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325 - 325 - SYSTEMS LEAD ENGINEER: EMERGENCY PLAN-POSITION SPECIFIC PROCEDURE

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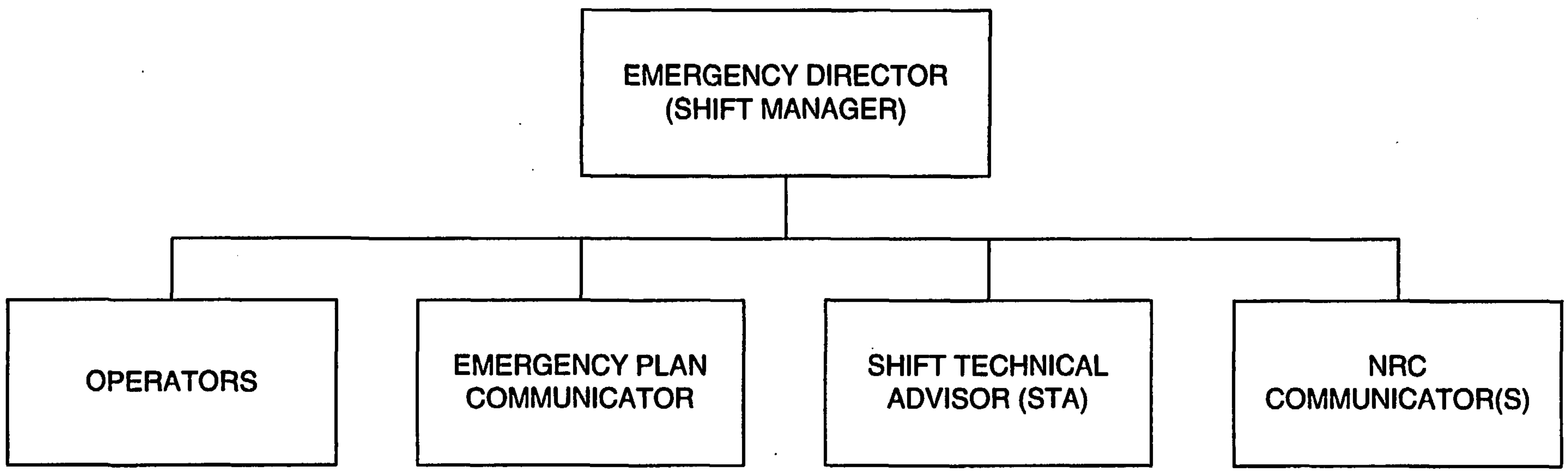
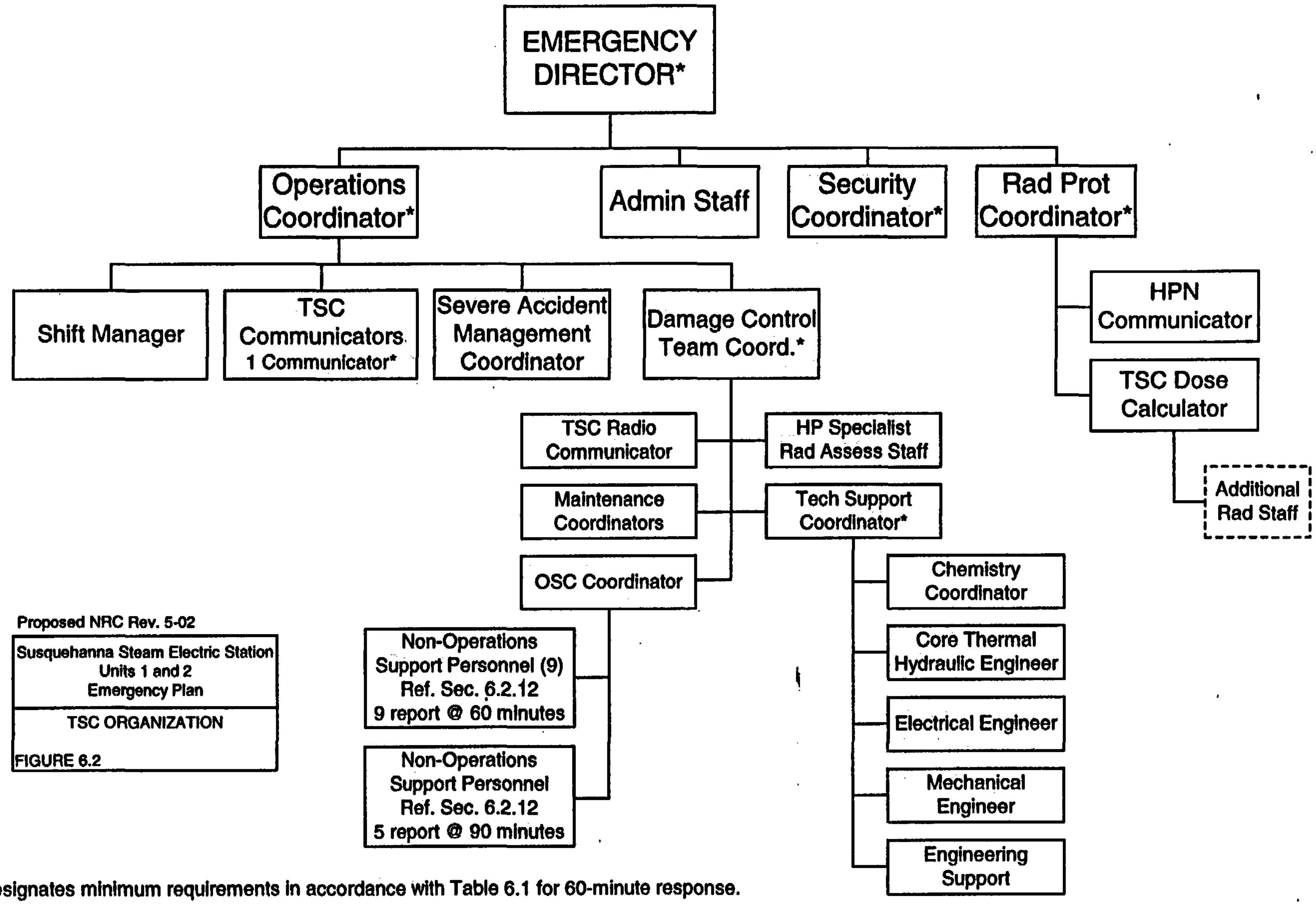
A045

EMERGENCY ORGANIZATION
CONTROL ROOM



TSC ORGANIZATION



Proposed NRC Rev. 5-02

Susquehanna Steam Electric Station
Units 1 and 2
Emergency Plan

TSC ORGANIZATION

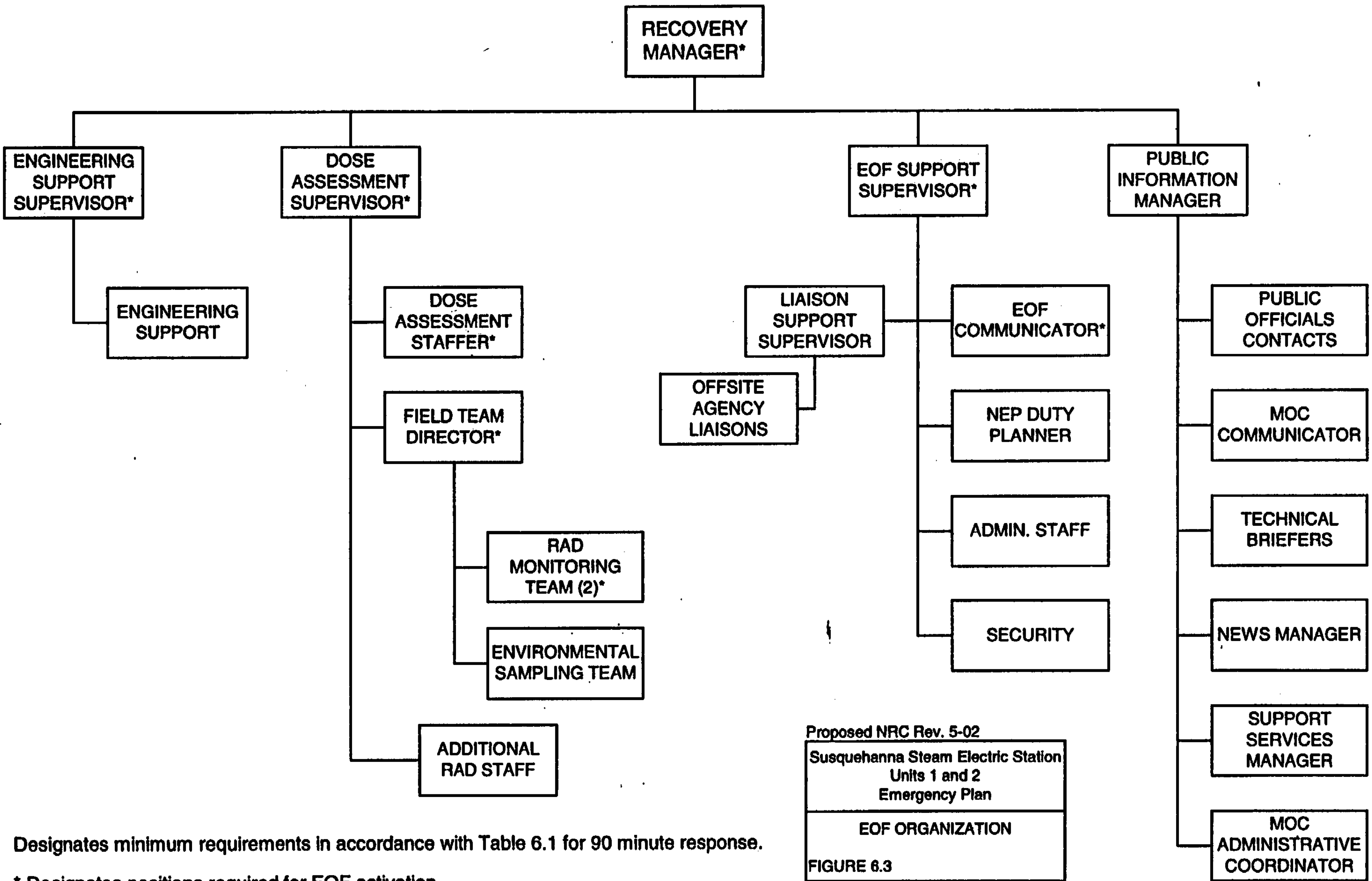
FIGURE 6.2

Designates minimum requirements in accordance with Table 6.1 for 60-minute response.

- - - - - Individuals may be located in the OSC, TSC, or Field.

* Designates positions required for TSC activation.

EOF ORGANIZATION



Designates minimum requirements in accordance with Table 6.1 for 90 minute response.

* Designates positions required for EOF activation.

Proposed NRC Rev. 5-02

Susquehanna Steam Electric Station Units 1 and 2 Emergency Plan
EOF ORGANIZATION FIGURE 6.3

DOSE PROJECTION WORKSHEET

GENERAL:

After initial activation, the following information should be kept current at all times. This information is used by the Health Physics Dose Calculator to perform dose projections in support of Protective Action Recommendations required to be made within 15 minutes of a General Emergency classification.

The Engineering Support function will provide the following information to the Dose Calculator in a timely manner to allow sufficient time to perform a dose projection.

The following information is the best estimate possible considering the time and information available. If the information is not provided, the Dose Calculator will use default values which may yield dose projections much more severe than actual conditions prompting a non-conservative Protective Action Recommendation.

