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AFFECTED DOCUMENT IPEC EMERGENCY PLANNING PROCEDURES

DOC #	REV #	TITLE	INSTRUCTIONS

THE FOLLOWING PROCEDURE(S) HAS BEEN REVISED, PLEASE REMOVE YOUR CURRENT COPY AND REPLACE WITH ATTACHED UPDATED REVISION:

IP-EP-360 REVISION 6

And General Record

EFFECTIVE DATE: 6/1/2020

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10CFR50.54(Q)(2) Review

Procedure/Document Number: IP-EP-360	**Revision: 6**
Equipment/Facility/Other: Indian Point Energy Center	
Title: Core Damage Assessment	

Part I. Description of Activity Being Reviewed (event or action, or series of actions that have the potential to affect the emergency plan or have the potential to affect the implementation of the emergency plan):

Procedure was revised, to reflect the requirement in the Post Unit 2 Shutdown Eplan (PSEP), as submitted to the NRC per LAR, license #NL-19-001. See attached matrix for changes made. Procedure will be effective on June 1, 2020,

Part II. Emergency Plan Sections Reviewed (List all emergency plan sections that were reviewed for this activity by number and title. IF THE ACTIVITY IN ITS ENTIRETY IS AN EMERGENCY PLAN CHANGE, EAL CHANGE OR EAL BASIS CHANGE, ENTER THE SCREENING PROCESS. NO 10CFR50.54(q)(2) DOCUMENTATION IS REQUIRED.

Part 1 Introduction:

Section A: Purpose

Part 2 Planning Standards and Criteria:

Section A: Assignment of Responsibility

Section B: Station Emergency Response Organization

Section H: Emergency Facilities and Equipment

Section I: Accident Assessment

Part III. Ability to Maintain the Emergency Plan (Answer the following questions related to impact on the ability to maintain the emergency plan):

1. Do any elements of the activity change information contained in the emergency plan (Section 3.0 Step 6)?
   YES ☐ NO ☒ **IF YES, enter screening process for that element**

2. Do any elements of the activity change an emergency classification Initiating Condition, Emergency Action Level (EAL), associated EAL note or associated EAL basis information or their underlying calculations or assumptions?
   YES ☐ NO ☒ **IF YES, enter screening process for that element**

3. Do any elements of the activity change the process or capability for alerting and notifying the public as described in the FEMA-approved Alert and Notification System design report?
   YES ☐ NO ☒ **IF YES, enter screening process for that element**

4. Do any elements of the activity change the Evacuation Time Estimate results or documentation?
   YES ☐ NO ☒ **IF YES, enter screening process for that element**

5. Do any elements of the activity change the Onshift Staffing Analysis results or documentation?
   YES ☐ NO ☒ **IF YES, enter screening process for that element**

10CFR50.54(Q)(2) Review

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Title: Core Damage Assessment	

Part IV. Maintaining the Emergency Plan Conclusion The questions in Part III do not represent the sum total of all conditions that may cause a change to or impact the ability to maintain the emergency plan. Originator and reviewer signatures in Part V document that a review of all elements of the proposed change have been considered for their impact on the ability to maintain the emergency plan and their potential to change the emergency plan.

1. Provide a brief conclusion that describes how the conditions as described in the emergency plan are maintained with this activity.
2. Check the box below when the 10CFR50.54(q)(2) review completes all actions for all elements of the activity – no 10CFR50.54(q)(3) screening or evaluation is required for any element. Otherwise, leave the checkbox blank.

☒ I have completed a review of this activity in accordance with 10CFR50.54(q)(2) and determined that the effectiveness of the emergency plan is maintained. This activity does not make any changes to the emergency plan. No further actions are required to screen or evaluate this activity under 10CFR50.54(q)(3).

Per Post Shutdown Emergency Plan (PSEP), Unit 3 CCR will be the active/running plant and Unit 2 will be at shut down. The changes made to this procedure (see attached matrix) reflects this requirement of the Post Unit 2 Shutdown Eplan, as submitted to the NRC (license # NL-19-001) and added a graph to support determining RVLIS level for Unit 3. The NRC has approved the PSEP per RA-20-040.

A review of this activity in accordance with 10 CFR 50.54(q)(2) has been completed and determined that the effectiveness of the PSEP is maintained. This revision aligns the procedure with the protocols of the post Unit 2 shutdown. None of the changes affect the ability to perform classifications, notifications, or PARs, it does not affect activation or staffing of the ERO, and all planning standard requirements are maintained. The changes made do not require a change to the Emergency Action Level scheme, On-shift Staffing study or the PSEP.

No further actions are required to screen or evaluate this activity under 10 CFR 50.54(q)(3).

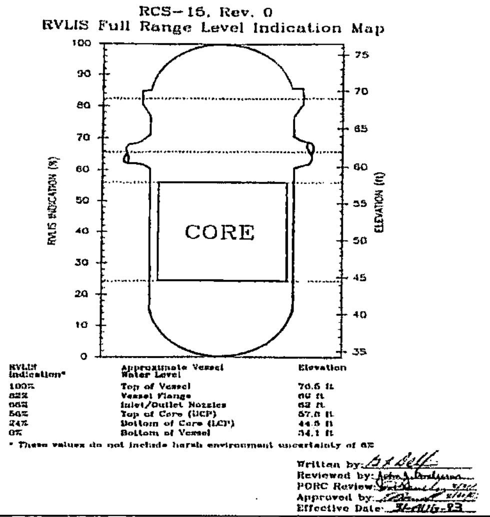
Part V. Signatures:

Preparer Name (Print) Rebecca A. Martin	Preparer Signature Rebecca A Martin	Date: 5/21/2020
(Optional) Reviewer Name (Print)	Reviewer Signature	Date:
Reviewer Name (Print) Timothy Garvey Nuclear EP Project Manager	Reviewer Signature Rebecca A. Martin for T. Garvey Approval Per Telecom	Date: 5/26/2020
Approver Name (Print) Frank Mitchell Emergency Planning Manager or designee	Approver Signature F Mitchell	Date: 5/26/2020

