# 15.6 Decrease in Reactor Coolant Inventory

## 15.6.1 Inadvertent Safety/Relief Valve Opening

This event is presented and analyzed in Subsection 15.1.4.

## 15.6.2 Failure of Small Line Carrying Primary Coolant Outside Containment

This event postulates a small steam or liquid line pipe break inside or outside the primary containment, but within a controlled release structure. To bound the event, it is assumed that a small instrument line, instantaneously and circumferentially, breaks at a location where it may not be able to be isolated and where detection is not automatic or apparent. This event is less limiting than the postulated events presented in Subsections 15.6.4, 15.6.5, and 15.6.6.

This postulated event represents the envelope evaluation for small line failure inside and outside the primary containment relative to sensitivity for detection.

### 15.6.2.1 Identification of Causes and Frequency Classification

#### 15.6.2.1.1 Identification of Causes

There is no specific event or circumstance identified which results in the failure of an instrument line. These lines are designed to high quality engineering standards and to seismic and environmental requirements. However, for the purpose of evaluating the consequences of a small line rupture, the failure of an instrument line is assumed to occur.

A circumferential rupture of an instrument line which is connected to the Primary Coolant System is postulated to occur outside the drywell, but inside the reactor building. This event could conceivably occur also in the drywell. However, the associated effects would not be as significant as those from the failure in the reactor building.

#### 15.6.2.1.2 Frequency Classification

This event is categorized as a limiting fault.

### 15.6.2.2 Sequence of Events and Systems Operations

#### 15.6.2.2.1 Sequence of Events

The leak may result in noticeable increases in radiation, temperature, humidity, or noise levels in the secondary containment or abnormal indications of actuations caused by the affected instrument.

Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolatable leak. The action consists of the orderly shutdown and depressurization of the reactor vessel.

#### 15.6.2.2.2 Systems Operation

A presentation of plant and reactor protection system action and ESF action is given in Sections 6.3, 7.3, and 7.6.

#### 15.6.2.2.3 The Effect of Single Failures and Operator Errors

There is no single failure or operator error that will significantly affect the system response to this event.

### 15.6.2.3 Core and System Performance

Instrument line breaks, because of their small size, are substantially less limiting from a core and systems performance standpoint than the events examined in Subsections 15.6.4, 15.6.5, and 15.6.6. Consequently, instrument line breaks are considered to be bounded specifically by the steamline break (Subsection 15.6.4). Details of this calculation, including those pertinent to core and system performance, are presented in Subsection 15.6.4.3.

#### 15.6.2.3.1 Input Parameters and Initial Conditions

All information concerning ECCS models employed, input parameters, and detailed results for a more limiting (steamline break) event are presented in Section 6.3.

#### 15.6.2.3.2 Results

No fuel damage or core uncovering occurs as a result of this accident. Similarly, instrument line breaks are within the spectrum considered in ECCS performance calculations presented in Subsection 6.3.3.

### 15.6.2.4 Barrier Performance

The following assumptions and conditions are the basis for the mass loss during the release period of this event:

(1) The instrument line releases coolant into the Reactor Building for a period of ten minutes at normal operating temperature and pressure. Following this 10-minute period, the operator is assumed to have isolated the event and taken steps to SCRAM the reactor to reduce reactor pressure over a period of 5.4 hours.

(2) The flow from the instrument line is limited by reactor pressure and a 0.64 cm diameter flow restricting orifice inside the drywell. The Moody critical blowdown model is applicable, and the flow is critical at the orifice (Reference 15.6-1).

The total integrated mass of fluid released into the Reactor Building is 13610 kg, with approximately 2270 kg being flashed to steam.

#### 15.6.2.5 Radiological Analysis

##### 15.6.2.5.1 General

The radiological analysis is based upon conservative assumptions considered acceptable to the NRC. Though the Standard Review Plan does not provide detailed guidance, the assumptions found in Table 15.6-1 assume that all of the iodine available in the flashed water is transported via the HVAC System or blowout panels to the environment without prior treatment by the Standby Gas Treatment System. Other isotopes in the water contribute only negligibly to the total dose.

##### 15.6.2.5.2 Fission Product Release

The iodine activity in the coolant is assumed to be at the maximum equilibrium Technical Specification limit (see Subsection 15.6.4.5.1.1, Case 1) for continuous operation. The iodine released to the Reactor Building atmosphere and to the environment are presented in Table 15.6-2.

##### 15.6.2.5.3 Results

Results of the analysis (Table 15.6-3) are within the 10% of 10CFR100 specified in the Standard Review Plan. COL applicants need to update the analysis to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information).

### 15.6.3 Steam Generator Tube Failure

This section is not applicable to the direct cycle BWR.

### 15.6.4 Steam System Piping Break Outside Containment

This event involves postulating a large steamline pipe break outside containment. It is assumed that the largest steamline, instantaneously and circumferentially breaks at a location downstream of the outermost isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line and actuate the necessary protective features. This postulated event represents the envelope evaluation of steamline failures outside containment.

#### 15.6.4.1 Identification of Causes and Frequency Classification

##### 15.6.4.1.1 Identification of Causes

A main steamline break is postulated without the cause being identified. These lines are designed to high quality engineering codes and standards, and to seismic and environmental requirements. However, for the purpose of evaluating the consequences of a postulated large steamline rupture, the failure of a main steamline is assumed to occur.

#### 15.6.4.1.2 Frequency Classification

This event is categorized as a limiting fault.

### 15.6.4.2 Sequence of Events and Systems Operation

#### 15.6.4.2.1 Sequence of Events

Accidents that result in the release of radioactive materials directly outside the containment are the result of postulated breaches in the reactor coolant pressure boundary or the steam power conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting event for breaks outside the containment is a complete severance of one of the four main steamlines. The sequence of events and approximate time required to reach the event is given in Table 15.6-4.

The reactor operator maintains reactor vessel water inventory and core cooling with the RCIC System or with one of the HPCF Systems. Without operator action, the RCIC and the HPCF Systems would initiate automatically on low water level following isolation of the mainsteam supply system (i.e., MSIV closure). The core remains covered throughout the accident and there is no fuel damage.

#### 15.6.4.2.2 Systems Operation

A postulated guillotine break of one of the four main steamlines outside the containment results in mass loss from both ends of the break. The flow from the upstream side is initially limited by the flow restrictor within the reactor vessel steam outlet nozzle. Flow from the downstream side is initially limited by the total area of the flow restrictors within the reactor vessel steam outlet nozzles for the three unbroken lines. Subsequent closure of the MSIVs further limits the flow when the valve area becomes less than the limiter area and finally terminates the mass loss when the full closure is reached.

Discussions of plant and reactor protection system action and ESF action are presented in Sections 6.3, 7.3 and 7.6.

#### 15.6.4.2.3 The Effect of Single Failures and Operator Errors

The steamline break outside the containment is a special case of the general LOCA break spectrum considered in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in Subsection 6.3.3.3. For the steamline break outside the containment, because the break is isolatable, either the RCIC System or one of the HPCF systems can provide adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either one of the HPCF systems or the RCIC System would still allow sufficient flow to keep the core covered with water (see Section 6.3 and Appendix 15A for analysis details).

#### 15.6.4.3 Core and System Performance

Quantitative results (including mathematical models, input parameters, and consideration of uncertainties) for this event are presented in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage.

##### 15.6.4.3.1 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 6.3-1.

##### 15.6.4.3.2 Results

There is no fuel damage as a consequence of this accident.

Refer to Section 6.3 for ECCS analysis.

#### 15.6.4.4 Barrier Performance

Because this break occurs outside the containment, barrier performance within the containment envelope is not applicable. Details of the results of this event can be found in Subsection 6.2.3.

The following assumptions and conditions are used in determining the mass loss from the primary system from the inception of the break to full closure of the MSIVs:

(1) The reactor is operating at the power level associated with maximum mass release.

(2) Nuclear system pressure is initially 7.17 MPa.

(3) An instantaneous circumferential break of the main steamline occurs.

(4) Isolation valves start to close at 0.5 s on high flow signal and are fully closed at 5.5 s.

(5) The Moody critical flow model (Reference 15.6-1) is applicable.

Initially, only steam will issue from the broken end of the steamline. The flow in each line is limited by critical flow at the limiter to a maximum of 200% of rated flow for each line. Rapid depressurization of the RPV causes the water level to rise, resulting in a steam-water mixture flowing from the break until the valves are closed. The total integrated mass leaving the RPV through the steamline break is 34,817 kg (21,949 kg of liquid and 12,868 kg of steam).

#### 15.6.4.5 Radiological Consequences

The radiological analysis for this accident is based on conservative assumptions considered to be acceptable to the NRC for the purposes of determining adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the “design basis analysis.”

A schematic of the release path is shown in Figure 15.6-1.

#### 15.6.4.5.1 Design Basis Analysis

The specific models, assumptions and the program used for computer evaluation are described in Reference 15.6-2. Specific values of parameters used in the evaluation are presented in Table 15.6-5.

General Compliance or Alternate Approach Statement (RG 1.5):

This guide provides assumptions acceptable to the NRC that may be utilized in evaluating the radiological consequences of a steamline break accident for a BWR.

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

(1) All regulatory position requirements implemented.

(2) Site boundary and LPZ $\chi/Q$ are in conformance with NRC Regulatory Guide 1.145.

Some of the models and conditions that are prescribed are inconsistent with actual physical phenomena. The impact of the conservative bias that is introduced is generally limited to plant design choices not within the scope of the ABWR standard design. The resultant dose is within regulatory limits.

##### 15.6.4.5.1.1 Fission Product Release from Fuel

There is no fuel damage as a result of this accident. The only activity available for release from the break is that which is present in the reactor coolant and steamlines prior to the break. This level of activity is consistent with an offgas release rate of 3.7 GBq/s for Case 1 and 14.8 GBq/s for Case 2 referenced to a 30 minute decay. The iodine concentration in the reactor coolant is:

| | MBq/g | |
|---|---|---|
| | **Case 1** | **Case 2** |
| I-131 | 0.001739 | 0.03515 |
| I-132 | 0.01536 | 0.30747 |
| I-133 | 0.01206 | 0.24161 |
| I-134 | 0.02634 | 0.52688 |
| I-135 | 0.01647 | 0.3293 |

Other isotopes of high intrinsic activity such as N-16 have been precluded due to their extremely short half lives.

#### 15.6.4.5.1.2 Fission Product Transport to the Environment

The transport pathway is a direct unfiltered release to the environment. The MSIV detection and closure time of 5.0 s (maximum MSIV closing time and instrument delay) results in a discharge of 12,870 kg of steam and 21,953 kg of liquid from the break. Assuming all the activity in this discharge becomes airborne, the release of activity to the environment is presented in Table 15.6-6.

#### 15.6.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.6-7 and are less than the guidelines of 10CFR100. COL applicants need to update the calculations to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information.).

## 15.6.5 Loss-of-Coolant Accident (Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary)—Inside Containment

This event postulates a spectrum of piping breaks inside containment varying in size, type, and location. The break type includes steam and/or liquid process system lines. This event is also assumed to be coincident with a safe shutdown earthquake (SSE) for the mechanical design of components.

The event has been analyzed quantitatively in Sections 6.3 (Emergency Core Cooling Systems), 6.2 (Containment Systems), 7.3 and 7.1 (Instrumentation and Controls), and 8.3 (Onsite Power Systems). Therefore, the following discussion provides only information not presented in the subject sections. All other information is cross-referenced.

The postulated event represents the envelope evaluation for liquid or steamline failures inside containment.

### 15.6.5.1 Identification of Causes and Frequency Classification

#### 15.6.5.1.1 Identification of Causes

There are no realistic, identifiable events which would result in a pipe break inside the containment of the magnitude required to cause a loss-of-coolant accident coincident with an SSE. The subject piping is of high quality, designed to construction industry codes and standards, and for seismic and environmental conditions. However, because such an accident provides an upper limit estimate for the resultant effects for this category of pipe breaks, it is evaluated without the causes being identified.

#### 15.6.5.1.2 Frequency Classification

This event is categorized as a limiting fault.

### 15.6.5.2 Sequence of Events and Systems Operation

#### 15.6.5.2.1 Sequence of Events

The sequence of events associated with this accident is presented in Table 6.3-2 for core system performance.

Following the pipe break and scram, the MSIV begins closing on the low level 1.5 signal. The low water level or high drywell pressure signal initiates RCIC, HPCF and RHR flooding systems.

#### 15.6.5.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator, after assuring that all rods have been inserted, should perform the following:

(1) Determine plant conditions by observing the annunciators.

(2) After observing that the ECCS flows are initiated, check that the diesel generators have started and are on standby condition and confirm that the Service Water System is operating in the LOCA mode.

(3) After the RHR System and other auxiliary systems are in proper operation, the operator should periodically monitor the oxygen concentration in the drywell and wetwell.

#### 15.6.5.2.3 Systems Operations

Accidents that could result in the release of radioactive fission products directly into the containment are the results of postulated nuclear system primary coolant pressure boundary pipe breaks. Possibilities for all pipe breaks, sizes and locations are presented in Sections 6.2 and 6.3, including the severance of main steamlines, ECCS lines, feedwater lines, or other process system lines. The most severe nuclear system effects and the greatest potential release of radioactive material to the containment result from a complete circumferential break of one of the two HPCF injection lines. The minimum required functions of any reactor and plant protection system are presented in Sections 6.2, 6.3, 7.3, 7.6 and 8.3, and Appendix 15A.

### 15.6.5.3 Core and System Performance

#### 15.6.5.3.1 Mathematical Models

The analytical methods and associated assumptions which are used in evaluating the consequences of this accident are considered to provide conservative assessment of the expected consequences of this improbable event.

The details of these calculations, their justification, and bases for the models are developed in Sections 6.3, 7.3, 7.6, 8.3 and Appendix 15A.

#### 15.6.5.3.2 Input Parameters and Initial Conditions

Input parameters and initial conditions used for the analysis of this event are presented in Table 6.3-1.

#### 15.6.5.3.3 Results

Results of this event are presented in detail in Section 6.3. The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause perforation of the fuel cladding. Therefore, no fuel damage results from this accident. Post-accident tracking instrumentation and control is assured.

Continued long-term core cooling is demonstrated. Radiological impact is minimized and within limits. Continued operator control and surveillance is examined and provided.

### 15.6.5.4 Barrier Performance

The design basis for the containment is to maintain its integrity and experience normal stresses after the instantaneous rupture of any primary system piping within the structure, while also accommodating the dynamic effects of the pipe break at the same time an SSE is also occurring. Therefore, any postulated LOCA does not result in exceeding the containment design limit (see Sections 3.8.2.3, 3.6, and 6.2 for details and results of the analyses).

### 15.6.5.5 Radiological Consequences

The source term due to postulated LOCAs is completely dominated by the reactor power level, especially for the iodine dose (which is the limiting factor in dose calculations). Therefore the results reported herein are valid for a wide range of BWR fuel, including SVEA-96 Optima2.

