53450N

# CONTROLLED DOCUMENT TRANSMITTAL

53450N

Transmittal#: 53450N Date: 07/30/2003 Creator: TRACY NELSON Page 1

**Description:** ISSUE OF 1 PMP-EPP SERIES PROCEDURE

**Distribution Group(s):** Procedures: EPP: PMP-2081-EPP-105

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| Document | Revision | Title |
|---|---|---|
| PROCEDURE | | SEE ATTACHED |

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A045

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# Search

## Results of Last Search

**Total of 1 items found**

PMP-2081-EPP-105 Properties Actions Edit

**Revision:** 005
**AEP Status:** Approved
**Title:** INITIAL CORE DAMAGE ASSESMENT
**Document Series:** Procedures
**Document Type:** Emergency Planning - Response
**Approval/Record Date:** 07/22/2003
**Effective Date:** 07/30/2003

Top of Page

# REVIEW AND APPROVAL TRACKING FORM

**Procedure Information:**

Number: PMP-2081-EPP-105 Rev. 5 Change: 0

Title: Core Damage Assessment

**Category (Select One Only):**

- ☐ Correction (Full Procedure)
- ☒ Change (Full Procedure) with Review of Change Only
- ☐ Correction (Page Substitution)
- ☐ Change (Page Substitution) with Review of Change Only
- ☐ Cancellation
- ☐ New Procedure or Change with Full Review
- ☐ Superseded (list superseding procedures): ______

**Associated Configuration Impact Assessments:**

Change Driver/CDI Tracking No(s).: ______ ☒ N/A

**Required Reviews:**

Cross-Discipline Reviews:

- X Chemistry
- ☐ Training
- ☐ Maintenance
- ☐ Work Control
- ☐ NDM
- ☒ Rx. Eng.
- ☐ Operations
- ☐ ______
- ☐ PA/PV
- ☐ ______
- ☐ Reg Affairs
- ☐ ______
- ☐ RP
- ☐ None Required

Programmatic Reviews:

- ☐ ALARA
- ☐ Reactivity Mgmt Team
- ☐ Component Engineering
- ☐ SPS (Safety & Health)
- ☐ Design Engineering
- ☐ Surveillance Section
- ☐ Emerg Oper Proc Grp
- ☐ System Engineering
- ☐ Environmental
- ☐ SOMS Administrator
- ☐ ISI/IST Coordinator
- ☐ ______
- ☐ Performance Assurance
- ☒ None Required

☒ Cognizant Org Review: BK Malloy Date: 5/12/03

☒ Technical Review: Darrell Schroeder Date: 5/12/03

**Concurrence:**

☐ Ops Director Concurrence: N/A Date: __/__/__

**Package Check:**

Updated Revision Summary attached? ☒ Yes

10 CFR 50.59 Requirements complete? Tracking No.: 2003-0842-00 ☒ Yes ☐ N/A

Implementation Plan developed? (Ref. Step 3.4.17) ☐ Yes ☒ N/A

Package Complete: Cindy Shaffner Date: 7/21/03

**Approvals:**

PORC Review Required: ☒ Yes ☐ No Mtg. No.: 4020 [signature]

Administrative Hold Status: ☐ Released ☐ ~~Reissued~~ ☒ N/A CR No.: N/A

Approval Authority Review/Approval: [signature] Date: 7/24/03

Expiration Date/Ending Activity N/A Effective Date: 7/30/03

**Periodic Review:**

Periodic Review conducted? (Data Sheet 5 Complete) ☐ Yes ☒ No

**Follow-up Actions:**

Commitment Database update requested in accordance with PMP-7100-CMP-001? ☐ Yes ☒ N/A

NDM notified of new records or changes to records that could affect record retention? ☐ Yes ☒ N/A

NDM Use Only

NUCLEAR DOCUMENT MANAGEMENT SECTION

JUL 30 2003

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**Office Information For Form Tracking Only – Not Part of Form**

This form is derived from the information in PMP-2010-PRC-002, Procedure Correction, Change, and Review, Rev. 10a, Data Sheet 1, Review and Approval Tracking Form.

