Attached is a draft Appendix I of the Accident Analysis handbook. This appendix states in its introduction:

The general approach to evaluate a dose from a criticality accident is:

1. Determine the fission yield (i.e., power history).

2. Determine the direct dose at the appropriate distance.

3. Determine fission product quantities.

4. Determine the source term for inhalation dose. This includes the determination of nuclides present in the released cloud at the time of exposure, including the effects of decay during transport.

5. Determine external beta and gamma immersion doses.

Guidance for these steps is provided in this appendix. *(emphasis added)*

A Task Team of the CSSG developed the guidance for items 1. and 2. above. This guidance has been reviewed by the entire CSSG and revised as appropriate and is included in this draft. Items 3, 4, and 5 are not criticality safety subject matter expert areas of expertise, but were taken largely from Handbook 3010-94 by the Safety Basis member of the Task Team and do not, in the judgment of the CSSG, provide the guidance stated above. Section I.5 of this draft addresses, or needs to address, these items. Specifically, 1) there are multiple references to withdrawn NRC Regulatory Guides from the early 1970's and it would seem that this information should be updated and 2) the statement, from section I.5.1 "With the availability of modern code systems and cross sections it is entirely feasible to calculate the fission products from a postulated criticality event" does not seem particularly helpful absent any mention of these codes. Finally, an example is provided in Section I-6 that shows the determination of the fission yields for a postulated accident. Similar information showing the determination of items 3, 4, and 5 would be helpful to the user.

However, this does not close this Tasking, nor is it the end for this Appendix or the Handbook. The CSSG will maintain involvement in the Handbook development process until finished.

APPENDIX I
CRITICALITY ACCIDENTS
Draft, January 15, 2015

I.1   Introduction

The purpose of this appendix is to provide guidance for the quantitative estimate of radiological doses to support the qualitative assignments of consequences for the DSA hazard evaluation as discussed in Section 2.4.5, Nuclear Criticality Hazard Evaluation, of this handbook. The general approach to evaluate a dose from a criticality accident is:

1. Determine the fission yield (i.e., power history).

2. Determine the direct dose at the appropriate distance.

3. Determine fission product quantities.

4. Determine the source term for inhalation dose. This includes the determination of nuclides present in the released cloud at the time of exposure, including the effects of decay during transport.

5. Determine external beta and gamma immersion doses.

Guidance for these steps is provided in this appendix.

Criticality accident hazards are unique to nuclear facilities and even then only to a subset of these facilities. This subset has a fissile material inventory that is significant, generally defined as exceeding the single parameter subcritical mass limits given in ANSI/ANS-8.1-2014, *American National Standard for Administrative Practices for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, and requiring specific criticality controls to reduce the likelihood of a criticality accident to an acceptable level. This acceptable level is generally one that results in the accident likelihood being judged to be incredible, at least for individual processes. For facilities requiring such process-specific criticality controls, the hazard categorization based on the criticality accident hazard would generally be Hazard Category (HC) 2, in accordance with DOE-STD-1027, Change Notice 1.

Commonly accepted terminology, as used in the criticality safety discipline, is found in report LA-11627-MS (Paxton, 1989). In particular the following two terms are important to the discussions in this Appendix:

> **Criticality Accident:** The release of energy as a result of accidentally producing a self-sustaining or divergent fission chain reaction.

> **Criticality Safety:** Protection from the consequences of a criticality accident, preferably by prevention of the accident. Encompasses procedures, training, and other precautions in addition to physical protection.

Process facility criticality accidents, are the subject of this Appendix. Criticality accidents associated with critical assemblies and reactors are outside the scope of ANSI/ANS-8.1-2014 and this Appendix.

