Attached are the staff's comments on the Appendix C of NEI 99-03 entitled, "CRH Dose Analysis: Regulatory Enhancements." Per previous discussions with members of the analysis subgroup of the task force associated with this effort, the NRC staff has stated that they would like this appendix to use language like the guidance provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." The task force agreed that the comments in this form would be useful. The attached provides our current thinking to date on how Appendix C might look.

Appendix A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LWR LOSS-OF-COOLANT ACCIDENT

The assumptions in this appendix are acceptable to the NRC staff for evaluating the radiological consequences of loss-of-coolant accidents (LOCAs) at light water reactors (LWRs). These assumptions supplement the guidance provided in the main body of this guide.

Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 defines LOCAs as those postulated accidents that result from a loss of coolant inventory at rates that exceed the capability of the reactor coolant makeup system. Leaks up to a double-ended rupture of the largest pipe of the reactor coolant system are included. The LOCA, as with all design basis accidents (DBAs), is a conservative surrogate accident that is intended to challenge selective aspects of the facility design. Analyses are performed using a spectrum of break sizes to evaluate fuel and ECCS performance. With regard to radiological consequences, a large-break LOCA is assumed as the design basis case for evaluating the performance of release mitigation systems and the containment and for evaluating the proposed siting of a facility.

SOURCE TERM ASSUMPTIONS

1. Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.

ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT

3. Acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the primary containment in PWRs or the drywell in BWRs are as follows:

3.1 The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to ensure mixing between the drywell to the wetwell.

3.2 Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. An acceptable model for removal of iodine and particulates is described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1).

3.3 Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP[14] (Ref. A-1) may be credited. An acceptable model for the removal of iodine and particulates is described in Chapter 6.5.2 of the SRP.

The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. On a case by case basis containment mixing rates determined by the cooldown rate in the sprayed region and the buoyancy-driven flow that results may be considered. [this issue being reviewed] The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown.

The SRP sets forth a maximum decontamination factor (DF) for elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached.

3.4 Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6).

3.5 Guidance for reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs is given in Section 6.5.5 of the SRP (Ref. A-1). For suppression pool solutions having pH less than 7, molecular iodine vapor should be conservatively assumed to evolve into the containment atmosphere. [TBD by Staff... outstanding issues to be resolved]

3.6 Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).

3.7 The primary containment (i.e., drywell and wetwell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications.

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[14] Note that Regulatory Guide 6.2.5, revision 2 erroneously states that twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for the leakage from the primary reactor system. This value should be fifty percent of the equilibrium radioactive iodine inventory. Revision 2 erroneously accounted twice for the iodine deposited on the wall of the containment.

3.8 If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity.

ASSUMPTIONS ON DUAL CONTAINMENTS

4. For facilities with dual containment systems, the acceptable assumptions related to the transport, reduction, and release of radioactive material in and from the secondary containment or enclosure buildings are as follows.

4.1 Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.

4.2 Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.

4.3 The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).

4.4 Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.

4.5 Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and particulates may be considered on a case-by-case basis. Similarly, deposition of particulate radioactivity in gas-filled lines may be considered on a case-by-case basis.

4.6 Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

ASSUMPTIONS ON ESF SYSTEM LEAKAGE

5. ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems (Ref. A-7). The radiological consequences from the postulated leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of leakage from ESF components outside the primary containment for BWRs and PWRs.

5.1 With the exception of noble gases, all the fission products released from the fuel to the containment should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the start of the accident. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are nonconservative with regard to the buildup of sump activity.

5.2 The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems out of service. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.

5.3 With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.

5.4 If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment:

$$FF = \frac{h_{f_1} - h_{f_2}}{h_{fg}}$$

Where: h_{f1} is the enthalpy of liquid at system design temperature and pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.

5.5 If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.

5.6 Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS

6. For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.

6.1 For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.

6.2 All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.

6.3 Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.

6.4 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.

6.5 A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser may be credited if the components and piping systems used in the release path are capable of performing their safety function during and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.

ASSUMPTION ON CONTAINMENT PURGING

7. The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).

Appendix A REFERENCES

A-1 USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800.

A-2 D.A. Powers et al, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," NUREG/CR-6189, USNRC, July 1996.

A-3 S.L. Humphreys et al., "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, USNRC, April 1998.

A-4 D.A. Powers and S.B. Burson, "A Simplified Model of Aerosol Removal by Containment Sprays," NUREG/CR-5966, USNRC, June 1993.

A-5 USNRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 2, March 1978.

A-6 USNRC, "Laboratory Testing of Nuclear Grade Activated Charcoal," Generic Letter 99-02, June 3, 1999.

A-7 USNRC, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991.

A-8 USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

A-9 J.E. Cline, "MSIV Leakage Iodine Transport Analysis," Letter Report dated March 26, 1991. (ADAMS Accession Number ML003683718)

A-10 USNRC, "Safety Evaluation of GE Topical Report, NEDC-31858P (Proprietary GE report), Revision 2, *BWROG Report for Increasing MSIV Leakage Limits and Elimination of Leakage Control Systems, September 1993*," letter dated March 3, 1999, ADAMS Accession Number 9903110303.

Appendix B

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A FUEL HANDLING ACCIDENT

This appendix provides assumptions acceptable to the staff for evaluating the radiological consequences of a fuel handling accident at light water reactors. These assumptions supplement the guidance provided in the main body of this guide.

1. SOURCE TERM

Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The following assumptions also apply.

1.1 The number of fuel rods damaged during the accident should be based on a conservative analysis that considers the most limiting case. This analysis should consider parameters such as the weight of the dropped heavy load or the weight of a dropped fuel assembly (plus any attached handling grapples), the height of the drop, and the compression, torsion, and shear stresses on the irradiated fuel rods. Damage to adjacent fuel assemblies, if applicable (e.g., events over the reactor vessel), should be considered.

1.2 The fission product release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. All the gap activity in the damaged rods is assumed to be instantaneously released. Radionuclides that should be considered include xenons, kryptons, and halogens.

1.3 The iodine gap inventory is composed of inorganic species (99.75%) and organic species (0.25%).

2. WATER DEPTH

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). This difference in decontamination factors for elemental (99.75%) and organic iodine (0.25%) species results in the iodine above the water being composed of 44% elemental and 56% organic species. If the depth of water is not 23 feet, the decontamination factor will have to be determined on a case-by-case method (Ref. B-1). Proposed increases in the pool DF above 200 will need to address re-evolution of the scrubbed iodine species over the accident duration and should be supported by empirical data.

For release pressures greater than 1200 psig and water depth less than 23 feet, the iodine decontamination factors will be less than those assumed in this guide and must be calculated on an individual case basis using assumptions comparable in conservatism to those of this guide.

3. NOBLE GASES

The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

4. FUEL HANDLING ACCIDENTS WITHIN THE FUEL BUILDING

For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff.

4.1 The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.

4.2 A reduction in the amount of radioactive material released from the fuel pool by engineered safety feature (ESF) filter systems may be taken into account provided these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2, B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system[1] should be determined and accounted for in the radioactivity release analyses.

4.3 The radioactivity release from the fuel pool should be assumed to be drawn into the ESF filtration system without mixing or dilution in the fuel building. If mixing can be demonstrated, credit for mixing and dilution may be considered on a case-by-case basis. This evaluation should consider the magnitude of the building volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the pool, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the pool and the exhaust plenums.

5. FUEL HANDLING ACCIDENTS WITHIN CONTAINMENT

For fuel handling accidents postulated to occur within the containment, the following assumptions are acceptable to the NRC staff.

5.1 If the containment is isolated[2] during fuel handling operations, no radiological consequences need to be analyzed.

5.2 If the containment is open during fuel handling operations, but designed to automatically isolate in the event of a fuel handling accident, the release duration should be based on

[1] These analyses should consider the time for the radioactivity concentration to reach levels corresponding to the monitor setpoint, instrument line sampling time, detector response time, diversion damper alignment time, and filter system actuation, as applicable.

[2] Containment *isolation* does not imply containment integrity as defined by technical specifications for non-shutdown modes. The term isolation is used here collectively to encompass both containment integrity and containment closure, typically in place during shutdown periods. To be credited in the analysis, the appropriate form of isolation should be addressed in technical specifications.

delays in radiation detection and completion of containment isolation. If it can be shown that containment isolation occurs before radioactivity is released to the environment,[1] no radiological consequences need to be analyzed.

5.3 If the containment is open during fuel handling operations (e.g., personnel air lock or equipment hatch is open),[3] the radioactive material that escapes from the reactor cavity pool to the containment is released to the environment over a 2-hour time period.

