A DOE-STD-3009 Hazard and Accident Analysis Methodology for Non-reactor Nuclear Facilities

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Abstract

This paper demonstrates the use of appropriate consequence evaluation criteria in conjunction with generic likelihood of occurrence data to produce consistent hazard analysis results for nonreactor nuclear facility Safety Analysis Reports (SAR). An additional objective is to demonstrate the use of generic likelihood of occurrence data as a means for deriving defendable accident sequence frequencies, thereby enabling the screening of potentially incredible events ($<10^{-6}$ per year) from the design basis accident envelope. Generic likelihood of occurrence data has been used successfully in performing SAR hazard and accident analyses for two nonreactor nuclear facilities at Sandia National Laboratories.

DOE-STD-3009-94 addresses and even encourages use of a qualitative binning technique for deriving and ranking nonreactor nuclear facility risks. However, qualitative techniques invariably lead to reviewer requests for more details associated with consequence or likelihood of occurrence bin assignments in the text of the SAR. Hazard analysis data displayed in simple worksheet format generally elicits questions about not only the assumptions behind the data, but also the quantitative bases for the assumptions themselves ("engineering judgment" may not be considered sufficient by some reviewers). This is especially true where the criteria for "qualitative" binning of likelihood of occurrence involves numerical ranges. Oftentimes reviewers want to see calculations or at least a discussion of event frequencies or failure probabilities to support likelihood of occurrence bin assignments. This may become a significant point of contention for events that have been binned as incredible.

This paper will show how the use of readily available generic data can avoid many of the reviewer questions that will inevitably arise from strictly qualitative analyses, while not significantly increasing the overall burden on the analyst.

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Introduction

DOE-STD-3009, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, (Reference 1) allows for largely qualitative methods for determining both consequence and frequency in deriving and ranking nonreactor nuclear facility risks. The DOE standard specifically states that "ranking (i.e., low, medium, and high) of estimated consequences and frequencies are (sic) based on judgment of the analysts" (Reference 2). It further states that a table such as that shown in Figure 1 (Reference 3) provides typical descriptions of consequence thresholds for binning. However, qualitative evaluation criteria are vulnerable to wide differences in interpretation. Moreover, when the evaluation criteria involve numerical probability ranges (such as for frequency binning), reviewers often

Descriptive Word	Description
No	Negligible onsite and offsite impact on people or the environs
Low	Minor onsite and negligible offsite impact on people or the environs
Moderate	Considerable onsite impact on people or the environs; only minor offsite impact
High	Considerable onsite and offsite impacts on people or the environs

Figure 1. Qualitative Severity Classification Table

request more details associated with bin assignments. They may question not only the assumptions behind the data, but the bases for the assumptions themselves. Reviewers may also request that calculations, or at least a discussion of event frequencies and failure probabilities to support likelihood of occurrence bin assignments, be incorporated into the SAR text.

This paper presents methods for consistent, defendable risk determinations. These methods include development of better definitions for consequence evaluation criteria and the use of generic data for initiating event frequencies, structure/system/component failure probabilities, and human error probability.

Hazard Analysis Consequence Evaluation Criteria

Although the standard suggests the use of a very simple set of consequence evaluation criteria as illustrated in Figure 1, such criteria are subject to individual reviewers' interpretations or perceptions. Qualifiers such as *negligible*, *minor*, and *considerable*, as well as the word *impact* itself, lead to subjective reviews. Additional qualification of these terms would improve the review process. Furthermore, the intent of the criteria in Figure 1 is to identify immediate human health effects from hazardous materials. Their primary purpose is to assess acute health risks, rather than potential latent risks (Reference 4) or the exceedance of regulatory compliance limits.

The use of appropriate consequence evaluation criteria will provide consistent risk assessment results. Such assessments are less subjective and thus more easily defended. Therefore, in order to avoid unnecessary debates over the qualitative terminology used in the consequence analysis guidelines shown in Figure 1, the Sandia National Laboratories nonreactor nuclear facilities have adopted the consequence evaluation criteria shown in Table 1.

For purposes of evaluating the potential consequences of human radiation exposures, the criteria of Table 1 would benefit from additional definition. The criteria for the public/collocated workers and facility workers in Table 1 can be readily translated into radiation dose thresholds using the data of Table 2, which shows the radiation exposure levels at which health effects appear in healthy adults.