Process facility criticality accidents have been few, both in the US and worldwide, 7 and 22, respectively; and fatalities have been similarly infrequent, 2 and 9, respectively (LA-13638,

McLaughlin, et. al., 2000). The most recent US criticality accident was in 1978 and worldwide in 1999. No accidents have resulted in any significant mechanical energy release, and radiation exposure is the only significant hazard. From criticality accidents seen to date, significant doses have only been associated with nearby workers, with insignificant exposures to co-located workers outside the facility, the public, or the environment.

When criticality accident likelihoods are judged to be non-trivial in a facility, then it is almost always concluded that a criticality accident alarm system is an appropriate safety system in accordance with ANSI/ANS-8.3, *Criticality Accident Alarm System*. Competing risks associated with the response to false alarms may, rarely, modify the decision as to when a criticality accident alarm should be installed. The use of a criticality accident alarm system will help to define the mitigated dose consequences, but will not be taken into account for unmitigated doses.

I.2   Regulatory Requirements, Recommendations, and Guidance.

DOE Order 420.1C, *Facility Safety*, and 10 CFR 830 Subpart B, *Safety Basis Requirements*, both include words reflecting the process analysis requirement from ANSI/ANS-8.1 § 4.1.2, Process Analysis:

> *"Before a new operation with fissionable material is begun, or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions."*

Credible accidents, including credible criticality accidents, are analyzed in the Documented Safety Analysis (DSA). It is assumed that controls for the prevention of criticality accidents are, or will be, in place. It is also assumed that the need for criticality alarms will be determined, and alarms installed, if justified. This Appendix also assumes that enough fissionable material is being handled in a manner such that any potential criticality accident is identified by the hazards evaluation. If the fissionable quantities are below the minimum subcritical limits in the appropriate ANS standard (ANSI/ANS-8.1-2014 or 8.15-2014) analysis of a criticality accident may not be needed, and in this instance would not be identified in the hazard evaluation.

I.2.1   Unmitigated Analysis

DOE-STD-3009 provides guidance on unmitigated accident analysis. In essence, unmitigated analysis means that controls intended to prevent or mitigate an accident are assumed not to function. In the case of a criticality accident, this means, in part, that the accident is assumed to happen with no mitigative features taken into account. An exception to this is that passive safety features that can be shown to survive the initiating event may be considered in the analysis. For example, if a seismic event causes a criticality accident in a shielded area (including building walls), and the shield can be shown to survive the event, then the effectiveness of the shield in mitigating worker accident doses can be accounted for in the unmitigated analysis. However, the inclusion of passive safety features "credited" for mitigation may result in the feature being designated as safety significant or safety class to ensure the feature meets the safety function.

For the analysis of criticality accidents that have the potential for lasting longer than an initial pulse, the accident duration should be limited to eight (8) hours (based on guidance from DOE-STD-3009).

DOE-STD-3009[1] requires the analysis of unmitigated accidents to determine the class of needed controls, that is, safety class or safety significant. For purposes of this Appendix, the need for safety class controls is based on the unmitigated consequences to the public at the site boundary, and the need for safety significant controls is based on the unmitigated consequences to the co-located worker. The co-located worker is 100 meters from the criticality accident for direct exposure calculations. For atmospheric dispersion, the co-located worker is 100 m from the building emission point (which is likely different than the direct exposure point). DOE-STD-3009 also has requirements related to selecting safety significant controls based on other criteria; however, these criteria are not addressed in this appendix. In general terms, if the dose to the public at the site boundary is less than 0.05 Sv (5 rem), safety class controls are not needed, and if the unmitigated dose exceeds 0.25 Sv (25 rem) then safety class controls are required. If the dose to the co-located worker at 100 meters is less than 1 Sv (100 rem), safety significant controls are not needed.

For the criticality accident analysis, DOE-HDBK-3010-94, Change Notice 1 provides estimates of fission yields, and is a recognized source for other accident analysis parameters. In addition, ANSI/ANS-8.23-2007, R2012, *Nuclear Criticality Accident Emergency Planning and Response,* requires potential criticality accident locations be identified, characterized and evaluated, including radiological dose predictions.

