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January 14, 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Changes to Emergency Plan Implementing Procedures –January 14, 2004

Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29

GNRO-2004/00003

Ladies & Gentlemen:

Entergy Operations, Inc. submits in accordance with 10CFR50 Appendix E, Section V changes to the following Emergency Plan Implementing Procedure(s):

10-S-01-35	Rev. 2

This letter does not contain any commitments.

Yours truly,

CAB/MJL
attachment: 1. Procedure 10-S-01-35

cc: (See Next Page)

cc:

Hoeg	T. L.	(GGNS Senior Resident)	(w/a)
Levanway	D. E.	(Wise Carter)	(w/a)
Reynolds	N. S.		(w/a)
Smith	L. J.	(Wise Carter)	(w/a)
Thomas	H. L.		(w/o)

U.S. Nuclear Regulatory Commission ATTN: Mr. Bruce Mallett (w/2) 611 Ryan Plaza Drive, Suite 400 Arlington, TX 76011-4005	ALL LETTERS
U.S. Nuclear Regulatory Commission ATTN: Mr. Bhalchandra Vaidya, NRR/DLPM (w/2) **ATTN: FOR ADDRESSEE ONLY** ATTN: U.S. Postal Delivery Address Only Mail Stop OWFN/7D-1 Washington, D.C. 20555-0001	ALL LETTERS – U.S. POSTAL SERVICE MAIL DELIVERY ADDRESS ONLY

PLANT OPERATIONS MANUAL

Volume 10 | 10-S-01-35
Section 01 | Revision: 2
Date: 12/16/03

EMERGENCY PLAN PROCEDURE

CORE DAMAGE ASSESSMENT

SAFETY RELATED

Prepared: William E. Leng Jr.

Reviewed: W A Russell
Technical

Concurred:
Responsible Manager

OSRC:

Approved: J Brad Edwards / M
Plant General Manager / Manager Emergency Preparedness

List of Effective Pages:

Pages 1-7

Attachment I-VI~~I~~ WEL 12-15-03

List of TCNs Incorporated:

Revision	TCN
0	None
1	None
2	None

Title: Core Damage Assessment	No.: 10-S-01-35	Revision: 2	50.59 Evaluation

I. OVERVIEW / SIGNATURES

Facility: Grand Gulf Nuclear Station

Document Reviewed: 10-S-01-35 **Change / Rev.** 2

System Designator(s) / Description: N/A

Description of Proposed Change:

Operating license commitment 2.c. (33) (c) was deleted in Amendment 158 of the Grand Gulf Nuclear Station (GGNS) operating license. This commitment previously specified requirements for the Post Accident Sampling System (PASS). This revision to the core damage assessment procedure removes the PASS system as an input to the procedure and incorporates the updated GE methodology described in NEDC-33045P to obtain core damage estimates.

If the proposed activity, in its entirety, involves any one of the criteria below, check the appropriate box, provide a justification/basis in the Description above, and forward to a Reviewer. No further 50.59 Review is required. If none of the criteria is applicable, continue with the 50.59 Review.

- ☐ The proposed activity is editorial/typographical as defined in Section 5.2.2.1.
- ☐ The proposed activity represents an "FSAR-only" change as allowed in Section 5.2.2.2______. (Insert item # from Section 5.2.2.2).

If further 50.59 Review is required, check the applicable review(s): (Only the sections indicated must be included in the Review.)

■	**SCREENING**	**Sections I, II, III and IV required**
☐	**50.59 EVALUATION EXEMPTION**	**Sections I, II, III, IV and V required**
☐	**50.59 EVALUATION (#: ________________)**	**Sections I, II, III, IV and VI required**

	Name (print) /	Signature /	Company /	Department /	Date
Preparer:	William E. Long Jr.	William E Long Jr	EOI	NE- SA	11-4-03
Reviewer:	W. A. Russell	W A Russell	EOI	O/S ~~DS~~ WR 11/5/03	11/5/03

OSRC: N/A

Chairman's Name (print) **Signature /** **Date**

(Required only for Programmatic Exclusion Screenings (see Section 5.8) and 50.59 Evaluations.)

List of Assisting / Contributing Personnel:

Name: ______________________ **Scope of Assistance:** ______________________

Section 1.01 *II. SCREENING*

Sect **Section 1.03** *Licensing Basis Document Review*

1. Does the proposed activity impact the facility or a procedure as described in any of the following Licensing Basis Documents? (Check "N/A" for those documents that are not applicable to the facility.)

	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Operating License	☐	■	
TS	☐	■	
NRC Orders	☐	■	
If "YES", obtain NRC approval prior to implementing the change by initiating an LBD change in accordance with NMM LI-113 (Reference 2.2.13). (See Section 5.1.13 for exceptions.)			

	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
FSAR	☐	■	
TS Bases	☐	■	
Technical Requirements Manual	☐	■	
Core Operating Limits Report	☐	■	
NRC Safety Evaluation Reports[1]	☐	■	
If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

	YES	NO	CHANGE # and/or SECTIONS TO BE REVISED
Quality Assurance Program Manual[2]	☐	■	
Emergency Plan[2]	☐	■	
Fire Protection Program[3] (includes the Fire Hazards Analysis)	☐	■	
Offsite Dose Calculations Manual	☐	■	
If "YES", evaluate any changes in accordance with the appropriate regulation AND initiate an LBD change in accordance with NMM LI-113 (Reference 2.2.13).			

2. Does the proposed activity involve a test or experiment not described in the FSAR? ☐ Yes

   If "YES", perform an Exemption Review per Section V OR perform a 50.59 Evaluation per Section VI. ■ No

3. Does the proposed activity potentially impact equipment, procedures, or facilities utilized for storing spent fuel at an Independent Spent Fuel Storage Installation? ☐ Yes ☐ No ■ N/A

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[1] If "YES", see Section 5.1.5. No LBD change is required.
[2] If "YES", *notify the responsible department and ensure a* 50.54 Evaluation is performed. Attach the 50.54 Evaluation.
[3] If "YES", evaluate the change in accordance with the requirements of the facility's Operating License Condition.