1.0 GENERAL INFORMATION

RX SHUTDOWN TIME: ________	**RELEASE START TIME:** ________	**RELEASE STOP TIME:** ________

2.0 TYPE OF RELEASE (select one)

MONITORED: ________	**UNMONITORED:** ________	**MONITORED & UNMONITORED:** ________

3.0 DBA ACCIDENT TYPES [ISOTOPIC DETERMINATION (CHOOSE ONE)]

UNKNOWN MIX (Containment Rad <5 R/hr)		
NORMAL COOLANT LEAK (Containment Rad <5 R/hr)		
LOCA (No Fuel Damage, Iodine Spike, Containment Rad <10 R/hr)		
LOCA CLAD FAILURE (Containment Rad 1.5E+02 - 5.0E+04 R/hr)		**_______% (Give Estimate)**
LOCA FUEL MELT (Containment Rad 8.0E+03 - 1.0E+06 R/hr)		**_______% (Give Estimate)**
FUEL HANDLING ACCIDENT		

NOTE: Quick methods to determine the isotopic mix/type of fuel damage and estimate percentages are located in HELP tabs entitled Core Damage Estimate I (Primary System Breach Inside Containment) and Core Damage Estimate II (Small or no Primary System Breach Inside Containment).

4.0 RELEASE TYPES (Select release type and go to release selected)

4.1 Drywell Release (circle one in each column)

Core Condition (Choose one, provide estimate in %)	Sprays	Hold-up Time	Treatment	Release Rate
Gap Release (Core uncovered for 15-30 minutes)	On	<1 Hour	Filtered	100%/hr
In Vessel Severe Damage (Core uncovered > 30 minutes)	Off	2-12 Hours	Unfiltered	100%/day
Vessel Melt Through		24 Hours		Design (1%/day)

4.2 Wetwell Release (circle one in each column)

Core Condition (Choose one, provide estimate in %)	Water Conditions	Hold-up Time	Treatment	Release Rate
Gap Release (Core uncovered for 15-30 minutes)	Subcooled	<1 Hour	Filtered	100%/hr
In Vessel Severe Damage (Core uncovered > 30 minutes)	Saturated	2-12 Hours	Unfiltered	100%/day
Vessel Melt Through		24 Hours		Design (1%/day)

4.3 **Secondary Containment Bypass Release (circle one in each column)**

Core Condition (Choose one, provide estimate in %)	**Treatment**	**Release Rate**
Gap Release (Core uncovered for 15-30 minutes)	Filtered	100%/hr
In Vessel Severe Damage (Core uncovered > 30 minutes)	Unfiltered	100%/day
Vessel Melt Through		Design (1%/day)

4.4 **Spent Fuel Pool Release (circle one in each column)**

Core Condition (Choose one)	**Accident Type**	**Hold-up Time**	**Treatment**	**Release Rate**
Gap Release (Core uncovered for 15-30 minutes)	Zircaloy Fire in One 3 Month Batch	<1 Hr	Filtered/ Sprays on	100%/hr
In Vessel Severe Damage (Core uncovered > 30 minutes)	Gap Release from One 3 Month Batch	2-12 Hrs.	Unfiltered/ Sprays off	100%/day
Vessel Melt Through	Gap Release from 15 Batches			

EMERGENCY CLASSIFICATION

CHECK ☑

1.0 TIMING OF CLASSIFICATION

☐ 1.1 UNUSUAL EVENT

An **UNUSUAL EVENT** shall be declared within 15 minutes of having information necessary to make a declaration.

☐ 1.2 ALERT

An **ALERT** shall be declared within 15 minutes of having information necessary to make a declaration.

☐ 1.3 SITE AREA EMERGENCY

A **SITE AREA EMERGENCY** shall be declared within 15 minutes of having information necessary to make a declaration.

☐ 1.4 GENERAL EMERGENCY

A **GENERAL EMERGENCY** shall be declared within 15 minutes of having information necessary to make a declaration.

CLASSIFICATION OF EMERGENCY CONDITIONS

USE OF EMERGENCY CLASSIFICATION MATRIX

NOTE: CONFIRM THAT INDICATORS AND/OR ALARMS REFLECT ACTUAL CONDITIONS PRIOR TO TAKING ACTION BASED ON THE INDICATOR OR ALARM.

The matrix is worded in a manner that assumes parameter values indicated are the actual conditions present in the plant.

The matrix is designed to make it possible to precisely classify an abnormal occurrence into the proper emergency classification based on detailed Emergency Action Level (EAL) descriptions. It is impossible to anticipate every abnormal occurrence. Therefore, before classifying any abnormal occurrence based on the EALs in the matrix, one should verify that the general conditions prevalent in-plant and offsite meet the general class description of the emergency classification. In addition, prior to classification, one should be aware of the ramifications in-plant and particularly offsite of that classification. Special consideration of offsite consequences should be made prior to declaring a **GENERAL EMERGENCY**.

CLASS DESCRIPTIONS

UNUSUAL EVENT - Events that are occurring 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.

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

SITE AREA EMERGENCY - Events that are occurring or have occurred which involve actual or imminent major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA Protective Action Guideline exposure levels except inside the emergency planning boundary.

GENERAL EMERGENCY - Events that are occurring or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Expectation is that releases will exceed EPA Protective Action Guideline exposure levels beyond the emergency planning boundary.

CATEGORY INDEX TO THE MATRIX FOR THE CLASSIFICATION OF EMERGENCY CONDITIONS TABLE OF CONTENTS

1 - AIRCRAFT/TRAIN ACTIVITY

UNUSUAL EVENT

EAL# 1.1 Aircraft crash or train derailment onsite as indicated by:

Visual observation or notification received by control room operator.

ALERT

EAL# 1.2 Aircraft or missile strikes a station structure as indicated by:

Direct observation or notification received by control room operator.

SITE AREA EMERGENCY

EAL# 1.3 Severe damage to safe shutdown equipment from aircraft crash or missile impact when not in cold shutdown, determined by:

(A *and* B *and* C)
A. Direct observation or notification received by control room operator.
and
B. Shift Supervisor evaluation.
and
C. Reactor Coolant temperature greater than 200°F as indicated on Panel 1C651 (2C651).

GENERAL EMERGENCY

EAL# 1.4 None.

2 - CONTROL ROOM EVACUATION

UNUSUAL EVENT

EAL# 2.1 None.

ALERT

EAL# 2.2 Control Room evacuation as indicated by:

(A *and* B)
A. Initiation of control room evacuation procedures.
and
B. Establishment of control of shutdown systems from local stations.

SITE AREA EMERGENCY

EAL# 2.3 Delayed Control Room Evacuation as indicated by:

(A *and* B)
A. Initiation of control room evacuation procedures.
and
B. Shutdown systems control at local stations not established within 15 minutes.

GENERAL EMERGENCY

EAL# 2.4 None.

3 - FUEL CLADDING DEGRADATION

UNUSUAL EVENT

EAL# 3.1 Core degradation as indicated by:

(A *or* B)

A. Valid Off-gas Pre-treatment Monitor high radiation alarm annunciation on Panel 1C651 (2C651) or indication on Panel 1C600 (2C600).

or

B. Reactor coolant activity, determined by sample analysis greater than or equal to 2 µCi/cc of I-131 equivalent.

ALERT

EAL# 3.2 Severe fuel cladding degradation as indicated by:

(A *or* B *or* C *or* D)

A. Valid Off-gas Pre-treatment monitor High-High radiation alarm annunciation on Panel 1C651 (2C651) or indication on Panel 1C600 (2C600).

or

B. Valid Reactor coolant activity greater than 300 µCi/cc of equivalent I-131, as determined by sample analysis.

or

C. Valid Main Steam Line High radiation trip annunciation or indication on Panel 1C651 (2C651).

or

D. Valid containment post accident monitor indication on Panel 1C601 (2C601) greater than 200 R/hr. (An 8R/hr correction factor must be added manually to the indication to offset a downscale error if primary containment temperature exceeds 225 degrees Fahrenheit. Reference EC-079-0521.)

(CONTINUED ON NEXT PAGE)

3 - FUEL CLADDING DEGRADATION (continued)

SITE AREA EMERGENCY

EAL# 3.3 Severely degraded core as indicated by:

(A *or* B)

A. Reactor coolant activity greater than 1,000 µCi/cc of equivalent I-131 as determined by sample analysis.

or

B. Valid containment post accident monitor indication on Panel 1C601 (2C601) greater than 400 R/hr. (An 8 R/hr correction factor must be added manually to the indication to offset a downscale error if primary containment temperature exceeds 225 degrees Fahrenheit. Reference EC-079-0521.)

(CONTINUED ON NEXT PAGE)

3 - FUEL CLADDING DEGRADATION (continued)

GENERAL EMERGENCY

EAL# 3.4.a Fuel cladding degradation. Loss of 2 out of 3 fission product barriers (fuel cladding and reactor coolant pressure boundary) with potential loss of the third barrier (primary containment) as indicated by:

(A *or* B)

A. (1 *and* 2)

1. Valid containment post accident monitor indication on Panel 1C601 (2C601) greater than 400 R/hr. (An 8 R/hr correction factor must be added manually to the indication to offset a downscale error if primary containment temperature exceeds 225 degrees Fahrenheit. Reference EC-079-0521.)

<u>and</u>

2. (a *or* b *or* c)
   a. Containment pressure greater than 40.4 PSIG, indicated on Panel 1C601 (2C601).

   <u>or</u>

   b. A visual inspection of the containment indicates a potential for loss of containment (e.g. anchorage or penetration failure, a crack in containment concrete at tendon).

   <u>or</u>

   c. Other indications of potential or actual loss of primary containment.

<u>or</u>

B. (1 *and* 2)

1. Reactor coolant activity greater than 1,000 µCi/cc of equivalent I-131 as determined by sample analysis.

<u>and</u>

2. Actual or potential failure of reactor coolant isolation valves to isolate a coolant leak outside containment as determined by valve position indication on Panel 1C601 (2C601) or visual inspection.

<u>OR</u>

EAL# 3.4.b Core melt as indicated by:

(A *and* B)

A. Valid containment post accident monitor indication on Panel 1C601 (2C601) greater than 2000 R/hr. (An 8 R/hr correction factor must be added manually to the indication to offset a downscale error if primary containment temperature exceeds 225 degrees Fahrenheit. Reference EC-079-0521.)

<u>and</u>

B. Containment high pressure indication or annunciation on Panel 1C601 (2C601).

4 - GENERAL

UNUSUAL EVENT

EAL# 4.1 Plant conditions exist that warrant increased awareness on the part of plant operating staff or state and/or local offsite authorities as indicated by:

Events that are occurring 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.

ALERT

EAL# 4.2 Other plant conditions exist that warrant precautionary activation of PPL, State, County, and local emergency centers as indicated by:

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

SITE AREA EMERGENCY

EAL# 4.3 Other plant conditions exist that warrant activation of emergency centers and monitoring teams or a precautionary notification to the public near the site as indicated by:

Events that are occurring or have occurred which involve actual or imminent major failures of plant functions needed for protection of the public. Any releases are not expected to exceed EPA Protective Action Guideline exposure levels except inside the emergency planning boundary.

GENERAL EMERGENCY

EAL# 4.4 Other plant conditions exist, from whatever, source, that make release of large amounts of radioactivity in a short time period available as indicated by:

Events that are occurring or have occurred which involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Expectation is that releases will exceed EPA Protective Action Guideline exposure levels beyond the emergency planning boundary.

5.- INJURED/CONTAMINATED PERSONNEL

UNUSUAL EVENT

EAL# 5.1 Transportation of externally contaminated injured individual from site to offsite medical facility as deemed appropriate by Shift Supervisor.

ALERT

EAL# 5.2 None.

SITE AREA EMERGENCY

EAL# 5.3 None.

GENERAL EMERGENCY

EAL# 5.4 None.

6 - IN-PLANT HIGH RADIATION

UNUSUAL EVENT

EAL# 6.1 Unanticipated or unplanned concentrations of airborne activity exist in normally accessible areas, which are not due to planned maintenance activities, as indicated by:

Concentrations exceed 500 times the DAC values of 10CFR20 Appendix B, Table I values for a single isotope, or for multiple isotopes where

$$\frac{C_A}{DAC_A}+\frac{C_B}{DAC_B}+\frac{C_C}{DAC_C}\cdots\frac{C_N}{DAC_N}\geq 500$$

ALERT

EAL# 6.2 Unexpected in-plant high radiation levels or airborne contamination which indicates a severe degradation in the control of radioactive material as indicated by:

Area Radiation Monitor reading 1000 times normal annunciation on Panel 1C601 (2C601) or indication on Panel 1C600 (2C600).