Additional guidance for estimating the bounding number of fissions in criticality accidents is provided in section I.3, below. This guidance is based on evolving understanding of the consequences of potential criticality accidents since DOE-HDBK-3010-94 was issued. Application of the DOE-HDBK-3010-94 methods and guidance is judged to provide a more conservative estimate, and may be useful as a scoping calculation to demonstrate that the total doses (direct, inhalation, and immersion) at the site boundary or at 100 m are not significant, in which case further evaluation would not be necessary.

For both existing and new facilities the (preliminary) DSA should provide information as to planned operations and facility layouts and features such as wall compositions and thicknesses. This information should be used with the techniques discussed in Section I.3 to estimate bounding, unmitigated fission yields.

I.3   Accident Fission Yields

As with many known hazards, the understanding of, and thus the control of, the criticality accident hazard has improved dramatically since this hazard was first introduced in the 1940s with the advent of the production of significant quantities of plutonium and enriched uranium. Both the causes, largely human factors, and the personnel effects, localized to within a few to several meters of the accident location, are now well understood. Nevertheless, risks will never vanish and due diligence should always be applied.

I.3.1   Fission Yields of Solution and Solution-Like Systems

History has shown that process criticality accidents have occurred almost exclusively in (hydrogenous) liquid media. The most common medium was fissile material in nitric acid,

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[1] When used without a 2-digit or 4-digit year number after "DOE-STD-3009", it refers to both the 1994 Change Notice 3 and 2014 versions of the DOE Standard. Otherwise, specific versions are referenced.

followed by an organic solution and then suspensions/slurries. The hydrogenous nature of the medium results in relatively slow fission excursions and insignificant likelihoods of mechanical (destructive) energy releases. The liquid nature of the medium results in a combination of instantaneous bubble generation and thermal expansion as the major feedback mechanisms for limiting the first-spike yield (fissions) of the excursion.

ANSI/ANS-8.23-2007, R2012, Appendix C, provides a comprehensive summary of data from criticality accident simulations in controlled environments and its application to estimating accident yields, both the first spike and the steady-state fission rate should the accident not self-terminate. The data cover broad ranges of key parameters, most importantly the solution volume and the reactivity insertion rate.

Two figures from reference documents, reproduced in ANS-8.23, Appendix C, are also reproduced here. Figure I-1 shows the variation in the specific yield of the first spike for prompt critical excursions from the literature data. As is shown, for all but very rapid excursions the specific fission yield is ~1 x 10^{15} fissions/liter.

Some of the process criticality accidents did not even reach the prompt critical state and thus had much smaller specific yields. However, this value of ~1 x 10^{15}/liter is judged to be a practical upper bound for a first spike yield for the purpose of accident analysis based on the experimental accident simulation data and as none of the process accidents exhibited specific yields statistically greater than this value. The data in the figure for the very short period excursions, <10 ms, that do show larger specific yields resulted from reactivity insertion rates that are likely not credible during process accident conditions.

Figure I-2 shows a curve judged to be a practical bounding envelope of the integrated specific fissions during the first 10 minutes subsequent to a prompt critical excursion that is neither self-terminating nor otherwise terminated. Application of the information in Figures I-1 and I-2 enables a realistic upper estimate to be made of both the first-spike dose and the integrated dose to workers during evacuation from a postulated process accident. It also enables the analyst to estimate the dose rate at various locations in order to make decisions as to immediate evacuation zone boundaries and appropriate muster locations. Finally, if this 10-minute time window is consistent with site emergency plans and procedures, then the fission yield curve in Figure I-2 would be appropriate for determining bounding co-located worker and public exposures prior to possible further personnel relocations/evacuations.