5.4 A reduction in the amount of radioactive material released from the containment by ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. B-2 and B-3). Delays in radiation detection, actuation of the ESF filtration system, or diversion of ventilation flow to the ESF filtration system should be determined and accounted for in the radioactivity release analyses.[1]

5.5 Credit for dilution or mixing of the activity released from the reactor cavity by natural or forced convection inside the containment may be considered on a case-by-case basis. Such credit is generally limited to 50% of the containment free volume. This evaluation should consider the magnitude of the containment volume and exhaust rate, the potential for bypass to the environment, the location of exhaust plenums relative to the surface of the reactor cavity, recirculation ventilation systems, and internal walls and floors that impede stream flow between the surface of the reactor cavity and the exhaust plenums.

[3] The staff will generally require that technical specifications allowing such operations include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such adminstrative controls will generally require that a dedicated individual be present, with necessary equipment available, to restore containment closure should a fuel handling accident occur. Radiological analyses should generally not credit this manual isolation.

Appendix B REFERENCES

B-1. G. Burley, "Evaluation of Fission Product Release and Transport," Staff Technical Paper, 1971. (NRC Accession number 8402080322 in ADAMS or PARS)

B-2. USNRC, "Design, Testing, and Maintenance Criteria for Postaccident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 2, March 1978.

B-3. USNRC, "Laboratory Testing of Nuclear Grade Activated Charcoal," Generic Letter 99-02, June 3, 1999.

Appendix C

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR ROD DROP ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod drop accident at BWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.

1. Assumptions acceptable to the NRC staff regarding core inventory are provided in Regulatory Position 3 of this guide. For the rod drop accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and on the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are released to the reactor coolant.

2. If no or minimal[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity (typically 4 μCi/gm DE I-131) allowed by the technical specifications.

3. The assumptions acceptable to the NRC staff that are related to the transport, reduction, and release of radioactive material from the fuel and the reactor coolant are as follows.

3.1 The activity released from the fuel from either the gap or from fuel pellets is assumed to be instantaneously mixed in the reactor coolant within the pressure vessel.

3.2 Credit should not be assumed for partitioning in the pressure vessel or for removal by the steam separators.

3.3 Of the activity released from the reactor coolant within the pressure vessel, 100% of the noble gases, 10% of the iodine, and 1% of the remaining radionuclides are assumed to reach the turbine and condensers.

3.4 Of the activity that reaches the turbine and condenser, 100% of the noble gases and 10% of the iodine are available for release to the environment. The turbine and condensers leak to the environment as a ground- level release at a rate of 1% per day[2] for a period of 24 hours, at which time the leakage is assumed to terminate. No credit should be assumed

[1] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining the dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

[2] If there are forced flow paths from the turbine or condenser, such as unisolated motor vacuum pumps or unprocessed air ejectors, the leakage rate should be assumed to be the flow rate associated with the most limiting of these paths. Credit for collection and processing of releases, such as by off gas or standby gas treatment, will be considered on a case-by-case basis.

for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

3.5 In lieu of the transport assumptions provided in paragraphs 3.2 through 3.4 above, a more mechanistic analysis may be used on a case-by-case basis. Such analyses account for the quantity of contaminated steam carried from the pressure vessel to the turbine and condensers based on a review of the minimum transport time from the pressure vessel to the first main steam isolation (MSIV) and considers MSIV closure time.

3.6 The iodine species released from the reactor coolant within the pressure vessel should be assumed to be 5% particulate, 91% elemental, and 4% organic.

Appendix D

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A BWR MAIN STEAM LINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line accident at BWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

2. If no or minimal[1] fuel damage is postulated for the limiting event, the released activity should be the maximum coolant activity allowed by technical specification. The iodine concentration in the primary coolant is assumed to correspond to the following two cases in the nuclear steam supply system vendor's standard technical specifications.

2.1 The concentration that is the maximum value (typically 4.0 μCi/gm DE I-131) permitted and corresponds to the conditions of an assumed pre-accident spike, and

2.1 The concentration that is the maximum equilibrium value (typically 0.2 μCi/gm DE I-131) permitted for continued full power operation.

3. The activity released from the fuel should be assumed to mix instantaneously and homogeneously in the reactor coolant. Noble gases should be assumed to enter the steam phase instantaneously.

TRANSPORT

4. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.

4.1 The main steam line isolation valves (MSIV) should be assumed to close in the maximum time allowed by technical specifications.

4.2 The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure.

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[1] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

4.3 All the radioactivity in the released coolant should be assumed to be released to the environment instantaneously as a ground-level release. No credit should be assumed for plateout, holdup, or dilution within facility buildings.

4.4 The iodine species released from the main steam line should be assumed to be 5% particulate, 91% elemental, and 4% organic.

Appendix E

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR STEAM GENERATOR TUBE RUPTURE ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a steam generator tube rupture accident at PWR light-water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by technical specification. Two cases of iodine spiking should be assumed.

2.1 A reactor transient has occurred prior to the postulated steam generator tube rupture (SGTR) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).

2.2 The primary system transient associated with the SGTR causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 335 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.

3. The activity released from the fuel, if any, should be assumed to be released instantaneously and homogeneously through the primary coolant.

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.

[2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

TRANSPORT

5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows:

5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. The leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

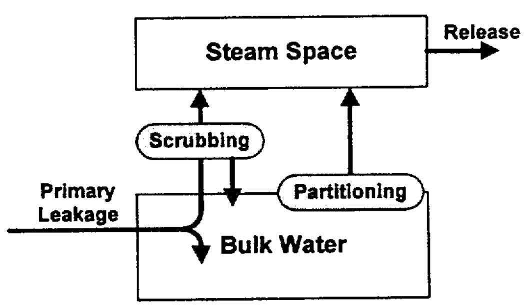
5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on room temperature liquid. Facility instrumentation used to determine leakage is typically located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

5.5 The transport model described in this section should be utilized for iodine and particulate releases from the steam generators. This model is shown in Figure E-1 and summarized below:

Figure E-1
Transport Model



5.5.1 A portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant.

- With regard to the unaffected steam generators used for plant cooldown, the primary-to-secondary leakage can be assumed to mix with the secondary water without flashing during periods of total tube submergence.

5.5.2 The primary to secondary leakage that immediately flashes to vapor will rise through the bulk water of the steam generator and enter the steam space. Credit may be taken for scrubbing in the generator, using the models in NUREG-0409, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident" (Ref. E-1), during periods of total submergence of the tubes.

5.5.3 The primary to secondary leakage that does not immediately flash is assumed to mix with the bulk water.

5.5.4 The radioactivity in the bulk water is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient.[3] A partition coefficient for iodine of 100 may be assumed.

5.6 Operating experience and analyses have shown that for some steam generator designs, tube uncovery may occur for a short period following any reactor trip (Ref. E-2). The potential impact of tube uncovery on the transport model parameters (e.g., flash fraction, scrubbing credit) needs to be considered. The impact of emergency operating procedure restoration strategies on steam generator water levels should be evaluated.

5.7 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

[3] *Partition Coefficient* is defined as:

$$PC = \frac{mass\ \ of\, I_2\ per\ unit\ mass\ of\ liquid}{mass\ \ of\, I_2\ per\ unit\ mass\ of\ gas}$$

Appendix E REFERENCES

E-1 USNRC, "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," NUREG-0409, May 1985.

E-2 USNRC, "Steam Generator Tube Rupture Analysis Deficiency," Information Notice 88-31, May 25, 1988.

Appendix F

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR MAIN STEAM LINE BREAK ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a main steam line break accident at PWR light water reactors. These assumptions supplement the guidance provided in the main body of this guide.[1]

SOURCE TERMS

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached. The fuel damage estimate should assume that the highest worth control rod is stuck at its fully withdrawn position.

2. If no or minimal[2] fuel damage is postulated for the limiting event, the activity released should be the maximum coolant activity allowed by the technical specifications. Two cases of iodine spiking should be assumed.

2.1 A reactor transient has occurred prior to the postulated main steam line break (MSLB) and has raised the primary coolant iodine concentration to the maximum value (typically 60 μCi/gm DE I-131) permitted by the technical specifications (i.e., a preaccident iodine spike case).

2.2 The primary system transient associated with the MSLB causes an iodine spike in the primary system. The increase in primary coolant iodine concentration is estimated using a spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant (expressed in curies per unit time) increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value (typically 1.0 μCi/gm DE I-131) specified in technical specifications (i.e., concurrent iodine spike case). A concurrent iodine spike need not be considered if fuel damage is postulated. The assumed iodine spike duration should be 8 hours. Shorter spike durations may be considered on a case-by-case basis if it can be shown that the activity released by the 8-hour spike exceeds that available for release from the fuel gap of all fuel pins.