While the mortality threshold is shown in Table 2 as 150 rem, this same threshold is given in NUREG/CR-4214 as 200 rem. On the basis of this data, a dose of 100 rem can be conservatively chosen as the threshold dose to produce immediately observable health effects in the off-site public or collocated workers, and 200 rem chosen as the threshold dose to effect a loss of life in facility workers. These thresholds represent a high consequence level for hazard evaluation binning purposes (Category A in Table 1). A conservative threshold of 25 rem can be chosen to represent only minor observable effects on members of the off-site public or collocated workers (e.g., blood count changes), but one that does not result in any permanent health effects. Similarly, 100 rem can be chosen to represent the threshold dose resulting in a "lost time injury" to facility workers, but one that does not result in any disability. These doses thus represent the thresholds for a low consequence level (Category C in Table 1). Doses below these thresholds represent no acute health consequence.

Table 1. Consequence Severity Categories

Category	Affected Population/Entity			
	Public	Environment	Collocated Worker	Facility Worker
A	Immediate health effects	Significant off-site contamination requiring cleanup	Immediate health effects	Loss of life
B	Potential for latent health effects	-Moderate to significant on-site contamination -Moderate off-site contamination	Potential for latent health effects	Severe injury or disability
C	Minor observable effects, but no permanent health effects	-Moderate contamination of facility -Minor on-site or off-site contamination	Minor observable effects, but no permanent health effects	Lost time injury, but no disability
D	No observable health effects	-Minor contamination of facility -Negligible on-site or off-site contamination	No observable health effects	Minor or no significant effects

Table 2. Radiation Exposure Level Effects in Healthy Adults

Effects	Acute Dose in Gy	g Dose in Rem
Blood count changes	0.50[a]	50
Vomiting (threshold)	1.00[a]	100[b]
Mortality (threshold)	1.50[a]	150
80-100% Mortality	>5.4[a]	>540

a. NCRP Report 98, “Guidance on Radiation Received in Space Activities,” NCRP, Bethesda, MD (1989)
b. NUREG/CR-4214, “Health Effects Models for Nuclear Power Plant Accident Consequence Analysis”

Based on the above, a moderate consequence level can then be represented by thresholds of 50 rem to the off-site public or collocated workers, and 150 rem to facility workers (Category B in Table 1). A threshold dose of 50 rem to the off-site public or collocated workers (midway between immediate health effects and no health effects) is conservatively assumed to result in latent health effects. A threshold dose of 150 rem for facility workers (midway between loss of life and no disabling effects) is conservatively assumed to result in severe injuries or disabilities.

Hazard Analysis Consequence Assessment

A particularly useful tool for assessing the potential off-site consequences of radiological hazards at the SNL nonreactor nuclear facilities is a database of single isotope airborne doses versus distance from the facility. This database (Reference 5) was generated in 1998 using the MACCS2 code (Reference 6) to perform a probabilistic evaluation that provided 95th percentile centerline doses for individual isotopes. With the aid of this database one can quickly assess the potential off-site dose associated with an assumed release of radiological material. Figure 2 shows how this database information is used to quickly assess the potential magnitude of an off-site radioactive material release.

Off-site Public Dose from Test Cell Fire It is assumed that a test unit containing 50 Ci of Pu-239 and combustible material is involved in a fire while being irradiated in a shielded test cell. If the cell ventilation system is not turned off, then airborne Pu-239 from the fire may be discharged from the facility ventilation exhaust stack, with the potential for an off-site radiological exposure. From the single isotope airborne doses versus distance database, the centerline dose for an elevated release of 1 Curie of Pu-239 at a distance of 3000 meters from the exhaust stack is approximately 0.7 rem. Discharge of all 50 Ci from the stack then results in a whole body dose of approximately 35 rem, which causes no permanent health effects in exposed members of the public.

Figure 2. Off-site Public Dose from Test Cell Fire

Hazard Analysis Frequency Evaluation Criteria

DOE-STD-3009 suggests that the use of binning criteria for frequency "should typically cover two orders of magnitude." Table 3, which closely resembles the example given in the standard, illustrates the frequency binning criteria used for SNL nonreactor nuclear facility hazard analysis.