SITE AREA EMERGENCY

EAL# 6.3 None.

GENERAL EMERGENCY

EAL# 6.4 None.

7 - LOSS OF AC POWER

UNUSUAL EVENT

EAL# 7.1 Loss of offsite power <u>or</u> loss of all onsite AC power supplies as indicated by:

(A *or* B)

A. Loss of power to Startup Transformer 10 <u>and</u> 20 annunciation or indication on Panel 0C653.

<u>or</u>

B. Failure of all diesel generators to start or synchronize to the emergency buses by indication or annunciation on Panel 0C653.

ALERT

EAL# 7.2 Loss of all offsite power <u>and</u> all onsite AC power supplies as indicated by:

(A *and* B)

A. Loss of power to Startup Transformer 10 <u>and</u> 20 annunciation or indication on Panel 0C653.

<u>and</u>

B. Failure of all diesel generators to start or synchronize to the emergency buses by annunciation or indication on Panel 0C653.

SITE AREA EMERGENCY

EAL# 7.3 Loss of all offsite power and loss of all onsite AC power supplies for greater than 15 minutes as indicated by:

(A *and* B *and* C)

A. Loss of offsite power.

<u>and</u>

B. Failure of <u>all</u> diesel generators to startup or synchronize to the emergency buses by indication or annunciation on 0C653.

<u>and</u>

C. The above conditions exist for greater than 15 minutes.

GENERAL EMERGENCY

EAL# 7.4 None.

8 - LOSS OF CONTROL ROOM ALARMS AND ANNUNCIATORS

UNUSUAL EVENT

EAL# 8.1 None.

ALERT

EAL# 8.2 Loss of all control room annunciators as indicated by:

In the opinion of the Shift Supervisor, all Control Room annunciators and the Plant Process Computer are lost, or insufficient annunciators are available to safely operate the unit(s) without supplemental observation of plant systems.

SITE AREA EMERGENCY

EAL# 8.3 All annunciators lost and plant transient initiated while annunciators are lost as indicated by:

(A *and* B)

A. In the opinion of the Shift Supervisor, all Control Room annunciators and the Plant Process Computer are lost, or insufficient annunciators are available to safely operate the unit(s) without supplemental observation of plant systems.

<u>and</u>

B. (1 *or* 2 *or* 3 *or* 4)

1. Low-Low reactor water level indication on Panel 1C651 (2C651) followed by ECCS initiation on Panel 1C601 (2C601).

<u>or</u>

2. Reactor coolant temperature change greater than 100°F per hour indication on recorder TR-1R006 on Panel 1C007 (2C007) (Reactor Building elevation 683').

<u>or</u>

3. High reactor pressure indication on Panel 1C651 (2C651) and followed by scram indication on Panel 1C651 (2C651).

<u>or</u>

4. Any indication that transient has occurred or is in progress.

GENERAL EMERGENCY

EAL# 8.4 None.

9 - LOSS OF DC POWER

UNUSUAL EVENT

EAL# 9.1 None.

ALERT

EAL# 9.2 Loss of onsite vital DC power as indicated by:

(A *and* B)

A. Less than 210 volts on the 250 VDC main distribution Panel buses, 1D652 (2D652) and 1D662 (2D662) as indicated by trouble alarms on Panel 1C651 (2C651).

and

B. Less than 105 volts on the 125 VDC main distribution buses 1D612 (2D612), 1D622 (2D622), 1D632 (2D632), and 1D642 (2D642) as indicated by trouble alarms on Panel 1C651 (2C651).

NOTE: Buses are not tripped on undervoltage condition.

SITE AREA EMERGENCY

EAL# 9.3 Loss of all vital onsite DC power sustained for greater than 15 minutes as indicated by:

(A *and* B *and* C)

A. Less than 210 volts on the 250 VDC main distribution Panel buses, 1D652 (2D652) and 1D662 (2D662) as indicated by trouble alarms on Panel 1C651 (2C651).

and

B. Less than 105 volts on the 125 VDC main distribution buses 1D612 (2D612), 1D622 (2D622), 1D632 (2D632), and 1D642 (2D642) as indicated by trouble alarms on Panel 1C651 (2C651).

and

C. The above condition exists for greater than 15 minutes.

NOTE: Buses are not tripped on undervoltage condition.

GENERAL EMERGENCY

EAL# 9.4 None.

10 - LOSS OF DECAY HEAT REMOVAL CAPABILITY

UNUSUAL EVENT

EAL# 10.1 None.

ALERT

EAL# 10.2 Inability to remove decay heat while in plant condition 4, inability to maintain the plant in cold shutdown as indicated by:

Inability to maintain reactor coolant temperature less than 200°F with the reactor mode switch in shutdown; exception is when testing per Special Test Exception TS 3.10.1 which allows maximum temperature of 212°F.

SITE AREA EMERGENCY

EAL# 10.3 Inability to remove decay heat while the plant is shutdown as indicated by:

(A *and* B *and* C)
A. Reactor Mode switch in shutdown.
<u>and</u>
B. Reactor Coolant System temperature greater than 200°F and rising.
<u>and</u>
C. Suppression Pool temperature greater than 120°F and rising.

GENERAL EMERGENCY

EAL# 10.4 Inability to remove decay heat while the plant is shutdown with possible release of large amounts of radioactivity as indicated by:

(A *and* B *and* C)
A. Reactor mode switch in shutdown.
<u>and</u>
B. Reactor coolant system temperature greater than 200°F and rising.
<u>and</u>
C. Suppression pool temperature greater than 290°F indicated on the computer output (MAT 12,13,14,15 or 16).

11 - LOSS OF REACTIVITY CONTROL

UNUSUAL EVENT

EAL# 11.1 Inadvertent Criticality as indicated by:

Unexpected increasing neutron flux indication on Panel 1C651 (2C651).

ALERT

EAL# 11.2 Failure of the Reactor Protection System or the Alternate Rod Insertion System to initiate and complete a scram that brings the reactor subcritical as indicated by:

(A *or* B) *and* (C *and* D *and* E)

A. Trip of at least one sub-channel in each trip system (RPS A and RPS B) as indicated by annunciators and trip status lights on Panel 1C651 (2C651).

or

B. Trip of both trip systems (ARI A and ARI B) as indicated by annunciators on Panel 1C601 (2C601).

and

C. Failure of control rods to insert, confirmed by the full core display indication on Panel 1C651 (2C651) or process computer indications.

and

D. Failure to bring the reactor subcritical confirmed by neutron count rate on the neutron monitoring indication on Panel 1C651 (2C651).

and

E. Reactor power >5% as indicated on Panel 1C651 (2C651).

(CONTINUED ON NEXT PAGE)

11 - LOSS OF REACTIVITY CONTROL (continued)

SITE AREA EMERGENCY

EAL# 11.3 Loss of functions needed to bring the reactor subcritical and loss of ability to bring the reactor to cold shutdown as indicated by:

(A *and* B *and* C *and* D)

A. Inability to insert sufficient control rods to bring the reactor subcritical as indicated by count rate on the neutron monitoring instrumentation on Panel 1C651 (2C651).

<u>and</u>

B. (1 *or* 2)
Failure of both loops of standby liquid control to inject into the vessel indicated by:
1. Low pump discharge pressure indication on Panel 1C601 (2C601).

<u>or</u>

2. Low flow indication on Panel 1C601 (2C601).

<u>and</u>

C. Reactor coolant temperature greater than 200°F, indicated on Panel 1C651 (2C651).

<u>and</u>

D. Reactor power >5% indicated on Panel 1C651 (2C651).

GENERAL EMERGENCY

EAL# 11.4 Loss of functions needed to bring the reactor subcritical and transient in progress that makes release of large amounts of radioactivity in a short period possible as indicated by:

(A *or* B) *and* (C *and* D)

A. Trip of at least one sub-channel in each trip system (RPS A and RPS B), indicated by annunciation or trip status lights on Panel 1C651 (2C651).

<u>or</u>

B. Trip of both systems (ARI A and ARI B) as indicated by annunciators on Panel 1C601 (2C601).

<u>and</u>

C. Loss of SLC system capability to inject, indicated by instrumentation on Panel 1C601 (2C601).

<u>and</u>

D. Reactor power greater than 25% of rated, indicated on Panel 1C651 (2C651).

12 - LOSS OF REACTOR VESSEL INVENTORY

UNUSUAL EVENT

EAL# 12.1 Valid initiation of an Emergency Core Cooling System (ECCS) System as indicated by:

(A *or* B)

A. Initiation of an ECCS System <u>and</u> low, low, low reactor water level (-129) annunciation or indication on Panel 1C651 (2C651).

<u>or</u>

B. Initiation of an ECCS System <u>and</u> High Drywell Pressure annunciation or indication on Panel 1C601 (2C601).

ALERT

EAL# 12.2 Reactor coolant system leak rate greater than 50 gpm as indicated by:

(A *or* B)

A. Drywell floor drain sump A or B Hi-Hi alarm on Panel 1C601 (2C601) <u>and</u> 2 or more drywell floor drain pumps continuously running as indicated on Panel 1C601 (2C601).

<u>or</u>

B. Other estimates of Reactor coolant system leakage indicating greater than 50 gpm.

SITE AREA EMERGENCY

EAL# 12.3 Known loss of coolant accident greater than make-up capacity as indicated by:

Water level below (and failure to return to) top of active fuel for greater than three minutes as indicated on fuel zone level indicator on Panel 1C601 (2C601).

(CONTINUED ON NEXT PAGE)

12 - LOSS OF REACTOR VESSEL INVENTORY (continued)

GENERAL EMERGENCY

EAL# 12.4.a Loss of coolant accident with possibility of imminent release of large amounts of radioactivity as indicated by:

Water level below (and failure to return to) top of active fuel for greater than 20 minutes as indicated on fuel zone level indicator on Panel 1C601 (2C601).

OR

EAL# 12.4.b Loss of Reactor Vessel inventory. Loss of 2 out of 3 fission product barriers (fuel cladding & reactor coolant pressure boundary) with potential loss of the third barrier (primary containment), as indicated by:

(A *or* B)

A. (1 *and* 2 *and* 3)

1. High drywell pressure annunciation or indication on Panel 1C601 (2C601).

   and

2. (a *or* b *or* c)
   - a. Containment pressure exceeds 40.4 PSIG as indicated on Panel 1C601 (2C601).

     or
   - b. A visual inspection of the containment indicates a potential or actual loss of containment (e.g. anchorage or penetration failure).

     or
   - c. Containment isolation valve(s) fail to close as indicated by valve position indication on Panel 1C601 (2C601).

   and

3. Reactor Vessel level drops below (and fails to return to) top of active fuel for greater than three minutes as indicated on fuel zone level indicator on Panel 1C601 (2C601).

or

B. (1 *and* 2)

1. Failure of reactor pressure vessel isolation valves to isolate coolant break outside containment as indicated by valve position indication on Panel 1C601 (2C601) or visual inspection.

   and

2. Reactor vessel level drops below (and fails to return to) top of active fuel for greater than three minutes as indicated on fuel zone level indicator on Panel 1C601 (2C601).

13 - NATURAL PHENOMENA

UNUSUAL EVENT

EAL# 13.1 Natural phenomenon occurrence as indicated by:

(A *or* B *or* C)
A. Tornado impact on site.
or
B. Hurricane impact on site.
or
C. Earthquake detected by seismic instrumentation systems on Panel 0C696.

ALERT

EAL# 13.2 Natural Phenomenon Occurrence as indicated by:

(A *or* B *or* C)
A. Tornado with reported wind velocities greater than 200 mph impacting on site.*
or
B. Reported hurricane or sustained winds greater than 70 mph.*
or
C. Earthquake at greater than operating basis earthquake (OBE) levels as indicated on Panel 0C696.

