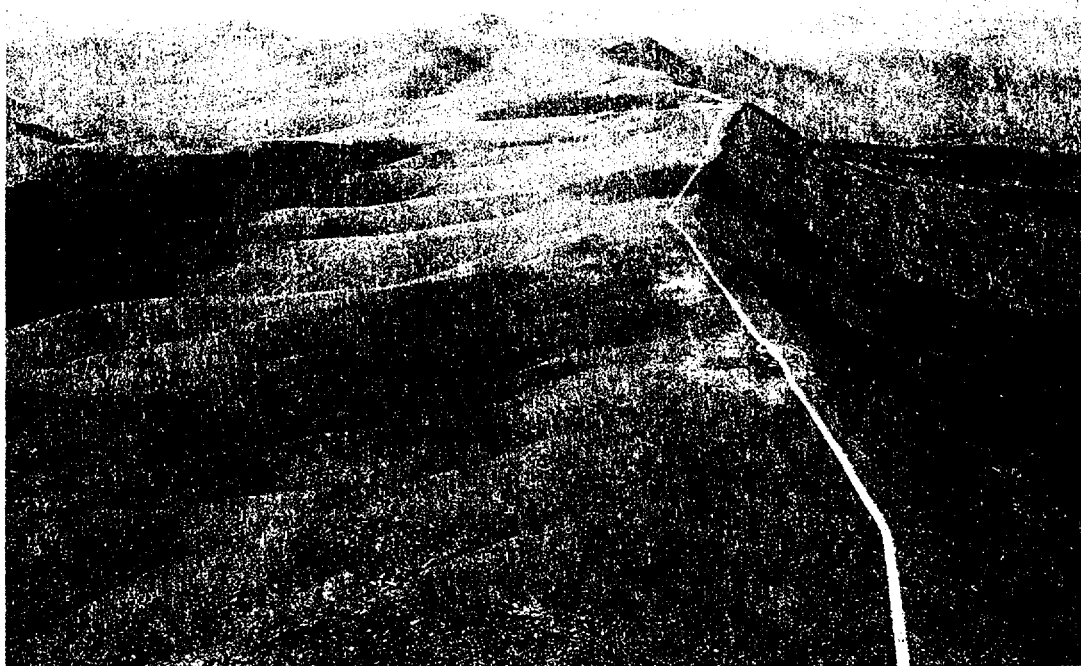


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# Preclosure Safety Analysis Guide



Prepared for:  
U.S. Department of Energy  
Yucca Mountain Site Characterization Office  
P.O. Box 30307  
North Las Vegas, Nevada 89036-0307

Prepared by:  
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Under Contract Number  
DE-AC08-01NV12101

**PRELIMINARY DRAFT  
INFORMATION ONLY**

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## ACRONYMS

ARF	airborne release fraction
BWR	boiling water reactor
CDE	committed dose equivalent
CEDE	committed effective dose equivalent
CI/FA	curies per fuel assembly
CR	cladding release fraction
CSNF	commercial spent nuclear fuel
DCF	dose conversion factor
DDE	deep-dose equivalent
DEP	deposition factor
DF	damage fraction
DOE	U.S. Department of Energy
DSNF	DOE spent nuclear fuel
EDE	effective dose equivalent
HEPA	high efficiency particulate air
LDE	lens dose equivalent
NRC	U.S. Nuclear Regulatory Commission
PWR	pressurized water reactor
RF	respirable fraction
SDE	skin dose equivalent
SNF	spent nuclear fuel
TEDE	total effective dose equivalent
WHB	Waste Handling Building
YMP	Yucca Mountain Site Characterization Project

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## 8. CONSEQUENCE ANALYSIS

### 8.1 INTRODUCTION

The consequence analysis demonstrates that the preclosure performance objectives for the repository operations area, specified in 10 CFR 63.111, have been met. Dose criteria specified in 10 CFR Part 63 and 10 CFR Part 20 (Table 8-1) specify the offsite and worker dose limits during normal operations and for Category 1 event sequences, and the offsite dose limits for Category 2 event sequences.

In this section, the methodology for calculating offsite doses for Category 1 and Category 2 event sequences and for calculating worker doses for Category 1 event sequences is presented. Because the regulatory limit for Category 2 event sequences is a total effective dose equivalent (TEDE) of 5 rem per event (10 CFR 63.111), the doses due to Category 2 event sequences are calculated on a per event basis. The regulatory limit for Category 1 event sequences is an annual TEDE of 15 mrem per year (10 CFR 63.111). Therefore, all Category 1 event sequence doses are calculated on a per year basis. It should be noted that the data presented in Section 8 is preliminary in nature and should not be interpreted as the final data to be used in the final design.

Four dose measures applicable to Category 1 and Category 2 event sequences include:

- **The TEDE**—For purposes of assessing doses to workers, the TEDE is equal to the sum of the deep-dose equivalent (DDE) (for external exposures) and the committed effective dose equivalent (CEDE) (for internal exposures) (10 CFR 63.2). For purposes of assessing doses to members of the public, the TEDE is equal to the sum of the effective dose equivalent (EDE) (for external exposures) and the CEDE (10 CFR 63.2). The CEDE is calculated using the effective inhalation dose conversion factor (DCF). The EDE is calculated using the effective air submersion DCF. For normal operations and Category 1 event sequences, the TEDE also includes ingestion and groundshine doses in addition to inhalation and submersion doses. In assessing compliance with the individual radiation protection standard, the DDE is replaced by the EDE per the U.S. Nuclear Regulatory Commission (NRC) guidance on the use of the DDE and EDE for external exposure (66 FR 55732).
- **The Highest of the Committed Dose Equivalents (CDEs) plus the DDE**—The organs evaluated to determine the highest CDEs are the lungs, breasts, gonads, red marrow, bone surface, thyroid, and remainder. The remainder is not an organ, but rather a weighted combination of the five remaining organs or tissues (e.g., liver, kidneys, spleen, and brain, but excluding skin, lens of the eye, and the extremities) receiving the highest doses (Eckerman et al. 1988). The DDE, which is added to the highest CDE, is equal to that used to calculate the TEDE. In assessing compliance with the individual radiation protection standard, the DDE is replaced by the EDE per the NRC guidance on the use of the DDE and EDE for external exposure (66 FR 55732).
- **Lens of the Eye**—In Federal Guidance Report 11 (Eckerman et al. 1988), only one lens of the eye DCF (i.e., Kr-83m) can be found. The lens dose equivalents (LDEs) are not calculated using the lens of the eye DCFs, as the lens of the eye DCFs given in Federal

Guidance Report 11 (Eckerman et al. 1988) are incomplete. However, in NUREG-1567 (NRC 2000), it is stated that compliance with the lens of the eye dose limit is achieved if the sum of the skin dose equivalent (SDE) and the TEDE does not exceed 15 rem.

- **Skin and Extremities**—The dose to the skin and extremities is only due to the air submersion pathway. SDEs are calculated using the DCFs for air submersion in Federal Guidance Report 12 (Eckerman and Ryman 1993).

Table 8-1. Dose Criteria for Category 1 and Category 2 Event Sequences

Event Sequence Type	Dose Type	Dose Criteria	
		Worker <sup>a</sup>	Offsite <sup>b</sup>
Category 1	1. Annual TEDE during normal operations and for Category 1 event sequences; 2. Aggregate TEDE for Category 1 event sequences	---	15 mrem/yr
Category 1	TEDE	5 rem/yr	100 mrem/yr
Category 1	CDE + DDE	50 rem/yr	---
Category 1	Lens of the Eye	15 rem/yr	---
Category 1	Skin & Extremities	50 rem/yr	---
Category 1	External Dose: Highest of DDE, Skin Dose, or Dose to Lens of the Eye	---	2 mrem/hr
Category 2	TEDE	---	5 rem/event
Category 2	CDE + DDE	---	50 rem/event
Category 2	Lens of the Eye	---	15 rem/event
Category 2	Skin & Extremities	---	50 rem/event

<sup>a</sup> 10 CFR 20.1201

<sup>b</sup> 10 CFR 20.1301, 10 CFR 63.111, and 10 CFR 63.204

## 8.2 EXAMPLES OF SOURCE TERMS

### 8.2.1 Source Terms for Commercial Spent Nuclear Fuel and Crud

**Commercial Spent Nuclear Fuel (CSNF)**—Examples of average and maximum source terms for pressurized water reactor (PWR) and boiling water reactor (BWR) CSNF can be found in *PWR Source Term Generation and Evaluation* (CRWMS M&O 1999a) and *BWR Source Term Generation and Evaluation* (CRWMS M&O 1999b). Future revisions to these documents should be referenced as needed.

The SAS2H sequence in SCALE V4.3 (CRWMS M&O 1997) was used to calculate the PWR and BWR source terms (CRWMS M&O 1999a and 1999b) for selected fuel assemblies as a function of assembly average burnup and cooling time. The prime functional module of the SAS2H code sequence utilized is the ORIGEN-S code. This code performs a point depletion and decay calculation of a selected fuel type with user-specified irradiation conditions and decay times. The resulting source terms are then extracted from the SAS2H output and used as input to consequence analysis.

For preclosure consequence analysis, source terms for PWR and BWR spent fuel assemblies with four different combinations of initial enrichment, burnup, and decay time should be considered:

Assembly	Percent	GWd/MTU	Years
Average PWR	4.0	48	25
Maximum PWR	5.0	75	5
Average BWR	3.5	40	25
Maximum BWR	5.0	75	5

**Example of Source Term Usage**—Average PWR fuel was selected for consequence analysis of all Category 1 event sequences because it was found to result in a higher offsite dose consequence as compared to average BWR fuel (BSC 2001). This result is generally attributed to a higher enrichment, burnup, and concentration of long-lived radionuclides in PWR fuel.

For Category 2 event sequences, either the maximum PWR fuel or the maximum BWR fuel was used to calculate maximum doses, depending on which source term results in the highest dose. For events occurring in spent fuel pools, maximum PWR fuel results in the largest offsite dose due to the larger inventory of radioactive gases compared to maximum BWR fuel (BSC 2001). For events involving particulate releases in a dry environment (e.g., hot cell), however, maximum BWR fuel results in a larger offsite dose than maximum PWR fuel because of the increased crud inventory (BSC 2001). For Category 2 event sequences, mean doses should also be calculated using the average PWR or average BWR fuel. These mean doses will be compared with the regulatory dose limits given in Table 8-1.

Radionuclide inventories in curies per fuel assembly (Ci/FA) for each nuclide and fuel type evaluated are presented in Table 8-2. These radionuclide inventories were used in site recommendation public and worker dose calculations, and they may be revised later for license application.

Table 8-2. Example of Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) Spent Nuclear Fuel (SNF) Radionuclide Inventories

Nuclide	Average PWR (Ci/FA)	Maximum PWR (Ci/FA)	Average BWR (Ci/FA)	Maximum BWR (Ci/FA)
Ac-227	1.61E-05	0.00E+00	0.00E+00	0.00E+00
Am-241	1.98E+03	8.71E+02	5.58E+02	2.66E+02
Am-242m	6.39E+00	1.02E+01	2.17E+00	3.40E+00
Am-243	2.20E+01	5.22E+01	5.35E+00	1.93E+01
C-14	3.32E-01	4.89E-01	1.75E-01	3.16E-01
Cd-113m	7.66E+00	3.82E+01	2.26E+00	1.39E+01
Cl-36	6.80E-03	9.69E-03	2.93E-03	4.99E-03
Cm-242	5.27E+00	3.43E+01	1.79E+00	1.13E+01
Cm-243	1.03E+01	3.83E+01	2.48E+00	1.12E+01
Cm-244	1.36E+03	1.12E+04	2.56E+02	3.95E+03
Cm-245	3.07E-01	1.41E+00	4.04E-02	3.54E-01
Cm-246	1.04E-01	8.38E-01	1.45E-02	2.97E-01
Co-60	3.13E+02	5.66E+03	4.40E+01	8.56E+02

Table 8-2. Example of Pressurized Water Reactor and Boiling Water Reactor Spent Nuclear Fuel Radionuclide Inventories (Continued)

Nuclide	Average PWR (Ci/FA)	Maximum PWR (Ci/FA)	Average BWR (Ci/FA)	Maximum BWR (Ci/FA)
Cs-134	2.52E+01	3.72E+04	6.32E+00	1.16E+04
Cs-135	3.50E-01	5.99E-01	1.39E-01	2.82E-01
Eu-155	5.16E+01	1.68E+03	1.64E+01	6.37E+02
Fe-55	3.47E+00	6.84E+02	1.09E+00	2.35E+02
H-3	1.14E+02	4.72E+02	3.95E+01	1.76E+02
I-129	2.20E-02	3.38E-02	7.43E-03	1.36E-02
Kr-85	1.13E+03	5.63E+03	3.81E+02	2.03E+03
Nb-93m	1.30E+01	4.54E+01	4.74E-01	1.22E+00
Nb-94	8.39E-01	1.27E+00	1.87E-02	3.39E-02
Ni-59	2.09E+00	2.78E+00	5.03E-01	7.80E-01
Ni-63	2.52E+02	4.16E+02	5.87E+01	1.16E+02
Np-237	2.47E-01	3.85E-01	6.89E-02	1.33E-01
Pa-231	2.97E-05	4.25E-05	1.39E-05	2.94E-05
Pb-210	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Pd-107	8.41E-02	1.45E-01	2.65E-02	5.70E-02
Pm-147	1.19E+02	2.34E+04	3.98E+01	7.46E+03
Pu-238	2.29E+03	6.16E+03	5.85E+02	2.11E+03
Pu-239	1.77E+02	1.85E+02	5.35E+01	5.36E+01
Pu-240	3.18E+02	3.90E+02	1.14E+02	1.48E+02
Pu-241	2.47E+04	7.91E+04	6.78E+03	2.25E+04
Pu-242	1.64E+00	3.01E+00	5.09E-01	1.26E+00
Ra-226	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ra-228	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ru-106	1.23E-02	1.27E+04	3.00E-03	3.29E+03
Sb-125	9.71E+00	2.05E+03	2.89E+00	6.21E+02
Se-79	4.57E-02	6.95E-02	1.59E-02	2.89E-02
Sm-147	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sm-151	2.11E+02	3.13E+02	5.39E+01	8.22E+01
Sn-126	3.85E-01	6.28E-01	1.27E-01	2.52E-01
Sr-90	2.72E+04	6.30E+04	9.54E+03	2.52E+04
Tc-99	8.99E+00	1.28E+01	3.20E+00	5.35E+00
Th-229	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Th-230	1.48E-04	3.56E-05	6.09E-05	2.05E-05
Th-232	0.00E+00	0.00E+00	0.00E+00	0.00E+00
U-232	2.05E-02	5.31E-02	4.64E-03	2.00E-02
U-233	4.07E-05	2.42E-05	1.14E-05	0.00E+00
U-234	6.77E-01	5.46E-01	2.49E-01	2.26E-01
U-235	7.36E-03	4.15E-03	2.62E-03	9.40E-04
U-236	1.72E-01	2.24E-01	6.26E-02	9.55E-02
U-238	1.48E-01	1.43E-01	6.32E-02	6.07E-02
Zr-93	8.94E-01	1.33E+00	3.38E-01	6.03E-01

Source: CRWMS M&amp;O 1999a; CRWMS M&amp;O 1999b.



**Crud**—Crud is a corrosion product that has been found on the exterior surface of spent nuclear fuel (SNF) assemblies due to irradiation and imperfect water chemistry control in the reactor coolant system. Crud can be released to the environment during an accident involving CSNF at the potential repository.

After decaying for five years, the nuclide species that have significant activity in the crud are Fe-55 and Co-60. Crud activities used for the average and maximum PWR and BWR assemblies can be based on values recommended in *Commercial SNF Accident Release Fractions* (CRWMS M&O 1999c). Future revisions to this document should be referenced as appropriate.

CSNF fuel assemblies have the following initial crud activities at the time of discharge from the reactor:

Radionuclide	PWR ( $\mu\text{Ci}/\text{cm}^2$ )	BWR ( $\mu\text{Ci}/\text{cm}^2$ )
Co-60	140	1254
Fe-55	5902	7415

These crud activities are bounding estimates based on analysis in *Commercial SNF Accident Release Fractions* (CRWMS M&O 1999c).

The crud surface activity for a given assembly is a function of time after discharge from the reactor. The time-dependent crud surface activity is based on the radioactive decay equation given in CRWMS M&O 1999a, Section 5.6. The radioactive decay equation is:  $N(t) = N(0) \exp(-t \times \ln 2 / t_{1/2})$ , where  $N(t)$  is the crud activity at time  $t$ ,  $N(0)$  is the crud activity at time 0,  $t_{1/2}$  is the radionuclide half-life in years, and  $t$  is the decay time in years.

The crud source term (Ci/FA) released to the environment, on a per assembly basis, is calculated as follows:

$$ST_{crud} = SA_{crud} \times A_{SFA} \times conv \quad (\text{Eq. 8-1})$$

where

$ST_{crud}$  = Crud source term (Ci/FA)

$SA_{crud}$  = Crud surface activity ( $\mu\text{Ci}/\text{cm}^2$ )

$A_{SFA}$  = Surface area per assembly ( $\text{cm}^2/\text{FA}$ )

$conv$  = Conversion factor ( $10^{-6} \text{ Ci}/\mu\text{Ci}$ )

CSNF fuel assemblies have the following surface areas,  $A_{SFA}$ :

PWR = 449,003  $\text{cm}^2/\text{assembly}$

BWR = 168,148  $\text{cm}^2/\text{assembly}$

These surface areas are bounding estimates based on the assemblies with the highest known surface areas, a South Texas PWR assembly (CRWMS M&O 1999a) and an ANF 9x9 JP-4 BWR assembly (CRWMS M&O 1999b).

Crud source terms for Category 1 event sequences were based on average PWR fuel with a 25-year decay time (BSC 2001). An example of crud source term calculations is given in Table 8-3.

Table 8-3. Example of Category 1 Crud Source Term Calculations

Fe-55 Included In Crud Calculation. (Y/N):	Y
Crud Decay Time (years):	25
Fe-55 PWR Surface Activity ( $\mu\text{Ci}/\text{cm}^2$ ) =	10.3 <sup>a</sup>
Fe-55 BWR Surface Activity ( $\mu\text{Ci}/\text{cm}^2$ ) =	13.0 <sup>a</sup>
Co-60 PWR Surface Activity ( $\mu\text{Ci}/\text{cm}^2$ ) =	2.6 <sup>a</sup>
Co-60 BWR Surface Activity ( $\mu\text{Ci}/\text{cm}^2$ ) =	23.4 <sup>a</sup>
Bounding PWR Surface Area ( $\text{cm}^2$ ) =	449,003
Bounding BWR Surface Area ( $\text{cm}^2$ ) =	168,148
Co-60 Half Life	5.271 yrs
Conversion factor (Bq per $\mu\text{Ci}$ ) =	3.70E+04
Conversion factor (rem per Sv) =	100
Crud Source	(Ci/FA)
Fe-55 PWR	4.60E+00
Fe-55 BWR	2.20E+00
Co-60 PWR	1.20E+00
Co-60 BWR	3.90E+00

<sup>a</sup> Values are corrected for half-life and average surface activity if there is more than one FA.

### 8.2.2 Source Terms for U.S. Department of Energy Spent Nuclear Fuel

The U.S. Department of Energy (DOE) SNF isotopic compositions, (DTN: MO0001SPADBE00.001) currently provided by the National Spent Nuclear Fuel Program for over 250 fuel types, are given in terms of the entire fuel inventories scheduled to be disposed of at the proposed repository. For example, if a fuel inventory were made up of 400 canisters, the isotopic data provided would be for all 400 canisters. Since the entire inventory of a given fuel type would not be involved in a potential canister breach event, a scaling factor should be used to adjust the total isotopic inventory to the amount involved in the event. The use of a scaling factor results in an average source term.

The National Spent Nuclear Fuel Program is developing average and bounding source terms for over 250 DOE fuel types using a template methodology (INEEL 2000). The bases for the template methodology are templates consisting of radionuclide inventories that have been precalculated using validated calculational methodologies for specific reactor types, fuel types, burnups, and decay times. The templates are first segregated by reactor and fuel type, which includes reactor moderator type, fuel cladding, fuel enrichment, and the fuel's beginning-of-life

heavy metal constituents. For each reactor and fuel type, a family of templates may be developed by parametrically varying burnup and decay times. Each template will be based on a depletion calculation with a given set of input conditions and assumptions that can be conservatively mapped to a fuel's reactor moderator type, fuel type, burnup value, and decay time. After choosing an appropriate template based on reactor and fuel type, one selects the specific template that bounds the burnup and decay time. The template radionuclides are then scaled to account for the fuel's mass and, if necessary, its burnup, in order to conservatively estimate the radionuclide inventories for the fuel.

### **8.2.3 Source Terms for Vitrified High-Level Radioactive Waste**

Post-irradiated isotopic concentrations for high-level radioactive waste forms from the Savannah River Site, Hanford, West Valley, and Idaho National Engineering and Environmental Laboratory are presented in Table 8-4. The isotopic concentrations in Table 8-4 were provided by the waste generators and documented in several topical reports. These data were used in site recommendation public and worker dose calculations, and they may be revised later for license application if the designs change. The methodology for calculating the HLW isotopic concentrations will not be discussed in this section.

### **8.2.4 Source Terms for Plutonium Disposition Waste Form**

The DOE plutonium disposition waste form program is currently on-hold and there is no decision regarding the final disposition of a small amount of plutonium that cannot be made into mixed oxide reactor fuel. Therefore, no source terms or dose calculations will be discussed in this section.

## **8.3 CATEGORY 1 OFFSITE DOSES**

### **8.3.1 GENII-S Dose Calculation Methodology**

The annual doses to a hypothetical subsistence farmer at the site boundary due to Category 1 event sequences and normal operational releases are calculated using the GENII-S (Leigh et al. 1993) dose calculation methodology described in BSC (2001, Attachment IV). Offsite public dose calculations for Category 1 event sequences consider all of the potential exposure pathways, including inhalation, ingestion, submersion, and groundshine.

Table 8-4. Example of High-Level Radioactive Waste Activity (Curies Per Canister)

Isotope	SRS	Hanford	West Valley	INEEL
Ac-227	0.00E+00	0.00E+00	9.07E-03	0.00E+00
Am-241	2.28E+01	5.72E+02	2.10E+02	2.61E+00
Am-242m	0.00E+00	0.00E+00	1.23E+00	0.00E+00
Am-243	0.00E+00	0.00E+00	1.36E+00	0.00E+00
Cd-113m	0.00E+00	1.18E+01	6.75E+00	0.00E+00
Ce-144	1.16E+02	3.50E+02	0.00E+00	1.23E+02
Cm-243	0.00E+00	0.00E+00	4.67E-01	0.00E+00
Cm-244	8.89E+01	1.03E+01	2.48E+01	5.48E-01
Co-60	8.80E+01	0.00E+00	1.57E+00	0.00E+00
Eu-154	4.14E+02	2.24E+02	2.51E+02	1.54E+02
Eu-155	2.40E+02	0.00E+00	0.00E+00	0.00E+00
Np-237	0.00E+00	2.00E-01	9.22E-02	0.00E+00
Pa-231	0.00E+00	0.00E+00	5.97E-02	0.00E+00
Pm-147	6.46E+03	1.06E+04	0.00E+00	4.09E+03
Pu-238	1.43E+03	7.40E-01	3.14E+01	8.60E+01
Pu-239	1.29E+01	1.41E+00	6.38E+00	8.92E-01
Pu-240	8.70E+00	5.46E-01	4.67E+00	8.27E-01
Pu-241	1.32E+03	2.03E+01	2.50E+02	1.61E+02
Th-228	0.00E+00	0.00E+00	3.43E-02	0.00E+00
U-232	0.00E+00	0.00E+00	2.68E-02	0.00E+00
I-129	0.00E+00	1.63E-05	0.00E+00	0.00E+00
Cs-134	6.28E+01	2.23E+02	3.78E+00	7.85E+02
Cs-137	3.87E+04	4.54E+04	2.52E+04	1.48E+04
Ru-106	7.40E+01	1.64E+02	1.82E-03	4.07E+01
Sr-90	4.28E+04	3.82E+04	2.41E+04	1.52E+04
Y-90	4.28E+04	3.82E+04	2.41E+04	1.52E+04

Source: CRWMS M&O 1999d, Attachment VI. The most recent revision to this calculation should be referenced.

NOTE: SRS = Savannah River Site; INEEL = Idaho National Engineering and Environmental Laboratory.

The total Category 1 annual dose is based on contributions from three sources: Category 1 event sequences, normal operational (routine) releases from the Waste Handling Building (WHB), and normal operational releases from the subsurface repository. The total Category 1 annual dose (mrem/yr) is generally described by the following equation:

$$D_{Cat.1}^{TOT} = D_{Cat.1}^{DBEs} + D_{NO}^{WHB} + D_{NO}^{Sub} \quad (\text{Eq. 8-2})$$

where

$$D_{Cat.1}^{DBEs} = \text{Annual dose due to all Category 1 event releases (mrem/yr)}$$

$$D_{NO}^{WHB} = \text{Annual dose due to normal operational releases from the WHB (mrem/yr)}$$

$D_{NO}^{Sub}$  = Annual dose due to normal operational releases from the subsurface repository (mrem/yr)

$D_{Cat.1}^{DBEs}$ ,  $D_{NO}^{WHB}$ , and  $D_{NO}^{Sub}$  are calculated using GENII-S. The radiological release (Ci/yr) estimates for each of these three components are used as input to GENII-S dose calculations. The annual releases due to normal operations in the WHB and normal operations in the subsurface repository have been obtained from CRWMS M&O (2000a and 2000b). Current references for annual releases should be used as appropriate. The annual release due to all Category 1 event sequences is calculated using the following equation:

$$R_{DBEs}^{TOT} = \sum_{i=1}^n R_i^{DBE} \cdot f_i \quad (\text{Eq. 8-3})$$

where

- $i$  = Index for a given Category 1 event sequence ( $i = 1, 2, \dots, n$ )
- $n$  = Total number of Category 1 event sequences
- $R_i^{DBE}$  = Radiological release due to event sequence  $i$  (Ci/event)
- $f_i$  = Frequency of event sequence  $i$  (events/yr)

To show compliance with 10 CFR 63.111, the calculated annualized (aggregate) dose,  $D_{Cat.1}^{TOT}$ , is compared with the regulatory dose limit of 15 mrem per year. In addition, the calculated TEDE due to each Category 1 event sequence and a combination of Category 1 event sequences that can occur in one single year are also compared with the regulatory dose limit of 15 mrem per year.

The external doses calculated using the GENII-S semi-infinite or finite plume model will be used to estimate the dose rates due to external exposures. The calculated dose rates will be compared with the 2 mrem per hour dose rate limit given in Table 8-1.

### 8.3.2 Dose Conversion Factors

For Category 1 dose assessment, inhalation DCFs are derived by the GENII-S code based on the dosimetric methodology from International Commission on Radiological Protection Publication 30 (ICRP 1979).

External DCFs for air submersion, water surface, soil surface, deep soil, and buried waste, used by GENII-S to calculate doses for Category 1 event sequences, are taken from input file GRDF.15 (BSC 2001, Attachment IV, Figure 3). This file is a modification to the original GENII-S input file GRDF.DAT, and incorporates the latest data from Federal Guidance Report No. 12 (Eckerman and Ryman 1993) and DTN: MO9912RIB00066.000.

Internal and external DCFs used in GENII-S were evaluated in *Dose Conversion Factor Analysis: Evaluation of GENII-S Dose Assessment Methods* (CRWMS M&O 1999e). This analysis found that, as compared to Federal Guidance Report No. 11 (Eckerman et al. 1988), GENII-S overestimates internal doses from some radionuclides and underestimates internal

doses from others, but concluded that the differences are acceptable considering the level of uncertainties inherent in the dose assessment process.

### 8.3.3 Atmospheric Dispersion Factors

Atmospheric dispersion factors are taken from *Calculations of Acute and Chronic "Chi/Q" Dispersion Estimates for a Surface Release* (CRWMS M&O 1999f). Normal operational releases are modeled as chronic releases while Category 1 event releases are modeled as acute releases. The acute exposure  $\chi/Q$  is based on a 2-hr exposure at the Exclusion Area Boundary (Regulatory Guide 1.145), which corresponds to the site boundary for the purpose of consequence analysis. The chronic exposure is based on the annual average, best-estimate exposure at the site boundary, in accordance with Regulatory Guide 1.111. Stack releases are not assumed in either the acute or chronic exposures.

The 50 percent acute  $\chi/Q$  and chronic  $\chi/Q$  are evaluated at either 8 km for subsurface releases or 11 km for surface releases. These distances should be reviewed and modified as necessary for the design used in license application.

A site boundary distance of 11-km should be used to calculate doses due to radiological releases from the WHB. This distance corresponds to the distance from the WHB ventilation exhaust shaft to the nearest point on the proposed Yucca Mountain Site Characterization Project (YMP) Withdrawal Area boundary (to the West) (DTN: MO0001YMP00001.000), which is the closest point that any member of the public could be standing or living at the time of a postulated radiological release. It is assumed that no persons will be allowed to live within the YMP Withdrawal Area and that administrative controls will be in place to evacuate any members of the public that could potentially be located within the YMP Withdrawal Area but outside of the Preclosure Controlled Area Boundary following an event sequence.

A site boundary distance of 8-km should be used to calculate potential doses due to radiological releases from the subsurface repository. This distance corresponds to the approximate distance between the potential repository and the nearest point of public access on the proposed YMP Withdrawal Area boundary (to the West) (DTN: MO0001YMP00001.000).

Atmospheric dispersion factors of  $2.98 \times 10^{-7}$  sec/m<sup>3</sup> and  $1.99 \times 10^{-7}$  sec/m<sup>3</sup> have been used to calculate offsite doses at 8 km and 11 km, respectively, due to chronic (normal operational) releases.

Atmospheric dispersion factors may be revised later for license application if the decision to use more recent meteorological data (up to 2001) is made. Most recent atmospheric dispersion data should be used in dose calculations for license application.

### 8.3.4 Breathing Rate

A breathing rate of  $2.662 \times 10^{-4}$  m<sup>3</sup>/s (266.2 cm<sup>3</sup>/s) is used by GENII-S to calculate Category 1 doses. This breathing rate is the GENII-S input parameter for reasonable representation cases (BSC 2001, Attachment IV, Figure 2).

### 8.3.5 Category 1 Event Sequences

The Category 1 dose assessment includes contributions from three sources:

- Routine radiological releases from the surface facilities
- Routine radiological releases from the subsurface facility
- Category 1 event sequences – event sequences expected to occur one or more times before permanent closure (i.e., frequency  $\geq 10^{-2}$  per year).

The total annual release from annual surface and subsurface routine releases (Ci/yr) and annualized releases from Category 1 event sequences should be calculated. The total annual release is then input to the offsite dose calculation performed using the GENII-S code (BSC 2001, Attachment IV).

### 8.3.6 Uncertainty in Consequence Analysis

Consequence analysis performed for normal operational releases and Category 1 event sequences will use either average or best-estimate input parameter values. The reasons for using average or best-estimate values are normal operations and Category 1 event sequences are expected to occur several times a year and aggregate doses are calculated for a combination of Category 1 event sequences that could occur in one single year. Therefore, each consequence analysis input parameter should be an average value that represents a waste form with different burnup values, a range of waste form damage conditions, or a range of weather conditions over one year period. For example, average source terms, annual average, best-estimate atmospheric dispersion factors, best-estimate release fractions, and best-estimate annual food consumption rates for a real member of the public will be used for Category 1 event sequences. Uncertainty analysis will not be performed for normal operational releases and Category 1 event sequences.

### 8.3.7 Category 1 Dose Calculation Examples

The offsite dose consequences for Category 1 event sequences are based on the methodology described in Section 8.2.1.

**Example of Calculating Annualized Releases**—The total estimated release from the surface facilities (4,010 Curies per year) is due entirely to Kr-85 releases from the WHB. This release was estimated based on the postulated failure of PWR and BWR spent fuel assemblies during normal handling operations (CRWMS M&O 2000a). The Waste Treatment Building is not expected to generate significant radiological emissions, based on current, best-available information (CRWMS M&O 2000a).

The total estimated releases from the subsurface facility, due to normal operations, are shown in Table 8-5 below:

Table 8-5. Example of Annual Releases from the Subsurface Due to Normal Operations

Routine Release - Subsurface (Ci/yr)		Half Life ( $T_{1/2}$ )
Activated Air		
N-16	2.909E-3	7.13 sec
Ar-41	5.728E+1	1.82 hr
Activated Dust		
N-16	1.189E-8	7.13 sec
Na-24	6.471E-3	14.96 hr
Al-28	3.963E-3	2.25 min
Si-31	7.170E-4	2.62 hr
K-42	8.041E-4	12.36 hr
Fe-55	1.492E-4	2.73 yr

This release was estimated based on the postulated activation of air and dust in the subsurface facilities during normal operations (CRWMS M&O 2000b). Subsurface releases are due to radionuclides generated by activation of air (Ar-41 and N-16) and dust (N-16, Na-24, Al-28, Si-31, K-42 and Fe-55). Nitrogen-16, Aluminum-28 and Potassium-42 are not considered in the GENII-S dose assessment because they are not included in the default GENII-S radionuclide libraries. However, their releases and half-lives are so small that their annual offsite dose contributions are insignificant. Iron-55 (Fe-55) is the only subsurface radionuclide released that has a half life measured in years (2.73), but its total Curie release ( $1.492 \times 10^{-4}$ ) is insignificant compared to Curie releases from Category 1 event sequences (BSC 2001, Attachment IX).

Fourteen Category 1 event sequences for CSNF have been identified (BSC 2001, p. IX-2). The Category 1 event frequency calculations were performed in BSC (2001, p. VII-5). Category 1 event releases are calculated based on the event frequencies (events per year) and event source terms (Ci/event) as described in BSC (2001, Attachments VII and VIII, respectively). Radiological releases due to Category 1 event sequences are annualized by multiplying the expected release from each event by the event frequency, as indicated in Equation (8-3). An example of annualized release calculations is shown in Table 8-6 for Cs-137.



Table 8-6. Example of Calculating Annualized Cesium-137 Release for Category 1 Event Sequences

Event No. (1)	Frequency (events/yr) (2)	Source Term (Ci/event) (3)	Annualized Release (Ci/yr) =(2) × (3)
1-01	2.34E-01	0	0
1-02	3.90E-02	0	0
1-03	4.22E-02	0	0
1-04	1.92E-01	0	0
1-05	4.10E-02	0	0
1-06	4.10E-02	0	0
1-07	4.10E-02	0	0
1-08	4.10E-02	0	0
1-09	4.10E-02	0	0
1-10	4.10E-02	3.29E-01	1.35E-02
1-11	4.10E-02	6.59E-01	2.70E-02
1-12	2.34E-01	1.65E-01	3.85E-02
1-13	2.34E-01	8.25E-02	1.93E-02
1-14	2.34E-01	1.65E-01	3.85E-02
Total			1.37E-01

The calculations shown in Table 8-6 should be repeated for all radionuclides in spent fuel or waste form of interest.

**Example of GENII-S Dose Calculation**—Using the aforementioned annual releases as input to GENII-S, the annual offsite TEDE dose is calculated for a receptor at the offsite boundary using the methodology described in BSC (2001, Attachment IV). An example for GENII-S dose calculation for normal operational releases or Category 1 event sequences from the surface or subsurface facility can be found in BSC (2001, Attachment IV). For these event sequences, the GENII-S input parameters are documented in BSC (2001, Attachment IV).

Because no more than 25 radionuclides can be input to GENII-S, the source term inputs must be divided into four computer runs. The source terms for four computer runs PRCHGS, PRCHP1, PRCHP2, and PRCHP3 are given in Table 8-7.

The GENII-S input data are taken as a reasonable representation case from biosphere analysis model reports. The GENII-S default parameters for the reasonable representation case can be found in BSC (2001, Attachment IV).

This calculation example used subsistence farmer's food consumption rates (BSC 2001, Attachment IV). A subsistence farmer is defined as the maximally exposed individual of the critical group who grows all of their own food using irrigation water from a well. An individual lives in the vicinity of Yucca Mountain and draws untreated ground water for drinking water supply. This individual also uses the ground water to irrigate crops and lawns and raise livestock. It was assumed that the groundwater is not contaminated during the preclosure period. Consequently, this individual will not be exposed to radiation resulting from ingestion of ground water. However, this individual will consume locally produced food; inhalation of resuspended

Table 8-7. Example of Source Terms for GENII-S Computer Runs

GENII-S Run	Radionuclide	Air Release (Ci/yr)
PRCHSF	Kr-85	4.01E+03
PRCHSB	Na-24	6.471E-03
	Si-31	7.17E-04
	Ar-41	5.728E+01
	Fe-55	1.492E-04
PRCHGS	H-3	1.46E+02
	Kr-85	1.45E+03
	I-129	2.82E-02
PRCHP1	Cl-36	3.39E-09
	Fe-55	7.72E-02
	Co-60	1.9656E-02
	Se-79	2.28E-08
	Zr-93	4.46E-07
	Nb-93m	6.49E-06
	Nb-94	4.19E-07
	Ru-106	6.14E-09
	Pd-107	4.20E-08
	Cd-113m	3.82E-06
	Sb-125	4.85E-06
	Sn-126	1.92E-07
	Pm-147	5.94E-05
	Sm-147	0
	Sm-151	1.05E-04
	Eu-154	3.35E-04
	Eu-155	2.57E-05
	Ra-226	0
	Pb-210	0
	Ra-228	0
	Cm-242	2.63E-06
	Cm-244	6.79E-04
	Cm-245	1.53E-07
	Cm-246	5.19E-08
	Cm-243	5.14E-06
PRCHP2	C-14	1.66E-07
	Ni-63	1.26E-04
	Sr-90	1.36E-02
	Tc-99	4.49E-06
	Cs-137	1.37E-01
	U-232	1.02E-08
	U-234	3.38E-07
	U-236	8.58E-08

Table 8-7. Example of Source Terms for GENII-S Computer Runs (Continued)

GENII-S Run	Radionuclide	Air Release (Ci/yr)
PRCHP2	Pa-231	1.48E-11
	Ac-227	8.03E-12
	Np-237	1.23E-07
	U-233	2.03E-11
	Th-229	0
	U-238	7.39E-08
	Pu-238	1.14E-03
	Pu-240	1.59E-04
	Am-241	9.88E-04
	Am-243	1.1E-05
	Pu-239	8.83E-05
	Ni-59	1.04E-06
	Cs-134	8.38E-05
	Cs-135	1.16E-06
	Th-230	7.39E-11
	Th-232	0
	U-235	3.67E-09
	Am-242m	3.19E-06
	Pu-242	8.18E-07
PRCHP3	Pu-241	1.23E-02
	Cs-134	8.38E-05

dust; and direct external exposure to contaminated soil. Domestic water was assumed to come from a well.

Data can be entered into GENII-S software through a series of interactive data input screens and by modifying input data files located in GENII-S directory. The GENII-S input parameters for this calculation were based on output data from the TSPA-VA (CRWMS M&O 1998), five Data Tracking Numbers (DTNs), and other source documents shown below:

- MO0003RIB00061.001. *Input Parameter Values for External and Inhalation Radiation Exposure Analysis.*
- MO9911RIB00064.000. *Environmental Transport Parameter Values for Dose Assessment.*

- MO9912RIB00066.000. *Parameter Values for Internal and External Dose Conversion Factors.*
- MO9912SPAING06.033. *Ingestion Exposure Pathway Parameters.*
- SN0002T0512299.003. *Revised Leaching Coefficients for GENII-S Code.*
- CRWMS M&O 1998. "Biosphere." Chapter 9 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document*. B000000000-01717-4301-00009 REV 01.
- CRWMS M&O 1999f. *Calculations of Acute and Chronic "Chi/Q" Dispersion Estimates for a Surface Release*. TDR-MGR-MM-000001 REV 00.
- CRWMS M&O 2000a. *Estimated Annual MGR Normal Radiological Release*. Input Transmittal RSO-SUF-99389.T.a.
- CRWMS M&O 2000c. *Non-Disruptive Event Biosphere Dose Conversion Factors*. ANL-MGR-MD-000009 REV 00.

Three data files in GENII-S were modified to accommodate the results of site-specific studies on the GENII-S input parameters. The original names of these three files are FTRANS.DAT, DEFAULT.IN, and GRDF.DAT.

FTRANS.DAT is the default GENII-S food transfer and soil leaching factor library. The soil leaching factors are important parameters for determining radionuclide buildup in soil. The food transfer factors relate concentrations of elements in soil to concentrations in farm products grown in that soil and concentrations in animal feed to concentrations in animal products. FTRANS.DAT was used in computer runs PRCHSF, PRCHGS, and PRCHP1. An updated FTRANS.DAT, renamed as FTRANRR.TXT, was developed and used in computer runs PRCHP2 and PRCHP3.

Examples of the GENII-S input parameters, the FTRANRR.TXT file, the food transfer and soil leaching factor library, the DEF\_RR.TXT file, and the GRAF.15 file can be obtained from BSC (2001, Attachment IV). These files were used in GENII-S dose calculations.

## **8.4 CATEGORY 2 OFFSITE DOSES**

Only inhalation and air submersion doses are considered in calculating offsite doses from Category 2 event sequences. The ingestion pathway, if it occurs, is a slow-to-develop pathway and is not considered an immediate threat to an exposed population in the same sense as airborne plume exposures. Therefore, the ingestion pathway is not included in the calculation of the radiological doses resulting from Category 2 event sequences for comparison against the dose criteria given in Table 8-1.

Exposure through the ingestion pathway occurs when radioactive materials that have been deposited offsite are ingested, either by eating crops grown in contaminated soil, or through inadvertent ingestion of trace amounts of contaminated soil, or through ingestion of

contaminated drinking water. Potential doses from the ingestion pathway are not included in the comparison to the regulatory dose limits because during the preclosure operations period there would be interdiction programs in place (to be established in a DOE emergency response program) to prevent the ingestion of contaminated food and water in the event of a Category 2 event sequence. The U.S. Environmental Protection Agency provides protective action guides for radiological protection guidance for federal, state, and local government in the *Manual of Protective Actions for Nuclear Incidents* (EPA 1980) that may be used for responding to a nuclear incident or radiological emergency. Protective action guides are defined as the projected dose to a standard man from an unplanned release of radioactive material at which a specific protective action to reduce or avoid that dose is warranted. The protective actions recommended to avoid or reduce the radiation dose are based on exposure pathway (i.e., inhalation, plume, ground deposition) and incident phase (i.e., early, intermediate, late). The *Criteria for the Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants* (Podolak et al. 1988) provides a basis for NRC licensees and state and local governments to develop radiological response plans and improve emergency preparedness. In this report it is stated that the basis for the choice of protective actions will be established. Further, the protective measures to be used will be established for the ingestion pathway, including the methods of protecting the public from consumption of contaminated foodstuffs. It is also stated that the plan shall identify procedures for detecting contamination, for estimating the dose commitment consequences of uncontrolled ingestion, and for imposing protection procedures such as impoundment, decontamination, processing, decay, product diversion, and preservation. Category 2 event sequences result in acute releases over a period of a few hours and the doses from these pathways (i.e., ingestion via crops, soil, or water) may be controlled by interdiction as needed, thus precluding ingestion, water immersion and contaminated soil (groundshine) source term pathways.