Installation?
(Check "N/A" if dry fuel storage is not applicable to the facility.)

If "YES", perform a 72.48 Review in accordance with NMM Procedure LI-112. (See Sections 1.5 and 5.3.1.5 of the EOI 10CFR50.59 Review Program Guidelines.)

B. Basis

Provide a clear, concise basis for the answers given in the applicable sections above. Explain why the proposed activity does or does not impact the Operating License / Technical Specifications and / or the FSAR and why the proposed activity does or does not involve a new test or experiment not previously described in the FSAR. Adequate basis must be provided within the Screening such that a third-party reviewer can reach the same conclusions. Simply stating that the change does not affect TS or the FSAR is not an acceptable basis. See EOI 50.59 Guidelines Section 5.6.6 for guidance.

Operating License

Operating license commitment 2.c. (33) (c) was deleted in Amendment 158 of the Grand Gulf Nuclear Station (GGNS) operating license. This commitment previously specified requirements for the Post Accident Sampling System (PASS). This revision is consistent with implementation of this amendment in that the core damage assessment is performed in accordance with the methodology considered in the Generic SER for BWROG Topical Report NEDO-32991, dated June 12, 2001 (ADAMS Accession Number ML011630016). Specifically this revision includes other real-time plant indicators to obtain more realistic core damage estimates. Therefore, the proposed activity does not impact the GGNS operating license. The Technical Specifications and the Environmental Protection Plan are not impacted by the revision.

Technical Specifications, Technical Requirements Manual and TS Basis

This procedure is not controlled by the TS, TRM, or TS bases. No changes to these documents are required as a result of the procedure changes.

NRC Orders

The NRC Orders issued at Grand Gulf are not affected. This procedure is not to be used for security reasons, which is the subject of Grand Gulf's current NRC Orders. This activity does not impact the Security Plan since it does not require the breaching security barriers. Therefore, the NRC Orders are not affected.

UFSAR

The use of the PASS system for obtaining samples for core damage assessment was removed from the UFSAR by LDC 2003-0095. The use of the PASS system is otherwise unchanged. Therefore, the UFSAR is not affected by the proposed change.

Core Operating Limits Report

This activity does not impact the COLR (GGNS Core Operating Limits Report). This updates the core damage assessment procedure. It does not impact the COLR since it makes no changes in the operation of the plant, does not impact the type of fuel used, and does not impact the limits defined in the COLR.

NRC Safety Evaluation Report

As discussed under Operating License above, this revision is consistent with Amendment 158 and the Generic BWROG SER performed for eliminating the use of PASS to assess core damage. These requirements supercede the core damage requirements described in NUREG 0737 and the

Quality Assurance Program Manual

This revision complies with all requirements of the Entergy Quality Assurance Program Manual, as applicable. This activity does not change any commitments contained in the QAPM. Therefore, this activity does not require a change to the QAPM. This activity complies with all requirements of the Entergy Quality Assurance Manual (QAPM). This does not affect any commitments contained in the QAPM.

Emergency Plan

Discretionary EAL 4.1 (GE) contains the requirement "Core damage is predicted to occur (within 2 hours). The revised procedure provides an enhanced assessment methodology which is expected to improve the ability to make the required projection. This enhancement does not affect the ability to meet this EAL requirement and does not otherwise impact the Emergency Plan. Therefore, the Emergency Plan is not affected.

Fire Protection Program

This revision does not impact the fire protection program. Therefore, no change is required.

Off Site Dose Calculations Manual

This revision does not impact routine releases or any equipment required to monitor offsite dose. Therefore, no changes to the ODCM are required.

C. References

Discuss the methodology for performing the LBD search. State the location of relevant licensing document information and explain the scope of the review such as electronic search criteria used (e.g., key words) or the general extent of manual searches per Section 5.3.6.4 of LI-101. NOTE: Ensure that electronic and manual searches are performed used controlled copies of the documents. If you have any questions, contact your site Licensing department.

LBDs / Documents reviewed via keyword search:	Keywords:
UFSAR, Technical Specifications, TRM, QAPM, SERs and supplements	Core damage, damage procedure, 0737, emergency

LBDs / Documents reviewed manually:

Amendment 158 SER, Generic BWROG SER

D. **Is the validity of this Review dependent on any other change?**

(See Section 5.3.4 of the EOI 10CFR50.59 Program Review Guidelines.)

☐ Yes
■ No

If "Yes," list the required changes.

III. ENVIRONMENTAL SCREENING

If any of the following questions is answered "yes," an Environmental Review must be performed in accordance with NMM Procedure EV-115, "Environmental Evaluations," and attached to this 50.59 Review. Consider both routine and non-routine (emergency) discharges when answering these questions.

Will the proposed Change being evaluated:

	Yes	No	
1.	☐	■	Involve a land disturbance of previously disturbed land areas in excess of one acre (i.e., grading activities, construction of buildings, excavations, reforestation, creation or removal of ponds)?
2.	☐	■	Involve a land disturbance of undisturbed land areas (i.e., grading activities, construction, excavations, reforestation, creating, or removing ponds)?
3.	☐	■	Involve dredging activities in a lake, river, pond, or stream?
4.	☐	■	Increase the amount of thermal heat being discharged to the river or lake?
5.	☐	■	Increase the concentration or quantity of chemicals being discharged to the river, lake, or air?
6.	☐	■	Discharge any chemicals new or different from that previously discharged?
7.	☐	■	Change the design or operation of the intake or discharge structures?
8.	☐	■	Modify the design or operation of the cooling tower that will change water or air flow characteristics?
9.	☐	■	Modify the design or operation of the plant that will change the path of an existing water discharge or that will result in a new water discharge?
10.	☐	■	Modify existing stationary fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
11.	☐	■	Involve the installation of stationary fuel burning equipment or use of portable fuel burning equipment (i.e., diesel fuel oil, butane, gasoline, propane, and kerosene)?[1]
12.	☐	■	Involve the installation or use of equipment that will result in an air emission discharge?
13.	☐	■	Involve the installation or modification of a stationary or mobile tank?
14.	☐	■	Involve the use or storage of oils or chemicals that could be directly released into the environment?
15.	☐	■	Involve burial or placement of any solid wastes in the site area that may effect runoff, surface water, or groundwater?