## REVISION SUMMARY

Number: PMP-2081-EPP-105 Revision: 5 Change: 0

Title: Core Damage Assessment

| # | Section or Step | Change/Reason For Change |
|---|---|---|
| 1. | Entire Procedure | Change: Reformatted to PMP-2010-PRC-001 required format. (Rev. 4 Steps/Sections shown in parentheses); deleted references to Post-Accident Sampling System and associated nomenclature throughout the procedure; marginal markings not used.<br>Reason: No longer necessary due to deletion of Post Accident Sampling System (PASS) requirements per TS Amendments 261 (Unit 1 and 244 (Unit 2); marginal markings not used due to extensive reformatting and other changes. |
| 2. | Procedure Title (Section 1) | Change: Removed "Initial" from the phrase Core Damage Assessment<br>Reason: Unnecessary word in title. |
| 3. | Section 1 Purpose and Scope (Section 2 Objective and Section 4 Responsibilities) | Change: Deleted references to fission products and containment hydrogen; incorporated responsibilities into this section and reassigned duties to the PET Reactor Physics Analyst (formerly a PET Chemist function).<br>Reason: No longer necessary due to deletion of Post Accident Sampling System (PASS) requirements per TS Amendments 261 (Unit 1 and 244 (Unit 2); TSC Chemistry position was eliminated--Rx Physics Analyst is qualified to perform this function. |
| 4. | Section 2 Definitions and Abbreviations | Change: New section.<br>Reason: Required for PMP format. |
| 5. | Section 3 Details (Section 5 Limitations and Precautions, and Section 6 Prerequisites) | Change: Removed references to PASS, samples, nuclides, hydrogen measurement, Core Damage Assessment computer program, core inventory, and other references to post-accident sampling.<br>Reason: No longer necessary due to deletion of Post Accident Sampling System (PASS) requirements per TS Amendments 261 (Unit 1 and 244 (Unit 2) |
| 6. | Step 3.4.1.c (formerly 7.1.3) | Change: Deleted all but the first sentence.<br>Reason: The remainder of the step described use of Appendix A.2 which was eliminated as redundant and therefore unnecessary. |
| 7. | Step 3.4.1.d (formerly 7.1.4) | Change: Deleted "core damage state(s)" and substituted "% core damage".<br>Reason: Allows for reporting percentage of core damage rather than a description of core damage state (e.g., <10% vs. normal coolant). |

## REVISION SUMMARY

Number: PMP-2081-EPP-105 Revision: 5 Change: 0

Title: Core Damage Assessment

| # | Section or Step | Change/Reason For Change |
|---|---|---|
| 8. | Step 3.4.2 | Change: Added clarification on location of Core Exit Thermocouple readings on PPC.<br>Reason: To aid in locating applicable PPC window. |
| 9. | Step 3.5 | Change: New section that directs the user to compare the two CDA estimates and report results based on the comparison.<br>Reason: To allow the Reactor Physics Analyst the flexibility to use engineering judgement in validation of results based upon the plant conditions and events in progress. |
| 10. | Step 3.6 | Change: New step that cycles the user to re-perform applicable steps of the procedure for subsequent core damage assessments.<br>Reason: Allows for subsequent core damage assessments during the emergency. |
| 11. | Attachment 1 (Appendix B) | Change: Renamed Appendix B as Attachment 1 for this revision.<br>Reason: Formatting per PMP-2010-PRC-001. |
| 12. | Data Sheet 1 | Change: New data sheet.<br>Reason: Incorporates data recording on one sheet instead of 2 (formerly Appendix A.1 and Attachment 1). |
| 13. | Data Sheet 2 | Change: New data sheet.<br>Reason: Incorporates elements of two former appendices (Appendix E.2 and E.3) to determine power correction factor on one work sheet; simplifies determination of power correction factor. |