Two methods can be used to calculate offsite doses for Category 2 event sequences. The first method uses GENII-S to calculate the offsite doses for Category 2 event sequences. However, the code does not calculate the SDE or the LDE. Therefore, separate spreadsheet calculations for the SDE and LDE are needed. The methodology for calculating the SDE and LDE is discussed in Section 8.4.1. The Category 2 event sequence dose calculations using GENII-S are similar to the Category 1 event dose calculations discussed in Section 8.3.7, except for the ingestion pathway input data. As discussed above, the ingestion dose is not calculated for Category 2 event sequences.

The second method uses the spreadsheet calculations entirely to calculate TEDE, CDE+DDE, SDE, and LDE. The methodology for calculating these doses is presented in the following sections.

#### **8.4.1 Category 2 Event Dose Calculation Methodology**

In Category 2 event sequences, radioactive materials are assumed to be released as a ground-level radioactive plume, which is dispersed en route to the site boundary and results in a 2-hr acute individual exposure. All potential radiation doses from inhalation and air submersion pathways are considered for Category 2 event sequences.

The TEDE dose measure is expressed as:

$$TEDE = CEDE + EDE = \sum_j D_{j,effective}^{inh} + \sum_j D_{j,effective}^{sub} \quad (\text{Eq. 8-4})$$

where

$$\begin{aligned} TEDE &= \text{TEDE (rem)} \\ CEDE &= \text{CEDE (rem)} \\ EDE &= \text{EDE (rem)} \\ D_{j,effective}^{inh} &= \text{Whole body effective inhalation dose from the } j^{\text{th}} \text{ isotope (rem)} \\ D_{j,effective}^{sub} &= \text{Whole body effective submersion dose from the } j^{\text{th}} \text{ isotope (rem)} \end{aligned}$$

The CDE+DDE dose measure is expressed as:

$$CDE_k + EDE_k = \sum_j D_{j,k}^{inh} + \sum_j D_{j,k}^{sub} \quad \text{where } k \neq \text{effective or skin} \quad (\text{Eq. 8-5})$$

where

$$\begin{aligned} CDE_k &= \text{CDE to the } k^{\text{th}} \text{ organ (rem)} \\ D_{j,k}^{inh} &= \text{Radiation dose from the } j^{\text{th}} \text{ isotope to the } k^{\text{th}} \text{ organ due to inhalation (rem)} \\ D_{j,k}^{sub} &= \text{Radiation dose from the } j^{\text{th}} \text{ isotope to the } k^{\text{th}} \text{ organ due to air submersion (rem)} \\ k &= \text{Organ index, where organs are gonads, breast, lungs, red marrow, bone surface, thyroid, and remainder} \end{aligned}$$

The inhalation and air submersion doses shown above can be further expressed as:

$$D_{j,k}^{inh} = ST_j \times FA \times \frac{\chi}{Q} \times BR \times conv \times DCF_{j,k}^{inh} \quad (\text{Eq. 8-6})$$

$$D_{j,k}^{sub} = ST_j \times FA \times \frac{\chi}{Q} \times conv \times DCF_{j,k}^{sub} \quad (\text{Eq. 8-7})$$

where

$$\begin{aligned} ST_j &= \text{Inventory source term release per FA or per canister for the } j^{\text{th}} \text{ isotope (Ci/FA or Ci/canister)} \\ FA &= \text{Number of fuel assemblies or canisters involved in the release (\# FAs or \# canisters)} \\ \frac{\chi}{Q} &= \text{Atmospheric dispersion factor (s/m}^3\text{)} \\ BR &= \text{Breathing rate (m}^3\text{/s)} \end{aligned}$$

$DCF_{j,k}^{inh}$  = Inhalation DCF of the  $j^{th}$  isotope for the  $k^{th}$  organ (Sv/Bq) (Eckerman et al. 1988)

$DCF_{j,k}^{sub}$  = Air submersion DCF of the  $j^{th}$  isotope for the  $k^{th}$  organ [(Sv-m<sup>3</sup>)/(Bq-s)] (Eckerman and Ryman 1993)

$conv$  = DCF unit conversion factor:  $3.7 \times 10^{-12}$  (rem-Bq)/(Ci-Sv) (Eckerman et al. 1988)

The SDE is equal to

$$SDE = \sum_j D_{j,skin}^{sub} \quad (\text{Eq. 8-8})$$

where

$SDE$  = SDE (rem)

$D_{j,skin}^{sub}$  = Radiation dose from the  $j^{th}$  isotope to the skin due to air submersion (rem)

In NRC 2000, it is stated that compliance with the LDE limit is achieved if the sum of the SDE and the TEDE does not exceed 15 rem. Therefore, the LDE may be expressed as

$$LDE = TEDE + SDE \quad (\text{Eq. 8-9})$$

where

$LDE$  = LDE (rem)

#### 8.4.2 Dose Conversion Factors

DCFs for inhalation are dependent on the chemical form of the radionuclide, which is represented by the lung clearance class (D = daily, W = weekly, Y = yearly) and the fractional uptake from the small intestine to blood (f<sub>1</sub>). Some isotopes have only one lung clearance class (e.g., H-3), whereas others have multiple lung clearance classes (e.g., Pu-239). For Category 2 inhalation dose assessment of radionuclides with multiple lung clearance classes, the lung clearance class corresponding to the oxide form of the radionuclide (Eckerman et al. 1988) is assumed. The inhalation DCFs utilized for Category 2 dose assessment are from Table 2.1 of Federal Guidance Report No.11 (Eckerman et al. 1988). The air submersion DCFs for gonads, breast, lungs, red marrow, bone surface, thyroid, remainder, effective (i.e., whole body), and skin for Category 2 dose assessment are from Table III.1 of Federal Guidance Report No. 12 (Eckerman and Ryman 1993). The inhalation and submersion DCFs used for Category 2 dose assessment are listed in BSC (2001, Attachment VIII).

#### 8.4.3 Atmospheric Dispersion Factors

Atmospheric dispersion factors are taken from *Calculations of Acute and Chronic "Chi/Q" Dispersion Estimates for a Surface Release* (CRWMS M&O 1999f). Category 2 event releases are modeled as acute releases. The maximum sector 99.5 percentile acute (0.5 percent exceedance)  $\chi/Q$  values are used to calculate the maximum doses and the

50 percentile acute  $\chi/Q$  values are used to calculate the mean doses. The selection of the maximum sector 99.5 percent  $\chi/Q$  value is based on it being larger than the 95 percent overall site  $\chi/Q$  value, per Regulatory Guide 1.145.

Atmospheric dispersion factors of  $2.94 \times 10^{-5}$  sec/m<sup>3</sup> and  $2.17 \times 10^{-5}$  sec/m<sup>3</sup> were used to calculate the maximum offsite doses (BSC 2001) at 8 km and 11 km, respectively, due to acute (Category 2 event) releases. These  $\chi/Q$  values are based on the 99.5 percentile values at each distance, which corresponds with Wind Sector 14 (West-Northwest to East-Southeast). The maximum doses can be used to account for the uncertainty/variability of the input parameters (see Section 8.4.6). The mean doses based on 50 percentile acute  $\chi/Q$  values should also be calculated.

Atmospheric dispersion factors may be revised later for license application if the decision to use more recent meteorological data (up to 2001) is made. Most recent atmospheric dispersion data should be used in dose calculations for license application.

#### **8.4.4 Breathing Rate**

For calculating offsite doses due to acute releases (Category 2 event sequences), a reference man breathing rate of  $3.3 \times 10^{-4}$  m<sup>3</sup>/s should be used (ICRP 1975).

#### **8.4.5 Category 2 Event Sequences**

Thirteen Category 2 event sequences for the surface facilities and one Category 2 event for the subsurface facility have been identified (BSC 2001, p. X-2). The Category 2 event frequency calculations were performed in BSC (2001, p. VII-5).

The Category 2 TEDE dose (Equation 8-4) to an individual receptor at the offsite boundary is calculated based on the methodology in Section 8.4.1. Category 2 doses are calculated on a per event basis for each of the Category 2 event sequences. Details of the Category 2 dose calculations are provided in BSC (2001, Attachment X).

#### **8.4.6 Uncertainty in Consequence Analysis**

Uncertainty analysis should be performed to account for uncertainty and variability in input parameter values. Three methods are used in the determination of uncertainty in consequence analysis.

**Method of Stacking of Conservatism** - The first method uses conservative or bounding values for all input parameters, which stacks conservatism on top of conservatism. This ensures that the calculated public and workers doses are conservative. For example, bounding event sequences, bounding source terms, and 99.5 percent maximum sector acute  $\chi/Q$  values can be used to calculate offsite doses for Category 2 event sequences.

**Using GENII-S** - The second method is the GENII-S Monte Carlo method. The "S" in the name of this consequence analysis code stands for SUNS (Sensitivity and Uncertainty Analysis Shell). For each of the variable input to GENII-S, a range of parameter values and distribution types are specified. The available distribution types in GENII-S include fixed, normal, lognormal,



triangular, uniform, and empirical. Sampling from the ranges and distributions assigned to the variable input is performed within a GENII-S routine using the Monte Carlo sampling technique. GENII-S variable input parameter ranges and distribution types are documented in CRWMS M&O (1998, Table 9-3). A GENII-S statistical run generates mean, minimum, and maximum values, as well as standard deviation for each output variable. The graphical output includes cumulative distribution functions, complementary cumulative distribution functions, and histogram. These features of GENII-S provide insights into the significance of uncertainties in consequence analyses. The GENII-S output variables include organ doses, total inhalation EDE, total ingestion EDE, internal EDE, external dose, and annual EDE. The mean doses calculated by GENII-S will be compared with the regulatory dose limits in Table 8-1.

Running GENII-S includes the following input information provided by the analyst:

- Scenario options – near-field or far-field; population or individual dose; acute or chronic
- Transport options – air; surface water; biotic; waste form degradation
- Exposure pathways and options – external (immersion in finite or infinite plume, ground exposure); inhalation; ingestion
- Select radionuclides - select up to 25 from a built-in radionuclide library
- Run options – select deterministic or statistical output
- Correlations between variables.

GENII-S input variables and values are discussed in Section 8.7.3.

**Using Excel with @RISK** – A spreadsheet model is created to multiply the factors such as source term, airborne release fraction (ARF), respirable fraction (RF), breathing rate, DCFs, and atmospheric dispersion factors. An uncertainty distribution is assigned to each factor in the consequence model. The @RISK program is used to perform a Monte Carlo or Latin Hypercube simulation to propagate all of the uncertain variables to the output. The results are tabular giving the mean, median, standard deviation, and percentiles. The results may be plotted as probability distribution functions, cumulative distribution functions or histograms using Excel graphics.

#### **8.4.7 Category 2 Dose Calculation Example**

An example of Category 2 dose calculation is Event No. 12, Shipping Cask Drop into Cask Unloading Pool (BSC 2001, Attachments X). An example of inhalation dose calculation for this event sequence is shown in Table 8-8.

Table 8-8. Inhalation Dose Calculation Example

Group	Source Term (rem/FA)	# of FA's	Release Fraction	Mitigation Factor	Breathing Rate (m <sup>3</sup> /sec)	$\chi/Q$ (sec/m <sup>3</sup> )	Offsite Dose (rem)	% of Total
Particulates	1.90E+12	68	1.50E-07	1.0E-02	3.3E-04	2.17E-05	1.38E-03	5.96
Noble Gas	1.36E+04	68	3.00E-01	1.0E+00	3.3E-04	2.17E-05	1.99E-03	8.57
Cesium	1.77E+09	68	2.00E-04	1.0E-02	3.3E-04	2.17E-05	1.73E-03	7.43
Strontium	6.03E+09	68	1.50E-07	1.0E-02	3.3E-04	2.17E-05	4.41E-06	0.02
Crud	1.24E+07	68	3.00E-01	1.0E-02	3.3E-04	2.17E-05	1.81E-02	78.02
Total							2.32E-02	100.00

NOTE: Inhalation dose to the whole body. Event: 2-12, Shipping cask drop into cask unloading pool. Fuel type: maximum BWR

In Table 8-8, source term (rem/FA) to each organ, for each nuclide group, are calculated by multiplying the crud inventory (Ci/FA) or the inventory in Table 8-2 by its respective inhalation DCF for the organ taken from Federal Guidance Report 11 (Eckerman et al. 1988). For example, in Table 8-8, the crud group includes Co-60 and Fe-55. Therefore, the crud source term (rem/FA) is equal to:

$$\text{Crud source term (rem/FA)} = \text{Co-60 inventory (Ci/FA)} \times \text{inhalation DCF}_{\text{Co-60}} (\text{Sv/Bq}) \times \text{conv} \\ + \text{Fe-55 inventory (Ci/FA)} \times \text{inhalation DCF}_{\text{Fe-55}} (\text{Sv/Bq}) \times \text{conv}$$

where conv is the DCF unit conversion factor:  $3.7 \times 10^{-12}$  (rem-Bq)/(Ci-Sv) (Eckerman et al. 1988). The 56 radionuclides listed in Table 8-2 are divided into four groups: particulates, noble gas, cesium, and strontium. The noble gas group includes H-3, I-129, and Kr-85. The cesium group includes Cs-134, Cs-135, and Cs-137. The strontium group includes Sr-90. The rest of radionuclides in Table 8-2 are included in the particulates group. The release fraction for each group is taken from CRWMS M&O (1999c). The mitigation factor is discussed in Section 8.7.

An example of an air submersion dose calculation including only H-3 and Kr-85 is shown in Table 8-9.

In Table 8-9, air submersion dose rates (rem-m<sup>3</sup>/FA-s) to each organ, for H-3 and Kr-85, are calculated by multiplying the inventories (Ci/FA) of H-3 and Kr-85 in Table 8-2 by its respective air submersion DCF for the organ taken from Federal Guidance Report 12 (Eckerman and Ryman 1993).

An example of a dose summary table is given in Table 8-10.

Table 8-9. Submersion Dose Calculation Example

Submersion Dose (Due to Noble Gases H-3 and Kr-85 Only)						
Event:	2-12	Shipping Cask Drop Into Cask Unloading Pool				
Fuel Type:	Maximum BWR					
	Maximum BWR Submersion Dose Rate (rem-m <sup>3</sup> /FA-s)	# of FA's	Release Fraction	Mitigation Factor	$\chi/Q$ (sec/m <sup>3</sup> )	Offsite Dose (rem)
(Column No.)	(1)	(2)	(3)	(4)	(5)	(6)=(1)*(2)*(3)*(4)*(5)
Gonad	8.79E-01	68	3.00E-01	1.00E+00	2.17E-05	3.89E-04
Breast	1.01E+00	68	3.00E-01	1.00E+00	2.17E-05	4.45E-04
Lung	8.58E-01	68	3.00E-01	1.00E+00	2.17E-05	3.80E-04
R Marrow	8.19E-01	68	3.00E-01	1.00E+00	2.17E-05	3.62E-04
B Surface	1.65E+00	68	3.00E-01	1.00E+00	2.17E-05	7.31E-04
Thyroid	8.86E-01	68	3.00E-01	1.00E+00	2.17E-05	3.92E-04
Remainder	8.19E-01	68	3.00E-01	1.00E+00	2.17E-05	3.62E-04
Whole Body	8.94E-01	68	3.00E-01	1.00E+00	2.17E-05	3.96E-04
Skin	9.91E+01	68	3.00E-01	1.00E+00	2.17E-05	4.39E-02
Eye Lens	0.00E+00	68	3.00E-01	1.00E+00	2.17E-05	0.00E+00

Table 8-10. Example of Dose Summary Table

Summary of Inhalation and Submersion Dose Calculations						
Event:	2-12	Shipping Cask Drop Into Cask Unloading Pool				
Fuel Type:	Maximum BWR					
Organ	Inhalation Dose (rem)	Dose Term for Regulation	Submersion Dose (rem)	Dose Term for Regulation	Sum of Inhalation and Submersion (rem)	Dose Term for Regulation
Gonad	5.43E-03		3.89E-04		5.82E-03	
Breast	8.95E-03		4.45E-04		9.40E-03	
Lung	1.10E-01	<MAX CDE	3.80E-04		1.10E-01	<MAX CDE
R Marrow	1.03E-02		3.62E-04		1.04E-02	
B Surface	2.75E-02		7.31E-04		2.82E-02	
Thyroid	1.98E-02		3.92E-04		2.02E-02	
Remainder	1.57E-02		3.62E-04		1.61E-02	
Whole Body	2.32E-02	<CEDE	3.96E-04	<DDE	2.36E-02	<TEDE=CEDE +DDE
Eye Lens	N/A		0.00E+00	<EYE		
Skin	N/A		4.39E-02	<SKIN		
					1.10E-01	<MAX CDE + DDE

NOTE: The maximum organ inhalation dose (CDE) is to the Thyroid.  
 Bolded results are added together to calculate TEDE (CEDE + DDE).

**RESULTS:**

Dose Term	Offsite Dose (rem)
TEDE =	2.36E-02
Max. CDE + DDE =	1.10E-01
Eye =	0.00E+00
Skin =	4.39E-02

**8.5 CATEGORY 1 WORKER DOSES AND EXPOSURES**

The expected annual dose to a non-involved worker located 100 meters from the release point is calculated for all Category 1 event releases, including normal operational releases from the surface and subsurface facilities. The dose calculation for the non-involved worker assumes that a single worker receives a chronic exposure, at a distance of 100 m from the release point, from potential Category 1 event sequences and normal operational releases in a single year. No credit is taken for worker training, administrative controls, or emergency response procedures to minimize worker exposures to Category 1 event sequences. In general, the annualized worker doses are based on following assumptions:

- Chronic exposure over a period of one year.
- Frequency weighted dose contributions from Category 1 event sequences.
- Only inhalation and submersion pathways are considered. Ingestion and ground contamination pathways are not included because there will be no crops produced onsite and the radiation protection program will prevent worker exposures to contaminated soil.
- Mitigated (high efficiency particulate air [HEPA]-filtered) particulate releases from the surface facilities.
- Chronic  $\chi/Q$  evaluated at a distance of 100 m from the WHB (surface release) or the subsurface facility (subsurface release) to the nearest non-involved worker.

An example of Category 1 non-involved worker dose calculations can be found in BSC (2001, Attachment V).

The occupational dose limits for adults are specified in 10 CFR 20.1201, which include:

- An annual limit of either (whichever is more limiting): TEDE of 5 rem, or the sum of the DDE and the CDE to any individual organ or tissue, other than the lens of the eye, of 50 rem.
- Annual limits to the lens of the eye and to the skin: an LDE of 15 rem, and a SDE of 50 rem.

The calculated worker doses should be compared with the above regulatory dose limits.

## 8.6 RELEASE FRACTIONS

### 8.6.1 Commercial Spent Nuclear Fuel Release Fractions

The total release fraction is defined as the fraction of total inventory of a given radionuclide that is released to the environment from a waste form following an event sequence (e.g., drop of a fuel element). The release fraction for CSNF is primarily a measure of the inventory of fuel particulates, gases and volatile species present in the plenum region (a.k.a. gap release) of a breached fuel element.

The total release fraction for calculating the source term released from Category 1 event sequences is a function of the cladding damage fraction (DF), cladding release fraction (CR), ARF, RF, and the local deposition factor (DEP):

$$\text{Total Release Fraction} = DF \times CR \times ARF \times RF \times DEP \quad (\text{Eq. 8-10})$$

The DF is the fraction of fuel rods that are assumed to fail by cladding breach during a event sequence. The CR is the fraction of the total radionuclide inventory in the gap between fuel elements and cladding. The ARF is the fraction of the total radionuclide inventory in damaged fuel rods that is released from breached cladding and is suspended in air as an aerosol following an event. The RF is the fraction of airborne radionuclide particles having an aerodynamic equivalent diameter of 10- $\mu\text{m}$  and less, which can be transported through air, inhaled into the human respiratory system, and contribute to the inhalation dose. The DEP is the fraction of the ARF that reaches the ventilation system after local deposition (i.e., plate-out and gravitational settling) within the WHB. The mitigation factor is the fraction of radionuclides that is released to the environment after escaping from the HEPA filters in the WHB ventilation system.

The ARF and RF parameters for Category 1 event sequences involving CSNF releases in air were based on *Commercial SNF Accident Release Fractions* (CRWMS M&O 1999c). The only exception for releases in air is the RF for Category 1 releases. In this case, a RF of 1.0 is assumed, which means all particle sizes are included in the dose calculation for the ingestion pathway. Particle sizes larger than respirable sizes could deposit on the ground and contribute to radiation doses through the ingestion pathway (i.e., human consumption of crops, fruits, and vegetables grown on the contaminated soil).

For events occurring in a spent fuel pool, an ARF equal to zero ( $ARF = 0$ ) is assumed for all particulate and volatile species. In these events, only the noble gases are released from the pool. The release fractions for CSNF releases in air and water for Category 1 and Category 2 dose assessments are shown in BSC (2001, Attachments IX and X, respectively).

The CSNF release fractions in air, for Category 2 dose assessment, are shown in Table 8-11.

It should be noted that the release fractions given in Table 8-11 are very conservative (e.g., crud release fraction of 0.3) and they are being revised to reduce conservatism.

Table 8-11. Example of CSNF Release Fractions in Air for Category 2 Event Sequences

Nuclide	Airborne Release Fraction	RF	Effective Release Fraction
H-3	0.3	1.0	0.3
Kr-85	0.3	1.0	0.3
I-129	0.3	1.0	0.3
Cs	2.0E-04	1.0	2.0E-04
Sr	3.0E-05	5.0E-03	1.5E-07
Ru	3.0E-05	5.0E-03	1.5E-07
Crud	1.0E+00	3.0E-01	3.0E-01
Fuel Fines	3.0E-05	5.0E-03	1.5E-07

NOTE: These release fractions are conservative estimates based analysis in *Commercial SNF Accident Release Fractions* (CRWMS M&O 1999c).

### 8.6.2 U.S. Department of Energy Spent Nuclear Fuel Release Fractions

There are a large number of DOE spent nuclear fuels (DSNFs) with different characteristics. These fuel types are placed in six event groups: stable metal, intact; stable metal, not intact; non-metal, intact; non-metal, intact; other, intact; other, not intact (CRWMS M&O 1999g). Since each event group has different physical and chemical characteristics, the release fractions are assigned to each event group. For each of these groups, a bounding fuel that bounds the offsite dose consequences of all fuels in that group was selected (CRWMS M&O 1999g). Screening analysis, to identify bounding event sequences from a list of Category 2 event sequences, has been performed in CRWMS M&O (1999h). To make the consequence analysis tractable, only event sequences involving bounding fuels in canisters were analyzed.

ARFs and RFs for DSNF can be found in CRWMS M&O (1999c and 1999g). Credit for a reduction in DSNF source terms due to retention of radionuclides in canisters can be taken by use of the canister leakpath factor. Canister leakpath factors for DOE SNF can be found in CRWMS M&O (2000d).

## 8.7 MITIGATION FACTOR

The mitigation factor refers to the mitigation of particulates provided by HEPA filters that are present in the WHB ventilation system. A mitigation factor of 0.01 should be applied to all particulate releases to calculate offsite doses. The mitigation factor of 0.01 corresponds to a particulate removal efficiency of 99 percent, which is consistent with the NRC-recommended credit for accident dose evaluations in Regulatory Guides 1.140 and 1.52.

Due to the high reliability of the WHB Heating, HVAC system, all event sequences that involved a failure of the HVAC system have been found to be beyond Category 2 event sequences (BSC 2001).

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## ACRONYMS

BI	Birnbaum Importance
EF	error factor
FV	Fussell-Vesely importance
IM	importance measure
LA	license application
LB	lower bound
LN	lognormal
NRC	U.S. Nuclear Regulatory Commission
PDF	probability density function
PLC	programmable logic controller
PRA	probabilistic risk analysis
PSA	preclosure safety analysis
RAW	risk-achievement worth
RRW	risk-reduction worth
SSC	structures, systems, and components
UB	upper bound

## 9. UNCERTAINTY AND SENSITIVITY ANALYSIS, GENERAL CONCEPTS, AND METHODS

### 9.1 INTRODUCTION

This section provides guidance on methods for identifying, quantifying, propagating, and interpreting uncertainties in event sequence frequency and consequence analyses. The material provides general concepts and methods for qualitatively and quantitatively assessing uncertainties associated with event sequence frequency analysis, or radiological consequence analysis.

### 9.2 OVERVIEW OF APPROACH

Two primary sources were used to develop this section: the *PRA Procedures Guide* (NRC 1983, Chapter 12) and the CPQRA guidelines (AIChE 1989). Additional information has been incorporated from several sources including Regulatory Guide 1.174, NUREG-0800 (NRC 1987, Chapter 19), and NUREG/BR-0184 (NRC 2001).

Calculations of probabilities, frequencies, source terms, and doses used in preclosure safety are often expressed as single numbers (i.e., point values) for simplicity and convenience of presentation. It is generally understood, however, that virtually every input parameter and every output value has uncertainty associated with it. When the point value represents the mean or expectation value of the quantity, it is often a sufficient parameter for decision making or compliance evaluation because the mean value represents a probability-weighted integration over the uncertainty range. When the mean of an output quantity like event sequence frequency is far (e.g., an order of magnitude or more) from a decision point like the frequency boundary between Category 1 and Category 2, then the analyst and U.S. Nuclear Regulatory Commission (NRC) has confidence that the sequence is properly categorized. But when the mean is only a factor of two or so from the boundary, the shape and range of the uncertainty distribution come into question. The probability that the true value of the frequency is in the other category may be unacceptable. In either case, an expression of uncertainty distribution is needed to support the decision-making. This section describes the process.

The geologic repository is a first-of-a-kind facility. There is no prototype or repository-specific test facility from to derive equipment performance information. The PSA must rely on generic or surrogate information. The application of such information to the repository introduces uncertainty because the exact equipment represented in information bases may not be used, and the physical and operational environments at the repository may not be represented in the surrogate information. Further, portions of the facility design may not be mature or finely detailed at the time of license application (LA). Such issues are sources of uncertainty.

Section 7.5 describes the processes for defining the uncertainty distributions for parameters that are inputs to the PSA fault-tree and event-tree modeling. Therefore, these sources of uncertainty are briefly mentioned in this section. This section concentrates on how uncertainties are identified, propagated through the analyses, and examined through sensitivity analyses.

Mathematically, the uncertainty in an input parameter is expressed by a probability distribution that represents the probability that a given value of, for example, an event sequence frequency, is

the true frequency. Each input parameter is expressed in terms of a probability density function (PDF) that represent the uncertainty in that parameter, and the calculation of an event sequence frequency requires the multiplication and addition of scores of input parameters. The output of the frequency quantification is also represented as a PDF that reflects not only the product or sum of all of the input median (or mean) values, but also the uncertainty distributions of all of the input PDFs. This effect is termed propagation of uncertainties.

The propagation of uncertainties can be performed by hand under certain conditions. Generally, the solution is too complex, however, so computer solutions are used. For the PSA, therefore, the Monte Carlo or Latin Hypercube methods will be used for most of the uncertainty analyses of event sequence frequencies. These techniques are embedded in the SAPHIRE workstation, but can also be performed in a Microsoft Excel spreadsheet using the @RISK add-in. In this regard, much of the real effort in uncertainty analysis is that of identifying and quantifying the sources of uncertainty and representing it as an appropriate, quantitative PDF.

In some instances, the source of uncertainty may not be amenable to being expressed as a PDF. For example, there may be uncertainty regarding the presence of a certain operational or environmental condition, or a design feature (e.g., does the power supply system have redundant trains?). In such cases, a sensitivity analysis may be performed to explore the significance of assuming one condition over another (e.g., calculate the event frequency with single-train and with two-train redundancy to evaluate the significance of the alternative design on the results).

Many of the concepts and methods of uncertainty analysis have been developed around the statistical properties of the Normal distribution. Section 7.5 describes other distributions that serve significant roles in the ability to quantify and propagate uncertainty. In particular, the lognormal (LN) distribution has become the workhorse for uncertainty analysis in probabilistic risk analysis (PRA) and, likewise, will have substantial application in the PSA.

Subsection 9.4 presents the details of the approach.

## **9.3 DETAILS OF APPROACH**

This section describes the basic approaches for applying and interpreting measures of uncertainty in the PSA. The discussion will include identification of sources of uncertainty and means to evaluate and interpret uncertainty, both qualitatively and quantitatively. The discussion includes guidance on when to use sensitivity analysis and importance analysis as means to evaluate the significance of sources of uncertainty.

### **9.3.1 Background**

The well known bell curve of the Normal Distribution is a typical example of an expression of the uncertainty, or variability, that is known to be present in a measured parameter. The bell curve is a PDF. Such variability is known as random or chance errors. There is a true, or most representative, value of some parameter (e.g., the tensile strength) of a certain kind of steel. Results of repeated tensile tests on several specimens are expected to give slightly different values, some above and some below some central value. The statistical analysis produces an expected value (the mean) and a measure of the dispersion, characterized by the variance (or the standard deviation). More recently, such chance variability is termed aleatory uncertainty.

Most analysts are familiar with basic statistical concepts that express random variability in measured parameters in terms of the number standard deviations from the mean, or the 95% confidence level. In risk analysis, the ratio of the 95th percentile to the median (which is also the mean for the normal distribution) is termed the error factor (EF). For the LN distribution, the mean is not equal to the median but is readily calculated from the median and the EF.

Similarly, most analysts are familiar with the concept of propagating uncertainties through any calculations that use two or more uncertain parameters (e.g., in adding two quantities, the standard deviation of the sum is calculated value as the root-of-sum-of-squares as the standard deviation of each input parameter, while the mean of the sum is the sum of the means of the input parameters). The greater the number of uncertain variables that are combined, the wider becomes the dispersion (uncertainty) in the output.

In fault tree and event sequence quantification, however, the end result may involve sums and products of many quantities having different kinds of probability distributions, other than the normal. These facts make the propagation of uncertainty more difficult and usually not amenable to an analytic (i.e., closed form) solution. Therefore, alternative methods must be applied, including approximations and computer simulation (e.g., Monte Carlo analysis).

The concept of epistemic uncertainty and means of dealing with it are not as well known to most analysts. The term encompasses many forms of knowledge uncertainty that can be considered in assessing the frequency and consequences. Epistemic uncertainties include, for example, model uncertainties, applicability of generic information and parameters, and effects of environmental factors, and human error rates.

For the PSA, analysts will have to make judgements on how to apply information (e.g., failure rates for equipment, human error rates, and radionuclide release fractions) that are adapted from various sources. The analysts must decide on whether or not to adjust the best-estimate value (mean or median) to suit repository conditions, alter the EF, or pool information from multiple sources.

Many of the parameters used in the PSA modeling are not amenable to direct physical measurement as in a laboratory, but are derived from operational (field) data in many instances. The parameters needed include equipment failure rates, human error rates, and equipment repair times. For example, to estimate a failure rate for a component or system like a gantry crane, the kind of information used is 1) a count of the number of failures and 2) the time in which the failures occurred. For most industrial components or systems, the information gathered is imprecise and subject to considerable uncertainties (i.e., the information is not collected during controlled experiments). The raw information may involve scouring operating logbooks, maintenance records, and estimates of operational time. Generally, there are only a few components of a given type at a given plant in the sample. Results of several different failure modes may be intermixed in the information. The preclosure safety analyst must establish some expression of uncertainty for the derived failure rate and its applicability to repository operations. The uncertainty in this instance will include aleatory and epistemic uncertainty (see Section 7.6).

In a few cases, component failure rate may be developed from reliability tests conducted by manufacturers or the military. For example, a large lot of solid-state devices are subjected to

operational tests. The number of failures is precisely known and the time-on-test is precisely known. Further, post-test examinations can reveal the precise mode or cause of each failure. In such instances, the variability in derived failure rate is aleatory uncertainty.

When the source of uncertainty cannot be described as a PDF, such as uncertainty in a design configuration, then sensitivity analyses may be used to examine the effect of alternative configurations.

With such PDFs defined for all parameters in the event probabilities in fault tree or fault tree models, the uncertainty can be propagated quantitatively by one of the various methods. In this guide, the Monte Carlo-Latin Hypercube computer-based approach is the primary method. Other methods are described.

### **9.3.2 Identifying Sources of Uncertainty in Models and Input Information**

Until the mid-1990s, uncertainty in risk analysis was considered to arise from three sources: parameter uncertainty, model uncertainty, and completeness uncertainty. In more recent risk analysis literature, the terms aleatory (chance) uncertainty, and epistemic (knowledge) uncertainty have been introduced. However, these terms are essentially a re-packaging of the prior concepts. These newer terms are not used in this section unless the distinction is important.

The parameter uncertainty is further divided into 1) randomness inherent in any measured quantity, and 2) applicability uncertainty (e.g., using generic failure rate data to a specific facility). Both of these sources of uncertainty can be quantified and propagated through frequency and consequence analyses. Further, the significance of such uncertainties can be evaluated through sensitivity analyses (e.g., by letting a given parameter go to an extreme value) and importance analyses.

Model uncertainty is treated more philosophically, wherein the analyst recognizes that models are abstractions of reality, and therefore quantification was limited. Model uncertainty for systems safety include the use of event trees and fault trees that may not realistically model dependent failures (e.g., by using a simple beta-factor model) or human interactions. Event free and fault free modeling are generally accepted methods and will not be subjected to any uncertainty analysis with respect to alternative models. The logic models will be checked for accuracy. Uncertainties in specific modeling elements, such as mapping the physical configuration of equipment or systems into the logic models and treatment of dependent failures, can be subjected to sensitivity analysis, as needed. Further, use of exponential failure model (constant failure rate) in estimating event probabilities and various human reliability models are abstractions that introduce uncertainty. These methods are generally accepted and are not subjected to uncertainty analysts. Modeling uncertainties in consequences include the source term, damage mechanisms, release fractions, and leak-path mechanisms, as well as environmental transport. These factors can be evaluated by sensitivity analyses.

Completeness uncertainty is also treated philosophically. It is the residual, or unknown, that may remain after performing an exhaustive, structured PSA. After examining the preclosure operations using the risk triplet questions (see Section 4) by a cognizant team of safety analysts and designers, such uncertainty should be low. The LA will represent to a high level of



confidence that credible Category 1 and Category 2 event sequences have been identified and treated. Further, the LA can discuss event sequences identified as Beyond Category 2 with respect to modeling assumptions and parameters used. This will provide more transparency to reviewers (e.g., the NRC) who can perform their own sensitivity analysis, if desired, to assure themselves that the list of credible sequences is complete.

Finally, it is possible that differences between analysts introduce another source of uncertainty. The application of this Guide, however, and associated Yucca Mountain Project procedures should essentially eliminate this source of uncertainty from the PSA.

### **9.3.2.1 Uncertainties in Input Parameters**

Section 7.5 discusses three sources of uncertainty that are associated with the input information used to quantify event probabilities and frequencies: random variability, uncertainty associated with information source, and uncertainty associated with applicability to repository facilities.

Similar types of parameter uncertainty are associated with consequence analyses.

Other sources of uncertainty in input parameters include the waste stream year-to-year loadings.

Such sources of uncertainty are amenable to quantification and propagation through the event sequence frequency and consequence analyses.

### **9.3.2.2 Uncertainties in Model Inputs and Modeling**

Uncertainty in model inputs relates to the level of detail on design and operations that is available. For the LA for construction authorization, it is anticipated that the level of detail will be limited. Principal operations and associated equipment will be defined, along with the degree of redundancy, dependence on power supplies, spatial relationships, and anticipated human interactions. Therefore, the PSA will require some imagination on the part of analysts, with concurrence of design personnel, to define potential hazards and event sequences, and to synthesize system fault trees and event sequences. This lack of certainty in design and operations becomes a source of modeling uncertainty, described below. The PSA to support the LA to Receive and Process will not have this source of uncertainty.

Uncertainties in modeling stem from generic and specific causes. The generic uncertainties stem from use of standard event probability models, such as constant failure rate, repair models, common-cause failure models, and human reliability models. In general, such sources of uncertainty will not be addressed in the PSA unless such modeling effects appear to affect the PSA results. If deemed necessary, sensitivity analyses must be performed to examine the results with alternative models.

Repository-specific modeling uncertainties stem from the representation of reality in the event sequence and fault tree logic models. For example, event sequence (event tree) analysis (Sections 7.1 and 10.1) may include dependencies between events whose conditional probabilities are estimated. The presence and nature of the dependency may be uncertain, and the associated conditional probability may be an assumption. Similarly, a fault tree models of a control systems might include a redundant train and might include components like PLCs that

are expected to be used, but which have not been completely specified in design documents. Such models may include a common-cause failure model for redundant components. Such modeling is potentially a source of significant uncertainty and stems, in part, from the uncertainty in the model inputs and level of design detail. For these kinds of modeling uncertainties, sensitivity analysis should be performed to demonstrate how significantly they affect the PSA results.

All bases for modeling will be documented so that design-dependent issues and assumption-dependent uncertainties can be identified and appropriate sensitivity analyses performed as necessary.

### **9.3.3 Representing Uncertainties in Input and Output Variables**

For the PSA, particularly for the LA for construction authorization, it is recommended that all event probabilities and frequencies be treated as LN distributions, except in cases where a normal distribution is more appropriate. The LN is a good fit to the distribution that result when several distributions are multiplied, as in an event sequence quantification. As described and illustrated in Section 7.5, the LN in inputs can be converted back and forth to other distributions that are better to work with analytically in event probability estimation (e.g., in an empirical Bayesian analyses). Further, the parameters of the LN are readily associated with the properties and tabulations of the normal distribution. The principal properties of the LN and normal distributions used in the PSA are described in the following subsections.

#### **9.3.3.1 Uncertainty Interval and Bounds**

The uncertainty in a variable  $x$  is described by a PDF that gives the probability  $p(x)dx$  that the true value of the variable is within the  $dx$  about  $x$ . The cumulative probability function is given as  $P(x_p)$ , defined as the probability that the true value of the variable is less than or equal to  $x_p$ .  $P(x_p)$  is the integral of  $p(x)dx$  between the lower limit of the distribution and the value  $x_p$ . The cumulative probability function,  $P(x_p)$ , is used to define the confidence interval, or range, for the input variable or calculated value.

Unless otherwise specified or required, uncertainty on input variables and calculated outputs of event sequence frequencies and consequences will always imply a 90 percent confidence interval (range). This means that 10 percent of the values of inputs or of results can fall outside of the interval. Generally, the PSA will use confidence intervals that span the range from the 5<sup>th</sup> percentile to the 95<sup>th</sup> percentile.

The bounding, or limiting, values that define the 90 percent confidence interval of a variable  $x$  are the values of  $x_{0.05}$  and  $x_{0.95}$ , where  $P(x_p) = 0.05$  and  $0.95$ , respectively. The median value of distribution occurs at the value  $x_{0.5}$  where  $P(x_p) = 0.5$ . The EF is defined as the ratio  $x_{0.95} / x_{0.5}$ .

These definitions apply irrespective of the particular form of PDF that is used. The following subsections describe how these definitions are applied to normal and LN distributions.

### 9.3.3.2 Properties of the Normal and Lognormal Distributions

**Normal Distribution**—For a variable  $y$ , the properties of the normal distribution apply. The normal distribution is a symmetric ranging from  $-\infty$  to  $+\infty$ . The measures of central tendency are numerical equal: mean ( $y$ ) = median ( $y$ ) = mode ( $y$ ). Dispersion about the mean ( $\mu$ ) is described by the variance ( $\sigma^2$ ) or the standard deviation ( $\sigma$ ).

The normal distribution is often expressed in normalized form in terms of a variable:

$$z = [y - \mu]/\sigma$$

The PDF for  $z$  has a mean value of 0.0 and a standard deviation of 1.0. The cumulative probability of the normal distribution from  $-\infty$  to a value  $z = Z$ , sometimes termed the normal curve of error function for  $Z$ , is amenable to direct integration and has been extensively tabulated and built into spreadsheet programs like Microsoft Excel. In the normalized form, the percentiles of the cumulative distribution are given as:

$Z_{0.5}$	=	0	median (and mean)
$Z_{0.05}$	=	-1.64	5th percentile
$Z_{0.84}$	=	1.00	84th percentile
$Z_{0.95}$	=	1.64	95th percentile
$Z_{0.99}$	=	2.33	99th percentile.

Note: For brevity, the  $Z$  values are shown only to two decimal places. For hand calculations, at least three to four places should be used. When using Microsoft Excel or SAPHIRE, the functions are built in and will display as many places as selected.

From the definition of  $z$ , the corresponding values of the variable  $y$  are given as:

$y_{0.50}$	=	$\mu$	median (and mean)
$y_{0.05}$	=	$\mu - 1.64\sigma$	5th percentile
$y_{0.84}$	=	$\mu + 1.00\sigma$	84th percentile
$y_{0.95}$	=	$\mu + 1.64\sigma$	95th percentile
$y_{0.99}$	=	$\mu + 2.33\sigma$	99th percentile.

The parameters  $\mu$  and  $\sigma$  characterize the normal distribution.  $\mu$  may be considered a location parameter (defines the central value) and  $\sigma$  may be considered a shape parameter that describes the degree of spreading or peaking in the distribution of the variable  $x$ . The larger the value of  $\sigma$ , the wider the distribution and the greater the uncertainty.

For the normal distribution, the EF, as defined above becomes:

$$EF_{\text{normal}} = 1 + 1.64\sigma/\mu.$$

The EF is not often directly used with the normal distribution. The application of the EF comes when estimating uncertainty ranges and assuming distributions.

For example, if the analyst (supported by design or vendors) believes that 90 percent of a failure rate for some component lie between a lower bound (LB) of  $1 \times 10^{-3}$  and an upper bound (UB) of  $5 \times 10^{-3}$ , and are normally distributed. Using the relationships above, the statement of belief gives the following:

$$\begin{aligned} \text{LB} &= y_{0.05} = 1 \times 10^{-3} = \mu - 1.64\sigma \\ \text{UB} &= y_{0.95} = 5 \times 10^{-3} = \mu + 1.64\sigma \end{aligned}$$

which yield

$$\begin{aligned} \mu &= 3 \times 10^{-3} \\ \sigma &= 1.2 \times 10^{-3} \end{aligned}$$

The EF becomes  $1.67 (5 \times 10^{-3} / 3 \times 10^{-3})$ , but is a derived quantity in this example. The EF has a more fundamental role for the LN distribution, however.

The properties of the normal can be applied to a lognormally distributed variable as described below.

**Lognormal Distribution**—The properties of a LN distribution for a random variable  $x$  is developed from the properties of the normal distribution for the transformed variable  $y = \ln(x)$ , where  $\ln(\cdot)$  is the natural logarithm. The LN distribution on  $x$ , the non-transformed variable is not symmetric. It ranges from  $x = 0$  to  $x = +\infty$ . The mean value of  $x$  is not equal to the median or mode.

Mathematical expressions for the parameters of the LN are somewhat complex, but they are derived from the properties of the normal distribution for the transformed variable  $y$ .