[1] See NMM Procedure EV-117, "Air Emissions Management Program," for guidance in answering this question.

Title: Core Damage Assessment	No.: 10-S-01-35	Revision: 2	50.59 Evaluation

IV. SECURITY PLAN SCREENING

If any of the following questions are answered "yes," a Security Plan review must be performed by the Security Department to determine actual impact to the Plan and the need for a change to the Plan.

A. Could the proposed activity being evaluated:

	Yes	No	
1.	☐	■	Add, delete, modify, or otherwise affect Security department responsibilities (e.g., including fire brigade, fire watch, and confined space rescue operations)?
2.	☐	■	Result in a breach to any security barrier(s) (e.g., HVAC ductwork, fences, doors, walls, ceilings, floors, penetrations, and ballistic barriers)?
3.	☐	■	Cause materials or equipment to be placed or installed within the Security Isolation Zone?
4.	☐	■	Affect security lighting by adding or deleting lights, structures, buildings, or temporary facilities?
5.	☐	■	Modify or otherwise affect the intrusion detection systems (e.g., E-fields, microwave, fiberoptics)?
6.	☐	■	Modify or otherwise affect the operation or field of view of the security cameras?
7.	☐	■	Modify or otherwise affect (block, move, alter) installed access control equipment, intrusion detection equipment, or other security equipment?
8.	☐	■	Modify or otherwise affect primary or secondary power supplies to access control equipment, intrusion detection equipment, other security equipment, or to the Central Alarm Station or the Secondary Alarm Station?
9.	☐	■	Modify or otherwise affect the facility's security-related signage or land vehicle barriers, including access roadways?
10.	☐	■	Modify or otherwise affect the facility's telephone or security radio systems?

The Security Department answers the following questions if one of the questions was answered "yes."

B. Is the Security Plan actually impacted by the proposed activity?

☐ Yes

☐ No

C. Is a change to the Security Plan required?

☐ Yes Changed # (optional) __________

☐ No

N/A		N/A		N/A
Name of Security Plan reviewer (print)	/	**Signature**	/	**Date**

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RPTS FORM

10CFR50.59 Review Required?	(X) Yes	If Yes, attach 50.59 Review Form
	( ) No	Not required per LI-101

Cross-Discipline review required?	(X) Yes	(Note affected Departments Below)
	( ) No	
Preparer Initials>>>	WEL	

Department Cross-Discipline Reviews Needed	**Signoff** (signed, electronic, telcon)
Emergency Planning	wsk
Nuclear Safety and Regulatory Affairs	RRG

Does this directive contain Tech Spec Triggers? ( ) YES (X) NO

REQUIREMENTS CROSS-REFERENCE LIST

Requirement Implemented by Directive		Directive Paragraph Number That Implements Requirement
Name	Paragraph Number	
NEDC-33045P[+]	5.0	*

+ - Endorsed by the NRC's Safety Evaluation dated June 12,2001, for BWROG Topical Report NEDO-32991 as referenced in GGNS License Amendment 158.

* - Covered by directive as a whole or by various paragraphs of the directive.

> NOTE
>
> The Component Database Change Request statement is applicable only to Volume 06 and 07 maintenance directives.

Component Database Change Request generated and the backup documentation available for setpoint and/or calibration data only ☐ Yes ☒ N/A CDBCR # ________

Current Revision Statement

This is complete rewrite of the previous revision to remove the use of the PASS system for core damage assessments and to incorporate an updated core damage assessment methodology described in NEDC 33045P.

1. DOCUMENT INFORMATION

Procedure/Document Number: 10-S-01-35	Proc./Doc. Revision: 2
Document Title: Core Damage Assessment	
Brief Description of Proposed Revision: This revision removes the Post Accident Sampling System (PASS) as an input to Core Damage Assessment procedure and incorporates the updated GE methodology described in NEDC-33045P to obtain core damage estimates.	

2. SCREENING

A.) Does the proposed revision require a change to the Emergency Plan, change a commitment in the E-Plan, or potentially decrease the effectiveness of the E-Plan?
____ **Yes** _X_ **No**

B.) Does the proposed revision change the Emergency Action Levels (EALs)?
____ **Yes** _X_ **No**

If the answer to either question is YES, then a 10CFR50.54(q) evaluation must be performed. See attachment 9.2. If the answer is NO provide justification below.

3. Justification for "NO" Answers

Reviewed the applicable portions of the Emergency Plan. . The information from PASS samples is not considered to be beneficial for accident management or emergency response during the early phases of a nuclear emergency. PASS samples are more appropriately used in the recovery phase of an emergency to formulate and validate long term plans. As discussed in the generic SER for BWROG Topical Report NEDO -32991 other real time plant indicators provide adequate information for assessing core damage, therefore this procedure revision does not decrease the effectiveness of the Emergency Plan. This procedure does not contain any EAL information and therefore does not change the GGNS EALs.

4. APPROVAL

Screening Prepared By: William E. [signature] 12-9-03
Signature Date

Screening Reviewed By: DKTownsend 12-9-03
Signature Date

Screening Approved By: [signature] 12-9-03
Signature Date

1.0 PURPOSE

The purpose of this procedure is to provide a simple qualitative evaluation of the overall status of the reactor core under severe accident conditions. The results of the evaluation support appropriate mitigative and protective actions while an event is in progress.

2.0 RESPONSIBILITIES

2.1 Accident Assessment Engineer - Is responsible for the performance of this procedure, core damage estimate calculations, and providing core damage assessment to the Offsite Emergency Coordinator.