For an unmitigated accident with an 8-hour duration there is scant data upon which to base total fission/liter estimates. The Hanford (1961), Novosibirsk (1997) and Tokai-Mura (1999) accidents are the only reported process accidents to have continued fissioning for at least 8 hours. No accident simulations, such as the CRAC series, were allowed to run for more than minutes. Based on these three accidents and the reality that the fission rate is (theoretically) expected to decrease over time, and did in these three accidents, one can only estimate the ratio of the 8-hour fissions to first-spike fissions as perhaps a factor of 30, based on engineering judgment and the observed Toki-Mura power history.
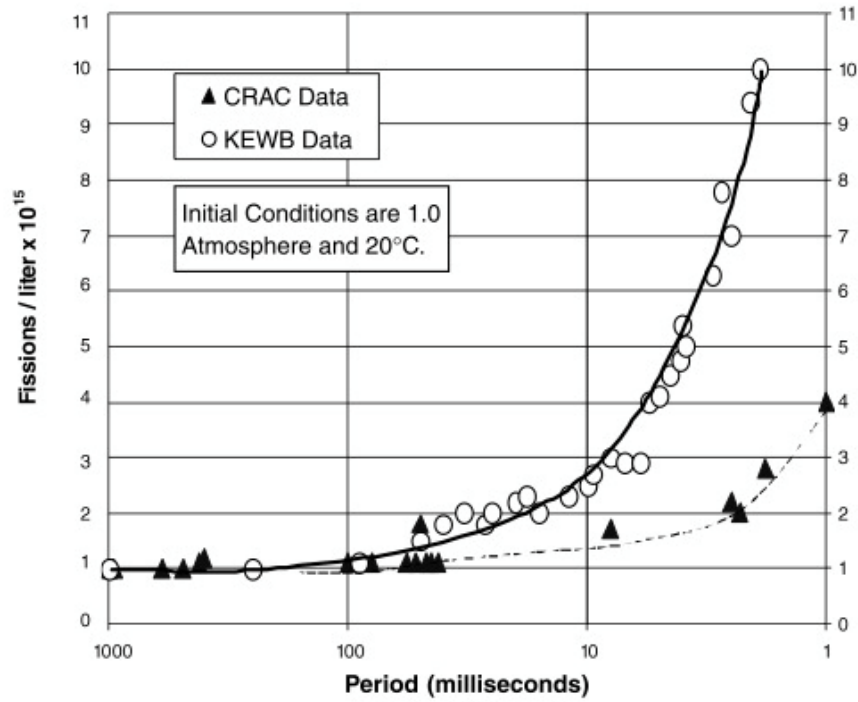
Figure I-1 – Specific fissions in first spike as a function of reactor period
Reactor period is the time required for power to increase by Euler's number (Napier's constant).
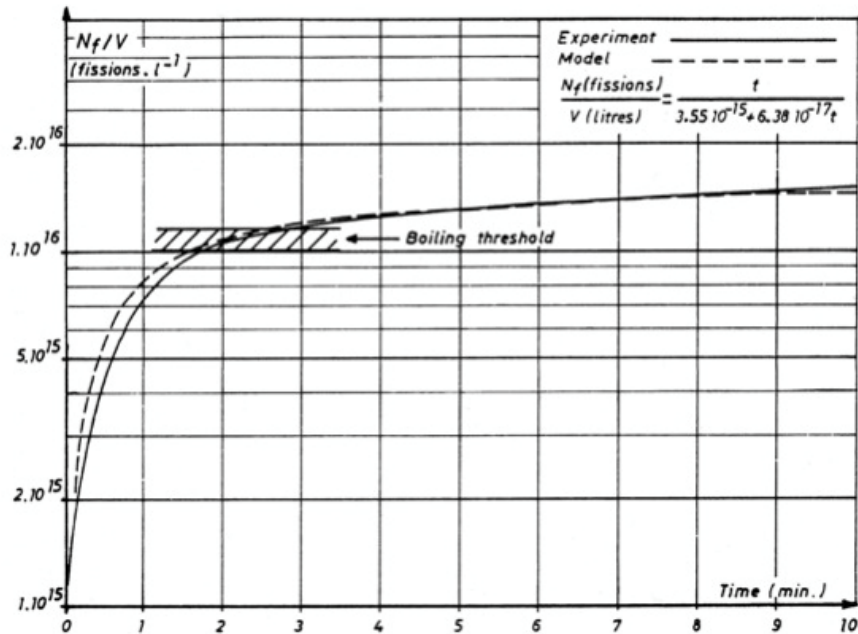(reproduced from Figure C.1 of ANSI/ANS 8.23-2007;R2012)

