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REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

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METHODS AND ASSUMPTIONS FOR EVALUATING RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS AT LIGHT-WATER NUCLEAR POWER REACTORS

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PURPOSE AND SCOPE

A. INTRODUCTION

This guide provides guidance to licensees of operating power reactors on acceptable methods and assumptions for performing evaluations of fission product releases and radiological consequences of several postulated light-water reactor design basis accidents. It describes the radiological sources; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and the content of submittals acceptable to the NRC staff.

In 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 50.34, "Contents of Applications; Technical Information," requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility. Applicants are also required by 10 CFR 50.34 to provide an analysis of the proposed site. In 10 CFR Part 100, "Reactor Site Criteria," Section 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," provides criteria for evaluating the radiological aspects of the proposed site. A footnote to 10 CFR 100.11 states that the fission product release assumed in these evaluations should be based on a major accident involving substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

General Design Criterion (GDC-19), "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 establishes criteria for a control room and requires means for remote plant shutdown. GDC-19 also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures more than 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (Ref. 1), is cited in 10 CFR Part 100 as a source of further guidance on these analyses. Although initially used only for siting evaluations, the TID-14844 source term has been used in other design basis applications, such as environmental qualification of equipment under 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and in some requirements related to Three Mile Island (TMI) as stated in NUREG-0737, "Clarification of TMI Action Plan Requirements" (Ref. 2). The analyses and evaluations required by 10 CFR 50.34 for an operating license are documented in the facility's final safety analysis report (FSAR). Fundamental assumptions that are design inputs, including the source term, are to be included in the FSAR and become part of the facility design basis.[1]

[1] As defined in 10 CFR 50.2, *design bases* means information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculation or experiments or both) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. The NRC considers the accident source term to be an integral part of the design basis because it sets forth specific values (or a range of values) for controlling parameters that constitute reference bounds for design.

Since the publication of TID-14844, significant advances have been made in understanding the timing, magnitude, and chemical form of fission product releases from severe nuclear power plant accidents. A holder of an operating license issued prior to January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued prior to January 10, 1997, is allowed by 10 CFR 50.67, "Accident Source Term," to voluntarily revise the accident source term used in design basis radiological consequence analyses. This guide is not applicable to facilities that use the alternative source term as described in 10 CFR 50.67, "Accident Source Term." Guidance for the alternative source term is provided in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 3).

This regulatory guide does not apply to applicants for a construction permit, a design certification, or a combined license who do not reference a standard design certification and who applied after January 10, 1997, nor to licensees authorized to use an alternative source term (AST) under 10 CFR 50.67. These applicants and licensees are required by regulation to calculate offsite dose in units of total effective dose equivalent (TEDE). TEDE criteria are expected to be used with the AST and not with results calculated according to TID-14844 (Ref. 1). Therefore, because this guide pertains to the TID-14844 source terms and the corresponding whole body and thyroid criteria, it does not apply to applicants and licensees who are required to use the TEDE criteria.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget (OMB), approval number 3150-3011. The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

An accident source term is intended to be representative of a major accident involving significant core damage and is typically postulated to occur in conjunction with a large loss-of-coolant accident (LOCA). Although the LOCA is typically the maximum credible accident, NRC staff experience in reviewing license applications has indicated the need to consider other accident sequences of lesser consequence but higher probability of occurrence. The design basis accidents (DBAs) were not intended to be actual event sequences, but rather were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. Although probabilistic risk assessments (PRAs) can provide useful insights into system performance and suggest changes in how the desired defense in depth is achieved, defense in depth continues to be an effective way to account for uncertainties in equipment and human performance. The NRC's policy statement on the use of PRA methods (Ref. 4) calls for the use of PRA technology in all regulatory matters in a manner that complements the NRC's deterministic approach and supports the traditional defense-in-depth philosophy.

The NRC's traditional methods for calculating the radiological consequences of design basis accidents were described in a series of regulatory guides and Standard Review Plan (SRP) chapters. That guidance was developed to be consistent with the release fractions and timing from the TID-14844 source term and the whole body and thyroid doses stated in 10 CFR 100.11. The guidance contained in this regulatory guide will supersede corresponding radiological analysis assumptions provided in other regulatory guides when used in conjunction with guidance that is in Regulatory Guide 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors." The affected guides will not be withdrawn as they may still be used at the option of licensees. Specifically, the guidance in Regulatory Guides 1.195 and 1.196 can be used instead of the following regulatory guides:

Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors" (Ref. 5)

Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors" (Ref. 6)

Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors" (Ref. 7)

Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors" (Ref. 8)

Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors" (Ref. 9)

This guide primarily addresses design basis accidents, such as those addressed in Chapter 15 of typical final safety analysis reports. This guide does not address all areas of potentially significant risk. Although this guide addresses fuel handling accidents, other events that could occur during shutdown operations are not currently addressed. The NRC staff has several ongoing initiatives involving risks of shutdown operations, extended burnup fuels, and risk-informing current regulations. The information in this guide may be revised in the future as NRC staff evaluations are completed and regulatory decisions on these issues are made.

C. REGULATORY POSITION

1. GENERIC CONSIDERATIONS

1.1 Safety Margins

The proposed uses of this guide and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that sufficient safety margins are maintained, including a margin to account for analysis uncertainties. Changes, or the net effects of multiple changes, that result in a reduction in safety margins may require prior NRC approval. Licensees may use 10 CFR 50.59 and its supporting guidance to assess safety margins related to facility modifications and changes to procedures that are described in the Updated Final Safety Analysis Report.

1.2 Defense in Depth

The proposed uses of this guide and the associated proposed facility modifications and changes to procedures should be evaluated to determine whether the proposed changes are consistent with the principle that adequate defense in depth is maintained to compensate for uncertainties in accident progression and analysis data. Consistency with the defense-in-depth philosophy is maintained if system redundancy, independence, and diversity are preserved commensurate with the expected frequency, consequences of challenges to the system, and uncertainties. In all cases, compliance with the General Design Criteria in Appendix A to 10 CFR Part 50 is essential for facilities to which GDC criteria apply. Modifications proposed for the facility generally should not create a need for compensatory programmatic activities, such as reliance on manual operator actions, use of potassium iodide as a prophylactic drug, or self-contained breathing apparatus.

Proposed modifications that seek to downgrade or remove required engineered safeguards equipment should be evaluated to be sure that the modification does not invalidate assumptions made in facility PRAs and does not adversely impact the facility's severe accident management program.

The radiological analyses provide a fundamental basis upon which a significant portion of the facility design is based. Additionally, many aspects of facility operation derive from radiological design analyses. Radiological analyses generally should be based on assumptions and inputs that are consistent with corresponding data used in other design basis safety analyses, radiological and nonradiological, unless these data would result in nonconservative results or otherwise conflict with the guidance in this guide.

