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3	RES DEPARTMENT MANAGER	RES (UNIT 3/IPEC ONLY)	45-4-A
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33	LONGO N (PLAN ONLY)	EMERGENCY SERVICES	ROCKLAND
34	GREENE D (PLAN ONLY)	DISASTER & CIVIL DEFENSE	ORANGE
35	RAMPOLLA M (PLAN ONLY)	OFFICE OF EMERG MANAGE	PUTNAM
41	SIMULATOR	TRAIN(UNIT 3/IPEC ONLY)	48-2-A
107	QA MANAGER	QA (UNIT 3/IPEC)	K-IP2-4302
319	L.GRANT (LRQ-OPS TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
354	L.GRANT(LRQ-OPS/TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
424	HULBERT TRACY(7COPIES)	(UNIT 3/IPEC ONLY)	#48
510	L.GRANT(LRQ-OPS/TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
511	L.GRANT(LRQ-OPS/TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
512	L.GRANT (LRQ-OPS TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
513	L.GRANT (LRQ-OPS TRAIN)	LRQ (UNIT 3/IPEC ONLY)	#48
517	PLANT MANAGER'S OFFICE	ADMIN/(UNIT 2/IPEC ONLY)	IP2
518	LAZAZZARO MIKE	UNIT 2(UNIT 2/IPEC ONLY)	IP2
520	CONTROL ROOM(D001 SERIES)	OPS (UNIT 2 & IPEC ONLY)	IP2
521	SIMULATOR	TRAIN (UNIT 2/IPEC ONLY)	IP2
522	NRC RESIDENT	US NRC(UNIT 2/IPEC ONLY)	88'ELV IP2
523	DIGIOVANNI TINA (UNIT 2)	REFERENCE LIBRARY/TRAIN	48-2-ALLE
524	BLAIR W.	IPEC LICENSING	K-IP2-4323
558	TORRES DAMARIS	R&D EEC BUILDING 2ND FL.	IP2

A045

ATTACHMENT 10.1 SMM CONTROLLED DOCUMENT TRANSMITTAL FORM

SITE MANAGEMENT MANUAL CONTROLLED DOCUMENT TRANSMITTAL FORM - PROCEDURES

Page 1 of 1

Entergy	CONTROLLED DOCUMENT TRANSMITTAL FORM - PROCEDURES

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********** FOLLOW ATTACHED INSTRUCTIONS **********

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Document #	Document Name Emergency Plan Implementing Procedure	New Rev. #/ Date	Old Rev. #/ Date	Instructions
TOC	Emergency Plan Implementing Procedures	12/18/03	12/10/03	Remove and Replace
IP-EP-360	Core Damage Assessment	Rev.0 12/18/03		Add New Procedure
IP-EP-310	Dose Assessment	Rev.3 12/10/03		Remove and Replace pages 9 - 18
	Unit 3 procedure holders			
IP-1028	Core Damage Assessment	VOID	Rev. 9 06/98	Remove from Binder
IP-1070	Inventory	VOID	Rev. 31 2/01	Remove from Binder

Indian Point Energy Center
Emergency Plan Implementing Procedures
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Indian Point Energy Center
Emergency Plan Implementing Procedures
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IP-1028 REV.9(VOID DATE 18-DEC-2003)

IS VOID

REASON FOR VOID: REPLACED BY IP-EP-360

Entergy

IP-1070 REV.31 (VOID DATE 30-DEC-2003)

IS VOID

REASON FOR VOID: REPLACED BY IP-EP-AD6

CORE DAMAGE ASSESSMENT

CONTROLLED

COPY # 25

Prepared by:	C. Kelly Walker Print Name	C. Kelly Signature	4/9/03 Date
Approval:	Frank Inzirillo Print Name	M.Z. Mule Signature for F. Inzirillo	12/16/03 Date

Effective Date: 12-18-03

Table of Contents

CORE DAMAGE ASSESSENT

1.0 PURPOSE

This guideline provides a methodology for the assessment of:

- The degree of damage to the fuel rod cladding that results in the release of the fission product inventory in the fuel rod gap space.
- The degree of core overheating that results in the release of the fission product inventory in the fuel pellets.
- The appropriate Emergency Action Level for off-site radiological protective actions based on the degree of damage to the reactor core.

This guideline should be used when the reactor is shutdown and either:

- Core temperatures are at or above 700°F, or
- Containment radiation level is at or above 1 R/hr

2.0 REFERENCES

2.1 WCAP-14696-A, Westinghouse Owners Group Core Damage Assessment Guideline, Rev. 1

2.2 "Containment Radiation Level Using Core Damage Assessment Guideline, Revision 1 (1996) For Specific Indian Point Unit 2 EAL Application: A Summary," by Dave Smith, 12/2000.

2.3 PGI-00467-00, 4/5/01 "Containment Radiation Monitor Response/Core Damage Assessment Procedure Support"

2.4 IP-CA-3, Hydrogen Flammability in Containment, Pg 2, Rev. 0

3.0 DEFINITIONS

None

4.0 RESPONSIBILITIES

4.1 Upon recognition of **EITHER** core exit thermocouple temperature(s) **> 700 °F OR** containment radiation levels **> 1 R/hr,** the Core Physics Engineer (Reactor Engineer) shall implement this procedure to assess the existence and extent of core damage.

4.2 The Core Physics Engineer (Reactor Engineer) shall immediately inform the Technical Assessment Coordinator /TSC Manager of the results of any core damage assessment performed.

5.0 DETAILS

NOTE:
Core Damage Estimate may be base on historical monitor readings. For Example: If core thermocouple readings were high 4 hours into an event but are now off-scale or inoperable use values and time after shutdown for when readings were valid.