Two specific analyses are provided for the evaluation of the radiological consequences of a design basis LOCA, one for offsite dose evaluations and the second for control room dose evaluations. Both analyses are based upon assumptions provided in Regulatory Guide 1.3 except where noted. The analysis is based upon a process flow diagram shown in Figure 15.6-2 and accident parameters specified in Table 15.6-8.

#### 15.6.5.5.1 Fission Product Release and Pathways to the Environment

Fission product releases are based upon Regulatory Guide 1.3, in that it is assumed that of the fission products found in the core, 100% of the noble gases and 50% of the iodines are released from the core. Of these iodines, 50% are assumed to plate out, leaving 25% of the total core inventory of iodine airborne and available for release. The chemical species differentiation for the iodine isotopes released to the containment atmosphere is assumed as specified in Regulatory Guide 1.3 as 91% elemental form, 4% organic form, and the remaining 5% as particulate form. Following the release of fission products to the containment atmosphere from the reactor pressure vessel, the fission products are subject to holdup and radioactive decay, removal processes, and leakage to other plant areas and to the environment.

Two specific pathways are analyzed in releasing fission products to the environment. The first pathway is leakage to the Reactor Building (secondary containment) via penetrations and engineered safety feature (ESF) components. This leakage pathway is assumed as not greater than an equivalent release of 0.5% by weight per day of the primary containment free air weight per plant Technical Specifications. The secondary containment is a multi-compartment self-contained structure maintained at negative pressure with respect to the environment, thereby providing a significant holdup volume for fission product releases. All leakage pathways from the primary containment, except the main steamlines and the feedwater lines, terminate in the Reactor Building. Leakage through the steamlines is treated separately below, and leakage through the feedwater lines is assumed negligible assuming the proper isolation and filling of the feedwater lines upstream of the primary containment through the feedwater system. Flow through the Reactor Building/secondary containment is directed via the Standby Gas Treatment System to the plant stack through HEPA and charcoal filters. Credit is taken for holdup, assuming 50% mixing in the secondary containment without plateout and other removal processes except filtration in the SGTS (Table 15.6-8). It is assumed that for the first 20 minutes after an isolation signal, the SGTS is drawing the Reactor Building down to negative pressures, and therefore all leakage during this time period is assumed without effective filtration. Following this 20 minute period, full filtration is assumed for the remainder of the period.

The removal process in the primary containment and for leakages from the primary containment is described in the following sections. Subsection 15.6.5.5.1.1 discusses reductions in airborne iodine due to water attrition, while Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3 discuss removal processes for leakages downstream of the MSIVs.

#### 15.6.5.5.1.1 Suppression Pool Scrubbing

The BWR suppression pool, though designed primarily as a pressure suppression mechanism for vessel blowdown, serves also as an excellent medium for the entrainment and capturing of all fission products except the noble gases. The design and operational characteristics of the BWR provide for a release pathway from the vessel and drywell into the suppression pool for all cases involving vessel depressurization and, therefore, for removal of fission products by scrubbing in the suppression pool. The NRC has accepted the fact that the suppression pool is capable of removing fission products and provides for credit to incorporate this phenomenon in design basis analysis by recourse to the requirements of Standard Review Plan 6.5.5. The requirements of SRP 6.5.5 state that any flow directed through the pool can be credited with a decontamination factor (DF) of 10 providing the requirements of Subsection II are met and that the total decontamination is a combination of the decontamination applied to flow through the pool to that fraction of the release which bypasses the pool. The following paragraphs describe the determination of the bypass fraction for the calculation of overall pool decontamination.

The requirements of Regulatory Guide 1.3 stipulate an instantaneous release of fission products from the vessel to the containment atmosphere. Coincident with an instantaneous release, under LOCA conditions, the BWR pressure vessel will be depressurized, resulting in the purging of the primary containment atmosphere to the suppression pool. This situation is shown in Figure 15.6-3, which shows the fractions of airborne particulate as a function of time in the drywell and wetwell airspaces, assuming a decontamination factor of 10 for that flow which is purged either through the horizontal vents or the safety/relief valves. The figure shows that the airborne inventory is reduced by almost a factor of ten within two minutes of the initiation of the blowdown event.

However, the application of the precepts of Regulatory Guide 1.3 do not indicate the most likely train of events in a core damage event, which is what is implied in the design basis release assumptions. Both Regulatory Guide 1.3 and its predecessor, TID-14844, are based upon non-mechanistic assumptions and devices and are in the process of being replaced. Therefore, consideration of a range of accident progressions beyond the rigidly narrow scope of Regulatory Guide 1.3 is given below to evaluate potential suppression pool bypass under more realistic conditions.

The basic assumption of this evaluation of suppression pool bypass conditions is that an event occurs which challenges the reactor core causing sufficient damage to release approximately half the fission product volatile iodines. Damage to the core is limited to this extent, implying the ability to recover core cooling and limit in-vessel damage. Such an assumption complies with the intent of design basis licensing, in that the exact means by which the core is challenged is not specified; but given the challenge, the response and adequacy of the plant design is tested. In addition, the assumption of resumption of core cooling and recovery with limited release is fully justifiable, since the ABWR incorporates multiple cooling modes with redundant safety grade cooling systems. Events leading to more significant core damage are not considered as design basis, since they assume massive damage with “multiple failures to the design safety systems.” Such events are of exceedingly low probability and are described and evaluated in Chapter 19. Therefore, broadly speaking, events which lead to the assumed damage can be divided into two categories, break and non-break. Break events are those through which primary coolant are released directly to the primary containment atmosphere, and non-break events are those in which the primary coolant boundary is not breached. Both types of events will be considered below to provide a bounding analysis for suppression pool bypass.

In considering the non-break events, core damage is primarily the result of failure to maintain proper core water level, resulting in uncovering the core with subsequent release of fission products upon overheating of the fuel rods. To consider the train of events in such a case, the MAAP code (see Appendix 19E for a description of the MAAP code) was used to model vessel response. Based upon the MAAP analysis, releases would begin shortly after core water level reaches the bottom of the core and would proceed rapidly. During this period, isolation of the Primary Coolant System and containment would have been automatically tripped on low water level and the MSIVs, as well as all the other isolation valves, would have tripped, effectively isolating all flow from the primary containment. Therefore, the released fission products would be exposed to three primary influences: (1) plateout and removal in the dryers and separators, (2) leakage from the MSIVs into the main steamlines, and (3) flow through the SRVs into the suppression pool.

The release of volatile fission products would occur over a period of 10–20 minutes, during which steam or hydrogen flow from the core region would be very small. Using an upper bound estimate of 2 kg/s of steam generation during this period, the vessel flushing rate would be once every ten minutes. Therefore, during this period, 0.13% of the flow would bypass the pool through MSIV leakage. The remaining fraction would be transported through the SRVs. Without recovery of cooling water after this period, significant damage would occur to the core beyond that of a design basis event. With the recovery of water, the energy generated from decay heat which would be evident in overall core temperature rise and core degradation would cause a rapid pulse of steam, resulting in purging of the pressure vessel of all airborne materials. Based upon the MAAP analysis, it is conservatively estimated that $9 \times 10^3$ kg of steam would be generated in a short period of time (on the order of minutes), resulting in a vessel purge rate of seven to eight complete exchanges. Therefore, effectively all fission products remaining airborne in the vessel or lines would be purged to the suppression pool. The effective pool bypass fraction would then be 0.13% for an integrated overall DF of 9.8 without credit for plateout or 4.9 with a factor of two plateouts.

The break case follows a similar logic. Initially, following a break, massive depressurization of the pressure vessel would occur, causing all non-condensables in the drywell to be purged into the wetwell air space through both the horizontal vents and the safety/relief valves. Isolation of the containment and associated lines would be automatically initiated on depressurization. Following this rapid depressurization, there would follow a period during which the water level in the vessel would drop to the bottom of the core, resulting in the eventual release of fission products from the core. Since in a break case the path of least resistance would be through the break, the fission products would be effectively purged to the drywell airspace. In this case, the temperatures and surface areas involved would provide adequate plateout areas to validate the Regulatory Guide 1.3 plateout factor of 2. Like the non-break case, the total release is limited, implying resumption of cooling and a massive release of steam upon resumption of cooling. In the case of reflood with a break, because of the large volume of the drywell, conservatively 80% of the drywell volume is purged during the reflood period. If complete mixing is assumed, which is reasonable because of the dynamic flows involved, it is then found that 55.6% of the airborne fission products are purged to the suppression pool in the few minutes needed to reflood the core. Therefore, in this case an integrated pool DF of 2 is calculated.

In summary, it is found that for DBA conditions, the suppression pool is capable of reducing the elemental and particulate airborne iodine inventory by a factor of 2. Credit is taken for the proper operation of redundant safety grade systems subject to the single-failure criteria.

#### 15.6.5.5.1.2 Main Steamline Modeling

The second potential release pathway is via the main steamline through leakage in the main steamline isolation valves. It is assumed that a pathway exists which permits the primary containment atmosphere, or in the non-break case pressure vessel air space, direct access to the main steamlines and that the MSIVs leak at the maximum technical specification. Furthermore, it is assumed that the most critical MSIV fails in the open position. Therefore, the total leakage through the steamlines is equal to the maximum technical specification for the plant.

The main steamlines are graded (Table 3.2-1) as Seismic Classification I Quality Group B from the pressure vessel interface to the outboard seismic restraint outboard of the downstream MSIV, thereby providing a qualified safety grade mitigation system for fission product leakage, which in this case is limited by the leakage criteria specific in the technical specifications for the MSIV. The primary purpose of this system is to stop any potential flow through the main steamlines. Downstream of the seismic restraint referred to above, the steamlines pass through the Reactor Building-Control Building interface into the steam tunnel located in the Control Building upper floor. This steam tunnel is a heavily-shielded Seismic Category I structure designed primarily to shield the Control Building complex. From the Control Building the steamlines pass through the Control Building-Turbine Building interface into the Turbine Building steam tunnel, which is a heavily shielded reinforced concrete structure designed to shield workers from main steamline radiation shine. The steamlines and their associated branch lines outboard of the last Reactor Building seismic restraint are Quality Group B structures. In addition, these lines and structures are required to be dynamically analyzed to SSE conditions (Subsection 3.2.5.3) which determine the flexibility and structural capabilities of the lines under hypothetical SSE conditions.

The analysis of leakage from the primary containment through the main steamlines involves the determination of (1) probable and alternate flow pathways, (2) physical conditions in the pathways, and (3) physical phenomena which affect the flow and concentration of fission products in the pathways. The most probable pathway for fission product transport from the main steamlines is found to be from the outboard MSIVs into the drain lines coming off the outboard MSIV and then into the Turbine Building to the main condenser. A secondary path is found along the main steamlines into the turbine though flow through this pathway as described below is a minor fraction of the flow through the drain lines. Consideration of the main steamlines and drain line complex downstream of the Reactor Building as a mitigative factor in the analysis of LOCA leakage is based upon the following determination.

(1) The main steamlines and drain lines are high quality lines inspected on a regular schedule.

(2) The main steamlines and drain lines are designed to meet SSE criteria and analyzed to dynamic loading criteria.

(3) The main steamlines and drain lines are enclosed in a shielded corridor which protects them from collateral damage in the event of an SSE. For those portions not enclosed in the steam tunnel complex, an as-built inspection is required to verify that no damage could be expected from other components and structures in a SSE.

(4) The main steamlines and drain lines are required under normal conditions to function to loads at temperature and pressure far exceeding the loads expected from an SSE. This capability inherent in the basic design of these components furnishes a level of toughness and flexibility to assure their survival under SSE conditions. A large database of experience in the survival of these types of components under actual earthquake conditions exists which proves this contention (Reference 15.6-4). In the case of the ABWR, further margin for survival can be expected, since the ABWR lines are designed through dynamic analysis to survive such events, whereas in the case of the actual experience database, the lines shown to survive were designed to lesser standards to meet only normally expected loads.

Therefore, based upon the facts above, the main steamlines and drain lines in the ABWR are used as mitigative components in the analysis of leakage from the MSIVs.

The analysis of leakage from the MSIVs follows the procedures and conditions specified in Reference 15.6-4. Two flow paths are analyzed for dose contributions. The first pathway through the drain lines is expected to dominate because of the incorporation of a safety grade isolation valve on the outboard drain line which will open the line for flow down the drain line under LOCA conditions. The second pathway through the main steamlines into the turbine is expected to carry less than 0.3% of the flow based upon a determination that the maximum leakage past the turbine stop valves with an open drain line would permit only 0.3% flow for the valves to operate within specification. Specific values used and results of the main steamline leakage analysis are given in Table 15.6-8.

The COL applicant will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain, and main condenser. See Subsection 15.6.7.1 for COL license information requirements.

#### 15.6.5.5.1.3 Condenser and Turbine Modeling

The condenser and turbine are modeled as detailed in Reference 15.6-4 with specific values used given in Table 15.6-8. Both volumes are modeled primarily as stagnant volumes, assuming the shutdown of all active components. Both turbine and condenser are used as mitigative volumes based upon the determination that such components designed to standard engineering practice are sufficiently strong to withstand SSE conditions due wholly to their design (Reference 15.6-4). The only requirement in the design of the condenser is that it be bolted to the building basemat to prevent walking during an earthquake. The turbine has no such restriction and may possibly move. The requirement on these components for purposes of mitigation is only that they survive as a volume and not that they provide functionality or leaktightness following an earthquake.

Release from the condenser/Turbine Building pathway is assumed via diffuse sources in the Turbine Building. The two major points of release in the Turbine Building are expected to be the truck doors at the far end of the Turbine Building and the maintenance panels located midway on the Turbine Building on the side opposite the service building. Releases are assumed to be ground level releases. See Subsection 15.6.5.5.3 for applicable meteorology.

The COL applicant will recalculate iodine removal credit on the basis of its design characteristics of main steamlines, drain, and main condenser. See Subsection 15.6.7.1 for COL license information requirements.

### 15.6.5.5.2 Control Room

The ABWR control room is physically integrated with the Reactor Building and Turbine Buildings and is located between these structures (Figure 15.6-4). During a LOCA, exposure to the operators will consist of contributions from airborne fission products entrained into the control room ventilation system and gamma shine from the Reactor Building and airborne fission products external to the Control Building. Of these contributions, the last two involving gamma shine are negligible, since the inhabited portions of the ABWR control room are physically located underground with sufficient shielding overhead (a minimum of 1.6 meters of concrete) and in the side walls (1.2 meters) to protect the operators from the normal steamline gamma shine. Such shielding is more than sufficient to protect the operators given any amount of airborne fission products.

Therefore, exposure to the operators will consist almost entirely of fission products entrained into the control room environment from the atmosphere. The ABWR control room uses a redundant safety grade HVAC System with 100 mm (four-inch) charcoal filters for removal of iodines and two wall-mounted automatically controlled intake vents. The locations of the vents are given in Figure 15.6-4. Because of the location of these vents, it cannot be assumed that at least one vent will be uncontaminated, given most conditions of meteorology. Therefore, no credit for dual intakes was taken. In addition, the location of these vents with respect to the potential release points shows that, given any wind flow condition, the vents may be contaminated only by a release from the Reactor Building or Turbine Building but not both. Nevertheless, for purposes of conservative calculations, it was arbitrarily assumed that for 30% of the time stagnant meteorological conditions were assumed such that the primary intake vent was contaminated by both sources. For the remaining 70% of the time, only the more significant source was assumed to contaminate the primary intake vent.