1.3 Scope of Required Analyses

1.3.1 Design Basis Radiological Analyses

A fundamental commitment required for application of the methodology in this guide is to perform an analysis of each applicable accident. The scope of accidents considered should include accidents mentioned in this guide applicable to a specific design (i.e., boiling-water reactors (BWRs) or pressurized-water reactors (PWRs)), supplemented by those in the FSAR and other licensee documents, as appropriate. Regulatory Positions 1.3.2 and 1.3.3 may be used to further define the scope and type of analyses performed.

The performance of these analyses will determine the limiting event with respect to offsite and control room dose. Some licensees have evaluated the control room dose only for the DBA LOCA, which is typically the limiting event for offsite radiological releases. The DBA LOCA is generally the large break (LB) LOCA event analysis. Other events may be analyzed as part of the design basis accident evaluation for the facility. Although these events may have been shown to be nonlimiting with respect to offsite dose, control room dose analyses for these events are needed to identify the limiting event for the GDC-19 control room dose design criterion.

There are several regulatory requirements for which compliance is demonstrated, in part, by the evaluation of the radiological consequences of design basis accidents. A plant's licensing bases may include, but are not limited to, the following.

- Environmental Qualification of Equipment (10 CFR 50.49)
- Control Room Habitability (GDC-19 of Appendix A to 10 CFR Part 50)
- Emergency Response Facility Habitability (Paragraph IV.E.8 of Appendix E to 10 CFR Part 50)
- Environmental Reports (10 CFR Part 51)
- Facility Siting (10 CFR 100.11)

There may be other areas in which the technical specification bases and various licensee commitments refer to specific evaluations. A plant's licensing bases may include, but are not limited to, the following from Reference 2, NUREG-0737.

- Post-Accident Access Shielding (NUREG-0737, II.B.2)
- Post-Accident Sampling Capability (NUREG-0737, II.B.3)
- Accident Monitoring Instrumentation (NUREG-0737, II.F.1)
- Leakage Control (NUREG-0737, III.D.1.1)
- Emergency Response Facilities (NUREG-0737, III.A.1.2)
- Control Room Habitability (NUREG-0737, III.D.3.4)

1.3.2 Re-Analysis Guidance

Facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the methodology utilized. The scope and extent of the re-evaluation will be a function of the specific proposed facility modification or the change in methodology. The NRC staff expects licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid. Generic analyses, such as those performed by owner groups or vendor topical reports, may be used provided the licensee justifies the applicability of the generic conclusions to the specific facility and implementation. Sensitivity analyses, discussed below, may also be an option. If affected design basis analyses are to be recalculated, all affected assumptions and inputs should be updated. Any license amendment request should describe the licensee's re-analysis effort and provide statements regarding the acceptability of the proposed implementation, including modifications, against each of the applicable analysis requirements and commitments identified in Regulatory Position 1.3.1 of this guide.

1.3.3 Use of Sensitivity or Scoping Analyses

It may be possible to demonstrate by sensitivity or scoping evaluations that existing analyses have sufficient margin and need not be recalculated. As used in this guide, a *sensitivity analysis* is an evaluation that considers how the overall results vary as an input parameter is varied. A *scoping analysis* is a brief evaluation that uses conservative, simple methods to show that the results of the analysis bound those obtainable from a more complete treatment. Sensitivity analyses are particularly applicable to suites of calculations that address diverse components or plant areas but are otherwise largely based on generic assumptions and inputs. Such cases might include postaccident vital area access dose calculations, shielding calculations, and equipment environmental qualification (integrated dose). It may be possible to identify a bounding case, re-analyze that case, and use the results to draw conclusions regarding the remainder of the analyses. If sensitivity or scoping analyses are used, the license amendment request should include a discussion of the analyses performed and the conclusions drawn. Scoping or sensitivity analyses should not constitute a significant part of the evaluations for the design basis exclusion area boundary (EAB), low population zone (LPZ), or control room dose unless there is a clear and defensible basis for doing so.

1.4 Risk Implications

This guide provides guidance only on the regulatory assumptions that licensees should make in their calculation of the radiological consequences of design basis accidents. These assumptions have no direct influence on the probability of the design basis accident initiator. These analyses assumptions cannot increase the core damage frequency (CDF) or the large early release frequency (LERF). However, facility modifications made possible by the use of this guide could have an impact on risk.

Consideration should be given to the risk impact of proposed implementations that seek to remove or downgrade the performance of previously required engineered safeguards equipment on the basis of the reduced postulated doses. The NRC staff may request risk information if there is a reason to question adequate protection of public health and safety.

The licensee may elect to use risk insights in support of proposed changes to the design basis that are not addressed in currently approved NRC staff positions. For guidance, refer to Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis” (Ref. 10).

1.5 Submittal Requirements

According to 10 CFR 50.90, an application for an amendment must fully describe the changes desired and follow, as far as applicable, the form prescribed for original applications. Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (Ref. 11), provides additional guidance. The NRC staff’s finding as to whether an amendment is to be approved or rejected is based in part on the licensee’s analyses, since it is these analyses that will become part of the design and licensing basis of the facility. The amendment request should describe the licensee’s analyses of the radiological and nonradiological impacts of the proposed modification in sufficient detail to support review by the NRC staff. The staff recommends that licensees submit affected FSAR pages annotated with changes that reflect the revised analyses or submit the actual calculation documentation.

If the licensee has used a current, approved version of an NRC-sponsored computer code, the NRC staff review can be made more efficient if the licensee identifies the code used and submits the inputs that the licensee used in the calculations made with that code. In many cases, this will reduce the need for NRC staff confirmatory analyses. This recommendation does not constitute a requirement that the licensee use NRC-sponsored computer codes.

1.6 Final Safety Analysis Report Requirements

Requirements for updating the facility's FSAR are in 10 CFR 50.71, "Maintenance of Records, Making of Reports." The regulations in 10 CFR 50.71(e) require that the FSAR be updated to include the effects of all changes made in the facility or procedures described in the FSAR and all safety analyses and evaluations performed by the licensee in support of approved license amendments or in support of conclusions that changes did not require a license amendment in accordance with 10 CFR 50.59. The affected radiological analysis descriptions in the FSAR should be updated to reflect the replacement of the design basis changes to the methodology and inputs. The analysis descriptions should contain sufficient detail to identify the methodologies used, significant assumptions and inputs, and numeric results. Regulatory Guide 1.70 (Ref. 11) provides additional guidance. The descriptions of superseded analyses should be removed from the FSAR in the interest of maintaining a clear design basis.

The NRC staff reviews licensee amendment requests to ensure the proposed change will maintain an adequate level of protection of public health and safety. The NRC staff accomplishes these reviews by evaluating the information submitted in the amendment request against the current plant design basis as documented in the FSAR, staff safety evaluation reports (SERs), regulatory guidance, other licensee commitments, and staff experience gained in approving similar requests for other plants. The NRC staff bases its finding that the amendment is acceptable on its assessment of the licensee's analysis, since it is the licensee's analysis that becomes part of the facility's design basis. Licensees should ensure that adequate information, including analysis assumptions, inputs, and methods, are presented in the submittal to support the staff's assessment. The NRC staff's assessment may include performance of independent analyses to confirm the licensee's conclusion. Licensees should expect an NRC staff effort to resolve critical differences in analysis assumptions, inputs, and methods used by the licensee and those deemed acceptable to the NRC staff.