## REVISION SUMMARY

Number: PMP-2081-EPP-105 Revision: 5 Change: 0

Title: Core Damage Assessment

| # | Section or Step | Change/Reason For Change |
|---|---|---|
| 14. | (Deleted Sections/Steps: 7.3, 7.4, 7.5, page 2 of Attachment 1, Appendix A.2, page 3 of Appendix A.3, Appendix C.1, C.2, C.3, C.4, C.5, C.6, D.1, D.2, D.3, D.4, D.5, D.6, D.7, D.8, E.1, E.2, E.3, E.4, E.5, E.6, F.1, F.2, F.3, F.4, F.5, and F.6) | Change: (Deleted sections/appendices)<br><br>Reason: (Steps or appendices not required due to PASS elimination or for the following reasons:<br><br>Step 7.5 which discussed 'final' CDA and engineering judgement is discussed as applicable in steps 3.5, 3.6, and Section 4 of this revision. Appendix A.2 was used in a method of core damage assessment that duplicated the method currently used in the containment radiation monitor methodology, therefore, it was unnecessary.) |

# Core Damage Assessment

| Information | | Effective Date: 9/30/03 |
|---|---|---|
| C. J. Graffenius | S. M. Partin | Emergency Planning |
| Writer | Owner | Cognizant Organization |

## TABLE OF CONTENTS

## 1 PURPOSE AND SCOPE

1.1 The purpose of this procedure is to provide a method to estimate the extent of core damage through measurement of core exit thermocouple temperature, water level within the pressure vessel, and containment radiation monitors. It is to be used for actual Emergency Plan response, as well as during Emergency Plan drills or exercises.

1.2 Discussion

- Estimations of post accident core damage can be determined through a correlation of containment atmosphere radiation monitor readings to the appropriate NRC category of core damage.
- Estimations of post accident core damage can be determined through a correlation of reactor coolant core exit thermocouple temperature to NRC fuel damage categories.

1.3 Responsibilities

- The TSC Reactor Physics Analyst will be responsible for assessment of core damage. Other Plant Evaluation Team members may act as an alternate in the event the Reactor Physics Analyst is not available.

## 2 DEFINITIONS AND ABBREVIATIONS

| Term | Meaning |
|---|---|
| CDA | Core Damage Assessment |
| CET | Core Exit Thermocouple |
| CRM | Containment Radiation Monitor |
| LOCA | Loss of Coolant Accident |
| NSS | Nuclear Steam Supply |
| PET | Plant Evaluation Team |
| PPC | Plant Process Computer |
| RCS | Reactor Coolant System |
| SEC | Site Emergency Coordinator |
| T/C or TC | Thermocouple |

| Term | Meaning |
|---|---|
| 1-VRA-1310 | Unit 1 High Range Upper Containment Radiation Monitor |
| 1-VRA-1410 | Unit 1 High Range Lower Containment Radiation Monitor |
| 2-VRA-2310 | Unit 2 High Range Upper Containment Radiation Monitor |
| 2-VRA-2410 | Unit 2 High Range Lower Containment Radiation Monitor |

# 3 DETAILS

## 3.1 Limitations

3.1.1 The results from this procedure have limited accuracy based on the assumptions made in the core damage assessment methodology. Each set of readings (containment high range radiation monitor and/or CET) describes a static event in the system. Multiple static sets of readings over an extended time period will give better indication of the dynamic event.

3.1.2 Depending on plant conditions one method of CDA may be more accurate than the other (i.e., containment radiation monitor readings may present more realistic data than CET data). In this case, once Section 3.4 of this procedure has been fully completed, the Reactor Physics Analyst may choose to return to the applicable steps of Section 3.4 for subsequent assessments of core damage.

## 3.2 Prerequisites (**either** of the conditions listed applies).

3.2.1 Any plant condition in which the operator would suspect defect or failed fuel, and an estimate of the amount of defect or failed fuel is required.

3.2.2 Any plant condition in which an operator would suspect a loss of reactor core cooling or knows reactor core cooling will no longer be maintained.

## 3.3 Background Information on Stages of Core Damage

3.3.1 No Damage

a. Indications of core damage in these categories are halogen spiking and tramp uranium where typically less than one percent of the total core inventory is released to the coolant.

3.3.2 Clad Rupture/Gas Gap Release

a. An increase in reactor coolant noble gas concentration will be observed.

b. In the case of a LOCA, containment radiation monitor indications will be elevated. Containment building pressure and temperature increases are additional indications of continuing leakage from the reactor coolant system.