IP-EP-360 Revision 6
REVISION MATRIX

Change No.	Page/Section	Previous Version	New Version	Editorial Change	Effect on 10 CFR 50.47(b) Planning Standards or NUREG-0654 program elements? Justify if NO.
1.	Page 3 Reference	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.	None	N	N – Removed reference for Unit 2 and updated numbering. Per Post Shutdown Emergency Plan, Unit 3 CCR will be the active/running plant and Unit 2 will be defueled and core damage assessment is not needed for Unit 2. This change reflects that requirement in the Post Unit 2 shut down Eplan, which is under an LAR. (license # NL-19-001) NRC approved per RA-20-040.
2.	Page 4 Section 5.1 b	b. Use H2-02 analyzer on Accident Assessment Panel (Unit 2) or Initiate performance of 3-SOP-SS-004, "containment Hydrogen Concentration Measurement System" (Unit 3).	b. Initiate performance of 3-SOP-SS-004, "containment Hydrogen Concentration Measurement System" (Unit 3).	N	N – removed Use H2-02 analyzer on Accident Assessment Panel (Unit 2) Per Post Shutdown Emergency Plan, Unit 3 CCR will be the active/running plant and Unit 2 will be defueled and core damage assessment is not needed for Unit 2. This change reflects that requirement in the Post Unit 2 shut down Eplan, which is under an LAR. (license # NL-19-001) NRC approved per RA-20-040.
3.	Page 5 Section 9.4	None	9.4 Attachment 2, RCS-15	Y	N – added an attachment to the procedure. This change only updated the attachment section. See Change #8 for justification.

4.	Page 9 Section 2.1	(Refer to PICS [Unit 2] or SPDS [Unit 3])	(Refer to PICS or SPDS [Unit 3])	N	N – removed Unit 2 reference. Per Post Shutdown Emergency Plan, Unit 3 CCR will be the active/running plant and Unit 2 will be defueled and core damage assessment is not needed for Unit 2. This change reflects that requirement in the Post Unit 2 shut down Eplan, which is under an LAR. (license # NL-19-001) NRC approved per RA-20-040.
5.	Page 12, section 3.1	3.1 Determine the following:	3.1 Determine the following: (see Attachment 4)	N	N – added "(see Attachment 4)" to this step to support in determining RVLIS indications. See justification in change #8.
6.	Page 14 Section 2.1	(Refer to PICS [Unit 2] or [Unit 3])	(Refer to PICS [Unit 3])	N	N - Removed Unit 2 reference. Per Post Shutdown Emergency Plan, Unit 3 CCR will be the active/running plant and Unit 2 will be defueled and core damage assessment is not needed for Unit 2. This change reflects that requirement in the Post Unit 2 shut down Eplan, which is under an LAR. (license # NL-19-001) NRC approved per RA-20-040.
7.	Page 20, Section 4.4.1	• Hydrogen burn in containment or affects of passive autocatalytic hydrogen recombination (Unit 2)	• Hydrogen burn in containment	N	N – Removed Unit 2 reference. Per Post Shutdown Emergency Plan, Unit 3 CCR will be the active/running plant and Unit 2 will be defueled and core damage assessment is not needed for Unit 2. This change reflects that requirement in the Post Unit 2 shut down Eplan, which is under an LAR. (license # NL-19-001) NRC approved per RA-20-040.

8.	Page 21 Attachment 4	None		N	N -. Per request of Rx Engineering who is the end user of this procedure, Attachment 4 RCS-15 graph was added to assistant in determining RVLIS indications. This graph is used by Rx Engineering during drills and finding the graph has been time consuming. Graph was added to procedure to reduce time spent looking for it. Adding the graph did not change the intent of the procedure and will aide in getting results faster and does not need any further evaluation

ATTACHMENT 10.2 **IPEC PROCEDURE REVIEW AND APPROVAL**

(Page 1 of 1)

Procedure Title: Core Damage Assessment

Procedure No: IP-EP-360 Existing Rev: 5 New Rev: 6 DRN/EC No: DRN-20-00308

Procedure Activity (MARK Applicable)	☐ Converted To IPEC, Replaces:	**Temporary Procedure Change (MARK Applicable)**
☐ NEW PROCEDURE ☐ GENERAL REVISION ☒ PARTIAL REVISION ☐ EDITORIAL REVISION ☐ VOID PROCEDURE ☐ SUPERSEDED	Unit 1 Procedure No: ______ Unit 2 Procedure No: ______ Unit 3 Procedure No: ______	☐ EDITORIAL Temporary Procedure Change ☐ ADVANCE Temporary Procedure Change ☐ CONDITIONAL Temporary Procedure Change Terminating Condition: ______
☐ RAPID REVISION	Document in Microsoft Word: ☐ Yes ☐ No	☐ VOID DRN/TPC No(s): ______

Revision Summary ☐ N/A - See Revision Summary Matrix

Implementation Requirements

Implementation Plan? ☐ Yes ☒ No Formal Training? ☒ Yes ☐ No Special Handling? ☐ Yes ☒ No

RPO Dept: Emergency Planning Writer **(Print Name/ Ext/Sign):** Rebecca A. Martin/x7106/ Rebecca A Martin

Review and Approval (Per Attachment 10.1, IPEC Review And Approval Requirements)

1. ☒ Technical Reviewer: Kevin Robinson / [signature] 5-14-2020
(Print Name/ Signature/ Date)

2. ☒ Cross-Disciplinary Reviewers:
Dept: Rx Eng Reviewer: Ray Williams / [signature] Digitally signed by Raymond Williams Date: 2020.05.26 07:52:29 -04'00'
(Print Name/ Signature/ Date)
Dept: ______ Reviewer: ______
(Print Name/ Signature/ Date)

3. ☒ **RPO- Responsibilities/Checklist:** F. Mitchell [signature] 5/15/2020
(Print Name/ Signature/ Date)
   - ☒ PAD required and is complete (PAD Approver and Reviewer qualifications have been verified)
   - ☐ Previous exclusion from further LI-100 Review is still valid
   - ☐ PAD not required due to type of change as defined in 4.6

4. ☐ Non-Intent Determination Complete: ______
(Print Name/ Signature/ Date)

NO change of purpose or scope
NO reduction in the level of nuclear safety
NO voiding or canceling of a procedure, unless requirements are incorporated into another procedure or the need for the procedure was eliminated via an alternate process.
NO change to less restrictive acceptance criteria
NO change to steps previously identified as commitment steps
NO deviation from the Quality Assurance Program Manual
NO change that may result in deviations from Technical Specifications, FSAR, plant design requirements or previously made commitments.

