Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

November 6, 2003
LIC-03-0151

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Reference: Docket No. 50-285

SUBJECT: Transmittal of Changes to Emergency Plan Implementing Procedures (EPIP)

In accordance with 10 CFR 50.54(q), 10 CFR 50, Appendix E, Section V, and 10 CFR 50.4(b)(5), please find EPIP change packages enclosed for the Document Control Desk (holder of Copy 165) and the NRC Region IV Plant Support Branch Secretary (holder of Copies 154 and 155).

The document update instructions and summary of changes are included on the Confirmation of Transmittal form (Form EP-1) attached to each controlled copy change package. Please return the Confirmation of Transmittal forms by December 26, 2003.

The revised documents included in the enclosed package are:
EPIP Index page 1 and 2 issued 10/28/03
EPIP TSC-8 R15 issued 10/28/03

If you have any questions regarding the enclosed changes, please contact Mr. Carl Simmons at (402) 533-6430.

Sincerely,

J. B. Herman
Manager
Nuclear Licensing

JBH/ckf

Enclosures

C: NRC Region IV Plant Support Branch Secretary (2 sets)
Alan Wang, NRC Project Manager (w/o enclosures)
J. G. Kramer, NRC Senior Resident Inspector (w/o enclosures)
Emergency Planning Department (w/o enclosures)

A045

FORM EP-1
Rev. 6 (04/29/99)

OMAHA PUBLIC POWER DISTRICT

Confirmation of Transmittal for Emergency Planning Documents/Information

☐ Radiological Emergency Response Plan (RERP)
☒ Emergency Plan Implementing Procedures (EPIP)
☐ Emergency Planning Forms (EPF)
☐ Emergency Planning Department Manual (EPDM)
☐ Other Emergency Planning Document(s)/ Information

Transmitted to:

Name: Document Control Desk — Copy No: 165 — Date: ______
Division of Reactor Safety — Copy No: 154
Attn: Senior Emergency Preparedness Inspector
Division of Reactor Safety — Copy No: 155
Attn: Senior Emergency Preparedness Inspector

The following document(s) / information are forwarded for your manual:

REMOVE SECTION	INSERT SECTION
EPIP Index pages 1 through 3 issued 07/29/03	EPIP Index page 1 and 2 issued 10/28/03
EPIP-TSC-8 R14 issued 01/19/01	EPIP TSC-8 R15 issued 10/28/03

Summary of Changes:
EPIP-TSC-8 was revised to add scale lines to Figure 1 Attachment 6.5 for ease of reading.

Supervisor - Emergency Planning

I hereby acknowledge receipt of the above documents/information and have included them in my assigned manuals.
Signature: ______ **Date:** ______

Please sign above and return by 12/26/03 to:

Beth Nagel
Fort Calhoun Station, FC-2-1
Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

NOTE: If the document(s)/information contained in this transmittal is no longer requested or needed by the recipient, or has been transferred to another individuals, please fill out the information below.

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Name: ______ Mailing Address: ______

EPIP

Document	Document Title	Revision/Date
EPIP-OSC-1	Emergency Classification	R35 05-02-02
EPIP-OSC-2	Command and Control Position Actions/Notifications	R42 05-28-03a
EPIP-OSC-9	Emergency Team Briefings	R7 12-09-99
EPIP-OSC-15	Communicator Actions	R22 10-24-00a
EPIP-OSC-21	Activation of the Operations Support Center	R12 10-29-02a
EPIP-TSC-1	Activation of the Technical Support Center	R24 06-19-03
EPIP-TSC-2	Catastrophic Flooding Preparations(R0 03-22-95) DELETED (05-09-95)REINSTATED	R4 07-29-03
EPIP-TSC-8	Core Damage Assessment	R15 10-28-03
EPIP-EOF-1	Activation of the Emergency Operations Facility	R13 10-29-02
EPIP-EOF-3	Offsite Monitoring	R19 07-29-03
EPIP-EOF-6	Dose Assessment	R32 01-23-02a
EPIP-EOF-7	Protective Action Guidelines	R14 04-15-03
EPIP-EOF-10	Warehouse Personnel Decontamination Station Operation	R10 01-13-00a
EPIP-EOF-11	Dosimetry Records, Exposure Extensions and Habitability	R20 07-02-03
EPIP-EOF-19	Recovery Actions	R8 07-17-03
EPIP-EOF-21	Potassium Iodide Issuance	R4 11-07-00
EPIP-EOF-23	Emergency Response Message System	R5 10-12-99
EPIP-EOF-24	EOF Backup Alert Notification System Activation	R3 09-09-99
EPIP-RR-11	Technical Support Center Director Actions	R14 02-29-00a
EPIP-RR-13	Reactor Safety Coordinator Actions	R14 12-09-99a
EPIP-RR-17	TSC Security Coordinator Actions	R15 12-10-02a
EPIP-RR-17A	TSC Administrative Logistics Coordinator Actions	R20 11-07-02a
EPIP-RR-19A	Operations Liaison Actions	R6 04-15-03a
EPIP-RR-21	Operations Support Center Director Actions	R13 08-28-03
EPIP-RR-21A	Maintenance Coordinator Actions	R4 11-30-99a