5.1 Determine the possible status of the reactor core using the following flowchart and perform the associated action.

High Level Core Damage Assessment Flowchart

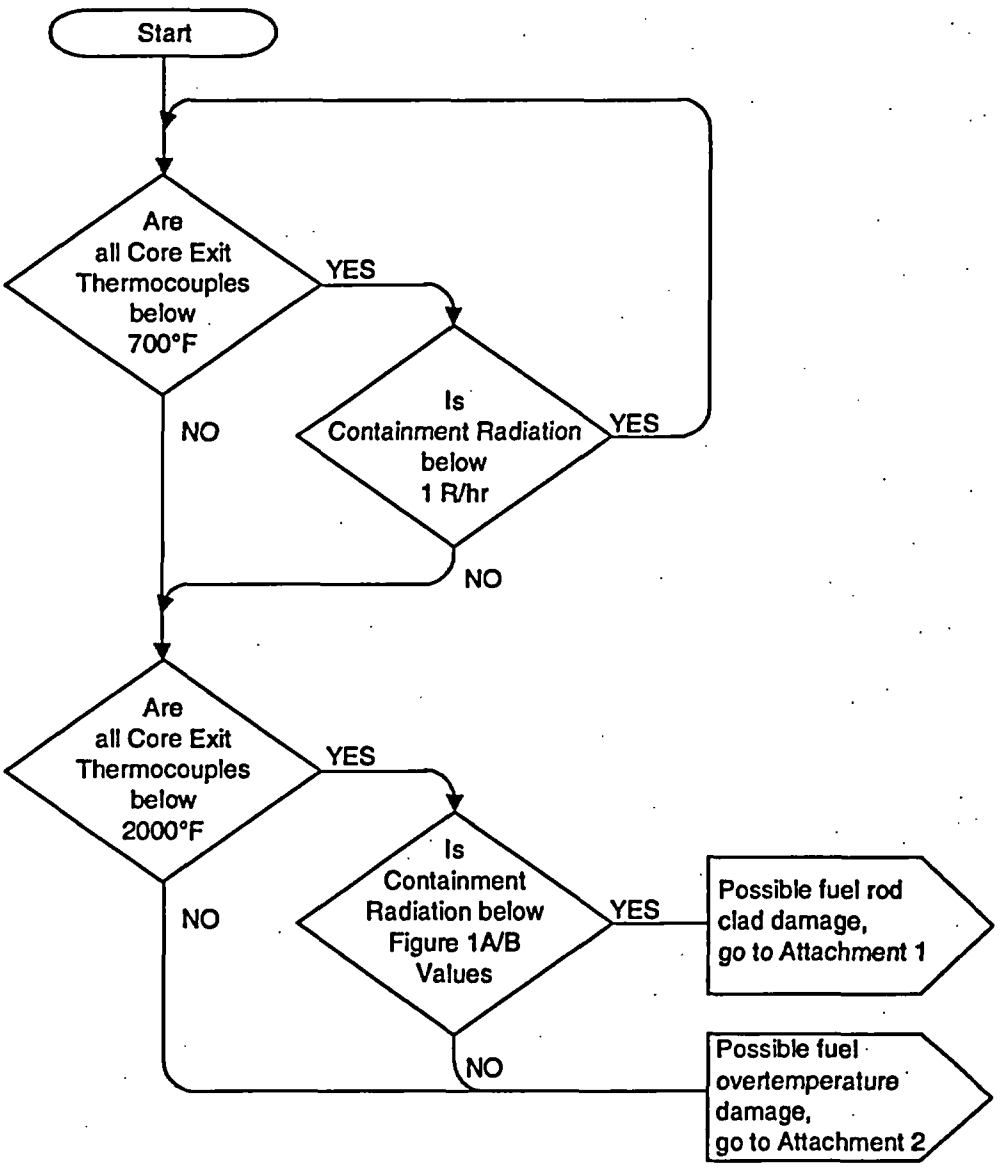


Figure 1A
Containment Radiation Level for 1% Fuel Overtemperature Flowchart
(0 to 6 hours after shutdown)

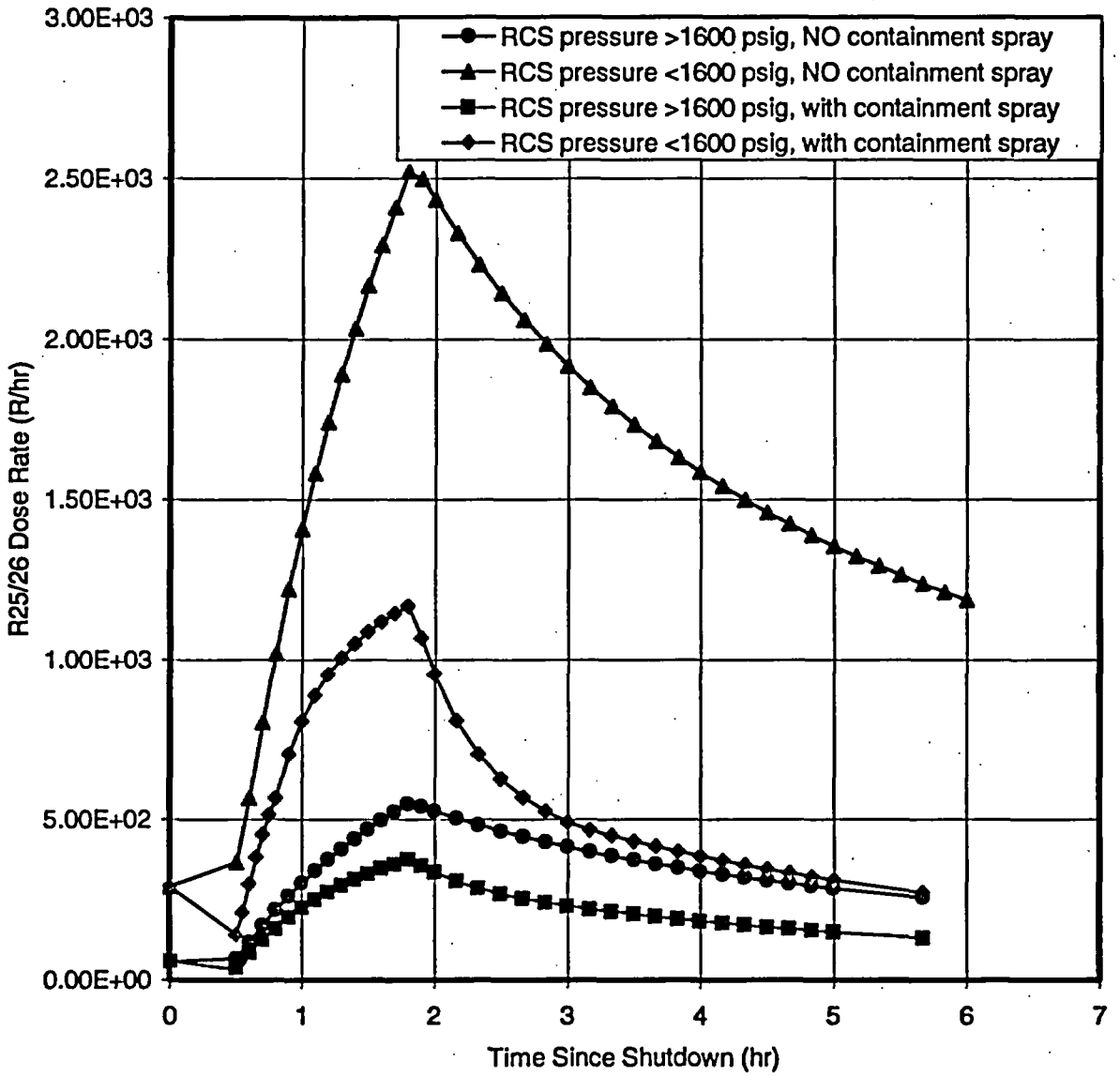
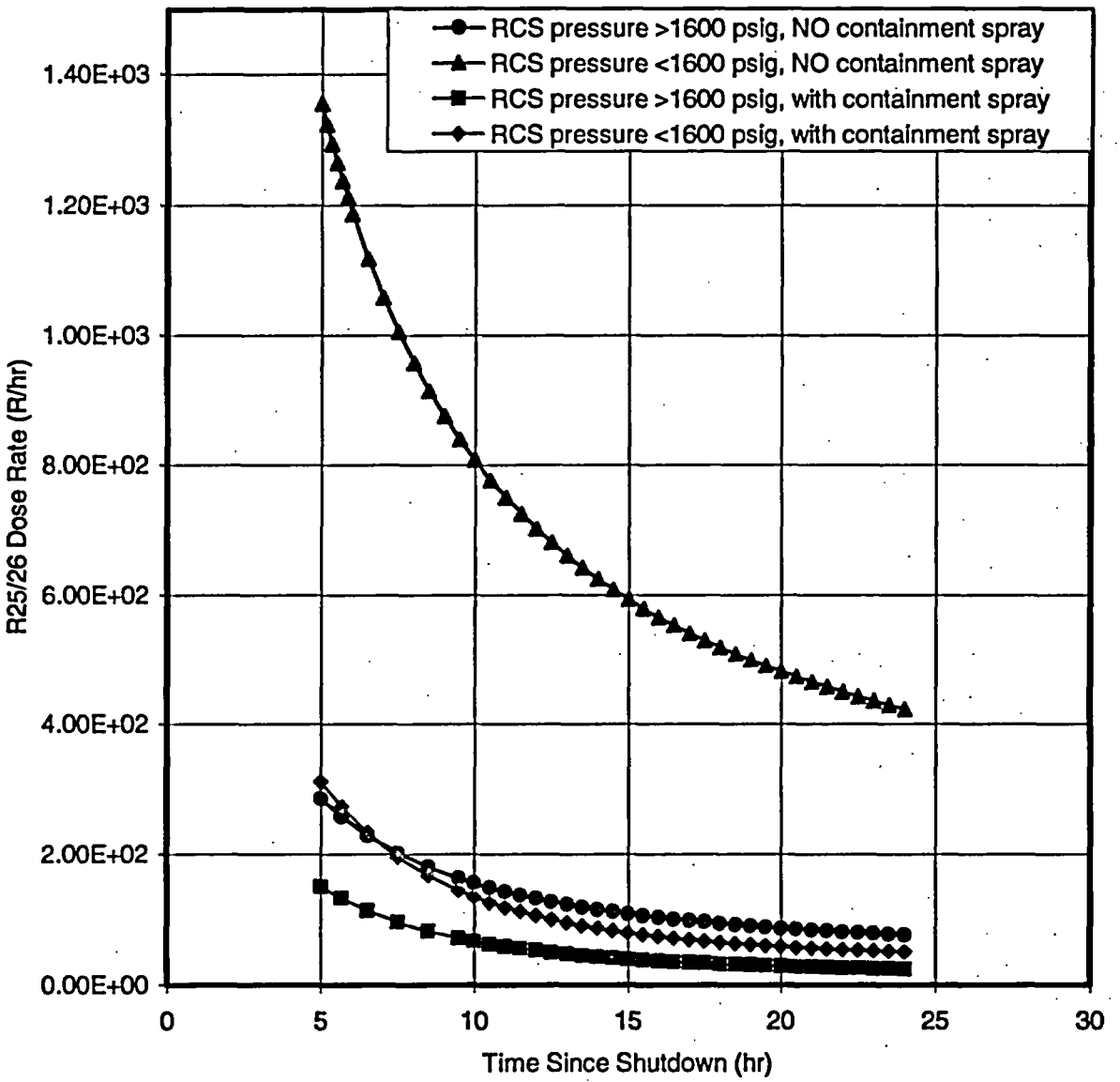