3.3.3 Grain Boundary

a. Temperatures in the RCS as indicated by incore thermocouples exceed saturation temperature as the level in the core drops and the fuel temperature increases.

b. Containment area monitor indications increase noticeably from normal levels. This indicates probable fuel cladding damage (failure) in the hotter regions of the core releasing fission products from the fuel pellets.

c. The coolant level in the core may continue to decrease if the water is boiling off. Fuel pellet overheating due to increasing temperatures causes additional fission products to diffuse out of the fuel pellets.

d. Containment spray system will be actuated to remove 99% of the elemental radioiodines and air particulates from containment.

3.3.4 Core Melt

a. Further decreases in coolant level result in increasing temperatures. The temperature of the upper portion of the core increases and can reach and exceed the melting point of the zircalloy cladding (typically, zircalloy melts at >2000°F).

b. Continued heating for a still longer period of time causes core uncovery, extensive core damage takes place and the upper, central portion of the core may begin melting.

c. The containment radiation monitors progressively increase and may saturate.

3.4 Estimation of Core Damage State

3.4.1 Estimation of Core Damage Based on Containment Radiation Monitor Indications.

a. Record the time of reactor shutdown, the time after shutdown the containment high range area radiation monitor indications were obtained, the containment radiation monitor number, and containment high range area radiation monitor indications on Data Sheet 1, Core Damage Assessment.

b. Divide the radiation monitor readings by the power correction factor determined in Data Sheet 2, Power Correction Factor Determination and record on Data Sheet 1, Core Damage Assessment.

c. Compare the corrected readings with Attachment 1, Estimation of Core Damage Based on CRM and CET Readings, to estimate the corresponding extent of core damage.

d. Based on containment radiation monitor readings, record the estimated core damage state(s) from Attachment 1, Estimation of Core Damage Based on CRM and CET Readings, on Data Sheet 1, Core Damage Assessment.

3.4.2 Estimation of Core Damage Based on CET Indications

**CAUTION:** If a large break LOCA is suspected or indicated, undetected core heat-up and flashing of cooling water during core recovery will occur. Thermocouple readings may rise sharply, then quench when core recovery commences. In this case, this section would yield <u>low estimates</u> of core damage.

**NOTE:** If a void develops in the upper internals area of the core, the core exit thermocouples may not be immersed in RCS water and can indicate lower temperatures than actually exist in the core. RVLIS (PPC location: **ER/ERDS Data Display/Reactor Vessel Level** in conjunction with # of RCPs running table located on the PPC at: **SPDS/I TREE EOP**) is used to measure RCS water level. The top of the core is at approximately 60% on the narrow range indication. This section yields damage estimates in NRC categories 5 through 10 and is most appropriate for core uncovery with a maximum temperature above the rapid oxidation temperature of 1800°F. A smooth core exit thermocouple trend recording and an uncovery duration 20 minutes or longer are indicators for a good prediction of clad oxidation.

a. Record the hottest CET temperature (on PPC/NSS/Nuclear Steam Parameters or PPC/NSS/T/C MAP1) on Data Sheet 1, Core Damage Assessment.

b. Based on Attachment 1, Estimation of Core Damage Based on CRM and CET Readings, record the estimated % core damage on Data Sheet 1, Core Damage Assessment.

3.5 Data Results

3.5.1 **IF** comparison of the two core damage estimates from Section 3.4 differ by less than 50%, **THEN** report the larger estimate.

3.5.2 **IF** comparison of the two core damage estimates from Section 3.4 differ by greater than 50%, **THEN** consider the validity of the CET estimate based on the CAUTION and NOTE prior to Step 3.4.2.a.

a. **IF** by the judgement of the Reactor Physics Analyst a determination can not be made as to which estimate is accurate, and the estimates differ by greater than 50%, **THEN** report the larger estimate.

3.5.3 Report CDA results to the SEC or Assistant SEC.

3.6 Subsequent CDAs

3.6.1 Repeat applicable steps of Section 3.4 and 3.5 as necessary.

## 4 FINAL CONDITIONS

4.1 Emergency has been terminated or the SEC has determined that CDA is no longer necessary.

4.2 Records generated from this procedure are to be turned over to the SEC.

## 5 REFERENCES

5.1 Use References:

5.1.1 Westinghouse Owner Group NUREG 0737, Item II.B.3 Post Accident Core Damage Assessment Revision 2, to WOG Methodology, November, 1984.