5. ☐ On-Shift Shift Manager/CRS: ______
(Print Name/ Signature/ Date)
6. ☐ User Validation: User: ______
7. ☐ Special Handling Requirements Understood: ______

CONTROLLED

CORE DAMAGE ASSESSMENT

Prepared by:	Rebecca A. Martin	Rebecca A Martin	5/21/2020
	Print Name	Signature	Date
Approval:	Frank J. Mitchell	Frank Mitchell	5/26/2020
	Print Name	Signature	Date

Effective Date: June 1, 2020

Table of Contents

CORE DAMAGE ASSESSENT

1.0 PURPOSE

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

2.3 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 Reactor Engineer shall implement this procedure to assess the existence and extent of core damage.

4.2 The Reactor Engineer shall immediately inform the Engineering Coordinator /TSC Manager of the results of any core damage assessment performed.

5.0 DETAILS

NOTE

- Core Damage Estimate may be based 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.
- The Core Damage Assessment may be performed as data becomes available. If data is unavailable for a given core damage methodology, then the affected step(s) can be NA'd.
- Containment Hi Range Radiation Monitor R-25 and R-26 bottom scale reading is approximately ~1 R/hr. Because of this scale limitation of R-25 and R-26, radiation monitors R-2, VC 80ft and R-7, VC Seal table should be used to observe an increasing trend towards 1 R/hr (1000 mr/hr), when assessing core damage using the "High level Core Damage Assessment Flowchart". Due to containment positions, R-2/R-7 readings of approximately 200 mr/hr, should relate to 1 R/hr on R-25/R-2.

5.1 If possible, check or obtain Containment Hydrogen Concentration by either:

a. Dispatching chemistry personnel to obtain sample or

b. Initiate performance of 3-SOP-SS-004, "containment Hydrogen Concentration Measurement System" (Unit 3).

5.2 Determine the possible status of the reactor core using the flowchart in Attachment 1 and perform the associated action.

6.0 INTERFACES

6.1 IP-EP-120, Emergency Classification

6.2 EN-EP-610, Technical Support Center

7.0 RECORDS

This procedure generates completed Fuel Rod Clad Damage (Attachment 2) and/or Fuel Over-temperature Damage (Attachment 3) worksheets.

8.0 REQUIREMENTS AND COMMITMENTS

None

9.0 ATTACHMENTS

9.1 Attachment 1, High Level Core Assessment Flowchart

Figure 1A, Containment Radiation Level for 1%. Fuel Over-temperature Release (0 to 6 hours after shutdown

Figure 1B, Containment Radiation Level for 1%. Fuel Over-temperature Release (>5 hours after shutdown)

9.2 Attachment 2, Fuel Rod Clad Damage

9.3 Attachment 3, Fuel Over-temperature Damage

9.4 Attachment 4, RCS-15

High Level Core Damage Assessment Flowchart

Attachment 1

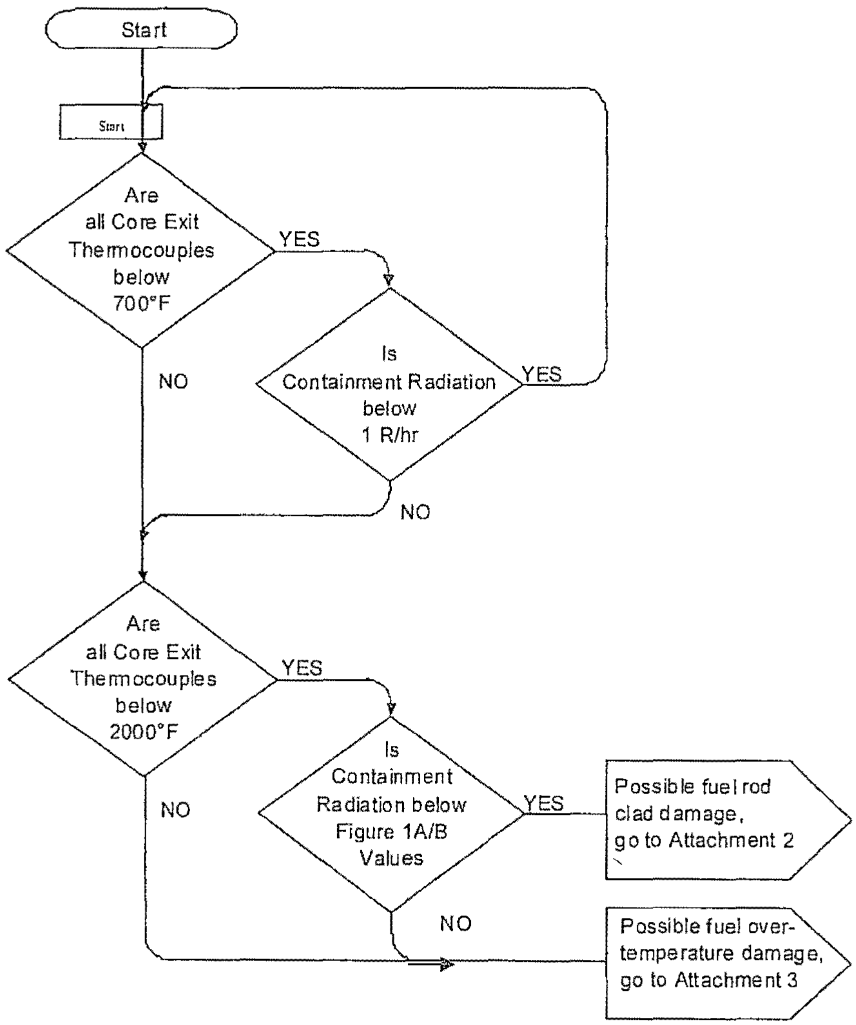


Figure 1A
Containment Radiation Level for 1% Fuel Overtemperature Release Flowchart

(0 to 6 hours after shutdown)

