U.S. Nuclear Regulatory Commission

Vol. 1, Rev. 1

RTM-91

Response Technical Manual

Division of Operational Assessment

Office for Analysis and Evaluation of Operational Data

NUREG/BR-0150, Vol. 1, Rev. 1

U.S. Nuclear Regulatory Commission

RTM-91

Response Technical Manual

April 1991

T. McKenna, K. Jackson, G. Brown, W. Reckley,
R. Hogan, J. Jolicoeur, K. Gant (ORNL), B. Burnett (FDA)

Division of Operational Assessment
Office for Analysis and Evaluation
of Operational Data

RESPONSE TECHNICAL MANUAL (RTM-91)
VOLUME 1, REV. 1

Table of Contents

Page

ACKNOWLEDGEMENTS

The following individuals contributed substantially to the body of this report: Robert Bores, NRC Region I, Anne Boland, NRC Region II, Eugene Bates, NRC Region IV, and Emilio Garcia, NRC Region V.

PREFACE

<u>This is not a controlled copy.</u>

This document is a training version of the RESPONSE TECHNICAL MANUAL. This version may be incomplete and/or out of date. Each of the NRC response organizations that has responsibility to use these procedures during response to an accident has controlled copies of this manual. Only controlled copies of the manual should be used during an accident.

Controlled copies of this manual are maintained by the Incident Response Branch, Office for Analysis and Evaluation of Operational Data, Division of Operational Assessment at NRC Headquarters and by the individual Emergency Response Coordinators at the regional offices. Please direct any questions concerning this manual to the Incident Response Branch, AEOD.

AEOD plans to publish updated versions of this manual and accompanying workbook (Volume 2) once every two years.

NOTE:

This Response Technical Manual (RTM-91) replaces RTM-90 Volume 1, NUREG/BR-0150, dated June 1990, as the Agency's basis for response technical training.

USE OF THE MANUAL

This manual contains procedures used by the NRC when responding to accidents. In using the procedures follow these rules:

1. Use only the procedures on which you are trained.

2. Complete each step in order.

3. Read all "CAUTIONS."

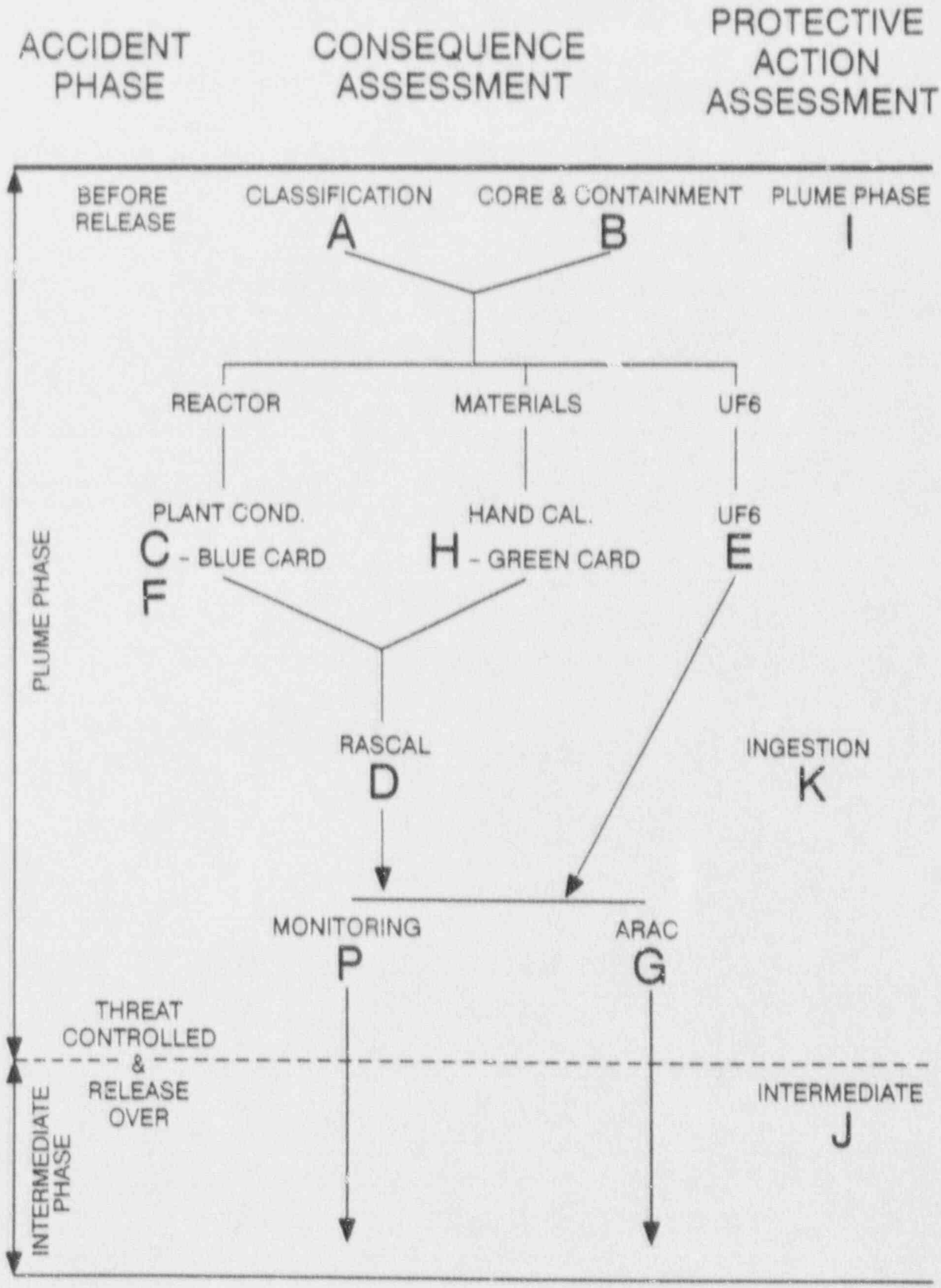
4. Read the "NOTES" if additional information is needed.

The diagram on page vi "Assessment Tool Overview" shows the typical sequence in which the major procedures would be used relative to the phases of an accident. Consequence Assessment Procedure Overview and Protective Action Assessment Procedure Overview (pages vii, viii) provide a summary of these procedures.

There are other procedures and reference materials in the manual that may also be helpful. The back cover provides a summary of the contents and provides "thumb" tabs for quick reference.

When in doubt, use the index.

ASSESSMENT TOOL OVERVIEW



CONSEQUENCE ASSESSMENT PROCEDURE OVERVIEW

Section A, Classification Assessment

Differentiate accidents requiring immediate protective actions (general emergencies) from those that can wait for further assessments (e.g., field measurements).

Section B, Core and Containment Assessment

Aids in assessing core and containment conditions.

Section C, Reactor Accident Consequence Assessment Based on Plant Conditions or Blue Card

Initially bound consequences (health effects and EPA PAG comparisons) based on possible reactor accident conditions.

Section D, Use of RASCAL

Standard NRC accident consequence / dose projection model (reactor conditions and isotopic releases).

Section E, UF6 Release Assessment

Assessment of consequence and protective actions for a UF6 accident.

Section F, Long Distance, Elevated Release and Rain Dose Projections for Severe Reactor Accidents

Dose projections out to great distances for severe reactor accidents that show the influence of elevated releases and rain.

Section G, Use of ARAC

Advanced transport model. Requires about an hour to produce end result and we must specify source term (release information). Can project to great distances.

Section H, Plume Dose and Source Term Hand Calculation or Green Card

Calculate dose or source term from field measurements.

Section P, DOE Radiological Field Measurements Assistance

How to get DOE field monitoring assistance. Use ASAP.

PROTECTIVE ACTION ASSESSMENT PROCEDURE OVERVIEW

Section I, Plume Phase General Population Protective Action Assessment

Immediate actions taken before or shortly after a release. Actions to protect the population from the plume - prevent early health effects and reduce dose above EPA PAGs.

Section J, Intermediate Phase General Population Protective Action Assessment (Relocation - Reentry)

Actions taken to relax or adjust protective actions taken using the plume phase (Section I) procedures.

Section K, Ingestion Pathway Protective Action Assessment

Actions to protect the public from ingestion of contaminated food or water.

OTHER PROCEDURES

Section L, Exposure Control for NRC During Rescue and Recovery Activities

Section M, Use of Potassium Iodide (KI) and Thyroid Monitoring

Section N, Use of NRC TLD System Direct Radiation Monitoring Network

Section O, Medical Assistance

Section Q, SI Units and Conversions

Section R, Putting Radiation in Perspective for the Public

Section S, Half-Life Table

Classification Assessment

Section A

A Classification Assessment

CLASSIFICATION ASSESSMENT

Objective

Assess licensee classification of the accident.

Guidance

Step 1

Conduct an assessment depending on whether the facility is a **reactor (A)** or **fuel cycle (B)** facility:

A. **Reactor**

NOTE:

Most reactors will use NUREG-0654 method.

NOTE:

If the licensee classification does not appear to be correct, check the licensee classification procedure and/or discuss with licensee.

- **NUREG-0654 Method** - use the NUREG-0654 Quick Assessment initially - consult the NUREG-0654 guidance if necessary.

END

THIS PAGE INTENTIONALLY LEFT BLANK

NUREG-0654 QUICK ASSESSMENT METHOD INDEX

NOTE:

The quick assessment chart is based on NUREG-0654. The items apply to both PWRs and BWRs unless noted.

TYPE	UNUSUAL EVENT	ALERT	SITE AREA	GENERAL
REACTIVITY CONTROL LOSS		Failure to completely shutdown reactor	Transient requiring shutdown and failure to shutdown	Transient requiring shutdown, failure to shutdown and - failure of ECCS or - indication of core damage
INVENTORY CONTROL LOSS	ECCS starts and injects water into reactor	Primary system leak rate > 50 gpm	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 temp. - coolant pres. - fuel temp. Loss of engineered safety feature system requiring shutdown by Tech Specs	Complete loss of function required for cold shutdown 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

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 gpm 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 shutdown
CONTROL ROOM		Evacuation of control room & control of shut-down system established	Evacuation of control room & control of shut-down system not established within 15 min.	
INSTRUMENTS & ALARMS	Loss of instru-ments & 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	

TYPE	UNUSUAL EVENT	ALERT	SITE AREA	GENERAL
FUEL CLADDING LOSS	>0.1% clad failure in 30 min.	> 1.0% cladding failure in 30 min. or 5% total clad failure	Degraded core with possible loss of coolable geometry	Actual or projected >100% cladding failure equivalent
	BWR high radioactivity in offgas or reactor coolant	BWR very high radioactivity in offgas (>5 Ci/s). Reactor coolant (>300 μCi/cc)		Any sequence that could lead to severe heatup of the core
	Coolant activity > Tech Spec			
PRIMARY SYSTEM BREAKS & LEAKS	Primary system leak > Tech Spec	Primary system leak rate > 50 gpm	Primary system leak > makeup system capability (LOCA)	Primary system leak > makeup and failure of ECCS
	PWR steam generator tube leak > Tech Specs	PWR rapid steam generator tube rupture(s) and - loss of offsite power or - leak > 300 gpm	PWR rapid steam generator tube rupture > 300 gpm & loss of offsite power	Events leading to prolonged core uncovery
	PWR rapid loss of secondary side pressure		PWR steam line break & steam generator tube rupture leak > 50 gpm and cladding failure	
	Stuck open code safety or power operated relief valves	PWR steam line break and steam generator tube leak > 10 gpm		
		BWR steam line break inside containment without MSIV closure	BWR steam line break outside containment without MSIV closure	Loss of 2 of 3 fission product barriers & potential loss of the 3rd barrier

TYPE	UNUSUAL EVENT	ALERT	SITE AREA	GENERAL
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 2 of 3 fission product barriers & potential loss of the 3rd 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 X instantaneous limits Inplant radiation levels > 1000 X normal	Whole body dose projection assuming adverse meteorological conditions indicate >50 mR/hr for 30 min. or 500 mR/hr 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
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

TYPE	UNUSUAL EVENT	ALERT	SITE AREA	GENERAL
FIRE	Plant fire lasting > 10 minutes 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
HAZARDS	Actual or projected hazards - earthquakes - floods - hurricane - tornado - explosion - gas release - aircraft crash - derailment	Actual or projected severe hazards	Severe natural phenomena or hazard and plant not in cold shutdown - any event > 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
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			

NOTE:

NUREG-0654 Appendix A

NOTE:

Material related to protective actions has been deleted. That information is superseded by Section I.

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 (staying inside) 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	Licensee Actions	State and/or Local Offsite Authority Actions
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.	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	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; <u>or</u> 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.

THIS PAGE INTENTIONALLY LEFT BLANK

Class	Licensee Actions	State and/or Local Offsite Authority Actions
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.	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	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 and 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

THIS PAGE INTENTIONALLY LEFT BLANK

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 <u>general emergency</u> 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 <u>general emergency</u> 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

THIS PAGE INTENTIONALLY LEFT BLANK

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.

BARRIER CHALLENGE OR LOSS METHOD

CAUTION:

> This method is only used if the "barrier method" of classification is used by the plant. Most plants use the NUREG-0654 method.

<u>Step 2</u>

CAUTION:

> Use redundant indications to verify or replace (if required) parameters listed on the table.

NOTE:

> - "Possible" indicates loss/challenge to a barrier but consideration of other parameters is required.
> - Use parameter trends. If a parameter is approaching emergency action level criteria and mitigation systems are unavailable, assume the barrier will be lost.

Using either the PWR (page A-27) or BWR (page A-28) "Barrier Challenge or Loss Indicator Matrix" to complete the following:

	Challenged	Loss
Cladding		
Reactor Coolant System (RCS)		
Containment		

<u>Step 3</u>

Classify the event using the following rules:

<u>Class</u>	<u>Rules</u>
Alert	Challenge to or loss of either fuel cladding <u>or</u> RCS
Site Area Emergency	Challenge to or loss of both fuel cladding <u>and</u> RCS <u>or</u> Any challenge to or loss of either fuel cladding <u>or</u> RCS <u>and</u> loss of containment
General Emergency	Loss of any two barriers <u>and</u> challenge to the third barrier <u>or</u> Severe loss of the cladding (core melt seqeuence)

PWR
BARRIER CHALLENGE OR LOSS INDICATOR MATRIX

	FUEL CLADDING		REACTOR COOLANT SYSTEM		CONTAINMENT	
	CHALLENGED	LOST	CHALLENGED	LOST	CHALLENGED	LOST
Core Exit Thermocouples	High	Very High	High	Very High	Very High	---
Reactor Vessel Level	Low	Very Low	Low	Very Low	Very Low	---
Reactor Coolant Leak Rate	---	---	>50 gpm	>Charging Capacity	---	---
Reactor Coolant Pressure	---	---	Very High or Low	Low	---	---
Steam Generator Tube Rupture	---	---	Yes	Possible	Possible	Possible
Reactor Coolant Activity	>1% Failed Fuel	>20% Failed Fuel	---	---	---	---
Containment Pressure & Temperature	---	---	Possible	Possible	High	>>Design Unexplained Decrease
Containment Radiation	High	Very High	High	Very High	Very High	---
Containment Sump Level	---	---	Possible	Possible	Increasing	Not Incr w/other LOCA Signs
Containment Hydrogen	---	High	---	High	High	Possible
Containment Isolation	---	---	---	---	Possible	Possible

BWR
BARRIER CHALLENGE OR LOSS INDICATOR MATRIX

	FUEL CLADDING		REACTOR COOLANT SYSTEM		CONTAINMENT	
	CHALLENGED	LOST	CHALLENGED	LOST	CHALLENGED	LOST
Reactor Vessel Level	Low	Very Low	Low	Very Low	Very Low	---
Offgas/ Steamline Radiation	High	Very High	Very High	---	---	---
Reactor Coolant Leak Rate	---	---	>50 gpm	---	---	---
Reactor Coolant Pressure	---	---	Very High or Low	Low	---	---
Reactor Coolant Activity	>1% Failed Fuel	>20% Failed Fuel	---	---	---	---
Drywell Pressure & Temperature	---	---	High	Very High	High	>>Design Unexplained Decrease
Drywell & Containment Radiation	High	Very High	High	Very High	Very High	---
Drywell & Containment Hydrogen	---	High	---	High	High	Possible
Suppression Pool Level	---	---	Possible	Possible	Possible	Possible
Suppression Pool Temperature	Possible	---	Possible	Possible	Possible	Possible
Containment Isolation	---	---	---	---	Possible	Possible

THIS PAGE INTENTIONALLY LEFT BLANK

Fuel Cycle and Material Incident Emergency Classes

NOTE:

> There are no Unusual Event or General Emergency classifications for non-reactors.

NOTE:

> Not all fuel cycle and materials facilities have implemented emergency plans; some facilities do not have plans because of the small quantity of material they handle. The classification procedure would not apply if there were no plans.

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 Police 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.

Fuel Cycle and Material Incident Emergency Classes - Continued

Class/Description	Offsite Consequences	Anticipated Responses
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 Police 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.

Core and Containment Assessment

Section B

B Core and Containment Assessment

CORE AND CONTAINMENT ASSESSMENT

Objective

Assess the level of reactor core damage or containment conditions.

Guidance

Step 1

Conduct an assessment using:

END

GENERAL CAUTIONS ON USE OF INSTRUMENTS FOR CORE DAMAGE ASSESSMENT

GENERAL CAUTION:

For core damage assessment, never use only one instrument as the basis.

CAUTIONS

PWR Core Exit Thermocouples (CET)

- Once the core is uncovered, CET readings will be considerably lower than the average and maximum core temperatures.

- CET should be used to confirm that there is insufficient injection of water to keep the core covered (see Saturation Table on page B-13). Based on an estimate of the time the core is uncovered, use Core Damage Progression Once Uncovered, page B-4, to estimate damage.

BWR Water Level

- High drywell temperature can cause the BWR reactor water level to read erroneously high due to (1) decreased density in the reference leg or (2) reference leg water flashing to steam if the reactor is depressurized.

  Source: NUREG/CR-2726

- Mechanical Yarway instruments may indicate a false on-scale water level of about 1 ft above the top of core if the actual water level fell below the lower end of the instrument range.

  Source: ORNL/NRC/LTR-90/18

Radiation Level

- Release may bypass the monitor.

- Monitors may be influenced by a source not intended to be monitored.

- Areas monitored may not be representative.

- 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.

  Source: NUREG-1228

FUEL HEATUP ONCE UNCOVERED

CAUTION:

> To estimate core damage progression, use "Core Damage Progression Once Uncovered," page B-4.

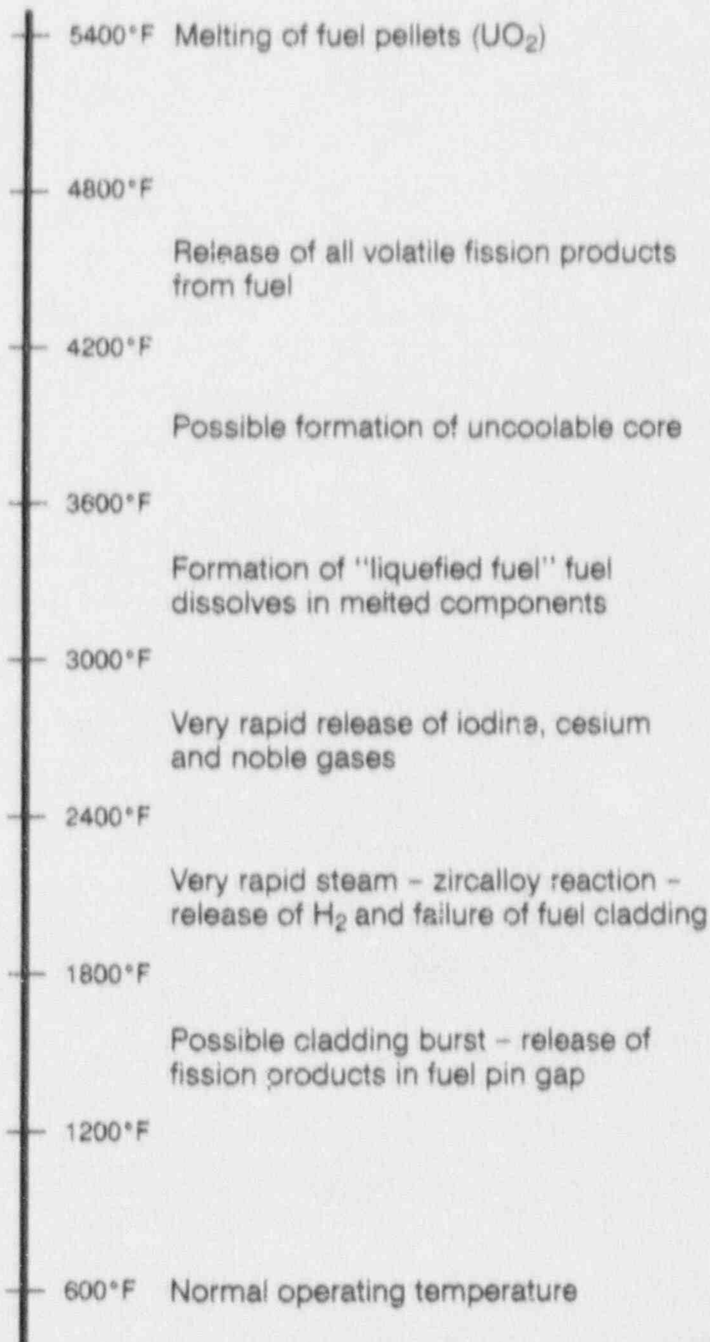
Step

While the top of the active core is uncovered, assume that the fuel will heat up at 1-2°F/sec. The increased core temperature will result in fuel pin damage as shown below.

Figure B-1

NOTE:

> These estimates are reasonable (factor of 2) if the core is uncovered within a few hours of shutdown (including failure to scram). If there is sufficient injection, core heatup may be stopped or slowed due to steam cooling. Steam cooling may not prevent core damage under accident conditions.



Source: NUREG-0900, NUREG/CR-4524, NUREG-0956

CORE DAMAGE PROGRESSION ONCE UNCOVERED

CAUTION:

> If core is severely damaged, it may not be in a coolable state even if covered again with water.

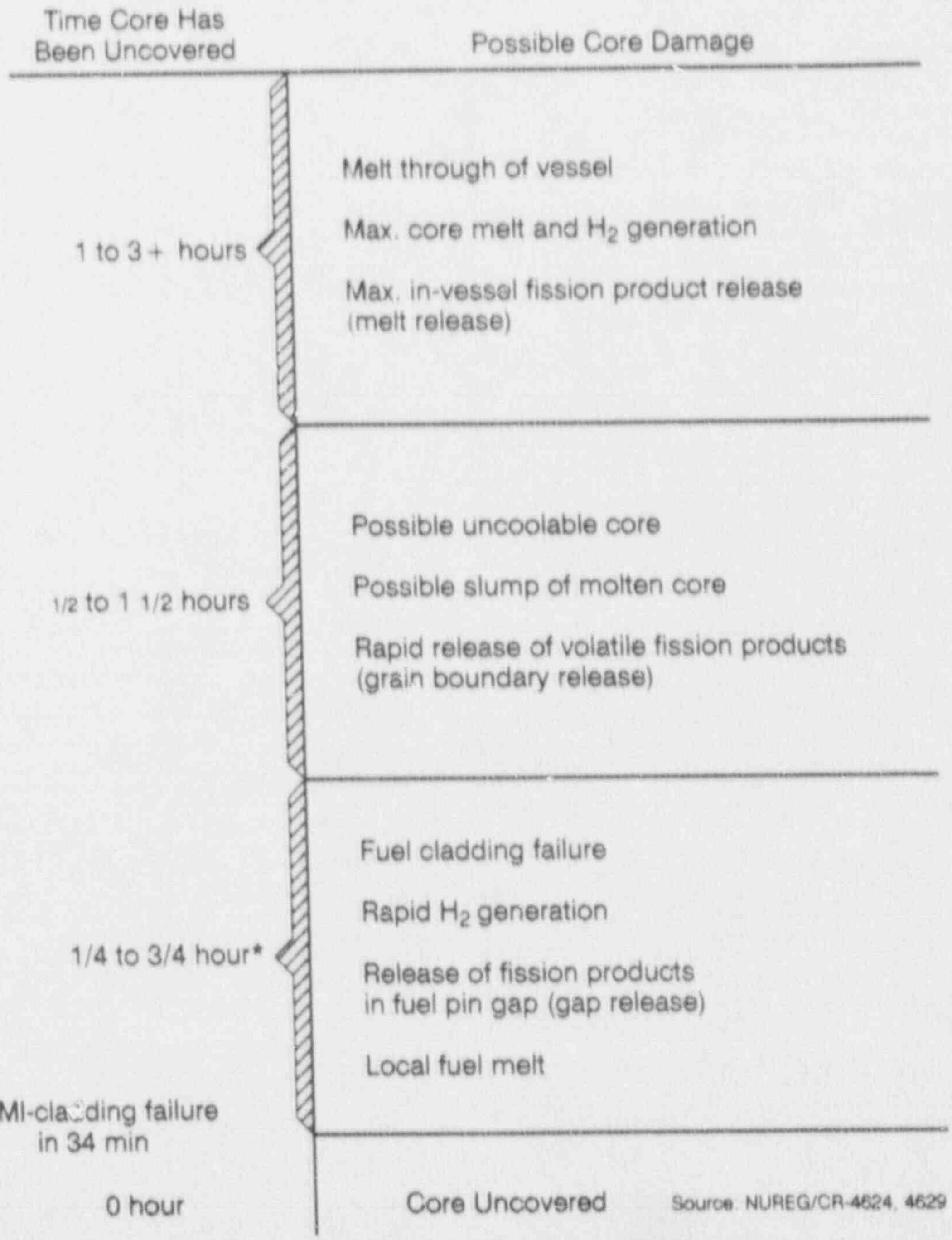
NOTE:

> If there is sufficient injection, core heatup may be stopped or slowed due to steam cooling. Steam cooling may not prevent core damage under accident conditions.

Step

Use the figure below as a guide:

Figure B-2



WATER INJECTION REQUIRED TO COOL CORE BY BOILING

CAUTION:

> These rates are those required to remove decay heat from a 3000 MW(t) plant <u>by boiling</u>. If there is a break requiring make up or injected water, more water than indicated will be required to both keep the core covered and cooled.

NOTE:

> These curves are based on a 3000 MW(t) plant operated at a constant power for an infinite period and then shutdown instantaneously. The decay heat power is based on ANS-5.1/N18.6. Assuming the injected water is at 80°F, these curves are within 5% for pressures between 14 psia to 2500 psia. These curves are within 20% for injected water temperatures up to 212°F.

Step 1

Use Figure B-3 (1/2 to 24 hours after shutdown), below, or Figure B-4 (1 to 30 days after shutdown), on the next page, to determine the minimum amount of water that must be injected to replace water lost by boiling (due to decay heat) for a 3000 MW(t) plant:

Figure B-3
INJECTION (gpm) REQUIRED TO REPLACE WATER LOST BY BOILING DUE TO DECAY HEAT FOR A 3000 MW(t) PLANT (1/2-24 HOURS AFTER SHUTDOWN)

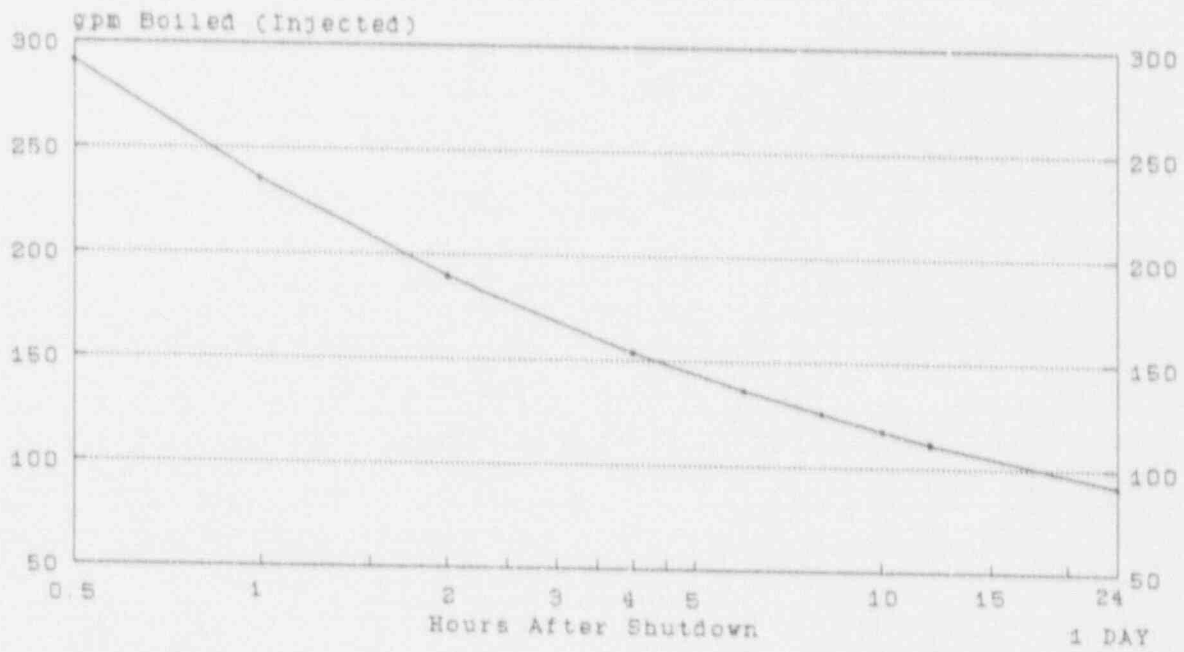
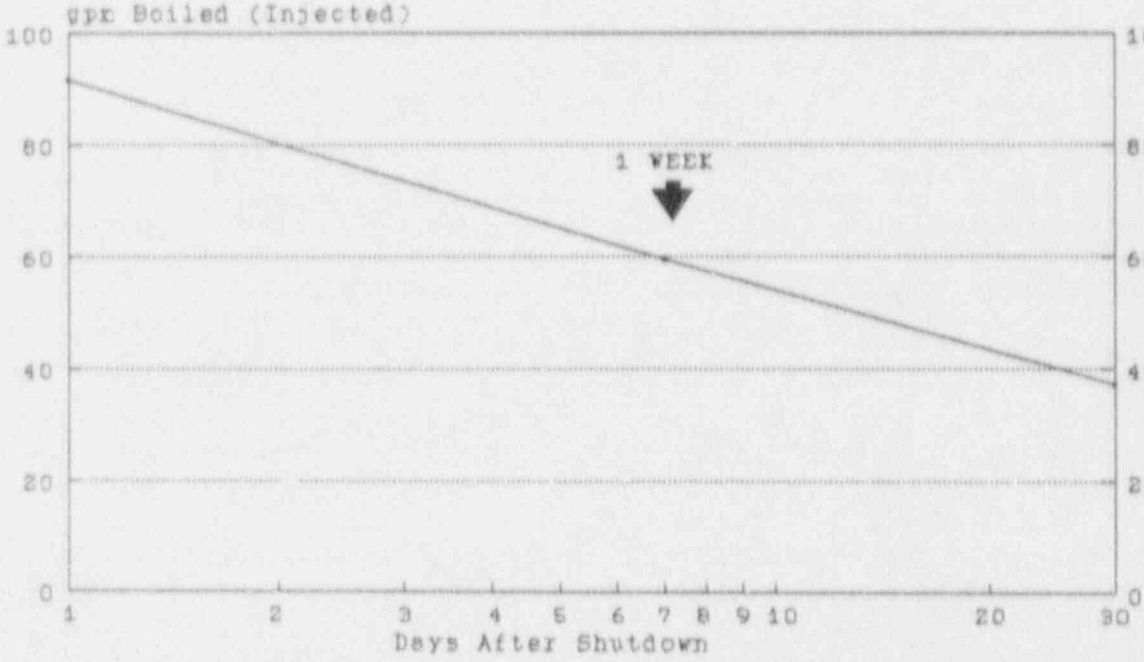


Figure B-4

INJECTION (gpm) REQUIRED TO REPLACE WATER LOST BY BOILING DUE TO DECAY HEAT FOR A 3000 MW(t) PLANT (1 to 30 DAYS AFTER SHUTDOWN)



Step 2

NOTE:

MW(t) ≅ 3 x MW(e)

Adjust for the size of plant where

$$gpm = \text{gpm from chart} \times \frac{\text{MW(t) of plant}}{\text{3000 MW(t)}}$$

CONTAINMENT RADIATION LEVELS VS CORE DAMAGE

CAUTION:

- The release from the core may bypass the containment, be retained in the primary system, or not be uniformly mixed. Therefore, a low containment radiation reading does not guarantee a lack of core damage. The levels of damage indicated by the figures should be considered minimum levels unless there are inconsistent monitor readings.
- Inconsistent readings may be due to the uneven mixing in containment (e.g., steam rising to top of dome). It may take hours for uniform mixing.

Step 1

NOTE:

Since the mix is most likely different than that assumed in the calibration of the monitor, the actual unshielded reading could differ by a factor of 10 - 100 for a shielded monitor.

Record:
Unshielded monitor reading(s) ________________.
Time after release into containment ____________.
Location of monitor(s) ____________________.
Monitor reading during normal operations ________.

Step 2

Confirm the containment monitor "sees" more than 50% of the shaded area shown in

PWR Figure B-5 page B-8
BWR Figure B-6 page B-9

If not, this procedure should not be used.

Step 3

Subtract the reading during normal operations from the unshielded monitor reading to get the absolute containment monitoring reading.

Step 4

Estimate the minimum core damage from containment radiation levels using

PWR	Figure B-7	page B-10
BWR Mark I and II	Figure B-8	page B-11
BWR Mark III	Figure B-9	page B-12

END

Verification Source: NUREG/CR-5775

Figure B-5

PWR MONITOR LOCATIONS

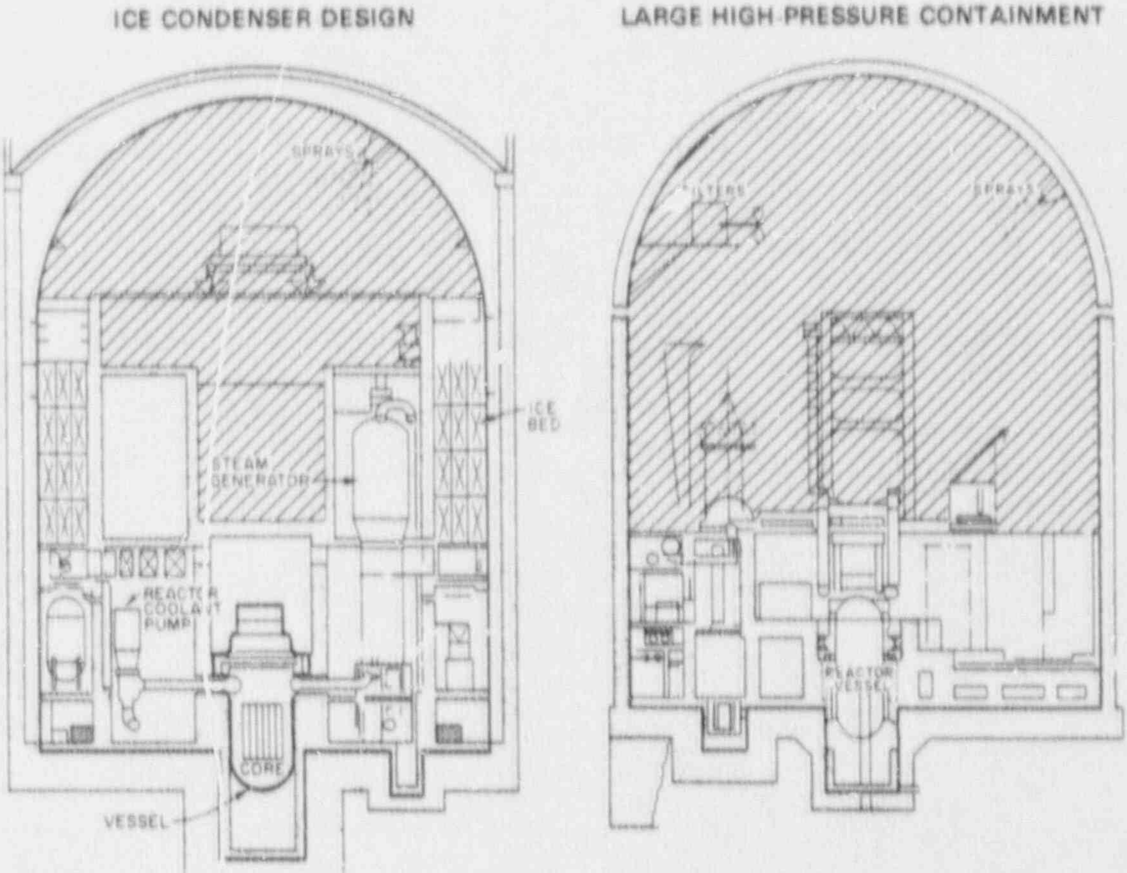
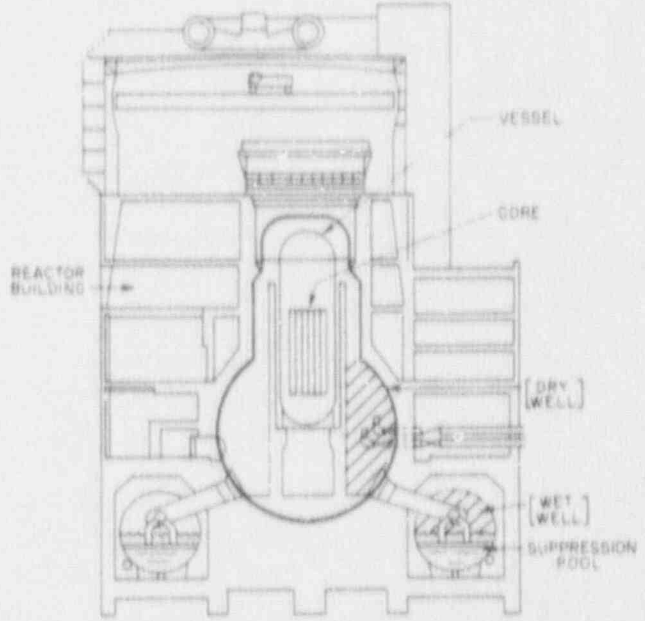


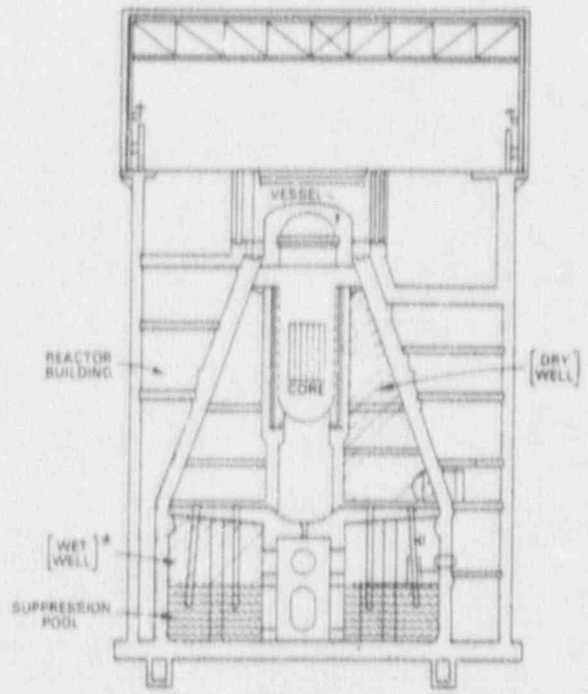
Figure B-6

BWR MONITOR LOCATIONS

BWR MARK I CONTAINMENT DESIGN



BWR MARK II CONTAINMENT DESIGN



BWR MARK III CONTAINMENT DESIGN

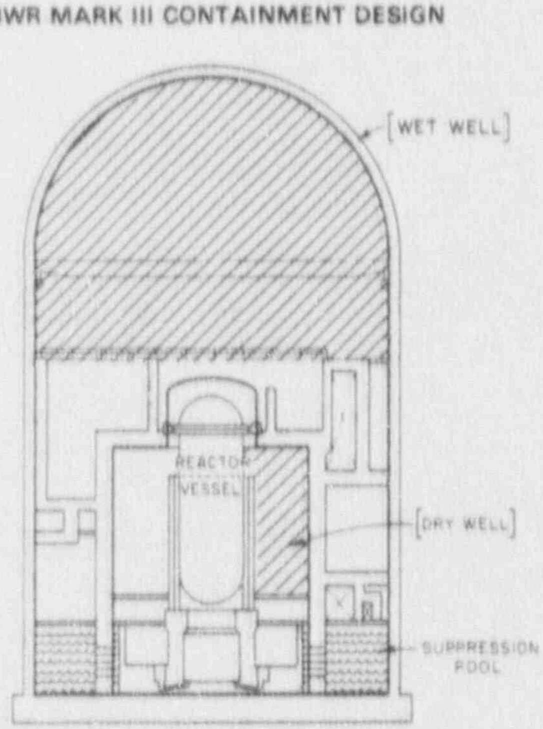
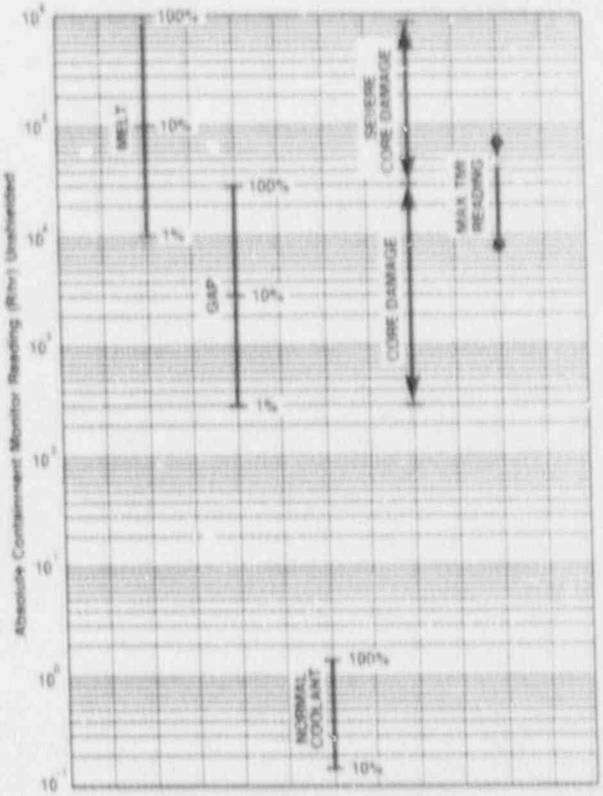
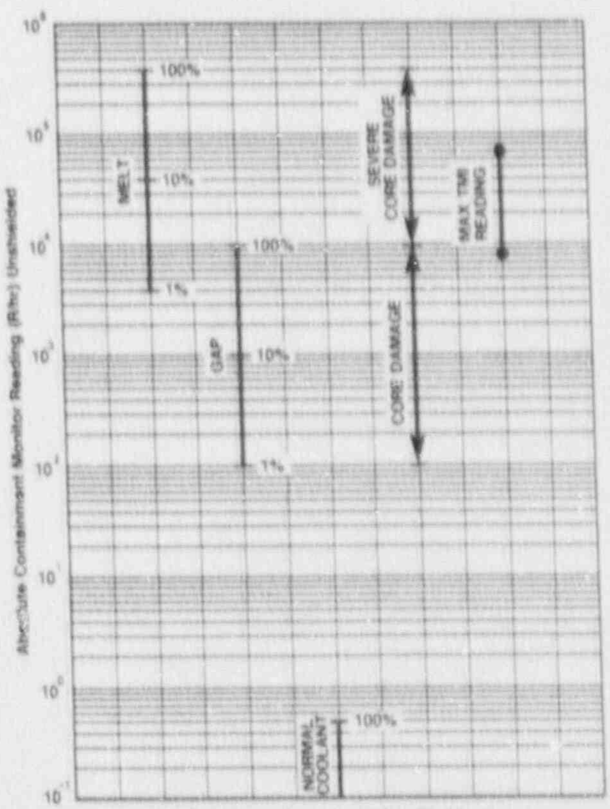


Figure B-7

PWR CONTAINMENT MONITOR RESPONSE



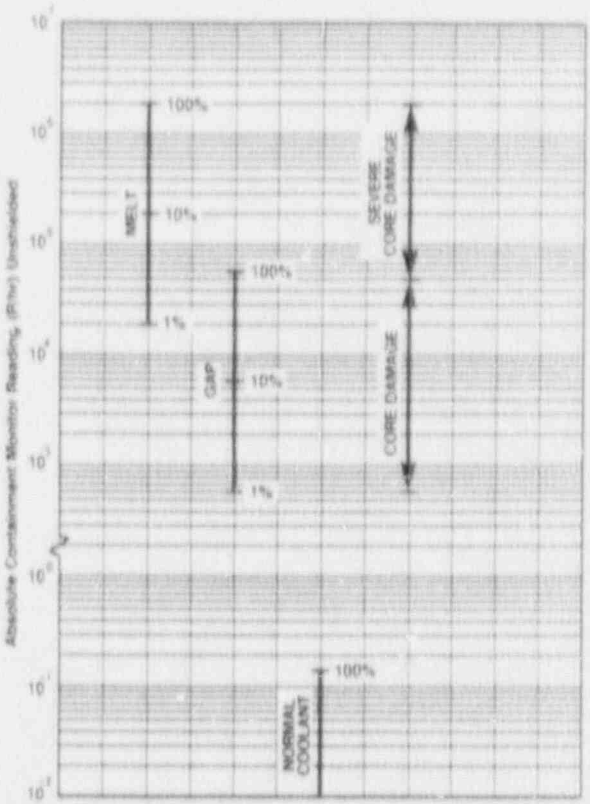
No-spray 1/2-2 hours After Shutdown
(PWR)



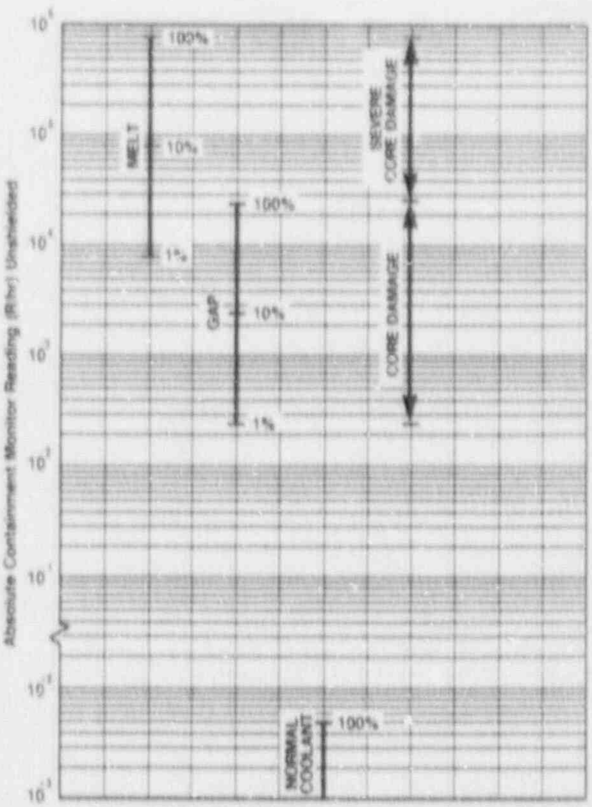
Spray-on 1/2-2 hours After Shutdown
(PWR)

Figure B-8

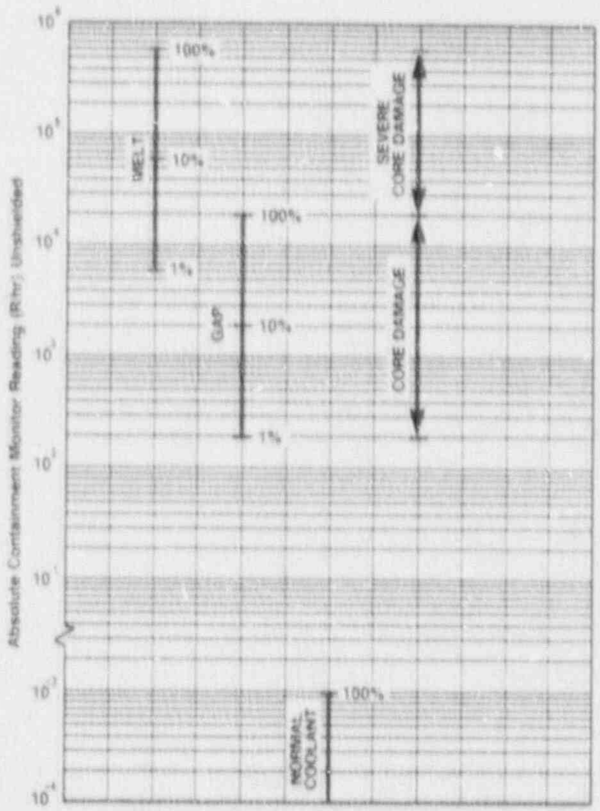
BWR MARK I & II CONTAINMENT MONITOR RESPONSE



Mark I & II Dry Well No-spray 1/2-2 hours After Shutdown
(BWR)



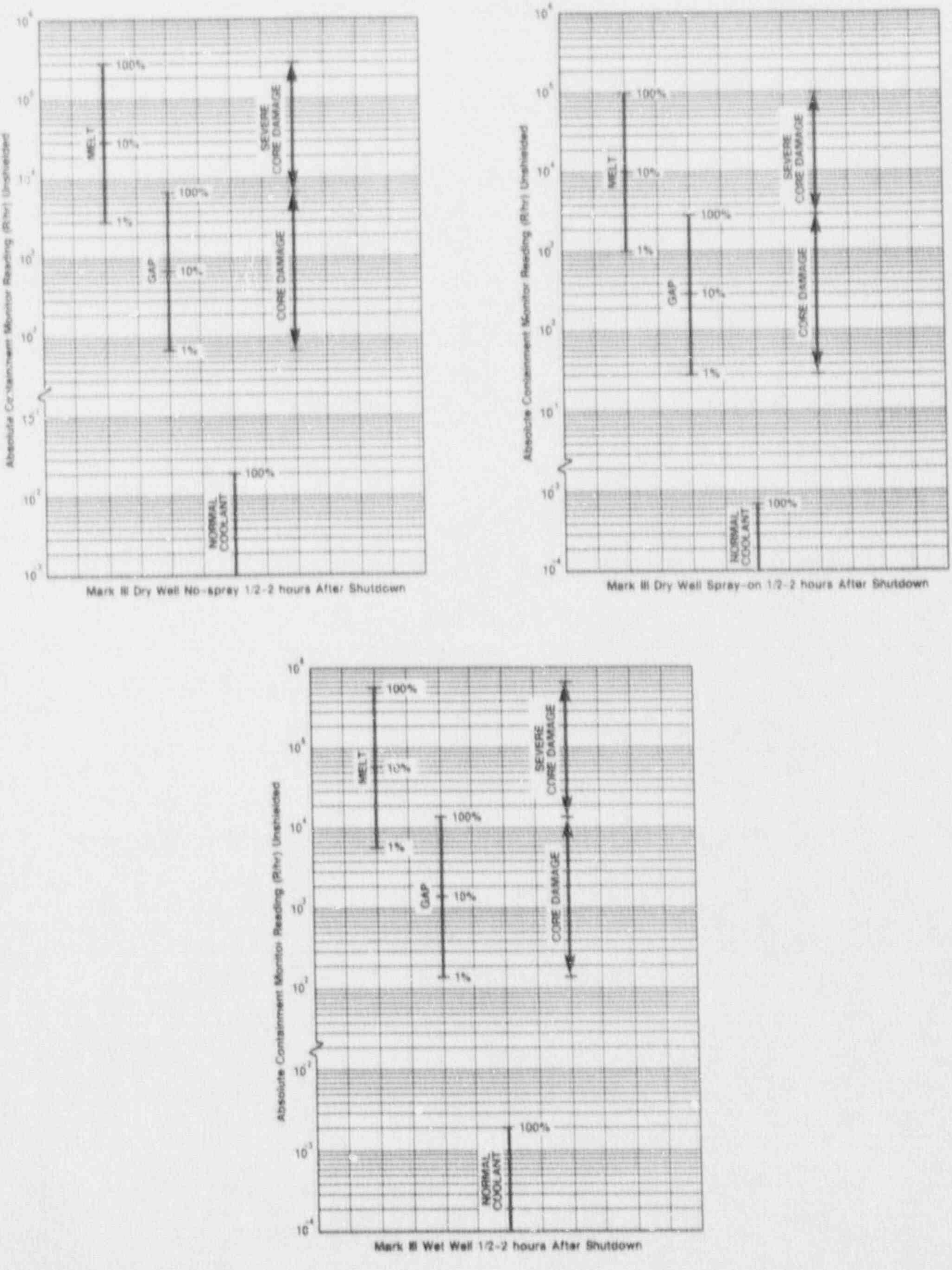
Mark I & II Dry Well Spray-on 1/2-2 hours After Shutdown
(BWR)



Mark I & II Wet Well 1/2-2 hours After Shutdown
(BWR)

Figure B-9

BWR MARK III CONTAINMENT MONITOR RESPONSE



Mark III Dry Well No-spray 1/2-2 hours After Shutdown

Mark III Dry Well Spray-on 1/2-2 hours After Shutdown

Mark III Wet Well 1/2-2 hours After Shutdown

SATURATION TABLE (P/T)

At the saturation temperature the coolant is boiling.

NOTE:

> Subcooling margin can be approximated by subtracting the coolant temperature from the saturation temperature at the given pressure.

Absolute Pressure (psia)	Saturation Temp. (F)
14.696	212.00
15.0	213.03
20.0	227.96
30.0	250.34
40.0	267.25
50.0	281.02
60.0	292.71
70.0	302.93
80.0	312.04
90.0	320.28
100.0	327.82
110.0	334.79
120.0	341.27
130.0	347.33
140.0	353.04
150.0	358.43
160.0	363.55
170.0	368.42
180.0	373.08
190.0	377.53
200.0	381.80
210.0	385.91
220.0	389.88
230.0	393.70
240.0	397.39
250.0	400.97
260.0	404.44
270.0	407.80
280.0	411.07
290.0	414.25
300.0	417.35
350.0	431.73
400.0	444.60
450.0	456.28
500.0	467.01
550.0	476.94
600.0	486.20
650.0	494.89
700.0	503.08
750.0	510.84
800.0	518.21
850.0	525.24
900.0	531.95
950.0	538.39
1000.0	544.58
1050.0	550.53
1100.0	561.82
1200.0	567.19
1250.0	572.38
1300.0	577.42
1350.0	582.32
1400.0	587.07
1450.0	591.70
1500.0	596.20
1550.0	600.59
1600.0	604.87
1650.0	609.05
1700.0	613.13
1750.0	617.12
1800.0	621.02
1850.0	624.83
1900.0	628.56
1950.0	632.22
2000.0	635.80
2100.0	642.76
2200.0	649.45
2300.0	655.89
2400.0	662.11
2500.0	668.11
2600.0	673.91
2700.0	679.53
2800.0	684.96
2900.0	690.22
3000.0	695.33
3100.0	700.28
3200.0	705.08
3208.2*	705.47

*Critical pressure

REF: 1967 ASME Steam Tables

COOLANT FISSION PRODUCT CONCENTRATIONS VS CORE DAMAGE

CAUTION:

> Coolant samples take hours and may not be representative of primary system concentrations (e.g., no flow through sample line).

CAUTION:

> The following assumptions were used in calculating the coolant concentrations and may be unrealistic for the acc'dent being analyzed:
>
> 1. All of the fuel pin gap and melt releases are assumed to be uniformly mixed in the coolant.
> 2. No dilution due to injection is assumed.
> 3. The tables over-estimate the concentrations for the long-lived fission products (Cs and Sr) in a new core.
> 4. Concentrations given are for 1/2 hour after shutdown. The half-life of the fission products should be considered in analyzing samples taken hours after shutdown.

NOTE:

> Plant-specific coolant system volume does not have a major influence on coolant concentrations (<20%).

Step

Compare reported coolant concentration with PWR or BWR Baseline Coolant Concentrations below and on the next page respectively:

PWR BASELINE COOLANT CONCENTRATIONS

Nuclide	Normal Concentration (μCi/g)	Concentration after Gap Release (μCi/g)	Concentration after Melt Release (μCi/g)	TMI Concentration +48 hrs (μCi/g)
I-131	4.5×10^{-2}	6.8×10^{3}	3.4×10^{5}	1.3×10^{4}
I-133	1.4×10^{-1}	1.4×10^{4}	6.8×10^{5}	6.5×10^{3}
I-135	2.6×10^{-1}	1.2×10^{4}	6.0×10^{5}	--
Cs-134	7.1×10^{-3}	1.5×10^{3}	3.0×10^{4}	6.3×10^{1}
Cs-137	9.4×10^{-3}	9.4×10^{2}	1.9×10^{4}	2.8×10^{2}
Sr-90	1.2×10^{-5}	NC	1.0×10^{3}	5.3×10^{0}

NC = Data not available

BWR BASELINE COOLANT CONCENTRATIONS

CAUTION:

> The following assumes that the release from the core is uniformly mixed in the <u>reactor coolant system</u> and <u>suppression pool</u>. If most of the core release is confined to the reactor coolant system, the concentrations could be 10 times higher.

Nuclide	Normal RCS Concentration (μCi/g)	Concentration after Gap Release (μCi/g)[1]	Concentration after Melt Release (μCi/g)[1]
I-131	2.2×10^{-3}	6.2×10^{2}	3.1×10^{4}
I-133	1.5×10^{-2}	1.2×10^{3}	6.2×10^{4}
I-135	2.2×10^{-2}	1.1×10^{3}	5.5×10^{4}
Cs-134	3.0×10^{-5}	1.4×10^{2}	2.7×10^{3}
Cs-137	8.0×10^{-5}	8.6×10^{1}	1.7×10^{3}
Sr-90	7.0×10^{-6}	NC	9.5×10^{1}

[1]In the RCS and suppression pool.

NC = Data not available

Assumption: Core Inventory, RTM-91, page C-83.
Core Release Fractions, RTM-91, page C-84.
Normal Coolant Concentration, ANSI/ANS-18.1, 1984

CONTAINMENT HYDROGEN VS CORE DAMAGE

CAUTION:

> Containment samples may require hours and may not be representative of the total hydrogen generated in the core due to incomplete mixing in the containment.

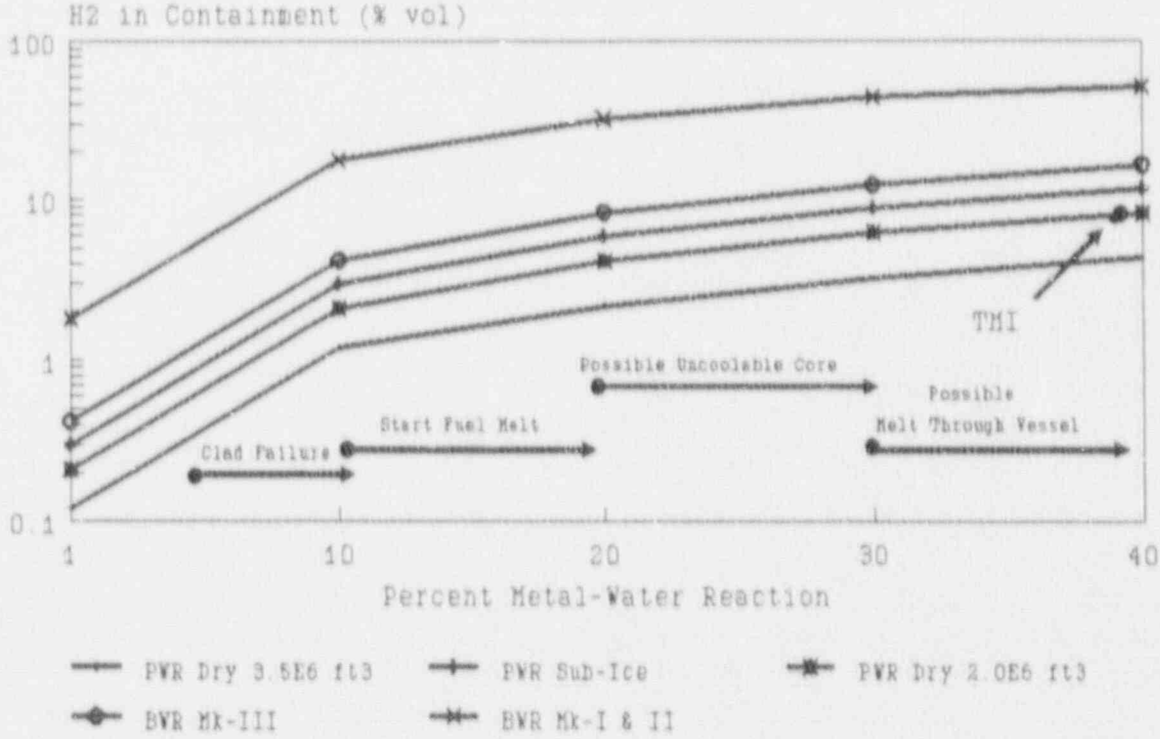
Step

Compare the estimated percentage of hydrogen in a wet sample with the figure below to estimate the core damage state.

NOTE:

> These curves are for wet samples. However, most H_2 samples are dry (steam removed). If a dry sample concentration is compared with the chart, it may considerably over-estimate the level of core damage.

Figure B-10



Source: NUREG/CR-2726, page 4-3.
Damage States, Source NUREG-4524, Vol. 5.
TMI Percentage, Source NUREG-1370.
NUREG/CR-4041.

HYDROGEN FLAMMABILITY

CAUTION:

The ranges may be larger for the high temperatures and pressures found in containments under accident conditions.

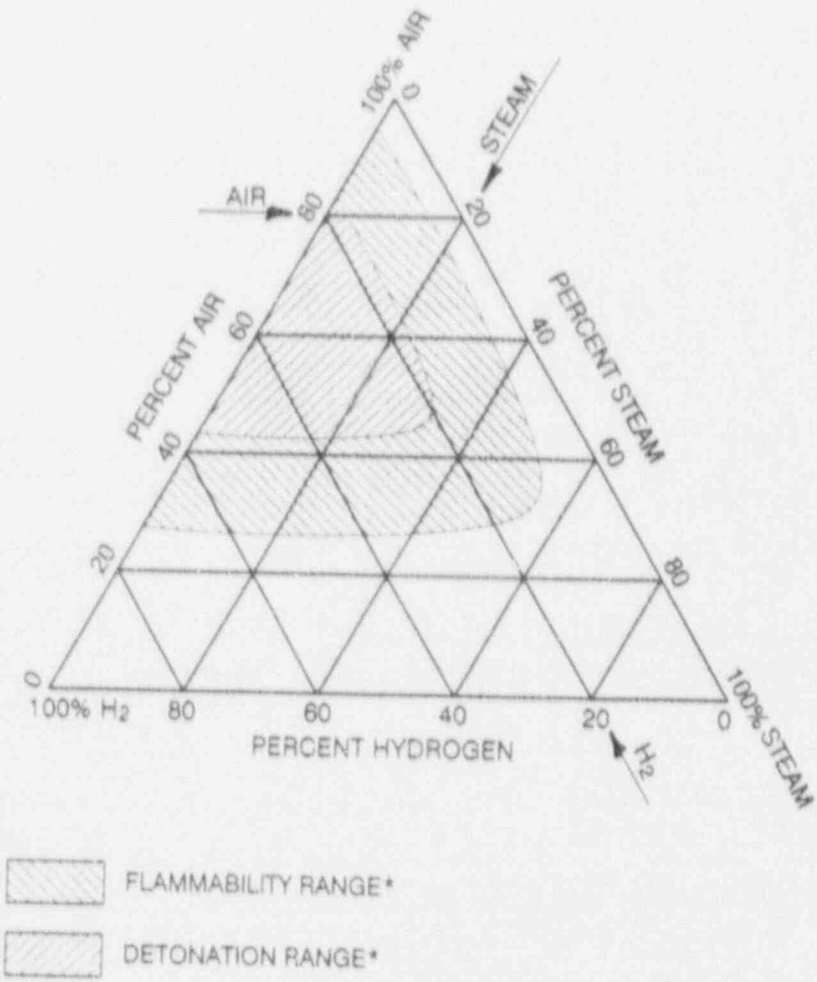
NOTE:

Hydrogen detonation is not considered a likely cause of containment failure in Dry or Sub-Atmosphere PWR unless it takes place when the containment is at very high pressure. Hydrogen detonation is not likely cause of BWR Mark I or II containment failures if the containment atmosphere is inert.

Step

Use figure below to determine flammability:

Figure B-11



Source: NUREG/CR-2726

Reactor Accident Consequence Assessment Based on Plant Conditions

Section C

C

Reactor Accident Consequence Assessment Based on Plant Conditions

REACTOR ACCIDENT CONSEQUENCE ASSESSMENT BASED ON PLANT CONDITIONS

Objective

Estimate offsite consequences based on the status of reactor conditions.

Guidance

NOTE:

- Assessments are accurate within a factor of 10-100, <u>only</u> if plant conditions are accurately represented.
- RASCAL 1.3b estimate may be a factor of 2 or 3 times higher due to different assumptions.
- Scientific notation used 5E+3 = 5×10^3
- General descriptions of methods and assumptions are found on pages C-81 through C-90.

Step 1

Proceed to the appropriate release type:

END

Source: NUREG-1228 revised to conform to NUREG-1150 findings.

Reactor Accident Consequence Overview

Containment Leakage

Core Condition	Containment Status	Mitigating System Status (*)	Acute Dose (rem) 1 Hour Release @ 1 mile (**) WB	Thy
MELT Release From Core 4500° F	Early Total Failure (<1 hr)	No Mitigation	1000+	10^5+
		Mitigated	250	10^4
	Late Total Failure (2-12 Hrs)		250	10^4
	Major Leakage (100%/day)		10	10^3
	Design Leakage		10^{-2}	1
Gap Release From Core 1500° F	Early Total Failure (<1 hr)	No Mitigation	50	10^4
		Mitigated	10	10^3
	Late Total Failure (2-12 Hrs)		5	10^3
	Major Leakage (100%/day)		10^{-1}	10
	Design Leakage		10^{-4}	10^{-2}

Steam Generator Tube Rupture
(1 Tube Full Pressure For 1 Hour)

Coolant Concentration	Generator Status	Acute Dose (rem) 1 hour Release @ 1 mile WB	Thy
100x Coolant Spike	No Partitioning	10^{-1}	2
	Partitioned	10^{-2}	10^{-1}
Normal Coolant		10^{-3}	10^{-2}

* Sprays, Pools, Filters

** 1hr. Cloud and inhalation
3 hrs ground

PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT RELEASE EVENT TREES

NOTE:

> A general description of the assumptions used in the event trees are found on page C-4.

Step 2

Select the appropriate plant conditions:

NOTE:

> Holdup time is the time from the release to the containment atmosphere to the start of the release to the environment.

Core condition (see Core Damage Progression Once Uncovered, B-4)
- gap release (1300-2000°F) ____________
- grain boundary (>3000°F) ____________
- melt (>4500°F) ____________

Containment sprays
- on ____________
- off ____________

Holdup time before release
- <1 hour ____________
- 2-12 hours ____________
- 24 hours ____________

Containment leak rate
- design (0.1%/day) ____________
- 100%/day (isolation failure) ____________
- 100%/hour (catastropic) ____________

Step 3

Go to appropriate event tree based on core condition.

Gap C-6
Grain Boundary C-8
Melt C-10

PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT EVENT TREES

GENERAL DESCRIPTION: The core is uncovered and the release fractions on page C-84 for a 1000 MWe plant typical of a gap (1300-2000°F), grain boundary (3000°F), or melt (4500°F) release from the core are assumed. For the vessel melt-through cases (ex-vessel melt), the strontium release fractions could be substantially increased. This could increase the projected whole body dose by 50%. The release is assumed to pass by a dry pathway through the primary system into the containment atmosphere. Particulates and aerosols airborne in containment are reduced by the fractions shown on page C-85 to account for the actions of sprays or natural processes for <1, 2-12, or 24 hours holdup in the containment before release. Releases from the containment are estimated for a 1 hour release at the release rates shown on page C-86 for 0.1%/day (design leakage); 100%/day (isolation valve seal failure); or 100%/hour (catastrophic failure > 1 sq ft).

Doses at 1 mile are calculated using the factors shown on page C-89 to estimate a 1 hour ground level release, 4 mph, and D stability wind conditions. Acute whole body dose includes 1 hour of cloud shine and acute (30 day committed) inhalation dose and 3 hours of ground shine. Noble gas decay is included. Thyroid doses are for adults and from inhalation only.