2.2 TSC Coordinator - Is responsible for monitoring plant conditions for indication of core damage.

3.0 REFERENCES

3.1 NEDC-33045P, "Methods of Estimating Core Damage in BWRs", Revision 0, General Electric Co., July 2001.

3.2 RTM-96, "Response Technical Manual", Volume 1, Revision 4, USNRC, March 1996.

3.3 Calculation XC-Q1111-99001, Plant Specific Technical Guidelines, Calculation for Plant-Specific Variables and Graphs, Revision 3.

3.4 05-S-01-EP-2, "RPV Control", Revision 34.

3.5 Calculation JC-Q1E61-R602-1, Drywell and Containment Hydrogen Concentration Loop Uncertainty, Revision 0.

4.0 ATTACHMENTS

4.1 ATTACHMENT I - Core Cooling Assessment
4.2 ATTACHMENT II - Drywell Radiation Levels

4.3 ATTACHMENT III - Containment Radiation Levels

4.4 ATTACHMENT IV - Volume-Weighted Average Hydrogen Concentration

4.5 ATTACHMENT V - Zirconium Oxidation Fraction

4.6 ATTACHMENT VI - Core Damage Tracking Data

5.0 DEFINITIONS

5.1 CORE DAMAGE - Widespread degradation of the fuel pellet or cladding fission product barriers due to inadequate core cooling. Core damage is characterized by the combination of cladding and overheating damage.

5.2 NO DAMAGE - A condition in which there is no positive indication of core damage has been detected. Fuel cladding temperatures below 1500°F and fission product releases are limited to the radionuclides normally present in the reactor coolant

5.3 CLADDING DAMAGE - Degradation of the fuel rod cladding permitting the release of gaseous and volatile fission products from the space between the fuel pellets and the fuel cladding. Significant cladding damage is expected to begin at temperatures of approximately 1500°F.

5.4 OVERHEATING DAMAGE - Damage to the fuel from temperatures high enough to release gaseous and volatile fission products from the fuel matrix but insufficient to melt the fuel.

5.5 FUEL MELT - Damage that occurs at higher temperatures and results in changes to the core geometry. This phase of core damage is not addressed by this procedure since no mitigative actions are based on the onset of melting and it is not expected that the condition would be easily detectable by plant personnel.

5.6 Minimum Steam Cooling RPV Water Level (MSCRWL) - The lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to prevent any clad temperature in the uncovered part of the core from exceeding 1500°F (-192 inches).

5.7 SPIKED COOLANT RELEASE - The release into containment of 100 times the non-noble gas fission products normally found in the coolant. Spikes are normally associated with a rapid shutdown or depressurization.

6.0 DETAILS

6.1 General Directions

6.1.1 The type and extent of core damage are estimated based on assessments of core cooling history, primary containment radiation, and primary containment hydrogen concentration. The initial screening identifies conditions in which core damage may be possible. Primary containment radiation levels provide the primary indicator with core cooling and hydrogen concentration providing confirmatory indications.

> NOTE
>
> Interpolate or extrapolate from the predicted levels given in Attachments II & III to account for decay between sub-criticality and when readings are taken.

6.1.2 Check for indication of possible core damage using Table 1 below. Make adjustments to the predicted radiation levels, as necessary.

Table 1
Indication of Possible Core Damage

Condition	Indications
Core Cooling Assessment	See Attachment I
Primary containment radiation.	Drywell and Containment radiation levels above the maximum predicted coolant release values for the Spiked Coolant release - See Attachments II and III.
Hydrogen generation	Drywell or containment hydrogen concentration above 0.8%.

6.1.3 The core damage assessment is expected to be performed many times during the course of an accident. Use Attachment VI to record the results of these assessments.

6.1.4 If core damage is indicated, evaluate the type and amount of damage using Table 2 below.

NOTE

Evaluate drywell and containment radiation levels independently.

Table 2
Preliminary Core Damage Assessment

Drywell/Containment Radiation (Attachments II and III)	**Possible Damage States**	**Action**
Below 1% Gap Release (cladding failure) radiation levels	Core damage unlikely	None
Between 1% Gap Release (cladding failure) radiation levels and 1% Overheating Release radiation levels.	Limited cladding damage and Overheating damage unlikely	Estimate the amount of cladding damage using Section 6.2.
Above 1% Overheating Release radiation levels.	Widespread cladding damage and Possible overheating damage	Estimate the amount of cladding damage using Section 6.2. and Estimate the amount of overheating damage using Section 6.3.
Above 100% Gap Release (cladding failure) radiation levels.	Widespread cladding damage and Widespread overheating damage	Verify the amount of cladding damage using Section 6.2. and Estimate the amount of overheating damage using Section 6.3.

6.2 Estimating Cladding Damage

6.2.1 Indications

a. Drywell or containment radiation level above the 1% Gap Release (cladding failure) radiation level.

b. Drywell or containment hydrogen concentration above 0.8%.

c. RPV water level below MSCRWL (-192 inches)

d. Inadequate core spray (HPCS not injecting and Low Pressure Core Spray flow less than design (7000 gpm)) and RPV water level below -217 inches

6.2.2 Procedure

a. Determine the predicted drywell and containment radiation levels corresponding to 100% Gap Release (cladding failure) from Attachments II and III using an appropriate time adjustment.

6.2.2 (Cont.)

b. Estimate the amount of cladding damage as follows:

> NOTE
>
> Perform separate calculations for the drywell and containment. Use the higher of the two results.

$$\%\,Cladding\ \ Damage = \frac{Indicated\ \ Radiation\ \ Level}{100\%\ Gap\ \mathrm{Re}lease\ \ Radiation\ \ Level} \times 100$$

> NOTE
>
> If igniter operation or other hydrogen depleting mechanisms have occurred (see 7.3.3), Attachment IV and VI will underestimate the amount of zirconium oxidation.

c. If Hydrogen is present, estimate the amount of zirconium oxidation using Attachment IV or V.

d. Refer to Section 7 for guidance on interpreting the damage assessment results.