5.2 Writing References:

5.2.1 Source References:

a. D. C. Cook Post Accident Core Damage Assessment Methodology (developed by AEPSC), August, 1984.

5.2.2 General References

a. None

| Characteristics of Categories for Fuel Damage | | | | | | |
|---|---|---|---|---|---|---|
| | Maximum Core Exit Thermocouple Temperature F° | Reference Radiation Monitor R/hr | Percent Damage | Source of Release | NRC Description | NRC Cat # |
| No Damage | Normal | Normal | < 1 | Gas Gap | No Damage | 1 |
| Rupture | Normal to 750 | Normal to 660 | < 10 | Gas Gap | Initial Cladding Failure | 2 |
| | 750 to 1300 | 660 to 990 | 10 – 50 | Gas Gap | Intermediate Cladding Failure | 3 |
| | 1300 to 1650 | 990 to 1325 | > 50 | Gas Gap | Major Cladding Failure | 4 |
| Oxidation | > 1650 | 1325 to 8.6E4 | < 10 | Fuel Pellet | Initial Fuel Pellet Overheating | 5 |
| | > 1650 | 8.6E4 to 1.7E5 | 10 – 50 | Fuel Pellet | Intermediate Fuel Pellet Overheating | 6 |
| | > 1650 | 1.7E5 to 3.4E5 | > 50 | Fuel Pellet | Major Fuel Pellet Overheating | 7 |
| Core Melt | > 1650 | 3.4E5 to 4.6E5 | < 10 | Fuel Pellet | Initial Fuel Pellet Melt | 8 |
| | > 1650 | 4.6E5 to 5.8E5 | 10 – 50 | Fuel Pellet | Intermediate Fuel Pellet Melt | 9 |
| | > 1650 | > 5.8E5 | > 50 | Fuel Pellet | Major Fuel Pellet Melt | 10 |

**CDA Based on Containment Radiation Readings**

Unit________

Time Reactor Shut Down________________

Power Correction Factor From Data Sheet 2_______

Corrected Reading = (Actual Reading ÷ Power Correction Factor)/100

| **Time Post-Accident, Δt Hours** | **Actual*** | | **Corrected** | | **Estimate of Core Damage (%)** |
|---|---|---|---|---|---|
| | **1-VRA-1310/ 2-VRA-2310 R/hr.** | **1-VRA-1410/ 2-VRA-2410 R/hr.** | **1-VRA-1310/ 2-VRA-2310 R/hr.** | **1-VRA-1410/ 2-VRA-2410 R/hr.** | |
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***PPC location: Unit 1 or Unit 2/ER/Dose Assessment**

**CDA Based on CET Readings**

| **Date/Time** | **Maximum CET Temp. °F** | **CET Location** | **Estimate of Core Damage (%)** |
|---|---|---|---|
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**PPC Location: NSS/Nuclear Steam Parameters/Hottest TC °F**

1 Power Correction Factor Calculations

1.1 Determination of Average Reactor Power

1.1.1 If reactor power has not changed by more than ± 10% for a period greater than thirty days, the power at the time of the shutdown can be used.

1.1.2 If the power has changed by more than ± 10% during the 4 or 30 days prior to the accident, an estimate must be made to establish the most representative power level. The thirty-day average power level is not necessarily the most representative indication. Weighted average power history is determined by summing the products of power level durations multiplied by the power levels and dividing by the total duration length. Perform this estimation for the prior four-day period and thirty-day period, using the following formula:

$$\text{Power Correction Factor} = \frac{(\text{days at power}_1 \text{ x \% power}_1) + (\text{days at power}_2 \text{ x power}_2) + \ldots}{\text{Total days considered in this history}}$$

| | Percent Power | Duration, Day |
|---|---|---|
| Prior four days | __________ | __________ |
| | __________ | __________ |
| | __________ | __________ |
| | __________ | __________ |
| Prior 30 days | __________ | __________ |
| | __________ | __________ |
| | __________ | __________ |
| | __________ | __________ |
| | __________ | __________ |

Four-day estimated reactor power at time of shutdown____________________

Thirty-day estimated reactor power at time of shutdown____________________

1.1.3 Select the power correction factor that is lower and record this value on Data Sheet 1, Core Damage Assessment.