[Figure C-1 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

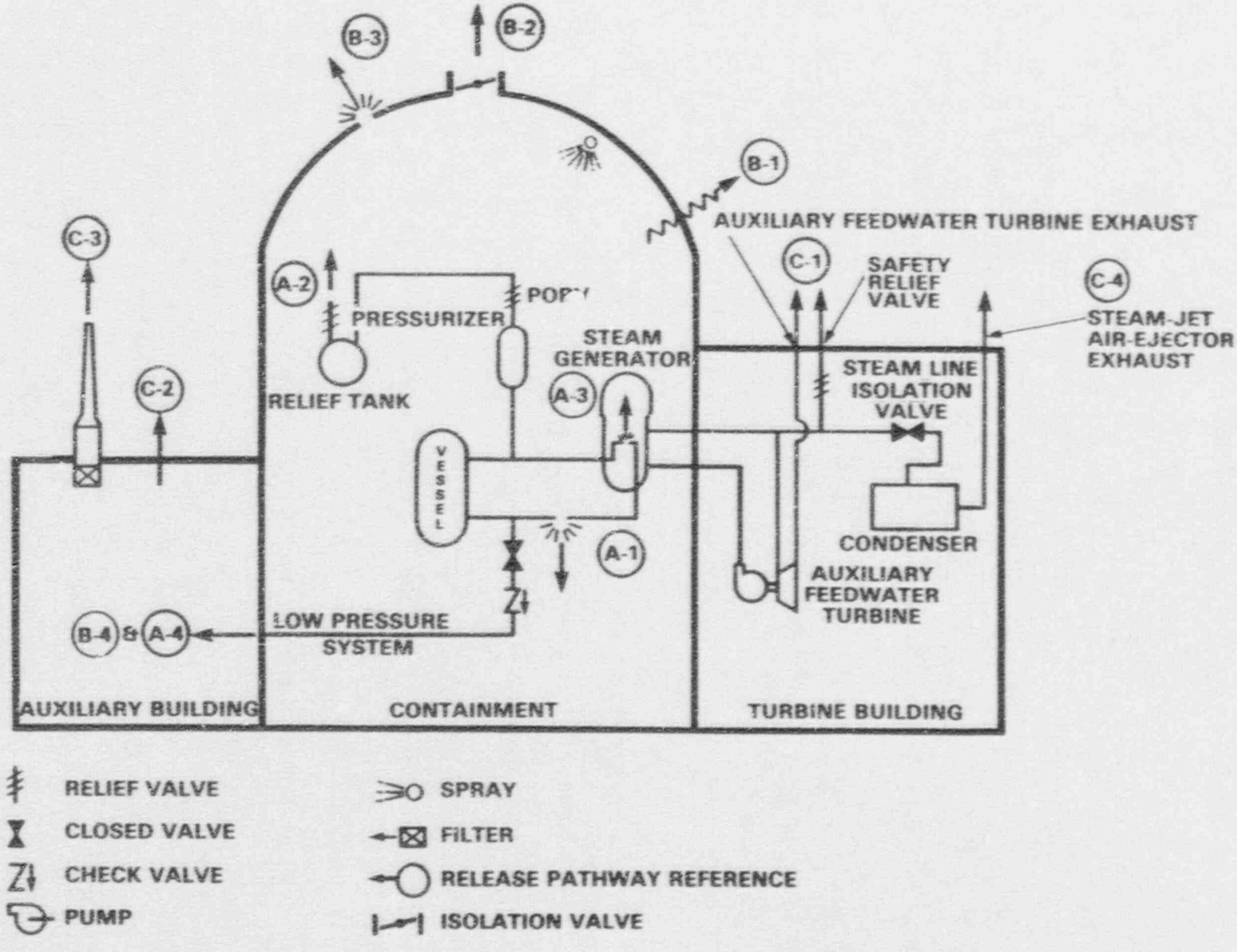
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic (>1 sq ft)
- B-4 Bypass

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

Figure C-1

PWR DRY CONTAINMENT SIMPLIFIED RELEASE PATHWAYS



GAP
PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT

Step 4

Record from event tree (Figure C-2 on page C-7):

Acute whole body dose @ 1 mile ____________
Thyroid dose @ 1 mile ____________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x $\frac{\text{\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-2

PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT RELEASE EVENT TREE FOR A GAP RELEASE FROM THE CORE

core condition	containment conditions	holdup time in containment	containment leak rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
GAP RELEASE FROM CORE	spray off	<1 hour	100%/hour	2E+01	2E+03
		2-12 hours	100%/hour	5E+00	4E+02
			100%/day	2E-01	2E+01
			design rate	2E-04	2E-02
		24 hours	100%/hour	1E+00	1E+02
			100%/day	5E-02	4E+00
			design rate	5E-05	4E-03
	spray on	<1 hour	100%/hour	5E+00	3E+02
		2-12 hours	100%/hour	3E+00	2E+02
			100%/day	1E-01	8E+00
			design rate	1E-04	8E-03
		24 hours	100%/hour	4E-01	2E+01
			100%/day	2E-02	8E-01
			design rate	2E-05	8E-04

GRAIN BOUNDARY
PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT

Step 4

Record from event tree (Figure C-3 on page C-9):

Acute whole body dose @ 1 mile ____________________
Thyroid dose @ 1 mile ____________________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x ________ plant MW(e) / 1000 MW(e) = __________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = __________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-3

PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT RELEASE EVENT TREE FOR A GRAIN BOUNDARY RELEASE FROM THE CORE

core condition	containment conditions	holdup time in containment	containment leak rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
GRAIN BOUNDARY RELEASE FROM CORE	spray off	<1 hour	100%/hour	4E+02	4E+04
		2-12 hours	100%/hour	1E+02	8E+03
			100%/day	4E+00	3E+02
			design rate	4E-03	3E-01
		24 hours	100%/hour	2E+01	2E+03
			100%/day	9E-01	8E+01
			design rate	9E-04	8E-02
	spray on	<1 hour	100%/hour	1E+02	6E+03
		2-12 hours	100%/hour	6E+01	4E+03
			100%/day	2E+00	2E+02
			design rate	2E-03	2E-01
		24 hours	100%/hour	7E+00	4E+02
			100%/day	3E-01	2E+01
			design rate	3E-04	2E-02

MELT
PWR LARGE DRY OR SUBATMOSPHERIC CONTAINEMNT

Step 4

Record from event tree (Figure C-4 on page C-11):

Acute whole body dose @ 1 mile ______________________
Thyroid dose @ 1 mile ______________________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ________
- Distance to which early health effects are possible (WB dose > 50 rem) ________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-4

PWR LARGE DRY OR SUBATMOSPHERIC CONTAINMENT RELEASE EVENT TREE FOR A MELT RELEASE FROM THE CORE

core condition	containment conditions	holdup time in containment	containment leak rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
MELT RELEASE FROM CORE	spray off	<1 hour	100%/hour	1E+03	1E+05
		2-12 hours	100%/hour	2E+02	2E+04
			100%/day	9E+00	8E+02
			design rate	9E-03	8E-01
		24 hours	100%/hour	6E+01	5E+03
			100%/day	2E+00	2E+02
			design rate	2E-03	2E-01
	spray on	<1 hour	100%/hour	2E+02	2E+04
		2-12 hours	100%/hour	1E+02	1E+04
			100%/day	5E+00	4E+02
			design rate	5E-03	4E-01
		24 hours	100%/hour	2E+01	1E+03
			100%/day	7E-01	4E+01
			design rate	7E-04	4E-02

PAGE INTENTIONALLY LEFT BLANK

PWR ICE CONDENSER CONTAINMENT
RELEASE EVENT TREES

NOTE:

> A general description of the assumptions used in the event trees are found on page C-14.

Step 2

CAUTION:

> If the ice condenser is bypassed or the ice is exhausted before the release from core, assume it is a PWR large, dry containment release (page C-3).

NOTE:

> Holdup time is the time from the release to the containment atmosphere to the start of the release to the environment.

Select the appropriate plant conditions:

Core condition (see Core Damage Progression Once Uncovered, B-4)
- gap release (1300-2000°F) ____________
- grain boundary (>3000°F) ____________
- melt (>4500°F) ____________

Containment sprays
- on ____________
- off ____________

Holdup time before release
- <1 hour ____________
- 2-12 hours ____________
- 24 hours ____________

Containment leak rate
- design (0.25%/day) ____________
- 100%/day ____________
- 100%/hour ____________

Recirculation fans ____________
- on (with recirculation) ____________
- off (once-through ice) ____________

Step 3

Go to appropriate event tree based on core condition.

Gap C-16
Grain Boundary C-18
Melt C-20

PWR ICE CONDENSER CONTAINMENT RELEASE EVENT TREES

GENERAL DESCRIPTION: The core is uncovered and the release fractions on page C-84 typical of a gap (1300-2000°F), grain boundary (3000°F), or melt (4500°F) release from the core are assumed. For the vessel melt-through cases (ex-vessel melt), the strontium release fractions could be substantially increased. This could increase the projected whole body dose by 50%. The release is assumed to pass by a dry pathway through the primary system into the containment atmosphere through the ice condenser. Particulates and aerosols airborne in the containment are reduced by the factors shown on page C-85 to account for the actions of sprays and natural processes for <1, 2-12 or 24 hours holdup in the containment atmosphere, removal due to one pass through the ice condenser (because of loss of the recirculation fans e.g., loss of AC) and for continual recirculation. If the ice condenser is bypassed or the ice is exhausted before the release form the core, the PWR large, dry containment release trees should be used (page C-3). Releases from the containment are estimated for 1 hour at the release rates shown on page C-86 for 0.25%/day (design leakage); 100%/day (isolation valve seal failure); or 100%/hour (catastrophic failures >1 sq ft).

Doses at 1 mile are calculated using the factors shown on page C-89 to estimate a 1 hour ground level release, 4 mph, and D stability wind conditions. Whole body dose includes 1 hour of cloud shine and acute (30 day committed) inhalation dose and 3 hours of ground shine. Noble gas decay is included. Thyroid doses are for adults and from inhalation only.

[Figure C-5 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

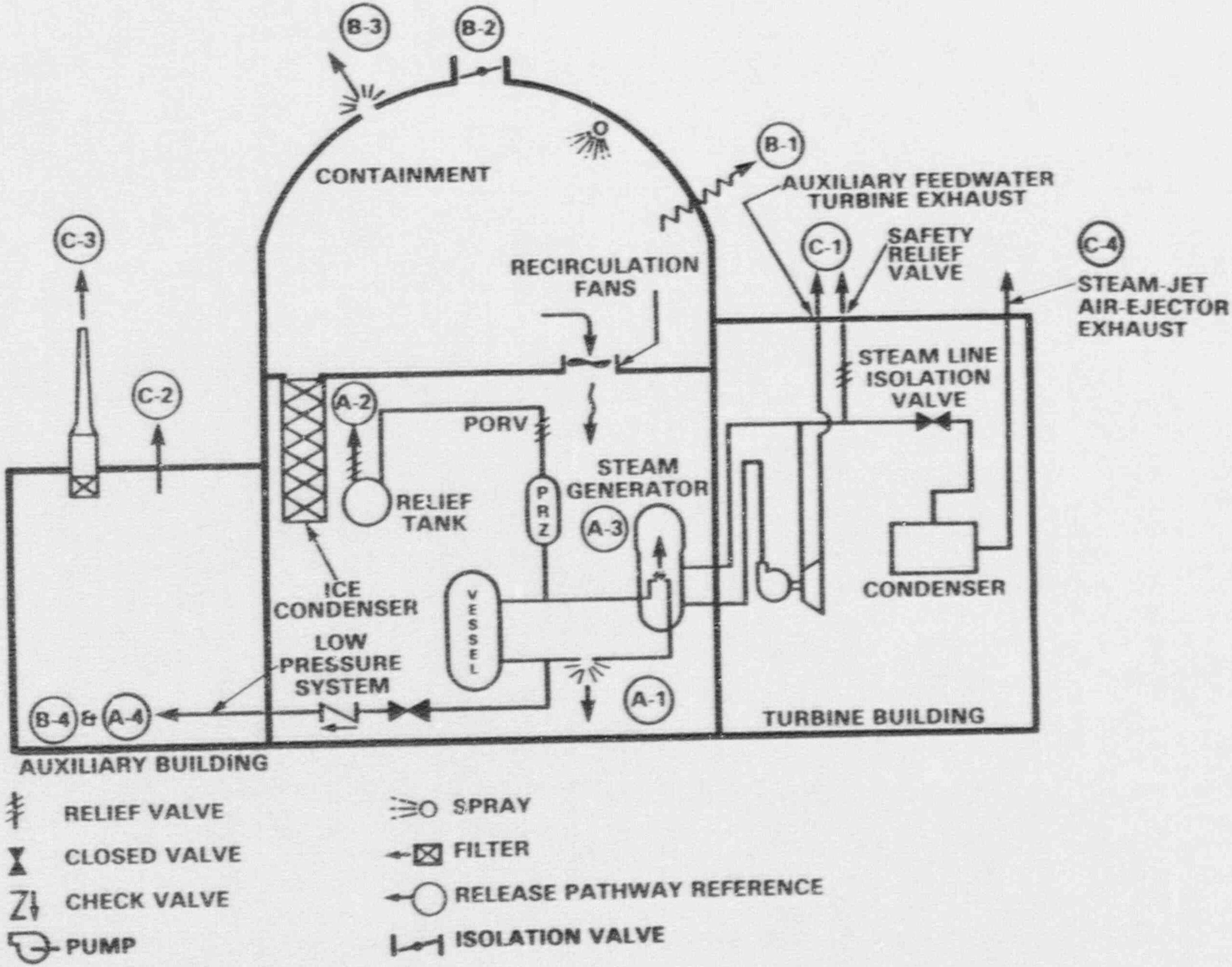
- B-1 Design leakage
- B-2 Small isolation valve seal failure
- B-3 Catastrophic (>1 sq ft)
- B-4 Bypass

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

Figure C-5

PWR ICE CONDENSER CONTAINMENT SIMPLIFIED RELEASE PATHWAYS



GAP
PWR ICE CONDENSER CONTAINMENT

Step 4

Record from event tree (Figure C-6 on page C-17):

Acute whole body dose @ 1 mile ____________________

Thyroid dose @ 1 mile ____________________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x $\frac{\text{plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Thyroid @ 1 mile ________ x $\frac{\text{plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ________________
- Distance to which early health effects are possible (WB dose > 50 rem) ________________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ________________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-6

PWR ICE CONDENSER CONTAINMENT RELEASE EVENT TREE FOR A GAP RELEASE FROM THE CORE

core and containment conditions	holdup time in containment	containment leak rate	whole-body dose (rem) at 1 mile once thru ice	thyroid dose (rem) at 1 mile once thru ice	whole-body dose (rem) at 1 mile with recir	thyroid dose (rem) at 1 mile with recir
GAP RELEASE FROM CORE — sprays off	<1 hour	100%/hour	1E+01	1E+03	7E+00	5E+02
	2-12 hours	100%/hour	3E+00	2E+02	2E+00	1E+02
		100%/day	1E-01	8E+00	6E-02	4E+00
		design rate	3E-04	2E-02	2E-04	1E-02
	24 hours	100%/hour	7E-01	5E+01	5E-01	2E+01
		100%/day	3E-02	2E+00	2E-02	1E+00
		design rate	7E-05	5E-03	5E-05	3E-03
GAP RELEASE FROM CORE — sprays on	<1 hour	100%/hour	4E+00	2E+02	3E+00	8E+01
	2-12 hours	100%/hour	2E+00	1E+02	1E+00	5E+01
		100%/day	6E-02	4E+00	4E-02	2E+00
		design rate	2E-04	1E-02	1E-04	5E-03
	24 hours	100%/hour	3E-01	1E+01	3E-01	1E+01
		100%/day	1E-02	4E-01	1E-02	4E-01
		design rate	3E-05	1E-03	3E-05	1E-03

GRAIN BOUNDARY
PWR ICE CONDENSER CONTAINMENT

Step 4

Record from event tree (Figure C-7 on page C-19):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x ________ $\frac{\text{plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Thyroid @ 1 mile ________ x ________ $\frac{\text{plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile _______ x 0.40 = _______ whole body @ 2 mile
Whole body @ 1 mile _______ x 0.09 = _______ whole body @ 5 mile
Whole body @ 1 mile _______ x 0.03 = _______ whole body @10 mile

Thyroid @ 1 mile _______ x 0.40 = _______ thyroid @ 2 mile
Thyroid @ 1 mile _______ x 0.09 = _______ thyroid @ 5 mile
Thyroid @ 1 mile _______ x 0.03 = _______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) __________
- Distance to which early health effects are possible (WB dose > 50 rem) __________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) __________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-7

PWR ICE CONDENSER CONTAINMENT RELEASE EVENT TREE FOR A GRAIN BOUNDARY RELEASE FROM THE CORE

GRAIN BOUNDARY RELEASE FROM CORE

containment conditions	holdup time in containment	containment leak rate	whole-body dose (rem) at 1 mile once thru ice	thyroid dose (rem) at 1 mile once thru ice	whole-body dose (rem) at 1 mile with recir	thyroid dose (rem) at 1 mile with recir
sprays off	<1 hour	100%/hour	2E+02	2E+04	1E+02	1E+04
	2-12 hours	100%/hour	5E+01	4E+03	3E+01	2E+03
		100%/day	2E+00	2E+02	1E+00	8E+01
		design rate	5E-03	4E-01	3E-03	2E-01
	24 hours	100%/hour	1E+01	1E+03	8E+00	5E+02
		100%/day	5E-01	4E+01	3E-01	2E+01
		design rate	1E-03	1E-01	9E-04	5E-02
sprays on	<1 hour	100%/hour	7E+01	3E+03	6E+01	2E+03
	2-12 hours	100%/hour	3E+01	2E+03	2E+01	1E+03
		100%/day	1E+00	8E+01	8E-01	4E+01
		design rate	3E-03	2E-01	2E-03	1E-01
	24 hours	100%/hour	6E+00	2E+02	6E+00	2E+02
		100%/day	2E-01	8E+00	2E-01	8E+00
		design rate	6E-04	2E-02	6E-04	2E-02

CORE MELT
PWR ICE CONDENSER CONTAINMENT

<u>Step 4</u>

Record from event tree (Figure C-8 on page C-21):

Acute whole body dose @ 1 mile ____________________
Thyroid dose @ 1 mile ____________________

<u>Step 5</u>

Adjust for reactor size:

Whole body @ 1 mile________ x $\frac{\text{\_\_\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

<u>Step 6</u>

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

<u>Step 7</u>

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile _______ x 0.40 = _______ whole body @ 2 mile
Whole body @ 1 mile _______ x 0.09 = _______ whole body @ 5 mile
Whole body @ 1 mile _______ x 0.03 = _______ whole body @10 mile

Thyroid @ 1 mile _______ x 0.40 = _______ thyroid @ 2 mile
Thyroid @ 1 mile _______ x 0.09 = _______ thyroid @ 5 mile
Thyroid @ 1 mile _______ x 0.03 = _______ thyroid @10 mile

<u>Step 8</u>

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ______________
- Distance to which early health effects are possible (WB dose > 50 rem) ______________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ______________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-8

PWR ICE CONDENSER CONTAINMENT RELEASE EVENT TREE FOR A MELT RELEASE FROM THE CORE

MELT RELEASE FROM CORE

containment conditions	holdup time in containment	containment leak rate	whole-body dose (rem) at 1 mile once thru ice	thyroid dose (rem) at 1 mile once thru ice	whole-body dose (rem) at 1 mile with recir	thyroid dose (rem) at 1 mile with recir
sprays off	<1 hour	100%/hour	6E+02	5E+04	3E+02	2E+04
	2-12 hours	100%/hour	1E+02	1E+04	7E+01	5E+03
		100%/day	5E+00	4E+02	3E+00	2E+02
		design rate	1E-02	1E+00	7E-03	5E-01
	24 hours	100%/hour	3E+01	2E+03	2E+01	1E+03
		100%/day	1E+00	1E+02	8E-01	5E+01
		design rate	3E-03	3E-01	2E-03	1E-01
sprays on	<1 hour	100%/hour	2E+02	8E+03	1E+02	4E+03
	2-12 hours	100%/hour	7E+01	5E+03	5E+01	2E+03
		100%/day	3E+00	2E+02	2E+00	1E+02
		design rate	7E-03	5E-01	5E-03	3E-01
	24 hours	100%/hour	1E+01	5E+02	1E+01	5E+02
		100%/day	5E-01	2E+01	5E-01	2E+01
		design rate	1E-03	5E-02	1E-03	5E-02

PAGE INTENTIONALLY LEFT BLANK

PWR STEAM GENERATOR TUBE RUPTURE RELEASE EVENT TREES

NOTE:

A general description of the assumptions used in the event trees are found on page C-24.

Step 2

CAUTION:

For a dry (primary and secondary) release pathway assume it is a BWR/PWR containment bypass on page C-61.

NOTE:

The steam generator condition is not **partitioned** if the tube break is above the secondary side water level or if the secondary side is full of primary coolant. Normally multiple tube failures and failures in once through steam generators are assumed not to be partitioned.

NOTE:

100X spike assumes all the non-noble fission products in normal coolant are 100 times higher. This can take place even if the core is covered and is not considered a severe condition.

Select the appropriate plant conditions:

Coolant concentration (see Core Damage Progression Once Uncovered, B-4)
- normal ______
- 100X spike ______
- gap release (1300-2000°F) ______
- grain boundary (3000°F) ______
- melt (4500°F) ______

Tube leak size
- 1 tube failure at high pressure (500 gpm) ______
- 1 charge pump flow (50 gpm) ______

Steam generator conditions
- partitioned ______
- not partitioned ______

Atmospheric release point
- safety valves ______
- condenser steam-jet air-ejector ______

Step 3

Go to appropriate event tree based on coolant and core conditions.

Normal and 100X spike (safety valve release only) . . . C-28
Gap C-30
Grain Boundary C-32
Melt C-34

PWR STEAM GENERATOR TUBE RUPTURE EVENT TREES

GENERAL DESCRIPTION: Two normal range coolant concentrations (normal and 100X spike) are assumed in addition to three accident concentrations. The 100X spike assumes all the normal non-noble fission product concentrations are 100X higher than normal. The accident concentrations assume that all of the gap, grain boundary, or melt core release fractions are contained in the coolant. The assumed coolant concentrations are found on page C-87. The coolant is assumed to be released by a steam generator tube rupture (SGTR) to the secondary side and then to the atmosphere by the safety relief valves or through the condenser and then the steam-jet air-ejector (Figure C-1 or C-5, pathway C-1 or C-4). The release rate for total failure of one SG tube at full pressure or for coolant (Figures C-1 and C-5, pathway A-3) being pushed out by one charging pump is assumed. The reduction of non-nobles by the factors shown on page C-85 are used. A 50% reduction of non-noble fission products, to account for secondary side dilution, is assumed. For a dry primary and secondary side release pathway, the BWR/PWR containment bypass should be assumed (page C-61.)

For multiple tube failures or once through steam generators normally assume that the release is not partitioned and multiply the dose by the number of tubes estimated to have failed.

Doses at 1 mile are calculated using the factors shown on C-89 to estimate a 1 hour ground level release, 4 mph, and D stability wind conditions. Whole body dose includes 1 hour of cloud shine and acute (30 day committed) inhalation dose and 3 hours of ground shine. Noble gas decay is included by assuming a 1 hour decay for the high-pressure release and a 6 hour decay for the low-pressure release. Thyroid doses are for adults and from inhalation only.

[Figure C-1 & C-5 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 sq ft)
- 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

Figure C-1

PWR DRY CONTAINMENT SIMPLIFIED RELEASE PATHWAYS

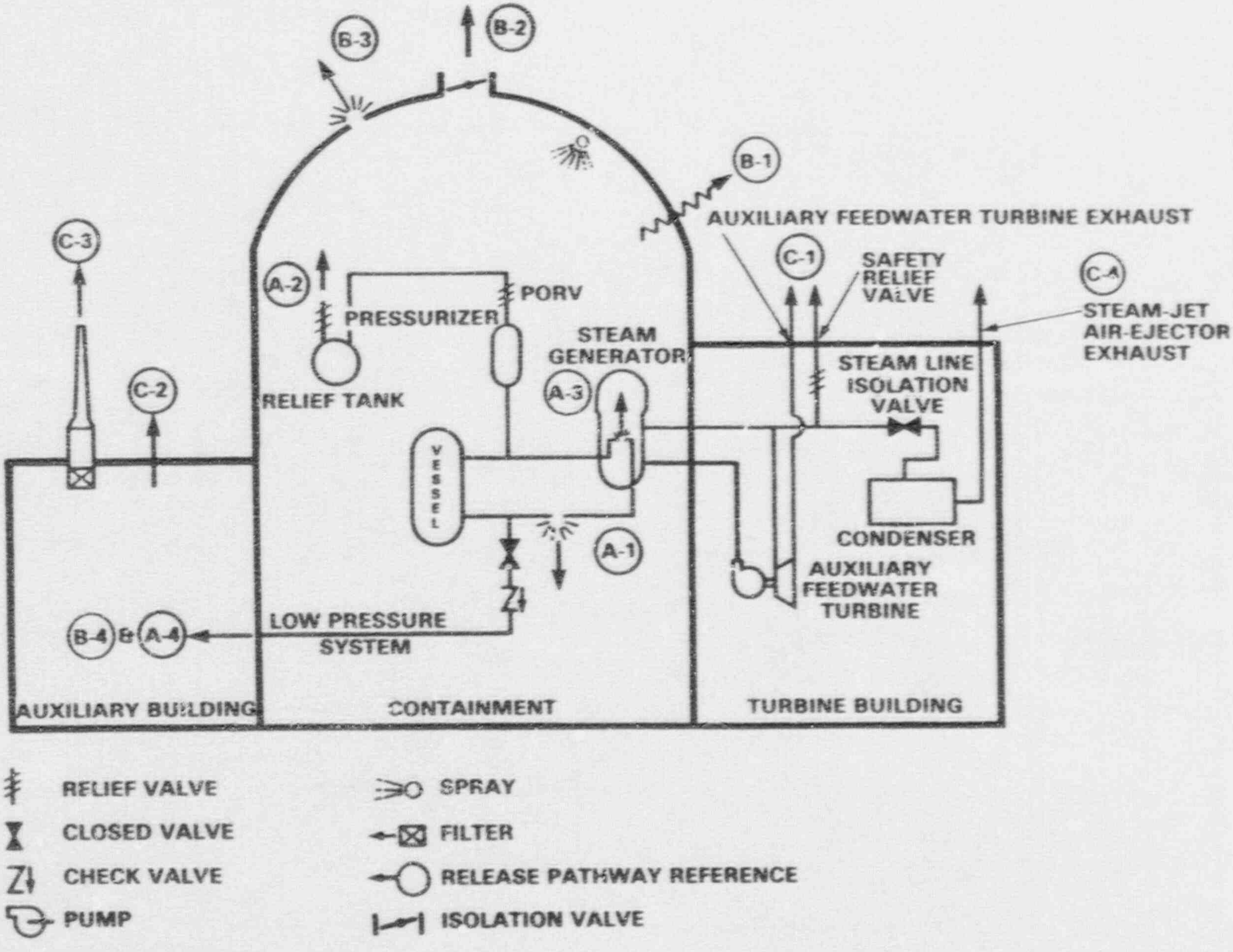
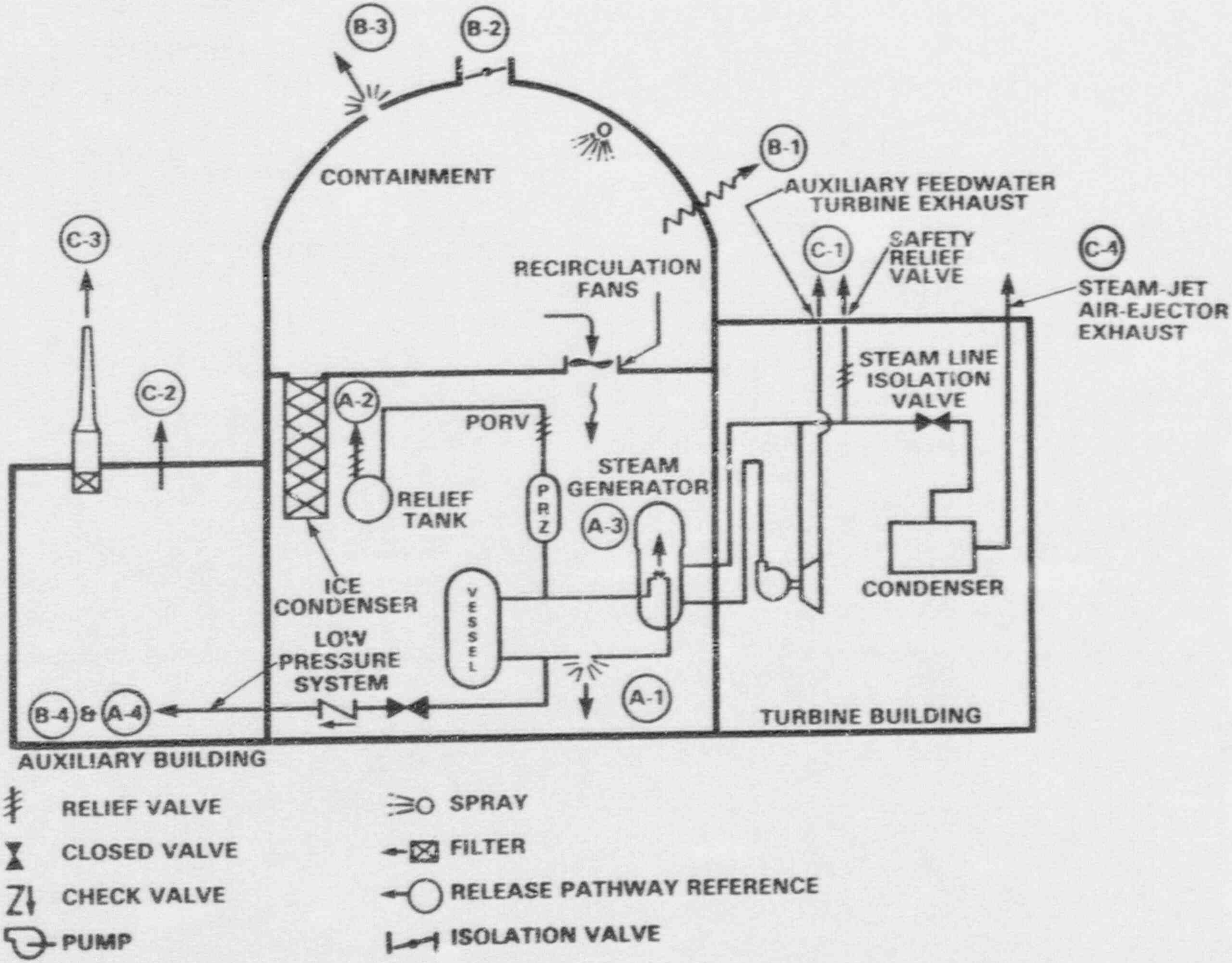


Figure C-5

PWR ICE CONDENSER CONTAINMENT SIMPLIFIED RELEASE PATHWAYS



PAGE INTENTIONALLY LEFT BLANK

NORMAL AND 100X SPIKE
STEAM GENERATOR TUBE RUPTURE

Step 4

Record from event tree (Figure C-9 on page C-29):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x $\frac{\_\_\_\_\_\text{ plant MW(e)}}{1000\text{ MW(e)}}$ = ________

Thyroid @ 1 mile ________ x $\frac{\_\_\_\_\_\text{ plant MW(e)}}{1000\text{ MW(e)}}$ = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile _______ x 0.40 = _______ whole body @ 2 mile
Whole body @ 1 mile _______ x 0.09 = _______ whole body @ 5 mile
Whole body @ 1 mile _______ x 0.03 = _______ whole body @10 mile

Thyroid @ 1 mile _______ x 0.40 = _______ thyroid @ 2 mile
Thyroid @ 1 mile _______ x 0.09 = _______ thyroid @ 5 mile
Thyroid @ 1 mile _______ x 0.03 = _______ thyroid @10 mile

Step 8

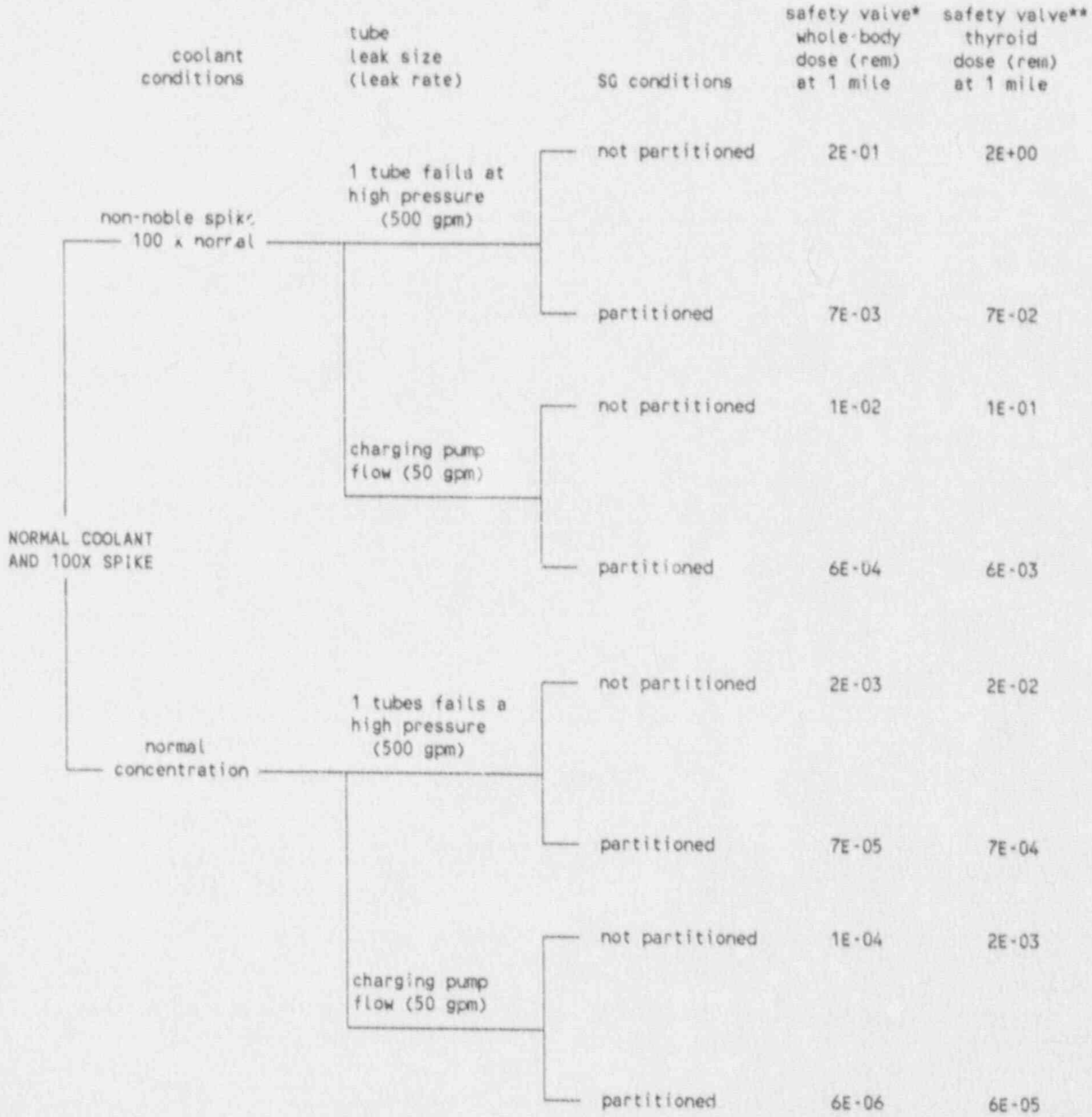
Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ______________
- Distance to which early health effects are possible (WB dose > 50 rem) ______________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ______________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-9

PWR STEAM GENERATOR TUBE RUPTURE EVENT TREE FOR A RELEASE OF NORMAL AND 100X SPIKE COOLANT



* Dose for releases by the air ejector will be a factor of 2 to 4 lower.
**Dose for releases by the air ejector will be about a factor of 50 lower.

GAP
STEAM GENERATOR TUBE RUPTURE

Step 4

Record from event tree (Figure C-10 on page C-31):

Acute whole body dose @ 1 mile ____________
Thyroid dose @ 1 mile ____________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x $\frac{\text{\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-10

PWR STEAM GENERATOR TUBE RUPTURE EVENT TREE FOR A RELEASE OF COOLANT CONTAMINATED WITH A GAP RELEASE FROM THE CORE

coolant conditions	tube leak size	SG conditions	safety valve whole-body dose (rem) at 1 mile	safety valve thyroid dose (rem) at 1 mile	steam jet air ejector whole-body dose (rem) at 1 mile	steam jet air ejector thyroid dose (rem) at 1 mile
GAP CONCENTRATION	1 tube fails at high presure (500 gpm)	not partitioned	2E+01	2E+03	1E+00	4E+01
		partitioned	2E+00	7E+01	9E-01	1E+00
	charging pump flow (50 gpm)	not partitioned	2E+00	2E+02	5E-02	3E+00
		partitioned	8E-02	6E+00	2E-02	1E-01

GRAIN BOUNDARY
STEAM GENERATOR TUBE RUPTURE

Step 4

Record from event tree (Figure C-11 on page C-33):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x $\frac{\text{\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile _______ x 0.40 = _______ whole body @ 2 mile
Whole body @ 1 mile _______ x 0.09 = _______ whole body @ 5 mile
Whole body @ 1 mile _______ x 0.03 = _______ whole body @10 mile

Thyroid @ 1 mile _______ x 0.40 = _______ thyroid @ 2 mile
Thyroid @ 1 mile _______ x 0.09 = _______ thyroid @ 5 mile
Thyroid @ 1 mile _______ x 0.03 = _______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ________
- Distance to which early health effects are possible (WB dose > 50 rem) ________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-11

PWR STEAM GENERATOR TUBE RUPTURE EVENT TREE FOR A RELEASE OF COOLANT CONTAMINATED WITH A GRAIN BOUNDARY RELEASE FROM THE CORE

coolant conditions	tube leak size	SG conditions	safety valve whole-body dose (rem) at 1 mile	safety valve thyroid dose (rem) at 1 mile	steam jet air ejector whole-body dose (rem) at 1 mile	steam jet air ejector thyroid dose (rem) at 1 mile
GRAIN BOUNDARY CONCENTRATION	1 tube fails at high presure (500 gpm)	not partitioned	4E+02	4E+04	2E+01	7E+02
		partitioned	3E+01	1E+03	1E+01	3E+01
	charging pump flow (50 gpm)	not partitioned	3E+01	3E+03	9E-01	6E+01
		partitioned	1E+00	1E+02	3E-01	2E+00

MELT
STEAM GENERATOR TUBE RUPTURE

Step 4

Record from event tree (Figure C-12 on page C-35):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x $\frac{\text{\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile _______ x 0.40 = _______ whole body @ 2 mile
Whole body @ 1 mile _______ x 0.09 = _______ whole body @ 5 mile
Whole body @ 1 mile _______ x 0.03 = _______ whole body @10 mile

Thyroid @ 1 mile _______ x 0.40 = _______ thyroid @ 2 mile
Thyroid @ 1 mile _______ x 0.09 = _______ thyroid @ 5 mile
Thyroid @ 1 mile _______ x 0.03 = _______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ______________
- Distance to which early health effects are possible (WB dose > 50 rem) ______________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ______________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-12

PWR STEAM GENERATOR TUBE RUPTURE EVENT TREE FOR A RELEASE OF COOLANT CONTAMINATED WITH A MELT RELEASE FROM THE CORE

coolant conditions	tube leak size	SG conditions	safety valve whole-body dose (rem) at 1 mile	safety valve thyroid dose (rem) at 1 mile	steam jet air ejector whole-body dose (rem) at 1 mile	steam jet air ejector thyroid dose (rem) at 1 mile
MELT CONCENTRATION	1 tube fails at high presure (500 gpm)	not partitioned	9E+02	9E+04	5E+01	2E+03
		partitioned	6E+01	3E+03	3E+01	7E+01
	charging pump flow (50 gpm)	not partitioned	8E+01	8E+03	2E+00	2E+02
		partitioned	4E+00	3E+02	7E-01	6E+00

PAGE INTENTIONALLY LEFT BLANK

BWR CONTAINMENT DRY-WELL LEAKAGE/FAILURE

NOTE:

> A general description of the assumptions used in the event trees are found on page C-38.

Step 2

NOTE:

> Holdup time is from the release to the containment to the start of the release to the environment.

Select the appropriate plant conditions:

- Core condition (see Core Damage Progression Once Uncovered, B-4)
  - gap release (1300-2000°F) ____________
  - grain boundary (>3000°F) ____________
  - melt (>4500°) ____________
- Dry-well containment sprays
  - on ____________
  - off ____________
- Holdup time in dry well before release
  - <1 hour ____________
  - 2-12 hours ____________
  - 24 hours ____________
- Dry-well leak rate
  - design (0.5%/day) ____________
  - 100%/day (isolation failure ____________
  - 100%/hour (catastropic) ____________
- Release through filters (SBGTS)
  - yes ____________
  - no ____________

Step 3

Go to appropriate event tree based on core condition.

Gap C-42
Grain Boundary C-44
Melt C-46

BWR CONTAINMENT DRY-WELL LEAKAGE RELEASE EVENT TREES

GENERAL DESCRIPTION: The core is uncovered and release fractions on page C-84 for a 1000 MWe plant typical of a gap (1300-2000°F), grain boundary (3000°F), or melt (4500°F) release from the core are assumed. For the vessel melt-through cases (ex-vessel melt) the strontium release fractions could be substantially increased. This could increase the projected whole body dose by 50%. The release is assumed to pass by a dry pathway through the primary system into the containment dry-well atmosphere without passing through the suppression pool. Particulates and aerosols airborne in the containment dry well are reduced by the factors shown on page C-85 to account for the actions of sprays and natural processes for <1, 2-12, or 24 hours holdup in the containment. Releases from the containment dry well are estimated for 0.5%/day (design leakage), 100%/day (isolation valve seal failure), or 100%/hour (catastrophic failure > 1 sq ft). The release can be filtered by the standby gas treatment system (SBGTS) (pathway C-2) before release or unfiltered (bypass the SBGTS).

Doses at 1 mile are calculated using the factors shown on page C-89 to estimate a 1 hour ground level release, 4 mph, and D stability wind conditions. Acute whole body dose includes 1 hour of cloud shine and acute (30 day committed) inhalation dose and 3 hours of ground shine. Noble gas decay included. Thyroid doses are for adults and from inhalation only.

[Figures C-13, C-14 and C-15 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)

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)

Figure C-13

BWR MARK I SIMPLIFIED RELEASE PATHWAYS

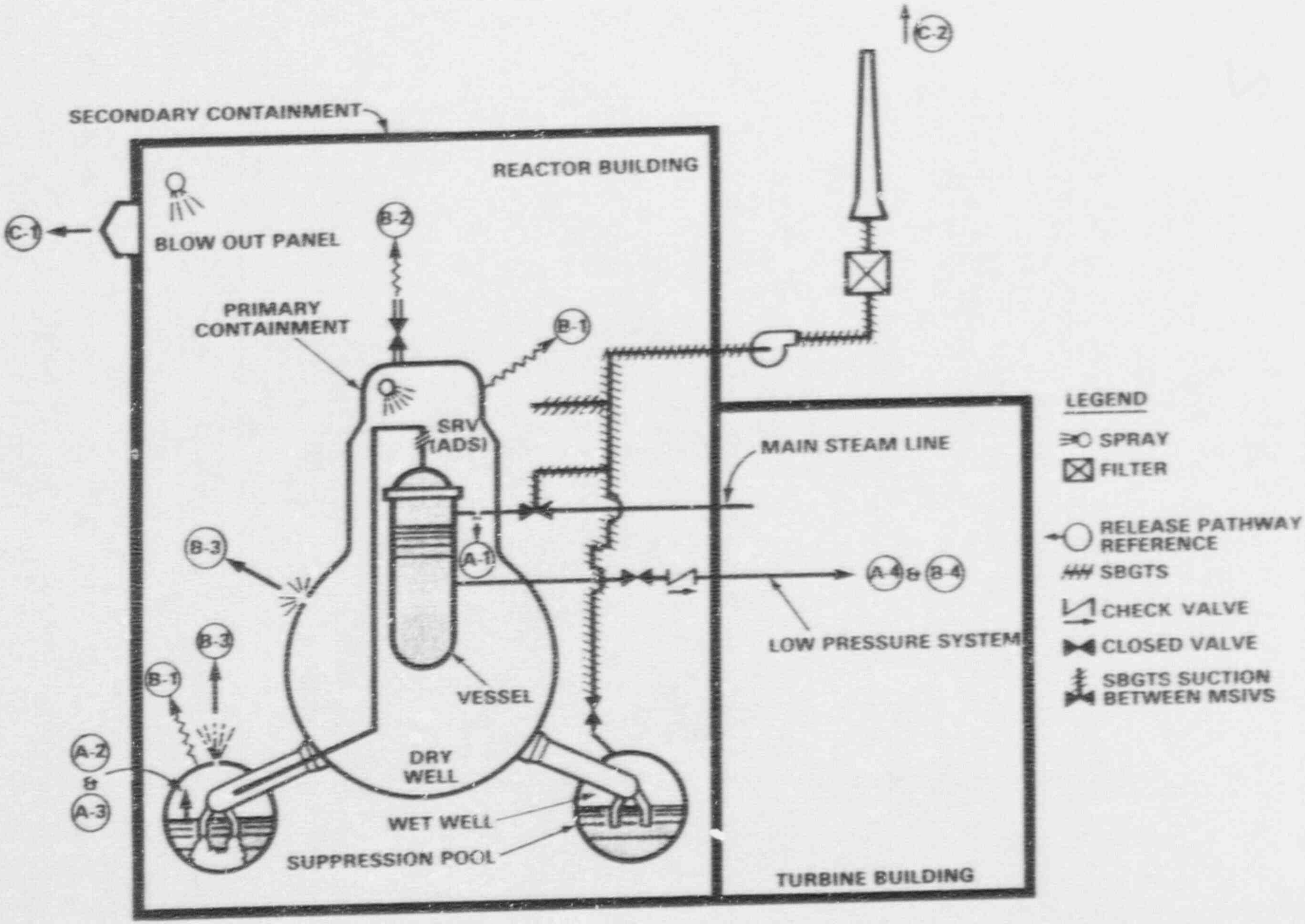


Figure C-14

BWR MARK II SIMPLIFIED RELEASE PATHWAYS

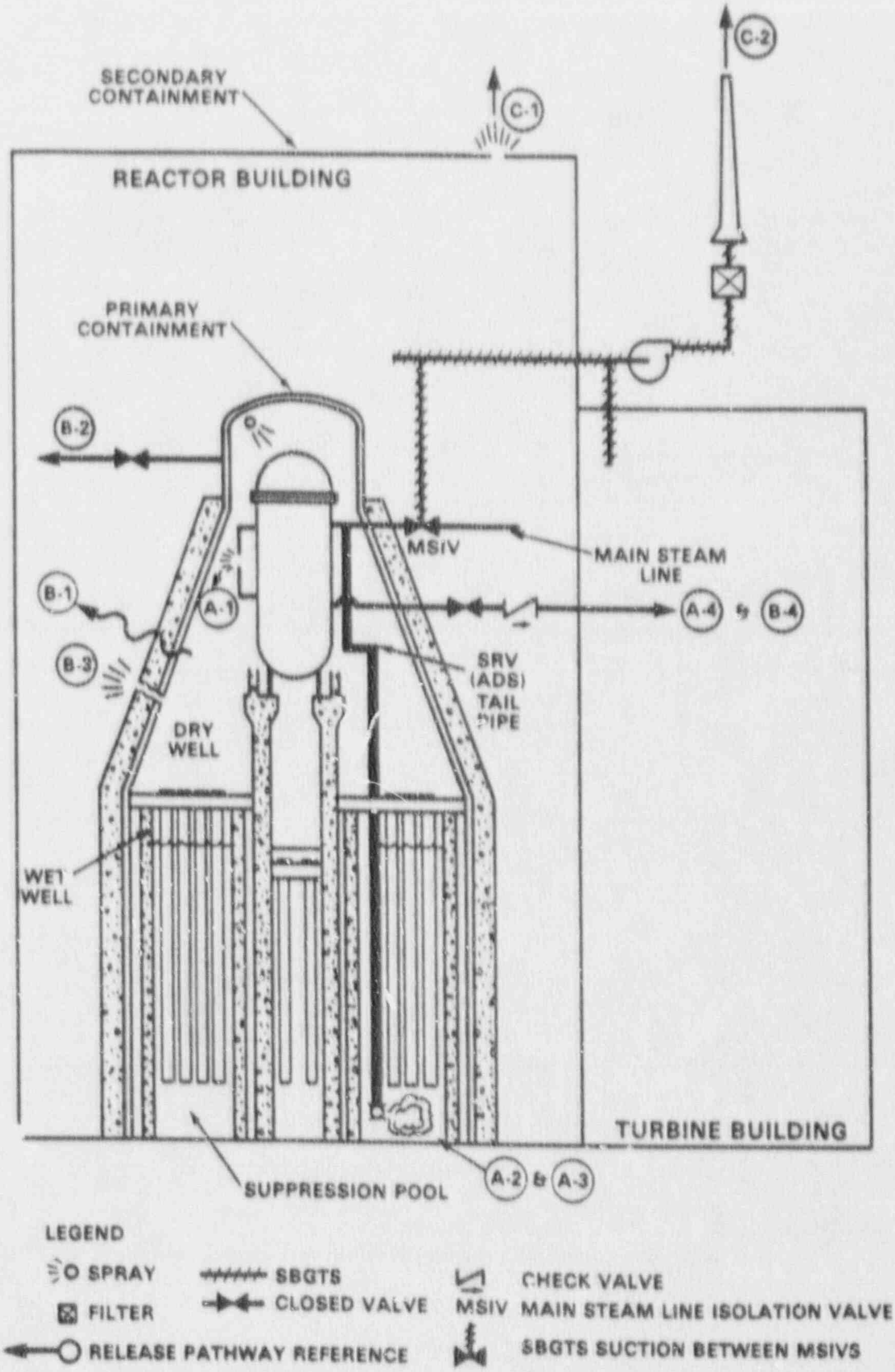
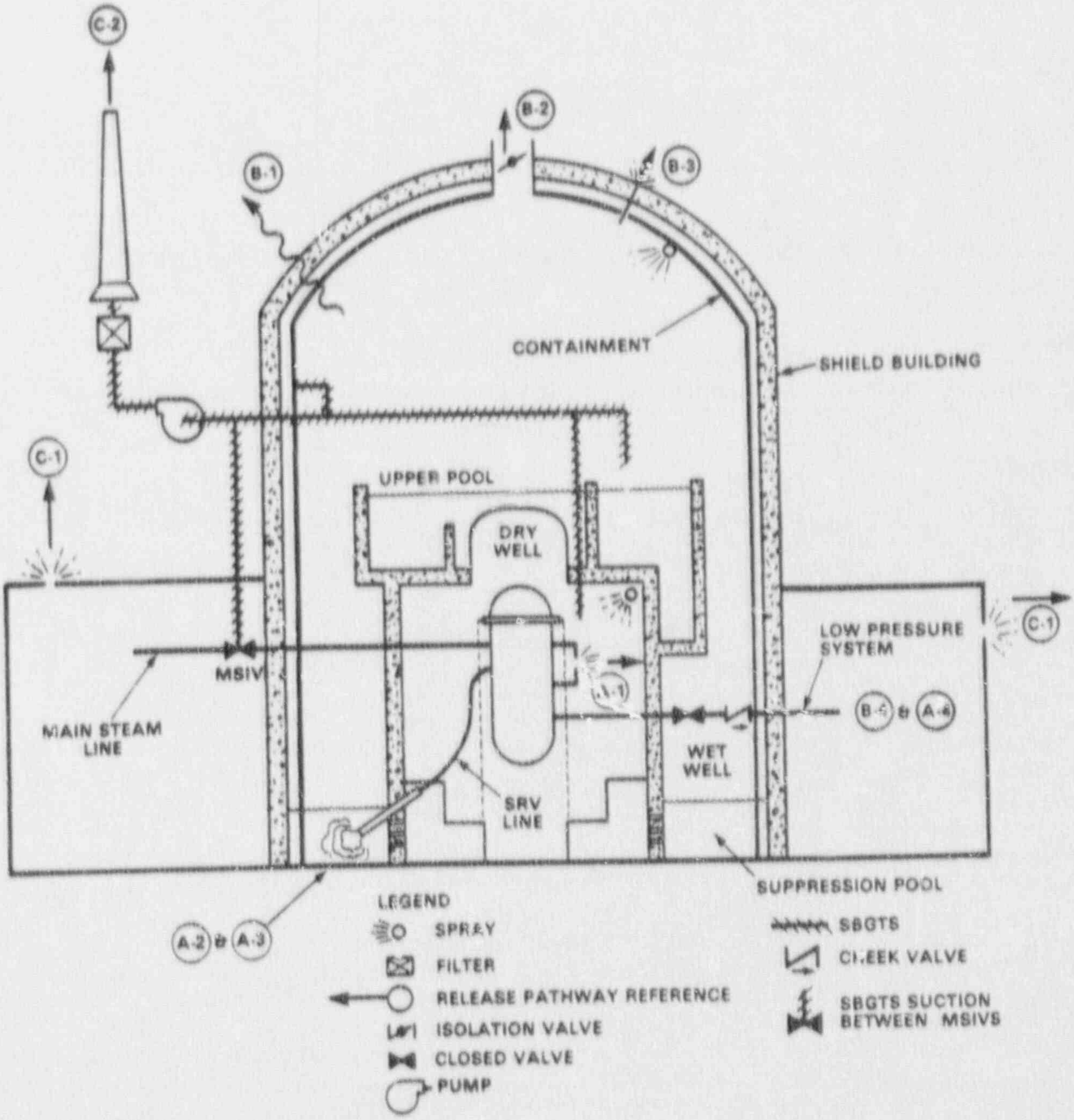


Figure C-15

BWR MARK III SIMPLIFIED RELEASE PATHWAYS



GAP
BWR DRY WELL

Step 4

Record from event tree (Figure C-16 on page C-43):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile__________ x $\frac{\text{\_\_\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Thyroid @ 1 mile __________ x $\frac{\text{\_\_\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ______________
- Distance to which early health effects are possible (WB dose > 50 rem) ______________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ______________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-16

BWR CONTAINMENT, DRY-WELL LEAKAGE/FAILURE RELEASE EVENT TREE FOR A GAP RELEASE FROM THE CORE

core and dry well conditions	holdup time in dry well	dry well leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
GAP RELEASE FROM CORE — sprays off	<1 hour	100%/hour	2E+01	2E+03	****	****
	2-12 hours	100%/hour	5E+00	4E+02	****	****
		100%/day	2E-01	2E+01	3E-02	2E-01
		design leak rate	9E-04	8E-02	1E-04	8E-04
	24 hours	100%/hour	1E+00	1E+02	****	****
		100%/day	5E-02	4E+00	9E-03	4E-02
		design leak rate	2E-04	2E-02	4E-05	2E-04
GAP RELEASE FROM CORE — sprays on	<1 hour	100%/hour	5E+00	3E+02	****	****
	2-12 hours	100%/hour	3E+00	2E+02	****	****
		100%/day	1E-01	8E+00	2E-02	8E-02
		design leak rate	5E-04	4E-02	1E-04	4E-04
	24 hours	100%/hour	4E-01	2E+01	****	****
		100%/day	2E-02	8E-01	8E-03	8E-03
		design leak rate	8E-05	4E-03	4E-05	4E-05

**** - No filtering (filters assumed to blow-out).

GRAIN BOUNDARY
BWR DRY WELL

Step 4

Record from event tree (Figure C-17 on page C-45):

Acute whole body dose @ 1 mile ____________________
Thyroid dose @ 1 mile ____________________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x $\frac{\text{\_\_\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ____________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = ____________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-17

BWR CONTAINMENT, DRY-WELL LEAKAGE/FAILURE RELEASE EVENT TREE FOR A GRAIN BOUNDARY RELEASE FROM THE CORE

GRAIN BOUNDARY RELEASE FROM CORE

core and dry well conditions	hold up time in dry well	dry well leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
sprays off	<1 hour	100%/hour	4E+02	4E+04	****	****
	2-12 hours	100%/hour	9E+01	8E+03	****	****
		100%/day	4E+00	3E+02	4E-01	3E+00
		design leak rate	2E-02	2E+00	2E-03	2E-02
	24 hours	100%/hour	2E+01	2E+03	****	****
		100%/day	9E-01	8E+01	1E-01	8E-01
		design leak rate	5E-03	4E-01	7E-04	4E-03
sprays on	<1 hour	100%/hour	1E+02	6E+03	****	****
	2-12 hours	100%/hour	5E+01	4E+03	****	****
		100%/day	2E+00	2E+02	4E-01	2E+00
		design leak rate	1E-02	8E-01	2E-03	8E-03
	24 hours	100%/hour	7E+00	[illegible]E+02	****	****
		100%/day	3E-01	2E+01	1E-01	2E-01
		design leak rate	2E-03	8E-02	7E-04	8E-04

**** - No filtering (filters assumed to blow-out)

CORE MELT
BWR DRY WELL

Step 4

Record from event tree (Figure C-18 on page C-47):

Acute whole body dose @ 1 mile ____________________
Thyroid dose @ 1 mile ____________________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ____________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ____________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-18

BWR CONTAINMENT, DRY-WELL LEAKAGE/FAILURE RELEASE EVENT TREE FOR A MELT RELEASE FROM THE CORE

core and dry well conditions	hold up time in dry well	dry well leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
MELT RELEASE FROM CORE — sprays off	<1 hour	100%/hour	1E+03	1E+05	****	****
	2-12 hours	100%/hour	2E+02	2E+04	****	****
		100%/day	9E+00	8E+02	9E-01	8E+00
		design leak rate	4E-02	4E+00	4E-03	4E-02
	24 hours	100%/hour	6E+01	5E+03	****	****
		100%/day	2E+00	2E+02	3E-01	2E+00
		design leak rate	1E-02	1E+00	2E-03	1E-02
MELT RELEASE FROM CORE — sprays on	<1 hour	100%/hour	2E+02	2E+04	****	****
	2-12 hours	100%/hour	1E+02	1E+04	****	****
		100%/day	5E+00	4E+02	8E-01	4E+00
		design leak rate	2E-02	2E+00	4E-03	2E-02
	24 hours	100%/hour	2E+01	1E+03	****	****
		100%/day	7E-01	4E+01	3E-01	4E-01
		design leak rate	3E-03	2E-01	1E-03	2E-03

**** - No filtering (filters assumed to blow-out)

PAGE INTENTIONALLY LEFT BLANK

BWR CONTAINMENT WET-WELL LEAKAGE/FAILURE

NOTE:

> A general description of the assumptions used in the event trees are found on page C-50.

Step 2

CAUTION:

> For bypass of the suppression pool or cases where more than decay heat are released to the pool, use BWR containment dry-well on page C-37.

Select the appropriate plant conditions:

NOTE:

> Holdup time is the time from the release to the containment atmosphere to the start of the release to the environment.

Core condition (see Core Damage Progression Once Uncovered, B-4)
- gap release (1300-2000°F) ____________
- grain boundary (3000°F) ____________
- melt (4500°F) ____________

Suppression pool conditions (see Saturation Table B-13)
- saturated ____________
- subcooled ____________

Holdup time in dry/wet well before release
- <1 hour ____________
- 2-12 hours ____________
- 24 hours ____________

Wet-well leak rate
- design (0.5%/day) ____________
- 100%/day (isolation failure) ____________
- 100%/hour (catastrophic) ____________

Release through filters (SBGTS)
- yes ____________
- no ____________

Step 3

Go to appropriate event tree based on coolant and core conditions.

Gap C-54
Grain Boundary C-56
Melt C-58

BWR CONTAINMENT WET-WELL LEAKAGE RELEASE EVENT TREES

GENERAL DESCRIPTION: The core is uncovered and the release fractions on page C-84 typical of a gap (1300-2000°F), grain boundary (3000°F), or melt (4500°F) release from the core are assumed. For the vessel melt-through cases (ex-vessel melt) the strontium release fractions could be substantially increased. This could increase the projected whole body dose by 50%. The release is assumed to pass through the suppression pool into the containment wet-well atmosphere. Particulates and aerosols airborne in the containment wet well are reduced by the factors shown on page C-85 to account for the scrubbing of the suppression pool under subcooled and saturated conditions. For bypass of the suppression pool, use BWR containment dry-well trees on page C-37. It is assumed that only decay heat is being released. If the reactor is not shut down, 10 to 20 times as much particulates could be released through the pool and under these conditions and therefore no scrubbing from the pool should be assumed (e.g., use dry well case). In addition, it is assumed that the particulates are reduced to account for natural depletion while held up in the containment. Releases from the containment wet well can be estimated for 0.5%/day (design leakage); 100%/day (isolation valve seal failure); or 100%/hour (catastrophic failure >1 sq ft). The release can be filtered by the SBGTS (pathway C-2) before release or bypass the SBGTS (pathway C-1).

Doses at 1 mile are calculated using the factors shown on C-89 to estimate a 1 hour ground level release, 4 mph, and D stability wind conditions. Acute whole body dose includes 1 hour of cloud shine and acute (30 day committed) inhalation dose and 3 hours of ground shine. Noble gas decay included. Thyroid doses are for adults and from inhalation only.

[Figures C-13, C-14 and C-15 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)

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)

Figure C-13

BWR MARK I SIMPLIFIED RELEASE PATHWAYS

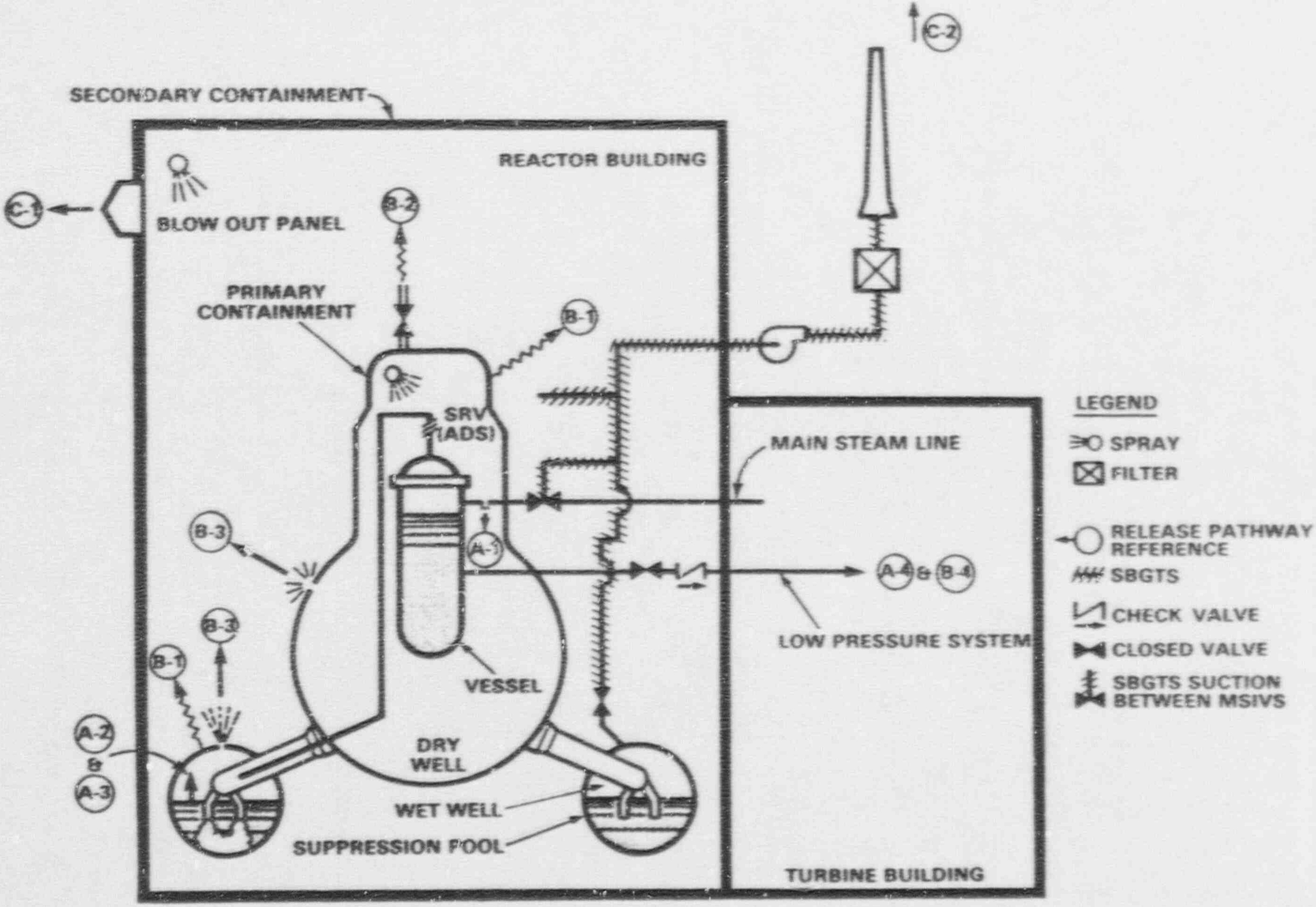


Figure C-14

BWR MARK II SIMPLIFIED RELEASE PATHWAYS

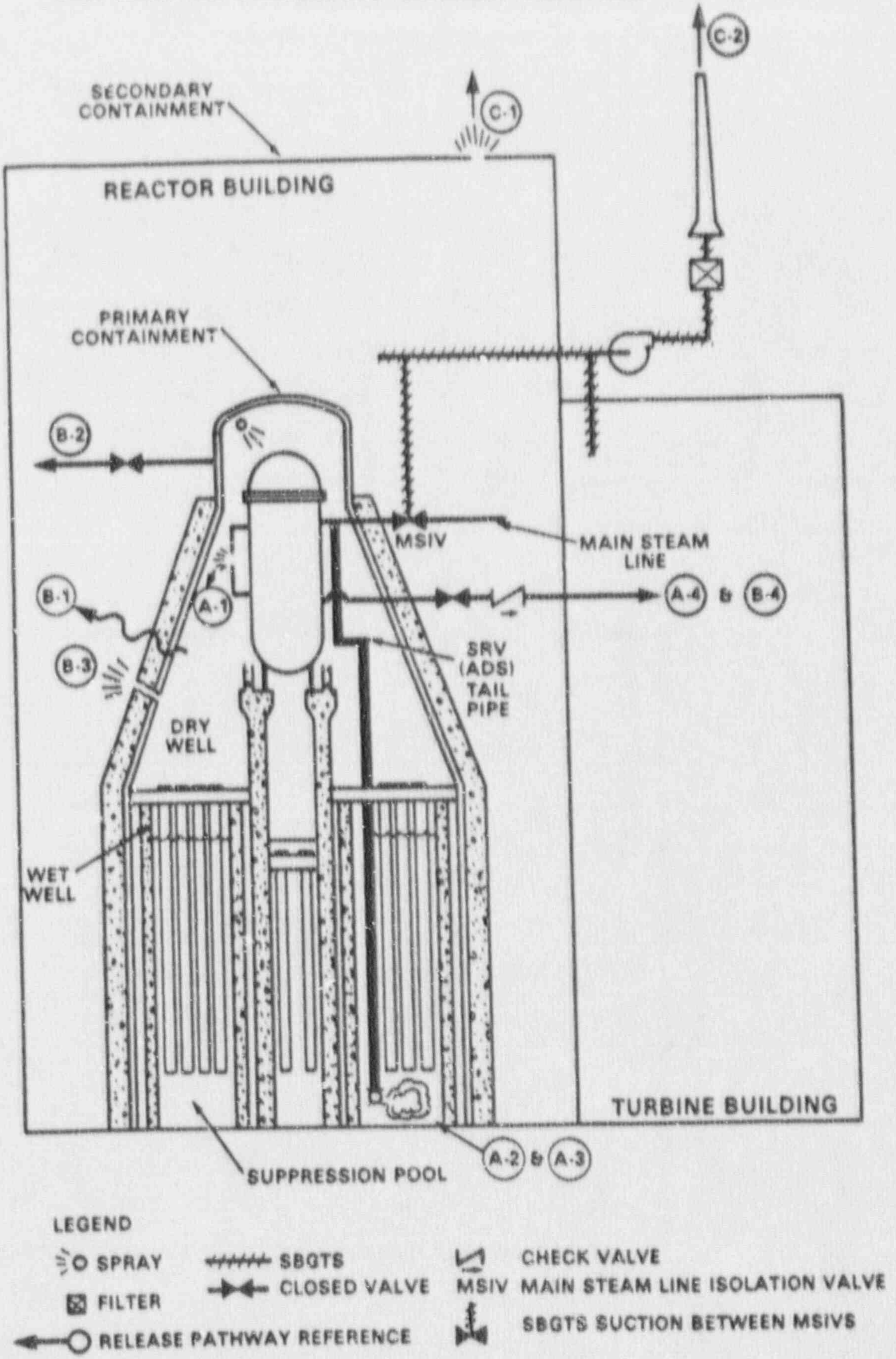
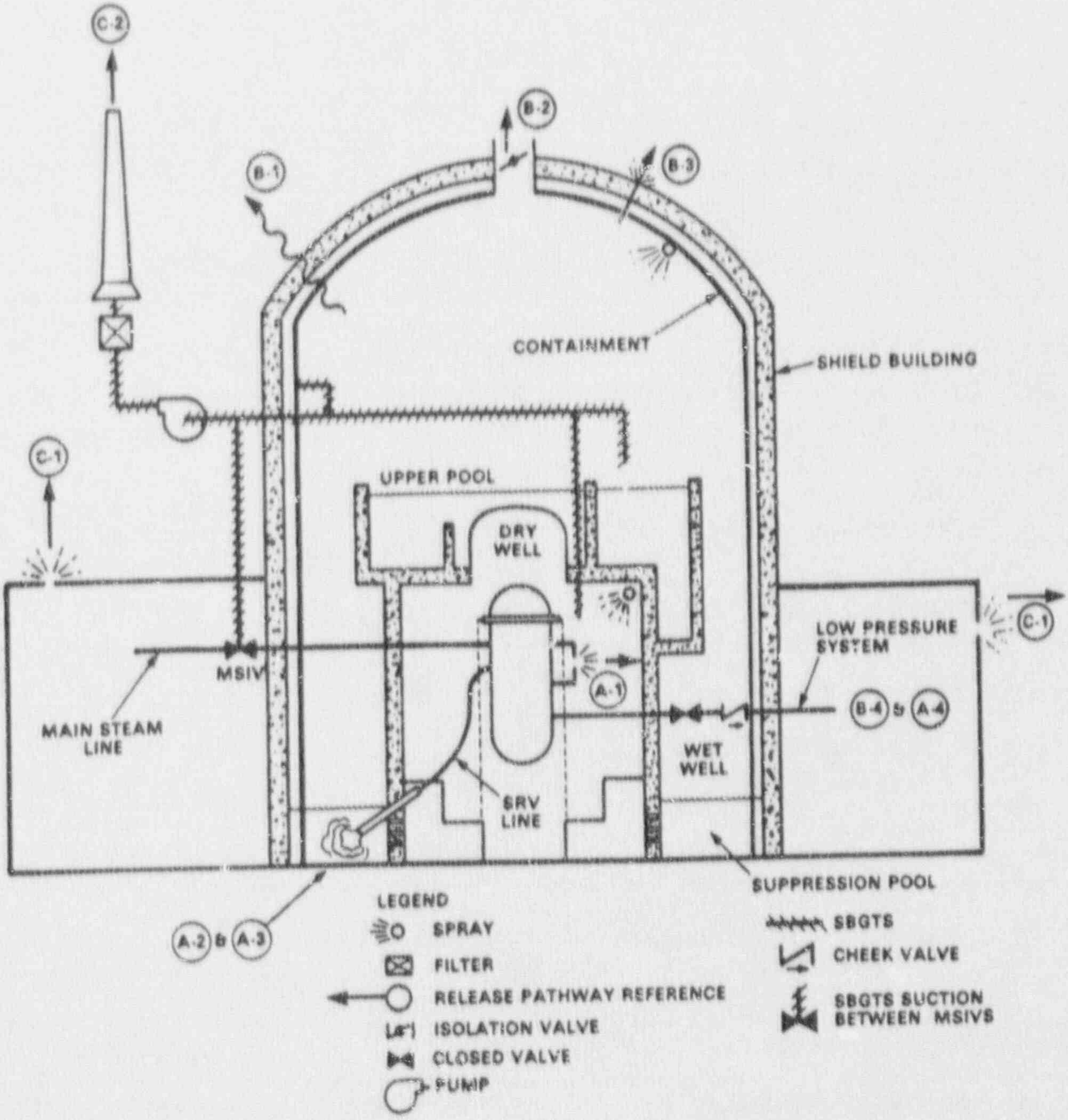


Figure C-15

BWR MARK III SIMPLIFIED RELEASE PATHWAYS



GAP
BWR WET WELL

Step 4

Record from event tree (Figure C-19 on page C-55):

Acute whole body dose @ 1 mile ____________
Thyroid dose @ 1 mile ____________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x ________ plant MW(e) / 1000 MW(e) = ________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ________
- Distance to which early health effects are possible (WB dose > 50 rem) ________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-19

BWR CONTAINMENT, WET-WELL LEAKAGE/FAILURE RELEASE EVENT TREE FOR A GAP RELEASE FORM THE CORE

GAP RELEASE FROM CORE

core and suppression pool conditions	hold up time in dry/wet well	wet well leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
sat	<1 hour	100%/hour	4E+00	2E+02	****	****
sat	2-12 hours	100%/hour	8E-01	2E+01	****	****
sat	2-12 hours	100%/day	3E-02	8E-01	2E-02	8E-03
sat	2-12 hours	design rate	2E-04	4E-03	1E-04	4E-05
sat	24 hours	100%/hour	3E-01	1E+01	****	****
sat	24 hours	100%/day	1E-02	4E-01	8E-03	4E-03
sat	24 hours	design rate	6E-05	2E-03	4E-05	2E-05
sub	<1 hour	100%/hour	3E+00	4E+01	****	****
sub	2-12 hours	100%/hour	7E-01	1E+01	****	****
sub	2-12 hours	100%/day	3E-02	4E-01	2E-02	4E-03
sub	2-12 hours	design rate	1E-04	2E-03	1E-04	2E-05
sub	24 hours	100%/hour	3E-01	1E+01	****	****
sub	24 hours	100%/day	1E-02	4E-01	8E-03	4E-03
sub	24 hours	design rate	6E-05	2E-03	4E-05	2E-05

**** - No filtering (filters assumed to blow-out)

GRAIN BOUNDARY
BWR WET WELL

Step 4

Record from event tree (Figure C-20 on page C-57):

Acute whole body dose @ 1 mile ____________________
Thyroid dose @ 1 mile ____________________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x $\frac{\text{\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Thyroid @ 1 mile ________ x $\frac{\text{\_\_\_\_\_\_ plant MW(e)}}{\text{1000 MW(e)}}$ = __________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = _______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = _______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = _______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = _______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = _______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = _______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) __________
- Distance to which early health effects are possible (WB dose > 50 rem) __________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) __________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-20

BWR CONTAINMENT WET-WELL LEAKAGE/FAILURE RELEASE EVENT TREE FOR A GRAIN BOUNDARY RELEASE FROM THE CORE

GRAIN BOUNDARY RELEASE FROM CORE

core and suppression pool conditions	hold up time in dry/wet well	wet well leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
sat	<1 hour	100%/hour	8E+01	4E+03	****	****
	2-12 hours	100%/hour	1E+01	4E+02	****	****
		100%/day	6E-01	2E+01	4E-01	2E-01
		design rate	3E-03	8E-02	2E-03	8E-04
	24 hours	100%/hour	6E+00	2E+02	****	****
		100%/day	2E-01	8E+00	1E-01	8E-02
		design rate	1E-03	4E-02	7E-04	4E-04
sub	<1 hour	100%/hour	5E+01	8E+02	****	****
	2-12 hours	100%/hour	1E+01	2E+02	****	****
		100%/day	5E-01	8E+00	4E-01	8E-02
		design rate	2E-03	4E-02	2E-03	4E-04
	24 hours	100%/hour	6E+00	2E+02	****	****
		100%/day	2E-01	8E+00	1E-01	8E-02
		design rate	1E-03	4E-02	7E-04	4E-04

**** - No filtering (filters assumed to blow-out)

CORE MELT
BWR WET WELL

Step 4

Record from event tree (Figure C-21 on page C-59):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ______________
- Distance to which early health effects are possible (WB dose > 50 rem) ______________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ______________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-21

BWR CONTAINMENT, WET-WELL LEAKAGE/FAILURE RELEASE EVENT TREE FOR A MELT RELEASE FROM THE CORE

MELT RELEASE FROM CORE

core and suppression pool conditions	hold up time in dry/wet well	wet well leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
sat	<1 hour	100%/hour	2E+02	1E+04	****	****
	2-12 hours	100%/hour	3E+01	1E+03	****	****
		100%/day	1E+00	4E+01	8E-01	4E-01
		design rate	6E-03	2E-01	4E-03	2E-03
	24 hours	100%/hour	1E+01	5E+02	****	****
		100%/day	5E-01	2E+01	3E-01	2E-01
		design rate	2E-03	1E-01	1E-03	1E-03
sub	<1 hour	100%/hour	1E+02	2E+03	****	****
	2-12 hours	100%/hour	2E+01	5E+02	****	****
		100%/day	1E+00	2E+01	8E-01	2E-01
		design rate	5E-03	1E-01	4E-03	1E-03
	24 hours	100%/hour	1E+01	5E+02	****	****
		100%/day	5E-01	2E+01	3E-01	2E-01
		design rate	2E-03	1E-01	1E-03	1E-03

**** - No filtering (filters assumed to blow-out)

PAGE INTENTIONALLY LEFT BLANK

BWR/PWR CONTAINMENT BYPASS

NOTE:

A general description of the assumptions used in the event trees are found on page C-62.

