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Incident Response Division

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of Operational Data

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Before You Begin

This manual contains simple methods for estimating the possible consequences of different kinds of radiological accidents. The resulting estimates will help officials determine or confirm where to recommend protective actions to the public. These methods should be used only by trained personnel who can interpret the calculations, table, and figures in this document.

There are two objectives for the use of this manual:

- To prevent early health effects (deaths and injuries) by
  - taking action before or shortly after a major (core damage) release from a light water reactor or nuclear material accident and
  - keeping the acute dose equivalent (due to both external exposure and inhalation) below the early health effects thresholds.
- To reduce the risk of delayed effects on health (primarily cancer and genetic effects) by implementing protective actions in accordance with EPA
  - early phase Protective Action Guides (PAGs) and
  - intermediate phase PAGs (both ingestion and relocation concerns).

This manual contains methods (procedures) used to perform the assessments necessary to meet the public protection objectives. The methods are in the approximate order that assessments will be performed and are located at the front of the manual. Later sections contain the tables, figures, worksheets, and reference material that are necessary or useful in performing the procedures.

Each assessment section contains at least one stand-alone procedure. The more complicated assessments involve the use of more than one method (a separate procedure) to complete the assessment. Each method is organized into purpose, discussion, and steps. The discussion may provide a summary of the steps, assumptions, cautions, and other relevant information. Method steps provide the instructions for the procedure.

This manual is conservative; that is, the results should overestimate the dose or result in actions at levels below those recommended in the guidance. *Do not add additional conservatism* (e.g., do not divide a guideline by 10 just to be safe). Adding additional conservatism will cause confusion and will make it difficult to compare the risk of the action to the risk avoided.

This manual is intended to be consistent with the guidance in the U.S. Environmental Protection Agency's May 1992 *Manual of Protective Action Guides and Protective Actions For Nuclear Incidents* (EPA 400-R-92-001).

The following suggestions will help in getting the best use of this manual.

- Use one of the charts in the following overview to guide the initial assessment.
- Read the remainder of this introduction and the discussion at the beginning of each section before performing assessments.
- Read each method completely before applying it.

Overview of Assessment Process

The following discussion and flow charts present an overview of the basic tasks in assessing a light water reactor (LWR) accident or a generic non-LWR accident. These charts can provide a starting point for the assessment.

LWR Accident Assessment

LWR accidents posing the greatest risk involve core damage and a prompt (within 24 h) release. Releases that pose the greatest risk will most likely be those that occur through an unmonitored pathway. The most effective protective action for a core damage accident is to evacuate near the plant (2-5 miles) before or shortly after the start of a release. Protective actions should be taken based on core conditions. Do not wait until a release is confirmed. The status of the containment is not normally considered because containment failure (leakage) is unpredictable under severe core damage conditions.

The basic assessment strategy for a severe (core damage) LWR accident is divided into three nonexclusive time periods:

Immediate and ongoing actions

If there is actual or projected severe core damage, recommend protective actions close to the site. The goal is to take protective actions before or shortly after the start of a release.

In plume (during a release)

Adjust the protective actions taken based on plant conditions and any results from monitoring in the plume. Estimate the inhalation dose to the public and emergency workers in the plume using dose rates and inhalation dose to dose-rate ratios. Note

that the thyroid dose from inhalation can be a hundred or more times higher than the dose from external exposure.

After plume passage

- Locate and evacuate areas with high deposition dose rates [hot spots, >500 mrem/h (early health effects) and >10 mrem/h (evacuation PAGs)]. The principal source of early and late phase dose after plume passage will be from the external exposure to material deposited on the ground. Inhalation dose from resuspension should not be a concern for LWR accidents.

- Locate areas where deposition dose rates will result in doses in excess of the intermediate phase relocation PAGs and relocate the people in those areas.

- Identify areas, based on deposition exposure rates and isotope concentrations, where ingestion may be a concern. Confirm where ingestion is of concern based on analysis of food, water and milk from the suspect areas.

Chart 1 shows the order in which the assessment should be performed for a LWR accident.

Generic Accident Assessment (for Non-LWR Accidents)

For an accident involving contamination (dispersion of radioactive material), the isotopic "mix" or composition of the definition must be identified in order to determine appropriate protective actions, establish emergency worker turn back limits, or assess environmental data. For some isotopes (e.g., Pu), inhalation doses (plume and resuspension of deposited material) could pose a threat of early health effects, even though the external exposure rates may be very low.

Chart 2 shows the order in which the assessment should be performed for a non-LWR accident.

Chart 1. LWR accident assessment tasks

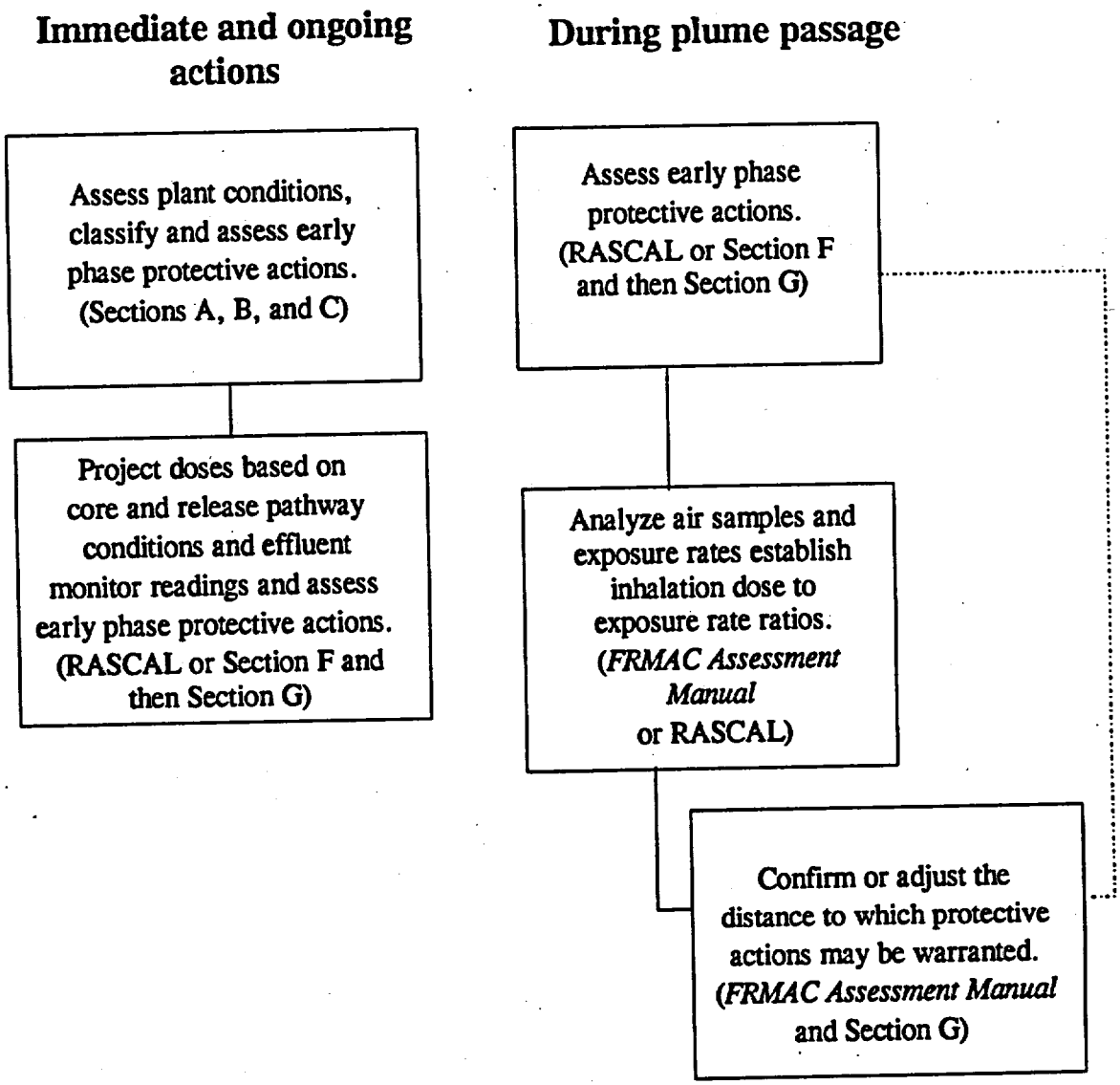


Chart 1. LWR accident assessment tasks (continued)

After plume passage

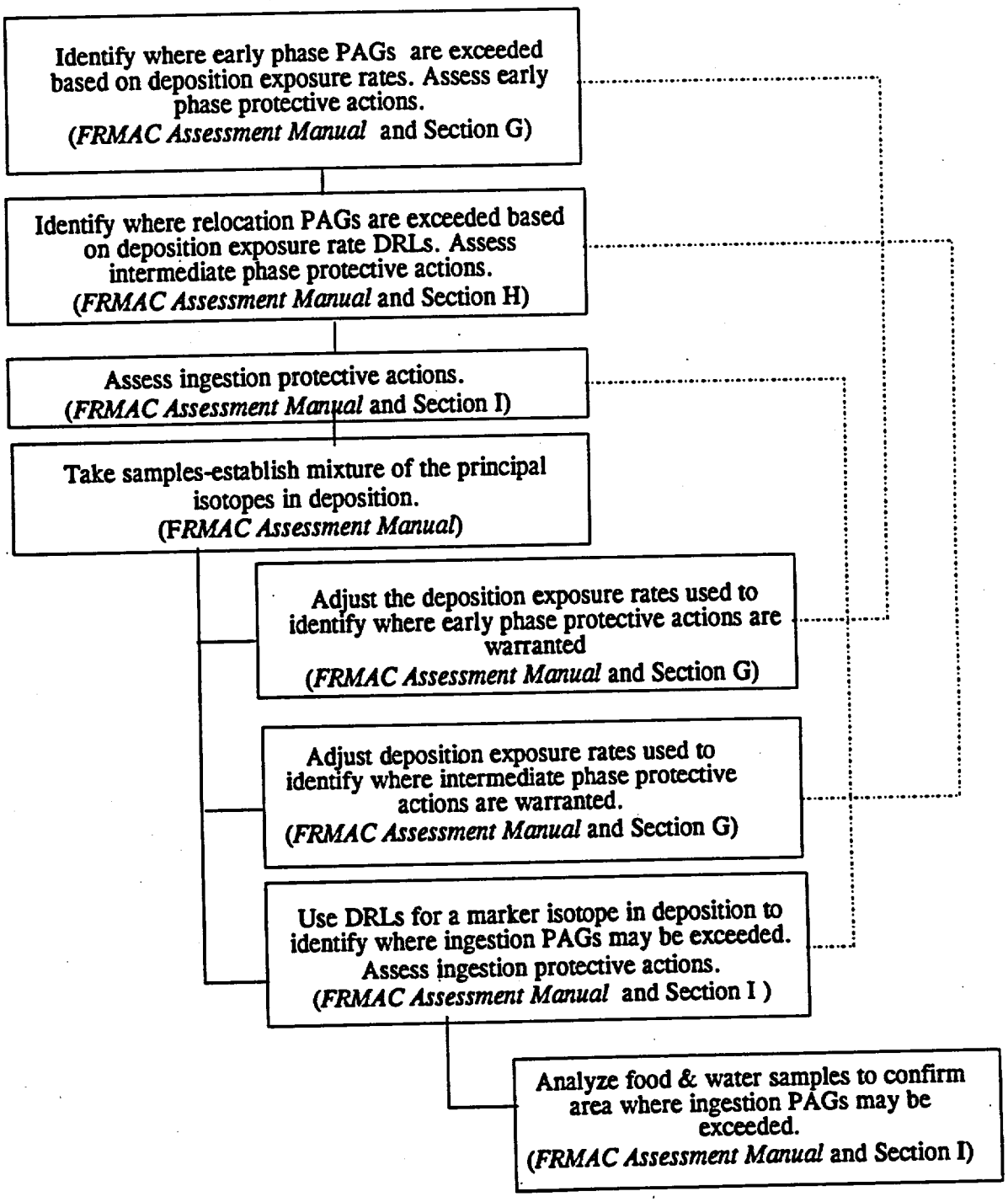


Chart 2. Generic accident assessment tasks (non-LWR accident)

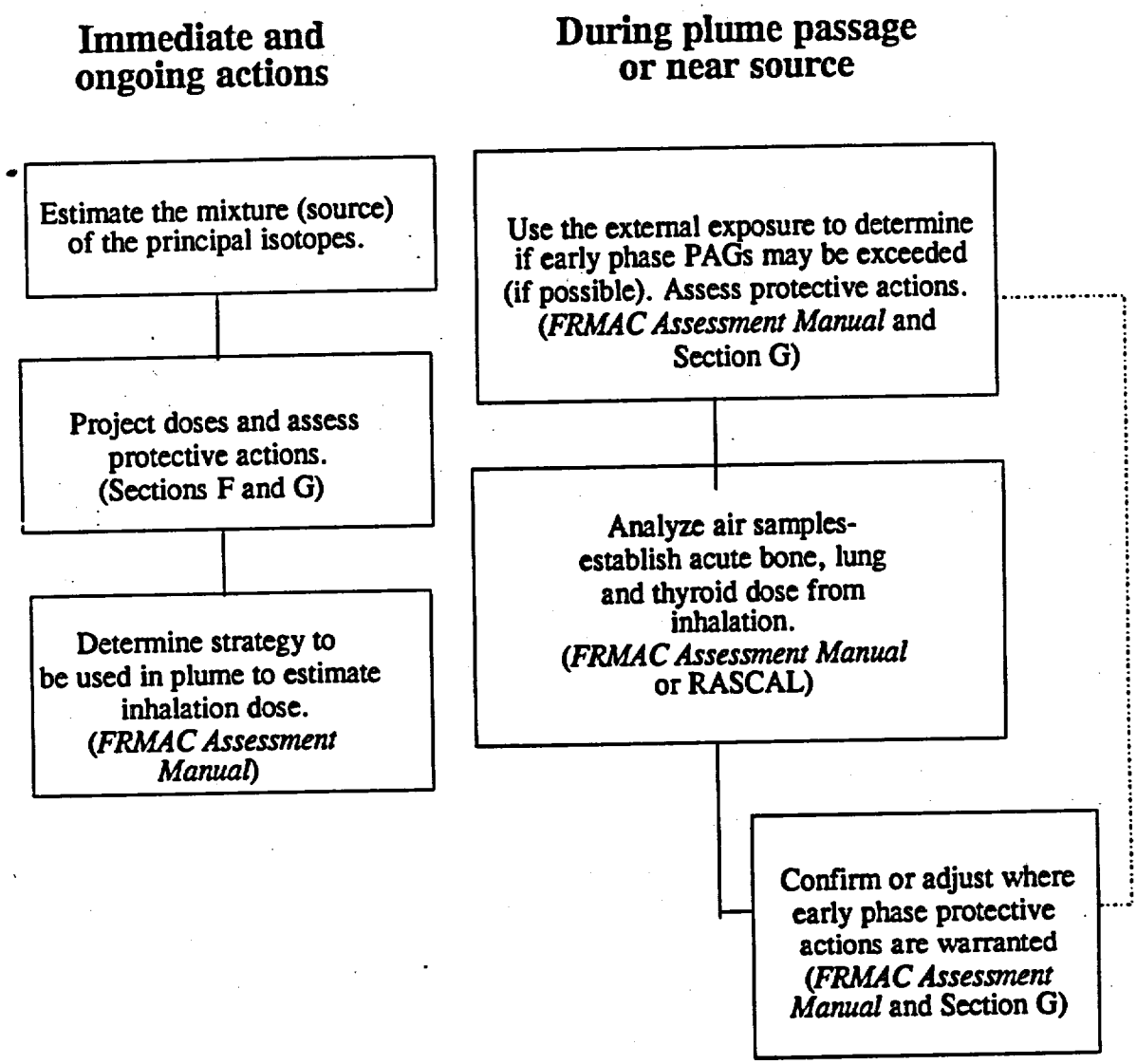
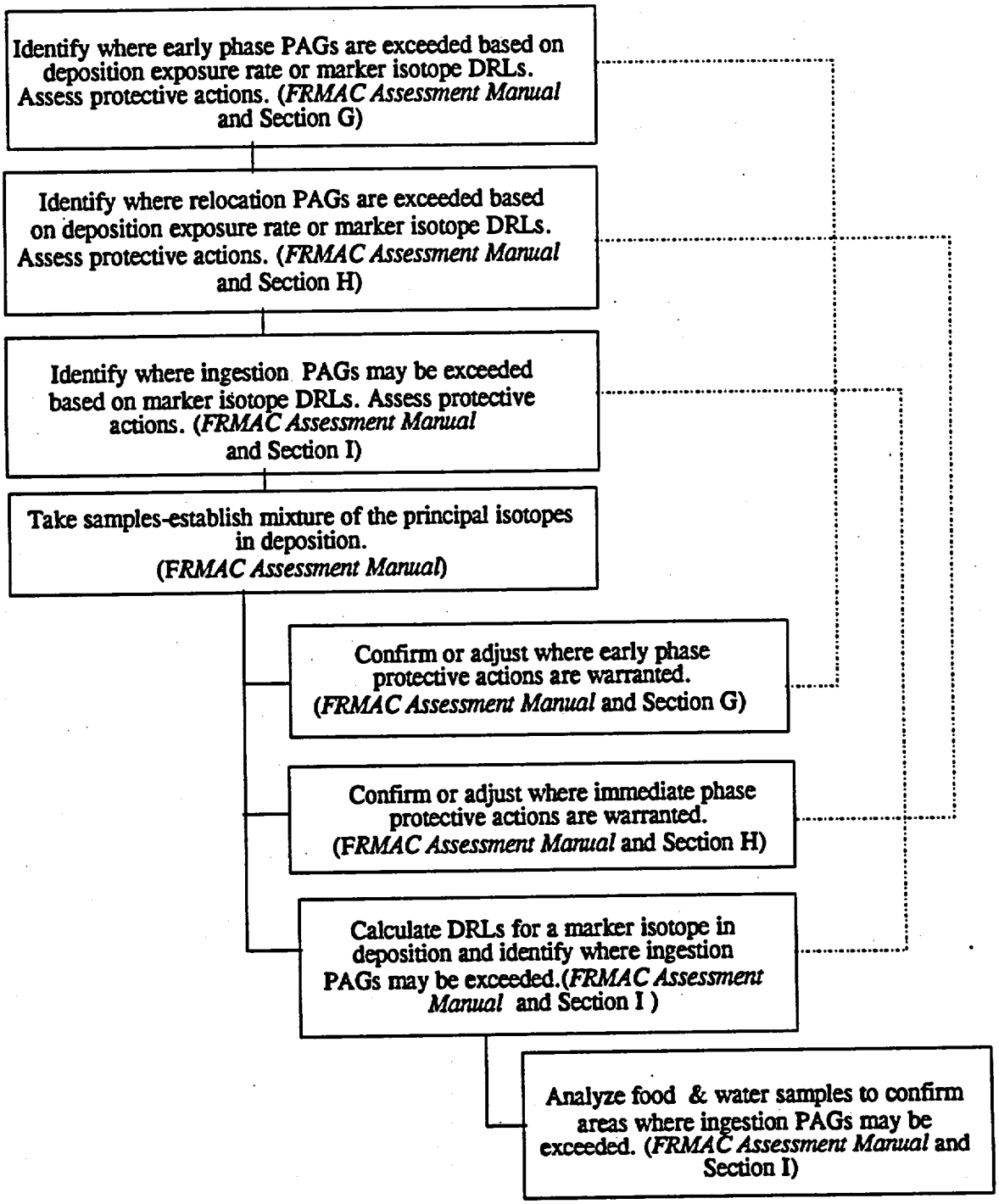


Chart 2. Generic accident assessment tasks (non-LWR accident) (continued)

After plume passage



A

SECTION A

REACTOR CORE DAMAGE ASSESSMENT

Section A
Quick Reference Guide

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Section A
Reactor Core Damage Assessment

Purpose

To assess the condition of a light water reactor core for use in classifying an accident (Section B), projecting possible consequences (Section C), and determining early phase protective actions (Section G).

Discussion

In assessing core conditions, do not lose sight of the big picture! Never use just one instrument as the basis for your assessment.

Core damage assessment is a continual process. The steps in this process are listed below in the approximate order that the needed information might be available. After completing any method in this section, the assessor must continue to monitor the core status for changes and must update core damage assessments for others performing related assessments.

The steps in this assessment are summarized below:

Step 1	Assess the status of critical safety functions for indications that the core may already be uncovered.
Step 2	Monitor for indications that the core may soon become uncovered.
Step 3	Project core damage if uncovered and inform those assessing consequences, classification, and protective actions.
Step 4	Monitor radiation levels to confirm and assess core damage.
Step 5	Continue to assess core damage.

Step 1

Assess the current status of the critical safety functions by answering the following questions. If any of the critical safety functions are not being met or are degraded, estimate when the core may be uncovered. If the core is projected to be uncovered, perform Steps 2 and 3.

- Is the plant subcritical (shutdown)? How is this confirmed?
- Is the core covered now and will it be in the long term? How is this confirmed?

- Is the amount of water being injected into the primary or secondary system sufficient to remove the decay heat? Use Method A.1 to confirm that there is sufficient injection of water.
- Is decay heat being removed to the environment? How is this confirmed?
- What is the status of the vital auxiliaries? DC power? AC power?

Step 2

Monitor the following indications for detecting imminent uncovering of the reactor core. Consider the reliability of the indications or instrument readings during accident conditions as discussed below.

For PWR

Core exit thermocouple (CET) readings and primary cooling system pressure can be used to evaluate whether the core will be uncovered. A loss of sub-cooling margin (Method A.2) indicates that sufficient water injection is not being provided to keep the core covered. If the core is uncovered, the CET readings will continue to rise but will be considerably lower than the actual average and maximum core temperatures. CET readings are not accurate after core damage.

In-vessel water level indication system can also be used as an indicator of potential uncovering of the core. Decreasing water levels can confirm that there is insufficient water injection to keep the core covered. Water level indications should be used only to detect trends because of the considerable (up to 30%) uncertainties in the measurements during accident conditions. This system is not reliable after core damage.

For BWR

Water level can be used under some accident conditions to confirm that insufficient water is being injected to protect the core and to estimate the time at which the core will be uncovered.

Consider the following limitations:

- The lower limit of the water measurement system is at or above the level at which core heat-up begins (20% uncovered).
- High drywell temperature (e.g., LOCA) can cause the BWR reactor water level to read erroneously high.

- During low pressure accidents, the BWR water level can read erroneously high.
- Mechanical Yarway instruments may indicate a false on-scale water level at about 1 ft above the top of core if the actual water level fell below the lower end of the instrument range.

If there are indications of imminent uncovering of the core, go to Step 3. If not, provide an assessment of critical safety functions and core status to those assessing the emergency classification (Section B), assessing early phase protective actions (Section G), or projecting consequences (Section C). Continue to monitor plant indicators (Step 1 and Step 2).

Step 3

If core is projected to be uncovered (Step 1) or there are indications that this is imminent (Step 2), use Method A.3 to determine projected times for the following core damage states:

Time to gap release from fuel: _____ h

Time to in-vessel core melt: _____ h

and

provide an assessment of the critical safety functions and core status to those assessing the emergency classification (Section B), assess early phase protective actions (Section G), or project consequences (Section C). If actual or projected core damage is detected, the accident should be classified as a General Emergency and protective actions should be considered in accordance with Section G of this manual. *Do not wait* for core damage to be confirmed.

Step 4

Monitor the radiation levels to attempt initial confirmation of core damage. Detection of very large increases (orders of magnitude) in radiation levels by radiation monitors (e.g., containment) can confirm actual core damage. If the release is into the containment, use Method A.4 to assess the level of damage. Compare with core damage estimate from Step 3. The following possibilities should be considered:

- The release may bypass the monitor.
- Monitors may be influenced by a source not intended to be monitored.
- Areas monitored may not be representative of the entire containment.

- Calibration assumptions may not match accident conditions.
- Shielding or other design factors may have been incorrectly considered.
- Monitor may show high, low, or center range if it fails.
- Monitor may be read incorrectly.

Reassess the emergency classification (Section B), early phase protective actions (Section G), or consequences (Section C). If actual or projected core damage is detected, the accident should be classified as a General Emergency and protective actions should be considered in accordance with Section G.

Step 5

After the core is uncovered, continue to evaluate the amount of core damage using the available information. The following methods may be used:

Evaluate core once uncovered . . . . . Method A.3
Evaluate containment radiation . . . . . Method A.4
Evaluate coolant concentrations . . . . . Method A.5
Evaluate containment hydrogen . . . . . Method A.6

Reassess the emergency classification (Section B), early phase protective actions (Section G), or consequences (Section C). If actual or projected core damage is detected, the accident should be classified as a General Emergency and protective actions should be considered in accordance with Section G. Note that these methods for estimating core damage can be time-consuming and may be unreliable. *Do not delay protective actions* by waiting for confirmation of core damage.

END

Source: NUREG-1228

Method A.1
Evaluation of Water Injection

Purpose

To determine the amount of water that must be injected into a LWR core to replace the water lost by boiling resulting from decay heat.