2. DOSE ANALYSIS MODELS

2.1 Radiological Consequences

The radiological consequences of an accident in a nuclear reactor depend on the quantity of the radioactive material that escapes to the environment or enters the control room. As the radioactivity is transported through the containment and other buildings, credit is given for several natural and engineered removal mechanisms. Within compartments, these removal mechanisms include sprays, natural deposition, leakage, natural and forced convection, filters, and suppression pools. This Regulatory Position describes an acceptable set of general equations used to model the transport and removal of fission products between compartments, the calculation of activities in the environment, and the calculation of offsite and compartment doses.

The equations contained within this Regulatory Position do not model the impact of daughter products (e.g., I-135 to Xe-135) that are due to the decay of the parent isotopes. Daughter products do not typically contribute significantly to the dose. If it is determined that they contribute significantly to the whole body, thyroid, or beta doses, daughter products should be considered and the equations provided in Regulatory Position 2 will need to be modified.

2.2 Activities in Compartments Without Inflow From the Environment

The following balance equation models the rate of change of activity of a nuclide in a compartment r. An example of a compartment that is typically without inflow from the environment is a reactor containment building.

$$\frac{da_r}{dt} = \sum_{\substack{k=1 \\ k \neq r}}^{M} \lambda_{tkr} a_k - \sum_{j=1}^{N} \lambda_{rj} a_r \tag{1}$$

where:

a_r	=	activity of a nuclide in compartment r at time t, Ci
a_k	=	activity of a nuclide in compartment k at time t, Ci
N	=	the number of removal processes
M	=	the number of compartments modeled
λ_{rj}	=	removal constant of the j^{th} removal process internal to compartment r, i.e., decay, plateout, filtration, spray in containment, flow out of compartment r, sec^{-1}
λ_{tkr}	=	transfer constant from region k to compartment r, i.e., flow rate from compartment k to compartment r divided by the volume of compartment k, sec^{-1}

For halogens, a more specific form of Equation 1 may be written to account for removal mechanisms that are chemically species-specific, e.g., filter efficiencies for particulate, elemental, and organic iodine. For these situations, Equation 1 can be rewritten to define the activity and removal constants on a per nuclide and species basis.

2.3 Activities in the Environment

Equation 1 is solved for the time-dependent activity in each compartment. The release rate from M compartments to the environment is given by Equation 2. The activity in the environment from each compartment is given by Equation 3.

$$R = \sum_{k=1}^{M} R_{ke} \tag{2}$$

$$R_{ke} = \frac{1}{V_k}\left[Q_{keF}\left(1 - f_{ke}\right) + Q_{keU}\right]a_k \tag{3}$$

where:

a_k = activity of compartment k, Ci
f_{ke} = filter removal efficiency fraction for a filter between compartment k and the environment, dimensionless
Q_{keF} = filtered flow from compartment k to the environment, m^3/sec
Q_{keU} = unfiltered flow from compartment k to the environment, m^3/sec
R = release rate of activity from M compartments to the environment, Ci/sec
R_{ke} = release rate of activity from compartment k to the environment, Ci/sec
V_k = free volume of compartment k, m^3

2.4 Activities in Compartments That Intake Only Outside Contaminated Air

Equation 4 models compartments that intake radioactivity transported to the compartment via only atmospheric dispersion. Control rooms or technical support centers that do not intake radioactivity directly from other buildings are examples of these compartments.

$$\frac{da_r}{dt} = \left[Q_{erF}(1 - f_{er}) + Q_{erU}\right]\sum_{\substack{k=1 \\ k \neq r}}^{M}\left(\chi / Q\right)_{ker} R_{ke} - \sum_{j=1}^{N}\lambda_{rj}a_r \tag{4}$$

where:

f_{er} = filter removal fraction for a filter between the environment and compartment r, dimensionless
Q_{erF} = filtered intake flow rate from the environment to compartment r, m^3/sec
Q_{erU} = unfiltered intake flow rate from the environment to the compartment r, m^3/sec
$(\chi/Q)_{ker}$ = atmospheric dispersion factor from compartment k to intake of compartment r, sec/m^3

Examples of removal process (λ_{rj}) typically modeled for control rooms are given below:

λ_{r1} = exhaust rate from the control room to the environment, $sec^{-1} = Q_E/V_r$, where Q_E is the exhaust flow rate from the control room to the environment ($Q_E = Q_{erF} + Q_{erU}$), m^3/sec, and V_r is the free volume of the control room, m^3
λ_{r2} = nuclide decay constant, sec^{-1}
λ_{r3} = recirculation removal rate, $sec^{-1} = (Q_R/V_r) \times f_R$ where Q_R is the recirculation flow rate in the control room, m^3/sec; V_r is the free volume of the control room, m^3; and f_R is the recirculation filter removal efficiency fraction, dimensionless.

2.5 Integrated Activity Released Into the Environment

The integrated activity (Curies) released into the environment over the time interval j from time t_0 to t_1, IAR_j, is given by the following equation. In calculating IAR_j, no credit is taken for cloud depletion by ground deposition or radioactive decay during transit to the exclusion area boundary or the LPZ outer boundary.

$$IAR_j = \int_{t_0}^{t_1} R\,dt \tag{5}$$

2.6 Integrated Activity in a Compartment

The integrated activity (Ci-sec) in a compartment k over the time interval j from time t_0 to t_1, IA_{kj}, is determined by the expression:

$$IA_{kj} = \int_{t_0}^{t_1} a_k\,dt \tag{6}$$

2.7 Offsite Doses

The following equations give the models used to calculate offsite doses. Equations for calculating thyroid and whole body doses are given.

Offsite thyroid doses are calculated using the equation:

$$D_{TH} = \sum_{i=1}^{N} (DCF_{TH})_i \sum_{j=1}^{T} (IAR)_{ij} (BR)_j (\chi/Q)_j \tag{7}$$

Assuming a semi-infinite cloud of photon emitters, offsite whole body doses are calculated using the equation:

$$D_{\gamma B} = \sum_{i=1}^{N} (DCF_{\gamma B})_i \sum_{j=1}^{T} (IAR)_{ij} (\chi/Q)_j \tag{8}$$

where:

$(BR)_j$	=	breathing rate during time interval j, m³/sec
D_{TH}	=	offsite thyroid dose via inhalation during time interval j, rem
$D_{\gamma B}$	=	offsite whole body dose during time interval j, rem
$(DCF_{TH})_i$	=	thyroid dose conversion factor via inhalation for nuclide i, rem/Ci
$(DCF_{\gamma B})_i$	=	photon body dose conversion factor for nuclide i, rem-m³/Ci-sec
$(IAR)_{ij}$	=	integrated activity of nuclide i released during the time interval j, Ci
N	=	number of nuclides
T	=	number of time intervals over which (IAR) is calculated
$(\chi/Q)_j$	=	offsite atmospheric dispersion factor during time interval j, sec/m³