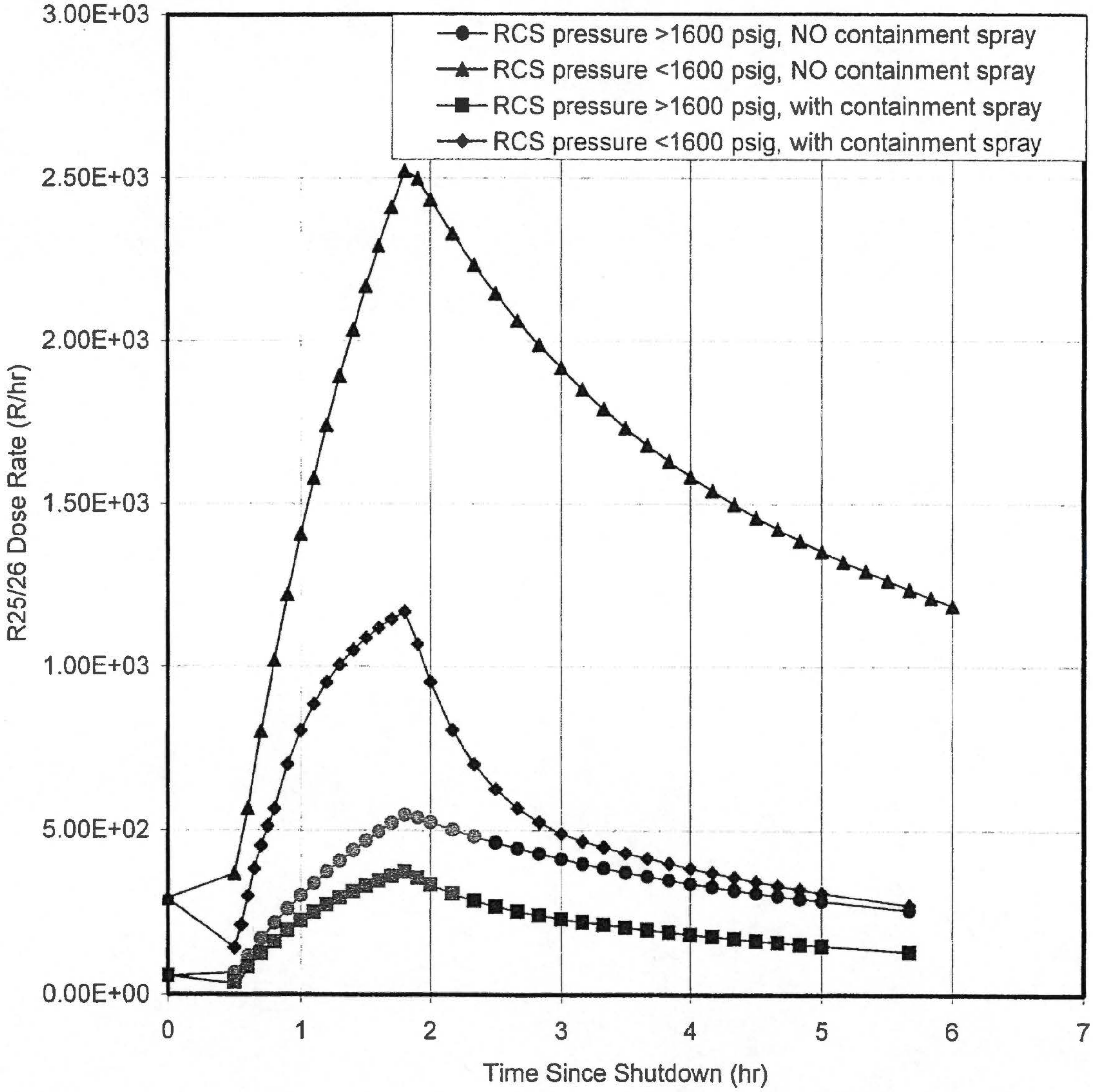
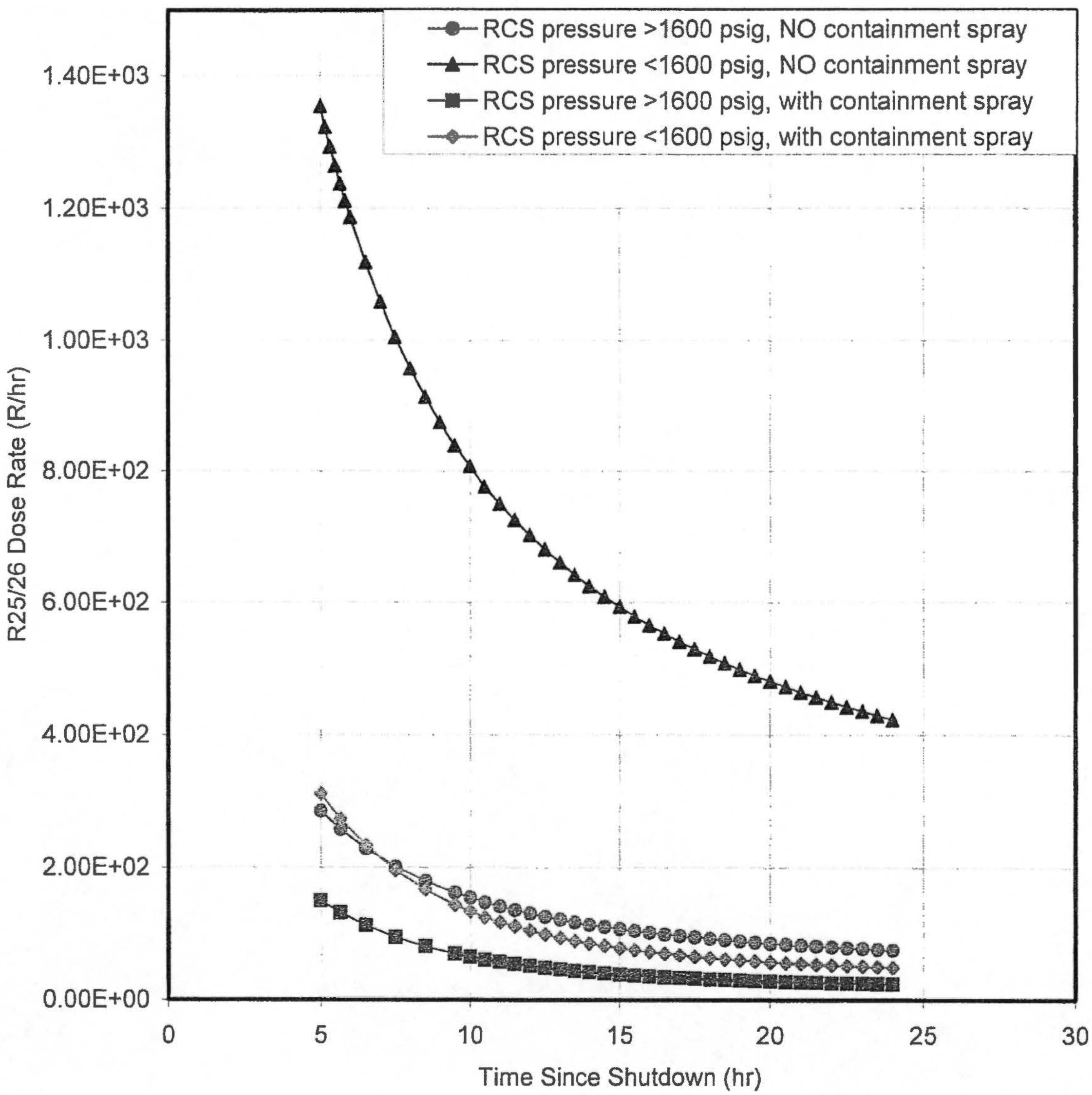