0-10% damage = Limited Damage

>10% damage = Widespread Damage

6.3 Estimating Overheating Damage

6.3.1 Indications

a. Drywell or containment radiation level above the 1% Overheating Release radiation level. (Attachments II and III).

b. Drywell or containment hydrogen concentration above 2.0%.

c. RPV water level below -217 inches and Core Spray flow less than design (7000 gpm) for more than 30 min.

6.3.2 Procedure

a. Determine the drywell and containment radiation levels corresponding to 100% Overheating Release from Attachments II and III.

b. Estimate the amount of overheating damage as follows:

> NOTE
>
> Perform separate calculations for the drywell and containment. Use the higher of the two results.

6.3.2 (Cont.)

$$\%OverheatingDamage = \frac{IndicatedRadiationLevel}{100\%Overheating\,ReleaseRadiationLevel} \times 100$$

NOTE

If igniter operation or other hydrogen depleting mechanisms have occurred (see 7.3.3), Attachment IV and VI will underestimate the amount of zirconium oxidation.

c. Estimate the amount of zirconium oxidation using Attachment IV or V. Use Attachment V if a more accurate estimate is needed.

d. Refer to Section 7 for guidance on interpreting the damage assessment results.

0-10% damage = Limited Damage

>10% damage = Widespread Damage

7.0 DISCUSSION

7.1 General

7.1.1 This procedure provides a general evaluation of the status of the core. The accuracy of the results is governed by source term uncertainties, instrumentation characteristics, and event-specific variations.

7.1.2 Use appropriate engineering judgment to reconcile any differences between assessment results and obtain a best estimate of core damage.

7.1.3 Due to the power distribution in the core during operation, core damage is expected to occur at different rates in different parts of the core. Overheating damage may exist in the hottest regions of the core while fuel in lower power regions remains undamaged.

7.2 Radiation Levels (Attachments II and III)

7.2.1 A radiation level equivalent to 1% of the gap release inventory is positive indication of cladding damage.

7.2.2 The cladding damage and overheating damage radiation ranges overlap. Within the overlap range, the relative amounts of each type of damage cannot be distinguished. Calculating separate values for each type of damage tends to overestimate both amounts.

7.2.3 A radiation level above the 1% overheating damage value is not positive indication of overheating damage since the same radiation level could be reached with only cladding damage. However, it is unlikely that widespread cladding damage would exist without at least some overheating damage. Above the 50% cladding damage radiation level, it may be assumed that overheating damage also exists.

7.2.4 The relative values of drywell and containment radiation levels are event-dependent. If a primary system break exists and steam is discharged into the drywell, drywell radiation levels will probably increase first. If no primary system break exists and steam is discharged through the SRVs, containment radiation level will probably increase first.

7.3 Hydrogen

7.3.1 Rapid hydrogen production begins at temperatures above the threshold for cladding damage but below the onset of significant overheating damage.

7.3.2 Detectable hydrogen is an indication of possible cladding damage.

7.3.3 Hydrogen removal mechanisms including igniter operation in the drywell or containment, spurious hydrogen burns, or containment venting are expected to significantly reduce the hydrogen concentration and reduce the containment oxygen concentration. Since containment oxygen concentrations are not available, it is impossible to assess the impact of these removal mechanisms on the Hydrogen concentration. As a result, hydrogen concentration can only be correlated to core damage when igniter operation or other removal mechanisms have not occurred.

7.3.4 If no hydrogen is detected, it is unlikely that overheating damage exists.

7.4 Uncertainties

7.4.1 Radiation level measurements may underestimate core damage if:

a. The primary containment or RPV has been vented.

b. Primary system isolations have been defeated to permit continued use of the main condenser under failure-to-scram conditions.

c. Primary containment integrity has been lost.

7.4.2 Radiation level measurements may overestimate core damage if:

a. The suppression pool has been bypassed.

b. Suppression pool water level is low.

7.4.3 Hydrogen concentration measurements may underestimate core damage if:

a. Hydrogen igniters have operated or a spurious hydrogen burn has occurred.

b. The primary containment has been vented.

c. Primary containment integrity has been lost.

d. Significant amounts of hydrogen remain trapped in the RPV.

7.4.4 Hydrogen concentration measurements may overestimate core damage if:

a. Significant amounts of hydrogen have been generated by radiolysis.

b. The hydrogen injection system is leaking.

c. Steam is present in the drywell but the drywell atmosphere is not at saturation conditions.