To describe a LN distribution, the analyst needs only two values such as the median value of  $x$  and a value for EF, or the median and UB, or the UB and LB. The latter is the 90 percent confidence range on  $x$ . The following relationships apply:

$$\begin{aligned} M(x) &= x_{0.50} = \text{UB}/\text{EF} = \text{LB} \times \text{EF} = (\text{UB} \times \text{LB})^{1/2} \\ \text{EF} &= \text{UB}/M = M/\text{LB} = (\text{UB}/\text{LB})^{1/2} \end{aligned}$$

These expressions relate to parameters for the distribution of the non-transformed variable,  $x$ . When the variable is transformed to  $y = \ln(x)$ , the parameters for the normal distribution on  $y$ , are derived as follow:

$$\begin{aligned} \mu_{\text{LN}} &= \ln(M), \text{ the location parameter (mean of the } y \text{ distribution)} \\ \sigma_{\text{LN}}^2 &= \ln(\text{EF})/Z_{0.95} = \ln(\text{EF})/1.64 \end{aligned}$$

With these parameters so defined, the mean of the LN distribution of the non-transformed variable,  $x$ , becomes:

$$\text{mean}(x) = \underline{x} = \exp[\mu_{LN} + \frac{1}{2} \sigma_{LN}^2],$$

which can be expressed alternatively as:

$$\text{mean}(x) = \underline{x} = M \times [EF]^a, \text{ where } a = 1/(2 Z_{0.95}) = 1/(3.29).$$

For example, the analyst (supported by design or vendors) believes that 90 percent of a failure rate for some component lies between a LB of  $1 \times 10^{-4}$  and a UB of  $1 \times 10^{-2}$ , and it is lognormally distributed (this range covers two orders of magnitude). Using the relationships above for the LN, the statement of belief gives:

$$LB = x_{0.05} = 1 \times 10^{-4}$$

$$UB = x_{0.95} = 1 \times 10^{-2}$$

which yield

$$M(x) = x_{0.50} = (UB \times LB)^{1/2} = (1 \times 10^{-2})(1 \times 10^{-4})^{1/2} = 1 \times 10^{-3}$$

$$EF = UB/M = 1 \times 10^{-2}/1 \times 10^{-3} = 10$$

$$\mu_{LN} = \ln(M) = \ln(1 \times 10^{-3}) = -6.907$$

$$\sigma_{LN}^2 = \ln(EF)/1.64 = \ln(10)/1.64 = 1.40$$

$$\underline{x} = \exp[\mu_{LN} + \frac{1}{2} \sigma_{LN}^2] = \exp[-6.907 + (1.40)^{1/2}] = 3.3 \times 10^{-3}.$$

In this case, with an EF of 10, the mean  $\underline{x}$  is a factor of 3.3 greater than the median. The UB is a factor of 3 above the mean.

The LN distribution is asymmetrical and, because the variable may range over several decades, the distribution presents interesting properties with respect to the relationship of mean to median, and mean to the upper 95 percent confidence limit. Table 9-1 illustrates how the parameter  $\sigma_{LN}$  varies with the EF (note that an EF of 30 indicates a factor of 900 for the ratio of UB/LB). The table also shows how the ratio of mean/median, mean/UB, and UB/mean vary with EF. It is noted that the ratio of mean/median ranges from about 1.1 to 8.5 for the range of EF shown. This indicates that the mean is within a factor of three or less of the median for EF less than 10. The ratio of UB/mean indicates that the mean is within a factor of about 2 to 4 of the upper bound over the range of EF shown.

The last column in Table 9-1, CDF(Mean), is the value of the CDF of the LN evaluated at the mean. This column indicates, somewhat paradoxically, that as the uncertainty increases, characterized by the EF ranging from 2 to 30, the probability that the true value exceeds the mean actually decreases. For example, for an EF of 2, the CDF is 0.54, meaning that there is 0.54 probability that the true value is less than or equal to the mean, and a probability of 0.45 that

the true value exceeds the mean. By contrast, for an EF of 10, the CDF is 0.84 and the complement probability is 0.16.

Because of the characteristics of the LN, the mean value is a suitable measure for binning event sequence frequencies or for evaluating consequences against regulatory limits.

### 9.3.3.3 Comparison of Output Values with Limits

The PSA will calculate two kinds of output quantities that must be used in the risk-informed performance-based compliance with 10 CFR Part 63. These variables represent frequency and consequence, respectively. Limits on frequency relate to the boundaries between Category 1 and Category 2, and between Category 2 and Beyond Category 2. Limits on consequences relate to the respective dose limits for the public and workers defined in 10 CFR 63.111.

The PSA will use mean values of frequency and doses. Thus, if the mean value of a dose is 4 rem total effective dose equivalent and therefore less than the limit of 5 rem, the result is compliant with the regulations.

Similarly, if the mean value of the frequency of an event sequence is less than  $1 \times 10^{-2}$  per yr, the sequence is considered to be Category 2. If the mean value of the frequency of an event sequence is less than  $1 \times 10^{-6}$  per yr, the sequence is considered beyond Category 2.

The uncertainty factors associated with frequency and consequence analyses should be quantified, however. If the PDF for the frequency or consequences of a given event sequence is shown to be lognormally distributed. Table 9-1 illustrates that the mean value is within a factor of 2 to 4 of the 95th percentile upper confidence bound. Therefore, there is confidence that event sequences will be appropriately categorized with respect to frequency.

As noted in Section 3, the LA for CA will use one-half the regulatory limit as guidance for estimating dose consequences. Therefore, if the mean value of an estimated dose is less than or equal to one-half of the regulatory limit, and the uncertainty in dose is shown to be lognormally distributed, there will be low probability of exceeding the regulatory limits.

Table 9-1 Properties of the Mean of a Lognormal Distribution

EF	Mean/Median	Mean/UB	UB/Mean	CDF(Mean)
2	1.09	0.55	1.83	0.54
3	1.25	0.42	2.40	0.59
5	1.61	0.32	3.10	0.68
10	2.66	0.27	3.75	0.84
30	8.48	0.28	3.54	0.98

### 9.3.4 Propagating Uncertainties in Frequency and Consequence Analyses

This subsection describes the principal means for propagating uncertainties. The discussion is based on frequency analyses. Consequence analyses are treated similarly.

The *PRA Procedures Guide* (NRC 1983) and CPQRA Guidelines (AIChE 2000) describe several methods that could be used for propagating uncertainties. It is noted that direct (analytical) integration of the multivariable probability distribution is generally not possible. Therefore, the favored techniques are numerical integration, which includes the discrete probability method and Monte Carlo simulation, and moments methods. For complex analyses, the moments methods also appear to be untractable unless the output moments are approximated by using a Taylor series expansion where only second-order terms are retained.

Since current desktop software like SAPHIRE, GENII-S, and Microsoft Excel with an @RISK add-in can perform Monte Carlo simulation, or the similar Latin Hypercube Analysis. Numerical analysis will be the primary technique to be used in frequency and consequence analysis for the PSA. The SAPHIRE workstation permits eleven forms of distribution functions for uncertainty.

The analyst must specify the mean and one other parameter, depending on the distribution selected. However, for some purposes, including checking of results, a Taylor-series approximation could be used. The following describes the basic steps for sequence uncertainty analysis. The users' manual for the particular program (e.g., SAPHIRE) should be consulted.

#### **9.3.4.1 Sequence Uncertainty Analysis**

The following are the principal steps for developing an event sequence analysis that includes propagation of uncertainty:

1. Construct the logic models (event tree, fault tree, and human reliability) for the initiating events and systems (see Sections 7.1, 7.2, and 7.3)
2. Obtain and quantify input information for each basic event and initiating event used in the logic models, including quantification of uncertainty distribution (see Section 7.5)
3. Perform sequence quantification analysis to generate minimal cutsets and point estimates (see Section 7.1)
4. Select method for uncertainty analysis. For simple sequences, it may be appropriate to use Microsoft Excel with the @RISK add-in.

If the event sequence quantification is performed in a program like SAPHIRE (Smith et al. 2000), however, it will be more efficient to use the uncertainty propagation that is built in. The analysis may be performed on individual sequences, group, or family. For example, it may be necessary to add the frequencies of two or more Category 1 sequences for the same or different initiating events for a given repository operations area. For either @RISK or SAPHIRE, the analyst must select either Monte Carlo or Latin hypercube, number of trials, and random number seed.

1. Obtain tabular and graphical outputs of uncertainty analysis. The output of sequence uncertainty analysis typically includes: mean, median, standard deviation, 5th and 95th percentiles, maximum and minimum (for the run of  $N$  samples), seed number, and sample size.

2. Interpret results. Examine acceptability of results and identify dominant contributors to sequence frequency and uncertainty in results

The sequence uncertainty analysis will be used primarily to generate the mean and EFs to evaluate sequence categorization and to demonstrate compliance with 10 CFR Part 63. Where appropriate, sequence sensitivity analysis will be performed.

#### **9.3.4.2 System (Fault Tree Top Event) Uncertainty Analysis**

The steps for evaluating system uncertainty are essentially the same as for the event sequence uncertainty analysis, and are not repeated here.

The system uncertainty analysis will be used primarily to generate insights into the dominant contributors to system reliability and safety performance. Where appropriate, system sensitivity analysis or IM will be obtained.

#### **9.3.4.3 Uncertainties in Consequences**

Section 8 describes the approach to consequence analyses for the PSA. Analyses for Category 1, Category 2, and Beyond Category 2 event sequences are described. The bases for identifying and treating uncertainties are discussed, including the use of conservative or bounding values, where appropriate.

### **9.3.5 Uncertainty Analysis versus Sensitivity Analysis**

Some sources of uncertainty cannot be analyzed by assigning a PDF, but insights on their significance may be investigated quantitatively through sensitivity analysis.

A sensitivity analysis for event sequence frequency analysis is performed by changing features of logic models, human reliability models, input parameter values, or the features of the physical facility or operations. In general, a sensitivity analysis examines rather large-scale changes such as:

- Changing the redundancy of a system (i.e., adding a train or deleting a train)
- Changing the probability of a basic event (hardware, software, or human failure) from a best-estimate probability to a bounding value (i.e., to 1.0 or to 0.0)
- Changing the failure logic by adding or deleting elements, such as adding or deleting an alarm that alert the human operator to take action (e.g., using AND logic).

Such changes are made one at a time in fault tree and event sequence quantification so the output can be compared to the baseline result.

In addition, a sensitivity analysis can be used the effect of changing the assumed probability distribution for an input parameter. If it is uncertain, for example, whether to use a LN or a normal distribution for a particular input, the alternative distributions are used and results compared.



A sensitivity analysis of consequences is performed by changing source term parameters (i.e., age and burnup, and fraction of inventory that is released) and the presence of mitigating features (i.e., high-efficiency particulate air filters and deposition). The process is essentially the same as for frequency analysis.

### 9.3.6 Importance Measures Analysis

Importance analysis may be regarded as a special form of sensitivity analysis. There are several standard definitions of IMs that have been applied in PRAs and regulatory evaluations. An IM analysis can be performed on fault trees or event sequence frequency quantification. Analysis of the standard IMs are performed automatically by programs such as SAPHIRE. Therefore, the discussion here is brief.

In fault tree analysis for a system, the top event represents the probability that the defined event will occur (e.g., HVAC fails to run and filter particulates for at least 24 hours). Suppose the fault tree analysis shows the mean probability is  $1 \times 10^{-4}$ , and lists all of the cutsets (products of basic event probabilities) above a cutoff probability of  $1 \times 10^{-7}$ . The value for all of the cutset and top event probabilities are subject to the values input for basic event probabilities.

The IM can be used to analyze the sensitivity of the result to the inputs in the following way:

1. The risk-achievement worth (RAW) of every basic event, with respect to output, is calculated by setting the probability of each basic event to 1.0 one-at-a-time while holding the baseline values of probabilities of all of the other basic events. The analysis program then produces a table, which ranks every basic event according to its RAW value. The RAW for a given structure, system, or component (SSC) represents the increase in system failure probability if the SSC is removed from the system model. The RAW, like the Birnbaum Importance (BI), may be interpreted as a measure of the margin of safety contributed by proper operation of the model element (i.e., the SSCs). The RAW process is similar to the take-away process described in Section 12.
2. The risk-reduction worth (RRW) of every basic event, with respect to output, is calculated by setting the probability of each basic event to 0.0 one-at-a-time while holding the baseline values of probabilities of all of the other basic events. The analysis program then produces a table, which ranks every basic event according to its RRW value. The RRW for a given SSC represents the decrease in system failure probability if the SSC is perfectly reliable.
3. The BI is calculated, in essence, by taking the difference between the RAW and RRW for each basic event. The analysis program produces a table, which ranks every basic event according to its BI value. The BI is interpreted as a measure of the margin of safety contributed by proper operation of the model elements (i.e., the SSCs). The BI is sometimes interpreted as the maintenance importance for a given SSC (i.e., the importance of keeping it operational). Because the RRW is usually small compared to the RAW, the BI is usually quite close to the RAW numerically.

4. Fussell-Vesely Importance (FV) is calculated, in essence, by taking the product of each basic event probability times its BI (there are other, more fundamental definitions of FV). The analysis program produces a table, which ranks every basic event according to its FV value. The FV illustrates the fraction of current risk (or top event probability) involving the failure of the model element (i.e., a particular SSC). The BI portion of the product gives the relative magnitude of the risk achievement by the presence of the model element and the event probability in the product gives a weight of the relative likelihood of having the SSC fail to achieve the risk reduction.

Such IMs provide insight to the dominant contributors to system failure and can be used to develop risk-informed maintenance, Quality Assurance, and training programs. Further, IM can be used to scope an uncertainty analysis where more attention is given to identifying and quantifying uncertainties in the basic events that have the dominant IMs.

When sequence quantification is performed by the fault-tree linking method (see Sections 7.1 and 7.2), the top event becomes the frequency that the sequence occurs. All of the sequence cutsets include the initiating event frequency times one or more basic event probabilities. Since the initiating event frequency is common to all, it can be set equal to 1.0 and the IMs of the remaining cutsets are evaluated as described above.

The application of IMs has been described in Regulatory Guide 1.174. In those applications, the baseline risk is a measure of integral risk like core-damage frequency of a reactor core damage that stems from multiple event sequences. The change in core-damage frequency is calculated using the RAW, RRW, BI, and FV. At the present, the PSA for the repository will not employ an integral risk measure. Therefore, the application of IMs will be limited to developing insights on SSC risk significance from fault-tree analysis and event sequence analysis.

### 9.3.7 Examples of Uncertainty Analyses

Information for this section is under development and will be provided later.

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## ACRONYMS

AFB	Air Force Base
ALARA	as low as is reasonably achievable
ATS	assembly transfer system
BLM	U.S. Bureau of Land Management
DOE	U.S. Department of Energy
ESD	event sequence diagram
FC	frequency category
GF	guaranteed failure
HCLPF	high confidence of low probability of failure
HEPA	high-efficiency particulate air
HVAC	heating, ventilation, and air-conditioning
LOSP	loss of offsite power
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NTS	Nevada Test Site
PDF	probability density functions
PGA	peak ground acceleration
PMF	probable maximum flood
PRA	probabilistic risk assessment
PSA	preclosure safety analysis
PSHA	probabilistic seismic hazards analysis
QA	quality assurance
QL	quality level
RF	random failure
SET	seismic event tree
SMA	seismic margins analysis
SNF	spent nuclear fuel
SNFA	spent nuclear fuel assembly
SSCs	structures, systems, and components
SSE	safe shutdown earthquake
TNT	2,4,6-Trinitrotoluene (explosive)
UBC	Uniform Building Code

**ACRONYMS (Continued)**

VGM	Vibratory ground motion
WHB	Waste Handling Building



## 10. EXTERNAL EVENTS

This section provides a bridge between the external events hazards analysis and the design basis such that no credible event sequences which do not meet 10 CFR Part 63 performance objectives can occur. Each section may have a different scope, approach, and length depending on the topic.

The sections will provide means to (a) identify those structures, systems, or components (SSCs) that need to withstand credible external events and thereby prevent a radiological release; and (b) describe methods to develop controls that prevent a credible release scenarios given the occurrence of the initiating event.

### 10.1 SEISMIC ANALYSIS

#### 10.1.1 Introduction

This section describes the bases and methods for analyzing the design of surface and subsurface repository facilities and waste package design for potential vulnerability to seismically-induced event sequences that could potentially lead to a criticality or radiological release. The safety strategy includes the prevention of any credible scenarios that could potentially lead to consequences that exceed the performance requirements of 10 CFR Part 63. This guide defines the steps in the analyses that, in many cases, link to other portions of the Preclosure Safety Analysis (PSA) and to the seismic design strategy that has been presented to the U.S. Nuclear Regulatory Commission (NRC) in a series of seismic topical reports.

Examples presented are based on hypothetical situations. None of the values of event sequence frequencies or doses should be taken as results applicable to a repository.

#### 10.1.2 Overview of Approach

##### 10.1.2.1 Background

**Precedents from Nuclear Power Plants**—The licensing basis for the repository, with respect to seismic design, adapts the principal tenets of regulatory precedence that the NRC has applied to nuclear power plants (NPPs) (which are regulated per 10 CFR Part 50). The fundamental licensing concept for NPPs includes the definition of a design basis earthquake (termed a safe shutdown earthquake [SSE]) and requirements stipulating that important to safety SSCs must be designed to withstand the vibratory motions associated with that earthquake. These important to safety SSCs are classified as a single category termed Seismic Category 1 SSCs.

The SSE is specified by the peak ground acceleration (PGA) and other characteristics of the vibratory ground motion, such as spectral acceleration and time-history, that become input parameters to the design of SSCs important to safety. Approved regulatory guides and industry codes and standards are applied in the design. It is deterministically argued that, for the NPPs, as long as an earthquake of intensity greater than the SSE does not occur at the site, there will be no seismically-induced accident sequences that cannot be prevented or mitigated such that the plant cannot be brought to a safe condition. Design principles and approaches for seismic hardening are provided in sections of the NPP standard review plan (NRC 1987).

The PGA and other parameters have previously been defined based on the largest historical earthquake for the site. Although the return period of the SSE for power plants can be defined, it was not used in the initial licensing basis for most NPPs. In more recent regulatory practices for NPPs, however, probabilistic seismic hazard analysis have been used to define the return period for the design basis earthquake and its vibratory motion characteristics. Seismic probabilistic risk assessments (PRAs), for example, use the return period or the frequency of exceedence of a spectrum of earthquakes as input to the risk analysis. The characteristics of the SSE are included in the spectrum of earthquakes. Instead, the risk analysis determines the annual probability (or frequency) of a seismically initiated sequence of events that cannot be brought to a safe condition (i.e., for NPPs, the core damage frequency that is attributed to seismic events is calculated).

**Monitored Geologic Repository Preclosure Seismic Strategy**—The precedent seismic design principles for NPPs have been adapted for the repository by requiring that all SSCs important to safety must withstand a design basis earthquake. The adaptation of this deterministic principle to the repository licensing basis is more complicated because it is applied in a two-tiered, risk-informed, performance criteria.

Seismic Topical Report No. 2 (YMP 1997), submitted by the DOE and tentatively approved by the NRC pending their receipt and approval of Topical Report No. 3, describes the rationale for this approach as well as design principles to be applied to surface facility SSCs, waste package SSCs, subsurface openings and ground support SSCs, and other SSCs in the subsurface facility. With certain exceptions as noted in Topical Report No. 2, the design of surface and subsurface SSCs will apply the principles of the *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987).

The two-tiered design basis earthquake approach, however, leads to several departures from a strict carryover of analyses from the power plant precedent and becomes somewhat more complex when applied to items important to safety that have been classified into three categories of quality level QL-1, QL-2, and QL-3 (see Section 12). Because of the two-tiered, risk-informed, approach, two design basis earthquakes are defined. For the repository, the PGA and ground motion characteristics are to be defined (via the forthcoming Topical Report No. 3) based on the probabilistic seismic hazards analysis (PSHA) for site-specific earthquakes having return periods that correspond to Frequency Category (FC)-1 and Frequency Category 2 (FC)-2 as defined in Topical Report No. 2 (YMP 1997). The design basis earthquake to be applied to the design of a given SSC depends on the magnitude of potential radiological exposures to the public or workers with respect to compliance with 10 CFR 63.111.

The safety case for the repository will be based on deterministic design basis applied to the repository SSCs. In this licensing basis, a specific SSC designed to withstand one of the respective design basis earthquakes is assumed able to perform its safety function during earthquakes of magnitudes up to, and including, that magnitude specified as the design basis earthquake for that SSC. However, the SSC is assumed to fail with a probability of 1.0 in the event of an earthquake that exceeds the magnitude of the design basis earthquake for that SSC. This approach may be characterized as assigning a step-function fragility factor to each SSC (i.e., the conditional probability of the SSC failing, given the occurrence of an earthquake, is 0.0 up to, and including, the vibratory ground motion (VBM) associated with the design basis

earthquake; however, the probability becomes 1.0 for higher VBM's). As described in Topical Report No. 2 (YMP 1997), however, it is implicit that there will be margins in the design of the SSC such that the actual probability of failure, given a design basis earthquake, will be a factor of 0.1 or less. This margin provides assurance, for example, that in the event of a FC-2 design basis earthquake at  $1 \times 10^{-4}$  per year, the annual probability of an offsite dose that exceeds the limits of 10 CFR 63.111(b) will be less than  $1 \times 10^{-5}$  per year. Although this probability is not demonstrated to be below the  $1 \times 10^{-6}$  per year probability threshold for screening out events, it is in accordance with regulatory precedence.

It is expected that additional analyses will be required, such as SSC fragility analyses or SMAs, as well as associated uncertainty analyses to demonstrate that high-consequence seismically-induced sequences are beyond design basis. It is further expected that this demonstration can be provided for a limited number of sequences and a limited number of SSCs; therefore, it will not be necessary to perform a comprehensive seismic PRA.

#### **10.1.2.2 Summary of Approach**

The PSA will address potential seismically-induced radiological release events through a comprehensive hazards analysis and limited event sequence analysis as a means to identify the SSCs important to safety that are required to withstand the vibratory ground motions associated with the respective FC-1 or FC-2 design basis earthquake for those SSCs.

The following analysis steps will be applied:

- Review available design descriptions and drawings.
- Review available hazards, event sequences, criticality scenarios, and consequence analyses for events initiated by internal events, internal fires, and internal floods.
- For each of the repository operational areas, define the scenarios by which radionuclides could be released, or a critical condition could result, due to events initiated by an earthquake as a result of the direct or indirect effects of seismically-induced failure of the SSCs.
- Use mean values for source terms to calculate the offsite and worker dose that could result from each hypothetical seismic scenario and the postulated failure of a given SSC. Calculate the dose with and without mitigation features, if mitigation is currently used in the design, or could be applied.
- Determine the design basis earthquake frequency category to apply to each SSC based on the approach described in Topical Report No. 2 (YMP 1997). Event sequence evaluations should include the considerations of potential criticality events, the importance of ensuring waste isolation, the importance of ensuring the ability to retrieve waste packages, and ALARA (i.e., as low as is reasonably achievable) principles and practices.

Several issues regarding this approach have not been resolved at this time. The detailed discussion of the approach identifies some of these issues.

### **10.1.3 Details of Approach**

#### **10.1.3.1 Design Requirement per Seismic Topical Report 2**

The DOE and NRC have established the principles for the seismic design of items important to safety in the repository in a topical report, *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997) (otherwise known as Topical Report No. 2). The report provides design principles for the following categories of SSCs: surface facilities, underground openings, other underground SSCs, and waste package. Seismic design principles are provided for both vibratory ground motion and ground fault displacement.

A project position paper, *Monitored Geologic Repository Seismic Design Requirements Strategy* (CRWMS M&O 1998), provided a method for applying the principles of Topical Report No. 2 to the design of the repository for SSCs that were previously classified as important to radiological safety. Although Topical Report No. 2 was written by the DOE and reviewed by the NRC according to the regulations of 10 CFR Part 60, the fundamental approach can be adapted to address 10 CFR Part 63, since the two-tiered Category 1 and Category 2 event sequence structure has been retained in 10 CFR Part 63.

The principle difference between 10 CFR Part 60 and 10 CFR Part 63 is the change from prescriptive requirements to performance-based requirements. Section 10.1.4.2 summarizes the impact of moving from 10 CFR Part 60 to 10 CFR Part 63. It has been concluded that the change has little substantive effect on the requirements for seismic design for preclosure safety.

Previously SSCs classified as important to safety (i.e., preclosure radiological safety) were designated by a single quality assurance classification level (QA-1). The application of Topical Report No. 2 required that each of the QA-1 SSCs be able to withstand a FC-1 or FC-2 design basis earthquake, depending on the potential magnitude of the resultant radiological doses. The process for the quality assurance classification of items important to safety has been revised to include three levels of QL-1, QL-2, and QL-3 (as described in Section 12). The application of Topical Report No. 2 becomes somewhat more complex because each SSC will carry a QL designation as well as a seismic classification (i.e., FC-1 or FC-2).

The seismic design requirements are expressed in terms of two design basis earthquakes that are characterized by their mean frequency (or return period) labeled as FC-1 and FC-2, respectively. For the vibratory ground motion design basis earthquakes, FC-1 is defined as having a mean annual probability of  $1 \times 10^{-3}$  per year, and FC-2 is defined as having a mean annual probability of  $1 \times 10^{-4}$  per year. The determination of the parameters associated with the intensity of the respective design basis earthquakes will be the subject of the forthcoming Topical Report No. 3. Topical Report No. 3 will provide the information associated with the intensity and characteristics such as peak acceleration and the resulting ground motion response spectra. This information is required to ensure that the structural design of SSCs can withstand the dynamic loads resulting from an earthquake.

In addition, for fault displacement design basis earthquakes, FC-1 is defined as having a mean annual probability of  $1 \times 10^{-4}$  per year, and FC-2 is defined as  $1 \times 10^{-5}$  per year. The principal design defense against fault displacement is fault avoidance as described in Topical Report No. 2 (YMP 1997).

Guidance from prior regulatory positions, such as the *Standard Review Plan for Spent Fuel Dry Storage Facility* (NRC 2000a), requires that subcriticality be maintained for all normal event, abnormal events and postulated accidents. These events include the effects of natural phenomena. Therefore, SSCs that might lead to criticality events were assigned the FC-2 design basis earthquake. Furthermore, the FC-2 design basis earthquake was assigned to SSCs that could significantly impair the ability of the repository to retrieve waste packages or degrade waste if the SSCs were damaged.

The principles of ALARA and defense-in-depth are addressed in the consideration of seismic classification. The final step in the seismic classification process provides a catch all where prudence or engineering judgement may dictate a more stringent seismic design, commensurate with cost-benefit considerations or throughput/availability considerations that may be more limiting than the radiological safety considerations.

#### **10.1.3.2 Impact of 10 CFR Part 63**

The principal difference between 10 CFR Part 60 and 10 CFR Part 63 is the change from prescriptive requirements to risk-informed, performance-based requirements. However, 10 CFR Part 63 retains the requirement stipulating that credible natural phenomena must be considered in the design of the repository. After comparing the respective requirements of the two regulations, it is concluded that there is no substantive difference in regulations that affect the requirements for seismic design for preclosure safety. Topical Report No. 2 remains applicable but regulatory references in Topical Report No. 2 will have to be updated to refer to the appropriate sections of 10 CFR Part 63 instead of 10 CFR Part 60. For example, the radiological performance requirements are now provided in 10 CFR 63.111 and references to 10 CFR Part 60 definitions for important to safety and the design basis event frequency categories (with their corresponding dose limits) must be replaced by counterpart references to 10 CFR Part 63.

A change in these regulations that may affect the application of Topical Report No. 2 involves the requirement that states that doses from Category 1 event sequences must be less than, or equal to, the final Environmental Protection Agency criteria for Yucca Mountain (15 mrem total effective dose equivalent (TEDE) per year). It was previously assumed in the preclosure Seismic Strategy that the 10 CFR Part 20 dose limit of 100 mrem would be applied to determine which SSCs are required to withstand a FC-1 design basis earthquake.

#### **10.1.3.3 Process for Assigning Design Basis Earthquake**

The following steps are applied to assign the appropriate design basis earthquake to SSCs:

1. Define the scenarios by which radionuclides could potentially be released by event sequences initiated by an earthquake. The postulated scenarios include the failures of SSCs directly associated with the handling, storing, or containment of radioactivity of high-level radioactive waste forms, SSCs that could interact with those SSCs

associated with the handling or storage of waste forms; the failure of fire protection systems, and radiation waste treatment systems.

2. The analysis may build on prior hazards analyses or event sequence analyses that have been developed for internal events, internal fires, internal floods, and criticality scenarios. As appropriate, to aid in identifying potential seismic scenarios, seismic event trees (SETs) may be constructed as described in Section 10.1.6.
3. Calculate the offsite dose that could result from each postulated failure of a given SSC and the resulting radiological release. Calculate doses with, and without, mitigation features, if mitigation is currently used in the design or could be applied.
4. Subject each SSC to the following dose comparisons:
  - a. If the individual offsite dose is greater than or equal to the 10 CFR 63.111(b)(2), then the SSC must be designed to withstand the vibratory ground motion of a FC-2 design basis earthquake. One guideline from 10 CFR 63.111(b)(2) is a dose less than 5 rem TEDE for the public.
  - b. If, however, the offsite dose is less than 10 CFR 63.111(b)(2) limits but greater than or equal to 10 CFR 63.111(b)(1) [which in turn references 10 CFR 63.111(a)] then the SSC must be designed to withstand the vibratory ground motion of a FC-1 design basis earthquake. The guidelines of 10 CFR 63.111(a) for Category 1 event sequences include paragraph (1), which requires meeting 10 CFR Part 20 limits. These limits include worker dose limits (per 10 CFR 20.1201) and annual doses of less than 100 mrem to the public, and paragraph (2) which presents a limit of an annual TEDE of less than 15 mrem to the public.
  - c. If both the offsite doses and worker doses are less than the 10 CFR 63.111(a) requirements for both workers and the public, then the SSC may be designated as non-seismic and designed accordingly (e.g., to the Universal Building Code).
5. For SSCs designated as non-seismic or as FC-1, examine the SSCs and determine if there are any radiological exposure, waste retrieval, or waste isolation issues that suggest designing to a more stringent category earthquake. If, after a cost analysis, it is shown that with only a small increase in costs that it is reasonable to design an SSC to a more stringent category earthquake and the redesign results in a reduction in dose, then this approach should be taken to promote ALARA.

The QA requirements associated with the seismic design will be considered at a later date as the graded QA program is developed.

#### **10.1.4 Examples of Application of Seismic Analyses**

The seismic methodology is demonstrated by application to a conceptual repository surface facility design. The first example is a scoping calculation applied to the structure of the Waste

Handling Building (WHB). The other examples represent conceptual handling and operations in the WHB.

The examples presented here (Section 10.1.5) are based on a evaluation of an earlier conceptual design. The examples are intended to illustrate how seismic classifications are assigned to SSCs. To support the LA submittal, however, the evaluations must be performed in a structured and thorough manner to ensure that potentially significant seismic vulnerabilities are identified. The SET described is one approach that will aid in the structured analysis. As noted, the seismic classification analysis should draw from prior analyses, including existing event trees non-seismic initiating events that can be modified to include seismic effects.

#### **10.1.4.1 Consequences of Hypothetical Waste Handling Building Collapse**

The structures of a waste handling building (WHB), including the outer shell and the structures of individual operations areas, have to be examined for two vulnerabilities to seismic radiological releases. First, the massive structural elements that could fall onto waste forms have to be examined as potential initiators of release scenarios. Second, these same structures may be required to provide confinement of releases initiated within the WHB by an earthquake. This section addresses the first issue by determining the appropriate design basis earthquake to assign to various structural elements such as the roof, walls, and foundation.

For this analysis, the WHB operational and staging areas are assumed to be full to maximum capacity in order to present the potentially largest radiological source term in the event of an earthquake. Further, the spent nuclear fuel assembly (SNFA) operations and staging areas are assumed to be full of pressurized water reactor or boiling water reactor spent-fuel assemblies, whichever is shown to produce the maximum source term. Where possible, falling structural elements having mass too small to breach a given waste form or to damage staging racks are eliminated from consideration. Otherwise, it is postulated that fragments of the roof or wall fall onto and damage the struck waste form or storage rack. The source term for the maximum inventory of the area is used, along with the release fractions and atmospheric transport parameters, to calculate the offsite and worker doses.

Table 10-1 presents several examples of offsite doses using conservative release fractions and atmospheric transport factors to a site boundary (assumed conservatively to be a distance of 5 km). All of the doses (which are unmitigated) exceed 5 rem. Therefore, it would be concluded from this hypothetical exercise that the roofs, walls, and foundation must be designed to withstand an FC-2 design basis earthquake. The application of this methodology, with current release fractions and a larger site boundary, is expected to change the evaluation.

#### **10.1.4.2 Seismically Induced Releases in Operations Area**

Within the WHB, a hypothetical Assembly Transfer System (ATS) receives and unloads SNFAs from transport casks. Several potential release scenarios have been identified for the ATS (as well as other operations) from hazards analysis and internal event sequence analysis. Event trees may be available for some of the initiating events. Each of the release scenarios is examined for potential initiation directly, or indirectly, in an earthquake. In addition, each operation (e.g., each lift, movement, or staging) is examined independently for potential direct or indirect

vulnerability to faults induced by the occurrence of an earthquake. The ATS may have several parallel operations that could each contain a maximum inventory when an earthquake occurs, so it can be assumed that parallel operations will fail concurrently during the earthquake. The source term is the sum from parallel operations that could have concurrent seismically induced releases.

The hypothetical offsite doses are shown in Table 10-2 for the seismic-induced failures of SSCs in several operations areas of the ATS. The doses are calculated using the same assumptions of conservative release fractions and atmospheric transport factors to the site boundary described in Section 10.1.5.1. For the example, that the hypothetical unmitigated doses are made to exceed 5 rem; therefore, in this example some of the SSCs will have to be classified for seismic design.

Table 10-3 presents the seismic classification for the alternative design strategies. In the confinement-mitigation strategy, it is assumed that the WHB (ATS) structure and the heating, ventilation, and air-conditioning (HVAC) and high-efficiency particulate air (HEPA) filter system(s) will be designed to withstand the FC-2 design basis earthquake. In this case, the seismic classification of waste-handling SSCs, such as transfer carts or cranes, for example, can be designated as FC-1 or Uniform Building Code. In the prevention strategy, it is assumed that the handling and staging equipment within the ATS will be designed to withstand the FC-2 design basis earthquake. In this case, no credit is taken for the mitigation effects of the HVAC/HEPA filter system and it is classified as Uniform Building Code. The structure of the WHB (ATS) would remain FC-2 because of the results of the analysis described in Section 10.1.5.1.

The examples presented here are based on a somewhat superficial evaluation of a previous conceptual design and are intended to illustrate how seismic classifications are ultimately assigned to each SSC. To support the LA submittal, however, the evaluations will be performed in a structured and thorough manner to ensure that potentially significant seismic vulnerabilities are identified. The seismic event tree described in the following section is one approach that will aid in the structured analysis. As noted previously, the seismic classification analysis should draw from prior analyses, including existing event tree analyses that can be modified to include seismic effects.

### **10.1.5 Development and Application of Seismic Event Trees**

Section 7.1 describes the techniques used in event tree construction and analyses for any event sequence initiated by any kind of internal or external event. The discussion in Section 7.1 includes the treatment of dependent failures between event tree headings and initiating events. The event tree modeling of such dependent failures represents failures that are induced, or made more probable, by the occurrence of a precursor event. Earthquakes, fires, floods, winds and tornadoes, and loss of offsite power are potentially significant because they can act as common-cause initiators that not only initiate an event sequence but can concurrently induce failure of mitigative SSCs that are in the design. The event tree format helps to define the potential event sequences and potential common-cause vulnerabilities. Such vulnerabilities, if resulting in an unacceptable dose, are identified and the associated SSCs are required to be hardened to withstand the common-cause initiator. In the case of earthquakes, each SSC important to safety is identified as having to withstand one of the two design basis earthquakes.



Table 10-1. Examples of Seismic Consequences of Roof Collapse onto Waste Forms (Hypothetical System)

Building Cell	Worst Case Waste Form	Max. Inventory (PWR/BWR)	VA Roof Material	Roof Height (m)	Roof Area, A (m^2)	Note 1	Note 2	Note 3			Note 4		
						Min. Roof Mass, M (MT)	Max. Roof Thickness [t=M/(d*A)] (m)	Conservative Offsite Dose - No Mitigation (rem)			Conservative Offsite Dose - with Mitigation (rem)		
								Canister	PWR	BWR	Canister	PWR	BWR
Waste Treatment			Steel	9.14	4,831								
Carrier Bay	Cask	52/122	Steel	3.35	1,691	33	3.20		>>5	>>5		1.8E-02	2.5E-02
ATS Cask Prep/Air Lock	Cask	78/183	Steel	21.64	260	5	3.22		>>5	>>5		6.4E-02	8.9E-02
Pool Area (Note 5)	SFA	792/1584	Steel	21.64	334	0.04	0.02		>>5	>>5		7.3E-01	8.3E-01
DC Hot Cell (dryer)	SFA	126/264	Concrete	15.24	130	0.06	0.24		>>5	>>5		8.6E-02	1.0E-01
DC Load/Decon (ATS)	DC	63/132	Concrete	8.23	146	20	74.51		25	31		7.5E-03	9.3E-03
CTS Corridor	Canister	10	Steel	18.29	752	1	0.18	>5			7.50E-03		
CTS Cask Prep/Decon	Canister	10	Steel	12.19	260	1	0.79	>5			5.10E-03		
CTS Lag Storage	Canister	40	Concrete	19.51	37	1	11.48	>>5			3.20E-02		
DC Load Area (CTS)	Canister	10	Concrete	19.51	272	1	1.57	>5			8.10E-03		
DC Handling Cell	DC	105/220	Concrete	18.29	1,208	9	4.06		>>5	>>5		8.3E-02	1.0E-01
DC Staging Area	DC	420/880	Concrete	18.29	637	9	7.70		>>5	>>5		2.9E-01	3.6E-01
DC Transfer/Load	DC	21/44	Concrete	9.14	242	18	40.64		>5	>5		7.5E-03	9.3E-03
WP Remediation	DC	21/44	Concrete	4.88	195	34	94.25		>5	>5		7.5E-03	9.3E-03

## Notes:

- 1 Minimum roof mass equivalent to a waste form drop from above the design height
- 2 Assumes roof is homogeneous with a density equal to steel or concrete
- 3 Deterministic TEDE dose with PWR/BWR/DHLW DBF source term, and no HEPA filtration
- 4 Deterministic TEDE dose with PWR/BWR/DHLW DBF source term, and single-stage HEPA filtration
- 5 Max inventory source term assumes no water is present in the pools

Calculation method: Using the design basis drop height for each waste form, equated the potential energy of the waste form mass dropping from its design basis height to the potential energy of the roof mass dropping from the roof height.

$$M \times g \times H = m \times g \times h$$

M = Roof mass

H = roof height

m = waste form mass

h = design basis drop height

Table 10-2. Example Seismic Calculations for Seismic Failure in Hypothetical Assembly Transfer System

ATS Location/Activity	Equipment	Potential DBE	SSC Failure	Equivalent Drop Height (in)	Source Term	Consequences, Rem	
					Number	With HEPAs	Without HEPAs
					PWR/BWR SFAs Seismic Event	PWR/BWR Seismic Event	PWR/BWR Seismic Event
<b>Receive Trans. Cask from CBS</b>	Cask transfer cart	Slapdown from transfer cart	Cask Tr. Cart	210	78/183	1.33E-2/1.53E-2	>5 / >5
A. Transfer from CHB		Collision with ATS Airlock Door					
<b>Cask Prep and Decon Room</b>	Cask transfer cart	Slapdown from transfer cart	Yoke	129	78/183	1.13E-2/1.26E-2	>5 / >5
A. Remote cask cavity gas sampling	Cask unloading area bridge crane	Cask drop from bridge crane into pit	Bridge Crane				
B. Cask venting	Cask prep manipulator	Handling equipment drops on cask	Lifting Fixtures				
C. Cask gas and water cool-down	Cask lid lifting fixture						
D. Outer lid removal	Dry cask lifting yoke						
E. Inner shield plug lifting fixture attachment	Large, small DPC lifting fixtures						
F. DPCs remotely sampled, vented, cooled	Wet cask lifting yoke						
G. DPC lifting fixture remotely attached							
<b>Cask Unloading Pool</b>	Large, small DPC overpacks	Handling equipment drops on cask	Lifting grapples				
A. Cask placed in pool, inner shield plug out	Large, small DPC lid severing tools	Handling equipment drops on DPC					
B. Cask inner shield plug removed	Wet assembly lifting grapple	Cask drop from bridge crane into pool	Bridge Crane	N/A	78/183	7.53E-3/7.40E-3	7.53E-3/7.40E-3
C. Cask containing DPC put into pool	Wet assembly transfer machine	Assembly drop onto pool floor, cask	Wet assembly transfer machine				
D. DPC removed from cask, put in overpack	Pool, downstream valves and drains						
E. DPC lid severed and removed							
F. Assemblies taken from DPC/cask to baskets							

Table 10-3. Example Seismic Classification of Structures, Systems, and Components for Hypothetical Assembly Transfer System

Location/Equipment	Equipment Identifier	SSC Seismic Failure	Seismic Frequency Category VA Design	Basis
Airlock Roof; walls	N/A	Concrete mass falls onto cask	FC-2	Heavy mass impacts cask > design basis & concurrent loss of confinement
Cask transfer cart	PU-CR-110	Slapdown from transfer cart; impact with cell walls	FC-1	Drop or impact on hard surface > cask design basis; credit for confinement
Cask transfer cart - control & motive systems	PU-CR-110	Collision with Airlock Door, breach cask; or breach building confinement	FC-1	Prevent uncontrolled motion of transfer cart to avoid collision with structures, other objects; avoid damage to airlock doors; credit for confinement
Airlock doors	N/A	Fail to maintain building confinement	FC-1	Loss of confinement for cell and waste handling building
Cask Preparation and Decontamination Room; Roof; walls	N/A	Concrete mass falls onto cask	FC-2	Heavy mass impacts cask > design basis & concurrent loss of confinement

Event trees for seismically induced sequences may be built on event trees developed previously for internal or external event initiators. The cause of the prior initiating event (e.g., drop, loss of offsite power [LOSP], and fire) in the SET is assumed to be an earthquake, however, rather than a random failure (RF). Alternatively, a SET may be developed from scratch.

In some cases, construction of an event sequence diagram (ESD) will aid in creation of a SET. The potential for common-cause failures among SSCs induced by the earthquake can be diagrammed on the ESD. These dependencies are accounted for in the structure of the SET. The ESDs are briefly described in a following paragraph.

Initially, the SET is applied qualitatively without the consideration of event frequencies with assumed dependencies between the initiating earthquake and mitigative events to define the vulnerabilities and the need for seismic hardening of SSCs. Consequences associated with each sequence of events are quantified; however, to designate the design basis earthquake associated with each SSC. After a given SSC is designed to withstand a given design basis earthquake, the failure dependency with the earthquake initiator is removed from the SET. In more advanced analyses, it may be necessary to evaluate seismic margins or to quantify the frequencies of seismic sequences using fragilities of the SSCs. This section describes both applications of SETs.

#### 10.1.5.1 Seismic Event Sequence Diagrams

An ESD is a less rigid structure than an event tree. It allows the systems analyst to respond to the question "what can happen?" in a brainstorming mode. Several examples of ESDs are described here to illustrate how they are constructed and modified to include earthquakes and intermediate events.