Step 2

Select the appropriate plant conditions:

Core condition (see Core Damage Progression Once Uncovered, B-4)
- gap release (1300-2000°F) ________
- grain boundary (3000°F) ________
- melt (4500°F) ________

Release path conditions
- filtered ________
- not filtered ________

Release rate
- typical design (0.1%/day) ________
- 100%/day ________
- 100%/hour ________

Step 3

Go to appropriate event tree based on coolant and core conditions.

Gap C-68
Grain Boundary C-70
Melt C-72

BWR/PWR CONTAINMENT BYPASS RELEASE EVENT TREES

GENERAL DESCRIPTION: The core is uncovered and release fractions on page C-84 for a 1000 MWe plant typical of a gap (1300-2000°F), grain boundary (3000°F), or melt (4500°F) release from the core are assumed. The release passes through a dry line that bypasses the containment. Particulates and aerosols in the release flow are reduced to account for plateout in the line. Releases through the line are calculated for the following release rates: 0.1%/day, 100%/day, or 100%/hour. At high pressure most interfacing failures could result in release of a large fraction of the available fission products in about an hour (100%/hr case).

Doses at 1 mile are calculated using the factors shown on page C-89 to estimate a 1 hour ground level release, 4 mph, and D stability wind conditions. Acute whole body dose includes 1 hour of cloud shine and acute (30 day committed) inhalation dose and 3 hours of ground shine. Noble gas decay is included by assuming 1 hour decay for 100%/hour release case and 6 hour decay for others. Thyroid doses are for adults and from inhalation only.

[Figure C-1 & C-5 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 sq ft)
- 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

[Figures C-13, C-14 and C-15 Key]

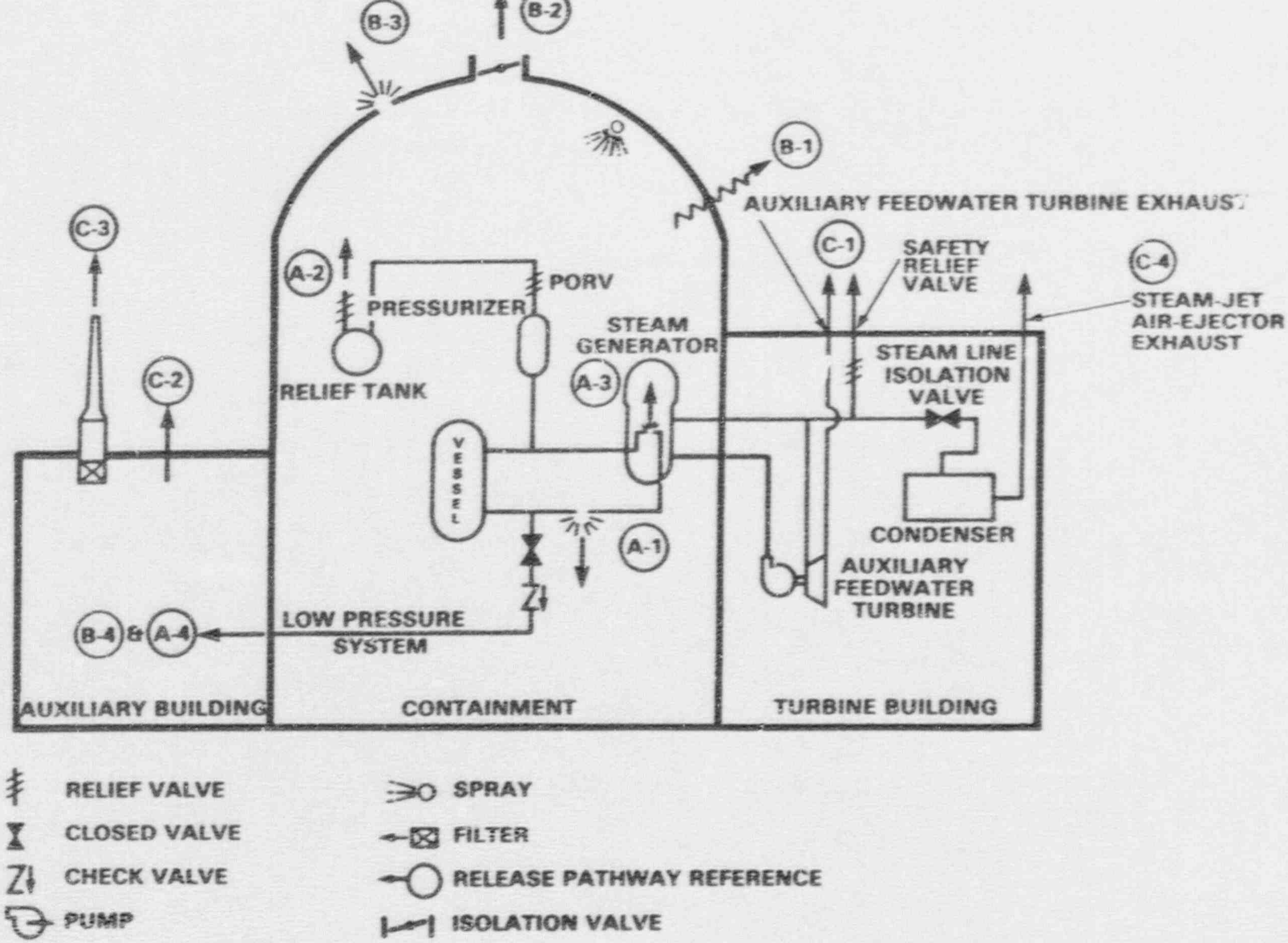
A Reactor Coolant System
- A-1 Breaks and leaks bypassing suppression pool
- A-2 Breaks and leaks through suppression pool
- A-3 Automatic depressurization system (ADS) and safety relief valves (SRV)

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)

Figure C-1

PWR DRY CONTAINMENT SIMPLIFIED RELEASE PATHWAYS



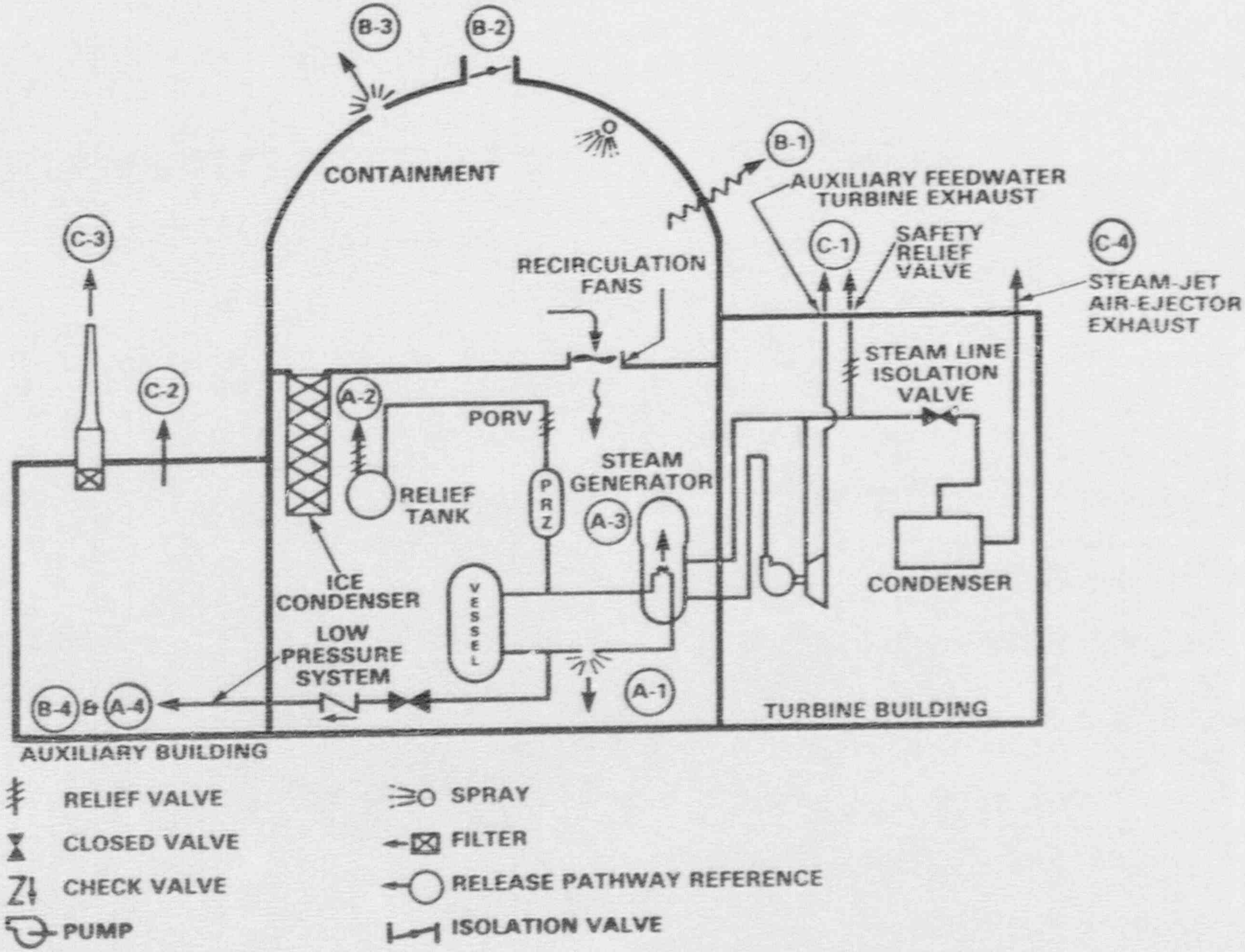


Figure C-5

PWR ICE CONDENSER CONTAINMENT RELEASE PATHWAYS

Figure C-13

BWR MARK I SIMPLIFIED RELEASE PATHWAYS

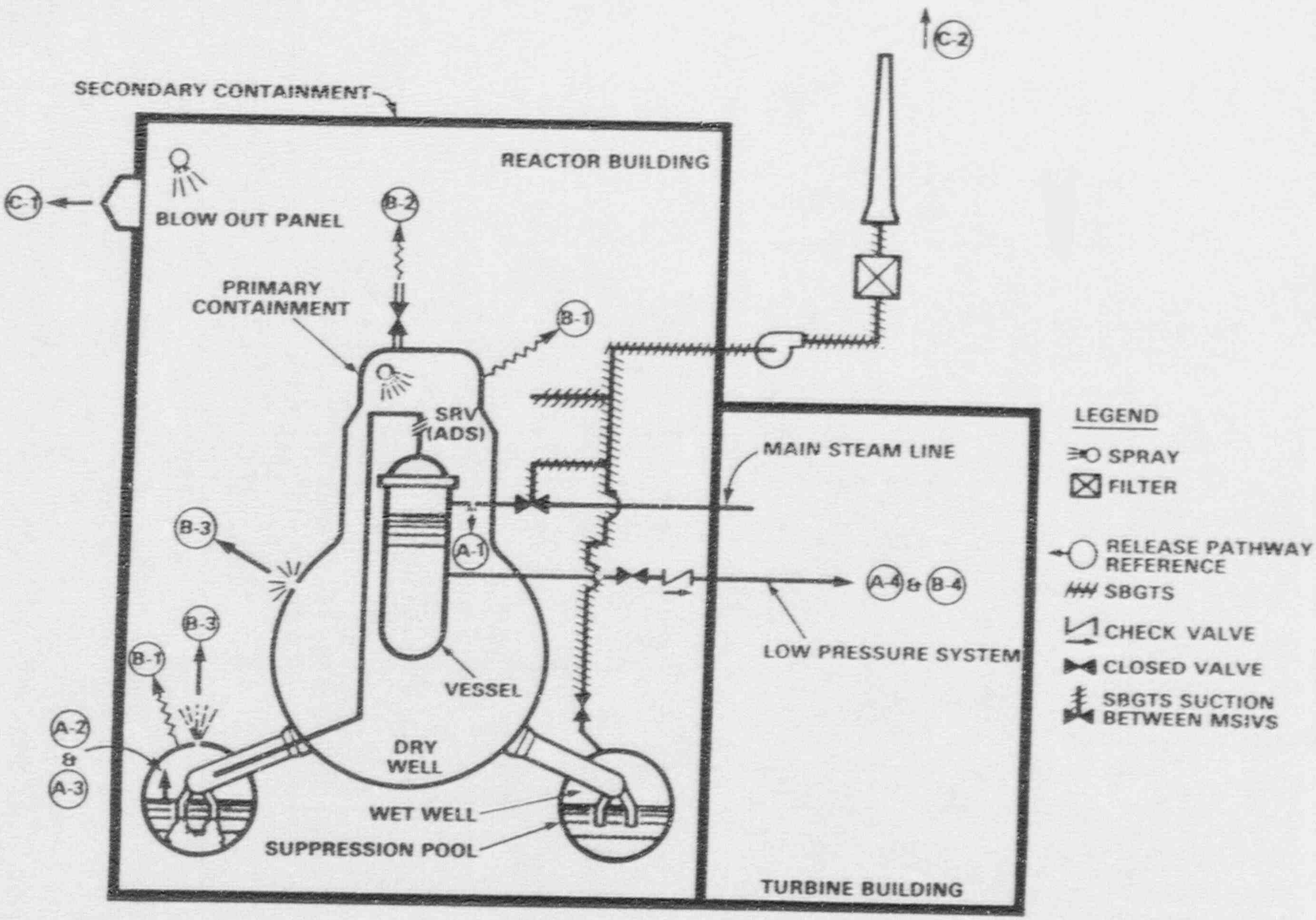


Figure C-14

BWR MARK II SIMPLIFIED RELEASE PATHWAYS

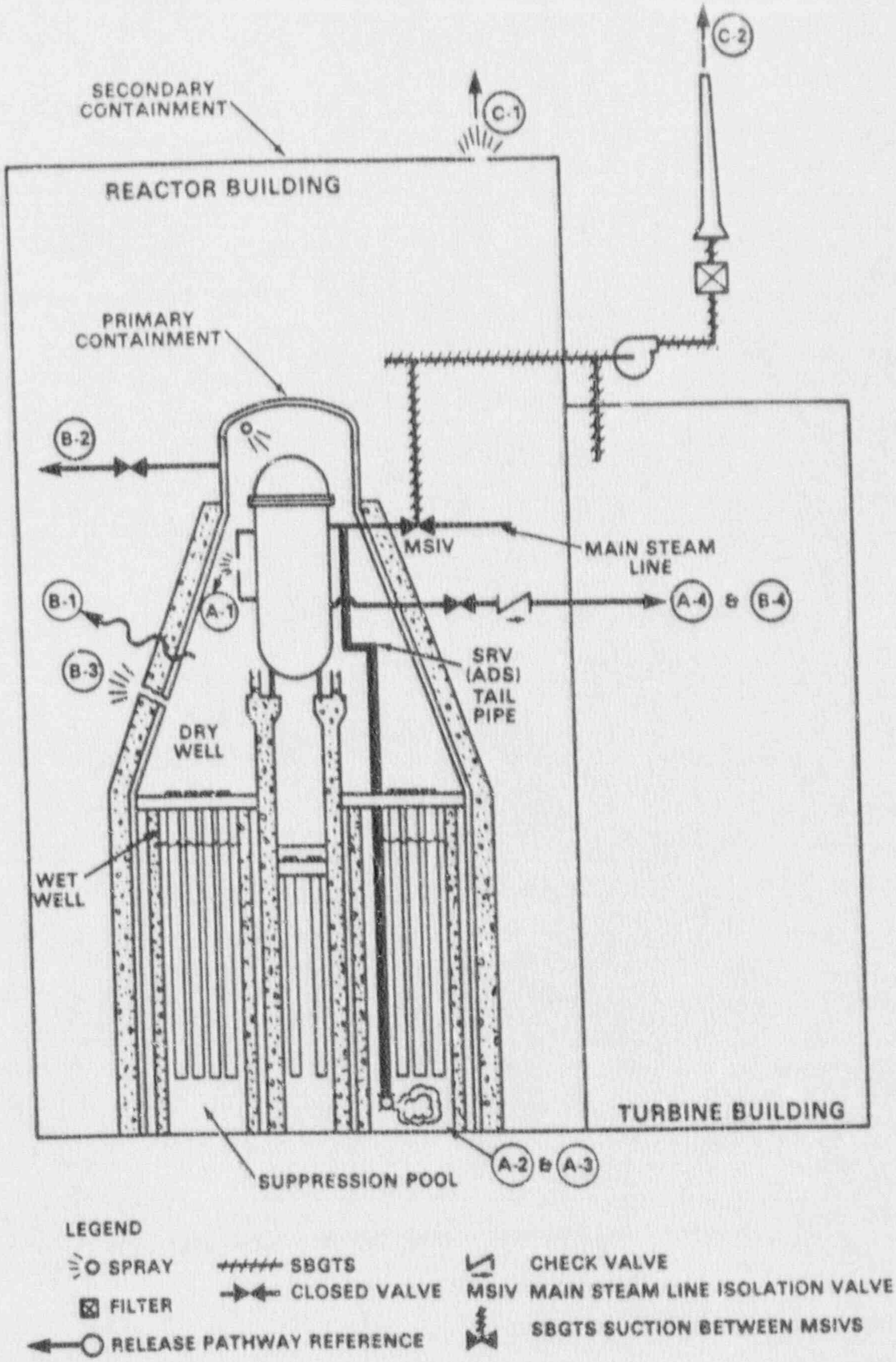
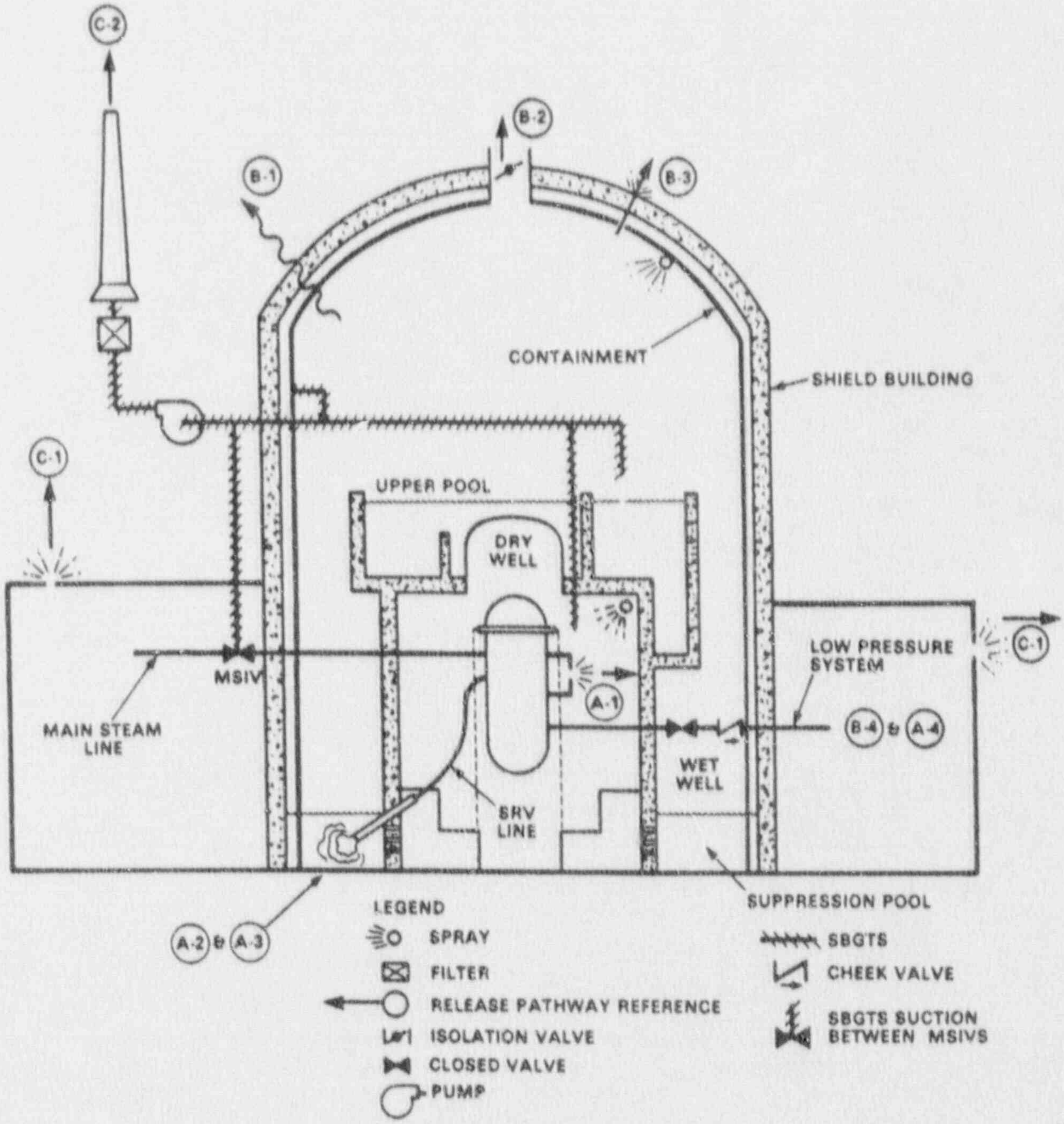


Figure C-15

BWR MARK III SIMPLIFIED RELEASE PATHWAYS



GAP
BYPASS

Step 4

Record from event tree (Figure C-22 on page C-69):

Acute whole body dose @ 1 mile ____________
Thyroid dose @ 1 mile ____________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x ________ plant MW(e) / 1000 MW(e) = ________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-22

PWR AND BWR CONTAINMENT BYPASS RELEASE EVENT TREE FOR A GAP RELEASE FROM THE CORE

core conditions	release conditions	release rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
GAP RELEASE FROM CORE	not filtered	100%/hour	2E+01	2E+03
		100%/day	8E-01	8E+01
		typical design rate	2E-03	2E-01
	filtered	100%/hour	****	*****
		100%/day	4E-02	2E+00
		typical design rate	1E-04	4E-03

**** - No filtering (filters assumed to blow-out)

GRAIN BOUNDARY
BYPASS

Step 4

Record from event tree (Figure C-23 on page C-71):

Acute whole body dose @ 1 mile ______________
Thyroid dose @ 1 mile ______________

Step 5

Adjust for reactor size:

Whole body @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ______________
- Distance to which early health effects are possible (WB dose > 50 rem) ______________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ______________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-23

PWR AND BWR CONTAINMENT BYPASS RELEASE EVENT TREE FOR A GRAIN BOUNDARY RELEASE FROM THE CORE

core conditions	release conditions	release rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
GRAIN BOUNDARY RELEASE FROM CORE	not filtered	100%/hour	4E+02	4E+04
		100%/day	2E+01	2E+03
		typical design rate	4E-02	4E+00
	filtered	100%/hour	****	****
		100%/day	7E-01	3E+01
		typical design rate	2E-03	8E-02

**** - No filtering (filters assumed to blow-out)

CORE MELT
BYPASS

Step 4

Record from event tree (Figure C-24 on page C-73):

Acute whole body dose @ 1 mile ____________
Thyroid dose @ 1 mile ____________

Step 5

Adjust for reactor size:

Whole body @ 1 mile________ x ________ plant MW(e) / 1000 MW(e) = ________

Thyroid @ 1 mile ________ x ________ plant MW(e) / 1000 MW(e) = ________

Step 6

If the plant has been shutdown 30 days or more, use the factors on page C-91 to adjust doses.

Step 7

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ______ x 0.40 = ______ whole body @ 2 mile
Whole body @ 1 mile ______ x 0.09 = ______ whole body @ 5 mile
Whole body @ 1 mile ______ x 0.03 = ______ whole body @10 mile

Thyroid @ 1 mile ______ x 0.40 = ______ thyroid @ 2 mile
Thyroid @ 1 mile ______ x 0.09 = ______ thyroid @ 5 mile
Thyroid @ 1 mile ______ x 0.03 = ______ thyroid @10 mile

Step 8

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ____________
- Distance to which early health effects are possible (WB dose > 50 rem) ____________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ____________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

Figure C-24

PWR AND BWR CONTAINMENT BYPASS RELEASE EVENT TREE FOR A MELT RELEASE FROM THE CORE

core conditions	release conditions	release rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
MELT RELEASE FROM CORE	not filtered	100%/hour	1E+03	1E+05
		100%/day	4E+01	4E+03
		typical design rate	1E-01	1E+01
	filtered	100%/hour	****	*****
		100%/day	2E+00	8E+01
		typical design rate	4E-03	2E-01

**** - No filtering (filters assumed to blow-out)

PAGE INTENTIONALLY LEFT BLANK

PWR & BWR SPENT FUEL POOL

Step 1

Select the appropriate conditions:

CAUTION:

> Zircaloy fire is possible in high density racks in any fuel discharged in the past 2 years and in a low density rack within 2 months of discharge. A gap release is possible after considerable prolong uncovery of the fuel in the pool.

NOTE:

> - A batch is the 1/3 of a core removed during refueling.
> - Holdup time is the time from release from the fuel to release to the environment.
> - A general description of the assumptions used in the event trees are found on page C-77.

Spent fuel condition:
- Gap release from one 3 month old batch ____________
- Gap release from 15 batches ____________
- Zircaloy fire in one 3 month old batch ____________

Release through filters:
- Yes ____________
- No ____________

Hold up time before release
- 1 hour ____________
- 2-12 hours ____________

Leak rate
- 100%/hour ____________
- 100%/day ____________

Step 2

Use event trees to estimate dose.

Zircaloy fire	1 batch	. . . .	Figure C-25, page C-78
Gap release	1 batch	. . . .	Figure C-26, page C-79
Gap release	15 batches	. . . .	Figure C-27, page C-80

Record from event trees:

Whole body dose @ 1 mile ______________________

Thyroid dose @ 1 mile ______________________

Step 4

Adjust for number of batches (reloads) in pool:

Step 5

Calculate doses at 2, 5 and 10 miles.

Whole body @ 1 mile ________ x 0.40 = ________ whole body @ 2 mile
Whole body @ 1 mile ________ x 0.09 = ________ whole body @ 5 mile
Whole body @ 1 mile ________ x 0.03 = ________ whole body @10 mile

Thyroid @ 1 mile ________ x 0.40 = ________ thyroid @ 2 mile
Thyroid @ 1 mile ________ x 0.09 = ________ thyroid @ 5 mile
Thyroid @ 1 mile ________ x 0.03 = ________ thyroid @10 mile

Step 6

Because of great uncertainty, do not use dose numbers in presenting results. Use the possible consequences as determined below:

- Distance to which early deaths are possible (WB dose > 150 rem) ________________
- Distance to which early health effects are possible (WB dose > 50 rem) ________________
- Distance to which PAG may be exceeded (WB dose > 1 rem or thyroid dose > 5 rem) ________________

Combine this assessment with the general description (following Step 3), input information (Step 2) and a markup of figure showing assumed release pathway(s).

BASIC DESCRIPTION

There are two basic cases. The Zircaloy fire is assumed to result in a release fraction similar to those for a core melt for the last discharged batch (3 month old).

The gap case assumes that the pool is drained and the cladding ruptures releasing the gaseous fission products.

The inventory for a Zircaloy fire is assumed to be 1 batch (1/3 core) of 3 month old fuel. The inventory for a gap release is for 1 batch of 3 month old fuel or 15 batches of 1 to 15 year old fuel.

The releases have been reduced to account for plateout and other reduction mechanisms.

The dose and reduction assumptions are the same as for the reactor cases.

Figure C-25

SPENT FUEL POOL RELEASE FOR A ZIRCALOY FIRE TO 1 BATCH OF FUEL

pool condition	filter conditions	holdup time no spray	building leak rate	whole-body dose (rem) at 1 mile	thyroid dose (rem) at 1 mile
ZIRCALOY FIRE IN ONE 3 MONTH OLD BATCH	not filtered	1 hour	100%/hour	6.00E+01	5.00E+01
			100%/day	2.40E+00	2.00E+00
		2-12 hours	100%/hour	2.40E+01	2.00E+01
			100%/day	9.60E-01	8.00E-01
	filtered	1 hour	100%/hour	6.00E-01	5.00E-01
			100%/day	2.40E-02	2.00E-02
		2-12 hours	100%/hour	2.40E-01	2.00E-01
			100%/day	9.60E-03	8.00E-03

Figure C-26

SPENT FUEL POOL RELEASE FROM A GAP RELEASE FROM 1 BATCH

GAP RELEASE FROM ONE 3 MONTH BATCH

pool conditions	holdup time	leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
sprays off	1 hour	100%/hour	2.00E+00	2.00E+00	****	****
		100%/day	8.00E-02	8.00E-02	8.00E-04	8.00E-04
	2-12 hours	100%/hour	8.00E-01	8.00E-01	****	****
		100%/day	3.20E-02	3.20E-02	3.20E-04	3.20E-04
sprays on	1 hour	100%/hour	1.00E+00	1.00E+00	****	****
		100%/day	4.00E-02	4.00E-02	4.00E-04	4.00E-04
	2-12 hours	100%/hour	6.00E-01	6.00E-01	****	****
		100%/day	2.40E-02	2.40E-02	2.40E-04	2.40E-04

**** - No filtering assumed at this leak rate.

Figure C-27

SPENT FUEL POOL RELEASE FOR A GAP RELEASE FROM TOTAL FUEL POOL (15 BATCHES)

	pool conditions	holdup time	leak rate	whole-body dose (rem) at 1 mile not filtered	thyroid dose (rem) at 1 mile not filtered	whole-body dose (rem) at 1 mile filtered	thyroid dose (rem) at 1 mile filtered
GAP RELEASE FROM 15 BATCHES	sprays off	1 hour	100%/hour	1.00E+01	1.00E+01	****	****
			100%/day	4.00E-01	4.00E-01	4.00E-03	4.00E-03
		2-12 hours	100%/hour	4.00E+00	4.00E+00	****	****
			100%/day	1.60E-01	1.60E-01	1.60E-03	1.60E-03
	sprays on	1 hour	100%/hour	5.00E+00	5.00E+00	****	****
			100%/day	2.00E-01	2.00E-01	2.00E-03	2.00E-03
		2-12 hours	100%/hour	3.00E+00	3.00E+00	****	****
			100%/day	1.20E-01	1.20E-01	1.20E-03	1.20E-03

**** - No filtering assumed at this leak rate.

GENERAL DESCRIPTION OF ASSUMPTIONS

BASIC METHOD

To make a first approximation of a severe accident source term, follow the event tree method below.

(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, gap, grain boundary or melt release.

(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 amount of the available fission product inventory actually released to the environment.

(5) Estimate the dose at 1 mi.

The event trees estimate source terms by:

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

for radionuclide i and n reduction mechanisms

where

FPI_i = element i core or coolant inventory

[assume a 1000 MWe core] - See Table 1

$$CRF_i = \frac{\text{element i released from core (based on core damage state)(Table 2)}}{\text{element i inventory in core}}$$

$$RDF_i = \frac{\text{element i after reduction mechanisms available for release (Table 3)}}{\text{element i before reduction mechanisms available for release}}$$

$$EF_i = \frac{\text{element i released (Table 4)}}{\text{element i available for release}}$$

The steam generator tube rupture case was somewhat different in that the starting point was an assumed coolant concentration (Table 5 or 6) that assumes all the material released from the core is contained in the coolant.

DOSE - The doses in the trees were calculated by taking the doses at 1 mile calculated by MESORAD for the stated assumptions in Table 7, and multiplying these doses by the reduction factor (RDF x EF) that accounts for the appropriate mitigating conditions (hold-up time, sprays, filters, and containment leak rate).

Table C-1

Fission product inventories (Ci/MWe)

Fission product	Inventory (Ci/MWe)
Kr-85	560
Kr-85m	24,000
Kr-87	47,000
Kr-88	68,000
Sr-89	94,000
Sr-90	3,700
Sr-91	110,000
Y-91	120,000
Mo-99	160,000
Ru-103	110,000
Ru-106	25,000
Te-129m	5,300
Te-131m	13,000
Te-132	120,000
Sb-127	6,100
Sb-129	33,000
I-131	85,000
I-132	120,000
I-133	170,000
I-134	190,000
I-135	150,000
Xe-131m	1,000
Xe-133	170,000
Xe-133m	6,000
Xe-135	34,000
Xe-138	170,000
Cs-134	7,500
Cs-136	3,000
Cs-137	4,700
Ba-140	160,000
La-140	160,000
Ce-144	85,000
Np-239	1.64×10^6

Source: WASH-1400

For end of cycle core, 1/2 hour after shutdown.

Table C-2

CORE RELEASE FRACTION ASSUMPTIONS (1-HOUR RELEASE)

Core Condition	Fuel Cladding Temperature	Fission Product	Assumed Release Fraction From Fuel
Fuel Pin Cladding Intact - Normal Leakage	600°F	Normal[4]	Tables C-5, C-6
Gap Release (Cladding Failure)	1300°-2100°F	Xe, Kr	0.03
		I	0.02
		Cs	0.05
		Te, Sb	1×10^{-4}
Grain Boundary Release	>3000°F	Xe, Kr	0.5
		I, Cs	0.5
		Te	0.1
		Sb	0.02
		Ba	0.01
		Mo	0.01
		Sr	1×10^{-3}
		Ru	1×10^{-4}
Core Melt (In-Vessel)[1,2]	>4500°F	Xe, Kr	1.0
		Cs	1.0
		I	1.0
		Sb	0.02
		Te[3]	0.3
		Ba	0.2
		Sr[3]	0.07
		Mo	0.1
		Ru	7×10^{-3}
		La	1×10^{-4}
		Y	1×10^{-4}
		Ce	1×10^{-4}
		Np	1×10^{-4}

[1] Based on Tables 4.8 and 4.9 of NUREG-0956

[2] For La, Y, Np, and Ce, the Sr release fraction was used, based on BMI-2104, Vol. VI, page 6-24 grouping, Battelle Columbus Laboratories, 1984

[3] Ex-vessel (melt-through) melt release fractions may be much larger (0.4 to 0.8).

[4] ANSI/ANS-18.1-1984

Table C-3

SUMMARY OF PARTICULATE/AEROSOL GASEOUS (UNLESS NOTED) REDUCTION MECHANISMS

Release Mechanism	Reduction Factor
Standby Gas Treatment System Filters:	
Dry-low pressure flow	0.01
Wet-high pressure flow (blowout)	1.0
Other Filters:	
Dry-low pressure flow	0.01
Wet-high pressure flow (blowout)	1.0
Suppression Pool Scrubbing:	
Slow steady flow (decay heat)	
Pool subcooled	0.01
Pool saturated	0.05
Pool bypass	1.00
Removal of Suspended Aerosols and Particulates:	
Natural processes (no sprays)	
<1 hour holdup time	0.20*
2- to 12-hour holdup time	0.04
24-hour holdup time	0.01
Sprays on	
<1 hour holdup time	0.03
2- to 12-hour holdup time	0.02
24-hour holdup time	0.002
Ice Condenser:	
One pass through condenser (no recircultion)	0.5
Continual recirculation through condenser (1 hour or more)	0.25
Ice bed exhausted before core damage	1.0
Primary System Retention (Plateout):	
Bypass accidents only	0.20*
SG Partitioning (Liquid Release from RCS):	
Partitioned	0.02
Not partitioned	0.50

SG = steam generator
RCS = reactor coolant system
Scurce: NUREG-1228
* = adjusted based on NUREG-1150

Table C-4

Release pathway	Escape fraction*
Primary containment failure/leakage	
Typical design leakage:	
PWR - large dry (0.1%/day)	4×10^{-5}
PWR - subatmospheric (0.1%/day)	4×10^{-5}
PWR - ice condenser (0.25%/day)	1×10^{-4}
BWRs (0.5%/day)	2×10^{-4}
Failure to isolate (100%/day):	
Failure of isolation valve seal	0.04
Castastrophic failures:	
1-hr puff release	1
Steam generator tube rupture	
1 tube at full pressure (coolant leak)	0.35
1 tube at low-pressure single charging pump flow (coolant leak)	0.03

*Fraction of containment volume or primary system coolant inventory released in 1 hour.

Table 5 PWR baseline coolant concentrations for various levels of core damage

Nuclide	Core inventory (1000 MWe) (Ci)	Normal concentration (µCi/g)	Gap release fraction	Gap coolant concentration (µCi/g)	Melt release fraction	Melt coolant concentration (µCi/g)	TMI concentration +48 hrs* (µCi/g)
Kr-85	$5.60x10^{5}$	$4.30x10^{-1}$	[illegible]$.00x10^{-2}$	$6.72x10^{1}$	1.00	$2.24x10^{3}$	-
Kr-85m	$2.40x10^{7}$	$1.60x10^{-1}$	$3.00x10^{-2}$	$2.88x10^{3}$	1.00	$9.60x10^{4}$	-
Xe-133	$1.70x10^{8}$	2.60	$3.00x10^{-2}$	$2.04x10^{4}$	1.00	$6.80x10^{5}$	-
Xe-135	$3.40x10^{7}$	$8.50x10^{-1}$	$3.00x10^{-2}$	$4.08x10^{3}$	1.00	$1.36x10^{5}$	-
I-131	$8.50x10^{7}$	$4.50x10^{-2}$	$2.00x10^{-2}$	$6.80x10^{3}$	1.00	$3.40x10^{5}$	$1.30x10^{4}$
I-133	$1.70x10^{8}$	$1.40x10^{-1}$	$2.00x10^{-2}$	$1.36x10^{4}$	1.00	$6.80x10^{5}$	$6.50x10^{3}$
I-135	$1.50x10^{8}$	$2.60x10^{-1}$	$2.00x10^{-2}$	$1.20x10^{4}$	1.00	$6.00x10^{5}$	-
Cs-134	$7.50x10^{6}$	$7.10x10^{-3}$	$5.00x10^{-2}$	$1.50x10^{3}$	1.00	$3.00x10^{4}$	$6.30x10^{1}$
Cs-137	[illegible]$.70x10^{6}$	$9.40x10^{-3}$	$5.00x10^{-2}$	$9.40x10^{2}$	1.00	$1.88x10^{4}$	$2.80x10^{2}$
Sr-90	[illegible]$0x10^{6}$	$1.20x10^{-5}$	0.00	$1.2x10^{-5}$	$7.00x10^{-2}$	$1.04x10^{3}$	5.30
Np-239	$1.60x10^{9}$	$2.20x10^{-3}$	0.00	$2.2x10^{-3}$	$1.00x10^{-4}$	$6.40x10^{2}$	-

Assumptions: WASH-1400 inventory.
ANSI/ANS 18.1-1984 normal coolant concentrations.
$2.5x10^{5}$ kg primary coolant inventory.

*Concentrations of sample taken 3/29/79, 1600 counted 3/30/79 (>48 hours after accident).

Table C-6

BWR COOLANT CONCENTRATIONS

Nuclide	Core inventory (1000 MWe) (Ci)	Normal concentration (μCi/g)	Gap release fraction	Gap coolant concentration (μCi/g)	Melt release fraction	Melt coolant concentration (μCi/g)
Kr-85	5.60×10^{5}	0.00	3.00×10^{-2}	9.88×10^{1}	1.00	3.29×10^{3}
Kr-85m	2.40×10^{7}	0.00	3.00×10^{-2}	4.24×10^{3}	1.00	1.41×10^{5}
Xe-133	1.70×10^{8}	0.00	3.00×10^{-2}	3.00×10^{4}	1.00	1.00×10^{6}
Xe-135	3.40×10^{7}	0.00	3.00×10^{-2}	6.00×10^{3}	1.00	2.00×10^{5}
I-131	8.50×10^{7}	2.20×10^{-3}	2.00×10^{-2}	1.00×10^{4}	1.00	5.00×10^{5}
I-133	1.70×10^{8}	1.50×10^{-2}	2.00×10^{-2}	2.00×10^{4}	1.00	1.00×10^{6}
I-135	1.50×10^{8}	2.20×10^{-2}	2.00×10^{-2}	1.76×10^{4}	1.00	8.82×10^{5}
Cs-134	7.50×10^{6}	3.00×10^{-5}	5.00×10^{-2}	2.21×10^{3}	1.00	4.41×10^{4}
Cs-137	4.70×10^{6}	8.00×10^{-5}	5.00×10^{-2}	1.38×10^{3}	1.00	2.76×10^{4}
Sr-90	3.70×10^{6}	7.00×10^{-6}	0.00	0.00	7.00×10^{-2}	1.52×10^{3}
Np-239	1.60×10^{9}	8.00×10^{-3}	0.00	0.00	1.00×10^{-4}	9.41×10^{2}

Assumptions: WASH-1400 inventory.
ANSI/ANS-18.1-1984 normal coolant concentrations.
1.7×10^{5} kg of coolant.

Table C-7

1-mile doses for release of various core and coolant inventories[1]

Type of release	Total whole-body dose, rem (hours after shutdown) [2, 3]					Thyroid inhalation dose, rem (hours after shutdown) [4]	
	0 hr	1 hr	6 hr	12 hr	24 hr	0 hr	6 hr
100% gap noble gases	$3x10^{0}$	$2.4x10^{0}$ [5]	$6x10^{-1}$ [5]	$3x10^{-1}$	$2.1x10^{-1}$ [5]	NC	NC
100% grain boundary noble gases	$5x10^{1}$	$4x10^{1}$ [5]	$1x10^{1}$ [5]	$5x10^{0}$	$3.5x10^{0}$ [5]	NC	NC
100% melt noble gases	$1x10^{2}$	$8x10^{1}$ [5]	$2x10^{1}$ [5]	$1x10^{1}$	$7x10^{0}$ [5]	NC	NC
100% gap particulates and aerosols	$2x10^{2}$	NC	$1x10^{2}$ [5]	NC	NC	$1.7x10^{4}$	$1x10^{4}$ [5]
100% grain boundary particulates and aerosols	$3x10^{3}$	NC	$2x10^{3}$ [5]	NC	NC	$3x10^{5}$	$2x10^{5}$ [5]
100% melt particulates and aerosols	$8x10^{3}$	NC	$5x10^{3}$ [5]	NC	NC	$6x10^{5}$	$5x10^{5}$ [5]
100% coolant normal	$1x10^{-2}$ [5]	NC	NC	NC	NC	$1x10^{-1}$ [5]	$9x10^{-2}$
100% coolant 100X spike[6]	$1x10^{0}$ [5]	NC	NC	NC	NC	$1x10^{1}$ [5]	NC

(1) 1-mile dose, 4 mph, D stability, ground level release, NRC MESORAD code.
(2) Hours after shutdown release is projected to start.
(3) Includes cloud shine, inhalation, and 3 hours of ground shine.
(4) Inhalation and adult thyroid only.
(5) Assumed in reactor event trees.
- For 0.5-, 6-, and 24-hour containment holdup cases, 1-, 6- and 24-hour noble gas factors were used, respectively.
- For SGTR high-pressure and low-pressure cases, 1-hour and 6-hour noble gas factors were used, respectively.
- For bypass 100%/hour and other cases, the 1-hour and 6-hour noble gas factors were used, respectively.

(6) Spike of all non-noble fission products.

NOTE: NC = not calculated.

Table C-8

Key to release pathway references on Figures 1, 5, 13, 14, and 15

Pathway		Description
		PWR - Figures 1 and 5
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 sq ft)
	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
		BWR - Figures 13, 14, and 15
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)
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)

SHUTDOWN REACTOR CORRECTION FACTORS

Adjust the whole body and thyroid dose by multiplying by the factors below for a reactor that has been shutdown for 30 or greater than 120 days at the time of the start of the release:

WHOLE BODY	SHUTDOWN TIME	
	30 DAYS	≥120 DAYS
Gap	.1	.1
Grain Boundary	.1	.1
Melt	.1	.05

THYROID	SHUTDOWN TIME	
	30 DAYS	≥120 DAYS
Gap	.05	.005
Grain Boundary	.1	.003
Melt	.5	.002

Use of RASCAL

Section D

D Use of RASCAL

USE OF RASCAL

Objective

Provide guidance on the use of the RASCAL model.

Guidance

Step 1

Complete RASCAL worksheet (page D-3).
NOTE:

- Use Whole Body (WB) Cloud Shine dose and Thyroid dose for comparison to EPA PAGs.
- Use total dose-acute bone (30 day) and thyroid dose for insights on early health for reactor accidents. For other accidents, lung dose may be most important.
- Make runs that will provide upper and lower bounds for the possible range of plant accident conditions (source terms), release, or meteorological conditions.
- For assessment of licensee release projection - i.e., field measurement results vs. release estimates - assume:
  * A 1-hour release at the projected rate.
  * Set "end of calculation time" to assure total plume passage at point of the field measurement.
  * Compare field measurement with cloud shine dose.

Step 2

Run the model and

1. Print a copy of the results.

2. Print a copy of "Input and Assumptions" (ensure the results and assumptions have the same time).

Step 3

Present the results:

1. Compare results against a topographical map to identify major terrain effects that were not considered.

2. Assure the great uncertainties are understood. Protective actions should never be limited to the areas shown in the model as requiring them. In general, actions should be recommended in all directions.

3. Discuss result in terms of the possibility of EPA PAG or early health effect thresholds being exceeded close or far from the plant. Specific dose numbers should be avoided.

END

PAGE INTENTIONALLY LEFT BLANK

RASCAL WORKSHEET

(Complete Sections I, II, and III)

Analysts ____________

Time/Date ____________

Title/Description ____

____

____

I. SOURCE TERM (USE METHOD A, B, OR C)

A. Isotopic Release

Release Height ____________ Start of Release ________
End of Release ________

Isotope	Rate (Ci/s or Bq/s)
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________
____________	____________

B. Gross Release Rate

Release Height ____________ Start of Release ________
End of Release ________

Gross Release Rate ____________
% Iodine ____________
% Noble Gas ____________

C. Based on Reactor Conditions (PWR or BWR)

Complete/worksheet for appropriate release pathway from "Reactor Consequence Assessment Based on Plant Conditions," Section C, and RASCAL will estimate source term.

II. TRANSPORT/DISPERSION

- End of calculation time ________ (Time of Dose Snap-Shot)
- Calculation Radius (10 or 25) (normally 10 mile) ________
- For up to 4 time steps provide:

	Step 1	Step 2	Step 3	Step 4
Start Time of Following Conditions	______	______	______	______
Wind Direction	______	______	______	______
Wind Speed	______	______	______	______
Stability Class	______	______	______	______
Mixing Height	______	______	______	______
Precipitation	______	______	______	______

III. DOSE/CONSEQUENCE CALCULATIONS REQUIRED

________ - Calculate whole body cloud shine dose and compare to EPA PAG

________ - Calculate thyroid dose and compare to EPA thyroid PAG

________ - For reactor accidents, calculate total bone-acute (30 day) and thyroid dose and compare to threshold of early health effects

________ - Calculate whole body cloud shine and compare to field monitoring external dose measurements

________ - For accidents where inhalation lung dose may dominate (e.g, Pu), calculate acute lung dose and compare to threshold of early health effects.

UF_6 Release Assessment

Section E

E UF$_6$ Release Assessment

UF_6 RELEASE ASSESSMENT

Objective

To provide guidance for the assessment of protective actions and consequences of UF_6 releases.

Guidance

CAUTION

> The chemical toxicity of a UF_6 release far outweighs the radiological hazard it may present. UF_6 is primarily a threat when heated above 200° F, because it undergoes a chemical reaction to form the two toxic chemicals, UO_2F_2 and hydrogen fluoride (HF). The primary action of UO_2F_2 is as a heavy metal poison toxic to the kidneys. THe HF fraction is very corrosive to skin and tissue and toxic when inhaled. Since it can be seen, smelled, and felt by an exposed individual, it is possible to avoid exposure to UF_6 by his own awareness. However, toxic levels can be reached in minutes, so immediate protective actions should be taken when a release is detected. Any individual who inhales a release should be observed for effects following the accident.

NOTE

> Lethal toxic chemical doses of natural and low-enriched uranium would not produce enough radiation to exceed the EPA PAGs (1 rem effective dose equivalent). High-enriched uranium, however, could pose radiological hazards at lower concentrations (see Table E-6).

Step 1

Assess the need for immediate protective action:

A. **RELEASE UNDERWAY**

IF IT CAN BE SMELLED immediately move away from the plume.

B. **POTENTIAL RELEASE**

Evacuate distance shown in Table E-1 (page E-2) if it can be completed before the plume arrives. Otherwise, recommend sheltering and any ad-hoc respiratory protection available.

Table E-1

EVACUATION DISTANCES
FROM NUREG-1140

LOW-ENRICHED UF6

Potential Release		Distance	
Tons	Kg	English	Metric
1/2	500	100 yd	100 m
5.5	5000	1/3 mi	500 m
14	12,500	1 mi	1.5 Km

HIGH-ENRICHED UF6

Potential Release		Distance	
Lbs	Kg	English	Metric
22	10	100 yd	100 m
220	100	1/3 mi	500 m

Step 2

Calculate the intake of uranium and concentration of hydrogen fluoride by one of the following methods, compare the results with the health effects levels indicated in Table E-4 (page E-4) and recommend appropriate action:

A. **RAPID ESTIMATION**

Multiply factors in Tables E-2 (page E-3) and E-3 (page E-3) by the total tons of heated UF_6 inventory available for release.

B. **MANUAL CALCULATION**

Atmospheric release, use Worksheet E-1, page E-6.
Enclosed release, use Worksheet E-2, page E-7.

Step 3

Calculate the CEDE dose if high-enriched UF_6 is involved and compare CEDE dose with the EPA PAGs in Section I (page I-4).

A. For rapid estimation of radiological dose at 1/4 mile use Table E-6 (page E-5).

B. For detailed calculation use the Section H method for U^{235}. Assume that the specific activity of U^{235} is:

$$6.5 \times 10^{-5} \text{ Ci/gm} \quad \text{(per 10 CFR 20)}$$

END

Table E-2

INTAKE OF URANIUM
From Release From Heated Tank
Per Ton of UF_6 Inventory in Tank
(mg)

Distance Meters	Distance Miles	Average Dispersion[1]	Very Poor Dispersion[2]
200		3.71	0.43
300		4.16	3.26
400	1/4	3.62	6.87
500		2.80	7.24
600		2.35	7.78
700		1.99	7.78
800	1/2	1.72	7.51
900		1.45	7.05
1000		1.18	6.51
1100		1.09	5.97
1200	3/4	1.08	5.52
1800	1	0.57	2.94

1 D Stability, elevated, wind speed 4.5 m/s.
2 F Stability, elevated, wind speed 1 m/s.

Table E-3

HYDROGEN FLUORIDE (HF) CONCENTRATION
From Release From Heated Tank
Per Ton of UF_6 Inventory in Tank
(mg/m^3)

Distance Meters	Distance Miles	Average Dispersion[1]	Very Poor Dispersion[2]
200		5.47	0.63
300		6.13	4.80
400	1/4	5.33	10.13
500		4.13	10.67
600		3.47	11.47
700		2.93	11.47
800	1/2	2.53	11.07
900		2.13	10.40
1000		1.73	9.60
1100		1.60	8.80
1200	3/4	1.47	8.13
1800	1	1.39	5.61

1 D Stability, elevated, wind speed 4.5 m/s.
2 F Stability, elevated, wind speed 1 m/s.

Source: NUREG-1140

Table E-4

HEALTH EFFECTS
FOR PRODUCTS OF UF_6 RELEASE

URANIUM INTAKE		CONCENTRATION HYDROGEN FLUORIDE	
Intake (mg)	Health Effect	CONC (mg/m^3)	Health Effect
<4.5	None	2.5	Odor, no effects
8.6	Transient renal damage	13	Irritation (15 minute)
		100	Unbearable for 1 min
45	Permanent renal damage	3,500	Lethal (15 minute)
243	Lethal Dose-50%		

30 ppm is immediately dangerous to life or health (NIOSH)

Table E-5

DISPERSION (X/Q)
Buoyant - No Building Wake

Distance		'D' Stability ($\bar{u}$=4.5 m/s)	'F' Stability ($\bar{u}$=1 m/s)
Meters	Miles	s/m^3	s/m^3
200		4.2E-05	5.0E-06
300		4.6E-05	3.6E-05
400	1/4	4.0E-05	6.8E-05
500		3.1E-05	8.6E-05
600		2.6E-05	9.9E-05
700		2.2E-05	8.6E-05
800	1/2	1.9E-05	8.4E-05
900		1.6E-05	8.0E-05
1,000		1.3E-05	7.2E-05
1,100		1.2E-05	6.7E-05
1,200	3/4	1.2E-05	6.2E-05
1,800	1	1.2E-05	5.7E-05
3,600	2	1.1E-06	4.8E-06

Table E-6

RADIOLOGICAL DOSE FOR HIGH-ENRICHED (93%) URANIUM
1/4 MILE, 'D' STABILITY, WIND SPEED 2 M/S

Gram	Pounds	Curies	Total CEDE (rem)
1,000	2	.065	1.2
2,000	4	.130	2.4
3,000	7	.195	3.7
4,000	9	.260	4.9
5,000	11	.325	6.1
6,000	13	.390	7.4
7,000	15	.455	8.7
8,000	18	.520	9.9
9,000	20	.585	11.1
10,000	22	.650	12.4
20,000	44	1.30	24.7
30,000	66	1.95	37.1
40,000	88	2.60	49.4
50,000	110	3.25	61.8
60,000	132	3.90	74.1
70,000	154	4.55	86.5
80,000	176	5.20	98.8
90,000	198	5.85	111.2
100,000	220	6.50	123.5

ASSUMPTION

Formula from 10 CFR Part 20 for Uranium:

$$SpA = (0.4 + 0.38E + 0.0034E^2) \times 10^{-6} \text{ Ci/g}$$

(where E = weight percent)

For 93% weight percent U^{235}:

$$SpA = (0.4 + 0.38(93) + 0.0034(8649) \times 10^{-6} \text{ Ci/g}$$

$$= 6.5 \times 10^{-5} \text{ Ci/g}$$

ATMOSPHERIC
UF_6 RELEASE
WORKSHEET E-1

CAUTION

Use the X/Q in Table 4 of this section. Do not use the Xu/Q in Section H.

RELEASE TO ATMOSPHERE

URANIUM INTAKE

(Quan UF6 Released) (mg) × (Chem Reac Ratio × Breathing Rate) (m3/s) × ⑤ (X/Q) (s/m3)

( ) × 1.65×10^{-4} × ( ) = mg

HF CONCENTRATION

(Quan UF6 Released) (mg) × ⑤ (X/Q) (s/m3) × (Chem. Reac. Ratio)

$$\frac{(\quad) \times (\quad) \times .20}{(900 \text{ seconds})} = \text{mg/m3}$$

(900 seconds)
(* 15 minutes)
(Release Duration)

(* Table E-4 assumed exposure duration)

⑤ Refers to Table No. for data

ENCLOSED
UF_6 RELEASE
WORKSHEET E-2

ROOM/BUILDING VOLUME

(Length) (m) × (Width) (m) × (Height) (m)

( ) × ( ) × ( ) = m3

CONVERSION FACTORS

ft x 0.305 = meters
cu ft x 0.028 = cu meter

URANIUM INTAKE

$$\frac{(\text{Quan UF6 Released})(\text{mg}) \times \left(\frac{\text{Chem. Reac. Rate} \times}{\text{Breathing Rate}}\right)(\text{m3/s}) \times (\text{Duration})(\text{seconds})}{(\text{Volume})(\text{m}^3)}$$

$$\frac{(\quad\quad) \times 1.65 \times 10^{-4} \times (\quad\quad)}{(\quad\quad)} = \quad \text{mg}$$

HF CONCENTRATION

$$\frac{(\text{Quan UF6 Released})(\text{mg}) \times \left(\text{Chem. Reac. Rate}\right)}{(\text{Volume})(\text{m}^3)}$$

$$\frac{(\quad\quad) \times (.20)}{(\quad\quad)} = \quad \text{mg/m3}$$

ASSUMPTIONS

1. From a ruptured, heated 14-ton cylinder of UF6, 75% would escape and hydrolyze to form UO_2F_2 and Hydrogen Fluoride (HF). Of this hydrolyzed amount, 50% would form airborne UO_2F_2 and 20% would form airborne HF.

2. Due to heat from the energy released in forming chemical compounds, the plume would become buoyant at 20 meters. At 200-300 meters, plume centerline height would be 20 meters (i.e., a ground release beyond 20 meters is impractical so no building wake effect is considered.) Solid (unheated) UF_6 is not likely to pose an airborne threat.

3. Breathing rate for intake of Uranium is the adult breathing rate of 3.3×10^{-4} m^3/s per EPA guidance. (NUREG-1140 uses 2.66×10^{-4} m^3/s).

4. For HF the health effects indicated in Table E-3 above are for direct exposure at indicated concentration levels for a 15 minute duration.

5. Calculations in Tables E-2 and E-3 are for a 30-minute release in an open field with a 1 cm/sec deposition velocity.

6. Specific Activity for 93% enriched UF_6 for Table E-5 calculations
   a. Uranium Sp A = $(0.4 + 0.38E + 0.0034E^2) \times 10^{-6}$ Ci/gm
      where E = weight percent (Source: 10 CFR 20)
      Sp A (93% U^{235}) = 6.5×10^{-5} Ci/gm

   b. Table H-2 of Section H was used to calculate dose in rems found in Table E-6.

7. Protective action recommendations based on NUREG-1140 page 104.

SOURCE: NUREG 1140 (1/88)

Long Distance, Elevated Release and
Rain Dose Projections for Severe Reactor Accidents

Section F

F. Long Distance, Elevated Release and Rain Dose Projections for Severe Reactor Accidents

LONG DISTANCE, ELEVATED RELEASE AND RAIN DOSE PROJECTIONS FOR SEVERE REACTOR ACCIDENTS

Objective

Insights on the influence of distance, rain and release elevation on offsite consequences.

Guidance

CAUTION:

> These estimates are very crude. It is very difficult or possibly impossible to accurately project dose under these conditions.

Step 1

Determine release conditions.

NOTE:

> Use the 30 meter release height for ground level releases; for highly elevated releases, estimate release height by observation. The maximum release height should be less than 1/2 of the atmospheric mixing level.

Release Elevation:

- Ground level (30 m) ________________
- 200 m ________________
- 500 m ________________

Light Rain or No Rain ________________

Step 2

CAUTION:

> Doses are for acute whole body (bone) except where noted. Doses represent exposure to all of the cloud shine and inhalation from a 1 hour plume and approximately 20 hours of the resulting ground shine with no sheltering. The doses are for the center line of a plume, for constant wind direction and do not consider building wake. Actual maximum doses should be considerably lower than those shown.

Go to appropriate case based on Figure F-1, page F-3.

Step 3

NOTE:

- Total acute dose should be used for insights on early health effects (see page H-36).
- The dose curves assume average meteorological conditions (D stability, 4 mph) and 30 m release height cases should be within a factor of 10 for A to F stability.
- Curves are based on RASCAL 1.3b runs.

Adjust results for different release and exposure durations.

Only ground level (30 m) releases can be adjusted. Multiply the whole body cloud shine and inhalation 30 m release height components by the release duration. Multiply the ground shine by the release duration and exposure period .
20 hours

Step 4

Examine widths of plumes using page F-20.

END

Figure F-1

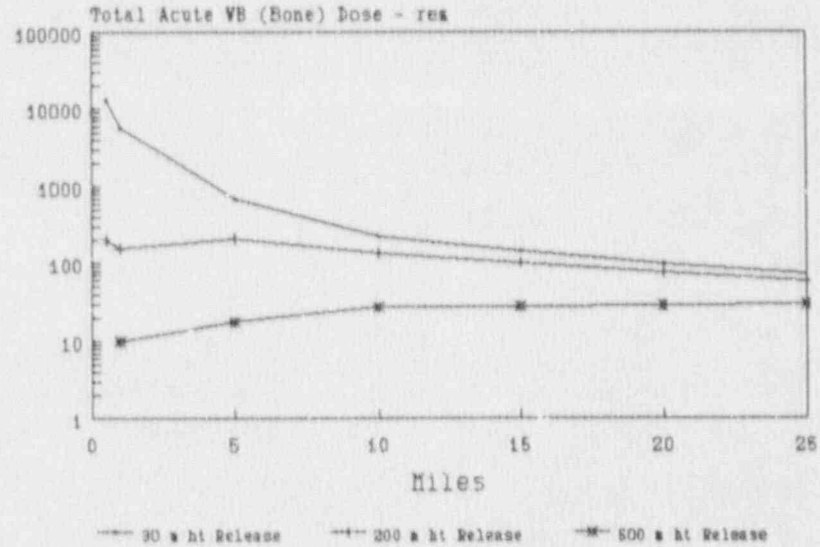
RELATIONSHIP OF ACCIDENT CONDITIONS TO DOSE CURVES

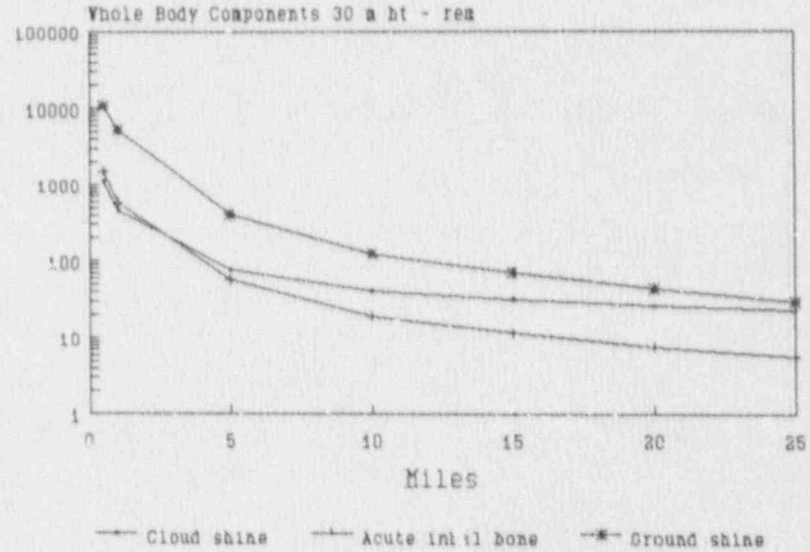
NOTE:

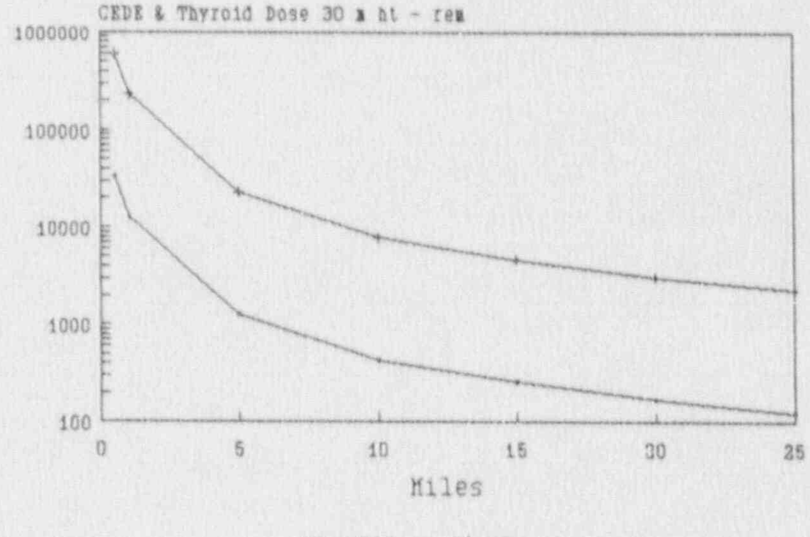
- Release is considered **mitigated** if it passes through a containment with sprays on, the suppression pool or filters.
- See B-4 for guidance on core damage assessment.