Infiltration of airborne contamination to the control room was considered negligible, owing to the pathway for access to the control room complex. Entry into the control room is via the Service Building and a labyrinth doorway entry system through double doors into the clean portions of the Service Building. From the Service Building, additional controlled access through double doors provides entry into the control room. In each of these entry/access door systems, positive pressure is maintained to vent infiltrated air to the outside and away from the control room complex. As such, no contamination is anticipated beyond the initial access entry way from which infiltrating air is purged to the environment.

Control room dose is based upon fission product releases modeled as described in Subsection 15.6.5.5.1 and the values presented in Table 15.6-8. Operator exposure was based upon those conditions given in Table 15.6-8 and occupancy factors as shown below derived from SRP 6.4. Meteorology was derived as is specified in Subsection 15.6.5.5.3.2.

| Time | Occupancy Factor |
|---|---|
| 0–1 day | 1.0 |
| 1–4 days | 0.6 |
| >4 days | 0.4 |

### 15.6.5.5.3 Meteorology

#### 15.6.5.5.3.1 Offsite Meteorology

Tier 2 involves the use of a generic U.S. site which does not specifically identify meteorological parameters adequate to define dispersion conditions for accident evaluation. Therefore, for the evaluation of offsite accident conditions, recourse was made to Regulatory Guides 1.145 and 1.3 for meteorological definitions. Specifically, the table found in Section C.2.g(4) of Regulatory Guide 1.3 was used to define the meteorological parameters for use with the models found in Regulatory Guide 1.145. All releases were defined as ground level incorporating building wake conditions using the minimum ABWR building cross section.

Unlike the other design basis accidents found in Chapter 15, the LOCA accident analysis requires the development of meteorological conditions over a 30 day period. To develop a bounding 30 day set of four $\chi$/Q dispersion parameters, recourse was made to Regulatory Guide 1.3 and the metrological prescription found under Subsection 2.g. From this prescription, the $\chi$/Q values for 30 days were “walked” in from a 4828 m LPZ to approximately 1140 meters where the 30 day thyroid dose became 3 Sv. By plotting these resulting four $\chi$/Q values on log-log paper a straight line curve was established from which a 2-hour 95% LPZ $\chi$/Q and annual average $\chi$/Q value were back fitted with a small factor of conservatism in the derivation so that the resultant integrated dose was less than 300 Rem. The resultant straight line plot and $\chi$/Q values are shown in Figure 15.6-6. The end points are the 95% 2-hour LPZ $\chi$/Q of 4.11E-04 and annual average (8760 hour) $\chi$/Q of 1.17E-06 from which the intermediate values given in Table 15.6-13 (shown as Chp 2 values) were derived as specified in Regulatory Guide 1.145.

#### 15.6.5.5.3.2 Control Room Meteorology

No specific acceptable method exists to calculate the meteorology for standard plant application for control room dose analysis. Unlike the offsite dose methodology, which is a relatively straight forward application of Regulatory Guides 1.3 and 1.145, the parameters and methods by which the control room intake concentrations can be calculated are poorly characterized and currently not codified in a usable form. Therefore, for application to the ABWR, a back-calculation was used to provide an estimate of the meteorological $\chi/Q$ dispersion parameters which would provide for the maximum acceptable dose under SRP 6.4. Since the calculation covers a period of 30 days, a variation in meteorological $\chi/Q$ was assumed for variations in wind direction and wind speed. The variation factors chosen were taken from Table 1 of Reference 15.6-3 and are shown below.

| Time Period | Murphy-Campe $\chi/Q$ Improvement Factor |
|---|---|
| 0–8 hours | 1.0 |
| 8–24 hours | 0.59 |
| 1–4 days | 0.375 |
| > 4 days | 0.165 |

Also, since the control room may be contaminated from two physically separated sources, the Reactor Building stack base or the Turbine Building truck doors, reference was made to the most recently published work of Ramsdell to evaluate the differences in $\chi/Q$ for releases from each source to the control building. Using the methodology given in References 15.6-5 and 15.6-6, it was determined that releases from the Turbine Building at 108 meters from the control room intake would be a factor of six lower in concentration for an equal release than releases from the Reactor Building stack base at 41 meters from the nearest Control Building intake. Therefore, a factor of six improvement in $\chi/Q$ was assumed for releases from the Turbine Building.

For application to specific site analysis, two methods exist for determination of control room dose. The first method is a one-on-one comparison of the $\chi/Q$ values in Table 15.6-14 to the site $\chi/Q$s. If the site $\chi/Q$s are for all values less than the values in Table 15.6-14, then the control room doses are less than regulatory requirements. If this is not true, then a site specific calculation needs to be performed for the site. For this purpose, an isotope-by-isotope release rate table is given in Tables 15.6-10 and 15.6-12, from which actual calculations can be made.

#### 15.6.5.5.4 Results

The results of this analysis are presented in Tables 15.6-13 and 15.6-14 for both offsite and control room dose evaluations and are within current regulatory guidelines. COL applicants need to update the analysis to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information).

## 15.6.6 Cleanup Water Line Break—Outside Containment

To evaluate liquid process line pipe breaks outside containment, the failure of a cleanup water line is assumed to evaluate the response of the plant design to this postulated event. The postulated break of the cleanup water line, representing the most significant liquid line outside containment, provides the envelope evaluation for this type of break. The break is assumed to be instantaneous, circumferential and downstream of the outermost isolation valve.

A more limiting event from a core performance evaluation standpoint (Feedwater Line Break—Inside Containment) has been quantitatively analyzed and is presented in Section 6.3. Therefore, the following discussion provides only new information not presented in Section 6.3. All other information is cross-referenced to appropriate Chapter 6 subsections.

### 15.6.6.1 Identification of Causes and Frequency Classification

#### 15.6.6.1.1 Identification of Causes

A cleanup water line break is assumed without the cause being identified. The subject piping is designed to high quality, to strict engineering codes and standards, and to seismic environmental requirements.

#### 15.6.6.1.2 Frequency Classification

This event is categorized as a limiting fault (liquid line break).

### 15.6.6.2 Sequence of Events and System Operation

#### 15.6.6.2.1 Sequence of Events

The sequence of events is presented in Table 15.6-15.

#### 15.6.6.2.2 Identification of Operator Actions

Because automatic actuation and operation of the ECCS is a system design basis, no operator actions are required. However, the operator should perform the following (shown for informational purposes only):

(1) determine that a line break has occurred

(2) ensure that if vessel water level is below level 3 that reactor has scrammed,

(3) monitor vessel water level and ensure actuation of ECCS as needed, and

(4) implement site radiation incident procedures.

These actions occur over an elapsed time of 3–4 hours.

#### 15.6.6.2.3 Systems Operation

It is assumed that the normally operating plant instrument and controls are functioning. Credit is taken for the actuation of the reactor isolation system and ECCS. The reactor protection system (safety/relief valves, ECCS, and control rod drive) and plant protection system (RHR heat exchangers) are assumed to function properly to assure a safe shutdown.

The ESF Systems and HPCF System are assumed to operate normally. RCIC will automatically isolate to high RCIC room temperature caused by steam escaping from the break prior to closing the isolation valves.

#### 15.6.6.2.4 The Effect of Single Failures and Operator Errors

The cleanup water line outside the containment is a special case of the general LOCA break spectrum presented in detail in Section 6.3. The general single-failure analysis for LOCAs is presented in detail in Subsection 6.3.3.3. For the cleanup water line break outside the containment, because the break is isolatable, either the RCIC System or one of the HPCF Systems provides adequate flow to the vessel to maintain core cooling and prevent fuel rod clad failure. A single failure of either one of the HPCF Systems or the RCIC System still provides sufficient flow to keep the core covered with water (see Section 6.3 and Appendix 15A for analysis details).

### 15.6.6.3 Core and System Performance

#### 15.6.6.3.1 Qualitative Summary

The accident evaluation qualitatively considered in this subsection is considered to be a conservative and envelope assessment of the consequences of the postulated failure (i.e., severance) of one the feedwater piping lines external to the containment. The accident is postulated to occur at the input parameters and initial conditions presented in Table 6.3-1.

#### 15.6.6.3.2 Qualitative Results

The cleanup water line break outside the containment is less limiting than either of the steamline breaks outside the containment (analysis presented in Sections 6.3 and/or 15.6.4), or the feedwater line break inside the containment (analysis presented in Subsections 6.3.3 and 15.6.5).

The reactor vessel is isolated on water level L1.5, and the RCIC and the HPCF Systems together restore the reactor water level to the normal elevation if needed. The fuel is covered throughout the transient and there are no pressure or temperature transients sufficient to cause fuel damage.

#### 15.6.6.3.3 Consideration of Uncertainties

This event was conservatively analyzed and uncertainties were adequately considered (see Section 6.3 for details).

### 15.6.6.4 Barrier Performance

Accidents that result in the release of radioactive materials outside the containment are the result of postulated breaches in the RCPB or the steam power-conversion system boundary. A break spectrum analysis for the complete range of reactor conditions indicates that the limiting fault event for breaks outside the containment is a complete severance of one of the main steamlines as presented in Subsection 15.6.4. The cleanup water system piping break is less severe than the main steamline break. Results of analysis of this event can be found in Subsections 6.2.3 or 6.2.4.

### 15.6.6.5 Radiological Consequences

#### 15.6.6.5.1 Design Basis Analysis

The NRC provides no specific regulatory guidelines for the evaluation of this accident; therefore, the analysis presented is based upon conservative assumptions considered acceptable to the NRC.

#### 15.6.6.5.2 Analysis

The analysis is based on a conservative assessment of this accident. The specific models, assumptions and the program used for computer evaluation are presented in Reference 15.6-2. Specific values of parameters used in the evaluation are presented in Table 15.6-16. A schematic diagram of the leakage path for this accident is shown in Figure 15.6-5.

##### 15.6.6.5.2.1 Fission Product Release

There is no fuel damage as a consequence of this accident.

At the initiation of this accident it is assumed that the total non-filtered inventory in both the regenerative and non-regenerative heat exchangers is released through the break. Inventory in the demineralizer is prevented from being released by back flow check valves from exiting that component. A break on the downstream side of the demineralizer would be bounded due to the demineralizer action compared to a break on the upstream side of the demineralizer.

Isolation of the CUW line is conservatively analyzed based upon actuation of the flow differential pressure instrumentation. This instrumentation has a built in 45 second time delay so that for the initial 45 seconds of the accident full flow through the CUW line subject to flow restriction by a 140 $cm^2$ flow restrictor located in the primary containment. After the initial 45 second flow, motor operated isolation valves will close over a period of 30 seconds. During this period of 75 seconds, flow of reactor water is assumed at the maximum equilibrium reactor water concentration given in Subsection 15.6.4.5.1.1, case 1, with flashing to steam at reactor temperature and pressure. In addition, iodine spiking based upon a differential reactor depressurization from 7.24 MPa to 6.69 MPa in 20 seconds and using the spiking source terms given in Table 15.6-16 is assumed. Noble gas activity in the reactor coolant is negligible and is therefore ignored in this analysis.

#### 15.6.6.5.2.2 Fission Product Transport to the Environment

The transport pathway consists of a tortuous path from the lowest levels of the reactor building through designed rupture disks to the pipe chases terminating with flow directed into the main steam tunnel. The main steam tunnel incorporates an over pressurization flow chimney which will route the flow finally to turbine building upper deck. Flow ejected into this area will most probably be entrained into the turbine building HVAC and directed to the plant stack. However credit for this pathway to the stack is not assumed and releases to the turbine building are considered environmental releases out turbine building doors. Because the release pathway experiences significant surface areas in the reactor building, steam tunnel and turbine building, a credit for iodine plateout of 0.5 is assumed.

#### 15.6.6.5.2.3 Results

The calculated exposures for the analysis are presented in Table 15.6-18 and are a small fraction of 10CFR100 guidelines. COL applicants need to update the calculations to conform to the as-designed plant and site-specific parameters (see Subsection 15.6.7.2 for COL license information.).

## 15.6.7 COL License Information

### 15.6.7.1 Iodine Removal Credit

The COL applicant will recalculate iodine removal credit as outlined in Subsections 15.6.5.5.1.2 and 15.6.5.5.1.3.

## 15.6.8 References

15.6-1 F.J. Moody, “Maximum Two-Phase Vessel Blowdown from Pipes”, ASME Paper Number 65-WA/HT-1, March 15, 1965.

15.6-2 H.A. Careway, V.D. Nguyen, and P.P. Stancavage, “Radiological Accident Evaluation - The CONAC03 Code”, December 1981 (NEDO-21143-1).

15.6-3 K.G. Murphy, and K.M. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19”, 13th ASC Air Cleaning Conference, June 1974.

15.6-4 L.S. Lee, “BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems”, February 1991 (NEDC-31858P).

15.6-5 J.V. Ramsdell, “Atmospheric Diffusion for Control Room Habitability Assessments”, May 1988 (NUREG/CR-5055).

15.6-6 Ramsdell, J.V., “Alternatives to Current Procedures Used to Estimate Concentrations in Building Wakes”, 21st DOE/NRC Nuclear Air Cleaning Conference, pgs 714-729.