* **Telephone numbers for the National Weather Bureau are located in the Emergency Telephone Directory.**

(CONTINUED ON NEXT PAGE)

13 - NATURAL PHENOMENA (continued)

SITE AREA EMERGENCY

EAL# 13.3 Severe natural phenomenon occurrence, with plant not in cold shutdown, as indicated by:

(A *and* B)

A. Reactor Coolant Temperature greater than 200°F as indicated on Panel 1C651 (2C651).

and

B. (1 *or* 2 *or* 3)

1. Reported hurricane or sustained winds greater than 80 mph.*

   or

2. Earthquake with greater than Safe Shutdown Earthquake (SSE) levels as indicated on Panel 0C696.

   or

3. Tornado with reported wind velocities greater than 220 mph impacting on site.*

GENERAL EMERGENCY

EAL# 13.4 None.

* **Telephone numbers for the National Weather Bureau are located in the Emergency Telephone Directory.**

14 - ONSITE FIRE/EXPLOSION

UNUSUAL EVENT

EAL# 14.1 Significant fire within the plant as indicated by:

(A *and* B)
A. Activation of fire brigade by Shift Supervisor.
and
B. Duration of fire longer than 15 minutes after time of notification.

OR

Explosion inside security protected area, with no significant damage to station facilities, as indicated by:

Visual observation or notification received by control room operator and Shift Supervisor evaluation.

ALERT

EAL# 14.2 On-site Fire/Explosion as indicated by:

(A *or* B)
A. Fire lasting more than 15 minutes and fire is in the vicinity of equipment required for safe shutdown of the plant and the fire is damaging or is threatening to damage the equipment due to heat, smoke, flame, or other hazard.
or
B. (1 *and* 2)
Explosion damage to facility affecting plant operation as determined by:
1. Direct observation or notification received by control room operator.
and
2. Shift Supervisor observation.

(CONTINUED ON NEXT PAGE)

14 - ONSITE FIRE/EXPLOSION (continued)

SITE AREA EMERGENCY

EAL# 14.3 Damage to safe shutdown equipment due to fire or explosion has occurred when plant is not in cold shutdown, and damage is causing or threatens malfunction of equipment required for safe shutdown of the plant as determined by:

(A *and* B *and* C)
A. Direct observation or notification received by control room operator.
and
B. Shift Supervisor evaluation.
and
C. Reactor Coolant Temperature greater than 200°F as indicated on Panel 1C651 (2C651).

GENERAL EMERGENCY

EAL# 14.4 None.

15 - RADIOLOGICAL EFFLUENT

UNUSUAL EVENT

EAL# 15.1 Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 2 times the Technical Requirements Manual limits for 60 minutes or longer.

EAL# 15.1 (1 or 2 or 3)

1. Valid Noble Gas vent stack monitor reading(s) that exceeds a total site release rate of 2.0E+6 µCi/min and that is sustained for 60 minutes or longer.

OR

2. Confirmed sample analyses for airborne releases indicates total site release rates at the site boundary with a release duration of 60 minutes or longer resulting in dose rates of:

   a) Noble gases >1000 mrem/year whole body, or
   b) Noble gases >6000 mrem/year skin, or
   c) I-131, I-133, H-3, and particulates with half lives >8 days >3000 mrem/year to any organ (inhalation pathways only).

OR

3. Confirmed sample analyses for liquid releases indicates concentrations with a release duration of 60 minutes or longer in excess of two time the Technical Requirements Manual liquid effluent limits.

(CONTINUED ON NEXT PAGE)

15 - RADIOLOGICAL EFFLUENT (continued)

ALERT

EAL# 15.2 Any unplanned release of gaseous or liquid radioactivity to the environment that exceeds 200 times Technical Requirement Manual limits for 15 minutes or longer.

EAL# 15.2 (1 or 2 or 3)

1. Valid Noble Gas vent stack monitor reading(s) that exceeds a total site release rate of 2E+8 µCi/min and that is sustained for 15 minutes or longer.

OR

2 Confirmed sample analyses for airborne releases indicates total site release rates at the site boundary for 15 minutes or longer resulting in dose rates of:

   a) Noble gases >1.0E+5 mrem/year whole body, or
   b) Noble gases >6.0E+5 mrem/year skin, or
   c) I-131, I-133, H-3, and particulates with half-lives >8 days >3.0E+5 mrem/year to any organ (inhalation pathways only).

OR

3. Confirmed sample analyses for liquid releases indicates concentrations in excess of 200 times the Technical Requirements Manual liquid effluent limits for 15 minutes or longer.

(CONTINUED ON NEXT PAGE)

15 - RADIOLOGICAL EFFLUENT (continued)

SITE AREA EMERGENCY

EAL# 15.3 Dose at the Emergency Plan boundary resulting from an actual or imminent release of gaseous radioactivity exceeds 100 mrem whole body TEDE or 500 mrem child thyroid CDE for the actual or projected duration of release.

EAL# 15.3 (1 or 2 or 3 or 4 or 5)

1. Valid Noble Gas vent stack monitor readings(s) that exceeds a total release rate 6.2E8 µCi/min for greater than 15 minutes and Dose Projections are not available.

   Note: If the required dose projection cannot be completed within the 15 minute period, then the declaration must be made based on a valid sustained monitor reading(s).

<u>OR</u>

2. Valid dose assessment using actual meteorology indicates projected doses greater than 100 mrem whole body TEDE or 500 mrem child thyroid CDE at or beyond the EPB.

<u>OR</u>

3. A valid reading sustained for 15 minutes or longer on the RMS perimeter radiation monitoring system greater than 100 mR/hr.

<u>OR</u>

4. Field survey results indicate Emergency Planning boundary dose rates exceeding 100 mR/hr expected to continue for more than one hour.

<u>OR</u>

5. Analyses of field survey samples indicate child thyroid dose commitment at the Emergency Planning Boundary of 500 mrem for one hour of inhalation.

(CONTINUED ON NEXT PAGE)

15 - RADIOLOGICAL EFFLUENT (continued)

GENERAL EMERGENCY

EAL# 15.4 Dose at the Emergency Planning Boundary resulting from an actual or imminent release of gaseous radioactivity exceeds 1000 mrem whole body TEDE or 5000 mrem child thyroid CDE for the actual or projected duration of the release using actual meteorology.

EAL# 15.4 (1 or 2 or 3 or 4 or 5)

1. Valid Noble Gas vent stack monitor readings(s) that exceed a total release rate of 6.2E9 µCi/min for greater that 15 minutes and Dose Projections are not available.

   Note: If the required dose projection cannot be completed within the 15 minute period, then the declaration must be made based on a valid sustained monitor reading(s).

<u>OR</u>

2. Valid dose assessment using actual meteorology indicates projected doses greater than 1000 mrem whole body TEDE or 5000 mrem child thyroid CDE at or beyond the EPB.

<u>OR</u>

3. A valid reading sustained for 15 minutes or longer on the RMS perimeter radiation monitoring system greater than 1000 mR/hr.

<u>OR</u>

4. Field survey results indicate Emergency Planning Boundary dose rates exceeding 1000 mR/hr expected to continue for more than one hour.

<u>OR</u>

5. Analyses of field survey samples indicate child thyroid dose commitment at the Emergency Planning Boundary of 5000 mrem for one hour of inhalation.

16 - SECURITY EVENT

UNUSUAL EVENT

EAL# 16.1 Security threat or attempted entry or attempted sabotage as indicated by:

(A *or* B *or* C)

A. A report from Security of a security threat, attempted entry, or attempted sabotage of the owner controlled area adjacent to the site.

or

B. Any attempted act of sabotage which is deemed legitimate in the judgment of the SHIFT SUPERVISOR/EMERGENCY DIRECTOR, and affects plant operation.

or

C. A site specific credible security threat notification.

ALERT

EAL# 16.2 Ongoing Security Compromise as indicated by:

(A *or* B)

A. A report from Security that a security compromise is at the site but no penetration of protected areas has occurred.

or

B. Any act of sabotage which results in an actual or potential substantial degradation of the level of safety of the plant as judged by the SHIFT SUPERVISOR/EMERGENCY DIRECTOR.

SITE AREA EMERGENCY

EAL# 16.3 An ongoing adversary event threatens imminent loss of physical control of plant as indicated by:

(A *or* B)

A. Report from Security that the security of the plant vital area is threatened by unauthorized (forcible) entry into the protected area.

or

B. Any act of sabotage which results in actual or likely major failures of plant functions needed for protection of the public as judged by the SHIFT SUPERVISOR/EMERGENCY DIRECTOR.

(CONTINUED ON NEXT PAGE)

16 - SECURITY EVENT (continued)

GENERAL EMERGENCY

EAL# 16.4 Loss of physical control of facilities as indicated by:

(A *or* B)

A. Report from Security that a loss of physical control of plant vital areas has occurred.

or

B. Any act of sabotage which results in imminent significant cladding failure or fuel melting with a potential for loss of containment integrity or the potential for release of significant amounts of radioactivity in a short time as judged by the SHIFT SUPERVISOR/EMERGENCY DIRECTOR.

17 - SPENT FUEL RELATED INCIDENT

UNUSUAL EVENT

EAL# 17.1 Unanticipated or unplanned concentrations of airborne activity exist in normally accessible areas, which is not due to planned maintenance activities, as indicated by:

Concentrations exceed 500 times the DAC values of 10CFR20 Appendix B, Table I values for a single isotope, or full multiple isotopes where

$$\frac{C_A}{DAC_A} + \frac{C_B}{DAC_B} + \frac{C_C}{DAC_C} \cdots \frac{C_N}{DAC_N} \geq 500$$

ALERT

EAL# 17.2 Unexpected in-plant high radiation levels or airborne contamination which indicates a severe fuel handling accident as indicated by:

Refuel floor area radiation monitor reading 1000 times normal annunciation on Panel 1C601 (2C601) or indication on Panel 1C600 (2C600).

(CONTINUED ON NEXT PAGE)

17 - SPENT FUEL RELATED INCIDENT (continued)

SITE AREA EMERGENCY

EAL# 17.3.a Major damage to irradiated fuel with actual or clear potential for significant release of radioactive material to the environment as indicated by:

(A *and* B)
A. Dropping, bumping, or otherwise rough handling of a new OR irradiated fuel bundle with irradiated fuel in the pool.
and
B. (1 *or* 2)
1. Refueling floor area radiation monitor reading 1000 times normal annunciation on Panel 1C601 (2C601) or indication on Panel 1C600 (2C600).
or
2. Reactor Building vent stack monitoring system high radiation annunciation or indication on Panel 0C630 or 0C677.

OR
EAL# 17.3.b Damage to irradiated fuel due to uncontrolled decrease in the fuel pool level to below the level of the fuel as indicated by:

(A *and* B)
A. (1 *or* 2)
1. Uncovering of irradiated fuel confirmation by verification of significant leakage from spent fuel pool.
or
2. Visual observation of water level below irradiated fuel in the pool.
and
B. (1 *or* 2)
1. Refueling floor area radiation monitor annunciation on Panel 1C651 (2C651) or indication on Panel 1C600 (2C600).
or
2. Reactor Building vent stack monitoring system high radiation annunciation or indication on Panel 0C630 or 0C677.

GENERAL EMERGENCY

EAL# 17.4 None.

18 - STEAM LINE BREAK

UNUSUAL EVENT

EAL# 18.1 None.

ALERT

EAL# 18.2 MSIV malfunction causing leakage as indicated by:

(A *and* B)
A. Valid MSIV closure signal or indication on Panel 1C601 (2C601).
and
B. (1 *or* 2)
1. Valid Main Steam Line flow indication on Panel 1C652 (2C652).
or
2. Valid Main Steam Line radiation indication on Panel 1C600 (2C600).