2.8 Compartment Doses

Compartment thyroid doses via inhalation pathway are calculated using the following equation:

$$(D_{TH})_k = \frac{1}{V_k}\sum_{i=1}^{N}(DCF_{TH})_i\sum_{j=1}^{T}(IA_k)_{ij}O_j(BR)_j \qquad (9)$$

Because of the finite size of a compartment, the whole body photon doses in a compartment caused by the radioactive cloud will be substantially less than the doses caused by immersion in an infinite cloud of photon emitters. The finite cloud photon doses are calculated using Murphy's method, which models the compartment as a hemisphere. The following equation is used:

$$(D_{\gamma B})_k = \frac{1}{GF_k V_k}\sum_{i=1}^{N}(DCF_{\gamma B})_i\sum_{j=1}^{T}(IA_k)_{ij}O_j \qquad (10)$$

The beta skin doses in a compartment are calculated using the following equation:

$$(D_{\beta S})_k = \frac{1}{V_k}\sum_{i=1}^{N}(DCF_{\beta S})_i\sum_{j=1}^{T}(IA_k)_{ij}O_j \qquad (11)$$

where:

BR_j	=	breathing rate in time interval j, m^3/sec
$(D_{TH})_k$	=	compartment k thyroid dose via inhalation, rem
$(D_{\gamma B})_k$	=	compartment k whole body dose, rem
$(D_{\beta S})_k$	=	compartment k beta skin dose, rem
$(DCF_{\beta S})_i$	=	beta skin dose conversion factor for nuclide i, rem-m³/Ci-sec
GF_k	=	dose reduction due to the compartment geometry correction factor $352/V_k^{0.338}$, or if V_k is defined in units of cubic feet the geometry factor is $1173/V_k^{0.338}$, dimensionless[2] (see Regulatory Position 4.2.7)
$(IA_k)_{ij}$	=	integrated activity concentration in compartment k, for nuclide i during time interval j, Ci-sec
O_j	=	compartment occupancy fraction during time interval j
T	=	number of time intervals over which (IA) is calculated
V_k	=	compartment k free volume, m^3

In addition to the dose contribution from the compartment airborne activity described above, the whole body dose contribution from external sources as listed in Regulatory Position 4.2.1 should be considered.

[2] Control room envelopes may be composed of more than one room or subcompartment. If those rooms contain shielding that blocks the majority (99% or greater) of whole body dose outside the room, the geometry factor is calculated using the free volume of the largest subcompartment.

3. ACCIDENT SOURCE TERM

This Regulatory Position 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 for 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, currently licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the emergency core cooling system (ECCS) evaluation uncertainty.[3] These parameters should be examined to maximize fission product inventory. 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 ORIGEN2 (Ref. 12) or ORIGEN-ARP (Ref. 13). Core inventory factors (Ci/MWt) provided in TID-14844 (Ref. 1) 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.[4] Further assumptions are in Appendix A to this guide. 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 shut down, e.g., a fuel handling accident, radioactive decay from the time of shutdown may be modeled.

3.2 Release Fractions

The core inventory release fractions, by radionuclide groups, for DBA LOCAs and non-LOCA DBAs where the fuel is melted and the cladding is breached are listed in Table 1 for BWRs and PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1.

For non-LOCA events, where only the cladding is breached, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 2. The release

[3] 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 1.00 (correlates to the licensed power level) may be used provided the proposed alternative value has been demonstrated to account for uncertainties caused by power level instrumentation error.

[4] Some plants evaluate the radiological consequences of a reactor head drop accident. For these analyses it is appropriate to use the core average inventory to assess the consequences of this accident.

fractions from Table 2 are used in conjunction with the calculated fission product inventory and the maximum core radial peaking factor.

Table 1[5]
BWR and PWR Core Inventory Fraction Released Into Containment Atmosphere

Group	Release Fraction
Noble Gases	1.0
Iodines[6]	0.5

Table 2[7]
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 Iodines	0.05

3.3 Timing of Release Phases

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

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

[6] If wall deposition by containment sprays is modeled as a time-dependent process, such as in Revision 2 of Standard Review Plan Section 6.5.2 (Ref. 14), 50% of the equilibrium radioactive iodine inventory is assumed released into the containment atmosphere and is available to be deposited on the walls on the containment. Elemental plateout using the time-dependent models in Revision 2 of Standard Review Plan Section 6.5.2 may be assumed.

If wall deposition by containment sprays is modeled as a time-independent process (instantaneous removal), such as in Revision 0 of Standard Review Plan Section 6.5.2, 50% of the equilibrium, radioactive iodine inventory is assumed released into the containment atmosphere. One-half of this iodine is assumed to be instantaneously deposited on the walls of the containment. The net value of core inventory available for release from containment would, therefore, be 25% of the equilibrium radioactive iodine inventory. Further iodine removal by wall deposition should not be assumed. For these assumptions a limitation of 10 hr^{-1} should be imposed on the elemental iodine spray removal lambda.

Please note that Revision 2 of SRP Section 6.5.2 erroneously implied that 25% 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 50% of the equilibrium radioactive iodine inventory when time-dependent wall deposition by spray is assumed. Using 50% prevents accounting twice for the iodine deposited on the wall of the containment.

[7] 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 rods with burnups that exceed 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.

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
Iodines	I

3.5 Chemical Form

Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 5% of the iodine released should be assumed to be particulate iodine, 91% elemental iodine, and 4% organic iodide. This includes releases from the gap and the fuel pellets. The same chemical form is assumed in releases from fuel pins in fuel handling accidents (FHAs) and from releases from the fuel pins through the reactor coolant system 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 A through H 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.

For the postulated main steam line break, steam generator tube rupture, and locked rotor accidents, the amount of fuel damage should be evaluated assuming that the highest worth control rod is stuck at its fully withdrawn position.

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

4. DOSE CALCULATIONAL METHODOLOGY

This Regulatory Position provides a dose calculational methodology that is acceptable to the NRC staff. It provides guidance on the calculation of offsite and onsite consequences and on offsite and control room acceptance criteria.

4.1 Offsite Dose Consequences

The following assumptions should be used in determining the doses for persons located at or beyond the boundary of the exclusion area (EAB):

4.1.1 The dose calculations should determine the thyroid and whole body doses.

4.1.2 The exposure-to-thyroid factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 15). Table 2.1 of Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 16), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "thyroid" should be used.[8]

4.1.3 For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.

4.1.4 The whole body doses should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 17), provides external conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses correspond to the whole body dose. The use of effective dose-conversion factors (DCFs) as a surrogate for whole body DCFs is appropriate because of the uniform body exposure associated with semi-infinite cloud dose modeling.

4.1.5 The whole body and thyroid doses should be determined for an individual at the most limiting EAB location. The maximum EAB dose for the first 2 hours following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria given in Table 4.

4.1.6 The whole body and thyroid doses should be determined for the most limiting receptor at the outer boundary of the LPZ and should be used in determining compliance with the dose criteria in Table 4.