Figure I-2 – Maximum specific fission yield resulting from criticality solution excursions in CRAC and Silene
(reproduced from Figure C.2 ANSI/ANS 8.23-2007;R2012)

The information contained in Figure I-2 has also been incorporated into the Nuclear Criticality Slide Rule that may also be used to estimate bounding fission yields (NUREG/CR-6504). Similar results will be attained.

I.3.2   Fission Yields of Non-Solution-Like Systems

As the world-wide accident history shows, non-solution process accidents are rare. In fact, if one reviews the circumstances leading up to the one reported non-solution accident, it is apparent that this accident was enabled by a working environment that condoned significant procedural violations in the interest of expediency. It is judged that this working environment would be rarely present to a similar extent in nuclear operations today, particularly in the U.S.

Thus, while criticality accidents in non-solution environments, be they metals or compounds/powders or storage operations, will never be 100% risk-free, credible accident scenarios should be readily foreseen and prevented by design. Such events may then be able to be shown be not credible per ANS-8.1 and DOE-STD-3007-2007, *Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Nonreactor Nuclear Facilities*.

I.3.2.1 Metals

For an unspecified criticality accident with uranium or plutonium in metal form and with no significant moderation, bounding first-spike yields consistent with known accidents, both process and critical experiment, are 1×10^{18} and 1×10^{16} fissions, respectively. For these large first-spike yields prompt shutdown due to thermal expansion and possibly melting would be expected. For lesser first-spike yields a delayed-critical fission reaction is bounding and judged to produce maximum 8-hour yields of 1×10^{19} and 1×10^{18} fissions respectively. Radiation exposures from metal accidents are known to be essentially all from direct neutrons and gamma rays with insignificant fission product releases. For fissile material operations conducted inside typical facilities with thick concrete walls there would be minimal exposures to co-located workers or the MOI from these bounding fission sources.

I.3.2.2 Moderated Solids, Dry Solids, and Large Arrays

These types of criticality accidents, although discussed in DOE-HDBK-3010-94, are now, based on accident experience, deemed so unlikely that they need not be further evaluated for the DSA hazard evaluation if determined to be not credible per the NCS analysis. Accident recovery operations, e.g., from a significant seismic event or fire, may warrant consideration of scenarios such as flooded fissile metal pieces or powders, but this is not a process operation, it cannot be accurately defined, and it is appropriately handled during the accident recovery process.

I.3.3   Fission Yields of Autocatalytic Accidents

This deals with a criticality accident where the reactivity initially increases as the fission reaction progresses, generally due to the effects of temperature and pressure causing material rearrangement within the fissioning medium. One early estimate of excursion yields in a specific facility postulated an unusual accident whose reactivity initially increased due to the initial energy release (Woodcock, 1965). This type of event has not been observed in accident history. However, if the accident being evaluated has the potential for self-propagation, this should be considered.

I.4 Evaluation of Direct Radiation Doses

The prompt dose depends only upon the number of fissions in the criticality accident, the distance from the accident site to the receptor, and the amount of intervening shielding material, such as self-attenuation within the fissioning medium or building walls. The Nuclear Criticality Slide Rule gives curves of unshielded dose as a function of distance, number of fissions, and time after the criticality accident. There is also information on the shielding effect of typical construction materials. This is preferred over the withdrawn NRC Regulatory Guides that have been historically used for DSA development as discussed in Section 6.2.2.2.4, Prompt (Direct) Dose.