(CONTINUED ON NEXT PAGE)

18 - STEAM LINE BREAK (continued)

SITE AREA EMERGENCY

EAL# 18.3 Steam line break occurs outside of containment without isolation as indicated by:

(A *or* B *or* C *or* D)

A. (1 *and* 2)

1. Failure of both MSIVs in the line with the leak to close as indicated by position indication on Panel 1C601 (2C601).

<u>and</u>

2. (a *or* b)

   a. High MSL flow annunciation on Panel 1C601 (2C601) or indication on Panel 1C652 (2C652).

   <u>or</u>

   b. Other indication of main steam leakage outside containment.

<u>or</u>

B. (1 *and* 2)

1. Failure of RCIC steam isolation valves HV-F008 and HV-F007 to close as indicated on Panel 1C601 (2C601).

<u>and</u>

2. (a *or* b *or* c *or* d *or* e *or* f)

   a. RCIC steamline pipe routing area high temperature annunciation on Panel 1C601 (2C601), or indication on Panel 1C614 (2C614).

   <u>or</u>

   b. RCIC equipment area high temperature annunciation on Panel 1C601 (2C601) or indication on Panel 1C614 (2C614).

   <u>or</u>

   c. RCIC steamline high flow annunciation on Panel 1C601 (2C601).

   <u>or</u>

   d. RCIC steamline tunnel ventilation high delta temperature annunciation on Panel 1C601 (2C601).

   <u>or</u>

   e. RCIC turbine exhaust diaphragm high pressure annunciation on Panel 1C601 (2C601).

   <u>or</u>

   f. Other indication of steam leakage from the RCIC system.

(CONTINUED ON NEXT PAGE)

18 - STEAM LINE BREAK (continued)

SITE AREA EMERGENCY (continued)

or

C. (1 *and* 2)

1. Failure of HPCI steam isolation valves HV-F002 and HV-F003 to close as indicated by position indicator on Panel 1C601 (2C601).

   and

2. (a *or* b *or* c *or* d *or* e *or* f)
   a. HPCI steamline pipe routing area high temperature annunciation on Panel 1C601 (2C601), or indication on Panel 1C614 (2C614).

      or

   b. HPCI equipment area high temperature annunciation on Panel 1C601 (2C601) or indication on Panel 1C614 (2C614).

      or

   c. HPCI steamline high flow annunciation on Panel 1C601 (2C601).

      or

   d. HPCI steamline tunnel ventilation high delta temperature annunciation on Panel 1C601 (2C601).

      or

   e. HPCI turbine exhaust diaphragm high pressure annunciation on Panel 1C601 (2C601).

      or

   f. Other indication of steam leakage from the HPCI system.

or

D. Any other un-isolatable steam line breaks.

GENERAL EMERGENCY

EAL# 18.4 None.

19 - TOXIC/FLAMMABLE GASES

UNUSUAL EVENT

EAL# 19.1 Nearby or onsite release of potentially harmful quantifies of toxic or flammable material as indicated by:

Visual observation or notification received by the control room operator.

ALERT

EAL# 19.2 Entry of toxic or flammable gases into the facility, with subsequent habitability problem as indicated by:

Visual observation, direct measurement, or notification received by the control room operator.

SITE AREA EMERGENCY

EAL# 19.3 Toxic or flammable gases enter vital areas, restricting access and restricted access constitutes a safety problem, as determined by:

(A *and* B)
A. Shift Supervisor's evaluation.
<u>and</u>
B. Visual observation, direct measurement, or notification received by control room operator.

GENERAL EMERGENCY

EAL# 19.4 None.

20 - TECHNICAL SPECIFICATION SAFETY LIMIT

UNUSUAL EVENT

EAL# 20.1 Abnormal occurrences which result in operator complying with any of the Technical Specification SAFETY LIMIT ACTION statements indicated by:

(A *or* B *or* C *or* D)

A. Exceeding THERMAL POWER, low pressure or low flow safety limit 2.1.1.1.

or

B. Exceeding THERMAL POWER, high pressure and high flow safety limit 2.1.1.2.

or

C. Exceeding REACTOR VESSEL WATER LEVEL safety limit 2.1.1.3.

or

D. Exceeding REACTOR COOLANT SYSTEM PRESSURE safety limit 2.1.2.

ALERT

EAL# 20.2 None.

SITE AREA EMERGENCY

EAL# 20.3 None.

GENERAL EMERGENCY

EAL# 20.4 None.

21 – DRY FUEL STORAGE

UNUSUAL EVENT

EAL# 21.1.a. Situations are occurring or have occurred during the transport of the irradiated spent fuel to the onsite storage facility, which jeopardize the integrity of the spent fuel or its container as indicated by:

(A *or* B)

A. Radiological readings exceed 2 R/hour at the external surface of any transfer cask or horizontal storage module.

or

B. Radiological readings exceed 1 R/hour one foot away from the external surface of any transfer cask or horizontal storage module.

OR

EAL# 21.1.b. Situations are occurring or have occurred at the irradiated spent fuel storage facility, which jeopardize the integrity of the dry cask storage system as indicated by:

(A *or* B)

A. Radiological readings exceed 2 R/hour at the external surface of any transfer cask or horizontal storage module.

or

B. Radiological readings exceed 1 R/hour one foot away from the external surface of any transfer cask or horizontal storage module.

ALERT

EAL# 21.2 None.

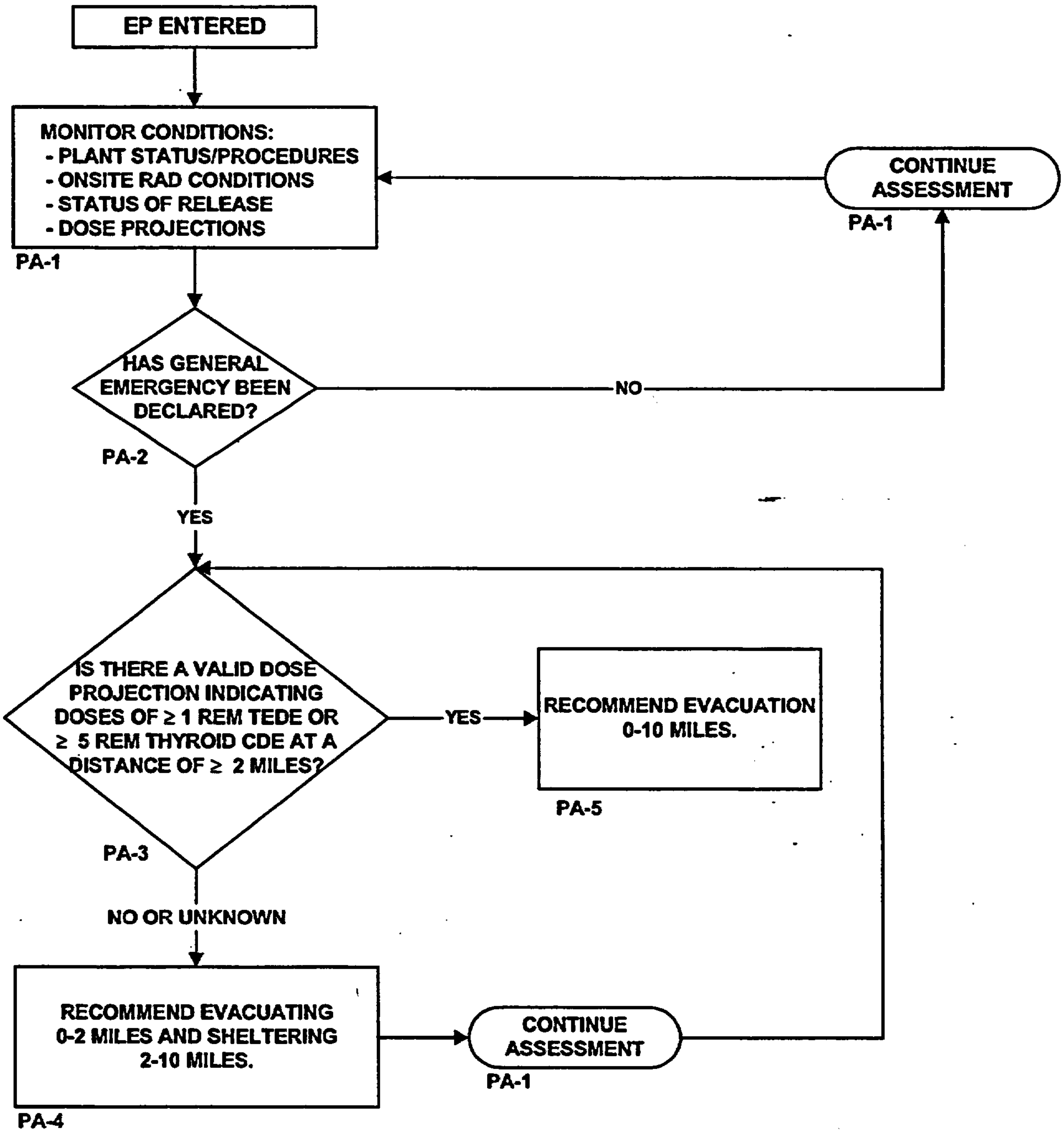
SITE AREA EMERGENCY

EAL# 21.3 None.

GENERAL EMERGENCY

EAL# 21.4 None

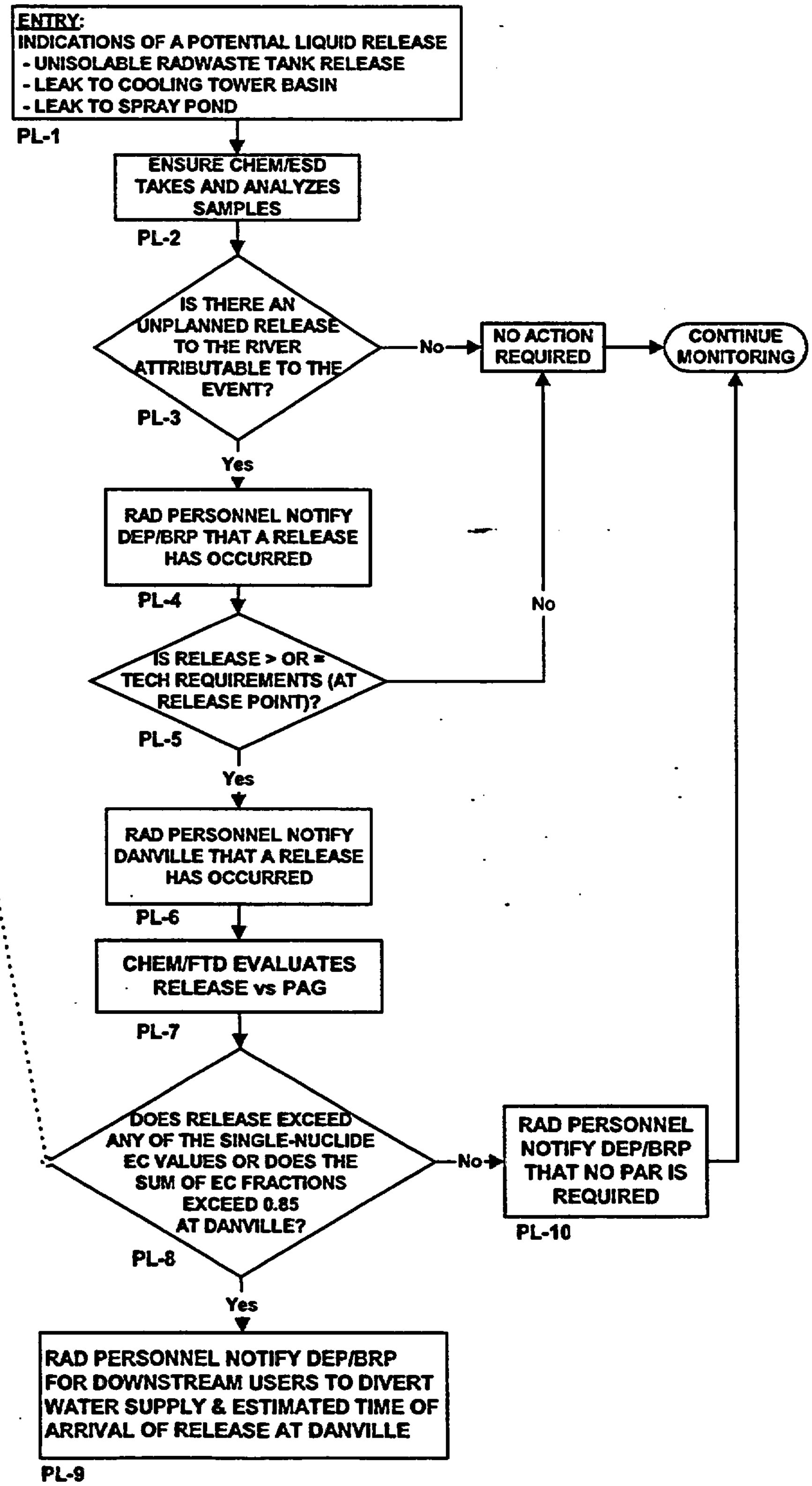
PAR AIRBORNE RELEASES



<u>NOTES</u>:

1. PA-# CAN BE USED TO REFER TO PROCEDURE STEPS FOR MORE DETAILED INFORMATION ON THE ACTION TO BE TAKEN.
2. DOSE PROJECTIONS DO <u>NOT</u> INCLUDE DOSE ALREADY RECEIVED.
3. TEDE - WHOLE BODY (TEDE) IS THE SUM OF EFFECTIVE DOSE EQUIVALENT RESULTING FROM EXPOSURE TO EXTERNAL SOURCES. THE COMMITTED EFFECTIVE DOSE EQUIVALENT (CEDE) FROM ALL SIGNIFICANT INHALATION PATHWAYS AND THE DOSE DUE TO GROUND DEPOSITION.
4. CDE - COMMITTED DOSE EQUIVALENT TO THE CHILD THYROID.