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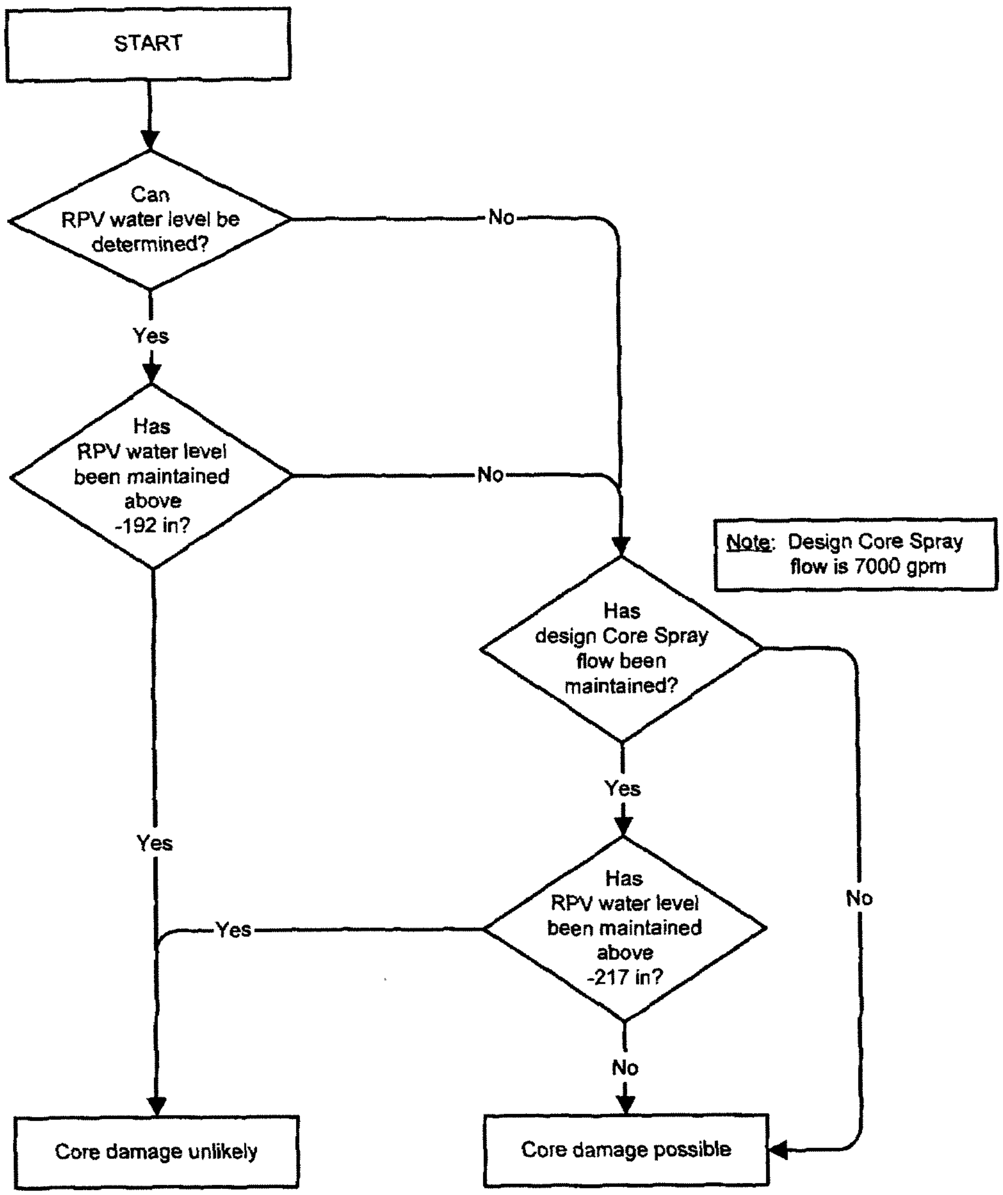
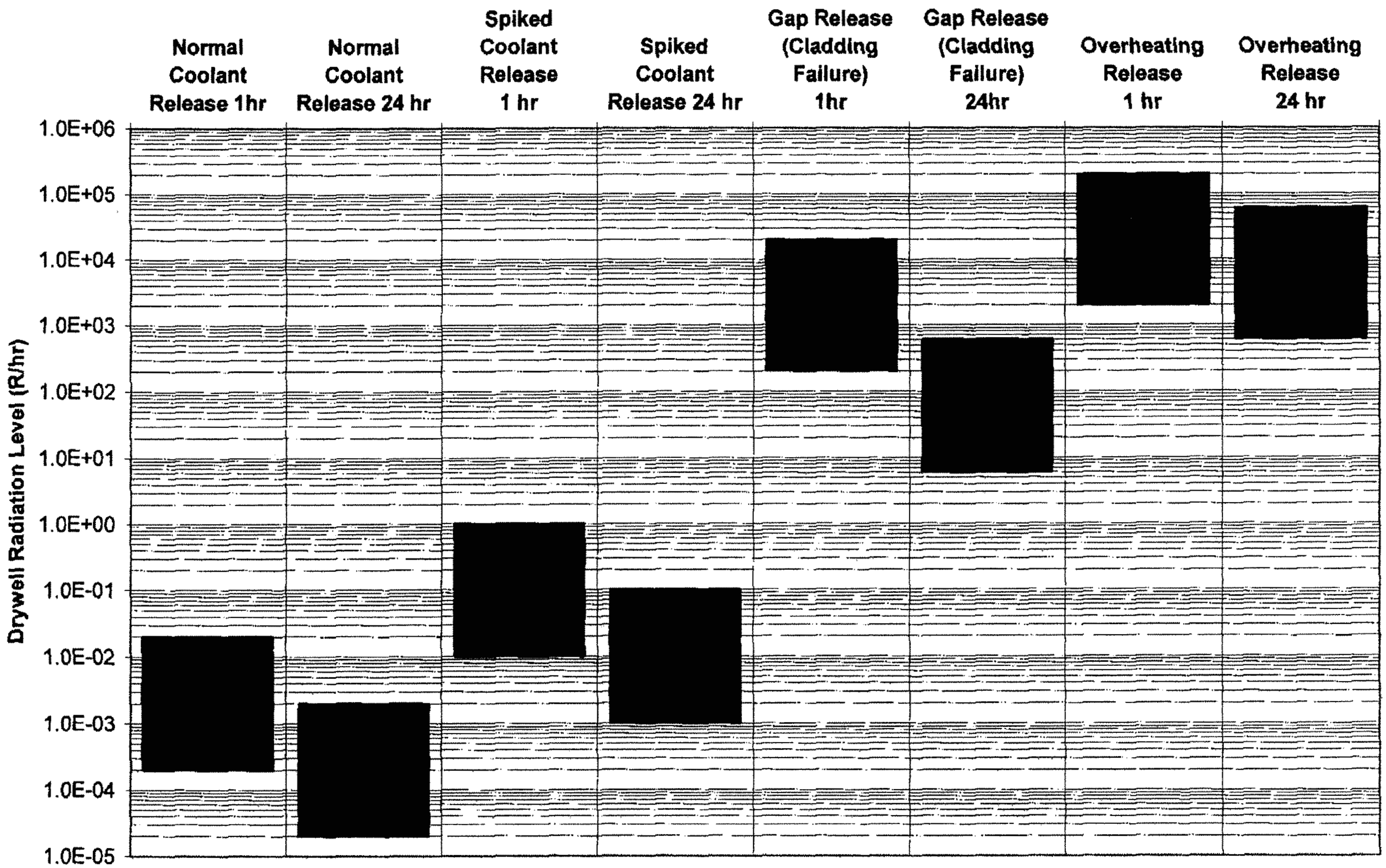
Attachment I	Page 1 of 1