Discussion

This method provides curves of the water injection rates required to remove decay heat *by boiling*. These curves are based on a 3000-MW(t) plant operated at a constant power for an infinite period and then shut down instantaneously. The decay heat power is based on ANSI/ANS-5.1. If the injected water is about 80°F (27°C), the curves are within 5% for pressures 14–2500 psia (0.1–17.2 MPa). The curves are valid within 20% for injected water temperatures up to 212°F (100°C).

If there is a break in a pipe requiring make-up water, more water than indicated here will be required to keep the core covered and cooled. If the core has been uncovered for an extended time (e.g., > 15–30 min), the fuel temperatures will have already increased significantly. In this case, additional injection water will be required to accommodate the heat from the Zr-H_2O reaction and allow the heat transfer necessary to return the fuel to equilibrium temperatures.

Step 1

Use Figs. A-1 and A-2 to determine the minimum amount of water that must be injected to replace water lost by boiling (resulting from decay heat) for a 3000-MW(t) plant based on the time since shutdown.

Time since reactor shutdown: _____ h or _____ days

Minimum required water injection: _____ gal/min

Step 2

Adjust this injection rate for the size of the plant using

$$inject_{plant} = inject_{3000} \times \frac{MW(t)_{plant}}{3000}$$

$$(\qquad \text{gal/min}) = (\qquad \text{gal/min}) \times \frac{[\qquad \text{MW(t)}]}{3000\ \text{MW(t)}}$$

where:

$inject_{3000}$	=	amount of injected water for 3000-MW(t) plant (from Fig. A-1 or A-2),
$MW(t)_{plant}$	=	size of plant in MW(t) [MW(t) ≅ 3 × MW(e)],
$inject_{plant}$	=	amount of injected water needed for this plant.

Step 3

If the core has been uncovered for 15-30 min or longer, increase the amount of water required to cool core by a factor of two to three.

END

Method A.2
Evaluation of Sub-Cooling Margin (Saturation Table)

Purpose

To determine if water at a given pressure and temperature is boiling and to calculate the sub-cooling margin. (This method is only useful for a PWR with primary system pressure and temperature instruments.)

Discussion

The sub-cooling margin can be approximated by subtracting the coolant temperature from the saturation temperature for the given primary system pressure. A coolant temperature taken from the core exit thermocouple reading greater than the saturation temperature indicates that the water in the core is boiling.

Step 1

Record the primary system pressure.

_____ psia or _____MPa

Step 2

Record the primary coolant temperature ($temp_{PWR}$) from the core exit thermocouple.

_____ °F or _____°C

Step 3

Using Table A-1, determine the saturation temperature ($temp_{sat}$) for the primary system pressure recorded above.

_____ °F or _____ °C

Step 4

Determine the sub-cooling margin using the following equation:

$$sub\text{-}cooling\ margin = temp_{sat} - temp_{PWR}$$

A negative sub-cooling margin in a PWR indicates that water is boiling in the reactor vessel and that the core may be uncovered.

END

Method A.3
Evaluation of Core Once Uncovered

Purpose

To estimate LWR core temperature and damage progression once the core is uncovered.

Discussion

A severely damaged core may not be in a coolable state, even if it is re-covered with water. Core temperature (core exit thermocouple) and primary system water temperature (delta T) indications cannot confirm a coolable core.

It can be assumed that the fuel in the core will heat up at 1–2°F/s (0.5–1.0°C/s) immediately after the top of an active core of a PWR is uncovered or 5–10 min after the top of an active core of a BWR is uncovered. These fuel heatup estimates are reasonable within a factor of two if the core is uncovered within a few hours of shutdown (including failure to scram) for a boil-down case (without injection). If there is injection, core heatup may be stopped or slowed because of steam cooling. However, steam cooling may not prevent core damage under accident conditions.

Step 1

Use Table A-2 to estimate or project the level of core damage based on the time the core is uncovered. Estimate average fuel temperature by assuming an increase of 1–2°F/s (0.5–1°C/s) once the core is uncovered.

END

Method A.4
Evaluation of Containment Radiation

Purpose

To assess the core damage based on the containment radiation monitor readings.

Discussion

This method uses containment radiation monitor readings to assess core damage; however, containment radiation monitor readings *cannot* confirm core damage in all cases. The release may bypass the containment, be retained in the primary system, be released over a long period of time, or not be uniformly mixed. Therefore, a low containment radiation reading does not guarantee a lack of core damage.

Confirm that the containment radiation monitor "sees" more than 50% of the shaded area shown in either Fig. A-3 (PWR) or Fig. A-4 (BWR). If not, this method should not be used to assess core damage.

These calculations should provide the maximum reading expected under the conditions stated. The calculations assume (1) a prompt release to containment of all the fission products in the coolant, spike, gap, or from in-vessel core melt; (2) uniform mixing in the containment; and (3) an unshielded monitor that can see most of the area shown in Fig. A-3 or Fig. A-4. Because the mix is most likely different from that assumed in the calibration of the monitor, the actual reading at the upper end of the scale could differ by a factor of 10–100 if a shielded detector is used for the higher radiation measurements.

The levels of damage indicated on Figs. A-5–A-12 should be considered minimum levels unless there are inconsistent monitor readings. Inconsistent readings may be caused by the uneven mixing in containment [e.g., steam rising to top of dome, not enough time for uniform mixing to occur (it may take hours)]. The values in the figures were generated using CONDOS II (NUREG/CR-2068).

Four types of releases are considered:

In-vessel core melt release—the release into containment of all the fission products expected to be released from a core that is partially melted (see Table C-4) after being uncovered for 30 min or more.

Gap release—the release into containment of all the fission products in the fuel pin gap (see Table C-4) after the fuel cladding has failed from being uncovered for more than 15 min.

Spiked coolant release—the release into containment of 100 times the non-noble gas fission products normally found in the coolant.

Typical (normal) coolant release—the release into containment of the fission products normally found in the coolant (see Tables C-2 and C-3).

Step 1

Record the following readings:

Normal radiation monitor reading: _____ R/h

Unshielded monitor reading: _____ R/h

Time of reading after release into containment: _____ h

Sprays: _____ on *or* _____ off

Step 2

Determine the absolute containment radiation rate by subtracting the radiation monitor reading during normal operations from the unshielded monitor reading after the accident.

absolute radiation rate = unshielded monitor reading − normal radiation monitor reading

(R/h) = (R/h) − (R/h)

Step 3

Estimate the core damage based on the absolute containment radiation rate calculated above, using the appropriate figure from the following list. The figures show the range of containment monitor readings assuming that the fission products associated with 1–100% of the level of core damage stated. It is assumed the release from the core is uniformly mixed in the containment and that the monitor is unshielded. Sprays are assumed to remove non-nobles to a location where the monitor cannot see them.

PWR (sprays on) . . . . . Fig. A-5
PWR (sprays off) . . . . . Fig. A-6
BWR Mark I & II dry well (sprays on) . . . . . Fig. A-7
BWR Mark I & II dry well (sprays off) . . . . . Fig. A-8
BWR Mark I & II wet well . . . . . Fig. A-9
BWR Mark III dry well (sprays on) . . . . . Fig. A-10
BWR Mark III dry well (sprays off) . . . . . Fig. A-11
BWR Mark III wet well . . . . . Fig. A-12

END

Source: NUREG/CR-5157 was used to confirm the core melt numbers.

Method A.5
Evaluation of Coolant Concentrations

Purpose

To assess the LWR core damage based on a coolant sample.

Discussion

Coolant concentrations should not be required to confirm core damage because they may take hours to draw and analyze and may not be representative of primary system concentrations (e.g., no flow through sample line).

This method of confirming core damage assumes that releases from the core are uniformly mixed in the coolant and that there is no dilution from injection. The baseline coolant concentrations are for 0.5 h after shutdown of a core that has been through at least one refueling cycle. The half-life of the fission products should be considered in analyzing samples. The plant-specific coolant system volume does not have a major influence on coolant concentrations (<20%).

For a BWR, it is assumed that the release from the core is uniformly mixed in the reactor coolant system and suppression pool. If most of the core release is confined to the reactor coolant system, the concentrations in the coolant could be up to 10 times higher.

Step 1

For PWRs and BWRs, compare the reported coolant concentrations with the baseline coolant concentrations in Table A-3 or Table A-4. These tables will overestimate the concentrations for the long-lived fission products (Cs and Sr) in a *new* core.

For other LWRs that have primary system coolant inventories considerably different than those assumed in Table A-3 (2.5×10^5 kg), adjust the Table A-3 baseline concentrations by multiplying each value by

$$\frac{2.5 \times 10^5 \text{ kg}}{\textit{reactor coolant inventory (kg)}}.$$

END

Method A.6
Evaluation of Containment Hydrogen

Purpose

To assess the core damage based on hydrogen concentrations in containment samples.

This method may be used to assess the core damage based on hydrogen concentrations in samples of the containment atmosphere. Hydrogen concentrations should not be relied upon to confirm core damage in all cases. Containment samples may require hours to collect and analyze and may not be representative of the total hydrogen generated in the core because of incomplete mixing in the containment or containment bypass.

Discussion

The hydrogen concentrations used in this method are for wet samples; however, most hydrogen samples are dry (steam removed). If a dry sample concentration is used, one may overestimate considerably the level of core damage. This method assumes that all hydrogen is released to the containment and is completely mixed in the containment atmosphere. The curves in Fig. A-13 are a function of containment size. The results of severe accident research (research supporting NUREG-1150) were examined to identify the least percentage of metal-water reaction associated with each core damage state. Higher percentages of metal-water reaction are possible for some accident sequences (e.g., Three Mile Island).

Step 1

Obtain an estimate from the facility of the average hydrogen wet sample concentration in the containment.

Hydrogen percentage: _____ %

Step 2

Use the hydrogen percentage and Fig. A-13 to estimate the percentage of metal-water reaction and possible levels of core damage for the appropriate reactor containment

type. Any of the core conditions shown on the y-axis below this percentage may be possible.

Percentage metal-water reaction: _____ %

Possible levels of core damage:

END

Table A-1. Saturation table

Absolute pressure		Saturation temperature	
(psia)	(MPa)	(°F)	(°C)
14.70	0.10	212.0	100.00
15	0.10	213.0	100.57
20	0.14	228.0	108.87
30	0.21	250.3	121.30
40	0.28	267.3	130.69
50	0.34	281.0	138.34
60	0.41	292.7	144.84
70	0.48	302.9	150.52
80	0.55	312.0	155.58
90	0.62	320.3	160.16
100	0.69	327.8	164.34
110	0.76	334.8	168.22
120	0.83	341.3	171.82
130	0.90	347.3	175.18
140	0.97	353.0	178.36
150	1.03	358.4	181.35
160	1.10	363.6	184.19
170	1.17	368.4	186.90
180	1.24	373.1	189.49
190	1.31	377.5	191.96
200	1.38	381.8	194.33
210	1.45	385.9	196.62
220	1.52	389.9	198.82
230	1.59	393.7	200.94
240	1.65	397.4	202.99
250	1.72	401.0	204.98
260	1.79	404.4	206.91
270	1.86	407.8	208.78
280	1.93	411.1	210.59
290	2.00	414.3	212.36
300	2.07	417.4	214.08
350	2.41	431.7	222.07
400	2.76	444.6	229.22
450	3.10	456.3	235.71
500	3.45	467.0	241.67

Table A-1. Saturation table (continued)

Absolute pressure		Saturation temperature	
(psia)	(MPa)	(°F)	(°C)
550	3.79	476.9	247.19
600	4.14	486.2	252.33
650	4.48	494.9	257.16
700	4.83	503.1	261.71
750	5.17	510.8	266.02
800	5.52	518.2	270.12
850	5.86	525.2	274.02
900	6.21	532.0	277.75
950	6.55	538.4	281.33
1000	6.89	544.6	284.77
1050	7.24	550.5	288.07
1100	7.58	556.3	291.27
1150	7.93	561.8	294.34
1200	8.27	567.2	297.33
1250	8.62	572.4	300.21
1300	8.96	577.4	303.01
1350	9.31	582.3	305.73
1400	9.65	587.1	308.37
1450	10.00	591.7	310.94
1500	10.34	596.2	313.44
1550	10.69	600.6	315.88
1600	11.03	604.9	318.26
1650	11.38	609.1	320.58
1700	11.72	613.1	322.85
1750	12.07	617.1	325.07
1800	12.41	621.0	327.23
1850	12.76	624.8	329.35
1900	13.10	628.6	331.42
1950	13.44	632.2	333.46
2000	13.79	635.8	335.44
2100	14.48	642.8	339.31
2200	15.17	649.5	343.03
2300	15.86	655.9	346.61
2400	16.55	662.1	350.06
2500	17.24	668.1	353.39

Table A-1. Saturation table (continued)

Absolute pressure		Saturation temperature	
(psia)	(MPa)	(°F)	(°C)
2600	17.93	673.9	356.62
2700	18.62	679.5	359.74
2800	19.31	685.0	362.76
2900	19.99	690.2	365.68
3000	20.68	695.3	368.52
3100	21.37	700.3	371.27
3200	22.06	705.1	373.93
3208.2[a]	22.12[a]	705.5	374.15

[a]Critical temperature.

Source: ASME 1993, Table 2, pp. 187–193.

Table A-2. Core damage vs. time that reactor core is uncovered

Time PWR or 20% of BWR active core is uncovered (h)	Core temperature (°F)	Core temperature (°C)	Possible core damage
0	>600	>315	• None
0.5 to 0.75	1800–2400	980–1300	• Local fuel melting • Burning of cladding with steam production (exothermic Zr-H_2O reaction with rapid H_2 generation) • Rapid fuel cladding failure (gap release from the core[a])
0.5 to 1.5	2400–4200	1300–2300	• Rapid release of volatile fission products (in-vessel severe core damage release from core[a]) • Possible relocation (slump) of molten core • Possible uncoolable core
1 to 3+	>4200	>2300	• Melt-through of vessel with possible containment failure and release of additional less-volatile fission products

[a]Table C-4 contains the assumed core release fractions for this release.

Sources: NUREG/CR-4245, NUREG/CR-4624, NUREG/CR-4629, NUREG/CR-5374, NUREG-0900, NUREG-0956, NUREG-1150, and NUREG-1465.

Table A-3. PWR baseline coolant concentration

Nuclide	Normal concentration (μCi/g)	Concentration after gap release (μCi/g)	Concentration after in-vessel melt (μCi/g)	TMI concentration + 48 h (μCi/g)[a]
^{131}I	4×10^{-2}	2×10^{4}	1×10^{5}	1.3×10^{4}
^{133}I	1×10^{-1}	3×10^{4}	2×10^{5}	6.5×10^{3}
^{135}I	2×10^{-1}	3×10^{4}	2×10^{5}	-
^{134}Cs	7×10^{-3}	2×10^{3}	8×10^{3}	6.3×10^{1}
^{137}Cs	9×10^{-3}	9×10^{2}	5×10^{3}	2.8×10^{2}
^{140}Ba	NC[b]	NC	3×10^{4}	-
^{90}Sr	1×10^{-5}	NC	1×10^{4}	5.3×10^{0}

[a]TMI coolant concentrations 48 h after the accident.
[b]NC = not calculated (data not available).

Source: (Normal coolant) ANSI/ANS 18.1, 1984, confirmed by NUREG/CR-4397, Table 2.1; (TMI) NUREG-600.

Table A-4. BWR baseline coolant concentration

Nuclide	Normal concentration (μCi/g)	Concentration after gap release (μCi/g)[a]	Concentration after in-vessel melt (μCi/g)[a]
^{131}I	2×10^{-3}	1×10^{3}	1×10^{4}
^{133}I	1×10^{-2}	3×10^{3}	2×10^{4}
^{135}I	2×10^{-2}	2×10^{3}	2×10^{4}
^{134}Cs	3×10^{-5}	1×10^{2}	6×10^{2}
^{137}Cs	8×10^{-5}	8×10^{1}	4×10^{2}
^{140}Ba	NC[b]	NC	2×10^{3}
^{90}Sr	7×10^{-6}	NC	1×10^{3}

[a]In the reactor coolant system and suppression pool.
[b]NC = not calculated (data not available).

Source: (Normal coolant) ANSI/ANS 18.1, 1984, confirmed by NUREG/CR-4245, Table 3.2.

Fig. A-1
Injection required to replace water lost by boiling resulting from decay heat for a 3000 MW(t) plant (0.5–24 h after shutdown).

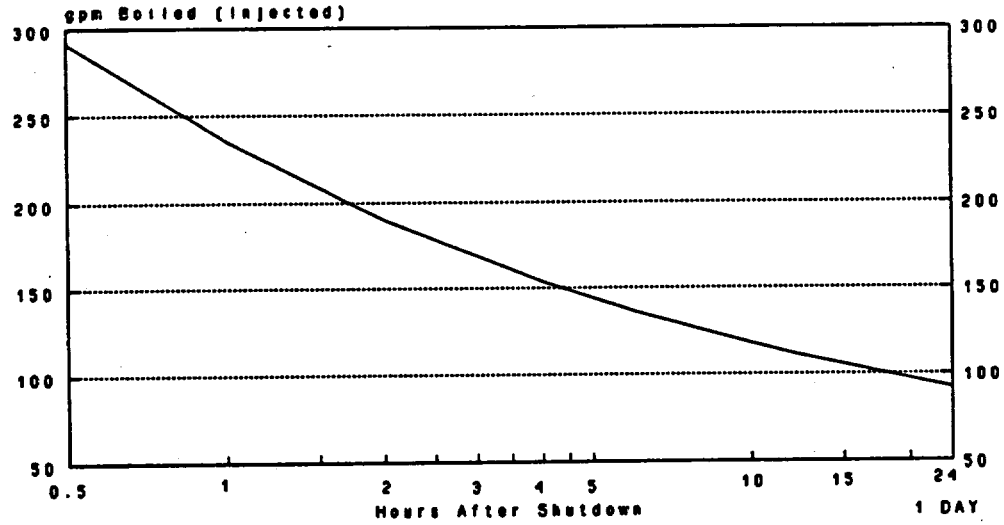


Fig. A-2
Injection required to replace water lost by boiling resulting from decay heat for a 3000 MW(t) plant (1–30 days after shutdown).

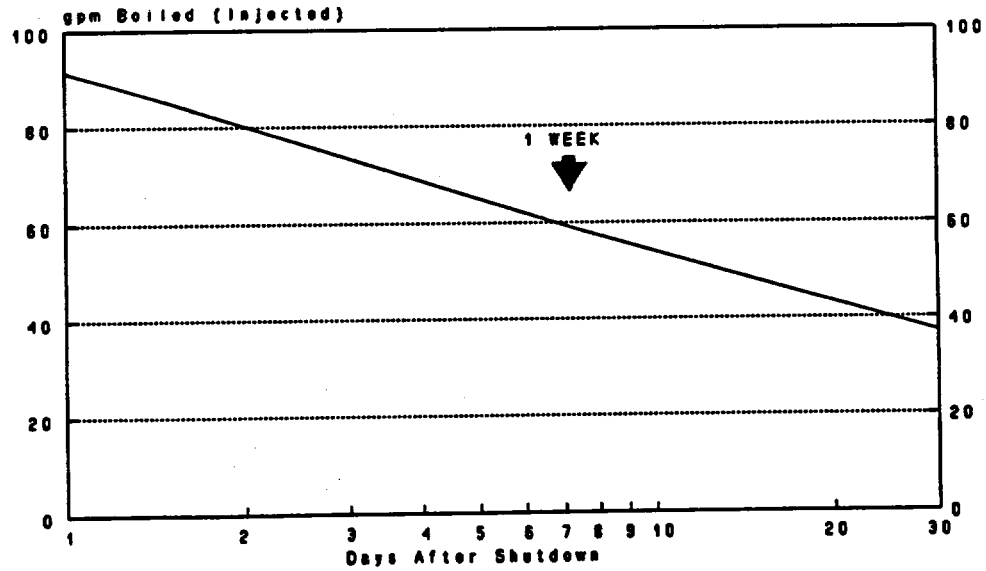


Fig. A-3
Areas assumed to be monitored in PWR containment.

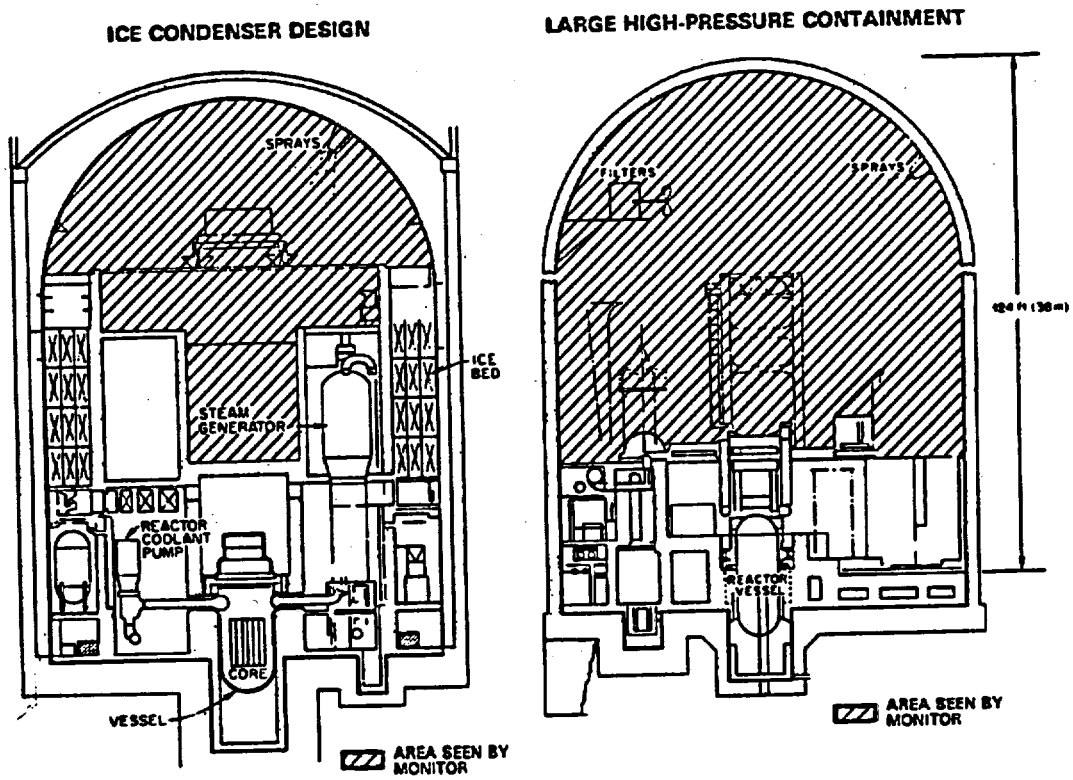


Fig. A-4
Areas assumed to be monitored in BWR containment.

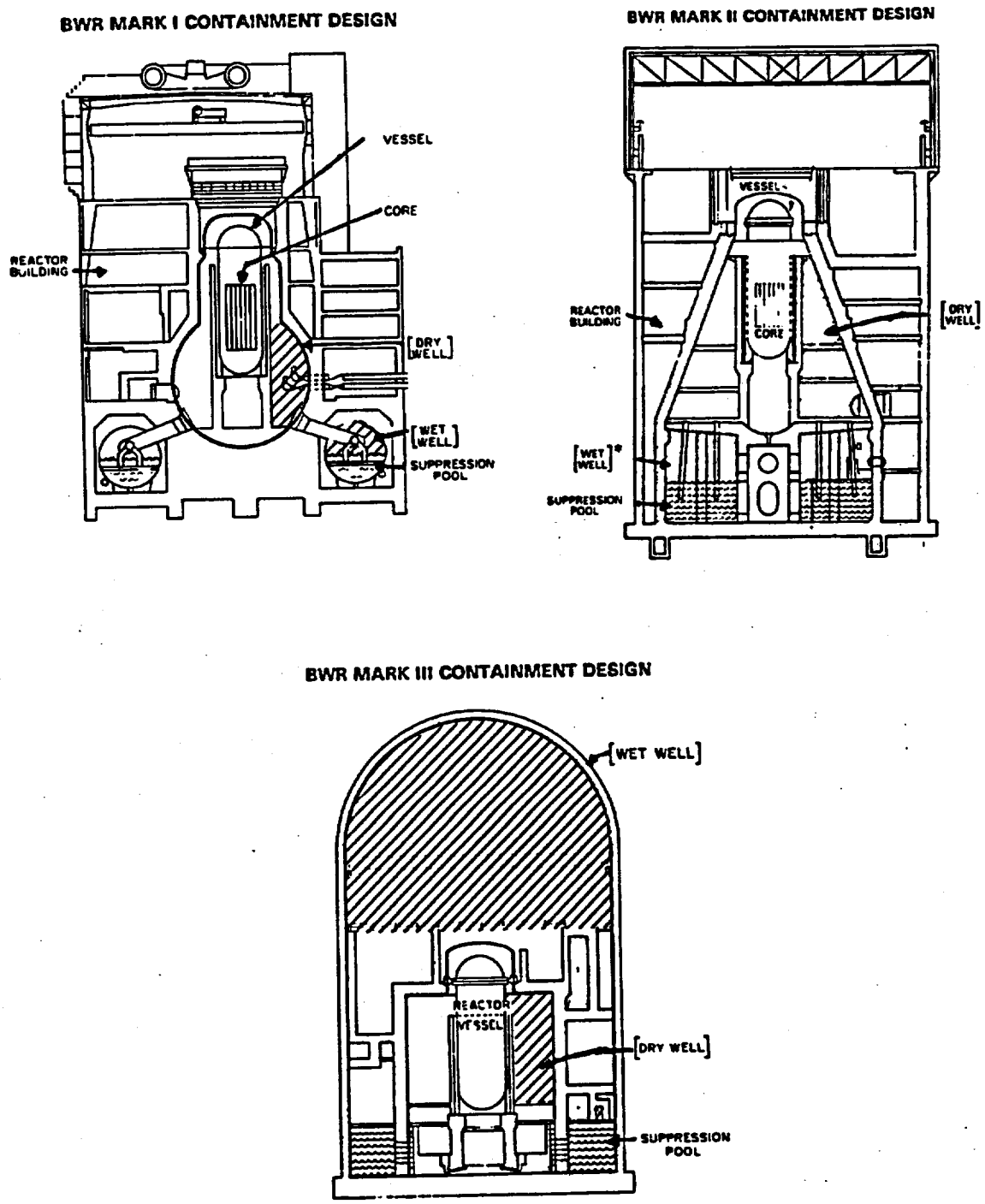


Fig. A-5
PWR containment monitor response (sprays on).