PAR LIQUID RELEASES



RADIONUCLIDE	EC Values (µCi/ml)
Co-60	3E-6
Sr-91	2E-5
Mo-99	2E-5
Te-132	9E-6
I-131	1E-6
I-133	7E-6
I-134	4E-4
I-135	3E-5
Cs-134	9E-7
Cs-136	6E-6
Cs-137	1E-6
Ba-139	2E-4
Ba-140	8E-6
Ba-141	3E-4
Np-239	2E-5

NOTES:
1. PL-# CAN BE USED TO REFER TO PROCEDURE STEPS FOR MORE DETAILED INFORMATION ON THE ACTION TO BE TAKEN.
2. CALLS TO DANVILLE ARE COURTESY INFORMATION CALLS ONLY. PROTECTIVE ACTION RECOMMENDATION CALLS MUST BE MADE BY DEP/BRP.

PUBLIC PROTECTIVE ACTION RECOMMENDATION GUIDE

AIRBORNE RELEASES

❑ PA-1 MONITOR CONDITIONS FOR PAR APPLICATION

The following conditions should be continuously evaluated to determine if a PAR should be implemented or changed:

- Plant status and prognosis for changes in conditions
- Onsite radiological conditions
- Status of actual or potential radioactive releases
- Offsite dose projections or actual offsite radiological conditions
- Escalation in Emergency Classification (i.e., General)

(Go to PA-2)

PA-2 HAS A GENERAL EMERGENCY BEEN DECLARED?

❑ **YES —** If a GENERAL EMERGENCY has been declared, a PAR must be made within 15 minutes of the emergency declaration. The PAR requirement is found in NUREG-0654. **(Go to PA-3)**

❑ **NO —** If a GENERAL EMERGENCY has not been declared, continue to monitor plant status, parameter trends, and prognosis for termination or escalation of the event. **(Go to PA-1)**

PA-3 IS THERE A VALID DOSE PROJECTION INDICATING DOSES OF ≥ 1 REM TEDE OR ≥ 5 REM CDE CHILD THYROID AT A DISTANCE OF > 2 MILES?

❑ **YES —** If the projected doses at 2 miles are ≥ 1 REM TEDE or ≥ 5 REM CDE child thyroid, then full evacuation (0-10 miles) is recommended. **(Go to PA-5)**

❑ **NO/UNKNOWN — (Go to PA-4)**

❑ PA-4 RECOMMEND EVACUATION 0-2 MILES; SHELTER 2-10 MILES

Limited Evacuation (0-2 miles) and sheltering is appropriate for events that are significant enough to cause a General Emergency classification and dose projections are low, unknown, or below full evacuation guidelines.

❑ PA-5 EVACUATE 0-10 MILES

Full evacuation of members of the general public is recommended at this point based on the emergency classification and dose projections.

LIQUID

❑ PL-1 ENTRY

This section is entered when there are indications of a potential unplanned radioactive liquid release.

Indications of potential unplanned releases include:

- an unisolable radwaste tank release
- leaks to cooling tower basin
- leak to spray pond

(Go to PL-2)

❑ PL-2 CHEMISTRY/ENVIRONMENTAL SAMPLING DIRECTOR (ESD) TAKES AND ANALYZES SAMPLE

(Go to PL-3)

PL-3 IS THERE AN UNPLANNED RELEASE TO THE RIVER?

❑ **YES —** An unplanned release to the river has occurred when event-related radioactive materials are released to the river that are not controlled by the release methodologies described in the ODCM and applicable Chemistry procedures.

(Go to PL-4)

❑ **NO —** If there is no unplanned release to the river, then no notifications are required and monitoring should continue.

❑ PL-4 RAD PERSONNEL NOTIFY DEP/BRP THAT A RELEASE HAS OCCURRED

Depending on which facility is activated, the notification to BRP will be made by the RPC (TSC), Dose Assessment Supervisor, or Radiological Liaison at the EOF.

DO NOT MAKE ANY PROTECTIVE ACTION RECOMMENDATIONS AT THIS TIME.

(Go to PL-5)

LIQUID (CONT'D)

PL-5 IS RELEASE ≥ TECHNICAL REQUIREMENTS LIMITS (AT THE RELEASE POINT)?

☐ **YES —** Releases are at or greater than Technical Requirements limits when Chemistry determines that the limits are exceeded based on methodologies described in the ODCM and applicable Chemistry procedures.

(Go to PL-6)

☐ **NO —** If the release is < Technical Requirements limits, then no further notifications are required and monitoring should continue.

☐ PL-6 RAD PERSONNEL NOTIFY DANVILLE THAT A RELEASE HAS OCCURRED

Depending on which facility is activated, the notification to Danville will be made by the RPC (TSC), Dose Assessment Supervisor, or Radiological Liaison at the EOF.

DO NOT MAKE ANY PROTECTIVE ACTION RECOMMENDATIONS AT THIS TIME.

(Go to PL-7)

☐ PL-7 CHEM/FTD EVALUATES RELEASE VERSUS PAGs

The results of the sample analysis are compared to the PAGs for radionuclides in drinking water. The analysis calculates the expected concentration at Danville, taking into account the dilution afforded by the river.

PL-8 DOES RELEASE EXCEED PAGs (AT DANVILLE)?

☐ **YES —** If a single isotope exceeds its effluent concentration (EC) value or the sum of EC fractions exceeds 0.85, then a protective action recommendation should be made for downstream water users (e.g., Danville) to **DIVERT DRINKING WATER** supply to a backup supply or terminate user intake until the release has passed.

(Go to PL-9)

☐ **NO —** If the PAGs are not exceeded, monitoring should continue and the State should be notified that no PAR for the liquid release is required.

(Go to PL-10)

LIQUID (CONT'D)

☐ PL-9 RAD PERSONNEL NOTIFY DEP/BRP OF PAR

Depending on which facility is activated, the PAR notification to DEP/BRP will be made by the RPC (TSC), Dose Assessment Supervisor, or Radiological Liaison at the EOF. **The PAR FORM shall be used to document the PAR.**

DO NOT COMMUNICATE THE PROTECTIVE ACTION RECOMMENDATION TO DANVILLE. THE DEP/BRP IS RESPONSIBLE FOR THIS COMMUNICATION AND ANY COMMUNICATION TO OTHER DRINKING WATER SUPPLIERS OR WATER USERS.

☐ PL-10 RAD PERSONNEL NOTIFY DEP/BRP

No PAR is required. Depending on which facility is activated, the RPC (TSC), Dose Assessment Supervisor, or Radiological Liaison at the EOF shall notify DEP/BRP that no PAR is required.

CORE DAMAGE ESTIMATE I
(Primary System Breach Inside Containment)

NOTE: It is important to quickly provide a status of the present situation and a prognosis on whether the situation is expected to degrade, improve, or remain the same, (i.e., within 5 to 10 minutes of a change in plant status).

1.0 INDICATORS USED

1.1 Containment Radiation

Use Attachment 1, A, B, or C, as applicable, to determine the amount and type of fuel damage using containment radiation monitors. These figures were taken from the US NRC Response Technical Manual, RTM-96. Obtain the containment radiation levels from SPDS or the Control Room indicators.

NOTE (1): Correction for the pre-release background radiation levels may be required as listed below.

Gap or In-Vessel Melt - The background radiation monitor value is normally low (≤ 4 R/hr) relative to 1% gap or in-vessel melt release. Consequently, the monitor reading does not require correction for background level in determining the type and amount of fuel damage. If the background radiation monitor reading is > 4 R/hr, the monitor reading should be corrected for the background level in determining the type and amount of fuel damage.

Spiked or Normal Coolant - The radiation monitor value requires correction for the background level. Correct the monitor reading to account for the normal background level in determining the type and amount of fuel damage.

NOTE (2): Containment radiation will go up if there is fuel damage. The increase will depend on the type of fuel damage, and whether or not there was a LOCA, Drywell and/or Wetwell sprays were used, and the amount of blowdown from the Reactor Vessel to the Suppression Pool.

In the case of a LOCA, the fuel damage estimate depends strongly on whether or not containment sprays are being used. Special care should be taken to confirm the operation of containment sprays.

1.2 **Containment Hydrogen**

Use Attachment 2, taken from the US NRC Response Technical Manual RTM-96, to determine the amount and type of fuel damage using Hydrogen Concentration. Obtain the containment Hydrogen levels from SPDS or the Control Room indicators.

NOTE: Containment Hydrogen will increase if there is a LOCA inside the containment and significant fuel damage.

1.3 **Coolant Fission Product Concentration vs. Core Damage**

Coolant sampling will indicate the amount of fuel damage, but in most cases, will take too long for use in dose projections. If PASS sample data becomes available, the Nuclear Fuels Engineer is responsible for assuring a fuel damage calculation based on the measured fission product inventories is performed. The results of this analysis should be compared to previous calculations using other methods.