3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity," for acceptable assumptions and methodologies for performing radiological analyses.

[2] The activity assumed in the analysis should be based on the activity associated with the projected fuel damage or the maximum technical specification values, whichever maximizes the radiological consequences. In determining dose equivalent I-131 (DE I-131), only the radioiodine associated with normal operations or iodine spikes should be included. Activity from projected fuel damage should not be included.

4. The chemical form of radioiodine released from the fuel should be assumed to be 5 percent particulate iodine, 91 percent element iodine, and 4 percent organic iodide. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

TRANSPORT [3]

5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.

5.1 For facilities that have not implemented alternative repair criteria (see Ref. F-1, DG-1074), the primary-to-secondary leak rate in the steam generators should be assumed to be the leak rate limiting condition for operation specified in the technical specifications. For facilities with traditional generator specifications (both per generator and total of all generators), the leakage should be apportioned between affected and unaffected steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of the parameter being converted. The ARC leak rate correlations are generally based on the collection of cooled liquid. Surveillance tests and facility instrumentation used to show compliance with leak rate technical specifications are typically based on cooled liquid. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

5.4 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

5.5 The transport model described in Regulatory Positions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates. During dryout in the faulted steam generator, all of the primary to secondary leakage is assumed to flash to vapor and released to the environment with no mitigation.

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[3] *Faulted* refers to the state of the steam generator in which the secondary side has been depressurized by a MSLB such that protective system response (main steam line isolation, reactor trip, safety injection, etc.) has occurred.

Appendix F REFERENCES

F-1 USNRC, "Steam Generator Tube Integrity," Draft Regulatory Guide DG-1074, December 1998.

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Appendix G

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR LOCKED ROTOR ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a locked rotor accident at PWR light water reactors.[1] These assumptions supplement the guidance provided in the main body of this guide.

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory and the release of radionuclides from the fuel are in Regulatory Position 3 of this regulatory guide. The release from the breached fuel is based on Regulatory Position 3.2 of this guide and the estimate of the number of fuel rods breached.

2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the main steam line break outside containment.

3. The activity released from the fuel should be assumed to be released instantaneously and homogeneously through the primary coolant.

4. The chemical form of radioiodine released from the fuel should be assumed to be 5 percent particulate iodine, 91 percent elemental iodine, and 4 percent organic iodide. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

RELEASE TRANSPORT

5. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material to the environment are as follows.

5.1 The primary-to-secondary leak rate in the steam generators should be assumed to be the leak-rate-limiting condition for operation specified in the technical specifications. The leakage should be apportioned between the steam generators in such a manner that the calculated dose is maximized.

5.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests are typically based on cool liquid. Facility instrumentation used to determine leakage is typically located on lines containing

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.

cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

5.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated.

5.4 The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

5.5 All noble gas radionuclides released from the primary system are assumed to be released to the environment without reduction or mitigation.

5.6 The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

Appendix H

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A PWR ROD EJECTION ACCIDENT

This appendix provides assumptions acceptable to the NRC staff for evaluating the radiological consequences of a rod ejection accident at PWR light water reactors.[1] These assumptions supplement the guidance provided in the main body of this guide.

SOURCE TERM

1. Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. For the rod ejection accident, the release from the breached fuel is based on the estimate of the number of fuel rods breached and the assumption that 10% of the core inventory of the noble gases and iodines is in the fuel gap. The release attributed to fuel melting is based on the fraction of the fuel that reaches or exceeds the initiation temperature for fuel melting and the assumption that 100% of the noble gases and 50% of the iodines contained in that fraction are available for release from containment.[2] For the secondary system release pathway, 100% of the noble gases and 50% of the iodines in that fraction are released to the reactor coolant.

2. If no fuel damage is postulated for the limiting event, a radiological analysis is not required as the consequences of this event are bounded by the consequences projected for the loss-of-coolant accident (LOCA) and the main steam line break.

3. Two release cases are to be considered. In the first, 100% of the activity released from the fuel should be assumed to be released instantaneously and homogeneously through the containment atmosphere. In the second, 100% of the activity released from the fuel should be assumed to be completely dissolved in the primary coolant and available for release to the secondary system.

4. The chemical form of radioiodine released to the containment atmosphere should be assumed to be 5% particulate iodine, 91% elemental iodine, and 4% organic iodide. If containment sprays do not actuate or are terminated prior to accumulating sump water, or if the containment sump pH is not controlled at values of 7 or greater, the iodine species should be evaluated on an individual case basis. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products.

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[1] Facilities licensed with, or applying for, alternative repair criteria (ARC) should use this section in conjunction with the guidance that is being developed in Draft Regulatory Guide DG-1074, "Steam Generator Tube Integrity" (USNRC, December 1998), for acceptable assumptions and methodologies for performing radiological analyses.

[2] Note that Regulatory Guide 1.77 erroneously states that twenty-five percent of the iodines contained in that fraction are available for release from containment and also allowed for credit for reduction or containment radioactivity by sprays. When sprays are credited this value should be fifty percent rather than twenty-five percent.

TRANSPORT FROM CONTAINMENT

6. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the containment are as follows.

6.1 A reduction in the amount of radioactive material available for leakage from the containment that is due to natural deposition, containment sprays, recirculating filter systems, dual containments, or other engineered safety features may be taken into account. Refer to Appendix A to this guide for guidance on acceptable methods and assumptions for evaluating these mechanisms.

6.2 The containment should be assumed to leak at the leak rate incorporated in the technical specifications at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident. Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing. Leakage from subatmospheric containments is assumed to be terminated when the containment is brought to a subatmospheric condition as defined in technical specifications.

TRANSPORT FROM SECONDARY SYSTEM

7. Assumptions acceptable to the NRC staff related to the transport, reduction, and release of radioactive material in and from the secondary system are as follows.

7.1 A leak rate equivalent to the primary-to-secondary leak rate limiting condition for operation specified in the technical specifications should be assumed to exist until shutdown cooling is in operation and releases from the steam generators have been terminated.

7.2 The density used in converting volumetric leak rates (e.g., gpm) to mass leak rates (e.g., lbm/hr) should be consistent with the basis of surveillance tests used to show compliance with leak rate technical specifications. These tests typically are based on cooled liquid. The facility's instrumentation used to determine leakage typically is located on lines containing cool liquids. In most cases, the density should be assumed to be 1.0 gm/cc (62.4 lbm/ft^3).

7.3 All noble gas radionuclides released to the secondary system are assumed to be released to the environment without reduction or mitigation.

7.4 The transport model described in assumptions 5.5 and 5.6 of Appendix E should be utilized for iodine and particulates.

Appendix I

Acronyms

AST	Alternative source term
BWR	Boiling water reactor
CDF	Core damage frequency
COLR	Core operating limits report
DBA	Design basis accident
DNBR	Departure from nucleate boiling ratio
EAB	Exclusion area boundary
EPA	Environmental Protection Agency
ESF	Engineered safety feature
FHA	Fuel handling accident
FSAR	Final safety analysis report
IPF	Iodine protection factor
LERF	Large early release fraction
LOCA	Loss-of-coolant accident
LPZ	Low population zone
MOX	Mixed oxide
MSLB	Main steam line break
NDT	Non-destructive testing
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RM	Radiation monitor
SG	Steam generator
SGTR	Steam generator tube rupture
TEDE	Total effective dose equivalent
TID	Technical information document
TMI	Three Mile Island

References for page 11

12. A.G. Croff, "A User's Manual for the ORIGEN 2 Computer Code," ORNL/TM-7175, Oak Ridge National Laboratory, July 1980.

13. S.M. Bowman and L.C. Leal, "The ORIGNARP Input Processor for ORIGEN-ARP," Appendix F7.A in *SCALE: A Modular Code System for Performing Standardized Analyses for Licensing Evaluation*, NUREG/CR-0200, USNRC, March 1997.

3. ACCIDENT SOURCE TERM

This section provides a source term that is acceptable to the NRC staff. It provides guidance on the fission product inventory, release fractions, timing of the release, radionuclide composition, chemical form and the fuel damage in non-loca DBAs.

3.1 Fission Product Inventory

The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty.[1] The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 (Ref. 12) or ORIGEN-ARP (Ref. 13). Core inventory factors (Ci/MWt) provided in TID14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.

For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.

No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions[2]

The core inventory release fractions, by radionuclide groups, for the DBA LOCAs are listed in Table 1 for BWRs and PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

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[1] The uncertainty factor used in determining the core inventory should be that value provided in Appendix K to 10 CFR Part 50, typically 1.02. A value lower than 1.02, but not less than the licensed power level may be used provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error. See Federal Register Notice Volume 65, Number 106, June 1, 2000 for details.

[2] The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.