4.1.7 No correction should be made for depletion of the effluent plume by deposition on the ground.

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[8] Licensees who use these dose conversion factors in accident calculations should determine whether this will impact the facility's technical specification definition for dose equivalent I-131.

Table 4
EAB and LPZ Accident Dose Criteria

Accident or Case	Dose Criteria (rem) Whole Body	Thyroid	Analysis Release Duration
LOCA	25 rem	300 rem	30 days for containment, ECCS, and MSIV (BWR) leakage
BWR Main Steam Line Break			Instantaneous puff
Fuel Damage or Pre-Accident Spike	25 rem	300 rem	
Equilibrium Iodine Activity	2.5 rem	30 rem	
BWR Rod Drop Accident	6.3 rem	75 rem	24 hours
PWR Steam Generator Tube[9] Rupture			Affected SG: time to isolate; Unaffected SG(s): until cold shutdown is established
Fuel Damage or Pre-Accident Spike	25 rem	300 rem	
Coincident Iodine Spike	2.5 rem	30 rem	
PWR Main Steam Line Break[9]			Until cold shutdown is established
Fuel Damage or Pre-Accident Spike	25 rem	300 rem	
Coincident Iodine Spike	2.5 rem	30 rem	
PWR Locked Rotor Accident	2.5 rem	30 rem	Until cold shutdown is established
PWR Rod Ejection Accident	6.3 rem	75 rem	30 days for containment pathway; until cold shutdown is established for secondary pathway
Fuel Handling Accident	6.3 rem	75 rem	2 hours

4.2 Control Room Dose Consequences

The following guidance should be used in determining the whole body, thyroid, and skin doses for persons located in the control room envelope.

4.2.1 The whole body, thyroid, and skin dose analyses should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include:

[9] For PWRs with steam generator alternative repair criteria, different dose criteria may apply to steam generator tube rupture and main steam line break analyses as suggested by guidance being developed in Draft Regulatory Guide DG-1074, "Steam Generator Integrity" (Ref. 18).

- Contamination of the control room envelope atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility,

- Contamination of the control room envelope atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope.

4.2.2 The radioactive material releases and radiation levels used in the control room envelope dose analysis should be determined using the same source term, in-plant transport, and release assumptions used for determining the EAB and the LPZ dose values, unless these assumptions would result in nonconservative results for the control room envelope.

4.2.3 The models used to transport radioactive material into and through the control room envelope,[10] and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.

4.2.4 Credit for engineered safety features that mitigate airborne radioactive material within the control room envelope may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 14); Regulatory Guide 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants" (Ref. 22); and Generic Letter 99-02 (Ref. 23) for guidance. The control room envelope design is often optimized for the DBA LOCA and the protection afforded for other accident sequences may not be as advantageous. In most designs, control room isolation is actuated by engineered safeguards feature (ESF) signals or radiation monitors (RMs). In some cases, the ESF signal is effective only for selected accidents, placing reliance on the RMs for the remaining accidents. Several aspects of RMs can delay the control room isolation, including the delay for activity to build up to concentrations equivalent to the alarm setpoint and the effects of different radionuclide accident isotopic mixes on monitor response.

4.2.5 Credit should generally not be taken for the use of personal protective equipment or use of potassium iodide (KI) as a thyroid prophylactic drug.

4.2.6 The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room envelope for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days

[10] The iodine protection factor (IPF) methodology of Reference 19 may not be adequately conservative for all DBAs and control room arrangements since it models a steady-state control room condition. Since many analysis parameters change over the duration of the event, the IPF methodology should only be used with caution. The NRC computer codes HABIT (Ref. 20) and RADTRAD (Ref. 21) incorporate suitable methodologies.

to 30 days.[11] For the duration of the event, the breathing rate of this individual should be assumed to be 3.5 x 10^{-4} cubic meters per second (Ref. 25).

4.2.7 Control room envelope doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The calculation should consider all radionuclides that are significant with regard to dose consequences and the release of radioactivity. The whole body dose from photons may be corrected for the difference between finite cloud geometry in the control room envelope and the semi-infinite cloud assumption used in calculating the dose conversion factors using a compartment geometry correction factor. This factor is incorporated in Equation 10 of Regulatory Position 2.8. This correction is not applied to the beta skin dose estimates, as the range of beta particles in air is less than the typical control room dimensions. The skin dose DCFs presented in Federal Guidance Report 12 (Ref. 17) are based on both photon and beta emissions. Doses should be calculated using the factors in the column headed "Skin" in Table III.1 of Federal Guidance Report 12.

4.3 Other Dose Consequences

The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2).

4.4 Offsite Acceptance Criteria

The radiological criteria for the EAB and the outer boundary of the LPZ are given in 10 CFR 100.11. These criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break LOCA. For events with a higher probability of occurrence, postulated EAB and LPZ doses should not exceed the criteria tabulated in Table 4. The criteria provided in Table 4 are the same criteria provided in the Standard Review Plan (Ref. 14).

4.5 Control Room Acceptance Criteria

The following guidelines may be used in lieu of those provided in SRP 6.4 (Ref. 14) when showing compliance with the dose guidelines in GDC-19 of Appendix A to 10 CFR Part 50. The following guidelines relax the thyroid and skin acceptance criteria from that given in SRP 6.4. Currently, 10 CFR 20.1201 limits organ dose to 50 rem annually. The release duration is specified in Table 4. The exposure period is 30 days for all accidents. The criterion in GDC-19 applies to all accidents.

[11] These occupancy factors are already included in the determination of the χ/Q values using the Murphy and Campe methodology (Ref. 19) and should not be credited twice. The ARCON96 Code (Ref. 24) does not incorporate these occupancy factors in the determination of the χ/Q values. Therefore, when using ARCON96 χ/Q values, occupancy factors should be included in the dose calculations.

Whole body	5 rem
Thyroid	50 rem
Skin	50 rem[12]

4.6 Other Dose Consequences Acceptance Criteria

The acceptance criteria for the various NUREG-0737 (Ref. 2) items generally reference GDC-19 from Appendix A to 10 CFR Part 50 or specify criteria derived from GDC-19. These criteria remain unchanged except for the thyroid and beta dose limits as stated in Regulatory Position 4.5.

Before the General Design Criteria were established in 10 CFR Part 50, these criteria existed in draft form. Some of the facilities that were licensed during this time period committed to various draft criterion for control room habitability. These commitments may be different from GDC-19. Application of this regulatory guide to those plants will be considered on a case by case basis.

5. ANALYSIS ASSUMPTIONS AND METHODOLOGY

5.1 General Considerations

5.1.1 Analysis Quality

The analyses required by 10 CFR 100.11 and GDC-19 in Appendix A to 10 CFR Part 50 and any re-analyses of these analyses required by 10 CFR 50.34 are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50.