Figure 1B
Containment Radiation Level for 1% Fuel Overtemperature Release

(>5 hours after shutdown)



Attachment 2
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 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:

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

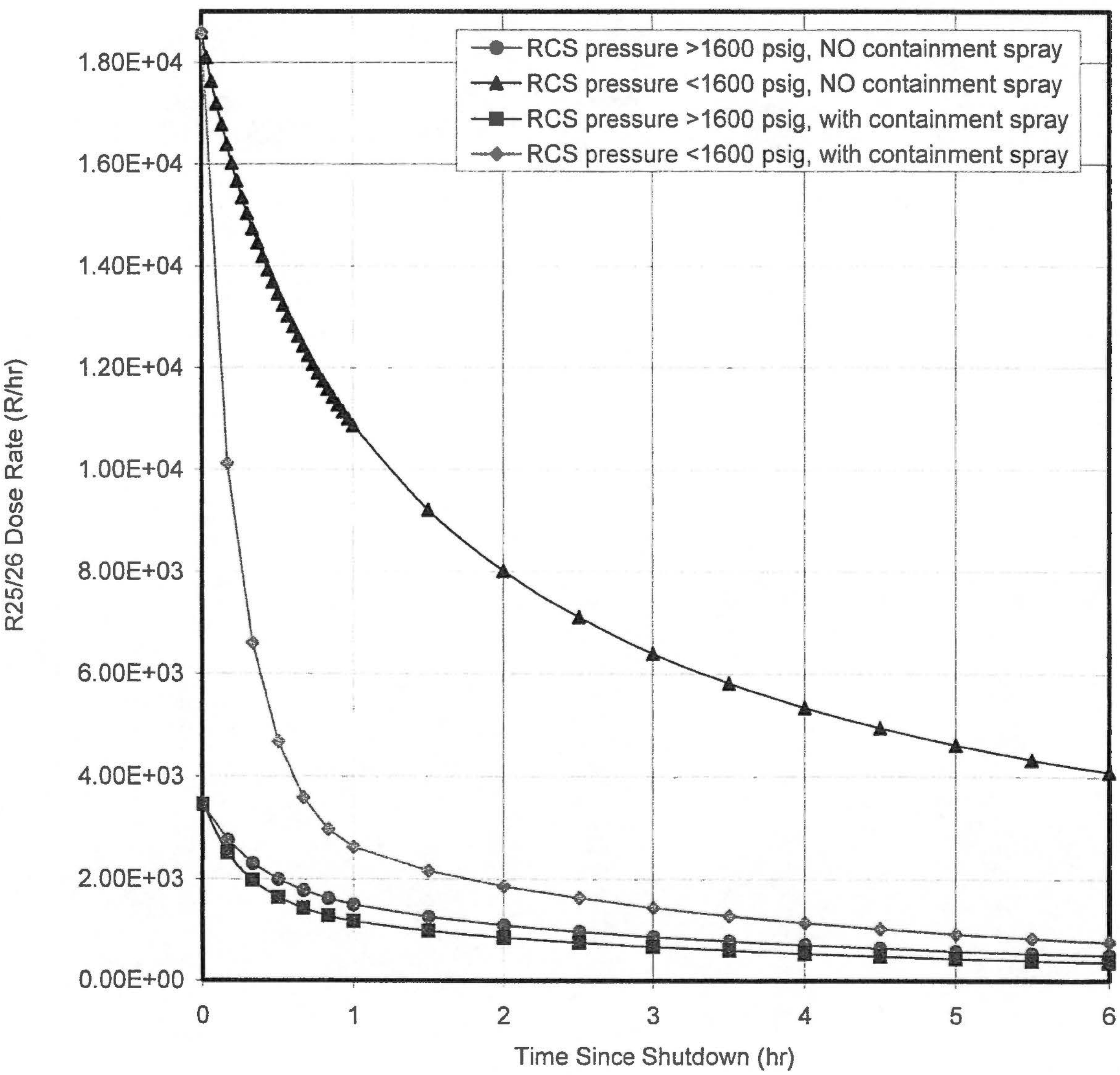
   2.3 For RCS pressure below 1600 psig:

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

Attachment 2
Fuel Rod Clad Damage
Sheet 2 of 5

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

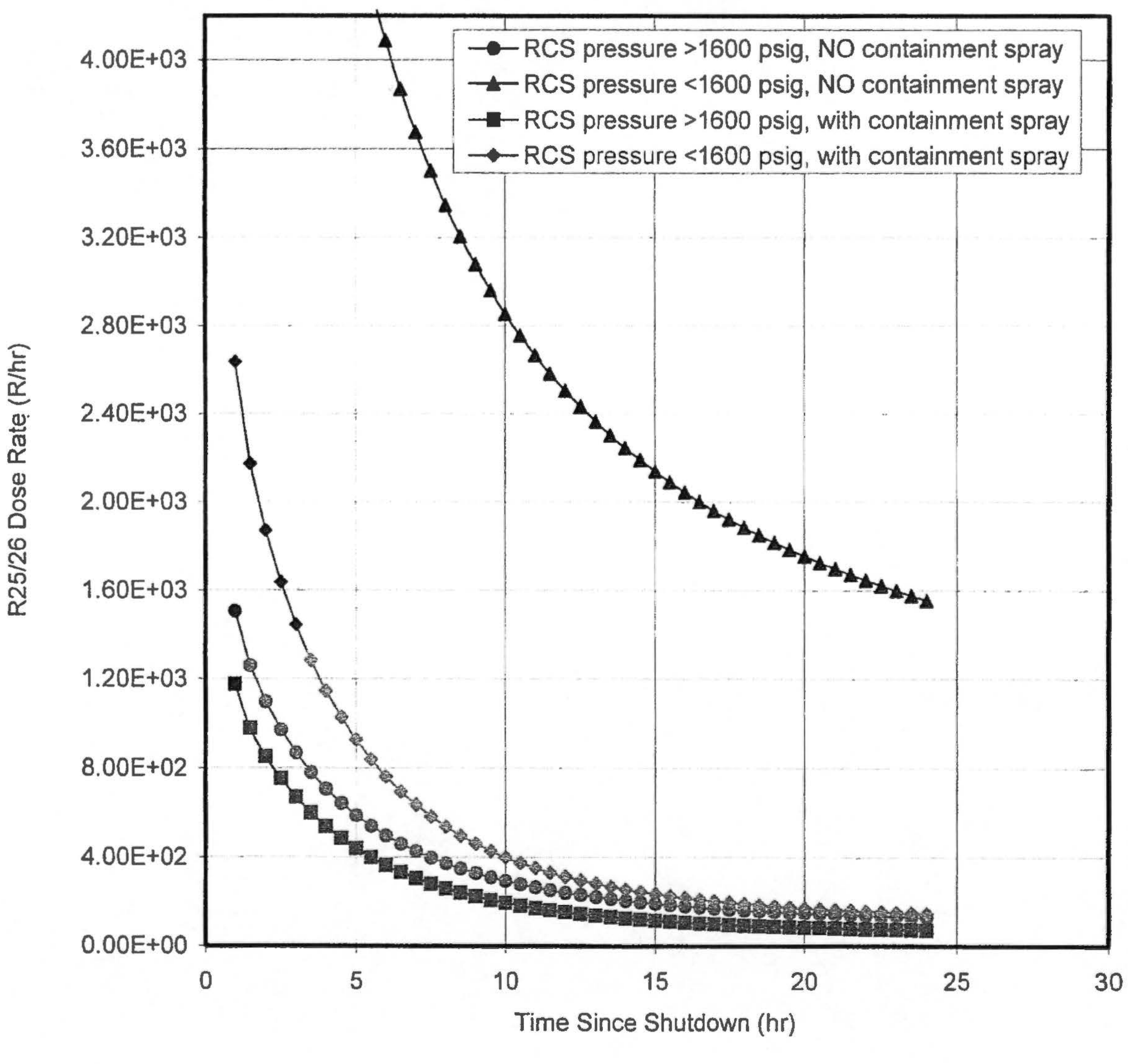
(0 to 6 hours after shutdown)



Attachment 2
Fuel Rod Clad Damage
Sheet 3 of 5

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

(> 1 hour after shutdown)



Attachment 2
Fuel Rod Clad Damage
Sheet 4 of 5

3. Confirm reasonableness of clad damage estimates.

3.1 Determine the following: (see Attachment 4)

- 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 2
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 Manager.

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

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

Attachment 3
Fuel Over-temperature 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 3]) J = __________
   - Number of CETs at or above 2000°F K = __________

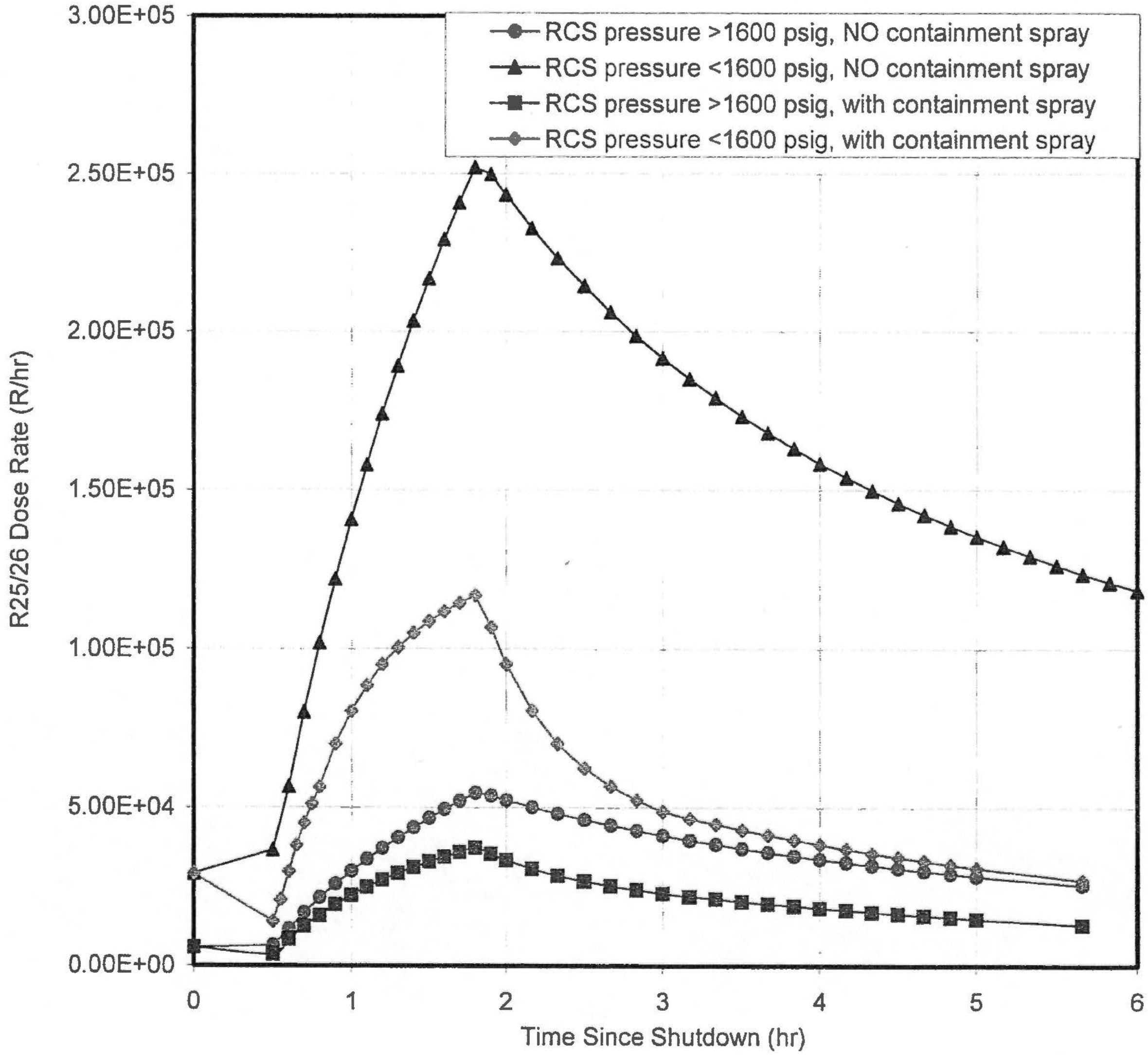
   2.2 Estimate fuel overtemperature damage (%):