Figure 1B
Containment Radiation Level for 1% Fuel Overtemperature Release
(>5 hours after shutdown)



6.0 INTERFACES

6.1 IP-EP-120, Emergency Classification

6.2 IP-EP-222, Unit 2 Technical Support Center

6.3 IP-EP-223, Unit 3 Technical Support Center

7.0 RECORDS

This procedure generates completed Fuel Rod Clad Damage (Attachment 1) and/or Fuel Overtemperature Damage (Attachment 2) worksheets.

8.0 REQUIREMENTS AND COMMITMENTS

None

9.0 ATTACHMENTS

9.1 Attachment 1, Fuel Rod Clad Damage

9.2 Attachment 2, Fuel Overtemperature Damage

Attachment 1
Fuel Rod Clad Damage
Sheet 1 of 5

1. Estimate fuel rod clad damage based on containment radiation (CRM) levels.

   1.1 Determine the following:

   - Time since shutdown (hr) _____
   - RCS pressure (psig) _____
   - Containment sprays operating (yes/no) _____

   1.2 Find the following containment radiation dose rates:

   - Containment radiation level (R/hr) for 100% clad damage (Figure 2A/B) A = _____
   - Current containment radiation level (R/hr) B = _____

   1.3 Estimate clad damage (%):

$$\%\ \text{Clad Damage}_{CRM} = \frac{B \times 100}{A} = \_\_\_\_\_$$

2. Estimate fuel rod clad damage based on Core Exit Thermocouples (CETs).

   2.1 Determine the following:

   - Total number of operable CETs. (Refer to PICS [Unit 2] or SPDS [Unit 3]) D = _____
   - Number of CETs at or above 1400°F E = _____
   - Number of CETs at or above1200°F F = _____

   2.2 For RCS pressure at or above 1600 psig:

$$\%\ \text{Clad Damage}_{CET} = \frac{E \times 100}{D} = \_\_\_\_\_$$

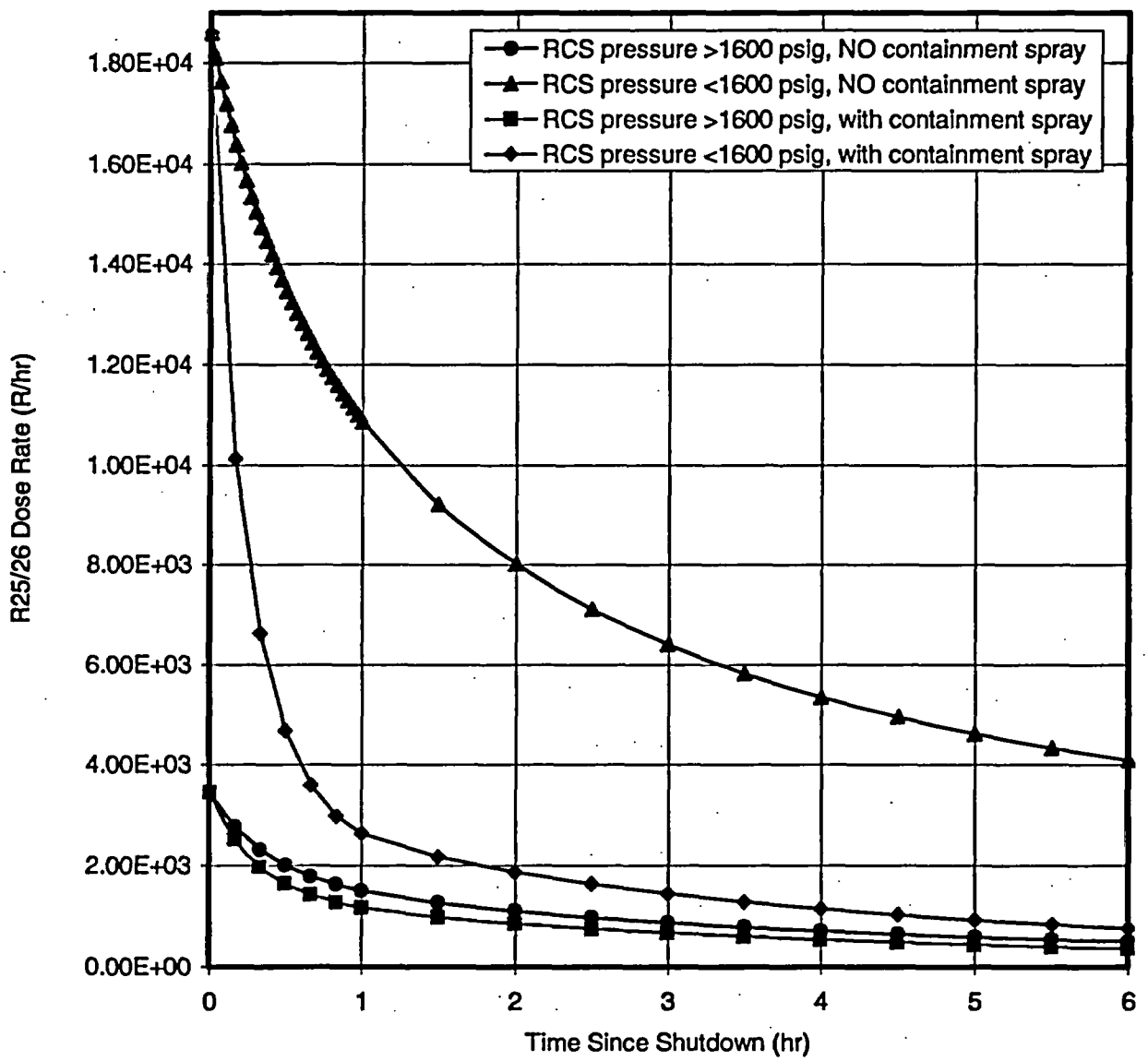
   2.3 For RCS pressure below 1600 psig::