Attachment I: Core Cooling Assessment

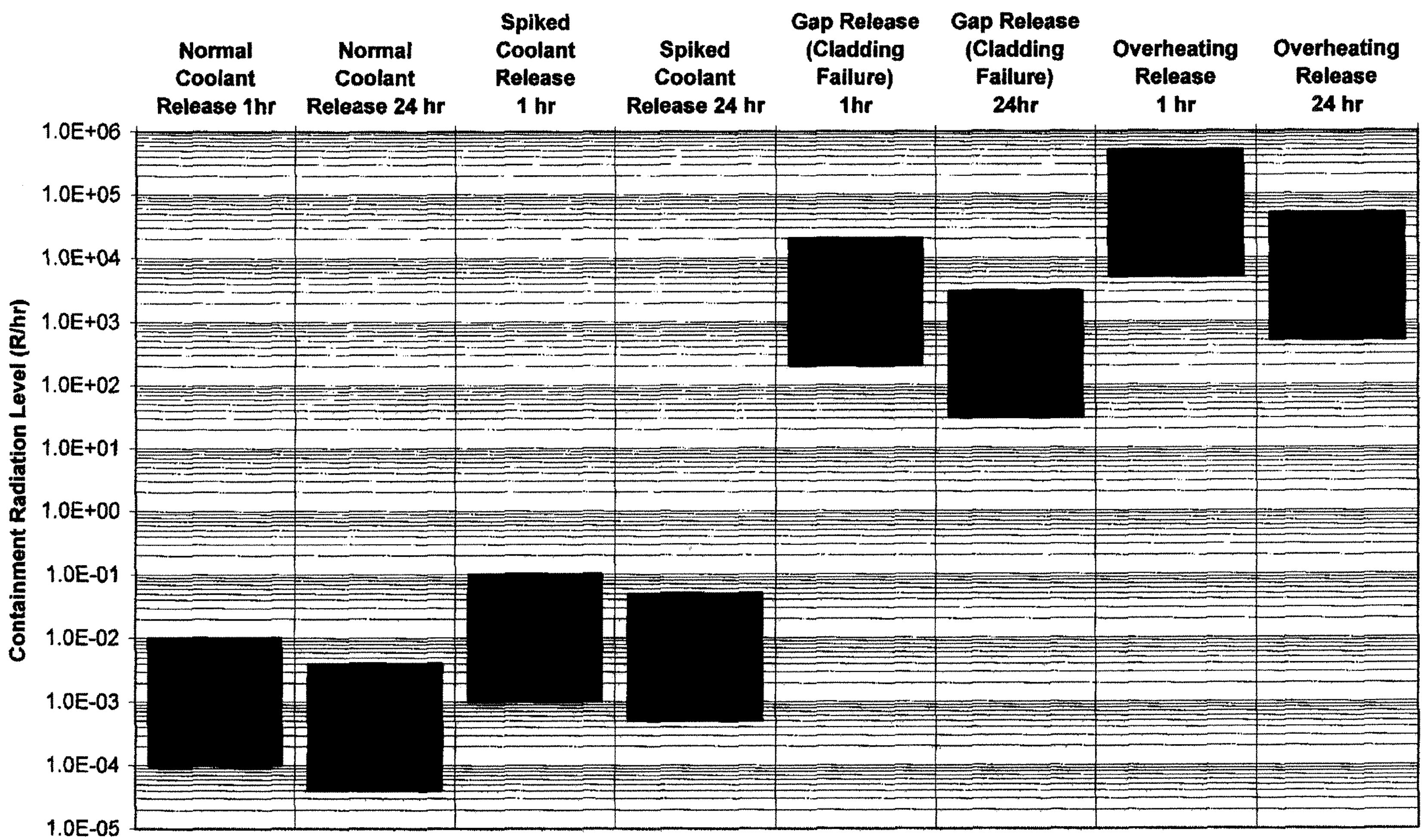
Attachment II

Drywell Radiation Levels



Shaded areas represent 1% (low value) to 100% (high value) of inventory release to the drywell

Attachment III
Containment Radiation Levels



Shaded areas represent 1% (low value) to 100% (high value) of inventory release to the containment

Attachment IV:

Volume-Weighted Average Hydrogen Concentration

Use this worksheet to calculate the volume-weighted average containment hydrogen concentration.