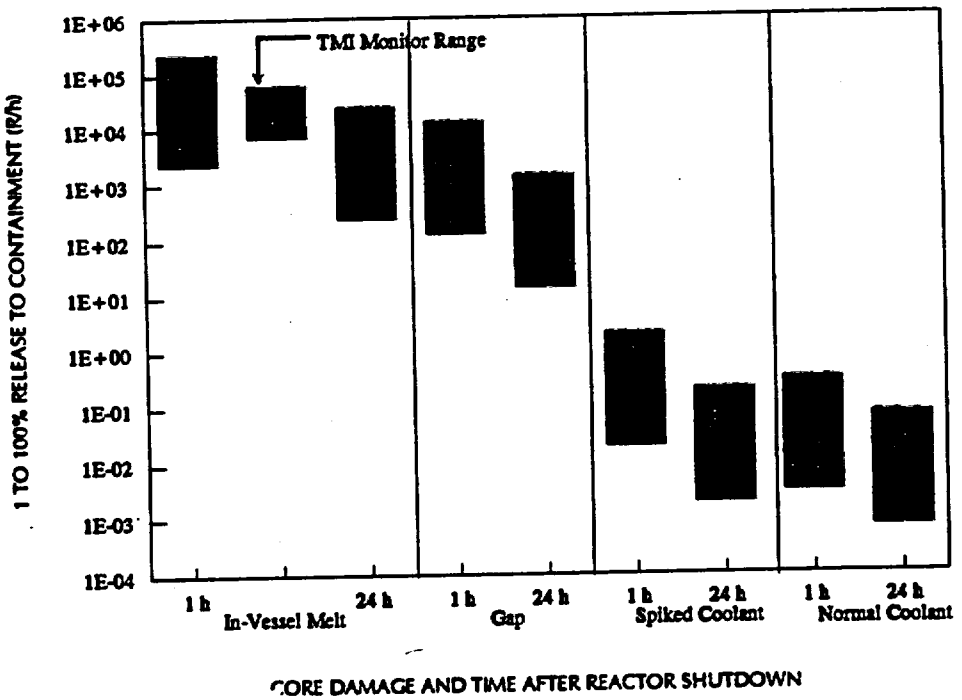


Fig. A-6
PWR containment monitor response (sprays off).

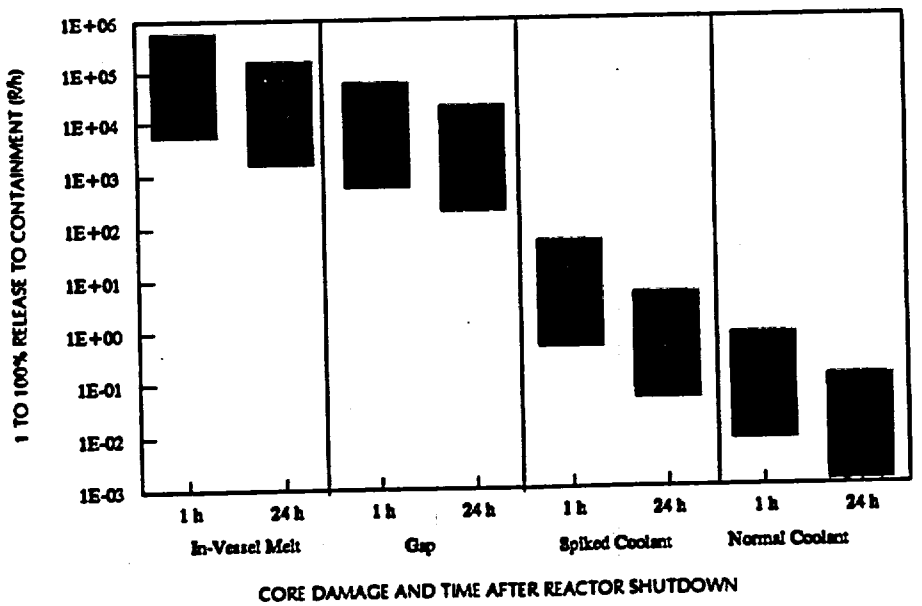


Fig. A-7
BWR Mark I & II drywell containment monitor response (sprays on).

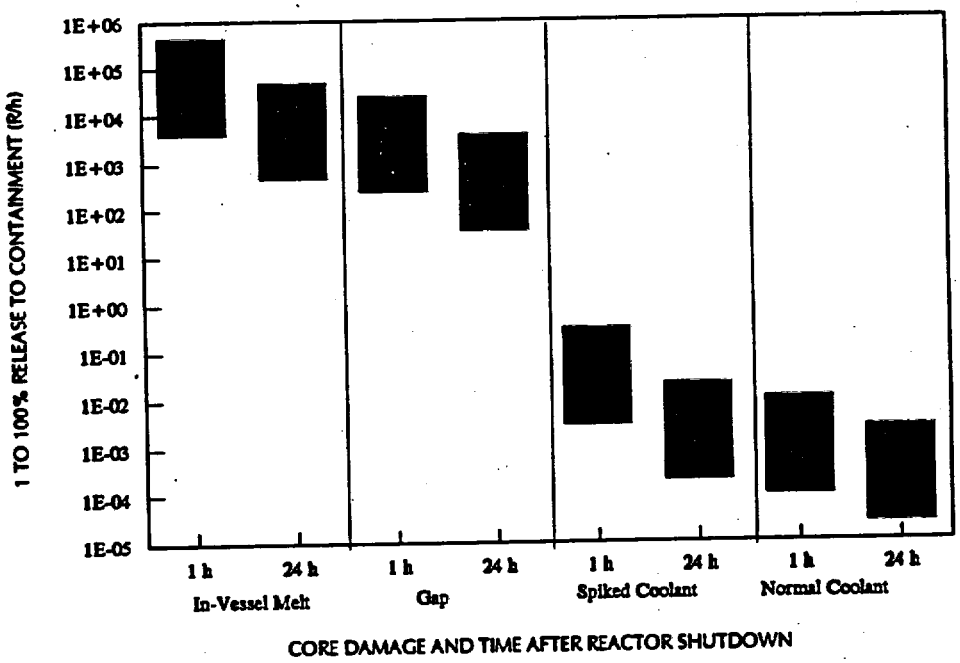


Fig. A-8
BWR Mark I & II drywell containment monitor response (sprays off).

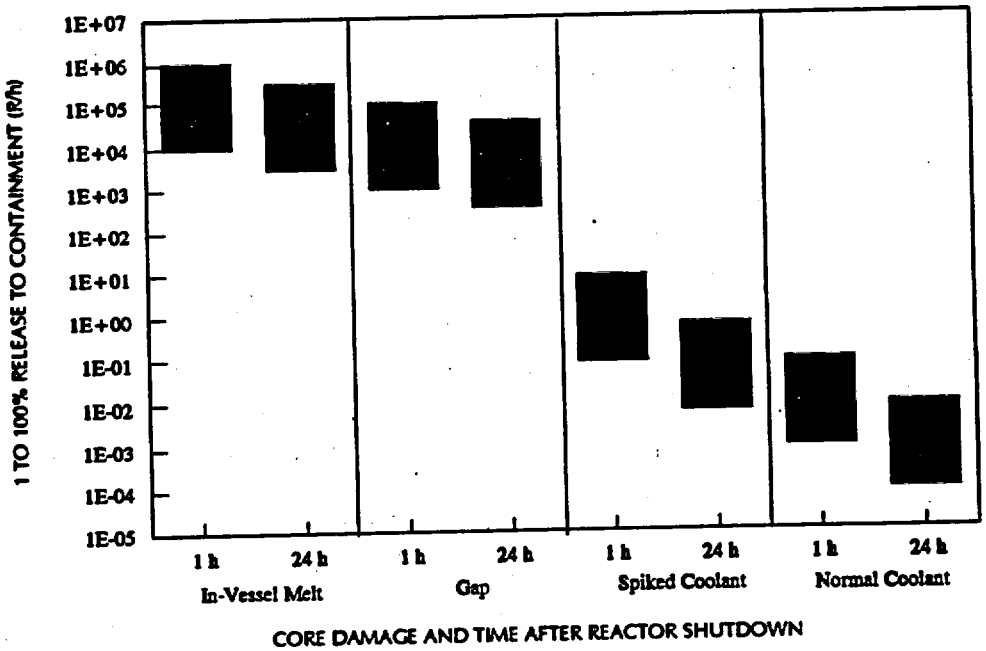


Fig. A-9
BWR Mark I & II wetwell containment monitor response.

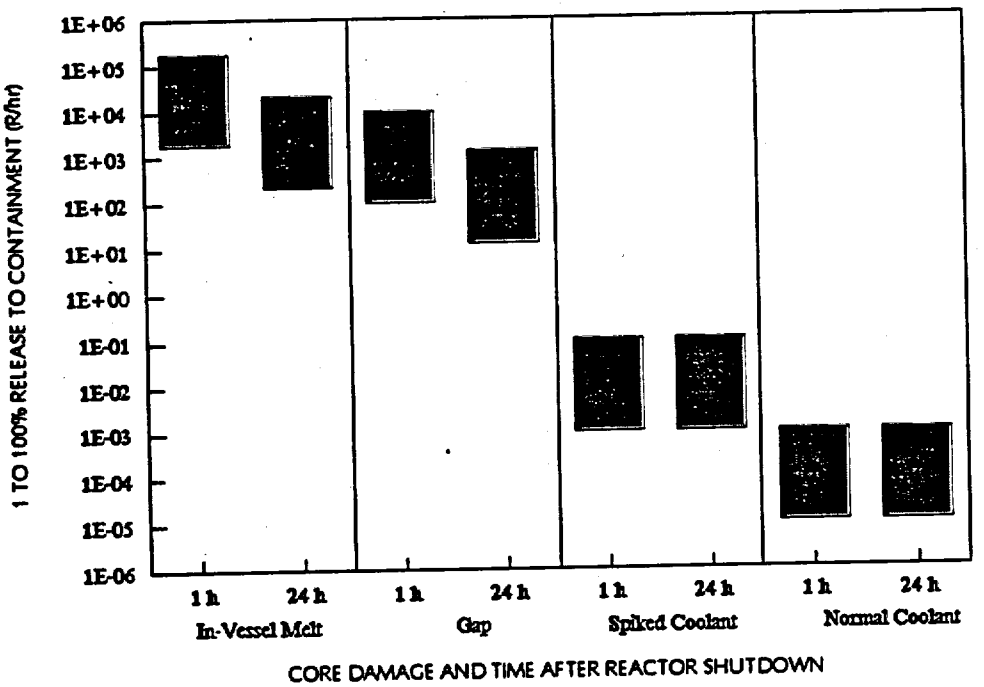


Fig. A-10
BWR Mark III drywell containment monitor response (sprays on).

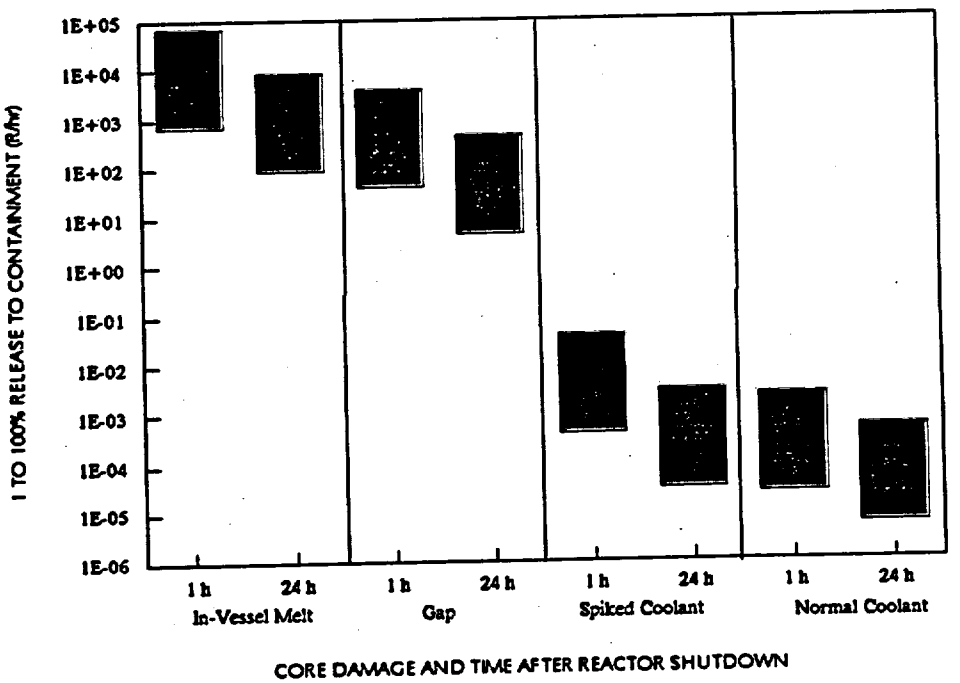


Fig. A-11
BWR Mark III drywell containment monitor response (sprays off).

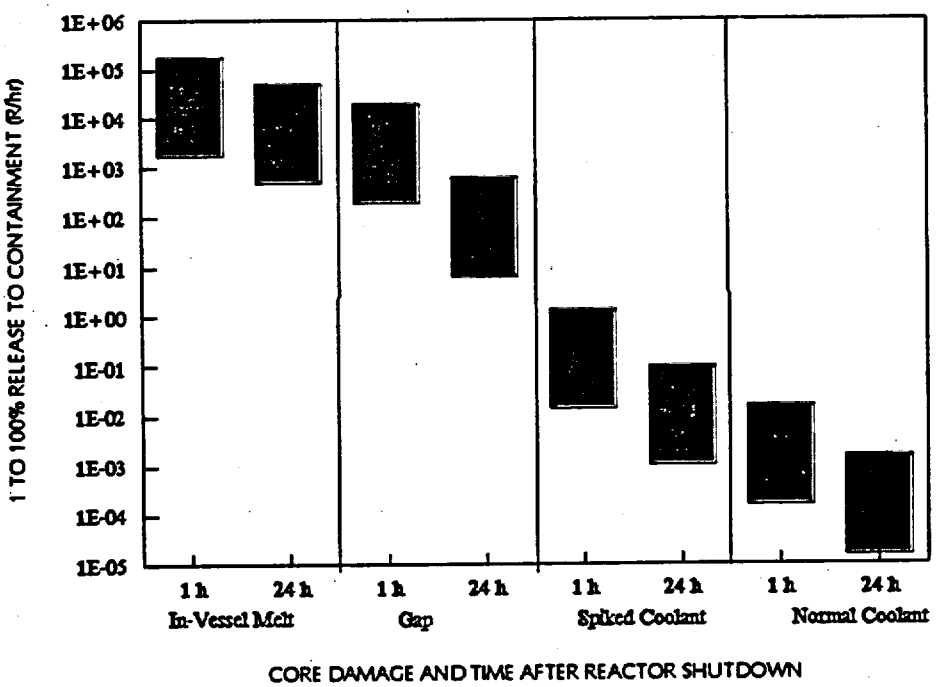


Fig. A-12
BWR Mark III wetwell containment monitor response.

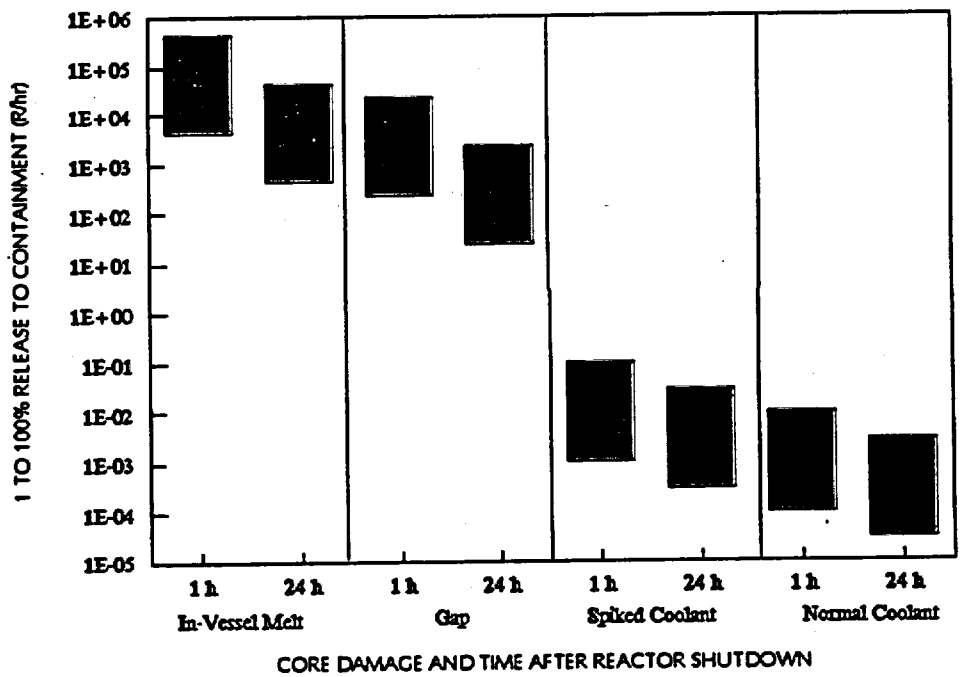
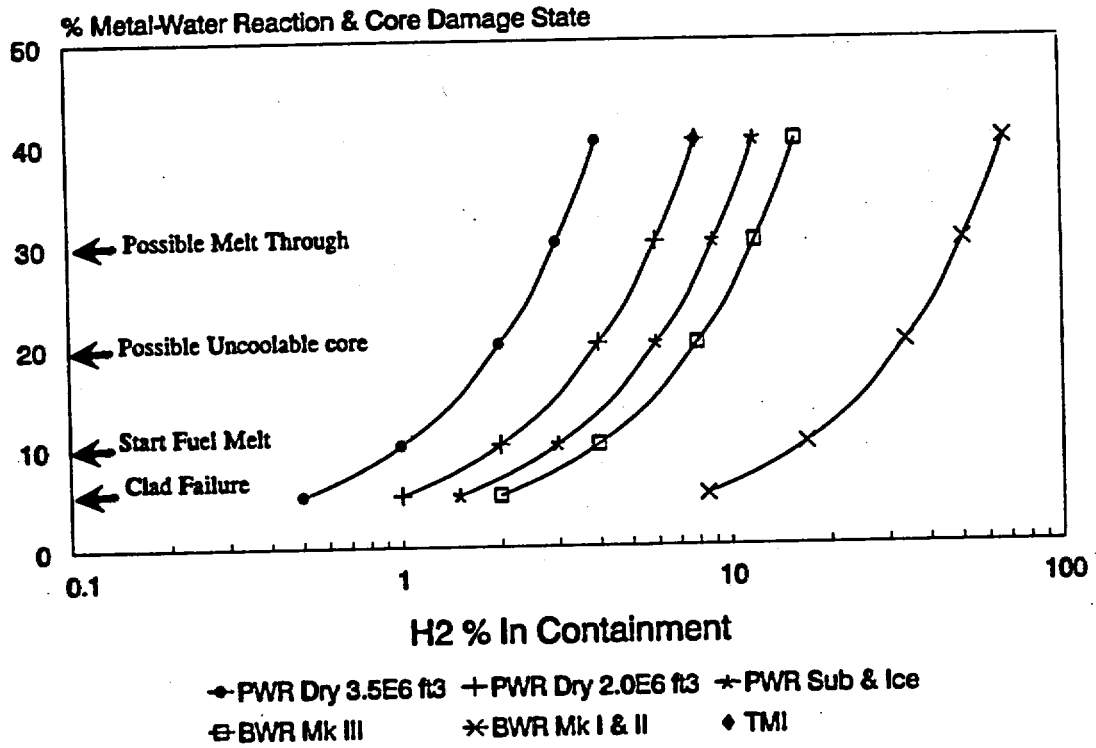


Fig. A-13
Percentage of H_2 in containment relative to core damage.



Sources: NUREG/CR-2726, p. 4-3; damage states, NUREG-4524, Vol. 5.; TMI percentage, NUREG-1370; NUREG/CR-4041; NUREG/CR-5567, Table 4.9, p. 71, confirms "dry" volume.

SECTION B

CLASSIFICATION ASSESSMENT

Section B
Quick Reference Guide

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Classification assessment	B-3
NUREG-0654 quick assessment chart	B-5
NUREG-0654 full guidance	B-7
NUMARC/NESP-007 emergency action level guidance	B-23
Fuel cycle and material facilities classification guidance	B-25
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Section B
Classification Assessment

Purpose

To verify the licensee's classification of the accident.

Discussion

This section provides methods for determining the appropriate classification of an accident at a nuclear power reactor or at a fuel cycle or material facility.

Differences in classification should be discussed with the licensee only if there is a clear conflict in classification. Questioning the licensee in other cases could slow the accident response.

Step 1

Assess the classification of the accident using one of the methods below. The method chosen will depend on the type of facility and the classification method the facility uses.

Reactor accident
- NUREG-0654 quick assessment . . . . . Method B.1
- NUREG-0654 full guidance . . . . . Method B.2
- NUMARC/NESP-007 assessment (barrier approach) . . . . . Method B.3

Fuel cycle and material facilities accident . . . . . Method B.4

END

Sources: NUREG-0654, NUMARC/NESP-007

Method B.1
NUREG-0654 Quick Assessment Chart

Purpose

To assess the classification of the accident when the facility uses a classification system based on the initiating conditions (IC) contained in NUREG-0654.

Discussion

This method uses a quick assessment chart containing the NUREG-0654 initiating conditions sorted by the critical safety function, fission product barriers, radiological releases, and other events for easy comparison with the accident condition.

Step 1

Use Table B-1 to determine the emergency classification.

Step 2

Compare the classification with the licensee's classification. If the licensee's classification does not appear to be correct, review the licensee's classification procedure before discussing your finding with the licensee. Resolve any differences in the interpretation of the plant conditions.

Step 3

If, after attempts to resolve any differences, it appears that the licensee is potentiallv underclassifying a General Emergency, ask the licensee to reevaluate.

Step 4

If the classification is determined to be General Emergency, assess protective actions using Section G.

END

Method B.2
NUREG-0654 Full Guidance

Purpose

To assess the classification of the accident using the NUREG-0654 guidance.

Discussion

This method is used to assess the classification of the accident when the facility uses a classification system based on the initiating conditions contained in NUREG-0654. It provides the complete description, as contained in NUREG-0654, of the initiating conditions that correspond to each of the emergency classes. This method should be used only if you are very familiar with NUREG-0654.

Step 1

Determine the emergency classification of the accident by locating the appropriate example initiating conditions that are listed for each of the emergency classes. (The appropriate appendix from NUREG-0654 is reprinted beginning on p. B-9. An index to the excerpted material appears below.)

Page
Notification of Unusual Event B-10
Alert B-13
Site Area Emergency B-16
General Emergency B-19

Step 2

Compare the classification with the licensee's classification. If the licensee's classification does not appear to be correct, review the licensee's classification procedure before discussing your finding with the licensee. Resolve any differences in the interpretation of the plant conditions.

Step 3

If, after attempts to resolve any differences, it appears that the licensee is potentially underclassifying a General Emergency, ask the licensee to reevaluate.

Step 4

If the classification is determined to be General Emergency, assess protective actions using Section G.