## Table 15.6-1 Instrument Line Break Accident Parameters

| | |
|---|---|
| **I** Data and assumptions used to estimate source terms | |
| A. Power level | 4005 MWt |
| B. Mass of fluid released | 13610 kg |
| C. Mass of fluid flashed to steam | 2270 kg |
| D. Duration of accident | 8 h |
| E. Number of bundles in core | 872 |
| **II** Data and assumptions used to estimate activity released | |
| A. Iodine water concentration | 15.6.4.5.1.1, case 1 |
| B. Iodine Spiking (MBq/bundle) | |
| I-131 | 7.77E+04 |
| I-132 | 1.18E+05 |
| I-133 | 1.85E+05 |
| I-134 | 2.00E+05 |
| I-135 | 1.78E+05 |
| C. Iodine plateout fraction | 50% |
| D. Reactor Building Flow rate | 200%/h |
| E. SGTS Filter Efficiency | None assumed |
| **III** Dispersion and Dose Data | |
| A. Meteorology | Table 15.6-3 |
| B. Boundary and LPZ distances | Table 15.6-3 |
| C. Method of Dose Calculation | Reference 15.6-2 |
| D. Dose conversion assumptions | Reference 15.6-2, RG 1.109, and ICRP 30 |
| E. Activity Inventory/releases | Table 15.6-2 |
| F. Dose Evaluations | Table 15.6-3 |

## Table 15.6-2 Instrument Line Break Accident Isotopic Inventory

| Reactor Building Inventory (Megabecquerel) | | | | | | |
|---|---|---|---|---|---|---|
| **Isotope** | **1- min** | **10-min** | **1-hour** | **2-hour** | **4-hour** | **8-hour** |
| I-131 | 3.77E+01 | 3.27E+02 | 2.60E+04 | 1.73E+04 | 1.38E+04 | 4.59E+00 |
| I-132 | 3.68E+02 | 3.11E+03 | 2.31E+05 | 1.44E+05 | 1.17E+05 | 1.17E+01 |
| I-133 | 2.59E+02 | 2.24E+03 | 1.75E+05 | 1.16E+05 | 9.29E+04 | 2.72E+01 |
| I-134 | 7.22E+02 | 5.92E+03 | 3.89E+05 | 2.26E+05 | 1.86E+05 | 2.65E+00 |
| I-135 | 3.77E+02 | 3.25E+03 | 2.52E+05 | 1.64E+05 | 1.32E+05 | 2.90E+01 |
| Total | 1.76E+03 | 1.48E+04 | 1.07E+06 | 6.68E+05 | 5.41E+05 | 7.52E+01 |
| **Isotopic Release to Environment (Megabecquerel)** | | | | | | |
| **Isotope** | **1- min** | **10-min** | **1-hour** | **2-hour** | **4-hour** | **8-hour** |
| I-131 | 6.36E–01 | 5.77E+01 | 2.77E+04 | 6.81E+04 | 1.27E+05 | 1.41E+05 |
| I-132 | 6.18E+00 | 5.51E+02 | 2.52E+05 | 5.96E+05 | 1.09E+06 | 1.19E+06 |
| I-133 | 4.37E+00 | 3.96E+02 | 1.87E+05 | 4.59E+05 | 8.51E+05 | 9.44E+05 |
| I-134 | 1.21E+01 | 1.06E+03 | 4.44E+05 | 9.92E+05 | 1.76E+06 | 1.90E+06 |
| I-135 | 6.36E+00 | 5.74E+02 | 2.71E+05 | 6.59E+05 | 1.21E+06 | 1.34E+06 |
| Total | 2.97E+01 | 2.64E+03 | 1.18E+06 | 2.77E+06 | 5.04E+06 | 5.51E+06 |

## Table 15.6-3 Instrument Line Break Accident Results

| Meteorology* and Dose Results | | | |
|---|---|---|---|
| **Meteorology ($s/m^3$)** | **Distance (m)** | **Thyroid Dose (Sv)** | **Whole Body Dose (Sv)** |
| 8.59E–03 | max | 3.0E–01 | 6.0E–03 |
| 1.37E-03 | Chp 2 | 4.8E–02 | 9.4E–04 |
| 2.19E–04 | 800 | 7.6E–03 | 1.5E–04 |
| 1.11E–04 | 1600 | 3.9E–03 | 7.9E–05 |
| 5.61E–05 | 3200 | 2.0E–03 | 4.0E–05 |
| 3.73E–05 | 4800 | 1.3E–03 | 2.6E–05 |

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

## Table 15.6-4 Sequence of Events for Steamline Break Outside Containment

| Time (s) | Event |
|---|---|
| 0 | Guillotine break of one main steamline outside primary containment. |
| ~0.5 | High steamline flow signal initiates closure of main steamline isolation valve |
| <1.0 | Reactor begins scram. |
| ≤5.0 | Main steamline isolation valves fully closed. |
| 38 | Safety/relief valves open on high vessel pressure. The valves open and close to maintain vessel pressure at approximately 7.58 MPa. |
| 30 | RCIC initiates on vessel low-water Level 2. |
| 50 | RCIC begins injection. |
| 199 | HPCF initiates on low water level. |
| 236 | One HPCF begins injection (the other HPCF is unavailable due to the single failure assumption). |
| 1–2 hours | Normal reactor cooldown procedure established. |

## Table 15.6-5 Steamline Break Accident Parameters

| | | |
|---|---|---|
| **I** | **Data and assumptions used to estimate source terms.** | |
| | A. Power Level | 4005 MWt |
| | B. Fuel damage | none |
| | C. Reactor coolant activity | Subsection 15.6.4.5 |
| | D. Steam mass released | 12,870 kg |
| | E. Water mass released | 21,953 kg |
| **II** | **Data and assumptions used to estimate activity released** | |
| | A. MSIV closure time (break time until fully closed) | 5.0 s |
| | B. Maximum release time | 2 h |
| **III** | **Dispersion and Dose Data** | |
| | A. Meteorology | Table 15.6-7 |
| | B. Boundary and LPZ distances | Table 15.6-7 |
| | C. Method of Dose Calculation | Reference 15.6-2 |
| | D. Dose conversion Assumptions | Reference 15.6-2, RG 1.109, and ICRP 30 |
| | E. Activity Inventory/release | Table 15.6-6 |
| | F. Dose Evaluations | Table 15.6-7 |

## Table 15.6-6 Main Steamline Break Accident Activity Released to Environment (megabecquerel)

| Isotope | Case 1 | Case 2 |
|---|---|---|
| I-131 | 7.29E+04 | 1.46E+06 |
| I-132 | 7.10E+05 | 1.42E+07 |
| I-133 | 5.00E+05 | 9.99E+06 |
| I-134 | 1.40E+06 | 2.79E+07 |
| I-135 | 7.29E+05 | 1.46E+07 |
| | | |
| Total Halogens | 3.41E+06 | 6.81E+07 |
| | | |
| KR-83M | 4.07E+02 | 2.44E+03 |
| KR-85M | 7.18E+02 | 4.29E+03 |
| KR-85 | 2.26E+00 | 1.36E+01 |
| KR-87 | 2.44E+03 | 1.47E+04 |
| KR-88 | 2.46E+03 | 1.48E+04 |
| KR-89 | 9.88E+03 | 5.92E+04 |
| KR-90 | 2.55E+03 | 1.55E+04 |
| XE-131M | 1.76E+00 | 1.06E+01 |
| XE-133M | 3.39E+01 | 2.04E+02 |
| XE-133 | 9.47E+02 | 5.70E+03 |
| XE-135M | 2.89E+03 | 1.74E+04 |
| XE-135 | 2.70E+03 | 1.62E+04 |
| XE-137 | 1.23E+04 | 7.40E+04 |
| XE-138 | 9.44E+03 | 5.66E+04 |
| XE-139 | 4.33E+03 | 2.59E+04 |
| | | |
| Total Noble Gases | 5.11E+04 | 3.07E+05 |

## Table 15.6-7 Main Steamline Break Meteorology* Parameters and Radiological Effects

| Meteorology ($s/m^3$) | Distance (m) | Thyroid Dose (Sv) | Whole Body Dose (Sv) |
|---|---|---|---|
| Case 1 | | | |
| 8.12E–03 | max | 3.0E–01 | 7.4E–03 |
| 1.37E–03 | Chp 2 | 2.6E–02 | 6.2E–04 |
| 1.18E–03 | 300 | 2.2E–02 | 5.4E–04 |
| 2.19E–04 | 800 | 4.1E–03 | 1.0E–04 |
| Case 2 | | | |
| 8.12E–03 | max | 3.0E+00 | 7.4E–02 |
| 1.37E–03 | Chp 2 | 5.1E–01 | 1.3E–02 |
| 1.18E–03 | 300 | 4.4E–01 | 1.1E–02 |
| 2.19E–04 | 800 | 8.1E–02 | 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 10% of 10CFR100 limits.

## Table 15.6-8 Loss of Coolant Accident Parameters

| | | |
|---|---|---|
| **I** | **Data and assumptions used to estimate source terms.** | |
| | A. Power Level | 4005 MWt |
| | B. Fraction of Core Inventory Released | |
| | Noble Gases | 100% |
| | Iodines | 50% |
| | C. Iodine Initial Plateout Fraction | 50% |
| | D. Iodine Chemical Species | |
| | Elemental | 91% |
| | Particulate | 5% |
| | Organic | 4% |
| | E. Suppression Pool Decontamination Factor–Section 15.6.5.5.1.1 | |
| | Noble Gas | 1 |
| | Organic Iodine | 1 |
| | Elemental Iodine | 2 |
| | Particulate | 2 |
| | Pool Bypass Area | 46.5 $cm^2$ |
| **II** | **Data and Assumptions used to estimate activity released.** | |
| | A. Primary Containment Leakage | |
| | (1) Penetration and ESF Equipment | 0.5%/day |
| | (2) MSIV Leakage (Total all lines) | 66.1 L/min |
| | B. Reactor Building Leakage | 150%/h |
| | (1) 0–20 min | 150%/h |
| | (2) >20 min | 50%/d |
| | (3) Mixing Efficiency | 50% |
| | C. SGTS | |
| | Filter Efficiency (15.2 cm) | 97% |
| | Drawdown Time | 20 min |
| | D. MSIV Leakage—see Reference 15.6-4 for standard parameters | |
| | Main Steamline Length | 47.9 m |
| | Drain Line length | 71.6 m |
| | Main Steamline IR/OR | 31.98/35.55 cm |
| | Drain Line IR/OR | 3.33/4.45 cm |
| | Main Steamline Insulation | 12.0 cm |
| | Drain Line Insulation | 6.5 cm |
| | Plateout and Resuspension Factors | Ref. 15.6-5 |

## Table 15.6-8 Loss of Coolant Accident Parameters (Continued)

| | | |
|---|---|---|
| | E. Condenser data | |
| | Free Air Volume | 6230 $m^3$ |
| | Fraction of Volume involved | 20% |
| | Leakage Rate | 11.6%/d |
| | Iodine Removal Factor | |
| | Elemental | 0.993 |
| | Particulate | 0.993 |
| | Organic | 0 |
| **III** | **Control Room Data** | |
| | A. Control Room Volumes | |
| | Total Free Air Volume | 5,509 $m^3$ |
| | Gamma Room Volume (room size) | 1,400 $m^3$ |
| | B. Recirculation Rates | |
| | Filtered Intake | 0.944 $m^3/s$ |
| | Unfiltered Intake | 0.0 |
| | Filtered Recirculation | 0.47 $m^3/s$ |
| | Filter Efficiency (100 mm) | 99% |
| **IV** | **Dispersion and Dose Data** | |
| | A. Meteorology | Sec 15.6.5.5.3<br>Tbls 15.6-13, 15.6-14 |
| | B. Dose Calculation Method (semi-infinite) | Ref 15.6-2 & 15.6-3, RG 1.109 |
| | C. Dose Conversion Assumptions | Ref 15.6-2, 15.6-3 |
| | D. Activity/Releases | Tbls 15.6-9, 15.6-10, 15.6-11, 15.6-12<br>Appendix 15F |
| | E. Dose Evaluation | Tbls 15.6-13,15.6-14 |

## Table 15.6-9 Iodine Activities

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| A. Primary Containment Inventory (megabecquerel) | | | | | | | | | | |
| I-131 | 5.2E+11 | 5.2E+11 | 5.2E+11 | 5.2E+11 | 4.8E+11 | 4.8E+11 | 4.8E+11 | 4.4E+11 | 3.4E+11 | 2.7E+10 |
| I-132 | 7.4E+11 | 7.0E+11 | 5.6E+11 | 4.1E+11 | 2.2E+12 | 6.7E+10 | 1.9E+10 | 5.2E+8 | 1.6E+11 | 0 |
| I-133 | 1.1E+12 | 1.0E+12 | 1.0E+12 | 1.0E+12 | 9.3E+11 | 8.1E+11 | 7.0E+11 | 4.8E+11 | 4.1E+10 | 2.8E+1 |
| I-134 | 1.1E+12 | 1.0E+12 | 5.2E+11 | 2.4E+11 | 4.8E+10 | 2.1E+9 | 8.9E+7 | 6.7E+3 | 0 | 0 |
| I-135 | 1.0E+12 | 1.0E+12 | 8.9E+11 | 8.1E+11 | 6.7E+11 | 4.4E+11 | 2.8E+11 | 7.8E+10 | 4.1E+7 | 0 |
| Total | 4.5E+12 | 4.3E+12 | 3.5E+12 | 3.0E+12 | 4.3E+12 | 1.8E+12 | 1.5E+12 | 1.0E+12 | 5.4E+11 | 2.7E+10 |
| B. Reactor Building Inventory (megabecquerel) | | | | | | | | | | |
| I-131 | 1.7E+6 | 1.5E+7 | 9.3E+7 | 1.9E+8 | 3.7E+8 | 7.0E+8 | 9.6E+8 | 1.4E+9 | 1.7E+9 | 1.3E+8 |
| I-132 | 2.5E+6 | 2.1E+7 | 1.0E+8 | 1.6E+8 | 1.7E+8 | 9.3E+7 | 3.7E+7 | 1.6E+6 | 7.8E–4 | 0 |
| I-133 | 3.6E+6 | 3.1E+7 | 1.9E+8 | 3.7E+8 | 7.0E+8 | 1.1E+9 | 1.4E+9 | 1.5E+9 | 2.0E+8 | 1.4E–1 |
| I-134 | 4.1E+6 | 3.0E+7 | 1.0E+8 | 9.3E+7 | 3.7E+7 | 2.9E+6 | 1.7E+5 | 2.1E+1 | 0 | 0 |
| I-135 | 3.4E+6 | 2.9E+7 | 1.7E+8 | 3.1E+8 | 4.8E+8 | 5.9E+8 | 5.6E+8 | 2.5E+8 | 1.9E+5 | 0 |
| Total | 1.5E+7 | 1.3E+8 | 6.5E+8 | 1.1E+9 | 1.8E+9 | 2.5E+9 | 2.9E+9 | 3.2E+9 | 1.9E+9 | 1.3E+8 |
| C.1 MSIV Pathway—Condenser Inventory (megabecquerel)—Elemental Iodine | | | | | | | | | | |
| I-131 | 0 | 0 | 7.8E+6 | 4.8E+7 | 2.0E+8 | 6.3E+8 | 1.1E+9 | 2.4E+9 | 4.1E+9 | 3.1E+7 |
| I-132 | 0 | 0 | 8.5E+6 | 4.1E+7 | 8.9E+7 | 8.5E+7 | 4.4E+7 | 2.6E+6 | 1.9E–3 | 0 |
| I-133 | 0 | 0 | 1.6E+7 | 9.6E+7 | 3.7E+8 | 1.0E+0 | 1.6E+9 | 2.4E+9 | 4.8E+8 | 3.3E–2 |
| I-134 | 0 | 0 | 8.1E+6 | 2.4E+7 | 2.0E+7 | 2.7E+6 | 2.0E+5 | 3.5E+1 | 0 | 0 |
| I-135 | 0 | 0 | 1.4E+7 | 8.1E+7 | 2.7E+8 | 5.6E+8 | 6.3E+8 | 4.1E+8 | 4.8E+5 | 0 |
| Total | 0 | 0 | 5.4E+7 | 2.9E+8 | 9.5E+8 | 2.3E+9 | 3.4E+9 | 5.2E+9 | 4.6E+9 | 3.1E+7 |
| C.2 MSIV Pathway—Condenser Inventory (megabecquerel)—Organic Iodine (Primary Containment) | | | | | | | | | | |
| I-131 | 0 | 0 | 6.7E+5 | 4.1E+6 | 1.7E+7 | 5.6E+7 | 9.3E+7 | 2.1E+8 | 5.9E+8 | 1.3E+8 |
| I-132 | 0 | 0 | 7.0E+5 | 3.4E+6 | 7.8E+6 | 7.0E+6 | 3.7E+6 | 2.3E+5 | 2.8E–4 | 0 |
| I-133 | 0 | 0 | 1.4E+6 | 8.1E+6 | 3.2E+7 | 8.9E+7 | 1.4E+8 | 2.1E+10 | 7.0E+7 | 1.4E–1 |
| I-134 | 0 | 0 | 7.0E+5 | 2.0E+6 | 1.7E+6 | 2.3E+5 | 1.7E+4 | 3.0E+0 | 0 | 0 |
| I-135 | 0 | 0 | 1.2E+6 | 6.7E+6 | 2.3E+7 | 4.8E+7 | 5.6E+7 | 3.6E+8 | 6.7E+4 | 0 |
| Total | 0 | 0 | 4.6E+6 | 2.4E+7 | 8.1E+7 | 2.0E+8 | 2.9E+8 | 2.2E+10 | 6.6E+8 | 1.3E+8 |