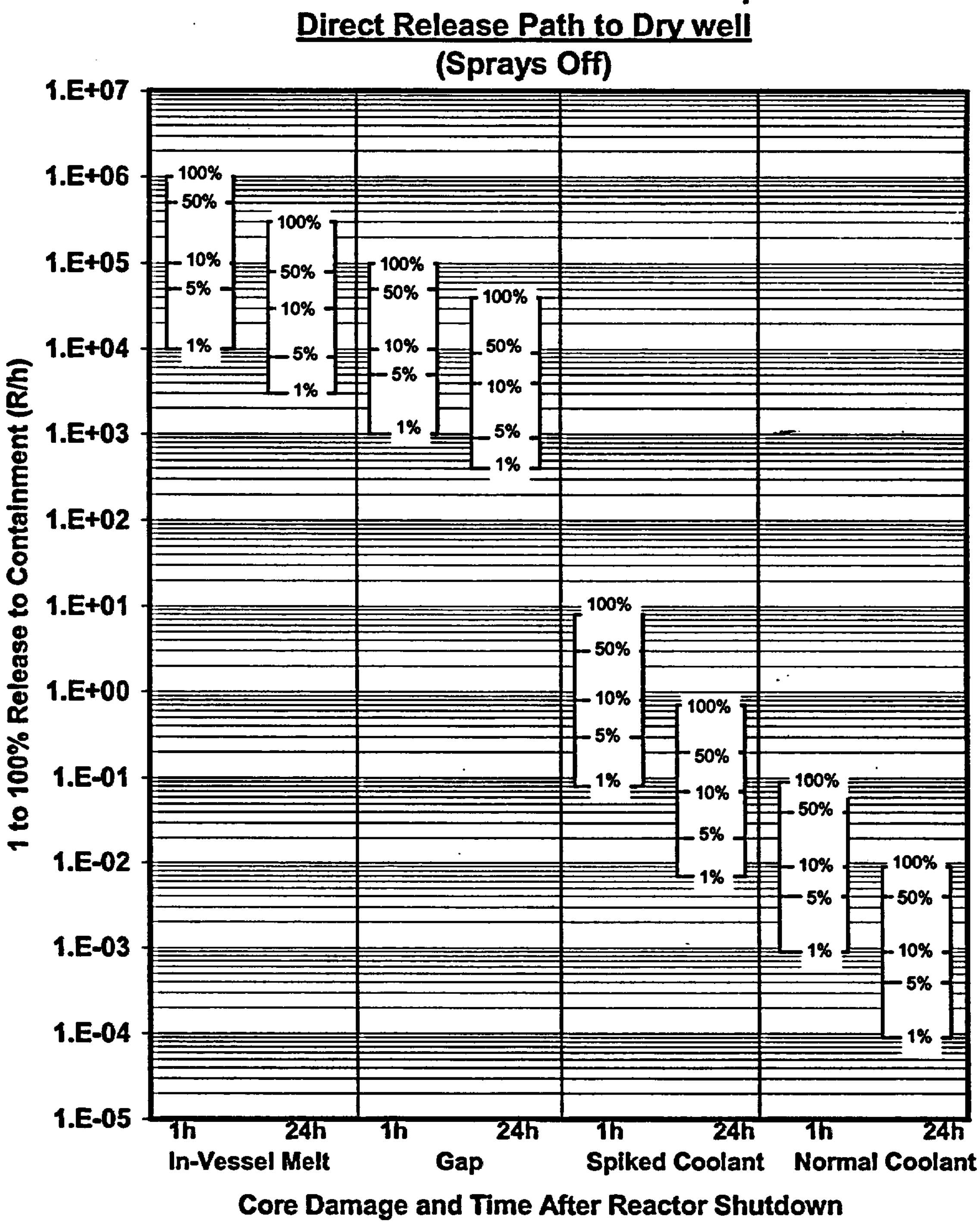
1.4 **Plant Transient Precipitating Fuel Damage**

If the core experienced a loss of coolant accident and is not covered within 15 minutes, refer to Attachment 3 taken from the US NRC Response Technical Manual RTM-96. The amount of time the core was uncovered can be determined using SPDS. Using the attached figures will provide an estimate of potential fuel damage. Coolant samples must be taken to accurately assess fuel damage.

The type of transient experienced by the reactor leading to fuel damage can be an indicator of the amount and type of fission products released.

- If the core experienced an overpower/pressure transient, a gap release may have occurred.
- If the core experienced a mechanical failure, which could produce flow blockage, there may be localized fuel melt.
- If the core experienced a mechanical perturbation, such as a seismic event or a large steam line break causing a large delta pressure across the core, a gap release could result.
- If the Reactor failed to shut down (ATWS) with a subsequent loss of cooling, there may be fuel melt.

Containment Radiation Monitor Response

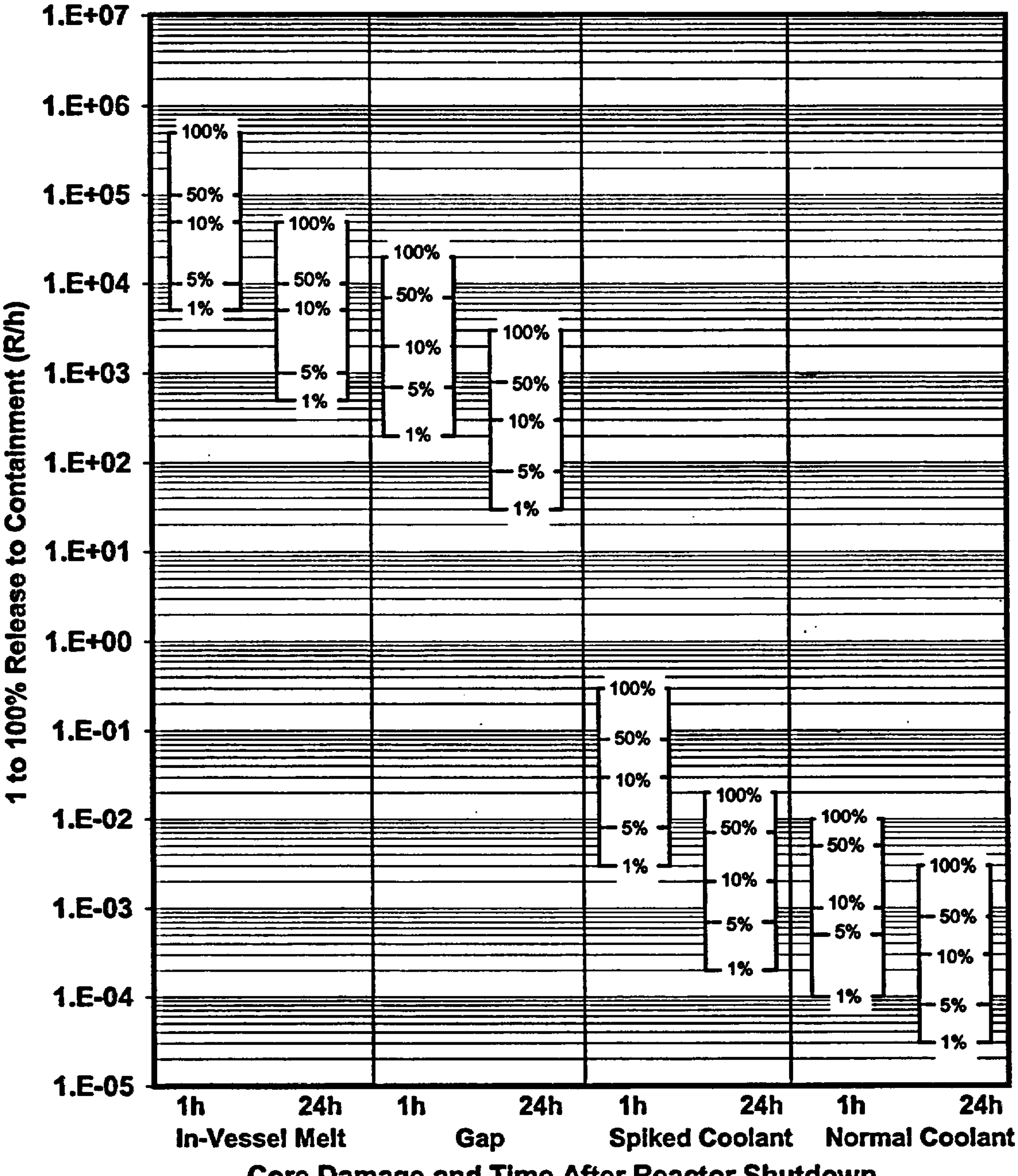


Note 1: This figure should be used only when there is a primary system breach inside containment and a direct release path to the Drywell.

Note 2: See Attachment 3 to determine if fuel melt occurred (core uncovered or fuel blockage).

ATTACHMENT 1A

Containment Radiation Monitor Response
Direct Release Path to Dry well
(Sprays On)

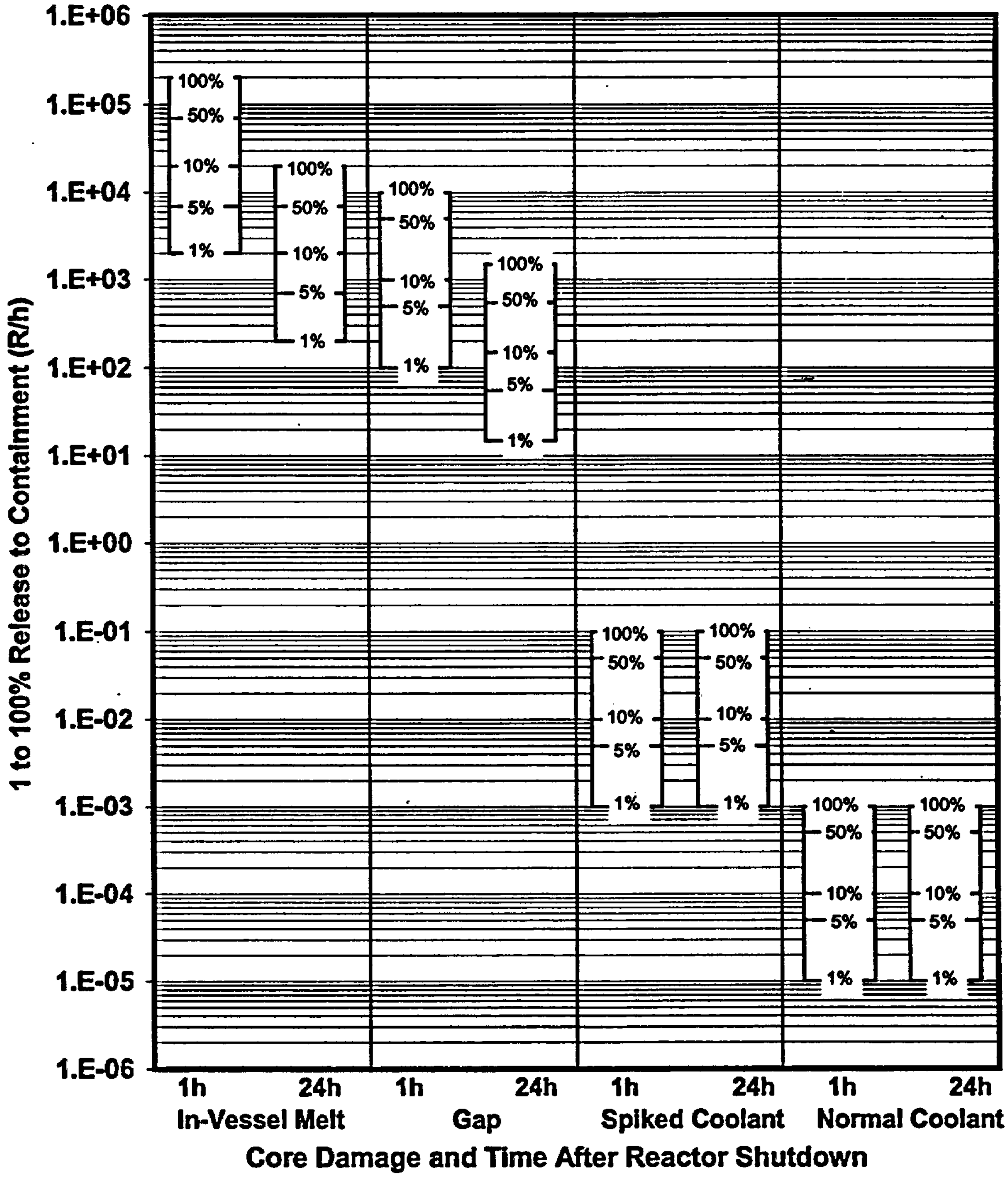


Note 1: This figure should be used only when there is a primary system breach inside containment and a direct release path to the Drywell.

Note 2: See Attachment 3 to determine if fuel melt occurred (core uncovered or fuel blockage).