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

Attachment 3
Fuel Over-temperature Damage
Sheet 2 of 7

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

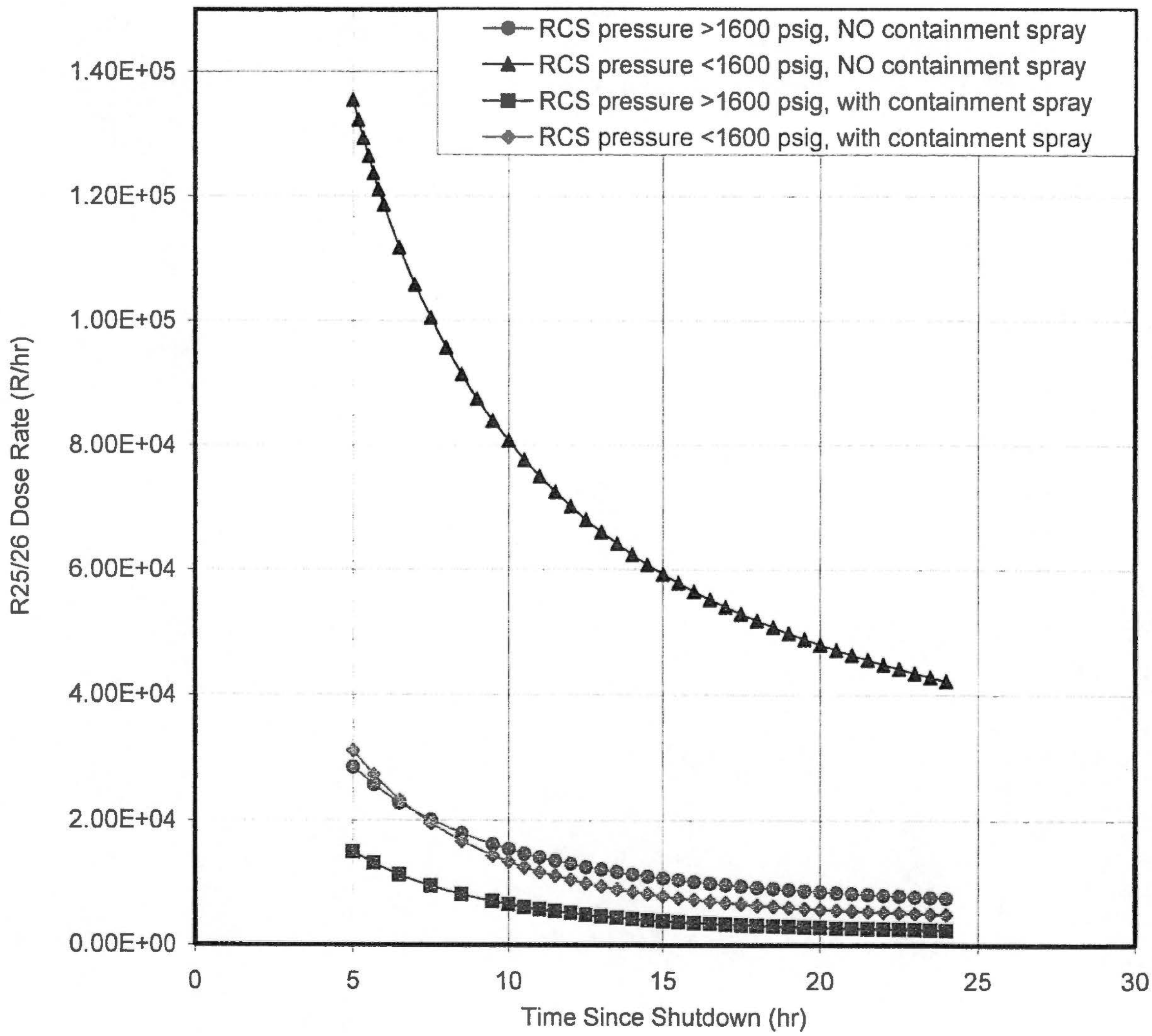
(0 to 6 hours after shutdown)



Attachment 3
Fuel Over-temperature Damage
Sheet 3 of 7

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

(>5 hours after shutdown)



Attachment 3
Fuel Over-temperature 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