END

NUREG-0654 Appendix 1

BASIS FOR EMERGENCY ACTION LEVELS FOR NUCLEAR POWER FACILITIES

Four classes of Emergency Action Levels are established which replace the classes in Regulatory Guide 1.101, each with associated examples of initiating conditions. The classes are:

Notification of Unusual Event

Alert

Site Area Emergency

General Emergency

The rationale for the notification and alert classes is to provide early and prompt notification of minor events which could lead to more serious consequences given operator error or equipment failure or which might be indicative of more serious conditions which are not yet fully realized. A gradation is provided to assure fuller response preparations for more serious indicators. The site area emergency class reflects conditions where some significant releases are likely or are occurring but where a core melt situation is not indicated based on current information. In this situation full mobilization of emergency personnel in the near site environs is indicated as well as dispatch of monitoring teams and associated communications. The general emergency class involves actual or imminent substantial core degradation or melting with the potential for loss of containment. ~~The immediate action for this class is sheltering (stay in place) rather than evacuation until an assessment can be made that (1) an evacuation is indicated and (2) an evacuation, if indicated, can be completed prior to significant release and transport of radioactive material to the affected areas.~~

The example initiating conditions listed after the immediate actions for each class are to form the basis for establishment by each licensee of the specific plant instrumentation readings (as applicable) which, if exceeded, will initiate the emergency class.

Potential NRC actions during various emergency classes are given in NUREG-0728, Report to Congress: NRC Incident Response Plan. The NRC response to any notification from a licensee will be related to, but not limited by, the licensee estimate of severity; NRC will consider such other factors as the degree of uncertainty and the lead times required to position NRC response personnel should something more serious develop.

Prompt notification of offsite authorities is intended to indicate within about 15 minutes for the unusual event class and sooner (consistent with the need for other emergency actions) for other classes. The time is measured from the time at which operators recognize that events have occurred which make declaration of an emergency class appropriate.

Class

NOTIFICATION OF UNUSUAL EVENT

Class Description

Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Purpose

Purpose of offsite notification is to (1) assure that the first step in any response later found to be necessary has been carried out, (2) bring the operating staff to a state of readiness, and (3) provide systematic handling of unusual events information and decisionmaking.

Licensee Actions

1. Promptly inform State and/or local offsite authorities of nature of unusual condition as soon as discovered
2. Augment on-shift resources as needed
3. Assess and respond
4. Escalate to a more severe class, if appropriate

or

5. Close out with verbal summary to offsite authorities; followed by written summary within 24 hours

State and/or Local Offsite Authority Actions

1. Provide fire or security assistance if requested
2. Escalate to a more severe class, if appropriate
3. Stand by until verbal closeout

EXAMPLE INITIATING CONDITIONS: NOTIFICATION OF UNUSUAL EVENT

1. Emergency Core Cooling System (ECCS) initiated and discharge to vessel
2. Radiological effluent technical specification limits exceeded
3. Fuel damage indication. Examples:
   a. High offgas at BWR air ejector monitor (greater than 500,000 µci/sec; corresponding to 16 isotopes decayed to 30 minutes; or an increase of 100,000 µci/sec within a 30 minute time period)
   b. High coolant activity sample (e.g., exceeding coolant technical specifications for iodine spike)
   c. Failed fuel monitor (PWR) indicates increase greater than 0.1% equivalent fuel failures within 30 minutes
4. Abnormal coolant temperature and/or pressure or abnormal fuel temperatures outside of technical specification limits
5. Exceeding either primary/secondary leak rate technical specification or primary system leak rate technical specification
6. Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure
7. Loss of offsite power or loss of onsite AC power capability
8. Loss of containment integrity requiring shutdown by technical specifications
9. Loss of engineered safety feature or fire protection system function requiring shutdown by technical specifications (e.g., because of malfunction, personnel error or procedural inadequacy)
10. Fire within the plant lasting more than 10 minutes
11. Indications or alarms on process or effluent parameters not functional in control room to an extent requiring plant shutdown or other significant loss of assessment or communication capability (e.g., plant computer, Safety Parameter Display System, all meteorological instrumentation)
12. Security threat or attempted entry or attempted sabotage
13. Natural phenomenon being experienced or projected beyond usual levels
    a. Any earthquake felt in-plant or detected on station seismic instrumentation
    b. 50 year floor or low water, tsunami, hurricane surge, seiche
    c. Any tornado on site
    d. Any hurricane

14. Other hazards being experienced or projected

    a. Aircraft crash on-site or unusual aircraft activity over facility

    b. Train derailment on-site

    c. Near or onsite explosion

    d. Near or onsite toxic or flammable gas release

    e. Turbine rotating component failure causing rapid plant shutdown

15. Other plant conditions exist that warrant increased awareness on the part of a plant operating staff or State and/or local offsite authorities or require plant shutdown under technical specification requirements or involve other than normal controlled shutdown (e.g., cooldown rate exceeding technical specification limits, pipe cracking found during operation)

16. Transportation of contaminated injured individual from site to offsite hospital

17. Rapid depressurization of PWR secondary side.

Class

ALERT

Class Description

Events are in process or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant. Any releases expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.

Purpose

Purpose of offsite alert is to (1) assure that emergency personnel are readily available to respond if situation becomes more serious or to perform confirmatory radiation monitoring if required, and (2) provide offsite authorities current status information.

Licensee Actions

1. Promptly inform State and/or local authorities of alert status and reason for alert as soon as discovered
2. Augment resources and activate on-site Technical Support Center and on-site operational support center. Bring Emergency Operations Facility (EOF) and other key emergency personnel to standby status
3. Assess and respond
4. Dispatch on-site monitoring teams and associated communications
5. Provide periodic plant status updates to offsite authorities (at least every 15 minutes)
6. Provide periodic meteorological assessments to offsite authorities and, if any releases are occurring, dose estimates for actual releases
7. Escalate to a more severe class, if appropriate
8. Close out or recommend reduction in emergency class by verbal summary to offsite authorities followed by written summary within 8 hours of closeout or class reduction

State and/or Local Offsite Authority Actions

1. Provide fire or security assistance if requested
2. Augment resources and bring primary response centers and EBS to standby status
3. Alert to standby status key emergency personnel including monitoring teams and associated communications
4. Provide confirmatory offsite radiation monitoring and ingestion pathway dose projections if actual releases substantially exceed technical specification limits
5. Escalate to a more severe class, if appropriate
6. Maintain alert status until verbal closeout or reduction of emergency class

EXAMPLE INITIATING CONDITIONS: ALERT

1. Severe loss of fuel cladding

   a. High offgas at BWR air ejector monitor (greater than 5 ci/sec; corresponding to 16 isotopes decayed 30 minutes)

   b. Very high coolant activity sample (e.g., 300 μci/cc equivalent of I-131)

   c. Failed fuel monitor (PWR) indicates increase greater than 1% fuel failures within 30 minutes or 5% total fuel failures.

2. Rapid gross failure of one steam generator tube with loss of offsite power

3. Rapid failure of steam generator tubes (e.g., several hundred gpm primary to secondary leak rate)

4. Steam line break with significant (e.g., greater than 10 gpm) primary to secondary leak rate (PWR) or MSIV malfunction causing leakage (BWR)

5. Primary coolant leak rate greater than 50 gpm

6. Radiation levels or airborne contamination which indicate a severe degradation in the control of radioactive materials (e.g., increase of factor of 1000 in direct radiation readings within facility)

7. Loss of offsite power <u>and</u> loss of all onsite AC power (see Site Area Emergency for extended loss)

8. Loss of all onsite DC power (See Site Area Emergency for extended loss)

9. Coolant pump seizure leading to fuel failure

10. Complete loss of any function needed for plant cold shutdown

11. Failure of the reactor protection system to initiate and complete a scram which brings the reactor subcritical

12. Fuel damage accident with release of radioactivity to containment or fuel handling building

13. Fire potentially affecting safety systems

14. Most or all alarms (annunciators) lost

15. Radiological effluents greater than 10 times technical specification instantaneous limits (an instantaneous rate which, if continued over 2 hours, would result in about 1 mr at the site boundary under average meteorological conditions)

16. Ongoing security compromise

17. Severe natural phenomena being experienced or projected
    a. Earthquake greater than OBE levels
    b. Flood, low water, tsunami, hurricane surge, seiche near design levels
    c. Any tornado striking facility
    d. Hurricane winds near design basis level
18. Other hazards being experienced or projected
    a. Aircraft crash on facility
    b. Missile impacts from whatever source on facility
    c. Known explosion damage to facility affecting plant operation
    d. Entry into facility environs of uncontrolled toxic or flammable gases
    e. Turbine failure causing casing penetration
19. Other plant conditions exist that warrant precautionary activation of technical support center and placing near-site Emergency Operations Facility and other key emergency personnel on standby
20. Evacuation of control room anticipated or required with control of shutdown systems established from local stations

Class

SITE AREA EMERGENCY

Class Description

Events are in process or have occurred which involve actual or likely major failures of plant functions needed for protection of the public. Any releases not expected to exceed EPA Protective Action Guideline exposure levels except near site boundary.

Purpose

Purpose of the site area emergency declaration is to (1) assure that response centers are manned, (2) assure that monitoring teams are dispatched, (3) assure that personnel required for evacuation of near-site areas are at duty stations if situation becomes more serious, (4) provide consultation with offsite authorities, and (5) provide updates for the public through offsite authorities.

Licensee Actions

1. Promptly inform State and/or local offsite authorities of site area emergency status and reason for emergency as soon as discovered
2. Augment resources by activating on-site Technical Support Center, on-site operational support center and near-site Emergency Operations Facility (EOF)
3. Assess and respond
4. Dispatch on-site and offsite monitoring teams and associated communications
5. Dedicate an individual for plant status updates to offsite authorities and periodic pressure briefings (perhaps joint with offsite authorities)
6. Make senior technical and management staff onsite available for consultation with NRC and State on a periodic basis
7. Provide meteorological and dose estimates to offsite authorities for actual releases via a dedicated individual or automated data transmission
8. Provide release and dose projections based on available plant condition information and foreseeable contingencies
9. Escalate to general emergency class, if appropriate

   or

10. Close out or recommend reduction in emergency class by briefing of offsite authorities at EOF and by phone followed by written summary within 8 hours of closeout or class reduction

State and/or Local Offsite Authority Actions

1. Provide any assistance requested
2. If sheltering near the site is desirable, activate public notification system within at least two miles of the plant
3. Provide public within at least about 10 miles periodic updates on emergency status
4. Augment resources by activating primary response centers
5. Dispatch key emergency personnel including monitoring teams and associated communications
6. Alert to standby status other emergency personnel (e.g., those needed for evacuation) and dispatch personnel to near-site duty stations
7. Provide offsite monitoring results to licensee, DOE and others and jointly assess them
8. Continuously assess information from licensee and offsite monitoring with regard to changes to protective actions already initiated for public and mobilizing evacuation resources
9. Recommend placing milk animals within 2 miles on stored feed and assess need to extend distance
10. Provide press briefings, perhaps with licensee
11. Escalate to general emergency class, if appropriate
12. Maintain site area emergency status until closeout or reduction of emergency class

EXAMPLE INITIATING CONDITIONS: SITE AREA EMERGENCY

1. Known loss of coolant accident greater than makeup pump capacity

2. Degraded core with possible loss of coolable geometry (indicators should include instrumentation to detect inadequate core cooling, coolant activity and/or containment radioactivity levels)

3. Rapid failure of steam generator tubes (several hundred gpm leakage) with loss of offsite power

4. BWR steam line break outside containment without isolation

5. PWR steam line break with greater than 50 gpm primary to secondary leakage and indication of fuel damage

6. Loss of offsite power and loss of onsite AC power for more than 15 minutes

7. Loss of all vital onsite DC power for more than 15 minutes

8. Complete loss of any function needed for plant hot shutdown

9. Transient requiring operation of shutdown systems with failure to scram (continued power generation but no core damage immediately evident)

10. Major damage to spent fuel in containment or fuel handling building (e.g., large object damages fuel or water loss below fuel level)

11. Fire compromising the functions of safety systems

12. Most or all alarms (annunciators) lost and plant transient initiated or in progress

13. a. Effluent monitors detect levels corresponding to greater than 50 mr/hr for 1/2 hour or greater than 500 mr/hr W.B. for two minutes (or five times these levels to the thyroid) at the site boundary for adverse meteorology

    b. These dose rates are projected based on other plant parameters (e.g., radiation level in containment with leak rate appropriate for existing containment pressure) or are measured in the environs

    c. EPA Protective Action Guidelines are projected to be exceeded outside the site boundary

14. Imminent loss of physical control of the plant

15. Severe natural phenomena being experienced or projected with plant not in cold shutdown

    a. Earthquake greater than SSE levels

b. Flood, low water, tsunami, hurricane surge, seiche greater than design levels or failure of protection of vital equipment at lower levels

c. Sustained winds or tornadoes in excess of design levels

16. Other hazards being experienced or projected with plant not in cold shutdown

a. Aircraft crash affecting vital structures by impact or fire

b. Severe damage to safe shutdown equipment from missiles or explosion

c. Entry of uncontrolled flammable gases into vital areas. Entry of uncontrolled toxic gases into vital areas where lack of access to the area constitutes a safety problem

17. Other plant conditions exist that warrant activation of emergency centers and monitoring teams or a precautionary notification to the public near the site

18. Evacuation of control room and control of shutdown systems not established from local stations in 15 minutes

Class

GENERAL EMERGENCY

Class Description

Events are in process or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.

Purpose

Purpose of the general emergency declaration is to (1) initiate predetermined protective actions for the public, (2) provide continuous assessment of information from licensee and offsite organization measurements, (3) initiate additional measures as indicated by actual or potential releases, (4) provide consultation with offsite authorities and (5) provide updates for the public through offsite authorities.

Licensee Actions

1. Promptly inform State and local offsite aut orities of general emergency status and reason for emergency as soon as discovered (Parallel notification of State/local)
2. Augment resources by activating on-site Technical Support Center, on-site operational support center and near-site Emergency Operations Facility (EOF)
3. Assess and respond
4. Dispatch on-site and offsite monitoring teams and associated communications
5. Dedicate an individual for plant status updates to offsite authorities and periodic press briefings (perhaps joint with offsite authorities)
6. Make senior technical and management staff onsite available for consultation with NRC and State on a periodic basis
7. Provide meteorological and dose estimates to offsite authorities for actual releases via a dedicated individual or automated data transmission
8. Provide release and dose projections based on available plant condition information and foreseeable contingencies
9. Close out or recommend reduction of emergency class by briefing of offsite authorities at EOF and by phone followed by written summary within 8 hours of closeout or class reduction

State and/or Local Offsite Authority Actions

1. Provide any assistance requested
2. Activate immediate public notification of emergency status and provide public periodic updates
3. Recommend sheltering for 2 mile radius and 5 miles downwind and assess need to extend distances. Consider advisability of evacuation (projected time available vs. estimated evacuation times)
4. Augment resources by activatin primary response centers
5. Dispatch key emergency personn including monitoring teams and associated communications
6. Dispatch other emergency personnel to duty stations within 5 mile radius and alert all others to standby status
7. Provide offsite monitoring results to licensee, DOE and others and jointly assess them
8. Continuously assess information from licensee and offsite monitoring with regard to changes to protective actions already initiated for public and mobilizing evacuation resources
9. Recommend placing milk animals within 10 miles on stored feed and assess need to extend distance
10. Provide press briefings, perhaps with licensee
11. Maintain general emergency status until closeout or reduction of emergency class

EXAMPLE INITIATING CONDITIONS: GENERAL EMERGENCY

1. a. Effluent monitors detect levels corresponding to 1 rem/hr W.B. or 5 rem/hr thyroid at the site boundary under actual meteorological conditions

   b. These dose rates are projected based on other plant parameters (e.g., radiation levels in containment with leak rate appropriate for existing containment pressure with some confirmation from effluent monitors) or are measured in the environs

   Note: Consider evacuation only within about 2 miles of the site boundary unless these site boundary levels are exceeded by a factor of 10 or projected to continue for 10 hours or EPA Protective Action Guideline exposure levels are predicted to be exceeded at longer distances

2. Loss of 2 of 3 fission product barriers with a potential loss of 3rd barrier, (e.g., loss of primary coolant boundary, clad failure, and high potential for loss of containment)

3. Loss of physical control of the facility

   Note: Consider 2 mile precautionary evacuation.

4. Other plant conditions exist, from whatever source, that make release of large amounts of radioactivity in a short time period possible, e.g., any core melt situation. See the specific PWR and BWR sequences below.

   Notes: a. For core melt sequences where significant releases from containment are not yet taking place and large amounts of fission products are not yet in the containment atmosphere, consider 2 mile precautionary evacuation. Consider 5 mile downwind evacuation (45° to 90° sector) if large amounts of fission products (greater than gap activity) are in the containment atmosphere. Recommend sheltering in other parts of the plume exposure Emergency Planning Zone under this circumstance.

   b. For core melt sequences where significant releases from containment are not yet taking place and containment failure leading to a direct atmospheric release is likely in the sequence but not imminent and large amounts of fission products in addition to noble gases are in the containment atmosphere, consider precautionary evacuation to 5 miles and 10 mile downwind evacuation (45° to 90° sector).

   c. For core melt sequences where large amounts of fission products other than noble gases are in the containment atmosphere and containment failure is judged imminent, recommend shelter for those areas where evacuation cannot be completed before transport of activity to that location.

d. As release information becomes available adjust these actions in accordance with dose projections, time available to evacuate and estimated evacuation times given current conditions.

5. Example PWR Sequences

   a. Small and large LOCA's with failure of ECCS to perform leading to severe core degradation or melt in from minutes to hours. Ultimate failure of containment likely for melt sequences. (Several hours likely to be available to complete protective actions unless containment is not isolated)

   b. Transient initiated by loss of feedwater and condensate systems (principal heat removal system) followed by failure of emergency feedwater system for extended period. Core melting possible in several hours. Ultimate failure of containment likely if core melts.

   c. Transient requiring operation of shutdown systems with failure to scram which results in core damage or additional failure of core cooling and makeup systems (which could lead to core melt)

   d. Failure of offsite and onsite power along with total loss of emergency feedwater makeup capability for several hours. Would lead to eventual core melt and likely failure of containment.

   e. Small LOCA and initially successful ECCS. Subsequent failure of containment heat removal systems over several hours could lead to core melt and likely failure of containment.

   NOTE: Most likely containment failure mode is melt-through with release of gases only for dry containment; quicker and larger releases likely for ice condenser containment for melt sequences. Quicker releases expected for failure of containment isolation system for any PWR.

6. Example BWR Sequences

   a. Transient (e.g., loss of offsite power) plus failure of requisite core shut down systems (e.g., scram). Could lead to core melt in several hours with containment failure likely. More severe consequences if pumps trip does not function.

   b. Small or large LOCA's with failure of ECCS to perform leading to core melt degradation or melt in minutes to hours. Loss of containment integrity may be imminent.

   c. Small or large LOCA occurs and containment performance is unsuccessful affecting longer term success of the ECCS. Could lead to core degradation or melt in several hours without containment boundary.

d. Shutdown occurs but requisite decay heat removal systems (e.g., RHR) or non-safety systems heat removal means are rendered unavailable. Core degradation or melt could occur in about ten hours with subsequent containment failure.

7. Any major internal or external events (e.g., fires, earthquakes, substantially beyond design basis) which could cause massive common damage to plant systems resulting in any of the above.

Method B.3
NUMARC/NESP-007 Emergency Action Level Guidance

Purpose

To assess the classification of an accident when the facility uses a classification system based on NUMARC/NESP-007 methodology (barrier approach).

Discussion

The Nuclear Energy Institute [formerly the Nuclear Management and Resources Council, Inc. (NUMARC)] methodology is contained in NUMARC/NESP-007, *Methodology for Development of Emergency Action Levels*. The emergency action level (EAL) methodology is generic and was intended to provide the logic for developing site-specific EALs. As a result, the utilities' EALs may reference site-specific procedures, indications, values, etc.

This methodology uses Recognition Categories for classifying the accident. For each of the Recognition Categories, a matrix is provided showing initiating conditions (ICs) and the corresponding emergency class. The IC matrices apply to both PWRs and BWRs. Refer to NUMARC/NESP-007 for specific examples of EALs.

All cases of severe core damage (loss of fuel cladding barrier) should be classified as a General Emergency.

Step 1

Determine the Recognition Category (A, H, S, or F, as shown below) that matches the existing plant condition.

Abnormal radiation levels/radiological effluent . . . . . A
Hazards and other conditions affecting plant safety . . . . . H
System malfunction . . . . . S
Fission product barrier degradation . . . . . F

Step 3

If the Recognition Category is F, go to Step 4.

If the Recognition Category is A, H, or S, use the applicable IC matrix for the Recognition Category to determine the IC and the corresponding emergency class. Then go to Step 5.

Category A . . . . . Table B-2
Category H . . . . . Table B-3
Category S . . . . . Table B-4

Step 4

If the Recognition Category is F, then use the fission product barrier degradation IC matrix (Table B-5) to determine the IC and the emergency class. To determine if the fission product barrier(s) is/are lost or potentially lost, refer to the barrier-based emergency action levels (EALs) listed in Table B-6 for BWRs and Table B-7 for PWRs. Match the observed plant parameters affecting each of the fuel, RCS, and containment fission product barriers to the EALs that are listed in these tables, and note whether each barrier is lost or potentially lost. Then return to Table B-5 and determine the classification from the listed barrier conditions.

Step 5

Compare the classification with the licensee's classification. If the licensee's classification does not appear to be correct, review the licensee's classification procedure before discussing your finding with the licensee. Resolve any differences in the interpretation of the plant conditions.

Step 6

If, after attempts to resolve any differences, it appears that the licensee is potentially underclassifying a General Emergency, ask the licensee to reevaluate.

Step 7

If the classification is determined to be General Emergency, assess protective actions using Section G.

END

Method B.4
Fuel Cycle and Material Facilities Classification Guidance

Purpose

This method is used to assess the classification of an accident at a fuel cycle or material facility.

Discussion

Emergency plans for fuel cycle and material facilities are not yet standardized. As licenses are renewed at facilities requiring emergency plans, a standardized classification system will be adopted. Some facilities do not have emergency plans because of the small quantity of material they handle. These classification descriptions would not apply to the facilities that do not have emergency plans.

Step 1

Use the classification descriptions in Table B-8 to determine the emergency classification of the accident. Note that there are no Unusual Event or General Emergency classifications for non-reactor facilities.

Step 2

Compare the classification with the licensee's classification. If the licensee's classification does not appear to be correct, review the licensee's classification procedure before discussing your finding with the licensee. Resolve any differences in the interpretation of the plant conditions.

Step 3

If, after attempts to resolve any differences, it appears that the licensee is potentially underclassifying a General Emergency, ask the licensee to reevaluate.

Step 4

If the classification is determined to be General Emergency, assess protective actions using Section G.

END

Table B-1. NUREG-0654 quick assessment chart

Type	Unusual Event	Alert	Site Area	General
Reactivity control loss		Failure to completely shut down the reactor	Transient requiring shutdown *and* failure to shutdown	Transient requiring shutdown, failure to shut down, *and* - failure of ECCS *or* - indication of core damage
Inventory control loss	ECCS starts and injects water into reactor	Primary system leak rate > 50 gal/min	Primary system leak > makeup capacity	Primary coolant system leak (LOCA) *and* failure of ECCS Any event leading to prolonged uncovery of core
Heat removal loss	Outside Tech Specs: - coolant temperature - coolant pressure - fuel temperature Loss of engineered safety feature system requiring shutdown by Tech Specs	Complete loss of function required for cold shutdown RC pump seizure leading to fuel cladding failure	Complete loss of function needed for hot shutdown	Decay heat removal systems (primary coolant or containment) failure for extended period PWR loss of main and auxiliary feedwater for an extended period

Table B-1. NUREG-0654 quick assessment chart (continued)

Type	Unusual Event	Alert	Site Area	General
Vital auxiliaries loss Power	AC loss onsite or offsite	AC loss onsite and offsite Vital DC loss onsite PWR loss of offsite power *and* rapid steam generator tube rupture	AC loss > 15 min onsite and offsite Vital DC loss onsite > 15 min PWR loss of offsite power *and* > 300 gal/min steam generator tube rupture	PWR AC loss onsite and offsite *and* loss of auxiliary feedwater for several hours BWR loss of onsite and offsite power *and* reactor not shut down
Control room		Evacuation of control room *and* control of shutdown system established	Evacuation of control room *and* control of shutdown system not established within 15 min	Loss of control of the facility
Instruments & alarms	Loss of instruments and alarms requiring shutdown by Tech Specs Significant loss of assessment capability	Loss of most control room alarms	Loss of most control room alarms *and* transient in progress	
Fuel cladding loss	>0.1% clad failure in 30 min BWR high radioactivity in offgas or reactor coolant Coolant activity > Tech Spec	>1.0% cladding failure in 30 min *or* 5% total clad failure BWR very high radioactivity in offgas (>5 Ci/s). Reactor coolant (>300 μCi/cc)	Degraded core with possible loss of coolable geometry	Actual or projected >100% cladding failure equivalent Any sequence that could lead to severe heatup of core

Table B-1. NUREG-0654 quick assessment chart (continued)

Type	Unusual Event	Alert	Site Area	General
Primary system breaks & leaks	Primary system leak > Tech Spec PWR steam generator tube leak > Tech Specs PWR rapid loss of secondary side pressure Stuck open code safety or power operated relief valves	Primary system leak rate > 50 gal/min PWR rapid steam generator tube rupture(s) *and* - loss of offsite power *or* - leak > 300 gal/min PWR steam line break *and* steam generator tube leak > 10 gal/min BWR steam line break inside containment without MSIV closure	Primary system leak > makeup system capability (LOCA) PWR rapid steam generator tube rupture > 300 gal/min & loss of offsite power PWR steam line break & steam generator tube rupture leak > 50 gal/min *and* cladding failure BWR steam line break outside containment without MSIV closure	Primary system leak > makeup *and* failure of ECCS Events leading to prolonged core uncovery Loss of two of three fission product barriers *and* potential loss of the third barrier
Containment loss	Loss of containment integrity requiring shutdown by Tech Specs	BWR steam line break inside containment without MSIV closure	BWR steam line break outside containment without MSIV closure	Loss of any two of three fission product barriers *and* potential loss of the third barrier BWR primary system leak *and* loss of containment integrity affecting success of ECCS
Radiological release	Effluent radiation release > Tech Specs	Offsite radiation release > 10 × instantaneous limits In-plant radiation levels > 1000 × normal	Whole body dose projection assuming adverse meteorological conditions indicate > 50 mR/h for 30 min *or* 500 mR/h for 2 min at site boundary	Actual measurements or dose projections under actual meteorological conditions indicate EPA PAGs will be exceeded at the site boundary Possible release of large amounts of radioactivity offsite

Table B-1. NUREG-0654 quick assessment chart (continued)

Type	Unusual Event	Alert	Site Area	General
Spent fuel accident		Spent fuel damage with radiation release in plant	Major damage to spent fuel Spent fuel pool water level below top of spent fuel	Dose projections or measurements indicate EPA PAGs will be exceeded at site boundary
Fire	Plant fire lasting > 10 min Loss of fire protection system requiring shutdown by Tech Specs	Fire potentially affecting safety systems	Fire compromising the functions of safety systems	Major fire that could cause massive common damage to plant systems leading to core melt
Security	Security threat Attempted entry Attempted sabotage	Ongoing security compromise	Imminent loss of control of the plant	Loss of control of the plant
Other hazards	Actual or projected hazards - earthquakes - floods - hurricanes - tornados - explosions - gas releases - aircraft crashes - derailment	Actual or projected severe hazards	Severe natural phenomena or hazard *and* plant not in cold shutdown - any event greater than design - damage to safety systems - flammable gas in vital areas	Major event which could cause massive common damage to plant systems resulting in core melt

Table B-1. NUREG-0654 quick assessment chart (continued)

Type	Unusual Event	Alert	Site Area	General
Activation of centers	Conditions warrant increased awareness	Conditions warrant activation of TSC	Conditions warrant activation of TSC or EOF	
Public notification			Conditions warrant notification of the public	
Medical	Transport of contaminated injured person to hospital			

Source: Adapted from NUREG-0654.