## Table 15.6-9  Iodine Activities (Continued)

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| C.3 MSIV Pathway—Condenser Inventory in Curies—Resuspended Organic | | | | | | | | | | |
| I-131 | 0 | 0 | 2.8E+3 | 5.6E+3 | 3.4E+4 | 8.9E+4 | 2.7E+5 | 9.3E+5 | 4.8E+7 | 9.6E+7 |
| I-132 | 0 | 0 | 2.2E+3 | 3.7E+3 | 8.5E+3 | 1.2E+4 | 5.9E+3 | 1.7E+3 | 0 | 0 |
| I-133 | 0 | 0 | 5.6E+3 | 1.1E+4 | 5.9E+4 | 1.5E+5 | 3.6E+5 | 9.3E+5 | 5.9E+6 | 5.2E–1 |
| I-134 | 0 | 0 | 1.4E+3 | 2.1E+3 | 1.3E+3 | 1.0E+3 | 5.6E+1 | 1.2E+0 | 0 | 0 |
| I-135 | 0 | 0 | 4.4E+3 | 8.5E+3 | 3.6E+4 | 7.4E+4 | 1.1E+5 | 1.6E+5 | 9.6E+3 | 0 |
| Total | 0 | 0 | 1.6E+4 | 3.1E+4 | 1.4E+5 | 3.2E+5 | 7.5E+5 | 2.0E+6 | 5.4E+7 | 9.6E+7 |
| C.4 Condenser Inventory (megabecquerel)—Combined | | | | | | | | | | |
| I-131 | 0 | 0 | 8.5E+6 | 5.6E+7 | 2.2E+8 | 7.0E+8 | 1.2E+9 | 2.6E+9 | 4.8E+9 | 2.6E+8 |
| I-132 | 0 | 0 | 9.3E+6 | 4.4E+7 | 9.6E+7 | 9.3E+7 | 4.8E+7 | 2.8E+6 | 2.2E–3 | 0 |
| I-133 | 0 | 0 | 1.7E+7 | 1.1E+8 | 4.1E+8 | 1.1E+9 | 1.7E+9 | 2.7E+9 | 5.6E+8 | 7.0E–1 |
| I-134 | 0 | 0 | 8.9E+6 | 2.6E+7 | 2.2E+7 | 2.9E+6 | 2.2E+5 | 4.1E+1 | 0 | 0 |
| I-135 | 0 | 0 | 1.5E+7 | 8.5E+7 | 2.9E+8 | 5.9E+8 | 7.0E+8 | 4.4E+8 | 5.6E+5 | 0 |
| Total | 0 | 0 | 5.9E+7 | 3.2E+8 | 1.0E+9 | 2.5E+9 | 3.7E+9 | 5.7E+9 | 5.4E+9 | 2.6E+8 |
| D.1 Control Room Inventory (megabecquerel) | | | | | | | | | | |
| I-131 | 8.4E-1 | 7.3E+1 | 1.4E+2 | 5.8E+1 | 2.0E+1 | 2.6E+1 | 2.2E+1 | 3.6E+1 | 3.2E+1 | 1.9E+0 |
| I-132 | 1.2E+0 | 1.0E+2 | 1.5E+2 | 4.6E+1 | 8.7E+0 | 3.4E+0 | 8.9E–1 | 4.0E–2 | 0 | 0 |
| I-133 | 1.8E+0 | 1.5E+2 | 2.8E+2 | 1.1E+2 | 3.7E+1 | 4.2E+1 | 3.2E+1 | 3.7E+1 | 3.8E+0 | 3.8E-9 |
| I-134 | 1.9E+0 | 1.5E+2 | 1.4E+2 | 2.8E+1 | 2.0E+0 | 1.1E–1 | 4.0E–3 | 5.4E–7 | 0 | 0 |
| I-135 | 1.7E+0 | 1.4E+2 | 2.4E+2 | 9.3E+1 | 2.6E+1 | 2.2E+1 | 1.3E+1 | 6.3E+0 | 3.6E–3 | 0 |
| Total | 7.4E+0 | 6.2E+2 | 9.5E+2 | 3.4E+2 | 9.3E+1 | 9.3E+1 | 6.8E+1 | 8.0E+1 | 3.5E+1 | 1.9E+0 |
| D.2 Control Room Integrated Activity (megabecquerel–seconds) | | | | | | | | | | |
| I-131 | 1.7E+1 | 1.5E+4 | 5.4E+5 | 3.3E+5 | 2.3E+5 | 3.0E+5 | 3.1E+5 | 1.3E+6 | 8.0E+6 | 1.5E+7 |
| I-132 | 2.5E+1 | 2.1E+4 | 6.7E+5 | 3.1E+5 | 1.5E+5 | 7.4E+4 | 2.4E+4 | 1.2E+4 | 0 | 0 |
| I-133 | 3.5E+1 | 3.2E+4 | 1.1E+6 | 6.5E+5 | 4.4E+5 | 5.2E+5 | 4.8E+5 | 1.6E+6 | 3.2E+6 | 2.0E+5 |
| I-134 | 3.9E+1 | 3.2E+4 | 8.1E+5 | 2.5E+5 | 6.4E+4 | 8.4E+3 | 4.1E+2 | 2.0E+1 | 0 | 0 |
| I-135 | 3.3E+1 | 3.0E+4 | 1.0E+6 | 5.6E+5 | 3.4E+5 | 3.2E+5 | 2.2E+5 | 4.1E+5 | 1.7E+5 | 0 |
| Total | 1.5E+2 | 1.3E+5 | 4.1E+6 | 2.1E+6 | 1.2E+6 | 1.2E+6 | 1.0E+6 | 3.3E+6 | 1.1E+7 | 1.5E+7 |

## Table 15.6-10 Iodine Activity Release to the Environment

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| A. Release from Reactor Building to Environment (megabecquerel) | | | | | | | | | | |
| I-131 | 2.9E+4 | 2.6E+6 | 9.6E+6 | 9.6E+6 | 1.0E+7 | 1.3E+7 | 1.7E+7 | 3.6E+7 | 1.9E+8 | 6.7E+8 |
| I-132 | 4.1E+4 | 3.7E+6 | 1.3E+7 | 1.3E+7 | 1.4E+7 | 1.4E+7 | 1.5E+7 | 1.5E+7 | 1.5E+7 | 1.5E+7 |
| I-133 | 5.9E+4 | 5.6E+6 | 2.0E+7 | 2.0E+7 | 2.1E+7 | 2.6E+7 | 3.3E+7 | 5.6E+7 | 1.2E+8 | 1.3E+8 |
| I-134 | 6.7E+4 | 5.6E+6 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 |
| I-135 | 5.6E+4 | 5.2E+6 | 1.9E+7 | 1.9E+7 | 2.0E+7 | 2.3E+7 | 2.6E+7 | 3.1E+7 | 3.5E+7 | 3.5E+7 |
| Total | 2.5E+5 | 2.3E+7 | 8.0E+7 | 8.1E+7 | 8.4E+7 | 9.5E+7 | 1.1E+8 | 1.6E+8 | 7.0E+8 | 8.6E+8 |
| B.1 MSIV Pathway Release to Environment—Elemental (megabecquerel) | | | | | | | | | | |
| I-131 | 0 | 0 | 5.6E+1 | 9.3E+2 | 9.3E+3 | 6.3E+4 | 1.8E+5 | 8.9E+5 | 1.0E+7 | 3.0E+7 |
| I-132 | 0 | 0 | 6.3E+1 | 8.5E+2 | 5.6E+3 | 1.8E+4 | 2.7E+4 | 3.3E+4 | 3.4E+4 | 3.4E+4 |
| I-133 | 0 | 0 | 1.1E+2 | 1.9E+3 | 1.7E+4 | 1.1E+5 | 3.0E+5 | 1.1E+6 | 4.8E+6 | 5.6E+6 |
| I-134 | 0 | 0 | 6.3E+1 | 6.3E+2 | 2.3E+3 | 3.6E+3 | 3.7E+3 | 3.7E+3 | 1.3E+2 | 3.7E+3 |
| I-135 | 0 | 0 | 1.0E+2 | 1.6E+3 | 1.3E+4 | 7.0E+4 | 1.6E+5 | 3.7E+5 | 5.6E+5 | 5.6E+5 |
| Total | 0 | 0 | 3.9E+2 | 5.9E+3 | 4.8E+4 | 2.7E+5 | 6.6E+5 | 2.4E+6 | 1.6E+7 | 3.6E+7 |
| B.2 MSIV Pathway Release to Environment—Organic (megabecquerel) | | | | | | | | | | |
| I-131 | 0 | 0 | 6.7E+2 | 1.1E+4 | 1.1E+5 | 7.8E+5 | 2.2E+6 | 1.1E+7 | 1.6E+8 | 1.4E+9 |
| I-132 | 0 | 0 | 7.4E+2 | 1.0E+4 | 6.7E+4 | 2.2E+5 | 3.3E+5 | 4.1E+5 | 4.1E+5 | 4.1E+5 |
| I-133 | 0 | 0 | 1.3E+3 | 2.3E+4 | 2.1E+5 | 1.4E+6 | 3.6E+6 | 1.4E+7 | 7.0E+7 | 8.1E+7 |
| I-134 | 0 | 0 | 7.8E+2 | 7.8E+3 | 2.8E+4 | 4.4E+4 | 4.4E+4 | 4.4E+4 | 4.4E+4 | 4.4E+4 |
| I-135 | 0 | 0 | 1.2E+3 | 1.9E+4 | 1.6E+5 | 8.5E+5 | 1.9E+6 | 4.4E+6 | 7.0E+6 | 7.0E+6 |
| Total | 0 | 0 | 4.7E+3 | 7.1E+4 | 5.8E+5 | 3.3E+6 | 8.0E+6 | 3.0E+7 | 2.4E+8 | 1.5E+9 |
| B.3 MSIV Pathway Release to Environment—Resuspended Organic (megabecquerel) | | | | | | | | | | |
| I-131 | 0 | 0 | 6.7E+0 | 2.7E+1 | 2.2E+2 | 1.4E+3 | 4.8E+3 | 3.7E+4 | 6.3E+6 | 5.2E+8 |
| I-132 | 0 | 0 | 5.6E+0 | 2.1E+1 | 8.1E+1 | 2.9E+2 | 4.4E+2 | 6.7E+2 | 7.0E+2 | 7.0E+2 |
| I-133 | 0 | 0 | 1.3E+1 | 5.2E+1 | 4.1E+2 | 2.4E+3 | 7.4E+3 | 4.4E+4 | 1.4E+6 | 3.3E+6 |
| I-134 | 0 | 0 | 4.1E+0 | 1.3E+1 | 2.8E+1 | 4.8E+1 | 5.6E+1 | 5.6E+1 | 5.6E+1 | 5.6E+1 |
| I-135 | 0 | 0 | 1.1E+1 | 4.1E+1 | 2.6E+2 | 1.4E+3 | 3.2E+3 | 1.1E+4 | 3.4E+4 | 3.5E+4 |
| Total | 0 | 0 | 4.1E+1 | 1.5E+2 | 1.0E+3 | 5.5E+3 | 1.6E+4 | 9.4E+4 | 7.7E+6 | 5.2E+8 |

## Table 15.6-10 Iodine Activity Release to the Environment (Continued)

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| B.4 Release from Condenser to Environment—Sum of B.1+B.2+B.3 (megabecquerel) | | | | | | | | | | |
| I-131 | 0 | 0 | 7.0E+2 | 1.2E+4 | 1.2E+5 | 8.5E+5 | 2.4E+6 | 1.2E+7 | 1.8E+8 | 2.0E+9 |
| I-132 | 0 | 0 | 8.1E+2 | 1.1E+4 | 7.0E+4 | 2.4E+5 | 3.5E+5 | 4.4E+5 | 4.4E+5 | 4.4E+5 |
| I-133 | 0 | 0 | 1.5E+3 | 2.4E+4 | 2.3E+5 | 1.5E+6 | 3.7E+6 | 1.5E+7 | 7.4E+7 | 8.9E+7 |
| I-134 | 0 | 0 | 8.5E+2 | 8.5E+3 | 3.0E+4 | 4.8E+4 | 4.8E+4 | 4.8E+4 | 4.8E+4 | 4.8E+4 |
| I-135 | 0 | 0 | 1.3E+3 | 2.1E+4 | 1.7E+5 | 9.3E+5 | 2.0E+6 | 5.2E+6 | 7.4E+6 | 7.4E+6 |
| Total | 0 | 0 | 5.1E+3 | 7.7E+4 | 6.2E+5 | 3.5E+6 | 8.5E+6 | 3.3E+7 | 2.6E+8 | 2.1E+9 |