I.5 Criticality Accident Source Terms

Chapter 5, Source Term Analysis, covers source term estimation in detail depending on the accident stress on the material. For criticality accidents, however, the source term is fundamentally defined by the number of fissions occurring. This specialized subject will therefore be covered as part of this appendix.

There are two main contributors to the criticality accident source term: fission products generated by the excursion, and releases from the fissionable/fissile material itself. Of the criticality accident fission products themselves, the major components of concern have historically been the noble gases (isotopes of krypton and xenon) and radioiodine, due to their propensity to become airborne and escape filtration. It is permissible, however, to realistically account for the decay products of isotopes, not merely to presume exposure to the radioisotopes initially released.

I.5.1 Fission Product Inventories

A criticality accident generates the same types of fission products contained in spent nuclear reactor fuel. These are the primary fission product isotopes along with the subsequent decay of the initial fission products into other radioactive isotopes that, in turn, continue the decay chain. The typical pattern for total fission product activity in a criticality accident is a decrease in activity by orders of magnitude in the first 30 seconds after the criticality accident terminates. This is due to the loss of high-energy, short half-life isotopes that decay almost immediately. The activity then continues to decrease at a slower rate, with the contributions from various elements and classes of elements changing due to the ongoing decay process. Figure I-3 provides an example of this. It depicts total activity and the activity due to noble gases, halogens (of which iodine is one), and solids over time from a criticality accident pulse of 1×10^{18} fissions.

The NRC, in withdrawn Regulatory Guides 3.33, 3.34 and 3.35, provided an estimate of "the radioactivity of significant nuclides released" for fuel reprocessing solutions, uranium solutions, and plutonium solutions. The criticality accident assumed had a 1×10^{18} fissions initial burst followed by 47 bursts of 1.9×10^{17} fissions each over the next 8 hours for a cumulative total of 1×10^{19} fissions. It is noted that this particular scenario has not been justified in any technical document. The significant nuclides noted were isotopes of krypton, xenon, and iodine. Their activity levels were based on the cumulative yield for the fission energy spectrum, an assumption noted as "very conservative" since it did not consider decay schemes for these nuclides.
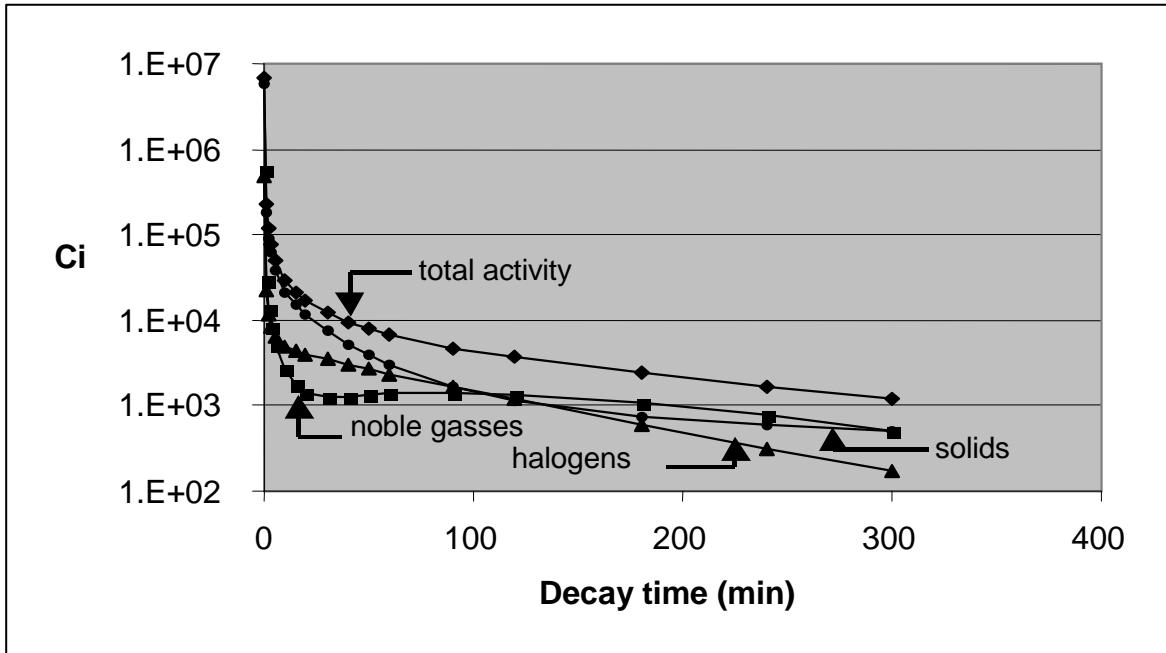
**Figure I-3: Fission Product Activity as a Function of Decay Time Following a
Criticality Accident Pulse**