Table B-2. Recognition Category A: Abnormal rad levels/radiological effluent initiating condition matrix

Unusual Event		Alert		Site Area Emergency		General Emergency	
AU1	Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds two times the radiological technical specifications for 60 min or longer. Op. modes: All	AA1	Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times the radiological technical specifications for 15 min or longer. Op. modes: All	AS1	Site boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 100 mR whole body or 500 mR child thyroid for the actual or projected duration of the release. Op. modes: All	AG1	Site boundary dose resulting from an actual or imminent release of gaseous radioactivity that exceeds 1000 mR whole body or 5000 mR child thyroid for the actual or projected duration of the release using actual meteorology. Op. modes: All
AU2	Unexpected increase in plant radiation levels or airborne concentration. Op. modes: All	AA2	Major damage to irradiated fuel or loss of water level that has or will result in the uncovering of irradiated fuel outside the reactor vessel. Op. modes: All				
		AA3	Release of radioactive material or increases in radiation levels within the facility that impedes operation of systems required to maintain safe operations or to establish or maintain cold shutdown. Op. modes: All				

Source: NUMARC/NESP-007, p. 5-3.

Table B-3. Recognition Category H: Hazards and other conditions affecting plant safety initiating condition matrix

Unusual Event	Alert	Site Area Emergency	General Emergency
HU1 Natural and destructive phenomena occurring within the protected area. Op. modes: All	HA1 Natural and destructive phenomena occurring within the plant vital area. Op. modes: All	HS1 Security event in plant vital area. Op. modes: All	HG1 Security event resulting in loss of ability to reach and maintain cold shutdown. Op. modes: All
HU2 Fire within protected area boundary not extinguished within 15 min of detection. Op. modes: All	HA2 Fire affecting the operability of plant safety systems required for the current operating mode. Op. modes: All	HS2 Control room evacuation has been initiated and plant control cannot be established. Op. modes: All	HG2 Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of a General Emergency. Op. modes: All
HU3 Release of toxic or flammable gases deemed detrimental to safe operation of the plant. Op. modes: All	HA3 Release of toxic or flammable gases within a facility structure which jeopardizes operation of systems required to establish or maintain cold shutdown. Op. modes: All	HS3 Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of a Site Area Emergency. Op. modes: All	
HU4 Confirmed security event which indicates a potential degradation in the level of safety of the plant. Op. modes: All	HA4 Security event in a plant protected area. Op. modes: All		
HU5 Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of an Unusual Event. Op. modes: All	HA5 Control room evacuation has been initiated. Op. modes: All		
	HA6 Other conditions existing which, in the judgement of the Emergency Director, warrant declaration of an Alert. Op. modes: All		

Source: NUMARC/NESP-007, p. 5-35.

Table B-4. Recognition Category S: System malfunction initiating condition matrix

Unusual Event	Alert	Site Area Emergency	General Emergency
SU1 Loss of all offsite power to essential buses for greater than 15 min. Op. modes: All	**SA1** Loss of all offsite power and loss of all onsite AC power to essential buses during cold shutdown or refueling mode. Op. modes: Cold shutdown, refueling, defueled	**SS1** Loss of all offsite power and loss of all onsite AC power. Op. modes: Power operation, hot standby, hot shutdown	**SG1** Prolonged loss of all offsite power and prolonged loss of all onsite AC power. Op. modes: Power operation, hot standby, hot shutdown
SU2 Inability to reach required shutdown within technical specification limits. Op. modes: Power operation, hot standby, hot shutdown	**SA2** Failure of reactor protection system instrumentation to complete or initiate an automatic reactor scram once a reactor protection system setpoint has been exceeded and manual scram was successful. Op. modes: Power operation, hot standby	**SS2** Failure of reactor protection system instrumentation to complete or initiate an automatic reactor scram once a reactor protection system setpoint has been exceeded and manual scram was NOT successful. Op. mode: Power operation.	**SG2** Failure of the reactor protection system to complete an automatic scram and manual scram was NOT successful AND there is indication of an extreme challenge to the ability to cool the core. Op. mode: Power operation
SU3 Unplanned loss of most or all safety system annunciators for greater than 15 min. Op. modes: Power operation, hot standby, hot shutdown	**SA3** Inability to maintain plant in cold shutdown. Op. modes: Cold shutdown, refueling	**SS3** Loss of all vital DC power. Op. modes: Power operation, hot standby, hot shutdown	
SU4 Fuel clad degradation. Op. modes: All	**SA4** Unplanned loss of all safety annunciators with transient in progress. Op. modes: Power operation, hot standby, hot shutdown	**SS4** Complete loss of function needed to achieve or maintain hot shutdown. Op. modes: Power operation, hot standby, hot shutdown	
SU5 RCS leakage. Op. modes: Power operation, hot standby, hot shutdown, cold shutdown		**SS5** Loss of water level that has or will uncover fuel in the reactor vessel. Op. modes: Cold shutdown, refueling	

Table B-4. Recognition Category S: System malfunction initiating condition matrix (continued)

Unusual Event	Alert	Site Area Emergency	General Emergency
SU6 Unplanned loss of all onsite or offsite communication capabilities. Op. modes: All SU7 Unplanned loss of required DC power during cold shutdown or refueling for greater than 15 min. Op. modes: Cold shutdown, refueling	SA5 AC power capability to essential buses reduced to a single power source for greater than 15 min such that any additional single failure would result in station blackout. Op. modes: Power operation, hot standby, hot shutdown	SS6 Inability to monitor a significant transient in progress. Op. modes: Power operation, hot standby, hot shutdown	

Source: NUMARC/NESP-007, p. 5-54.

Table B-5. Recognition Category F: Fission product barrier degradation initiating condition matrix[a,b]
(See Table B-6 for BWR example EALs and Table B-7 for PWR example EALs.)

Unusual Event	Alert	Site Area Emergency	General Emergency
FU1 ANY loss or ANY potential loss of containment. Op. modes: Power operation, hot standby/startup (BWR), hot shutdown	FA1 ANY loss or ANY potential loss of EITHER fuel clad OR RCS. Op. modes: Power operation, hot standby/startup (BWR), hot shutdown	FS1 Loss of BOTH fuel clad AND RCS OR Potential loss of BOTH fuel clad AND RCS OR Potential loss of EITHER fuel clad OR RCS, and loss of ANY additional barrier Op. modes: Power operation, hot standby/startup (BWR), hot shutdown	FG1 Loss of ANY two barriers AND Potential loss of third barrier. Op. modes: Power operation, hot standby/startup (BWR), hot shutdown

[a]Although the logic used for these initiating conditions appears overly complex, it is necessary to reflect the following considerations:

- The fuel clad barrier and the RCS barrier are weighted more heavily than the containment barrier (see Sections 3.4 and 3.8 of NUMARC/NESP-007 for more information on this point). Unusual Event ICs associated with RCS and fuel clad barriers are addressed under system malfunction ICs.
- At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from a General Emergency. For example, if fuel clad barrier and RCS barrier "loss" EALs existed, this would indicate to the Emergency Director that, in addition to offsite dose assessments, continual assessments of radioactive inventory and containment integrity must be focused on. If, on the other hand, both fuel clad barrier and RCS barrier "potential loss" EALs existed, the Emergency Director would have more assurance that there was no immediate need to escalate to a General Emergency.
- The ability to escalate to higher emergency classes as an event gets worse must be maintained. For example, steadily increasing RCS leakage would represent an increasing risk to public health and safety.

[b]Be capable of addressing event dynamics. Thus, the EAL reference tables (Figs. B-6 and B-7) state that imminent (i.e., within 1 to 3 h) loss or potential loss should result in a classification as if the affected threshold(s) are already exceeded, particularly for the higher emergency classes.

Source: NUMARC/NESP-007, p. 5-17.

Table B-6. BWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers

Determine which combination of the three barriers is lost or has a potential loss and use the following key to classify the event. Also, an event or multiple events could occur, which result in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT (i.e., within 1 to 3 h). In this IMMINENT LOSS situation, use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		PRIMARY CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
1. Primary coolant activity levels		1. RCS leak rate		1. Drywell pressure	
Coolant activity GREATER THAN (site-specific) value	Not applicable	(Site-specific) indication of main steamline break	RCS leakage GREATER THAN 50 gal/min inside the drywell	Rapid unexplained decrease following initial increase	(Site-specific) psig and increasing
OR		OR	OR	OR	OR
			Unisolable primary system leakage outside drywell as indicated by area temp or area rad alarm	Drywell pressure response not consistent with LOCA conditions	Explosive mixture exists
		OR		OR	

Table B-6. BWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		PRIMARY CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
2. Reactor vessel water level		2. Drywell pressure		2. Containment isolation valve after containment isolation	
Level LESS THAN (site-specific) value	Level LESS THAN (site-specific) value	Pressure GREATER THAN (site-specific) psig	Not applicable	Failure of both valves in any one line to close AND downstream pathway to the environmental exists	Not applicable
OR		OR		OR	
				Intentional venting per EOPs	Not applicable
				OR	
				Unisolable primary system leakage outside dry well as indicated by area temp or area rad alarm	
				OR	

Table B-6. BWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad and RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS and loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		PRIMARY CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
3. Drywell radiation monitoring		3. Drywell radiation monitoring		3. Significant radioactive inventory in containment	
Drywell radiation monitor reading GREATER THAN (site-specific) R/h	Not applicable	Drywell radiation monitor reading GREATER THAN (site specific) R/h	Not applicable	Not applicable	Containment radiation monitor reading GREATER THAN (site-specific) R/h
		4. Reactor vessel water level		4. Reactor vessel water level	
4. Other (site-specific) indications		Level LESS THAN (site-specific) value	Not applicable	Not applicable	Reactor vessel water level LESS THAN (site-specific) value and the maximum core uncovery time limit is in the UNSAFE region
(Site-specific) as applicable	(Site-specific) as applicable	OR		OR	
OR					

Table B-6. BWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad and RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS and loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		PRIMARY CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
5. Emergency Director judgement		5. Other (site-specific) indications		5. Other (site-specific) indications	
Any condition in the judgement of the Emergency Director that indicates loss or potential loss of the FUEL CLAD barrier.		(Site-specific) as applicable	(Site-specific) as applicable	(Site-specific) as applicable	(Site-specific) as applicable
		OR		OR	
		6. Emergency Director judgement		6. Emergency Director judgement	
		Any condition in the judgement of the Emergency Director that indicates loss or potential loss of the RCS barrier.		Any condition in the judgement of the Emergency Director that indicates loss or potential loss of the CONTAINMENT barrier.	

Source: NUMARC/NESP-007, pp. 5-18, 5-19.

Table B-7. PWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

Determine which combination of the three barriers is lost or has a potential loss and use the following key to classify the event. Also, an event or multiple events could occur, resulting in the conclusion that exceeding the loss or potential loss thresholds is IMMINENT (i.e., within 1 to 3 h). In this IMMINENT LOSS situation, use judgement and classify as if the thresholds are exceeded.

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad or RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
1. Critical safety function status		1. Critical safety function status		1. Critical safety function status	
Core-cooling-red	Core cooling-orange OR Heat sink-red	Not applicable	RCS integrity-red OR Heat sink-red	Not applicable	Containment-red
OR		OR		OR	

Table B-7. PWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad or RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
2. Primary coolant activity level		2. RCS leak rate		2. Containment pressure	
Coolant activity GREATER THAN (site-specific) value	Not applicable	GREATER THAN available makeup capacity as indicated by a loss of RCS subcooling	Unisolable leak exceeding the capacity of one charging pump in the normal charging mode	Rapid unexplained decrease following initial increase	(Site-specific) psig and increasing
OR		OR		OR	OR
				Containment pressure or sump level response not consistent with LOCA conditions	Explosive mixture exists
					OR
					Containment pressure GREATER THAN containment depressurization system setpoint with less than one full train of depressurization equipment operating
				OR	

Table B-7. PWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
3. Core exit thermocouple readings		3. SG tube rupture		3. Containment isolation valves status after containment isolation	
GREATER THAN (site-specific) degree F	GREATER THAN (site-specific) degree F	(Site-specific) indication that a SG is ruptured and has non-insoluble secondary line break OR (Site-specific) indication that a SG is ruptured and a prolonged release of contaminated secondary coolant is occurring from the affected SG to the environment	(Site-specific) indication that a SG is ruptured and the primary-to-secondary leak rate exceeds the capacity of one charging pump in the normal charging mode.	Valve(s) not closed AND Downstream pathway to the environment exists	Not applicable
OR				OR	
4. Reactor vessel water level				4. SG secondary side release with primary-to-secondary leakage	
Not applicable	Level LESS THAN (site-specific) value			Release of secondary side to atmosphere with primary-to-secondary leakage GREATER THAN tech spec allowable	Not applicable
OR		OR		OR	
		4. Containment radiation monitoring			
		Containment radiation monitor reading GREATER THAN (site-specific) R/h	Not applicable		
		OR			

Table B-7. PWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
5. Containment radiation monitoring		5. Other (site-specific)		5. Significant radioactive inventory in containment	
Containment radiation monitor reading GREATER THAN (site-specific) R/h	Not applicable	(Site-specific) as applicable	(Site-specific) as applicable	Not applicable	Containment radiation monitor reading GREATER THAN (site-specific) R/h
OR		OR		OR	
6. Other (site-specific) indications		6. Emergency Director judgement		6. Core exit thermocouple reading	
(Site-specific) as applicable	(Site-specific) as applicable	Any condition in the judgement of the Emergency Director that indicates loss or potential loss of the RCS barrier		Not applicable	Core exit thermocouples in excess of 1200°F AND restoration procedures not effective within 15 min OR Core exit thermocouples in excess of 700°F with reactor vessel level below top of active fuel and restoration procedures not effective within 15 min.
OR					OR

Table B-7. PWR emergency action level fission product barrier reference table thresholds for loss or potential loss of barriers (continued)

UNUSUAL EVENT	ALERT	SITE AREA EMERGENCY	GENERAL EMERGENCY
ANY loss or ANY potential loss of containment	ANY loss or ANY potential loss of EITHER fuel clad OR RCS	Loss of BOTH fuel clad AND RCS, OR potential loss of BOTH fuel clad AND RCS, OR potential loss of EITHER fuel clad OR RCS AND loss of ANY additional barrier	Loss of ANY two barriers AND potential loss of third barrier

FUEL CLAD BARRIER EXAMPLE EALS		RCS BARRIER EXAMPLE EALS		CONTAINMENT BARRIER EXAMPLE EALS	
LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS	LOSS	POTENTIAL LOSS
7. Emergency Director judgement Any condition in the judgement of the Emergency Director that indicates loss or potential loss of FUEL CLAD barrier				7. Other (site-specific) indications (Site-specific) as applicable	(Site-specific) as applicable
				OR	
				8. Emergency Director judgement Any condition in the judgement of the Emergency Director that indicates loss or potential loss of the CONTAINMENT barrier	

Source: NUMARC/NESP-007, pp. 5-25-5-27.

Table B-8. Event classification for fuel cycle and material facilities

Class/Description	Offsite Consequences	Anticipated Responses
Alert		
Events may occur, are in progress, or have occurred that could lead to a release of radioactive material but the release is not expected to require a response by offsite response organizations to protect persons offsite.	Possible minor releases well below EPA PAG exposure levels. Environmental sampling and some offsite monitoring may be required.	Licensee emergency response personnel secure operations, stop any releases and perform monitoring. State and local organizations notified, inspectors dispatched. Fire department, ambulance and law enforcement respond as required to support onsite response. NRC notified, Regional Operations Center activated and inspectors or site team dispatched. HQ may activate Operations Center. DOE medical support and/or monitoring may be requested.
Site Area Emergency		
Events may occur, are in progress, or have occurred that could lead to a significant release of radioactive material and could require a response by offsite response organizations to protect persons offsite.	Significant release possibly approaching EPA PAG exposure levels. Radiation and contamination levels may require restricting areas offsite. Environmental sampling and offsite monitoring required.	Licensee emergency response personnel secure operations, stop the release, perform monitoring and regain control of radioactive material. State and local organizations notified, emergency personnel respond to site, assess situation, assist monitoring activities and advise the public as required. Fire department, ambulance and law enforcement personnel respond to mitigate consequences, restrict public access to affected areas and support onsite response as required. NRC notified, Operations Center activated and site team dispatched. DOE monitoring support requested. DOE medical support may be requested if required.

SECTION C

REACTOR CONSEQUENCE ASSESSMENT BASED ON PLANT CONDITIONS

Section C
Quick Reference Guide

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Section C
Reactor Accident Consequence Assessment Based on Plant Conditions

Purpose

To estimate offsite consequences based on the status of the reactor core and release pathway conditions.

Discussion

This procedure allows projection of selected radiation doses from reactor accidents before the release of radioactive material or when the release occurs through an unmonitored pathway. These estimates can be used to confirm or modify protective action recommendations. Table C-1 provides a quick summary of the offsite consequences for various reactor conditions.

Step 1

Use Method C.1 to assess the consequences of the accident.

Step 2

Report your assessment of the possible consequences of the reactor accident and the assumptions behind the assessment.

END

Source: NUREG-1228. Some values in this document have been revised to conform to NUREG-1150 findings.

Method C.1
Reactor Accident Consequence Assessment Using Event Trees

Purpose

To estimate offsite consequences based on the status of the reactor core and on release pathway conditions.

Discussion

This method uses event trees containing precalculated dose estimates to determine the offsite consequences of a reactor accident. This method is designed to provide a best estimate of the dose when the source term is not known (before a release or for a release through an unmonitored pathway). These calculations consider only the plant, release, and atmospheric conditions that have a major (greater than a factor of 10) impact on dose.

Consequence assessments in this method are based on a best estimate of the maximum total acute bone marrow dose (TABD) and maximum thyroid dose (plume center line) to an individual, assuming average weather conditions, a 1-h release, and no sheltering or protection. TABD is considered the most sensitive indication for the onset of early non-thyroid health effects. Thyroid dose is calculated because it provides an indication of the distances at which the EPA early phase PAGs may be exceeded.

Doses include the external and inhalation dose from the passing plume and the dose from exposure to contaminated ground for 24 h. The plant conditions and release conditions having the greatest impact on dose are considered. The dose estimates should be within a factor of 10–100 *if* the plant, release height, and rain conditions are accurately represented.

The steps in this assessment are summarized below:

Step 1	Review assumptions and select release pathway.
Step 2	Locate event tree and determine projected dose.
Step 3	Record doses from event tree.
Step 4	Adjust doses for reactor size.
Step 5	Adjust doses for release duration.
Step 6	Correct doses for reactor shutdown time.
Step 7	Correct dose estimate for distance, release elevation, and rain.
Step 8	Determine distance at which selected consequences are possible.
Step 9	Combine consequence projection and release description for presentation.

Step 1

Review the event tree assumptions, p. C-11, and select the potential release pathway to be considered. Identify the release pathway on the appropriate figure.

PWR dry containment . . . . . Fig. C-1
PWR ice condenser containment . . . . . Fig. C-2
BWR Mark I containment . . . . . Fig. C-3
BWR Mark II containment . . . . . Fig. C-4
BWR Mark III containment . . . . . Fig. C-5

Step 2

Select the appropriate type of release. Once the release type has been selected, locate the corresponding event tree, select appropriate plant conditions, and determine the projected doses.

PWR large dry or subatmospheric containment release
- Gap release . . . . . Fig. C-6
- In-vessel core melt . . . . . Fig. C-7

PWR ice condenser containment release
(If the ice condenser is bypassed or the the ice is exhausted before the release from the core, treat the event as a PWR large dry containment release.)
- Gap release . . . . . Fig. C-8
- In-vessel core melt . . . . . Fig. C-9

PWR steam generator tube rupture (SGTR) release
(If the primary and secondary release pathway are dry, treat the event as a PWR/BWR containment bypass release.)

Normal coolant . . . . . Fig. C-10
Spiked coolant, 100 × non-nobles . . . . . Fig. C-11
Gap release . . . . . Fig. C-12
In-vessel core melt . . . . . Fig. C-13

BWR containment drywell release

Gap release . . . . . Fig. C-14
In-vessel core melt . . . . . Fig. C-15

BWR containment wetwell release
(If the suppression pool is bypassed or if more heat than decay heat is released to the pool or if the suppression pool is actively boiling, treat the event as a BWR containment drywell release.)

Gap release . . . . . Fig. C-16
In-vessel core melt . . . . . Fig. C-17

BWR/PWR containment bypass release

Gap release . . . . . Fig. C-18
In-vessel core melt . . . . . Fig. C-19

Step 3

Record the following doses for a 1-h release from a 1000-MW(e) reactor from the appropriate event tree:

TABD @ 1 mile: _____ rem
Thyroid dose @ 1 mile: _____ rem

Step 4

Adjust doses for reactor size.

Size of reactor: _____ MW(e)

$$(TABD\ @\ 1\ mile)_{reactor} = (TABD\ @\ 1\ mile)_{1000\ MW(e)} \times (reactor\ size)/1000$$

(rem) = (rem) × [MW(e)]/[1000 MW(e)]

$$(thyroid\ dose\ @\ 1\ mile)_{reactor} = (thyroid\ dose\ @\ 1\ mile)_{1000\ MW(e)} \times (reactor\ size)/1000$$

(rem) = (rem) × [MW(e)]/[1000 MW(e)]

Step 5

Adjust the doses for different release durations by multiplying the dose by the release duration in hours. (Do not assume more than a 1-h release for the 100%/h release cases; 1 h is the the maximum possible release time for these cases.)

(TABD @ 1 mile) = (TABD @ 1 mile for 1-h release) × release duration

(rem) = (rem/h) × (h)

(thyroid dose @ 1 mile) = (thyroid dose @ 1 mile for 1 h release) × release duration

(rem) = (rem/h) × (h)

Step 6

If the reactor has been shutdown 1 day or more, use the shutdown time correction factors to reduce the adjusted dose. Choose the appropriate factor from the identified graphs, choosing the curve corresponding to the holdup time.

Acute bone dose after gap release . . . . . Fig. C-20
Acute bone dose after in-vessel severe core damage release . . . . . Fig. C-21
Thyroid dose after release (all core conditions) . . . . . Fig. C-22

TABD shutdown correction factor: _____
Thyroid dose shutdown correction factor: _____

(TABD @ 1 mile)$_{corrected}$ = (TABD @ 1 mile) × shutdown correction factor

(rem)$_{corrected}$ = (rem) × ( )

(thyroid dose @ 1 mile)$_{corrected}$ = (thyroid dose @ 1 mile) × shutdown correction factor

(rem)$_{corrected}$ = (rem) × ( )

Step 7

Estimate the dose at other distances and adjust dose if there has been rain or the release was elevated. Doses at 1, 2, 5, 10, and 25 miles for these conditions can be projected using Method F.5, "Adjusting Dose Projections to Consider Distance, Elevation, and Rain."