For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 2. The release fractions from Table 2 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. For non-LOCA DBAs where fuel melt is postulated the core inventory release fractions, by radionuclide groups is listed in Table 1 for BWRs and PWRs.

Table 1
BWR And PWR Core Inventory Fraction Released Into Containment Atmosphere

Group	Release Fraction
Noble Gases	1.0
Halogens[3]	0.5

Table 2[4]
Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05

3.3 Timing of Release Phases

For LOCA DBAs the core activity released is assumed to be immediately available for release. For non-LOCA DBAs in which fuel damage is projected, the activity available for

[3] If containment sprays are not modeled mechanistically, such as in Regulatory Guide 6.5.2, revision 2, one half of the equilibrium radioactive iodine inventory may be assumed to be deposited on the walls of the containment. The net value of core inventory available for release from containment would, therefore, be 0.25 for a non mechanistic spray representation. Please note that Regulatory Guide 6.5.2, revision 2 erroneously states that twenty-five percent of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for the leakage from the primary reactor system. This value should be fifty percent of the equilibrium radioactive iodine inventory. Revision 2 erroneously accounted twice for the iodine deposited on the wall of the containment.

[4] The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for burnups exceeding 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.

release from the fuel is assumed to be immediately available for release from containment or the building where the fuel is damaged.

3.4 Radionuclide Composition

Table 3 lists the elements in each radionuclide group that should be considered in design basis analyses.

Table 3
Radionuclide Groups

Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 5 percent of the iodine released should be assumed to be particulate iodine, 91 percent elemental iodine, and 4 percent organic iodide. This includes releases from the gap and the fuel pellets. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.

3.6 Fuel Damage in Non-LOCA DBAs

The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.

The amount of fuel damage caused by a FHA is addressed in Appendix B of this guide.

January 4, 2001

POTENTIAL MODIFICATIONS TO ARCON96 TO TREAT HIGH-VELOCITY VENT RELEASES

Disclaimer note: The following is an initial draft effort. As such it does not imply what the ultimate outcome may be and it may contain errors. Many issues require resolution. Decisions related to these and/or other issues that may be identified could result in significantly different results than may be implied below, including a decision that there is not an adequate basis for revising the ARCON methodology. Further testing, verification, resolution of some apparent inconsistencies, etc., are needed.

BACKGROUND

The ARCON computer code (Ramsdell and Simonen 1997) was developed as an alternative to the Murphy-Campe (Murphy and Campe 1974) method of calculating atmospheric dispersion factors (X/Q) for control room habitability assessments. The code calculates relative concentration values (X/Qs) for ground-level and stack releases using standard Gaussian equations. It has a third release option, vent releases. In the vent release option, X/Qs for vent releases are calculated by averaging the X/Qs for ground-level and stack release using the procedure described in Regulatory Guide 1.111 (NRC 1977). That procedure was developed to estimate X/Qs at the EAB and LPZ for releases from vents with low to moderate vertical velocities (10 to 15 m/s).

The ARCON code makes several significant departures from the Murphy-Campe procedure. It calculates X/Qs from hourly meteorological data and averages the hourly X/Qs, accounting for changes in meteorological conditions, to get X/Qs for periods ranging from one to 720 hours in duration. It includes corrections to the dispersion factors to account for dispersion under low wind speed conditions and in building wakes. It considers stack releases explicitly, and it has the vent release option.

Following release of the ARCON code, problems have arisen in two related areas of code application. The first problem area has been the use of the ARCON code for control room habitability assessments for stack releases. In these applications, the distance from the stack to the control room intake is generally small compared to the stack height. Therefore, the stack effluent does not have time to disperse down to intake level before being transported beyond the intake. As a result, the dispersion factors calculated by ARCON are typically essentially zero ($<<10^{-10}$ s/m^3). The second problem area is the use of ARCON for calculation of dispersion factors for vents with high velocity releases, for example, for releases from main steam safety valves (MSSVs) and atmospheric dump valves (ADVs). These releases have significant plume rise that carries the effluents well above control room intake level for almost all meteorological conditions. Again the dispersion factors calculated by ARCON are typically essentially zero.

Atmospheric dispersion factors of zero pose problems from a regulatory standpoint if there are physical mechanisms that can carry contaminated effluent from a stack or vent to the control room intake. Two such mechanisms can be postulated. The first is dispersion under calm

Attachment 3

conditions (mean wind velocity = 0.0), and the other involves wind direction meander including wind direction reversals. These mechanisms are not represented in ARCON or other computer codes that rely on straight-line Gaussian dispersion models. However, ARCON can be modified to represent either or both mechanisms. Frenkiel derived a model that can be used to compute X/Q for any wind speed, including zero, and the standard straight-line Gaussian model can be used for meandering conditions and wind reversal if the distance is interpreted as distance traveled before arriving at the intake rather than the straight-line distance from the release point to the intake. In either case, the key to estimating X/Q is determining plume rise.

LOW WIND SPEED DISPERSION MODELS

In ARCON, ground-level X/Qs for stack releases are calculated using the standard straight-line Gaussian plume model

$$\chi / Q = \frac{1}{\pi \sigma_y \sigma_z U} \exp\left[-0.5 \left(\frac{h_s + \Delta h}{\sigma_z} \right)^2 \right] \tag{1}$$

where σ_y and σ_z are atmospheric dispersion parameters that are a function of atmospheric stability and distance. For all stability classes, X/Q is zero directly beneath the release point because the exponential term is zero. As the distance increases, the dispersion parameters increase. Near the release point, the exponential term increases faster than the remainder of the term on the right side of Equation (1) decreases. The exponential term asymptotically approaches a value of 1.0, and as a result there is a distance at which the ground-level concentration reaches a maximum; as the distance continues to increase, the concentration decreases. This model can be used for low wind speeds, but not for calm winds because the equation becomes undefined if the wind speed is zero. As a practical matter, the equation should not be used for wind speeds less than about 1 m/s.

Atmospheric dispersion does not cease when the wind is apparently calm. Equation (1) becomes undefined in calm winds because it is only a partial solution of the governing equations. A portion of the complete solution for the governing equations is eliminated by the assumptions leading to Equation (1). Frenkiel (1953) used a different set of assumptions in solving the governing equations and arrived at a solution that remains defined for calm winds.

FRENKIEL'S MODEL

Frenkiel's model is well behaved in low wind speed conditions and gives finite X/Q values for calm wind (mean wind velocity = 0). As wind speed increases, X/Qs increase to a maximum value for a wind speed in the 1 to 2 m/s range and then decrease as the wind speed continues to increase. The model is (ASPP 1984, Eq 6.260 following correction)

$$\chi/Q = \frac{\sigma_u \exp\left(-\frac{U^2}{2\sigma_u^2}\right)}{(2\pi)^{3/2}\sigma_v\sigma_w r^2}\left[1 + \left(\frac{\pi}{2}\right)^{1/2}\frac{Ux}{\sigma_u r}\exp\left(\frac{U^2x^2}{2\sigma_u^2 r^2}\right)\mathrm{erfc}\left(-\frac{1}{2^{1/2}}\frac{Ux}{\sigma_u r}\right)\right] \quad (2)$$

where σ_u, σ_v, and σ_w are along wind, cross wind, and vertical turbulence measures, respectively (m/s) and r is a pseudo-diagonal distance from a point directly above the release point to the intake. For calm winds Equation (2) has a simple form that is similar to the standard Gaussian puff model. It is

$$\chi / Q = \frac{\sigma_u}{(2\pi)^{3/2}\sigma_v\sigma_w r^2} \quad (3)$$

The definition of r is

$$r^2 = x^2 + \left(\frac{\sigma_u}{\sigma_v}\right)^2 y^2 + \left(\frac{\sigma_u}{\sigma_w}\right)^2 z^2 \quad (4)$$

for positions under the center line of a plume, y = 0.0, and $z = h_s + \Delta h$, thus

$$r^2 = x^2 + \left(\frac{\sigma_u}{\sigma_w}\right)^2 (h_s + \Delta h)^2 \quad (5)$$

Values of σ_u and σ_w may be estimated from the wind speed and, perhaps, stability. Published data on low wind turbulence and data collected during dispersion experiments suggest that reasonable estimates of the turbulence parameters required by the Frenkiel's model can be made. A cursory review of atmospheric turbulence data indicates that an equation of the form

$$\sigma = (a^2 + b^2U^2)^{1/2} \quad (6)$$

can be used to estimate both σ_u and σ_w, providing appropriate values are selected for a and b. However, this approach has not been peer reviewed or published.