Hazard Analysis Frequency Assessment

In assessing the frequency associated with a particular hazard consequence, the analyst must generally define an event sequence that associates the hazard with an undesired outcome. Rather than guess at the overall sequence frequency category, a defendable assessment can be readily obtained by applying generic failure probabilities to individual sequence elements.

Table 3. Frequency Categories

Category	Category Descriptor	Frequency (per year)	Description
I	Normal Operations	$F \geq 1$	Normal Operations
II	Likely or Anticipated	$1 > F \geq 10^{-2}$	Incidents that may occur several times during the lifetime of the facility. (Incidents that commonly occur)
III	Unlikely	$10^{-2} > F \geq 10^{-4}$	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this frequency category include Uniform Building Code-level earthquake, 100-year flood, maximum wind gust, etc.
IV	Very Unlikely	$10^{-4} > F \geq 10^{-6}$	Accidents that will probably not occur during the life cycle of the facility.
V	Extremely Unlikely	$10^{-6} > F$	All other accidents. Accidents too unlikely to be considered in the design basis. Some accidents in this frequency category may be evaluated as beyond design basis accidents.

Failure probabilities appropriate for use in nonreactor nuclear facility hazard analyses can be found as indicated in the following references:

- *Qualitative Methods for Assessing Risk*, Mahn, J.A., G.W. Hannaman, and P.M. Kryska, SAND95-0320, Sandia National Laboratories, April 1995.
  - generic initiating events (per year)
  - generic equipment failure rates (per hour)
  - system level failure rates (per demand)
  - human error probabilities
  - generic structure failure rates (per demand)

- *Hazard and Barrier Analysis Guidance Document*, U.S. Department of Energy, Office of Operating Experience Analysis and Feedback, November 1996.
  - electrical and electromechanical equipment failure rates (per hour)
  - mechanical equipment failure rates (per hour)
  - human error probabilities

- *Savannah River Site Generic Data Base Development (U)*, WSRC-TR-93-262. Westinghouse Savannah River Company, June 1993.
    - water system equipment failure rates
    - chemical process system equipment failure rates
    - compressed gas system equipment failure rates
    - HVAC/exhaust system equipment failure rates
    - electrical distribution system equipment failure rates
    - instrumentation and control system equipment failure rates

The failure probability data provided in these documents is representative of typical U.S. facilities designed and constructed under consensus codes and standards, using commercially available equipment, and operated by knowledgeable personnel. As such, it will assist the analyst in arriving at a reasonable assessment of the overall hazard scenario frequency. However, because these data sources are necessarily limited in scope, failure probabilities for some event sequence elements will occasionally need to be based on engineering judgment. (Other more detailed failure probability databases may also exist that are suitable for this analysis.)

Figure 3 illustrates the use of this data for assessing the frequencies of two hazardous events. (Note that the sequence events in these examples are assumed to be independent. In choosing to use this approach, the analyst should assess the validity of such an assumption.) It can be seen from this quick frequency assessment that the appropriate frequency categories associated with the worst-case, unmitigated consequences for events ST-1 and F-1 are IV and III, respectively. Documenting this analysis (by means of the table shown in Figure 3) will avoid the need for assembling this information after the fact in response to reviewers' comments.

To select the minimal set of accidents for further analysis, it is necessary to identify the worst credible consequence and the highest risk scenarios associated with each hazard type from the hazard analysis. These then constitute the representative bounding accidents for the facility, for which more detailed source term and event tree analyses (Figure 4) are performed in order to derive final consequence and frequency bin assignments. The type of evaluation shown in Figure 3 makes the selection of representative bounding accidents simpler and less prone to errors in judgment. Another benefit is that much of the data required for calculating event tree branch frequencies has already been assembled.