$$\%\ \text{Clad Damage}_{CET} = \frac{F \times 100}{D} = \_\_\_\_\_$$

Attachment 1
Fuel Rod Clad Damage
Sheet 2 of 5

Figure 2A
Containment Radiation Level for 100% Clad Damage Release

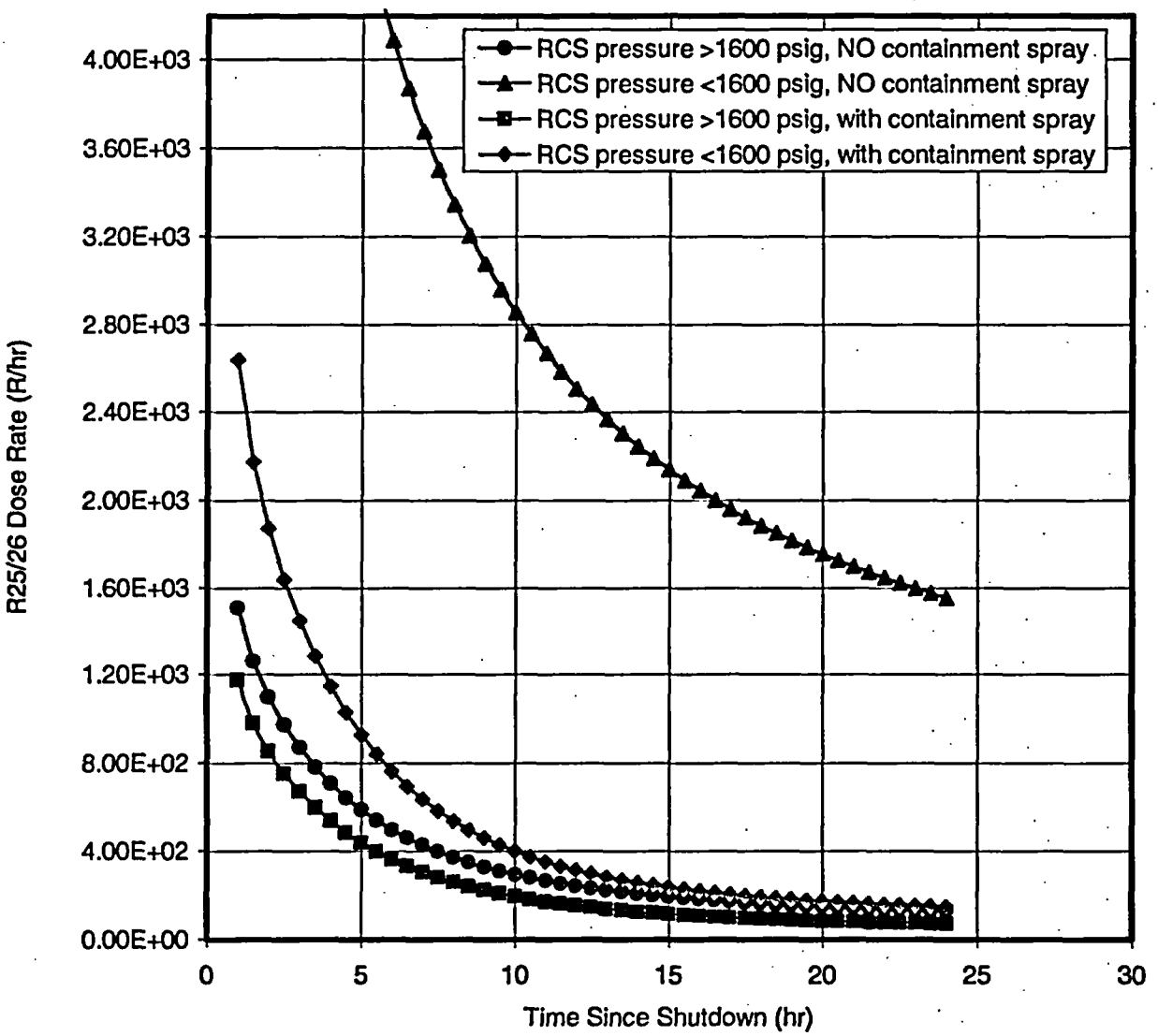
(0 to 6 hours after shutdown)



Attachment 1
Fuel Rod Clad Damage
Sheet 3 of 5

Figure 2B
Containment Radiation Level for 100% Clad Damage Release

(> 1 hour after shutdown)



Attachment 1
Fuel Rod Clad Damage
Sheet 4 of 5

3. Confirm reasonableness of clad damage estimates.

3.1 Determine the following:

- Containment hydrogen concentration (vol. %) _____
- RVILS reading (%) _____
- RCS saturation temperature (°F) _____
- Hot leg RTD temperature (°F) _____

3.2 Compare estimated clad damage to expected response by answering the following questions (yes/no)

- Is containment hydrogen concentration less than 0.5%? _____
- Is RVLIS between 64% and 47%? _____
- Is hot leg RTD between T_{sat} and 650°F? _____
- Is the absolute difference (% Diff) between estimated containment radiation clad damage and estimated core exit thermocouple clad damage less than 50%? _____

$$\%\ Diff_{diff} = \frac{|\%\ Clad\ Damage_{CRM} - \%\ Clad\ damage_{CET}|}{\%\ Clad\ Damage_{CRM}} \times 100$$

3.3 If all of the answers to the questions in Step 3.2 are YES, the expected response has been obtained; continue at Step 4.

3.4 If any answer to the questions in Step 3.2 is NO, the expected response has not been obtained; determine if the deviation can be explained from either:

3.4.1 Accident progression:

- Injection of water to the RCS
- Bleed paths from the RCS
- Direct radiation to the containment radiation monitors

Attachment 1
Fuel Rod Clad Damage
Sheet 5 of 5

3.4.2 Conservatisms in the predictive model:

- Fuel burnup
- Fission product retention in the RCS
- Fission product removal from containment

4. Report findings

4.1 If clad damage estimates have increased by more than 1% in the past 30 minutes

OR

Estimates exceed 2% clad damage

Then report possible impact on emergency classification based upon Emergency Action Level thresholds to the Emergency Plant Manager/Plant Operations Manger.

4.2 Report clad damage estimate to the Technical Assessment Coordinator/TSC Manager.

5. Return to Step 5.1 of the procedure to continue assessment of the reactor core.

Attachment 2
Fuel Overtemperature Damage
Sheet 1 of 7

1. Estimate Fuel Overtemperature Damage Based on Containment Radiation (CRM) Levels.

   1.1 Determine the following:

   - Time since shutdown (hr) _____
   - RCS pressure (psig) _____
   - Containment sprays operating (yes/no) _____

   1.2 Find the following containment radiation dose rates:

   - Containment radiation level (R/hr) for 100% core overtemperature damage (Figure 3A/B) G = _____
   - Current containment radiation level (R/hr) H = _____