Step 8

Because of the great uncertainty in these dose projections, do not use dose values when presenting results. Instead, use the results of Step 7 to identify the distances to which certain consequences might be possible and fill in the blanks below. (When dealing with elevated releases under these assumptions, the maximum dose will occur more than 1 mile downwind from the plant.)

Distance to which early deaths are possible
(TABD > 220 rem) _____ mile
Distance to which vomiting and diarrhea are possible
(TABD > 50 rem) _____ mile
Distance to which EPA early phase PAG may be exceeded
(thyroid dose > 5 rem) _____ mile

Step 9

Combine this assessment with the general description of the release, information on plant conditions, and a markup of one of the reactor diagrams (Figs. C-1–C-5) showing assumed release pathway(s).

END

Assumptions for Reactor Accident Consequence Event Trees

Core Conditions

Four different core conditions can be assumed. These conditions span the entire range of possible core damage states. The amount of fission products assumed to be released is approximately the mean value calculated for a range of core damage accidents.

The first two core conditions, (1) normal coolant leakage and (2) spiked coolant leakage, are used for steam generator tube rupture (SGTR) accidents that do not involve core damage. Normal coolant concentrations are based on an ANSI standard and are shown in Tables C-2 and C-3. Spiked coolant assumes all the non-nobles in the normal coolant increase by a factor of 100 to estimate the maximum spiking sometimes seen with rapid shutdown or depressurization of the primary system.

The two remaining core conditions are based on the amount of core damage: (3) A gap release assumes that all fuel pins have failed, releasing the gaseous fission products contained in the fuel pin gap; and (4) in-vessel core melt assumes that the entire core has melted, releasing a mixture of isotopes believed to be representative for most core melt accidents.[1] The assumed core release fractions are shown in Table C-4.

Release Pathways and Conditions

Figures C-1–C-5 show the simplified release pathways for PWR large dry, PWR ice condenser, BWR Mark I, BWR Mark II, and BWR Mark III containments. For each containment release pathway, the mechanisms that will substantially reduce the release are considered (e.g., containment sprays). The effectiveness of the reduction mechanism used is representative for a range of assumptions. The reduction factors assumed for each reduction mechanism are listed in Table C-5.

A PWR dry containment release and BWR drywell containment release assume a release into the containment which in turn leaks to the atmosphere. The effectiveness of sprays or natural processes (plate-out) can be considered. For the BWR drywell containment release, it is assumed that the majority of the release bypasses the suppression pool. In this containment, the amount of released material may be reduced if it passes through the standby gas treatment system filters.

[1]Previous editions of the *Response Technical Manual* included a fifth case, vessel melt-through. Although vessel melt-through would release additional fission products and increase the projected doses, the protective action recommendations resulting from this situation would be the same as those from the in-vessel core melt case.

A PWR ice condenser containment release assumes either a single pass through the ice (because of fan failure or major containment failure) or recirculation through the ice. Credit for sprays and natural processes can also be taken. If the ice is depleted before core damage occurs, then the PWR dry containment release pathway should be used.

A BWR wetwell containment release assumes a release through the suppression pool. If the release bypasses the suppression pool, then the BWR drywell release pathway should be used. Credit may be taken for a release through the standby gas treatment system filters.

A PWR SGTR release assumes contaminated coolant leaks through the rupture. Steam generator partitioning can be considered as a reduction mechanism. The effectiveness of the condenser may also be considered for releases out of the steam-jet air ejector. If the primary system is dry, then the containment bypass release pathway should be used.

A PWR/BWR containment bypass release assumes a release through a dry pathway from the primary system out of the containment. Only plateout on pipes and filtering (if established) in the release pathway can be considered.

Release Rates

The release rates were chosen to provide estimates for the total range of possible rates. The assumed release rates and resulting escape fractions are listed in Table C-6.

Containment leakage rates include (1) catastrophic failure, releasing most of the fission products promptly (in about 1 h for a 1 ft^2 hole at design pressure), (2) 100%/day, which is a traditional assumption for a failure to isolate containment, and (3) design leakage.

The SGTR leakage rates are for failure of one tube at full pressure (500 gal/min) and for the failure of one tube at low-pressure with coolant being pushed out of the break by one charging pump (50 gal/min).

Dose Calculation

Doses at 1 mile are calculated with RASCAL 2.1 assuming a 1-h ground level release, building wake, and average meteorological conditions (4 mph, no rain, and D stability). Total acute bone dose (TABD) includes 1 h of cloudshine, acute (30-day committed) inhalation dose, and 24 h of groundshine. Radioactive decay and in-growth are included. Thyroid doses are for adults from inhalation only.

Basic Source Term Calculation Method

The following is a summary of the method used to approximate severe accident source terms; see NUREG-1228 for a full description.

(1) Estimate the amount of fission products in the core.
(2) Estimate the fraction of the fission product inventory released from the core for a normal coolant, spiked coolant, gap release, or in-vessel core melt.
(3) Estimate the fraction of the fission product inventory released from the core that is removed on the way to the environment.
(4) Estimate the fraction of the available fission product inventory actually released to the environment.

Source estimation for the event trees calculations was done using the following formula:

$$Source\ Term_i = FPI_i \times CRF_i \times \prod_{j=1}^{n} RDF_{(i,j)} \times EF_i$$

for radionuclide *i* and *n* reduction mechanisms.

$$FPI_i\ (Tables\ C\text{-}2,\ C\text{-}3,\ C\text{-}7) = Isotope\ inventory\ (coolant,\ core)$$

$$RDF_{ij}\ (Table\ C\text{-}5) = \frac{Ci\ available\ for\ release\ following\ reduction\ mechanism\ j}{Ci\ before\ reduction\ mechanism\ j}$$

$$CRF_i\ (Table\ C\text{-}4) = \frac{Amount\ of\ isotope\ i\ released\ out\ of\ core}{Core\ inventory\ of\ isotope\ i}$$

$$EF_i\ (Table\ C\text{-}6) = \frac{Ci\ released\ to\ environment}{Ci\ available\ for\ release}$$

[The maximum reduction allowed (minimum value of IIRDF) is 0.001.]

Table C-1. Possible offsite consequences within a few hours after the start of a release

Conditions			Results	
Core	Release		Distance at which EPA early phase PAGs exceeded (miles)	Threshold for acute health effects[a] exceeded
Spikes or no damage	• Lesser coolant release (50 gal/min)		None	None
	• Major coolant release (>500 gal/min)		<2	None
Gap release (uncovered 15–30 min)	• Design leak in containment		None	None
	• Major leak in containment		<5	None
	• Total failure of containment		>10	Thyroid effects
Melted or severely heated (uncovered >30 min)	• Design leak in containment		<2	None
	• Major leak in containment		>10	Thyroid effects
	Total containment failure >2 h after release from vessel	Mitigated[b] or unmitigated	>10	Radiation sickness
	Total containment failure <2 h after release from vessel	Mitigated[b]	>10	Radiation sickness
		Unmitigated	>10	Deaths

[a]Effects from high dose rates delivered over a short period of exposure (see Table G-2) at the site boundary.

[b]Mitigated by sprays, filters and/or through pool. Releases with hold-up times >2 h are assumed to be mitigated by plateout in containment.

Table C-2. PWR coolant concentrations

Nuclide	Normal concentration (Ci/g)
^{3}H	1.0E−06
^{54}Mn	1.6E−09
^{58}Co	4.6E−09
^{60}Co	5.3E−10
^{85}Kr	4.3E−07
^{85m}Kr	9.6E−10
^{87}Kr	5.2E−12
^{88}Kr	6.4E−09
^{89}Sr	4.7E−09
^{90}Sr	7.5E−09
^{91}Sr	1.6E−07
^{91}Y	1.5E−07
^{99}Mo	2.8E−07
^{99}Tc	1.4E−10
^{103}Ru	1.2E−11
^{106}Ru	9.6E−08
^{129m}Te	1.9E−10
^{131m}Te	1.5E−09
^{132}Te	1.7E−09
^{127}Sb	0
^{129}Sb	0
^{131}I	4.5E−08
^{132}I	2.1E−07
^{133}I	1.4E−07
^{134}I	3.4E−07
^{135}I	2.6E−07
^{131m}Xe	7.3E−07
^{133}Xe	2.6E−06
^{133m}Xe	7.0E−08
^{135}Xe	8.5E−07
^{138}Xe	1.2E−07
^{134}Cs	7.1E−09
^{136}Cs	8.7E−10
^{137}Cs	9.4E−09
^{140}Ba	1.3E−08
^{140}La	2.5E−08
^{144}Ce	4.0E−09
^{239}Np	2.2E−09

Source: ANSI/ANS 18.1, 1984.

Table C-3. BWR coolant concentrations

Nuclide	Normal concentration (Ci/g)
^{3}H	1.0E−08
^{54}Mn	7.0E−11
^{58}Co	2.0E−10
^{60}Co	4.0E−10
^{85}Kr	0
^{85m}Kr	0
^{87}Kr	0
^{88}Kr	0
^{89}Sr	1.0E−10
^{90}Sr	7.0E−12
^{91}Sr	4.0E−09
^{91}Y	4.0E−11
^{99}Mo	2.0E−09
^{99}Tc	2.0E−09
^{103}Ru	2.0E−11
^{106}Ru	3.0E−12
^{129m}Te	4.0E−11
^{131m}Te	1.0E−10
^{132}Te	1.0E−11
^{127}Sb	0
^{129}Sb	0
^{131}I	2.2E−09
^{132}I	2.2E−08
^{133}I	1.5E−08
^{134}I	4.3E−08
^{135}I	2.2E−08
^{131m}Xe	0
^{133}Xe	0
^{133m}Xe	0
^{135}Xe	0
^{138}Xe	0
^{134}Cs	3.0E−11
^{136}Cs	2.0E−11
^{137}Cs	8.0E−11
^{140}Ba	4.0E−11
^{140}La	4.0E−10
^{144}Ce	3.0E−12
^{239}Np	8.0E−09

Source: ANSI/ANS 18.1, 1984.

Table C-4. Core release fractions (CRF)[a]

Core condition	Fuel cladding temperature (°F)	Element	Core release fraction
Fuel pin cladding intact (normal leakage)	600		Normal coolant[b]
Spikes resulting from rapid shutdown or depressurization (core remains covered)	600		100× normal coolant[c]
Gap release (cladding failure, core uncovered 15–30 min)	1300–2100	Xe, Kr	0.05
		I	0.05
		Cs	0.05
In-vessel severe core damage (core uncovered >30 min)	>3000	Xe, Kr	0.95
		I, Br	0.35
		Cs	0.25
		Te, Sb, Se	0.15
		Ba	0.04
		Sr	0.03
		Ce, Np, Pu	0.01
		Ru, Mo, Tc, Rh, Pd	0.008
		La, Y, Pm, Zr, Nd, Eu, Nb, Pr, Sm	0.002
Vessel melt through[d]	>3000	Xe, Kr	0.95
		I, Br	0.64
		Cs	0.64
		Te, Sb, Se	0.44
		Ba	0.14
		Sr	0.15
		Ce, Np, Pu	0.03
		Ru, Mo, Tc, Rh, Pd	0.012
		La, Y, Pm, Zr, Nd, Eu, Nb, Pr, Sm	0.017

[a]The core release fraction is the fraction of each element that is assumed to be released from the core for different core damage states [CRF = (Ci released from core)/(Ci in core)]. It is assumed that the entire core is in one state. The fractions are mean estimates for the range of core damage accidents.

[b]Coolant concentration assuming the core remains covered. See Tables C-2 and C-3 for normal concentrations. Normal concentration is based on ANSI/ANS-18:1, 1984.

[c]Spikes assume that all the non-noble concentrations are 100 times higher than normal. A 100 times increase is a reasonable upper bound if the core remains covered.

[d]Assume that the core melts through the vessel before the start of the release.

Source: NUREG-1465, Table 3.12.

Table C-5. Reduction factors (RDF)

Reduction mechanism	Reduction factor[a]
Standby gas treatment system filters	
Dry-low pressure flow	0.01
Wet-high pressure flow (blowout)	1.00
Other filters	
Dry-low pressure flow	0.01
Wet-high pressure flow (blowout)	1.00
Suppression pool scrubbing	
Slow study flow (decay heat)	0.01
Pool subcooled	0.05
Pool saturated	1.00
Pool bypass	
Removal of suspended aerosols and particulates	
Natural processes (no sprays)	0.75[b]
<1 h holdup time	0.36[b]
2- to 12-h holdup time	0.03[b]
24-h holdup time	
Sprays on	0.03
<1 h holdup time	0.02
2 to 12-h holdup time	0.01[b]
24-h holdup time	
Ice condenser	
One pass through condenser (no recirculation)	0.50
Continual recirculation through condenser (1 h or more)	0.25
Ice bed exhausted before core damage	1.00
Primary system retention (plateout)	
Bypass accidents only	0.20[b]
Steam generator partitioning (liquid release from reactor cooling system)	
Partitioned	0.02
Not partitioned	0.50
Air ejector	0.02

[a]This list contains representative reduction factors [RDF = (Ci available for release after reduction mechanism)/(Ci available for release before reduction mechanism)] for the mechanisms that should have the greatest impact on fission products traveling from the core to the environment. These RDFs are for fission products carried by a dry gas stream (gas or steam) except for SGTR partitioning. *These mechanisms apply only to non-noble gases.* The total reduction can be estimated by multiplying the RDFs together. However, the minimum RDF allowed is 0.001.

[b]Values adjusted to be representative of NUREG-1150.

Source: NUREG-1228, except values noted by *b*.

Table C-6. Escape fractions (EF)

Release pathway	Escape fraction[a]
Primary containment failure leakage	
Typical design leakage at design pressure	
PWR—large dry (0.1%/day)	4E−05
PWR—subatmospheric (0.1%/day)	4E−05
PWR—ice condenser (0.25%/day)	1E−04
BWRs (0.5%/day)	2E−04
Failure to isolate (100%/day)	
Failure of isolation valve seal	0.04
Catastrophic failures	
1-h puff release	1.00
Steam generator tube rupture	
1 tube at full pressure (coolant leak 500 gal/min)	0.35
1 tube at low-pressure, single charging pump flow (coolant leak 50 gal/min)	0.03

[a]Fraction of containment volume or primary system coolant inventory released in 1 h.

Source: NUREG-1228, p. 4-37.

Table C-7. Fission and activation product inventory (FPI) in LWR core about 30 min after shutdown[a]

Fission product	Inventory [Ci/MW(e)]	Inventory [Ci/1000 MW(e)]
$^{85}Kr^{b}$	5.6E+02	5.6E+05
$^{85m}Kr^{b}$	2.4E+04	2.4E+07
$^{87}Kr^{b}$	4.7E+04	4.7E+07
$^{88}Kr^{b}$	6.8E+04	6.8E+07
^{86}Rb	2.6E+01	2.6E+04
$^{89}Sr^{b}$	9.4E+04	9.4E+07
$^{90}Sr^{b}$	3.7E+03	3.7E+06
$^{91}Sr^{b}$	1.1E+05	1.1E+08
^{90}Y	3.9E+03	3.9E+06
$^{91}Y^{b}$	1.2E+05	1.2E+08
^{95}Zr	1.5E+05	1.5E+08
^{97}Zr	1.5E+05	1.5E+08
^{95}Nb	1.5E+05	1.5E+08
$^{99}Mo^{b}$	1.6E+05	1.6E+08
^{99m}Tc	1.4E+05	1.4E+08
$^{103}Ru^{b}$	1.1E+05	1.1E+08
^{105}Ru	7.2E+04	7.2E+07
^{106}Ru	2.5E+04	2.5E+07
^{105}Rh	4.9E+04	4.9E+07
^{127}Te	5.9E+03	5.9E+06
^{127m}Te	1.1E+03	1.1E+06
^{129}Te	3.1E+04	3.1E+07
$^{129m}Te^{b}$	5.3E+03	5.3E+06
$^{131m}Te^{b}$	1.3E+04	1.3E+07
$^{132}Te^{b}$	1.2E+05	1.2E+08
$^{127}Sb^{b}$	6.1E+03	6.1E+06
$^{129}Sb^{b}$	3.3E+04	3.3E+07
$^{131}I^{b}$	8.5E+04	8.5E+07
$^{132}I^{b}$	1.2E+05	1.2E+08
$^{133}I^{b}$	1.7E+05	1.7E+08
$^{134}I^{b}$	1.9E+05	1.9E+08
$^{135}I^{b}$	1.5E+05	1.5E+08
$^{131m}Xe^{b}$	1.0E+03	1.0E+06
$^{133}Xe^{b}$	1.7E+05	1.7E+08
$^{133m}Xe^{b}$	6.0E+03	6.0E+06

Table C-7. Fission and activation product inventory (FPI) in LWR core about 30 min after shutdown (continued)

Fission product	Inventory [Ci/MW(e)]	Inventory [Ci/1000 MW(e)]
$^{135}Xe^{b}$	3.4E+04	3.4E+07
$^{138}Xe^{b}$	1.7E+05	1.7E+08
$^{134}Cs^{b}$	7.5E+03	7.5E+06
$^{136}Cs^{b}$	3.0E+03	3.0E+06
$^{137}Cs^{b}$	4.7E+03	4.7E+06
$^{140}Ba^{b}$	1.6E+05	1.6E+08
$^{140}La^{b}$	1.6E+05	1.6E+08
^{141}Ce	1.5E+05	1.5E+08
^{143}Ce	1.3E+05	1.3E+08
$^{144}Ce^{b}$	8.5E+04	8.5E+07
^{143}Pr	1.3E+05	1.3E+08
^{137}Nd	6.0E+04	6.0E+07
$^{239}Np^{b}$	1.6E+06	1.6E+09
^{238}Pu	5.7E+01	5.7E+04
^{239}Pu	2.1E+01	2.1E+04
^{240}Pu	2.1E+01	2.1E+04
^{241}Pu	3.4E+03	3.4E+06
^{241}Am	1.7E+00	1.7E+03
^{242}Cm	5.0E+02	5.0E+05
^{244}Cm	2.3E+01	2.3E+04

[a]It is assumed that the core is at equilibrium [i.e., has been operating for at least one fueling cycle (18 months)]. This assumption could overestimate the inventory of long-lived fission products for a new core. Only the fission products with half-lives greater than 30 min are considered.

[b]Fission products that should be considered in assessments because they are either a major contributor to early phase dose or they are likely to be released (noble gases).

Source: WASH-1400, Table VI-3-1.

Fig. C-1
PWR dry containment simplified release pathways.

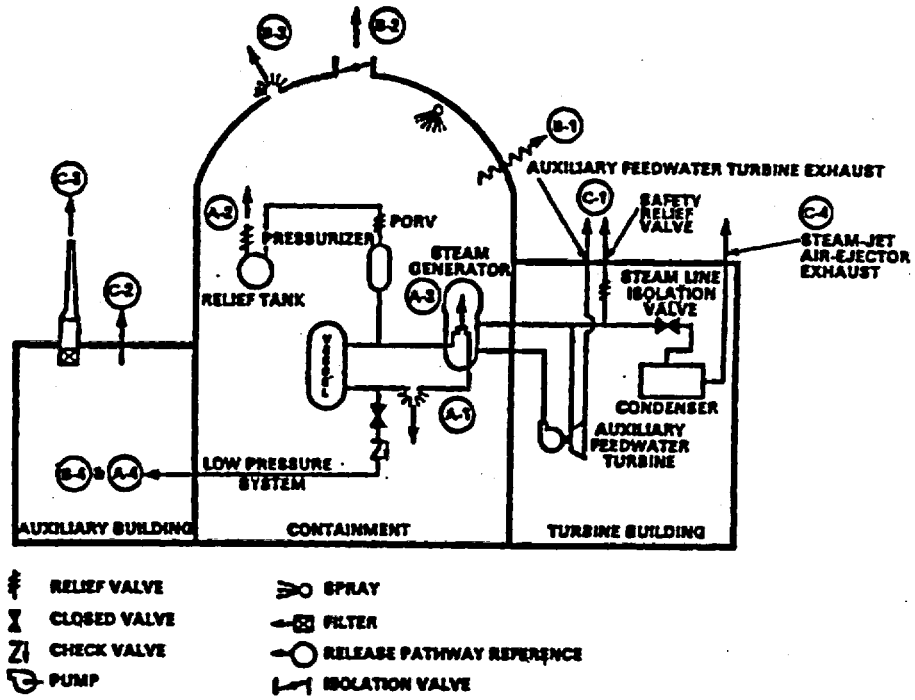


Figure key:

A Reactor coolant system
- A-1 Breaks and leaks
- A-2 Power-operated relief valves (PORVs)
- A-3 Steam generator tube rupture
- A-4 Bypass (failure into low-pressure steam)

B Containment
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic (> 1 ft^2)
- B-4 Bypass

C Other
- C-1 Secondary side relief/safety valve or turbine exhaust
- C-2 Building leakage—unfiltered
- C-3 Building leakage—filtered
- C-4 Condenser steam-jet air-ejector

Fig. C-2
PWR ice condenser containment simplified release pathways.

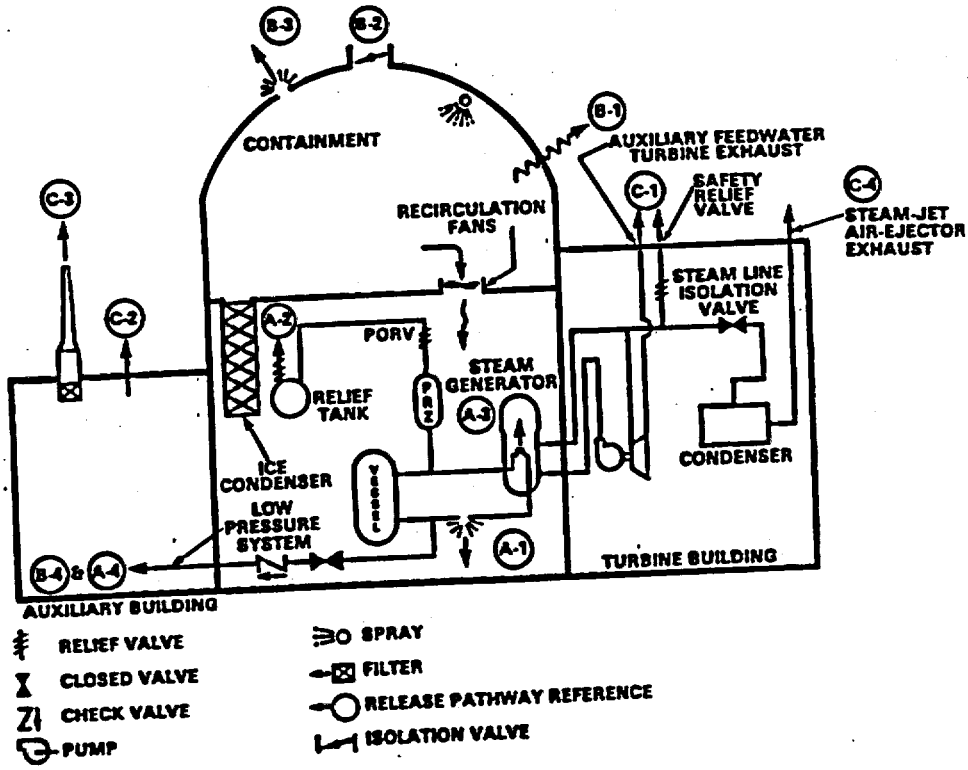


Figure key:

A Reactor coolant system
- A-1 Breaks and leaks
- A-2 Power-operated relief valves (PORVs)
- A-3 Steam generator tube rupture
- A-4 Bypass (failure into low-pressure steam)

B Containment
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic (> 1 ft^2)
- B-4 Bypass

C Other
- C-1 Secondary side relief/safety valve or turbine exhaust
- C-2 Building leakage—unfiltered
- C-3 Building leakage—filtered
- C-4 Condenser steam-jet air-ejector

Fig. C-3
BWR Mark I simplified release pathways.