At moderate and high wind speeds, the X/Qs predicted by Frenkiel's model decrease proportional to about $u^{-3/2}$ (not including the effect of wind speed on plume rise). This decrease is more rapid than other models. In the common straight-line Gaussian models, X/Q decreases proportional to u^{-1}. Consequently, it would seem appropriate to limit application of the model to calm and near calm conditions.

 DRAFT

RECIRCULATION MODEL

The recirculation model is basically the straight-line Gaussian model given by Equation (1) except that the distance used to determine model parameters (σ_y, σ_z, and Δh) is no longer the downwind distance. Instead, the wind direction is sufficiently variable that effluent returns to the vicinity of the release point, and the distance traveled is assumed to be equal to the distance to the maximum of the X/Q vs distance curve for the standard straight-line model. The distance to the maximum in the X/Q vs distance curve depends on initial release height, plume rise, and atmospheric stability. In addition, the maximum may be reached before or after the plume reaches its equilibrium height. Therefore, it is necessary to search for the maximum X/Q.

The assumption that the plume returns to the vicinity of the release point is reasonable for calm and nearly calm winds. As the wind speed increases, the assumption becomes less tenable. As with the Frenkiel model, there is wind speed above which the model should not be used. Selection of the limiting wind speed is likely to be highly subjective.

PLUME RISE

Both Frenkiel's model and the recirculation model require plume rise estimates. Briggs (1984) provides useful equations for limiting plume rise in stable (PG stability classes E through F) and neutral atmospheric conditions (PG stability class D). However, the equation he provides for use in unstable conditions (PG stability classes A through C) requires information not readily available from licensee's meteorological systems.

The rise of plumes from MSSVs and ADVs does not appear to be specifically addressed in the literature. However, the plume rise equations of Briggs are derived from a combination of theoretical bases and experimental data that should be reasonably applicable to rise of plumes from these vents provided that the vents are at roof-top level and that the vent is uncapped and directed upward.

Near the source, Briggs gives following equation (Briggs 1984, Eq. 8.57) for plume rise near the source

$$\Delta h = \left(\frac{3}{\beta_1^{\,2}} \frac{F_m}{U^2} x + \frac{3}{2\beta_2^{\,2}} \frac{F_b}{U^3} x^2 \right)^{1/3} \qquad \textbf{(7)}$$

where
Δh = plume rise (m)
F_m = momentum flux parameter (m^4/s^2)
β_1 = dimensionless entrainment constant related to momentum
U = wind speed at release height
x = distance from the release point (m)
F_b = buoyancy flux parameter (m^4/s^3)
β_2 = dimensionless entrainment constant related to buoyancy.

Briggs uses a value of 0.6 for β_1 and calculates β_2 as (Briggs 1984, Eq. 8.46)

$$\beta_2 = 0.4 + 1.2\frac{U}{w_0} \tag{8}$$

where w_0 is the effluent exit velocity. The momentum flux parameter, F_m, is the momentum flux of the effluent at the vent divided by $\pi\rho_a$ where ρ_a is the density of air (kg/m^3). Thus, F_m is

$$F_m = \frac{\rho_o V_o}{\pi\rho_a} w_o \tag{9}$$

where
ρ_o = effluent density after expansion to atmospheric pressure (kg/m^3)
V_o = volumetric release rate (m^3/s)
w_o = effluent vertical velocity (m/s).

Similarly, the buoyancy flux parameter is the buoyant flux divided by $\pi\rho_a$. It is

$$F_b = g\frac{(\rho_a - \rho_o)V_o}{\pi\rho_a} \tag{10}$$

Although plume rise estimated by Eq. 10 continues indefinitely, Briggs provides equations to estimate maximum plume rise for rise into a stable layer and rise limited by ambient turbulence.

For stable atmospheric conditions, the rise of buoyancy dominated plumes is limited (Briggs 1984, Eq. 8.71) to

$$\Delta h_{max} = 2.6\left(\frac{F_b}{Us^2}\right)^{1/3} \tag{11}$$

where s is the Brunt-Väisälä frequency, given by

$$s = \frac{g}{\theta_a}\frac{\partial\theta_a}{\partial z} \tag{12}$$

where g is gravitational acceleration, 9.8 m/s^2, and θ_a is the potential temperature of the air and z is the height above ground. Typical values of s are 0.00049 s^{-2} for E stability, 0.0013 s^{-2} for F stability, and 0.002 s^{-2} for G stability. Buoyant plume rise in neutral conditions is limited by ambient turbulence to (Briggs 1984, Eq. 8.97)

$$\Delta h_{max} = 1.2\left(\frac{F_b}{Uu^{*2}}\right)^{3/5}(h_s + \Delta h)^{2/5} \tag{13}$$

where u* is a scaling velocity related to atmospheric turbulence. For most purposes, u* is proportional to the wind speed and surface roughness with a constant of proportionality that is a function of surface roughness and height above ground. Typical values of the constant for nuclear power plant sites range from about 10 to 20. Buoyant rise in unstable conditions should be greater than the rise in neutral conditions.

According to Briggs, even moderately warm plumes should ultimately be buoyancy dominated. Nevertheless, the following equations that can be used to estimate the maximum rise of momentum-dominated plumes are given for completeness. For stable atmospheric conditions, the rise of a momentum-dominated plume is limited to (Briggs 1984, Eq. 8.66)

$$\Delta h_{max} = 2.44\left(\frac{F_m}{s}\right)^{1/4} \qquad (14)$$

For momentum-dominated plumes in neutral conditions, plume rise is limited to (Briggs 1984, Eqs. 8.87 and 8.96)

$$\Delta h_{max} = \beta_2^{-6/7}\left(\frac{F_m}{U}\right)^{3/7}\left[\frac{0.6(h_s + \Delta h)}{u^{*3}}\right]^{1/7} \qquad (15)$$

Equations (13) and (15) both involve plume rise in a manner that precludes a general closed form solution. Approximate solutions are readily obtainable if $h_s << \Delta h$ or $h_s >> \Delta h$. In the case of high temperature, high velocity vent releases neither approximation is appropriate. However, when the equations are solved iteratively, the solutions converge rapidly.

Neither Ray Hosker (NOAA Atmospheric Transport and Dispersion Division) nor David Wilson (Univ. of Alberta) finds fault with the basic notion of using Briggs plume rise equations for releases from main steam system isolation valves or atmospheric dump valves. However, they expressed uneasiness with using the full plume rise calculated by Briggs equations for vents adjacent to or on the sides of buildings. Wilson suggested that some adjustment could be made to the entrainment constants in the equations to account for building effects. Neither Hosker nor Wilson is aware of anything in recent literature that addresses these issues.

VENT RELEASE PARAMETERS

The plume rise equations contain a buoyance flux parameter (F_b) and a momentum flux parameter (F_m). For most releases, these parameters are easily calculated from the air temperature, and the effluent temperature, stack flow, and stack radius. For high-temperature steam releases from the vents under consideration, the vent acts as a throttle. As the steam enters the air it expands to atmospheric pressure. Steam tables are needed to estimate the temperature and density of the effluent after expansion. It will be necessary to either require users to calculate F_b and F_m and enter them with other data or program steam tables into ARCON.

Precise estimates of the density of the steam will require air temperature and pressure. Neither air temperature or pressure is included in ARCON meteorological data sets. Temperature is included in the standard NRC meteorological data format, but pressure isn't.

PRELIMINARY COMPUTATIONAL INSIGHTS

Industry[a] provided NRC with a characterization of the thermodynamic conditions, mass flow rates, and velocities for steam discharges from ADVs, PORVs, and MSSVs for design-basis steam generator tube rupture events for three reactor types. These reactors were a 1973 vintage Two-Loop Westinghouse Plant, a 1973 vintage B&W plant, and the ABB-CE System 80+ ALWR. The System 80+ discharges were characterized at three stages of the SGTR event. Some additional information was provided that indicated that characteristics of potential discharges from a mid-1970s vintage ABB/CE plant are similar to those from a B&W plant. The discharge characteristics are summarized in Table 1.