Core Condition	Containment Status	Mitigation Status	Page	Case
Melted (1 to 3 hrs after core is uncovered)	Early Total Failure/Bypass (100%/hr)	Not Mitigated	F-4	1
Melted (1 to 3 hrs after core is uncovered)	Early Total Failure/Bypass (100%/hr)	Mitigated	F-6	2
Melted (1 to 3 hrs after core is uncovered)	Late (2-12 Hours) Total Failure (100%/hr)		F-6	2
Melted (1 to 3 hrs after core is uncovered)	Major Leakage (100%/day)		F-8	3
Melted (1 to 3 hrs after core is uncovered)	Design Leakage		F-10	4
Gap Release from Core (1/2 to 3/4 hour after core is uncovered)	Early Total Failure/Bypass (100%/hr)	Not Mitigated	F-12	5
Gap Release from Core (1/2 to 3/4 hour after core is uncovered)	Early Total Failure/Bypass (100%/hr)	Mitigated	F-14	6
Gap Release from Core (1/2 to 3/4 hour after core is uncovered)	Late (2-12 Hours) Total Failure (100%/hr)		F-16	7
Gap Release from Core (1/2 to 3/4 hour after core is uncovered)	Major Leakage (100%/day)		F-18	8

CASE 1 - NO RAIN

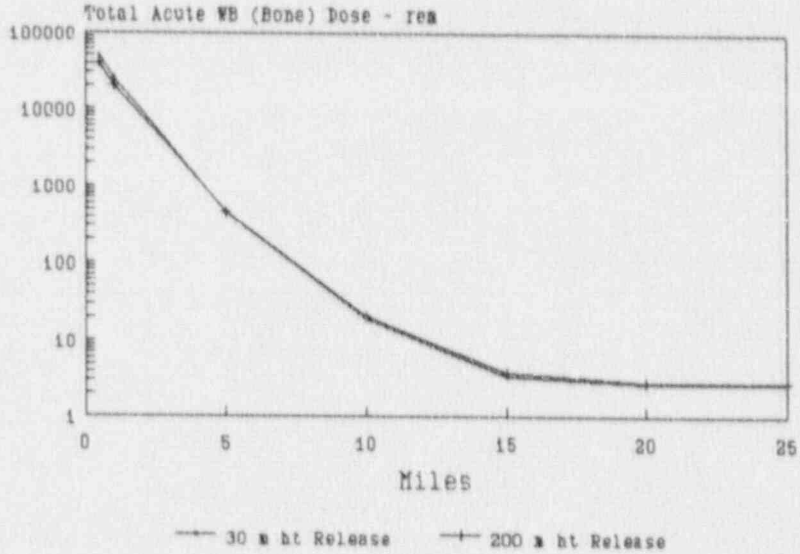


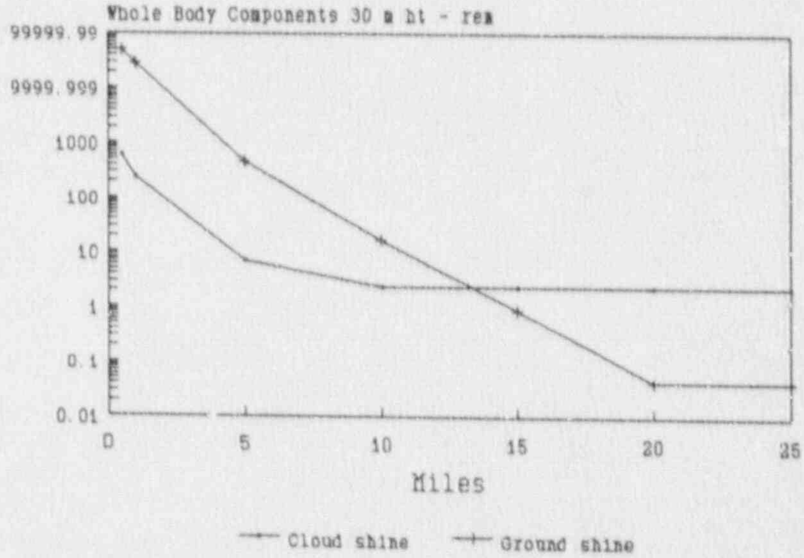


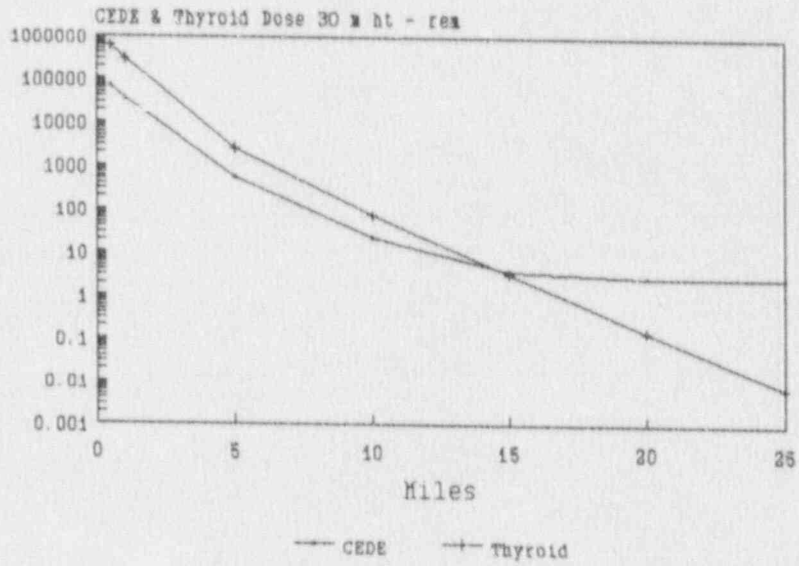


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 1 - RAIN

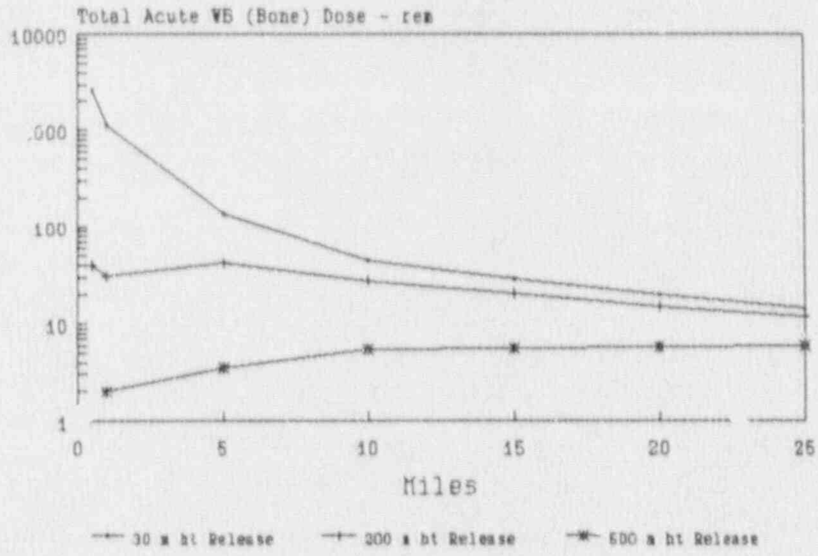


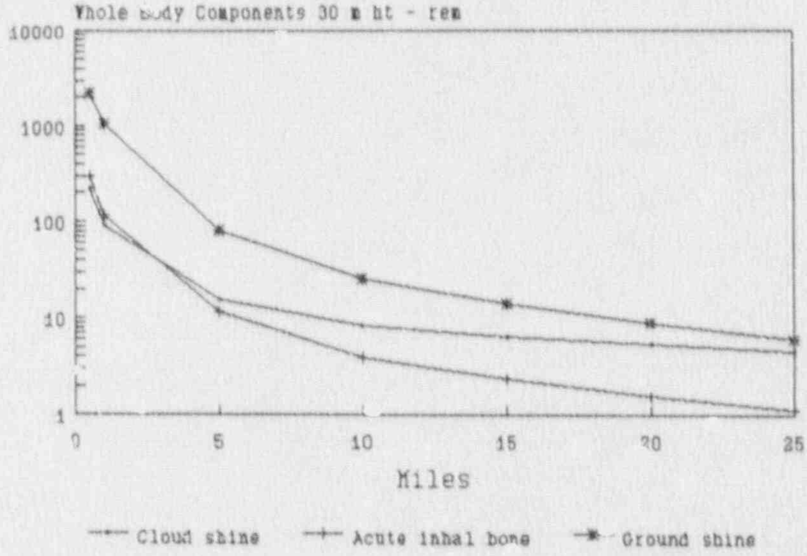


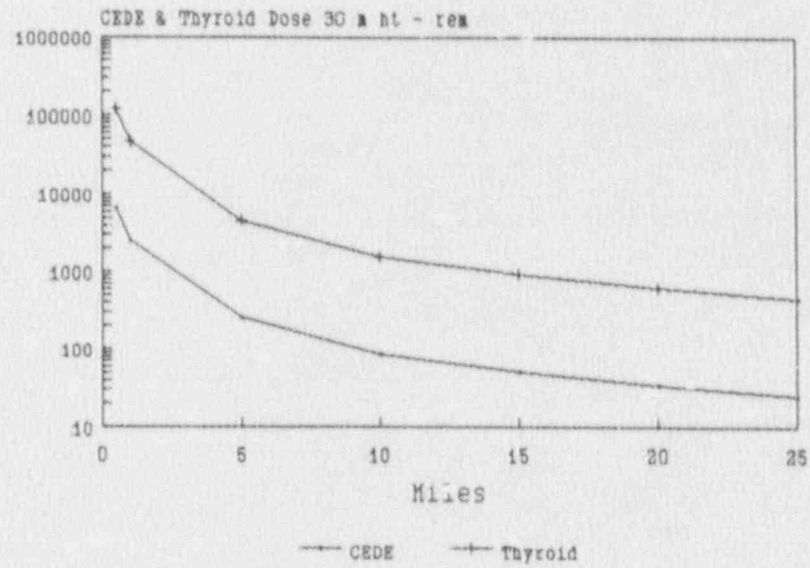


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 2 - NO RAIN

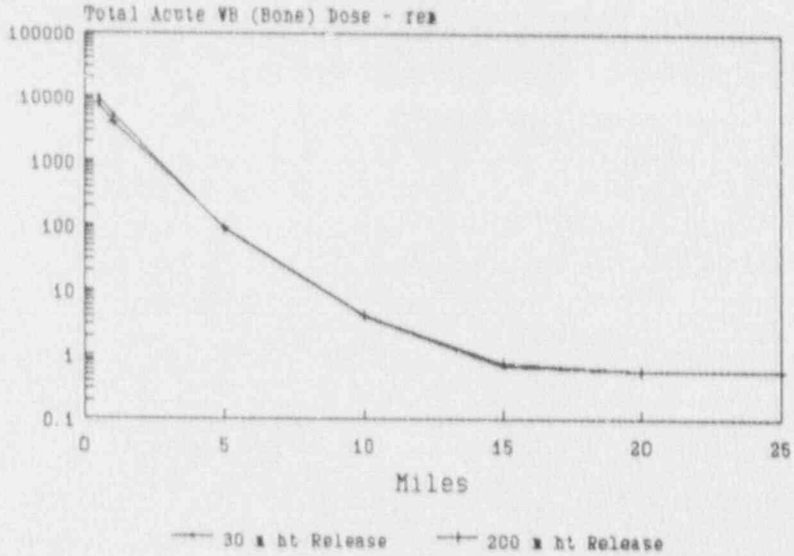


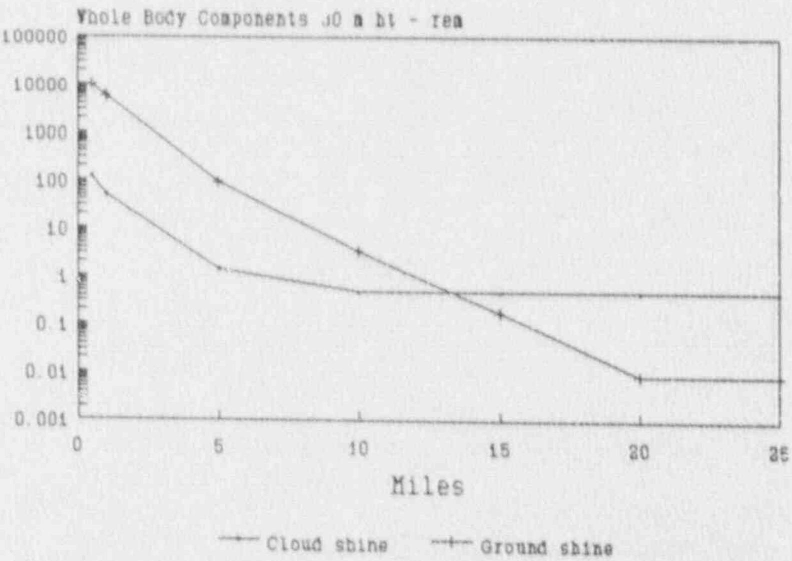


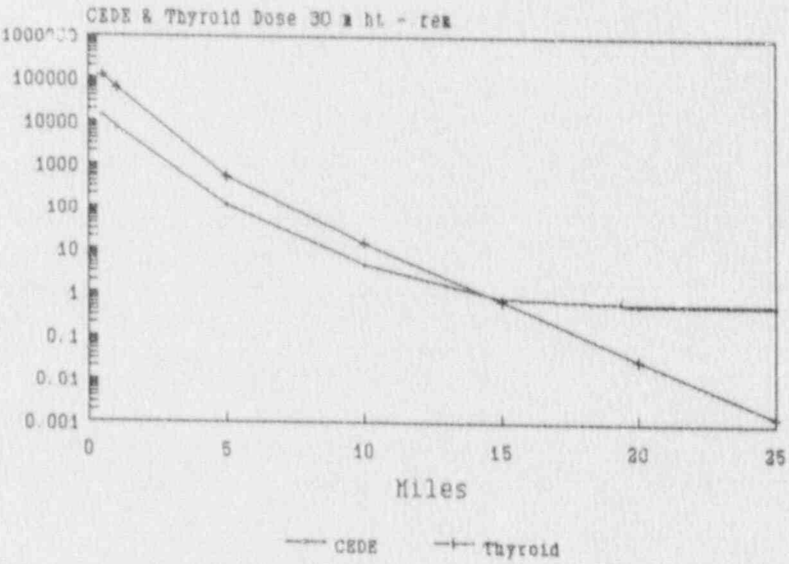


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 2 - RAIN

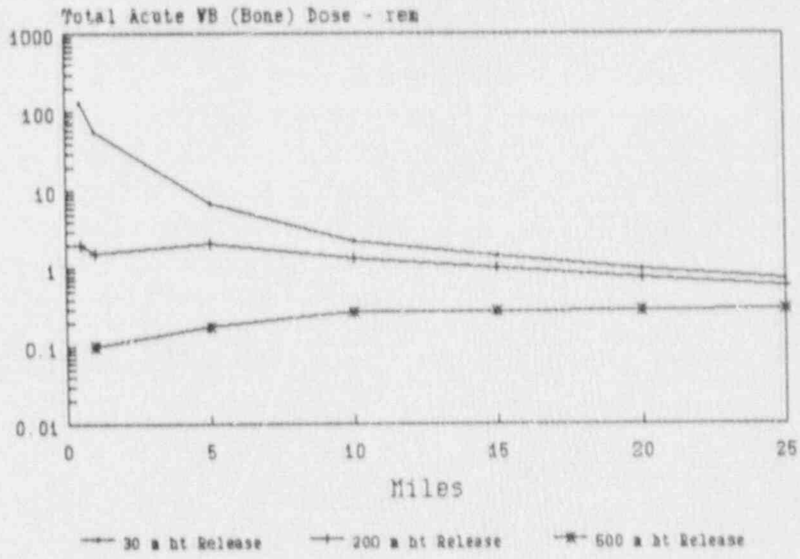


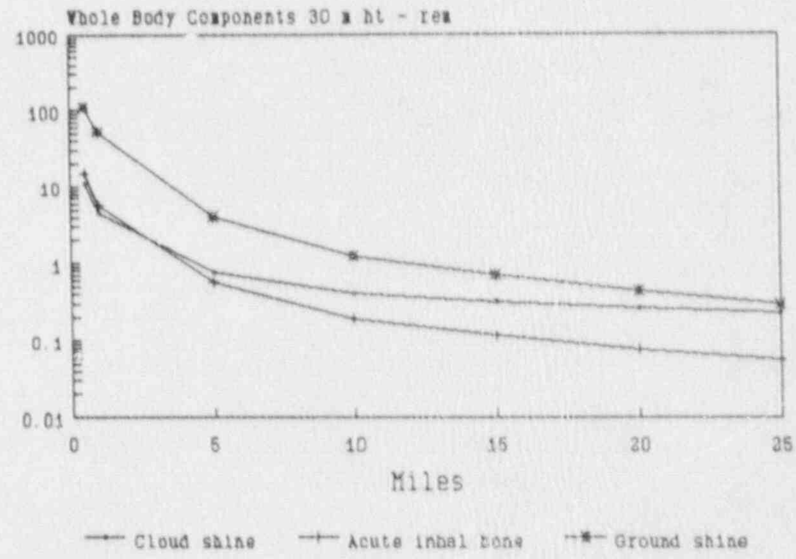


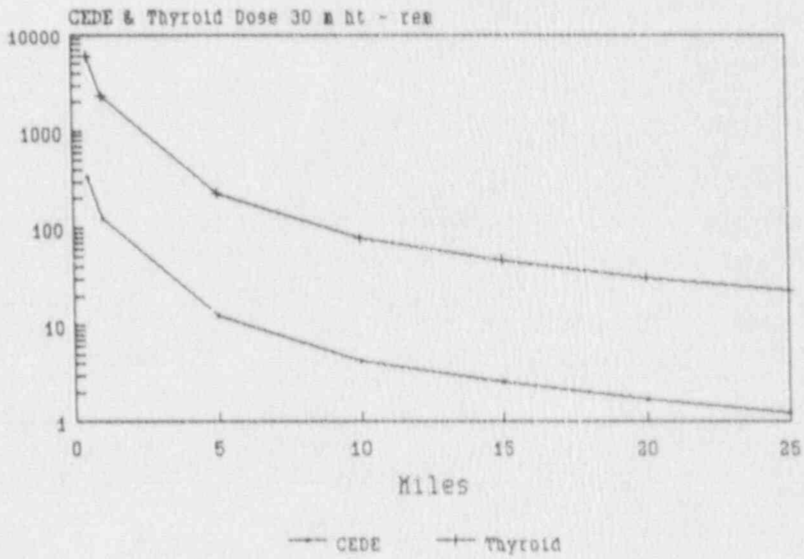


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 3 - NO RAIN

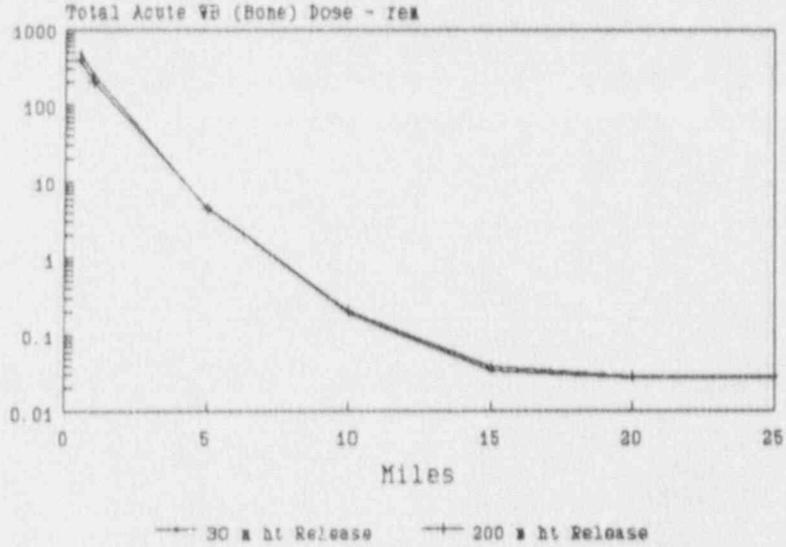


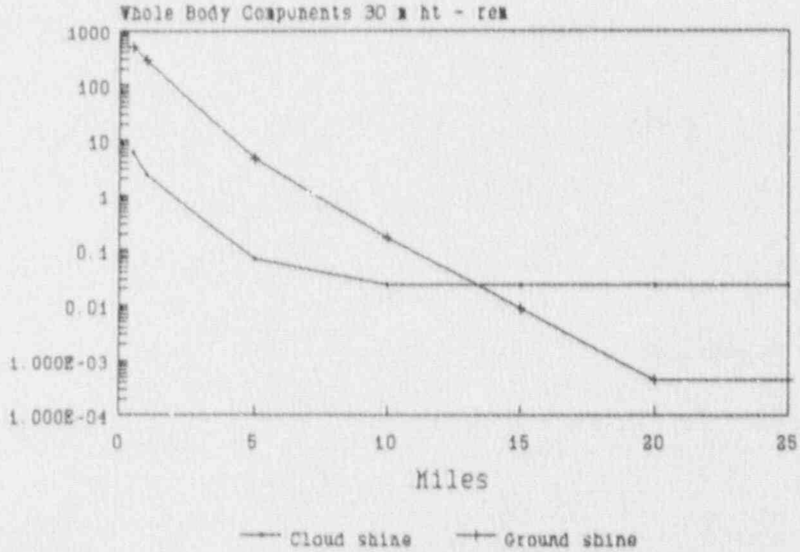


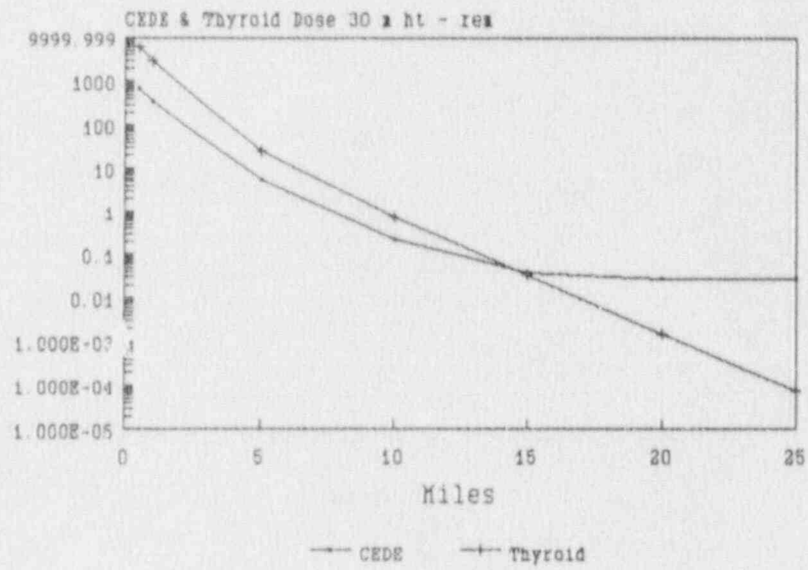


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 3 - RAIN

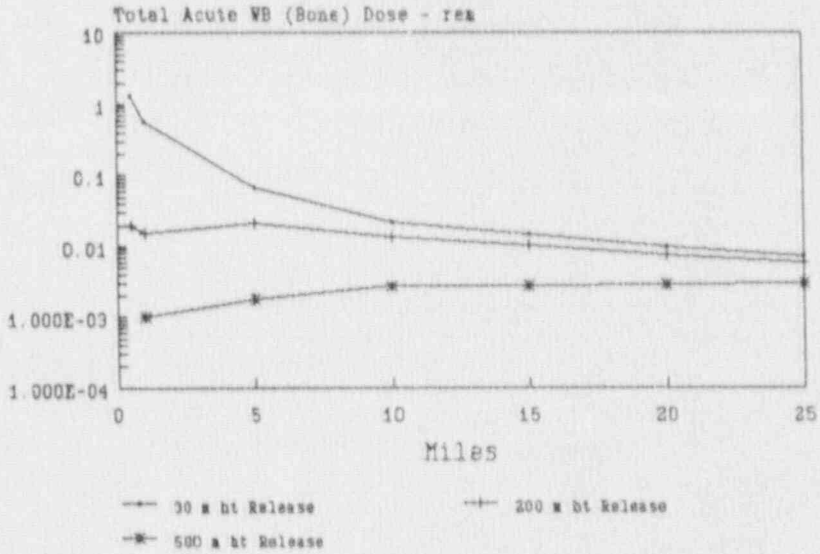


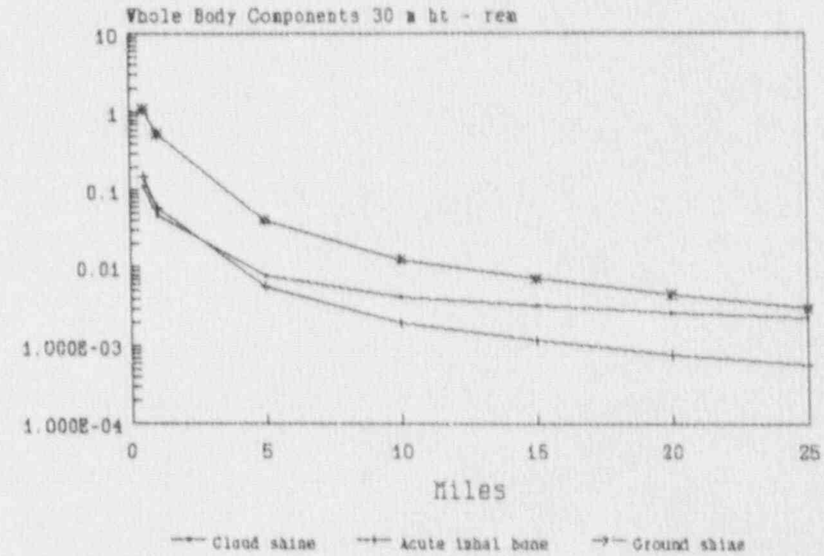


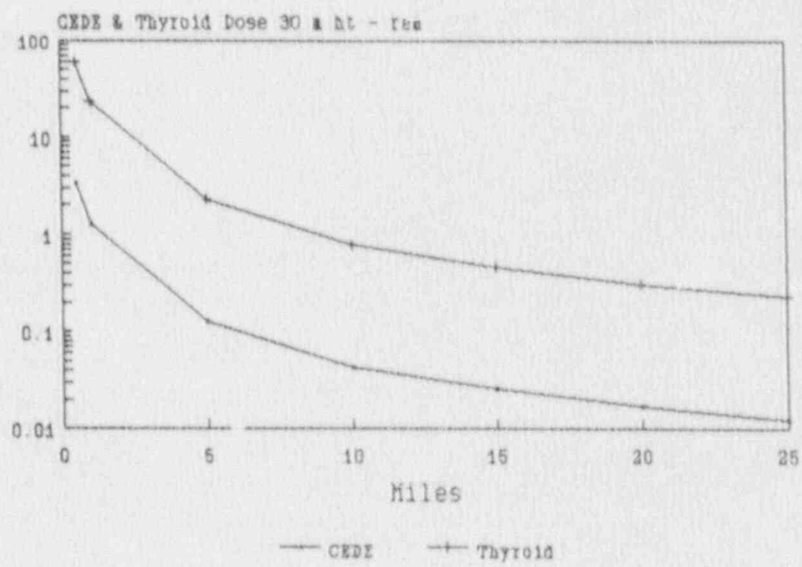


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 4 - NO RAIN

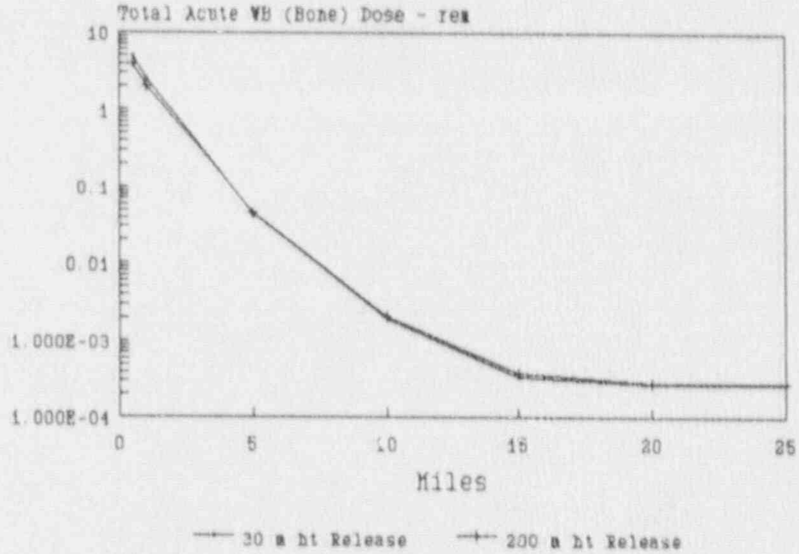


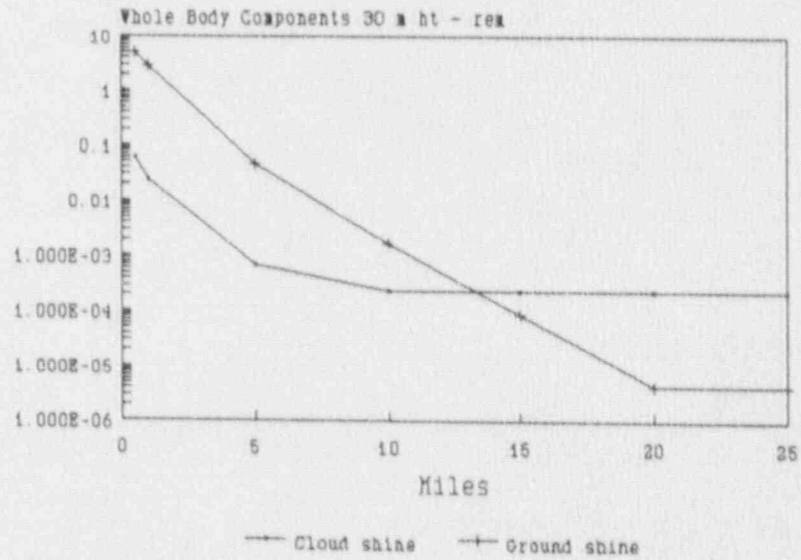


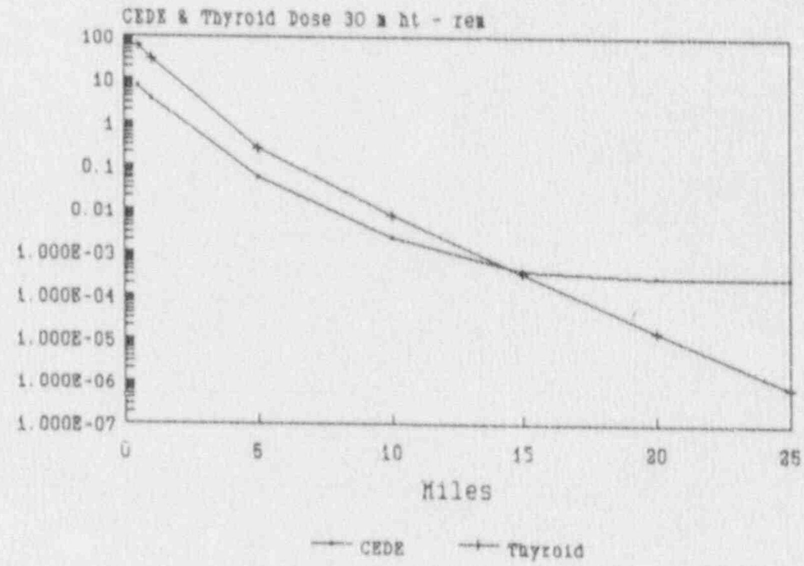


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 4 - RAIN

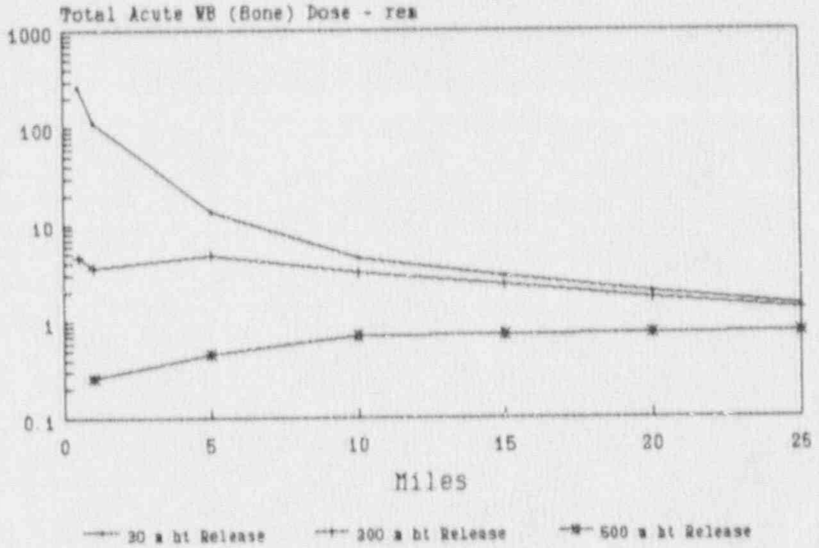


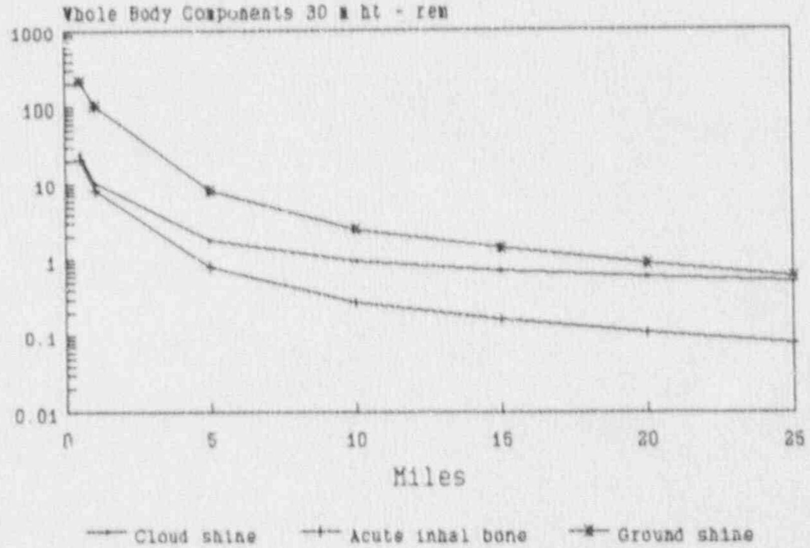


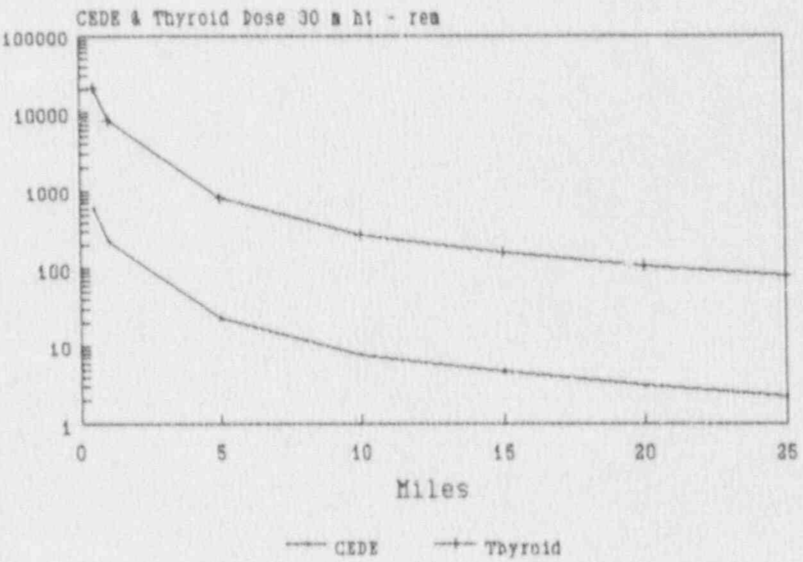


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 5 - NO RAIN

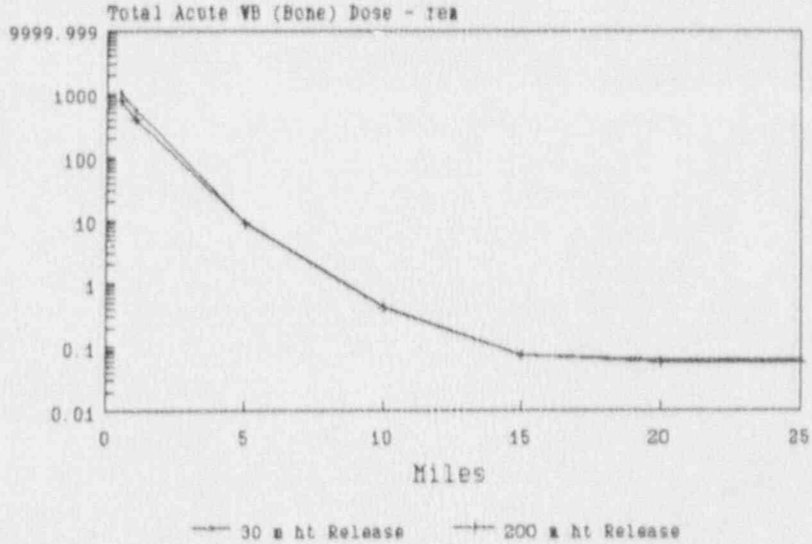


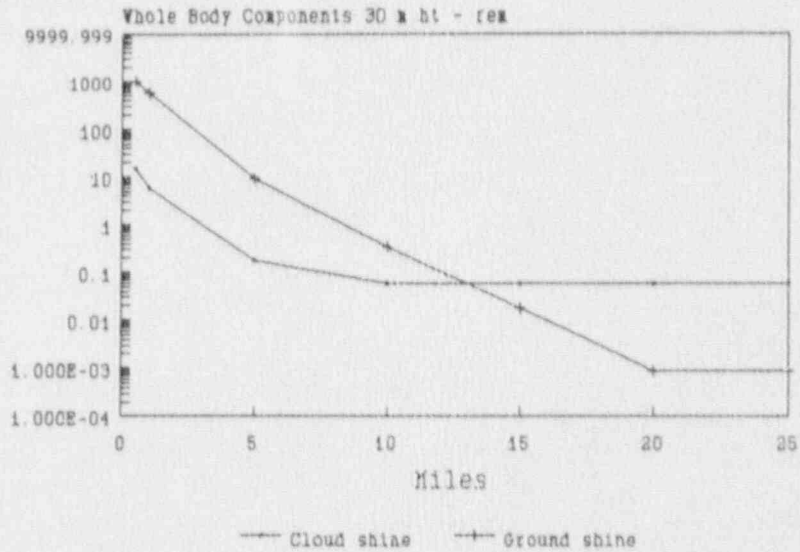


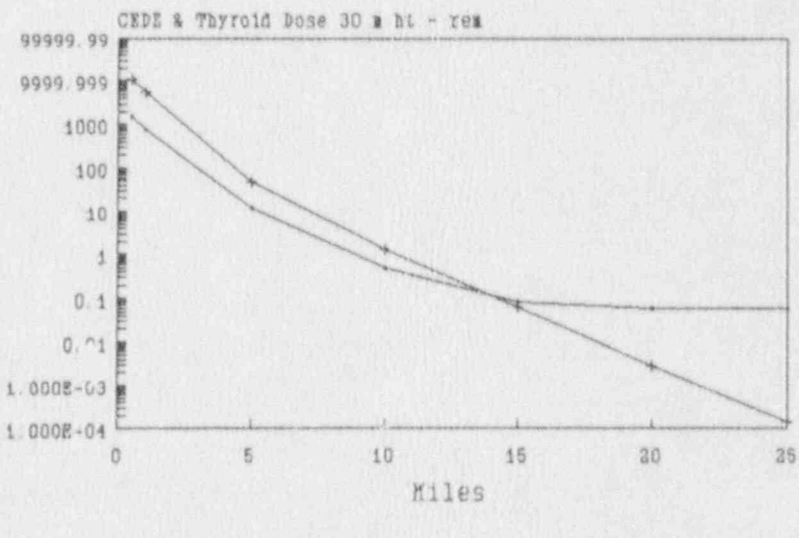


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 5 - RAIN

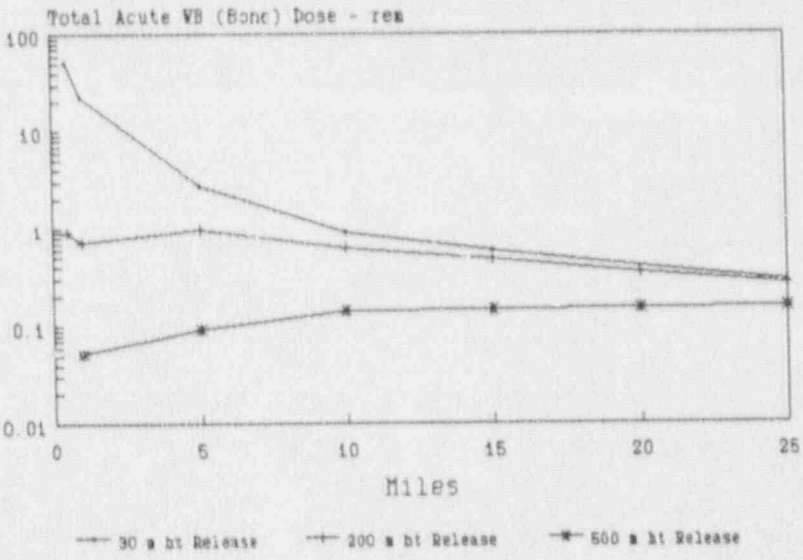


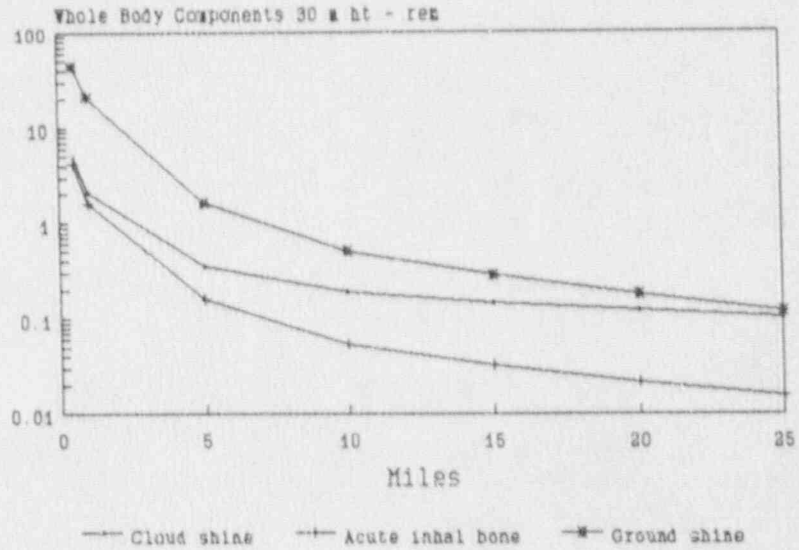


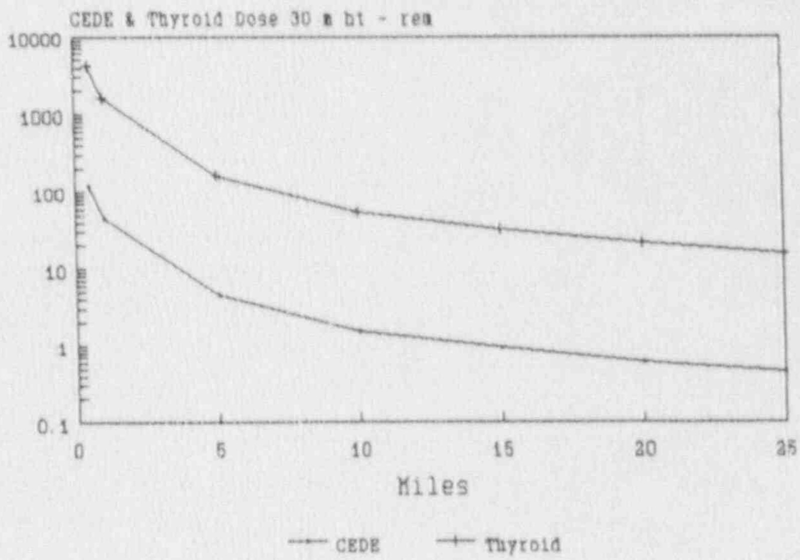


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 6 - NO RAIN

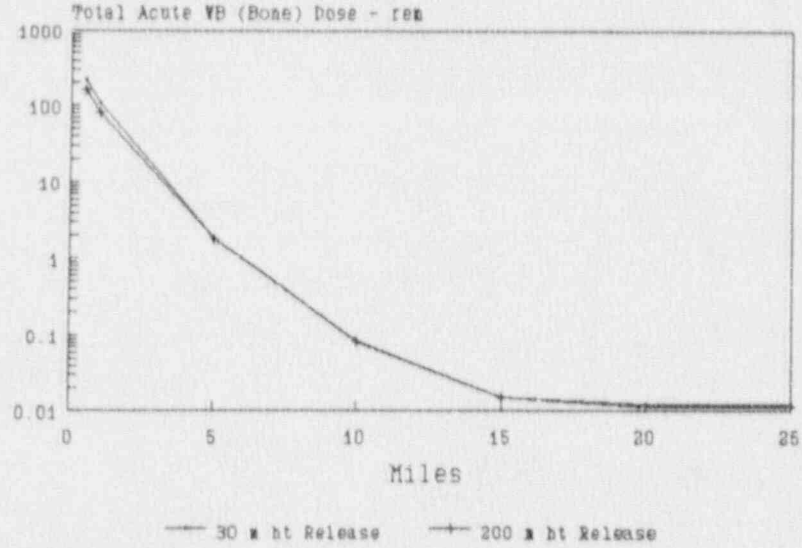


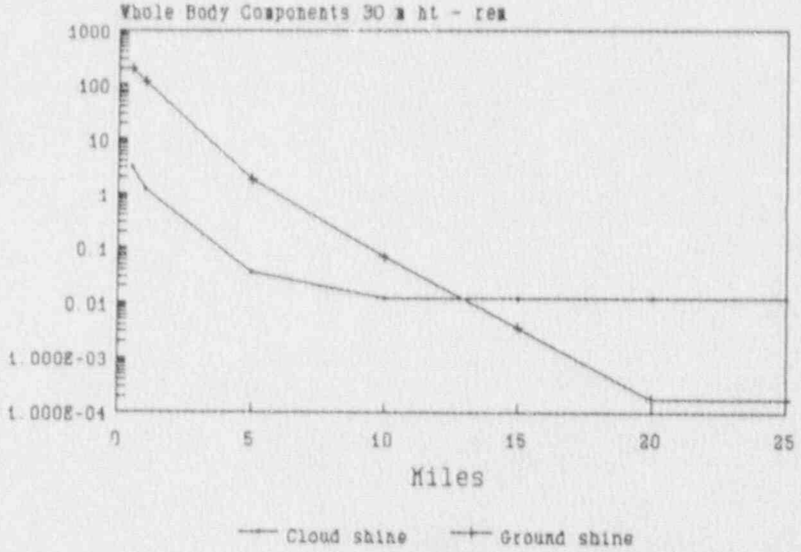


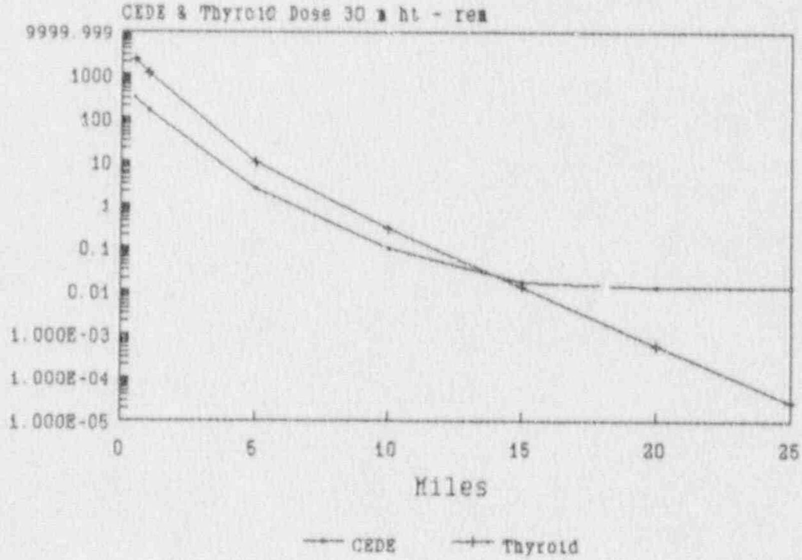


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 6 - RAIN

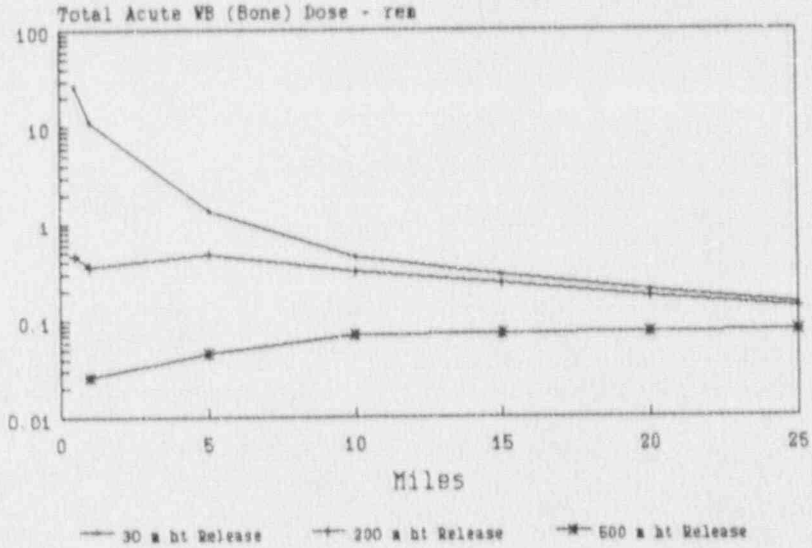


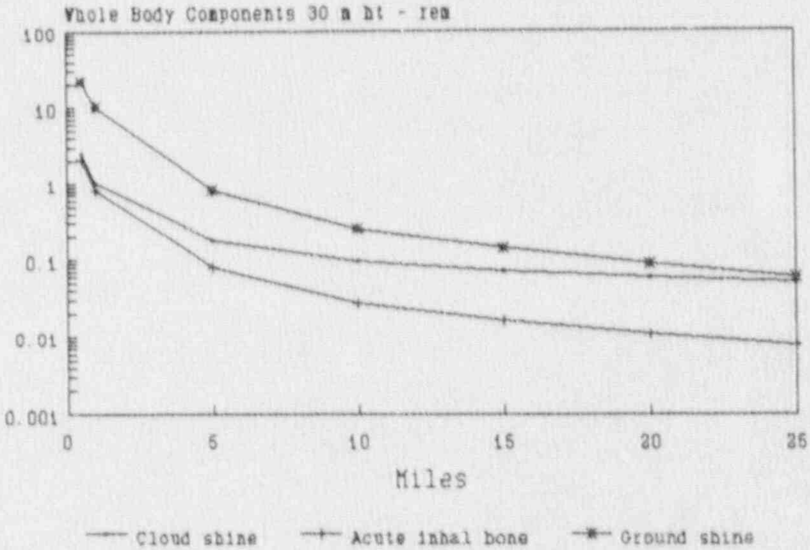


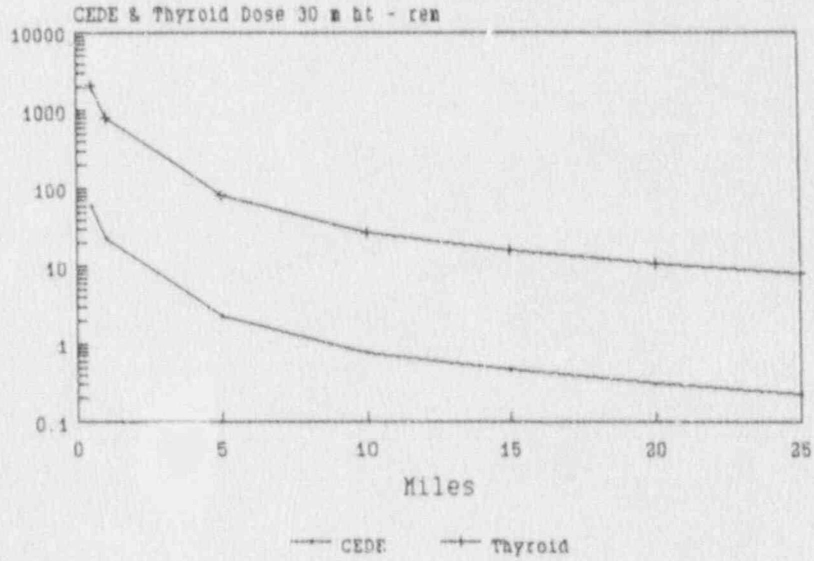


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 7 - NO RAIN

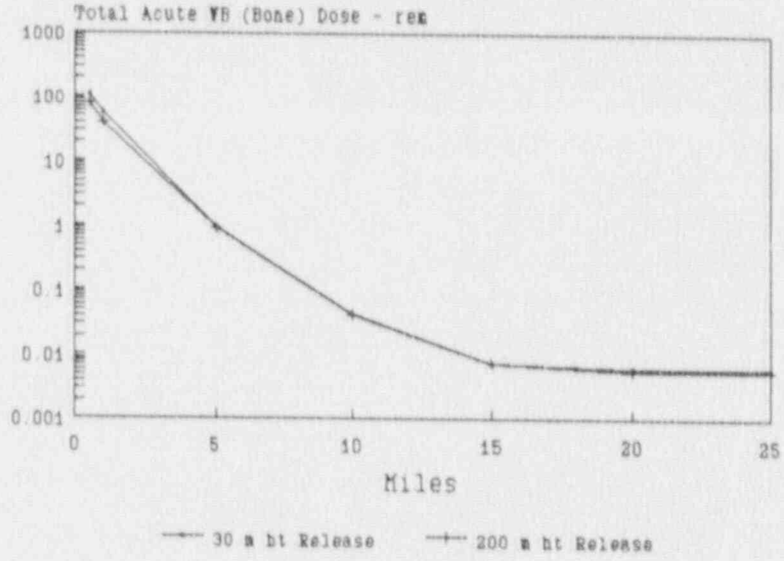


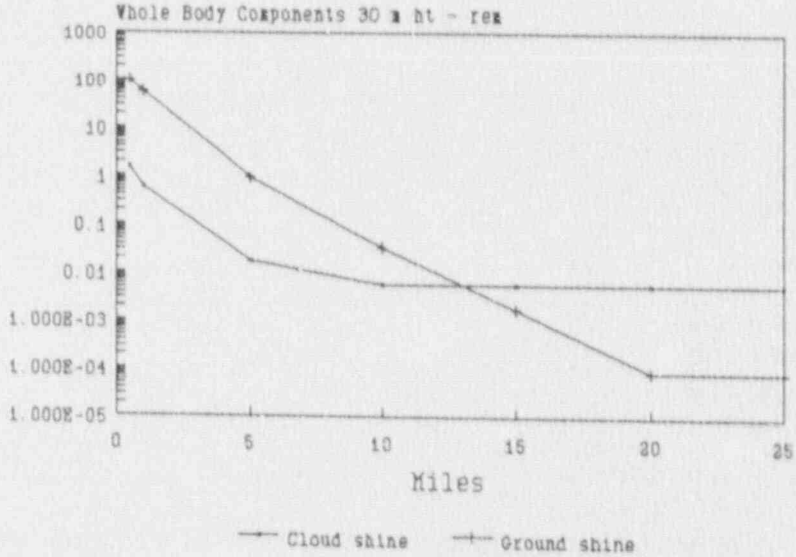


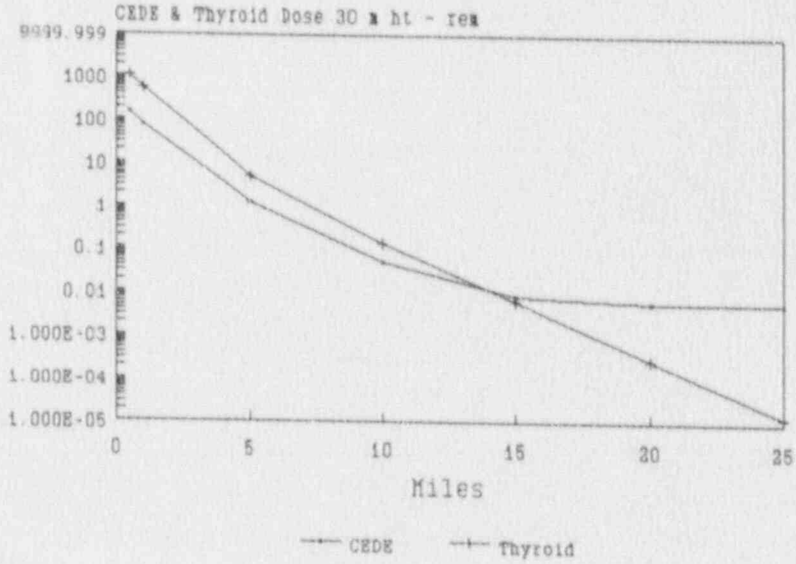


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 7 - RAIN

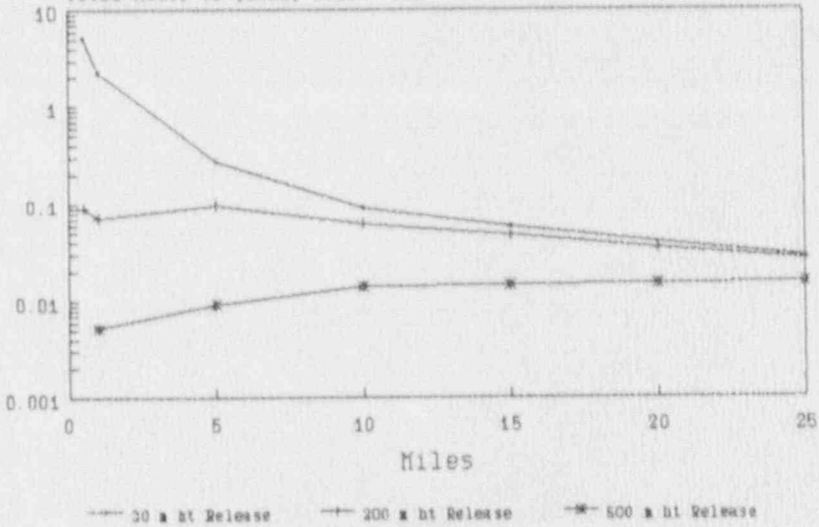


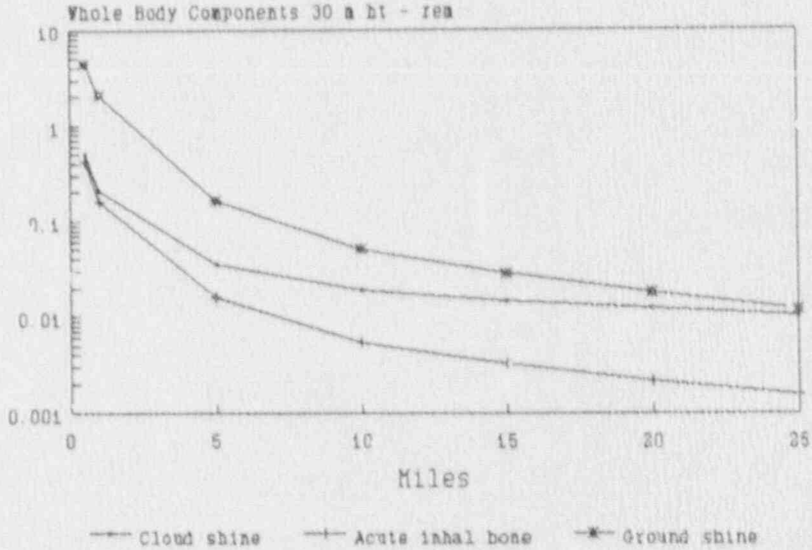


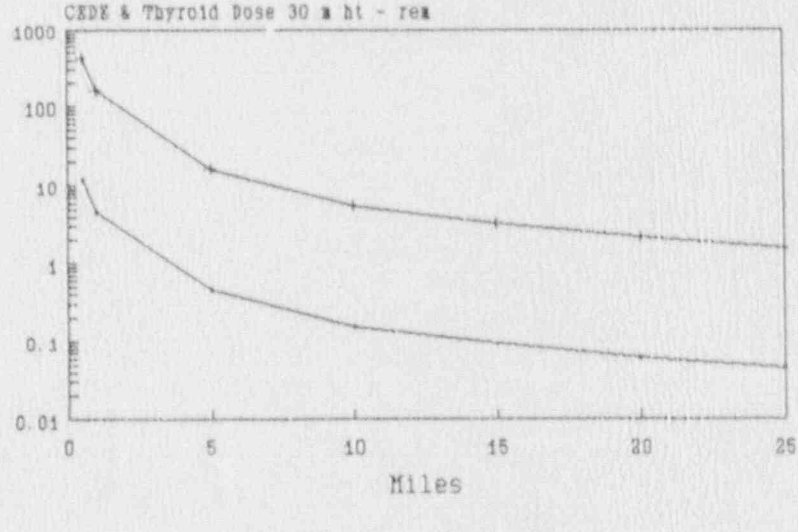


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 8 - NO RAIN

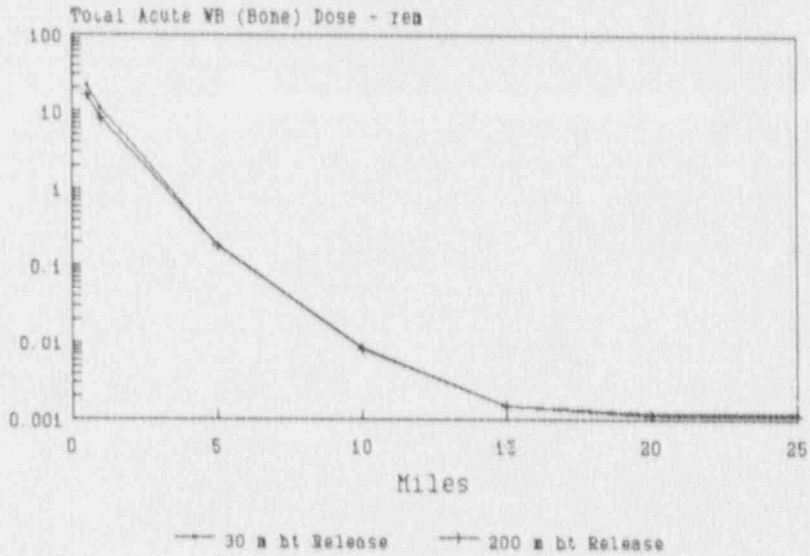


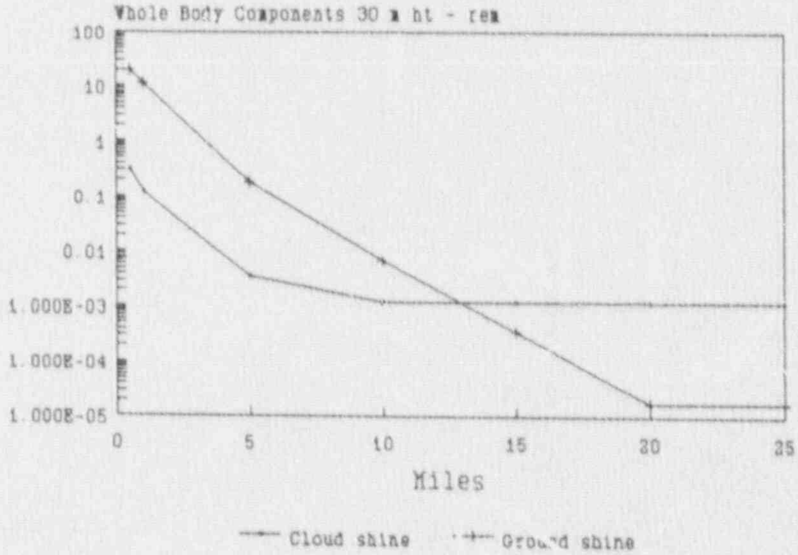


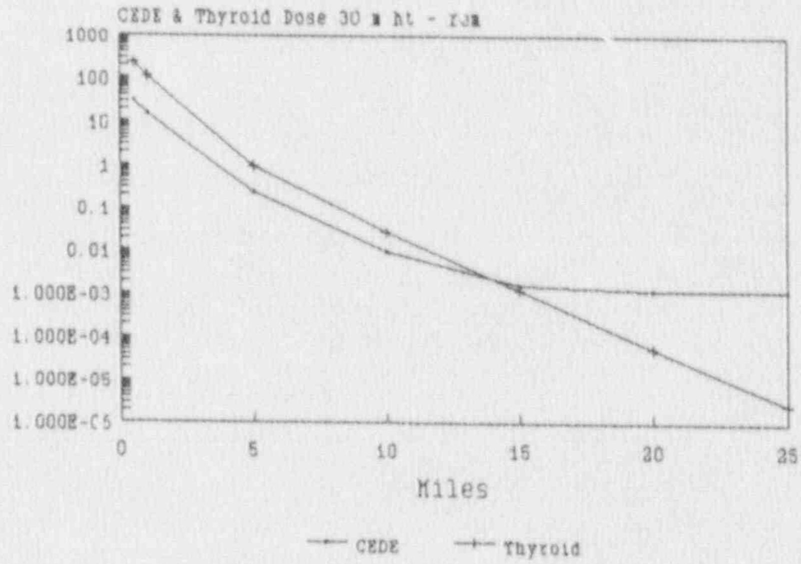


NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

CASE 8 - RAIN







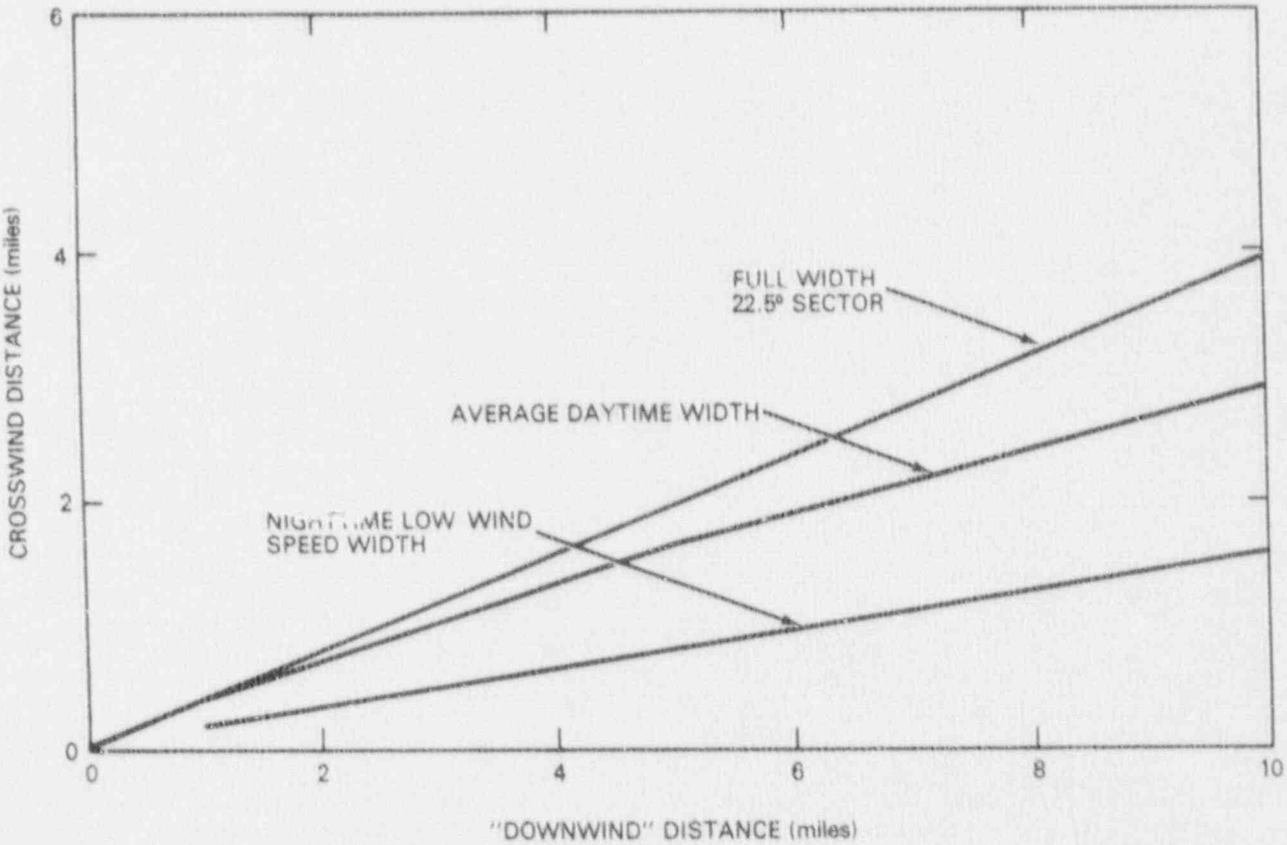
NOTE: 1 hour cloud shine and inhalation; 20 hours of ground shine

Figure F-20

PLUME WIDTHS

NOTE:

Outside the widths shown, the doses would be 1/100 of those on the curves in this Section. The plumes in this Section correspond to the average daytime width.



Theoretical full widths of puffs at one percent of maximum concentration vs distance along track of puff.

Source: NUREG-1062

Use of ARAC

Section G

G. Use of ARAC

USE OF ARAC

Objective

To access the Atmospheric Release Advisory Capability (ARAC) transport model.

Guidance

Step 1

NOTE:

ARAC will normally be contacted by NRC Headquarters. ARAC can only model releases containing up to 9 nuclides. Use field montioring, licensee estimates or RASCAL to estimate source term. When using RASCAL to estimate source term, run the "Reactor Accident Condition" source term option for the projected plant conditions and get the release estimate from the "Computed Source Term" output option. Concentrate on the following:

I-131	Te-131m	Kr-88
I-132	Te-132	Xe-133
I-133	Cs-134	
I-134	Cs-136	
I-135	Cs-137	

Estimate the source term (release content, duration, location).

Step 2

NOTE:

- Results can be expected in 1 hour after all information is provided.
- ARAC cannot project 30 day acute doses.
- Assume that all non-nobles form 1 micron-diameter (10^{-6}m) particulates.

Complete "ARAC Notification Form" (page G-3). Refer to Table G-1 (page G-5) for plant latitude and longitude.

Step 3

Call (415) 422-9100 (FTS 532-9100) and provide information. After normal working hours, tell contact to activate ARAC and provide your name and a call back number.

Step 4

Use attached reference (page G-7) to interpret results.

NOTE:

To convert from Z time [Z time is equivalent to Universal Coordinated Time (UTC) which replaced Greenwich Mean Time (GTM)], subtract the number of hours shown below from Z time:

	Standard	Daylight
Eastern	5	4
Central	6	5
Mountain	7	6
Pacific	8	7

Step 5

Inform DOE HQ that ARAC assistance has been requested:

- At NRC Headquarters - Through the Government Liaison Officer (GLO).
- Region - Through DOE representative at the FRMAC or through HQ GLO or Protective Measures Team.

END

ARAC NOTIFICATION FORM

[ARAC EMGCY PHONE NO. (415) 422-9100]
[FTS 532-9100]

1. NAME OF FACILITY WITH PROBLEM ____________________

2. PERSON SUPPLYING THIS FORM'S INFO ____________________

3. PHONE NUMBER FOR CALL BACK ____________________

4. FACSIMILE PHONE # FOR TRANSMISSION OF ARAC PLOTS ____________________

5. LATITUDE AND LONGITUDE OF BLDG. WITH PROBLEM OR STACK RELEASING MATERIAL ____________________

6. START TIME OF RELEASE ____________ (TIME ZONE) ____________

7. DURATION (IF RELEASE HAS ENDED) ____________

8. TYPE OF RELEASE EVENT: check applicable box(es)

   PUFF ☐ PLUME ☐ STACK ☐ GROUND ☐

   ROOF VENT ☐ OTHER ____________________

9. ELEVATION OF RELEASE (METERS ABOVE GROUND LEVEL): ____________

10. RELEASE CHARACTERISTICS:

   STACK / VENT / OPENING DIMENSIONS ____________________

   EFFLUENT EXIT TEMPERATURE ____________________

   EFFLUENT EXIT VELOCITY ____________________

11. NUCLIDES OF IMMEDIATE CONCERN (MAX OF NINE): check appropriate boxe(s)

   a.) I-131 ☐ b.) I-132 ☐ c.) I-133 ☐ d.) I-135 ☐ e.) Te-132 ☐ (I-132)

   f.) Sr-89 ☐ g.) Ba-140 ☐ (La-140) h.) Cs-134 ☐ i.) Cs-136 ☐ j.) Cs-137 ☐ (Ba-137)

   k.) Sr-90 ☐ l.) Ru-103 ☐ (Rh-103) m.) KR-88 ☐ n.) Xe-133 ☐

   0.) Others: (list) ____________________

CONTD. OVER ➠

12. DOSE AND PATHWAYS: check appropriate box(es)

1.) CUMMULATIVE EFFECTIVE DOSE EQUIVALENT (CEDE) via Inhalation ☐

2.) 1st YEAR EFFECTIVE WHOLE BODY via Inhalation ☐

3.) CEDE via Immersion (Cloud Shine) ☐

4.) CHILD'S THYROID via Inhalation ☐

5.) OTHER: ____________________

13. DOSE PROJECTIONS REQUESTED: (E.G. a4 = I-131 dose to child's thyroid via inhalation; m3 = Kr 88 cumulative effective dose equivalent via cloud shine)

A. ________; B. ________; C. ________; D. ________; E. ________

14. HQs DoE NOTIFIED? YES ☐ Time: ____________ NO ☐

Table G-1

LIST OF NUCLEAR POWER PLANT LATITUDES / LONGITUDES

PLANT	STATE	LATITUDE	LONGITUDE
ARKANSAS	AR	35:18'36"	95:13'51"
BEAVER VALLEY	PA	40:37'19"	80:26'02"
BELLEFONTE	AL	34:42'32"	85:55'38"
BIG ROCK POINT	MI	45:21'33"	85:11'41"
BRAIDWOOD	IL	41:14'37"	88:13'44"
BROWNS FERRY	AL	34:42'15"	87:07'07"
BRUNSWICK	NC	33:57'30"	78:00'38"
BYRON	IL	42:04'30"	89:16'55"
CALLAWAY	MO	38:45'40"	91:46'54"
CALVERT CLIFFS	MD	38:26'05"	76:26'31"
CATAWBA	SC	35:03'05"	81:04'10"
CLINTON	IL	40:10'19"	88:50'03"
COMANCHE PEAK	TX	32:17'52"	97:47'06"
COOPER	NE	40:21'43"	95:38'28"
CRYSTAL RIVER	FL	28:57'26"	82:41'56"
DAVIS-BESSE	OH	41:35'50"	83:05'11"
D. C. COOK	MI	41:58'34"	86:33'59"
DIABLO CANYON	CA	35:12'42"	120:51'16"
DRESDEN	IL	41:23'23"	88:16'16"
DUANE ARNOLD	IA	42:06'02"	91:46'38"
FARLEY	AL	31:13'22"	85:06'45"
FERMI	MI	41:57'48"	83:15'31"
FITZPATRICK	NY	43:31'26"	76:23'54"
FORT CALHOUN	NE	41:31'15"	96:04'36"
FORT ST. VRAIN	CO	40:14'41"	104:52'28"
GINNA	NY	43:16'40"	77:18'32"
GRAND GULF	MS	32:00'27"	91:02'53"
HADDAM NECK	CT	41:28'55"	72:29'57"
HARRIS	NC	35:38'00"	78:57'22"
HATCH	GA	31:56'03"	82:20'40"
HOPE CREEK	NJ	39:28'04"	75:32'17"
INDIAN POINT	NY	41:15'17"	73:57'09"
KEWAUNEE	WI	44:20'35"	87:32'10"
LaCROSSE	WI	43:33'30"	91:13'50"
LaSALLE	IL	41:14'38"	88:40'15"
LIMERICK	PA	40:13'12"	75:35'24"
MAINE YANKEE	ME	43:57'02"	69:41'46"
McGUIRE	NC	35:25'56"	80:56'54"
MILLSTONE	CT	41:18'31"	72:10'05"
MONTICELLO	MN	45:20'00"	93:50'54"
NINE MILE POINT	NY	43:31'20"	76:24'36"
NORTH ANNA	VA	38:03'39"	77:47'26"
OCONEE	SC	34:47'30"	82:53'55"
OYSTER CREEK	NJ	39:48'51"	74:12'23"

Table G-1 - continued

PLANT	STATE	LATTITUDE	LONGITUDE
PALISADES	MI	42:19'20"	86:18'55"
PALO VERDE	AZ	33:23'23"	112:51'43"
PEACH BOTTOM	PA	39:45'32"	76:16'09"
PERRY	OH	41:48'04"	81:08'36"
PILGRIM	MA	41:56'40"	70:34'46"
POINT BEACH	WI	44:16'51"	87:32'10"
PRAIRIE ISLAND	MN	44:37'19"	92:37'59"
QUAD CITIES	IL	41:43'34"	90:18'36"
RANCHO SECO	CA	38:20'40"	121:07'12"
RIVER BEND	LA	30:45'26"	91:19'54"
ROBINSON	SC	34:24'09"	80:90'31"
SALEM	NJ	39:27'46"	75:32'09"
SAN ONOFRE	CA	33:22'13"	117:33'25"
SEABROOK	NH	42:53'53"	70:51'05"
SEQUOYAH	TN	35:13'24"	85:05'16"
SHOREHAM	NY	40:57'30"	72:52'00"
SOUTH TEXAS	TX	28:47'42"	96:02'53"
ST. LUCIE	FL	27:20'55"	80:14'47"
SUMMER	SC	34:17'45"	81:19'13"
SURRY	VA	37:09'56"	76:41'54"
SUSQUEHANNA	PA	41:05'30"	76:08'55"
THREE MILE ISLAND	PA	40:09'11"	76:43'30"
TROJAN	OR	46:02'27"	122:53'04"
TURKEY POINT	FL	25:26'06"	80:19'53"
VERMONT YANKEE	VT	42:46'49"	72:30'57"
VOGTLE	GA	33:08'31"	81:45'53"
WATERFORD	LA	29:59'42"	90:28'16"
WATTS BAR	TN	35:36'10"	84:47'25"
WNP-2	WA	46:28'17"	119:19'59"
WOLF CREEK	KS	38:14'20"	95:41'20"
YANKEE ROWE	MA	42:43'41"	72:55'44"
ZION	IL	42:26'44"	87:48'08"

ARAC REFERENCE

Atmospheric Release Advisory Capability

The Atmospheric Release Advisory Capability service, initiated in 1972 by the Atomic Energy Commission (now Department of Energy), is housed at Lawrence Livermore National Laboratory (LLNL). ARAC provides support to emergency response teams during accidents/incidents involving nuclear material.