## Table 15.6-11 Noble Gas Activities

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| A. Primary Containment Inventory (megabecquerel) | | | | | | | | | | |
| Kr-83m | 4.4E+11 | 4.4E+11 | 3.2E+11 | 2.2E+11 | 1.0E+11 | 2.3E+10 | 5.2E+9 | 5.9E+7 | 1.3E–4 | 0 |
| Kr-85 | 4.4E+10 | 4.4E+10 | 4.4E+10 | 4.4E+10 | 4.4E+10 | 4.4E+10 | 4.4E+10 | 4.4E+10 | 4.1E+10 | 3.1E+10 |
| Kr-85m | 1.0E+12 | 9.6E+11 | 8.5E+11 | 7.4E+11 | 5.2E+11 | 2.8E+11 | 1.5E+11 | 2.2E+10 | 2.4E+5 | 0 |
| Kr-87 | 1.9E+12 | 1.7E+12 | 1.1E+12 | 6.3E+11 | 2.1E+11 | 2.4E+10 | 2.7E+9 | 3.7E+6 | 0 | 0 |
| Kr-88 | 2.7E+12 | 2.6E+12 | 2.1E+12 | 1.7E+12 | 1.0E+12 | 3.7E+11 | 1.4E+11 | 7.0E+9 | 1.2E+2 | 0 |
| Kr-89 | 2.7E+12 | 3.7E+11 | 7.0E+6 | 1.4E+1 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-131m | 2.3E+10 | 2.3E+10 | 2.3E+10 | 2.3E+10 | 2.3E+10 | 2.3E+10 | 2.3E+10 | 2.2E+10 | 1.8E+10 | 2.9E+9 |
| Xe-133 | 8.1E+12 | 8.1E+12 | 8.1E+12 | 8.1E+12 | 8.1E+12 | 7.8E+12 | 7.8E+12 | 7.0E+12 | 4.4E+12 | 1.1E+11 |
| Xe-133m | 3.4E+11 | 3.4E+11 | 3.4E+11 | 3.3E+11 | 3.3E+11 | 3.1E+11 | 2.9E+11 | 2.5E+11 | 9.6E+10 | 2.4E+7 |
| Xe-135 | 1.1E+12 | 1.0E+12 | 1.0E+12 | 9.3E+11 | 7.8E+11 | 5.9E+11 | 4.1E+11 | 1.7E+11 | 7.0E+8 | 0 |
| Xe-135m | 1.5E+12 | 1.0E+12 | 1.1E+11 | 7.8E+9 | 3.7E+7 | 9.6E+2 | 2.4E–2 | 0 | 0 | 0 |
| Xe-137 | 5.9E+12 | 1.2E+12 | 1.3E+8 | 2.4E+3 | 8.1E–7 | 0 | 0 | 0 | 0 | 0 |
| Xe-138 | 6.7E+12 | 4.1E+12 | 3.6E+11 | 2.0E+10 | 5.6E+7 | 4.4E+2 | 3.6E–3 | 0 | 0 | 0 |
| Total | 3.2E+13 | 2.2E+13 | 1.4E+13 | 1.3E+13 | 1.1E+13 | 9.4E+12 | 8.8E+12 | 7.5E+12 | 4.6E+12 | 1.4E+11 |
| B. Reactor Building Inventory (megabecquerel) | | | | | | | | | | |
| Kr-83m | 1.6E+6 | 1.3E+7 | 5.9E+7 | 8.5E+7 | 7.8E+7 | 3.3E+7 | 1.0E+7 | 1.9E+5 | 0 | 0 |
| Kr-85 | 1.5E+5 | 1.3E+6 | 8.1E+6 | 1.7E+7 | 3.4E+7 | 6.3E+7 | 8.9E+7 | 1.4E+8 | 2.1E+8 | 1.6E+8 |
| Kr-85m | 3.4E+6 | 2.9E+7 | 1.6E+8 | 2.8E+8 | 4.1E+8 | 4.1E+8 | 2.9E+8 | 7.0E+7 | 1.2E+3 | 0 |
| Kr-87 | 6.7E+6 | 5.2E+7 | 2.1E+8 | 2.4E+8 | 1.6E+8 | 3.4E+7 | 5.2E+6 | 1.2E+4 | 0 | 0 |
| Kr-88 | 9.3E+6 | 7.8E+7 | 4.1E+8 | 6.3E+8 | 7.4E+8 | 5.2E+8 | 2.7E+8 | 2.2E+7 | 5.9E–1 | 0 |
| Kr-89 | 9.3E+6 | 1.1E+7 | 1.3E+3 | 5.6E–3 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-131m | 8.1E+4 | 7.0E+5 | 4.4E+6 | 8.9E+6 | 1.7E+7 | 3.2E+7 | 4.4E+7 | 7.0E+7 | 8.9E+7 | 1.4E+7 |
| Xe-133 | 2.8E+7 | 2.4E+8 | 1.5E+9 | 3.1E+9 | 5.9E+9 | 1.1E+10 | 1.5E+10 | 2.3E+10 | 2.3E+10 | 5.6E+8 |
| Xe-133m | 1.2E+6 | 1.0E+7 | 6.3E+7 | 1.3E+8 | 2.4E+8 | 4.4E+8 | 5.6E+8 | 7.8E+8 | 4.8E+8 | 1.2E+5 |
| Xe-135 | 3.6E+6 | 3.1E+7 | 1.8E+8 | 3.5E+8 | 5.9E+8 | 8.1E+8 | 8.5E+8 | 5.6E+8 | 3.5E+6 | 0 |
| Xe-135m | 5.2E+6 | 2.9E+7 | 2.0E+7 | 3.0E+6 | 2.9E+4 | 1.3E+0 | 4.8E–5 | 0 | 0 | 0 |
| Xe-137 | 2.0E+7 | 3.4E+7 | 2.5E+4 | 9.3E–1 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-138 | 2.2E+7 | 1.2E+8 | 6.7E+7 | 7.4E+6 | 4.1E+4 | 6.3E–1 | 0 | 0 | 0 | 0 |
| Total | 1.1E+8 | 6.5E+8 | 2.7E+9 | 4.9E+9 | 8.2E+9 | 1.3E+10 | 1.7E+10 | 2.4E+10 | 2.4E+10 | 7.2E+8 |

## Table 15.6-11 Noble Gas Activities (Continued)

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| C. Condenser Inventory (megabecquerel) | | | | | | | | | | |
| Kr-83m | 0 | 0 | 5.6E+6 | 2.4E+7 | 4.8E+7 | 3.3E+7 | 1.3E+7 | 3.5E+5 | 0 | 0 |
| Kr-85 | 0 | 0 | 7.8E+5 | 4.8E+6 | 2.0E+7 | 6.3E+7 | 1.1E+8 | 2.6E+8 | 9.6E+8 | 2.0E+9 |
| Kr-85m | 0 | 0 | 1.4E+7 | 7.8E+7 | 2.4E+8 | 4.1E+8 | 3.7E+8 | 1.3E+8 | 5.6E+3 | 0 |
| Kr-87 | 0 | 0 | 1.9E+7 | 7.0E+7 | 9.6E+7 | 3.4E+7 | 6.7E+6 | 2.2E+4 | 0 | 0 |
| Kr-88 | 0 | 0 | 3.6E+7 | 1.8E+8 | 4.4E+8 | 5.2E+8 | 3.5E+8 | 4.1E+7 | 2.7E+0 | 0 |
| Kr-89 | 0 | 0 | 1.2E+2 | 1.5E–3 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-131m | 0 | 0 | 4.1E+5 | 2.6E+6 | 1.0E+7 | 3.3E+7 | 5.6E+7 | 1.3E+8 | 4.1E+8 | 1.9E+8 |
| Xe-133 | 0 | 0 | 1.4E+8 | 8.9E+8 | 3.6E+9 | 1.1E+10 | 1.9E+10 | 4.1E+10 | 1.0E+11 | 7.4E+9 |
| Xe-133m | 0 | 0 | 5.9E+6 | 3.6E+7 | 1.4E+8 | 4.4E+8 | 7.4E+8 | 1.4E+9 | 2.1E+9 | 1.6E+6 |
| Xe-135 | 0 | 0 | 1.7E+7 | 1.0E+8 | 3.5E+8 | 8.1E+8 | 1.1E+9 | 1.0E+9 | 1.6E+7 | 1.0E–13 |
| Xe-135m | 0 | 0 | 1.9E+6 | 8.5E+5 | 1.7E+4 | 1.4E+0 | 6.3E–5 | 0 | 0 | 0 |
| Xe-137 | 0 | 0 | 2.3E+3 | 2.7E–1 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-138 | 0 | 0 | 6.3E+6 | 2.1E+6 | 2.5E+4 | 6.3E–1 | 0 | 0 | 0 | 0 |
| Total | 0 | 0.0E+0 | 2.5E+8 | 1.4E+9 | 4.9E+9 | 1.3E+10 | 2.2E+10 | 4.4E+10 | 1.1E+11 | 9.6E+9 |
| D.1 Control Room Inventory (megabecquerel) | | | | | | | | | | |
| Kr-83m | 7.7E+1 | 6.4E+3 | 1.3E+4 | 1.1E+4 | 1.0E+4 | 5.4E+3 | 1.1E+3 | 2.2E+1 | 0 | 0 |
| Kr-85 | 7.5E+0 | 6.6E+2 | 1.9E+3 | 2.2E+3 | 4.5E+3 | 1.0E+4 | 9.6E+3 | 1.6E+4 | 1.7E+4 | 6.4E+3 |
| Kr-85m | 1.7E+2 | 1.4E+4 | 3.6E+4 | 3.6E+4 | 5.3E+4 | 6.5E+4 | 3.2E+4 | 8.1E+3 | 9.8E-2 | 0 |
| Kr-87 | 3.2E+2 | 2.6E+4 | 4.7E+4 | 3.2E+4 | 2.2E+4 | 5.6E+3 | 5.8E+2 | 1.4E+0 | 0 | 0 |
| Kr-88 | 4.5E+2 | 3.8E+4 | 8.9E+4 | 8.2E+4 | 1.0E+5 | 8.6E+4 | 3.0E+4 | 2.6E+3 | 4.7E-5 | 0 |
| Kr-89 | 4.5E+2 | 5.6E+3 | 2.9E-1 | 7.0E-7 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-131m | 3.9E+0 | 3.4E+2 | 9.9E+2 | 1.2E+3 | 2.3E+3 | 5.3E+3 | 4.9E+3 | 8.0E+3 | 7.1E+3 | 5.9E+2 |
| Xe-133 | 1.4E+3 | 1.2E+5 | 3.4E+5 | 4.0E+5 | 8.0E+5 | 1.8E+6 | 1.6E+6 | 2.6E+6 | 1.8E+6 | 2.3E+4 |
| Xe-133m | 5.7E+1 | 5.0E+3 | 1.4E+4 | 1.7E+4 | 3.3E+4 | 7.1E+4 | 6.3E+4 | 9.1E+4 | 3.8E+4 | 4.9E+0 |
| Xe-135 | 1.8E+2 | 1.5E+4 | 4.1E+4 | 4.5E+4 | 7.8E+4 | 1.3E+5 | 9.2E+4 | 6.2E+4 | 2.8E+2 | 0 |
| Xe-135m | 2.5E+2 | 1.5E+4 | 4.6E+3 | 3.8E+2 | 3.9E+0 | 2.2E-4 | 5.1E-9 | 0 | 0 | 0 |
| Xe-137 | 1.0E+3 | 1.7E+4 | 5.6E+0 | 1.2E-4 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-138 | 1.1E+3 | 6.2E+4 | 1.5E+4 | 9.7E+2 | 5.6E+0 | 1.0E-4 | 7.8E-10 | 0 | 0 | 0 |
| Total | 5.4E+3 | 3.2E+5 | 6.1E+5 | 6.3E+5 | 1.1E+6 | 2.2E+6 | 1.9E+6 | 2.8E+6 | 1.9E+6 | 3.0E+4 |

## Table 15.6-11 Noble Gas Activities (Continued)

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| D.2 Control Room Integrated Inventory (megabecquerel-seconds) | | | | | | | | | | |
| Kr-83m | 1.6E+3 | 1.3E+6 | 4.8E+7 | 4.2E+7 | 7.8E+7 | 1.2E+8 | 3.7E+7 | 1.3E+7 | 0 | 0 |
| Kr-85 | 1.5E+2 | 1.4E+5 | 5.7E+6 | 7.1E+6 | 2.3E+7 | 1.1E+8 | 1.3E+8 | 5.6E+8 | 3.9E+9 | 1.6E+10 |
| Kr-85m | 3.3E+3 | 3.0E+6 | 1.2E+8 | 1.3E+8 | 3.2E+8 | 9.0E+8 | 6.3E+8 | 7.8E+8 | 1.5E+8 | 0 |
| Kr-87 | 6.4E+3 | 5.4E+6 | 1.8E+8 | 1.4E+8 | 1.9E+8 | 1.8E+8 | 3.0E+7 | 4.2E+6 | 0 | 0 |
| Kr-88 | 9.1E+3 | 8.0E+6 | 3.0E+8 | 3.0E+8 | 6.6E+8 | 1.4E+9 | 7.1E+8 | 4.9E+8 | 3.0E+7 | 0 |
| Kr-89 | 9.6E+3 | 2.2E+6 | 2.9E+6 | 8.0E+1 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-131m | 7.9E+1 | 7.1E+4 | 3.0E+6 | 3.7E+6 | 1.2E+7 | 5.5E+7 | 6.8E+7 | 2.8E+8 | 1.8E+9 | 3.6E+9 |
| Xe-133 | 2.7E+4 | 2.5E+7 | 1.0E+9 | 1.3E+9 | 4.2E+9 | 1.9E+10 | 2.3E+10 | 9.3E+10 | 5.2E+11 | 5.2E+11 |
| Xe-133m | 1.1E+3 | 1.0E+6 | 4.3E+7 | 5.3E+7 | 1.7E+8 | 7.5E+8 | 9.0E+8 | 3.4E+9 | 1.4E+10 | 4.9E+9 |
| Xe-135 | 3.5E+3 | 3.2E+6 | 1.3E+8 | 1.5E+8 | 4.4E+8 | 1.6E+9 | 1.5E+9 | 3.4E+9 | 2.4E+9 | 0 |
| Xe-135m | 5.0E+3 | 3.4E+6 | 4.8E+7 | 5.9E+6 | 5.9E+5 | 5.9E+3 | 2.9E-1 | 0 | 0 | 0 |
| Xe-137 | 2.1E+4 | 6.0E+6 | 1.2E+7 | 1.8E+3 | 0 | 0 | 0 | 0 | 0 | 0 |
| Xe-138 | 2.2E+4 | 1.4E+7 | 1.9E+8 | 1.8E+7 | 1.3E+6 | 7.6E+3 | 1.2E-1 | 0 | 0 | 0 |
| Total | 1.1E+5 | 7.3E+7 | 2.1E+9 | 2.1E+9 | 6.1E+9 | 2.4E+10 | 2.7E+10 | 1.0E+11 | 5.4E+11 | 5.5E+11 |