Table 1. Test Case Steam Discharge Characteristics for SGTR Events

Reactor	Rel. Ht (m)	Intake Ht (m)	Intake Dist. (m)	Cont. Ht (m)	Cont. Dist. (m)	Vert. Vel. (m/s)	F_b (m^4/s^3)	F_m (m^4/s^2)
Westinghouse	29.0	25.3	76	61.9	19.8	456	101	3,110
B&W	11.2	19	40	39.3	21	67.5	6.75	57.8
ABB/CE	17.2	10.7	62.4	39.3	21	67.5	6.75	57.8
ALWR Stage 1	13.7	22.5	35.0	53.0	24	457	432	46,300
ALWR Stage 2	13.7	22.5	35.0	53.0	24	131	181	5,680
ALWR Stage 3	13.7	22.5	35.0	53.0	24	12	16.5	47.4

The release height, intake height, and containment heights listed in Table 1 are all above grade. When the intake height is less than the release height, the difference in heights increases the effective release height. When the intake height is greater than the release height, the difference decreases the effective release height. In each case in Table 1, the release height is well below the height of the containment building. The difference between the containment height and the release height is related to the plume rise required for vent releases to clear the building and building wake. This factor has not been included in modeling of atmospheric dilution factors, but it is likely to be a significant factor if it can be incorporated appropriately. Plume rise less than this difference in release and containment building heights indicates the plume is likely to mix within the building wake and impact the intake even though the plume may initially rise above the intake. Plumes with rise less than a factor of 2 greater than the difference are likely to be entrained within the building wake on occasion. The intake distance is the horizontal distance between the release point and intake. It is a straight-line distance that does

[a] K.O. Cozens, Nuclear Energy Institute, email to J. J. Hayes, NRC, June 8, 2000.

not include the effects of intervening structures. The containment distance is the distance of the release point from the containment building. It may be a significant factor in determining plume rise, but has not been incorporated in plume rise modeling.

PLUME RISE

Briggs plume rise equations (ASPP 1984, Eq 8.71 and 8.97) were used to estimate a maximum plume rise for each discharge characterization for Pasquill-Gifford Stability Classes A through G and wind speeds from 0.5 m/s to 10 m/s. The results of these calculations are presented in Table 2. The following assumptions were made in making the calculations:

1) maximum plume rise is limited to 1,000 m
2) maximum plume rise for neutral conditions may be used for unstable conditions
3) during stable conditions, plume rise is limited to the smaller of the rises calculated for neutral and stable conditions
4) the characteristic wind speed (u*) may be estimated as U/12.

These assumptions are reasonable, but by no means the only assumptions that might have been made. In addition, the values of F_b and F_m provided by industry were used without full verification. These values appear reasonable.

The computational results shown in Table 2 indicate that there are likely to be conditions in which the discharge steam clears the structures in the vicinity of the release point and that there are other conditions in which the discharge steam may not clear the structures. For example, the test case ALWR Stage 1 discharges are likely to clear structures for almost all conditions, but CE, B&W, and stage 3 ALWR vent releases may not clear structures in moderate to high winds. The bold numbers in Table 2 indicate maximum plume rise less than twice the difference in release point and containment building heights, and the numbers in bold italics indicate maximum plume rise less than the difference in heights.

Table 2. Maximum Plume Rise as a Function of Reactor Type, Wind Speed, and Stability

Reactor	Stability Class	Wind Speed (m/s) 0.5	1.0	2.0	4.0	7.0	10.0
Westinghouse	A-D	1000.0	1000.0	1000.0	483.6	102.6	**42.3**
	E	171.2	135.9	107.8	85.6	71.0	**42.3**
	F	123.7	98.1	77.9	**61.8**	**51.3**	**42.3**
	G	107.1	85.0	67.5	**53.6**	**44.4**	**39.5**
CE and B&W	A-D	1000.0	1000.0	255.8	**37.1**	[redacted]	[redacted]
	E	69.5	**55.1**	**43.8**	**34.7**	[redacted]	[redacted]
	F	**50.2**	**39.8**	[redacted]	[redacted]	[redacted]	[redacted]
	G	**43.5**	**34.5**	[redacted]	[redacted]	[redacted]	[redacted]

ALWR Stage 1	A-D	1000.0	1000.0	1000.0	1000.0	372.0	132.8
	E	275.9	219.0	173.8	138.0	114.5	101.7
	F	199.3	158.2	125.6	99.7	82.7	**73.4**
	G	172.7	137.0	108.8	86.3	**71.6**	**63.6**
ALWR Stage 2	A-D	1000.0	1000.0	1000.0	840.4	164.2	**61.1**
	E	207.9	165.0	131.0	104.0	86.3	**61.1**
	F	150.2	119.2	94.6	**75.1**	**62.3**	**55.3**
	G	130.1	103.3	82.0	**65.1**	**54.0**	**47.9**
ALWR Stage 3	A-D	1000.0	1000.0	615.7	84.0	[redacted]	[redacted]
	E	93.6	**74.3**	**58.9**	**46.8**	[redacted]	[redacted]
	F	**67.6**	**53.7**	**42.6**	[redacted]	[redacted]	[redacted]
	G	**58.6**	**46.5**	[redacted]	[redacted]	[redacted]	[redacted]

The computational results shown in Table 2 do not take into account any potential effects of buildings on plume rise. In moderate and high winds, buildings may increase atmospheric turbulence and thereby reduce plume rise. Similarly, the results do not consider locations of vents other than on the top of the highest structure in the building complex. In particular, there is no assurance that the plume-rise estimates in Table 2 are reasonable for vents located on the sides of buildings well below the roof line.

ATMOSPHERIC DILUTION FACTORS

Control room atmospheric dilution factors were calculated for a range of wind speed and atmospheric stability classes to examine the variation of the dilution factors with these important meteorological parameters. The atmospheric dilution factors calculated by the Frenkiel model are listed in Table 3, and those calculated using the recirculation model are listed in Table 4.

Table 3. Atmospheric Dilution Factors Calculated by the Frenkiel Model

		Wind Speed (m/s)					
	Stability Class	0.5	1.0	2.0	4.0	7.0	10.0
Westinghouse	A-D	1.57E-07	5.11E-08	8.34E-09	8.84E-09	8.99E-08	3.42E-07
	E	5.22E-06	2.71E-06	7.04E-07	2.78E-07	1.86E-07	3.42E-07
	F	9.90E-06	5.14E-06	1.34E-06	5.28E-07	3.53E-07	3.42E-07
	G	1.31E-05	6.82E-06	1.78E-06	7.01E-07	4.68E-07	3.91E-07
B&W	A-D	8.01E-07	2.70E-07	7.78E-07	2.72E-05	2.19E-02	7.49E-03
	E	2.48E-04	1.71E-04	6.13E-05	3.39E-05	2.19E-02	7.49E-03
	F	5.62E-04	4.27E-04	1.73E-04	1.14E-04	2.19E-02	7.49E-03
	G	8.23E-04	6.58E-04	2.87E-04	2.17E-04	2.19E-02	7.49E-03
ABB/CE	A-D	1.22E-06	4.12E-07	1.10E-06	1.71E-05	1.41E-04	3.95E-04
	E	2.64E-04	1.66E-04	5.15E-05	2.28E-05	1.41E-04	3.95E-04
	F	5.04E-04	3.27E-04	1.04E-04	4.66E-05	1.41E-04	3.95E-04

	G	6.68E-04	4.38E-04	1.40E-04	6.39E-05	1.41E-04	3.95E-04
ALWR Stage 1	A-D	7.03E-07	2.36E-07	3.90E-08	9.74E-09	3.53E-08	2.31E-07
	E	9.98E-06	5.63E-06	1.59E-06	6.85E-07	4.97E-07	4.40E-07
	F	1.99E-05	1.15E-05	3.36E-06	1.50E-06	1.12E-06	1.01E-06
	G	2.72E-05	1.60E-05	4.74E-06	2.15E-06	1.63E-06	1.49E-06
ALWR Stage 2	A-D	7.03E-07	2.36E-07	3.90E-08	1.39E-08	2.13E-07	1.67E-06
	E	1.82E-05	1.05E-05	3.05E-06	1.35E-06	1.00E-06	1.67E-06
	F	3.69E-05	2.20E-05	6.65E-06	3.08E-06	2.38E-06	2.21E-06
	G	5.08E-05	3.08E-05	9.54E-06	4.52E-06	3.57E-06	3.37E-06
ALWR Stage 3	A-D	7.03E-07	2.36E-07	1.06E-07	2.30E-06	1.55E-04	2.76E-02
	E	1.08E-04	6.91E-05	2.28E-05	1.16E-05	1.55E-04	2.76E-02
	F	2.36E-04	1.63E-04	5.89E-05	3.33E-04	1.55E-04	2.76E-02
	G	3.38E-04	2.43E-04	9.28E-05	5.62E-05	1.55E-04	2.76E-02