ARAC provides the user with computer model estimates of the distribution of contamination resulting from an accident which releases nuclear material to the atmosphere. ARAC products include computer generated estimates of the location and contamination levels of deposited radiological material and radiation dose to exposed population in the surrounding areas. Until time and equipment permit completion of extensive radiation surveys and bioassays, ARAC projections will assist in assessing the potential impact of an accident and in identifying areas for initial investigation by response force radiological teams.

In the event of an accident that may or already has released nuclear material to the atmosphere at an NRC facility, the ARAC center will be alerted by either the HQ Protective Measures Team (PMT), Radiological Assessment (RAM), or Site Team Protective Measures Coordinator (PMC) via telephone using the ARAC Notification Form shown on page G-3. ARAC should be contacted directly by calling ARAC's emergency number, (415) 422-9100 (commercial) or 532-9100 (FTS).

During normal working hours (currently 0730-1615 local [Pacific] time, Monday-Friday, excluding holidays), initial estimates of the extent of contamination can be ready for transmission via telefacsimile from ARAC within 30 minutes. This assumes that ARAC has received notification of the accident/incident and the information required to make a calculation via the ARAC notification form (page G-3).

Calls to the ARAC emergency phone number during other than normal working hours will be answered by the LLNL Fire Dispatcher. The LLNL Fire Dispatcher should be given the name and phone number of the individual requesting the ARAC support, the latitude and longitude of the problem site (for nuclear power plants, see pages G-5 and G-6), the general nature of the blem, and an ARAC call-out should be requested. Responses outside normal working hours are subject to an additional 60-90 minutes delay and availability of personnel. Every effort should be made to provide updated or supplementary information to the ARAC Center as soon as it is available. (Note that ARAC can generate projection plots to match a given map scale [e.g., 1:50,000] for ease of overlaying the projected deposition pattern.) As soon as available the ARAC dose projection plots will be transmitted to the PMT, RAM and/or the PMC via telefacsimile to their CCITT Group 3 telefacsimile machine.

The following provides information regarding the sample ARAC projection shown on page G-9 and provides some insight into interpreting the ARAC product:

(A) The triangular "tick" marks along the upper and right side of the graphical plot display are distance scales in feet and refer to the scale depicted in "I" on the same plot. Immediately to the right of the graphical plot will be an area containing notes describing the graphical ARAC plots, lines B-I on page G-9.

(B) This is a title line for the graphical plot.

(C) Denotes the date and time that this specific computer model projection was produced.

(D) This section of four lines of text is reserved for general amplifying remarks about the computer estimation.

(E) This line identifies either the dose or deposition plot once again and the line below gives the dose integration period or total deposition period time as appropriate (NOTE: All times will be shown as "Z" time. "Z" is equivalent to Universal Coordinated Time [UCT] which has replaced the more familiar Greenwich Mean Time [GMT]).

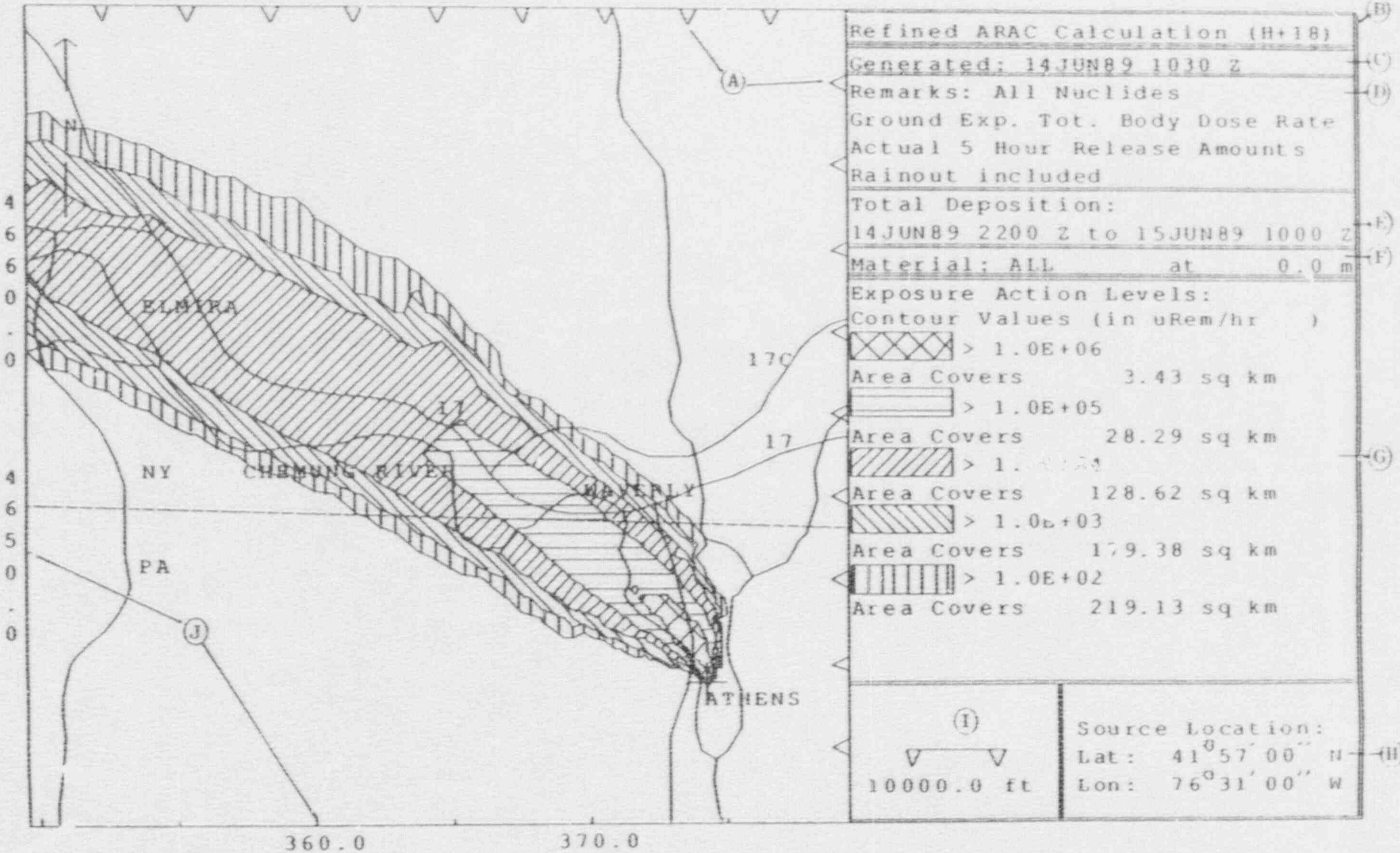
(F) This line of text gives the radiological material modelled and the height above ground level (AGL) at which the contour levels are calculated and displayed.

(G) This area will show the numerical value and units of specific computer projection isopleths of dose or deposition on the ground. These isopleth values will be keyed to Protection Action Guides (PAGs) levels as calculated for that particular plot when such PAGs are finalized by EPA. Additionally the total area enclosed by any particular isopleth will be shown in the units of square kilometers. Note that the area given will encompass the area of all higher levels shown (e.g., the area given for exceeding 1.0E+05 mrem/hr is the sum of the area covered by the >1.0E+05 and the >1.0E+06 mrem/hr contour patterns).

(H) The latitude and longitude of the release point or projected release point is depicted here in degrees, minutes, and seconds.

(I) This is the foot scale referred to previously.

(J) The left and bottom of the graphical plot area have a Universal Transverse Mercator (UTM) projection scale in kilometers and tens of kilometers, e.g., the distance between 360.0 and 370.0 on the lower border of the graphical plot is 10 kilometers.


Refined ARAC Calculation (H+18)
Generated: 14JUN89 1030 Z
Remarks: All Nuclides
Ground Exp. Tot. Body Dose Rate
Actual 5 Hour Release Amounts
Rainout included
Total Deposition:
14JUN89 2200 Z to 15JUN89 1000 Z
Material: ALL at 0.0 m
Exposure Action Levels:
Contour Values (in uRem/hr)
> 1.0E+06
Area Covers 3.43 sq km
> 1.0E+05
Area Covers 28.29 sq km
> 1.
Area Covers 128.62 sq km
> 1.0E+03
Area Covers 179.38 sq km
> 1.0E+02
Area Covers 219.13 sq km
10000.0 ft
Source Location:
Lat: 41°57′00″ N
Lon: 76°31′00″ W
A
B
C
D
E
F
G
H
I
J
ELMIRA
NY
PA
CHEMUNG RIVER
WAVERLY
ATHENS
17C
17
N
4660.0
4650.0
360.0
370.0

Plume Dose and Source Term Hand Calculation

Section H

H Plume Dose and Source Term Hand Calculation

PLUME DOSE AND SOURCE TERM HAND CALCULATION

Objective

Provide quick hand calculation methods.

Guidance

NOTE:

> A list of all tables, figures and worksheets is found on page H-3. Assumptions are listed on page H-47.

Step 1

Use one of the following to estimate dose or source term:

NOTE:

> Table H-1 (page H-4) can be used to estimate stability class and wind speed.

NOTE:

> The fire release cases can be considered representative of the worst possible accidents.

A. Down Wind Projection Based on Estimated Quantity Released

1. Quick Estimate Using Average Meteorological Conditions:

	Page WB, Bone, Lung CEDE	Page Thyroid
Release initiated by **fire**		
Close-in (1/4-mile)	H-10	H-37
1-mile	H-20	H-37
Maximum release **(100% to atmosphere)**		
Close-in (1/4-mile)	H-5	H-37
1-mile	H-15	H-37

2. Manual Calculation for **Isotopic Release Rate**, Various Meteorological Conditions and Distance:

   - Worksheet H-1 (page H-41)

B. Dose Projections Based on **Air or Ground Concentrations**

   - Worksheet H-2 (page H-43)

C. Down Wind Projections Based on **Field Radiation Readings**

   - Worksheet H-3 (page H-45)

D. **Source Term Based on Field Radiation Readings**

- Worksheet H-3 (page H-45)

E. **Releases in Enclosed Areas**

- Worksheet H-4 (page H-46)

Step 2

To consider shielding multiply doses by appropriate shielding factors in Table H-8 (page H-34) (immersion) or Table H-9 (page H-35) (ground).

Step 3

NOTE:

Acute dose in used for insights to early health effects.

Determine consequences:

- Compare the lung inhalation dose with the lung early heatlh effects thresholds in Table H-10 (page H-36).
- Compare thyroid inhalation dose and the sum of the immersion, ground shine, and CEDE with thyroid and whole body EPA PAGs respectively on page I-4.
- Compare with sum of the whole body immersion, ground shine and acute bone marrow inhalation dose with the whole body and bone marrow early health effects thresholds in Table H-10 (page H-36) or as shown in Figure H-1, page H-39.

Step 4

Present results.

Ensure everyone understands the great uncertainties of these projections.

END

LIST OF TABLES AND ASSUMPTIONS

Table H-1

RELATION OF WEATHER CONDITIONS TO STABILITY CLASSES

Beaufort Scale for Windspeed

Observations	Windspeed m/sec
Smoke rises vertically	0.3
Smoke drift gives direction but wind not felt on face	1
Wind felt on face, leaves rustle vane moved by wind	2-3
Leaves and twigs in constant motion wind extends flag	4-5
Moves dust, loose paper and small branches	6-7
Small trees in leaf begin to sway	8-9
Large branches in motion; high wires whistle	10-12

Stability Classes

Atmospheric Condition	Stability Class
Low wind speed (<2 m/s) at night	E-F
Typical	D
Partly cloudy with low winds or Sunny with high winds	B-C
Very low (<2 m/s) windspeed on a bright summer day	A

Table H-2

<1/4 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE

$\frac{\text{rem}}{\text{Ci release}}$

Radio-nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
H-3(H2O)	1.0E-05	9.3E-06	9.3E-06	0.0	0.0
C-14(ORG)	3.3E-04	1.3E-04	1.3E-04	0.0	0.0
Na-22	1.2E-03	NC	NC	1.5E-04	5.6E-05
Na-24	1.9E-04	1.2E-04	7.0E-04	3.2E-04	9.7E-05
P-32	2.4E-03	2.7E-03	1.3E-02	0.0	0.0
P-33	3.6E-04	NC	NC	0.0	0.0
S-35	3.9E-04	2.3E-05	1.4E-03	0.0	0.0
Cl-36	3.4E-03	NC	NC	1.2E-14	2.6E-14
K-40	1.9E-03	NC	NC	1.1E-05	3.7E-06
K-42	2.1E-04	NC	NC	1.9E-05	6.5E-06
Ca-45	1.0E-03	NC	NC	3.9E-17	2.1E-16
Sc-46	4.6E-03	NC	NC	1.4E-04	5.2E-05
Ti-44	7.1E-02	NC	NC	4.9E-06	2.5E-06
V-48	1.6E-03	NC	NC	2.0E-04	7.4E-05
Cr-51	5.2E-05	1.0E-05	1.7E-04	2.1E-06	9.0E-07
Mn-54	1.0E-03	4.5E-04	1.4E-03	5.8E-05	2.3E-05
Mn-56	5.9E-05	NC	NC	1.2E-04	4.4E-05
Fe-55	4.2E-04	9.3E-06	1.7E-04	3.2E-11	1.2E-10
Co-58	1.7E-03	2.0E-04	2.6E-03	6.7E-05	2.6E-05
Fe-59	2.3E-03	7.0E-04	4.6E-03	8.2E-05	2.9E-05
Co-60	3.4E-02	5.1E-04	7.5E-03	1.7E-04	6.1E-05
Ni-63	4.9E-04	1.8E-05	5.4E-04	0.0	0.0
Cu-64	4.3E-05	NC	NC	1.3E-05	5.2E-06
Zn-65	3.2E-03	2.7E-04	1.4E-03	4.0E-05	1.5E-05
Ga-68	2.2E-05	3.3E-06	1.2E-04	6.3E-05	2.6E-05
Ge-68+Ga-68[6]	8.1E-03	0.0	0.0	6.3E-05	2.6E-05
Se-75	1.3E-03	3.0E-04	1.3E-03	2.4E-05	1.1E-05
Kr-85	NC	NC	NC	1.5E-07	6.1E-08
Kr-85m	NC	NC	NC	9.6E-06	4.3E-06
Kr-87	NC	NC	NC	6.0E-05	2.0E-05
Kr-88+Rb-88[6]	NC	NC	NC	2.0E-04	6.3E-05
Rb-86	1.0E-03	1.0E-03	1.7E-03	6.6E-06	2.4E-06
Rb-87	5.1E-04	NC	NC	0.0	0.0
Rb-88	1.3E-05	NC	NC	4.8E-05	1.6E-05
Sr-89	6.5E-03	1.3E-03	1.8E-02	9.5E-09	3.7E-09
Sr-90	2.0E-01	2.7E-03	3.7E-02	0.0	0.0
Sr-91	2.6E-04	NC	NC	4.8E-05	1.8E-05
Y-90	1.3E-03	1.6E-04	5.4E-03	0.0	0.0
Y-91	7.7E-03	7.5E-04	1.9E-02	2.5E-07	8.9E-08
Y-91m	5.7E-06	NC	NC	3.5E-05	1.5E-05

* = 1st hour
NC = not calculated

Table H-2 Continued

<1/4 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE

rem / Ci release

Radio-nuclide	Inhalation[1] CEDE[4]	Acute[2] Bone Marrow	Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Zr-93	5.0E-02	NC	NC	0.0	0.0
Zr-95	3.7E-03	1.6E-03	5.5E-03	5.0E-05	2.0E-05
Nb-94	6.5E-02	NC	NC	1.1E-04	4.3E-05
Nb-95	9.1E-04	2.3E-04	2.3E-03	5.3E-05	2.1E-05
Mo-99	6.2E-04	2.2E-04	2.5E-03	1.0E-05	4.2E-06
Tc-99	1.3E-03	2.3E-05	3.2E-03	2.3E-11	1.1E-11
Tc-99m	5.1E-06	1.8E-06	1.8E-05	7.3E-06	3.3E-06
Rh-103	NC	NC	NC	NC	NC
Ru-103	1.4E-03	2.3E-04	4.0E-03	3.2E-05	1.3E-05
Ru-105	7.1E-05	NC	NC	5.3E-05	2.2E-05
Rh-106	NC	NC	NC	1.4E-05	5.6E-06
Ru-106+Rh-106[6]	7.5E-02	1.3E-03	5.2E-02	1.4E-05	5.6E-06
Ag-110m	1.3E-02	9.3E-04	7.5E-03	1.9E-04	7.2E-05
Cd-109+Ag-109m[6]	1.8E-02	NC	NC	1.7E-07	1.1E-07
Cd-113m	2.4E-01	NC	NC	0.0	0.0
In-114m	1.4E-02	NC	NC	5.8E-06	2.4E-06
Sn-113	1.7E-03	NC	NC	3.5E-07	1.8E-07
Sn-123	5.1E-03	NC	NC	4.8E-07	1.8E-07
Sn-126+Sb-126m[6]	1.6E-02	NC	NC	1.9E-04	7.7E-05
Sb-124	3.9E-03	7.5E-04	1.3E-02	1.3E-04	4.8E-05
Sb-126	1.8E-03	NC	NC	1.9E-04	7.6E-05
Sb-126m	5.3E-06	0.0	0.0	1.1E-04	4.3E-05
Sb-127	9.5E-04	NC	NC	4.5E-05	1.8E-05
Sb-129	1.0E-04	NC	NC	9.9E-05	3.8E-05
Te-127	5.0E-05	NC	NC	3.2E-07	1.4E-07
Te-127m	3.4E-03	1.4E-03	8.7E-03	4.5E-08	3.6E-08
Te-129	1.4E-05	NC	NC	3.5E-06	1.5E-06
Te-129m+Te-129[6]	3.8E-03	2.4E-03	1.5E-02	4.6E-06	1.8E-06
Te-131	7.5E-05	NC	NC	2.7E-05	1.1E-05
Te-131m	1.0E-03	1.3E-04	1.3E-03	9.8E-05	3.7E-05
Te-132	1.5E-03	1.6E-04	8.1E-04	1.3E-05	5.9E-06
I-125	3.8E-03	9.9E-06	5.8E-05	1.3E-07	1.2E-07
I-129	2.7E-02	1.1E-05	1.3E-04	1.2E-07	1.2E-07
I-131	5.2E-03	3.1E-05	3.8E-04	2.5E-05	1.1E-05
I-132	6.0E-05	8.1E-06	1.6E-04	1.6E-04	6.1E-05
I-133	9.2E-04	1.6E-05	4.8E-04	4.1E-05	1.7E-05
I-134	2.1E-05	3.5E-06	8.1E-05	1.8E-04	6.9E-05
I-135+Xe-135m[6]	1.9E-04	1.3E-05	2.6E-04	1.4E-04	5.1E-05
Xe-131m	NC	NC	NC	2.6E-07	1.6E-07
Xe-133	NC	NC	NC	1.3E-06	6.7E-07

* = 1st hour
NC = not calculated

Table H-2 Continued

<1/4 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE

$\frac{\text{rem}}{\text{Ci release}}$

Radio-nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Xe-133m	NC	NC	NC	1.6E-06	7.5E-07
Xe-135	NC	NC	NC	1.6E-05	7.0E-06
Xe-135m	NC	NC	NC	2.8E-05	1.2E-05
Xe-138	NC	NC	NC	8.3E-05	2.8E-05
Cs-134	7.2E-03	1.3E-03	1.6E-03	1.1E-04	4.2E-05
Cs-135	7.1E-04	1.2E-04	2.4E-04	0.0	0.0
Cs-136	1.1E-03	8.7E-04	1.2E-03	1.5E-04	5.7E-05
Ba-137m	NC	NC	NC	4.1E-05	1.6E-05
Cs-137+Ba-137m[6]	5.0E-03	8.1E-04	1.3E-03	4.1E-05	1.6E-05
Ba-133	1.2E-03	NC	NC	2.2E-05	1.0E-05
Cs-138	1.6E-05	NC	NC	1.7E-04	5.8E-05
Ba-140	5.9E-04	5.8E-04	9.2E-04	1.2E-05	5.2E-06
La-140	7.6E-04	2.7E-04	2.4E-03	1.6E-04	5.7E-05
Ce-141	1.4E-03	1.1E-04	4.9E-03	4.2E-06	1.9E-06
Ce-144+Pr-144[6]	5.9E-02	8.1E-04	4.7E-02	3.3E-06	1.2E-06
Pr-144	6.8E-06	NC	NC	2.4E-06	8.1E-07
Pr-144m	NC	NC	NC	8.9E-08	6.8E-08
Pm-145	4.8E-03	NC	NC	3.3E-07	2.2E-07
Pm-147	6.1E-03	8.7E-05	2.3E-03	1.9E-10	8.5E-11
Sm-147	1.2E+01	NC	NC	0.0	0.0
Sm-151	4.7E-03	2.7E-05	6.4E-04	6.4E-12	1.4E-11
Eu-152	3.5E-02	5.3E-04	6.4E-03	7.7E-05	2.9E-05
Eu-154	4.5E-02	7.0E-04	1.1E-02	8.5E-05	3.2E-05
Eu-155	6.5E-03	1.2E-04	2.1E-03	2.4E-06	1.2E-06
Gd-153	3.7E-03	NC	NC	3.3E-06	1.7E-06
Tb-160	3.9E-03	NC	NC	7.4E-05	2.8E-05
Ho-166m	1.2E-01	NC	NC	1.1E-04	4.3E-05
Tm-170	4.1E-03	NC	NC	1.6E-07	8.6E-08
Yb-169	1.3E-03	2.9E-04	4.1E-03	1.4E-05	6.6E-06
Hf-172	5.0E-02	NC	NC	NC	NC
Hf-181	2.4E-03	NC	NC	3.4E-05	1.5E-05
Ta-182	7.0E-03	NC	NC	8.6E-05	3.2E-05
W-187	9.7E-05	NC	NC	3.1E-05	1.3E-05
Ir-192	4.4E-03	4.8E-04	8.7E-03	5.3E-05	2.3E-05
Au-198	5.1E-04	NC	NC	2.7E-05	1.1E-05
Hg-203	1.1E-03	NC	NC	1.4E-05	6.4E-06
Tl-204	3.8E-04	NC	NC	3.7E-08	1.9E-08
Pb-210	2.1E+00	1.7E-03	6.3E-04	2.9E-08	1.9E-08
Bi-207	3.1E-03	NC	NC	1.0E-04	4.0E-05
Bi-210	3.1E-02	NC	NC	0.0	0.0

* = 1st hour

NC = not calculated

Table H-2 Continued

<1/4 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE

rem / Ci release

Radio-nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Po-210	1.5E+00	4.9E-02	1.6E+00	5.9E-10	2.3E-10
Ra-226	1.3E+00	5.8E-03	1.5E+00	4.0E-07	1.8E-07
Ac-227	1.0E+03	NC	NC	6.0E-09	2.8E-09
Ac-228	4.8E-02	NC	NC	6.4E-05	2.4E-05
Th-227	2.5E+00	2.9E-01	4.8E+00	6.3E-06	2.8E-06
Th-228	5.4E+01	5.8E-01	9.9E+00	9.6E-08	4.5E-08
Th-230	5.1E+01	9.9E-02	1.7E+00	1.5E-08	7.7E-09
Th-231	1.4E-04	NC	NC	4.5E-07	2.3E-07
Th-232	2.6E+02	8.7E-02	1.5E+00	5.8E-09	3.4E-09
Pa-231	2.0E+02	6.4E-02	1.9E+00	1.9E-06	8.2E-07
U-232	1.0E+02	1.0E-02	2.1E+00	9.0E-09	5.2E-09
Pa-233	1.5E-03	NC	NC	1.3E-05	5.8E-06
U-233	2.1E+01	NC	NC	1.1E-08	5.3E-09
U-234	2.1E+01	9.3E-03	1.7E+00	4.1E-09	2.7E-09
U-235	1.9E+01	8.4E-03	1.6E+00	9.0E-06	4.0E-06
U-236	2.0E+01	8.7E-03	1.7E+00	2.5E-09	2.0E-09
U-238	1.9E+01	8.4E-03	1.6E+00	2.1E-09	1.7E-09
Np-237	8.5E+01	6.4E-02	1.5E+00	9.9E-07	4.8E-07
Pu-236	2.3E+01	NC	NC	1.6E-09	1.9E-09
Pu-238	6.1E+01	7.5E-02	2.0E+00	7.9E-10	1.4E-09
Np-239	3.9E-04	5.2E-05	1.4E-03	9.4E-06	4.2E-06
Pu-239	6.7E+01	7.0E-02	1.9E+00	2.9E-09	1.7E-09
Pu-240	6.7E+01	7.0E-02	1.9E+00	8.1E-10	1.3E-09
Pu-241	1.3E+00	1.4E-05	3.7E-04	0.0	0.0
Pu-242	6.4E+01	7.0E-02	1.8E+00	7.4E-10	1.1E-09
Am-241	7.0E+01	7.5E-02	1.7E+00	5.5E-07	3.2E-07
Am-242m	6.7E+01	NC	NC	1.2E-08	8.8E-09
Am-243	6.9E+01	7.6E-02	1.7E+00	1.9E-06	9.4E-07
Cm-242	2.7E+00	8.1E-02	1.8E+00	8.5E-10	1.6E-09
Cm-243	4.8E+01	8.1E-02	1.9E+00	7.2E-06	3.2E-06
Cm-244	3.9E+01	8.1E-02	1.9E+00	6.2E-10	1.3E-09
Cm-245	7.1E+01	NC	NC	3.6E-06	1.6E-06
Cf-252	2.5E+01	1.5E+00	1.9E+01	8.0E-10	1.2E-09

NOTES:

1 ICRP No. 23 Reference Man with breathing rate of 1.2 cu m per hour (light activity).

2 Acute dose is the comitted dose for 30 days.

3 Cloud shine is the dose from immersion in a semi-infinite cloud.

4 CEDE is Comitted Effective Dose Equivalent.

* = 1st hour
NC = not calculated

5 Whole Body and Red Bone Marrow doses are considered approximately equal.

6 Dose from short-lived daughter products are included.

Formulas used are same as those on Worksheet H-1, Page H-41.

* = 1st hour
NC = not calculated

Table H-2-F

FIRE INVOLVING 1 CURIE
<1/4 MILE DOWN WIND DOSE

$\frac{\text{rem}}{\text{Ci in Fire}}$

Radio-nuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
H-3(H2O)	5.0E-01	5.0E-06	4.6E-06	4.6E-06	0.0	0.0
C-14(ORG)	1.0E-02	3.3E-06	1.3E-06	1.3E-06	0.0	0.0
Na-22	1.0E-02	1.2E-05	NC	NC	1.5E-06	5.6E-07
Na-24	1.0E-02	1.9E-06	1.2E-06	7.0E-06	3.2E-06	9.7E-07
P-32	5.0E-01	1.2E-03	1.3E-03	6.4E-03	0.0	0.0
P-33	5.0E-01	1.8E-04	NC	NC	0.0	0.0
S-35	5.0E-01	1.9E-04	1.2E-05	7.0E-04	0.0	0.0
Cl-36	5.0E-01	1.7E-03	NC	NC	5.9E-15	1.3E-14
K-40	5.0E-01	9.7E-04	NC	NC	5.4E-06	1.8E-06
K-42	1.0E-02	2.1E-06	NC	NC	1.9E-07	6.5E-08
Ca-45	1.0E-02	1.0E-05	NC	NC	3.9E-19	2.1E-18
Sc-46	1.0E-02	4.6E-05	NC	NC	1.4E-06	5.2E-07
Ti-44	1.0E-02	7.1E-04	NC	NC	4.9E-08	2.5E-08
V-48	1.0E-02	1.6E-05	NC	NC	2.0E-06	7.4E-07
Cr-51	1.0E-02	5.2E-07	1.0E-07	1.7E-06	2.1E-08	9.0E-09
Mn-54	1.0E-02	1.0E-05	4.5E-06	1.4E-05	5.8E-07	2.3E-07
Mn-56	1.0E-02	5.9E-07	NC	NC	1.2E-06	4.4E-07
Fe-55	1.0E-02	4.2E-06	9.3E-08	1.7E-06	3.2E-13	1.2E-12
Co-58	1.0E-03	1.7E-06	2.0E-07	2.6E-06	6.7E-08	2.6E-08
Fe-59	1.0E-02	2.3E-05	7.0E-06	4.6E-05	8.2E-07	2.9E-07
Co-60	1.0E-03	3.4E-05	5.1E-07	7.5E-06	1.7E-07	6.1E-08
Ni-63	1.0E-02	4.9E-06	1.8E-07	5.4E-06	0.0	0.0
Cu-64	1.0E-02	4.3E-07	NC	NC	1.3E-07	5.2E-08
Zn-65	1.0E-02	3.2E-05	2.7E-06	1.4E-05	4.0E-07	1.5E-07
Ga-68	NC	NC	NC	NC	NC	NC
Ge-68+Ga-68[6]	1.0E-02	8.1E-05	0.0	0.0	6.3E-07	2.6E-07
Se-75	1.0E-02	1.3E-05	3.0E-06	1.3E-05	2.4E-07	1.1E-07
Kr-85	1.0E+00	NC	NC	NC	1.5E-07	6.1E-08
Kr-85m	1.0E+00	NC	NC	NC	9.6E-06	4.3E-06
Kr-87	1.0E+00	NC	NC	NC	6.0E-05	2.0E-05
Kr-88+Rb-88[6]	1.0E+00	NC	NC	NC	2.0E-04	6.3E-05
Rb-86	1.0E-02	1.0E-05	1.0E-05	1.7E-05	6.6E-08	2.4E-08
Rb-87	1.0E-02	5.1E-06	NC	NC	0.0	0.0
Rb-88	1.0E-02	1.3E-07	NC	NC	4.8E-07	1.6E-07
Sr-89	1.0E-02	6.5E-05	1.3E-05	1.8E-04	9.5E-11	3.7E-11
Sr-90	1.0E-02	2.0E-03	2.7E-05	3.7E-04	0.0	0.0
Sr-91	1.0E-02	2.6E-06	NC	NC	4.8E-07	1.8E-07
Y-90	1.0E-02	1.3E-05	1.6E-06	5.4E-05	0.0	0.0
Y-91	1.0E-02	7.7E-05	7.5E-06	1.9E-04	2.5E-09	8.9E-10

* = 1st hour
NC = not calculated

Table H-2-F Continued

FIRE INVOLVING 1 CURIE
<1/4 MILE DOWN WIND DOSE

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Y-91m	1.0E-02	5.7E-08	NC	NC	3.5E-07	1.5E-07
Zr-93	1.0E-02	5.0E-04	NC	NC	0.0	0.0
Zr-95	1.0E-02	3.7E-05	1.6E-05	5.5E-05	5.0E-07	2.0E-07
Nb-94	1.0E-02	6.5E-04	NC	NC	1.1E-06	4.3E-07
Nb-95	1.0E-02	9.1E-06	2.3E-06	2.3E-05	5.3E-07	2.1E-07
Mo-99	1.0E-02	6.2E-06	2.2E-06	2.5E-05	1.0E-07	2.1E-07
Tc-99	1.0E-02	1.3E-05	2.3E-07	3.2E-05	1.0E-07	4.2E-08
Tc-99m	1.0E-02	5.1E-08	1.8E-08	1.8E-07	2.[illegible]-13	1.1E-13
Rh-103	NC	NC	NC	NC	NC	NC
Ru-103	1.0E-02	1.4E-05	2.3E-06	4.0E-05	3.2E-07	1.3E-07
Ru-105	1.0E-02	7.1E-07	NC	NC	5.3E-07	2.2E-07
Rh-106	NC	NC	NC	NC	NC	NC
Ru-106+Rh-106[6]	1.0E-02	7.5E-04	1.3E-05	5.2E-04	1.4E-07	5.6E-08
Ag-110m	1.0E-02	1.3E-04	9.3E-06	7.5E-05	1.9E-06	7.2E-07
Cd-109+Ag-109m[6]	1.0E-02	1.8E-04	NC	NC	1.6E-09	1.1E-09
Cd-113m	1.0E-02	2.4E-03	NC	NC	0.0	0.0
In-114m	1.0E-02	1.4E-04	NC	NC	5.8E-08	2.4E-08
Sn-113	1.0E-02	1.7E-05	NC	NC	3.5E-09	1.8E-09
Sn-123	1.0E-02	5.1E-05	NC	NC	4.8E-09	1.8E-09
Sn-126+Sb-126m[6]	1.0E-02	1.6E-04	NC	NC	1.9E-06	7.7E-07
Sb-124	1.0E-02	3.9E-05	7.5E-06	1.3E-04	1.3E-06	4.8E-07
Sb-126	1.0E-02	1.8E-05	NC	NC	1.9E-06	7.6E-07
Sb-126m	1.0E-02	5.3E-08	0.0	0.0	1.1E-06	4.3E-07
Sb-127	1.0E-02	9.5E-06	NC	NC	4.5E-07	1.8E-07
Sb-129	1.0E-02	1.0E-06	NC	NC	9.9E-07	3.8E-07
Te-127	1.0E-02	5.0E-07	NC	NC	3.2E-09	1.4E-09
Te-127m	1.0E-02	3.4E-05	1.4E-05	8.7E-05	4.5E-10	3.6E-10
Te-129	1.0E-02	1.4E-07	NC	NC	3.5E-08	1.5E-08
Te-129m+Te-129[6]	1.0E-02	3.8E-05	2.4E-05	1.5E-04	4.6E-08	1.8E-08
Te-131	1.0E-02	7.5E-07	NC	NC	2.7E-07	1.1E-07
Te-131m	1.0E-02	1.0E-05	1.3E-06	1.3E-05	9.8E-07	3.7E-07
Te-132	1.0E-02	1.5E-05	1.6E-06	8.1E-06	1.3E-07	5.9E-08
I-125	5.0E-01	1.9E-03	4.9E-06	2.9E-05	6.6E-08	6.0E-08
I-129	5.0E-01	1.4E-02	5.5E-06	6.4E-05	5.8E-08	6.0E-08
I-131	5.0E-01	2.6E-03	1.6E-05	1.9E-04	1.3E-05	5.4E-06
I-132	5.0E-01	3.0E-05	4.1E-06	7.8E-05	7.8E-05	3.1E-05
I-133	5.0E-01	4.6E-04	7.8E-06	2.4E-04	2.0E-05	8.3E-06
I-134	5.0E-01	1.0E-05	1.8E-06	4.1E-05	9.1E-05	3.5E-05
I-135+Xe-135m[6]	5.0E-01	9.6E-05	6.4E-06	1.3E-04	7.0E-05	2.5E-05

* = 1st hour
NC = not calculated

Table H-2-F Continued

FIRE INVOLVING 1 CURIE
<1/4 MILE DOWN WIND DOSE

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Xe-131m	1.0E+00	NC	NC	NC	2.6E-07	1.6E-07
Xe-133	1.0E+00	NC	NC	NC	1.3E-06	6.7E-07
Xe-133m	1.0E+00	NC	NC	NC	1.6E-06	7.5E-07
Xe-135	1.0E+00	NC	NC	NC	1.6E-05	7.0E-06
Xe-135m	1.0E+00	NC	NC	NC	2.8E-05	1.2E-05
Xe-138	1.0E+00	NC	NC	NC	8.3E-05	2.8E-05
Cs-134	1.0E-02	7.3E-05	1.3E-05	1.6E-05	1.1E-06	4.2E-07
Cs-135	1.0E-02	7.1E-06	1.2E-06	2.4E-06	0.0	0.0
Cs-136	1.0E-02	1.1E-05	8.7E-06	1.2E-05	1.5E-06	5.7E-07
Ba-137m	1.0E-02	NC	NC	NC	4.1E-07	1.6E-07
Cs-137+Ba-137m[6]	1.0E-02	5.0E-05	8.1E-06	1.3E-05	4.1E-07	1.6E-07
Ba-133	1.0E-02	1.2E-05	NC	NC	2.2E-07	1.0E-07
Cs-138	1.0E-02	1.6E-07	NC	NC	1.7E-06	5.8E-07
Ba-140	1.0E-02	5.9E-06	5.8E-06	9.3E-06	1.2E-07	5.2E-08
La-140	NC	NC	NC	NC	NC	NC
Ce-141	1.0E-02	1.4E-05	1.1E-06	4.9E-05	4.2E-08	1.9E-08
Ce-144+Pr-144[6]	1.0E-02	5.9E-04	8.1E-06	4.7E-04	3.3E-08	1.2E-08
Pr-144	NC	NC	NC	NC	NC	NC
Pr-144m	NC	NC	NC	NC	NC	NC
Pm-145	1.0E-02	4.8E-05	NC	NC	3.3E-09	2.2E-09
Pm-147	1.0E-02	6.1E-05	8.7E-07	2.3E-05	1.9E-12	8.5E-13
Sm-147	1.0E-02	1.2E-01	NC	NC	0.0	0.0
Sm-151	1.0E-02	4.7E-05	2.7E-07	6.4E-06	6.4E-14	1.4E-13
Eu-152	1.0E-02	3.5E-04	5.3E-06	6.4E-05	7.7E-07	2.9E-07
Eu-154	1.0E-02	4.5E-04	7.0E-06	1.1E-04	8.5E-07	3.2E-07
Eu-155	1.0E-02	6.5E-05	1.2E-06	2.1E-05	2.4E-08	1.2E-08
Gd-153	1.0E-02	3.7E-05	NC	NC	3.3E-08	1.7E-08
Tb-160	1.0E-02	3.9E-05	NC	NC	7.4E-07	2.8E-07
Ho-166m	1.0E-02	1.2E-03	NC	NC	1.1E-06	4.3E-07
Tm-170	1.0E-02	4.1E-05	NC	NC	1.6E-09	8.6E-10
Yb-169	1.0E-02	1.3E-05	2.9E-06	4.1E-05	1.4E-07	6.6E-08
Hf-172	1.0E-02	5.0E-04	NC	NC	NC	NC
Hf-181	1.0E-02	2.4E-05	NC	NC	3.4E-07	1.5E-07
Ta-182	1.0E-03	7.0E-06	NC	NC	8.6E-08	3.2E-08
W-187	1.0E-02	9.7E-07	NC	NC	3.1E-07	1.3E-07
Ir-192	1.0E-03	4.4E-06	4.8E-07	8.7E-06	5.3E-08	2.3E-08
Au-198	1.0E-02	5.1E-06	NC	NC	2.7E-07	1.1E-07
Hg-203	1.0E-02	1.1E-05	NC	NC	1.4E-07	6.4E-08
Tl-204	1.0E-02	3.8E-06	NC	NC	3.7E-10	1.9E-10

* = 1st hour
NC = not calculated

Table H-2-F Continued

FIRE INVOLVING 1 CURIE
<1/4 MILE DOWN WIND DOSE

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire[6]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Pb-210	1.0E-02	2.1E-02	1.7E-05	6.3E-06	2.9E-10	1.9E-10
Bi-207	1.0E-02	3.1E-05	NC	NC	1.0E-06	4.0E-07
Bi-210	1.0E-02	3.1E-04	NC	NC	0.0	0.0
Po-210	1.0E-02	1.5E-02	4.9E-04	1.6E-02	5.9E-12	2.3E-12
Ra-226	1.0E-03	1.3E-03	5.8E-06	1.5E-03	4.0E-10	1.8E-10
Ac-227	1.0E-03	1.0E+00	NC	NC	6.0E-12	2.8E-12
Ac-228	1.0E-03	4.8E-05	NC	NC	6.4E-08	2.4E-08
Th-227	1.0E-03	2.5E-03	2.9E-04	4.8E-03	6.3E-09	2.8E-09
Th-228	1.0E-03	5.4E-02	5.8E-04	9.9E-03	9.6E-11	4.5E-11
Th-230	1.0E-03	5.1E-02	9.9E-05	1.7E-03	1.5E-11	7.7E-12
Th-231	1.0E-03	1.4E-07	NC	NC	4.5E-10	2.3E-10
Th-232	1.0E-03	2.6E-01	8.7E-05	1.5E-03	5.8E-12	3.4E-12
Pa-231	1.0E-03	2.0E-01	6.4E-05	1.9E-03	1.9E-09	8.2E-10
U-232	1.0E-03	1.0E-01	1.0E-05	2.1E-03	9.0E-12	5.2E-12
Pa-233	1.0E-03	1.5E-06	NC	NC	1.3E-08	5.8E-09
U-233	1.0E-03	2.1E-02	NC	NC	1.1E-11	5.3E-12
U-234	1.0E-03	2.1E-02	9.3E-06	1.7E-03	4.1E-12	2.7E-12
U-235	1.0E-03	1.9E-02	8.4E-06	1.6E-03	9.0E-09	4.0E-09
U-236	1.0E-03	2.0E-02	8.7E-06	1.7E-03	2.5E-12	2.0E-12
U-238	1.0E-03	1.9E-02	8.4E-06	1.6E-03	2.1E-12	1.7E-12
Np-237	1.0E-03	8.5E-02	6.4E-05	1.5E-03	9.9E-10	4.8E-10
Pu-236	1.0E-03	2.3E-02	NC	NC	1.6E-12	1.9E-12
Pu-238	1.0E-03	6.1E-02	7.5E-05	2.0E-03	7.9E-13	1.4E-12
Np-239	1.0E-03	3.9E-07	5.2E-08	1.4E-06	9.4E-09	4.2E-09
Pu-239	1.0E-03	6.7E-02	7.0E-05	1.9E-03	2.9E-12	1.7E-12
Pu-240	1.0E-03	6.7E-02	7.0E-05	1.9E-03	8.1E-13	1.3E-12
Pu-241	1.0E-03	1.3E-03	1.4E-08	3.7E-07	0.0	0.0
Pu-242	1.0E-03	6.4E-02	7.0E-05	1.8E-03	7.4E-13	1.1E-12
Am-241	1.0E-03	7.0E-02	7.5E-05	1.7E-03	5.5E-10	3.2E-10
Am-242m	1.0E-03	6.7E-02	NC	NC	1.2E-11	8.8E-12
Am-243	1.0E-03	6.9E-02	7.6E-05	1.7E-03	1.9E-09	9.4E-10
Cm-242	1.0E-03	2.7E-03	8.1E-05	1.8E-03	8.5E-13	1.6E-12
Cm-243	1.0E-03	4.8E-02	8.1E-05	1.9E-03	7.2E-09	3.2E-09
Cm-244	1.0E-03	3.9E-02	8.1E-05	1.9E-03	6.2E-13	1.3E-12
Cm-245	1.0E-03	7.1E-02	NC	NC	3.6E-09	1.6E-09
Cf-252	1.0E-03	2.5E-02	1.5E-03	1.9E-02	8.0E-13	1.2E-12

* = 1st hour
NC = not calculated

NOTES:

1 ICRP No. 23 Reference Man with breathing rate of 1.2 cu m per hour (light activity).

2 Acute dosc is the comitted dose for 30 days.

3 Cloud shine is the dose from immersion in a semi-infinite cloud.

4 CEDE is Comitted Effective Dose Equivalent.

5 Whole Body and Red Bone Marrow doses are considered approximately equal.

6 Dose from short-lived daughter products are included.

7 Fraction = (Ci released)/(Ci in fire) -From NUREG-1140.

Formulas used are the same as those on Worksheet H-1, page H-41.

* = 1st hour
NC = not calculated

Table H-3

1 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

$\frac{\text{rem}}{\text{Ci release}}$

Radio-nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
H-3(H2O)	5.4E-07	5.0E-07	5.0E-07	0.0	0.0
C-14(ORG)	1.8E-05	7.2E-06	7.2E-06	0.0	0.0
Na-22	6.5E-05	NC	NC	8.0E-06	3.1E-06
Na-24	1.0E-05	6.6E-06	3.7E-05	1.7E-05	5.3E-06
P-32	1.3E-04	1.4E-04	6.9E-04	0.0	0.0
P-33	2.0E-05	NC	NC	0.0	0.0
S-35	2.1E-05	1.2E-06	7.5E-05	0.0	0.0
Cl-36	1.9E-04	NC	NC	6.3E-16	1.4E-15
K-40	1.0E-04	NC	NC	5.8E-07	2.0E-07
K-42	1.1E-05	NC	NC	1.0E-06	3.6E-07
Ca-45	5.6E-05	NC	NC	2.1E-18	1.1E-17
Sc-46	2.5E-04	NC	NC	7.5E-06	2.9E-06
Ti-44	3.8E-03	NC	NC	2.6E-07	1.4E-07
V-48	8.6E-05	NC	NC	1.1E-05	4.1E-06
Cr-51	2.8E-06	5.6E-07	9.4E-06	1.1E-07	4.9E-08
Mn-54	5.6E-05	2.4E-05	7.5E-05	3.1E-06	1.2E-06
Mn-56	3.2E-06	NC	NC	6.7E-06	2.4E-06
Fe-55	2.3E-05	5.0E-07	9.4E-06	1.7E-12	6.8E-12
Co-58	9.2E-05	1.1E-05	1.4E-04	3.6E-06	1.5E-06
Fe-59	1.2E-04	3.7E-05	2.5E-04	4.4E-06	1.6E-06
Co-60	1.8E-03	2.7E-05	4.1E-04	9.3E-06	3.4E-06
Ni-63	2.6E-05	9.7E-07	2.9E-05	0.0	0.0
Cu-64	2.3E-06	NC	NC	6.8E-07	2.9E-07
Zn-65	1.7E-04	1.5E-05	7.5E-05	2.2E-06	8.1E-07
Ga-68	1.2E-06	1.8E-07	6.6E-06	3.4E-06	1.4E-06
Ge-68+Ga-68[6]	4.4E-04	0.0	0.0	3.4E-06	1.4E-06
Se-75	7.1E-05	1.6E-05	6.9E-05	1.3E-06	5.9E-07
Kr-85	NC	NC	NC	8.0E-09	3.4E-09
Kr-85m	NC	NC	NC	5.2E-07	2.3E-07
Kr-87	NC	NC	NC	3.2E-06	1.1E-06
Kr-88+Rb-88[6]	NC	NC	NC	1.1E-05	3.5E-06
Rb-86	5.6E-05	5.6E-05	9.4E-05	3.5E-07	1.3E-07
Rb-87	2.7E-05	NC	NC	0.0	0.0
Rb-88	7.1E-07	NC	NC	2.6E-06	8.6E-07
Sr-89	3.5E-04	7.2E-05	9.7E-04	5.1E-10	2.0E-10
Sr-90	1.1E-02	1.5E-04	2.0E-03	0.0	0.0
Sr-91	1.4E-05	NC	NC	2.6E-06	1.0E-06
Y-90	7.1E-05	8.7E-06	2.9E-04	0.0	0.0
Y-91	4.1E-04	4.1E-05	1.0E-03	1.3E-08	4.9E-09

* = 1st hour
NC = not calculated

Table H-3 Continued

1 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

rem / Ci release

Radio-nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Y-91m	3.1E-07	NC	NC	1.9E-06	8.0E-07
Zr-93	2.7E-03	NC	NC	0.0	0.0
Zr-95	2.0E-04	8.4E-05	2.9E-04	2.7E-06	1.1E-06
Nb-94	3.5E-03	NC	NC	5.9E-06	2.3E-06
Nb-95	4.9E-05	1.2E-05	1.2E-04	2.8E-06	1.1E-06
Mo-99	3.3E-05	1.2E-05	1.3E-04	5.6E-07	2.3E-07
Tc-99	7.0E-05	1.2E-06	1.7E-04	1.2E-12	6.0E-13
Tc-99m	2.7E-07	9.7E-08	9.7E-07	3.9E-07	1.8E-07
Rh-103	NC	NC	NC	NC	NC
Ru-103	7.6E-05	1.2E-05	2.2E-04	1.7E-06	7.3E-07
Ru-105	3.8E-06	NC	NC	2.9E-06	1.2E-06
Rh-106	NC	NC	NC	7.5E-07	3.1E-07
Ru-106+Rh-106[6]	4.0E-03	7.2E-05	2.8E-03	7.5E-07	3.1E-07
Ag-110m	6.8E-04	5.0E-05	4.1E-04	1.0E-05	4.0E-06
Cd-109+Ag-109m[6]	9.6E-04	NC	NC	9.1E-09	6.3E-09
Cd-113m	1.3E-02	NC	NC	0.0	0.0
In-114m	7.5E-04	NC	NC	3.1E-07	1.3E-07
Sn-113	9.0E-05	NC	NC	1.9E-08	9.7E-09
Sn-123	2.7E-04	NC	[illegible]	2.6E-08	9.7E-09
Sn-126+Sb-126m[6]	8.4E-04	NC	[illegible]	1.0E-05	4.2E-06
Sb-124	2.1E-04	4.1E-05	6.9E-04	7.2E-06	2.6E-06
Sb-126	9.9E-05	NC	NC	1.0E-05	4.2E-06
Sb-126m	2.9E-07	0.0	0.0	5.8E-06	2.4E-06
Sb-127	5.1E-05	NC	NC	2.4E-06	1.0E-06
Sb-129	5.4E-06	NC	NC	5.4E-06	2.1E-06
Te-127	2.7E-06	NC	NC	1.7E-08	7.5E-09
Te-127m	1.8E-04	7.5E-05	4.7E-04	2.5E-09	2.0E-09
Te-129	7.6E-07	NC	NC	1.9E-07	8.1E-08
Te-129m+Te-129[6]	2.0E-04	1.3E-04	8.1E-04	2.5E-07	1.0E-07
Te-131	4.0E-06	NC	NC	1.5E-06	6.1E-07
Te-131m	5.4E-05	7.2E-06	6.9E-05	5.3E-06	2.1E-06
Te-132	8.0E-05	8.7E-06	4.4E-05	7.1E-07	3.2E-07
I-125	2.0E-04	5.3E-07	3.1E-06	7.1E-09	6.6E-09
I-129	1.5E-03	5.9E-07	6.9E-06	6.3E-09	6.6E-09
I-131	2.8E-04	1.7E-06	2.0E-05	1.4E-06	5.9E-07
I-132	3.2E-06	4.4E-07	8.4E-06	8.5E-06	3.4E-06
I-133	4.9E-05	8.4E-07	2.6E-05	2.2E-06	9.1E-07
I-134	1.1E-06	1.9E-07	4.4E-06	9.8E-06	3.8E-06
I-135+Xe-135m[6]	1.0E-05	6.9E-07	1.4E-05	7.5E-06	2.8E-06

* = 1st hour
NC = not calculated

Table H-3 Continued

1 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

$\frac{\text{rem}}{\text{Ci release}}$

	Inhalation1				
Radio-nuclide	CEDE[4]	Acute[2] Bone Marrow	Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Xe-131m	NC	NC	NC	1.4E-08	8.9E-09
Xe-133	NC	NC	NC	6.8E-08	3.7E-08
Xe-133m	NC	NC	NC	8.6E-08	4.1E-08
Xe-135	NC	NC	NC	8.6E-07	3.8E-07
Xe-135m	NC	NC	NC	1.5E-06	6.5E-07
Xe-138	NC	NC	NC	4.5E-06	1.6E-06
Cs-134	3.9E-04	7.2E-05	8.7E-05	5.7E-06	2.3E-06
Cs-135	3.8E-05	6.6E-06	1.3E-05	0.0	0.0
Cs-136	6.2E-05	4.7E-05	6.2E-05	8.0E-06	3.1E-06
Ba-137m	NC	NC	NC	2.2E-06	9.0E-07
Cs-137+Ba-137m[6]	2.7E-04	4.4E-05	6.9E-05	2.2E-06	9.0E-07
Ba-133	6.6E-05	NC	NC	1.2E-06	5.5E-07
Cs-138	8.5E-07	NC	NC	9.2E-06	3.2E-06
Ba-140	3.2E-05	3.1E-05	5.0E-05	6.6E-07	2.8E-07
La-140	4.1E-05	1.4E-05	1.3E-04	8.9E-06	3.2E-06
Ce-141	7.6E-05	5.9E-06	2.6E-04	2.3E-07	1.0E-07
Ce-144+Pr-144[6]	3.2E-03	4.4E-05	2.5E-03	1.8E-07	6.7E-08
Pr-144	3.7E-07	NC	NC	1.3E-07	4.4E-08
Pr-144m	NC	NC	NC	4.8E-09	3.7E-09
Pm-145	2.6E-04	NC	NC	1.8E-08	1.2E-08
Pm-147	3.3E-04	4.7E-06	1.2E-04	1.0E-11	4.7E-12
Sm-147	6.3E-01	NC	NC	0.0	0.0
Sm-151	2.5E-04	1.5E-06	3.4E-05	3.4E-13	7.7E-13
Eu-152	1.9E-03	2.8E-05	3.4E-04	4.2E-06	1.6E-06
Eu-154	2.4E-03	3.7E-05	5.9E-04	4.6E-06	1.7E-06
Eu-155	3.5E-04	6.6E-06	1.2E-04	1.3E-07	6.4E-08
Gd-153	2.0E-04	NC	NC	1.8E-07	9.5E-08
Tb-160	2.1E-04	NC	NC	4.0E-06	1.5E-06
Ho-166m	6.5E-03	NC	NC	5.8E-06	2.4E-06
Tm-170	2.2E-04	NC	NC	8.8E-09	4.8E-09
Yb-169	6.8E-05	1.6E-05	2.2E-04	7.5E-07	3.6E-07
Hf-172	2.7E-03	NC	NC	NC	NC
Hf-181	1.3E-04	NC	NC	1.9E-06	8.1E-07
Ta-182	3.8E-04	NC	NC	4.7E-06	1.7E-06
W-187	5.2E-06	NC	NC	1.7E-06	7.0E-07
Ir-192	2.4E-04	2.6E-05	4.7E-04	2.9E-06	1.3E-06
Au-198	2.8E-05	NC	NC	1.4E-06	6.3E-07
Hg-203	6.2E-05	NC	NC	7.8E-07	3.5E-07
Tl-204	2.0E-05	NC	NC	2.0E-09	1.0E-09

* = 1st hour
NC = not calculated

Table H-3 Continued

1 MILE DOWN WIND ADULT DOSE FOR 1 CURIE RELEASE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

rem / Ci release

Radio-nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Pb-210	1.1E-01	9.4E-05	3.4E-05	1.5E-09	1.1E-09
Bi-207	1.7E-04	NC	NC	5.6E-06	2.2E-06
Bi-210	1.7E-03	NC	NC	0.0	0.0
Po-210	7.9E-02	2.7E-03	8.4E-02	3.2E-11	1.3E-11
Ra-226	7.2E-02	3.1E-04	8.1E-02	2.2E-08	9.9E-09
Ac-227	5.6E+01	NC	NC	3.2E-10	1.6E-10
Ac-228	2.6E-03	NC	NC	3.4E-06	1.3E-06
Th-227	1.4E-01	1.6E-02	2.6E-01	3.4E-07	1.6E-07
Th-228	2.9E+00	3.1E-02	5.3E-01	5.2E-09	2.5E-09
Th-230	2.7E+00	5.3E-03	9.4E-02	8.2E-10	4.3E-10
Th-231	7.4E-06	NC	NC	2.5E-08	1.3E-08
Th-232	1.4E+01	4.7E-03	8.1E-02	3.1E-10	1.9E-10
Pa-231	1.1E+01	3.4E-03	1.0E-01	1.0E-07	4.5E-08
U-232	5.6E+00	5.6E-04	1.1E-01	4.9E-10	2.9E-10
Pa-233	8.0E-05	NC	NC	7.1E-07	3.2E-07
U-233	1.1E+00	NC	NC	6.0E-10	2.9E-10
U-234	1.1E+00	5.0E-04	9.4E-02	2.2E-10	1.5E-10
U-235	1.0E+00	4.5E-04	8.8E-02	4.8E-07	2.2E-07
U-236	1.1E+00	4.7E-04	9.1E-02	1.4E-10	1.1E-10
U-238	1.0E+00	4.5E-04	8.5E-02	1.1E-10	9.2E-11
Np-237	4.6E+00	3.4E-03	8.1E-02	5.4E-08	2.7E-08
Pu-236	1.2E+00	NC	NC	8.4E-11	1.0E-10
Pu-238	3.3E+00	4.1E-03	1.1E-01	4.2E-11	7.4E-11
Np-239	2.1E-05	2.8E-06	7.5E-05	5.1E-07	2.3E-07
Pu-239	3.6E+00	3.7E-03	1.0E-01	1.6E-10	9.2E-11
Pu-240	3.6E+00	3.7E-03	1.0E-01	4.4E-11	7.3E-11
Pu-241	7.0E-02	7.5E-07	2.0E-05	0.0	0.0
Pu-242	3.5E+00	3.7E-03	9.7E-02	4.0E-11	6.2E-11
Am-241	3.7E+00	4.1E-03	9.4E-02	3.0E-08	1.8E-08
Am-242m	3.6E+00	NC	NC	6.2E-10	4.9E-10
Am-243	3.7E+00	4.1E-03	9.1E-02	1.0E-07	5.1E-08
Cm-242	1.5E-01	4.4E-03	9.7E-02	4.6E-11	8.6E-11
Cm-243	2.6E+00	4.4E-03	1.0E-01	3.9E-07	1.8E-07
Cm-244	2.1E+00	4.4E-03	1.0E-01	3.3E-11	7.3E-11
Cm-245	3.8E+00	NC	NC	1.9E-07	9.1E-08
Cf-252	1.3E+00	8.1E-02	1.0E+00	4.3E-11	6.8E-11

* = 1st hour
NC = not calculated

NOTES:

1 ICRP No. 23 Reference Man with breathing rate of 1.2 cu m per hour (light activity).

2 Acute dose is the comitted dose for 30 days.

3 Cloud shine is the dose from immersion in a semi-infinite cloud.

4 CEDE is Comitted Effective Dose Equivalent.

5 Whole Body and Red Bone Marrow doses are considered approximately equal.

6 Dose from short-lived daughter products are included.

7 Fraction = (Ci released)/(Ci in fire) -From NUREG-1140.

Formulas used are the same as those on Worksheet H-1, page H-41.

* = 1st hour
NC = not calculated

Table H-3-F

FIRE INVOLVING 1 CURIE
1 MILE DOWN WIND DOSE
AVERAGE MET. CONDITIONS (D STABILITY 4 MPH WIND SPEED)

$\frac{\text{rem}}{\text{Ci in Fire}}$

Radionuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
H-3(H2O)	5.0E-01	2.7E-07	2.5E-07	2.5E-07	0.0	0.0
C-14(ORG)	1.0E-02	1.8E-07	7.2E-08	7.2E-08	0.0	0.0
Na-22	1.0E-02	6.5E-07	NC	NC	8.0E-08	3.1E-08
Na-24	1.0E-02	1.0E-07	6.6E-08	3.7E-07	1.7E-07	5.3E-08
P-32	5.0E-01	6.5E-05	7.2E-05	3.4E-04	0.0	0.0
P-33	5.0E-01	9.8E-06	NC	NC	0.0	0.0
S-35	5.0E-01	1.0E-05	6.2E-07	3.7E-05	0.0	0.0
Cl-36	5.0E-01	9.3E-05	NC	NC	3.2E-16	7.2E-16
K-40	5.0E-01	5.2E-05	NC	NC	2.9E-07	1.0E-07
K-42	1.0E-02	1.1E-07	NC	NC	1.0E-08	3.6E-09
Ca-45	1.0E-02	5.6E-07	NC	NC	2.1E-20	1.1E-19
Sc-46	1.0E-02	2.5E-06	NC	NC	7 5E-08	2.9E-08
Ti-44	1.0E-02	3.8E-05	NC	NC	2.6E-09	1.4E-09
V-48	1.0E-02	8.6E-07	NC	NC	1.1E-07	4.1E-08
Cr-51	1.0E-02	2.8E-08	5.6E-09	9.4E-08	1.1E-09	4.9E-10
Mn-54	1.0E-02	5.6E-07	2.4E-07	7.5E-07	3.1E-08	1.2E-08
Mn-56	1.0E-02	3.2E-08	NC	NC	6.7E-08	2.4E-08
Fe-55	1.0E-02	2.3E-07	5.0E-09	9.4E-08	1.7E-14	6.8E-14
Co-58	1.0E-03	9.2E-08	1.1E-08	1.4E-07	3.6E-09	1.5E-09
Fe-59	1.0E-02	1.2E-06	3.7E-07	2.5E-06	4.4E-08	1.6E-08
Co-60	1.0E-03	1.8E-06	2.7E-08	4.1E-07	9.3E-09	3.4E-09
Ni-63	1.0E-02	2.6E-07	9.7E-09	2.9E-07	0.0	0.0
Cu-64	1.0E-02	2.3E-08	NC	NC	6.8E-09	2.9E-09
Zn-65	1.0E-02	1.7E-06	1.5E-07	7.5E-07	2.2E-08	8.1E-09
Ga-68	NC	NC	NC	NC	NC	NC
Ge-68+Ga-68[6]	1.0E-02	4.4E-06	0.0	0.0	3.4E-08	1.4E-08
Se-75	1.0E-02	7.1E-07	1.6E-07	6.9E-07	1.3E-08	5.9E-09
Kr-85	1.0E+00	NC	NC	NC	8.0E-09	3.4E-09
Kr-85m	1.0E+00	NC	NC	NC	5.2E-07	2.3E-07
Kr-87	1.0E+00	NC	NC	NC	3.2E-06	1.1E-06
Kr-88+Rb-88[6]	1.0E+00	NC	NC	NC	1.1E-05	3.5E-06
Rb-86	1.0E-02	5.6E-07	5.6E-07	9.4E-07	3.5E-09	1.3E-09
Rb-87	1.0E-02	2.7E-07	NC	NC	0.0	0.0
Rb-88	1.0E-02	7.1E-09	NC	NC	2.6E-08	8.6E-09
Sr-89	1.0E-02	3.5E-06	7.2E-07	9.7E-06	5.1E-12	2.0E-12
Sr-90	1.0E-02	1.1E-04	1.5E-06	2.0E-05	0.0	0.0
Sr-91	1.0E-02	1.4E-07	NC	NC	2.6E-08	1.0E-08

* = 1st hour
NC = not calculated

Table H-3-F Continued

FIRE INVOLVING 1 CURIE
1 MILE DOWN WIND DOSE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Y-90	1.0E-02	7.1E-07	8.7E-08	2.9E-06	0.0	0.0
Y-91	1.0E-02	4.1E-06	4.1E-07	1.0E-05	1.3E-10	4.9E-11
Y-91m	1.0E-02	3.1E-09	NC	NC	1.9E-08	8.0E-09
Zr-93	1.0E-02	2.7E-05	NC	NC	0.0	0.0
Zr-95	1.0E-02	2.0E-06	8.4E-07	2.9E-06	2.7E-08	1.1E-08
Nb-94	1.0E-02	3.5E-05	NC	NC	5.9E-08	2.3E-08
Nb-95	1.0E-02	4.9E-07	1.2E-07	1.2E-06	2.8E-08	1.1E-08
Mo-99	1.0E-02	3.3E-07	1.2E-07	1.3E-06	5.6E-09	2.3E-09
Tc-99	1.0E-02	7.0E-07	1.2E-08	1.7E-06	1.2E-14	6.0E-15
Tc-99m	1.0E-02	2.7E-09	9.7E-10	9.7E-09	3.9E-09	1.8E-09
Rh-103	NC	NC	NC	NC	NC	NC
Ru-103	1.0E-02	7.6E-07	1.2E-07	2.2E-06	1.7E-08	7.3E-09
Ru-105	1.0E-02	3.8E-08	NC	NC	2.9E-08	1.2E-08
Rh-106	NC	NC	NC	NC	NC	NC
Ru-106+Rh-106[6]	1.0E-02	4.0E-05	7.2E-07	2.8E-05	7.5E-09	3.1E-09
Ag-110m	1.0E-02	6.8E-06	5.0E-07	4.1E-06	1.0E-07	4.0E-08
Cd-109+Ag-109m[6]	1.0E-02	9.6E-06	NC	NC	9.1E-11	6.3E-11
Cd-113m	1.0E-02	1.3E-04	NC	NC	0.0	0.0
In-114m	1.0E-02	7.5E-06	NC	NC	3.1E-09	1.3E-09
Sn-113	1.0E-02	9.0E-07	NC	NC	1.9E-10	9.7E-11
Sn-123	1.0E-02	2.7E-06	NC	NC	2.6E-10	9.7E-11
Sn-126+Sb-126m[6]	1.0E-02	8.4E-06	NC	NC	1.0E-07	4.2E-08
Sb-124	1.0E-02	2.1E-06	4.1E-07	6.9E-06	7.2E-08	2.6E-08
Sb-126	1.0E-02	9.9E-07	NC	NC	1.0E-07	4.2E-08
Sb-126m	1.0E-02	2.9E-09	0.0	0.0	5.8E-08	2.4E-08
Sb-127	1.0E-02	5.1E-07	NC	NC	2.4E-08	1.0E-08
Sb-129	1.0E-02	5.4E-08	NC	NC	5.4E-08	2.1E-08
Te-127	1.0E-02	2.7E-08	NC	NC	1.7E-10	7.5E-11
Te-127m	1.0E-02	1.8E-06	7.5E-07	4.7E-06	2.5E-11	2.0E-11
Te-129	1.0E-02	7.6E-09	NC	NC	1.9E-09	8.1E-10
Te-129m+Te-129[6]	1.0E-02	2.0E-06	1.3E-06	8.1E-06	2.5E-09	1.0E-09
Te-131	1.0E-02	4.0E-08	NC	NC	1.5E-08	6.1E-09
Te-131m	1.0E-02	5.4E-07	7.2E-08	6.9E-07	5.3E-08	2.1E-08
Te-132	1.0E-02	8.0E-07	8.7E-08	4.4E-07	7.1E-09	3.2E-09
I-125	5.0E-01	1.0E-04	2.7E-07	1.6E-06	3.6E-09	3.3E-09
I-129	5.0E-01	7.3E-04	3.0E-07	3.4E-06	3.1E-09	3.3E-09
I-131	5.0E-01	1.4E-04	8.4E-07	1.0E-05	6.8E-07	2.9E-07

* = 1st hour
NC = not calculated

Table H-3-F Continued

FIRE INVOLVING 1 CURIE
1 MILE DOWN WIND DOSE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
I-132	5.0E-01	1.6E-06	2.2E-07	4.2E-06	4.2E-06	1.7E-06
I-133	5.0E-01	2.5E-05	4.2E-07	1.3E-05	1.1E-06	4.5E-07
I-134	5.0E-01	5.5E-07	9.5E-08	2.2E-06	4.9E-06	1.9E-06
I-135+Xe-135m[6]	5.0E-01	5.2E-06	3.4E-07	6.9E-06	3.8E-06	1.4E-06
Xe-131m	1.0E+00	NC	NC	NC	1.4E-08	8.9E-09
Xe-133	1.0E+00	NC	NC	NC	6.8E-08	3.7E-08
Xe-133m	1.0E+00	NC	NC	NC	8.6E-08	4.1E-08
Xe-135	1.0E+00	NC	NC	NC	8.6E-07	3.8E-07
Xe-135m	1.0E+00	NC	NC	NC	1.5E-06	6.5E-07
Xe-138	1.0E+00	NC	NC	NC	4.5E-06	1.6E-06
Cs-134	1.0E-02	3.9E-06	7.2E-07	8.7E-07	5.7E-08	2.3E-08
Cs-135	1.0E-02	3.8E-07	6.6E-08	1.3E-07	0.0	0.0
Cs-136	1.0E-02	6.2E-07	4.7E-07	6.2E-07	8.0E-08	3.1E-08
Ba-137m	1.0E-02	NC	NC	NC	2.2E-08	9.0E-09
Cs-137+Ba-137m[6]	1.0E-02	2.7E-06	4.4E-07	6.9E-07	2.2E-08	9.0E-09
Ba-133	1.0E-02	6.6E-07	NC	NC	1.2E-08	5.5E-09
Cs-138	1.0E-02	8.5E-09	NC	NC	9.2E-08	3.2E-08
Ba-140	1.0E-02	3.2E-07	3.1E-07	5.0E-07	6.6E-09	2.8E-09
La-140	NC	NC	NC	NC	NC	NC
Ce-141	1.0E-02	7.6E-07	5.9E-08	2.6E-06	2.3E-09	1.0E-09
Ce-144+Pr-144[6]	1.0E-02	3.2E-05	4.4E-07	2.5E-05	1.8E-09	6.7E-10
Pr-144	NC	NC	NC	NC	NC	NC
Pr-144m	NC	NC	NC	NC	NC	NC
Pm-145	1.0E-02	2.6E-06	NC	NC	1.8E-10	1.2E-10
Pm-147	1.0E-02	3.3E-06	4.7E-08	1.2E-06	1.0E-13	4.7E-14
Sm-147	1.0E-02	6.3E-03	NC	NC	0.0	0.0
Sm-151	1.0E-02	2.5E-06	1.5E-08	3.4E-07	3.4E-15	7.7E-15
Eu-152	1.0E-02	1.9E-05	2.8E-07	3.4E-06	4.2E-08	1.6E-08
Eu-154	1.0E-02	2.4E-05	3.7E-07	5.9E-06	4.6E-08	1.7E-08
Eu-155	1.0E-02	3.5E-06	6.6E-08	1.2E-06	1.3E-09	6.4E-10
Gd-153	1.0E-02	2.0E-06	NC	NC	1.8E-09	9.5E-10
Tb-160	1.0E-02	2.1E-06	NC	NC	4.0E-08	1.5E-08
Ho-166m	1.0E-02	6.5E-05	NC	NC	5.8E-08	2.4E-08
Tm-170	1.0E-02	2.2E-06	NC	NC	8.8E-11	4.8E-11
Yb-169	1.0E-02	6.8E-07	1.6E-07	2.2E-06	7.5E-09	3.6E-09
Hf-172	1.0E-02	2.7E-05	NC	NC	NC	NC
Hf-181	1.0E-02	1.3E-06	NC	NC	1.9E-08	8.1E-09

* = 1st hour
NC = not calculated

Table H-3-F Continued

FIRE INVOLVING 1 CURIE
1 MILE DOWN WIND DOSE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire[7]	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
Ta-182	1.0E-03	3.8E-07	NC	NC	4.7E-09	1.7E-09
W-187	1.0E-02	5.2E-08	NC	NC	1.7E-08	7.0E-09
Ir-192	1.0E-03	2.4E-07	2.6E-08	4.7E-07	2.9E-09	1.3E-09
Au-198	1.0E-02	2.8E-07	NC	NC	1.4E-08	6.3E-09
Hg-203	1.0E-02	6.2E-07	NC	NC	7.8E-09	3.5E-09
Tl-204	1.0E-02	2.0E-07	NC	NC	2.0E-11	1.0E-11
Pb-210	1.0E-02	1.1E-03	9.4E-07	3.4E-07	1.5E-11	1.1E-11
Bi-207	1.0E-02	1.7E-06	NC	NC	5.6E-08	2.2E-08
Bi-210	1.0E-02	1.7E-05	NC	NC	0.0	0.0
Po-210	1.0E-02	7.9E-04	2.7E-05	8.4E-04	3.2E-13	1.3E-13
Ra-226	1.0E-03	7.2E-05	3.1E-07	8.1E-05	2.2E-11	9.9E-12
Ac-227	1.0E-03	5.6E-02	NC	NC	3.2E-13	1.6E-13
Ac-228	1.0E-03	2.6E-06	NC	NC	3.4E-09	1.3E-09
Th-227	1.0E-03	1.4E-04	1.6E-05	2.6E-04	3.4E-10	1.6E-10
Th-228	1.0E-03	2.9E-03	3.1E-05	5.3E-04	5.2E-12	2.5E-12
Th-230	1.0E-03	2.7E-03	5.3E-06	9.4E-05	8.2E-13	4.3E-13
Th-231	1.0E-03	7.4E-09	NC	NC	2.5E-11	1.3E-11
Th-232	1.0E-03	1.4E-02	4.7E-06	8.1E-05	3.1E-13	1.9E-13
Pa-231	1.0E-03	1.1E-02	3.4E-06	1.0E-04	1.0E-10	4.5E-11
U-232	1.0E-03	5.6E-03	5.6E-07	1.1E-04	4.9E-13	2.9E-13
Pa-233	1.0E-03	8.0E-08	NC	NC	7.1E-10	3.2E-10
U-233	1.0E-03	1.1E-03	NC	NC	6.0E-13	2.9E-13
U-234	1.0E-03	1.1E-03	5.0E-07	9.4E-05	2.2E-13	1.5E-13
U-235	1.0E-03	1.0E-03	4.5E-07	8.8E-05	4.8E-10	2.2E-10
U-236	1.0E-03	1.1E-03	4.7E-07	9.1E-05	1.4E-13	1.1E-13
U-238	1.0E-03	1.0E-03	4.5E-07	8.5E-05	1.1E-13	9.2E-14
Np-237	1.0E-03	4.6E-03	3.4E-06	8.1E-05	5.4E-11	2.7E-11
Pu-236	1.0E-03	1.2E-03	NC	NC	8.4E-14	1.0E-13
Pu-238	1.0E-03	3.3E-03	4.1E-06	1.1E-04	4.2E-14	7.4E-14
Np-239	1.0E-03	2.1E-08	2.8E-09	7.5E-08	5.1E-10	2.3E-10
Pu-239	1.0E-03	3.6E-03	3.7E-06	1.0E-04	1.6E-13	9.2E-14
Pu-240	1.0E-03	3.6E-03	3.7E-06	1.0E-04	4.4E-14	7.3E-14
Pu-241	1.0E-03	7.0E-05	7.5E-10	2.0E-08	0.0	0.0
Pu-242	1.0E-03	3.5E-03	3.7E-06	9.7E-05	4.0E-14	6.2E-14
Am-241	1.0E-03	3.7E-03	4.1E-06	9.4E-05	3.0E-11	1.8E-11
Am-242m	1.0E-03	3.6E-03	NC	NC	6.2E-13	4.9E-13
Am-243	1.0E-03	3.7E-03	4.1E-06	9.1E-05	1.0E-10	5.1E-11

* = 1st hour
NC = not calculated

Table H-3-F Continued

FIRE INVOLVING 1 CURIE
1 MILE DOWN WIND DOSE
AVERAGE MET. CONDITIONS (D STABILITY, 4 MPH WIND SPEED)

rem / Ci in Fire

Radio-nuclide	Fraction Released By a Fire	Inhalation[1]			Cloud Shine[3] W.Body[5]	Ground Shine W.Body*
		CEDE[4]	Acute[2] Bone Marrow	Acute[2] Lung		
Cm-242	1.0E-03	1.5E-04	4.4E-06	9.7E-05	4.6E-14	8.6E-14
Cm-243	1.0E-03	2.6E-03	4.4E-06	1.0E-04	3.9E-10	1.8E-10
Cm-244	1.0E-03	2.1E-03	4.4E-06	1.0E-04	3.3E-14	7.3E-14
Cm-245	1.0E-03	3.8E-03	NC	NC	1.9E-10	9.1E-11
Cf-252	1.0E-03	1.3E-03	8.1E-05	1.0E-03	4.3E-14	6.8E-14

NOTES:

1 ICRP No. 23 Reference Man with breathing rate of 1.2 cu m per hour (light activity).

2 Acute dose is the comitted dose for 30 days.

3 Cloud shine is the dose from immersion in a semi-infinite cloud.

4 CEDE is Comitted Effective Dose Equivalent.

5 Whole Body and Red Bone Marrow doses are considered approximately equal.

6 Dose from short-lived daughter products are included.

7 Fraction = (Ci released)/(Ci in fire) -From NUREG-1140.

Formulas used are the same as those on Worksheet H-1, page H-41.