Event No.	Description	Sequence Events	Individual Event Probabilities	Accident Frequency (per year)	Freq. Bin
ST-1	Cask breach, no contents released (35 cask transfer operations, 1 hour duration each)	A. Forklift accident (1) driving accident (human error) (2) equipment failure (3) dropping of load (human error) B. Cask lid breach (1) failure to secure lid (human error) (2) structural failure C. Failure of workers to evacuate from extremely high radiation field (human error)	A(1) 1E-3 per demand[a] A(2) 1E-5 per hour[b] A(3) 5E-5 per operation[c] B(1) 1E-3 per demand[a] B(2) 5E-2 per demand[d] C. 1E-3 per demand[a]	A(1) x B(2) = 2E-3 A(2) x B(2) = 2E-5 A(3) x B(2) = 9E-5 A(1) x B(1) = 4E-6 A(1) x B(2) x C = 2E-6	III IV IV IV IV
F-1	Combustible material fire in test cell (12 test units per year; 8-hour irradiation)	A. Fire (1) Test unit electrical equipment short (2) Failure of test unit reactive chemical barrier (3) Unanticipated chemical reaction (human error) B. Cell ventilation not turned off (1) Failure of cell heat sensor channel (2) Failure of control relay to terminate cell ventilation	A(1) 1E-6 per hour[e] A(2) 1E-3 per year[f] A(3) 1E-3 per demand[a] B(1) 5E-2 per demand[g] B(2) 1E-4 per demand[h]	A(1) x B(1) = 4.8E-6 A(2) x B(1) = 5E-5 A(3) x B(1) = 6E-4 A(3) x B(2) = 1.2E-6	IV IV III IV

a. Qualitative Methods for Assessing Risk, SAND95-0320, p. 26
b. Qualitative Methods for Assessing Risk, p. 24
c. Hazard and Barrier Analysis Guidance Document, p. B-9
d. Qualitative Methods for Assessing Risk, p. 27
e. Hazard and Barrier Analysis Guidance Document, p. A-15
f. Qualitative Methods for Assessing Risk, p. 23
g. Qualitative Methods for Assessing Risk, p. 25
h. Savannah River Site Generic Data Base WSRC-TR-93-262, p. 183

Figure 3. Assessment of Hazard Scenario Frequencies

Co-60 source cask transfer by forklift (no. of operations per year)	**Forklift operator maintains adequate control of forklift (prob.)**	**Forklift equipment functions properly (prob.)**	**Cask structural integrity not affec-ted by forklift accident (prob.)**	**Cask lid bolts were installed properly (prob.)**	**Cask lid bolts do not all break in forklift accident (prob.)**	**Co-60 sources are not ejected from cask and damaged (prob.)**	**Branch Frequency (per year)**	**Freq. Bin**	**Outcome**
35	1	1	NA	NA	NA	NA	35	I	No accident
		1E-5	1	1	1	NA	3.5E-4	III	No radiation hazard
					5E-2	1	1.8E-5	IV	Collimated radiation beam
						5E-2	8.8E-7	V	Unshielded radiation source and possible contamination
				1E-3	NA	1	3.5E-7	V	Collimated radiation beam
						5E-2	1.8E-8	V	Unshielded radiation source and possible contamination
			5E-2	NA	NA	1	1.8E-5	IV	Collimated radiation beam
						5E-2	8.8E-7	V	Unshielded radiation source and possible contamination
	1E-4		1	1	1	NA	3.5E-3	III	No radiation hazard
					5E-2	1	1.8E-4	III	Collimated radiation source
						5E-2	8.8E-6	IV	Unshielded radiation source and possible contamination
				1E-3	NA	1	3.5E-6	IV	Collimated radiation source
						5E-2	1.8E-7	V	Unshielded radiation source and possible contamination
			5E-2	NA	NA	1	1.8E-4	III	Collimated radiation source
						5E-2	8.8E-6	IV	Unshielded radiation source and possible contamination

Figure 4. Event Tree Analysis Diagram for a Source Transfer Cask Breach Accident

References

1. *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, DOE-STD-3009-94, U.S. Department of Energy, Washington, D.C., July 1994.
2. Ibid., p. 39
3. Ibid., p. 51
4. Ibid., p. xix
5. Memorandum, R.E. Naegeli to Distribution, *Transmittal of MACCS2 Single Isotope Airborne Dose Versus Distance Database*, Sandia National Laboratories, January 9, 1998.
6. D.I. Chanin and M.L. Young, *Code Manual for MACCS2: Volume 1, User's Guide*, SAND97-0594 (NUREG/CR-6613), Sandia National Laboratories, March 1997.