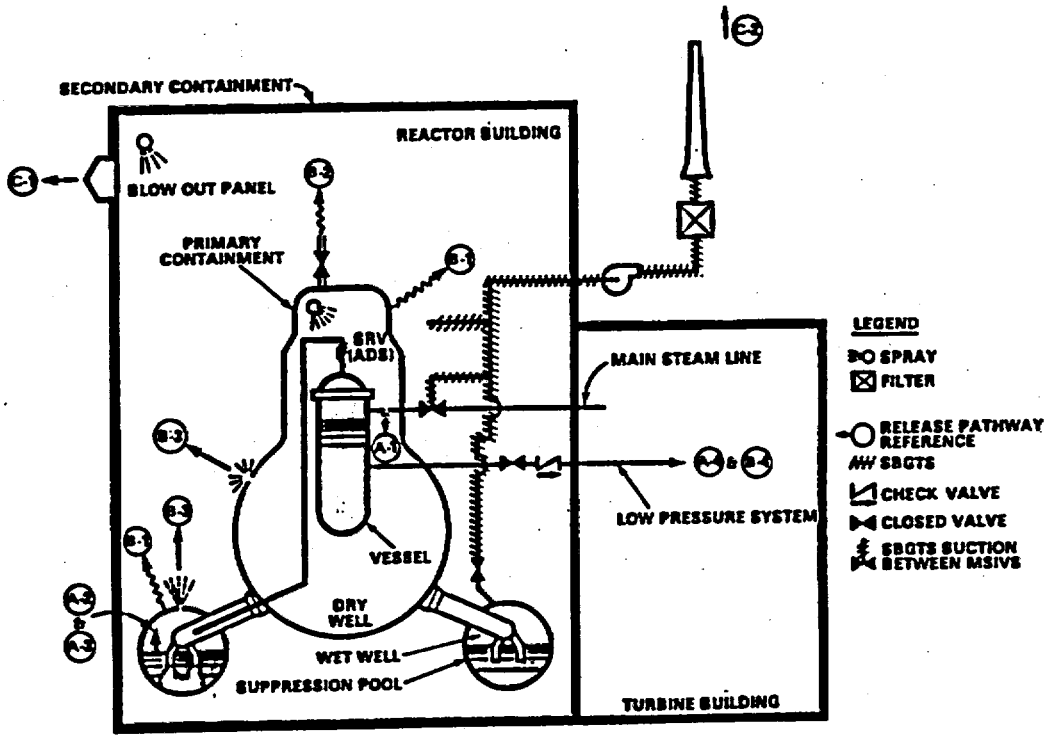


Figure key:

A Reactor coolant system
- A-1 Breaks and leaks bypassing suppression pool
- A-2 Breaks and leaks through suppression pool
- A-3 Automatic depressurization system (ADS) and safety relief valves (SRV)
- A-4 Bypass (interface LOCA)

B Containment
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic
- B-4 Bypass

C Other
- C-1 Building leakage—unfiltered
- C-2 Standby gas treatment system (SBGTS)

Fig. C-4
BWR Mark II simplified release pathways.

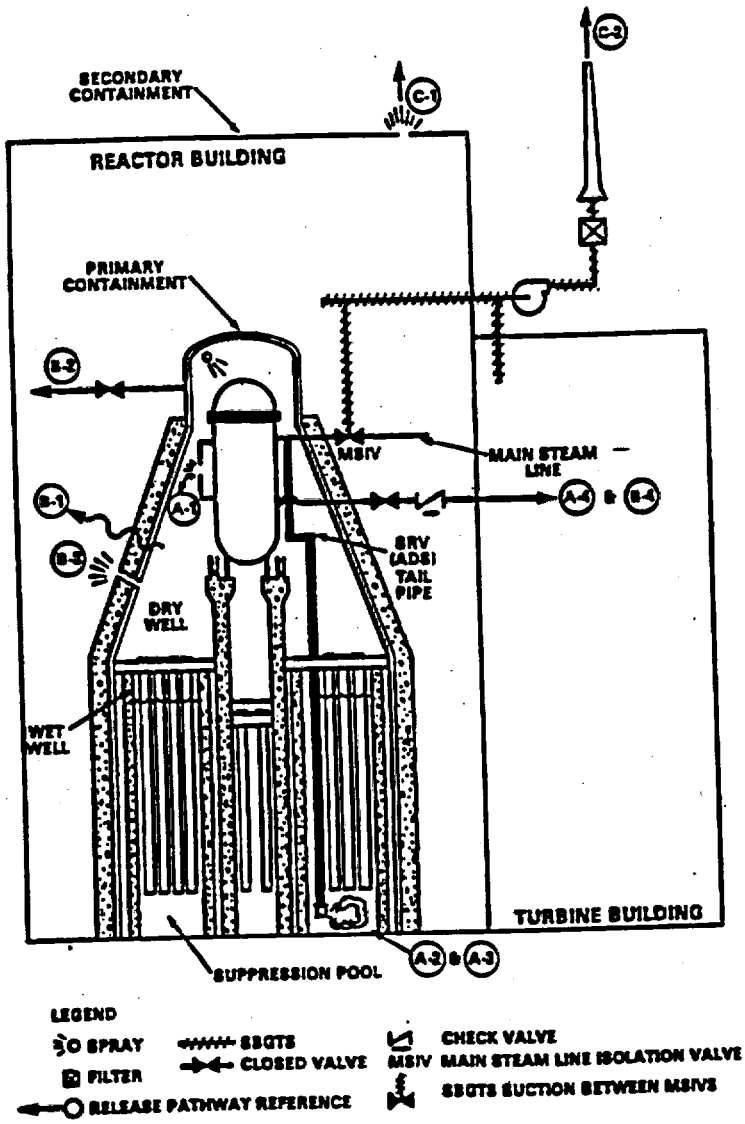


Figure key:

A Reactor coolant system
- A-1 Breaks and leaks bypassing suppression pool
- A-2 Breaks and leaks through suppression pool
- A-3 Automatic depressurization system (ADS) and safety relief valves (SRV)
- A-4 Bypass (interface LOCA)

B Containment
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic
- B-4 Bypass

C Other
- C-1 Building leakage—unfiltered
- C-2 Standby gas treatment system (SBGTS)

Fig. C-5
BWR Mark III simplified release pathways.

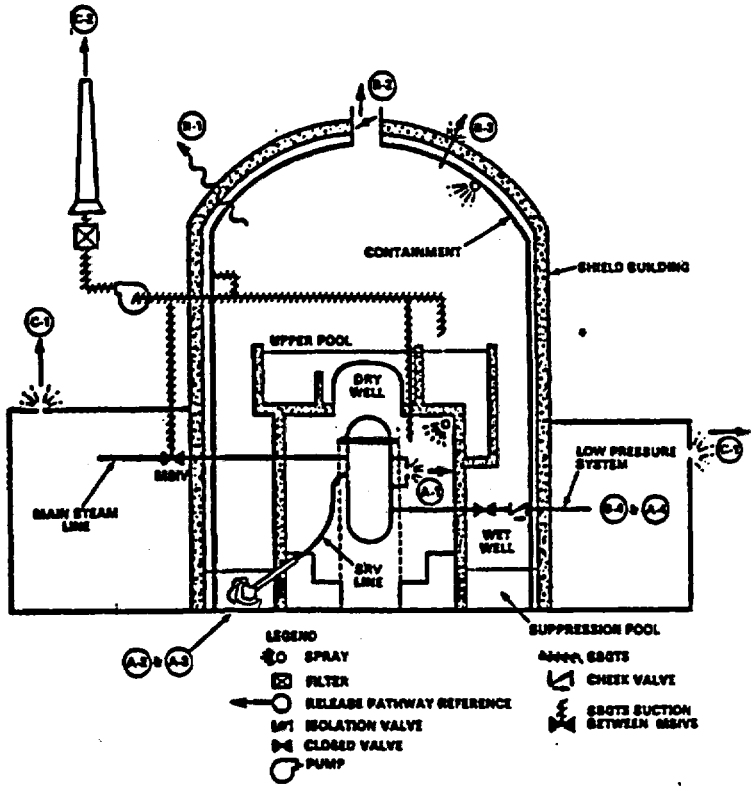


Figure key:

A Reactor coolant system
- A-1 Breaks and leaks bypassing suppression pool
- A-2 Breaks and leaks through suppression pool
- A-3 Automatic depressurization system (ADS) and safety relief valves (SRV)
- A-4 Bypass (interface LOCA)

B Containment
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic
- B-4 Bypass

C Other
- C-1 Building leakage—unfiltered
- C-2 Standby gas treatment system (SBGTS)

Fig. C-6
Dose for PWR large dry or subatmospheric containment release for a gap release.

Core	Containment			Dose[a] at 1 mile (rem)	
Condition	Conditions	Holdup time	Leak rate	TABD	Thyroid
gap release (uncovered 15–30 min)	spray off	≤1 h	100%/h	2E+02	1E+04
		2–12 h	100%/h	6E+01	5E+03
			100%/day	2E+00	2E+02
			design rate	2E−03	2E−01
		>12 h	100%/h	4E+00	4E+02
			100%/day	1E−01	1E+01
			design rate	<1E−03	1E−02
	spray on	≤1 h	100%/h	2E+01	4E+02
		2–12 h	100%/h	7E+00	3E+02
			100%/day	3E−01	1E+01
			design rate	<1E−03	1E−02
		>12 h	100%/h	2E+00	1E+02
			100%/day	6E−02	5E+00
			design rate	<1E−03	5E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The acute bone dose includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

Fig. C-7
Dose for PWR large dry or subatmospheric containment release for an in-vessel core melt release.

Core	Containment			Dose[a] at 1 mile (rem)	
Condition	Condition	Holdup time	Leak rate	TABD	Thyroid
in-vessel core melt (uncovered >30 min)	spray off	≤1 h	100%/h	2E+03	9E+04
		2–12 h	100%/h	6E+02	4E+04
			100%/day	2E+01	2E+03
			design rate	2E−02	2E+00
		>12 h	100%/h	4E+01	3E+03
			100%/day	2E+00	1E+02
			design rate	2E−03	1E−01
	spray on	≤1 h	100%/h	3E+02	3E+03
		2–12 h	100%/h	1E+02	2E+03
			100%/day	4E+00	9E+01
			design rate	4E−03	9E−02
		>12 h	100%/h	2E+01	9E+02
			100%/day	8E−01	4E+01
			design rate	<1E−03	4E−02

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

Fig. C-8
Dose for PWR ice condenser containment release for a gap release.

Core	Containment			Dose[a] at 1 mile (rem)			
				Once through ice (fans off)		With recir. (fans on)	
Condition	Conditions	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
gap release (uncovered 15-30 min)	sprays off	≤1 h	100%/h	1E+02	6E+03	[b]	[b]
		2-12 h	100%/h	3E+01	3E+03	[b]	[b]
			100%/day	1E+00	1E+02	7E−01	5E+01
			design rate	1E−03	1E+01	<1E−03	5E−02
		12 h	100%/h	2E+00	2E+02	[b]	[b]
			100%/day	8E−02	7E+00	5E−02	4E+00
			design rate	<1E−03	7E−03	<1E−03	4E−03
	sprays on	≤1 h	100%/h	2E+01	2E+02	[b]	[b]
		2-12 h	100%/h	6E+00	1E+02	[b]	[b]
			100%/day	2E−01	6E+00	2E−01	3E+00
			design rate	<1E−03	6E−03	<1E−03	3E−03
		>12 h	100%/h	1E+00	6E+01	[b]	[b]
			100%/day	4E−02	2E+00	3E−02	1E+00
			design rate	<1E−03	2E−03	<1E−03	1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]Because of the high release rate, it is assumed that there is no recirculation through ice.

Fig. C-9
Dose for PWR ice condenser containment release for an in-vessel core melt release.

Core	Containment			Dose[a] at 1 mile (rem)			
				Once through ice (fans off)		With recir. (fans on)	
Condition	Conditions	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
in-vessel core melt (uncovered >30 min)	sprays off	≤1 h	100%/h	1E+03	4E+04	[b]	[b]
		2-12 h	100%/h	3E+02	2E+04	[b]	[b]
			100%/day	1E+01	8E+02	8E+00	4E+02
			design rate	1E−02	8E−01	8E−03	4E−01
		>12 h	100%/h	2E+01	1E+03	[b]	[b]
			100%/day	9E−01	5E+01	7E−01	3E+01
			design rate	<1E−03	6E−02	<1E−03	3E−02
	sprays on	≤1 h	100%/h	3E+02	2E+03	[b]	[b]
		2-12 h	100%/h	9E+01	1E+03	[b]	[b]
			100%/day	4E+00	4E+01	3E+00	2E+01
			design rate	4E−03	4E−02	3E−03	2E−02
		>12 h	100%/h	1E+01	5E+02	[b]	[b]
			100%/day	6E−01	2E+01	5E−01	9E+00
			design rate	<1E−03	2E−02	<1E−03	1E−02

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The acute bone dose includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]Because of the high release rate, it is assumed that there is no recirculation through ice.

Fig. C-10
Dose for PWR steam generator tube rupture with a release of normal coolant.

Core/coolant	Steam generator		Dose[a] at 1 mile (rem)			
			Safety valve		Steam jet air ejector	
Conditions	Leak rate	Conditions	TABD	Thyroid	TABD	Thyroid
no core damage, normal coolant	1 tube fails at full pressure (500 gal/min)	not partitioned	<1E−03	9E−03	<1E−03	<1E−03
		partitioned	<1E−03	<1E−03	<1E−03	<1E−03
	1 pump fails at low pressure (50 gal/min)	not partitioned	<1E−03	<1E−03	<1E−03	<1E−03
		partitioned	<1E−03	<1E−03	<1E−03	<1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

Fig. C-11
Dose for PWR steam generator tube rupture with a release of spiked coolant (non-noble spike 100 × normal).

Core/coolant	Steam generator		Dose[a] at 1 mile (rem)			
			Safety valve		Steam jet air ejector	
Condition	Leak rate	Conditions	TABD	Thyroid	TABD	Thyroid
no core damage, coolant 100 × normal non-noble spike	1 tube fails at full pressure (500 gal/min)	not partitioned	6E−02	9E−01	1E−03	2E−02
		partitioned	2E−03	4E−02	<1E−03	2E−03
	1 pump injects at low pressure (50 gal/min)	not partitioned	5E−03	8E−02	<1E−03	2E−03
		partitioned	<1E−03	3E−03	<1E−03	<1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

Fig. C-12
Dose for PWR steam generator tube rupture with a release of coolant contaminated due to a gap release from the core.

Core/coolant	Steam generator		Dose[a] at 1 mile (rem)			
			Safety valve		Steam jet air ejector	
Condition	Leak rate	Conditions	TABD	Thyroid	TABD	Thyroid
gap release concentration (uncovered 15-30 min)	1 tube fails at full pressure (500 gal/min)	not partitioned	6E+01	3E+03	8E+00	5E+01
		partitioned	9E+00	1E+02	7E+00	5E+00
	1 pump injects at low pressure (50 gal/min)	not partitioned	5E+00	2E+02	7E−01	5E+00
		partitioned	8E−01	9E+00	6E−01	5E−01

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

Fig. C-13
Dose for PWR steam generator tube rupture with a release of coolant contaminated due to in-vessel core melt.

Core/coolant	Steam generator		Dose[a] at 1 mile (rem)			
			Safety valve		Steam jet air ejector	
Condition	Leak rate	Conditions	TABD	Thyroid	TABD	Thyroid
in-vessel core melt release concentration (uncovered >30 min)	1 tube fails at full pressure (500 gal/min)	not partitioned	6E+02	2E+04	1E+02	4E+02
		partitioned	2E+02	8E+02	1E+02	4E+01
	1 pump injects at low pressure (50 gal/min)	not partitioned	5E+01	2E+03	1E+01	3E+01
		partitioned	1E+01	7E+01	1E+01	4E+00

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

Fig. C-14
Dose for BWR containment drywell release for a gap release.

Core	Containment			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Condition	Conditions	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
gap release (uncovered 15-30 min)	sprays off	≤1 h	100%/h	2E+02	1E+04	[b]	[b]
		2-12 h	100%/h	6E+01	5E+03	[b]	[b]
			100%/day	2E+00	2E+02	2E−01	2E+00
			design rate	2E−03	2E−01	<1E−03	2E−03
		>12 h	100%/h	4E+00	4E+02	[b]	[b]
			100%/day	1E−01	1E+01	2E−02	1E−01
			design rate	<1E−03	1E−02	<1E−03	<1E−03
	sprays on	≤1 h	100%/h	2E+01	4E+02	[b]	[b]
		2-12 h	100%/h	7E+00	3E+02	[b]	[b]
			100%/day	3E−01	1E+01	2E−01	1E−01
			design rate	<1E−03	1E−02	<1E−03	<1E−03
		>12 h	100%/h	2E+00	1E+02	[b]	[b]
			100%/day	6E−02	5E+00	2E−02	5E−02
			design rate	<1E−03	5E−03	<1E−03	<1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering; filters are assumed to blow out.

Fig. C-15
Dose for BWR containment drywell release for an in-vessel core melt release.

Core	Containment			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Condition	Conditions	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
in-vessel core melt (uncovered >30 min)	sprays off	≤1 h	100%/h	2E+03	9E+04	[b]	[b]
		2-12 h	100%/h	6E+02	4E+04	[b]	[b]
			100%/day	2E+01	2E+03	3E+00	2E+01
			design rate	2E−02	2E+00	3E−03	2E−02
		>12 h	100%/h	4E+01	3E+03	[b]	[b]
			100%/day	2E+00	1E+02	4E−01	1E+00
			design rate	2E−03	1E−01	<1E−03	1E−03
	sprays on	≤1 h	100%/h	3E+02	3E+03	[b]	[b]
		2-12 h	100%/h	1E+02	2E+03	[b]	[b]
			100%/day	4E+00	9E+01	3E+00	9E−01
			design rate	4E−03	9E−02	3E−03	<1E−03
		>12 h	100%/h	2E+01	9E+02	[b]	[b]
			100%/day	7E−01	4E+01	4E−01	4E−01
			design rate	<1E−03	4E−02	<1E−03	<1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering; filters are assumed to blow out.

Fig. C-16
Dose for BWR containment wetwell release for a gap release.

Core	Containment			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Condition	Supression pool conditions	Holdup time in dry/wet well	Wet well leak rate	TABD	Thyroid	TABD	Thyroid
gap release (uncovered 15-30 min)	saturated (no sprays)	≤1 h	100%/h	2E+01	6E+02	[b]	[b]
		2-12 h	100%/h	7E+00	3E+02	[b]	[b]
			100%/day	3E−01	1E+01	2E−01	1E−01
			design rate	<1E−03	1E−02	<1E−03	<1E−03
		>12 h	100%/h	6E−01	2E+01	[b]	[b]
			100%/day	3E−02	7E−01	2E−02	7E−03
			design rate	<1E−03	<1E−03	<1E−03	<1E−03
	subcooled (no sprays)	≤1 h	100%/h	1E+01	1E+02	[b]	[b]
		2-12 h	100%/h	5E+00	5E+01	[b]	[b]
			100%/day	2E−01	2E+00	2E−01	2E−02
			design rate	<1E−03	2E−03	<1E−03	<1E−03
		>12 h	100%/h	6E−01	1E+01	[b]	[b]
			100%/day	2E−02	5E−01	2E−02	5E−03
			design rate	<1E−03	<1E−03	<1E−03	<1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering; filters are assumed to blow out.

Fig. C-17
Dose for BWR containment wetwell release for an in-vessel core melt release.

Core	Containment			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Condition	Suppression pool conditions	Holdup time in dry/wet well	Wet well leak rate	TABD	Thyroid	TABD	Thyroid
in-vessel core melt (uncovered >30 min)	saturated (no sprays)	≤1 h	100%/h	3E+02	4E+03	[b]	[b]
		2-12 h	100%/h	1E+02	2E+03	[b]	[b]
			100%/day	4E+00	8E+01	3E+00	8E−01
			design rate	4E−03	8E−02	3E−03	<1E−03
		>12 h	100%/h	1E+01	1E+02	[b]	[b]
			100%/day	4E−01	5E+00	4E−01	6E−02
			design rate	<1E−03	6E−03	<1E−03	<1E−03
	subcooled (no sprays)	≤1 h	100%/h	2E+02	9E+02	[b]	[b]
		2-12 h	100%/h	8E+01	4E+02	[b]	[b]
			100%/day	3E+00	2E+01	3E+00	2E−01
			design rate	3E−03	2E−02	3E−03	<1E−03
		>12 h	100%/h	1E+01	9E+01	[b]	[b]
			100%/day	4E−01	4E+00	4E−01	5E−02
			design rate	<1E−03	4E−03	<1E−03	<1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering; filters are assumed to blow out.

Fig. C-18
Dose for BWR/PWR containment bypass release for a gap release.

Core	Containment		Dose[a] at 1 mile (rem)	
Condition	Release conditions	Release rate	TABD	Thyroid
gap release (uncovered 15–30 min)	not filtered	100%/h	8E+01	3E+03
		100%/day	3E+00	1E+02
		0.1%/day	3E−03	1E−01
	filtered	100%/h	[b]	[b]
		100%/day	8E−01	1E+00
		0.1%/day	<1E−03	1E−03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering; filters are assumed to blow out.

Fig. C-19
Dose for BWR/PWR containment bypass release for an in-vessel core melt release.

Core	Containment		Bone dose[a] at 1 mile (rem)	
Condition	Release conditions	Release rate	TABD	Thyroid
in-vessel core melt (uncovered >30 min)	not filtered	100%/h	9E+02	2E+04
		100%/day	4E+01	9E+02
		0.1%/day	4E−02	1E+00
	filtered	100%/h	[b]	[b]
		100%/day	2E+01	9E+00
		0.1%/day	2E−02	1E−02

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The total acute bone dose (TABD) includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering; filters are assumed to blow out.

Fig. C-20
Shutdown time correction factors for total acute bone dose (TABD) after gap release.

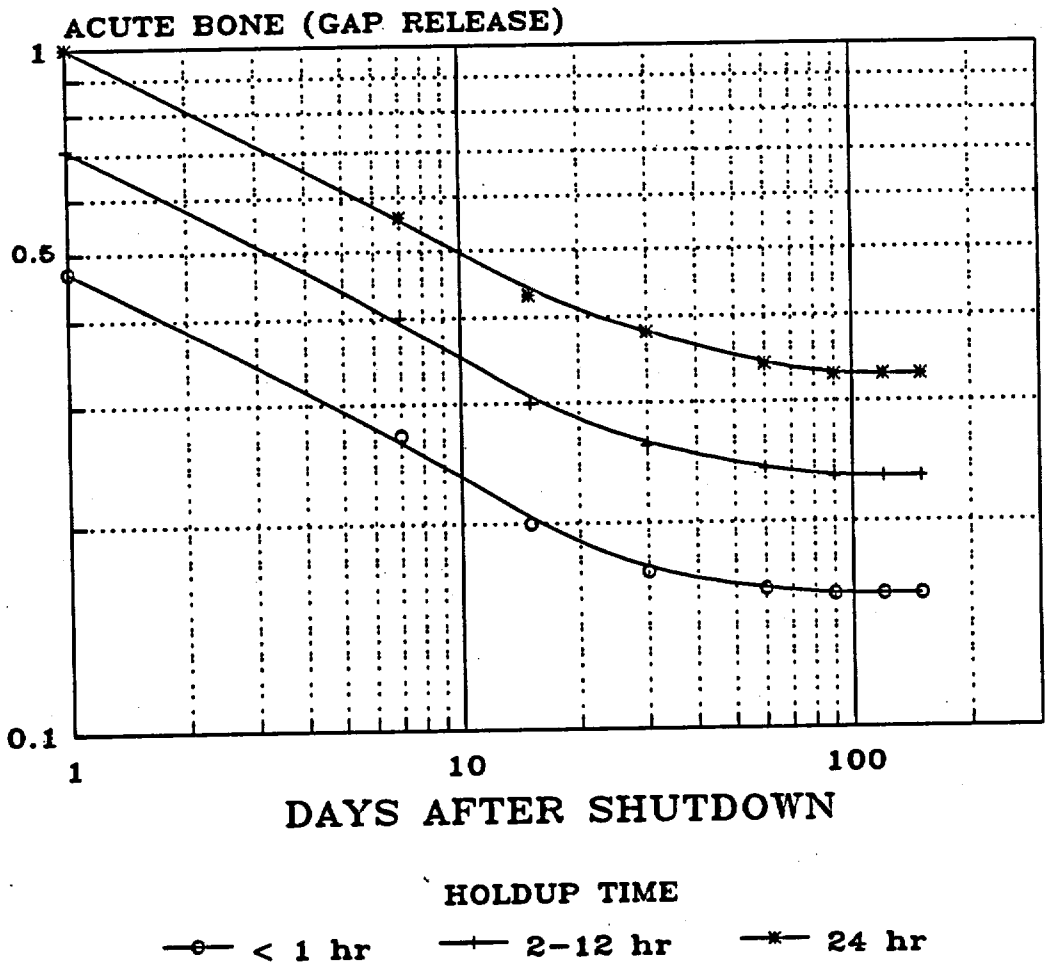


Fig. C-21
Shutdown time correction factors for total acute bone dose (TABD) after in-vessel core melt release.