EPIP-RR-22	Protective Measures Coordinator/Manager Actions	R23 09-09-03
EPIP-RR-22A	Chemistry Coordinator Actions	R6 12-07-01
EPIP-RR-25	EOF Dose Assessment Coordinator Actions	R21 05-15-03
EPIP-RR-28	OSC Accountability and Dosimetry Technician Actions	R8 09-25-01a
EPIP-RR-29	EOF Administrative Logistics Manager Actions	R20 11-07-02
EPIP-RR-39	Control Room Medical Responder Actions	R0 03-27-01a
EPIP-RR-63	EOF Dose Assessment Assistant Actions	R10 11-19-01
EPIP-RR-66	Communication Specialist Actions	R8 08-31-99
EPIP-RR-72	Field Team Specialist Actions	R13 07-09-02
EPIP-RR-87	Radiation Protection Coordinator Actions	R9 08-28-03
EPIP-RR-90	EOF/TSC CHP Communication Actions	R0 10-24-00

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Fort Calhoun Station
Unit No. 1

EPIP-TSC-8

EMERGENCY PLAN IMPLEMENTING PROCEDURE

Title: CORE DAMAGE ASSESSMENT

FC-68 Number: EC 33002

Reason for Change: Add lines to graph for ease of reading.

Requestor: M. Reller

Preparer: M. Reller

CORE UNCOVERY PREDICTIONS / CORE DAMAGE ASSESSMENTS

NON-SAFETY RELATED

1. PURPOSE

1.1 This procedure provides methods for trending Reactor Coolant System (RCS) volume above the active core during Loss of Coolant Accidents (LOCA), methods for estimating the time to core uncovery, and performing core damage assessments.

2. REFERENCES/COMMITMENTS DOCUMENTS

2.1 Development of the Comprehensive Procedure Guideline for Core Damage Assessment, Task 467, July 1983

2.2 FC-0204-98, Source Terms Used in Emergency Plan Core Damage Assessment Spreadsheet (TSC-8)

2.3 EA-FC-90-94, Fuel Handling Accident and Bounding Source Term

2.4 NUREG 1465, Accident Source Terms for Light - Water Nuclear Power Plants

2.5 Calculation Number FC06727, Estimated Containment 994' Elevation Volume vs. Level for EPIP-TSC-8

3. DEFINITIONS

3.1 Time to Core Uncovery - A time calculated based on a snapshot of plant parameters that estimates the time remaining before the start of core uncovery (Reactor Vessel Level at the top of the active core).

3.2 Core Damage Assessment - The categorization of a core damage into four major types as follows: no fuel damage, fuel cladding failures, fuel pellet overheating and fuel pellet melting. The three later categories are delineated as initial, intermediate, and major. Each of the ten categories of core damage can be characterized by the type of fuel damage, the corresponding temperature range and the mechanism of fission product release.

3.3 Category of no fuel damage - that which is characterized by the release of fission products through the release mechanisms of Halogen Spiking and tramp uranium fission. The release source is the Gas Gap and the characteristic isotopes are I-131, Cs-137, and Rb-88.

3.4 Category of Fuel Cladding Failure - that which is characterized by the release of fission products through the release mechanisms of burst and Gas Gap diffusion. The release source is the Gas Gap. The characteristic fission products are noble gases and halogens (Xe-131m, Xe-133, I-131 and I-133) because they are volatile and can migrate quickly through the fuel pellet and gas gap for release following cladding rupture. These isotopes are volatile in the Temperature range of 1300-1800°F.