* = 1st hour
NC = not calculated

Table H-4

DILUTION FACTORS
$(Xu/Q)m^{-2}$

Miles	Stability Class A	Stability Class B	Stability Class C	Stability Class D	Stability Class E	Stability Class F
≤.25[1,2]	1.0 E-3	1.0 E-3	1.0 E-3	1.0 E-3	1.0 E-3	1.0 E-3
0.50[1]	6.0 E-6	2.7 E-5	6.0 E-5	1.7 E-4	3.5 E-4	8.0 E-4
1	1.0 E-6	6.0 E-6	1.7 E-5	5.4 E-5	1.1 E-4	2.5 E-4
2	7.0 E-7	1.5 E-6	5.0 E-6	2.0 E-5	4.0 E-5	1.0 E-4
3	4.5 E-7	6.5 E-7	2.2 E-6	1.2 E-5	2.5 E-5	6.0 E-5
4	3.5 E-7	4.5 E-7	1.2 E-6	8.0 E-6	1.6 E-5	4.0 E-5
5	3.0 E-7	4.0 E-7	9.5 E-7	5.0 E-6	1.1 E-5	3.0 E-5
7[1]	2.2 E-7	2.6 E-7	5.2 E-7	3.0 E-6	8.0 E-6	2.0 E-5
10[1]	1.7 E-7	2.2 E-7	3.0 E-7	2.0 E-6	5.0 E-6	1.2 E-5
12[1]	1.4 E-7	1.8 E-7	2.5 E-7	1.5 E-6	3.5 E-6	9.0 E-6
15[1]	1.2 E-7	1.5 E-7	2.0 E-7	1.0 E-6	2.6 E-6	7.0 E-6
20[1]	9.5 E-8	1.1 E-7	1.7 E-7	7.0 E-7	2.0 E-6	5.0 E-6
25[1]	8.0 E-8	9.0 E-8	1.3 E-7	4.5 E-7	1.4 E-6	4.0 E-6
30[1]	6.4 E-8	8.0 E-8	1.0 E-7	3.5 E-7	1.1 E-6	3.5 E-6
35[1]	5.8 E-8	7.0 E-8	9.5 E-8	3.0 E-7	9.0 E-7	3.0 E-6
40[1]	5.0 E-8	6.5 E-8	8.5 E-8	2.5 E-7	8.0 E-7	2.5 E-6
50[1]	4.4 E-8	6.0 E-8	7.1 E-8	1.8 E-7	6.0 E-7	2.0 E-6
60[1]	4.0 E-8	5.0 E-8	6.1 E-8	1.5 E-7	5.0 E-7	1.6 E-6

NOTES:
[1] These factors are very unreliable.
[2] This considers building wake effects but still are considered very unreliable.

Source:
- "Workbook of Atmospheric Dispersion Estimates," TURNER 1970 and Regulatory Guide 1.145.
- 1.0E-3 for <1/4 mile assumes building wake dominates and is based on NUREG/CR-5055.

Table H-5

ADULT DOSE FACTORS

$\frac{rem/hr}{Ci/m3}$

Radio-Nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow[5]	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]
H-3(H2O)	6.4E+01	5.9E+01	5.9E+01	0.0
C-14(ORG)	2.1E+03	8.5E+02	8.5E+02	0.0
Na-22	7.7E+03	NC	NC	1.1E+03
Na-24	1.2E+03	7.8E+02	4.4E+03	2.5E+03
P-32	1.6E+04	1.7E+04	8.1E+04	0.0
P-33	2.3E+03	NC	NC	0.0
S-35	2.5E+03	1.5E+02	8.9E+03	0.0
Cl-36	2.2E+04	NC	NC	9.0E-08
K-40	1.2E+04	NC	NC	8.3E+01
K-42	1.4E+03	NC	NC	1.5E+02
Ca-45	6.6E+03	NC	NC	3.0E-10
Sc-46	3.0E+04	NC	NC	1.1E+03
Ti-44	4.5E+05	NC	NC	3.8E+01
V-48	1.0E+04	NC	NC	1.6E+03
Cr-51	3.3E+02	6.7E+01	1.1E+03	1.6E+01
Mn-54	6.7E+03	2.8E+03	8.9E+03	4.5E+02
Mn-56	3.8E+02	NC	NC	9.6E+02
Fe-55	2.7E+03	5.9E+01	1.1E+03	2.4E-04
Co-58	1.1E+04	1.3E+03	1.7E+04	5.2E+02
Fe-59	1.5E+04	4.4E+03	2.9E+04	6.3E+02
Co-60	2.2E+05	3.3E+03	4.8E+04	1.3E+03
Ni-63	3.1E+0[illegible]	1.1E+02	3.4E+03	0.0
Cu-64	2.8E+02	NC	NC	9.7E+01
Zn-65	2.0E+04	1.7E+03	8.9E+03	3.1E+02
Ga-68	1.4E+02	2.1E+01	7.8E+02	4.8E+02
Ge-68+Ga-68	5.2E+04	NC	NC	4.8E+02[6]
Se-75	8.5E+03	1.9E+03	8.1E+03	1.9E+02
Kr-85	NC	NC	NC	1.1E+00
Kr-85m	NC	NC	NC	7.4E+01
Kr-87	NC	NC	NC	4.6E+02
Kr-88+Rb-88	NC	NC	NC	1.5E+03[6]
Rb-86	6.6E+03	6.7E+03	1.1E+04	5.1E+01
Rb-87	3.2E+03	NC	NC	0.0
Rb-88	8.4E+01	NC	NC	3.7E+02
Sr-89	4.1E+04	8.5E+03	1.1E+05	7.3E-02
Sr-90	1.3E+06	1.7E+04	2.3E+05	0.0
Sr-91	1.7E+03	NC	NC	3.7E+02
Y-90	8.4E+03	1.0E+03	3.4E+04	0.0
Y-91	4.9E+04	4.8E+03	1.2E+05	1.9E+00
Y-91m	3.6E+01	NC	NC	2.7E+02

NC = not calculated

Table H-5 Continued

ADULT DOSE FACTORS

rem/hr
Ci/m3

Radio-Nuclide	Inhalation[1] CEDE[4]	Inhalation[1] Acute[2] Bone Marrow[5]	Inhalation[1] Acute[2] Lung	Cloud Shine[3] W.Body[5]
Zr-93	3.2E+05	NC	NC	0.0
Zr-95	2.4E+04	1.0E+04	3.5E+04	3.9E+02
Nb-94	4.1E+05	NC	NC	8.4E+02
Nb-95	5.8E+03	1.5E+03	1.5E+04	4.1E+02
Mo-99	4.0E+03	1.4E+03	1.6E+04	8.0E+01
Tc-99	8.3E+03	1.4E+02	2.0E+04	1.7E-04
Tc-99m	3.3E+01	1.1E+01	1.1E+02	5.6E+01
Ru-103	9.0E+03	1.5E+03	2.6E+04	2.5E+02
Ru-105	4.6E+02	NC	NC	4.1E+02
Rh-106	NC	NC	NC	1.1E+02
Ru-106+Rh-106	4.8E+05	8.5E+03	3.3E+05	1.1E+02[6]
Ag-110m	8.0E+04	5.9E+03	4.8E+04	1.4E+03
Cd-109+Ag-109m	1.1E+05	NC	NC	1.3E-00[6]
Cd-113m	1.5E+06	NC	NC	0.0
In-114m	8.9E+04	NC	NC	4.5E+01
Sn-113	1.1E+04	NC	NC	2.7E+00
Sn-123	3.3E+04	NC	NC	3.7E+00
Sn-126+Sb-126m	1.0E+05	NC	NC	1.4E+03[6]
Sb-124	2.5E+04	4.8E+03	8.1E+04	1.0E+03
Sb-126	1.2E+04	NC	NC	1.4E+03
Sb-126m	3.4E+01	NC	NC	8.2E+02
Sb-127	6.0E+03	NC	NC	3.4E+02
Sb-129	6.4E+02	NC	NC	7.7E+02
Te-127	3.2E+02	NC	NC	2.4E+00
Te-127m	2.1E+04	8.9E+03	5.5E+04	3.5E-01
Te-129	9.0E+01	NC	NC	2.7E+01
Te-129m+Te-129	2.4E+04	1.5E+04	9.6E+04	3.5E+01[6]
Te-131	4.8E+02	NC	NC	2.1E+02
Te-131m	6.4E+03	8.5E+02	8.1E+03	7.5E+02
Te-132	9.4E+03	1.0E+03	5.2E+03	1.0E+02
I-125	2.4E+04	6.3E+01	3.7E+02	1.0E+00
I-129	1.7E+05	7.0E+01	8.1E+02	8.9E-01
I-131	3.3E+04	2.0E+02	2.4E+03	1.9E+02
I-132	3.8E+02	5.2E+01	1.0E+03	1.2E+03
I-133	5.8E+03	1.0E+02	3.0E+03	3.1E+02
I-134	1.3E+02	2.3E+01	5.2E+02	1.4E+03
I-135+Xe-135m	1.2E+03	8.1E+01	1.6E+03	1.1E+03[6]
Xe-131m	NC	NC	NC	2.0E+00
Xe-133	NC	NC	NC	9.7E+00
Xe-133m	NC	NC	NC	1.2E+01

NC = not calculated

Table H-5 Continued

ADULT DOSE FACTORS

rem/hr
Ci/m3

Radio-Nuclide	Inhalation[1] CEDE[4]	Acute[2] Bone Marrow[5]	Acute[2] Lung	Cloud Shine[3] W.Body[5]
Xe-135	NC	NC	NC	1.2E+02
Xe-135m	NC	NC	NC	2.2E+02
Xe-138	NC	NC	NC	6.4E+02
Cs-134	4.6E+04	8.5E+03	1.0E+04	8.2E+02
Cs-135	4.6E+03	7.8E+02	1.5E+03	0.0
Cs-136	7.3E+03	5.6E+03	7.4E+03	1.1E+03
Ba-137m	NC	NC	NC	3.1E+02
Cs-137+Ba-137m	3.2E+04	5.2E+03	8.1E+03	3.1E+02[6]
Ba-133	7.8E+03	NC	NC	1.7E+02
Cs-138	1.0E+02	NC	NC	1.3E+03
Ba-140	3.7E+03	3.7E+03	5.9E+03	9.4E+01
La-140	4.8E+03	1.7E+03	1.6E+04	1.3E+03
Ce-141	9.0E+03	7.0E+02	3.1E+04	3.2E+01
Ce-144+Pr-144	3.7E+05	5.2E+03	3.0E+05	2.5E+01[6]
Pr-144	4.3E+01	NC	NC	1.8E+01
Pr-144m	NC	NC	NC	6.9E-01
Pm-145	3.0E+04	NC	NC	2.5E+00
Pm-147	3.9E+04	5.6E+02	1.4E+04	1.4E-03
Sm-147	7.5E+07	NC	NC	0.0
Sm-151	3.0E+04	1.7E+02	4.1E+03	4.9E-05
Eu-152	2.2E+05	3.4E+03	4.1E+04	5.9E+02
Eu-154	2.9E+05	4.4E+03	7.0E+04	6.6E+02
Eu-155	4.1E+04	7.8E+02	1.4E+04	1.8E+01
Gd-153	2.4E+04	NC	NC	2.5E+01
Tb-160	2.5E+04	NC	NC	5.7E+02
Ho-166m	7.7E+05	NC	NC	8.2E+02
Tm-170	2.6E+04	NC	NC	1.3E+00
Yb-169	8.1E+03	1.9E+03	2.6E+04	1.1E+02
Hf-172	3.2E+05	NC	NC	NC
Hf-181	1.5E+04	NC	NC	2.7E+02
Ta-182	4.5E+04	NC	NC	6.6E+02
W-187	6.2E+02	NC	NC	2.4E+02
Ir-192	2.8E+04	3.0E+03	5.5E+04	4.1E+02
Au-198	3.3E+03	NC	NC	2.1E+02
Hg-203	7.3E+03	NC	NC	1.1E+02
Tl-204	2.4E+03	NC	NC	2.8E-01
Pb-210	1.4E+07	1.1E+04	4.0E+03	2.2E-01
Bi-207	2.0E+04	NC	NC	8.0E+02
Bi-210	2.0E+05	NC	NC	0.0
Po-210	9.4E+06	3.1E+05	1.0E+07	4.5E-03

NC = not calculated

Table H-5 Continued

ADULT DOSE FACTORS

$\frac{rem/hr}{Ci/m3}$

Radio-Nuclide	Inhalation[1] CEDE[4]	Acute[2] Bone Marrow[5]	Acute[2] Lung	Cloud Shine[3] W.Body[5]
Ra-226	8.6E+06	3.7E+04	9.6E+06	3.1E+00
Ac-227	6.7E+09	NC	NC	4.6E-02
Ac-228	3.1E+05	NC	NC	4.9E+02
Th-227	1.6E+07	1.9E+06	3.1E+07	4.9E+01
Th-228	3.4E+08	3.7E+06	6.3E+07	7.4E-01
Th-230	3.3E+08	6.3E+05	1.1E+07	1.2E-01
Th-231	8.8E+02	NC	NC	3.5E+00
Th-232	1.6E+09	5.6E+05	9.6E+06	4.5E-02
Pa-231	1.3E+09	4.1E+05	1.2E+07	1.4E+01
U-232	6.6E+08	6.7E+04	1.3E+07	6.9E-02
Pa-233	9.5E+03	NC	NC	1.0E+02
U-233	1.4E+08	NC	NC	8.5E-02
U-234	1.3E+08	5.9E+04	1.1E+07	3.1E-02
U-235	1.2E+08	5.3E+04	1.0E+07	6.9E+01
U-236	1.3E+08	5.6E+04	1.1E+07	2.0E-02
U-238	1.2E+08	5.3E+04	1.0E+07	1.6E-02
Np-237	5.4E+08	4.1E+05	9.6E+06	7.6E+00
Pu-236	1.4E+08	NC	NC	1.2E-02
Pu-238	3.9E+08	4.8E+05	1.3E+07	6.1E-03
Np-239	2.5E+03	3.3E+02	8.9E+03	7.2E+01
Pu-239	4.3E+08	4.4E+05	1.2E+07	2.2E-02
Pu-240	4.3E+08	4.4E+05	1.2E+07	6.2E-03
Pu-241	8.3E+06	8.9E+01	2.4E+03	0.0
Pu-242	4.1E+08	4.4E+05	1.1E+07	5.7E-03
Am-241	4.4E+08	4.8E+05	1.1E+07	4.3E+00
Am-242m	4.3E+08	NC	NC	8.9E-02
Am-243	4.4E+08	4.8E+05	1.1E+07	1.4E+01
Cm-242	1.7E+07	5.2E+05	1.1E+07	6.5E-03
Cm-243	3.1E+08	5.2E+05	1.2E+07	5.6E+01
Cm-244	2.5E+08	5.2E+05	1.2E+07	4.8E-03
Cm-245	4.6E+08	NC	NC	2.7E+01
Cf-252	1.6E+08	9.6E+06	1.2E+08	6.2E-03

NOTES:

1 ICRP No. 23 Reference Man with breathing rate of 1.2 cu m per hour (light activity). Effects of daughter products are included in inhalation dose.

2 Acute dose is the comitted dose for 30 days.

3 Cloud shine is the dose from immersion in a semi-infinite cloud.

4 CEDE is Comitted Effective Dose Equivalent.

NC = not calculated

[5] Whole Body and Red Bone Marrow doses are considered approximately equal.

[6] Dose from short-lived daughter products are included.

Source: Immersion - DOE/EH-0070 "External Dose Rate Conversion Factors for Calculation of Dose to the Public", July 1988

Inhalation - <u>Inhalation Dose (ICRP-30)</u>, "Doses From Intake of Radionuclide by Adults and Young People", National Radiological Protection Board, NRPB-R162, 1985

NC = not calculated

Table H-6

GROUND CONCENTRATION FACTORS; GROUND LEVEL RELEASE; NO RAIN; AVERAGE MET. CONDITIONS (D STABILITY, WIND SPEED 2 M/S) FOR A 1 CURIE RELEASE

$\frac{Ci/m2}{Ci\ released}$

MILES	Ci/m^2
.5	7.8 E-07
1	2.9 E-07
2	1.1 E-07
3	7.1 E-08
4	4.0 E-08
5	2.9 E-08
6	2.2 E-08
7	1.9 E-08
8	1.4 E-08
9	1.1 E-08
10	9.7 E-09
15	5.9 E-09
20	3.9 E-09
25	2.8 E-09

Source: RASCAL 1.3a runs.

Table H-7

GROUND SHINE
DOSE FACTORS[1]
AT 1 METER, GROUND ROUGHNESS CONSIDERED
rem/hr
Ci/m2

Nuclide	Factor	Nuclide	Factor
H-3(H2O)	0.0	Y-91	1.7E-02
C-14(ORG)	0.0	Y-91m	2.8E+00
Na-22	1.1E+01	Zr-93	0.0
Na-24	1.8E+01	Zr-95	3.8E+00
P-32	0.0	Nb-94	8.1E+00
P-33	0.0	Nb-95	4.0E+00
S-35	0.0	Mo-99	8.0E-01
Cl-36	5.0E-09	Tc-99	2.1E-06
K-40	7.0E-01	Tc-99m	6.2E-01
K-42	1.2E+00	Ru-103	2.5E+00
Ca-45	4.0E-11	Ru-105	4.1E+00
Sc-46	9.9E+00	Rh-106	1.1E+00
Ti-44	4.7E-01	Ru-106+Rh-106[2]	1.1E+00
V-48	1.4E+01	Ag-110m	1.4E+01
Cr-51	1.7E-01	Cd-109+Ag-109m[2]	2.2E-02
Mn-54	4.3E+00	Cd-113m	0.0
Mn-56	8.3E+00	In-114m	4.6E-01
Fe-55	2.4E-05	Sn-113	3.4E-02
Co-58	5.0E+00	Sn-123	3.3E-02
Fe-59	5.6E+00	Sn-126+Sb-126m[2]	1.5E+01
Co-60	1.2E+01	Sb-124	9.1E+00
Ni-63	0.0	Sb-126	1.4E+01
Cu-64	9.9E-01	Sb-126m	8.2E+00
Zn-65	2.8E+00	Sb-127	3.5E+00
Ga-68	5.0E+00	Sb-129	7.2E+00
Ge-68+Ga-68[2]	5.0E+00	Te-127	2.6E-02
Se-75	2.0E+00	Te-127m	6.9E-03
Kr-85	1.2E-02	Te-129	2.8E-01
Kr-85m	8.1E-01	Te-129m+Te-129[2]	3.5E-01
Kr-87	3.8E+00	Te-131	2.1E+00
Kr-88+Rb-88[2]	1.2E+01	Te-131m	7.1E+00
Rb-86	4.6E-01	Te-132	1.1E+00
Rb-87	0.0	I-125	2.3E-02
Rb-88	3.0E+00	I-129	2.3E-02
Sr-89	7.0E-04	I-131	2.0E+00
Sr-90	0.0	I-132	1.2E+01
Sr-91	3.5E+00	I-133	3.1E+00
Y-90	0.0	I-134	1.3E+01

Table H-7 - Continued

GROUND SHINE
DOSE FACTORS[1]
AT 1 METER, GROUND ROUGHNESS CONSIDERED

$\frac{rem/hr}{Ci/m2}$

Nuclide	Factor	Nuclide	Factor
I-135+Xe-135m[2]	9.6E+00	Tl-204	3.5E-03
Xe-131m	3.1E-02	Pb-210	3.7E-03
Xe-133	1.3E-01	Bi-207	7.6E+00
Xe-133m	1.4E-01	Bi-210	0.0
Xe-135	1.3E+00	Po-210	4.4E-05
Xe-135m	2.2E+00	Ra-226	3.4E-02
Xe-138	5.4E+00	Ac-227	5.4E-04
Cs-134	8.0E+00	Ac-228	4.6E+00
Cs-135	0.0	Th-227	5.4E-01
Cs-136	1.1E+01	Th-228	8.6E-03
Ba-137m	3.1E+00	Th-230	1.5E-03
Cs-137+Ba-137m[2]	3.1E+00	Th-231	4.3E-02
Ba-133	1.9E+00	Th-232	6.4E-04
Cs-138	1.1E+01	Pa-231	1.6E-01
Ba-140	9.8E-01	U-232	9.9E-04
La-140	1.1E+01	Pa-233	1.1E+00
Ce-141	3.6E-01	U-233	1.0E-03
Ce-144+Pr-144[2]	2.3E-01	U-234	5.2E-04
Pr-144	1.5E-01	U-235	7.6E-01
Pr-144m	1.3E-02	U-236	3.7E-04
Pm-145	4.3E-02	U-238	3.2E-04
Pm-147	1.6E-05	Np-237	9.2E-02
Sm-147	0.0	Pu-236	3.6E-04
Sm-151	2.7E-06	Pu-238	2.6E-04
Eu-152	5.5E+00	Np-239	8.0E-01
Eu-154	6.0E+00	Pu-239	3.2E-04
Eu-155	2.2E-01	Pu-240	2.5E-04
Gd-153	3.3E-01	Pu-241	0.0
Tb-160	5.3E+00	Pu-242	2.2E-04
Ho-166m	8.2E+00	Am-241	6.2E-02
Tm-170	1.6E-02	Am-242m	1.7E-03
Yb-169	1.3E+00	Am-243	1.8E-01
Hf-172	NC	Cm-242	3.0E-04
Hf-181	2.8E+00	Cm-243	6.2E-01
Ta-182	6.0E+00	Cm-244	2.5E-04
W-187	2.4E+00	Cm-245	3.1E-01
Ir-192	4.4E+00	Cf-252	2.3E-04
Au-198	2.2E+00		

Dose factors from sources are multiplied by 1/2 to consider ground surface.

Dose from short-lived daughter products are included.

urce: DOE/EH-0070 "External Dose Rate Conversion Factors for Calculation of Dose to the Public", July 1988

Table H-8

REPRESENTATIVE SHIELDING FACTORS FROM GAMMA CLOUD SOURCE

Structure or Location	Shielding Factor[a]	Representative Range
Outside	1.0	--
Vehicles	1.0	--
Wood-Frame House[b] (no basement)	0.9	--
Basement of Wood House	0.6	0.1 to 0.7[c]
Masonry House (no basement)	0.6	0.4 to 0.7[c]
Basement of Masonry House	0.4	0.1 to 0.5[c]
Large Office or Industrial Building	0.2	0.1 to 0.3[c,d]

[a] The ratio of the interior dose to the exterior dose.
[b] A wood frame house with brick or stone veneer is approximately equivalent to a masonry house for shielding purposes.
[c] This range is mainly due to different wall materials and different geometries.
[d] The reduction factor depends on where the personnel are located within the building (e.g., the basement or an inside room).

Source: NUREG-1062 and EGG-1183-?670, December 1975.

Table H-9

REPRESENTATIVE SHIELDING FACTORS FOR SURFACE DEPOSITION

Structure or Location	Representative Shielding Factor[a]
Cars on fully decontaminated 50-ft road	.25
Trains	.4
One- and two-story wood-frame house (no basement)	.4[b]
One- and two-story block and brick house (no basement)	.2[b]
House basement, one or two walls fully exposed:	.1[b]
One story, less than 2 ft of basement, walls exposed	.05[b]
Two stories, less than 2 ft of basement, walls exposed	.03[b]
Three- or four-story structures, 5000 to 10,000 sq. ft. per floor:	
First and second floors	.05[b]
Basement	.01[b]
Multi-story structures, >10,000 sq. ft. per floor:	
Upper floors	.01[b]
Basement	.005[b]

[a] The ratio of the interior dose to the exterior dose.
[b] Away from doors and windows.

Source: NUREG-1062 and EGG-1183-1670, December 1975

Table H-10

EARLY HEALTH EFFECTS OF EXPOSURE TO RADIATION

Exposure Rate (r/hr)	Dose Threshold[e] (rad)	Organ	Health Effects
>6	50	Whole Body[a]	Vomiting and diarrhea might occur
>6	150	Whole Body[a]	50% vomiting and diarrhea occurs
>6	> 3000	Thyroid	Hypothyroidism
1000	700	Lung	Deaths might occur
100	2000	Lung	Deaths might occur
50	4000	Lung	Deaths might occur
10	16000	Lung	Deaths might occur
5	31000	Lung	Deaths might occur

			Type of Treatment	
			Minimal[b]	Supportive[c]
1000	150	Bone Marrow (WB[a])	Deaths might occur[d]	
1000	230	Bone Marrow (WB[a])		Deaths might occur[d]
5	220	Bone Marrow (WB[a])	Deaths might occur[d]	
5	330	Bone Marrow (WB[a])		Deaths might occur[d]
5	440	Bone Marrow (WB[a])	50% deaths	
5	660	Bone Marrow (WB[a])		50% deaths
1	500	Bone Marrow (WB[a])	Deaths might occur[d]	
1	750	Bone Marrow (WB[a])		Deaths might occur[d]
1	1000	Bone Marrow (WB[a])	50% deaths	
1	1500	Bone Marrow (WB[a])		50% deaths

[a] Whole Body calculated elsewhere in this manual is approximately equal to Bone Marrow dose.
[b] Minimal treatment is no medical treatment beyond basic first aid.
[c] Supportive treatment is the maximum medical treatment available, but does not include bone marrow transplant.
[d] This is a threshold. Actual occurrences at this level unlikely.
[e] The threshold for effects are total absorbed dose and rate dependent.

Source: NUREG/CR-4214, Rev. 1, 4/89

Table H-11

THYROID DOSES
≤ ¼ MILE DOWN WIND ACUTE ADULT DOSES

	Inhalation	
	Maximum Release	Fire Initiated Release
Nuclide	rem / Ci released	rem[1] / Ci in fire
I-125	4.8E-02	2.4E-02
I-129	1.4E-01	7.0E-02
I-131	1.5E-01	7.1E-02
I-132	9 8E-04	4.9E-04
I-133	2.7E-02	1.4E-02
I-134	1.6E-04	8.0E-05
I-135	4.6E-03	2.3E-03

THYROID DOSES
1 MILE DOWN WIND ACUTE ADULT DOSES - AVERAGE MET. CONDITIONS
(D STABILITY, WIND SPEED 2 M/S)

	Inhalation	
	Maximum Release	Fire Initiated Release
Nuclide	rem / Ci released	rem[1] / Ci in fire
I-125	2.6E-03	1.3E-03
I-129	7.6E-03	3.8E-03
J-131	8.3E-03	4.2E-03
I-132	5.3E-05	2.7E-05
I-133	1.5E-03	7.5E-04
I-134	8.6E-06	4.3E-06
1-135	2.5E-04	1.3E-04

THYROID DOSE FACTORS
ACUTE ADULT INHALATION
rem/hr / Ci/m3

Nuclide	Inhalation
I-125	3.5E+05
I-129	1.0E+06
I-131	1.1E+06
I-132	7.0E+03
I-133	1.9E+05
I-134	1.1E+03
1-135	3.3E+04

[1] Assumes a .5 release fraction.

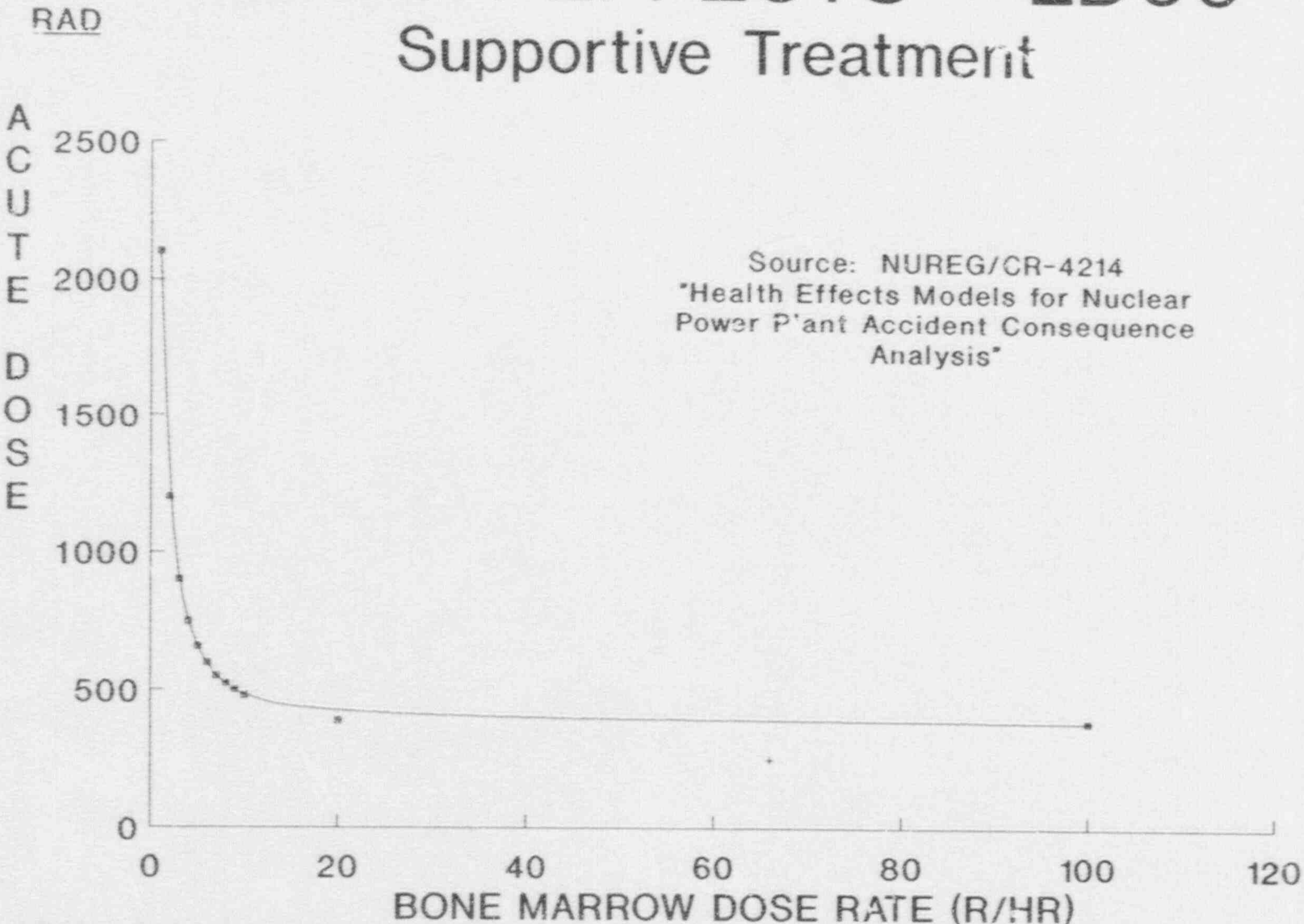
Table H-11 - Continued

Source: Immersion - DOE/EH-0070 "External Dose Rate Conversion Factors for Calculation of Dose to the Public", July 1988

Inhalation - Inhalation Dose (ICRP-30), "Doses From Intake of Radionuclide by Adults and Young People", National Radiological Protection Board, NRPB-R162, 1985

HEALTH EFFECTS - LD50[1]

Supportive Treatment



1 LD50 = 50% of people die within 30-60 days.

THIS PAGE INTENTIONALLY LEFT BLANK

WORKSHEET H-1
DOSE BASED ON ISOTOPIC RELEASE RATE

IMMERSION:

(Release Rate)* (Ci/s)	④ (xu/Q) (m^{-2})	⑪ (Thyroid) ⑤ (Dose Factor) $\frac{(rem)(m^3)}{(Ci)(hr)}$	(Release Duration) (hr)

$$\frac{(\quad) \times (\quad) \times (\quad) \times (\quad)}{(\quad)\ (m/s)\ (\text{Wind Speed})} = \quad (rem)$$

INHALATION:

(Release Rate)* (Ci/s)	④ (xu/Q) (m^{-2})	⑪ (Thyroid) ⑤ (Dose Factor) $\frac{(rem)(m^3)}{(Ci)(hr)}$	(Release Duration) (hr)

$$\frac{(\quad) \times (\quad) \times (\quad) \times (\quad)}{(\quad)\ (m/s)\ (\text{Wind Speed})} = \quad (rem)$$

GROUND:

(Ci released)* (Ci)	⑥ (Ground Concentration Factor) $\frac{(Ci/m^2)}{Ci\ released}$	⑦ (Dose Factor) $\frac{(rem)(m^2)}{(Ci)(hr)}$	(Exposure Duration) (hr)

( ) × ( ) × ( ) × ( )

= (rem)

NOTE: See other side for conversion chart

○ = *referenced Table No.*

* = Fire Release Rate = $\frac{\text{Ci Available} \times \text{Release Fraction By Fire (page H-11)}}{\text{Release Duration(s)}}$

WIND SPEED CONVERSION

MPH	M/S
1	0.5
2	0.9
3	1.3
4	1.8
5	2.2
6	2.7
7	3.1
8	3.6
9	4.0
10	4.5
11	4.9
12	5.4
13	5.8
14	6.3
15	6.7
16	7.2
17	7.6
18	8.0
19	8.5
20	9.0

METRIC TO ENGLISH

Centimeters	x 0.3937	= Inches
Meters	x 3.2808	= Feet
Kilometers	x 0.6214	= Miles
Square meters	x 10.764	= Square feet
Cubic meters	x 35.315	= Cubic feet

ENGLISH TO METRIC

Inches	x 2.54	= Centimeters
Feet	x 0.3048	= Meters
Miles	x 1.609	= Kilometers
Square feet	x 0.093	= Square meters
Cubic feet	x .028	= Cubic meters

WORKSHEET H-2
DOSE BASED ON AIR AND GROUND CONCENTRATIONS

IMMERSION:

(11) *(Thyroid)
(6)

(Air Concentration) x (Dose Factor) x (Time)

(Ci/m^3) $\frac{(rem)(m^3)}{(Ci)(hr)}$ (hr)

( ) x ( ) x ( ) = (rem)

INHALATION:

(11) *(Thyroid)
(6)

(Air Concentration) x (Dose Factor) x (Time)

(Ci/m^3) $\frac{(rem)(m^3)}{(Ci)(hr)}$ (hr)

( ) x ( ) x ( ) = (rem)

GROUND:

(7)

(Ground Concentration) x (Dose Factor) x (Time)

(Ci/m^2) $\frac{(rem)(m^2)}{(Ci)(hr)}$ (hr)

( ) x ( ) x ( )

= (rem)

○ = *referenced Table No.*

THIS PAGE INTENTIONALLY LEFT BLANK

WORKSHEET H-3

PROJECTIONS DOWNWIND FROM FIELD MEASUREMENTS

(4)

xu/Q (at projected distance) x dose rate
(m^{-2}) (units as measured)

$$\frac{(\quad) \times (\quad)}{(\quad)} =$$

(m^{-2})
xu/Q (at measurement point)
(4)

Dose rate at projected location
(Units as measured)

SOURCE TERM BASED ON FIELD MEASUREMENTS

(Air Concentration) x (Wind Speed)
(Ci/m^3) (m/s)

$$\frac{(\quad) \times (\quad)}{(\quad)} = \quad Ci/s$$

(m^{-2})
(xu/Q)
(4)

Note: Measurements should be taken

a) at plume center line
b) >2 miles from plant
c) 15-30 minute average

○ = *referenced Table No.*

WORKSHEET H-4
INSIDE ENCLOSURE
AIR CONCENTRATION

ENCLOSURE VOLUME

(Length) (Feet)		(Width) (Feet)		(Height) (Feet)		(Metric conversion) (m^3/ft^3)		
( )	×	( )	×	( )	×	(.028)	=	(cu meters)

Use formulas below to Calculate Concentration

FIRE INITIATED RELEASE CONCENTRATION

(Total Curies Available) (Ci) ( ) × (Fire Release Fraction) (2F) ( )

divided by

( ) (m^3) (Enclosure Volume)

= (Ci/m^3)

Go to Worksheet #2 to calculate dose

MAXIMUM RELEASE CONCENTRATION

(Total Curies Available) (Ci) ( )

divided by

( ) (m^3) (Enclosure Volume)

= (Ci/m^3)

Go to Worksheet #2 to calculate dose

○ = *referenced Table No.*

BASIS AND ASSUMPTIONS

1. Radionuclide Selection

   The lists include all radionuclides licensed by the NRC plus those expected to be present in a reactor accident

2. Doses

   a. Adult acute inhalation is 30-day integrated absorbed doses.

   b. Adult breathing rate of 1.2 m^3/hr is used for inhalation doses (source: EPA). Light activity ICRP-23.

   c. Red bone marrow doses from DOE/EH-0070 are used as whole body doses for immersion and ground shine.

   d. Immersion dose is for a semi-infinite cloud.

3. Plume Transport Condition Assumptions

   a. 'D' stability
   b. 2.2 m/sec wind speed (2 mph)
   c. distances of 1/4 and 1 mile
   d. 1 Ci release
   e. 1 hour duration of exposure
   f. no precipitation
   g. ground level release

4. Compounds Inhaled

   Where the NRPB Dose Factors provided a choice in lung clearance classes (i.e., D, W, and Y), the most conservative factor was selected.

   Dose equivalent quality factors were included in the tables. For an acute dose the high LET was multiplied by quality factor of 10 and added to the low LET factor as described in NUREG/CR-5247. A quality factor of 10 was used instead of 20 to better represent the acute dose.

5. The effects of daughter products were included in the inhalation factors but only in the immersion or ground shine factors where noted.

Plume Phase General Population Protective Action Assessment

Section I

I

Plume Phase General Population Protective Action Assessment

PLUME PHASE GENERAL POPULATION PROTECTIVE ACTION ASSESSMENT

Objective

Assessing public (offsite) protective actions.

Guidance

CAUTION:

> Do not allow offsite officials to await an NRC assessment before taking action.

CAUTION:

> Do not interfere with or question offsite actions if an announcement of protective action recommendations for the nearby population is imminent or has begun, because the process may be delayed. If further action appears appropriate, the NRC recommendation should be made after the initial actions have been started.

CAUTION:

> Use "UF_6 Release Assessment," Section E, for assessment of protective actions involving UF_6.

NOTE:

> For reactor accidents, licensees' recommend actions and offsite officials act on the recommendations. There are provisions for prompt decision making and warning of the public 24 hours a day. Revising the preplanning actions may delay the response. The NRC is not part of this process.

Step 1

NOTE:

> Only the emergency plans for power reactors contain protective action criteria.

Review the protective action criteria in the licensee and State or local reactor emergency plans and procedures.

Step 2

Assess the protective actions for reactor accidents involving severe core damage or loss of control using the chart on page I-3 or for accidents involving radiation releases using the EPA guidance on page I-4.

Step 3

If we are in agreement with the recommendations and actions, inform the ET/DSO. If not, proceed to Step 4.

PLUME PHASE GENERAL POPULATION PROTECTIVE ACTION ASSESSMENT

Objective

Assessing public (offsite) protective actions.

Guidance

CAUTION:

> Do not allow offsite officials to await an NRC assessment before taking action.

CAUTION:

> Do not interfere with or question offsite actions if an announcement of protective action recommendations for the nearby population is imminent or has begun, because the process may be delayed. If further action appears appropriate, the NRC recommendation should be made after the initial actions have been started.

CAUTION:

> Use "UF_6 Release Assessment," Section E, for assessment of protective actions involving UF_6.

NOTE:

> For reactor accidents, licensees' recommend actions and offsite officials act on the recommendations. There are provisions for prompt decision making and warning of the public 24 hours a day. Revising the preplanning actions may delay the response. The NRC is not part of this process.

Step 1

NOTE:

> Only the emergency plans for power reactors contain protective action criteria.

Review the protective action criteria in the licensee and State or local reactor emergency plans and procedures.

Step 2

Assess the protective actions for reactor accidents involving severe core damage or loss of control using the chart on page I-3 or for accidents involving radiation releases using the EPA guidance on page I-4.

Step 3

If we are in agreement with the recommendations and actions, inform the ET/DSO. If not, proceed to Step 4.

Step 4

CAUTION:

If offsite technical contacts cannot be contacted in a reasonable period of time, proceed to the next step.

Discuss assessment differences with licensee and/or off-site technical contacts and determine basis for their protective action recommendation.

Step 5

Prepare a briefing for the ET or DSO using the following checklist:

_____ comparison of the recommendations or actions of the licensee, off-site officials and your assessment

_____ the results of your discussions with licensee and off-site officials (or the fact that these parties could not be contacted)

_____ the title, name, and phone number of the licensee and off-site protective action decision makers

_____ existing emergency plan or procedure criteria

_____ conditions that affected licensee or off-site decisions:

- local weather
- population affected (see page I-5, reactor population data)
- populations needing special consideration (e.g., industrial facilities, beaches, hospitals, prisons, etc.)

_____ the bounds of your consequence assessment

_____ offsite radiological conditions, if known

Step 6

Provide assessment verbally to the ET/DSO immediately, followup with a written assessment.

Step 7

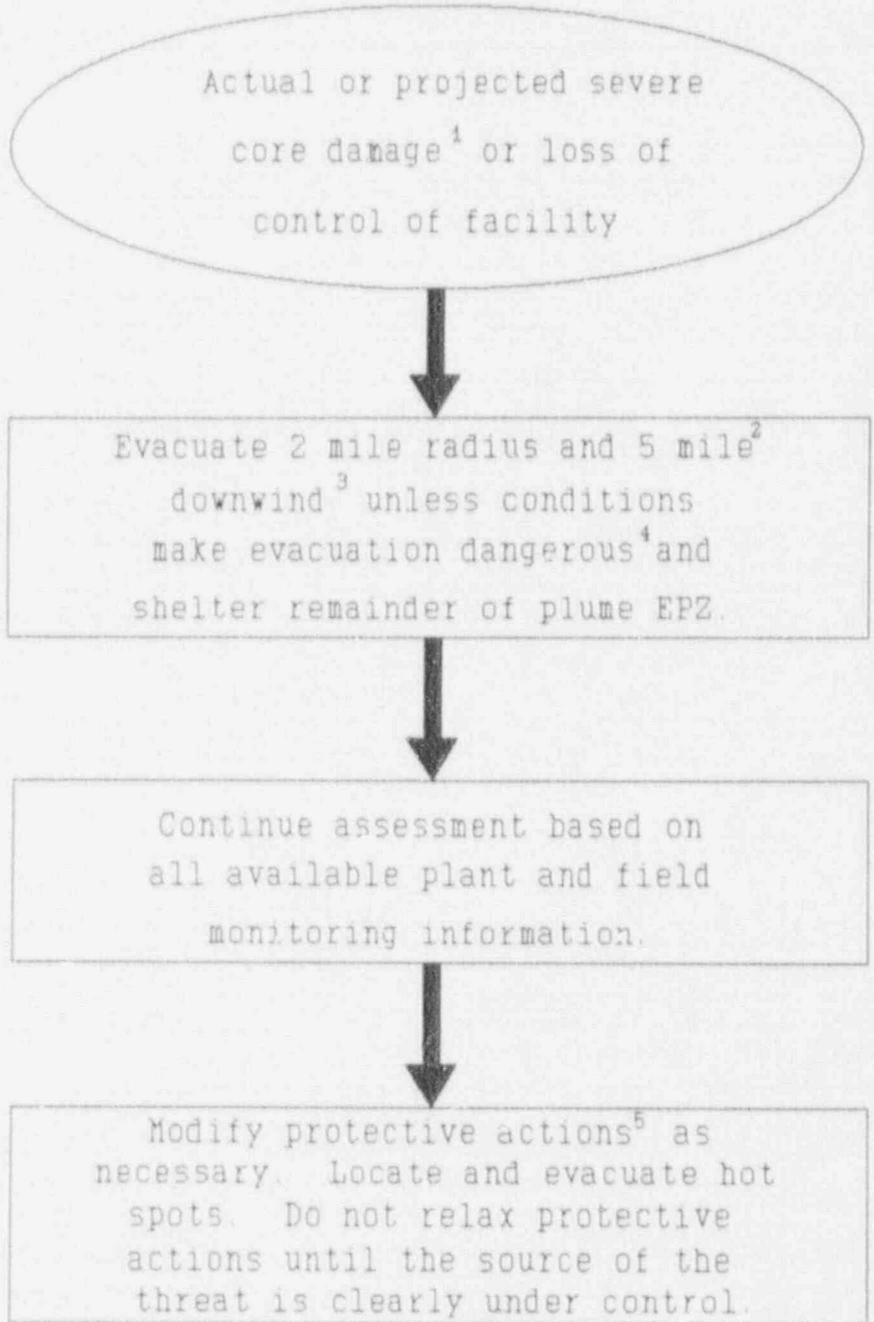
Consult and brief the representatives of other Federal agencies as time permits.

Step 8

Consider "Intermediate Phase General Population Protective Action Assessment," Section J-1.

END

SEVERE CORE DAMAGE OR LOSS OF CONTROL OF FACILITY
PUBLIC PROTECTIVE ACTIONS



1 - Severe core damage is indicated by (1) loss of critical functions required for core protection (e.g., loss of injection given a LOCA); (2) high core temperatures (PWR) or partially covered core (BWR); (3) very high radiation levels in area or process monitors.

2 - Distances are approximate - actual distances will be preplanned based on local conditions.

3 - During preparation for evacuation, people should shelter if possible.

4 - Such as very dangerous travel conditions or immobile infirmed population.

5 - Consider EPA PAGs (page I-4).

RECOMMENDED PROTECTIVE ACTIONS TO REDUCE WHOLE BODY AND THYROID DOSE FROM EXPOSURE TO A GASEOUS PLUME

CAUTION:

For non-reactor accidents involving releases of isotopes for which inhalation dose to organs other than the thyroid may dominate, use CEDE in place of whole body dose.

NOTE:

This EPA guidance may be revised shortly.

Projected Dose (rem) to the Population	Recommended Actions[a]	Comments
Whole body <1 Thyroid <5	No planned protective actions.[b] State may issue an advisory to seek shelter and await further instructions. Monitor environmental radiation levels.	Previously recommended protective actions may be reconsidered or terminated.
Whole body 1 to <5 Thyroid 5 to <25	Seek shelter as a minimum. Consider evacuation. Evacuate unless constraints make it impractical. Monitor environmental radiation levels. Control access.	If constraints exist, special consideration should be given for evacuation of children and pregnant women.
Whole body 5 and above Thyroid 25 and above	Conduct mandatory evacuation. Monitor environmental radiation levels and adjust area for mandatory evacuation based on these levels. Control access.	Seeking shelter would be an alternative if evacuation were not immediately possible.

[a] These actions are recommended for planning purposes. Protective action decisions at the time of the incident must take existing conditions into consideration.

[b] At the time of the incident, officials may implement low-impact protective actions in keeping with the principle of maintaining radiation exposures as low as reasonably achievable.

Source: EPA PAG Manual 89

SUMMARY OF PERMANENT AND TRANSIENT POPULATION AT NUCLEAR POWER REACTORS

CAUTION:

These data are 10 years old.

SITE	PERMANENT POPULATION* 0-2 mile	0-5 mile	0-10 mile	TRANSIENT POPULATION** 0-10 mile
ARKANSAS	853	7320	25394	6000
BEAVER VALLEY	3076	16658	142268	3400
BELLEFONTE	309	4696	25050	2437
BIG ROCK POINT	269	4368	9274	
BRAIDWOOD	3545	11490	26015	8105
BROWNS FERRY	148	2414	27678	19600
BRUNSWICK	711	4373	10583	21000
BYRON	371	7140	21393	43762
CALLAWAY	82	632	5759	4545
CALVERT CLIFFS	241	3501	19972	1150
CATAWBA	340	1058	81423	46879
CLINTON	48	918	12666	28472
COMANCHE PEAK	29	2684	10731	8918
COOPER STATION	40	830	5417	3000
CRYSTAL RIVER	0	825	13595	1010
D C COOK	723	12364	53755	16089
DAVIS-BESSE	1030	2572	16427	
DIABLO CANYON	10	57	18099	53700
DRESDEN	613	7498	39289	5900
DUANE ARNOLD	235	3821	79323	
FARLEY	27	1577	10681	1420
FERMI	3004	13460	71517	
FITZPATRICK	242	3909	35155	20790
FORT CALHOUN	207	7666	15254	871
FORT ST VRAIN	237	3373	16373	350
GINNA	930	9979	39162	5863
GRAND GULF	180	2025	7255	2873
HADDAM NECK	2345	12129	74080	29415
HARRIS	110	1545	15795	11000
HATCH	107	894	5312	150
HOPE CREEK	0	1209	22556	5539
INDIAN POINT	15165	74755	240455	92852
KEWAUNEE	163	1600	11086	
LASALLE	130	1145	13913	3130
LACROSSE	512	2001	7307	
LIMERICK	4349	100364	164870	23165
MAINE YANKEE	372	5663	28730	42338
MCGUIRE	420	4189	46233	31178
MILLSTONE	5176	48648	110166	83129
MONTICELLO	279	7611	20153	
NINE MILE POINT	242	3909	35155	20790
NORTH ANNA	225	1639	8688	1166

*Primary source of permanent population: Estimates of 1982 population developed by NRC staff.

**Transient population: Based on information obtained from FSARs, E Plans, NUREG/CR-1856 (1981) and on licensee estimates. Transient population data are considered to include a large degree of uncertainty.

SITE	PERMANENT POPULATION* 0-2 mile	0-5 mile	0-10 mile	TRANSIENT POPULATION** 0-10 mile
OCONEE	401	4670	50841	20000
OYSTER CREEK	4700	14950	71440	73676
PALISADES	959	5203	32773	
PALO VERDE	10	205	761	4000
PEACH BOTTOM	512	6153	28647	9858
PERRY	1882	17238	71902	53271
PILGRIM	1716	15249	41401	83085
POINT BEACH	239	1256	20994	1200
PRAIRIE ISLAND	290	4228	21462	
QUAD CITIES	224	5740	36445	12035
RANCHO SECO	126	691	8552	2234
RIVERBEND	601	4053	22872	13700
ROBINSON	1164	10435	26908	5000
SAINT LUCIE	210	9417	94854	40000
SALEM	0	1209	22556	5539
SAN ONOFRE	3650	28450	57150	25900
SEABROOK	6040	32060	100720	116988
SEQUOYAH	890	7503	38972	24000
SHOREHAM	6681	27681	108804	21420
SOUTH TEXAS	4	268	2550	4622
SUMMER	220	1883	8869	2000
SURRY	49	1399	73411	63755
SUSQUEHANNA	1177	13317	51232	3720
THREE MILE ISLAND	2331	27466	161509	6335
TROJAN	537	7507	65346	1500
TURKEY POINT	0	30	92664	4500
VERMONT YANKEE	2086	9231	31909	3544
VOGTLE	517	1133	2669	200
WATERFORD	914	13756	60009	7000
WATTS BAR	209	2696	13916	8000
WOLF CREEK	24	3698	5520	1100
WNP-2	0	80	1338	11824
YANKEE ROWE	204	1748	24718	2443
ZION	12981	59247	245006	65750

*Primary source of permanent population: Estimates of 1982 population developed by NRC staff.

**Transient population: Based on information obtained from FSARs, E Plans, NUREG/CR-1856 (1981) and on licensee estimates. Transient population data are considered to include a large degree of uncertainty.

Intermediate Phase General Population Protective Action Assessment (Relocation - Reentry)

Section J

J

Intermediate Phase General Population Protective Action Assessment (Relocation — Reentry)

INTERMEDIATE PHASE GENERAL POPULATION PROTECTIVE ACTION ASSESSMENT (RELOCATION - REENTRY)

Objective

Provide guidance on the actions to be taken once the source of the threat (or release) is under control, if requested by State or local authorities.

Guidance

Step 1

Assure the source of the release or threat is under control. If not, do not consider relaxing protective actions. Consider:

CAUTION:

Once the core has been severely damaged it may not be coolable even if covered by water.

- Is there radioactive material inventory capable of being released with offsite consequences?

- Are the barriers to a release threatened:

  * fire
  * facility controlled by others
  * hydrogen or other explosive gas
  * core melt through
  * pressure build up (decay heat)
  * direct containment heating (high pressure melt through)
  * Isolation failure

- Is the reactor shutdown (subcritical) and can it go critical?

- Is the reactor core being cooled?

Step 2

Establish a group composed of representatives of NRC, EPA, FEMA, DOE, USDA and FDA at the FRMAC to recommend to the Director of Site Operations (DSO) intermediate phase actions.

NOTE:

- EPA has developed the EPA Manual of Protective Actions for Nuclear Incidents, January 1990, which is the basis for the following guidance. These steps are also indicated in Figure 1, page J-4.
- EPA should be consulted.

Step 3

NOTE:

For reactor accidents, the dose from ground shine appears to be far more important than the inhalation dose from resuspension; however, this must be confirmed by sampling.

Based on environmental data, determine the areas where the projected one-year dose will exceed 2 rem and relocate persons from those areas, with priority given persons in the highest exposure rate areas using the Intermediate Phase Dose Projection Considerations, page J-5.

Step 4

Allow evacuated persons to return immediately if exposure rates are not in excess of twice the normal background before the accident (typical background is 10μR/hr). If resuspended activity could drift into the occupied areas, a buffer zone should be established to restrict occupancy until the situation is analyzed and dose projections are confirmed.

Step 5

Determine the location of the isodose line corresponding to the Intermediate Phase relocation PAG (Table 1, page J-6), establish the boundary of the restricted zone, and relocate any persons who still reside within the zone. Evacuated persons whose residence is in the area between the boundary of the plume deposition and the boundary to the restricted zone may return gradually as confidence is gained regarding the projected dose in the area.

NOTE:

It is an objective of the intermediate phase PAGs (Table 1) to assure that 1) doses in any single year after the first year will not exceed 0.5 rem, and 2) the cumulative dose over 50 years will not exceed 5 rem EDE. For reactor accidents, the 2 rem projected dose in the first year is expected to meet all of these objectives.

Step 6

Evaluate the dose reduction effectiveness of simple decontamination techniques and of sheltering due to partial occupancy of residences and work places. Results of these evaluations may influence recommendations for reducing exposure rates for persons who are not relocated from areas near, but outside, the restricted zone.

Step 7

Establish a mechanism for controlling access to and egress from the restricted zone. Typically this would be accomplished through control points at roadway accesses to the restricted zone.

NOTE:

Individuals permitted to reenter a restricted zone should not exceed 5 rem/year EDE.

Step 8

Establish monitoring and decontamination stations to support control of the restricted zone.

NOTE:

Significant levels of surface contamination may not be detected at control points because of high background levels. Additional monitoring may be required in low background areas.

Step 9

Implement simple decontamination techniques in contaminated areas outside the restricted zone, with priorities for areas with higher exposure rates and for residences of pregnant women. Refer to "General Guidance on Contamination Control," page J-10.

Step 10

Collect data needed to establish long-term radiation protection criteria for recovery and data to determine the effectiveness of various decontamination or other recovery techniques.

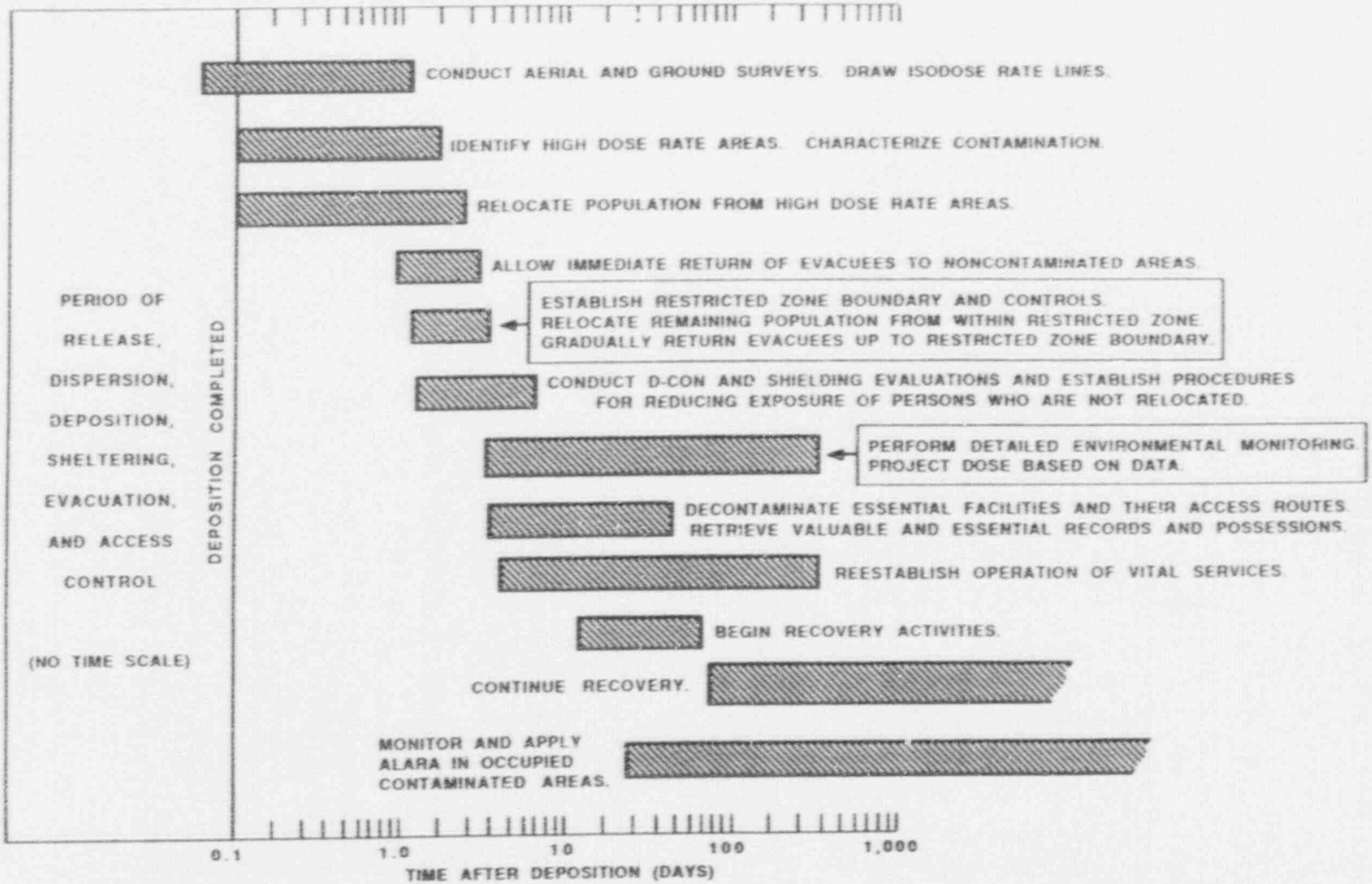
Step 11

Begin operations to recover contaminated property in the restricted zone.

END

Figure 1

POTENTIAL TIME FRAME OF RESPONSE TO A NUCLEAR INCIDENT



INTERMEDIATE PHASE DOSE PROJECTION CONSIDERATIONS

NOTE:

Environmental measurements and data analysis will be conducted by DOE at the FRMAC.

1. Conduct measurements and take samples to determine the principal radionuclides and amounts contributing to ground shine and their relative abundance.

NOTE:

Extensive measurements may be required due to considerable variations in deposition.

If only primary field monitoring data is available a crude estimate of the radionuclides could be made by using the ratios shown in "Typical 2nd Day Mix of Ground Contamination Resulting from a Major Reactor Accident," page K-6.

2. Determine the 1st year, 2nd year and 50 year integrated dose (include decay and weathering) from ground deposition.

Use Table 2, page J-7 to calculate the dose for each radionuclide.

NOTE:

If withdrawal of contaminated food or water from use would, in itself, create a health risk, the committed effective dose equivalent from ingestion should be added to the integrated dose.

3. Perform air sampling to determine magnitude of inhalation dose due to resuspension and calculated inhalation dose for the 1st and 2nd years.

   Table 3, page J-8, provides 1st and 2nd year dose factors for inhalation dose assuming weathering based on initial air concentrations taken early in the accident.

4. Sum the integrated ground shine dose projection from Step 2 and inhalation dose projection from Step 3 and compare to relocation PAGs (Table 1, page J-6).

5. Multiply the measured ground concentration (pCi/m^2) by the skin beta dose conversion factors with weathering (Table 4, page J-9) to project skin dose. Compare these levels with 50 times the relocation PAGs in Table 1, page J-6 (see Comment in Table).

CAUTION:

Skin dose is very difficult to measure directly and should be estimated based on ground concentrations.

Table 1

PROTECTIVE ACTION GUIDES FOR EXPOSURE TO DEPOSITED RADIOACTIVITY DURING THE INTERMEDIATE PHASE OF A NUCLEAR INCIDENT

Protective Action	PAG (projected dose)[a]	Comments
Relocate the general population.[b]	≥2 rem	Beta dose to skin may be up to 50 times higher.
Apply simple dose reduction techniques.[c]	<2 rem	These protective actions should be taken to reduce doses to as low as practicable levels.

[a] The projected sum of effective dose equivalent from external gamma radiation and committed effective dose equivalent from inhalation of resuspended materials, from exposure or intake during the first year. Projected dose refers to the dose that would be received in the absence of shielding from structures or the application of dose reduction techniques. These PAGs may not provide adequate protection from some long-lived radionuclides.

[b] Persons previously evacuated from areas outside the relocation zone defined by this PAG may return to occupy their residences. Cases involving relocation of persons at high risk from such action (e.g., patients under intensive care) should be evaluated individually.

[c] Simple dose reduction techniques include scrubbing and/or flushing hard surfaces, soaking or plowing soil, minor removal of soil from spots where radioactive materials have concentrated, and spending more time than usual indoors or in other low exposure rate areas.

Table 2

GAMMA EXPOSURE RATE AND EFFECTIVE DOSE EQUIVALENT (CORRECTED FOR RADIOACTIVE DECAY AND WEATHERING) DUE TO AN INITIAL UNIFORM CONCENTRATION OF 1 pCi/m² ON GROUND SURFACE

Radionuclide	Half-life hours	Initial exposure[a] rate at 1 m (mR/h per pCi/m²)	Integrated dose (weathering factor included)[b]		
			year 1 (mrem per pCi/m²)	year 2 (mrem per pCi/m²)	9-50 year (mrem per pCi/m²)
Zr-95	1.54E+03	1.2E-08	3.3E-05	4.0E-07	3.4E-05
Nb-95	8.41E+02	1.3E-08	(b)	(b)	(b)
Ru-103	9.44E+02	8.2E-09	7.1E-06	0	7.1E-06
Ru-106	8.84E+03	3.4E-09	1.2E-05	3.7E-06	1.8E-05
Te-132	7.82E+01	4.0E-09	3.2E-06	0	3.2E-06
I-131	1.93E+02	6.6E-09	1.3E-06	0	1.3E-06
I-132	2.30E+00	3.7E-08	(b)	(b)	(b)
I-133	2.08E+01	1.0E-08	2.1E-07	0	2.1E-07
I-135	6.61E+00	2.4E-08	1.6E-07	0	1.6E-07
Cs-134	1.81E+04	2.6E-08	1.0E-04	4.7E-05	2.4E-04
Cs-137	2.65E+05	1.0E-08	4.5E-05	2.9E-05	6.1E-04
Ba-140	3.07E+02	3.2E-09	1.1E-05	0	1.1E-05
La-140	4.02E+01	3.5E-08	(b)	(b)	(b)

[a]Estimated exposure rate at 1 meter above contaminated ground surface. Based on data from reference (DO-88).

[b]Radionuclides that have short-lived daughters (Zr/Nb-95, Te/I-132, Ru/Rh-106, Cs-137/Ba-137m, Ba/La-140) are assumed to quickly reach equilibrium. The integrated dose factors listed are the effective gamma dose due to the parent and the daughter.

Source: DOE/EH-0070, July 1988

Table 3

DOSE CONVERSION FACTORS FOR INHALATION OF RESUSPENDED MATERIAL

		Committed effective dose equivalent from specified exposure periods based on an initial concentration of one pCi/m^3 in air (with and without weathering)			
		Committed dose[a] considering radioactive decay and weathering (mrem per pCi/m^3)		Committed dose[a] considering radioactive decay only (mrem per pCi/m^3)	
Radionuclide[b]	Lung class[c]	year 1	year 2	year 1	year 2
Sr-90/Y-90	Y/Y	9.3E 0	5.5E 0	1.4E+1	1.3E+1
Zr-95/Nb-95	Y/Y	6.8E-2	0	7.9E-2	0
Ru-103	Y	1.3E-2	0	1.5E-2	0
Ru-106/Rh-106	Y/D	2.8E 0	1.0E 0	3.7E 0	1.9E 0
Te-132/I-132	W/D	1.3E-3	1.9E-5	1.3E-3	1.9E-5
I-131	D	1.1E-2	0	1.1E-2	0
Cs-134	D	3.2E-1	1.5E-1	4.1E-1	3.0E-1
Cs-137/Ba-137m	D/D	2.4E-1	1.4E-1	3.3E-1	3.2E-1
Ba-140/La-140	D/W	4.5E-3	0	4.7E-3	0
Ce-144/Pr-144	Y/Y	2.0E 0	4.2E-1	2.7E 0	9.8E-1

[a]Calculated using dose factors from Table 2.1. EPA-520/1-88-020, Federal Guidance Report-11

[b]Short lived daughters are not listed separately because the entries include the dose from both the daughter and the parent. These factors are based on the concentration of the parent only, at the beginning of the exposure period.