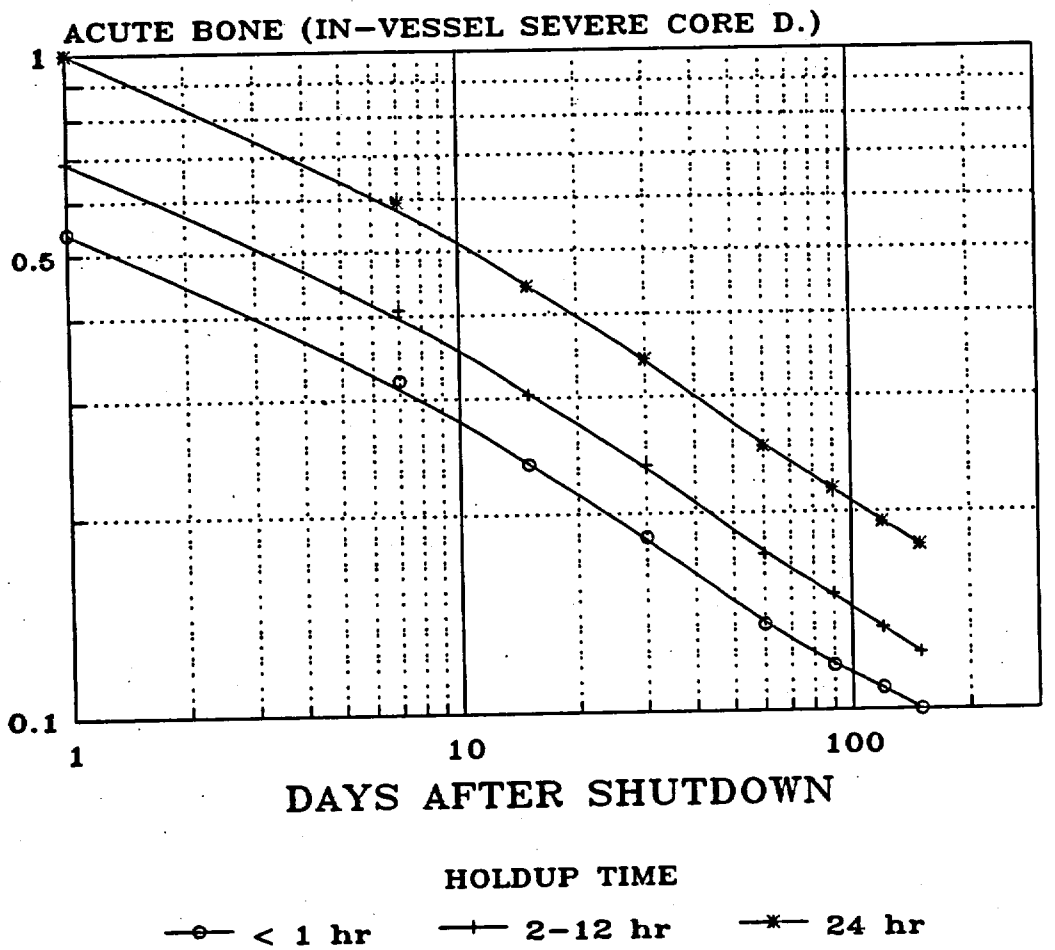
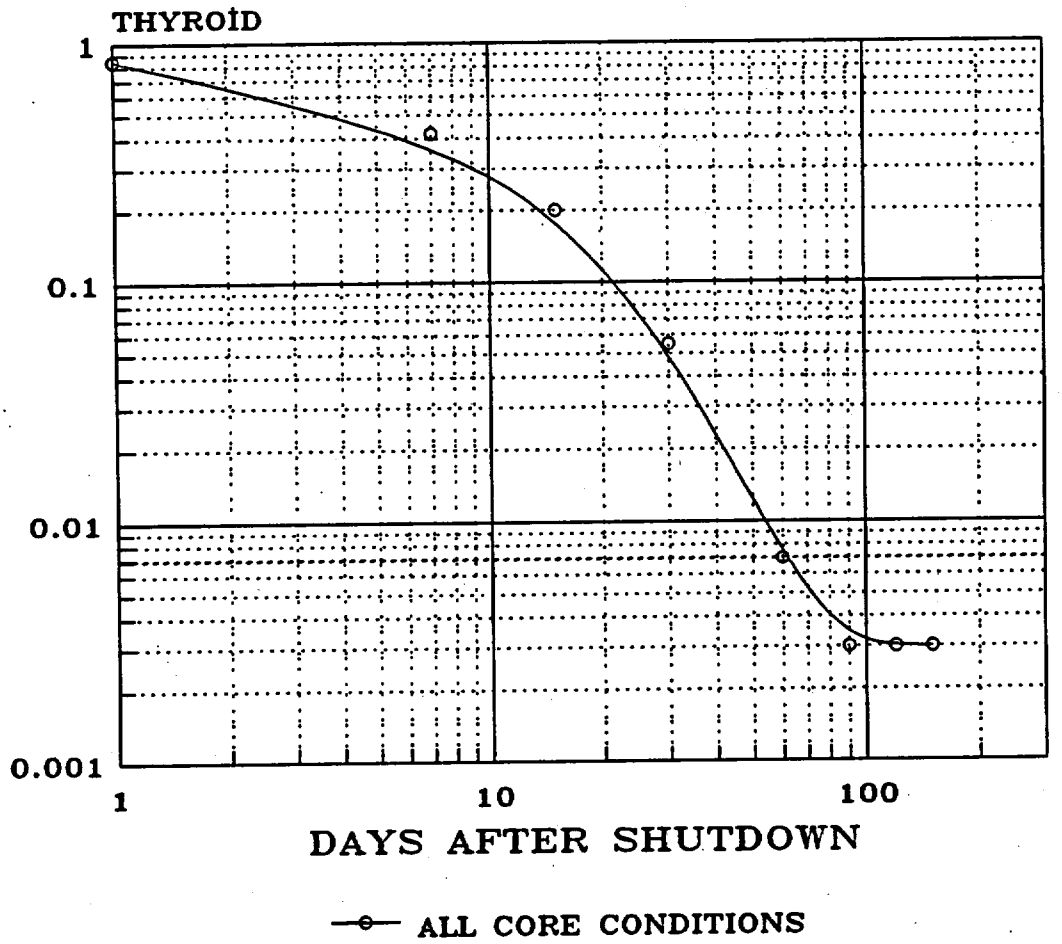


Fig. C-22
Shutdown time correction factors for thyroid dose after release (all core conditions).



SECTION D

SPENT FUEL POOL DAMAGE AND CONSEQUENCE ASSESSMENT

Section D
Quick Reference Guide

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Section D
Spent Fuel Pool Damage and Consequence Assessment

Purpose

To assess accidents involving loss of coolant to a spent fuel pool.

Discussion

Accidents involving the loss of coolant in the spent fuel pool may have offsite consequences because of damage to the fuel from overheating. Two types of damage may occur: (1) a Zircaloy cladding fire resulting in substantial release of fission products from recently discharged fuel and (2) cladding failure with release of the fission products in the fuel pin gap.

Fuel damage may be prevented if 100–250 gal/min of water can be sprayed on the pool, beginning within 1 h of draining the pool. This flow rate can be achieved with fire hoses. Use Fig. D-1 to estimate the dose from direct radiation from a drained pool (this estimate may be needed to protect those responding near the pool).

Step 1

Estimate the time to drain the pool (or boil off the water) using Table D-1.

Step 2

Estimate potential spent fuel damage. Consider the following:

- The spent fuel pool must be virtually drained for substantial damage to occur. Pools are considered coolable as long as 20% of the fuel is covered.

- Cladding failure with release of the fission products in the fuel pin gap is possible within 2 h to several days after the pool is drained. It is assumed that the pin will heat up before failure, releasing about 5% of the volatile fission products (i.e., typical gap release fractions).

- After the pool has been drained, a Zircaloy cladding fire resulting in release of a substantial amount of the volatile fission products (in-vessel core melt release fraction—i.e., 25% of cesium) is possible in BWR fuel for 30–250 days after shutdown (30-180 days for PWR). A Zircaloy cladding fire is likely to propagate to adjacent fuel bundles discharged within the last 2 years.

Step 3

Estimate potential offsite consequences using Method D.1.

Step 4

Report your assessment of the possible consequences of the reactor accident and the assumptions behind the assessment.

END

Sources: NUREG/CR-0649, NUREG-1353.

Method D.1
Spent Fuel Pool Accident Consequence Assessment Using Event Trees

Purpose

To estimate offsite consequences based on the status and age of the fuel in the spent fuel pool and on release pathway conditions.

Discussion

This method uses event trees containing precalculated dose estimates to determine the offsite consequences of a release from damaged fuel in a spent fuel pool. This method is designed to provide a best estimate of the dose when the source term is not known (before a release or for a release through an unmonitored pathway, such as a building pressure valve). These calculations consider only the fuel conditions, release, and atmospheric conditions that have a major (greater than a factor of 10) impact on dose.

Consequence assessments in this method are based on a best estimate of the maximum total acute bone marrow dose (TABD) and maximum thyroid dose (plume center line) to an individual, assuming average weather conditions, a 1-h release, and no sheltering or protection. TABD is considered the most sensitive indication for the onset of early non-thyroid health effects. Thyroid dose is calculated because it provides an indication of the distances at which the EPA early phase PAGs may be exceeded.

Doses were calculated using RASCAL 2.1 and include the external and inhalation dose from the passing plume and the dose from exposure to contaminated ground for 24 h. The dose estimates should be within a factor of 10–100 *if* the spent fuel pool and rain conditions are accurately represented.

The steps in this assessment are summarized below:

Step 1	Locate event tree and determine projected dose.
Step 2	Record doses from event tree.
Step 3	Adjust doses for number of batches if necessary.
Step 4	Adjust doses for release duration.
Step 5	Correct dose estimate for distance, release elevation, and rain.
Step 6	Determine distance at which selected consequences are possible.
Step 7	Combine consequence projection and release description for presentation.

Step 1

Select the type of release to be considered, locate the corresponding event tree, select appropriate pool and release conditions, and determine the projected doses. Doses can be adjusted later for the number of batches.

Zircaloy fire from one 3-month-old batch . . . . . Fig. D-2
Gap release from one 3-month-old batch . . . . . Fig. D-3
Gap release from 15 1-year-old batches . . . . . Fig. D-4

Step 2

Record the following doses for a 1-h release from the appropriate event tree:

TABD at 1 mile: _____ rem
Thyroid dose at 1 mile: _____ rem

Step 3

Adjust doses for the number of batches in pool. Multiply the TABD at 1 mile and the thyroid dose at 1 mile from Fig. D-2 and Fig. D-3 by the number of batches (reloads) in pool. (A batch is one-third of the core, the amount typically removed during refueling.) Note that the calculation in Fig. D-4 is for *15 batches*, instead of a single batch. In that case, this step may not be needed.

(TABD at 1 mile) = (TABD at 1 mile for 1 batch) × number of batches in pool

(rem) = (rem) × ( )

(thyroid dose at 1 mile) = (thyroid dose at 1 mile for 1 batch) × number of batches in pool

(rem) = (rem) × ( )

Step 4

Adjust the doses for different release durations by multiplying the dose by the release duration in hours. (Do not assume more than a 1-h release for the 100%/h release cases; 1 h is the maximum possible release time for these cases.)

(TABD at 1 mile) = (TABD at 1 mile for 1-h release) × release duration

(rem) = (rem/h) × (h)

(thyroid dose at 1 mile) = (thyroid dose at 1 mile for 1-h release) × release duration

(rem) = (rem/h) × (h)

Step 5

Estimate the dose at 1, 2, 5, 10, and 25 miles for a ground level or elevated release with or without rain, as appropriate, using Method F.5, "Adjusting Dose Projections to Consider Distance, Elevation, and Rain."

Step 6

Because of the great uncertainty, do not use dose numbers when presenting results. Instead, use the results of Step 5 to identify the distances to which certain consequences might be possible and fill in the blanks below. (When dealing with elevated releases under these assumptions, the maximum dose will be further than 1 mile away from the plant.)

Distance to which early deaths are possible
(TABD > 220 rem): _____ miles
Distance to which vomiting and diarrhea are possible
(TABD > 50 rem): _____ miles
Distance to which EPA early phase PAG may be exceeded
(thyroid dose > 5 rem) _____ miles

Step 7

Combine this assessment with the general description of the release.

END

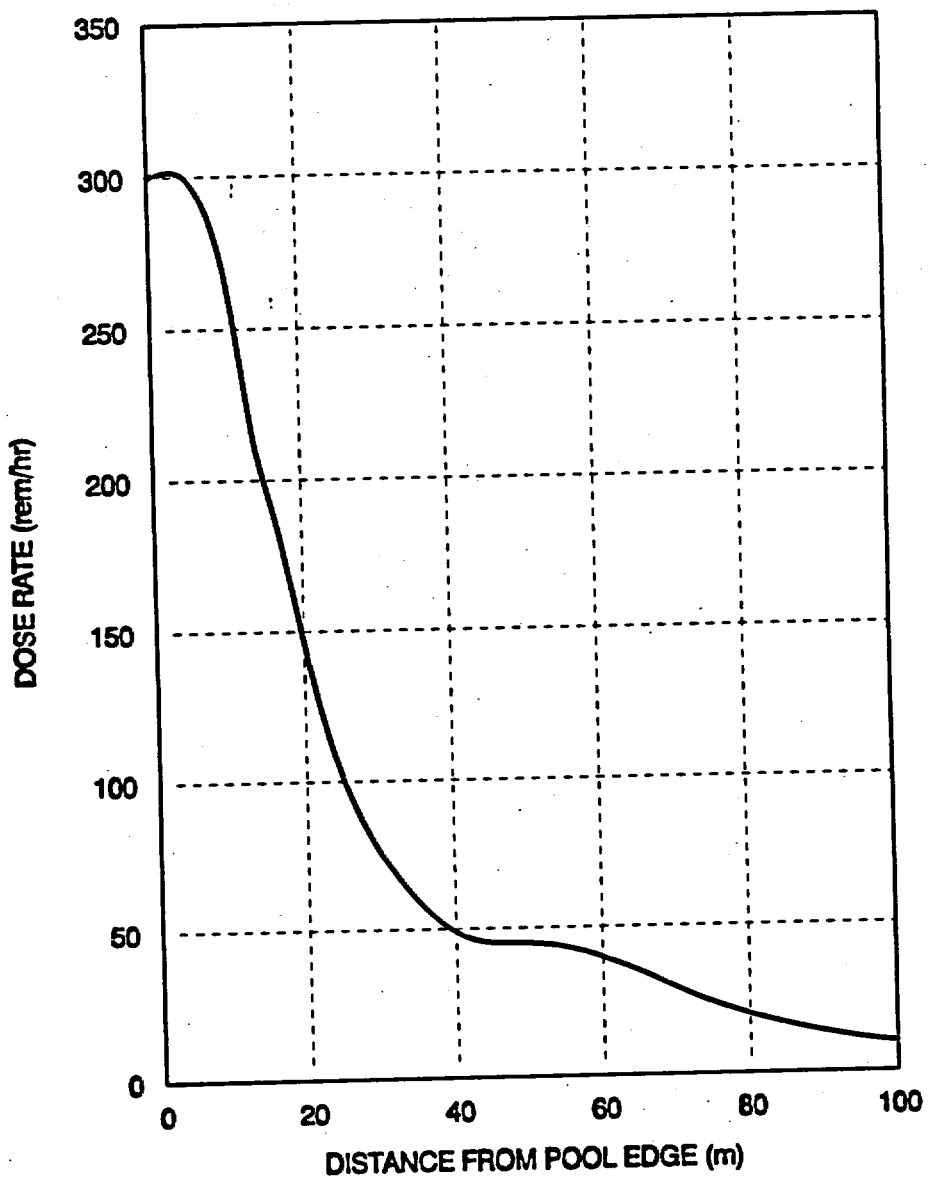
Table D-1. Heatup and boil-dry times for a typical spent fuel pool

	One-third of core recently discharged + 20 years of accumulated discharges			Full core recently discharged + 20 years of accumulated discharges		
Days after shutdown[a]	Time to heat from 125°F to 212°F[b] (h)	Time to boil off water[c] (h)	Water to make up boil-off (gal/min)	Time to heat from 150°F to 212°F[b] (h)	Time to boil off water[c] (h)	Water to make up boil-off (gal/min)
5	11.2	125.0	31.9	3.1	49.3	81.0
10	13.9	154.9	25.8	4.1	63.8	62.6
30	19.0	212.2	18.8	6.1	95.8	41.7
45	21.8	242.8	16.4	7.4	115.5	34.5
65	24.3	270.4	14.8	8.6	135.2	29.5
100	27.5	306.5	13.0	10.5	164.2	24.3
150	32.0	357.1	11.2	13.6	212.6	18.8
200	35.1	391.2	10.2	16.1	251.9	15.8
250	37.2	414.5	9.6	18.1	282.6	14.1
300	38.4	428.3	9.3	19.3	302.4	13.2
350	39.2	437.5	9.1	20.2	316.6	12.6
365	39.3	438.6	9,1	20.4	318.4	12.5

[a]Days after shutdown of core recently discharged.
[b]52°C to 100°C.
[c]To drain the pool.

Source: NUREG-1353.

Fig. D-1
Whole body gamma ground level dose rate from drained spent fuel pool.[a]



[a]30 days after one fuel core discharged. Rest of material is 1, 2, or 3 years after discharge.

Source: NUREG/CR-0649.

Fig. D-2
BWR/PWR spent fuel pool release event tree for a Zircaloy fire in one 3-month-old batch of fuel.

Fuel	Release			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Conditions	Pool condition	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
Zircaloy fire from one 3-month-old batch	sprays off	≤1 h	100%/h	3E+01	8E+01	[b]	[b]
			100%/day	1E+00	3E+00	1E-02	3E-02
		2-12 h	100%/h	1E+01	4E+01	[b]	[b]
			100%/day	5E-01	2E+00	5E-03	2E-02
	sprays on	≤1 h	100%/h	1E+00	3E+00	[b]	[b]
			100%/day	4E-02	1E-01	<1E-03	1E-03
		2-12 h	100%/h	7E-01	2E+00	[b]	[b]
			100%/day	3E-02	8E-02	<1E-03	<1E-03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The acute bone dose includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering is assumed at this leak rate.

Fig. D-3
BWR/PWR spent fuel pool release event tree
for a gap release from one 3-month-old batch of fuel.

Fuel	Release			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Condition	Pool conditions	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
gap release from one 3-month-old batch	sprays off	≤1 h	100%/h	3E+00	1E+01	[b]	[b]
			100%/day	1E-01	5E-01	1E-03	5E-03
		2-12 h	100%/h	2E+00	6E+00	[b]	[b]
			100%/day	6E-02	3E-01	<1E-03	3E-03
	sprays on	≤1 h	100%/h	1E-01	5E-01	[b]	[b]
			100%/day	5E-03	2E-02	<1E-03	<1E-03
		2-12 h	100%/h	9E-02	4E-01	[b]	[b]
			100%/day	4E-03	1E-02	<1E-03	<1E-03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The acute bone dose includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering is assumed at this leak rate.

Fig. D-4
BWR/PWR spent fuel pool release event tree for a gap release from 15 1-year-old batches of fuel.

Fuel	Release			Dose[a] at 1 mile (rem)			
				Not filtered		Filtered	
Condition	Pool conditions	Holdup time	Leak rate	TABD	Thyroid	TABD	Thyroid
gap release from 15 1-year-old batches	sprays off	≤1 h	100%/h	3E+01	1E+02	[b]	[b]
			100%/day	1E+00	4E+00	1E-02	4E-02
		2-12 h	100%/h	1E+01	5E+01	[b]	[b]
			100%/day	5E-01	2E+00	5E-03	2E-02
	sprays on	≤1 h	100%/h	1E+00	4E+00	[b]	[b]
			100%/day	4E-02	2E-01	<1E-03	2E-03
		2-12 h	100%/h	7E-01	3E+00	[b]	[b]
			100%/day	3E-02	1E-01	<1E-03	1E-03

[a]Dose calculations reflect a 1-h ground level release with average meteorological conditions (D stability, 4 mph, no rain) and the effect of building wake. The acute bone dose includes 1 h of inhalation, 1 h of cloudshine, and 24 h of groundshine to an adult performing normal activities. The thyroid dose includes 1 h of inhalation exposure to an adult.

[b]No filtering is assumed at this leak rate.

SECTION E

URANIUM HEXAFLUORIDE RELEASE ASSESSMENT

Section E
Quick Reference Guide

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Section E
Uranium Hexafluoride Release Assessment

Purpose

To assess the possible consequences of a uranium hexafluoride (UF_6) release and determine the need for protective actions.

Discussion

UF_6 is a readily dispersible form of uranium that is produced in uranium conversion plants, shipped to uranium enrichment plants, enriched in the ^{235}U isotope, and then shipped to fuel fabrication plants to be processed into nuclear fuel. A large accidental release is most likely when large, hot cylinders of UF_6 are handled at a processing facility or a large cylinder is involved in a fire.

Released UF_6 gas reacts vigorously with water vapor in the air, producing uranyl fluoride (UO_2F_2), hydrogen fluoride (HF), and excess heat. If there is sufficient humidity, the reaction products may be hydrates of UO_2F_2 and HF-H_2O fog, which are seen as a white cloud. The chemical toxicity of a UF_6 release dominates the radiological risks. The permissible exposure levels for soluble uranium compounds are based on chemical toxicity. UO_2F_2 is a particulate that is very soluble in the lungs, and the uranium acts as a heavy metal poison that can affect the kidneys. The hydrogen fluoride is an acid vapor that can cause acid burns on the skin or lungs if it is concentrated. Toxic levels can be reached in minutes, so immediate protective actions should be taken when a release is possible. Do not let your assessment impede ongoing protective actions at the site.

Chemically lethal or toxic airborne releases of natural and low-enriched soluble uranium would not produce enough radiation to exceed the PAGs beyond the area where there is a chemical hazard. After making any immediate protective action recommendations, estimate consequences attributable to chemical toxicity before considering any radiological threat.

If the release is underway, avoid contact with the plume consisting of UF_6 and its reaction products. A highly concentrated plume of UF_6 may be visible and immediately irritating to the lungs. Stay out of low areas. Evacuation out of the plume and/or sheltering may be appropriate. Cooling the source of the leak and misting the plume with water will significantly reduce the amount of material that becomes airborne.

The reaction of the UF_6 with water vapor produces HF, which is an extremely corrosive acid. Inhaling less than 1 gram of soluble uranium may be fatal, and contact of HF with the skin may cause burns. Anyone who contacts the plume should be examined for HF burns and low-level radioactive contamination and observed for several days following the accident for any delayed health effects resulting from renal uptake of soluble uranium.

The steps in this procedure are summarized below:

Step 1	Determine the amount and form of UF_6 available for release.
Step 2	Assess the need for immediate protective actions.
Step 3	Project the uranium intake and HF concentration downwind.
Step 4	Evaluate the potential health effects attributable to chemical toxicity and determine protective actions.
Step 5	Project the committed effective dose equivalent downwind.
Step 6	Compare committed effective dose equivalent to EPA PAGs.
Step 7	Recommend or adjust protective action recommendations.

Step 1

Determine the amount and physical form (gas or liquid) of UF_6 available for release, using information from the licensee or by checking the cylinder type in Table E-1. Liquid UF_6 will vaporize, but not all of the UF_6 will become airborne. UF_6 gas may be found in heated cylinders, releases in fires, or when the gas is being transferred.

Step 2

Assess the need for immediate protective actions. If the amount of UF_6 available for release is less than 0.5 metric ton (500 kg), recommend protective actions up to 0.5 mile (800 m). For a larger quantity of UF_6, recommend protective actions up to 1.0 mile (1600 m).[1] If the release has occurred, is occurring, or seems immediately imminent, sheltering may be an appropriate initial action until responders determine when and where evacuation is appropriate.

Step 3

Estimate the potential toxic effects of the incident, based on projected integrated intake of soluble uranium (IU_{sol}) and the projected HF concentration (χ_{HF}) at downwind distances of interest. Roughly estimate the uranium intake and HF

[1]NUREG-1140 considers the rupture outdoors of a heated "14-ton" UF_6 cylinder, releasing 9,500 kg UF_6, to be the maximum credible accident and finds a 1-mile evacuation appropriate to prevent fatalities and permanent injuries. Values for 0.5 ton were taken from Fig. E-2 for HF concentrations in the 16–24 mg/m^3 range.

concentration from a liquid UF_6 release using default parameters in Method E.1. To calculate projected values for a UF_6 gas release or for a liquid release using incident-specific information or other meteorological conditions, use Method E.2.

Step 4

Compare the projected integrated intake of soluble uranium (IU_{sol}) and the projected HF concentration (χ_{HF}) with the health effects values indicated in Table E-2 and Table E-3, respectively. Evaluate the potential health effects. Consider the duration of the HF exposure. If the duration of the exposure is short with low concentrations, there may be no significant effects. If the exposure is such that there may be significant health effects, recommend protective measures and postpone the evaluation of any radiological impact.

Figure E-1 (based on the same assumptions as those in Method E.1) provides an indication of the distance downwind to which the Immediately Dangerous to Life and Health (IDLH) concentration of HF might be reached. The distance at which the PAGs might be exceeded for *highly enriched* uranium are also indicated.

Step 5

Estimate the committed effective dose equivalent ($H_{E,50}$) for the desired downwind distance. Use Method E.3 to estimate this value for the desired distance and enrichment level under default assumptions for a liquid UF_6 release, or use Method E.4 to project the dose for a UF_6 gas release or to consider incident-specific parameters or other meteorology for a liquid release.

Step 6

Compare the committed effective dose equivalent ($H_{E,50}$) to the EPA early phase PAGs in Table G-1 (or to any State-specific PAGs).

Step 7

Recommend appropriate protective actions or adjust protective action recommendations based on comparisons in Step 4 and Step 6. Discuss these recommendations with licensee. Consult with other Federal agencies (EPA, HHS, and USDA and DOE for gaseous diffusion plants) if time permits.

END

Sources: NUREG-1140, NUREG-1391, DOT P 5800.5, ORO-651.