Historical practice for DOE DSAs has been to use the information in Regulatory Guides 3.33,
3.34, and 3.35 for all criticality accidents, simply scaling the results to reflect total fission yields
less than 1×10^{19} fissions (and eliminating the 8-hr duration for single spike criticality accidents).
However, the Regulatory Guides have been withdrawn. With the availability of modern code
systems and cross sections, it is entirely feasible to calculate the fission products from a
postulated criticality event.

I.5.2   Particulate Release and Health Related Parameters

As presented in Chapter 5, the Airborne Release Fraction (ARF) and Respirable Fraction (RF) are
major parameters in determining the amount of radioactive or other hazardous material released
in an accident. These parameters are normally evaluated by comparing a given phenomenology
to available experimental data. Unfortunately, no direct criticality accident release experiments
have ever been conducted. Further, the fission yields assigned are intended to bound fission
product formation as opposed to realistically estimating the physical changes experienced by the
fissionable/fissile material. Accordingly, criticality accident release fractions have been
developed only in a general sense without attempting to extrapolate them back to detailed
phenomenological modeling. The majority of the effort expended in developing them has also
focused on the fission product release. However, as related to release of particulates due to
melting of metal, boiling of a solution, heating of powders, or energetic dispersal of a powder, the
recommended ARFs and RFs are based on experimental data for those types of changes in the
materials.

As presented in Chapter 6, Section 6.2, Radiological Consequence Assessment, the dose a person
might receive from the fission products released by a criticality accident depends on many
factors. These are discussed in ICRP-68, *Dose Coefficients for Intakes of Radionuclides by
Workers*.

Release estimates for solution criticality accidents derive from NRC Regulatory Guides 3.33, 3.34, and 3.35. In these guides, the NRC established three assumptions. First, all noble gases are assumed to be released from solution and subsequently leave the facility. Second, it assumed that 25 percent of radioiodine ultimately escapes from the facility, either because only that much escapes from solution or because only that much of this element does not react with physical surfaces within the facility. Third, it was generically assumed that the criticality accident terminates when 25 percent of the available solution evaporates. The bounding ARF for boiling liquid is 2×10^{-3} (see Chapter 5). Applying the 25 percent factor to this ARF yields an effective release fraction of 5×10^{-4}, which the NRC originally applied to the base matrix of fissile plutonium in solution in Regulatory Guide 3.35.

The values cited above have been reiterated in NUREG/CR-6410, which also formally extended the 5×10^{-4} release fraction to the seven significant isotopes (Sr-91, Sr-92, Ru-106, Cs-137, Ba-139, Ba-140, and Ce-143). That document further noted that the 5×10^{-4} value is considered "applicable to all non-volatile compounds in the liquid." The portion of the actinides released is assumed to be proportional to the mass of actinides in the solution as the actinides are released through the spray caused by the bursting bubbles that reach the surface of the solution. The mass of the actinides depends upon their concentration in the solution.