   1.3 Estimate fuel overtemperature damage (%):

$$\%\ \text{Core Damage}_{CRM} = \frac{H \times 100}{G} = \_\_\_\_\_$$

2. Estimate fuel overtemperature damage based on Core Exit Thermocouple (CETs).

   2.1 Determine the following:

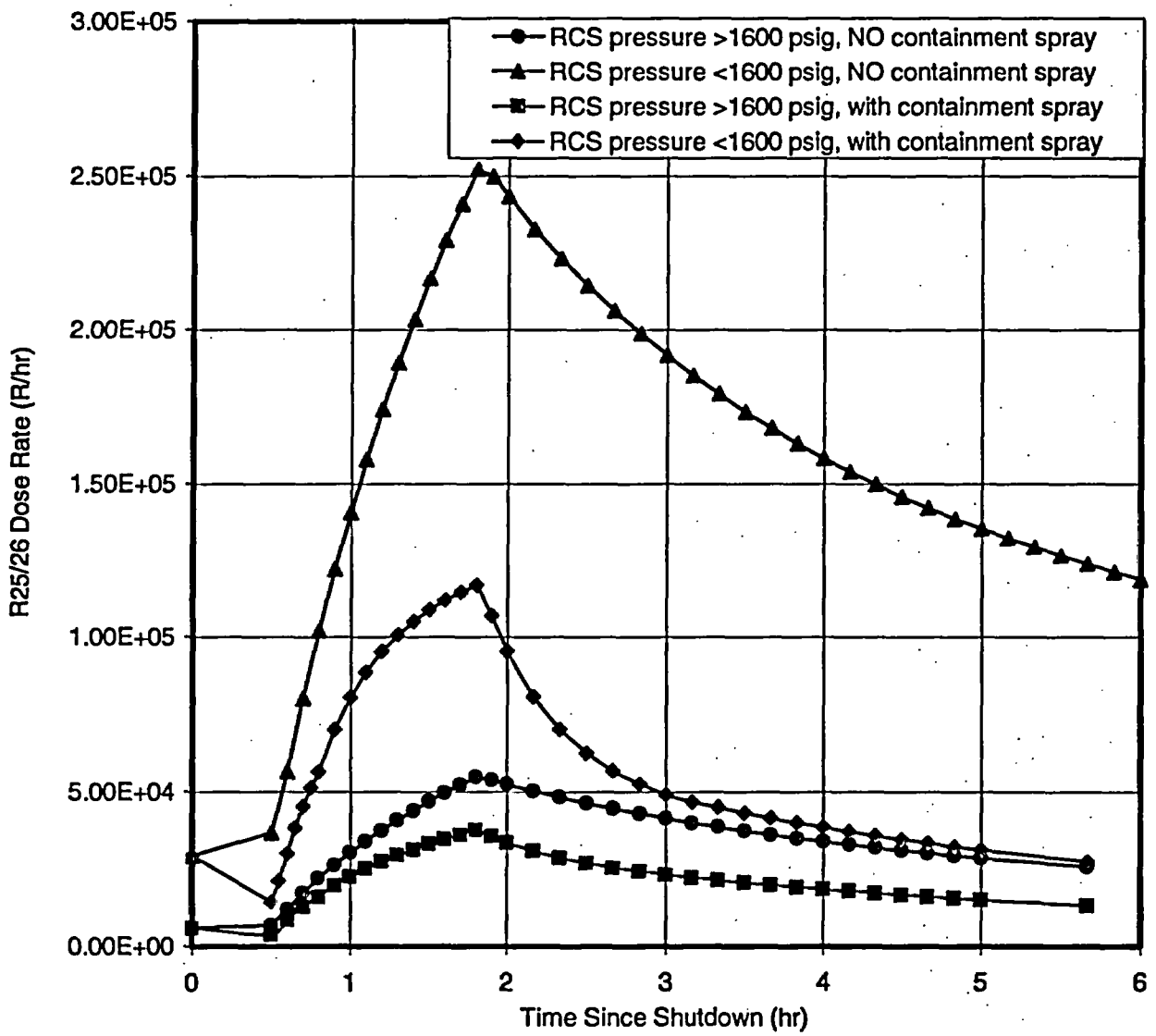
   - Total number of operable CETs. (Refer to PICS [Unit 2] or SPDS [Unit 3]) J = _____
   - Number of CETs at or above 2000°F K = _____

   2.2 Estimate fuel overtemperature damage (%):

$$\%\ \text{Core Damage}_{CET} = \frac{K \times 100}{J} = \_\_\_\_\_$$

Attachment 2
Fuel Overtemperature Damage
Sheet 2 of 7

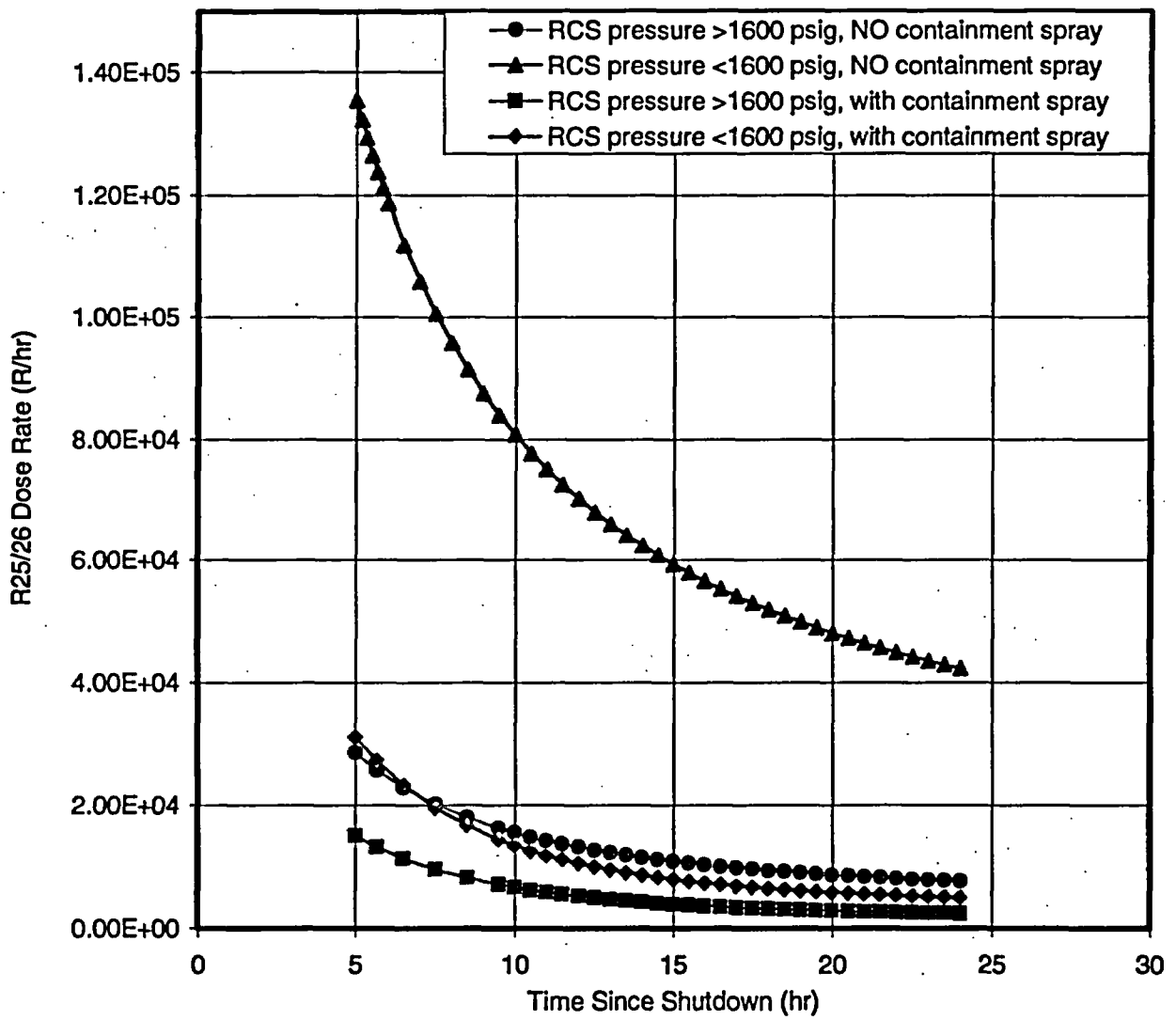
Figure 3A
Containment Radiation Level for 100% Fuel Overtemperature Release
(0 to 6 hours after shutdown)



Attachment 2
Fuel Overtemperature Damage
Sheet 3 of 7

Figure 3B
Containment Radiation Level for 100% Fuel Overtemperature Release

(>5 hours after shutdown)



Attachment 2
Fuel Overtemperature Damage
Sheet 4 of 7

3. Estimate fuel overtemperature damage based on containment hydrogen (Hyd) concentration.

   3.1 Determine the following:

   - RCS pressure (psig) _____
   - Current containment hydrogen concentration (vol. %) L = _____
   - Predicted containment hydrogen concentration at 100% core overtemperature, Table 2 (vol. %) M = _____