These design basis analyses were structured to provide a conservative set of assumptions to test the performance of one or more aspects of the facility design. Many physical processes and phenomena are represented by conservative, bounding assumptions rather than modeled directly. The staff has selected assumptions and models that provide an appropriate and prudent safety margin against unpredicted events in the course of an accident and compensate for large uncertainties in facility parameters, accident progression, radioactive material transport, and atmospheric dispersion. Licensees should exercise caution in proposing deviations based on data from a specific accident sequence since the DBAs were never intended to represent any specific accident sequence—the proposed deviation may not be conservative for other accident sequences.

5.1.2 Credit for Engineered Safeguard Features

Credit may be taken for accident mitigation features that are classified as safety related, are required to be operable by technical specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly

[12] Credit for the beta radiation shielding afforded by special protective clothing and eye protedtion is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. These limits are design criteria and are not to be interpreted as acceptable occupational doses.

addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences. Design basis delays in actuation of these features should be considered, especially for features that rely on manual operator intervention.

5.1.3 Assignment of Numeric Input Values

The numeric values that are chosen as inputs to the dose analyses required by regulations and described in Regulatory Position 5.1.1 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be nonconservative in another portion of the same analysis. For example, assuming minimum containment system spray flow is usually conservative for estimating iodine scrubbing, but in many cases may be nonconservative when determining sump pH. Sensitivity analyses may be needed to determine the appropriate value to use. As a conservative alternative, the limiting value applicable to each portion of the analysis may be used in the evaluation of that portion. A single value may not be applicable for a parameter for the duration of the event, particularly for parameters affected by changes in density. For parameters addressed by technical specifications, the value used in the analysis should be that specified in the technical specifications.[13] If a range of values or a tolerance band is specified, the value that would result in a conservative postulated dose should be used. If the parameter is based on the results of less frequent surveillance testing, e.g., steam generator nondestructive testing (NDT), consideration should be given to the degradation that may occur between periodic tests in establishing the analysis value.

5.2 Accident-Specific Assumptions

The appendices to this regulatory guide provide accident-specific assumptions that are acceptable to the staff for performing analyses that are described in Regulatory Position 5.1.1. Licensees should review their license basis documents for guidance pertaining to the analysis of radiological design basis accidents other than those provided in this guide. Licensees should analyze the DBAs that are affected by the specific proposed changes to the facility or to the radiological analyses.

The NRC staff has determined that the analysis assumptions in the appendices to this guide provide an integrated approach to performing the individual analyses and generally expects licensees to address each assumption or to propose acceptable alternatives. Such alternatives may be justifiable on the basis of plant-specific considerations or updated technical analyses. Although licensees are free to propose alternatives to these assumptions for consideration by the NRC staff, licensees should avoid use of previously approved staff positions that would adversely affect the consistency among the assumptions in this guide.

The NRC is committed to using probabilistic risk analysis (PRA) insights in its regulatory activities and will consider licensee proposals for changes in analysis assumptions that reflect risk

[13] Note that for some parameters, the technical specification value may be adjusted for analysis purposes by factors provided in other regulatory guidance. For example, ESF filter efficiencies are based on the guidance in Regulatory Guide 1.52 (Ref. 22) and in Generic Letter 99-02 (Ref. 23) rather than the surveillance test criteria in the technical specifications. Generally, these adjustments address possible changes in the parameter between scheduled surveillance tests.

insights. The staff will not approve proposals that would reduce the defense in depth deemed necessary to provide adequate protection for public health and safety. In some cases, this defense in depth compensates for uncertainties in the PRA analyses and addresses accident considerations not adequately addressed by the core damage frequency (CDF) and large early release frequency (LERF) surrogate indicators of overall risk.

5.3 Meteorology Assumptions

Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide provided such values remain relevant to the particular accident, its release points, and receptor location. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3, 1.4, and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" (Refs. 5, 6, and 26), and in the Murphy-Campe paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19" (Ref. 19).

Regulatory Guide 1.145 (Ref. 26) and Regulatory Guide 1.XXX, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants" (Ref. 27) should be used if the FSAR χ/Q values are to be revised or if values are to be determined for new release points or receptor distances. For stack releases, fumigation should be considered where applicable for the EAB and LPZ. For the EAB, the assumed fumigation period should be timed to be included in the 2-hour time period. The NRC computer code PAVAN (Ref. 28) implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. Regulatory Guide 1.XXX provides guidance on determining control room χ/Q values. The NRC computer code ARCON96 (Ref. 24) may be used in determining control room χ/Q values. Meteorological data collected in accordance with the site-specific meteorological measurements program described in the facility FSAR should be used in generating accident χ/Q values. Additional guidance is provided in Regulatory Guide 1.23, "Onsite Meteorological Programs" (Ref. 29).

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide. No backfitting is intended or approved in connection with the issuance of this guide.

Except when an applicant or licensee proposes an acceptable alternative method for complying with the specified portions of the NRC's regulations, the methods described in this guide will be used by the NRC staff in the evaluation of radiological consequences of design basis accidents at light-water nuclear power reactors for which the construction permit or license application is docketed after the issue date of this guide and at plants for which the licensee voluntarily commits to the provisions of this guide.

REFERENCES

{See the inside front cover of this guide for information on obtaining NRC documents.}

1. J.J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," USAEC TID-14844, U.S. Atomic Energy Commission (now USNRC), 1962.

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

3. USNRC, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183, July 2000.

4. USNRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal Register*, Volume 60, page 42622 (60 FR 42622) August 16, 1995.

5. USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.3, Revision 2, June 1974.

6. USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.4, Revision 2, June 1974.

7. USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors," Regulatory Guide 1.5, March 1971.

8. USNRC, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," Regulatory Guide 1.25, March 1972.

9. USNRC, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Regulatory Guide 1.77, May 1974.

10. USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.

11. USNRC, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.70, Revision 3, November 1978.

12. A.G. Croff, "A User's Manual for the ORIGEN2 Computer Code," ORNL/TM-7175, Oak Ridge National Laboratory, July 1980.

13. S.M. Bowman and L.C. Leal, “The ORIGENARP 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.

14. USNRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” NUREG-0800, September 1981 (or updates of specific sections).

15. ICRP, “Limits for Intakes of Radionuclides by Workers,” ICRP Publication 30, 1979.

16. K.F. Eckerman et al., “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

17. K.F. Eckerman and J.C. Ryman, “External Exposure to Radionuclides in Air, Water, and Soil,” Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

18. USNRC, “Steam Generator Tube Integrity,” Draft Regulatory Guide DG-1074, December 1998.

19. K.G. Murphy and K.W. Campe, “Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19,” published in *Proceedings of 13th AEC Air Cleaning Conference*, Atomic Energy Commission (now USNRC), August 1974.

20. USNRC, “Computer Codes for Evaluation of Control Room Habitability (HABIT V1.1),” Supplement 1 to NUREG/CR-6210, November 1998.

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

22. USNRC, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.52, Revision 3, June 2001.

23. USNRC, “Laboratory Testing of Nuclear-Grade Activated Charcoal,” NRC Generic Letter 99-02, June 3, 1999.

24. J.V. Ramsdell and C.A. Simonen, “Atmospheric Relative Concentrations in Building Wakes,” NUREG-6331, Revision 1, USNRC, May 1997.