## Table 15.6-12 Noble Gas Activity Release to Environment

| Isotope | 1 min | 10 min | 1 h | 2 h | 4 h | 8 h | 12 h | 1 day | 4 days | 30 days |
|---|---|---|---|---|---|---|---|---|---|---|
| A. Reactor Building Release to Environment (megabecquerel) | | | | | | | | | | |
| Kr-83m | 2.7E+4 | 2.3E+6 | 9.3E+6 | 1.2E+7 | 1.9E+7 | 2.8E+7 | 3.2E+7 | 3.3E+7 | 3.3E+7 | 3.3E+7 |
| Kr-85 | 2.6E+3 | 2.3E+5 | 1.0E+6 | 1.5E+6 | 3.6E+6 | 1.2E+7 | 2.4E+7 | 8.1E+7 | 6.7E+8 | 5.6E+9 |
| Kr-85m | 5.6E+4 | 5.2E+6 | 2.1E+7 | 3.1E+7 | 5.9E+7 | 1.3E+8 | 1.9E+8 | 2.7E+8 | 2.9E+8 | 2.9E+8 |
| Kr-87 | 1.1E+5 | 9.3E+6 | 3.6E+7 | 4.4E+7 | 6.3E+7 | 7.8E+7 | 8.1E+7 | 8.1E+7 | 8.1E+7 | 8.1E+7 |
| Kr-88 | 1.6E+5 | 1.4E+7 | 5.6E+7 | 7.8E+7 | 1.4E+8 | 2.5E+8 | 3.1E+8 | 3.6E+8 | 3.7E+8 | 3.7E+8 |
| Kr-89 | 1.7E+5 | 4.8E+6 | 6.7E+6 | 6.7E+6 | 6.7E+6 | 6.7E+6 | 6.7E+6 | 6.7E+6 | 6.7E+6 | 6.7E+6 |
| Xe-131m | 1.3E+3 | 1.2E+5 | 5.2E+5 | 7.8E+5 | 1.9E+6 | 5.9E+6 | 1.3E+7 | 4.1E+7 | 3.0E+8 | 1.4E+9 |
| Xe-133 | 4.8E+5 | 4.1E+7 | 1.8E+8 | 2.8E+8 | 6.7E+8 | 2.1E+9 | 4.4E+9 | 1.4E+10 | 8.9E+10 | 2.5E+11 |
| Xe-133m | 2.0E+4 | 1.8E+6 | 7.4E+6 | 1.1E+7 | 2.7E+7 | 8.5E+7 | 1.7E+8 | 5.2E+8 | 2.6E+9 | 4.1E+9 |
| Xe-135 | 5.9E+4 | 5.6E+6 | 2.3E+7 | 3.4E+7 | 7.4E+7 | 1.9E+8 | 3.3E+8 | 6.7E+8 | 1.0E+9 | 1.0E+9 |
| Xe-135m | 8.5E+4 | 5.9E+6 | 1.7E+7 | 1.8E+7 | 1.8E+7 | 1.8E+7 | 1.8E+7 | 1.8E+7 | 1.8E+7 | 1.8E+7 |
| Xe-137 | 3.7E+5 | 1.3E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 | 1.9E+7 |
| Xe-138 | 3.7E+5 | 2.6E+7 | 7.4E+7 | 7.4E+7 | 7.4E+7 | 7.4E+7 | 7.4E+7 | 7.4E+7 | 7.4E+7 | 7.4E+7 |
| Total | 1.9E+6 | 1.3E+8 | 4.3E+8 | 5.7E+8 | 1.2E+9 | 3.0E+9 | 5.7E+9 | 1.6E+10 | 9.4E+10 | 2.6E+11 |
| B. Condenser Release to Environment (megabecquerel) | | | | | | | | | | |
| Kr-83m | 0 | 0 | 5.9E+3 | 7.8E+4 | 4.4E+5 | 1.3E+6 | 1.7E+6 | 1.9E+6 | 1.9E+6 | 1.9E+6 |
| Kr-85 | 0 | 0 | 7.4E+2 | 1.3E+4 | 1.3E+5 | 9.3E+5 | 2.6E+6 | 1.3E+7 | 2.3E+8 | 5.9E+9 |
| Kr-85m | 0 | 0 | 1.5E+4 | 2.3E+5 | 1.8E+6 | 8.5E+6 | 1.6E+7 | 3.0E+7 | 3.6E+7 | 3.6E+7 |
| Kr-87 | 0 | 0 | 2.0E+4 | 2.4E+5 | 1.1E+6 | 2.4E+6 | 2.7E+6 | 2.8E+6 | 2.8E+6 | 2.8E+6 |
| Kr-88 | 0 | 0 | 3.7E+4 | 5.6E+5 | 3.7E+6 | 1.4E+7 | 2.3E+7 | 3.1E+7 | 3.2E+7 | 3.2E+7 |
| Kr-89 | 0 | 0 | 4.1E+0 | 4.1E+0 | 4.1E+0 | 4.1E+0 | 4.1E+0 | 4.1E+0 | 4.1E+0 | 4.1E+0 |
| Xe-131m | 0 | 0 | 4.1E+2 | 6.7E+3 | 6.7E+4 | 4.8E+5 | 1.3E+6 | 6.7E+6 | 1.0E+8 | 1.3E+9 |
| Xe-133 | 0 | 0 | 1.4E+5 | 2.4E+6 | 2.3E+7 | 1.6E+8 | 4.4E+8 | 2.2E+9 | 3.0E+10 | 1.8E+11 |
| Xe-133m | 0 | 0 | 5.6E+3 | 1.0E+5 | 9.3E+5 | 6.7E+6 | 1.8E+7 | 8.1E+7 | 8.1E+8 | 2.0E+9 |
| Xe-135 | 0 | 0 | 1.7E+4 | 2.7E+5 | 2.4E+6 | 1.4E+7 | 3.3E+7 | 9.6E+7 | 2.0E+8 | 2.0E+8 |
| Xe-135m | 0 | 0 | 2.9E+3 | 1.0E+4 | 1.3E+4 | 1.3E+4 | 1.3E+4 | 1.3E+4 | 1.3E+4 | 1.3E+4 |
| Xe-137 | 0 | 0 | 3.4E+1 | 3.5E+1 | 3.5E+1 | 3.5E+1 | 3.5E+1 | 3.5E+1 | 3.5E+1 | 3.5E+1 |
| Xe-138 | 0 | 0 | 1.0E+4 | 3.2E+4 | 3.7E+4 | 3.7E+4 | 3.7E+4 | 3.7E+4 | 3.7E+4 | 3.7E+4 |
| Total | 0 | 0 | 2.5E+5 | 3.9E+6 | 3.4E+7 | 2.1E+8 | 5.4E+8 | 2.5E+9 | 3.2E+10 | 1.9E+11 |

## Table 15.6-13 Loss of Coolant Accident Meteorology and Offsite Dose Results

| Site Boundary Dose Results | | | | |
|---|---|---|---|---|
| | **Meteorology* ($s/m^3$)** | **Dist (m)** | **Thyroid Dose (Sv)** | **Whole Body Dose (Sv)** |
| | 2.18E–03 | max | 3 | 6.4E–02 |
| | 1.37E–03 | Chp 2 | 1.9E+00 | 4.1E–02 |
| | 1.18E–03 | 300 | 1.6E+00 | 3.5E–02 |
| | 2.19E–04 | 800 | 3.0E–01 | 6.5E–02 |

* "Max" = maximum meteorology to meet 10CFR100 limitation.

| Low Population Zone Boundary Dose Results | | | | |
|---|---|---|---|---|
| **Time (h)** | **Meteorology ($s/m^3$)** | **Dist (m)** | **Thyroid Dose (Sv)** | **Whole Body Dose (Sv)** |
| 8 | 3.73E–05 | 4828 | 7.3E–02 | 2.5E–03 |
| 24 | 1.21E–05 | | 9.9E–02 | 3.5E–03 |
| 96 | 4.27E–06 | | 2.0E–01 | 4.9E–03 |
| 720 | 9.09E–07 | | 3.4E–01 | 5.9E–03 |
| 8 | 1.56E-04 | Chp 2 | 3.1E–01 | 1.0E–02 |
| 24 | 9.61E-05 | | 5.1E–01 | 1.8E–02 |
| 96 | 3.36E-05 | | 1.3E+00 | 2.9E–02 |
| 720 | 7.42E-06 | | 2.4E+00 | 3.8E–02 |

## Table 15.6-14 Loss of Coolant Accident Meteorology and Control Room Dose Results

| Time (h) | Meteorology* ($s/m^3$) | Thyroid† (Sv) | Whole Body† (Sv) | Beta† (Sv) |
|---|---|---|---|---|
| 0–8 h | 3.10E–03 | 3.60E-02 | 3.50E-03 | 4.20E-02 |
| 8–24 h | 1.83E–03 | 7.20E-02 | 9.00E-03 | 1.33E-01 |
| 1–4 days | 1.16E–03 | 1.66E-01 | 1.96E-02 | 3.20E-01 |
| 4–30 days | 5.12E–04 | 2.76E-01 | 2.67E-02 | 4.47E-01 |

* See Subsection 15.6.5.5.3.2 for description of meteorology. Values are for dispersion from Reactor Building. Dispersion values for releases from Turbine Building are a factor of six less than Reactor Building dispersion values.

† These values are cumulative from the beginning to the end of period in the first column.

### Table 15.6-15 Sequence of Events for Cleanup Line Break Outside Containment

| Time (s) | Event |
|---|---|
| 0 | Clean up water line break occurs |
| 0+ | Check valves on clean up water line to feedwater line isolate<br>Differential pressure instrumentation initiates delay sequence |
| 45 | Differential pressure instrumentation actuates isolation valves |
| 75 | Isolation valves complete closure and isolation |
| 1-2 hour | Normal reactor shutdown and cooldown procedure |

### Table 15.6-16 Cleanup Line Break Accident Parameters

| | | | |
|---|---|---|---|
| I | Data and assumptions used to estimate source terms. | | |
| | A. | Power level | 4005 MWt |
| | B. | Number of bundles in core | 872 |
| | C. | Mass of fluid released | $2.8 \times 10^7$ g |
| | D. | Mass of fluid flashed to steam | $9.9 \times 10^6$ g |
| | E. | Duration of accident | < 2 h |
| II | Data and assumptions used to estimate activity released | | |
| | A. | Iodine water concentration | 15.6.4.5.1.1, case 1 |
| | B. | Iodine spiking | Table 15.6-1, IIB. |
| | C. | Iodine plateout fraction | 50% |
| | D. | Building release rate | direct to environment |
| | E. | SGTS filter efficiency | none assumed |
| III | Dispersion and Dose Data | | |
| | A. | Meteorology | Table 15.6-18 |
| | B. | Boundary distance | Table 15.6-18 |
| | C. | Method of dose calculation | Reference 15.6-2 |
| | D. | Dose conversion assumptions | Reference 15.6-2, RG 1.109, and ICRP 30 |
| | E. | Activity releases | Table 15.6-17 |
| | F. | Dose evaluations | Table 15.6-18 |

### Table 15.6-17 Clean Up Water Line Break Isotopic Releases (megabecquerel)

| Isotope | Release |
|---|---|
| I-131 | 8.1E+4 |
| I-132 | 1.9E+5 |
| I-133 | 2.3E+5 |
| I-134 | 3.2E+5 |
| I-135 | 2.5E+5 |

### Table 15.6-18 Clean Up Water Line Break Meteorology* and Dose Results

| Meteorology ($s/m^3$) | Distance (m) | Thyroid Dose (Sv) | Whole Body Dose (Sv) |
|---|---|---|---|
| 2.29E-02 | max | 3.0E–1 | 2.8E–3 |
| 1.37E-03 | Chp 2 | 1.8E–02 | 1.7E–4 |
| 1.18E-03 | 300 | 1.5E–02 | 1.5E–4 |
| 2.19E-04 | 800 | 2.8E–03 | 2.7E–5 |