Table 2 – Core Overtemperature Estimate Based on Containment Hydrogen Concentration

RCS Pressure (psig)	Water Injection into RCS?	Predicted Containment Hydrogen Concentration from Figure 4 (vol. %)
Below 1050	Yes	CH2
	No	CH3
At or above1050	Yes	CH4
	No	CH3

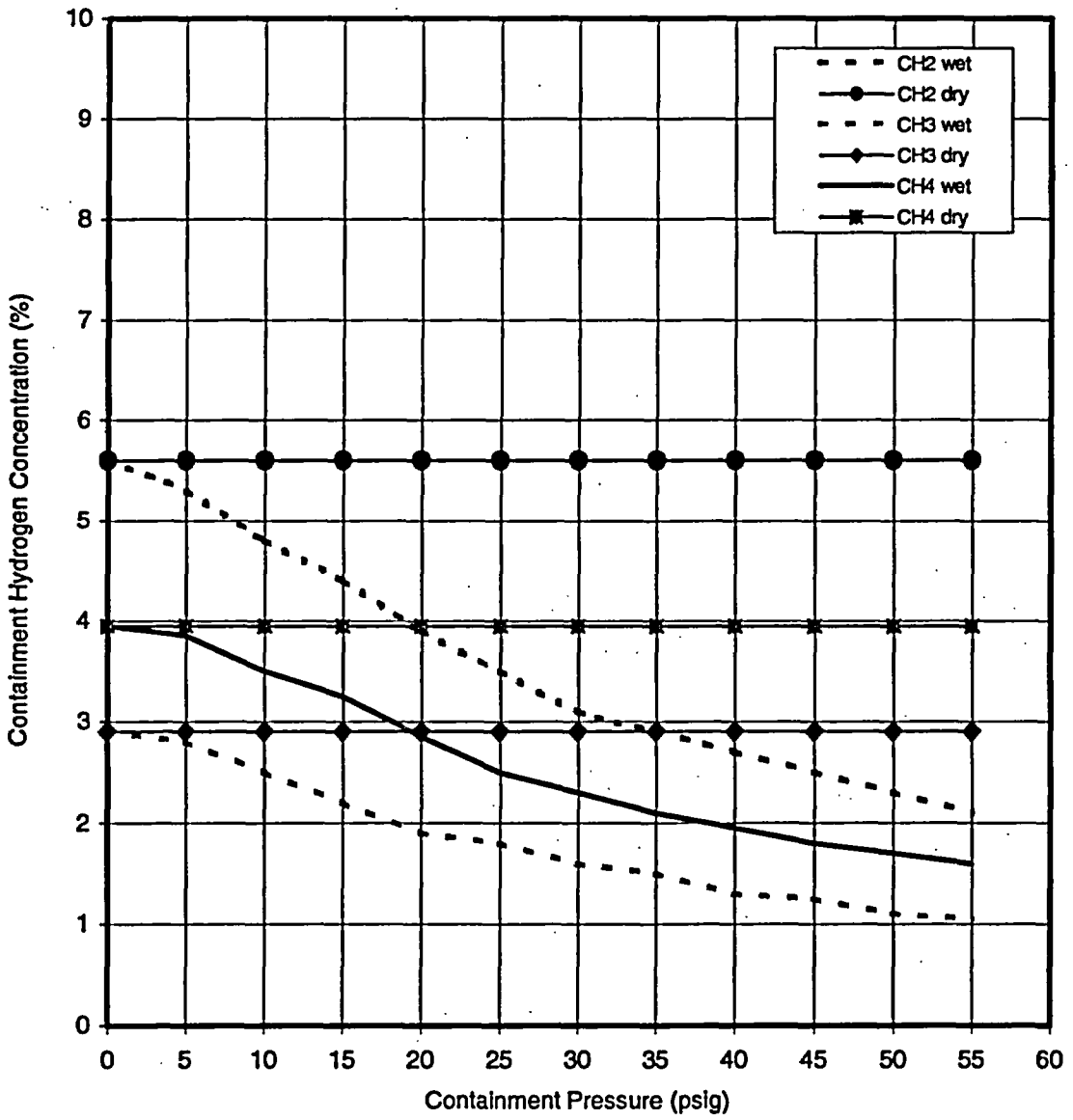
   3.2 Estimate fuel overtemperature damage (%):

$$\%\ \text{Core Damage}_{Hyd} = \frac{L \times 100}{M} =$$ _____

Attachment 2
Fuel Overtemperature Damage
Sheet 5 of 7

Figure 4
Predicted Containment Hydrogen Concentration
for 100% Fuel Overtemperature

Note: The wet hydrogen curves are used when superheated conditions inside containment exist or when a manual sample is used.



Attachment 2
Fuel Overtemperature Damage
Sheet 6 of 7

4. Confirm reasonableness of fuel overtemperature damage estimates.

   4.1 Determine the following:

   - RVILS reading (%) _____
   - Hot leg RTD temperature (°F) _____

   4.2 Compare estimated core damage to expected response by answering the following questions (yes/no)

   - Is RVLIS below 47%? _____
   - Is hot leg RTD at or above 650°F? _____
   - Is the absolute difference (% Diff) between estimated containment radiation core damage and estimated core exit thermocouple core damage less than 50%? _____

$$\%\ Diff_{diff} = \frac{|\%\ Core\ Damage_{CRM} - \%\ Core\ damage_{CET}|}{\%\ Core\ Damage_{CRM}} \times 100$$

   - Is the absolute difference (% Diff) between estimated containment hydrogen core damage and estimated radiation core damage less than 25%? _____

$$\%\ Diff_{diff} = \frac{|\%\ Core\ Damage_{Hyd} - \%\ Core\ damage_{CRM}|}{\%\ Core\ Damage_{Hyd}} \times 100$$

   - Is the absolute difference (% Diff) between estimated containment hydrogen core damage and estimated core exit thermocouple core damage less than 25%? _____

$$\%\ Diff_{diff} = \frac{|\%\ Core\ Damage_{Hyd} - \%\ Core\ damage_{CET}|}{\%\ Core\ Damage_{Hyd}} \times 100$$

Attachment 2
Fuel Overtemperature Damage
Sheet 7 of 7

4.3 If all of the answers to the questions in Step 4.2 are YES, the expected response has been obtained; continue at Step 6.

4.4 If any answer to the questions in Step 4.2 is NO, the expected response has not been obtained; determine if the deviation can be explained from either:

4.4.1 Accident progression:

- Injection of water to the RCS
- Bleed paths from the RCS
- Direct radiation to the containment radiation monitors
- Hydrogen burn in containment or affects of passive autocatalytic hydrogen recombination (Unit 2)

4.4.2 Conservatisms in the predictive model:

- Fuel burnup
- Fission product retention in the RCS
- Fission product removal from containment

5. Report fuel overtemperature estimate to the Technical Assessment Coordinator/TSC Manager.

6. Return to Step 5.1 of the procedure to continue assessment of the reactor core.

Attachment 9.1

Site Boundary Xμ/Q by Pasquill Stability Category

Cross Valley (Wind Direction from 210° – 339° or Wind Speed ≥ 4 m/s)