3.5 Category of Fuel Pellet Overheat - that which is characterized by the release of radioactivity through gain boundary diffusion and by diffusion from within the UO_2 grains. Grain boundary diffusion begins above 2450° F. Moderately volatile isotopes of cesium, rubidium and tellurium (Cs-134, Rb-88, Te-129, and Te-132) are characteristic of this type of damage.

3.6 Category of Fuel Pellet Melt - occurs at greater temperatures (2550-3450°F), reactions begin between the solid UO_2 and the solid metallic zircaloy, melting of the control rod materials, and melting of zirconium. At these temperatures greater amounts of tellurium are released. Alkali metals, such as barium, volatize as well as rare earths and actinides such as lanthanum and protactinium. These include the isotopes of Sr-89, Ba-140, La-140, La-142, and Pr-144.

3.7 Activity Ratios - theoretical calculations employed to determine typical ratios for isotopes of a fission product in the Gas Gap or the Fuel Pellet. Comparison of the ratios obtained from sample data with these calculated values determines the source of the fission product release. These ratios are made by comparing the noble gas/iodine isotopes in the sample to Xe-133/I-131 in the sample.

4. PREREQUISITES

4.1 This procedure is intended to be used during Loss of Coolant Accidents and other plant transients that may lead to core damage.

NOTE: Steps in this procedure may be done out of order or be completed concurrently.

NOTE: HJTCs may be deenergized per OI-RC-2A RCS Fill and Drain Operations causing RVLMS to indicate 100% while in reduced RCS inventory conditions. While the HJTCs are deenergized, use LI-119 and LI-197 and Attachment 6.1 -Figure #1, RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level for determining RCS volumes. Energizing HJTCs is acceptable for short periods to determine if a void exists in the Reactor Vessel per the precautions in OI-RC-2A.

5. PROCEDURE

NOTE: Trending the RCS Inventory above the active Core should be done if a significant decrease is observed in the RCS Inventory.

NOTE: Trending the RCS Inventory above the active Core may be performed using the Excel Spreadsheet "RCS_INV(R0)". This spreadsheet contains worksheets "RCS Inventory Data" and "RCS Inventory Chart" which are equivalent to Attachment 6.1 - Trending RCS Inventory above the Core.

5.1 Trend the RCS inventory volume above the core using Attachment 6.1 - Trending RCS Inventory above the Core.

NOTE: The time to core uncovery is determined using two methods. The first (normal) method, Attachment 6.2, determines a Net Loss/Gain in ft^3/min and uses this rate as the basis to determine the time remaining for Reactor Vessel Level to drop to the top of the active core. The second method, Attachment 6.3, uses Steaming Rate and RCS density to determine the boil-off rate in ft^3/min.

NOTE: The second method (Attachment 6.3) is used when the Reactor Vessel level is below that of any suspected leaks, the inventory is being lost is due to RCS boil-off, and it is determined that no make up flow is reaching the reactor vessel.

NOTE: When the plant has been in Cold Shutdown Condition (Mode 4) or Refueling Shutdown Condition (Mode 5), the time to Core Uncovery may be available from the Shutdown Safety Advisor or the Shift Technical Advisor, if not determine the time to Core Uncovery using one of methods below.

5.2 During Loss of Coolant Accidents, estimate the time to core uncovery by one of the following methods:

NOTE: Estimating the time to core uncovery, using the normal method, may be performed using the EXCEL spreadsheet "RCS_INV(R0)." This spreadsheet contains a worksheet labeled "Uncvy_Vol Chg" which is equivalent to Attachment 6.2 - Using the Change in RCS Volume vs. Time To Estimate the Time to Core Uncovery.

5.2.1 First (normal) method, complete Attachment 6.2 - Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery.

NOTE: Estimating the time to core uncovery, using the second method, may be performed using the EXCEL spreadsheet "RCS_INV(R0)". This spreadsheet contains a worksheet labeled "Uncvy_Stm Rate" which is equivalent to Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery.