I.6   Criticality Accident Example

This example is based on the assumption that the selected accident is the "bounding" hypothetical accident after a thorough review of all fissile operations in the facility. Generally, the bounding accident would involve the largest volume in a hydrogenous, liquid environment coupled with making the conservative assumption that the system reaches the prompt critical state such that the information in Figure I-1 would be applicable. This would lead to the largest number of fissions and thus be "bounding." Rare, extenuating circumstances such as larger accidents that are more remote or in shielded areas may result in a "bounding" accident that does not coincide with the largest number of fissions.

This example assumes that there is a fissile solution that inadvertently and very rapidly accumulates in a 150-liter vessel/volume and that the system just reaches the prompt critical state as the vessel becomes full. Thus, conservatively assuming that the prompt critical state is reached and that the system remains critical, and then applying the information from Figure I-2, we can calculate the first spike, 10-minute, and 8-hour fissions as:

First spike yield = 150 liters x 1×10^{15} fissions/liter = 1.5×10^{17} fissions.

10-minute yield = 150 liters x 1.5×10^{16} fissions/liter = 2.25×10^{18} fissions.

8-hour yield = 1.5×10^{17} x 30 = 4.5×10^{18} fissions.

I.7 References

ANSI/ANS-8.1-2014, *American National Standard for Administrative Practices for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, American Nuclear Society, La Grange Park, IL.

ANSI/ANS-8.3-1997;R2003;R2012 (R=Reaffirmed), *Criticality Accident Alarm System,* American Nuclear Society, La Grange Park, IL.

ANSI/ANS 8.7-1998, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials,* American Nuclear Society, La Grange Park, IL.

ANSI/ANS-8.15-1998, *American National Standard for Nuclear Criticality Control of Special Actinide Elements*, American Nuclear Society, La Grange Park, IL.

ANSI/ANS-8.19-2014, *American National Standard for Administrative Practices for Nuclear Criticality Safet*y, American Nuclear Society, La Grange Park, IL.

ANSI/ANS-8.23-2007;R2012 (R=Reaffirmed), *Nuclear Criticality Accident Emergency Planning and Response*, American Nuclear Society, La Grange Park, IL.

ANSI/ANS-8.24-2007,R2012 (R=Reaffirmed), *Validation of Neutron Transport Methods for Nuclear Criticality Safety Calculations,* American Nuclear Society, La Grange Park, IL.

DOE-STD-3007-2007 (or successor document), *Guidelines for Preparing Criticality Safety Evaluations at Department of Energy Nonreactor Nuclear Facilities.*

McLaughlin, et. al., *A Review of Criticality Accidents (2000 Revision)*, Los Alamos National Laboratory Report LA-13638, May 2000.

NRC Regulatory Guide 3.33, 1977 (withdrawn), *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant*, U.S. Nuclear Regulatory Commission, Washington, DC.

NRC Regulatory Guide 3.34, 1979 (withdrawn). *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant*, U.S. Nuclear Regulatory Commission, Washington, DC.

NRC Regulatory Guide 3.35, 1979 (withdrawn), *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing and Fuel Fabrication Plant*, U.S. Nuclear Regulatory Commission, Washington, DC.

NUREG/CR-6504, *An Updated Nuclear Criticality Slide Rule, Functional Slide Rule*, Vol/ 2, ORNL/TM-13322/V2, Oak Ridge National Laboratory for U.S. Nuclear Regulatory Commission, Washington, DC, April 1998.

ORNL 2011, *Scale: A comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1, June 2011, Oak Ridge National Laboratory, Oak Ridge, TN. This report is available from the Radiation Safety Information Computational Center at ORNL as CCC-785.

Paxton, *Glossary of Nuclear Criticality Safety Terms*, Los Alamos National Laboratory Report LA-11627-MS, October 1989.

Woodcock, *Potential Magnitude of Criticality Accidents*, Safeguards Division, Atomic Health and Safety Board, Risley, UK, 1965.