Table 4. Atmospheric Dilution Factors Calculated by the Recirculation Model

	Stability Class	Wind Speed (m/s) 0.5	1.0	2.0	4.0	7.0	10.0
Westinghouse	A	1.52E-06	9.07E-07	1.78E-06	3.08E-06	4.30E-06	7.68E-06
	B	5.00E-07	2.50E-07	1.00E-07	7.10E-07	2.17E-06	7.10E-06
	C	0.00E-00	1.36E-07	6.78E-08	7.36E-08	1.69E-06	6.30E-06
	D	0.00E-00	0.00E-00	2.63E-38	1.31E-14	9.16E-07	4.63E-06
	E	1.88E-06	1.91E-06	1.87E-06	1.78E-06	1.66E-06	3.98E-06
	F	1.87E-06	2.26E-06	2.46E-06	2.52E-06	2.47E-06	2.90E-06
	G	3.45E-07	1.02E-06	1.59E-06	1.78E-06	1.85E-06	1.86E-06
B&W	A	3.13E-06	8.07E-06	5.70E-05	7.80E-04	7.26E-04	4.45E-04
	B	5.13E-07	6.53E-07	1.47E-05	9.58E-04	1.24E-03	6.69E-04
	C	2.78E-07	1.39E-07	1.56E-06	7.51E-04	2.30E-03	1.02E-03
	D	0.00E-00	4.54E-24	1.18E-09	3.06E-05	4.98E-03	1.07E-03
	E	3.84E-05	3.71E-05	3.51E-05	3.36E-05	8.58E-03	8.18E-04
	F	7.27E-05	7.54E-05	7.59E-05	7.59E-05	1.54E-02	1.08E-03
	G	6.80E-05	7.92E-05	8.69E-05	9.97E-05	4.14E-03	9.43E-04
ABB/CE	A	2.97E-06	6.82E-06	1.51E-05	3.71E-05	5.98E-05	1.00E-04
	B	4.98E-07	5.63E-07	4.89E-06	2.31E-05	6.73E-05	9.79E-05
	C	2.70E-07	1.35E-07	7.44E-07	1.57E-05	6.21E-05	9.51E-05
	D	0.00E-00	1.68E-25	1.47E-10	1.15E-05	5.29E-05	8.34E-05
	E	2.22E-05	1.92E-05	1.60E-05	1.28E-05	5.07E-05	8.35E-05
	F	3.15E-05	2.84E-05	2.42E-05	1.95E-05	4.27E-05	7.03E-05
	G	2.28E-05	2.20E-05	1.98E-05	1.69E-05	3.41E-05	5.84E-05
ALWR Stage 1	A	1.54E-06	7.72E-07	5.15E-07	7.73E-07	1.02E-06	1.64E-06
	B	5.13E-07	2.57E-07	1.28E-07	7.76E-08	2.27E-07	1.15E-07
	C	0.00E-00	0.00E-00	6.95E-08	3.48E-08	1.46E-07	8.73E-07
	D	0.00E-00	0.00E-00	0.00E-00	9.81E-13	3.69E-08	4.38E-07
	E	2.82E-07	4.95E-07	5.66E-07	6.04E-07	6.27E-07	6.35E-07
	F	1.01E-07	3.85E-07	6.79E-07	8.48E-07	9.77E-07	1.05E-06
	G	0.00E-00	0.00E-00	2.17E-07	5.42E-07	7.26E-07	8.30E-07
ALWR Stage 2	A	1.54E-06	7.72E-07	1.27E-06	2.51E-06	4.31E-06	6.69E-06
	B	5.13E-07	2.57E-07	1.28E-07	4.17E-07	1.67E-06	5.59E-06
	C	0.00E-00	0.00E-00	6.95E-08	2.51E-08	7.95E-07	4.86E-06
	D	0.00E-00	0.00E-00	0.00E-00	7.66E-18	3.54E-07	3.46E-06
	E	1.22E-06	1.34E-06	1.43E-06	1.48E-06	1.50E-06	2.90E-06
	F	1.08E-06	1.68E-06	2.07E-06	2.43E-06	2.67E-06	2.81E-06
	G	0.00E+00	6.69E-07	1.41E-06	1.89E-06	2.29E-06	2.55E-06
ALWR Stage 3	A	1.67E-06	4.06E-06	1.30E-05	1.34E-03	8.80E-04	4.95E-04
	B	5.13E-07	2.57E-07	2.57E-06	2.02E-03	1.45E-03	7.04E-04
	C	2.78E-07	1.39E-07	1.55E-07	2.70E-03	2.45E-03	8.12E-04
	D	0.00E-00	1.56E-39	2.28E-13	3.72E-06	3.99E-03	4.13E-04
	E	1.64E-05	1.65E-05	1.61E-05	1.55E-05	5.00E-03	1.39E-04

F	2.82E-05	3.11E-05	3.30E-05	3.40E-05	1.10E-04	2.83E-06
G	2.32E-05	2.87E-05	3.43E-05	4.00E-05	9.10E-05	9.03E-03

For the Frenkiel model, the effective release height was assumed to be equal to the maximum plume rise plus the release height minus the intake height. The effective release height for the Westinghouse vent releases is approximately 4 m greater than the maximum plume rise, and the effective release height for the ABB/CE vent release is about 8 m greater than the plume rise. For the remaining vents releases, the effective release height is 8 or 9 m lower than the maximum plume rise. In some cases (B&W at 7 and 10 m/s, and ALWR Stage 3 at 10 m/s) the effective release height is near zero. The atmospheric dilution factors in these cases are much larger than the atmospheric dilution factors at lower wind speeds. As the wind speed increases above the speed at which the effective release height becomes zero, the atmospheric dilution factor decreases as if the release were a ground-level release. Note that the Frenkiel model may be used at all wind speeds, not just low wind speeds.

The effective release heights for the recirculation model calculations were also the maximum plume rise (Equation 13, 14, or 15, as appropriate) plus the release height minus the intake height. For each vent type in low-wind speed, neutral stability (D) conditions, the effective release height was sufficient that the maximum in the X/Q vs distance curve was not reached within 20 km of the release point. Similarly for ALWR stage 1 and stage 2 releases, the vertical dispersion under extremely stable (G) conditions at low wind speeds (0.5 m/s) is sufficiently small that the X/Q maximum was not reached. In all of these cases, the atmospheric dilution factors at 20 km, which are zero or near zero, are listed in Table 4. For the remaining cases, the table lists the maximum dilution factor in the X/Q vs distance curve unless the maximum occurs at a distance less than the distance between the release point and the intake. If the maximum in the X/Q vs distance curve occurs between the release point and the intake, the X/Q is calculated at the distance to the intake.

For wind speeds above about 4 m/s for unstable (A, B, C) conditions and about 5 or 6 m/s for neutral (D) and stable (E, F, G) conditions, the atmospheric dilution factors are about the same as atmospheric dilution factors for ground-level releases. However, these wind speeds are probably greater that the maximum wind speed for which a recirculation model is appropriate.

95TH PERCENTILE ATMOSPHERIC DILUTION FACTORS

Tables 3 and 4 present atmospheric dilution factors calculated by the Frenkiel and recirculation models without considering the frequencies of various combinations of atmospheric conditions. A copy of the of the ARCON96 source code was modified to use the Frenkiel and recirculation models for vent and elevation releases to permit evaluation of these models in a regulatory setting. The modified code has undergone limited testing, but the testing has not been sufficient to ensure that the results are correct.

In the initial set of modifications, the Frenkiel and recirculation models were added to ARCON. The Frenkiel model was used for wind speeds less than U_f, and the recirculation model was used for wind speeds greater than U_f but less than U_r. The existing elevated plume model was used for wind speeds above U_r. Using 1.0 m/s for U_f and 3.0 m/s for U_r, the modified code was run with five different sets of meteorological data for each of set of vent characteristics and vent/intake geometry. The results of these calculations are summarized in Table 5.

Table 5. ARCON00x Estimate of 95th Percentile X/Q Using A Combination of the Frenkiel and Recirculation Models

					System 80+ ALWR		
Meteor. Data	Year	West.	B&W	CE	Stage 1	Stage 2	Stage 3
Site A	1991	2.42E-06	2.93E-03	6.92E-05	8.16E-07	2.28E-06	4.17E-03
Site B	1988	1.86E-06	7.03E-04	5.06E-05	6.45E-07	1.97E-06	8.45E-04
Site C	1993	2.35E-06	4.41E-03	1.75E-03	4.39E-05	7.89E-05	6.32E-04
Site D	1995	7.22E-06	4.27E-03	1.41E-03	3.29E-05	6.27E-05	5.36E-04
Site E	1996	2.33E-06	4.54E-04	4.98E-05	6.67E-07	2.06E-06	1.19E-03

As expected, the 95th percentile X/Qs are sensitive to the vent release characteristics and release point-intake geometry. The sensitivity to vent release characteristics is seen in the variation of X/Qs for the ALWR releases. The geometry for these releases is the same. Sensitivity to release point/intake geometry is seen by comparing the X/Qs for the B&W and CE vent releases. These releases have the same characteristics, only the geometry is different.