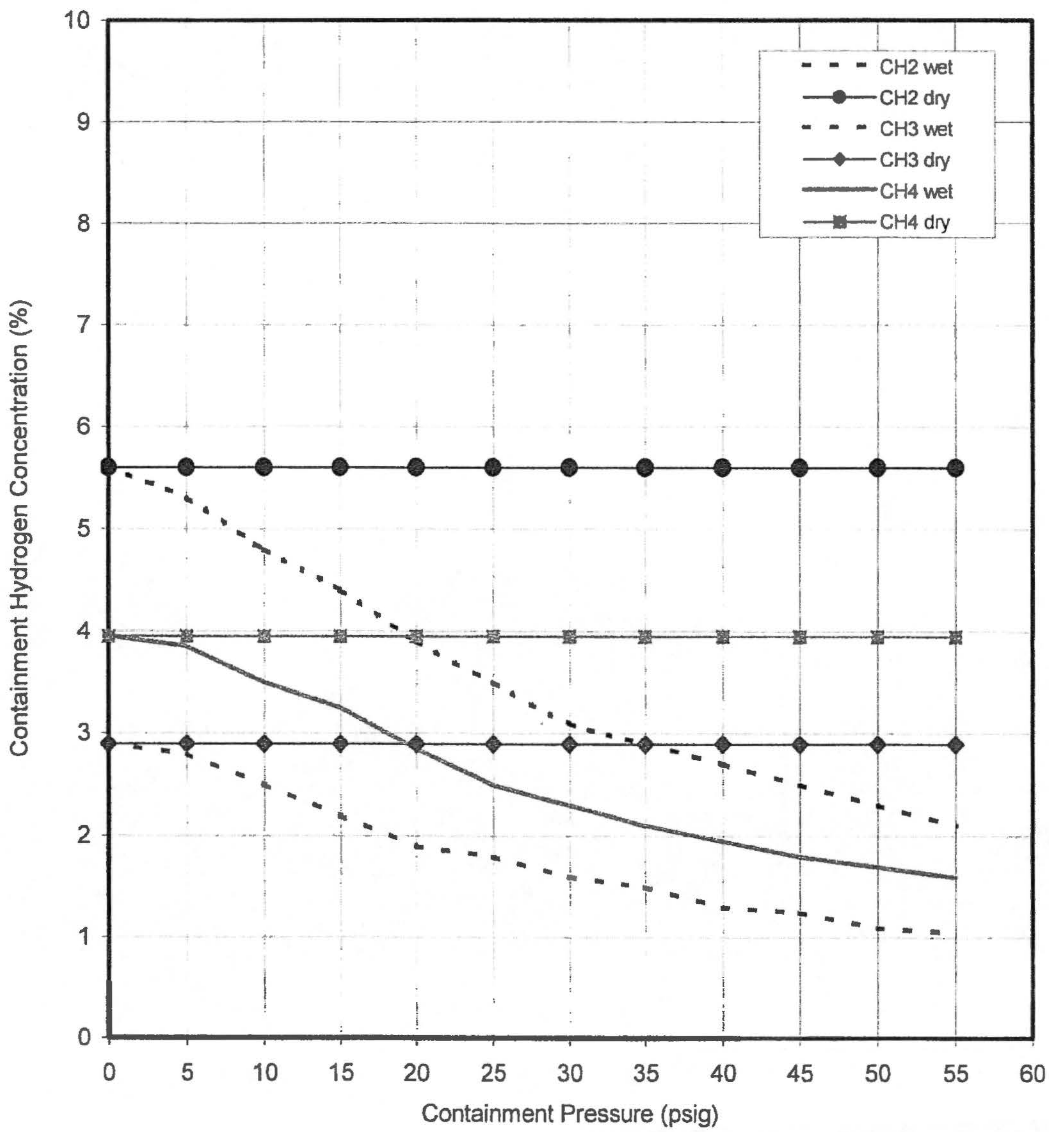
3.2 Estimate fuel overtemperature damage (%):

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

Attachment 3
Fuel Over-temperature 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 3
Fuel Over-temperature 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%? ________

$$\% \text{ Diff}_{\text{diff}} = \frac{|\% \text{ Core Damage}_{\text{CRM}} - \% \text{ Core damage}_{\text{CET}}|}{\% \text{ Core Damage}_{\text{CRM}}} \times 100$$

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

$$\% \text{ Diff}_{\text{diff}} = \frac{|\% \text{ Core Damage}_{\text{Hyd}} - \% \text{ Core damage}_{\text{CRM}}|}{\% \text{ Core Damage}_{\text{Hyd}}} \times 100$$

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

$$\% \text{ Diff}_{\text{diff}} = \frac{|\% \text{ Core Damage}_{\text{Hyd}} - \% \text{ Core damage}_{\text{CET}}|}{\% \text{ Core Damage}_{\text{Hyd}}} \times 100$$

Attachment 3
Fuel Over-temperature 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

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 Engineering Coordinator/TSC Manager.

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

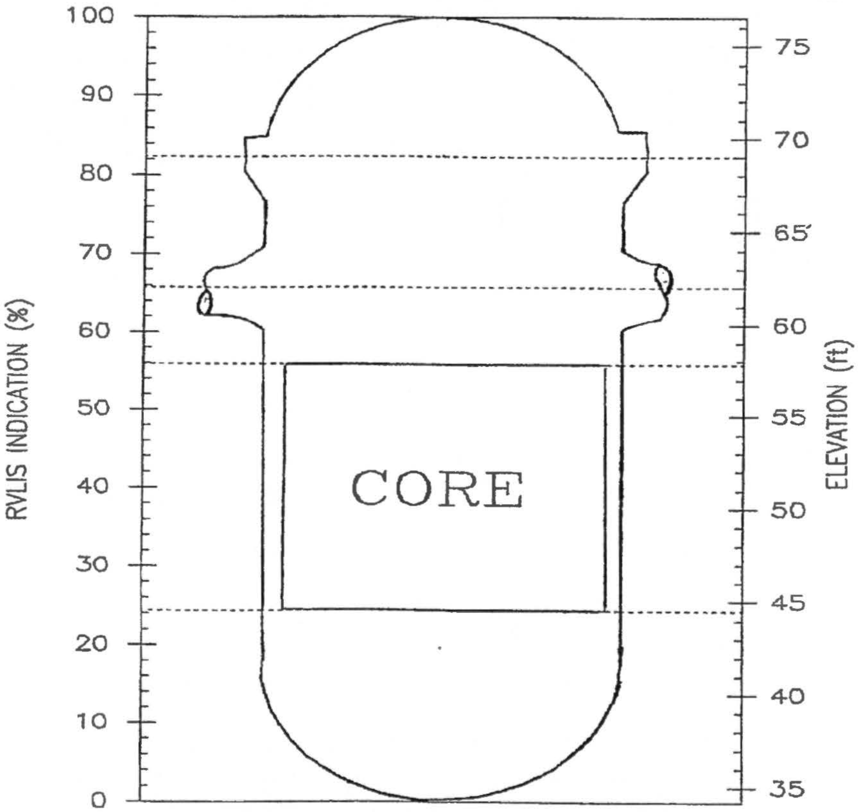
Attachment 3

RCS-15

Sheet 1 of 1

RCS−15, Rev. 0

RVLIS Full Range Level Indication Map



RVLIS Indication*	Approximate Vessel Water Level	Elevation
100%	Top of Vessel	76.5 ft
82%	Vessel Flange	69 ft
66%	Inlet/Outlet Nozzles	62 ft
56%	Top of Core (UCP)	57.8 ft
24%	Bottom of Core (LCP)	44.5 ft
0%	Bottom of Vessel	34.1 ft

* These values do not include harsh environment uncertainty of 6%

Written by: B J Delf

Reviewed by: John A. Anderson

PORC Review: [signature] 8/31/

Approved by: [signature] 8/31/

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