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

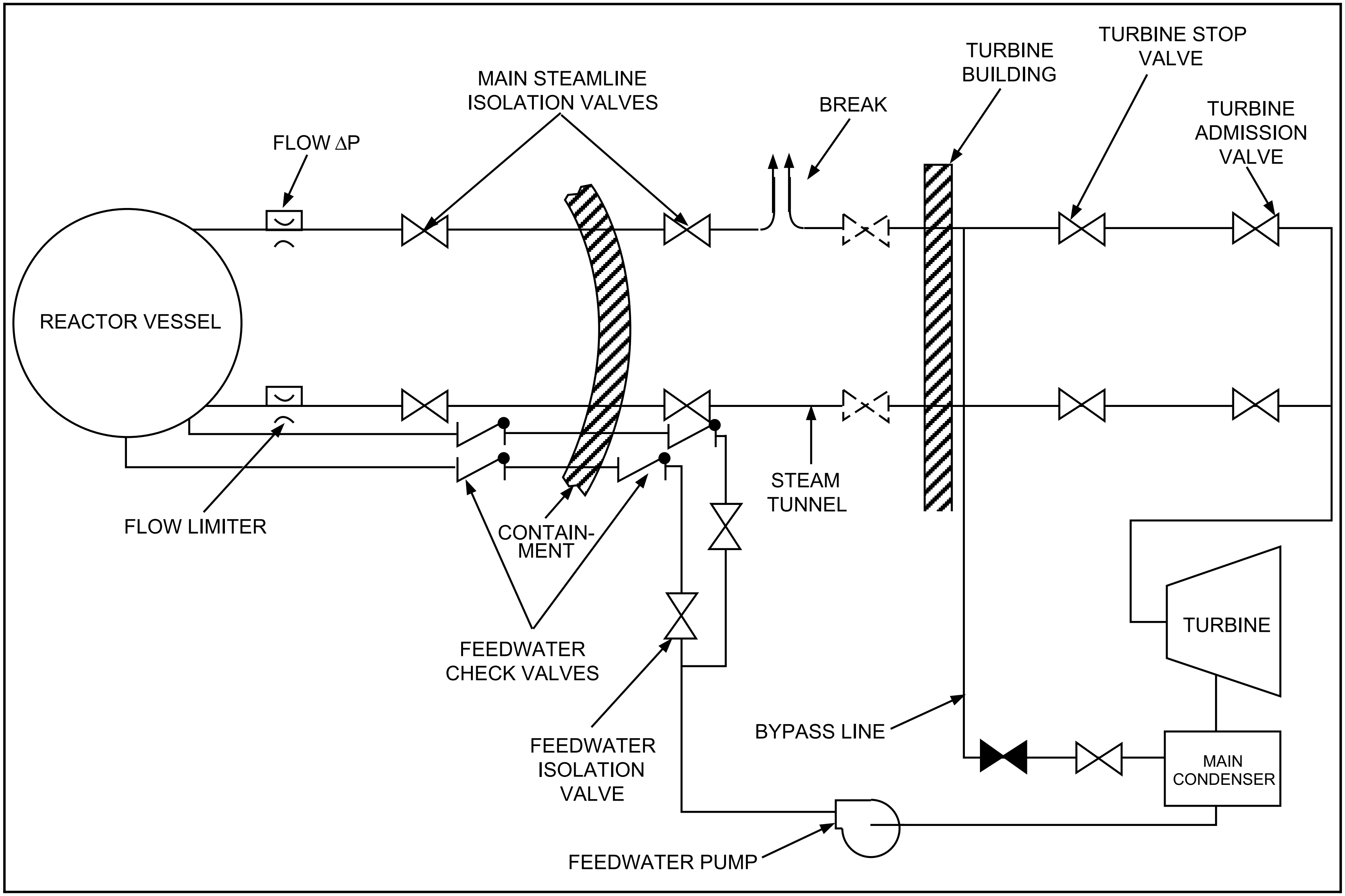


**Figure 15.6-1 Steam Flow Schematic for Steam Break Outside Containment**

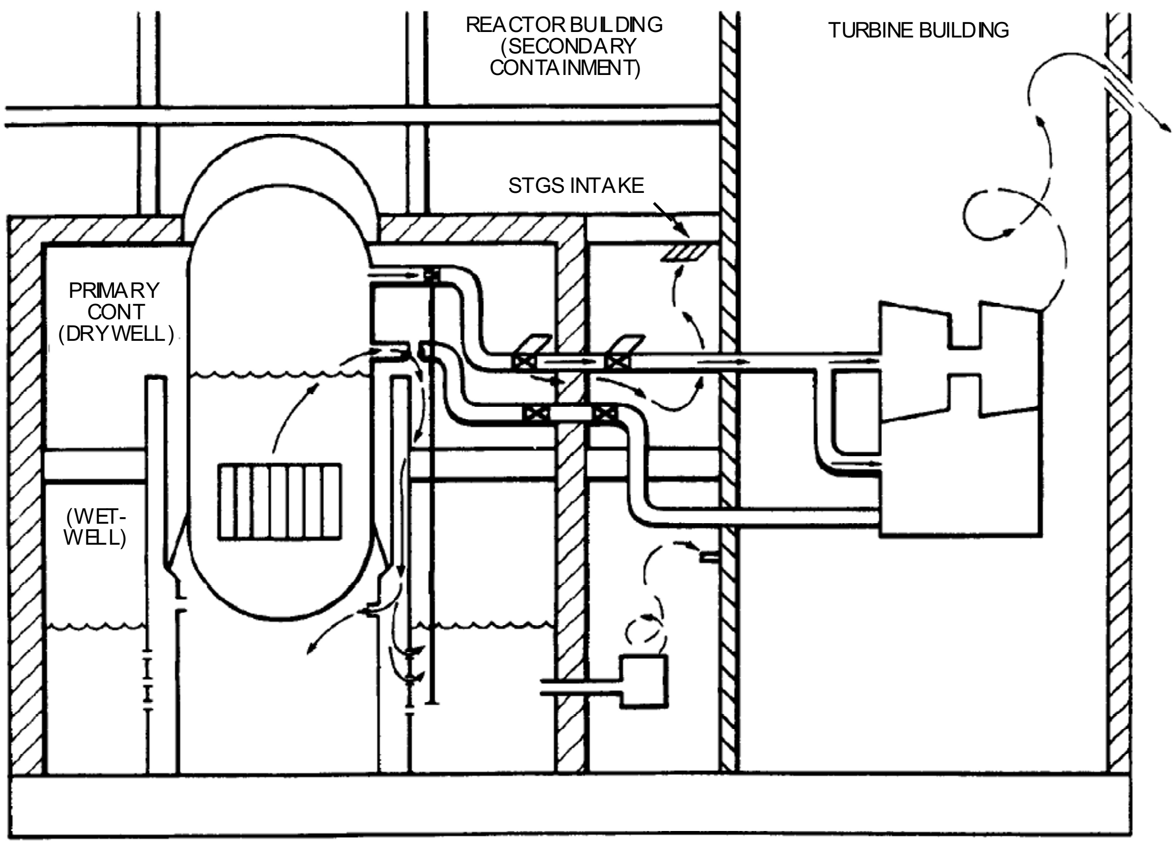


**Figure 15.6-2 LOCA Radiological Analysis**

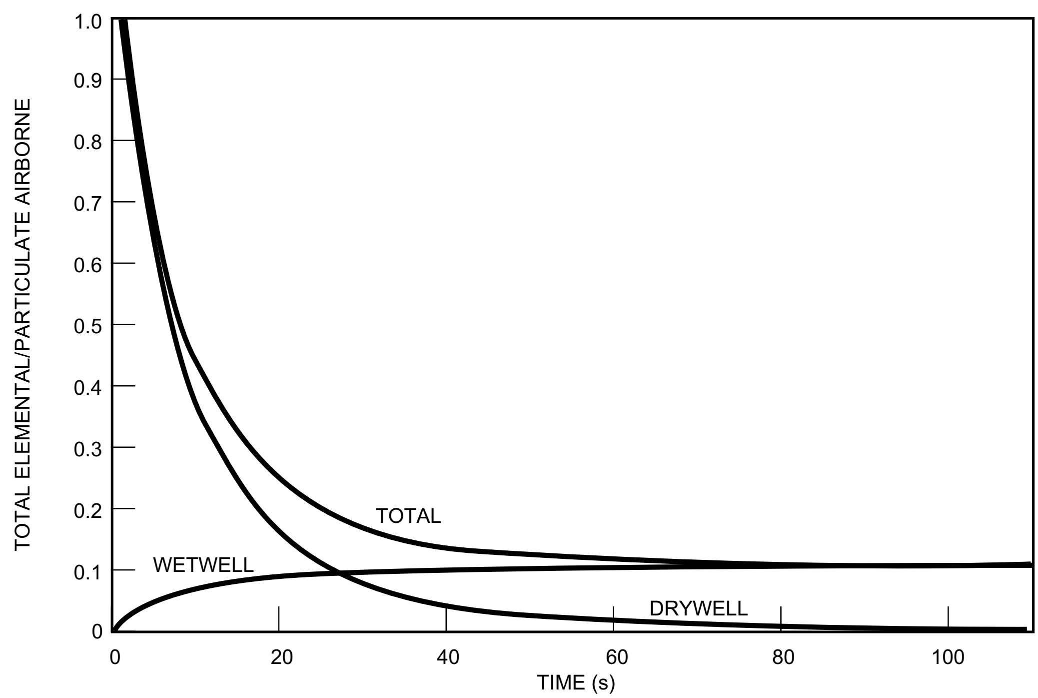


**Figure 15.6-3 Airborne Iodine in Primary Containment During Blowdown Phase**

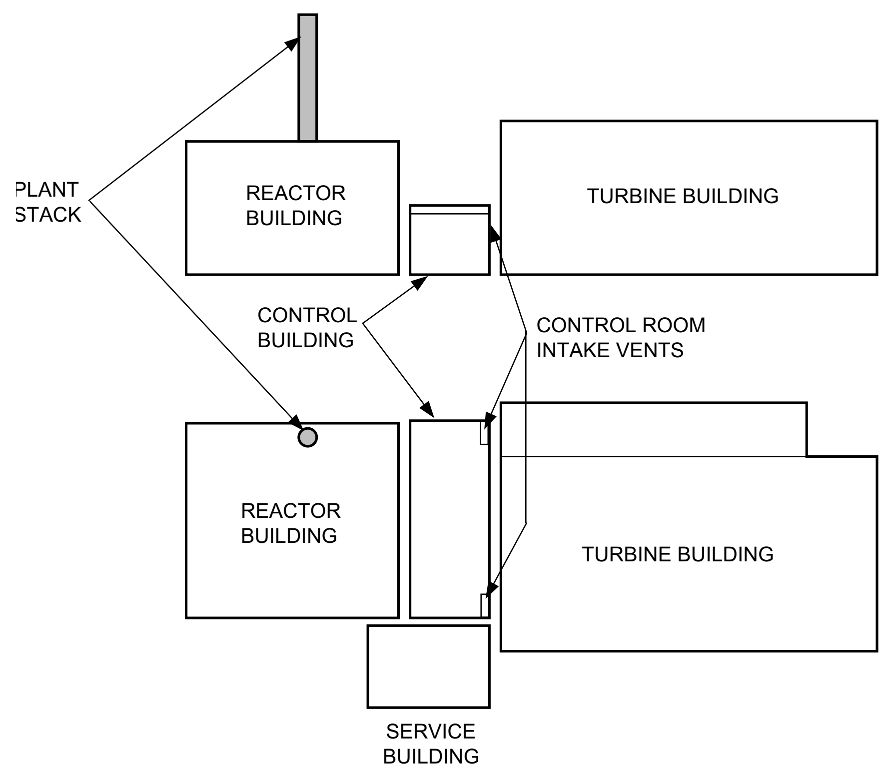


**Figure 15.6-4 ABWR Plant Layout**

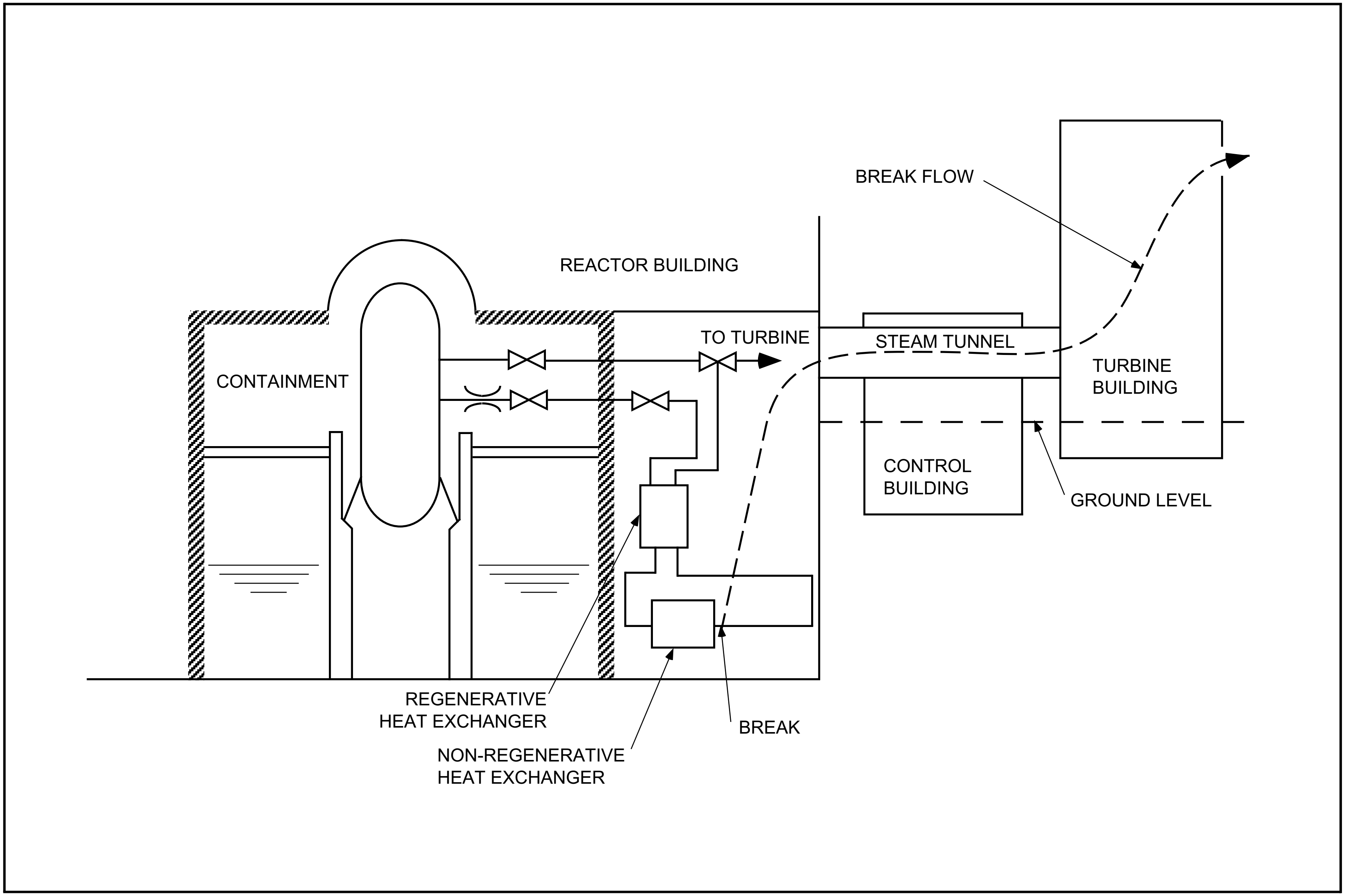


**Figure 15.6-5 Leakage Path for Clean Up Water Line Break**

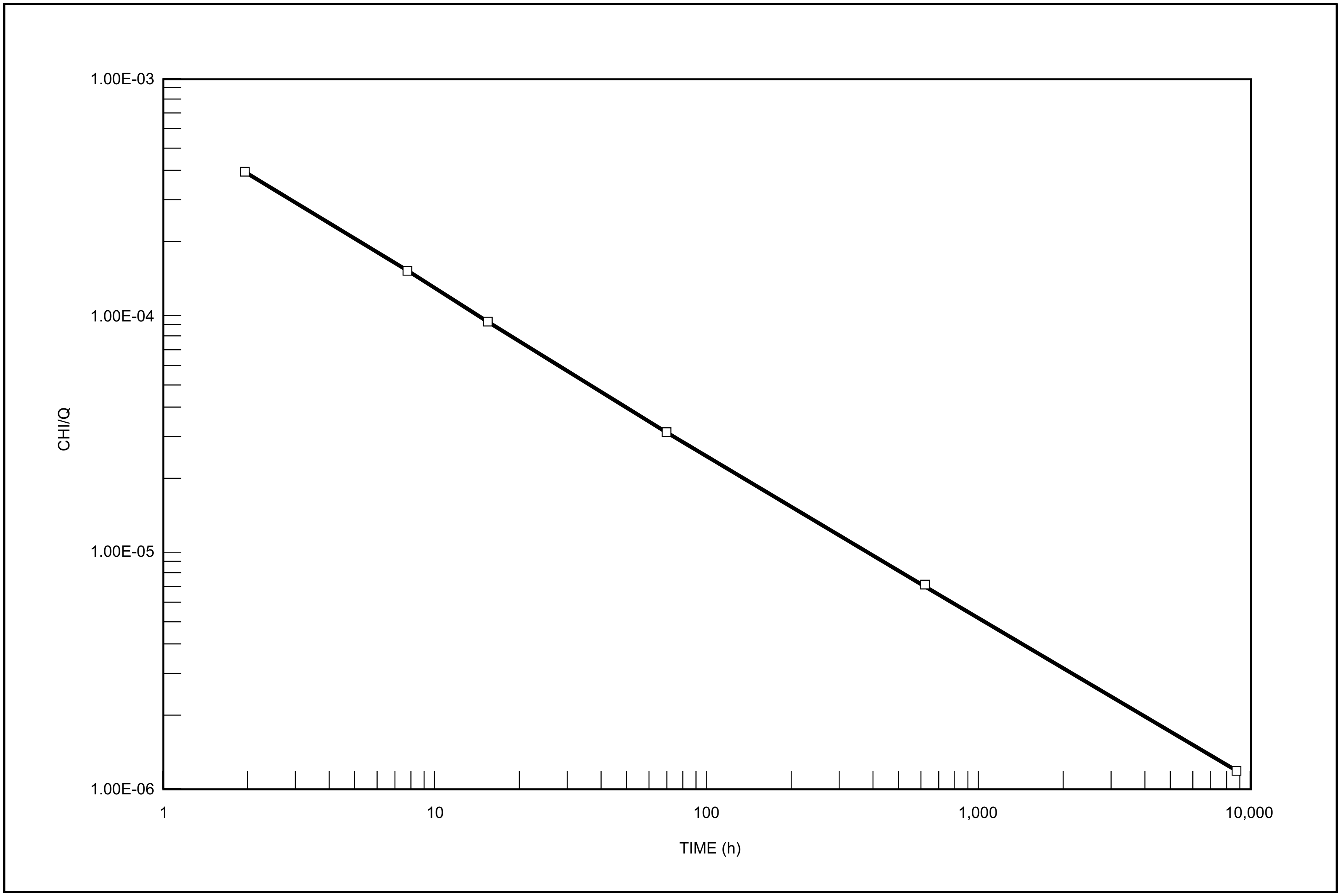


**Figure 15.6-6 ABWR Limiting LPZ CHI/Q**