[c]The lung clearance class chosen was the one which results in the highest dose conversion factor.

Table 4

SKIN BETA DOSE CONVERSION FACTORS FOR DEPOSITED RADIONUCLIDES(a)

Radionuclides	Dose conversion factors[b] (mrem per pCi/m^2)	
	Radioactive decay plus weathering	Radioactive decay only
Co-58	1.2E-7	1.4E-7
Co-60	4.2E-7	5.6E-7
Rb-86	6.3E-5	5.7E-5
Sr-89	1.5E-4	1.6E-4
Sr-90	1.2E-5	1.7E-5
Y-90	2.2E-4	2.9E-4
Y-91	1.6E-4	1.9E-4
Zr-95	7.2E-7	8.3E-7
Nb-95	6.1E-7	7.4E-7
Mo-99	4.4E-6	4.6E-6
Tc-99m	7.7E-9	7.7E-9
Ru-103	6.8E-7	7.8E-7
Ru-106[c]	6.4E-7	8.7E-7
Rh-105	6.5E-8	6.6E-8
Sb-127	3.4E-6	3.4E-6
Te-127	1.0E-6	1.0E-6
Te-127m	7.8E-7	9.5E-7
Te-129	5.0E-7	5.0E-7
Te-129m	3.4E-5	3.6E-5
Te-131m	2.9E-7	2.9E-7
Te-132	5.4E-9	5.4E-9
I-131	8.5E-7	8.7E-7
I-132	5.0E-5	5.0E-5
Cs-134	2.6E-5	3.3E-5
Cs-136[c]	1.4E-7	3.7E-7
Cs-137[c]	2.1E-5	2.9E-5
Ba-140	9.1E-6	9.6E-6
La-140	1.2E-5	1.3E-5
Ce-141	6.6E-7	7.1E-7
Ce-143	2.3E-6	2.3E-6
Ce-144[c]	8.7E-7	1.1E-6
Pr-143	1.3E-5	1.4E-5
Nd-147	4.3E-6	4.5E-6
Np-239	3.4E-8	3.4E-8
Am-241	4.6E-8	6.4E-8

[a] Based on data from reference AR-89.

[b] Dose equivalent integrated for a one-year exposure at one meter height plus the estimated dose to the skin from materials deposited on the skin as a result of being in the contaminated area.

[c] Contributions from short-lived (one hour or less) decay products are included in dose factors for the parent radionuclides (i.e., Rh-106, Ba-136, Ba-137, and Pr-144).

GENERAL GUIDANCE ON CONTAMINATION CONTROL

For emergency situations, the following general guidance regarding surface contamination is recommended:

1. Do not delay urgent medical care for decontamination efforts or for time-consuming protection of attendants.

2. Do not waste effort trying to contain contaminated wash water.

3. Do not allow monitoring and decontamination to delay evacuation from high or potentially high exposure rate areas.

4. (Optional provision, for use only if a major contaminating event occurs, and rapid early screening is needed.) After plume passage, it may be necessary to establish emergency contamination screening stations in areas not qualifying as low background areas. Such areas should be less than 5 mR/h gamma exposure rate. These screening stations should be used only during the early phase and for major releases of particulate materials to the atmosphere to monitor persons emerging from possible high exposure areas, provide simple (rapid) decontamination if needed, and make decisions on whether to send them for special care or to a monitoring and decontamination station in a lower background area. Table 5 provides guidance on surface contamination levels for use if such centers are needed.

5. Establish monitoring and personnel decontamination (bathing) facilities at evacuation centers or other locations in low background areas (less than 0.1 mR/h). Encourage evacuated persons who were exposed in areas where inhalation of particulate materials would have warranted evacuation to bathe, change clothes, wash clothes, and wash other exposed surfaces such as cars and trucks and their contents and then report to these centers for monitoring. Table 6 provides recommended surface contamination guidance for use at these centers.

6. After the restricted zone is established, set up monitoring and decontamination stations at exits from the restricted zone. Because of the probably high background radiation levels at these locations, low levels of contamination may be undetectable. If contamination levels are undetectable, then they probably do not exceed those in some unrestricted areas occupied by the exposed population and no decontamination is required. Nevertheless, these individuals should be advised to bathe and change clothes at their first opportunity and certainly within the next 24 hours. If, after decontamination at the boundary of the restricted zone stations, persons still exceed the limits for this station, they should be sent for further decontamination or for medical or other special attention. As an alternative to decontamination, contaminated items other than persons or animals may be retained in the restricted zone for radioactive decay.

7. Establish auxiliary monitoring and decontamination stations in low background areas (background less than 0.1 mR/h). These stations should be used to achieve ALARA surface contamination levels. Table 6 provides surface contamination screening levels for use at those stations.

Table 5

RECOMMENDED SURFACE CONTAMINATION SCREENING LEVELS FOR EMERGENCY SCREENING OF PERSONS AND OTHER SURFACES AT SCREENING OR MONITORING STATIONS IN HIGH BACKGROUND RADIATION AREAS (0.1 mR/h TO 5 mR/H GAMMA EXPOSURE)*

Condition	Geiger-counter shielded-window reading	Recommended action
Before decontamination	<2X bkgd and <0.5 mR/h above background	Unconditional release
	>2X bkgd or >0.5 mR/h above background	Decontaminate. Equipment may be stored or disposed of as appropriate.
After decontamination	<2X bkgd and <0.5 mR/h above background	Unconditional release
	>2X bkgd or >0.5 mR/h above background	Continue to decontaminate or refer to low back-ground monitoring and d-con station. Equipment may also be stored for decay or disposed of as appropriate.

*Monitoring stations in these high exposure rate areas are for use only during the early phase of an incident involving major atmospheric releases of particulate. Otherwise use Table 6.

Table 6

RECOMMENDED SURFACE CONTAMINATION SCREENING LEVELS FOR PERSONS AND OTHER SURFACES AT MONITORING STATIONS IN LOW BACKGROUND RADIATION AREAS (<0.1 mR/h GAMMA EXPOSURE RATE)

Condition	Geiger counter thin window[a] reading	Recommended action
Before decontamination	<2X bkgd	Unconditional release
	>2X bkgd	Decontaminate
After simple[b] decontamination effort	<2X bkgd	Unconditional release
	>2X bkgd	Full decontamination[c]
After full[c] decontamination effort	<2X bkgd	Unconditional release
	>2X bkgd	Continue to d-con persons
	<0.5 mR/h[d]	Release animals and equipment
After additional full decontamination effort	<2X bkgd	Unconditional release
	>2X bkgd	Send persons for special evaluation
	<0.5 mR/h[d]	Release animals and equipment
	>0.5 mR/h[d]	Refer, or use informed judgment on further control of animals and equipment

[a] Window thickness of approximately 30mg/cm^2 is acceptable. Recommended limits for open window readings are expressed as twice the existing background (including background) in the area where measurements are being made. Corresponding levels, expressed in units related to instrument designations, may be adopted for convenience. Levels higher than twice background (not to exceed the meter reading corresponding to 0.1 mR/h) may be used to speed the monitoring of evacuees in very low background areas.

[b] Flushing with water and wiping is an example of a simple decontamination effort.

[c] Washing or scrubbing with soap or solvent followed by flushing is an example of a full decontamination effort.

[d] Closed shield reading including background.

Ingestion Pathway Protective Action Assessment

Section K

K Ingestion Pathway Protective Action Assessment

INGESTION PATHWAY PROTECTIVE ACTION ASSESSMENT

Objective

Assess ingestion pathway protective actions, if requested by State or local authorities.

Guidance

CAUTION:

> For accidents potentially involving large release of iodine, cesium, etc. (e.g., General Emergency), milk producing animals within 10 miles should be immeuiately placed on stored feed if possible.

Step 1

Obtain HHS (FDA), USDA, and/or EPA (water only) expertise at either Headquarters (through the Protective Measures Team) or from the HHS, USDA, or EPA representatives at the site (FRMAC).

Step 2

NOTE:

> - Use verified measurements, not dose model projections.
> - Rain may increase deposition in some locations.
> - Use of gross gamma readings should be avoided if (1) the ratio of nuclides is not constant or (2) only a small number of samples have been analyzed in the laboratory.
> - This procedure is based on FDA Ingestion Pathway Guidance as reproduced in EPA Manual of Protective Actions for Nuclear Incidents, Chapter 3 and Appendix D, January 1990 except where noted.

Gather information on analyzed ingestion pathway samples from DOE (FRMAC), State and local governments.

Step 3

CAUTION:

> Levels of cesium in milk approaching the "response level" should cause surveillance of meat. If both Cs-134 and Cs-137 are equally present as might be expected in reactor accidents, the response levels should be reduced by a factor of 2.

CAUTION:

> There is a delay in appearance of radioactivity in milk. The peak concentration occurs at 3 days for I-131 and 6-8 days for Cs and Sr after ingestion of contaminated feed. In order to calculate the maximum concentration that the sample would build-up to, the concentration at the time of the sampling should be divided by the ratio of the unit concentration at the time of sampling to the unit maximum concentration on Figure K-1, page K-24.

NOTE:

> A severe reactor accident may release a complex mixture of isotopes -- look for them. See Table K-1, page K-6.

NOTE:

> Derived response level (DRL) is a calculated radionuclide concentration in food, milk, and water, which, if ingested without any protective actions, would result in a projected dose commitment equivalent to the preventive or emergency PAGs.

Determine the most important contributions to the ingestion pathway, using Table K-1, page K-6, as the starting point for a severe reactor accident. (I-131 will be important short term and Sr-90, Cs-134, etc., long term.) Sum the ratios of the measured value for each nuclide present to the individual derived preventive or emergency response level (DRL) in Tables K-2 through K-7 for each food of interest (Σ(nuclides/DRL)). Sum the ratios for all foods. If the total is greater than one, the FDA ingestion PAGs (Table K-8, page K-17) are exceeded and the actions in Table K-9 (pages K-18 and K-19) would be warranted.

DRL by Pathway	Table	Page
FDA Grass-Cow-Milk Preventive DRL	K-2 (see CAUTION)	K-7
FDA Grass-Cow-Milk Emergency DRL	K-3 (see CAUTION)	K-8
FEMA 5 day Drinking Water DRL	K-4	K-9
FEMA 1 year Drinking Water DRL	K-5	K-11
FEMA 30 day Vegetable DRL	K-6	K-13
FEMA 1 year Vegetable DRL	K-7	K-15

Step 4

Consideration of other foods or radionuclides:

Calculate the response levels for other radionuclides using the daily consumptions in Table 10, page K-20 and dose factors in Table 11, beginning on page K-21.

$$\text{Derived Response Level } (\mu Ci/kg) = \frac{\text{PAG, (mrem)}}{\left[\begin{array}{c}\text{Consumption}\\ \text{(kg/day)}\end{array}\right] \times \left[\begin{array}{l}\text{Days of Intake}\\ \text{(see discussion}\\ \text{below)}\end{array}\right] \times \left[\begin{array}{l}\text{Dose Conversion}\\ \text{Factor}\\ (\text{mrem}/\mu Ci)\end{array}\right]}$$

NOTE:

> The dose factors in Table K-11 are adult effective dose equivalent based on ICRP-30 taken from EPA-520/1-88-020, 9-88. These factors may be different from those used elsewhere in the document. They should only be used for isotopes not covered elsewhere.

Determination of Days of Intake

The days of intake determines the total quantity of food ingested.

Assessment of the days of intake to be used in the calculation of derived response levels should consider all the factors that influence the length of time that contaminated food would be consumed, such as; the food, the food distribution system, the population of concern, and the radionuclide.

The days of intake is determine by the shorter of the radionuclide mean life (T_M) or the mean effective life (T_{ME}) or some other arbitrary time limit (e.g., 30 days or 1 year). See NOTE (page K-4) for discussion of T_M and T_{ME}.

a. Radioactive decay will limit the days of intake for the short half-life radionuclides. If there is no weathering, and the activity is reduced only by radioactive decay, the mean life for the radionuclide of concern should be used to determine the days of intake (T_M = 1.44 x Radiological Half-Life) See NOTE (page K-4): Example A.

b. As an initial assessment, a 14 day weathering half-life may be assumed for foods being harvested on a daily basis.

c. When both radioactive decay and weathering are present, the mean effective life for the radionuclide of concern should be used to determine the days of intake. (T_{ME} = 1.44 x Effective Half-Life) See NOTE (page K-4): Example B.

d. For foods sold in retail markets the days of intake is limited by the quantity purchased at a time (e.g., 1 week or 30 days).

e. For foods that may be stored for some period of time before they are consumed, a decay correction is required.

f. Adjustments are required for population groups with special dietary intakes.

NOTE

(The days of intake are referred to as "effective days of intake" in the "FDA Ingestion Pathway Guidance" and as "effective ingestion time" in FEMA REP-13.)

The <u>mean life</u> of a radionuclide is a useful concept which permits the calculation of a single value of the derived response level for each nuclide.

For a given quantity of a radionuclide, the total number of disintegrations that occur during the life of that quantity is equal to the number of atoms initially present, and is determined by:

$N = A/\lambda$ Where: N = number of atoms present
λ = decay constant (s^{-1})
A = activity ($d\ s^{-1}$)

The above expression leads to the definition of mean life (or average life). Mean life (T_M) is the name given to the quantity $1/\lambda$ and is defined by the relationship: total disintegrations = initial activity x mean life.

From above, $N/A = 1/\lambda$ and the decay constant (λ) is related to the half-life (T_H) by:

$\lambda = 0.693/T_H$.

Therefore, <u>mean life</u> is: $T_M = 1/\lambda = T_H/0.693 = 1.44 \times T_H$.

When both radiological decay and weathering are acting to reduce the radioactivity on a crop, the <u>effective half-life</u> (T_E) of the radioactivity is given by:

$$T_E = \frac{T_R \times T_W}{T_R + T_W}$$

Where: T_R is the radiological half-life and
T_W is the weathering half-life.

The <u>mean effective life</u> (T_{ME}) is the name given to the mean life when the radiological decay and weathering is present and the radiological half-life is replaced with the effective half-life.

<u>Mean effective life</u> $T_{ME} = 1.44 \times T_E$

The mean life (or mean effective life) is used to assess the days of intake as follows:

Example A: I-131 without weathering

Radiological half-life (T_R) of 8.05 days.

Days of intake = mean life = 1.44 x T_H = 1.44 x 8.05
= 11.6 days

Example B: I-131 with weathering

Assume weathering half-life (T_W) of 14 days.

Radiological half-life (T_R) of 8.05 days.

Since both radiological decay and weathering are present the effective half-life (T_E) is used to determine the days of intake.

The effective half-life is:

$$T_E = \frac{T_R \times T_W}{T_R + T_W} = \frac{8.05 \times 14}{8.05 + 14} = 5.1 \text{ days.}$$

The days of intake from the combined action of both radiological decay and weathering are:

Days of intake = mean effective life = 1.44 x T_E
= 1.44 x 5.1
= 7.3 days

Care must be used in calculating the mean life for certain short-lived radionuclides which have long-lived parent nuclides. In these cases, if the parent nuclide is present in the contamination, the mean life for the short-lived radionuclide will appear to be the same as that for the long-lived parent nuclide.

When considering water, if the impact of parameters affecting concentration (other than radioactive decay) are unknown, the mean life should be used to assess the days of intake.

Source: Bruce Burnett, FDA, 3/91

END

Table K-1

TYPICAL 2ND DAY MIX OF GROUND CONTAMINATION RESULTING FROM A MAJOR REACTOR ACCIDENT

FRACTION OF GROUND CONTAMINATION				
		LEVEL OF CORE DAMAGE (TEMPERATURE)		
ISOTOPE	HALF[1] LIFE	GAP FRACTION (1200°F)	GRAIN BOUNDARY FRACTION (3000°F)	MELT FRACTION[2] (4500°F)
SR-89	50 D	0.000	0.003	0.023
SR-90	28 Y	0.000	0.000	0.001
SR-91	9.5 H	0.000	0.000	0.006
MO-99	66 H	0.000	0.015	0.056
RU-103	39 D	0.000	0.000	0.003
RU-106	368 D	0.000	0.000	0.001
TE-129m	33 D	0.000	0.002	0.003
TE-132	78 H	0.003	0.110	0.127
I-131	8 D	0.398	0.389	0.299
I-133	20 H	0.375	0.366	0.281
I-135	6.6 H	0.047	0.046	0.035
CS-134	2 Y	0.088	0.034	0.026
CS-136	13 D	0.035	0.014	0.011
CS-137	30 Y	0.055	0.022	0.017
BA-140	12.7 D	0.000	0.000	0.112

Notes:

1 H = hour, D = day, Y = year.
2 Other isotopes such as NP-239, Zr-95, Y-91, Sb-127 would also be present in lesser amounts.

Source: U.S. NRC based on NUREG-1228 and NUREG-1150.

Table K-2

FDA GRASS - COW - MILK PREVENTIVE DERIVED RESPONSE LEVELS

Derived response levels for grass-cow-milk pathway equivalent to Preventive PAG dose commitment of 1.5 rem thyroid, 0.5 whole body or red bone marrow to infant[1]

Response levels for Preventive PAG	I-131[2]	Cs-134[3]	Cs-137[3]	Sr-90	Sr-89
Initial activity area deposition (μCi/m²)	0.13	2	3	0.5	8
Forage concentration[4] (μCi/kg)	0.05	0.8	1.3	0.18	3
Peak milk acitivity (μCi/l)	0.015	0.15	0.24	0.009	0.14
Total intake (μCi)	0.09	4	7	0.2	2.6

[1]Newborn infant includes fetus (pregnant women) as critical segment of population for iodine-131. For other radionuclides, "infant" refers to child less than 1 year of age.

[2]From fallout, iodine-131 is the only radioiodine of significance with respect to milk contamination beyond the first day. In case of a reactor accident, the cumulative intake of iodine-133 via milk is about 2 percent of iodine-131, assuming equivalent deposition.

[3]Intake of cesium via the meat-man pathway for adult may exceed that of the milk pathway; therefore, such levels in milk should cause surveillance and protective actions for meat, as appropriate. If both Cs-134 and Cs-137 are equally present, as might be expected in reactor accidents, the response levels should be reduced by a factor of 2.

[4]Fresh weight.

Source: HHS Publication FDA 82-8196, 8-82, p. 17 (reproduced in EPA PAG Manual, EPA 520.1-75-001A, 1990).

Table K-3
FDA GRASS - COW - MILK EMERGENCY DERIVED RESPONSE LEVELS

Derived response levels for grass-cow-milk pathway equivalent to emergency PAG dose commitment of 15 rem thyroid, 5 rem whole body or red bone marrow

Response levels for emergency PAG	I - 131[3]		Cs - 134[5]		Cs - 137[5]		Sr - 90		Sr - 89	
	Infant[1]	Adult	Infant[2]	Adult	Infant[2]	Adult	Infant[2]	Adult	Infant[2]	Adult
Initial activity area deposition ($\mu Ci/m^2$)	1.3	18	20	40	30	50	5	20	80	1600
Forage concentration (μCi/kg)[4]	0.5	7	8	17	13	19	1.8	8	30	700
Peak milk activity (μCi/ℓ)	0.15	2	1.5	3	2.4	4	0.09	0.4	1.4	30
Total intake (μCi)	0.9	10	40	70	70	80	2	7	26	400

[1]Newborn infant includes fetus (pregnant women) as critical segment of population for iodine-131.

[2]"Infant" refers to child less than 1 year of age.

[3]From fallout, Iodine-131 is the only radioiodine of significance with respect to milk contamination beyond first day. In case of a reactor accident, the cumulative intake of iodine-133 via milk is about 2 percent of Iodine-131 assuming equivalent deposition.

[4]Fresh weight.

[5]Intake of cesium via the meat-man pathway for adult may exceed that of the milk pathway: therefore, such levels in milk should cause surveillance and protective actions for meat, as appropriate. If both Cs-134 and Cs-137 are equally present as might be expected for reactor accidents, the response levels should be reduced by a factor of 2.

Table K-4

5 DAY DERIVED PREVENTIVE RESPONSE LEVELS FOR DRINKING WATER[a,b]

		Initial Water Concentration Equivalent to the Preventive PAG Ingestion Dose Commitment[b]			
Nuclide	Organ[c]	Adult[d] (μCi/liter)	Teenager[d] (μCi/liter)	Child[d] (μCi/liter)	Infant[d] (μCi/liter)
I-131	Th	1.2E-1[e]	1.1E-1	4.5E-2	2.5E-2
I-132	Th	1.2E+1	1.4E+1	4.9E+0	3.5E-1
I-133	Th	2.4E+0	2.6E+0	9.1E-1	5.8E-1
Rb-86	Lv	2.6E+0	5.6E+0	1.9E+0	7.1E-1
Cs-134	Lv	7.4E-1	3.6E-1	1.9E-1	9.4E-1
Cs-136	Lv	2.2E+0	2.4E+0	1.3E+0	9.3E-1
Cs-137	Lv	8.2E-1	4.8E-1	2.2E-1	1.6E+0
Te-127m	Kd	1.8E+0	1.9E+0	8.8E-1	7.8E-1
Te-131m	GI	2.4E+0	3.1E+0	2.9E+0	4.4E+0
Te-132	GI	1.0E+0	1.6E+0	2.5E+0	2.7E+0
Sb-127[f]	GI	1.1E+0	1.7E+0	2.6E+0	4.8E+0
Sr-89	Bo	4.3E+0	1.7E-1	5.6E-2	5.9E-1
Sr-90	Bo	7.1E-2	8.6E-3	4.2E-3	4.5E-2
Ba-140	GI	1.4E+0	1.9E+0	9.8E-1	7.4E-1
Mo-99	Kd	9.1E+0	9.4E+0	4.6E+0	5.9E+0
Ru-103	GI	2.4E+0	3.5E+0	3.9E+0	6.4E+0
Ru-106	GI	2.8E-1	3.8E-1	3.9E-1	6.1E-1
Rh-105[f]	GI	1.2E+1	1.8E+1	2.9E+1	5.4E+1
Co-58	GI	3.4E+0	5.5E+0	7.0E+0	1.3E+1
Co-60	GI	1.2E+0	2.0E+0	2.4E+0	4.3E+0
Y-90	GI	4.9E-1	6.3E-1	6.1E-1	9.3E-1
Y-91	GI	6.6E-1	8.9E-1	9.2E-1	1.4E+0
Zr-95	GI	1.7E+0	2.4E+0	2.8+0	4.6E+0
Zr-97	GI	3.4E+0	4.0E+0	3.3E+0	4.9E+0
Nb-95	GI	2.5E+0	3.8E+0	4.6E+0	8.0E+0
La-140	GI	6.1E-1	8.3E-1	8.2E-1	1.3E+0
Ce-141	GI	2.2E+0	3.0E+0	3.0E+0	4.7E+0
Ce-143	GI	4.0E+0	5.1E+0	4.7E+0	7.1E+0
Ce-144	GI	3.0E-1	4.1E-1	4.2E-1	6.5E-1
Pr-143	GI	1.4E+0	1.9E+0	1.9E+0	2.9E+0
Nd-147	GI	1.7E+0	2.2E+0	2.3E+0	3.6E+0
Np-239	GI	4.5E+0	5.8E+0	5.5E+0	8.4E+0
Pu-238[f]	Bo	7.5E-3	9.9E-3	4.8E-3	6.9E-3
Pu-239[f]	Bo	6.4E-3	8.4E-3	3.4E-3	5.8E-3
Pu-240[f]	Bo	6.4E-3	8.4E-3	3.4E-3	5.8E-3
Pu-241[f]	Bo	3.1E-1	4.0E-1	2.0E-1	2.8E-1
Am-241[f]	Bo	1.2E-3	2.0E-2	7.8E-4	1.1E-3
Cm-242[f]	Bo	5.9E-2	7.8E-2	3.8E-2	5.3E-2
Cm-244[f]	Bo	2.5E-3	3.2E-3	1.6E-3	2.3E-3

Table K-4 - Continued

Notes:

[a] Assumes a contaminated water ingestion period equivalent to the shorter time interval of the radionuclide mean lifetime (assuming no weathering) or 5 days. Water is ingested at the rates given in NRC Reg. Guide 1.109 for the maximum exposed individual (see page K-20).

[b] The derived response level for each radionuclide is capable of producing the preventive PAG dose. Therefore, if more than one radionuclide is present in the sample, the sum of ratios technique must be used to estimate the individual radionuclide concentrations that are permissible, e.g.,

$$\frac{\text{Conc A}}{\text{Response Level A}} + \ldots + \frac{\text{Conc X}}{\text{Response Level X}} = \leq 1$$

[c] Th=thyroid, Lv=liver, Kd=kidney, Bo=bone, WB=whole body, GI=gastrointestinal tract. These are the critical organs for the corresponding radionuclides.

[d] Calculated concentrations may vary if calculation assumptions concerning ingestion rates and dose conversion factors are different from those presented in NRC Reg. Guide 1.109.

[e] $1.2E-1 = 1.2 \times 10^{-1} = 0.12$.

[f] Adult dose conversion factors (DCFs) were obtained from ICRP-30; dose conversion factors for other age groups were estimated by multiplying these adult DCFs by DCF ratios $\left(\frac{\text{other age group}}{\text{adult}}\right)$ presented in NRC Reg. Guide 1.109 for other nuclides having similar critical organs and retention times.

Source: FEMA REP-13, 1990, pp. 2-5, 2-6.

Table K-5

1 YEAR DERIVED PREVENTIVE RESPONSE LEVELS FOR DRINKING WATER[a]

		Initial Water Concentration Equivalent to the Preventive PAG Ingestion Dose Commitment[b]			
Nuclide	Organ[c]	Adult[d] (µCi/liter)	Teenager[d] (µCi/liter)	Child[d] (µCi/liter)	Infant[d] (µCi/liter)
I-131	Th	6.3E-2[e]	5.7E-2	2.4E-2	1.3E-2
I-132	Th	1.2E+1	1.4E+1	4.9E+0	3.5E-1
I-133	Th	2.4E+0	2.6E+0	9.1E-1	5.8E-1
Rb-86	Lv	6.4E-1	1.4E+0	4.7E-1	1.8E-1
Cs-134	Lv	1.2E-2	5.8E-3	3.0E-3	1.5E-2
Cs-136	Lv	7.6E-1	8.2E-1	4.3E-1	3.2E-1
Cs-137	Lv	1.1E-2	6.6E-3	3.0E-3	2.2E-2
Te-127m	Kd	8.5E-2	8.5E-2	4.0E-2	3.6E-2
Te-129	Kd	4.0E+4	3.1E+4	9.1E+2	5.2E+2
Te-131m	GI	2.4E+0	3.1E+0	2.9E+0	4.4E+0
Te-132[f]	GI	1.0E+0	1.6E+0	2.5E+0	2.7E+0
Sb-127[f]	GI	9.8E-1	1.5E+0	2.4E+0	4.4E+0
Sr-89	Bo	4.1E-1	1.6E-2	5.3E-3	5.6E-2
Sr-90	Bo	9.9E-4	1.2E-4	5.8E-5	6.2E-4
Ba-140	GI	4.7E-1	6.5E-1	3.4E-1	2.6E-1
Mo-99	Kd	9.1E+0	9.4E+0	4.6E+0	5.9E+0
Ru-103	GI	3.0E-1	4.3E-1	4.9E-1	7.9E-1
Ru-106[f]	GI	5.1E-3	6.9E-3	7.2E-3	1.1E-2
Rh-105[f]	GI	1.2E+1	1.8E+1	2.9E+1	5.4E+1
Co-58	GI	2.4E-1	3.8E-1	4.9E-1	8.8E-1
Co-60	GI	1.8E-2	2.8E-2	3.6E-2	6.3E-2
Y-90	GI	6.8E-3	8.8E-3	8.5E-3	1.3E-2
Y-91	GI	5.5E-2	7.5E-2	7.7E-2	1.2E-1
Zr-95	GI	1.3E-1	1.9E-1	2.1E-1	3.5E-1
Zr-97	GI	3.4E+0	4.0E+0	3.3E+0	4.9E+0
La-140	GI	2.1E-1	2.9E-1	2.9E-1	4.5E-1
Ce-141	GI	3.2E-1	4.4E-1	4.5E-1	7.0E-1
Ce-143	GI	4.0E+0	5.1E+0	4.7E+0	7.1E+0
Ce-144	GI	5.9E-3	7.9E-3	8.2E-3	1.3E-2
Pr-143	GI	4.6E-1	6.1E-1	6.2E-1	9.6E-1
Nd-147	GI	6.5E-1	8.9E-1	9.1E-1	1.4E+0
Np-239[f]	GI	4.5E+0	5.8E+0	5.5E+0	8.4E+0
Pu-238[f]	Bo	1.0E-4	1.4E-4	6.5E-5	9.6E-5
Pu-239[f]	Bo	8.8E-5	1.2E-4	5.4E-5	8.0E-5
Pu-240[f]	Bo	8.8E-5	1.2E-4	5.4E-5	8.0E-5
Pu-241[f]	Bo	4.4E-3	5.6E-3	2.8E-3	4.0E-3
Am-241[f]	Bo	1.7E-5	2.2E-5	1.1E-5	1.5E-5
Cm-242[f]	Bo	2.1E-3	2.7E-3	1.3E-3	1.8E-3
Cm-244[f]	Bo	3.5E-5	4.5E-5	2.2E-5	3.2E-5

Table K-5 - Continued

Notes:

[a] Assumes a contaminated water ingestion period equivalent to the shorter time interval of the radionuclide mean lifetime (assuming no weathering) or 365 days. Water is ingested at the rates given in NRC Reg. Guide 1.109 for the maximum exposed individual (see page K-20).

[b] The derived response level for each radionuclide is capable of producing the preventive PAG dose. Therefore, if more than one radionuclide is present in the sample, the sum of ratios technique must be used to estimate the individual radionuclide concentrations that are permissible, e.g.,

$$\frac{\text{Conc A}}{\text{Response Level A}} + \ldots + \frac{\text{Conc X}}{\text{Response Level X}} = \leq 1$$

[c] Th=thyroid, Lv=liver, Kd=kidney, Bo=bone, WB=whole body, GI=gastrointestinal tract. These are the critical organs for the corresponding radionuclides.

[d] Calculated concentrations may vary if calculation assumptions concerning ingestion rates and dose conversion factors are different from those presented in NRC Reg. Guide 1.109.

[e] 6.3E-2 = 6.3×10^{-2} = 0.063.

[f] Adult dose conversion factors (DCFs) were obtained from ICRP-30; dose conversion factors for other age groups were estimated by multiplying these adult DCFs by DCF ratios $\left(\frac{\text{other age group}}{\text{adult}}\right)$ presented in NRC Reg. Guide 1.109 for other nuclides having similar critical organs and retention times.

Source: FEMA REP-13, 1990, pp. 2-8, 2-9.

Table K-6

30 DAY DERIVED PREVENTIVE RESPONSE LEVELS FOR VEGETABLE[a,b]

	Adult[c]		[illegible][c]		Child[c]	
Nuclide	Produce[d] µCi/kg	Leafy[e] µCi/kg	Produc[illegible] µCi/[illegible]	[illegible]eafv [illegible]	Produce µCi/kg	Leafy µCi/kg
I-131	1.3E-1[f]	9.9E-1	6.4E-2	9.3E-1	3.3E-2	6.7E-1
I-132	2.0E+1	1.6E+2	1.3E+1	1.9E+2	5.6E+0	1.1E+2
I-133	3.5E+0	2.8+1	2.2E+0	3.2E+1	9.4E-1	1.9E+1
Rb-86	1.8E+0	1.4E+1	2.2E+0	3.1E+1	9.0E-1	1.8E+1
Cs-134	2.6E-1	2.1E+0	7.5E-2	1.1E+0	4.7E-2	9.5E-1
Cs-136	1.8E+0	1.4E+1	1.1E+0	1.6E+1	7.0E-1	1.4E+1
Cs-137	2.9E-1	2.3E+0	9.6E-2	1.4E+0	5.4E-2	1.1E+0
Te-127m	7.6E-1	6.0E+0	4.4E-1	6.3E+0	2.5E-1	5.1E+0
Te-131m	3.6E+0	2.9E+1	2.7E+0	3.8E+1	3.0E+0	6.1E+1
Te-132	1.7E+0	1.3E+1	1.5E+0	2.2E+1	2.9E+0	5.8E+1
Sb-127[g]	1.6E+0	1.3E+1	1.5E+0	2.1E+1	2.8E+0	5.7E+1
Sr-89	2.0E+0	1.6E+1	4.6E-2	6.6E-1	1.9E-2	3.8E-1
Sr-90	2.5E-2	2.0E-1	1.7E-3	2.5E-2	1.0E-3	2.1E-2
Ba-140	1.1E+0	8.7E+0	8.6E-1	1.2E+1	5.5E-1	1.1E+1
Mo-99	1.5E+1	1.2E+2	8.7E+0	1.3E+2	5.1E+0	1.0E+2
Ru-103	1.2E+0	9.8E+0	1.0E+0	1.5E+1	1.4E+0	2.9E+1
Ru-106	1.0E-1	8.2E-1	8.1E-2	1.2E+0	1.0E-1	2.1E+0
Rh-105[g]	1.8E+1	1.4E+2	1.6E+1	2.3E+2	3.1E+1	6.4E+2
Co-58	1.5E+0	1.2E+1	1.4E+0	2.0E+1	2.2E+0	4.4E+1
Co-60	4.4E-1	3.5E+0	4.0E-1	5.7E+0	6.0E-1	1.2E+1
Y-90	1.7E-1	1.4E+0	1.3E-1	1.8E+0	1.5E-1	3.0E+0
Y-91	3.1E-1	2.4E+0	2.4E-1	3.4E+0	3.0E-1	6.0E+0
Zr-95	7.5E-1	5.9E+0	6.3E-1	9.1E+0	8.7E-1	1.8E+1
Zr-97	5.0E+0	3.9E+1	3.4E+0	4.9E+1	3.4E+0	6.9E+1
Nb-95	1.3E+0	1.0E+1	1.2E+0	1.7E+1	1.7E+0	3.5E+1
La-140	5.0E-1	3.9E+0	3.8E-1	5.5E+0	4.7E-1	9.5E+0
Ce-141	1.2E+0	9.4E+0	9.3E-1	1.3E+1	1.2E+0	2.4E+1
Ce-143	6.1E+0	4.8E+1	4.4E+0	6.4E+1	5.0E+0	1.0E+2
Ce-144	1.1E-1	9.0E-1	8.8E-2	1.3E+0	1.1E-1	2.2E+0
Pr-143	1.1E+0	8.6E+0	8.4E-1	1.2E+1	1.0E+0	2.1E+1
Nd-147	1.4E+0	1.1E+1	1.1E+0	1.6E+1	1.4E+0	2.8E+1
Np-239	7.1E+0	5.6E+1	5.2E+0	7.6E+1	6.1E+0	1.2E+2
Pu-238[g]	2.6E-3	2.0E-2	2.0E-3	2.9E-2	1.2E-3	2.4E-2
Pu-239[g]	2.2E-3	1.8E-2	1.7E-3	2.4E-2	9.7E-4	2.0E-2
Pu-240[g]	2.2E-3	1.8E-2	1.7E-3	2.4E-2	9.7E-4	2.0E-2
Pu-241[g]	1.1E-1	8.6E-1	8.0E-2	1.2E+0	4.9E-2	9.9E-1
Am-241[g]	4.3E-4	3.4E-3	3.2E-4	4.7E-3	1.9E-4	3.9E-3
Cm-242[g]	2.3E-2	1.8E-1	1.7E-2	2.5E-1	1.0E-2	2.1E-1
Cm-244[g]	8.8E-4	6.9E-3	6.5E-4	9.4E-3	3.9E-4	7.8E-3

Table K-6 - Continued

Notes:

[a] Assumes an ingestion period equivalent to the shorter time interval of the radionuclide mean **effective** lifetime (includes weathering and radioactive decay) or 30 days. Leafy vegetables and other produce are ingested at the rates given in NRC Reg. Guide 1.109 (see page K-20) for the maximum exposed individual. Produce and leafy vegetables are assumed to remain in the garden or field until the time of consumption.

[b] The derived response level for each radionuclide is capable of producing the preventive PAG dose. Therefore, if more than one radionuclide is present in the sample, the sum of ratios technique must be used to estimate the individual radionuclide concentrations that are permissible, e.g.,

$$\frac{\text{Conc A}}{\text{Response Level A}} + \ldots + \frac{\text{Conc X}}{\text{Response Level X}} = \leq 1$$

[c] Calculated concentrations may vary if calculation assumptions concerning ingestion rates and dose conversion factors are different from those presented in NRC Reg. Guide 1.109.

[d] Produce = Non-leafy vegetables, fruits, and grains.

[e] $1.3E-1 = 1.3 \times 10^{-1} = 0.13$.

[f] Adult dose conversion factors (DCFs) were obtained from ICRP-30; dose conversion factors for other age groups were estimated by multiplying these adult DCFs by DCF ratios $\left(\frac{\text{other age group}}{\text{adult}}\right)$ presented in NRC Reg. Guide 1.109 for other nuclides having similar critical organs and retention times.

Source: FEMA REP-13, 1990, pp. 2-12, 2-13.

Table K-7

1 YEAR DERIVED PREVENTIVE RESPONSE LEVELS FOR VEGETABLE[a,b]

	Adult[c]		Teen[c]		Child[c]	
Nuclide	Produce[d] µCi/kg	Leafy[e] µCi/kg	Produce µCi/kg	Leafy µCi/kg	Produce µCi/kg	Leafy µCi/kg
I-131	8.9E-2[f]	7.0E-1	4.6E-2	6.6E-1	2.3E-2	4.7E-1
I-132	1.7E+1	1.4E+2	1.1E+1	1.6E+2	4.8E+0	9.8E+1
I-133	3.4E+0	2.7E+1	2.1E+0	3.0E+1	9.0E-1	1.8E+1
Rb-86	9.1E-1	7.1E+0	1.1E+0	1.6E+1	4.6E-1	9.4E+0
Cs-134	1.7E-2	1.3E-1	4.7E-3	6.8E-2	2.9E-3	5.9E-2
Cs-136	1.1E+0	8.4E+0	6.7E-1	9.6E+0	4.3E-1	8.6E+0
Cs-137	1.6E-2	1.3E-1	5.4E-3	7.7E-2	3.0E-3	6.1E-2
Te-127m	1.2E-1	9.4E-1	6.9E-2	9.9E-1	4.0E-2	8.1E-1
Te-131	3.4E+0	2.7E+1	2.5E+0	3.6E+1	2.8E+0	5.7E+1
Te-132	1.4E+0	1.1E+1	1.3E+0	1.9E+1	2.4E+0	5.0E+1
Sb-127[g]	1.4E+0	1.1E-1	1.2E+0	1.8E+1	2.4E+0	4.8E+1
Sr-89	5.7E-1	4.5E+0	1.3E-2	1.8E-1	5.2E-3	1.1E-1
Sr-90	1.4E-3	1.1E-2	9.6E-5	1.4E-3	5.7E-5	1.2E-3
Ba-140	6.7E-1	5.3E+0	5.2E-1	7.5E+0	3.4E-1	6.8E+0
Mo-99	1.3E+1	1.0E+2	7.6E+0	1.1E+2	4.5E+0	9.1E+1
Ru-103	4.2E-1	3.3E+0	3.5E-1	5.0E+0	4.8E-1	9.7E+0
Ru-106	7.2E-3	5.7E-2	5.6E-3	8.1E-2	7.1E-3	1.4E-1
Rh-105[g]	1.7E+1	1.3E+2	1.5E+1	2.2E+2	2.9E+1	5.9E+2
Co-58	3.3E-1	2.6E+0	3.1E-1	4.4E+0	4.8E-1	9.7E+0
Co-60	2.6E-2	2.0E-1	2.3E-2	3.3E-1	3.5E-2	7.1E-1
Y-90	9.6E-3	7.5E-2	7.1E-3	1.0E-1	8.3E-3	1.7E-1
Y-91	7.8E-2	6.2E-1	6.0E-2	8.7E-1	7.6E-2	1.5E+0
Zr-95	1.8E-1	1.4E+0	1.5E-1	2.2E+0	2.1E-1	4.2E+0
Zr-97	4.8E+0	3.8E+1	3.3E+0	4.7E+1	3.3E+0	6.7E+1
Nb-95	4.9E-1	3.8E+0	4.3E-1	6.2E+0	6.3E-1	1.3E+1
La-140	3.0E-1	2.4E+0	2.3E-1	3.4E+0	2.8E-1	5.8E+0
Ce-141	4.6E-1	3.6E+0	3.6E-1	5.2E+0	4.5E-1	9.1E+0
Ce-143	5.7E+0	4.5E+1	4.1E+0	6.0E+1	4.7E+0	9.5E+1
Ce-144	8.3E-3	6.5E-2	6.4E-3	9.3E-2	8.0E-3	1.6E-1
Pr-143	6.5E-1	5.1E+0	5.0E-1	7.2E+0	6.2E-1	1.2E+1
Nd-147	9.2E-1	7.3E+0	7.2E-1	1.0E+1	9.0E-1	1.8E+1
Np-239	6.3E+0	5.0E+1	4.7E+0	6.7E+1	5.5E+0	1.1E+2
Pu-238[g]	1.4E-4	1.1E-3	1.1E-4	1.6E-3	6.4E-5	1.3E-3
Pu-239[g]	1.2E-4	9.8E-4	9.3E-5	1.3E-3	5.4E-5	1.1E-3
Pu-240[g]	1.2E-4	9.8E-4	9.3E-5	1.3E-3	5.4E-5	1.1E-3
Pu-241[g]	6.2E-3	4.8E-2	4.5E-3	6.5E-2	2.7E-3	5.6E-2
Am-241[g]	2.4E-5	1.9E-4	1.8E-5	2.6E-4	1.1E-5	2.2E-4
Cm-242[g]	2.9E-3	2.3E-2	2.2E-3	3.1E-2	1.3E-3	2.6E-2
Cm-244[g]	4.9E-5	3.9E-4	3.7E-5	5.3E-4	2.2E-5	4.4E-4

Table K-7 - Continued

1 YEAR DERIVED PREVENTIVE RESPONSE LEVELS FOR VEGETABLE

Notes:

[a] Assumes an ingestion period equivalent to the shorter time interval of the radionuclide mean lifetime (assumes no weathering) or 365 days. Leafy vegetables and other produce are ingested at the rates given in NRC Reg. Guide 1.109 (see page K-20) for the maximum exposed individual. Also, assumes that the produce and leafy vegetables are harvested shortly after a contaminating event.

[b] The derived response level for each radionuclide is capable of producing the preventive PAG dose. Therefore, if more than one radionuclide is present in the sample, the sum of ratios technique must be used to estimate the individual radionuclide concentrations that are permissible, e.g.,

$$\frac{\text{Conc A}}{\text{Response Level A}} + \ldots + \frac{\text{Conc X}}{\text{Response Level X}} = \leq 1$$

[c] Calculated concentrations may vary if calculation assumptions concerning ingestion rates and dose conversion factors are different from those presented in NRC Reg. Guide 1.109.

[d] Produce = Non-leafy vegetables, fruits, and grains.

[e] Leafy = leafy vegetables.

[f] 8.9E-2 = 8.9×10^{-2} = 0.089.

[g] Adult dose conversion factors (DCFs) were obtained from ICRP-30; dose conversion factors for other age groups were estimated by multiplying these adult DCFs by DCF ratios $\left(\frac{\text{other age group}}{\text{adult}}\right)$ presented in NRC Reg. Guide 1.109 for other nuclides having similar critical organs and retention times.

Source: FEMA REP-13, 1990, pp. 2-10, 2-11.

Table K-8

FDA PROTECTIVE ACTION GUIDES FOR INGESTION OF CONTAMINATED FOODS

PAG	Organ of Interest	Projected Dose Commitment in Rem
Preventive[1] (Lower Impact)	Whole body, bone marrow, or any other organ	0.5
	Thyroid	1.5
Emergency[2] (High Impact)	Whole body, bone marrow, or any other organ	5
	Thyroid	15

1 Preventive PAGs are applicable to situations where protective actions causing minimal impact on the food supply are appropriate. A preventive PAG establishes a level at which responsible officials should take protective actions having minimal impact to prevent or reduce the radioactive contamination of food or animal feed.

2 Emergency PAGs are applicable to incidents where protective actions of greater impact on the food supply are justified because of the projected health hazards. An emergency PAG establishes a level at which responsible officials should isolate food containing radioactivity to prevents its introduction into commerce, and at which the responsible officials must determine whether condemnation or another disposition is appropriate.

Table K-9

INGESTION PROTECTIVE ACTIONS
PREVENTIVE PAG

Pasture:

- Removal of lactating (milk-producing) dairy animals from pasture and substitution of uncontaminated feed.

- Substitute uncontaminated water.

Milk:

- Withhold contaminated milk from market to allow decay.

- Diversion of fluid milk for production of dry whole milk, non-fat dry milk, butter, cheese or evaporated milk.

Fruits and Vegetables:

- Wash, brush, scrub or peel to remove surface contamination.

- Preserve by canning, freezing, dehydration or storage to permit decay.

Grains:

- Milling

- Polishing

Other Foods:

- Processing to remove surface contamination.

Table K-9 - Continued

INGESTION PROTECTIVE ACTIONS
EMERGENCY PAG

Isolate food to prevent introduction into commerce and determine whether condemnation or other disposition is appropriate.

Before taking action consider the following factors:

- Availability of other protective actions.
- Relative proportion of contaminated food in total diet.
- Importance of the food and availability of uncontaminated substitutes.
- Contribution of radioisotopes in other foods to total dose.
- Time and effect required to implement corrective action.
- Exposure to food processing workers.

Table K-10

DAILY CONSUMPTION FROM EPA PAG MANUAL, 1989, APP. D, P. 15

Food	Average consumption for the general population g/day	% of total diet
Milk, cream, cheese, ice cream[a]	567.5	27.2
Fats, oils	54.5	2.6
Flour, cereal	90.8	4.3
Bakery products	149.8	7.2
Meat	217.9	10.4
Poultry	54.5	2.6
Fish and shellfish	22.7	1.1
Eggs	54.5	2.6
Sugar, syrups, honey, molasses, etc.	72.6	3.5
Potatoes, sweet potatoes	104.4	5.0
Vegetables (excluding potatoes) fresh	145.3	7.0
Vegetables canned, frozen, dried	77.2	3.7
Vegetables juice (single strength)	9.1	0.4
Fruit, fresh	163.4	7.8
Fruit canned, frozen, dried	36.3	1.7
Fruit juice (single strength)	45.4	2.2
Other beverages (soft drinks, coffee, alcoholic bvgs.)	177.1	8.5
Soup and gravies (mostly condensed)	36.3	1.7
Nuts and peanut butter	9.1	0.4
Total	2088.1	99.9

[a]Expressed as calcium equivalent; that is, the quantity of whole fluid milk to which dairy products are equivalent in calcium content.

(From the U.S. Department of Agriculture Household Food Consumption Survey, 1965-1966)

AVERAGE INDIVIDUAL CONSUMPTION FROM REGULATORY GUIDE 1.109

Pathway	Child	Teen	Adult
Fruits, vegetables, & grain (kg/yr)	200	240	190
Milk (ℓ/yr)	170	200	110
Meat & poultry (kg/yr)	37	59	95
Fish (kg/yr)	2.2	5.2	6.9
Seafood (kg/yr)	0.33	0.75	1.0
Drinking water (ℓ/yr)	260	260	260
Shoreline recreation (hr/yr)	9.5	47	8.3
Inhalation (m^3/yr)	3700	8000	8000

Table K-11

ADULT INGESTION DOSE CONVERSION FACTORS
mrem/µCi
COMMITTED (50 YEAR) DOSE EQUIVALENT

Nuclide	R. Marrow	Thyroid	Effective
H-3	6.29E-02	6.29E-02	6.29E-02
C-14	2.07E+00	2.07E+00	2.07E+00
Na-22	1.55E+01	9.25E+00	1.15E+01
Na-24	1.37E+00	9.62E-01	1.41E+00
P-32	2.96E+01	2.41E+00	8.51E+00
P-33	1.81E+00	3.44E-01	8.88E-01
S-35	2.81E-01	2.81E-01	4.44E-01
Cl-36	2.92E+00	2.92E+00	3.00E+00
K-40	1.81E+01	1.78E+01	1.85E+01
K-42	7.40E-01	7.40E-01	1.11E+00
Ca-45	1.26E+01	2.07E-01	3.15E+00
Sc-46	1.48E+00	2.81E-02	6.29E+00
Ti-44	1.11E+01	8.51E+00	2.29E+01
V-48	2.18E+00	4.44E-02	8.51E+00
Cr-51	4.44E-02	1.37E-02	1.44E-01
Mn-54	1.78E+00	4.81E-01	2.74E+00
Mn-56	8.88E-02	8.88E-03	9.62E-01
Fe-55	3.70E-01	4.07E-01	5.92E-01
Fe-59	3.11E+00	2.22E+00	6.66E+00
Co-60	4.81E+00	2.89E+00	9.99E+00
Ni-63	3.15E-01	3.15E-01	5.55E-01
Cu-64	7.03E-02	4.07E-02	4.44E-01
Zn-65	1.66E+01	1.18E+01	1.44E+01
Ga-68	2.15E-02	9.62E-04	3.40E-01
Se-75	7.40E+00	4.07E+00	9.62E+00
Rb-86	1.37E+01	7.77E+00	9.25E+00
Sr-89	1.18E+01	8.88E-01	9.25E+00
Sr-90	7.03E+02	5.55E+00	1.41E+02
Sr-91	3.70E-01	8.88E-02	2.48E+00
Y-91	2.41E-02	4.44E-04	9.25E+00
Zr-93	2.74E+00	2.70E-04	1.63E+00
Zr-95	7.77E-01	3.03E-02	3.70E+00
Nb-94	2.70E+00	4.44E-01	7.03E+00
Nb-95	7.03E-01	4.07E-02	2.55E+00
Mo-99	1.96E+00	5.92E-01	3.03E+00
Tc-99	2.22E-01	5.92E+00	1.44E+00
Tc-99m	2.29E-02	3.11E-01	5.92E-02
Ru-103	5.92E-01	2.29E-01	3.03E+00
Ru-105	8.51E-02	6.66E-03	1.04E+00
Ru-106	5.18E+00	5.18E+00	2.74E+01
Ag-110m	3.48E+00	6.66E-01	1.07E+01
Cd-109	1.37E+00	9.99E-01	1.29E+01
Cd-113	1.26E+01	1.26E+01	1.59E+02
In-114m	1.29E+01	4.07E-01	1.70E+01

Table K-11 - Continued

ADULT INGESTION DOSE CONVERSION FACTORS
mrem/μCi
COMMITTED (50 YEAR) DOSE EQUIVALENT

Nuclide	R. Marrow	Thyroid	Effective
Sn-113	6.29E-01	7.77E-02	3.07E+00
Sn-123	8.88E-01	1.15E-01	8.14E+00
Sn-126	9.99E+00	2.03E+00	1.92E+01
Sb-124	2.26E+00	4.07E-01	9.62E+00
Sb-126	2.66E+00	3.70E-01	9.99E+00
Te-127m	2.00E+01	3.48E-01	8.14E+00
Te-129m	1.29E+01	5.55E-01	1.04E+01
Te-131m	8.88E-01	1.55E+02	8.88E+00
I-125	2.52E-01	1.26E+03	3.70E+01
I-129	8.14E-01	8.88E+03	2.74E+02
I-131	3.48E-01	1.74E+03	5.18E+01
I-135	1.33E-01	6.29E+01	2.22E+00
Cs-134	6.66E+01	6.29E+01	7.03E+01
Cs-136	1.07E+01	9.99E+00	1.11E+01
Cs-137	4.81E+01	4.44E+01	4.81E+01
Ba-133	5.18E+00	7.40E-01	3.37E+00
Ba-140	1.59E+00	1.92E-01	9.25E+00
La-140	1.04E+00	2.37E-02	8.14E+00
Ce-144	3.29E-01	1.89E-02	2.07E+01
Pm-145	1.96E-01	1.67E-03	4.44E-01
Pm-147	7.40E-02	1.15E-07	1.04E+00
Sm-151	9.99E-02	1.18E-07	3.70E-01
Eu-152	3.37E+00	2.44E-01	6.29E+00
Eu-154	4.07E+00	2.11E-01	9.25E+00
Eu-155	5.55E-01	6.29E-03	1.52E+00
Gd-153	2.96E-01	7.77E-04	1.15E+00
Tb-160	9.25E-01	1.55E-02	6.66E+00
Ho-166m	3.00E+00	2.03E-01	7.77E+00
Tm-170	1.11E-01	1.48E-03	5.18E+00
Yb-169	5.92E-01	1.33E-03	3.00E+00
Hf-172	3.48E+00	2.26E-01	4.44E+00
Hf-181	6.66E-01	1.33E-02	4.44E+00
Ta-182	1.11E+00	3.33E-02	6.29E+00
W-187	2.15E-01	2.85E-03	2.74E+00
Ir-192	9.25E-01	1.37E-01	5.55E+00
Au-198	3.15E-01	6.66E-02	4.07E+00
Hg-203	3.44E-01	7.77E-02	2.29E+00
Tl-204	2.41E+00	2.41E+00	3.33E+00
Pb-210	5.18E+03	4.44E+02	5.18E+03
Bi-207	1.26E+00	4.44E-02	5.18E+00
Bi-210	7.03E-02	7.03E-02	6.29E+00
Po-210	3.03E+02	3.03E+02	1.89E+03
Ra-226	2.18E+03	3.37E+02	1.30E+03
Ra-227	1.59E-01	6.66E-03	2.26E-01
Ac-227	2.00E+04	2.78E-01	1.41E+04
Ac-228	9.99E-01	3.44E-03	2.15E+00

Table K-11 - Continued

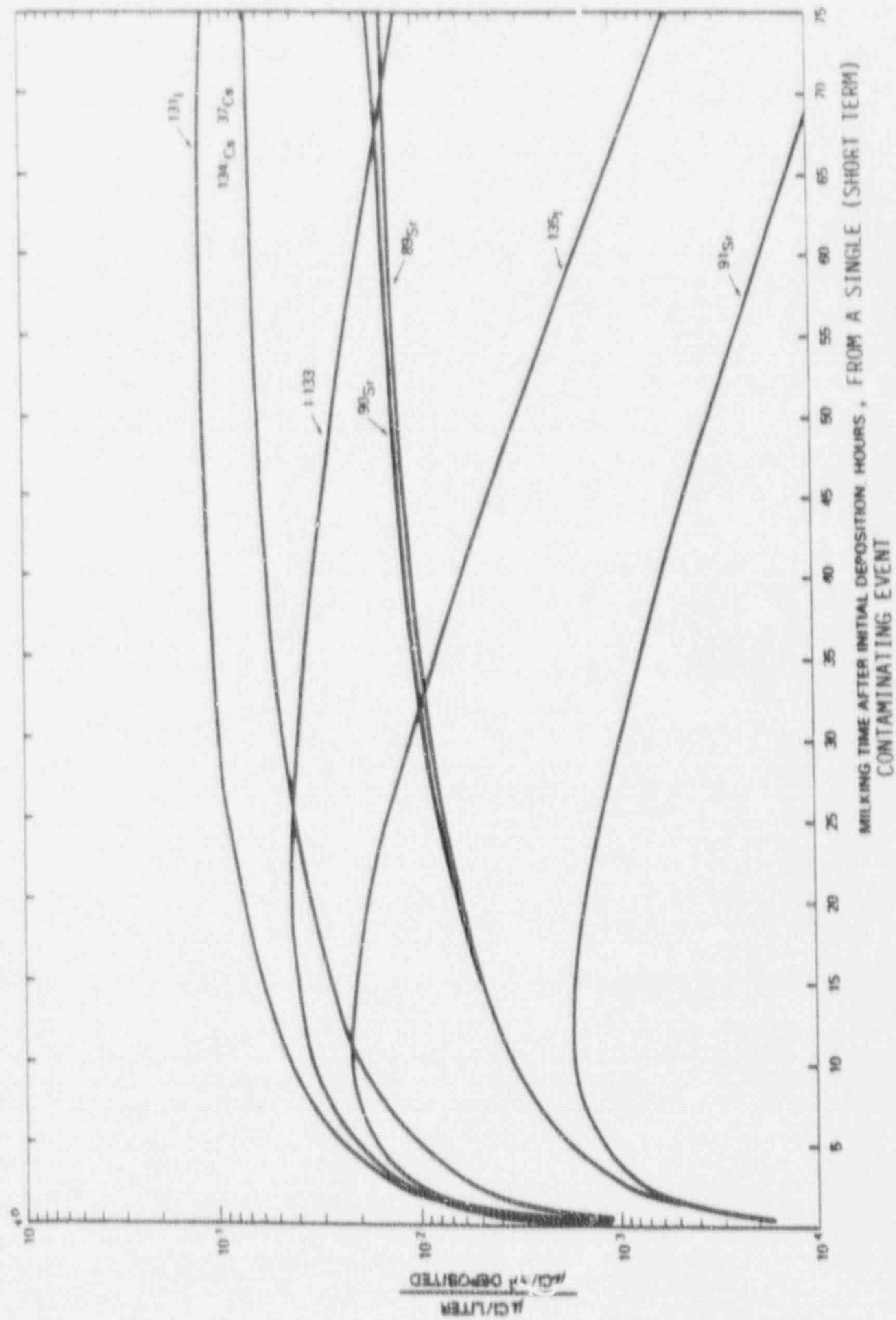
ADULT INGESTION DOSE CONVERSION FACTORS
mrem/μCi
COMMITTED (50 YEAR) DOSE EQUIVALENT

Nuclide	R. Marrow	Thyroid	Effective
Th-227	2.07E+01	4.44E-01	3.70E+01
Th-228	7.03E+02	8.51E+00	3.70E+02
Th-230	1.04E+03	2.52E+00	5.18E+02
Th-232	5.18E+03	4.44E+00	2.70E+03
Pa-231	2.11E+04	2.33E-01	1.04E+04
U-232	1.52E+03	3.00E+01	1.30E+03
U-234	2.66E+02	9.25E+00	2.81E+02
U-235	2.52E+02	8.88E+00	2.63E+02
U-238	2.52E+02	8.51E+00	2.52E+02
Np-237	7.77E+03	4.07E-01	4.44E+03
Np-239	1.70E-01	7.40E-04	3.26E+00
Pu-236	1.63E+03	5.55E-02	1.15E+03
Pu-238	4.44E+03	2.92E-02	3.18E+03
Pu-239	5.18E+03	2.74E-02	3.52E+03
Pu-240	5.18E+03	2.78E-02	3.52E+03
Pu-241	9.99E+01	3.70E-04	6.66E+01
Pu-242	4.81E+03	2.66E-02	3.33E+03
Am-241	5.18E+03	4.81E-02	3.63E+03
Am-242m	5.18E+03	1.37E-02	3.52E+03
Am-243	5.18E+03	2.52E-01	3.59E+03
Cm-242	1.30E+02	3.26E-02	1.15E+02
Cm-243	3.63E+03	1.15E-01	2.48E+03
Cm-244	2.89E+03	3.11E-02	2.00E+03
Cm-245	5.18E+03	1.11E-01	3.70E+03
Cf-252	1.70E+03	9.62E-01	1.07E+03

Source: Adapted from EPA-520/1-88-020, 1988, pp. 155-179.

Figure K-1

MILK CONCENTRATION VS. TIME AFTER DEPOSITION



Source: FEMA REP-12, 1987, pp. D-4 thru D-6.

Exposure Control for NRC During
Rescue and Recovery Activities

Section L

L

Exposure Control for NRC During Rescue and Recovery Activities

EXPOSURE CONTROL FOR NRC DURING RESCUE AND RECOVERY ACTIVITIES

Objective

Respond to questions concerning protective actions for NRC emergency workers.

Guidance

Step 1

Review NRC Manual Chapter 0524, Appendix Part 6, "Guidance for Emergency Exposure Control During Rescue and Recovery Activities," pages L-3 through L-7.

Step 2

Assure that all parties understand that:

1. All personnel not part of an emergency response organization should take the same protective actions recommended to the general population.

2. Emergency actions required for life saving or limiting the possible consequences of an accident may require doses in excess of the NRC regulatory limits (10 CFR Part 20).

3. Doses received should be recorded.

Step 3

Consider:

1. How critical is the function? Have alternatives been considered?

2. Is emergency worker dose being continuously monitored (personal dosimeter or monitoring instruments)? Is projected dose (commitment) known well enough to ensure that exposure received does not significantly exceed projections?

CAUTION:

> Some plant areas may require very high range monitoring instruments (1000 R/hr).

3. Can communications be maintained during mission?

4. Is preplanning possible to minimize dose?

5. Are volunteers familiar with the task to be done and capable of doing it?

6. Are provisions to minimize thyroid dose being taken (e.g., KI, shelter, normal or improvised respiratory protection).

7. Were the general concepts of NRC Manual Chapter 0524 considered?

8. Doses should be limited to the guidance in the EPA PAGS (see Table L-1, page L-8).

END

PART 6

GUIDANCE FOR EMERGENCY EXPOSURE CONTROL DURING RESCUE AND RECOVERY ACTIVITIES

A. Purpose. The emergency action guidance promulgated in this part provides instructions and background information for NRC employees who are involved in activities during an emergency to use in determining appropriate actions for the rescue and recovery of persons and the protection of health and property.

B. General Considerations.

1. The guiding principle for performing rescue and recovery operations is to minimize the risk of injury to those persons involved in these operations; to control radiation exposures, consistent with the immediate objective of saving human life; to recover deceased victims, and to protect health and preserve property. Performing rescue and recovery operations requires the exercise of prompt judgement to determine multiple hazards that may be involved with the operations and alternate methods of accomplishing these operations. Sound judgement and flexibility of action are crucial to the success of any type of emergency action.

2. To avoid restricting actions that may be necessary to save lives, these instructions do not establish a rigid upper limit of exposure but rather leave judgment up to the official in charge of emergency operations to determine the amount of exposure that should be permitted to perform the emergency mission.

3. The official in charge of the rescue or recovery activity shall carefully examine any proposed action involving further radiation exposure by weighing the risks of radiation exposure, actual or potential, against the benefits to be gained. The magnitude of the expected dose, biological consequences related to this dose, and the number of people involved are the essential elements to be evaluated in making a risk determination. The following table lists some of the biological effects associated with various radiation doses.

BIOLOGICAL EFFECTS OF ACUTE RADIATION EXPOSURES

Radiation Dose	Acute Effect
10 rads, whole body	Elevated number of chromosome aberrations; no clinical injury or symptoms
25-50 rads, whole body	Changes in number of blood cells
100 rads, whole body	Mild radiation sickness - nausea, vomiting, and fatigue are possible
400 rads, whole body	Probably will result in death for 50% of exposed and untreated population
600-900 rads, locally to eye	Cataract probable
1000 rads, to skin	Erythema and blistering
2000-5000 rads, whole body	Death in 3-10 days from intestinal tract damage
10,000 rads or more, whole body	Death within 48 hours owing to central nervous system damage

4. In these instructions the criteria for accident situations that involve saving lives differs from the criteria for recovering deceased victims or saving property. In the latter instances, the amount of expected dose received by persons should be controlled as much as possible within occupational limits. In this context, the use of potassium iodide should be considered to minimize thyroid doses. Scientific data shows that to maximize the benefit of dose reduction, potassium iodide must be taken several hours before the planned exposure.

C. Emergency Situations. Specific dose criteria and judgement factors are set forth for the three categories of risk-benefit considerations: actions that involve (1) saving human life, (2) recovering deceased victims, and (3) protecting health and property. These actions should be coordinated with those described in the licensee's emergency plans or other existing plans to avoid any appreciable differences between the dose criteria and judgment factors used by the NRC and the licensee. These actions should not be limited to the rescue of NRC employees and NRC contractors alone, but should apply to licensee employees, contractors, and visitors as well.

1. Saving Human Life.

   a. Where there is reasonable expectation that an individual is alive within the affected areas during an emergency, the course of action to be pursued should be determined by the official on site having responsibility for emergency actions.

   b. The offical on site having responsibility for the emergency action shall determine the amount of exposure for this type of emergency action. The official should immediately evaluate the situation and establish the exposure limit for the rescue mission based on the following:

      (1) Consideration of the inherent risks, including --

         (a) The reliability of the prediction of radiation injury. This reliability cannot be any greater than the reliability of the estimation of the dose. Therefore, consideration should be given to limits of error associated with the specific instruments and techniques used to estimate the dose rate. This is especially crucial when the estimated dose approximates 100 rems or more.

         (b) The dose expected in performing the action must be weighed in terms of the effects of acute external whole-body exposure and entry of radioactive material into the body. The official in charge of emergency action at the incident scene may permit volunteers to receive an expected dose up to, but not in excess of, 75 rems.

      (2) Current assessment of the degree and nature of the hazard, and the capability of reducing inherent risk from that hazard through appropriate mechanisms such as the use of protective equipment, remote manipulation equipment, or similar means.

   c. When making a decision to perform the action, the risk to rescue personnel should be weighed against the probability of success of the rescue action.

   d. Any rescue action that may involve substantial personal risk must be performed by volunteers, and each emergency worker must be advised of the known or estimated extent of the risk before participation. Preference should be given to volunteers who meet the following criteria:

      (1) Over the effective reproductive age of 45 years.

      (2) Physically fit and not overweight.

      (3) In good physical condition as determined by recent physical examination (e.g., no adverse heart conditions).

2. Recovering Deceased Victims.

a. Accidents that involve recovering deceased victims require criteria different from those for saving lives. Because the element of time is no longer a critical factor, the recovery of deceased victims should be well planned. The amount of radiation exposure received by persons in these recovery operations must be within existing occupational exposure guidelines, that is, 3 rems per quarter or 5 (N-18) rems, whichever is more limiting.

b. In those situations where the bodies are located in areas that are inaccessible because of high direct radiation fields, and where the recovery mission would result in exposure in excess of occupational exposure standards contained in this appendix, special remote recovery devices should be used to retrieve the bodies.

c. In special circumstances where it is impossible to recover bodies without the entry of emergency workers into the area, the individual in charge of the recovery mission may determine it necessary to exceed the quarterly occupational exposure standards contained in this appendix. The planned whole-body exposure of an individual participating in the recovery should not exceed 12 rems total for the year or 5 (N-18) rems, whichever is the more limiting.

3. Protecting Health and Property. Where the risk (probability and magnitude) of the radiation hazard either bears significantly on the state of health of people or may result in loss of property, so that immediate remedial action is required, the following criteria apply:

a. When the official in charge of emergency action on site deems it essential to reduce a hazard potential to acceptable levels or to prevent a substantial loss of property, a planned exposure up to, but not to exceed, 12 rems for the year or 5 (N-18) rems, whichever is more limiting, may be received by individuals participating in the operation.

However, the official in charge of emergency action at the incident scene may elect, under special circumstances, to waive these limits and permit volunteers to receive an exposure up to, but not to exceed, 25 rems.

b. Where the potential risk of radiation hazard following the nuclear incident is such that life would be in jeopardy, or that there would be severe effects on health of the public or loss of property inimicable to the public safety, the criteria under Section C.1 of this part for the saving of human life apply.

D. Implementation. Each regional office shall establish a plan to implement this guidance for emergency response during rescue and recovery activities. This plan shall --

1. Identify an individual by position or title who can authorize emergency workers to receive doses in excess of 10 CFR Part 20 limits. The regional administrator in each region should be designated as the lead person in charge of emergency action for the respective region.

2. Provide to all appropriate regional personnel periodic training (with written exams) on the emergency exposure procedures to be followed during rescue and recovery activities.

Table L-1

RECOMMENDED PROTECTIVE ACTIONS TO REDUCE WHOLE BODY AND THYROID DOSE FROM EXPOSURE TO A GASEOUS PLUME

Projected Dose (rem) to Emergency Team Workers	Recommended Actions[a]	Comments
Whole body 25 Thyroid 125	Control exposure of emergency team members to these levels except for lifesaving missions. (Appropriate controls for emergency workers include time limitations, respirators, and stable iodine.)	Although respirators and stable iodine should be used where effective to control dose to emergency team workers, thyroid dose may not be a limiting factor for lifesaving missions.
Whole body 75	Control exposures of emergency team members performing lifesaving missions to this level. (Control of time of exposure will be most effective.)	

[a] These actions are recommended for planning purposes. Protective action decisions at the time of the incident must take existing conditions into consideration.

Source: EPA PAG MANUAL, 1989

Use of Potassium Iodide (KI) and Thyroid Monitoring

Section M

M Use of Potassium Iodide (KI) and Thyroid Monitoring

USE OF POTASSIUM IODIDE (KI) AND THYROID MONITORING

Objective

Provide information on the use of potassium iodide (KI).

Guidance

Step 1

CAUTION:

KI is not an adequate substitute for prompt evacuation or sheltering of the general population near a plant for a severe reactor accident. KI should not be distributed to the general population if it will delay other immediate protective actions.

NOTE:

The decision to use KI to protect the public lies with the State and local health authorities.

Consult with the representatives of other Federal agencies (FEMA, HHS, EPA).

Step 2

Refer to the following information to answer questions:

1. Federal Position

   Recommend the distribution of KI for thyroid blocking to emergency workers, site personnel, and to institutionalized individuals who might have difficulty evacuating during an emergency. This recommendation is based on the individuals receiving an estimated projected dose commitment of 25 rem to the thyroid.

2. Dosage

   The FDA recommended dosage of KI is 130 mg/day for adults and children above one year, and 65 mg/day for children below one year of age (130 mg of KI contains 100 mg of stable iodine).

   To be most effective, KI should be taken shortly before or just after the intake of radioactive iodine. KI should be administered for at least 3 days after an acute exposure as it takes approximately 48 hours for most of the radioiodine to be excreted in the urine.

3. Effectiveness

The effectiveness of KI in blocking uptake of radioiodine by the thyroid depends strongly on the timing of the dose of KI relative to the exposure to radioactive iodine. KI is most effective when taken just before or within 1-2 hours after exposure to radioiodine. Taking the recommended dosages of KI just before or at the time of exposure can provide greater than 90% blocking of radioactive iodine uptake by the thyroid (see Figure M-1, page M-3). If KI is taken approximately 3 to 4 hours after acute exposure, about 50% blocking could still occur. Once radioactive iodine has concentrated in the thyroid, KI is not very effective at removing it.

4. Thyroid Monitoring

Field determination of iodine-131 uptake can be made by holding a gamma detector in a horizontal position immediately below an individual's adam's apple. From the count rate an approximation of the thyroid uptake can be obtained. Several detectors have been evaluated using this technique and the results are given in the Table below. Assume: Adult Thyroid Dose (rads) = 6.5 x I-131 μCi. (Source: FDA-83-8211, Preparedness and Response in Radiation Accidents.)