Drywell hydrogen concentration (%): H_{dw} = ________________%

Containment hydrogen concentration (%): H_{sc} = ________________%

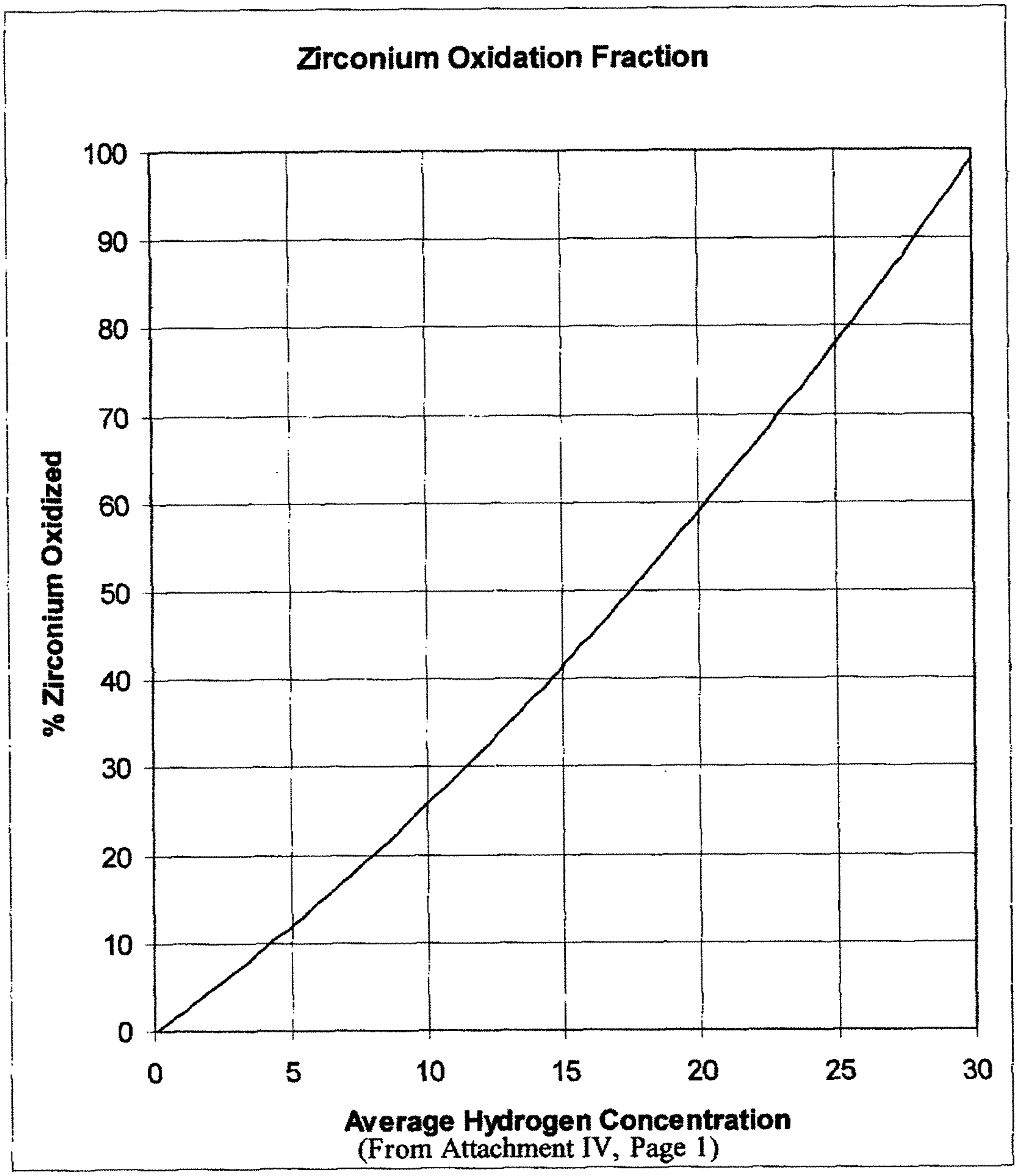
Drywell airspace free volume (ft^3): V_{dw} = 270,000 ft^3

Containment airspace free volume (ft^3): V_{sc} = 1,400,000 ft^3

Volume-weighted average hydrogen concentration:

$$H_{AVG} = \frac{H_{dw}(270{,}000) + H_{sc}(1.4E06)}{1.67E6} = \text{\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_}\ \%$$

Determine % Zr Oxidation from the graph on page 2 of this attachment.



Attachment IV: Volume-Weighted Average Hydrogen Concentration

Attachment V
Zirconium Oxidation Fraction

Use this worksheet to verify estimates obtained from Attachment IV.

1. Calculate the mass of oxidized zirconium based on the drywell hydrogen concentration:

Drywell hydrogen concentration (%):	H_{dw} = ________________%
Drywell airspace free volume** (ft^3):	V_{dw} = 270,000 ft^3
Drywell pressure (psig):	P_{dw} = ________________psig
Drywell temperature (°F):	T_{dw} = ________________°F
Saturation pressure for T_{dw} (psia):	P_{sat}^{dw} = ________________psia

Mass of oxidized zirconium in the drywell:

$$m_{Zr}^{dw} = 0.04242 \times \frac{H_{dw}(270{,}000)(P_{dw} + 14.7 - P_{sat}^{dw})}{(460 + T_{dw})} = \text{\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_ lbm}$$

2. Calculate the mass of oxidized zirconium based on the containment hydrogen concentration:

Containment hydrogen concentration:	H_c = ________________%
Containment airspace free volume**:	V_c = 1,400,000 ft^3
Containment pressure	P_c = ________________psig
Containment temperature	T_c = ________________°F
Saturation pressure for T	P_{sat}^{c} = ________________psia

Mass of oxidized zirconium in the containment:

$$m_{Zr}^{c} = 0.04242 \times \frac{H_c(1.4E06)(P_c + 14.7 - P_{sat}^{c})}{(460 + T_c)} = \text{\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_ lbm}$$

3. Calculate the total zirconium oxidation fraction:

$$F_{Zr} = \frac{m_{Zr}^{dw} + m_{Zr}^{c}}{71007} \times 100 = \text{\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_\_ \%}$$

** - Values based on normal water inventory. Adjust to account for significant flooding of either the drywell or containment.

Attachment VI
Core Damage Tracking Data

DATE: ______________

Cladding Damage

Time (24 hr)	Cladding Damage Indicated? (6.2.1)	100% Gap Release (cladding failure) Radiation Level (6.2.2.a) DW/Cont	Radiation levels (R/hr) DW/Cont	Cladding Damage Assessment (%) (6.2.2.b)	Hydrogen Concentration (%) DW/Cont	Zirconium Oxidation (%) (6.2.2.c)	Overall Damage Estimate (%)[1] (6.2.2.d)
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		

[1] 0-10% damage = Limited Damage; >10% damage = Widespread Damage

Attachment VI
Core Damage Tracking Data

DATE:__________________

Overheating Damage

Time (24 hr)	Overheating Damage Indicated? (6.3.1)	100% Damage Radiation Level (6.3.2.a) DW/Cont	Radiation levels (R/hr) DW/Cont	Overheating Damage Assessment (%) (6.3.2.b)	Hydrogen Concentration (%) DW/Cont	Zirconium Oxidation (%) (6.3.2.c)	Overall Damage Estimate (%)[1] (6.3.2.d)
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		
	YES / NO	/	/		/		

[1] 0-10% damage = Limited Damage; >10% damage = Widespread Damage