Figure 10-1 illustrates a simple ESD for a drop of a waste form (e.g., a spent nuclear fuel [SNF] assembly) in the WHB. The initiating event is the Drop. The immediate “what can happen?” is assumed (or known) to be a release of radioactivity from breached fuel rod cladding. With the knowledge that the purpose of the hot cell structure of the WHB and its HVAC/HEPA filters is to confine and filter any releases, respectively, it is recognized that the confinement function may be functioning or not functioning. If it not functioning (another “what can happen?”), the release may be assumed to go directly to the environment, bypassing the HVAC/HEPA filters. The consequence of this sequence of events (drop-release-not confined) is not mitigated and may not meet the regulatory dose limits. If the confinement is functioning, another “what can happen?” is the failure of the HVAC/HEPA filters to filter the release. The consequences of this sequence (drop-release-confined-not filtered) is also not mitigated and may exceed the regulatory dose limits. As discussed in Section 10.1.6.2, an event tree is developed from the ESD of Figure 10-1 as shown in Figure 10-5. Except for the event heading “Release” that is assumed to be guaranteed for a drop of an SNF assembly, all of the event headings are assumed to be independent, so the potential failure modes are random mechanical, software, or human failures.

Figure 10-2 illustrates how the example ESD is modified for an earthquake initiator. The top portion of the diagram from “Drop” to the right includes the same events as the base case. An event “No drop” has been added to illustrate that the earthquake may not induce a drop for the particular lifting device (e.g., especially if it has been designed to withstand an earthquake of a given magnitude). This diagram also indicates by dashed lines the possibility that the earthquake may directly and concurrently induce failure of the safety functions of the confinement structure and the HVAC/HEPA filter system. The consequences for the various sequences are assumed to be the same as for the internally-initiated drop, although each situation has to be evaluated for potential exacerbating factors brought about by the earthquake (e.g., previously trapped particulates in the HEPA filter might be released if the HEPA filter is failed by the earthquake, thus giving a higher consequence than the base case). The event tree for this case, now a SET, is described in Section 10.1.6.2.

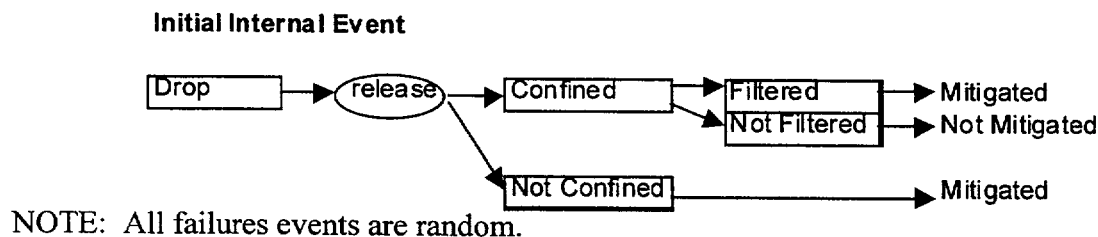
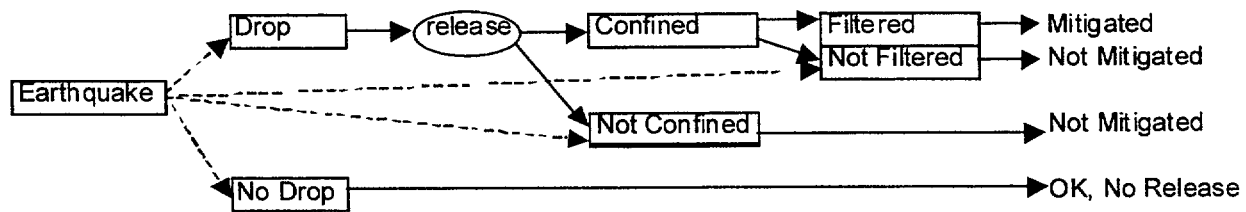


Figure 10-1. Event Sequence Diagram for Internal Initiated Drop Event

**Earthquake initiator**

NOTE: Failures events may be random or seismic. Potential seismic interactions/failures indicated by dashed lines.

Figure 10-2. Event Sequence Diagram for Seismic Initiated Drop Event

More complex seismic scenarios can be modeled in ESDs before developing SETs. Figure 10-3, for example, illustrates an ESD for internal fire-induced sequences. This ESD is seen to have the same structure as the earthquake-only case presented in Figure 10-1. However, it is unlikely that a single fire, confined to a given locale in the WHB, will be able to concurrently cause a drop and induce failure of the other safety functions. The internal-fire ESD is modified in Figure 10-4 to include an earthquake as an initiating event. Now the fire (or several fires) may be caused by the earthquake and the fire(s), in turn, may induce the drop and other failures of safety systems. However, the earthquake may concurrently cause the drop and/or other failures as well as initiate the fire. Potential consequences include radiological as well as non-radiological releases. The latter are not considered in the seismic classification of SSCs. A SET for the earthquake-fire cases is not developed in this guide, as it would be too general and speculative. As necessary, the PSA team will develop SETs specific to the WHB and other operational areas.

Similar ESDs and SETs could be developed for loss of off-site power LOSP-induced sequences. This ESD would have a similar structure as the earthquake-only case in Figure 10-2, except that there are not LOSP-induced failures of the confinement. Depending on the design of the lifting device, HVAC/HEPA filters, and the electrical supply system, however, it is may be possible for a LOSP event to concurrently cause a drop and induce failure of the HVAC/HEPA filters. The earthquake-LOSP cases is not developed in this guide, as it would be too general and speculative. As necessary, the PSA team will develop SETs specific to the WHB and other operational areas.

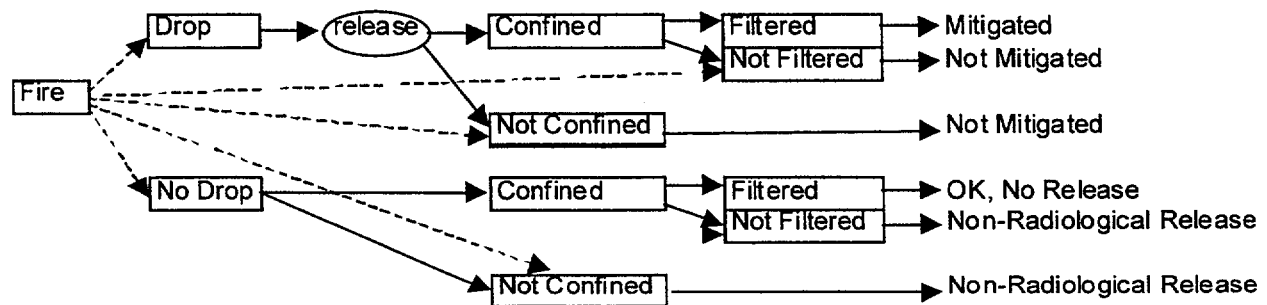
Further, an ESD on each SET could be developed to analyze indirect effects of seismic induced failures of SSC that are not important to safety by themselves. Such SSCs, however, may be located in such a way, or function in such a way, that a seismic event could cause the SSC to interact with one or more other SSC that are important to safety and, thereby, cause a loss of its safety function. (Such seismic interactions are known as two-over-one situations in NPP regulations, meaning that a Seismic Category 1 SSC (safety-related) is vulnerable to a seismic-induced fall or impact by a Non-Seismic-Category 1 component.) Logically, the seismic ESD is similar to that of the seismic fire-induced case since there is potential spatial interaction between items directly affected by the earthquake and one or more SSCs important to safety.

#### 10.1.5.2 Generic, Initial Seismic Event Tree

Figure 10-5 presents an example of a simple event tree structure for a hypothetical sequence of events associated with a facility that handles a radioactive waste form. The hypothetical

operation includes a crane that lifts and transports a particular waste form. Should the crane drop the waste form from a height exceeding its design basis, the waste form shell (i.e., its containment barrier) may breach and its contents (e.g., spent fuel assemblies) may breach, releasing radioactivity to the interior of a hot cell. If the hot cell structure remains intact, and if the HVAC ducting and HEPA filter remain intact, any releases are vented in a controlled manner. An accidental release scenario can be generated by a sequence of independent events (e.g., random failures or human errors) by seismically induced events, or by a combination of independent and seismically induced failures.

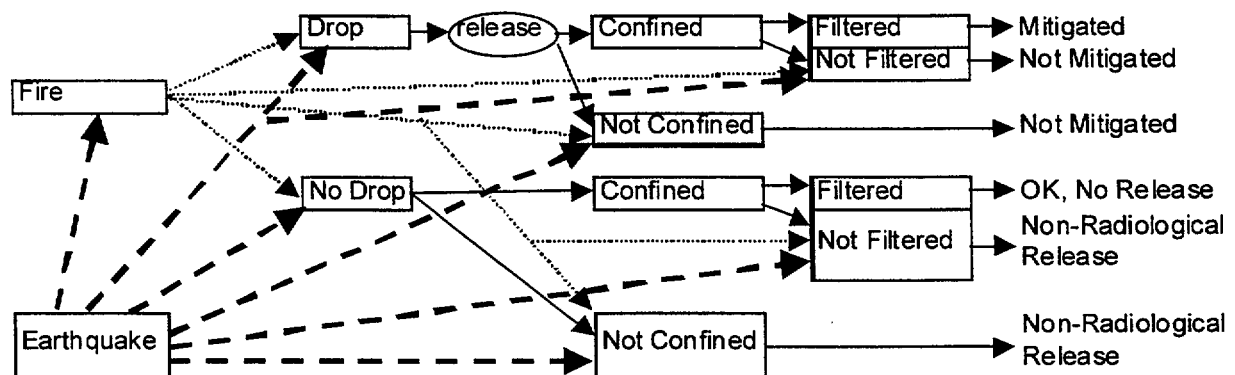
#### Fire initiator



NOTE: Failures events may be random or fire induced. Potential fire induced interactions/failures indicated by dashed lines.

Figure 10-3. Event Sequence Diagram for Fire Initiated Drop Event

#### Seismic Initiated Fire



NOTE: Failures events may be random, seismic, or fire induced. Potential fire induced failures indicated by light-dotted lines. Potential seismic induced interactions/failures indicated by heavy-dashed lines.

Figure 10-4. Event Sequence Diagram for Seismic & Fire Initiated Drop Event

Initiating Event: Earthquake	Crane Maintains Functional (1)	No Drop or Breach of WP (3)	Spent Fuel Remains Intact (4)	Confine- ment Maintained in Hot Cell (5)	HVAC Remains Intact and Functional (2)	Seq. No.	Source Term	Offsite Conse- quence (rem)
	yes	NA	NA	NA	NA	1	none	0
	GF or RF	GF	yes	yes	yes	2	C/SC, mitigated	2.00E-03
					GF or RF	3	C/SC, not mitigated	2
				GF or RF	GF - bypass	4	C/SC, not mitigated	2
		no	yes	yes		5	SNF inventory, mitigated	6.00E-03
					GF or RF	6	SNF inventory, not mitigated	6
				GF or RF	GF - bypass	7	SNF inventory, not mitigated	6

## NOTES:

Potential seismic failures in hypothetical facility include:

- (1) the failure of a crane lifting a spent fuel waste package inside a waste handling building,
- (2) damage to the building ventilation (filtration) system,
- (3) the drop and breach of the waste package,
- (4) damage to the spent fuel,
- (5) partitioning of a fraction of the radionuclide inventory to the building atmosphere,
- (6) release of some radioactive material through the damaged ventilation (filtration) system, and
- (7) exposure of an individual (either a worker or a member of the public) to the released radioactive material.

Bypass = failure of hotcell structure allows airborne radiation to bypass the HVAC ducting and HEPA filters; C/SC = crud, surface contamination, or both; GF = guaranteed failure, dependent on precursor event or on initiating event; N/A = Not asked, precursor event preclude; RF = random failure; SNF Inventory = radionuclides from inside fuel rods, surface crud, and any contamination from waste package.

Figure 10-5. Baseline Seismic Event Tree for Load Drop

Although not illustrated in Figure 10-5, an earthquake may be able to initiate a fire inside the WHB and at the same time cause failure of the fire-protection system and other systems. This scenario could evolve into a sequence of events that leads to a release of radioactivity. Similarly, an earthquake may cause a flood or a spurious actuation of the fire protection system that, in conjunction with seismic failure of geometry controls, could lead to a criticality condition. The seismic analysis for preclosure safety must be exhaustive in identifying such possibilities.

Figure 10-5 is developed for a hypothetical facility as a baseline event tree that will be modified to illustrate how seismic classifications are developed. Initially, an earthquake of unspecified intensity or frequency is represented as the initiating event. The event tree in Figure 10-5 includes five events shown across the top of the figure. These event labels are known as the event headings. The logic diagram shows a single line for the initiating event (earthquake), but allows for two branches for each challenge to each of the event headings. The two branches represent, respectively, upward (yes) when the heading event is successful (or TRUE) and downward (no) when the heading event is FALSE (e.g., the function fails or is unavailable).

The failure criteria for the function have been precisely defined so that the meaning of the “no” branch is unambiguous.

In Figure 10-5, the causes of the various “no” branches are indicated as RF (random failure), independent of the occurrence of the earthquake, or GF (guaranteed failure) due to the occurrence of a preceding event (either a random event or the initiating earthquake). A special case of GF is indicated by GF-bypass to indicate that failure to maintain hot-cell structural integrity guarantees failure of the HVAC/HEPA filter function because the radioactive air is vented via other pathway(s). When an event succeeds, such as Crane Maintains Functions, some potential succeeding events cannot occur or are irrelevant. Such branches are indicated by “NA” to indicate that the branch-point question is not asked, or is not applicable for that event heading in that sequence of events.

The cause of the crane failure may be GF (dependent on the earthquake) or RF (an independent event). The GF cases are discussed in Section 7.1). In this example, it is assumed that any drop of the waste form results in its breach, so the breach is labeled as GF. Depending on the assumptions, all or a portion of the SNF assemblies could remain intact. This event could be correlated to the height of the drop, but for this illustration it is assumed to be an independent, random event with “yes” and “no” branches. The probability of the “no” branch can be varied in sensitivity analyses.

Tracing through a particular path in Figure 10-5 arrives at an End State that represents the severity of, or absence of, a release of radioactivity to the environment. Event sequences 1 through 7 are described as follows

- **Sequence 1**—An earthquake occurs but the crane maintains its functions and does not drop the waste form. Therefore, all other event headings are irrelevant and are labeled “NA.” Since there is no release in this sequence, there is no source term nor consequences.
- **Sequence 2**—An earthquake occurs and the crane fails to maintain its functions and drops the waste form. The cause of the crane failure may be GF, dependent on the earthquake, or RF, an independent event. In this sequence all of the SNF assemblies remain intact and all other event headings function normally (all yes). The only potential source term in this scenario might be contaminants from inside the breached waste form and/or crud that has been freed from the surfaces of the SNF assemblies. Since the HVAC/HEPA filter is functioning normally, the resulting release would be limited to whatever gaseous contaminant that might have been contained in the initial waste form. As discussed in Sections 10.1.4 and 10.1.5, the magnitude of dose from this source term determines important to safety classification and the seismic classification of the crane function.
- **Sequence 3**—An earthquake occurs and the crane fails to maintain its functions and drops the waste form. The cause of the crane failure may be GF, dependent on the earthquake, or RF, an independent event. In this sequence all of the SNF assemblies remain intact and all other event headings function normally (all yes) except the HVAC/HEPA filters. The HVAC/HEPA filters may fail dependently because of the earthquake (GF) or independently (RF). The only potential source term in this scenario might be



contaminants from inside the breached waste form and/or crud that has been freed from the surfaces of the SNF assemblies. Since the HVAC/HEPA filters are not functioning normally, however, the resulting release would include volatile and particulate matter, as well as gaseous contaminants and/or crud released from surfaces of the SNF assemblies or interior of the waste form. The unmitigated dose would be expected to be higher than that of Sequence 2.

- **Sequence 4**—An earthquake occurs and the crane fails to maintain its functions and drops the waste form. The cause of the crane failure may be GF, dependent on the earthquake, or RF, an independent event. In this sequence all of the SNF assemblies remain intact. The confinement of the hot cell fails to be maintained due to a dependent failure because of the earthquake (GF) or independently (RF). In this case the failure of the HVAC/HEPA filters is labeled as GF-bypass, representing the dependency on maintaining the hot cell confinement and controlled pathway through the HVAC. The only potential source term in this scenario might be contaminants from inside the breached waste form and/or crud that has been freed from the surfaces of the SNF assemblies. Because the HVAC/HEPA filters are not providing filtration, the resulting release would include volatile and particulate matter, as well as gaseous contaminants and/or crud released from surfaces of the SNF assemblies or interior of the waste form. The unmitigated dose would be similar to that of Sequence 3.
- **Sequence 5**—An earthquake occurs and the crane fails to maintain its functions and drops the waste form. The cause of the crane failure may be GF, dependent on the earthquake, or RF, an independent event. In this sequence, the SNF assemblies do not remain intact. The remaining part of the sequence is similar to Sequence 2. The potential source term in this scenario might be the radionuclide contents of the breached fuel rods, in addition to contaminants from inside the breached waste form and/or crud that has been freed from the surfaces of the SNF assemblies. Since the HVAC/HEPA filters are functioning normally, the resulting release would be gases. The mitigated dose would be similar to, but larger than, expected to be that of Sequence 2.
- **Sequence 6**—An earthquake occurs and the crane fails to maintain its functions and drops the waste form. The cause of the crane failure may be GF, dependent on the earthquake, or RF, an independent event. In this sequence, the SNF assemblies do not remain intact. The remaining part of the sequence is similar to Sequence 3. The HVAC/HEPA filters may fail dependently because of the earthquake (GF) or independently (RF). The potential source term in this scenario might be the radionuclide contents of the breached fuel rods in addition to contaminants from inside the breached waste form and/or crud that has been freed from the surfaces of the SNF assemblies. Since the HVAC/HEPA filters are not functioning normally, however, the resulting release would include volatile and particulate matter, as well as gases, contaminants, and/or crud released from surfaces of the SNF assemblies or interior of the waste form. The unmitigated dose would be expected to be higher than that of Sequence 5.
- **Sequence 7**—An earthquake occurs and the crane fails to maintain its functions and drops the waste form. The cause of the crane failure may be GF, dependent on the earthquake, or RF, an independent event. In this sequence, the SNF assemblies do not remain intact.

The remaining part of the sequence is similar to Sequence 4. The confinement of the hot cell fails to be maintained due to a dependent failure because of the earthquake (GF) or independently (RF). In this case the failure of the HVAC/HEPA filters is labeled as GF-bypass representing the dependency on maintaining the hot cell confinement and controlled pathway through the HVAC. Since the HVAC/HEPA filters are not functioning normally, the resulting release would include volatile and particulate, as well as gases, contaminants, and/or crud released from surfaces of the SNF assemblies or interior of the waste form. The unmitigated dose would be similar to that of Sequence 6.

### 10.1.5.3 Applying Seismic Event Tree in Seismic Classification

The SET defines potentially seismic-initiated or exacerbated event sequences. The source term and resulting consequences for each event sequence determines its importance to safety and its seismic classification. The sequences are examined one at a time to identify which, if any, of the SSCs associated with the event headings have to be hardened to withstand a design basis earthquake per the design approach of Seismic Topical Report No. 2 (YMP 1997). For illustration purposes, the following hypothetical offsite dose consequences are assumed for the sequences defined in Figure 10-5:

Sequence Number	Offsite Consequences (rem)
1	0
2	0.002
3	2
4	2
5	0.006
6	6
7	6

- **Sequence 1**—SSCs require no seismic classification because the initiating earthquake is not strong enough to cause SSC failure or radiological release.
- **Sequence 2**—The dose is less than the 15 mrem limit for Category 1 event sequences so the crane function and the no-breach function of waste form are not important to safety and, therefore, these SSCs do not require seismic classification.
- **Sequence 3**—The dose is greater than 15 mrem, but less than 5 rem, so the HVAC/HEPA filter function is important to safety and must withstand an FC-1 design basis earthquake. With the HVAC/HEPA filter designed as FC-1, the crane function remains not important to safety and does not require seismic classification.
- **Sequence 4**—The dose is greater than 15 mrem, but less than 5 rem, so the hot cell confinement function is important to safety and must withstand an FC-1 design basis earthquake. With the hot cell designed as FC-1, the crane function remains not important to safety and does not require seismic classification.
- **Sequence 5**—The dose is less than 15 mrem, so the crane function is not important to safety and, therefore, does not require seismic classification.

- **Sequence 6**—The dose is greater than 5 rem, so the HVAC/HEPA filter function is important to safety and must withstand an FC-2 design basis earthquake. With the HVAC designed as FC-2, the crane function remains not important to safety and does not require seismic classification.
- **Sequence 7**—The earthquake causes a failure of the confinement and a bypass of the HVAC/HEPA filter system. The unmitigated dose is greater than 5 rem, so the hot cell confinement function is important to safety and must withstand an FC-2 design basis earthquake.

This example illustrates the application of the deterministic approach to the selection of design basis earthquakes for SSCs important to safety. The application of ESDs and/or SETs ensures a more structured analysis to identify the SSCs that need to withstand the design basis earthquakes. These tools help in carrying out the approach described in Sections 10.1.4 and 10.1.5.

This safety case will assume that SSCs designed to a given design basis earthquake will not fail under earthquake conditions up to, and including, the design basis earthquake as a direct result of the earthquake. The deterministic approach is also termed the Step Function Fragility approach to indicate that the conditional probability of failure of an SSC is zero for earthquakes of intensity less than or equal to its design basis and one for any earthquake that exceeds the design basis earthquake. The following section describes how seismic sequence frequencies could be quantified using this approach. Section 10.1.8 describes how seismic sequence frequencies would be quantified using other fragility functions.

#### **10.1.6 Frequency Quantification of Seismic Sequences Using Deterministic, Step-Function Fragility Functions or Seismic Margins Analysis**

The safety case will assume that SSCs designed to a given design basis earthquake will not fail under earthquake conditions up to, and including, the design basis earthquake as a direct result of the earthquake. If the SSCs do fail as result of the earthquake, the consequences will be within the consequence regulatory limits. It is possible, however, that independent failures of one or more SSCs could occur during an earthquake and lead to doses beyond the regulatory limits; the frequencies of such sequences, however, will have to be less than  $1 \times 10^{-6}/\text{yr}$  (i.e., beyond category 2 event sequences). This should not be too difficult since it is essentially the intersection of two independent events: an earthquake having a frequency of  $1 \times 10^{-3}$  per year for FC-1 (or  $1 \times 10^{-4}$  per year for FC-2) and an internal event sequence involving the independent failure of one or more SSCs that prevent or mitigate a release within a short time (e.g., 24 hours) after the earthquake.

**Case 1**—Figure 10-6 presents an event tree that illustrates the case where all SSCs are hardened to withstand an FC-1 design basis earthquake. The probability that the crane drops the waste form within 24 hours after the earthquake due to random failures is conservatively shown to be  $1 \times 10^{-3}$  (or  $1 \times 10^{-5}$  per lift for the lift that is in progress). Similarly, the probability that the confinement function of the hot cell and the HVAC/HEPA filters are unavailable in the release scenarios is  $1 \times 10^{-6}$  and  $1 \times 10^{-5}$ , respectively. It is seen that the frequency of all of the release scenarios resulting in doses of 2 rem or 6 rem is less than approximately  $1 \times 10^{-11}$  per year. This

frequency is too low to be considered credible. This conclusion is not surprising since the design was hardened.

**Case 2**—Figure 10-7 presents an event tree that illustrates the case where all SSCs are hardened to withstand an FC-1 design basis earthquake. However, an earthquake of a magnitude slightly larger than the FC-1 earthquake (and having a slightly lower frequency; assumed to be  $9 \times 10^{-4}$  per year) is assumed to occur. All of the SSCs (crane and lifting devices, confinement structure, and HVAC/HEPA filters fail dependently) (GF with a probability of 1.0). This results in sequences 1, 2, 3, 5, and 6 of Figure 10-6 being removed from the SET shown in Figure 10-7. The frequency of one release scenario in Figure 10-7 (Sequence 4) exceeds the Category 1 dose limits is  $9 \times 10^{-6}$  per year, (i.e., it is a credible event). However, its frequency makes it a Category 2 event sequence and its dose is less than the Category 2 limits, therefore, no seismic classification is required. The frequency of Sequence 7 is  $9 \times 10^{-4}$  which is also a Category 2 event sequence but the dose exceeds the Category 2 dose limits. Since this result is unacceptable, it indicates that some of the SSCs contributing to Sequence 7 must be hardened to withstand the FC-2 design basis earthquake.

Similarly, the SET can be modified to examine the effects of hardening some components (like the crane and HVAC/HEPA filter) to FC-1 and the hot cell confinement to FC-2. The consequences and event sequence frequencies are examples of how design bases conform with Seismic Topical Report No. 2 (YMP 1997).

While the NRC has tentatively concurred with Topical Report No. 2, they may request a demonstration of the seismic design margins that contribute to the risk-reduction factors. This demonstration will require confirmatory design analyses in the form of fragility factors or SMAs. These design analyses are beyond the scope of this PSA guide but are briefly described in Section 10.1.6. The application of such analyses in seismic sequence analyses is discussed in the following section.

#### **10.1.7 Frequency Quantification of Seismic Sequences Using Fragility Functions or Seismic Margins Analysis**

The purpose of this type of analysis is to provide a confirmatory demonstration that the Category 2 dose limits will not be exceeded in credible seismic sequences. Two approaches can be used: a seismic PRA or an SMA. Section 10.1.8.2 describes this method and the application of SETs. A seismic PRA may be performed later as part of a general PRA for the repository, if desired. The steps in a seismic PRA are briefly summarized in Section 10.1.8.1.

##### **10.1.7.1 Seismic Probabilistic Risk Assessment Analysis**

In a seismic PRA approach, three separate analyses are combined through probabilistic analyses (a convolution process) to estimate the frequencies of individual event sequences:

- **Preclosure Safety Seismic Systems Analysis**—This analysis, described in this chapter of the PSA guide, applies the methodology of event tree analysis to identify the seismically-induced sequences of events and the SSCs that come into play, and fault tree analysis to model the specific failure modes of the SSCs that come into play. The

combinations of earthquake initiator and concurrent failures of one or more SSCs are modeled in a top-event fault tree that provides

Initiating Event Earthquake	Crane Maintains Functional (1)	No Drop or Breach of WP (3)	Spent Fuel Remains Intact (4)	Confinement Maintained in Hot Cell (5)	HVAC Remains Intact and Functional (2)	Seq. No.	Source Term	Offsite Consequence (rem)	Frequency
1.00E-03	yes	NA	NA	NA	NA	1	none	0	9.99E-04
9.99E-01	RF	GF	yes	yes	yes	2	C/SC, mitigated	2.00E-03	1.00E-08
1.00E-03	1	1	0.01	1.00E+00	1.00E+00	3	C/SC, not mitigated	2	1.00E-13
				RF	GF-bypass	4	C/SC, not mitigated	2	1.00E-14
				1.00E-06	1				
		no	yes	yes		5	SNF inventory; mitigated	6.00E-03	9.90E-07
		0.99	1.00E+00	1.00E+00		6	SNF inventory; not mitigated	6	9.90E-12
				RF	GF-bypass	7	SNF inventory; not mitigated	6	9.90E-13
				1.00E-06	1				

## NOTES:

Assume crane and lifting devices, hotcell structure, and HVAC designed to withstand FC-1 earthquake.

Numbered potential seismic failures identified in Figure 10-8.

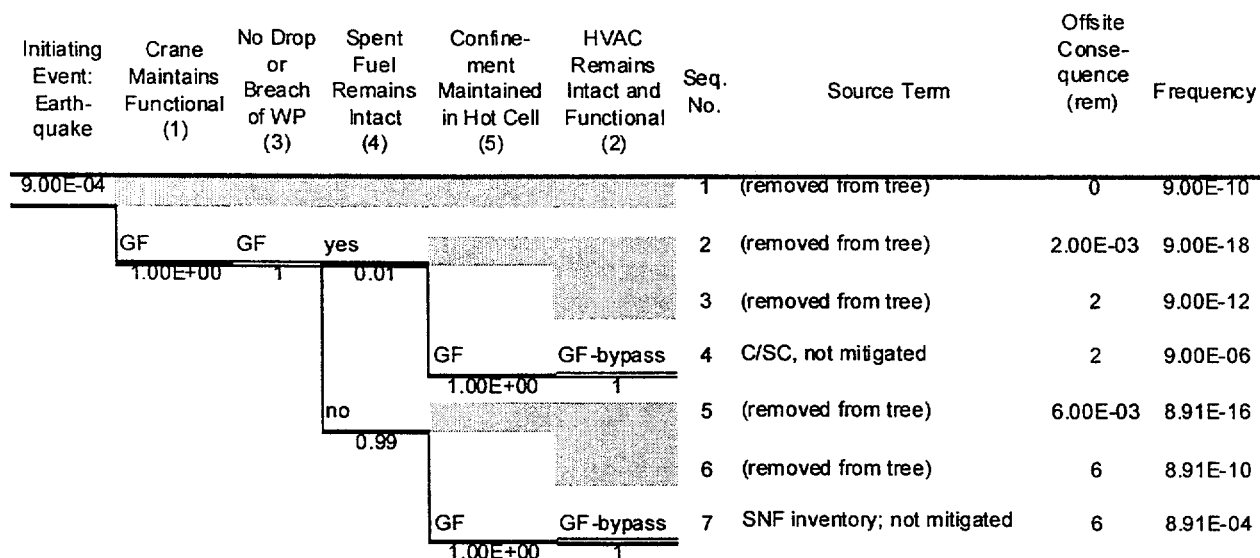
Bypass = failure of hotcell structure allows airborne radiation to bypass the HVAC ducting and HEPA filters; C/SC = crud, surface contamination, or both; GF = guaranteed failure, dependent on precursor event or on initiating event; N/A = Not asked, precursor event preclude; RF = random failure; SNF Inventory = radionuclides from inside fuel rods, surface crud, and any contamination from waste package.

Figure 10-6. Seismic Event Tree – Structures, Systems, and Components Hardened to Frequency Category 1 Earthquake

the appropriate logic model for the combinations of specific subsystems or components that have to fail during the earthquake to obtain the undesired release scenario. A PRA workstation program such as SAPHIRE (Smith et al. 2000) provides features to model the dependent, common-cause failures associated with earthquakes among multiple components in a system. The outputs of such analyses are seismic cutsets that include the various combinations of seismically induced failures and RFs.

The analysis includes the potential interactions with other SSCs (i.e., potential failure modes of a given SSC due to the impact of other SSCs in close proximity or overhead). In addition, the analyses will identify any potential seismically-initiated fire (or flooding) scenarios that can propagate into a release sequence or a criticality.

These analyses are performed by PSA personnel with the assistance of system/component design personnel.



#### NOTES:

Assume crane and lifting devices, hotcell structure, and HVAC designed to withstand FC-1 earthquake.

Numbered potential seismic failures identified in Figure 10-8.

Earthquake exceeding VBM of FC-1 earthquake occurs at  $9 \times 10^{-4}$ ; assume step function fragility for SSCs hardened to FC-1.

Bypass = failure of hotcell structure allows airborne radiation to bypass the HVAC ducting and HEPA filters; C/SC = crud, surface contamination, or both; GF = guaranteed failure, dependent on precursor event or on initiating event; N/A = Not asked, precursor event preclude; RF = random failure; SNF Inventory = radionuclides from inside fuel rods, surface crud, and any contamination from waste package.

Figure 10-7. Seismic Event Tree – Structures, Systems, and Components Hardened to Frequency Category 1 Earthquake, but Earthquake greater than Frequency Category 1

- SSC Fragility Analysis**—This analysis is applied to SSCs that appear in the seismic event sequences to determine the conditional probability of failure (or loss of safety function) given the occurrence of a peak vibratory ground motion of a given magnitude. Figure 10-8 is an example of the format of a fragility curve. It was generated as a family of lognormal distributions represented by a median value of acceleration at which a particular SSC fails to perform its safety function and through the use of uncertainty factors representing the random (aleatory) and knowledge (epistemic) uncertainty in the seismic response of the SSC.

The fragility function for each SSC is displayed as a set of three S-curves that are plots of probability (ranging from 0 to 1) versus the PGA associated with an earthquake (the acceleration ranges from fraction of g to tens of g's). The three curves represent, from top to bottom, the upper 95<sup>th</sup> percent, median, and lower 5<sup>th</sup> percent confidence levels. The abscissa of the fragility function of a given SSC is dependent upon its elevation and location within the structure (e.g., WHB) because the amplifications of motion due to

the lever-arm above the ground and the damping associated with the structure and mountings are factored in. This analysis is performed by structural analysts and/or fragility specialists. Figure 10-8 displays the format of fragility curves (this figure does not represent a real SSC). The three principal curves represent, from top to bottom, the upper 95<sup>th</sup> percent, median, and lower 5<sup>th</sup> percent confidence levels.

- **Probabilistic Seismic Hazards Analysis**—This analysis is the subject of Seismic Topical Reports Nos. 1 and 3. Topical Report No. 1 describes the method; Topical Report No. 3 provides the results. The PSHA results are in the form of a family of curves representing the annual probability (or frequency) of exceedence versus PGA at the repository site. The SMA is represented by a family of curves that include the median, mean, 5<sup>th</sup> percent Confidence Level, and 95<sup>th</sup> percent Confidence Level. This analysis is performed by a team of PSHA experts.
- **Seismic Event Sequence Frequency Analysis**—The three analyses are brought together in a probabilistic analyses, such as a Monte Carlo analysis or a Latin Hypercube analysis. Each sequence from the seismic sequence analysis is represented by one or more cutsets consisting of the product of the initiating event and the conditional failure of all of the contributing SSC.

Since the frequency of the initiating event and the fragility are both represented as probability density functions (PDFs), the product function will result in a PDF. The product PDF is obtained by a convolution analysis that is most easily performed numerically using a Monte Carlo-Latin Hypercube analysis. The output PDF provides the median, mean, 5<sup>th</sup> percent, and 95<sup>th</sup> percent bounds for each seismic sequence. The PDFs of several seismic release event sequences, having similar consequence levels, can be combined probabilistically to obtain a PDF for group of scenarios, which could all occur given an earthquake.

These analyses are performed by PSA personnel.

The seismic PRA approach is seen to be rather complex. Such analysis is not warranted for the LA submittal for construction authorization.

#### **10.1.7.2 Seismic Margins Analysis**

SMA has some similarity to seismic PRA. SMA makes use of the fragility factors and event sequence analysis, but the overall analyses are simpler than seismic PRA. The seismic hazards PDF is not directly used in the analysis, nor is an event sequence seismic PDF produced.

The goal of a SMA, as developed for NPPs, is to demonstrate that there is a high confidence of low probability of failure (HCLPF) to achieve a safe condition after the occurrence of a seismic margins earthquake (SME). The SME is stronger than a design basis earthquake. For NPPs, the safe condition is a safe shutdown and the design basis earthquake is the SSE. The SME is generally specified to be twice the SSE (i.e., for a 1-g SSE, the SME would be a 2-g PGA) for NPPs. The annual probability of exceedence of the SME is not used directly in the SMA, but it is usually assumed or shown to be at least an order of magnitude less probable than the SSE.

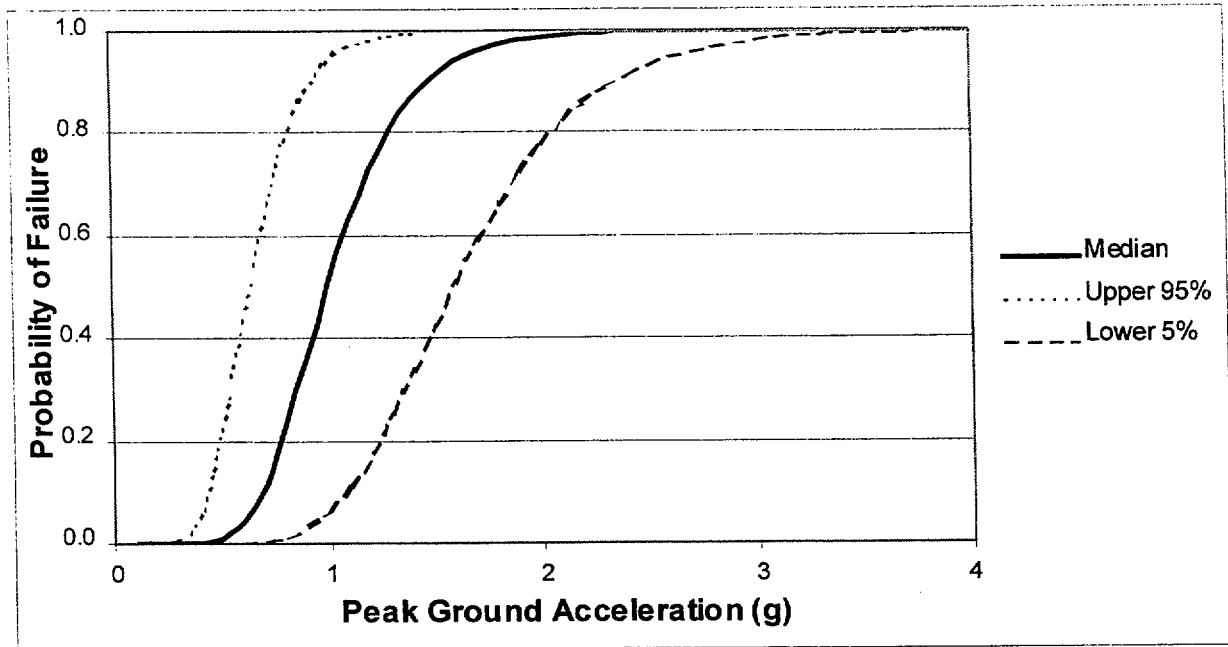


Figure 10-8. Example Fragility Curves

The SMA employs two of the three types of analyses used in seismic PRA, with some differences, as described in the following paragraphs:

- **Preclosure Safety Seismic Systems Analysis**—The approach described in Sections 10.1.4 and 10.1.5 are applied to identify the SSCs that must be designed to withstand, respectively, the FC-1 or FC-2 design basis earthquake. Designers of the affected SSCs apply appropriate methods, codes, and standards to ensure that the SSCs will perform the respective safety function while withstanding a design basis earthquake.
- **Seismic Margin Earthquake Definition**—Following the precedence of NPPs, the SME is defined to have a PGA that is twice that of the FC-2 design basis earthquake. The goal of the SMA is to demonstrate that there is a High Confidence of Low Probability of Failure (HCLPF). This HCLPF is a probabilistic measure that a seismically induced event sequence will not result in an offsite dose that exceed 10 CFR 63.111(b)(2) i.e., not exceed the dose limits for Category 2 event sequences. It is expected that no SMA will be performed for those SSCs designed to withstand an FC-1 design basis earthquake because such SSCs are already backed up by SSCs that withstand the FC-2 design basis earthquake and such an analysis is not in keeping with the philosophy of the SMA developed for NPPs.
- **SSC Fragility and HCLPF Analysis**—
  1. **Fragility analyses**—These analyses are performed for the SSCs that appear in the seismic event sequences to determine the conditional probability of failure (or loss of safety function) given the occurrence of a peak vibratory ground motion of a given magnitude. The fragility function for each SSC is displayed as a set of three



S-curves that are plots of probability (ranging from 0 to 1) versus the PGA associated with an earthquake. See Figure 10-8 and the discussion in Section 10.1.8.1.

2. **HCLPF analysis**—This analysis uses the upper 95<sup>th</sup> percent fragility curve for each SSC. The PGA of the SME is used as input on the abscissa of the 95<sup>th</sup> percent fragility curve to determine the conditional probability of failure. Using NRC precedence for SMA, if the probability of failure is less than or equal to 0.05 at the 95<sup>th</sup> percent confidence, it is considered to have HCLPF. The HCLPF value for a given SSC is expressed as the PGA, in units of “g,” at which the SSC would fail to perform its safety function with a probability of 0.05 at the 95<sup>th</sup> percent confidence level. Such analyses are performed by structural analysts and/or fragility specialists.

The step in performing a seismic margins analysis systems analysis as described in NUREG/CR-5632 (NRC 1990). As in a PRA approach, the systems analysis must define accident sequences for a range of feasible initiating events. Event sequences and consequence analyses may have been performed previously for internal-event or seismically induced initiator. Although the baseline safety analysis may have treated each of these initiating events as an independent event (i.e., drops of waste forms, loss of offsite power, fires inside the WHB, or runaway transporter), each initiating event is examined to determine if it could also be initiated by an earthquake. In addition, each event heading in the event tree for each initiator is examined for potential vulnerability to an earthquake. The vulnerability includes not only the failure of the front-line prevention or mitigation SSCs, but also the support systems (e.g., electrical power supplies, instrumentation, and control systems) and human actions that may be influenced by the occurrence of an earthquake.

The overall probability of an event sequence following an earthquake will account for dependent and independent failures. However, basic events are screened to delete the low probability independent events and dependent events that do not contribute significantly to the seismic sequences. The bases for the screening are described later.

**Modification of System Fault Trees**—The fault tree logic models for SSCs that appear in event tree headings are modified to include the potential seismically-induced failures. Each basic event (see Section 7.2) is assumed initially to be a candidate for both seismically-induced and independent (random) failures. If not already treated, the fault trees are modified to include potential common-cause failures and human induced failures. Preliminary, a non-quantitative evaluation of the cutsets is performed to check the fault-tree logic. For reactor SMA systems analyses, such fault trees are generally too complex, so screening and/or pruning is performed. Such extensive screening (or the criteria for screening) may not be necessary for the event sequences and simpler repository systems.

**Screening of SMA System Fault Trees**—The first seismic screening for a commercial nuclear power plant applies SMA guidelines (NRC 1986) that include generic HCLPF values (given in terms of PGA) for various categories of equipment. The applicability of the generic HCLPF values has to be verified by the systems and fragility analysts based on guidance and notes attached to the table of generic HCLPF values. The SSE for a typical commercial nuclear

reactor plant in the Eastern U.S. is 0.15 g and, therefore, the SME is 0.3 g (NRC 1990). Many commercial NPP SSCs have HCLPF values greater than 0.3 g and would be screened out of system fault trees. The generic databases indicate that for a SME in the range of 0.3 g to 0.5 g, most equipment has to be evaluated on a plant-specific basis. In many cases, verification of adequate equipment anchorages is sufficient to eliminate concern about the failure of a given piece of equipment during a SME.

For an SME greater than 0.5 g, however, virtually no SSCs can be screened out for the generic commercial nuclear reactor plants. Since the SME for the repository site may exceed 0.5 g, the generic HCLPF values may not provide the bases for screening. Instead, site-specific HCLPF values will have to be derived for repository SSCs.

In addition to direct seismic failure, some safety-significant SSCs in the system fault tree could be vulnerable to failure by secondary system interaction effects. These interaction effects could include failure of structural walls, falling equipment, relay chatter in the controls, or failure of electrical supply systems. Unless the specific interaction can be ruled out from design criteria or drawings, they may have to be kept in the fault trees until they are eliminated by as-built considerations such as a facility walkdown.

The next screening process aims to eliminate low-probability, non-seismic events. Since the event sequences of interest are all conditional on the unlikely occurrence of an SME, there is no point in being concerned about unlikely random events or combinations of events. Since the FC-2 design basis earthquake for the repository has an annual probability of  $1 \times 10^{-4}$ , any random event having a probability of less than approximately  $1 \times 10^{-3}$  would result in a sequence below the Category 2 threshold. Therefore, events with probability less than approximately  $1 \times 10^{-3}$  should be screened out (i.e., deleted from the fault tree model) unless there is some other reasons for retaining them, as discussed in the following paragraph.