Sheet 1 of 2

Sector	Wind From	Distance (Meters)	Pasquill Categories						
			A	B	C	D	E	F	G
1*	169° to 190°	2977	5.5 E-7	9.0 E-7	5.7 E-6	2.1 E-5	4.3 E-5	1.1 E-4	2.0 E-4
2*	191° to 213°	3234	5.2 E-7	1.0 E-6	5.0 E-6	1.9 E-5	3.9 E-5	9.6 E-5	1.8 E-4
3	214° to 235°	716	3.6 E-6	2.0 E-5	5.3 E-5	1.5 E-4	2.7 E-4	4.9 E-4	7.1 E-4
4	236° to 258°	701	3.7 E-6	2.0 E-5	5.4 E-5	1.6 E-4	2.7 E-4	5.0 E-4	7.2 E-4
5	259° to 280°	762	3.2 E-6	1.8 E-5	4.8 E-5	1.4 E-4	2.5 E-4	4.7 E-4	6.8 E-4
6	281° to 303°	625	4.7 E-6	2.5 E-5	6.4 E-5	1.8 E-4	3.1 E-4	5.5 E-4	7.9 E-4
7	304° to 325°	610	4.9 E-6	2.6 E-5	6.6 E-5	1.9 E-4	3.2 E-4	5.6 E-4	8.0 E-4
8	326° to 348°	701	3.7 E-6	2.0 E-5	5.4 E-5	1.6 E-4	2.7 E-4	5.0 E-4	7.2 E-5
9	349° to 10°	1006	2.1 E-6	1.0 E-5	3.2 E-5	9.9 E-5	1.8 E-4	3.6 E-4	5.4 E-4
10	11° to 33°	1006	2.1 E-6	1.0 E-5	3.2 E-5	9.9 E-5	1.8 E-4	3.6 E-4	5.4 E-4
11	34° to 55°	488	7.7 E-6	3.6 E-5	8.8 E-5	2.5 E-4	4.0 E-4	6.7 E-4	9.2 E-4
12*	56° to 78°	2349	6.6 E-7	1.5 E-6	8.3 E-6	3.0 E-5	6.0 E-5	1.4 E-4	2.6 E-4
13*	79° to 100°	1802	8.1 E-7	3.2 E-6	1.3 E-5	4.3 E-5	8.5 E-5	1.9 E-4	3.3 E-4
14*	101° to 123°	1689	9.0 E-7	3.7 E-6	1.4 E-5	4.8 E-5	9.2 E-5	2.0 E-4	3.5 E-4
15*	124° to 145°	1432	1.2 E-6	5.1 E-6	1.9 E-5	6.1 E-5	1.2 E-4	2.4 E-4	4.0 E-4
16*	146° to 168°	1416	1.2 E-6	5.2 E-6	1.9 E-5	6.2 E-5	1.2 E-4	2.5 E-4	4.0 E-4

* Plume for these sectors goes over the water before it touches public or private land. Site boundary in these cases is taken to be the landfall point at the sector center.

Attachment 9.1
Sheet 2 of 2

Site Boundary Xμ/Q by Pasquill Stability Category

Up Valley Plumes (wind speed <4 m/s) Wind Direction from 102° – 209°(1)

Pasquill Categories						
A	B	C	D	E	F	G
5.2 E-7	1.0 E-6	5.0 E-6	1.9 E-5	3.9 E-5	9.6 E-5	1.8 E-4

Site Boundary Xμ/Q by Pasquill Stability Category

Down Valley Plumes (wind speed <4 m/s) Wind Direction from 340° – 101°(2)

Pasquill Categories						
A	B	C	D	E	F	G
3.7 E-6	1.0 E-5	3.2 E-5	9.9 E-5	1.8 E-4	3.6 E-4	5.4 E-4

(1) Plume centerline will always cross the site boundary at sector 2. Therefore, the sector 2 Xμ/Q values are used.

(2) Plume centerline will cross the site boundary at either sector 8 (Pasquill category A) or sector 10 (for Pasquill category B – G)

Attachment 9.2

Xμ/Q Values for other Distances

Sheet 1 of 1

Sector	Distance (Meters)	Pasquill Categories						
		A	B	C	D	E	F	G
1.0	1608	9.5 E-7	4.0 E-6	1.5 E-5	5.0 E-5	9.0 E-5	2.1 E-4	3.4 E-4
1.5	2412	6.3 E-7	2.1 E-6	1.1 E-5	5.4 E-5	5.4 E-5	1.3 E-4	2.2 E-4
2.0	3216	5.2 E-7	8.3 E-7	5.0 E-6	1.9 E-5	3.9 E-5	9.6 E-5	1.8 E-4
2.5	4020	4.4 E-7	5.8 E-7	3.5 E-6	1.4 E-5	3.7 E-5	7.0 E-5	1.7 E-4
3.0	4824	3.6 E-7	5.0 E-7	2.8 E-6	1.0 E-5	2.2 E-5	5.7 E-5	1.3 E-4
3.5	5628	3.2 E-7	4.2 E-7	2.0 E-6	8.1 E-6	1.8 E-5	4.7 E-5	1.1 E-4
4.0	6432	2.8 E-7	3.7 E-7	1.6 E-6	6.8 E-6	1.5 E-5	4.0 E-5	9.4 E-5
4.5	7236	2.6 E-7	3.5 E-7	1.4 E-6	5.8 E-6	1.3 E-5	3.5 E-5	7.3 E-5
5.0	8040	2.4 E-7	3.2 E-7	1.2 E-6	5.1 E-6	1.1 E-5	3.1 E-5	6.7 E-5
5.5	8844	2.1 E-7	3.1 E-7	9.9 E-7	4.4 E-6	1.0 E-5	2.8 E-5	5.9 E-5
6.0	9648	2.0 E-7	2.7 E-7	8.3 E-7	3.8 E-6	9.1 E-6	2.5 E-5	5.4 E-5
6.5	10452	1.9 E-7	2.5 E-7	7.5 E-7	3.5 E-6	8.2 E-6	2.3 E-5	5.0 E-5
7.0	11256	1.8 E-7	2.4 E-7	6.7 E-7	3.2 E-6	7.5 E-6	2.1 E-5	4.7 E-5
7.5	12060	1.7 E-7	2.3 E-7	6.1 E-7	3.0 E-6	6.9 E-6	1.9 E-6	4.3 E-5
8.0	12864	1.6 E-7	2.2 E-7	5.5 E-7	2.7 E-6	6.3 E-6	1.8 E-5	4.1 E-5
8.5	13668	1.5 E-7	2.1 E-7	5.0 E-7	2.5 E-6	5.8 E-6	1.7 E-5	3.8 E-5
9.0	14472	1.5 E-7	2.0 E-7	4.6 E-7	2.3 E-6	5.5 E-6	1.6 E-5	3.6 E-5
9.5	15276	1.4 E-7	1.9 E-7	4.2 E-7	2.1 E-6	5.4 E-6	1.5 E-5	3.4 E-5
10.0	16080	1.4 E-7	1.8 E-7	4.0 E-7	2.1 E-6	5.3 E-6	1.5 E-5	3.4 E-5

Attachment 9.3
2, 5 and 10-Mile Xμ/Q Values
Sheet 1 of 1

PASQUILL CATEGORY	Xμ/Q		
	2 MILE	5 MILE	10 MILE
A	5.2E-7	2.4E-7	1.4E-7
B	8.3E-7	3.2E-7	1.8E-7
C	5.0E-6	1.2E-6	4.0E-7
D	1.9E-5	5.1E-6	2.1E-6
E	3.9E-5	1.1E-5	5.3E-6
F	9.6E-5	3.1E-5	1.5E-5
G	1.8E-4	6.7E-5	3.4E-5