APPROXIMATE GAMMA INSTRUMENT RESPONSE TO RADIOIODINE IN THE THYROID

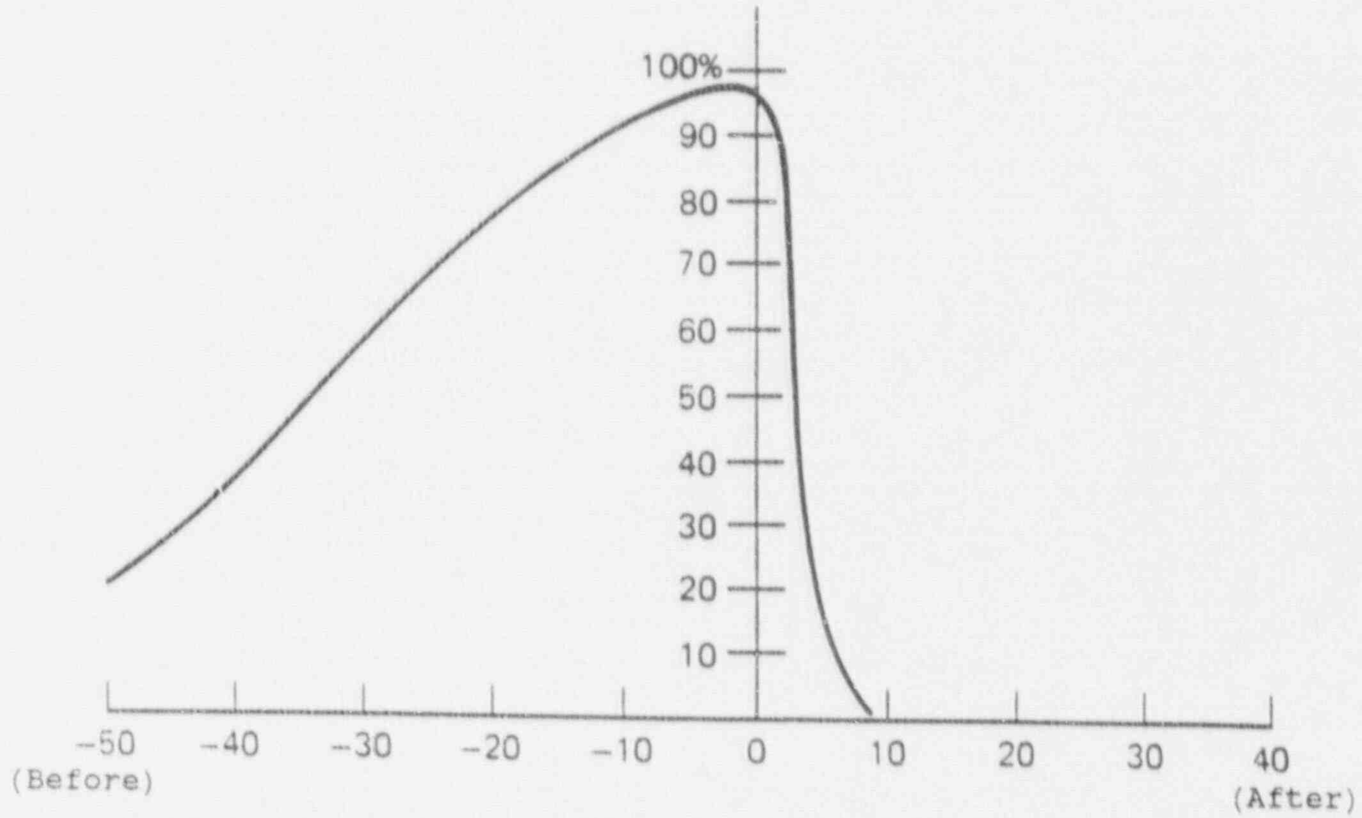
Instrument	Gamma Response at Surface of Neck of Adult Thyroid (20 gm) (cpm/μCi in gland)
GM instrument (CDV-700)	
OCD-D-103 tube[a]	Approximately 50
Victoreen 6306 tube	Approximately 500
End-window GM instrument[b]	Approximately 150
Ludlum model 44-3 gamma scintillator[c]	Approximately 4900

[a]Beta shield closed (approximately 1000 mg/cm^2)
[b]End-window facing the neck
[c]End of probe facing the neck

END

Figure M-1

PERCENT OF THYROID BLOCKING AFFORDED BY 130 mg of KI (100 mg OF IODINE)



TIME BEFORE OR AFTER INTAKE OF RADIOACTIVE IODINE WHEN KI IS TAKEN

Ref. Figure 4.3 in L.A. Il'in, G. V. Arkhangel'skaya, Yu.O. Konstantinov, and I.A. Likhtavev, Radioactive Iodine in the Problem of Radiation Safety, US AEC, June 1974 (translation from Russian book Radioaklivnyi iod u probleme radiatsionnoi bezopasnosti, Atomizdat, 1972).

Use of the NRC TLD Direct Radiation Monitoring Network

Section N

N Use of the NRC TLD Direct Radiation Monitoring Network

USE OF THE NRC TLD DIRECT RADIATION MONITORING NETWORK

Proceed to appropriate procedure:

ACTIVATION OF NRC TLD SUPPORT

Objective

Activation of and initial coordination with the Region I, TLD Direct Radiation Monitoring Network during a radiological emergency.

Guidance

NOTE:

> Region I maintains a network of TLDs around each commercial power reactor. The Region I staff who maintain this capability can provide TLD support for any radiological emergency to include non-reactor emergencies where TLDs may be effective in estimating offsite release over the course of the accident or supplemental personnel dosimetry. See page N-4 for a further description of this capability.

CAUTION:

> The TLDs (UD 801) currently used have a practical lower limit of detection of several mR integrated exposure for short exposure periods. An exposure of 1 mR represents about four days of background radiation.

CAUTION:

> NRC TLDs have not been qualified at present time under NVLAP criteria for personnel dosimetry.

CAUTION:

> TLDs cannot be used in making protective action decisions.

Step 1

Consider activation of Region I TLD support if there is a:

1. Potential for a release involving a risk to public health or involving considerable public concern.
2. Potential for a release during recovery from an accident (e.g., fission products in containment).
3. Site Area or General Emergency at a reactor.
4. Need for supplemental personnel dosimetry.

Step 2

Contact the Region I Dosimetry Group during normal hours (A) or after hours (B) by:

A. **Normal hours:** calling Region I at 215-337-5000 and asking for the Chief, Effluents Radiation Protection Section or his/her supervisor.

or

B. **After hours:** calling the NRC Headquarters Operations Center (HOO) on 301-951-0550 and asking for the Regional Duty Officer (RDO). Ask the RDO to place you in contact with Chief, Effluents Radiation Protection Section or his/her supervisor.

Step 3

Consult with Region I Chief, Effluents Radiation Protection Section or supervisor to discuss:

- Event location and status.
- Assistance required.
- Estimated time of arrival.
- Methods of coordination upon arrival at site location (e.g., through the HOO).
- Location where Region I personnel can operate and avoid unnecessary exposure of annealed TLDs.
- Methods to advise Region I personnel on radiological conditions upon arrival in general area of accident.
- Points of contact and health physics support needs.
- How to formally request the assistance if the NRC response organization has not been activated.

Step 4

Formally request Region I TLD support during activation of the NRC response organization (A) or when the NRC response organization is not activated (B) by:

A. **NRC Response Organization Activated (Standby, Initial or Expanded Activation):** calling the Headquarters Operations Center (301-951-0550) and providing the request to the Executive Team or Executive Team member.

or

B. **NRC Response Organization Not Activated:** as agreed upon in Step 3.

NRC TLD DIRECT RADIATION MONITORING NETWORK AND ITS USE DURING RADIOLOGICAL EMERGENCIES

Introduction and Purpose

The NRC TLD Direct Radiation Monitoring Network is a resource that can be used during and following a radiological emergency. The particular use(s) of the Network is dependent on the timing and duration of any releases and the response time of the NRC Dosimetry Group (Region I) to the incident site. The Dosimetry Group is defined as the Region I Dosimetry Specialist, his/her supervisor (Chief, Effluents Radiation Protection Section) and a dosimetry assistant. A dosimetry assistant is another Region I individual who is knowledgeable of the NRC TLD Network and system.

The overall purpose of the NRC TLD Network and any augmentation of it during a radiological emergency is to provide a geographical profile of radiological exposures or incremental exposures around a site, should significant radiological releases occur. In addition, the NRC TLD Network will be useful in confirming the licensee's initial and periodic dose assessments. Such confirmations will be important since the public's confidence in the licensee's measurements is likely to degrade in the event of an emergency.

The following sections discuss particular uses of the Network depending on the time at which Region I can respond and the phase of emergency releases at that time.

A. Uses of the NRC TLD Direct Radiation Monitoring Network During Emergencies

1. Arrival of Region I Dosimetry Group prior to the release or prior to anticipated significant additional releases.

   Many of the analyzed accidents, including those categorized as severe accidents, result in releases that start hours or days following accident initiation, if significant releases occur at all. In such cases the Region I Dosimetry Group could augment the routine network TLDs around the affected reactor by adding TLDs at the existing and/or at additional locations.

   Any augmentation would be at the direction of the Protective Measures Coordinator of the affected Region's Incident Response Site Team or his designee. The purposes for augmentation may be several.

   a) The current Network TLDs may not afford an optimal spacial distribution for monitoring the accident situation. Additional TLDs may be desirable in other sectors, at closer or greater distances, or at nearby population centers or other areas of interest. Those TLDs would supplement the Network TLDs, and those of the licensee and State(s).

Note: 1) See NRC Inspection Manual, Chapter 1420, Appendix, Section 2," General Siting and Placement Criteria", for a description of the general TLD siting philosophy employed in establishing the NRC TLD Direct Radiation Monitoring Network locations.

2) See NUREG-0837, "NRC TLD Direct Radiation Monitoring Network," (any volume or number) for a brief description of the TLD siting around any specific site.

b) Additional NRC TLDs may be desirable for purposes of providing additional quality of measurement checks on other TLDs in place, e.g., those of the licensee, State(s) or other agencies. (Those devices may also be subject to changes in response due to energy or exposure condition changes during an accident). In this case NRC TLDs could be co-located with those of the other entities and exchanged on compatible time schedules.

c) The augmentation may be for purposes of providing supplemental measurements at the locations of the Network TLDs and at locations included in a) and b) above. By posting a duplicate set of dosimeters at these locations, measurements can be made at weekly intervals (or more frequently, if the releases are significant), without disturbing the in-place dosimetry.

Note: The limitation of the TLD system must be recognized when deciding the frequency and intervals at which these supplemental dosimeters are exchanged. The UD 801 TLDs in use have a practical lower limit of detection of several mR integrated exposure for short exposure times. (An exposure of 1 mR represents about four days of background radiation at most sites.) Additionally, if releases are on-going or periodic during the exposure interval, exposure to the TLDs in-transit during the exchange could mask the exposure at the posted field locations. This masking could occur even if the TLDs are provided some shielding while in transit. Nevertheless, the supplemental TLDs could be used to assess upper limits to exposures at locations during the exposure intervals.

As a practical matter, the Protective Measures Team and NRC response management should be aware that the TLD system is a monitoring system not designed to be used in making protective action recommendations. Exposures received by the TLDs have already occurred and cannot be used to project future "dose savings" from radioactive plumes by taking protective actions. (TLD measurements subsequent to the first

TLD exchange following plume exposure can provide an indication of relative ground deposition levels.) TLDs provide the profile (or incremental profiles) of radiological exposure over the geographical area.

2. Arrival of Region I Dosimetry Group at the site after all significant releases have occurred.

Arrival of the Dosimetry Group with additional dosimeters at the accident site following all anticipated significant releases should prompt the following actions from the Protective Measures Coordinator or designee. If the releases were contaminating, that is, iodines or particulates were released in quantities sufficient to result in detectable surface contamination, it would be greatly desirable to exchange the Network TLDs expeditiously. A contaminating release would likely have contaminated the in-place TLD plastic holders and to a lesser extent the weatherproof protective pouch containing the NRC dosimetry. Because of the close proximity of these contaminated surfaces to the dosimetry, this contamination may affect the TLD measurements to a much greater extent than would contamination on the ground. Estimates will be made of the factors and uncertainties affecting the TLD measurements.

If the release has not been contaminating, the Protective Measures Coordinator should still consider exchanging the Network TLDs as soon as practicable following all likely significant releases. This exchange is desirable to allow the development of the geographical exposure profiles while minimizing additional exposure to the TLDs from ambient background levels following plume passage.

The NRC TLDs should <u>not</u> be removed from the field without replacing them. The need for information from the TLDs should <u>not</u> outweigh the need for maintaining continuous monitoring capability. This concern is to maintain either the original TLDs or their replacements at each location throughout the entire interval of interest following an accident, assuring that no monitoring gap has occurred. Any such gap in monitoring would leave open the possibility of, or suspicion of, releases or exposures during the unmonitored interval.

The geographical exposure profiles will be utilized by the NRC, FRMAC and/or other groups in assessing estimated total population exposures, using information available on protective actions taken and the timing of such actions. The NRC TLDs should also be compared to other devices co-located with them in order to correlate and expand the total number of measurements available for this assessment.

3. Use of NRC TLDs as supplemental personnel exposure monitoring devices

The configuration of the Panasonic UD 801 dosimeters with the two lithium borate elements allows the use of these TLDs for supplementing the routine personnel monitoring devices. The 14 mg/cm^2 "open window" also enables an assessment to be made of "non-penetrating" vs. "penetrating" exposure. The NRC TLDs have not been qualified at the present time under NVLAP criteria for personnel dosimetry. Nevertheless, these devices could be utilized to provide certain measurements that may assist in exposure control for NRC personnel. Several examples follow.

a) The devices could be used as supplemental monitors for specific emergency missions. An individual assigned to cover plant activities involving a potential high radiation field could be given a TLD, along with routine dosimetry (assigned TLD or film badge) and pocket dosimeters. Upon completion of the assignment or shift, the NRC TLD could be read and could be used as an indicator to determine whether the exposure received warrants removal of the individual from further radiation work until the routine, assigned dosimetry badge is read. Conservative allowances could be made for use of the NRC TLD, e.g., lower level of detection of 20 mR, uncertainty ±50%. The "open" window capability of the NRC TLD would also provide a better indication of significant beta exposure than would pocket chambers.

b) The NRC TLDs could be used to provide relative exposures to various portions of an individual's body. For example, exposure to extremities (feet or hands), head, etc., could be compared to that received at the location of individual's routine, assigned monitoring device.

c) The NRC TLDs could be employed to indicate that an NRC employee did <u>not</u> exceed a given level of exposure while in an area unlikely to have high radiation levels, or while on site only for a short time. (Following the TMI-2 accident, these same TLDs were used for this purpose by the NRC for Headquarters individuals on site for short times.) The NRC TLD provides some assurance that the individual did not exceed exposure levels based on conservative assumptions (See para. 3.a above).

B. <u>Relationship between the Region I Dosimetry Group and the NRC Incident Response Site Team</u>

The detailed description of the Dosimetry Group/Site Team interaction is described in the Protective Measures Team Procedures PMT-413 and PMT-414. In brief, the technical lead for operating the TLD system, determining the response of the system, evaluating the calibration, limitations and uncertainties of the system, following up with post-accident assessments of the system, etc., resides with the Region I Dosimetry Specialist and the Region I Chief, Effluents Radiation Protection Section. The emergency response use of the TLD system, augmentation of existing Network TLDs, personnel monitoring use, timing and frequency of TLD exchanges, etc., is under the direction of the Site Team's Protective Measures Coordinator or designee. It is expected that close coordination between these lead individuals will ensure the availability of technically valid data on a time scale that is commensurate with the incident response needs.

In application, the Dosimetry Specialist and his management will decide whether it is advisable to ship the manual reader to the incident site. The Region I Dosimetry Specialist can be expected to accompany the reader (if sent), and the annealed augmentation TLDs to the incident site. (If the manual reader is not brought to the incident site, the Dosimetry Specialist will remain in Region I with the automatic reader and equipment and prepare TLDs for the site, read exposed TLDs, interpret measurements, and report them to the Protective Measures Team.) The Region I Dosimetry Group will also bring site maps, if available, showing the locations of the Network TLDs and the current driving routes for TLD exchange. At the site, the Dosimetry Team and a health physicist (generally also from Region I) will perform the dosimetry exchange as directed by the Protective Measures Coordinator or designee. The health physicist is expected to make exposure and contamination measurements at each location (as directed) and to monitor the radiation levels necessary for the protection of the TLD exchange team in the event of a significant release. The dosimetry assistant will assure that the TLDs are transported to Region I for analysis or are read near site with the manual reader. The Dosimetry Specialist and his management will ensure that all appropriate reports are promptly prepared and disseminated with uncertainties and limitations clearly stated. The Protective Measures Coordinator or designee may provide technical input to the Dosimetry Specialist and suggest other report formats or evaluations to better serve the NRC in the incident response.

C. <u>Calibration Activities</u>

Following exposure of the NRC TLDs (from the Network, augmentation or personnel use) arrangements should be made by the Dosimetry Group with the National Institute of Standards and Technology (NIST) (formerly NBS) or agency with similar capability to expose a representative sample of the NRC TLDs to the spectra of nuclides and exposure configurations encountered during the accident. The TLDs in this way can be calibrated after the field exposures, to the same or approximately same energies and conditions. The field values can then be adjusted to better reflect the actual exposures and corresponding uncertainties measured during the accident. These activities serve to narrow the assumptions about the dosimetry response characteristics discussed in paragraph A.3 of this document. (It should be noted that representative samples of all types of TLDs and instrumentation used during the emergency should be similarly evaluated).

Other corrections should be made to the NRC TLD data, including the following. a) Normal ambient background exposure during the time the Direct Radiation Monitoring Network TLDs were in the field should be estimated (based on past TLD measurements at those locations) and subtracted. Note: For non-network locations, estimates of backgrounds can be based on licensee or State measurements, if the relative responses between the NRC and licensee/State TLDs have been determined; or based on comparisons with the EG&G AMS preoperational overflight data. b) In-transit exposure based on the Network control TLDs should be estimated and subtracted. c) In-transit exposure during the exchange following the accident should be estimated (based on appropriate controls) and subtracted. d) If contamination of the TLDs occurred in the field, estimates of the contribution of near-proximity contamination to that of the plume exposure should be made based on a measure of contamination levels, time of exposure and correlation of contamination levels to exposure rates to the TLD. e) TLDs placed in the field following a contaminating release may require correlation between TLD measured exposure levels and ground contamination levels. It should be noted that the energy changes in the gamma spectrum following deposition may require additional energy calibrations at NIST or the calibration facility. The Dosimetry Group should work closely with the FRMAC in these activities.

D. Technical Background

The NRC TLD Direct Radiation Monitoring Network is employed by the Nuclear Regulatory Commission to routinely measure the radiation doses at selected locations around all licensed NRC power reactor facilities and a number of other sites. The monitoring period over which the doses are accumulated is normally a calendar quarter, but this may be shortened if conditions warrant it. Such conditions include unexpected releases of radioactive materials from the reactor, such as during an accident. The program is conducted for the NRC by the Dosimetry Specialist and Laboratory Assistant in the Effluents Radiation Protection Section (ERPS), Facilities Radiological Safety and Safeguards Branch (FRSSB), Division of Radiation Safety and Safeguards (DRSS), Region I. NRC uses participating States and contractor personnel to physically exchange the dosimeters in the field.

The NRC TLD Direct Radiation Monitoring Network utilizes a Panasonic Model UD 710A automatic reader and Model UD 801 4-element thermoluminescent dosimeters (TLDs). The NRC also has a Panasonic Model UD 702E manual TLD reader which can be transported to remote locations and set up to anneal and read TLDs at locations near the site of interest. Both the automatic reader and manual reader can be connected to a micro-computer, to collect, analyze, and store the data and perform desired calculations. The automatic reader is routinely used to anneal, and read all of the field TLDs (3000 to 4000) on a quarterly basis. The manual reader is not routinely used in the NRC TLD Direct Radiation Monitoring Network, but is periodically checked to assure it will operate properly, if needed.

The UD 801 TLDs have two lithium borate ($Li_2B_4O_7$:Cu) and two calcium sulfate, ($CaSO_4$:Tm) elements. The lithium borate elements have a flat, near tissue equivalent photon energy response. The calcium sulfate, while much more sensitive than the lithium borate, is much more photon energy dependent at photon energies below 100 keV. To reduce this energy dependence, lead filters are used to cover the calcium sulfate elements. The total filtration over the calcium sulfate elements is 700 mg/cm². One lithium borate element has an "open" window (14 mg/cm²) and the second is covered by 150 mg/cm² plastic filters.

For routine environmental monitoring, all four elements of each dosimeter are read. Each element has its own calibration factor which is used in computing the dose measured by the dosimeter. Because the $CaSO_4$:Tm elements are much more sensitive to photon radiation than the lithium borate elements and show little residual readout after annealing, the $CaSO_4$:Tm elements are more precise for routine environmental measurements. Current NRC practice is to use only these two elements in computing the quarterly exposures for each field TLD location.

The lithium borate elements, while less sensitive and having a larger residual reading following annealing, are much less energy dependent. The NRC routinely reviews the level of the $CaSO_4$:Tm measured exposure and the ratio of the $CaSO_4$:Tm to $Li_2B_4O_7$:Cu measurements to ascertain any shifts in this ratio from that routinely seen in the environment. Any ratio shifts result in investigations of anomalous releases or unusual exposures of the TLD. (The ratio changes with photon energies below 100 keV and with direction of incidence of the radiation on the TLD. In addition, the compensation of the filters for perpendicularly incident photons does not result in a response curve which is completely independent for all photon energies.) If the approximate energies of the exposure can be determined, then the dosimeters can be calibrated for these energies after the readout and the measured exposures can be appropriately adjusted. (Ref. NUREG-0837; NUREG/CR-3775)

PAGE INTENTIONALLY LEFT BLANK

REGION I DOSIMETRY GROUP - USE OF NRC TLD DIRECT RADIATION MONITORING NETWORK DURING A RADIOLOGICAL EMERGENCY

01 PURPOSE

The purpose of this procedure is to describe the mechanism for activating the NRC TLD Direct Radiation Monitoring Network (TLD Program) during a radiological emergency, assign specific responsibilities for implementing and coordinating its use during an emergency, and provide considerations and actions for implementation for Region 1 personnel.

02 REFERENCES

PMT-413 Activation of NRC TLD Support

PMT-415 Site Team Health Physicist Support to Region 1 Dosimetry Group

Region I DRSS Instruction 0860.5 "NRC TLD Direct Radiation Monitoring Program Manual", Part VIII - Emergency Response

03 ACTIONS

03.1 Activation

In the event of a radiological emergency and if the emergency situation appears to warrant activation of the TLD program, the Protective Measures Manager/Coordinator (PMM/PMC) of the NRC Response Team should contact the Region I Dosimetry Specialist and his/her supervisor, Chief, Effluents Radiation Protection Section, (if available). The specific topics of discussion should include the status of the event; whether releases have occurred or will likely occur; the intended purpose of TLD augmentation (e.g., exchange of the Network TLDs, monitoring additional locations, posting additional TLDs at current Network locations, supplementing personnel monitoring devices, etc.); the estimated time of arrival, specific site destination, and contact point; the personnel and equipment responding; and support needs for the Dosimetry Group at the site. The Dosimetry Group consists of the Dosimetry Specialist and supervisor, and one or more dosimetry assistants (other individuals knowledgeable of the NRC TLD program). The Dosimetry Specialist would likely remain with the manual reader, if used, or in Region I if the manual reader is not taken to the site. The dosimetry assistant will assist in preparing to respond to the site and will respond with the equipment.

> Note: The PMM/PMC has responsibility for determining what TLD augmentation will occur and on what schedule. The Region I Dosimetry Group is responsible for the technical aspects of the implementation and will advise the PMM/PMC in technical areas and in data interpretation.

03.2 Preparation

After informing his/her management of the request for emergency dosimetry, the Dosimetry Specialist and dosimetry assistant should immediately prepare dosimetry and equipment for travel to the site vicinity. (See Region I DRSS Instruction 0860.5, Part VIII.)

Note: Dispatch to the site will not occur (other than at Region I) until officially requested by a member of the HQ Executive Team through Region I management.

The decision to take the manual TLD reader (Model UD 702E) to the site will be made by the Dosimetry Specialist and his/her management. This decision will be based, in part, on the ability to get freshly annealed TLDs to the site and return of exposed dosimeters to Region I on a timely basis. It is preferable to bring the exposed TLDs back to Region I and use the Model UD 710A automatic reader and associated computer equipment for all measurements. If, however, the situation is such that timely transport to and from the site are difficult, consideration should be given to moving the Model UD 702E manual reader to the site. In addition, if the TLDs are expected to be used in personnel monitoring, faster turn-around in terms of measurements may necessitate moving the manual reader to the site.

03.2.1 Materials and Equipment

The number of TLDs to be annealed and moved to the site will depend on their intended use as indicated by the Protective Measures Manager/Coordinator. The number ranges from a replacement set of Network TLDs for the site plus in-transit controls, to a full Network replacement set for the site plus a second set for supplemental measurements, approximately 50 TLDs for additional environmental locations, 50 TLDs for personnel use, adequate control TLDs and TLDs for standardizing the manual reader at the site. The Dosimetry Laboratory maintains 100 TLDs designated for emergency response, in addition to the second set of TLDs for each site (if the latter are not in transit to or from the site as part of the quarterly exchange process). After determining the intended use, the Dosimetry Group should anneal the appropriate number of TLDs and prepare them for shipment (with controls). (At this time it will not be necessary to bag and label each TLD, since the situation at the site may dictate other actions by the time the dosimetry reaches the site.)

The following additional equipment should be prepared as the situation dictates.

- the Manual Reader (Model UD 702E)/computer/associated software (if decision made to take reader to site)
- supply of bags for sealing field TLDs
- bag sealers
- tags for field TLDs
- site map showing current locations of Network TLDs, if available
- site TLD driving routes for exchanging TLDs
- NRC TLD measurement histories for site (NUREG-0837 data)

- AMS preoperational or operational overflight radiological reports/maps, if available
- uranium placque source for field calibrating manual reader at site
- tape (for sealing bags in the field)
- shields (2) for reducing in-transit exposures during TLD exchanges
- booties (disposable)
- latex gloves (box)
- TLD badge clips (if use is intended for personnel monitoring) and labels
- scissors
- smear papers/glassine envelopes
- stapler w/staples

The Dosimetry Laboratory has assembled an emergency kit containing the generic supplies and a listing of pertinent other equipment/supplies to be obtained at the time of an emergency.

03.2.2 Deployment to Site

Upon completion of the preparatory activities described above and a request for TLD support is received from the Executive Team, the Dosimetry Group (including a Region I health physicist) should travel to the site area in an expedited manner.

The mode of travel and ETA should be provided to the NRC Protective Measures Team. Directions for contacting NRC Site Team and location for a near site TLD operating base should be obtained from the NRC Protective Measures (typically through the responding Region's Base Team)

Note: If travel is by air, hand-carry the annealed TLDs. DO NOT HAVE X-RAYED AT AIRPORTS.

03.3 Arrival at Near Site Location

Upon arrival at the designated location, the Dosimetry Group should contact the PMC through the HOO and arrange for a conference to discuss the specifics of the TLD operations intended for the site. Requests for support for the Dosimetry Group (communications, health physics support if NOT from Region I, etc.) should be made to the PMC. Additionally, arrangements should be discussed relative to retrieving the control TLD maintained by the State or TLD exchange contractors for the batch of NRC Network TLDs currently in the field. With the PMC or designee the Dosimetry Group should determine where current Network TLDs should be augmented. A map with locations of the current NRC TLD Network, licensee, and State's dosimetry indicated would be very useful. When directed by PMC/designee, the dosimetry assistant begins the augmentation or exchange process with assigned health physicist.

Note: Exchanges should not be done if significant releases are in progress. Discussions with PMC/designee should include precautions to be taken if releases should occur while the dosimetry team is in the field.

If the manual reader was brought to the site, the Dosimetry Specialist should set up and test the equipment. The Dosimetry Specialist will usually remain with the manual reader (or in Region I if the automatic reader is to be used) and will not be available for the field exchanges or augmentation of the Network TLDs. Additionally, a series of calibration dosimeters should be exposed using the uranium placque source to ensure the reader is operating properly. The calibration range should extend as high or higher than the anticipated exposures from the field/personnel dosimeters. (Additional calibration dosimeters should be read 24 hours after exposure with the field/personnel exposed TLDs. If the emergency situation demands that the dosimeters be read earlier than 24 hours, they should be read and the fade times recorded. The fade correction can be estimated later.)

03.4.1 Addition of New Field Monitoring Locations

Prior to leaving the TLD operations base, the Dosimetry Team and the PMC/designee should have decided on the number and locations of each new monitoring location and marked these general locations on a map. The driving directions for optimum use of resources should be preplanned; dosimeters bagged with appropriate labels and placed in a shield in the vehicle; means for attaching new dosimeters (tape, stapler, extra bags, etc.) loaded; ladder/step stool loaded; and survey instruments (ionization chamber, pancake GM and microR meter) source checked and loaded. Just before departing to begin TLD exchange/augmentation, contact should be made with the PMC/designee to ascertain current plant conditions/likelihood of near-term releases. If radio communications are available with the Site Team, they should be tested.

At each general location sited by map, a specific location (telephone pole, small tree, etc.) must be used. The TLDs should be located 8-10 feet above ground and placed so they are least conspicuous to passers-by to make the TLDs less susceptible to vandalism. Prior to selection of the specific site, a survey of the ambient radiation levels at waist height should be done of the general area (10-20 ft radius) with the microR meter to ensure that the selected site does not have an inherent radiation anomaly. The identification of specific site needs to be recorded along with the driving directions and map indication, ambient radiation levels of and the time of posting the TLD at the site. This process is repeated for each additional location.

Note: If given a warning that a release has occurred while in the field and the release will affect the area of the Dosimetry Team, or if the health physicist measures significant plume activity, the team should leave the area immediately. (Exposure to the plume will affect the in-transit dose to dosimeters in the vehicle. Such exposure should be minimized.) Inform the PMC/designee of your actions and await his/her instructions after clearing the plume area.

03.4.2 Exchange of TLDs

If directed to exchange field TLDs, prepare dosimeters as in Section 03.4.1. Determine from the PMC/designee whether a contaminating release has occurred. If not, then the exchange process is similar to the installation of new TLD locations discussed above, except that the retrieved TLDs should be stored in a shielded container (preferably a different one than used for the annealed TLDs) in the vehicle. Again the measurement of the ambient radiation level at each location would be helpful and should be recorded. Proceed according to the existing driving route for exchange, unless directed to do otherwise.

If a contaminating release has occurred or was suspected, additional actions are required to prevent contamination of personnel, vehicle, and TLDs; to determine the extent of contamination of nearest surfaces to TLD; to remove TLDs from contaminated environment as soon as practical; and to install annealed dosimetry in uncontaminated areas or cages.

At each location the area should be surveyed for contamination after donning protective booties and gloves; the existing cage top should be surveyed to determine measurable contamination; and the cage then smear tested and gross counted for contamination with the GM. If contamination is found, the bag containing the TLD in the cage should be assumed to be contaminated. Upon removal from the cage the bag should be surveyed. The dosimetry bag should then be carefully cut and the dosimetry and tag put into a clean bag, taped shut and inserted into the shield for transit. The removed "contaminated" bag should be put in a larger bag and stored in the vehicle away from the TLDs. (One or more of the more highly contaminated bags should be stored separately for resurvey and definitive isotopic analysis upon return, such that an estimate of correlation between contamination level and exposure rate can be made. An annealed TLD inserted into this bag will also provide correlation information.) All survey and smear measurement results must be recorded with times associated with them.

If the TLD cages were contaminated, the annealed TLDs should <u>not</u> be put into the same cage, unless it has been decontaminated. New field support devices or means should be used to post the annealed TLDs. The above process is repeated at each TLD location.

Plant situation changes received by radio, if available, or as detected by the health physicist may require recontact with the PMC/designee to ascertain any changes in instructions during the TLD exchange.

03.5 <u>Return to TLD Operations Base</u>

Upon completion of the TLD exchange/augmentation route (or if directed by the PMC/designee during the route) return to the TLD operations base. The contaminated TLD bags and smear samples selected for analysis should be separated for this purpose. The remaining TLD bags, used gloves, booties, and other contaminated materials should be gathered together and held for disposal. The PMC/designee should be contacted by the health physicist to arrange for analysis of the smears and designated bags and for appropriate disposal of contaminated wastes. The assigned health physicist should assist in segregation and transport of these materials. The Dosimetry Team should again survey the retrieved TLDs for possible contamination.

The dosimeters should then be stored in a shield with controls for reading after 24 hours (unless directed otherwise by the PMC because of perceived urgency of data). If the manual reader was not brought to the site, hand carrying of the TLDs to Region I should be arranged and coordinated with the Site Team.

The TLDs are then read at Region I (or near site with the manual reader) after assuring that the reader is properly standardized.

03.6 Personnel Monitoring

If the PMC determines that the NRC TLDs are to be used to supplement personnel monitoring dosimetry, the Dosimetry Team will anneal the requested numbers of TLDs, place them in badge clips for personnel monitoring and affix appropriate identification labels on each badge. The badges will then be turned over to the PMC/designee for Site Team use. The PMC/designee is responsible for dosimetry issuance and personnel dosimetry records completion. Any unusual exposure configurations, radiation fields, energies, etc., should be reported to the Dosimetry Team in order to better interpret the measurements. Some factors may be estimated through post-exposure calibrations.

To the extent possible, the TLDs should be read after 24 hours following exposure.

03.7 Calibration of TLDs to Unusual Exposure Conditions

Releases of radioactive materials from nuclear plants as a result of an accident usually have photon and beta energy spectra which deviate from that normally measured in the environment. Consequently, monitoring devices, such as TLDs, survey instruments, etc., may respond differently than during routine calibration.

For TLDs used in the NRC Network, the Dosimetry Specialist will arrange for a calibration of a sample of the UD 801 TLDs by the National Institute of Science and Technology (NIST), Radiological and Environmental Sciences Laboratory (RESL) at Idaho Falls, or other qualified entity, under exposure conditions simulating those experienced during the accident. The Headquarters Executive Team should be informed through the Protective Measures Team and Director of Site Operations of these calibration needs such that funding can be appropriately arranged. The requests could also be made through the Federal Response Center (FRC) if the Federal Radiological Emergency Response Plan (FRERP) has been implemented.

The Dosimetry Team with assistance from the Protective Measures Team needs to document the exposure conditions to the extent possible and to define the scope of the calibration effort desired from the calibration facility. The information available through the Protective Measures Team and from field surveys and smears which should be helpful include:

- approximate nuclide mix of release to which TLDs were exposed;
- exposure configuration (overhead plume/immersion plume, nearby contamination on cages/TLD bags, etc.);
- nuclide mix of contamination;
- effect of ground contamination on TLD exposure; and
- beta/gamma vs gamma exposure to TLDs.

Dose estimates using the calibration factors should be incorporated into the assessments made by the Federal Radiological Monitoring and Assessment Center (FRMAC) as part of the federal (FRERP) response. The Dosimetry Team should work closely with the FRMAC in the data evaluation effort.

04 REPORTS

The Dosimetry Specialist should issue preliminary reports of the TLD measurements as soon as the TLD data become available. The reports should indicate the uncertainties and precautions to be used when using the report results (energy response changes, exposure configurations, etc.) and should indicate the status of efforts to get more definitive assessments of the results. These reports will be provided to the PMC/designee and to the FRMAC.

Final TLD assessment information should come from the FRMAC along with all other assessment information.

PAGE INTENTIONALLY LEFT BLANK

SITE TEAM HEALTH PHYSICISTS SUPPORT TO REGION 1 DOSIMETRY GROUP

01 PURPOSE

The purpose of this procedure is to describe the responsibilities and management of the health physicist assigned to the Dosimetry Team for the exchange of or field augmentation of the NRC TLD Direct Radiation Monitoring Network.

02 REFERENCES

PMT-413 Activation of NRC TLD Support

PMT-414 "Region I Dosimetry Group, Use of the NRC Direct Radiation Monitoring Network During a Radiological Emergency"

03 EQUIPMENT NEEDED

- Ionization Chamber (Eberline R0 2 or equivalent)
- Pancake type GM detector (shielded)
- MicroR meter
- Personnel dosimetry for Dosimetry Team

04 ACTIONS

04.1 <u>Management Chain</u>

The health physicist is generally a Region I Radiation Specialist assigned to respond with the Dosimetry Group by Region I management. The health physicist (1) provides personnel radiological protection during the NRC TLD field augmentation or exchange process; (2) supports the Region I dosimetry assistant during the TLD exchange process by providing radiological survey and contamination control support; and (3) provides general assistance for the exchange process. For purpose of personnel radiological protection, the health physicist reports to the PMC/designee. The dosimetry assistant directs the health physicist in aspects of obtaining technically valid radiological measurements necessary to relate TLD measurements to exposure conditions.

Note: (1) Radiological safety concerns always take precedence over technical support priorities.
(2) Contamination control is the responsibility of all team members.

04.2 <u>Field Augmentation or Exchange</u>

During the field exchange or augmentation of NRC TLDs, the health physicist should assist the Region I dosimetry assistant in following the driving route or map. At each <u>new</u> location, the health physicist should perform a survey of the general area (about 20 ft. radius of the proposed specific location) at waist height with a microR meter to document the ambient radiation level and ensure no local radiological anomaly exists. If a contaminating release had occurred, then a surface level survey (with HP-210 or equivalent detector) should be performed to document relative ground deposition levels. Additionally, measurements of existing TLD cage contamination levels and TLD bag contamination levels should be made with the survey instrument and with a smear paper.

Note: Remember that proper contamination control measures need be used for personnel (gloves, booties, etc.), as well as to prevent contamination spread/cross-contamination of the dosimetry! Additionally, the detected presence of a radiological plume should warrant immediate contact with the PMC/designee and leaving the area unless specifically directed otherwise by the PMC/designee!

Measurements should be clearly documented as to times, locations, and activity levels. Smears should be individually bagged after field measurement for later isotopic analyses, if warranted. An ionization chamber (RO 2, or equivalent) should be used to measure dose rates, if significant.

If dosimetry cages are found to be contaminated, the TLDs should be removed from the bags and carefully inserted individually with their location tags into clean bags and taped shut. They should then be put into a shield for transport back to the TLD operations base and stored until read or sent to NRC Region I. Contaminated bags and materials should be stored separately and as far from the TLDs as possible in the vehicle. A number of the contaminated bags should be segregated for later gamma spectral analyses for additional data which may assist in data interpretation. Contaminated cages may have to be removed from the current sites before installing freshly annealed TLDs (in some cases, other suitable TLD locations can be found in the same area).

04.3 Return to the Dosimetry Operations Area

Upon completion of the assignment and return to the dosimetry operations base, the health physicist should assist in unloading the exchanged TLDs and carefully surveying them for any residual contamination. Contaminated materials, not needed for analysis, should be disposed in accordance with the instructions of the PMC/designee. The health physicist may then be released (following any necessary decontamination of the vehicle and equipment) for further assignments by the PMC/designee.

Medical Assistance

Section O

O Medical Assistance

MEDICAL ASSISTANCE

Objective

Getting medical guidance for someone at the scene of the emergency or at the hospital.

Guidance

CAUTION:

Do not give medical advice yourself. You will be personally liable for giving medical advice.

Step

NOTE:

Reactor sites have identified medical facilities capable of responding to radiation emergencies.

If medical advice is requested, direct the caller to call the Radiation Emergency Assistance Center/Training Site (REAC/TS).

Day	(615)576-3131; FTS 626-3131
24 hour Hospital Disaster Network	(615)481-1000

NOTE:

Radiation Emergency Assistance Center/Training Site (REAC/TS): REAC/TS is a DOE response asset that maintains a radiological emergency response team consisting of physicians, nurses, health physicists, coordinators, and necessary support personnel. It is on 24-hour call to provide first-line responders with consultative or direct medical and radiological assistance at the REAC/TS facility or at the accident site. They have expertise in, and is equipped to conduct; (1) medical and radiological triage; (2) decontamination procedures and therapies for external contamination and internally deposited radionuclides, including DTPA chelation therapy; (3) diagnostic and prognostic assessments of radiation induced injuries; and (4) radiation dose estimates by methods that include cytogenetic analysis, bioassay, and in vivo counting.

DOE Radiological Field Monitoring Assistance

Section P

P DOE Radiological Field Monitoring Assistance

DOE RADIOLOGICAL FIELD MONITORING ASSISTANCE

Objective

Get DOE radiological monitoring help.

Guidance

CAUTION:

> NRC is not qualified to conduct extensive radiological monitoring.

NOTE:

> - If the State has not requested DOE assistance, the NRC should request DOE monitoring if required to assure public safety.
> - It may take hours to get DOE assistance, so request DOE help early. It is better to call off a request than not to have help if needed.

Step 1

Determine the assistance required (A - RAP, B - FRMAC or C - AMS):

NOTE:

> See page P-3 for a further description of each capability.

A. **Radiological Assistance Program (RAP)**

- Prompt and limited monitoring with limited assessment of results

B. **Federal Radiological Monitoring Assessment Center (FRMAC)**

- Limited or extensive monitoring and assessment of results

C. **Aerial Measuring System (AMS)**

- Prompt airborne monitoring and FRMAC followup

Step 2

Complete the "Emergency Information for DOE" form (page P-5).

Step 3

Request help by:

RAP Call DOE Regional Coordination Office - shown in chart on page P-4. After calling DOE, call the NRC Operations Center (301-951-0550) and inform a Government Liaison or Response Coordination Team (RCT) member of your request.

FRMAC or AMS Call the NRC Operations Center (301-951-0550) and provide your request and information gathered in Step 2 to a Government Liaison or Response Coordination Team (Incident Response Branch) member. They should call the DOE Emergency Operations Center (202-586-8100) and coordinate your request. Provide a call back number because the FRMAC Director (DOE Las Vegas) will call the NRC Operations Center for further information within 1 hour of being notified.

DESCRIPTION OF DOE RADIOLOGICAL MONITORING CAPABILITIES

Radiological Assistance Program (RAP).

Eight DOE Regional Coordinating Offices for Radiological Assistance provide radiological assistance on request in any region of the country. The radiological assistance teams are the front lines of Federal assistance under the Federal Radiological Emergency Response Plan (FRERP). RAP teams will generally be requested quite early in an event by the State and/or the NRC. RAP teams typically are composed of 4 to 6 people. They are capable of conducting gross gamma, alpha, and beta monitoring. They will require only 1/2 to 2 hours to prepare to respond; response time will depend on travel time to the location of accident. The RAP team will also act as an advance team for coordination of further DOE assistance (FRMAC or AMS).

Page P-4 shows the DOE Regional Coordinating Offices for Radiological Assistance and geographical areas of responsibility. RAP teams can be requested through these offices directly.

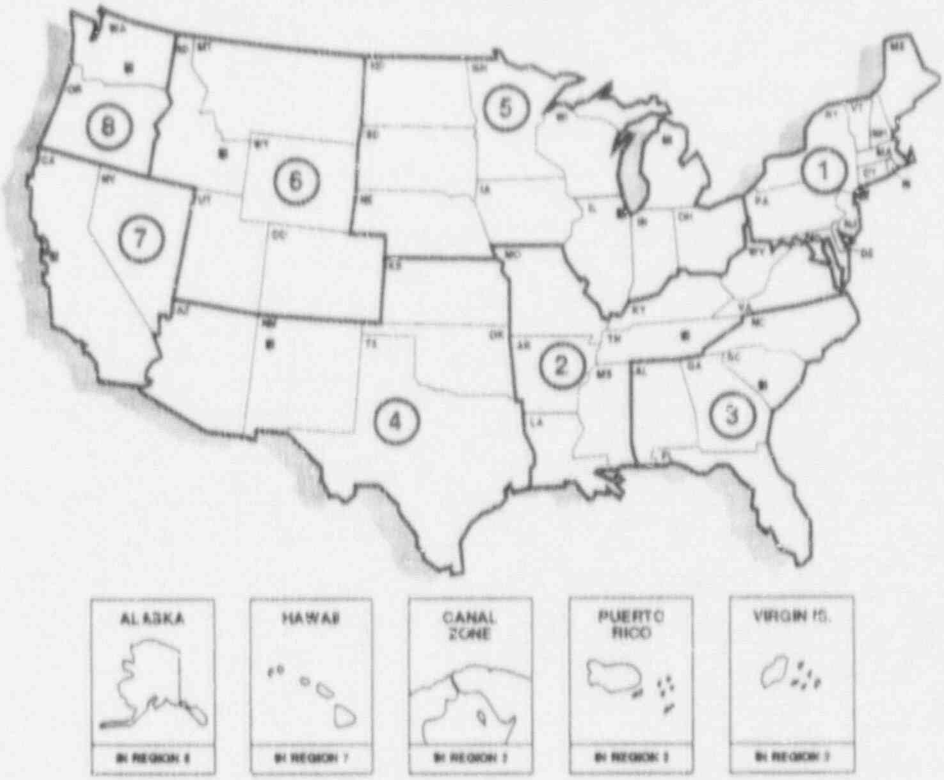
Federal Radiological Monitoring and Assessment Center (FRMAC).

A FRMAC is a center from which DOE coordinates the monitoring and assessment efforts of Federal agencies. The FRMAC is usually established near the site of the accident. The size and complexity of the FRMAC will depend on the incident. The FRMAC will coordinate the monitoring conducted by all Federal agencies. The FRMAC can provide the most extensive monitoring and assessment capabilities available in the U.S. It will require about 24 hours for a fully operational FRMAC to be established. DOE has delegated the responsibility for establishing the FRMAC to the DOE Nevada Operations Office. EG&G, a DOE contractor, provides support for the FRMAC.

Aerial Measuring System (AMS).

The AMS is an airborne radiological detection system that can respond to radiological incidents and perform plume tracking, radiation surveys, and radiation mapping over large areas around the site of an incident. EG&G operates AMS for DOE. Most of the aerial monitoring assets are located in Las Vegas, Nevada, but there are also some capabilities based at Andrews Air Force Base, near Washington, D.C. The AMS will be integrated into the FRMAC once it is established.

GEOGRAPHICAL AREAS OF RESPONSIBILITY



Region	Regional Coordinating Office	Post Office Address	Telephone for Assistance
1	Brookhaven Area Office	S&EP Division, Brookhaven National Lab, Upton, Long Island, NY 11973	(516)282-2200 FTS 666-2200
2	Oak Ridge Operations Office	U.S. DOE, P.O. Box 2001, Oak Ridge, TN 37831	(615)576-1005 FTS 626-1005 or (615)525-7885
3	Savannah River Operations Office	U.S. DOE, P.O. Box A, Aiken, SC 29801	(803)725-3333 FTS 239-3333
4	Albuquerque Operations Office	RAP Program Manager U.S. DOE, P.O. Box 5400, Albuquerque, NM 87115	(505)845-4667 FTS 845-4667
5	Chicago Operations Office	E&SH Division, U.S. DOE 9800 S. Cass Avenue Argonne, Illinois 60439	Duty Hours: (708)972-4800 FTS 972-4800 Off Hours: (708)972-5731 FTS 972-5731
6	Idaho Operations Office	U.S. DOE, 785 DOE Place, Idaho Falls, Idaho 83402	(208)526-1515 FTS 583-1515
7	San Francisco Operations Office	U.S. DOE, 1333 Broadway, Oakland, California 94612	(415)273-4237 FTS 536-4237
8	Richland Operations Office	U.S. DOE, P.O. Box 550, Richland, WA 99352	(509)373-3800 FTS 440-3800

Time of Call__________

EMERGENCY INFORMATION FOR DOE

1. Federal notifications made by NRC

	Time	Radiological Contact	Phone Number
EPA ______	______	______	______
USDA ______	______	______	______
HHS ______	______	______	______
FEMA ______	______	______	______
Other______	______	______	______

2. State notifications made by NRC

State	Agency or Dept.	Time	Contact	Phone Number
______	______	______	______	______
______	______	______	______	______
______	______	______	______	______
______	______	______	______	______

3. Potential airport(s) for FRMAC

Name __________ Location __________
Contact __________
Phone Number __________

4. Weather at site

General __________
Precipitation __________ Wind Speed __________
Temperature __________ Wind Direction __________

5. Phone status

Emergency Class __________ Time __________
Release Characteristics or Potential __________

6. EOF

Location __________

Licensee Contact __________
Phone Number __________

7. NRC Site Team

Contact (Protective Measures Coordinator) __________
Phone Number __________

8. Resources Requested

_____a. RAP
_____b. Small FRMAC
_____c. Full FRMAC
_____d. AMS (includes limited FRMAC)
_____e. ARAC

9. Deployment Schedule - Arrival Time

NRC Site Team to EOF ____________________
NRC FRMAC Team to FRMAC - Staff __________
TLD __________
Van __________

State - Staff ____________________
Monitoring Teams ____________________

DOE - RAP ____________________
AMS ____________________
Advance Party ____________________
ARAC ____________________
FRMAC ____________________

10. NRC Caller's Name ____________________
NRC Caller's Phone Number ____________________

SI Units and Conversions

Section Q

Q

SI Units and Conversions

SI UNITS AND CONVERSIONS

Refer to pages Q-2 and Q-3

SI UNITS AND CONVERSIONS

To Convert	Into	Multiply By
acres	sq. meters	4,047.00
acres	sq. miles	1.562×10^{-3}
centimeters	inches	0.39
cubic centimeters	ml	1.00
cubic feet	gallons (U.S. liq.)	7.48
cubic feet	quarts (U.S. liq.)	29.92
days	seconds	86,400.00
feet	centimeters	30.48
feet	kilometers	3.05×10^{-4}
gallons	cu feet	0.13
gallons	liters	3.78
gallons/min	cu ft/hr	8.02
inches	centimeters	2.54
kilograms	pounds	2.20
kilowatts	Btu/min	56.92
kilowatt-hrs	Btu	3,413.00
liters	cu cm	1,000.00
liters	cu feet	0.035
liters	gallons (U.S. liq.)	0.26
meters	centimeters	100.00
meters	feet	3.28
meters	miles (stat.)	6.21×10^{-4}
meters/sec	miles/hr	2.24
pounds of water	cu feet	0.016
pounds of water	cu inches	27.68
pounds of water	gallons	0.12
quarts (liq.)	gallons	0.25
square centimeters	sq inches	0.15
square feet	sq cms	929.00
square inches	sq cms	6.45
square kilometers	sq cms	10^{10}
square kilometers	sq miles	0.39
tons (metric)	pounds	2,205.00

Water at Atmospheric Conditions
14.7 psig & 80°F

7.48 gallons = 1 cu ft

62.4 pounds = 1 cu ft

8.33 pounds = 1 gallon

SI UNITS AND CONVERSIONS

TEMPERATURE

Celsius	Fahrenheit
3000°C	5432°F
2500°C	4532°F
2000°C	3632°F
1500°C	2732°F
1000°C	1832°F
800°C	1472°F
600°C	1112°F
400°C	752°F
200°C	392°F
100°C	212°F
50°C	122°F
0°C	32°F
-17.8°C	0°F

PRESSURE (Pascal)

1 Pa = 1.45×10^{-4} psi

7000 Pa ≅ 1 psi

1M Pa = 145 psi

SPEED

1 m/s ≅ 2 mph

VOLUME

1 m^3 = 10^3 ℓ

1 cc (cm^3) = 1 mℓ

1 cc ≅ 1 gram water

3785 cc/gal 7.48 gal/ft^3

AREA

1 km^2 = 10^6 m^2

1 m^2 ≅ 11 ft^2

RADIATION

DOSE

rem	sievert
0.1 mrem	1 μSv
1 mrem	10 μSv
10 mrem	100 μSv (0.1 mSv)
100 mrem	1 mSv
500 mrem	5 mSv
1 rem	10 mSv
5 rem	50 mSv
10 rem	100 mSv
25 rem	250 mSv
50 rem	500 mSv
100 rem	1 Sv

ABSORBED ENERGY

100 rad = 1 Gy (gray)

AMOUNT

curie	becquerel
1 pCi	37 mBq
27 pCi	1 Bq
1 nCi	37 Bq
27 nCi	1 kBq
1 μCi	37 kBq
27 μCi	1 MBq
1 mCi	37 MBq
27 mCi	1 GBq
1 Ci	37 GBq
27 Ci	1 TBq
1 kCi	37 TBq
27 kCi	1 PBq
1 MCi	37 PBq

SI UNITS PREFIXES:

E	exa	10^{18}	M	mega	10^6	μ	micro	10^{-6}
P	peta	10^{15}	k	kilo	10^3	n	nano	10^{-9}
T	tera	10^{12}	c	centi	10^{-2}	p	pico	10^{-12}
G	giga	10^9	m	milli	10^{-3}			

Putting Radiation in Perspective for the Public

Section R

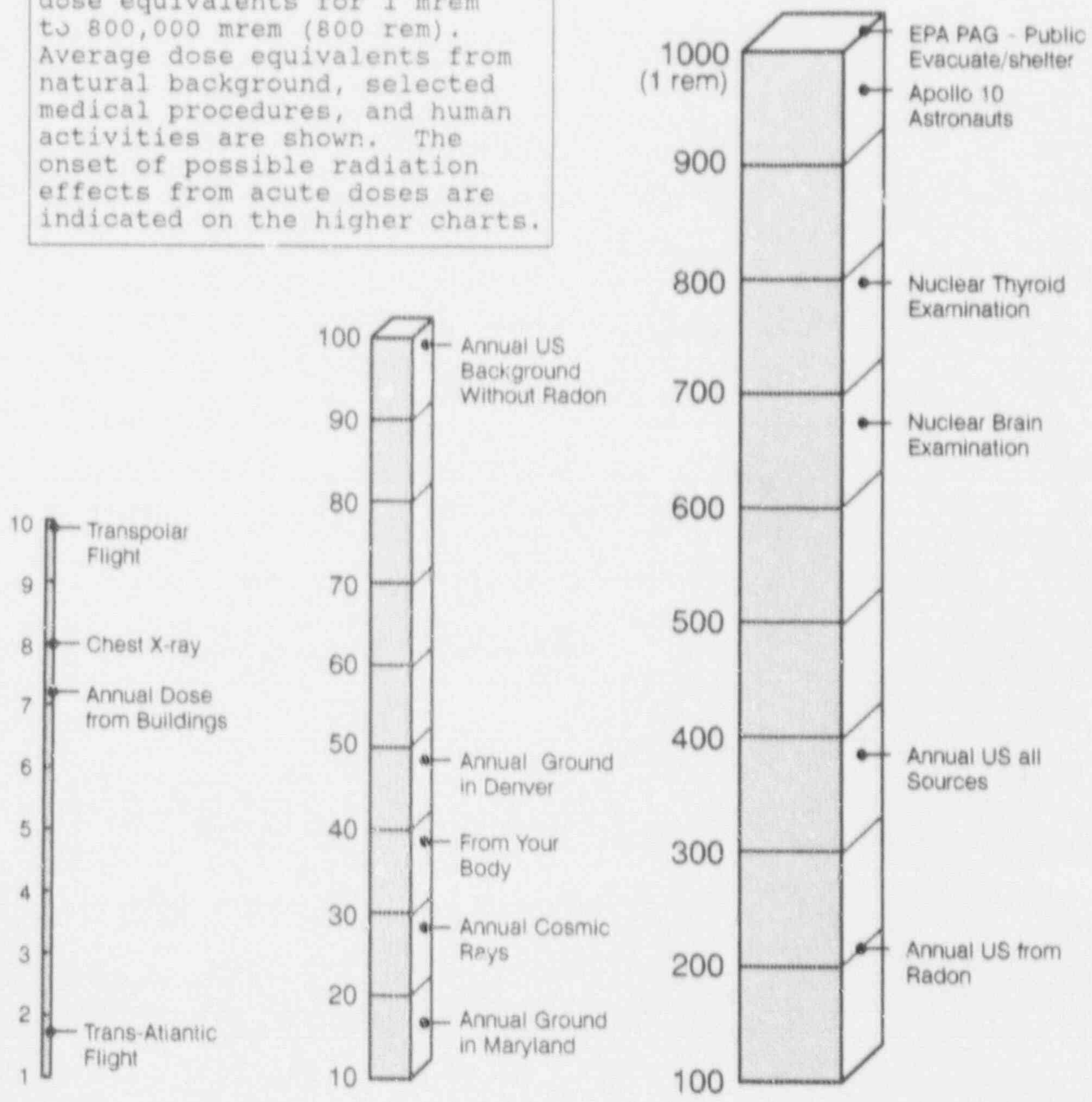
R Putting Radiation in Perspective for the Public

PUTTING RADIATION IN PERSPECTIVE FOR THE PUBLIC
(mrem)

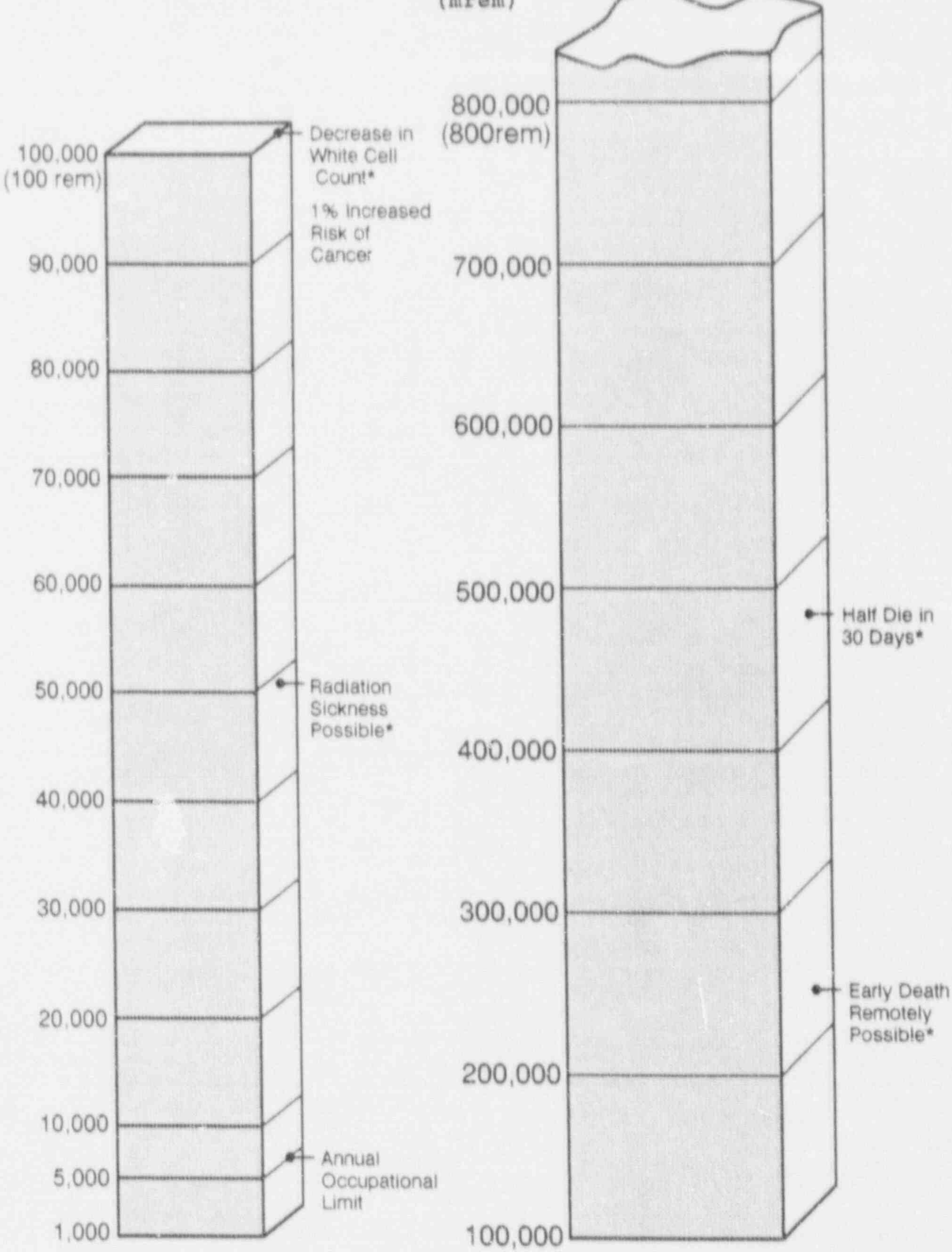
Refer to pages R-2 and R-3.

PUTTING RADIATION IN PERSPECTIVE FOR THE PUBLIC
(mrem)

This chart displays effective dose equivalents for 1 mrem to 800,000 mrem (800 rem). Average dose equivalents from natural background, selected medical procedures, and human activities are shown. The onset of possible radiation effects from acute doses are indicated on the higher charts.



PUTTING RADIATION IN PERSPECTIVE FOR THE PUBLIC
(mrem)



* Onset of possible radiation effects due to doses received over a short time period at high dose rates (acute doses).

Half-Life Table

Section 8

S Half-Life Table

HALF-LIFE TABLE

Element Names, Atomic Numbers, and Half-Lives[a]
for Selected Radionuclides

Radio-nuclide	Element Name	Atomic Number	Half-Life	Radio-nuclide	Element Name	Atomic Number	Half-Life
H-3	hydrogen	1	12.28 Y	Rb-86	rubidium	37	18.66 D
C-14	carbon	6	5.73E3 Y	Rb-88	rubidium	37	17.8 M
Na-22	sodium	11	2.602 Y	Sr-89	strontium	38	50.55 D
Na-24	sodium	11	15.00 H	Sr-90	strontium	38	28.6 Y
P-32	phosphorus	15	14.29 D	Sr-91	strontium	38	9.5 H
P-33	phosphorus	15	25.4 D	Y-91	yttrium	39	58.51 D
S-35	sulfur	16	87.44 D	Y-91m	yttrium	39	49.71 M
Cl-36	chlorine	17	3.01E5 Y	Zr-93	zirconium	40	1.53E6 Y
K-40	potassium	19	1.277E9 Y	Zr-95	zirconium	40	64.02 D
K-42	potassium	19	12.36 H	Nb-94	niobium	41	2.03E4 Y
Ca-45	calcium	20	162.7 D	Nb-95	niobium	41	35.06 D
Sc-46	scandium	21	83.80 D	Mo-99	molybdenum	42	66.02 H
Ti-44	titanium	22	47.3 Y	Tc-99	technetium	43	2.13E5 Y
V-48	vanadium	23	15.971 D	Tc-99m	technetium	43	6.02 H
Cr-51	chromium	24	27.704 D	Ru-103	ruthenium	44	39.35 D
Mn-54	manganese	25	312.7 D	Ru-105	ruthenium	44	4.44 H
Mn-56	manganese	25	2.5785 H	Ru-106	ruthenium	44	368.2 D
Fe-55	iron	26	2.7 Y	Rh-103m	rhodium	45	56.119 M
Fe-59	iron	26	44.63 D	Rh-105m	rhodium	45	45 S
Co-60	cobalt	27	5.271 Y	Rh-106	rhodium	45	29.92 S
Ni-63	nickel	28	100.1 Y	Ag-109m	silver	47	39.6 S
Cu-64	copper	29	12.701 H	Ag-110	silver	47	24.57 S
Zn-65	zinc	30	244.4 D	Ag-110m	silver	47	249.85 D
Ga-68	gallium	31	68.0 M	Cd-109	cadmium	48	464 D
Ge-68	germanium	32	288 D	Cd-113m	cadmium	48	13.7 Y
Se-75	selenium	34	119.78 D	In-114	indium	49	71.9 S
Kr-85	krypton	36	10.72 Y	In-114m	indium	49	49.51 D
Kr-85m	krypton	36	4.48 H	Sn-113	tin	50	115.1 D
Kr-87	krypton	36	76.3 M	Sn-123	tin	50	129.2 D
Kr-88	krypton	36	2.84 H	Sn-126	tin	50	1.0E5 Y

[a]S = second   5.73E3 = 5.73×10^3
M = minute
D = day
H = hour
Y = year

HALF-LIFE TABLE - Continued

Radio-nuclide	Element Name	Atomic Number	Half-Life	Radio-nuclide	Element Name	Atomic Number	Half-Life
Sb-124	antimony	51	60.20 D	Sm-151	samarium	62	90 Y
Sb-126	antimony	51	12.4 D	Eu-152	europium	63	13.6 Y
Sb-126m	antimony	51	19.0 M	Eu-154	europium	63	8.8 Y
Sb-127	antimony	51	3.85 D	Eu-155	europium	63	4.96 Y
Sb-129	antimony	51	4.40 H	Gd-153	gadolinium	64	241.6 D
Te-127m	tellurium	52	109 D	Tb-160	terbium	65	72.3 D
Te-129	tellurium	52	6[illegible] M	Ho-166m	holmium	67	1.20E3 Y
Te-129m	tellurium	52	33.6 D	Tm-170	thulium	69	128.6 D
Te-131m	tellurium	52	30 H	Yb-169	ytterbium	70	31.97 D
Te-132	tellurium	52	78.2 H	Hf-181	hafnium	72	42.39 D
I-125	iodine	53	60.14 D	Ta-182	tantalum	73	114.74 D
I-125	iodine	53	1.57E7 Y	W-187	tungsten	74	23.83 H
I-131	iodine	53	8.040 D	Ir-192	iridium	77	74.02 D
I-132	iodine	53	2.30 H	Au-198	gold	79	2.696 D
I-133	iodine	53	20.8 H	Hg-203	mercury	80	46.60 D
I-134	iodine	53	52.6 M	Tl-204	thallium	81	3.779 Y
I-135	iodine	53	6.61 H	Pb-210	lead	82	22.26 Y
Xe-131m	xenon	54	11.84 D	Bi-207	bismuth	83	33.4 Y
Xe-133	xenon	54	5.245 D	Bi-210	bismuth	83	5.013 D
Xe-133m	xenon	54	2.19 D	Po-210	polonium	84	138.378 D
Xe-135	xenon	54	9.11 H	Fr-223	francium	87	21.8 M
Xe-135m	xenon	54	15.36 M	Ra-226	radium	88	1600 Y
Xe-138	xenon	54	14.13 M	Ac-227	actinium	89	21.773 Y
Cs-134	cesium	55	2.062 Y	Ac-228	actinium	89	6.13 H
Cs-136	cesium	55	13.16 D	Th-227	thorium	90	18.718 D
Cs-137	cesium	55	30.17 Y	Th-228	thorium	90	1.9132 Y
Cs-138	cesium	55	32.2 M	Th-230	thorium	90	7.7E4 Y
Ba-133	barium	56	10.5 Y	Th-232	thorium	90	1.405E10 Y
Ba-137m	barium	56	2.552 M	Pa-231	protactinium	91	3.276E4 Y
Ba-140	barium	56	12.789 D	U-232	uranium	92	72 Y
Ce-141	cerium	58	32.50 D	U-233	uranium	92	1.592E5 Y
Ce-144	cerium	58	284.3 D	U-234	uranium	92	2.445E5 Y
Pr-144	praseodymium	59	17.28 M	U-235	uranium	92	7.038E8 Y
Pm-145	promethium	61	17.7 Y	U-238	uranium	92	4.468E9 Y
Pm-147	promethium	61	2.6234 Y	Np-237	neptunium	93	2.14E6 Y

HALF-LIVE TABLE - Continued

Radio-nuclide	Element Name	Atomic Number	Half-Life	Radio-nuclide	Element Name	Atomic Number	Half-Life
Np-239	neptunium	93	2.355 D	Am-242m	americium	95	152 Y
Pu-236	plutonium	94	2.851 Y	Am-243	americium	95	7.38E3 Y
Pu-238	plutonium	94	87.75 Y	Cm-242	curium	96	163.2 D
Pu-239	plutonium	94	24131 Y	Cm-243	curium	96	28.5 Y
Pu-240	plutonium	94	6537 Y	Cm-244	curium	96	18.11 Y
Pu-241	plutonium	94	14.4 Y	Cm-245	curium	96	8.5E3 Y
Pu-242	plutonium	94	3.758E5 Y	Cf-252	californium	98	2.639 Y
Am-241	americium	95	432.2 Y				

Acronyms/Abbreviations List

Section T

T

Acronyms/Abbreviations List

SELECTED ACRONYMS/ABBREVIATIONS LIST

AC	alternating current
ADS	Automatic Depressurization System (BWR)
AGL	above ground level
AMS	Aerial Measuring System (DOE)
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARAC	Atmospheric Release Advisory Capability (DOE)
ASME	American Society of Mechanical Engineers
Bq	becquerel (unity of radioactivity)
BWR	boiling water reactor
C	Celsius (unit of temperature)
cc	cubic centimeter, cm^3
CEDC	committed effective dose equivalent
CET	core exit thermocouples
CFR	Code of Federal Regulations
Ci	curie (unity of radioactivity)
CRF	Core Release Fraction
DC	direct current
DCF	dose conversion factor
DPTA	Diethylenetriaminepentaacetate
DOE	Department of Energy
DRL	derived response level
DSO	Director of Site Operations (NRC)
ECCS	emergency core cooling system
EF	escape fraction
EG&G	EG&G, Inc. (DOE support contractor)
ENS	Emergency Notification System
EOF	Emergency Operations Facility
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
ET	Executive Team (NRC)
ETA	estimated time of arrival
F	Fahrenheit (temperature scale)
FDA	Food and Drug Administration (HHS)
FPI	core fission product inventory
FRMAC	Federal Radiological Monitoring and Assessment Center
FSAR	Final Safety Analysis Report
ft	foot
gal	gallon
GL	Government Liaison Team (NRC)
GLO	Government Liaison Officer (NRC)
GMT	Greenwich mean time
gpm	gallons per minute
H2O	water
HHS	Department of Health and Human Services
HOO	Headquarters Operations Officer (NRC)
HP	health physics
HPN	Health Physics Network
HQ	headquarters
hr	hour

ICRP	International Commission on Radiological Protection
IRC	Incident Response Center
K	Kelvin (temperature scale)
l	liter
lbs	pounds
LD_{50}	Median Lethal Dose
LOCA	loss of cooling accident
m	meter
MARK	Model of BWR reactor
min	minute
mr	millirad
MSIV	main steam isolation valve
MW(e)	megawatt (electric)
MW(t)	megawatt (thermal) (MW(t)=3x(MW(e))
NRC	National Response Center (USCG)
NRC	Nuclear Regulatory Commission
NV	Nevada Operations Office (DOE)
OBE	operating basis earthquake
ORG	organic
P	pressure
Pa	Pascal (unit of pressure)
PAG	protective action guide/guideline(s)
PAR	protective action recommendation(s)
PMC	Protective Measures Coordinator (NRC)
PMT	Protective Measures Team (NRC)
PORV	power-operated relief valve
PRZ	pressurizer
psi	pounds per square inch
psia	pounds per square inch absolute (psia psig + 14.7 lbs)
psig	pounds per square inch gauge
PWR	pressurized water reactor
R	roentgen (unit of radiation exposure)
RAM	Radiological Assessment Manager (NRC)
RAP	Radiological Assistance Program (DOE)
RASCAL	computer code (NRC) for predicting accident consequences
RCS	reactor coolant system
RDF	reduction factor
REAC/TS	Radiation Emergency Assistance Center/Training Site (DOE)
RHRS	residual heat removal system
Rx	reactor
s	second
SBGTS	standby gas treatment system
sec	second
SFO	Senior FEMA Official
SG	steam generator
SGTR	steam generator tube rupture
SRV	safety relief valve
SSE	safe shutdown earthquake
Sv	Sievert (unit of dose)
TLD	thermoluminescent dosimeter
TMI	Three Mile Island Nuclear Power Station
TSC	Technical Support Center

UCT	universal coordinated time
USDA	U.S. Department of Agriculture
UTM	universal transverse mercator
WB, W.B.	whole body (radiation exposure or dose)

Index

Index

INDEX