Generally, only those equipment failure modes or human interactions with a probability greater than  $1 \times 10^{-3}$  that could exacerbate an event sequence are kept in the logic model. However, a failure with a probability less than  $1 \times 10^{-3}$  may be retained in the model if the failure could result in failures of one train each in multiple (different) systems or if the failure could result in failures of multiple trains in the same system. This screening rule was developed for reactor plant SMAs and may not be applicable to a repository.

The simplified system fault trees are then quantified using the appropriate repository database and mission times for the non-seismic events using a fault-tree program (e.g., the SAPHIRE program [Smith et al. 2000]). Cutsets to order 3 and 4 with probabilities to  $1 \times 10^{-6}$  are generated. The higher order and low probability cutsets are examined to see if any events were overlooked in the screening process. Such leftovers, if any, are subjected to the screening rules to further simplify the system fault trees.

**Fault Tree Construction for Seismic Margins Event Sequences**—A sequence fault tree is constructed for each seismic sequence that has been identified as having consequences that would exceed the 10 CFR Part 63 performance requirements if not mitigated by one or more SSCs that can withstand the design basis earthquake. The top event is the occurrence of a sequence that exceeds the performance requirements. The initiating event (earthquake) and other

events are input to the top event through an AND gate. The sequence fault tree is then analyzed to identify the minimal cutsets for the sequence. Each cutset will include, in addition to the initiating earthquake, one or more seismically-induced failures or random events. The random events include equipment failures, human errors, and common-cause failures.

The list of minimal cutsets for each sequence reveals the seismically-induced failures that can result in the top event. Seismic failures may appear in the cutsets as singles or doubles.

**Seismic Margins Evaluation**—Since 10 CFR Part 63 does not include an integral risk measure, the seismic margins evaluation can be applied to each sequence whose consequences exceed 10 CFR 63.111(2).

The probability of the initiating event is assumed to 1.0. The objective of the analysis is to demonstrate that a suitably low conditional probability of an unacceptable release exists, given an earthquake of the magnitude of the SME (specified here as vibratory ground motion having peak ground acceleration of  $g_{SME}$ ). Because the components having a HCLPF greater than  $g_{SME}$  were already screened out of the model, the remaining potential seismically-induced events could potentially cause the unwanted release. However, it is desired that none of the events will appear in cutsets as singles, but only in doubles or higher-order cutsets.

In SMA for NPPs that have complex sequence analyses involving fault trees from several preventive and mitigation systems, it is typical to apply additional screening rules to the sequence cutsets to keep only cutsets that meet the following criteria:

1. Single events, either seismic or random
2. Double cutsets that contain at least one seismic failure
3. Triple cutsets that contain at least one seismic failure
4. Quadruple cutsets containing at least one seismic failure and a conditional event that effectively reduces the cutset to a triple.

After applying these screening rules, a reduced cutset list is obtained. The laws of Boolean algebra (mainly the absorption law) are applied so that many cutsets containing the same seismic failure, or RF, are absorbed into the lowest order cutset containing that failure event. The result is the final Boolean expression for a given sequence or the top event probability that is being considered (e.g., the sum of several sequences that give the unwanted consequence).

The final Boolean expression is examined for the final conditional probability. As long as there is no single-event seismic term (having an HCLPF less than  $g_{SME}$ ) in the final Boolean expression and such seismic failures appear only in double or triples whose non-seismic probability is less than  $1 \times 10^{-3}$ , it can be concluded that the sequence probability meets the seismic margins criteria. If the seismic margin is adequate, it is generally concluded that the only path for the unwanted sequence(s) will result from RFs (often from human action or common-cause failure). The likelihood for a human error, for example, may be assumed to be quite high following a SME.

## **10.2 FLOODING**

### **10.2.1 Purpose**

This section defines the methods to determine the potential for flooding and the extent of any flood protection required for SSCs important to radiation safety or waste isolation.

### **10.2.2 Scope**

This section provides guidance for the interpretation and use of NRC guidance and industry standards to determine the applicability of flooding for the repository. Areas to be considered include historical flooding, intense precipitation, upper level of possible flood conditions, facility design to determine whether flood effects need to be considered in plant design or emergency procedures, and the extent of any flood protection required for SSCs necessary for preclosure safety and waste isolation.

### **10.2.3 Overview of Approach**

Compliance with 10 CFR 63.21(c)(1)(iii), NRC guidance provided by NUREG-0800 (NRC 1987, Chapter 2), and industry standards are used for guidance for acceptable criteria for license application.

### **10.2.4 Details of Approach**

#### **10.2.4.1 Potential for Flooding**

The flood history and the potential for flooding are to be reviewed for applicability. This section should provide a summary and identification of the flood-producing phenomena applicable to the site or region considered in establishing the flood design bases for SSCs. Phenomena to be examined include the following:

- Stream flooding (probable maximum flood [PMF])
- Surges
- Seiches
- Tsunami
- Seismically induced dam failures
- Flooding caused by landslides
- Ice loadings from water bodies.

Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*, provides previous NRC guidance for estimating the design basis for flooding considering the worst single phenomena and combinations of less severe phenomena. Regulatory Guide 1.29 identifies the SSCs and Regulatory Guide 1.102 describes acceptable flood protection to prevent the SSCs from being adversely affected.

#### 10.2.4.2 Determination of Maximum Water Level

ANSI/ANS-2.8-1992, *American National Standard for Determining Design Basis Flooding at Power Reactor Sites* (ANSI/ANS 1992), defines the PMF as:

The hypothetical flood (peak discharge, volume, and hydrograph shape) that is considered to be the most severe reasonably possible, based on comprehensive hydrometeorological application of probable maximum precipitation and other hydrologic factors favorable for maximum flood runoff such as sequential storms and snowmelt.

The same ANSI/ANS standard also defines the probable maximum precipitation as follows:

The estimated depth of precipitation for a given duration, drainage area, and time of year for which there is virtually no risk of exceedance. The probable maximum precipitation for a given duration and drainage area approximates the maximum that is physically possible within the limits of contemporary hydrometeorological knowledge and techniques.

These definitions describe events that are the most severe or the greatest physically possible for a specific site. The probability of occurrence of these events should be extremely low. By conservatively assuming the occurrence frequency is greater than  $1 \times 10^{-6}$ , rainfall-related flooding events will be evaluated as Category 2 event sequences. The following steps should be followed to perform a flood analysis:

- Review and comply with the Yucca Mountain Standard Review Plan (when issued).
- Identify flooding events that are applicable to the site or operating facilities. Candidate events are documented in the *MGR External Events Hazards Analysis* (CRWMS M&O 2000a) and *Monitored Geologic Repository Internal Hazards Analysis* (CRWMS M&O 2000b). The list of applicable internal and external events was screened at a high level for gross credibility, applicability to preclosure, radiological safety, and applicability to the Yucca Mountain location. Stream flooding, surges, seiches, tsunami, seismically induced dam failures, flooding caused by landslides, and ice loadings from water bodies are to be considered as required by NUREG-0800 (NRC 1987, Section 2.4.2).
- Identify historical flooding at the site and the region of the site. Identify the types of flood-producing phenomena that are considered in establishing the flood design bases for important to safety design features (e.g., stream flooding, surges, seiches, tsunami, dam failure, flooding caused by landslides, and ice loadings from water bodies).
- Consider 10 CFR Part 50, Appendix A, *General Design Criterion 2, Design Bases for Protection Against Natural Phenomena*, as it relates to important to safety SSCs being designed to withstand the effects of applicable external flooding initiating events.

- Consider 10 CFR Part 100, *Reactor Site Criteria*, as it relates to identifying and evaluating specific criteria for flood history, flood design considerations, effects of local intense precipitation.
- Consider Regulatory Guide 1.29, *Seismic Design Classification*, for estimating design basis flooding considering the worst single phenomena and combinations of less severe phenomena. This guide identifies important to safety SSCs and Regulatory Guide 1.102, *Flood Protection for Nuclear Power Plants*, describes acceptable flood protection to prevent important to safety facilities from being adversely affected.
- Consider publications of the U.S. Geological Survey, National Oceanic and Atmospheric Administration, Soil Conservation Service, Corps of Engineers, applicable State and river basin authorities, and other similar agencies relating to hydrological characteristics and extreme events in the region.
- A sample statement of an NRC evaluation of water level findings is provided in NUREG-0800 (NRC 1987, Section 2.4.2.IV). This sample statement will provide a sample of the results the NRC must be able to confirm.

#### **10.2.4.3 Probable Maximum Flood on Streams and Rivers**

The PMF as defined in Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*, should be evaluated to establish the stream and river flooding design basis referred to in 10 CFR Part 50, Appendix A, *General Design Criterion 2*. The probable maximum precipitation should also be evaluated for the roofs of important to safety structures. One of the following three conditions must be met:

1. The elevation attained by the PMF establishes a required protection level to be used in the design of the facility.
2. The elevation attained by the PMF is not controlling; the design basis flood protection level is established by another flood phenomenon (e.g., the probable maximum hurricane).
3. The site is dry (i.e., the site is well above the elevation attained by a PMF).

For condition 3, the site grade must be well above the NRC-determined PMF water levels. The evaluation of the adequacy of the margin (difference in flood and site elevations) is generally a matter of engineering judgement. This judgement is based on the confidence in the flood level estimate and the degree of conservatism in each parameter used in the estimate.

The following documents may be used as appropriate to determine the PMF data acceptability:

- Regulatory Guide 1.59 provides guidance for estimating the PMF design basis.
- Regulatory Guide 1.29 identifies the important to safety SSCs.

- Regulatory Guide 1.102 describes acceptable flood protection to prevent important to safety facilities from being adversely affected.

#### **10.2.4.4 Potential Dam Failures, Surge, Seiche, and Tsunami Flooding**

Justify the conclusion (as found in the *MGR External Events Hazards Analysis* [CRWMS M&O 2000a]) that there are no water control failure, surge, seiche, or tsunami flooding to evaluate for the Site.

#### **10.2.5 Flooding Protection Requirements**

Review and consider, as appropriate, SSCs relied upon for plant flood protection whose failure could result in uncontrolled release of significant radioactivity to assure conformance with 10 CFR Part 63.

If flood protection is required for any important to safety SSCs, consider, as appropriate, 10 CFR 50.55a requirements for SSCs to be designed and constructed to quality standards commensurate with the importance of the safety function to be performed.

NUREG-0800 (NRC 1987) requires a determination as to which SSCs are important to safety and should be protected against floods or flooded conditions. A failure modes and effects analysis may be performed to determine that the flooding consequences resulting from failures of such liquid-containing systems close to essential equipment will not preclude required functions of safety systems.

#### **10.2.6 Consequence Analysis**

For each applicable credible event, either individually or in combination with other events in an accident sequence, perform: (1) a frequency analysis that demonstrates the event is not credible (2) a nuclear safety analysis that demonstrates a radiological release does not occur as a result of the event (3) a consequence analysis that demonstrates that the radiological consequences of the event are within regulatory requirements or that identifies required preventative or mitigative SSCs that ensure the radiological consequences are within regulatory requirements.

For each of the credible events identified, dose assessments will be performed to show compliance to 10 CFR Part 63 requirements as applicable.

The frequency analysis of an event determines if the event is credible. If not credible, no quantitative dose limits are promulgated by 10 CFR Part 63, no further analysis is required, and there is no impact to other repository design or licensing organizations. Event sequences that are beyond Category 2 will be tracked to show safety design margin. If the event is determined to be credible, it is categorized based on the 10 CFR Part 63 definition, and an analysis is performed to determine if the dose limits associated with the applicable event category can be met.

Category 1 event sequences (i.e., frequency greater than  $1 \times 10^{-2}$  events per year) require that the sum of annual doses, exposures, and releases do not exceed limits specified in 10 CFR Part 63 for the public and 10 CFR Part 20 for occupational workers. Category 2 event sequences (i.e., frequency less than  $1 \times 10^{-2}$  and greater than  $1 \times 10^{-6}$  events per year) require that the

consequences of a specific Category 2 event sequence not exceed dose limits as specified in 10 CFR Part 63 for the public beyond the preclosure controlled area.

The consequence analysis determines if the calculated doses are within the applicable regulatory limits. If the calculated dose exceeds applicable limits, SSCs important to radiological safety are designated, new requirements are allocated to the system, assumptions are revised, the design configuration is revised (if necessary), and the dose is recalculated and again compared to applicable regulatory dose limits.

## **10.3 WINDS AND TORNADOS**

### **10.3.1 Purpose**

This analysis defines the methods to determine the design basis wind speed and design basis tornado (including tornado-generated missile) and the protection required for SSCs important to preclosure safety and waste isolation. These parameters will serve as the basis for the requirements that are developed to mitigate the effect of design basis winds and a design basis tornado on the structural stability of the repository surface facilities. The primary systems that will contain requirements developed from this section include the WHB, Waste Treatment Building, and Carrier Preparation Building.

### **10.3.2 Scope**

This section provides guidance on the interpretation and use of NRC guidance and industry standards to determine the applicability of external storm events for the repository. A preliminary hazard analysis performed in 1996 for the repository screened out the majority of the postulated external storm events (CRWMS M&O 1996). The analysis was unable to screen out extreme winds and tornado-related events.

### **10.3.3 Overview of Approach**

Compliance with 10 CFR 63.21(c)(1)(iii), NRC guidance provided by NUREG-0800 (NRC 1987, Chapter 3) and industry standards are used for guidance for acceptable design basis/requirements for the LA submittal.

### **10.3.4 Details of Approach**

#### **10.3.4.1 Methodology**

Currently, no regulations for design basis tornadoes or extreme winds have been promulgated for high-level radioactive waste sites. Therefore, the criteria used to develop the repository design basis for those phenomena have been adopted from the guidance provided in NUREG-0800 (NRC 1987).

#### **10.3.4.2 Extreme Winds**

The typical method for demonstrating compliance of the design of structures that have to withstand the effects of extreme winds is provided in NUREG-0800 (NRC 1987, Sections 2.3.1



and 3.3.1). This NUREG states that the 100-year return period “fastest mile of wind,” including the vertical velocity distribution and gust factor, should be used as the design and operating bases (NRC 1987, Section 2.3.1) and be based on applicable ANSI building code requirements, with suitable corrections for local conditions. The current standard published by the American National Standards Institute is ASCE 7-98, *Minimum Design Loads for Buildings and Other Structures* (ASCE 2000). The basic wind speed defined in this document is a 3-second gust with an annual probability of 0.02 of being equaled or exceeded (50-year mean recurrence interval). Regional data can be used to determine the basic wind speed.

The most recent site-specific wind speed data must be used in preclosure safety analyses. Wind speed data is being collected near the repository site, and includes observed maximum daily one-second gust and one-minute wind speed at nine locations for the period from 1993 to 1996 (CRWMS M&O 1997). The magnitude of the 50-year and 100-year return wind speeds were also estimated from this site-specific data. The example data shown in Table 10-4 corresponds to the location with the highest value in the meteorological monitoring network.

Table 10-4. Example Maximum Estimated and Observed Wind Speeds near Yucca Mountain, Nevada

	Wind Speed (m/sec)[mph]	
	50-year (3 second gust)	100-year (1 minute)
Observed	40.22 [90]	33.16 [74]
Estimated	54.11 [121]	48.47 [109]

Source: CRWMS M&O 1997

### 10.3.4.3 Tornado

Regulatory Guide 1.76 provides guidance for the selection of a design basis tornado for the three regions applicable to the continental United States. The design basis tornado characteristics for the applicable region can be used in the design of important to safety SSCs. A design basis **tornado** with less conservative parameter values than the regional values given in Regulatory Guide 1.76 can be selected. If the less conservative design basis tornado is selected for use the MGR, a comprehensive analysis must be performed to justify the selection. The subjects in the following paragraphs should be considered when determining the design basis tornado for the Yucca Mountain site.

The method for demonstrating compliance of the design of structures that have to withstand the effects of design basis tornadoes is provided in NUREG-0800 (NRC 1987). This guide states that facilities must be designed so that facilities remain in a safe condition in the event of the most severe tornado that can reasonably be predicted to occur at a site as a result of severe meteorological conditions.

The design basis tornado characteristics provided by Regulatory Guide 1.76 are shown in Table 10-5. Using these properties, it is possible to develop a definition for a design basis tornado in terms of the six tornado parameters and use analytical techniques for estimating values of these parameters for purposes of design with an adequate level of conservatism.

Table 10-5. Design Basis Tornado Characteristics

Region**	Maximum Speed* (mph)	Rotational Speed* (mph)	Translational Speed (mph)		Radius of Maximum Rotational Speed (feet)	Pressure Drop (psi)	Rate of Pressure Drop (psi/sec)
			Max	Min			
I	360	290	70	5	150	3.0	2.0
II	300	240	60	5	150	2.25	1.2
III	240	190	50	5	150	1.5	0.6

Source: Regulatory Guide 1.76

\* The maximum wind speed is the sum of the rotational speed component and the maximum translational speed component.

\*\* Region refers to the three tornado intensity regions within the contiguous United States as listed in Figure 1 of Regulatory Guide 1.76. Region III applies to the area surrounding Yucca Mountain.

The design basis tornado characteristics in Table 10-5 are based on a tornado that has a probability of occurrence of  $10^{-7}$  per year (Ramsdell and Andrews 1986).

Subsequent to the issuance of Regulatory Guide 1.76, the American Nuclear Society, through the American Nuclear Standards Institute, published ANSI/ANS-2.3-1983, *Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites* (ANSI 1983). This publication established guidelines to estimate the frequency of occurrence and the magnitude of parameters associated with tornadoes, hurricanes, and other extreme winds. Figures were presented that illustrated the regionalized tornado wind speed corresponding to a given probability. The information is summarized in Table 10-6. Although this publication expired in 1993, it represented the current state of knowledge on tornado and extreme wind characteristics at the time of publication (ANSI 1983).

Table 10-6. Tornado Wind Speed (miles per hour) by Region

Region *	Probability of Occurrence per Year		
	$10^{-5}$	$10^{-6}$	$10^{-7}$
I	200	260	320
II	150	200	250
III	100	140	180

Source: *Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites* (ANSI 1983)

\* Region III applies to the area surrounding Yucca Mountain.

In 1986, the NRC issued new guidance on tornado strike and intensity probabilities in NUREG/CR-4461, *Tornado Climatology of the Contiguous United States* (Ramsdell and Andrews 1986). The new guidance was based on 30 years of data contained in the National Severe Storms Forecast Center tornado database from the period of January 1, 1954 through December 31, 1983. The report contains tornado characteristics including the number of occurrences, frequencies of occurrence, and average dimensions. Values are provided for 5-degree latitude and longitude boxes for the contiguous United States.

Table 10-7 lists example wind speeds provided for  $10^{-5}$ ,  $10^{-6}$ , and  $10^{-7}$  per year probabilities of occurrence for the 5-degree latitude and longitude box containing the repository site. Both the nominal (expected) value and the value associated with the upper end of the 90 percent confidence interval for strike probabilities are shown. Statistically, this latter value is interpreted as the maximum value in a range that has a 90 percent chance of containing the true strike probability. The wind speed value ( $10^{-6}$  per year) selected for the repository is 189 miles per hour.

Table 10-7. Example Tornado Wind Speed for 5-Degree Latitude and Longitude Box Containing Yucca Mountain, Nevada

	Strike Probability of Occurrence per Year		
	$10^{-5}$	$10^{-6}$	$10^{-7}$
Nominal Wind Speed* (mph)	-	131	Not provided
Upper 90% Wind Speed* (mph)	151	189	189

Source: Ramsdell and Andrews (1986)

\* Wind speed is the sum of the translational and rotational components.

The pressure drop and the rate of pressure drop associated with a maximum tornado vortex (funnel) impacting a structure are two additional design basis tornado requirements needed for repository structural analyses. These values were not provided with the example estimated speed shown in Table 10-7.

Table 10-5 shows Regulatory Guide 1.76 pressure terms corresponding to the wind speeds given for the three regions of the United States. The table lists maximum tornado wind speeds, rotational speeds, maximum and minimum translation speeds, an assumed vortex radius of 150 feet, and the corresponding pressure drop and rate of pressure drop. The maximum pressure drop (pounds-force per square-inch) values can be calculated from the total and translation speeds using the methodology presented in ANSI/ANS-2.3-1983 (ANSI 1983):

#### 10.3.4.4 Tornado-Generated Missiles

The typical method for demonstrating compliance with the design of structures that have to withstand the effects of tornado-generated missiles is provided in Sections 3.5.1.4 (*Missiles Generated By Natural Phenomena*) and 3.5.3 (*Barrier Design Procedures*) of NUREG-0800 (NRC 1987). Important-to-safety equipment must be protected, as required, against damage from missiles that might be generated by the design basis tornado.

NUREG-0800 (NRC 1987, Section 3.5.1.4) requires that at least three objects (missiles) must be postulated: a massive high kinetic energy missile that deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. The NUREG identifies two missile spectrums that will satisfy these criteria. Spectrum I missiles include: an 1,800 Kg automobile, a 125 Kg 8-inch armor-piercing artillery shell, and a 1-inch solid steel sphere. The impact speed required is 35 percent of the maximum horizontal wind speed of the design basis tornado. The first two missiles are to impact at normal incidence, the last to impinge upon barrier openings in the most

damaging directions. Spectrum II missiles may be used as an alternative to Spectrum I missiles. Spectrum II missiles and associated horizontal speed are shown in Table 10-8.

Table 10-8. Spectrum II Missiles

Missile	Mass (Kg)	Dimensions (m)	Velocity (m/sec)*
A. Wood Plank	52	0.092 x 0.289 x 3.66	58
B. 6-inch Sch 40 pipe	130	0.168D x 4.58	10
C. 1-inch Steel Rod	4	0.0254D x 0.915	8
D. Utility Pole	510	0.343D x 10.68	26
E. 12-inch Sch 40 pipe	340	0.32D x 4.58	7
F. Automobile	1,810	5 x 2 x 1.3	41

Source: NRC (1987).

NUREG-0800 (NRC 1987).

\* Associated with Region III, Regulatory Guide 1.76.

Vertical velocities of 70 percent of the postulated horizontal velocities are used in both spectra except for the small missile in Spectrum I or missile C in Spectrum II. These missiles should have the same velocity in all directions. Missiles A, B, C and E are to be considered at all elevations and missiles D and F at elevations up to 30 feet above all grade levels within ½ mile of the facility structures.

### 10.3.5 Wind and Tornado Protection Requirements

The typical method showing compliance with the protection of SSCs important to radiological safety and waste isolation that have to withstand the effects of extreme winds and tornadoes is provided in NUREG-0800 (NRC 1987, Sections 3.5.2) and Regulatory Guide 1.117. SSCs to be protected from externally generated missiles include all SSCs that have been provided to ensure radiological safety and waste isolation. Based on their relation to safety, SSCs are identified as requiring protection from externally generated missiles if a missile could prevent the intended safety function, or if as a result of a missile impact on a non-important to safety SSC, its failure could degrade the intended safety function of a important to safety SSC.

The primary repository surface facilities to be considered for protection should include the Carrier Preparation Building, the Waste Treatment Building, the WHB, and any other facility structure that contains radioactive material. The QL-1 systems within identified affected structures must be protected against extreme winds, tornadic winds, and tornado-generated missiles because of the potential to cause a radiological release. It is necessary to demonstrate that failure of any structure or component will not affect the capability of other structures or components to perform their required safety function(s).

### 10.3.6 Consequence Analysis

Identify those SSCs that must withstand the effects of extreme winds, tornadoes, and tornado generated-missiles. An analysis must determine the design basis tornado frequency for repository design. This will ensure that a tornado-initiated sequence resulting in a radioactive

release that exceeds 10 CFR Part 63 performance objectives is categorized as non-credible event (having a frequency less than the Category 2 event sequence frequency cutoff).

The information provided in NUREG/CR-4461 (Ramsdell and Andrews 1986) represents the latest guidance on design basis tornadoes published by the NRC. This guidance will be used to determine the maximum wind speed of the design basis tornado. This value represents the upper end of the 90<sup>th</sup> percent confidence interval and reduces the uncertainty due to limited data sets associated with tornadic phenomena in western regions.

Either the Spectrum I or Spectrum II missile spectrum may be used to design the repository surface facilities for tornado generated missiles. Example design basis requirements for tornado generated missiles are shown in Table 10-9. Future analyses may determine whether the wind speed of the design basis tornado is sufficient to generate missiles with the entire generic missile spectrum. Portions of the missiles spectrum may be removed from the design basis missile spectrum if it is determined that they are not applicable to the location of the repository surface facilities.

#### 10.4 LIGHTNING AND EXTREME WEATHER

Lightning and extreme weather are natural phenomena that will be assumed to occur at least yearly. Lightning is a large-scale high-tension natural electric discharge in the atmosphere. When lightning strikes a building, a transporter, or an electrical component, the consequences may be a localized temperature increase, a loss-of-offsite power, or a short circuit. In addition, a lightning strike may initiate a fire.

Lightning and/or extreme weather are (at a minimum) to be analyzed as potential initiators for a loss-of-offsite power event and an internal fire. Analysis of lightning- and extreme weather-initiated event sequences is expected to demonstrate that there are no credible release scenarios that result from a lightning strike.

#### 10.5 WILDLAND FIRES

The objective of this section is to outline an approach for analysis of an external-initiated fire and related hazards identification for repository facilities sufficient to minimize the potential for: (1) damage from a wildland fire or related event; (2) an external-initiated fire that causes an unacceptable onsite or offsite release of hazardous or radiological material that will threaten the health and safety of employees, the public or the environment; and (3) critical process controls and safety class systems being damaged as a result of an external-initiated fire and related events.

The *MGR External Events Hazards Analysis* (CRWMS M&O 2000a) has previously identified wildland fire as a potential hazard to the repository. An analysis of the wildland fire hazard should consider the potential event sequences that could result in fire or fire-related damage to the repository facilities. The area surrounding the repository site is mostly barren with no trees and small, low-growing vegetation. The *Standard for the Protection of Life and Property from Wildfire*, NFPA 299 (NFPA 1997), should be used as guidance for site protection from wildfire. This standard presents minimum planning criteria for the protection of life and property from wildfire. Practices such as defensible spaces around buildings NFPA 299 (NFPA 1997) and use of noncombustible building materials may reduce the probability of damage from wildfire.

The section of the PSA that provides the analysis of fires should provide a comprehensive list of wildland fire hazards such that mitigation and response plans can adequately address the hazard, including adequate water supplies, reliable means of taking water to the fire, access route for emergency response vehicles, and personnel trained in wildland fire-fighting.

The hazards from a wildland fire can be reduced by the use of a fire protection program and design requirements for facilities that are designed and constructed to meet the applicable building code and National Fire Protection Association codes and standards, or exceeding them (when necessary to meet safety objectives).

The fire hazards analyses for the repository facilities will consider the hazards of wildland fires. The conclusions of the fire hazards analyses will be integrated into design basis and beyond design basis accident conditions.

## **10.6 OTHER EXTERNAL EVENTS**

### **10.6.1 Loss-of-Offsite Power**

The likelihood of loss-of-offsite power should be estimated for the Yucca Mountain site based on historical information for the region.

Loss-of-offsite power events at the repository may be likely to occur one or more times during the preclosure operations. Therefore, loss-of-offsite power will be a Category 1 initiating event sequence. The strategy for this event is to prevent credible release scenarios by design. Repository SSCs important to safety may be designed to fail safe during a loss-of-offsite power event. Important-to-safety cranes may also be designed in accordance with NUREG-0554 (NRC 1979) to preclude single point failures.

Emergency backup power sources and redundant offsite power lines/sources may be used to ensure continuous power is supplied to SSCs important to safety. The repository design may also include features such as external lightning rods to protect against a lightning-initiated loss-of-offsite power event.

Using the frequency of loss-of-offsite power event trees (see Section 7.1) and fault trees (see Section 7.2) should be developed. Event sequences determined from this study can be used to identify possible credible event sequences that result in offsite releases.

### **10.6.2 Aircraft**

#### **10.6.2.1 Purpose**

This guide recommends methods to determine aircraft hazard potential and the extent of any protection required for SSCs important to radiation safety or waste isolation.

Aircraft crashes were determined to be potentially applicable to the potential repository at Yucca Mountain in the *MGR External Hazards Analysis* (CRWMS M&O 2000a). This determination was conservatively based on limited knowledge of the flight data in the area of concern and the crash data on aircraft of the type flying near the repository. An aircraft frequency analysis will

meet the requirements of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987). The aircraft frequency analysis will establish the frequency of aircraft crashes into radioactive material control facilities at the repository. The results of that analysis will determine if an aircraft crash event is credible and warrants performing consequence analyses needed to quantify the risk of public exposure to radioactive materials.

#### **10.6.2.2 Scope**

The recommended approach to be used in performing aircraft crash analysis for the repository surface facilities is presented in this section. The NRC requires a determination of the probability of an aircraft crash and a consequence analysis if the probability exceeds allowable dose limits. Meeting this requirement could involve up to three phases: development of a vicinity map, crash frequency analyses, and, if necessary, a consequence analysis. This section focuses on the approach to be used for the first two phases. Example calculations are described to illustrate how these methods could be applied. All information used in these examples is subject to change.

#### **10.6.2.3 Overview of Approach**

Nuclear waste repository licensing requirements are defined in 10 CFR Part 63 wherein events with probabilities greater than 1 in 10,000 (based on an expected surface facility lifetime of 100 years) are considered credible event sequences. This probability limit equates to an event sequence frequency of  $1.0 \times 10^{-6}$  per year, which is used to determine if an aircraft crash event is credible and, if so, the consequences of the event must be evaluated. If the event is not credible and the NRC accepts this conclusion, no further analysis is required.

The Nellis Air Force Base (AFB) is located near Las Vegas. The potential repository site is located in the northwestern section of the NTS. The Nevada Test and Training Range includes the Nellis Air Force Range and surrounding military operations areas. The Nevada Test and Training Range surrounds the NTS on the east, north, and west sides. As such, it must be shown that aircraft flying from the base to the range and other aircraft related activities in the range and other surrounding areas would not result in a safety issue that cannot be prevented or mitigated. In addition, civilian air traffic to and from the Las Vegas McCarran International Airport and other smaller airports must be considered.

The airspace in Southern Nevada includes military operations areas, restricted areas, and general aviation areas. The responsibility for managing the airspace within these area is delegated to the United States Air Force, DOE, and the Federal Aviation Administration as appropriate. Air traffic outside of restricted areas is routed on commercial airways and military training routes. Air traffic within restricted areas does not follow specific flight corridors, but fly mission-specific routes resulting in a random distribution of aircraft within these areas. The Air Force has agreements with DOE for use of the NTS airspace for ingress and egress of the Nevada Test and Training Range.

Military air traffic is mainly composed of fighter and attack aircraft such as F-15s, F-16s, and A-10s. During large training exercises, other types of aircraft including bombers, tankers,

helicopters, and other U.S. and non-U.S. fighter aircraft are added to the air traffic. Military aircraft generally fly in normal cruise mode during ingress and egress of the test and training ranges but can perform some preliminary tactical maneuvers such as g-awareness exercises during this phase.

The typical military training missions within the military operations areas and restricted areas include basic flight maneuvers, air combat maneuvers, day/night weapons delivery, and large multi-aircraft exercises. Several types of ordnance are carried on these aircraft. Military Training Routes are used for low-altitude and navigational training.

#### **10.6.2.4 Details of Approach**

NUREG-0800 (NRC 1987) includes three acceptance criteria, and meeting all three would eliminate the need for crash frequency analyses. Because of the extensive military traffic in the vicinity of the potential repository, each of the three criteria may not be met and frequency analysis is performed as part of phase 2. Various analytical models used to determine the crash frequencies of different aircraft-related activities involve a determination of effective target areas, crash rates, number of flights, flight corridor widths and potential crash areas.

Because use of the NUREG-0800 (NRC 1987) airway model may be conservative for application to a repository at Yucca Mountain, another model should be evaluated to provide a comparative analysis. Suggested models are provided in the following sections.

##### **10.6.2.4.1 Use of NUREG-0800**

The analytical model in NUREG-800 (NRC 1987, Section 3.5.1.6.III.2) is best suited for calculating crash frequency of aircraft that fly well-defined routes such as Federal Aviation Administration commercial airways and jet routes. This NUREG addresses aircraft hazards to NPPs; however, this same methodology can be applied to other nuclear facilities. The NUREG includes proximity criteria, which, if met, would dismiss the event by inspection. If the proximity criteria are not met, a detailed review of the aircraft hazards must be performed.

The NUREG defines a process to be used by the NRC staff in reviewing the applicant's assessment of aircraft hazards. This process includes models for determining the probability of an aircraft crash at the repository site from airways, airports, and designated airspaces. The total aircraft hazard probability at the repository equals the sum of the individual probabilities obtained from these models. An example aircraft crash hazard defined in the *MGR External Events Hazards Analysis* (CRWMS M&O 2000a) involves military aircraft flying through the R4808N restricted airspace (DMA 1995) over the NTS which includes the site of the repository surface facilities. These aircraft are at high altitudes in an enroute/inflight phase while inside the R4808N airspace. Although they are not flying in standard Federal Aviation Administration airways, they fly within specifically defined areas. The model provided in NUREG-0800 (NRC 1987) for airways can be used to approximate the crash frequency and determine if the event is credible.



#### **10.6.2.4.2 Uniform Overflight Density Model**

The Uniform Overflight Density Model (Kimura et al. 1998) is better suited for calculating crash frequencies when aircraft overfly a defined area along various routes as in the case of aircraft in transit the NTS between Nellis AFB and the Nevada Test and Training Range. The route taken by pilots when traversing the NTS depends on the particular training mission and location on the Nevada Test and Training Range where this training will be executed. A crash hit frequency analysis (Kimura et al. 1998) for the NTS was completed as part of a DOE facility safety analysis. A model was developed in that analysis to address U.S. Air Force overflights of the DOE R-4808N restricted area. The Uniform Overflight Density Model can be applied to the repository surface facilities to provide a comparative analysis to the NUREG Airways Model.

In this model, the aircraft crash frequency equals the product of the number of aircraft, which overfly a particular area, the probability that the aircraft crashes in that particular area, and the probability that the aircraft hits a facility in this particular area. The Uniform Overflight Density Model is developed in detail in *Crash Hit Frequency Analysis of Aircraft Overflights of the Nevada Test Site (NTS) and the Device Assembly Facility (DAF)* (Kimura et al. 1998), and the resultant equation for the special case of the NTS is shown in Equation 15 of Kimura et al. (1998).

#### **10.6.2.4.3 NUREG Airport Model**

NUREG-0800 (NRC 1987) provides a model for determining the crash frequency associated with aircraft takeoff and landings from civilian and military airports. However, the NRC has determined that the crash frequency is negligible beyond 10 miles from the end of a runway (NRC 2000b). There are no airports or airstrips within 10 miles of the potential repository surface facility. If this situation were to change, an analysis using the NUREG Airport Model would be performed.

#### **10.6.2.4.4 Designated Airspace Model**

A variation of the Uniform Overflight Density Model can be used to determine the crash frequency of events involving military aircraft while flying in the Nevada Test and Training Range that have not been screened out during the vicinity map development phase. This variation was developed by the Private Fuel Storage Limited Liability Company for their aircraft crash analysis of a nuclear fuel storage facility in Utah, also located near a U.S. Air Force test and training range (Private Fuel Storage 2000). The crash impact hazard for each altitude band for each range sector is shown in Private Fuel Storage (2000).

The variation from the Uniform Overflight Density Model involves developing the crash rate per year. The Private Fuel Storage model develops an annual crash rate per square mile and multiplies it to the site specific cutout area. The cutout area is bounded by the edge of a specific range and an arc, centered on the nuclear facility, with a radius equal to the maximum distance wherein loss of pilot control is possible.

### 10.6.2.5 Regulatory Requirements

NUREG-0800 (NRC 1987) defines the requirements for reviewing the adequacy of an applicant's aircraft hazard evaluation. Additionally, nuclear waste repository licensing requirements are defined in 10 CFR Part 63 wherein events with probabilities greater than 1 in 10,000 (based on an expected surface facility lifetime of 100 years) are considered credible events. This probability limit equates to an event frequency of  $1.0 \times 10^{-6}$  per year, which is used to determine if an aircraft crash event is credible and, if so, the consequences of the event must be evaluated. If the event is not credible and the NRC accepts this conclusion, no further analysis is required.

The results of the repository aircraft crash frequency analysis will be compared with an evaluation criterion that determines if a crash hit event is credible. A crash hit event is defined as an aircraft impacting a potential repository surface radiological control facility that has sufficient radionuclide inventory to exceed the proposed rule 10 CFR Part 63 dose limits if released. The event is not credible and needs no further analysis if it meets the following criteria:

The Crash Hit Frequency of an aircraft into a radiological control facility from aircraft shall be less than  $1 \times 10^{-6}$  per year. Category 2 event sequences are defined in 10 CFR Part 63 as other natural and human-induced events that have at least one chance in 10,000 of occurring before permanent closure of the repository. The performance requirement for retrieval is assumed to require 100 years from the time of initial SNF/high-level radioactive waste receipts until permanent closure of the repository.

#### 10.6.2.5.1 Example of Application of NUREG-0800 Proximity Criteria

NUREG-0800 (NRC 1987, Section 3.5.1.6.II) defines proximity criteria, which are applied to the various types of aircraft flying in the regional airspace surrounding the repository surface facility. These criteria, identified as requirements in the NUREG, are listed below. According to the NUREG, the probability is considered below the threshold for further evaluation if the distances from the plant (repository surface facilities) meet all of these requirements:

1. The plant-to-airport distance,  $D$ , is between 5 and 10 statute miles, and the projected annual number of operations is less than  $500 D^2$  or the plant-to-airport distance  $D$  is greater than 10 statute miles, and the projected annual number of operations is less than  $1,000 D^2$ .
2. The plant is at least 5 statute miles from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1,000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.
3. The plant is at least 2 statute miles beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

For an example of an aircraft analysis that was performed for the Yucca Mountain site (see *MGR Aircraft Crash Frequency Analysis*; CRWMS M&O 1999a).

#### **10.6.2.5.2 Defining the Vicinity for Crash Frequency Analysis**

A preliminary crash frequency analysis completed in late 1998 focused on military aircraft traversing the NTS en-route from the Nellis AFB to the Nevada Test and Training Range. After review of this analysis by the NRC, it was agreed that extended evaluations over a wider area were needed to determine if other aircraft-related activities could impact the frequency of a crash into the potential repository surface facilities. Military operations areas are located on the east and north sides of the Nellis Air Force Range.

The area extending south below Las Vegas, Nevada, north to Tonopah, Nevada, and east beyond Highway 93 and west to Death Valley, California should define the vicinity for detailed frequency analysis.

The approach to be used to establish this vicinity involves screening of non-credible events using both qualitative and quantitative methods. Prepare a description of the various areas and airfields within the area, the aircraft flying in those areas, and the activities being performed by these aircraft. This includes describing military test and training missions conducted in each sub-range and military operations area within the Nevada Test and Training Range and determining the distances from these areas to the proposed repository surface facilities. The description includes relevant characteristics of each aircraft and ordnance carried by these aircraft. These characteristics will include the aircraft glide ratios and the maximum range of missile ordnance.

The qualitative evaluation will compare the distance a malfunctioning aircraft or missile needs to travel to reach and impact the repository surface facility and the capability of such aircraft or missile to traverse this distance. The distance to be traveled depends on which part of the Nevada Test and Training Range the aircraft is located when the malfunction occurs. Because of the long distances involved, it is expected that many events will be screened out through this evaluation. The qualitative evaluation with regard to airports will also look at the distances involved and most will be screened out through this evaluation.

Quantitative evaluations may be conducted using the same crash frequency models to be used in the more detailed analysis but with best available information for the model parameters. Screening will be done on those events that are sufficiently below the frequency limit such that when all such events are combined, the cumulative frequency of these events will not impact the results of detailed analysis.

### **10.6.3 Military and Industrial Hazards**

#### **10.6.3.1 Scope**

Example impacts due to nearby installations and operations were evaluated in the *Preliminary MGR Hazards Analysis* (CRWMS M&O 1996). This determination was conservatively based on limited knowledge of the potential activities ongoing on or off the NTS. It is intended that the Industrial/Military Activity-Initiated Accident Screening Analysis methodology provided herein will meet the requirements of the *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987) in establishing whether this external event can be

screened from further consideration or must be included as an initiator in the development of event sequences for the repository.

### **10.6.3.2 Methodology**

The approach recommended is defined in NUREG-0800 (NRC 1987). NUREG-0800 (NRC 1987, Sections 2.2.1 and 2.2.2) address identification of potential hazards in the vicinity of a nuclear facility. The methodology involves identifying facilities within specified criteria, describing these facilities, describing the nature and extent of activities conducted, and providing statistical data about hazardous materials used at these facilities. The criteria in NUREG-0800 (NRC 1987, Section 3.1) include:

- Identified facilities and activities within eight kilometers (five miles) of the plant should be reviewed. Facilities and activities at greater distances should be considered if they otherwise have the potential for affecting plant important to safety features.
- Any facilities which meet the above criteria have to be evaluated in accordance with other sections of NUREG-0800 listed below, as appropriate:
  - Section 2.2.3, Evaluation of Potential Accidents
  - Section 3.5.1.5, Site Proximity Missiles (Except Aircraft)
  - Section 3.5.1.6, Aircraft Hazards

Hazards from an aircraft crash are covered in a separate section of this document while hazards from objects/ordnance falling from aircraft will be covered in this section.

Initiating events may be screened from further consideration if they have a frequency that is less than  $1.0 \times 10^{-6}$  per year (i.e., they are beyond Category 2 event sequence frequencies) or they have no impact on the repository due to the combination of the event magnitude (e.g., minimal overpressure and temperature) and distance from the repository. Types of events that are screened include: explosions, fires, chemical releases, and objects or ordnance falling from aircraft.

Specific evaluations of the overpressure from an explosion and the frequency of a radiological release from the WHB due to dropped objects or ordnance are to be performed to demonstrate that these events can be screened from further event sequence consideration based on either their inability to cause a radiological release or their low frequency of occurrence.

### **10.6.3.3 Application of NUREG-0800 Proximity Criteria**

NUREG-0800 (NRC 1987, Section 2.2.1.III.1) specifies that all identified facilities and activities within 8 kilometers (5 miles) of the plant shall be reviewed. When applying this criterion, surface and subsurface facilities should be considered. DTN: MO9907YMP97128.001 shows the relationship of the surface facilities and the current extent of the subsurface facility with a five-mile perimeter drawn around the both the surface and subsurface areas.

#### 10.6.3.4 Application of NUREG-0800 Plant-Affecting Criteria

NUREG-0800 (NRC 1987, Section 2.2.1.III.1) specifies that facilities and activities at distances greater than five miles should be considered if they otherwise have the potential for affecting plant important to safety features. The area surrounding the 5-mile perimeter includes the balance of the land withdrawal area, the balance of the NTS, Air Force land, and U.S. Bureau of Land Management (BLM) land. The land withdrawal area extends another 2 miles to the west and eight miles to the south. The NTS extends over 30 miles to the north and over 20 miles to the east of the land withdrawal area. The Air Force land is part of the Nellie Air Force Range which extends over 50 miles to the north of the withdrawal area. BLM land extends beyond the withdrawal area to the west and south and includes U.S. Highway 95 providing the major route between Las Vegas and Reno, Nevada. The potential for transportation accidents and Air Force dropped objects affecting the important to safety features should be covered.

