Drop Accident Radiological Results Fission Product Releases (megabecquerel)

Isotope	Release to Reactor Building	Release to Environment
I-131	4.0E+5	3.8E+5
Kr-85	4.1E+8	3.7E+8
Xe-131m	2.0E+5	1.9E+5
Xe-133	4.8E+3	4.4E+3

Meteorology* and Dose Results

Meteorology (s/m^3)	Distance (m)	Thyroid (Sv)	Whole Body (Sv)
1.84E–02	max	7.4E–1	1.3E–3
1.37E-02	Chp 2	5.6E–2	1.0E–4
1.18E–03	300	4.8E–2	8.5E–5
2.19E–04	800	8.9E–3	1.6E–5

* Meteorology calculated using Regulatory Guide 1.145 for ground level 1.0 m/s, F stability release. “Max” = maximum meteorology to meet 25% of 10CFR100 limitation.

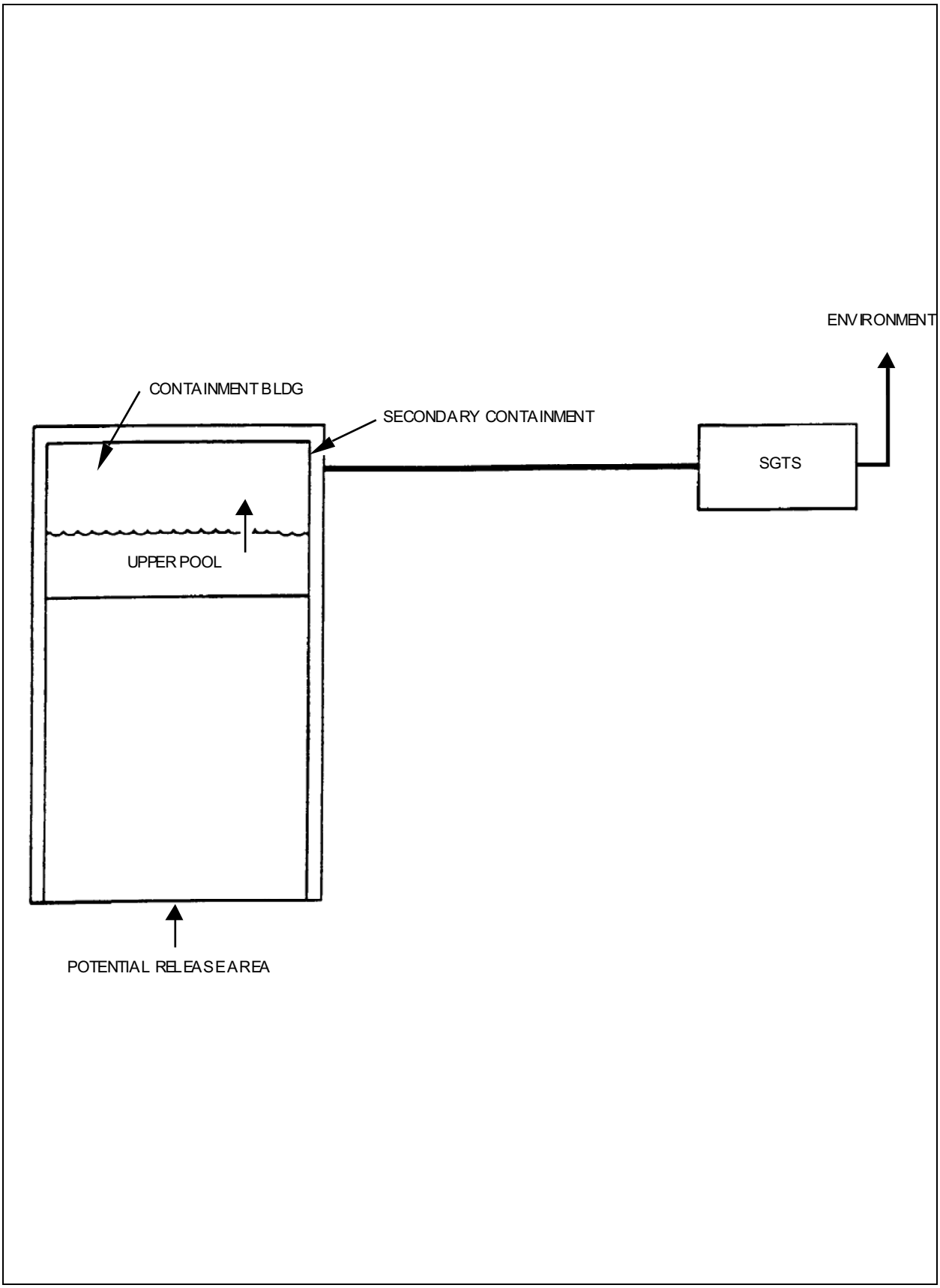

Figure 15.7-1 Leakage Path for Fuel-Handling Accident

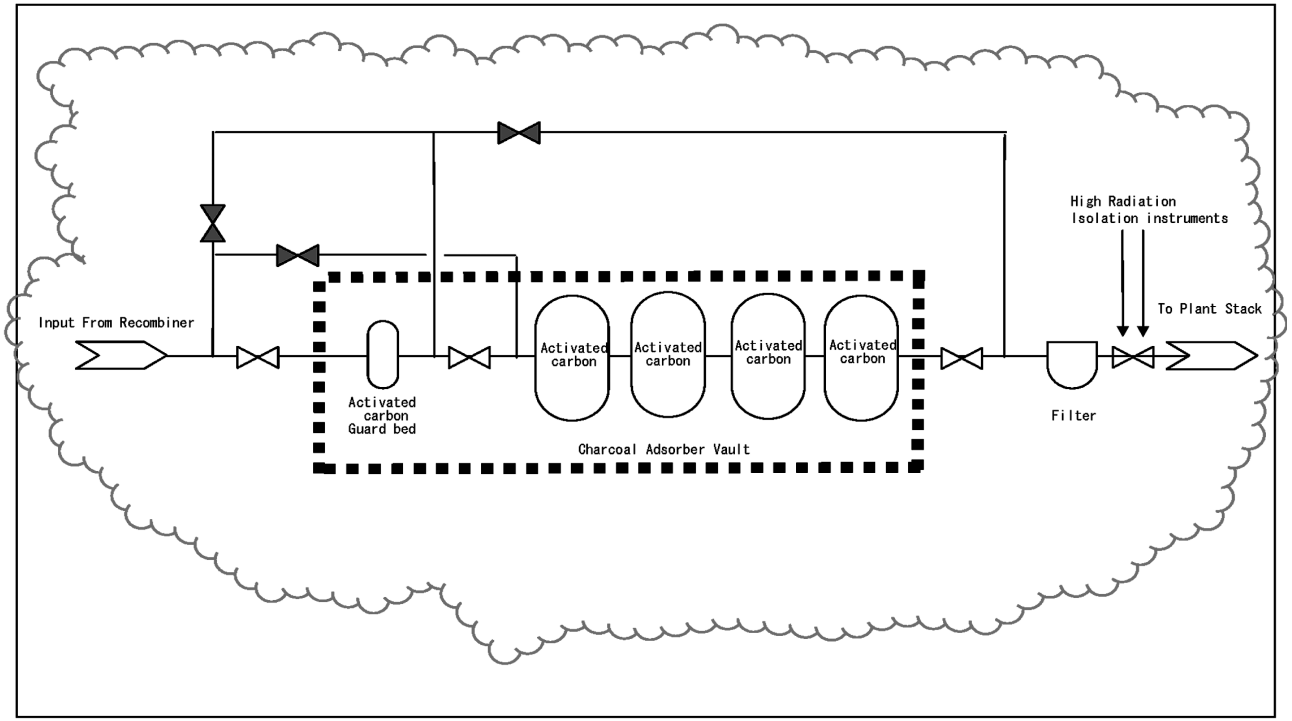

Figure 15.7-2 Offgas System (See Subsection 11.3)