25. ICRP, “Report of Committee II on Permissible Dose for Internal Radiation,” ICRP Publication 2, 1959.

26. USNRC, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Regulatory Guide 1.145, Revision 1, November 1982.

27. USNRC, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Regulatory Guide 1.XXX, [issue date].

28. T.J. Bander, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG-2858, USNRC, November 1982.

29. USNRC, "Onsite Meteorological Programs," Regulatory Guide 1.23, February 1972.

Appendix A

ASSUMPTIONS FOR EVALUATING THE RADIOLOGICAL CONSEQUENCES OF LWR LOSS-OF-COOLANT ACCIDENTS

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 emergency core cooling system 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.

1. SOURCE TERM ASSUMPTIONS

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

2. ASSUMPTIONS ON TRANSPORT IN PRIMARY CONTAINMENT

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.

2.1 At the start of the accident, 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. 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.

2.2 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).

2.3 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[1]

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[1] Footnote 6 of Regulatory Position 3.2 provides further details concerning assumptions applicable for crediting spray removal.

(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 region volume 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. 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 maximum decontamination factor (DF) for elemental iodine is based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine in the containment atmosphere remaining in equilibrium with the dissolved iodine in the containment water. This equilibrium is determined by the effective iodine partition coefficient. If the methodology in Revision 0 of Chapter 6.5.2 of the SRP (Ref. A-1) is used, the maximum iodine activity in primary containment atmosphere is the iodine activity, described in Regulatory Position 3.2, before the application of the 50% reduction assumed instantaneously deposited on the walls of containment.

2.4 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-2 and A-3).

2.5 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.

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

2.7 The primary containment (e.g., 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.

2.8 If the primary containment is 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. Iodine spikes need not be considered.

3. ASSUMPTIONS ON DUAL CONTAINMENTS

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.

3.1 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 2 1/2 times the height of any adjacent structure.

3.2 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.

3.3 The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on a case-by-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 either 5% or 95% of the total number 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% of the time).

3.4 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.

3.5 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.

3.6 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-2) and Generic Letter 99-02 (Ref. A-3).

4. ASSUMPTIONS ON ESF SYSTEM LEAKAGE

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-4). 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.

4.1 It is assumed that 50% of the core iodine inventory, based on the maximum reactor power level, is mixed instantaneously and homogeneously in the primary containment sump water (in PWRs) or the 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.

4.2 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-5), would require declaring such systems out of service. The factor of two multiplier is used to account for increased leakage in these systems over the duration of the accident and between surveillances or leakage checks. 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.

4.3 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.

4.4 If the temperature of the leakage is less than 212°F or the calculated FF 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.

4.5 The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. 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-2) and Generic Letter 99-02 (Ref. A-3).

5. ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRS

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.

5.1 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 Assumption 2 of this appendix). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.

5.2 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.

5.3 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.

5.4 In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in Section 5.5 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.

5.5 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, including the treatment of air ejector effluent by offgas systems, 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-6 and A-7 provide guidance on acceptable models.

6. ASSUMPTION ON CONTAINMENT PURGING

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-2) and Generic Letter 99-02 (Ref. A-3).

Appendix A REFERENCES

A-1 USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981 (or updates of specific sections).

A-2 USNRC, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 3, June 2001.

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

A-4 USNRC, "Potential Radioactive Leakage to Tank Vented to Atmosphere," Information Notice 91-56, September 19, 1991.

A-5 USNRC, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

A-6 J.E. Cline, "MSIV Leakage Iodine Transport Analysis," Report, March 26, 1991. (ADAMS Accession Number ML003683718)

A-7 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 ML003683734, NUDOCS 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 iodines.

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

1.4 The radioactive material available for release is assumed to be from the assemblies with the peak inventory. The fission product inventory for the peak assembly represents an upper limit value. The inventory should be calculated assuming the maximum achievable operational power history at the end of core life immediately preceding shutdown. This inventory calculation should include appropriate assembly peaking factors.

2. WATER DEPTH

If the depth of water above the damaged fuel is 23 feet or greater, the decontamination factor for the elemental and organic species are 400 and 1, respectively, giving an overall effective decontamination factor (DF) of 200 (i.e., 99.5% of the total iodine released from the damaged rods is retained by the water). If the depth of water is not at least 23 feet, the DF will have to be determined on a case-by-case basis (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 1,200 psig, the iodine DFs will be less than those assumed in this guide and must be calculated on a case-by-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).

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. The release rate is generally assumed to be a linear or exponential function over this 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.

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[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.

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 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 for the isolated pathway.

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. The release rate is generally assumed to be a linear or exponential function over this 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] Technical specifications that allow such operations usually include administrative controls to close the airlock, hatch, or open penetrations within 30 minutes. Such adminstrative controls 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 NUDOCS in NRC’s Public Document Room.)

B-2 USNRC, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.52, Revision 3, June 2001.

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 in Regulatory Position 3 of this guide. The activity released from the breached, but unmelted, 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 activity release attributed to fuel melting should be based upon the fraction of the fuel material that reaches or exceeds fuel melting temperature. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines contained in that fuel material 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 a pre-accident spike of 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 and/or from the 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 and 10% of the iodine are assumed to reach the turbine and condensers.

3.4 Of the activity that reaches the turbine and condensers, 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 for dilution or holdup within the turbine building. Radioactive decay during holdup in the turbine and condenser may be assumed.

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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 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 mechanical 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 offgas or standby gas treatment, will be considered on a case-by-case basis.

3.5 In lieu of the transport assumptions provided in Sections 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 valve (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. The release from the turbine and condenser should be assumed to be 97% elemental and 3% 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.

1. SOURCE TERM

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.

1.1 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.

1.1.1 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

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

1.2 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.

2. TRANSPORT

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

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

2.2 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.

[1] Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. 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.

2.3 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.

2.4 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]

1. SOURCE TERM

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.

1.1 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.

1.1.1 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 at full power operations (i.e., a pre-accident iodine spike case).

1.1.2 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 pins assumed to have defects.

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

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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] Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. 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.

1.3 The specific activity in the steam generator liquid at the onset of the SGTR should be assumed to be at the maximum value permitted by secondary activity technical specifications (typically 0.1 µCi/gm DE I-131).

1.4 Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.

2. TRANSPORT

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

2.1 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.

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

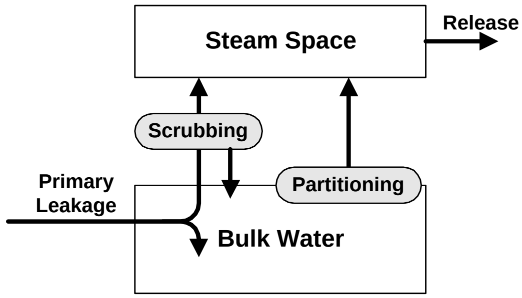
2.3 The primary-to-secondary leakage should be assumed to continue until the primary system pressure is less than the secondary system pressure, or until the temperature of the leakage is less than 100°C (212°F). The release of radioactivity from the unaffected steam generators should be assumed to continue until shutdown cooling is in operation and releases from the steam generators have been terminated. The release of radioactivity from the affected steam generator should be assumed to continue until shutdown cooling is operating and releases from the steam generator have been terminated, or the steam generator is isolated from the environment such that no release is possible, whichever occurs first.