Descriptions to be included in this study are the NTS facilities/activities and their potential to impact the repository; facilities/activities on BLM land, and facilities/activities on the Nellis Air Force Range.

#### 10.6.3.5 Parametric Evaluation of Potential Explosions

Some of the NTS and Nellis Air Force Range facilities handle high-explosive materials; in addition, events such as transportation and industrial accidents may result in explosions. The overpressure generated by an explosion (i.e., detonation) is a function of the amount of explosive material involved and the distance between the site of the explosion and the repository.

A methodology is given in Regulatory Guide 1.91, *Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants*, for evaluating the safe distance from a postulated explosion.

For example, setting the safe radius equal to the five-mile (26,400 ft) criterion (NRC 1987, Section 3.1) and using the methodology in Regulatory Guide 1.91, a value for the mass of explosive can be calculated, and that value compared with any of the explosive inventories currently associated with NTS facilities and any transportation or industrial explosive sources. Data from the Lake Denmark Explosion that occurred at the Picatinny Arsenal in 1926 (Kinney and Graham 1985, p. 13) should also be reviewed for determination of a safe distance from a large-scale explosion.

#### 10.6.3.6 Objects Dropped from Aircraft

Objects inadvertently dropped from aircraft can be screened from further event sequence consideration by demonstrating that the frequency of a release of radiological material is less than  $1.0 \times 10^{-6}$  per year. An event tree (see Section 7.1) can be used to evaluate the frequency of a dropped object causing a radiological release. Some events to consider in the construction of the event tree and sample numbers are listed below:

- **Frequency of Ordnance Drop (per sortie)**—Dropped objects are defined in SAIC (1991, p. 2-48) as items such as screws, bolts, and coverplates. The frequency at which

objects are dropped from military aircraft is given as 1.5 drops per 1,000 sorties (SAIC 1991, p. 2-48).

- **Number of Sorties that Overfly the NTS**—The number of sorties that overfly the NTS is 18,910 per year. This is the 95 percent confidence value as calculated in *MGR Aircraft Crash Frequency Analysis* (CRWMS M&O 1999a, p. V-2).
- **Fraction of Sorties that Fly in the Vicinity of the WHB**—It is estimated that no more than 2 percent ( $2.0 \times 10^{-2}$ ) of the total sorties that overfly the NTS fly within a 6-mile by 6-mile box centered on the WHB.
- **Probability of an Object being Dropped in the Vicinity of the WHB**—If it is assumed that drop frequency is uniform with respect to the flight path of an aircraft, then the conditional probability that a dropped object falls while an aircraft is within the 6-mile by 6-mile box is the ratio of the flight path length within the box to the total flight path length.
- **Probability of an Object Dropped in the Vicinity of the WHB Striking the WHB**—The conditional probability of an object dropped within the 6-mile by 6-mile box centered on the WHB actually striking the building is equal to the ratio of the WHB footprint ( $0.01 \text{ mi}^2$ ) to the footprint of the 6-mile by 6-mile box.

$$(e) = \frac{0.01 \text{ mi}^2}{(6 \text{ miles}) \times (6 \text{ miles})} = 2.78 \times 10^{-4} \quad (\text{Eq. 10-10})$$

- **Probability of an Object that Hits WHB Striking Nuclear Material**—To cause a radiological release, a dropped object that hits the WHB must strike nuclear material; of which the conditional probability is equal to the ratio the available strike area of nuclear material to the area of the WHB footprint.

A calculation that can be reviewed for examples of the treatment of objects dropped from aircraft is *Industrial/Military Activity-Initiated Accident Screening Analysis* (CRWMS M&O 1999b).

#### 10.6.3.7 Ordnance Dropped from Aircraft

Ordnance inadvertently dropped from aircraft can be screened from further event sequence consideration by demonstrating that the frequency of a release of radiological material is less than  $1.0 \times 10^{-6}$  per year. An event tree (see Section 7.1) can be used to evaluate the frequency of dropped ordnance causing a radiological release. Some events to consider in the construction of the event tree and example numbers are listed below:

- **Frequency of Ordnance Drop (per sortie)**—The frequency at which armaments are dropped from military aircraft is given as 0.005 ( $5.0 \times 10^{-3}$ ) drops per 1,000 sorties (SAIC 1991, p. 2-48).

- **Number of Sorties that Overfly the NTS**—The number of sorties that overfly the NTS is 18,910 per year. This is the 95 percent confidence value as calculated in CRWMS M&O (1999a, p. V-2).
- **Fraction of Sorties that Fly in the Vicinity of the WHB**—It is estimated that no more than 2 percent ( $2.0 \times 10^{-2}$ ) of the total sorties that overfly the NTS fly within a 6-mile by 6-mile box centered on the WHB.
- **Probability of Ordnance being Dropped in the Vicinity of the WHB**—For dropped ordnance to affect the WHB (and potentially cause a release of nuclear material) it must fall off of the aircraft as its flight path passes near the WHB. If it is assumed that drop frequency is uniform with respect to the flight path of an aircraft, then the conditional probability that dropped ordnance falls while an aircraft is within the 6-mile by 6-mile box is the ratio of the flight path length within the box to the total flight path length.
- **Fraction of Sorties that Fly in the Vicinity of the WHB with Live Ordnance**—Only 10 percent ( $1.0 \times 10^{-1}$ ) of the sorties flown in the vicinity of the WHB carry live ordnance. Based on “Nellis Airspace and Crash Data for Yucca Mountain” (Tullman 1997), most aircraft flying through the western NTS (i.e., near the repository) are armed with simulated ordnance.
- **Fraction of Sorties that Fly in the Vicinity of the WHB with armed ordnance that could be dropped**—Restrictions imposed by the Air Force on NTS overflights forbid overflight of the NTS with armed live ordnance, unless the ordnance is carried internally and bomb bay doors are confirmed closed (i.e., in a configuration where a drop cannot occur) (Irving 1997). Based on these restrictions and engineering judgement the conditional probability of an inadvertent drop of armed live ordnance is 0.01 ( $1.0 \times 10^{-2}$ ).
- **Fraction of Dropped Live Ordnance that Explodes Upon Impact**—It is conservatively assumed that any ordnance that is live (i.e., not simulated) and armed will explode upon impact.
- **Probability that Dropped Ordnance Effects the WHB**—The conditional probability of dropped ordnance affecting nuclear material inside the WHB is dependent upon whether or not the ordnance explodes on impact, the probability of which depends on whether the ordnance is live (as opposed to simulated), and whether the ordnance, if live, is armed.

A calculation that can be reviewed for examples of the treatment of ordnance dropped from aircraft is *Industrial/Military Activity-Initiated Accident Screening Analysis* (CRWMS M&O 1999b).

#### 10.6.3.8 Transportation

This section presents a general discussion for assessing transportation hazards. The following paragraphs are considered as an example.

U.S. Highway 95 and roads on the NTS are used to haul large quantities of explosives, munitions, propellants, hazardous materials, and radioactive materials. At its closest point, U.S. Highway 95 is approximately 13 miles from the repository surface facilities (DMA 1995).

Most transportation of hazardous materials on the NTS occurs on roads located at least 15 miles from the repository surface facilities (DMA 1995). The Lathrop Wells Road, which traverses the southeastern area of the proposed repository withdrawal area, is used to support testing in the X and Y tunnels. Assuming that materials are transported onto the NTS via the Lathrop Wells Road for this testing, the transport vehicles will be approximately 10 miles from the repository surface facilities (DMA 1995).

There are no transportation railroad lines within 20 miles of the repository surface facilities (DMA 1995).

These transportation routes are expected to be sufficiently distant from the repository to preclude adverse effects of transportation accidents resulting from explosions. Specific evaluations of the overpressure from an explosion on a nearby transportation route are to be performed to demonstrate that these events can be screened from further event sequence consideration. These distances are also sufficiently distant from the repository to preclude adverse effects of fires associated with transportation accidents. In the case of toxic releases, the NRC regulatory position for evaluating the habitability of a NPP control room can be found in Regulatory Guide 1.78.

#### **10.6.3.9 Evaluation of Application of NUREG-0800, Plant-Affecting Criteria, to Independent Spent Fuel Storage Installations and Nuclear Power Plants**

The Safety Analysis Report (Chapter 2.2, Nearby Industrial, Transportation, and Military Facilities) of other NRC-licensed facilities should be reviewed to see if there are any cases where detailed analyses were performed to assess facilities outside the NUREG-0800 (NRC 1987) five-mile evaluation limit. A couple of examples to consider include:

- Idaho National Engineering and Environmental Laboratory TMI-2 Independent Spent Fuel Storage Installation (INEL 1996)
- Rancho Seco Independent Spent Fuel Storage Installation (Shelter 1993).

#### **10.6.3.10 Consequence Analysis**

For each applicable credible, either individually or in combination with other events in an event sequence, perform: (1) a frequency analysis that demonstrates the event is not credible (2) a nuclear safety analysis that demonstrates a radiological release does not occur as a result of the event (3) a consequence analysis that demonstrates that the radiological consequences of the event are within regulatory requirements or that identifies required preventative or mitigative SSCs that ensure the radiological consequences are within regulatory requirements.

For each of the credible events identified, dose assessments will be performed to show compliance to requirements as applicable.



The frequency analysis of an event determines if the event is credible. If not credible, no quantitative dose limits are promulgated by 10 CFR Part 63, no further analysis is required, and there is no impact to other repository design or licensing organizations. Event sequences that are beyond Category 2 will be tracked to show safety design margin. If the event is determined to be credible, it is categorized based on the 10 CFR Part 63 definition, and an analysis is performed to determine if the dose limits associated with the applicable event category can be met.

Category 1 event sequences (i.e., frequency greater than  $1 \times 10^{-2}$  events per year) require that the sum of annual doses, exposures, and releases do not exceed limits specified in 10 CFR Part 63 for the public and 10 CFR Part 20 for occupational workers. Category 2 event sequences (i.e., frequency less than  $1 \times 10^{-2}$  and greater than  $1 \times 10^{-6}$  events per year) require that the consequences of a specific Category 2 event sequence not exceed dose limits as specified in 10 CFR Part 63 for the public beyond the preclosure controlled area.

The consequence analysis determines if the calculated doses are within applicable limits. If the calculated dose exceeds applicable limits, SSCs important to radiological safety are designated, new requirements are allocated to the system, assumptions are revised, the design configuration is revised (if necessary), and the dose is recalculated and again compared to applicable dose limits.

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### **10.7.3 Source Data**

MO9907YMP97128.001. Land Use and Ownership within a 25-Mile Radius of Yucca Mountain. Submittal date: 07/07/1999.

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## ACRONYMS

BWR	boiling water reactor
DC	disposal container
MCNP	<i>MCNP-A General Monte Carlo N-Particle Transport Code</i>
PWR	pressurized water reactor
SNF	spent nuclear fuel
SSC	systems, structures, and components

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## 11. CRITICALITY

### 11.1 INTRODUCTION

As part of preclosure safety analysis to support the license application submittal, a preclosure criticality risk analysis is performed to calculate the risks associated with criticality hazards and to prevent and control criticality during surface and subsurface spent nuclear fuel (SNF) handling and high-level radioactive waste operations at the repository. Criticality hazards are hazards associated with handling fissionable materials contained in SNF or other waste forms. The hazards are identified in Section 6.2. The results of the hazard analyses, which identify the systems, structures, components (SSCs), and processes that have the potential for criticality events, serve as input to the criticality risk analysis.

### 11.2 REGULATORY REQUIREMENTS

Regulatory requirements relating to criticality safety are described in 10 CFR 63.112(e)(6), which requires the preclosure safety analysis to include means to prevent and control criticality. The regulatory requirements applicable to operations involving SNF assemblies or canisters in transportation casks are derived from 10 CFR Part 71, which will not be addressed in Section 11. It is assumed that the transportation casks have been designed in accordance with applicable Certificates of Compliance and have met the criticality safety requirements in 10 CFR Part 71.

### 11.3 REPOSITORY PRECLOSURE CRITICALITY SAFETY STRATEGY

The repository preclosure criticality safety strategy is described in *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001). The strategy to prevent criticality is to rely, where practicable, on equipment design that uses passive-engineered controls (such as geometry control or fixed neutron absorbers) rather than on administrative controls. Where passive engineering controls alone are not practical or sufficient, administrative controls on fissionable material mass or other reliable and verifiable reactivity control methods, such as minimum burnup requirements on commercial SNF, will be established.

The repository preclosure criticality safety strategy also relies on a defense-in-depth approach. The defense-in-depth approach involves taking advantage of the natural site and engineering design features. These features are expected to reduce the probability of a preclosure criticality event to below the regulatory thresholds established in 10 CFR Part 63. The natural site and engineering design features accounted for in the preclosure criticality safety strategy will be identified and credited in the criticality risk analysis.

### 11.4 CRITICALITY RISK ANALYSIS PROCESS

Criticality is attained when the effective neutron multiplication factor,  $k_{\text{eff}}$ , of a system of fissionable material in a given geometry becomes equal to or greater than unity. Conversely, subcriticality is defined by a  $k_{\text{eff}}$  less than unity. When designing a system (e.g., a waste package) to be subcritical, it must be demonstrated that the calculated  $k_{\text{eff}}$  conservatively represents the true neutron multiplication of the system. This is accomplished by choosing a

value below unity where nuclear criticality is assumed to occur. This value is known as the upper subcritical limit.

The process for evaluating the risk of the potential criticality events associated with the handling waste forms containing fissionable materials is shown in a flow chart (Figure 11-1). The flow chart is composed of the following steps:

1. Examine the results from the hazard analyses.
2. Identify the SSCs and processes associated with criticality hazards.
3. Identify the types of waste forms that could achieve criticality.
4. Develop a criticality event tree.
5. Quantify the criticality event tree.
6. If an event sequence frequency is less than  $1.0 \times 10^{-6}$  per year, the event sequence is screened from additional analysis per the requirements of 10 CFR Part 63.
7. If an event sequence frequency is equal to, or greater than,  $1.0 \times 10^{-6}$  per year, perform criticality analysis to determine  $k_{\text{eff}}$ , where  $k_{\text{eff}}$  is the effective neutron multiplication factor.
8. If the calculated  $k_{\text{eff}}$  is less than the upper subcritical limit, the event sequence has no potential for criticality and additional analysis is not required.
9. If the calculated  $k_{\text{eff}}$  is equal to, or greater than, the upper subcritical limit, initiate an appropriate redesign of the SSCs and process, make the appropriate modifications to the criticality event tree, and then repeat steps 5 through 9 until the calculated  $k_{\text{eff}}$  is below the upper subcritical limit.
10. If practical modifications to the SSCs and processes have been evaluated, the event sequence frequency still exceeds  $1.0 \times 10^{-6}$  per year, and the calculated  $k_{\text{eff}}$  still equals or exceeds the upper subcritical limit, a consequence analysis of the event sequence will be performed to calculate the doses to members of the public and workers. The calculated doses will be compared with the regulatory dose limits (Table 8-1). It will be shown that the calculated doses are below the regulatory dose limits.

In steps 8, 9, and 10, the upper subcritical limit is defined as:

$$\text{Upper Subcritical Limit} = 1 - \text{bias} - \text{bias uncertainty} - \text{administrative margin}$$

The upper subcritical limit should not be confused with the critical limit that is defined as one minus the bias and the bias uncertainty. The bias is determined by comparison of criticality calculation results using a code such as *MCNP-A General Monte Carlo N-Particle Transport*



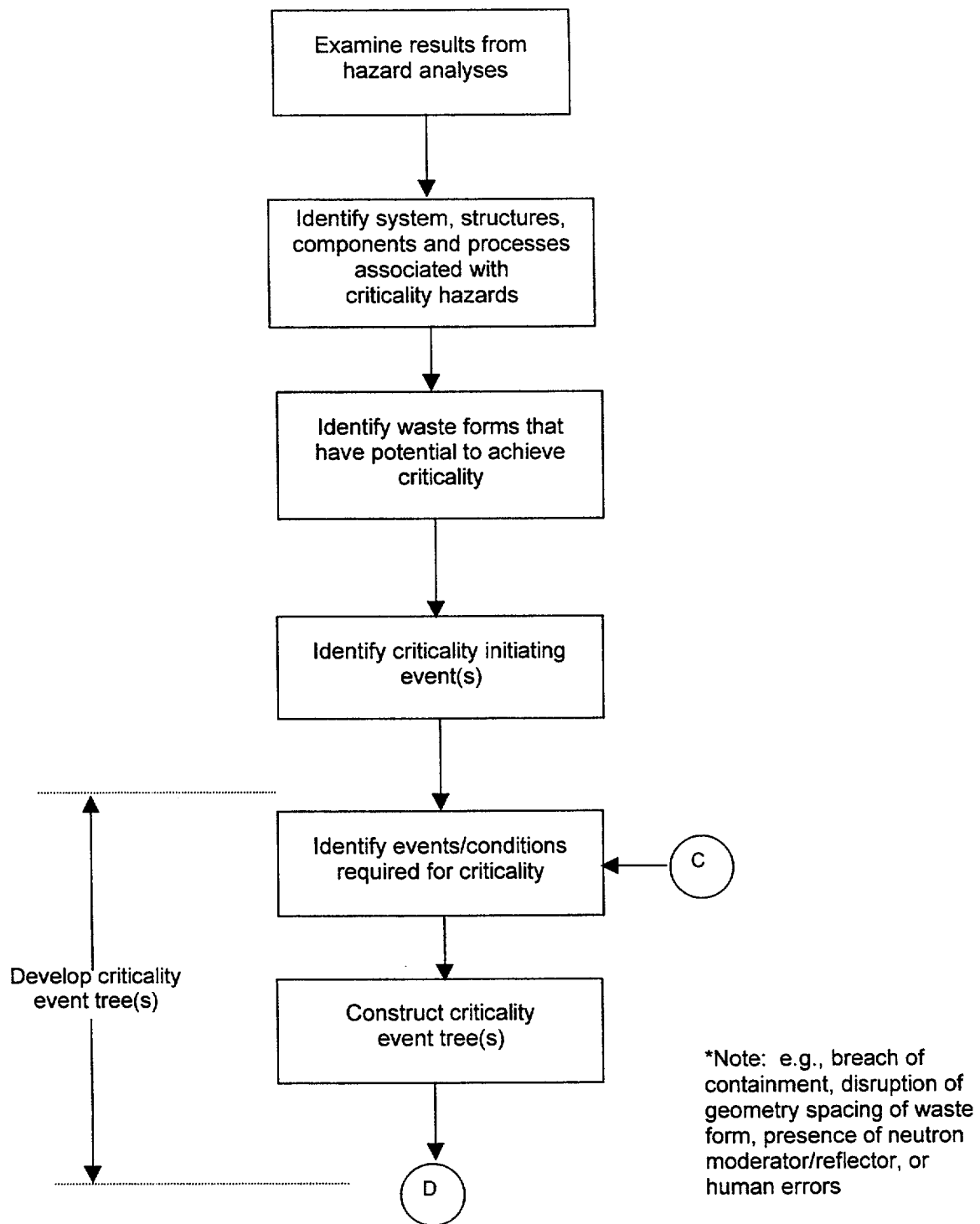


Figure 11-1. Criticality Risk Analysis Flow Chart

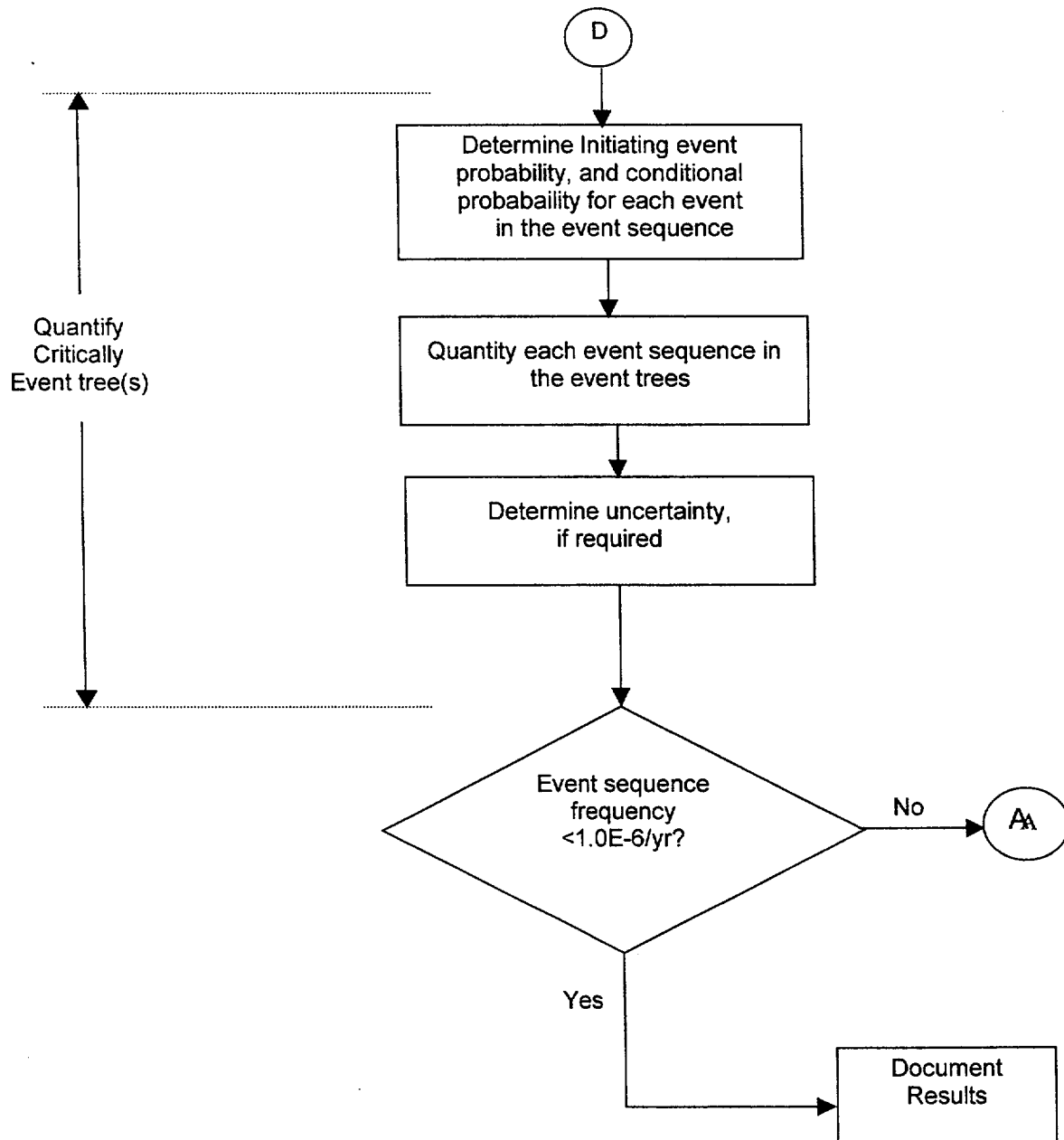


Figure 11-1. Criticality Risk Analysis Flow Chart (Continued)

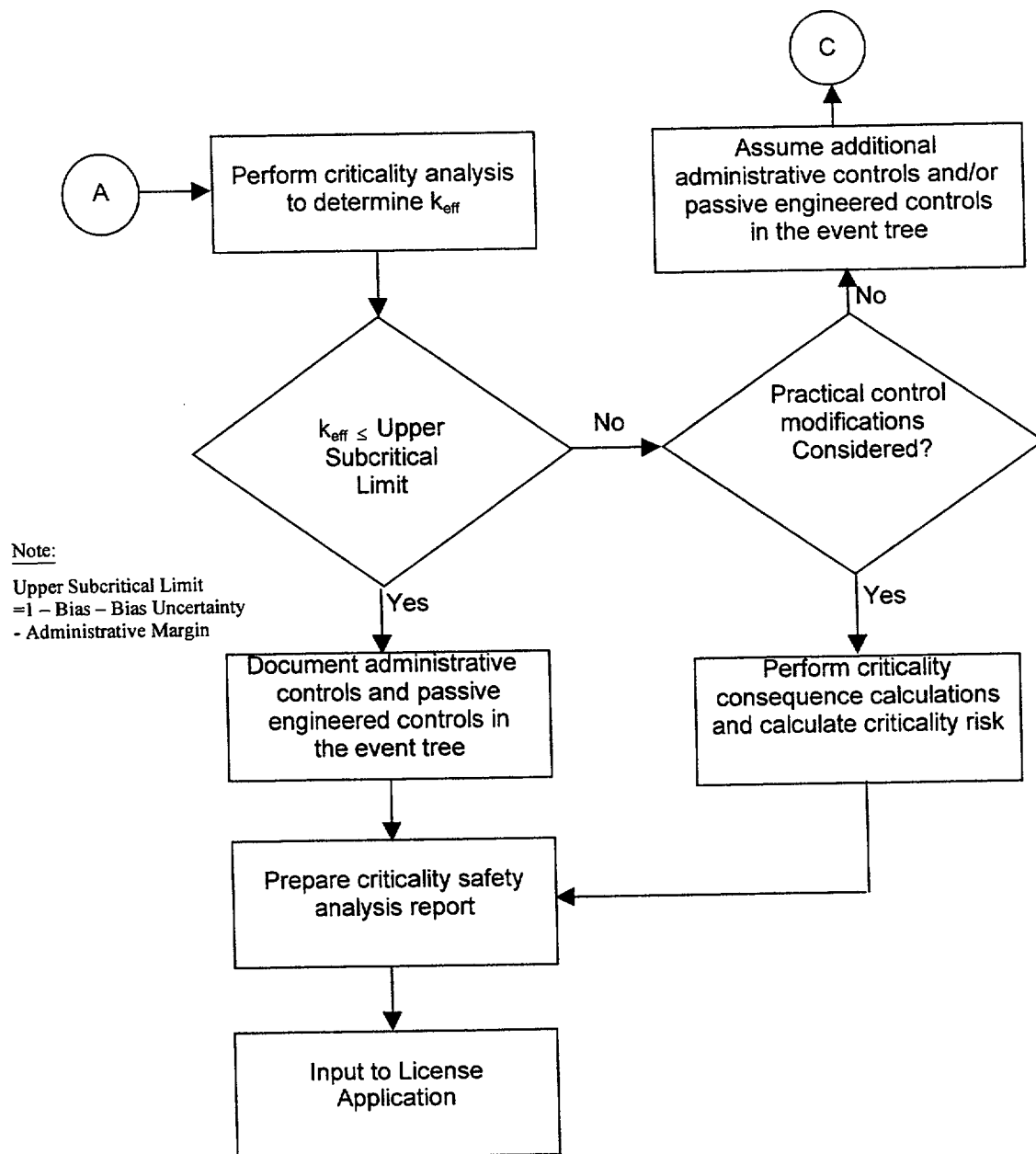


Figure 11-1. Criticality Risk Analysis Flow Chart (Continued)

*Code* (MCNP; Briesmeister 1997) to critical benchmarks. The administrative margin is an additional arbitrary margin applied to ensure subcriticality.

#### **11.4.1 Identify the Systems, Structures, Components, and Processes Associated with Criticality Hazards**

The results of hazard analyses will be reviewed to identify the SSCs and processes associated with the generic criticality hazards.

As an example, the following repository functional areas and processes identified from the site recommendation design would, at a minimum, be considered for criticality risk analysis:

- Waste Receipt and Carrier/Cask Transport
- Waste Handling Building - Carrier Preparation Area
- Waste Handling Building - Carrier Bay
- Waste Handling Building - Transfer Cell (including any waste form staging areas)
- Waste Handling Building - Disposal Container (DC) Handling Cell
- Waste Handling Building - Waste Package Remediation Cell
- Waste Handling Building - Waste Package Transporter Load Cell
- Surface storage facility (vault) (for the storage of steel containers that contain pressurized water reactor [PWR] and boiling water reactor [BWR] SNF assemblies), if a cold repository design is selected.
- Subsurface transport, emplacement, and monitoring.

#### **11.4.2 Identify Those Waste Forms That Have the Potential to Achieve Criticality**

Key factors in determining the potential for waste form criticality are the configuration and the quantity and type of fissionable material it contains. Only a fraction of the waste forms in the currently-identified inventory to be received at the repository have the potential to achieve criticality. These waste forms have medium or high reactivity with the characteristics of a high initial enrichment or low burnup. These include some PWR and BWR fuels and canisters with U.S. Department of Energy SNFs (e.g., Fast Flux Test Facility [i.e., FFTF], Enrico Fermi, Training Research Isotope General Atomic [i.e., TRIGA], Shippingport PWR and Light Water Breeder Reactor [i.e., LWBR], Ft. St. Vrain and N-Reactor SNF). Other waste forms such as high-level radioactive waste glass, contain insufficient fissionable material and, therefore, have no potential for criticality. The fraction of each of these waste forms in annual throughput should be determined and included in the criticality event sequence frequency calculation.

### 11.4.3 Develop Criticality Event Trees

Three potential preclosure criticality events were identified in *Preclosure Design Basis Events Related to Waste Packages* (CRWMS M&O 2000, p. 60). The events identified in this analysis were:

1. Alteration of geometry events
2. Introduction of neutron moderator or reflector events
3. Waste form misload events

These potential events are briefly discussed in the following sections.

#### 11.4.3.1 Event Trees for Alteration of Geometry Events

Initiating events resulting in the drop, tipover and slapdown, or collision of the waste forms or waste form containers could occur during various repository processes. For an event sequence to have the potential for causing a criticality in a waste form container or a waste form handling area, a number of events and conditions must occur. The events and conditions that must occur include the following:

1. The waste form container (transportation cask, canister, or waste package) must be breached.
2. The geometric spacing or configuration of the waste form and the neutron absorber materials of the waste form, waste form container, or waste form handling area must be disrupted or changed. The configuration of a waste form may change after a drop or tip-over/slap-down (e.g., fuel scrap could fall out of a scrap basket in a multi-canister overpack [i.e., MCO] after a drop and slap-down).
3. A significant source of neutron moderator or reflector material must be present at the location where the initiating event occurs. In addition, the neutron moderator and reflector material must be able to remain in close proximity to the waste form.
4. The waste form must have the potential to achieve criticality.

#### 11.4.3.2 Event Trees for Introduction of Neutron Moderator and Reflector Material Events

Initiating events resulting in the introduction of large quantities of a neutron moderator or reflector material (e.g., hydrogenous materials such as water) could occur in the repository surface facility. The current surface facility design philosophy is to preclude any water source in the waste form and DC handling areas. However, the potential for flooding events, such as a pipe break or an inadvertent actuation of the fire suppression system, will be assessed. The conditional probability of internal flooding could be determined using the fault tree methodology described in Section 7.3.

### 11.4.3.3 Event Trees for Misload Events

Methods for loading waste forms into a DC have been developed such that subcriticality is ensured during the preclosure period. This method may be referred to as the Criticality Loading Curve Evaluation. This evaluation established a simple criterion, in terms of the available waste form information, to determine if a waste form can be loaded into a given DC. The available waste form information used in the determination for commercial SNF consists of four components: bundle identification number, initial enrichment, assembly average discharge exposure, and decay time. With this information, a curve is developed for each commercial SNF (PWR, BWR, and mixed oxide) that represents the required minimum-average assembly burnup as a function of the initial-average assembly enrichment allowable for loading into a particular type of DC. For assemblies that fall below this curve, additional means of reactivity control are utilized. Such reactivity control increases the margin to criticality, either by addition of disposable control rods within the assembly or loading of these assemblies into an alternate DC, which is smaller and possesses a higher negative reactivity component inherent in its design. Adhering to this process ensures that a subcritical configuration exists. However, human errors during loading of the waste form into a DC could potentially lead to a criticality event.

Initiating events resulting in the waste form misload of a waste package or waste form storage area could occur. During the loading of waste forms into a DC, selection and conceptual human errors could occur resulting in an assembly misload event. A selection error simply represents an unintentional selection of the wrong item while trying to select the correct one. The conceptual error represents intentionally selecting the wrong item based on the erroneous belief that it is the correct item. The following paragraphs provide a brief list of the potential human errors that could result in a waste form misload.

- Based on the characterization of the waste form removed from transportation casks, the operator decides what type of DC it is to be loaded into. Deciding on an inappropriate DC type or selecting a wrong DC type is a human error.
- Some fuel assemblies will require the insertion of a neutron absorber rod assembly for permanent disposal. It is expected that these fuel assemblies will be shipped with a neutron absorber rod assembly already in place. For those assemblies that require the insertion of a neutron absorber rod assembly, but are not shipped with one in place, the following two human error scenarios are possible: the operator fails to identify the assembly as requiring the insertion of the neutron absorber assembly, or having identified the assembly as requiring the insertion of the neutron absorber assembly, the operator fails to insert it.
- An additional human error scenario is possible for those fuel assemblies that will require the insertion of a neutron absorber rod assembly for permanent disposal. It is expected that these fuel assemblies will be shipped with an inadequate neutron absorber rod assembly. The operator could fail to recognize that the assembly has an inadequate neutron absorber assembly and, therefore, fail to replace it with an acceptable assembly.
- The selection of the waste form to be placed in the DC is another opportunity for human error. The operator could select an incorrect waste form, or after selecting the correct

waste form for the DC, make a manipulation error with the crane and transfer the wrong one. Only the misload of medium or high reactivity waste forms into a DC would have the potential for criticality.

- After placing the waste forms into the DC, the operator performs a physical verification to ensure that correct waste forms were loaded. Failure to detect a misloaded SNF assembly in the DC during the physical verification process is another possible human error.

Human error probability can be reduced through the use of an independent checker during the physical verification task. Additionally, passive engineering controls have been utilized in the design of the DC internals such that some waste form types will not fit into a DC for which it was not intended. The analysis of human reliability is discussed in Section 7.4.

#### 11.4.4 Quantify the Criticality Event Trees

The event sequence frequency analysis will be performed using the methods in Section 7. Quantification of the initiating event frequency, probabilities of branches, and the event sequence frequency should be performed in accordance with Section 7.1.4.

#### 11.4.5 Perform Criticality Analysis to Determine $k_{\text{eff}}$

The criticality analysis methods for criticality analysis for the repository are detailed for the preclosure period in the *Preclosure Criticality Analysis Process Report* (CRWMS M&O 1999). This report provides the design requirements; applicable regulations, codes, and standards; summary descriptions of the types of computational tools to be used; and the types of analyses to be performed.

Criticality analyses will be performed when an event sequence frequency is determined to equal or exceed  $1.0 \times 10^{-6}$  per year. These criticality analyses will be performed to determine the sequence configuration  $k_{\text{eff}}$  using criticality analysis tools such as the MCNP code (Briesmeister 1997). MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron transport, and has the capability of calculating  $k_{\text{eff}}$  for generalized systems containing fissionable material. MCNP uses the isotopic compositions of the materials, a detailed representation of the geometry, and a set of nuclear information libraries to calculate  $k_{\text{eff}}$  for the system. The nuclear information libraries used by MCNP are comprised of continuous-energy cross sections of materials. A full set of these material cross sections has been evaluated and is provided through the MCNP code package.

If the calculated  $k_{\text{eff}}$  is below the upper subcritical limit, the system is considered to be subcritical and the event sequence is screened out.

#### 11.4.6 Perform Criticality Consequence Analyses

If the criticality event sequence evaluation results in a frequency equal to or exceeding the regulatory threshold of  $1.0 \times 10^{-6}$  per year, and practical steps have been taken to reduce the calculated  $k_{\text{eff}}$  of sequence configuration to below the upper subcritical limit, then it is necessary to determine the consequences of the criticality event. The doses to members of the public and

workers will be calculated using the methods presented in Section 8. The frequency calculated for the event sequence will determine whether the event sequence is a Category 1 or Category 2 event sequence. The calculated doses will be compared with the regulatory dose limits (Table 8-1). It will be shown that the calculated doses are below the regulatory dose limits.

#### **11.4.7 Document the Results**

A criticality safety analysis will be prepared to document the results of the criticality risk analysis. The documentation of the results of the criticality event tree evaluations should include the frequency of each event sequence identified. If administrative controls and natural site and engineered design features are relied on to reduce the criticality event sequence frequency to below the regulatory threshold of  $1.0 \times 10^{-6}$  per year, these controls and features should be documented through the SSCs system description documents.

### **11.5 REFERENCES**

#### **11.5.1 Documents Cited**

Briesmeister, J.F., ed. 1997. *MCNP-A General Monte Carlo N-Particle Transport Code*. LA-12625-M, Version 4B. Los Alamos, New Mexico: Los Alamos National Laboratory. ACC: MOL.19980624.0328.

BSC (Bechtel SAIC Company) 2001. *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation*. TDR-MGR-SE-000009 REV 00 ICN 03. Las Vegas, Nevada: Bechtel SAIC Company. ACC: MOL.20010705.0172.

CRWMS M&O 1999. *Preclosure Criticality Analysis Process Report*. B00000000-01717-5705-00132 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990930.0102.

CRWMS M&O 2000. *Preclosure Design Basis Events Related to Waste Packages*. ANL-MGR-MD-000012 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000725.0015.

#### **11.5.2 Codes, Standards, Regulations, and Procedures**

10 CFR 63. Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada. Readily available.

10 CFR 71. Energy: Packaging and Transportation of Radioactive Material. Readily available.



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## ACRONYMS

CQ	conventional quality
HLW	high-level radioactive waste
NRC	U.S. Nuclear Regulatory Commission
PRA	probabilistic risk assessment
PSA	preclosure safety analysis
QA	quality assurance
QL	quality level
SSCs	structures, systems, and components
TEDE	total effective dose equivalent

## 12. QUALITY ASSURANCE CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY

### 12.1 INTRODUCTION

This section provides an overview of the risk-informed quality assurance (QA) classification process for structures, systems, and components (SSCs) of a high-level radioactive waste (HLW) repository that are important to safety as required by 10 CFR 63.112(e). The screening criteria are based on the risk-informed, performance-based philosophy embodied in the regulation. SSCs are assigned one of four Quality Level (QL) classifications based on the risk significance of the SSC for the purpose of applying the QA program criteria of 10 CFR 63.142. A discussion is provided on how the QA classification process fits into the design process and the overall development of a preclosure safety analysis (PSA). Background information related to the evolution of the QLs and the associated criteria is provided, as well as a discussion of the application of the risk-informed screening criteria through a functional failure analysis.

The QA classification process is applied to the SSCs of a HLW repository that are important to preclosure safety and important to postclosure waste isolation. This section addresses the preclosure portion of the QA classification process; the identification of features important to waste isolation is not included.

QL classifications (QL-1, QL-2, QL-3, or Conventional Quality [CQ]) are assigned to SSCs that represent the relative importance of an SSC to the health and safety of the public or to the radiological safety of workers. The QA classifications will be used to develop a graded application of the QA program requirements. The QA classifications are derived in a risk-informed, performance-based framework. The QA classification process will support a license application for the construction authorization for a HLW repository.

The governing regulation for a HLW repository is 10 CFR Part 63, which states, in part:

[The U.S. Department of Energy] DOE is required by 63.21(c)(20) to include in its safety analysis report a description of the quality assurance program to be applied to all structures, systems, and components important to safety, to design and characterization of the barriers important to waste isolation, and to related activities (63.142(a))

The quality assurance program must control activities affecting the quality of the identified structures, systems, and components, to an extent consistent with their importance to safety. (63.142(c)(1))

The phrase "to an extent consistent with their importance to safety" recognizes that the QA program for a HLW repository should impose graded quality requirements that would be applied to SSCs commensurate with their relative importance to radiological safety.

Importance to safety is defined in 10 CFR 63.2 as:

Important to safety, with reference to structures, systems, and components, means those engineered features of the geologic repository operations area whose function is:

- (1) To provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of § 63.111(b)(1) for Category 1 event sequences; or
- (2) To prevent or mitigate Category 2 event sequences that could result in radiological exposures exceeding the values specified at § 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site.

The U.S. Nuclear Regulatory Commission (NRC) has published guidelines for applying risk-informed decision making to graded QA program controls for nuclear power plants. The NRC has presented the definition and principles of risk-informed decision making in nuclear power plant regulatory guidance using the results of a probabilistic risk assessment (PRA).

In NUREG-0800 (NRC 1987, Section 19.0), it is stated that "...the decision making process will use the results of the risk analysis in a manner that complements traditional engineering approaches, supports the defense-in-depth philosophy, and preserves safety margins. Thus, risk analysis will inform, but it will not determine regulatory decisions" (NRC 1987). The NRC applies the principles for risk-informed decision making to graded QA program controls for nuclear power plants in Regulatory Guide 1.176. Although the risk measures for nuclear power plants (e.g., core-damage frequency) are not relevant to a HLW repository, the principles provide general guidance for such a repository.

The NRC approach establishes a risk-informed framework for defining a multiple-level QA classification process for SSCs important to safety. The multiple-level QA classification will set the stage for implementation of graded QA program controls. For a HLW repository, the items important to safety are identified and classified using the PSA that employs elements of PRA supplemented by applicable regulatory and industry precedents. The SSCs that are subject to 10 CFR 63.142 QA requirements will be classified as QL-1, QL-2, or QL-3, and SSCs that are not subject to 10 CFR 63.142 QA requirements will be classified as CQ.

## **12.2 SUMMARY OF THE PRECLOSURE SAFETY ANALYSIS PROCESS**

Because 10 CFR Part 63 has been developed as a risk-informed, performance-based rule, the Yucca Mountain Site Characterization Project is adopting a risk-informed, performance-based approach for developing the PSA. Figure 12-1 illustrates the overall process for developing a PSA and defining the event sequences for a HLW repository. While parts of the safety strategy for a HLW repository will be based on deterministic principles and industry and regulatory precedents, much of the safety evaluation of the preclosure operations will apply some of the techniques used in PRA. Section 4 presents a more comprehensive overview, and details of PSA analyses are presented in Sections 7-11.

The development of the PSA is an iterative process that evolves as the repository design develops, as site characteristics are more fully defined, and as operational features are identified. As this evolution progresses (and consistent with the current state of development) potential internal and external hazards are identified, event sequences are developed, frequency assessments are performed, event sequences are categorized, and potential resultant consequences are evaluated. Additional information regarding these steps is described in Table 12-1. If the consequences of an event sequence do not meet the regulatory requirements, preventative or mitigative design features and administrative controls are implemented until the event sequence risk is reduced to meet the performance objectives. Sensitivity and uncertainty analyses are performed, as required, to demonstrate that the categorization of the sequence and potential consequences meet the preclosure performance objectives. The SSCs important to safety are identified through the use of these sensitivity and uncertainty analyses and the use of the QA classification process. The identification of SSCs important to safety is documented in classification analyses and is subject to independent checking and an interdisciplinary review.