Attachment 9.4

Reuter-Stokes Location Xμ/Q Values

Sheet 1 of 1

Sector Monitor Distance (m)		Stability Class						
		A	B	C	D	E	F	G
1	3226	5.3E-7	8.4E-7	5.1E-6	1.9E-5	4.0E-5	9.8E-5	1.8E-4
2	3379	5.2E-7	8.3E-7	5.0E-6	1.8E-5	3.9E-5	9.7E-5	1.7E-4
3	2574	6.3E-7	1.2E-6	7.3E-6	2.6E-5	5.3E-5	1.2E-4	2.4E-4
4	1448	1.2E-6	4.6E-6	1.8E-5	6.1E-5	1.1E-4	2.4E-4	3.9E-4
5	1287	1.4E-6	6.4E-6	2.3E-5	7.3E-5	1.4E-4	2.8E-4	4.4E-4
6	643	4.3E-6	2.2E-5	6.0E-5	1.8E-4	3.0E-4	5.5E-4	7.7E-4
7	643	4.3E-6	2.2E-5	6.0E-5	1.8E-4	3.0E-4	5.5E-4	7.7E-4
8	804	2.9E-6	1.7E-5	4.5E-5	1.3E-4	2.4E-4	4.5E-4	6.6E-4
9	1126	1.8E-6	8.5E-6	2.6E-5	8.1E-5	1.5E-4	3.2E-4	4.9E-4
10	1287	1.4E-6	6.4E-6	2.3E-5	7.3E-5	1.4E-4	2.8E-4	4.4E-4
11	1287	1.4E-6	6.4E-6	2.3E-5	7.3E-5	1.4E-4	2.8E-4	4.4E-4
12	2494	6.4E-7	1.3E-6	7.5E-6	2.7E-5	5.6E-5	1.2E-4	2.4E-4
13	1870	8.0E-7	2.7E-6	1.2E-5	4.2E-5	8.1E-5	1.8E-4	3.2E-4
14	1870	8.0E-7	2.7E-6	1.2E-5	4.2E-5	8.1E-5	1.8E-4	3.2E-4
15	1648	9.4E-7	3.9E-6	1.5E-5	5.0E-5	9.7E-5	2.1E-4	3.6E-4
16	1770	8.4E-7	3.3E-6	1.3E-5	4.5E-5	8.8E-5	1.9E-4	3.4E-4

Attachment 9.5

Accident Monitoring of Noble Gas Concentrations in the Plant Vent

Sheet 1 of 2

NOTES

1. The Operations Support Center (OSC) H.P Team Leader / Rad. Protection Coordinator will determine which reading to obtain first; plant vent or back-up plant vent monitoring.
2. Locations and equipment may be different from Unit 2 to Unit 3

1.0 Radiation readings may be obtained on the plant vent by the following:

1.1 Follow the provisions used by the OSC to plan and track team assignments.

1.2 Use a telescoping radiation monitoring instrument (e.g. teletector or equivalent) to perform this function.

1.3 AAs requested by OSC Health Physics (HP) Team Leader or Control Room (CR), REPORT radiation levels.

1.4 Proceed to the Containment Airlock area.

1.5 Using the fan-building wall for shielding, obtain radiation readings by Vapor Containment purge and exhaust ducts.

CAUTION

The door leading out to the plant vent area may lock when closed. To prevent being trapped in the plant vent area, BLOCK OPEN THE DOOR prior to going to the plant vent area.

1.6 Proceed through the door to the plant vent area.

1.7 Obtain radiation readings at the following locations:

1.7.1 6 feet from the plant vent 10 feet above the floor.

1.7.2 Contact with the plant vent 10 feet above the floor.

1.8 Notify the OSC or CR that radiation readings have been obtained and follow instructions as directed.

Attachment 9.5

Accident Monitoring of Noble Gas Concentrations in the Plant Vent

Sheet 2 of 2

2.0 Backup plant vent monitoring readings may be obtained by the following:

2.1 Follow the provisions used by the OSC to plan and track team assignments.

2.2 Proceed to the Auxiliary Building (PAB) Post Accident (PASS) Plant Vent Sample Cave

2.3 Ensure that the RMS-2 meter is positioned on top of the PASS plant vent shield.

2.4 Ensure that the RMS-2 detector is positioned on the floor of the PASS plant vent shield near the gas-sampling bulb.

2.5 Ensure that detector is connected properly to meter with the cable run through the 1-inch hole in the top of the PASS plant vent shield.

2.6 Ensure that the meter is energized by A/C and the power is on.

2.7 With the shield door closed, Establish recirculation flow of plant vent gases through the Pass plant vent piping according to RE-CS-040.

2.8 After recirculation is equilibrated (about 5 minutes)

2.9 Record backup plant vent readings from the RMS-2 monitor.

2.10 Using a hand held meter, OBTAIN a background radiation reading outside of the PASS plant vent shield.

2.11 Report RMS-2 readings to the OSC or CR and FOLLOW instructions as directed.

Attachment 9.6

Manual Calculation of Thyroid CDE

Sheet 1 of 1

Thyroid Committed Dose Equivalent (CDE) for Entergy personnel using actual or estimated data for radioiodine concentrations and stay times.

Calculation:

The following Dose Conversion Factors should be used to determine thyroid CDE based on airborne radioiodine concentration:

$$\text{MIX}^{[1]}\ \text{DCF} = 4.00\text{E}08\ \frac{\text{mRem/hr}}{\mu\text{Ci/cc}}$$

$$\text{I-131 DCF} = 1.30\text{E}09\ \frac{\text{mRem/hr}}{\mu\text{Ci/cc}}$$

$$\text{I-132 DCF} = 7.50\text{E}06\ \frac{\text{mRem/hr}}{\mu\text{Ci/cc}}$$

$$\text{I-133 DCF} = 2.20\text{E}08\ \frac{\text{mRem/hr}}{\mu\text{Ci/cc}}$$

$$\text{I-134 DCF} = 1.30\text{E}06\ \frac{\text{mRem/hr}}{\mu\text{Ci/cc}}$$

$$\text{I-135 DCF} = 3.80\text{E}07\ \frac{\text{mRem/hr}}{\mu\text{Ci/cc}}$$

[1]To be used for the first 24 hours after shutdown when the radioiodine mix is not known. The I-131 DCF is to be used for times greater than 24 hours after shutdown when the radioiodine mix is not known.

IF thyroid CDE is expected to exceed 5 Rem for any Entergy personnel, THEN the ORM should RECOMMEND to the ED that KI be issued to these individuals.

Attachment 9.7
Discussion
Sheet 1 of 2

The following instrumentation/methodology can be used to determine the noble gas release rate.

- Plant vent monitor-low range (Direct Readout)
- Plant vent monitor-high range (Direct Readout)
- Plant vent survey-hand held instrument or remote readout
- Isotopic analysis of sample taken from release point.
- Condenser air ejector monitor (Direct Readout).
- Main steam line monitors.
- Back calculating a release rate based on actual field radiological data.
- Containment radiation monitors R-25 and R-26 to measure the source term within containment and to estimate potential releases from containment.
- Potential exposure to the population if a future release of the existing containment source term occurs, utilizing the following information:
  1. Containment pressure relief line contains three isolation valves (one in containment and two outside).
  2. Containment purge system contains two isolation valves on the Inlet Duct (one in containment and one outside).
  3. Containment purge system contains two isolation valves on the Exhaust Duct (one in containment and one outside).
  4. Weld Channel (WC) and Isolation Valve Seal Water System (IVSWS) are pressurized to ensure that during accident conditions a pressure build up to AT LEAST 50 psi in containment would NOT cause a leak of radioactive material to the environment as long as the isolation valves remained in the closed position.

Attachment 9.6
Discussion
Sheet 2 of 2

5. WITHOUT WC AND IVSWS, BUT with isolation valves closed, the containment leak rate is expected to be LESS THAN 0.1% of the containment volume per day (Tech Spec) WITH a pressure buildup to 50 psi inside containment. At lower pressures the leak rate would be smaller, approaching zero as the pressure differential approaches zero.

6. Containment Volume = **2.6×10^6 ft^3 = 7.4×10^{10} cc**

7. For IP2 and Post-Steam Generator Tube Rupture (SGTR) cooldown using blowdown situations, the determination of the gaseous release rate from the blowdown flash tank shall be accomplished by determining the noble gas concentration in the faulted SG blowdown (Chem sample µCi/cc) AND the blowdown rate (GPM).