The variation of 95th percentile X/Qs with meteorological data sets is not consistent. The Site C and Site D data sets give significantly larger X/Q for the B&W, CE and ALWR Stage 1 and 2 vent releases than the other three data sets. In contrast, the Site C and Site D data sets give lower X/Qs for the ALWR Stage 3 releases than the other three data sets. This inconsistency needs to be investigated.

A second modification was made to ARCON to permit the code to be using only the recirculation model. The results of running the code in this manner are presented in Table 6. The recirculation model was used when the wind speed was less than 3.0 m/s and a wind speed of 1.0 m/s was assumed for all hours with wind speeds less than 1.0 m/s. In general, the 95th percentile X/Qs are of same order of magnitude as those calculated with a combination of the Frenkiel and recirculation models, but slightly smaller. However in several cases (primarily with the Site C and Site D data sets), the elimination of the Frenkiel model reduced 95th percentile X/Qs by more than an order of magnitude. In no case did a 95th percentile X/Q increase.

Elimination of the Frenkiel model significantly reduced the variability of 95th percentile X/Qs associated with changes in meteorological data sets. However, it should be noted that there is still considerable variability in the X/Qs for the B&W and ALWR Stage 3 vent releases. This variability may be associated with the release point/intake geometry. In both cases, the release point is below the intake and F_b and F_m are relatively small. The F_b and F_m for the CE releases are the same as for the B&W releases, but the CE release point is above the intake. The X/Qs for these releases show much less variability than the X/Qs for the B&W releases.

Table 6. ARCON00x Estimate of 95th Percentile X/Q Using Only the Recirculation Model

					System 80+ ALWR		
Meteor. Data	Year	West.	B&W	CE	Stage 1	Stage 2	Stage 3
Site A	1991	2.27E-06	1.64E-03	5.34E-05	6.48E-07	1.98E-06	4.17E-03
Site B	1988	1.85E-06	3.49E-04	4.95E-05	6.10E-07	1.66E-06	8.45E-04
Site C	1993	2.27E-06	2.97E-04	5.76E-05	5.72E-07	1.78E-06	3.43E-04
Site D	1995	2.38E-06	2.91E-04	5.97E-05	5.75E-07	1.74E-06	3.24E-05
Site E	1996	2.33E-06	3.54E-04	4.96E-05	6.51E-07	1.99E-06	1.19E-03

One more modification was made to the ARCON code. In this modification, the smaller of the transition plume rise (Equation 7) and the maximum plume rise was used in calculating the effective release height. The results of calculations with this version of the code are listed in Table 7. The 95th percentile X/Qs are the same as or slightly larger than those listed in Table 6. The largest increase is less than a factor of 2.

Table 7. ARCON00x Estimate of 95th Percentile X/Q Recirculation Model with Transition Plume Rise

					System 80+ ALWR		
Meteor. Data	Year	West.	B&W	CE	Stage 1	Stage 2	Stage 3
Site A	1991	2.32E-06	1.64E-03	5.34E-05	6.60E-07	1.98E-06	4.17E-03
Site B	1988	2.01E-06	4.85E-04	4.95E-05	6.20E-07	1.73E-06	8.45E-04
Site C	1993	2.33E-06	4.19E-04	5.76E-05	6.12E-07	1.81E-06	6.33E-04
Site D	1995	2.42E-06	4.07E-04	5.97E-05	6.12E-07	1.78E-06	4.44E-04
Site E	1996	2.33E-06	3.55E-04	4.96E-05	6.51E-07	1.99E-06	1.19E-03

MODEL SENSITIVITY TO MINIMUM WIND SPEED AND MAXIMUM WIND SPEED FOR RECIRCULATION

The ARCON version using the recirculation without transition plume rise was run for CE vent releases with the Site D data set to test the sensitivity of the 95th percentile X/Qs to variations in the minimum wind speed and the maximum wind speed for recirculation. The results of these calculations, which are summarized in Table 8, show limited sensitivity to variation of the minimum wind speed. They show almost no sensitivity to variation in the maximum speed for use of the recirculation model.

Table 8. Sensitivity of 95th Percentile X/Qs for CE Vent Releases to Variation of Minimum Wind Speed and Maximum Recirculation Wind Speed

	Maximum Recirculation Wind Speed (m/s)				
Minimum Wind Speed (m/s)	2.0	2.5	3.0	3.5	4.0
0.5	6.54E-05		6.54E-05		6.54E-05
1.0	5.97E-05	5.97E-05	5.97E-05	5.97E-05	5.99E-05
1.5	5.53E-05		5.53E-05		5.55E-05

A second set of calculations was run for B&W vent releases using the Site B data set. The results of these calculations, which are presented in Table 9, are the reverse of the calculations for the CE releases made with the Site D data set. There is a small variation of the 95th percentile X/Qs associated with variation of the maximum wind speed for use of the recirculation model and no change associated with variation of the minimum wind speed.

Table 9. Sensitivity of 95th Percentile X/Qs for B&W Vent Releases to Variation of Minimum Wind Speed and Maximum Recirculation Wind Speed.

	Maximum Recirculation Wind Speed (m/s)				
Minimum Wind Speed (m/s)	2.0	2.5	3.0	3.5	4.0
0.5	3.95E-04		3.49E-04		3.61E-04
1.0	3.95E-04	3.90E-04	3.49E-04	3.45E-04	3.61E-04
1.5	3.95E-04		3.49E-04		3.61E-04

The reasons for the different outcomes of these two sets of calculations need to be explored. However, it is clear that the model sensitivity to these two parameters is not great.

ISSUES TO BE RESOLVED

The preliminary evaluation of potential modifications of ARCON96 to enable the code to adequately handle high-velocity vent releases clearly indicates that such modifications are possible. It also indicates that there are a number of technical and regulatory issues that should be resolved before a new version of ARCON is produced. The following list contains the issues that come to mind at this time; there may be other issues that arise in time.

- Is it reasonable to use the recirculation model for all wind speeds below the upper limit for application of the model? If so, what wind speed should be used as a default in place of calm and nearly calm winds? (A default speed of 0.5 m/s seems reasonable.) Wind direction is not an issue for either this model or the Frenkiel model because the models maybe applied independent of direction.

- What is the appropriate upper wind speed for use of the recirculation model? (A speed in the 2-3 m/s range seems reasonable.)

- If the Frenkiel model is used, what is the appropriate upper wind speed for use of the Frenkiel model? (A speed in the 1-2 m/s range seems reasonable.)

- Because the recirculation model involves a search for the maximum X/Q, several questions arise related to the search. How precise should the estimate of the maximum X/Q be? Can the search be stopped when the maximum is bracketed and the probable error in X/Q is less than 1% or 5% of the highest calculated value? How far downwind should the search for the maximum value proceed? Can the search be terminated at 10 or 20 km if the peak hasn't been reached? If the search is terminated on the basis of distance, should the X/Q at the maximum distance be used?

- Is it reasonable to assume that plume rise for unstable conditions is at least as great as the rise in neutral conditions?

- Under some combinations of release and meteorological conditions, the limiting plume rise for stable conditions is slightly larger than the limiting rise for neutral conditions. In stable conditions, should the limiting rise for the neutral conditions be compared with the limiting rise for the stable condition, and the smaller rise be considered limiting?

- Is the use of transition plume rise in combination with maximum plume rise warranted? Or is the maximum plume rise sufficient?

- It sure would be nice to have some observational data on plume rise from MSSVs and ADVs for a variety of plants and meteorological conditions.

- How far above the highest building does the plume have to rise to be considered an elevated plume?

- If plume rise is calculated for MSSVs and ADVs, should it be routinely allowed for stack releases and other vents (low velocity)?

- Should an approximation to steam tables be included in ARCON, or is it appropriate to require ARCON users to enter the temperature and density of the steam after expansion to atmospheric pressure? Temperature is not included in the ARCON meteorological data set. It is in the NRC standard data format, but I don't recall seeing it very often. Is it acceptable to use an average air temperature and density in plume rise calculations?

- How should releases below intakes be treated if transition plume rise is included in the code? Should the recirculation model be used at all if the plume rise at the distance from the release point to the intake plus 3 (4, 5,6,...?) sigma z is less than the intake height minus the release height?

- Would it be appropriate to subject the approach to an external peer review before completion of the model and code development? If so, would it be appropriate to include industry representatives on the peer review panel?

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4. Industry's ultimate concern is that the current regulatory guidance of ignoring plume rise effects in modeling high-energy steam releases is generally too conservative. We appreciate the concern that the combination of the Briggs plume rise equation and the Gaussian atmospheric dispersion model may not calculate accurate estimates close to the source for high-energy releases. However, industry still supports developing either a mechanistic or deterministic method that recognizes the tradeoff between the maximum plume rise that occurs during low wind speed conditions and the maximum dilution that occurs during high wind speed conditions.

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Attachment 4

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