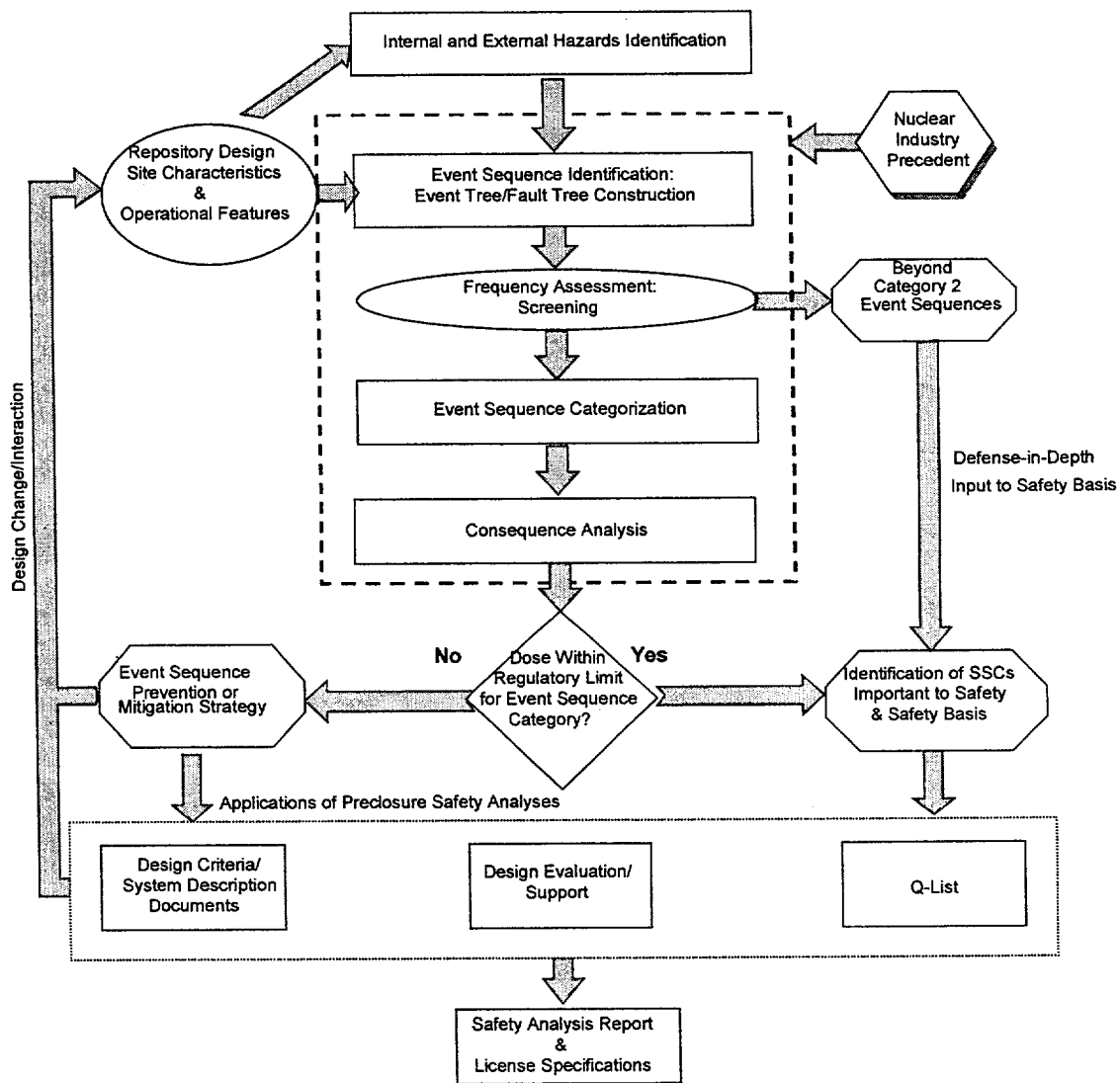


Figure 12-1. Preclosure Safety Analysis

Table 12-1. Development of a Preclosure Safety Analysis

<b>Internal and External Hazards Identification</b>	Hazards analysis is a systematic identification and evaluation of naturally occurring and human-induced hazards. To ensure completeness, the analysis begins with checklists of generic categories of hazards that have been developed for safety and risk analyses of nuclear power plants, spent fuel storage facilities, fuel cycle facilities, and spent fuel transportation. The generic hazards are screened to determine which hazards, either internal or external to the repository facilities, are applicable. The purpose of the hazards analysis is to identify energy-source categories that can potentially interact with a waste form (e.g., collision or crushing, chemical contamination or flooding, explosion or implosion, fire, radiation, thermal, natural phenomena, and potential criticality). Initially, qualitative evaluations are applied to screen out inapplicable or incredible* hazards. Potentially credible** external hazards that are not eliminated in the initial qualitative screening are subjected to quantitative analyses to screen them out, if possible. Otherwise, they are incorporated into the design bases to prevent initiation of a radiological release that would exceed preclosure performance objectives (e.g., earthquakes, winds, tornadoes, and loss of offsite power). Potentially credible internal hazards that are not eliminated in the qualitative screening are evaluated further in accident sequence analyses.
<b>Event Sequence Identification</b>	Potential accident scenarios (or event sequences) may be displayed in the form of event trees that include an initiating event (from an identified hazard) and one or more enabling events that must occur to result in a release of radioactivity, a criticality, or an abnormal exposure of a worker. The event tree format provides a framework for estimating the event sequence frequency by displaying the frequency of the initiating event and the conditional probabilities of contributing (enabling) events. Where necessary and appropriate, fault-tree analysis is used to estimate the frequency of an initiating event or probability of an enabling event. Potential criticality event sequences are subjected to specialized analyses to demonstrate that sufficient design and operational controls will be in place to ensure that the frequency of an accidental criticality will be below $1 \times 10^{-6}$ events per year (for a 100-year preclosure period).
<b>Frequency Assessment Screening and Event Sequence Categorization</b>	<p>The frequency (or annual probability of occurrence) is estimated for each event sequence. The frequencies of initiating events for internal hazards are estimated from the annual frequency of each operation multiplied by the conditional probability of the initiating event per operation. For example, the frequency of a potential canister drop is estimated by the product of the frequency of canister lifts (i.e., the number per year) and the conditional probability of dropping the canister per lift. Uncertainties in data and models will be addressed in the frequency analyses.</p> <p>This analysis results in a categorization of each event sequence according to its mean frequency as either Category 1, Category 2, or Beyond Category 2. This frequency categorization is important because it establishes which portion of the Preclosure Performance Objectives of 10 CFR 63.111 must be met for each sequence. Note that the frequency categorization is based on the mean frequency of an entire sequence of events and not just the frequency of the initiating event.</p>
<b>Consequence Analysis</b>	In this portion of the PSA the potential mean consequences are calculated for Category 1 and Category 2 event sequences and compared against the regulatory limits of 10 CFR 63.111. For Category 1 event sequences, which includes chronic releases from normal operations, consequences are evaluated from relevant pathways as potential contributors to chronic exposures and are aggregated. For Category 2 event sequences, consequences are evaluated for relevant pathways for each sequence as an acute exposure. Uncertainties will be addressed in the consequence analyses.

\* Incredible (derived from 10 CFR Part 63) is defined such that the event sequence has less than a one in 10,000 chance of occurring before permanent closure (10 CFR 63.2).

\*\* Credible (also derived from 10 CFR Part 63) is defined such that the event sequence has at least one chance in 10,000 of occurring before permanent closure.

The PSA will demonstrate with reasonable assurance that HLW can be received, handled, packaged, stored, emplaced, and retrieved without exceeding regulatory dose limits. The results of the PSA will be included in the license application. The PSA will also include a consideration of the means for providing radiation protection to workers, for detection and suppression of fires, for control of radioactive wastes and effluents, and for implementation of criticality safety principles. The PSA results will demonstrate the ability of SSCs important to safety to perform their intended safety functions based on reliability requirements. The SSCs identified as important to safety are identified on the *Q-List*, YMP/90-55Q (YMP 2001).

### 12.3 DEVELOPMENT OF A RISK-INFORMED QUALITY ASSURANCE CLASSIFICATION PROCESS

The QA requirements in 10 CFR Part 50, Appendix B, were imposed on the safety-related SSCs in commercial nuclear reactor power plants. The safety-related SSCs for nuclear power plants include complex systems that are relied upon to ensure the integrity of the reactor coolant pressure boundary, ensure the capability to shut down the reactor, and prevent or mitigate the consequences of accidents having significant radiological consequences.

Initially, full Appendix B requirements were imposed on the safety-related SSCs of nuclear power plants on a deterministic basis. However, regulation of nuclear power plants has transitioned from the traditional, deterministic basis to a risk-informed basis. The implementation of NUREG-0800 (NRC 1987, Section 19.0), and Regulatory Guides 1.174 and 1.176 establishes the bases for graded QA requirements for a given nuclear power plant. Through the use of PRA, both safety-related and non-safety-related SSCs are classified into four risk-informed categories that provide the bases for QA grading of SSCs in a nuclear power plant. This graded approach for identifying risk significance provides general guidance for applying risk-informed QA classification to a HLW repository. However, this approach is not directly applicable since, compared to a nuclear power plant, a HLW repository is a low-energy, nonvolatile-hazard facility. The HLW repository important to safety features are basically passive with a limited need for automatic safety response to events. No short-term operator actions are expected to be required to meet the preclosure performance objectives. Also, no major risk-significant receipt, handling, packaging, storage, emplacement, or retrieval events have been identified in the site recommendation safety assessments that have been performed (BSC 2001). Therefore, NUREG-0800 (NRC 1987) and Regulatory Guides 1.174 and 1.176 should not be applied directly to a HLW repository, but do provide conceptual guidance to establish a risk-informed QA classification process. Consistent with this guidance, levels of risk significance (similar to the risk-informed safety significance for a nuclear power plant) can be developed using the event sequence frequency and consequence performance objectives of 10 CFR Part 63. Event sequence definitions and performance objectives from 10 CFR Part 63 provide bases for establishing the risk significance of SSCs.

From a risk perspective, the performance objectives of 10 CFR 63.111(b)(1) and (2) can be interpreted to define acceptable and unacceptable risk regions in terms of event sequence frequency (Category 1 or Category 2) and consequences (performance objectives for Category 1 and Category 2 event sequences). While the absolute risk associated with the performance objectives is small, relative risk regions can be established to identify the relative significance of an SSC. The relative risk regions derived from the 10 CFR Part 63 event sequence

categorization and performance objectives are summarized in Table 12-2. Note that the event sequence frequencies are derived from the definitions of Category 1 and Category 2 event sequences based on a 100-year preclosure period. The use of a 100-year preclosure period results in a frequency of  $1 \times 10^{-2}$  events per year for event sequences that have at least one chance in the life of the facility and a frequency of  $1 \times 10^{-6}$  events per year for event sequences that have one chance in 10,000 for the life of the facility. The consequence values are derived from 10 CFR 63.204 (15 mrem), 10 CFR 63.111(b)(2) (5 mrem) and 10 CFR 20.1301(a)(1) (100 mrem).

Table 12-2. Relative Risk Significance Based on 10 CFR Part 63

Event Sequence Frequency	Consequences (Dose [TEDE])		
	$15 \text{ mrem} \leq d < 100 \text{ mrem}$	$d \geq 100 \text{ mrem}$	$d \geq 5 \text{ rem}$
$\geq 10^{-2}$ per yr	Medium	High	
$\geq 10^{-6}$ per yr	Low	Medium	High

d = dose, TEDE = total effective dose equivalent.

An event sequence complies with the performance objectives when the event sequence frequency and consequences are within the boundaries of the acceptable risk region. Any event sequence with a frequency-consequence ordered-pair outside of these regions is an unacceptable risk. Figure 12-2 displays the acceptable risk regions based on the 10 CFR Part 63 preclosure performance objectives. For simplicity, only the mean Total Effective Dose Equivalent (TEDE) offsite preclosure performance objectives from 10 CFR Part 63 are displayed.

The region in Figure 12-2 between the frequencies of  $1 \times 10^{-2}$  events per year and  $1 \times 10^{-6}$  events per year that is bounded by the vertical consequence line at 5 rem is derived from the 10 CFR 63.2 definition of Category 1 and 2 event sequences and the Category 2 preclosure performance objectives. Category 1 event sequences are defined in 10 CFR 63.2 as:

Those event sequences that are expected to occur one or more times before permanent closure of the geologic repository area...

Assuming a preclosure operating period of 100 years, the sequence frequency separating Category 1 and Category 2 event sequences becomes 1 per 100 years, or a frequency of occurrence greater than or equal to  $1 \times 10^{-2}$  events per year. Event sequences at or above a frequency of  $1 \times 10^{-2}$  events per year are Category 1 event sequences. Those event sequences below a frequency of  $1 \times 10^{-2}$  events per year are Category 2 event sequences. Category 2 event sequences are defined in 10 CFR 63.2 as:

Other event sequences that have at least one chance in 10,000 of occurring before permanent closure...

With an assumed 100 year period before permanent closure, the frequency above which event sequences are defined as Category 2 is  $(1/10,000)/(100 \text{ years})$ , or  $1 \times 10^{-6}$  events per year. Event sequences with frequencies greater than or equal to  $1 \times 10^{-6}$  events per year, but less than  $1 \times 10^{-2}$  events per year, are defined as Category 2 event sequences.



The Category 2 performance objective stated in 10 CFR 63.111 is 5 rem per event sequence. This objective is represented in Figure 12-2 as a vertical line at 5 rem ending at frequencies of  $1 \times 10^{-2}$  and  $1 \times 10^{-6}$  events per year.

An isorisk line drawn with a slope of 15 mrem per year, ending with a vertical line drawn at 15 mrem, is illustrated in Figure 12-2 for Category 1 event sequences. The points on this isorisk line (i.e., the event sequence frequencies multiplied by the doses) have a constant risk of 15 mrem per year. This risk measure is based on the annual dose performance objective stated in 10 CFR 63.111(a)(2) for the aggregation of Category 1 radiation releases. The truncation of the isorisk line with a vertical line at a dose of 15 mrem reflects an interpretation of the Category 1 performance objective that states that no single event sequence shall exceed 15 mrem in any year.

A second isorisk line with a slope of 100 mrem per year is illustrated in Figure 12-2 for Category 1 event sequences. This line represents the annual dose objective of 100 mrem year from 10 CFR 63.111(a)(1) and the description of the numerical guide given in 10 CFR 63.111(b)(1). The 10 CFR 63.111(a)(1) guidance states that the requirements of 10 CFR Part 20 must be met for Category 1 event sequences; the Part 20 requirements state that the total annual dose cannot exceed 100 mrem per year. The truncation of the isorisk line with a vertical line at a dose of 100 mrem reflects an interpretation of this Category 1 performance objective that states that no single event sequence shall exceed 100 mrem in any year. For the purposes of QA classification, if the results of an SSC functional failure analysis (as explained in Section 5) produce a dose greater than 15 mrem along the 15-mrem isorisk line, the SSC will be classified as QL-2. Similarly, if the results of an SSC functional failure analysis produce a dose greater than 100 mrem along the 100-mrem isorisk line, the SSC will be classified as QL-1.

For simplicity, only the TEDE offsite preclosure performance objectives from 10 CFR Part 63 are displayed in this figure.

A high frequency (more than  $1 \times 10^{-2}$  events per year) low consequence (less than the 10 CFR 63.204 dose limits) risk region can be established that recognizes the potential for the occurrence of a low-risk-significant important to safety SSC. These SSCs can also provide margin (or defense-in-depth) for Category 1 event sequences and provide confidence that normal operations can be maintained while minimizing offsite radiological consequences. The low consequence threshold can be established based on trivial offsite consequences (taken to be less than one percent of the 10 CFR 63.204 dose limits). The updated relative risk significance is presented in Table 12-3.

The resultant relative risk is presented graphically in Figure 12-3, where:

- Risk Region I - High Relative Risk
- Risk Region II - Medium Relative Risk
- Risk Region III - Low Relative Risk
- Risk Region IV - Very Low (or no) Relative Risk

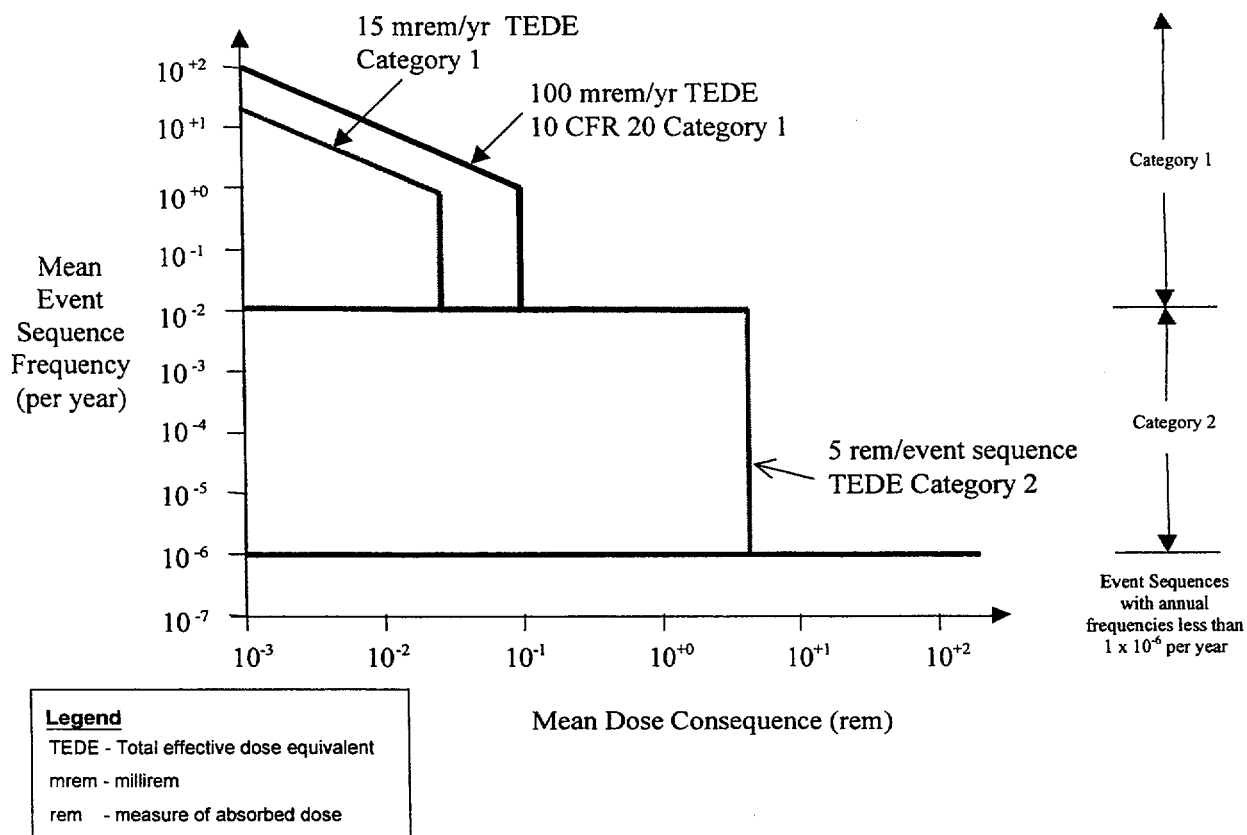


Figure 12-2. Compliance or Acceptable Risk Region Based on 10 CFR Part 63

Table 12-3. Updated Relative Risk Significance

Event Sequence Frequency	Consequences (Dose [TEDE])				
	$d < 0.15 \text{ mrem}$	$0.15 \text{ mrem} \leq d < 15 \text{ mrem}$	$15 \text{ mrem} \leq d < 100 \text{ mrem}$	$d \geq 100 \text{ mrem}$	$d \geq 5 \text{ rem}$
$\geq 10^{-2}$ per yr	Very Low	Low	Medium	High	
$\geq 10^{-6}$ per yr		Very Low	Low	Medium	High

d = dose, TEDE = total effective dose equivalent

The risk regions in Figure 12-3 can then be used to define regions of comparable risk significance. Since the level of QA requirements applied should be consistent with the risk significance of the SSC based on a risk-informed, performance-based approach, the QLs (where comparable QA controls are applied) can be established, as presented in Table 12-4.

Table 12-4. Quality Levels Based on Risk Significance

Event Sequence Frequency	Consequences (Dose [TEDE])				
	$d < 0.15$ mrem	$0.15 \text{ mrem} \leq d < 15$ mrem	$15 \text{ mrem} \leq d < 100$ mrem	$d \geq 100$ mrem	$d \geq 5$ rem
$\geq 10^{-2}$ per yr	CQ	QL-3	QL-2	QL-1	
$\geq 10^{-6}$ per yr		CQ	QL-3	QL-2	QL-1

d = dose, TEDE = total effective dose equivalent

Where:

- QL-1 - Quality level associated with highest relative risk significance, full application of 10 CFR 63.142 QA criteria
- QL-2 - Quality level associated with medium relative risk significance; selected application of 10 CFR 63.142 QA criteria, but less than full application of quality requirements
- QL-3 - Quality level associated with low relative risk significance; limited application of 10 CFR 63.142 QA criteria; less rigorous application of quality requirements than those of QL-2
- CQ - Quality level associated with little or no relative risk significance; application of 10 CFR 63.142 criteria not required

The acceptable risk regions are used as a basis to define areas of successively lower relative risk for QA classification of each SSC important to safety. The QA classification process starts after HLW repository event sequences identified by a PSA have been shown to comply with 10 CFR Part 63. The classification of each SSC is then determined through a sensitivity analysis (called an SSC functional failure analysis). This analysis recalculates the frequency and consequences for that event sequence assuming that a specific SSC fails to perform its defined function (i.e., the SSC is hypothetically removed from the event sequence). This classification analysis identifies the importance to safety of the SSC in satisfying the 10 CFR Part 63 radiological performance objectives.

In order to implement a QA classification process, QA classification criteria must be established for each QL consistent with the relative risk regions and the associated functional failure analysis. The quantitative QA classification criteria are derived from the 10 CFR Part 63 preclosure performance objectives for offsite doses and worker exposures. The event sequences are categorized into one of three frequency ranges: Category 1 event sequences, Category 2 event sequences, and beyond Category 2 event sequences (i.e., incredible). The risk-informed, performance-based criteria are established through the use of limiting dose performance coupled with Category 1 and Category 2 event frequency categorizations. The resultant QA classification criteria are summarized in Table 12-5. These risk-informed regulatory performance objectives are an essential element in the QA classification. Models and the results of the PSA are used to

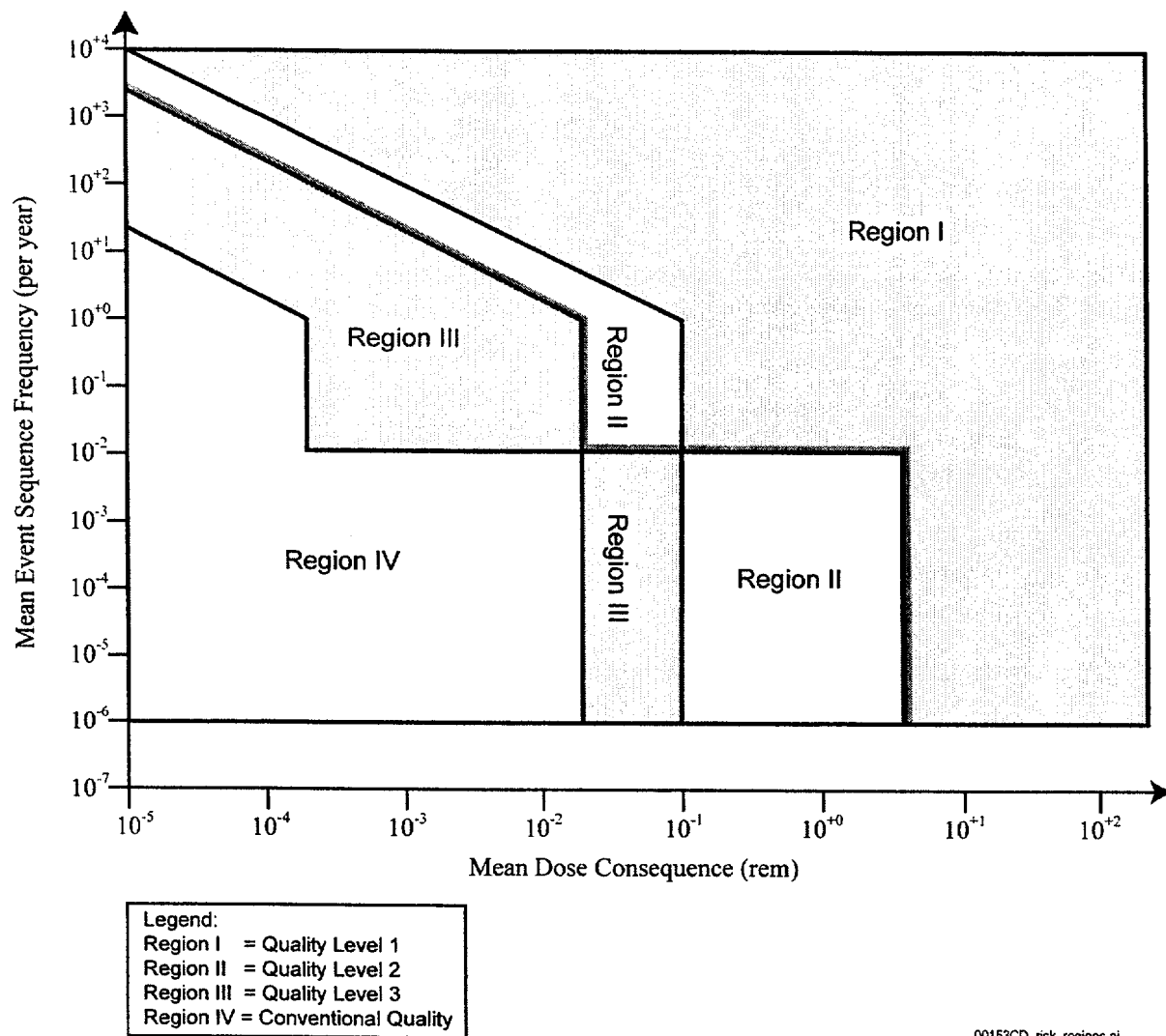


Figure 12-3. Example Quality Assurance Classification Risk Regions Developed from 10 CFR Part 63 Performance Objectives

assess the change in frequency or consequences that occur when a given SSC is assumed to be unavailable. A QL is assigned to an SSC in accordance with the risk region that contains the location of an event sequence, based on frequency and consequence, following the performance of the functional failure analysis for each SSC.

When evaluating the risk significance of an SSC in event sequences, the following factors should be considered:

- Is the SSC required to function to support another SSC that has been classified as important to safety?
- Does the SSC provide input to support a required action?

- Is there an interaction (e.g., seismic) in a portion of the event sequence that must be considered?
- Does the event sequence result in a fire that may impact the performance of an SSC in an event sequence?
- Are failure probabilities based on meeting licensing specifications or technical specifications (i.e., should licensing specification instrumentation be classified consistent with its role in the event sequence evaluation)?

Table 12-5. Risk Significant Screening Criteria

Criteria	Discussion	Basis
<b>QL-1</b>		
Frequency – Category 1 Consequence potential > 100 mrem TEDE	Regulatory Performance Objective Risk: High (High frequency, Medium consequence)	10 CFR 63.2 definition  10 CFR Part 20 limits (invoked by 10 CFR 63.111(a)(1) performance objective)
Frequency – Category 2 Consequence potential > 5 rem TEDE	Regulatory Performance Objective Risk: High (Low frequency, High consequence)	10 CFR 63.2 definition 10 CFR 63.111(b)(2) performance objective
<b>QL-2</b>		
Frequency – Category 1 Consequence potential > 15 mrem TEDE	Regulatory Performance Objective Risk: Medium (High frequency, Low consequence)	10 CFR 63.2 definition 10 CFR 63.204 performance objective
Frequency – Category 2 Consequence potential > 100 mrem TEDE	Risk judgment Risk: Medium (Low frequency, medium consequence)	10 CFR 63.2 definition 10 CFR Part 20 limit
<b>QL-3</b>		
Frequency – Category 1 Consequence potential > 0.15 mrem TEDE	Risk judgment classification for SSCs involved in event sequence Risk: Low (High frequency, low consequence) Supports minimization of normal releases	Engineering judgment to establish trivial offsite consequences (1% of category regulatory limit)
Frequency – Category 2 Consequence potential > 15 mrem TEDE	Risk judgment classification for SSCs involved in event sequence Risk: Low (Low frequency, low consequence)	Engineering judgment to establish trivial offsite consequences for associated event frequency (used QL-2 risk level for Category 1 event sequences)
Frequency – Category 1 Consequence potential > 5 rem to worker	Regulatory Performance Objective Highlights SSCs that contribute to worker dose reduction during category 1 event sequences	10 CFR 63.2 Regulatory Guide 8.8 10 CFR Part 20 limit (invoked by 10 CFR 63.111(a)(1) performance objective)

TEDE = total effective dose equivalent

Many of these factors are derived from industry and regulatory precedents. When evaluating the safety significance of an SSC, a robust, systematic approach is required that ensures that the

relevant factors have been considered. Industry and regulatory guidance such as the Regulatory Guides (e.g., 1.26, 1.29, 1.97, 1.143, 1.189), NUREGs (e.g., NUREG-0800 [NRC 1987]), and standards (e.g., ANSI/ANS-55.1-1992, American Society of Mechanical Engineers, and American National Standards Institute radioactive waste standards) can be used as guidance when evaluating the risk significance of an SSC. The use of precedent as guidance supports a risk-informed QA classification approach by linking the risk determined for an SSC to an identified hazard. Preclosure safety will be based on the SSCs that are required to meet the preclosure safety performance objectives. These SSCs are identified as important to safety in accordance with 10 CFR 63.2. In addition, defense-in-depth SSCs may also be included. By definition, defense-in-depth SSCs are not required in order to demonstrate compliance with preclosure performance objectives and are, therefore, classified as CQ.

## **12.4 QUALITY ASSURANCE CLASSIFICATION PROCESS**

The risk-informed, performance-based QA classification process is presented in Figure 12-4. Because the development of the PSA is an iterative process linked to the development of the design, so is the QA classification process. The PSAs that are based on, and consistent with, the level of development of a repository design provide inputs to the classification of SSCs. Using this information, each SSC is evaluated at the QL-1, QL-2, and QL-3 decision blocks depicted in Figure 12-4. At each decision block, each event sequence (that will become part of the licensing basis) involving the SSC being classified is considered. The QL-1, QL-2, and QL-3 screening criteria are applied sequentially to determine the appropriate QL for each SSC. The first positive response in a decision block yields the QL for the SSC. If the SSC passes through the three decision blocks, it is assigned a classification of CQ.

The QL decision criteria can be answered only with the support of PSA elements: hazards analyses for external and internal events, event sequence analyses, radiological consequence analyses, or discipline-specific analyses such as criticality analyses. Functional failure analyses are used to identify the relative risk significance of SSCs in event sequences.

The resultant SSC classifications are presented in the Q-List. Future updates to the Q-List will reflect the criteria prescribed in Table 12-5 and the process described in Figure 12-4.

## **12.5 FUNCTIONAL FAILURE ANALYSIS IN RISK INFORMED QUALITY ASSURANCE CLASSIFICATION**

The QA classification process is applied to individual SSCs in sequential order. The process is different for Category 1 and Category 2 event sequences due to the differences in the regulatory performance objectives. The steps in the QA classification process for Category 1 event sequences involve evaluating both the consequences of normal operations and Category 1 event sequences, which are then summed to estimate an annual dose. The dose summation provides aggregated consequences prior to permanent closure to show compliance with the performance objectives stated in 10 CFR 63.111(b)(1). Category 2 event sequences are assessed on an individual basis; the process has fewer steps than for Category 1 event sequences.

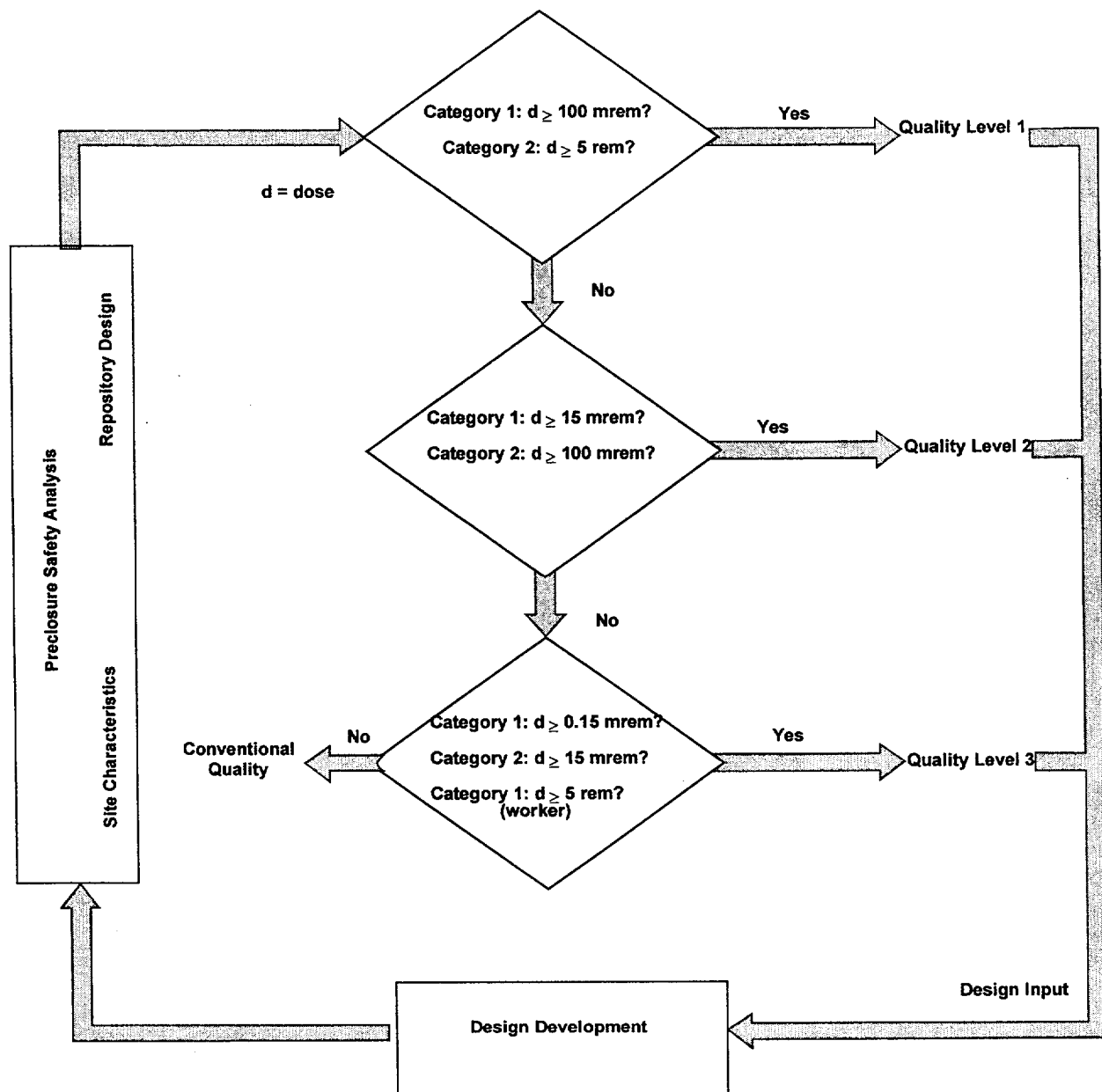


Figure 12-4. Risk Informed Quality Assurance Classification Process

## 12.6 RISK-INFORMED QUALITY ASSURANCE CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS ASSOCIATED WITH CATEGORY 1 EVENT SEQUENCES

As noted previously, before the QA classification process is applied, Category 1 event sequences will have been demonstrated to comply with the performance objectives. Demonstrating compliance for Category 1 event sequences takes into consideration contributions from normal surface and subsurface operational and offsite radiological releases from Category 1 event sequences in the annualized dose estimates. These contributions to the annualized dose estimate are part of the aggregation of exposures required by 10 CFR Part 63.

Compliance with the Category 1 annual dose objectives is performed as follows:

1. Demonstrate that the aggregate radiation exposures and the aggregate radiation levels in both restricted and unrestricted areas, and the aggregate releases of radioactive materials to unrestricted areas from normal operations in addition to the Category 1 event sequences meet the annual dose performance objectives in 10 CFR 63.111(a).
2. Demonstrate that the radiation exposure and the radiation levels in both restricted and unrestricted areas and releases of radioactive materials to unrestricted areas for each Category 1 event sequence meet the annual dose performance objectives in 10 CFR 63.111(a).
3. Demonstrate that the radiation exposure and the radiation levels in both restricted and unrestricted areas, and releases of radioactive material to unrestricted areas resulting from combinations of two or more Category 1 event sequences that could occur in a single year with a mean frequency greater than or equal to a frequency of  $1 \times 10^{-2}$  events per year, meet the annual offsite dose performance objectives in 10 CFR 63.111(a).

The methodology for Category 1 event sequence compliance demonstration is described in the following section.

## 12.7 COMPLIANCE DEMONSTRATION FOR NORMAL OPERATION RELEASES AND CATEGORY 1 EVENT SEQUENCES

For compliance demonstration of Category 1 event sequences and normal operating releases from the surface and subsurface facilities, the annual offsite dose is calculated as follows:

$$D_{\text{Cat1}} = D_{\text{norm}} + \sum F_i D_i \quad (\text{Eq. 12-1})$$

where

$D_{\text{Cat1}}$	=	total annual offsite doses from Category 1 event sequences and normal operations (mrem per yr)
$D_{\text{norm}}$	=	the expected annual offsite dose from surface and subsurface normal releases (mrem per yr)
$F_i$	=	frequency for event sequence $i$ (per year)
$D_i$	=	offsite dose for $i^{\text{th}}$ Category 1 event sequences (mrem)
$\sum F_i D_i$	=	sum of the frequency-weighted offsite doses for the Category 1 event sequences in any given year, summed over sequences $i = 1$ to $n$ , where $n$ is the number of Category 1 event sequences (mrem per yr)

Compliance is demonstrated when the performance objectives in 10 CFR 63.111(a) are met (e.g., when  $D_{\text{Cat1}}$  is demonstrated to be less than 15 mrem per yr TEDE).



Further, to ensure that no single event sequence could exceed the performance objectives in 10 CFR 63.111(a); compliance is also assessed for each individual Category 1 event sequence as follows:

$$D_i < 15 \text{ mrem} \quad (\text{Eq. 12-2})$$

where

$$D_i = \text{offsite TEDE dose for individual Category 1 event sequence } i \text{ (mrem)}$$

Total offsite doses are estimated for compliance demonstration of combinations of two or more Category 1 event sequences. The combinations of event sequences that can occur with an estimated mean frequency of  $1 \times 10^{-2}$  events per year or greater are selected from the list of Category 1 event sequences. The offsite doses for each event sequence in the combination are summed and then compared to the 10 CFR 63.111(a) performance objectives. The doses from the combined event sequences that could credibly occur are evaluated as follows to demonstrate compliance:

$$D_{Comb} = \Sigma D_i \quad (\text{Eq. 12-3})$$

where

$$D_{Comb} = \text{total offsite dose for credible combinations of individual Category 1 event sequences (mrem)}$$

$$D_i = \text{offsite dose for the } i^{\text{th}} \text{ Category 1 event sequence considered in the combination (mrem)}$$

Compliance is demonstrated when the dose from the combination of two or more Category 1 event sequences that could credibly occur is shown to be less than 15 mrem per yr TEDE.

## 12.8 PROCESS FOR RISK-INFORMED QUALITY ASSURANCE CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS IN CATEGORY 1 EVENT SEQUENCES

The QA classification of a particular SSC may require several different functional failure analysis iterations. The first analysis is applied to the SSCs involved in individual Category 1 event sequences. The second analysis is applied to those same SSCs that may also appear in credible combinations of Category 1 event sequences. For the classification analysis of each SSC, the offsite doses that result after the assumed functional failure of the SSC are evaluated in the following manner:

$$D_{cc} = [D_{norm} + \Sigma F_i D_i] + D_e^* \quad (\text{Eq. 12-4})$$

where

$$D_{norm} \text{ and } \Sigma F_i D_i \text{ are defined previously}$$

- $D_{ce}$  = offsite classification event dose with the assumed functional failure of the SSC (i.e., the SSC is hypothetically removed from the sequence being evaluated) (mrem per yr)
- $D_e$  = dose from an event sequence that includes the SSC being classified; however the mitigation function of the SSC is assumed to have failed (the sequence is assumed to occur no more than once per year) (mrem per yr)

**NOTE:** The Category 1 event sequences are assumed to have an estimated mean frequency of less than once per year. However, for the purpose of QA classification, these event sequences are assumed to have occurred every year and are not annualized.

The offsite classification event dose ( $D_{ce}$ ) represents the potential dose for Category 1 event sequences if the SSC under study was assumed to fail when called upon to mitigate consequences. The increase in dose resulting from the removal of the SSC is  $D_e$ . The  $D_e$  is added to aggregate offsite doses of  $D_{norm} + \sum F_i D_i$ . This approach provides a risk-informed basis for classifying each SSC. If  $D_{ce}$  exceeds 15 mrem per year TEDE but is less than 100 mrem per year TEDE, the SSC is classified as QL-2. If  $D_{ce}$  exceeds 100 mrem per year TEDE, the SSC is classified as QL-1.

If the SSC is also part of a credible combination of two or more Category 1 event sequences that could occur at a mean frequency of  $1 \times 10^{-2}$  events per year or greater, the QA classification process proceeds considering the unavailability of each SSC that has a mitigation function for that combination of event sequences. The annual offsite dose resulting from the combination of credible Category 1 event sequences after the assumed functional failure of an SSC will determine the QA classification of that SSC. If  $D_{Comb}$  exceeds 15 mrem per year TEDE but is less than 100 mrem per year TEDE, the SSC is classified as QL-2; if  $D_{Comb}$  exceeds 100 mrem per year TEDE, the SSC is classified as QL-1.

If applicable, each SSC is also evaluated for QA classification according to Category 2 event sequence criteria. The final QA classification for each SSC is the highest classification level resulting from the QA classification process used for Category 1 and Category 2 event sequence evaluations.

The steps for performing the QA classification evaluation of SSCs in Category 1 event sequences are summarized in Table 12-6.

## **12.9 RISK-INFORMED QUALITY ASSURANCE CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS ASSOCIATED WITH CATEGORY 2 EVENT SEQUENCES**

As discussed previously, before the QA classification process is begun, event sequences will have been shown to meet the preclosure performance objectives stated in 10 CFR Part 63. Compliance analyses for Category 2 event sequences will demonstrate, for each event sequence assessed individually, that the offsite dose is less than 5 rem TEDE (10 CFR 63.111(b)(2)). The classification analyses reassess the dose after an SSC functional failure is performed. If the dose

exceeds 5 rem, the SSC is classified as QL-1. If the dose is less than 5 rem but greater than 100 mrem, the SSC is classified as QL-2. If the dose is less than 100 mrem but greater than 15 mrem, the SSC is classified as QL-3. Otherwise, the SSC is not subject to the requirements of the 10 CFR Part 63 QA program (i.e., is classified as CQ).

Figure 12-5 illustrates an example of the QA classification process using the functional failure analysis. Initially, an event sequence is identified and demonstrated to comply with the performance objectives. For example, the square in Figure 12-5 represents an event sequence with a mean frequency of  $5 \times 10^{-5}$  events per year and an offsite dose of  $5 \times 10^{-3}$  rem TEDE. This event sequence complies with the performance objectives based on the performance of one or more SSCs important to safety. After the assumed removal of an SSC considered in the original event sequence evaluation, the consequences are re-estimated. In the illustration in Figure 12-5, the consequence is now shown to be 6 rem TEDE (a value used for illustration only) after the assumed functional failure of the SSC under consideration. This dose exceeds the 10 CFR 63.111(b)(2) performance objective of 5 rem TEDE. The resulting event sequence (represented as a circle in Figure 12-5) now has a mean frequency of  $5 \times 10^{-5}$  events per year and dose of 6 rem TEDE. This analysis demonstrates that the SSC under consideration is important to safety and that without it the event sequence would be in Risk Region I. In this illustration, the SSC under consideration would be classified as QL-1.