ATTACHMENT 1B

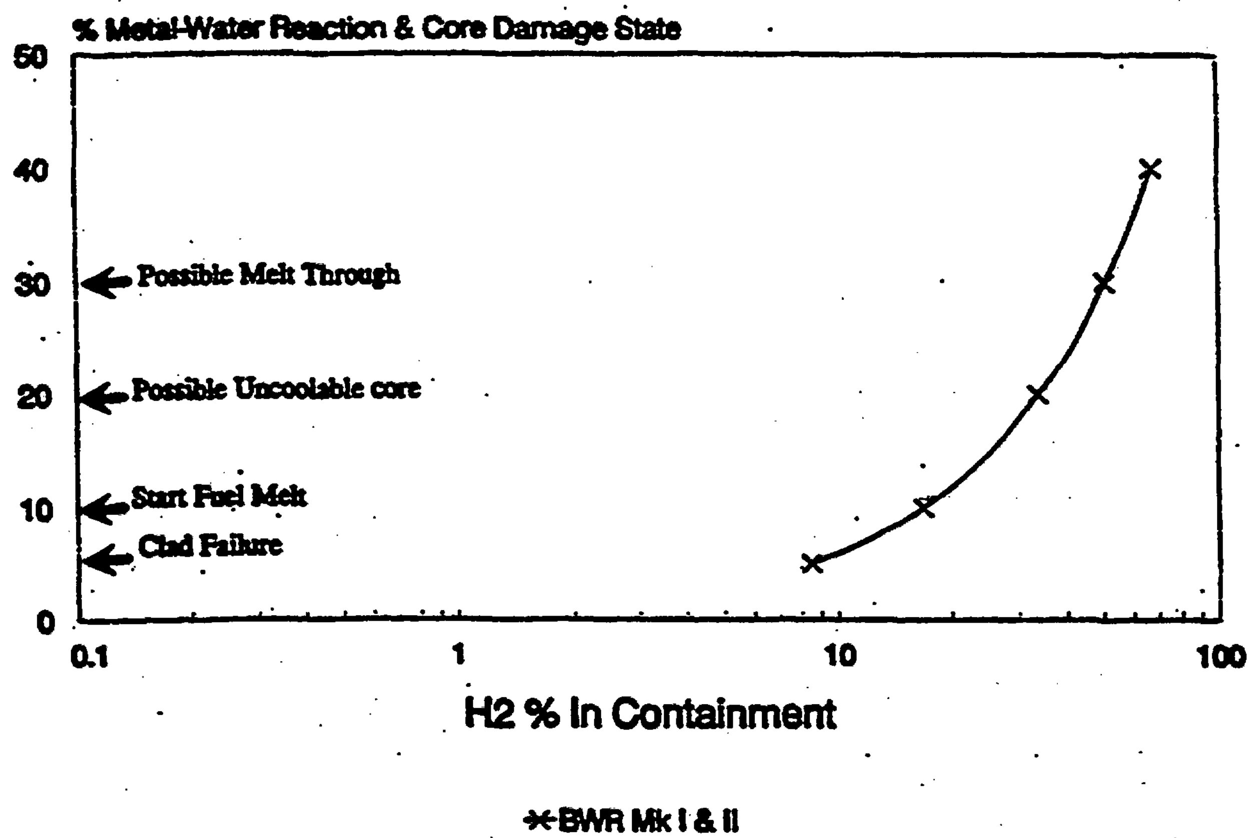
Containment Radiation Monitor Response
<u>Direct Release to Wetwell and Not to Drywell</u>



Note 1: This figure should be used only when there is a primary system breach inside containment and a direct release path to the Wetwell without a primary release to the Drywell.
Note 2: See Attachment 3 to determine if fuel melt occurred (core uncovered or fuel blockage).

<u>ATTACHMENT 1C</u>

CONTAINMENT HYDROGEN VS CORE DAMAGE



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

ATTACHMENT 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 by boiling. 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.

CAUTION:

If the core has been uncovered, the fuel temperature will have increased significantly. Additional flow will be required to accommodate the heat transfer necessary to return to equilibrium fuel temperature.

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.

ATTACHMENT 3
(Page 1 of 4)

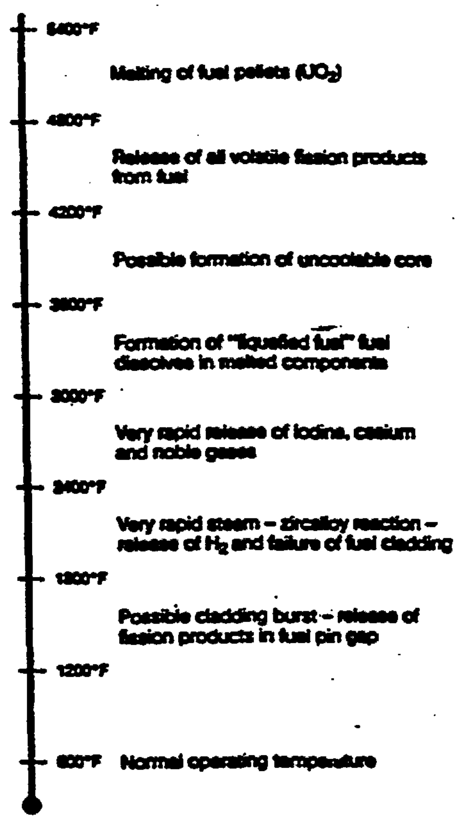
WATER INJECTION REQUIRED TO COOL CORE BY BOILING

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

ATTACHMENT 3
(Page 2 of 4)

CAUTION: **If the 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.

WATER INJECTION REQUIRED TO COOL CORE BY BOILING

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)

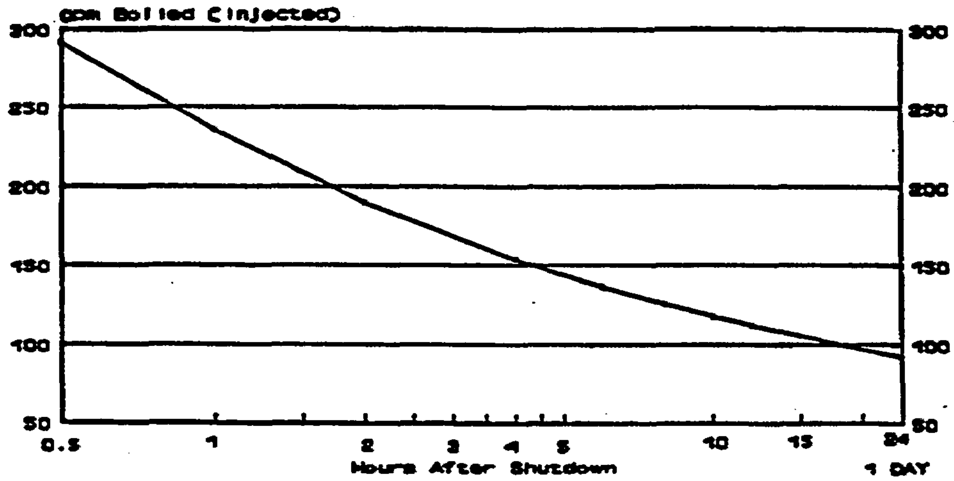
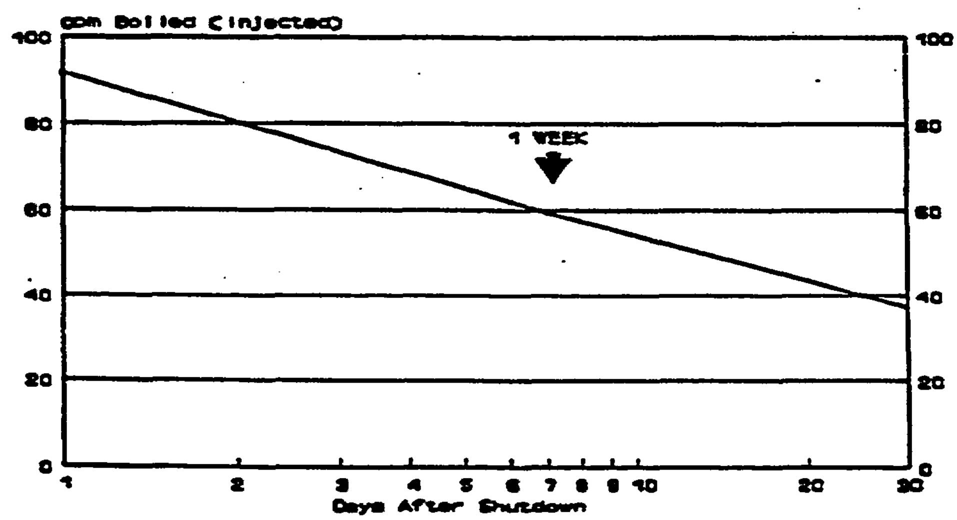


Figure B-4

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)



ATTACHMENT 3
(Page 3 of 4)

WATER INJECTION REQUIRED TO COOL CORE BY BOILING

Core damage vs. time that reactor core is uncovered

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

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

ATTACHMENT 3
(Page 4 of 4)

CORE DAMAGE ESTIMATE II

(Small or no primary system breach inside Containment)

This instruction provides a method of estimating the percentage of fuel that has failed using the Containment Post-Accident Radiation Monitor (CPARM) readings on panel 1C601 (2C601) during an accident. Since the Containment Post-Accident Radiation Monitor readings are readily available, this calculation provides a quick assessment of core damage. This estimate only applies if there is a small or no primary system breach within containment.

1.0 LIMITATIONS OF THE METHOD

1.1 This procedure will only determine qualitatively the amount of fuel damage. The method uses Containment Post-Accident Radiation Monitor Readings to calculate the percentage of failed fuel during an accident where the fission products are released from the fuel rod cladding. The methodology is based on assumptions with large uncertainties that can significantly affect the results.

1.2 To use this method, the accident scenario up to the time of the Containment Post-Accident Radiation Monitor Reading must be well understood to estimate the fuel temperatures required by this procedure.

1.3 In addition, a Containment Post-Accident Radiation Monitor Reading and the time the reading was obtained must be available.

2.0 RESPONSIBILITIES

2.1 The Nuclear Fuels Engineer, Lead Technical Support Engineer, or designee collects information and makes estimates and determinations described in this procedure.

3.0 INSTRUCTIONS

3.1 Determine if Cladding Failure, Fuel Overheat, or Fuel Melt has occurred:

3.1.1 Cladding Failure is expected if peak cladding temperature remains less that 2200°F, but the Containment Post-Accident Radiation Monitor readings have increased.

3.1.2 Fuel Overheat is expected if peak cladding temperature exceeds 2200°F, but the maximum volume-averaged fuel pellet temperature remains less that 4500°F.

3.1.3 Fuel Melt is expected if any volume-averaged fuel pellet temperature exceeds 4500°F.

3.2 Since the fuel melt temperatures are dependent on the event progression, specific guidelines cannot be given to cover all scenarios. Some judgment will have to be made or specific temperature calculations will have to be performed during the event. However, the following provides guidelines for a few known scenarios.

3.2.1 If a main steamline high radiation trip causes the scram and the core remains covered, usually cladding failure can be assumed and is possibly due to debris fretting, short term DNB, or PCI. However, if channel flow blockage is suspected, overheat or melting may occur.

3.2.2 For loss-of-inventory-after-the-reactor-is-shutdown scenarios, use Attachment 3 to Tab 4 to estimate if Fuel Melt has occurred.

3.3 Determine the Time After Reactor Shutdown that a Containment Post-Accident Radiation Monitor Reading was obtained.

3.4 Determine if the event has resulted in a primary system breach inside primary containment (increase in drywell pressure/temperature and inventory makeup to the vessel is required to maintain level in the vessel). If the total primary system water released to the drywell is equivalent to less than 9,000 gallons or no primary system breach has occurred inside primary containment, use Figure 1. Otherwise, use Core Damage Estimate I (Tab 4).

Note: The 9,000 gallon value is about 10% of the fluid volume of the reactor vessel and primary piping (main steam, reactor recirculation, and feedwater).

3.5 Determine Fraction of Fuel Failed (FFF) as follows:

$$FFF = \frac{CPARM\ Reading}{Expected\ 100\%\ Fuel\ Failure\ CPARM\ Reading} \times 100$$

FIGURE 1
CONTAINMENT HIGH RANGE RADIATION MONITOR READINGS THAT ARE EXPECTED WITH 100% OF THE FUEL FAILED FOR AN EVENT WITH NO PRIMARY SYSTEM BREACH INSIDE CONTAINMENT