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

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

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

Figure E-1
Transport Model



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

- During periods of steam generator dryout, all of the primary-to-secondary leakage is assumed to flash to vapor and be released to the environment with no mitigation.

- 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. During periods of uncovery, a flash fraction should be determined.

2.6.2 The 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.

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

2.6.4 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.

2.7 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.

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[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]

1. SOURCE TERMS

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.

1.1 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.

1.1.1 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 at full power operations (i.e., a pre-accident iodine spike case).

1.1.2 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 pins assumed to have defects.

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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" (Ref. F-1), for acceptable assumptions and methodologies for performing radiological analyses.

[2] Minimal fuel damage is defined as an amount of damage that will yield reactor coolant system activity concentration levels less than the maximum technical specification limits. 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.

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

1.3 The specific activity in the steam generator liquid at the onset of the MSLB should be assumed to be at the maximum value permitted by secondary activity technical specifications (typically 0.1 µCi/gm DE I-131).

1.4 Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

2. TRANSPORT

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

2.1 The bulk water in the faulted[3] steam generator is assumed to rapidly blow down to the environment. The duration of the blowdown is obtained from thermal-hydraulic analysis codes. The activity in the faulted steam generator bulk water is assumed released to the environment without mitigation.

2.2 For facilities that have not implemented alternative repair criteria (ARC) (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 steam 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. For example, for a four-loop facility with a limiting condition for operation of 500 gpd for any one generator not to exceed 1 gpm from all generators, it would be appropriate to assign 500 gpd to the faulted generator and 313 gpd to each of the unaffected generators.

For facilities that have implemented ARC, the primary-to-secondary leak rate in the faulted steam generator should be assumed to be the maximum accident-induced leakage derived from the repair criteria and burst correlations. For the unaffected steam generators, the leak rate limiting condition for operation specified in the technical specifications is equally apportioned between the unaffected steam generators.

[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.

2.3 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).

2.4 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.

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

2.6 The transport model described in this section should be utilized for releases from the steam generators.

2.6.1 The primary-to-secondary leakage to the faulted steam generator is assumed to flash to vapor and be released to the environment with no mitigation.

2.6.2 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. If the tubes are uncovered, a portion of the primary-to-secondary leakage will flash to vapor, based on the thermodynamic conditions in the reactor and secondary coolant.

- The 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 unaffected generators, using the models in NUREG-0409, “Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident” (Ref. F-2), during periods of total submergence of the tubes.

- The leakage to the unaffected generators that does not immediately flash is assumed to mix with the bulk water.

- The radioactivity in the bulk water of the unaffected generators is assumed to become vapor at a rate that is the function of the steaming rate and the partition coefficient. A partition coefficient[4] for iodine of 100 may be assumed.

2.7 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. F-3). 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.

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[4] *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 F REFERENCES

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

F-2 USNRC, “Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident,” NUREG-0409, May 1985.

F-3 USNRC, “Steam Generator Tube Rupture Analysis Deficiency,” Information Notice 88-31, May 25, 1988.

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.

1. SOURCE TERM

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.

1.1 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.

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

1.3 Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic. These fractions apply to iodine released as a result of fuel damage and to iodine released during normal operations, including iodine spiking.

2. RELEASE TRANSPORT

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

2.1 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.

2.2 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

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[1] Facilities licensed with, or applying for, alternate 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.

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).

2.3 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.

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

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

2.6 The transport model described in assumptions 2.6 and 2.7 of Appendix E should be used for iodine.

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. Two release paths are considered: (1) release via containment leakage and (2) release via the secondary plant. Each release path is evaluated independently as if it were the only pathway available. The consequences of this event are acceptable if the dose from each path considered separately is less than the acceptance criterion in Table 4 in Regulatory Guide 1.195.

1. SOURCE TERM

Assumptions acceptable to the NRC staff regarding core inventory are in Regulatory Position 3 of this guide. The activity released from the breached but unmelted 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 activity release attributed to fuel melting should be based on the fraction of the fuel material that reaches or exceeds fuel melting temperature. For this fuel material fraction, the assumption is that 100% of the noble gases and 25% of the iodines contained in that fraction are available for release from containment. For the secondary system release pathway, 100% of the noble gases and 50% of the iodines contained in that fuel material fraction are released to the reactor coolant.

1.1 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.

1.2 In the first release case, 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.

1.3 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. Evaluations of pH should consider the effect of acids created during the rod ejection accident event, e.g., pyrolysis and radiolysis products.

1.4 Iodine releases from the steam generators to the environment should be assumed to be 97% elemental and 3% organic.

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[1] Facilities licensed with, or applying for, alternate 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. TRANSPORT FROM CONTAINMENT

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

2.1 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.

2.2 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.

3. TRANSPORT FROM SECONDARY SYSTEM

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.

3.1 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.

3.2 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).

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

3.4 The transport model described in assumptions 2.6 and 2.7 of Appendix E should be used for iodine.

APPENDIX I

ACRONYMS

AST	Alternative source term
ARC	Alternative repair criteria
BWR	Boiling water reactor
CDF	Core damage frequency
CEDE	Committed effective dose equivalent
COLR	Core operating limits report
DBA	Design basis accident
DCF	Dose conversion factor
DE	Dose equivalent
DF	Decontamination factor
DNBR	Departure from nucleate boiling ratio
EAB	Exclusion area boundary
ECCS	Emergency core cooling system
EPA	Environmental Protection Agency
ESF	Engineered safety feature
FF	Flash fraction
FHA	Fuel handling accident
FSAR	Final safety analysis report
GDC	General Design Criteria (in Appendix A to 10 CFR Part 50)
gpm	Gallon per minute
gpd	Gallon per day
IPF	Iodine protection factor
LBLOCA	Large break loss-of-coolant accident
LERF	Large early release fraction
LOCA	Loss-of-coolant accident
LPZ	Low population zone
LWR	Light-water reactor
MOX	Mixed oxide
MSIV	Main steam isolation valve
MSLB	Main steam line break
NDT	Nondestructive testing
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RCS	Reactor cooling system
RM	Radiation monitor
SER	Safety evaluation report
SGTR	Steam generator tube rupture
SRP	Standard review plan
TEDE	Total effective dose equivalent
TID	Technical information document
TMI	Three Mile Island

REGULATORY ANALYSIS

A daft regulatory analysis was published with the draft of this guide when it was issued for public comment (DG-1113, January 2002). No changes were necessary, so a separate value/impact statement for this regulatory guide has not been prepared. A copy of DG-1113 and the value/impact statement is available for inspection or copying for a fee in the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD. An electronic version of DG-1113 is available in the NRC's Electronic Reading Room under accession number ML020160023.