If the same SSC appears in more than one event sequence, regardless of whether in Category 1 or Category 2 regions, an appropriate functional failure analysis would be performed. The analysis would be completed for each affected event sequence. The highest QA classification resulting from these analyses would be assigned to the SSC.

Table 12-6. Steps for Performing a Quality Assurance Classification Analysis of Structures, Systems, and Components Associated with Category 1

1. Calculate  $D_{\text{norm}}$ , the annual offsite dose from surface and subsurface normal operations.
2. Calculate  $D_i$ , the offsite dose from a Category 1 event sequence  $i$ .
3. Calculate  $\Sigma F_i D_i$ , the frequency-weighted offsite dose sum of the Category 1 event sequences [i.e.,  $i = 1$  to  $n$  sequences].
4. Identify event sequences that include the SSC being classified and its associated offsite dose  $D_e$  (after the SSC is assumed to be removed).
5. Perform an SSC functional failure analysis for each event sequence that includes the SSC being classified using classification event dose  $D_{\text{cc}}$  where  $D_{\text{cc}} = [D_{\text{norm}} + \Sigma F_i D_i] + D_e$ .
6. Perform an SSC functional failure analysis for each Category 1 combination of event sequences that includes the SSC being classified (the  $D_e$  from each event sequence is included).
7. Classify SSCs based on highest classification level identified based on the functional failure analysis from each Category 1 event sequence or Category 1 combination that includes the SSC being evaluated. If applicable, each SSC is also evaluated for classification according to Category 2 event sequences and consideration of event sequences with annual frequencies less than  $1 \times 10^{-6}$  events per year.

The final SSC QA classification is the highest classification level resulting from applying the QA classification objectives from Category 1, Category 2 and event sequences that are at a lower event sequence frequency than the Category 2 event sequence frequency.

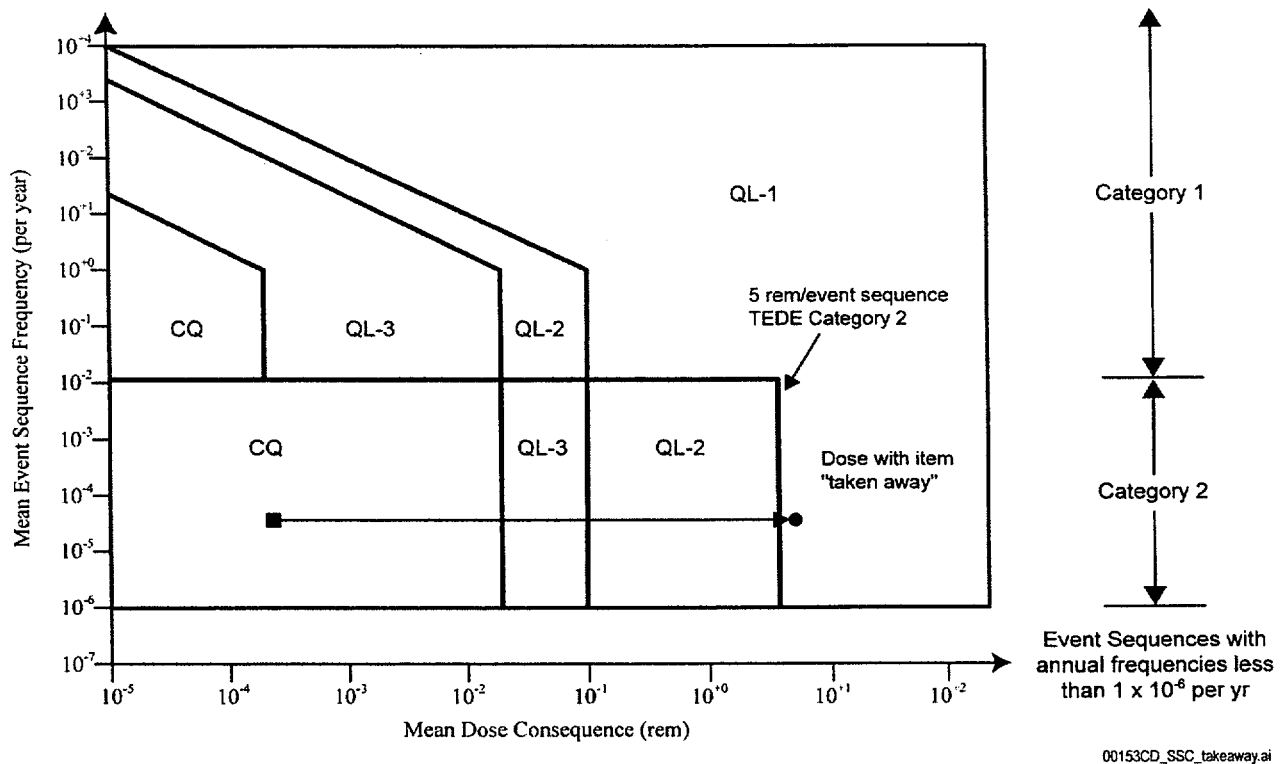


Figure 12-5. Example of the Quality Assurance Classification Process for a Category 2 Event Sequence

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## ACRONYMS

PSA	preclosure safety analysis
SAR	safety analysis report
SSCs	structures, systems, and components

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## **13. SELECTION OF 10 CFR 63.2 DESIGN BASES FOR STRUCTURES, SYSTEMS, AND COMPONENTS IMPORTANT TO SAFETY**

### **13.1 INTRODUCTION**

This section describes the development of the design bases for structures, system, and components (SSCs) important to preclosure safety. Background information from 10 CFR Part 63 will be presented followed by a general discussion of the methodology for developing the design bases for SSCs and an example.

### **13.2 BACKGROUND INFORMATION**

Important to safety, SSCs will have design bases established as defined in 10 CFR 63.2:

Design bases means that 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 constraints derived from generally accepted "state-of-the-art" practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include:

- (1) Estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved; and
- (2) Estimates of severe external human-induced events to be used for deriving design bases, that will be based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.

The values for the controlling parameters for external events include (1) estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved; and (2) estimates of severe external human-induced events to be used for deriving design bases, that will be based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.

When describing the content of application, 10 CFR 63.21 states that the safety analysis must include among other items:

- (3) (ii) The design criteria used and their relationship to the preclosure and postclosure performance objectives specified at §63.111(b), §63.113(b) and §63.113(c); and (iii) The design bases and their relation to the design criteria.

Design basis requirements are developed from the geologic repository Category 1 and Category 2 event sequences that are identified through a preclosure safety analysis (PSA) (as discussed in Section 7.6 of this guide). The Category 1 and 2 event sequences are evaluated against their respective dose performance objectives. SSC safety functions are identified from these event sequences.

SSCs involved in the event sequences that are required to prevent or mitigate offsite dose from exceeding the 10 CFR Part 63 preclosure performance objectives are selected as important to safety. A design basis is developed for each of these SSCs. The design basis describes the SSC safety function. Design criteria are established for each safety function. Design criteria are bounding values for controlling specific values or ranges of values chosen for controlling parameters as reference bounds for the design. The relationship of each SSC design criteria to the design bases must be tied directly to the preclosure performance objectives specified in 10 CFR 63.111(b).

A distinction is made between the repository design bases and 10 CFR 63.2 design bases for SSCs. SSCs that comprise the repository design will have design bases. However, only SSCs that are important to safety will have design bases developed per 10 CFR 63.2. These 10 CFR 63.2 design bases are a subset of the licensing bases and are required pursuant to 10 CFR 63.112 to be included in the safety analysis report (SAR). The SAR will set forth a safety assessment of the 10 CFR 63.2 repository design bases. Both 10 CFR 63.2 design bases and supporting design information are subjected to design control and other quality assurance criteria of 10 CFR 63.142 as applicable to the quality level classification of the SSCs. The 10 CFR 63.2 design bases and supporting information contained in the SAR are controlled in accordance with 10 CFR 63.44.

Figure 13-1 shows the relationship of 10 CFR 63.2 design bases to the repository design bases and the SAR.

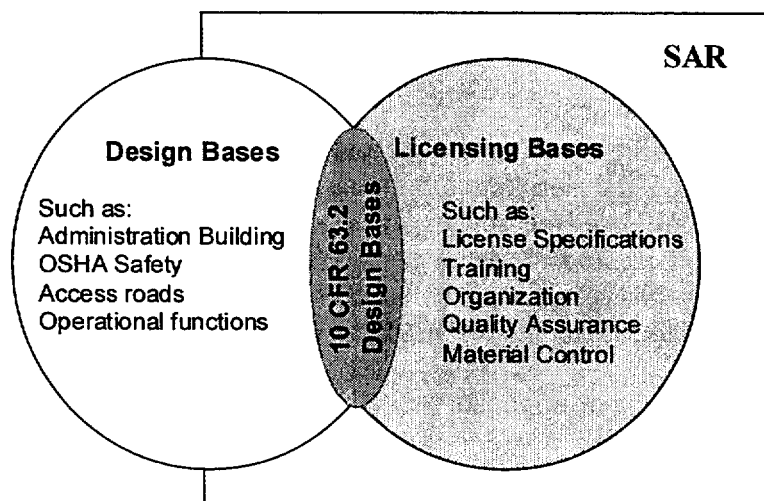


Figure 13-1. Relationship of Repository Design Bases to 10 CFR 63.2 Design Bases and the Safety Analysis Report (SAR)

The white circle represents the design bases for each SSC comprising the repository facility. The dark gray circle represents the complete licensing bases presented in the SAR. The crosshatched region depicts those SSC that are important to safety and have 10 CFR 63.2 design bases.

The assessment of the important to safety SSCs should be in sufficient detail so that the U.S. Nuclear Regulatory Commission staff can make an independent determination that there is reasonable assurance that safe operation will be achieved. Underlying 10 CFR 63.2 design bases is supporting design that includes design inputs, design analyses, and design output documents. Supporting design information may be contained in the SAR or other documents some of which are docketed and some that are retained by the licensee.

### 13.3 METHOD FOR DEVELOPING 10 CFR 63.2 DESIGN BASES

The basic steps in developing the 10 CFR 63.2 design bases are illustrated in Figure 13-2 and summarized as follows:

1. Identify, through a PSA, the Category 1 and 2 event sequences that are derived from internal, external, and manmade hazards (see Sections 6, 7, 10, and 11).

The list of Category 1 and 2 events form a part of the repository licensing bases and will appear in the SAR. Each Category 1 or 2 event sequence contains SSCs modeled within the PSA to assess the likelihood and consequences of an event sequence.

This list may change as the design matures. Changes in this list will result in a reassessment of the design bases and design criteria. Design iterations, design improvements, or modifications can lead to changes in this list throughout the licensing process and beyond.

2. Identify those SSCs important to safety, based on the Category 1 and 2 event sequences (see Section 12).

Every SSC on this list will require the establishment of 10 CFR 63.2 design bases and design criteria.

3. Select an important-to-safety SSC from the above list and identify the Category 1 and Category 2 event sequences that contain that SSC.

Category 1 and Category 2 event sequence compliance with 10 CFR Part 63 performance requirements are significantly different. Category 1 compliance assessments are based on annual performance requirements that require an aggregation of releases to unrestricted areas, as described in Section 8. Category 2 event sequence compliance assessment is on a per event basis. No aggregations of release are to be done for Category 2 event sequences. Because of these compliance differences, it will be easier to start with SSCs involved in Category 2 event sequences. Develop the design bases and design criteria for these SSCs first.

4. Identify the design criteria by safety function, from the selected Category 2 event sequences for the selected SSC.

Beginning with an SSC identified from Category 2 event sequences, establish the design criteria and safety functions or design bases. Document the relationship between design criteria and design bases and the performance objectives that are met by the SSC in the event sequence under study. Maintain this relationship by SSC within a database or some other organizational tool.

To provide confidence in the repository preclosure design, in some cases, the PSA includes evaluation of the consequences of selected sequences that are below the Category 2 frequency threshold, using the best estimate conditions. The purpose of this evaluation is to ensure that such an event sequence with unacceptable consequences is not arbitrarily excluded, based on probability. Any features added to mitigate the consequences of such events would not be considered as important to safety. Such analysis will not be part of the safety case, but has the potential of providing additional confidence in the repository performance.

5. Select the Category 2 bounding 10 CFR 63.2 design criteria for each safety function identified for the selected SSC.
6. Repeat Steps 3 through 5 until the important-to-safety SSCs have 10 CFR 63.2 design bases developed for Category 2 event sequences.
7. Select an SSC that is important to safety for meeting Category 1 performance criteria.
8. Examine the Category 1 event sequences containing this SSC and develop 10 CFR 63.2 design bases and design criteria.
9. Select the bounding design bases and design criteria for this SSC that will meet the performance objectives.
10. After the bounding Category 1 design bases are established for an SSC, review the Category 2 design bases established and optimize the Category 1 and 2 design bases such that the most limiting performance objectives from 10 CFR 63.111 are met.
11. Repeat Steps 7 through 10 until SSCs in Category 1 event sequences have 10 CFR 63.2 design bases established.
12. Review the total set of bounding design criteria for each important-to-safety SSC for completeness and consistency.

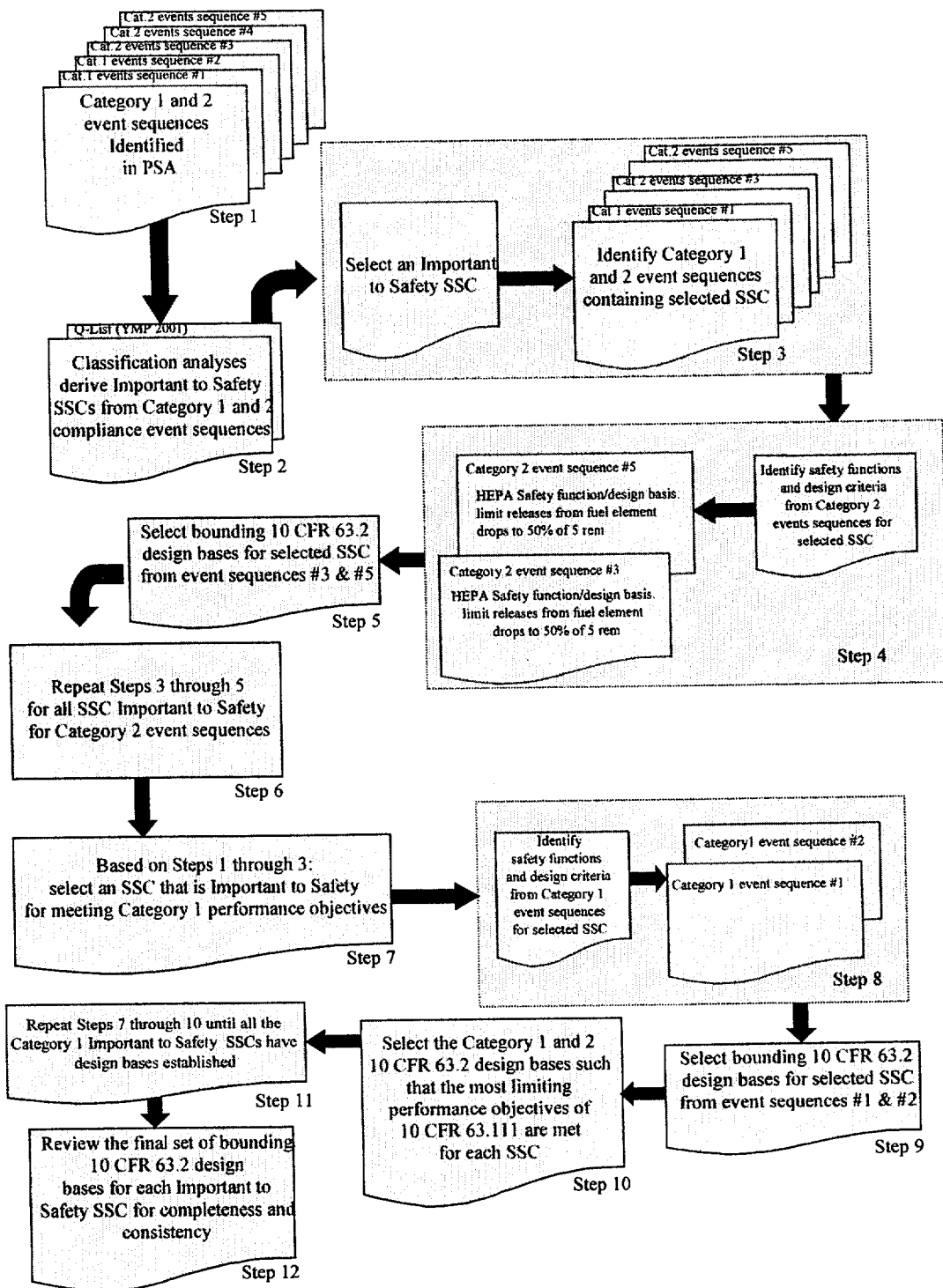


Figure 13-2. Basic Steps in Developing 10 CFR 63.2 Design Bases

### 13.4 EXAMPLES

[Information for this section is under development and will be provided later.]

## **13.5 REFERENCES**

### **13.5.1 Documents Cited**

YMP (Yucca Mountain Site Characterization Project) 2001. *Q-List*. YMP/90-55Q, Rev. 7. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.20010409.0366.

### **13.5.2 Codes, Standards, Regulations, and Procedures**

10 CFR 63. 2002. Energy: Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada. Readily available.

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## ACRONYMS

ALARA	as low as is reasonably achievable
BC2	beyond Category 2
HVAC	heating, ventilation, and air-conditioning
NRC	U.S. Nuclear Regulatory Commission
PSA	Preclosure Safety Analysis
SSCs	structures, systems, and components
YMP	Yucca Mountain Site Characterization Project

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## 14. DOCUMENTATION AND PREPARATION OF LICENSE APPLICATION

### 14.1 PRECLOSURE SAFETY ANALYSIS DOCUMENTATION

The documents that are required to provide the input for the Preclosure Safety Analysis (PSA) are described in the following sections.

#### 14.1.1 Internal and External Hazards Analyses

The internal and external hazards analyses document the potential hazards that may be present at a repository at Yucca Mountain. These analyses should be consistent with the 10 CFR 63.2 repository design bases. The process used to identify hazards is described in Section 6.

The internal hazards analysis describes the potential hazards that are related to the design and operation of the repository during the preclosure period, including potential chemical and criticality hazards. The external hazards analysis describes the spectrum of potential external events and natural phenomena. The credible hazards (i.e., initiating events) during the preclosure period will be based on historical data or U.S. Nuclear Regulatory Commission (NRC) regulatory guidance.

#### 14.1.2 Categorization of Event Frequencies

Using the results of the internal and external hazards analyses, potential event sequences will be categorized as Category 1, Category 2, or beyond Category 2 (BC2) event sequences by analysis. The analyses will be continually updated as design develops to ensure the categorization, design, and hazards identification are consistent with license application design. The evaluation of potential hazards will be captured in analyses and calculations such as:

**Aircraft Hazards Assessment**—The credibility of aircraft hazards within the selected vicinity of the general repository operations area, consistent with NUREG-0800 (NRC 1987) and other appropriate NRC guidance, will be analyzed using current flight information. The appropriateness of flight information for the repository operating period is evaluated, and future efforts, if required, to support aircraft hazards categorization are identified.

**Wind and Tornado Analysis**—The design basis tornado and wind loadings will be identified in accordance with nuclear and regulatory guidance (e.g., NUREG-0800 [NRC 1987] and Regulatory Guide 1.76). The tornado missile spectrum for the Yucca Mountain Site Characterization Project (YMP) will be developed. The features and controls that are required to ensure that design basis tornado and winds will not result in a radiological release that exceeds regulatory requirements will be identified.

**Industrial and Military Hazards Assessment**—The potential industrial and military hazards for potential consideration in the repository design consistent with nuclear industry precedents (e.g., NUREG-0800 [NRC 1987]) will be identified and evaluated. Any features or controls required to screen or mitigate industrial and military hazards will be identified.

**Rainstorm and Flooding Analysis**—The rainstorm and flooding criteria is determined in accordance with accepted nuclear precedents (e.g., NUREG-0800 [NRC 1987]). Any features or

controls required to ensure that a rainstorm or flooding event does not result in a radiological release will be identified.

**Seismic Analysis**—The appropriate SSC seismic design criteria consistent with regulatory requirements and the YMP seismic topical report will be identified. The SSC seismic criteria to ensure that regulatory requirements are met in the event of a seismic event will be identified.

**Fire Sequence Analysis**—Using the results of the facility fire hazards analyses, credible fires will be evaluated as initiating events for the potential to result in a radiological release. Potential fire scenarios should include surface, subsurface, and external fires (e.g., range fires and lightning-initiated fires). Any features or controls that ensure that fires do not result in a radiological release that exceeds regulatory requirements will be identified.

**Loss of Power**—Loss of power as an initiating event will be evaluated. It should be demonstrated that loss of power does not result in a radiological release that exceeds regulatory requirements. Any features or controls that are required to ensure loss of power does not result in releases that exceed regulatory requirements will be identified.

**Waste Handling Fault Tree Analysis**—The reliability of handling systems for use in event trees that include handling branches will be determined.

**Component Failure and Reliability Analysis Database**—Industry failure rate information to support the development of event trees will be collected and analyzed. Justification for appropriateness of failure rates for use at YMP will be included. An uncertainty analysis of the failure rates that is appropriate to support categorization of event sequences will be included.

#### **14.1.3 Consequence Analysis**

Potential consequences from Category 1 and 2 event sequences will be evaluated to demonstrate any radiological releases meet regulatory requirements. Consequence analyses will be available to support the SSC classification. Mean consequences, including the associated uncertainty distribution, will be calculated using the GENII-S computer code (Leigh et al. 1993) or other methods. Any features and controls that are required to limit radiological consequences to within regulatory limits will be identified. The potential radiological consequences of the selected BC2 event sequences will be determined. The basis will be provided for BC2 event sequences that are selected for evaluation. BC2 event sequences will be selected for evaluation to gain risk insights into the design and support identification of defense in depth features. Commercial spent nuclear fuel release fractions analysis will be available. Preclosure safety requirements for atmospheric dispersion factors will be developed. Atmospheric dispersion factors based on site meteorological data through 2001 will be developed. Site data collection will continue to support updates to the dispersion factor calculation. Dispersion factor calculation will be used to support calculations of mean and upper bound consequences from event sequences.

#### **14.1.4 Preclosure Safety Analysis**

The PSA will be consistent with 10 CFR Part 63 requirements, the Yucca Mountain Review Plan (when issued by the NRC), and other NRC guidance and interactions. The PSA will demonstrate

compliance with PSA regulatory requirements; summarize hazards analyses, event categorization, consequence analyses, worker dose, ALARA (i.e., as low as is reasonably achievable), radiation protection program, preclosure criticality, and classification processes and results.

**Preclosure Safety Analysis Guide**—The PSA guide will present the project approach to developing a PSA. The guide will identify PSA project interfaces and responsibilities. It will describe processes for performing hazards analyses and for developing event trees, fault trees, and event scenarios. Guidance on performing uncertainty analyses, Category 1 and 2 consequence analysis approaches, and developing classification analyses will be included. Also, discussions on the integration of PSA work performed in other project areas should be included (e.g., design requirements and criticality). Examples of products to be used in preparation of the PSA include:

- Internal hazards analysis
- External hazards analysis
- Aircraft hazards analysis
- Wind, tornado, and tornado missile analysis
- Industrial and military hazards analysis
- Rainstorm and flooding analysis
- Seismic analysis
- Fire sequence analysis
- Loss of power analysis
- Failure rate and reliability data analysis
- Categorization of event sequences analysis
- Consequence analysis
- BC2 evaluation plan
- Atmospheric dispersion factors calculation
- 10 CFR 63.2 design bases report
- Classification analysis updates (consistent with design needs) and *Q-List* updates

These products will support the final major PSA input into the license application design.

#### **14.1.5 Identify Structures, Systems, and Components to Prevent or Mitigate Event**

10 CFR 63.2 design bases, classification analyses, and *Q-List* (YMP 2001) will be maintained consistent with the latest regulatory requirements, Yucca Mountain Review Plan (when issued by NRC), safety analyses, and design concepts. SSCs will be identified and maintained for the *Q-List* and the selection and implementation of quality assurance requirements consistent the risks identified in the preclosure safety analyses. 10 CFR 63.2 design bases report and *Q-List* will be maintained by integrating with the design organizations and updating the products, as appropriate, to ensure consistency between the products and the design.

The classification analyses determine the quality level classification of SSCs based on their role in meeting radiological safety requirements. The *Q-List* is a tabulation of the repository SSCs and their respective quality level classification.

## **14.2 PRECLOSURE SAFETY ANALYSIS AND LICENSE APPLICATION SUBMITTAL**

The license application submittal will be prepared in accordance with "Management Plan for Development of the Yucca Mountain License Application" (in preparation).

[Information for this section is under development and will be provided later.]

## **14.3 REFERENCES**

### **14.3.1 Documents Cited**

Leigh, C.D.; Thompson, B.M.; Campbell, J.E.; Longsine, D.E.; Kennedy, R.A.; and Napier, B.A. 1993. *User's Guide for GENII-S: A Code for Statistical and Deterministic Simulations of Radiation Doses to Humans from Radionuclides in the Environment*. SAND91-0561. Albuquerque, New Mexico: Sandia National Laboratories. ACC: MOL.20010721.0031.

NRC (U.S. Nuclear Regulatory Commission) 1987. *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*. NUREG-0800. LWR Edition. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 203894.

YMP (Yucca Mountain Site Characterization Project) 2001. *Q-List*. YMP/90-55Q, Rev. 7. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.20010409.0366.

### **14.3.2 Codes, Standards, Regulations, and Procedures**

10 CFR 63. 2002. Energy: Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada.

Regulatory Guide 1.76, Rev. 0. 1974. *Design Basis Tornado for Nuclear Power Plants*. Washington, D.C.: U.S. Atomic Energy Commission. TIC: 2717.

## **GLOSSARY**

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## GLOSSARY

Term	Definition	Reference
Acceptance Limit	A value that accounts for uncertainty about the conditions to which structures, systems, and components will be subjected and accounts for variability in the properties of component materials. This value provides a margin for unacceptable conditions. Whether the value is a maximum or a minimum depends on the type of variable being discussed.	
Analysis	A documented study, mathematical process, or evaluation defining, investigating, validating, solving, or reviewing: an engineering problem using formula or computer code for the resulting engineering parameters; the development of design inputs; translation of design input into design output; or performance of engineered structures, systems, and components.	AP-3.12 Q
Assumption	A statement or proposition that is taken to be true or representative in the absence of direct confirming data or evidence.	AP-SIII.9Q
Breach	An opening in a transportation cask, spent nuclear fuel canister, disposal container, waste package, or drip shield caused by corrosion or mechanical stress.	
Calculation	A documented study, mathematical process, or evaluation defining, investigating, validating, solving, or reviewing: an engineering problem using formula or computer code for the resulting engineering parameters; the development of design inputs; translation of design input into design output; or performance of engineered structures, systems, and components.	AP-3.12Q
Calculated Value	Values used as input assumptions for safety analyses or evaluations, or which result from the performance of a safety analysis or evaluation, and which the U.S. Nuclear Regulatory Commission has accepted during its review of a license application.	
Certification	The act of determining, verifying, and attesting in writing to the achievement or compliance with specified requirements.	DOE 2000
Codes and Standards	Applicable industry codes and standards are those codes and standards applicable to the design of the structures, systems, and components	
Computation	A mathematical process of solving a problem by formula or computer code for the resulting engineering or scientific parameters. See the definition of Calculation.	
Confinement	To keep within limits; restrict.	
Committed Effective Dose Equivalent	The sum of products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to those organs or tissues. The committed dose equivalent means the dose equivalent to organs or tissues of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake.	10 CFR 20
Conventional Quality	Conventional quality items (i.e., not QL-1, QL-2, or QL-3) are not subject to the requirements of the <i>Quality Assurance Requirements Description</i> (DOE 2000). Program management controls are applied commensurate with regulatory requirements, industry standards, local codes, and good engineering practices.	

Term	Definition	Reference
Conservative	In developing and applying mathematical models of physical systems, choices can be made regarding assumptions, approximations, data values, and data distributions. If these choices are made so that the resulting models and the estimates produced by them tend to make the estimated performance of a safety system worse than might actually be expected, the choices made are considered conservative or pessimistic. If the development and application of the model are such that the estimated performance tends to be better than might actually be expected the choices made are considered optimistic.	Eisenberg et al. 1999
Consequence	Result or effect.	
Containment	The confinement of radioactive waste within a designated boundary.	10 CFR 63.2
Credible Event	An event or event sequence having a probability of occurrence of at least 1 in 10,000 prior to the final closure of the repository.	
Criticality	The condition in which nuclear fuel sustains a chain reaction. It occurs when the effective neutron multiplication factor (the number of fissions in one generation divided by the number of fissions in the preceding generation) of a system equals one.	
Data	As it pertains to Supplement III, information developed as a result of scientific investigation activities associated with site characterization of the Yucca Mountain repository or the results of reducing, manipulating, or interpreting data after its field or laboratory acquisition to prepare it for use in analyses, models, or calculations used in performance assessment, integrated safety analyses, the design process, performance confirmation, or other similar work using data as an input.	Document Action Request D813 to the <i>Quality Assurance Requirements and Description</i> document (DOE 2000)
Defense-In-Depth	(1) A design strategy based on a system of multiple, independent, and redundant barriers, designed to ensure that failure in any one barrier does not result in failure of the entire system.  (2) A term used to describe a system of multiple barriers that mitigate uncertainties in conditions, processes, and events.	DOE 2001
	An element of the U.S. Nuclear Regulatory Commission safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges.	NRC 1998
Design Agency	The design agency is the organization that performs design activities particularly those associated with design analysis and calculations. The design agency performs design activities at the direction of and under the responsibility of the design authority.	
Design Authority	The design authority is the organization responsible for establishing the design requirements and ensuring that design output documents accurately reflect the design requirements. The design authority is responsible for the design control and ultimate technical adequacy of the design processes.	



Term	Definition	Reference
Design Bases	Design bases are statements that refer to design requirements for structures, systems, and components and identify why the requirement exists, why it is specified in a particular manner, and why a specified value is used. The design bases provide information that identifies the specific functions performed by the structures, systems, and components of a facility and the specified range of values chosen for controlling the parameters that are the referenced boundaries for the design of the structures, systems, and components.	
10 CFR 63.2 Design Bases	That 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 constraints derived from generally accepted state-of-the-art practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a SSC must meet its functional goals. The values for controlling parameters for external events include: <ul style="list-style-type: none"> <li>(1) Estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved</li> <li>(2) Estimates of severe external human-induced events, to be used for deriving design bases that will be based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.</li> </ul>	10 CFR 63.2
Design Criteria	Design criteria consist of the standards, codes, laws, regulations, general discipline design criteria, event sequences, and hazards that shall be used as a basis for acceptance of design for structures, systems, and components to satisfy requirements.	10 CFR 63.112(f)
Design Input	Design inputs shall be defined as the design requirements, supporting design bases, applied design criteria, and any other design parameters, conditions, boundaries, limits, or values used to develop and complete design configuration(s) and design output documents.	DOE 2000, ASME NOG-1-1995
Design Output	Design outputs shall be defined as the drawings, specifications, and other design documents prepared to present the design configuration(s) of structures, systems and components that satisfy design inputs.	DOE 2000, ASME NOG-1-1995
Design Requirement	Detail design requirements are engineering technical requirements (determined by design processes) that define, for example, the functions, capabilities, capacities, physical size, configurations, dimensions, performance parameters, limits, and setpoints, that are developed and specified by the design authority for structures, systems, and components to satisfy the mission design input requirements. The detail design requirements are the result (often iterative) of design processes.  (Example: Lateral load resisting systems elements for surface structure will be designed to withstand 100 mph wind loads.)	

Term	Definition	Reference
Design Verification	Documented, traceable measures (e.g., design review, alternate calculation, and qualification testing) applied to a design package or technical output by qualified individuals or groups other than those who performed the original design work. These measures verify the technical validity, adequacy, and completeness of a design package or technical output in context with the total design, natural or engineered barrier system, or integrated technical work.	AP-3.20Q
Disposal	The emplacement of radioactive waste in a geologic repository with the intent of leaving it there permanently.	10 CFR 63.2
	The emplacement of radioactive material into the Yucca Mountain disposal system with the intent of isolating it for as long as reasonably possible and with no intent of recovering the material. Disposal of radioactive material in the Yucca Mountain disposal system begins when the ramps and other opening into the Yucca Mountain repository are sealed.	10 CFR 63.302
Documentary Material	<p>(1) Any information that a party, potential party, or interested governmental participant intends to use or to cite in support of their position in the proceeding for a license to receive and possess high-level radioactive waste at a geologic repository operations area pursuant to 10 CFR Part 60 or 10 CFR Part 63.</p> <p>(2) Any information that is known to, in the possession of, or developed by the party that is relevant to, but does not support, that information or that party's position.</p> <p>(3) Reports and studies, prepared by or on behalf of the potential party, interested governmental participant, or party, including related circulated drafts, relevant to the license application and the issues set forth in Regulatory Guide 3.69, Topical Guidelines, regardless of whether they will be relied upon or cited by a party. The scope of documentary material will be guided by the topical guidelines in the applicable U.S. Nuclear Regulatory Commission Regulatory Guides.</p>	10 CFR 2.1001
Electrical One-Line Diagrams	Electrical One Line Diagrams are diagrams of single lines showing the electrical power sources, distribution busses, major loads, and associated circuit breakers. Electrical one-line diagrams may be generated based on the general system description and the design information required to perform the safety analyses, such that no additional supporting information will be required.	
Evaluation	<p>(1) To examine and judge carefully.</p> <p>(2) To form an opinion about.</p> <p>(3) To determine the significance, worth, or condition, usually by careful appraisal and study.</p>	
Event Sequence	A series of actions or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to radiation exposure. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. Those event sequences that are expected to occur one or more times before permanent closure of repository are referred to as Category 1 event sequences. Event sequences that have at least one chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences.	10 CFR 63.2

Term	Definition	Reference
General Arrangement Drawings	General arrangement drawings provide an overall view of a structure, component, or area showing the arrangement of major structural features and major equipment. Only overall dimensions are included. General arrangement drawings may be generated based on the general system description and the design information required to perform the safety analyses such that no additional supporting information will be required.	
General System Description	A general system description provides a summary of the system functions, operations, the system design, concept of operations, and a description of system interfaces, such as in Section 1 of the System Description Documents. This description should include a discussion on any special construction or fabrication techniques, unique testing programs or special design and analysis procedures used for the structures, systems, and components, as applicable.	
Geologic Repository Operations Area	A high-level radioactive waste facility that is part of a geologic repository, including surface and subsurface areas, where waste handling activities are conducted.	10 CFR 63.2
Handling Diagrams	Handling diagrams depict major handling paths and sequence of operations at a summary level (e.g., fuel movement in the Waste Handling Building). Handling diagrams may be generated based on the general system description and the design information required to perform the safety analyses, such that no additional supporting information will be required.	
Important to Safety	With reference to structures, systems, and components, engineered features of the geologic repository operations area, that: <ul style="list-style-type: none"> <li>(1) Provide reasonable assurance that high-level radioactive waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of 10 CFR 63.111(b)(1) for Category 1 event sequences</li> <li>(2) Prevent or mitigate Category 2 event sequences that could result in doses exceeding the values specified 10 CFR 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site.</li> </ul>	10 CFR 63.2
Initiating Event	A natural or human induced event that causes an event sequence.	10 CFR 63.2
Licensing Basis	The currently effective requirements imposed on the facility including the requirements at the time the initial license was applied for and granted, together with requirements subsequently imposed. The licensing bases are contained in U.S. Nuclear Regulatory Commission regulations, orders, license conditions, exemptions, and licensee commitments contained in the safety analysis report and other docketed licensee correspondence.	
Margin	Margin is the difference between the calculated event sequence dose and the prescribed regulatory compliance limit, which provides confidence that the repository design features can adequately protect public health and safety and the environment from any uncontrolled radiological event.	
Model	A representation of a system, process, or phenomenon, along with any hypotheses required to describe the process or system or to explain the phenomenon, often mathematically.	DOE 2000
	Model development typically progresses from conceptual to mathematical models. Mathematical model development typically progresses from process, to abstraction, and to system models.	AP-SIII.10Q
Negligible	So small, unimportant, or of so little consequence as to warrant little or no attention.	

Term	Definition	Reference
Piping and Instrumentation Diagrams (Process and Instrumentation Diagrams)	Piping and instrumentation diagrams are diagrams showing only major flow paths, equipment, and instrumentation (e.g., pumps, tanks, ion exchangers, major valves, and instrumentation used for operation). Interfaces with other systems and seismic and quality interfaces shall be included on the piping and instrumentation diagrams. These piping and instrumentation diagrams may be generated based on the general system description and the design information required to perform the safety analyses, such that no additional supporting information will be required.	
Permanent Closure	The final backfilling of the underground facility, if appropriate, and the sealing of shafts, ramps, and boreholes.	10 CFR 63.2
Postclosure	Refers to the period of time after permanent closure of the repository system.	
Preclosure	Refers to the period of time before and during permanent closure of the repository system.	
Preclosure Safety Analysis	A systematic examination of the site; the design; and the potential hazards, initiating events, and event sequences; and their consequences (e.g., radiological exposures to workers and the public). The analysis identifies structures, systems, and components important to safety.	10 CFR 63.2
Qualification (Personnel)	The capabilities gained through education, training, or experience that qualify an individual to perform a required function.	DOE 2000
Quality Level 1	[Later]	
Quality Level 2	[Later]	
Quality Level 3	[Later]	
Reasonable Assurance	The test of compliance with the standards and criteria. This concept recognizes that absolute assurance of compliance is neither possible nor required.	Eisenberg et al. 1999
Regulatory Commitment	An explicit statement made to ensure compliance, agreed to or volunteered by the U.S. Department of Energy, to take a specific action. Regulatory commitments are made in written correspondence with the U.S. Nuclear Regulatory Commission or the U.S. Environmental Protection Agency.	AP-REG-005, Sections 3.1 and 3.10
Regulatory Limit	Limit specified by U.S. Nuclear Regulatory Commission regulations or other regulatory requirements document (e.g., SRP, Regulatory Guides, and NUREGs). Whether the value is a maximum or a minimum depends on the type of variable being discussed.	
Regulatory Margin	Regulatory margin is the difference between the event sequence dose and the prescribed regulatory compliance limit.	
Retrieval	The act of permanently removing radioactive waste from the underground location at which the waste had been previously emplaced for disposal.	10 CFR 63.2
Risk	(1) The probability that an undesirable event will occur, multiplied by the consequences of the undesirable event. (2) Expected (mean) value of the consequences of an undesirable process or event.	DOE 2001
Risk Informed	An approach to regulatory decision-making whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operation issues commensurate with their importance to public health and safety.	NRC 1998
Risk Insights	The results and findings that come from risk assessments.	NRC 1998

Term	Definition	Reference
Safety Case	The logic, analyses, and calculations that show that the repository system would meet performance objectives.	CRWMS M&O 2001
Significant	More than a minimal increase in the consequences of an accident (e.g., the margin to the regulatory limit is eroded by more than ten percent).	NEI 2000
Technical Information	Information available from drawings, specifications, calculations, analyses, reactor operational records, fabrication records, construction records, other design basis documents, regulatory or program requirements documents, or consensus codes and standards that describe physical, performance, operational, or nuclear characteristics or requirements.	Document Action Request D813 to the <i>Quality Assurance Requirements and Description</i> document (DOE 2000)
Total Effective Dose Equivalent	For purposes of assessing doses to workers, the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). For purposes of assessing doses to members of the public (including the reasonably maximally exposed individual), total effective dose equivalent means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).	10 CFR 63.2
Traceability	The ability to trace the history, application, or location of an item, data, or sample using recorded documentation.  Traceability exists when there is an unbroken chain linking the result of an assessment (e.g., final dose calculation) with models, assumptions, expert opinions, and data used in the formulation of the result.	DOE 2000
Transparent	A document (e.g., a calculation, analysis, or model) is transparent if it is sufficiently detailed as to purpose, method, assumptions, inputs, conclusions, references, and units such that a person technically qualified in the subject can understand the document and ensure its adequacy without recourse to the originator.	DOE 2000, Section 3.2.2
Uncertainty	The interval above and below the measurement, parameter, or result that contains the true value at a given confidence level.	AP-SIII.9Q
	There are two types of uncertainty: 1) Stochastic (or aleatory) uncertainty is caused by the random variability in a process or phenomenon 2) State-of-knowledge (or epistemic) uncertainty results from a lack of complete information about physical phenomena.  State-of-knowledge uncertainty is further divided into: i) Parameter uncertainty, which results from imperfect knowledge about the inputs to analytical models ii) Model uncertainty, which is caused by imperfect models of physical systems, resulting from simplifying assumptions or an incomplete identification of the system modeled iii) Completeness uncertainty, which refers to the uncertainty as to whether the important physical phenomena, relationships (coupling), and events have been considered.	Eisenberg et al. 1999
Unrestricted Area	Areas where access is neither limited nor controlled by the licensee.	10 CFR 63.2

Term	Definition	Reference
Validation	A process used to establish confidence that a conceptual model (as represented in a mathematical model, software, or analysis) adequately represents the phenomenon, process, or system in question.	DOE 2000
	A process carried out by comparison of model predictions with field observations and experimental measurements. A model is considered validated when sufficient testing has been performed to ensure an acceptable level of predictive accuracy over the range of conditions over which the model may be applied.	Eisenberg et al. 1999
Verification	The act of reviewing, inspecting, testing, checking, auditing, or otherwise determining and documenting whether items, processes, services, or documents conform to specified requirements.	DOE 2000
	A process of assuring that the implementation of a mathematical model (in the form of a computer code) is free of coding errors, and that the numerical schemes used are within the bounds of required accuracy. The process consists of following established Quality Assurance procedures during the development of the code, comparison of the code with analytic solutions, and comparison with results from other codes.	Eisenberg et al. 1999
Waste Form	The radioactive waste materials and any encapsulating or stabilizing matrix.	10 CFR 63.2

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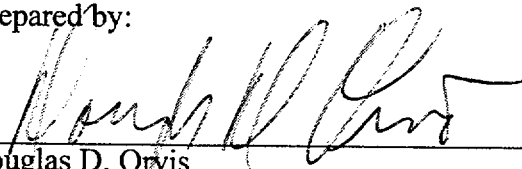
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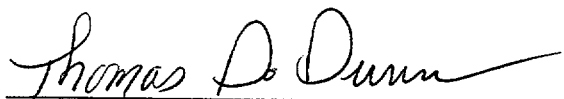
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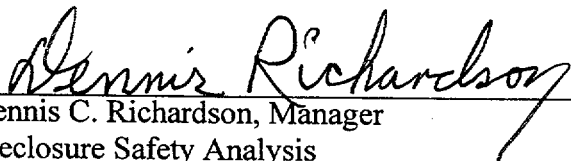
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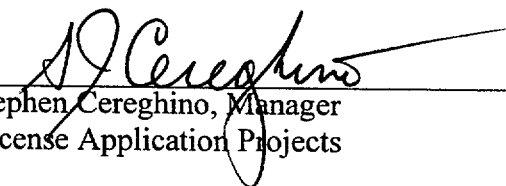
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