5.2.2 Second (Steaming Rate) method, complete Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery.

NOTE: The prior power history for the last 30 days may be recorded on the EXCEL spreadsheet "Cor_dam(R0) .xls", worksheet "Prior Power History". The worksheet "Prior Power History" is equivalent to Attachment 6.4 - Prior 30-Day Power History.

5.3 Determine the power history for the last 30 days from the time of reactor trip/shutdown by completing Attachment 6.4 - Prior 30-Day Power History.

NOTE: Core Damage Assessments should be done when the Fuel Cladding Fission Product Barrier Criteria has been exceeded per EPIP-OSC-1, Attachment 6.3, Three Fission Product Barrier Criteria or at the discretion of the Command and Control position.

5.4 Complete Attachment 6.5 - Assessment of Core Damage Using Containment Radiation Dose Rates.

5.5 For slow transients only, complete Attachment 6.6 - Assessment of Core Damage using CETs.

NOTE: The total quantity of fission products available at different locations in the Containment may be changing due to transient plant conditions. Samples of the Reactor Coolant System / Discharge of the Low Pressure Safety Injection Pump and the Containment Atmosphere used for core damage assessments should be obtained within a minimum time and under stabilized plant conditions when possible.

5.6 Prepare to do a Core Damage assessment by Radiological Analysis of samples as follows:

5.6.1 Verify that plant conditions are stabilized as much as possible that plant conditions can support sampling operations for a Reactor Coolant System or Discharge of the Low Pressure Safety Injection Pump sample and/or a Containment Atmosphere sample.

5.6.2 Before the sampling team is dispatched to collect the required samples, hold a briefing with the sampling team and coordinate the collection of data that needs to be obtained when samples are isolated per Attachment 6.7, Worksheet #1 and #2.

NOTE: Manual core damage calculations may be performed using the EXCEL spreadsheet "Cor_dam(R0).xls". This spreadsheet "Cor_dam.xls" and its worksheets are equivalent to Attachment 6.7 - Manual Core Damage Assessment Calculations.

5.7 Complete Attachment 6.7 - Manual Core Damage Assessment Calculations.

5.8 Using the completed Attachment 6.7, make a conclusion on the extent of core damage using following three parameters:

5.8.1 Identification of the fission product isotopes that characterize the Reactor Coolant System / LPSI Pump and Containment Atmosphere Samples.

5.8.2 Identification of the source of the fission products (the Gas Gap or the Fuel Pellet) from Attachment 6.7 - Worksheets #4 and #5.

5.8.3 The quantity of the fission products available for release to the environment expressed as a percent of source inventory on Attachment 6.7 - Worksheet #7.

NOTE: Knowledgeable judgement is used to compare the three parameters with the definitions of the 10 NRC categories of fuel damage found on Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage. Core Damage is not anticipated to take place uniformly. Therefore, when evaluating the three parameters listed above, the procedure is anticipated to yield a combination of one or more of the 10 categories defined on Attachment 6.8. The categories will exist simultaneously. The type of core damage is described in the 10 NRC categories listed on Attachment 6.8. The degree of core damage is described as the percent of the fission products in the source inventory that is now available in the Containment for release to the environment.

5.9 Based on the conclusions above, circle the categories of fuel damage on Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage.

6. <u>ATTACHMENTS</u>

6.1 Trending RCS Volume above the Core

6.2 Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery

6.3 Using Steaming Rate to Estimate the Time to Core Uncovery

6.4 Prior 30 Day Power History

6.5 Assessment of Core Damage Using Containment Radiation Dose Rates

6.6 Assessment of Core Damage using CETs

6.7 Manual Core Damage Assessment Calculations

6.8 Clad Damage Characteristics of NRC Categories of Fuel Damage

Attachment 6.1 - Trending RCS Inventory above the Core (Page 1 of 4)

1. On Worksheet 1 - RCS Inventory Data Sheet, record the Time, Pressurizer Level and RVLMS.

2. Record the RCS Volume above the Core on Worksheet 1 using the information from Figure 1, RCS Inventory above the Core vs. Pressurizer/RVLMS Level.

3. Plot the RCS Volume above the Core on Worksheet 2, Plot of RCS Volume above the Core vs. Time.

Attachment 6.1 - Trending RCS Inventory above the Core (Page 2 of 4)

Worksheet 1, RCS Inventory Data Sheet

Time	Pressurizer Level %	RVLMS %	Volume above the Core (ft^3)

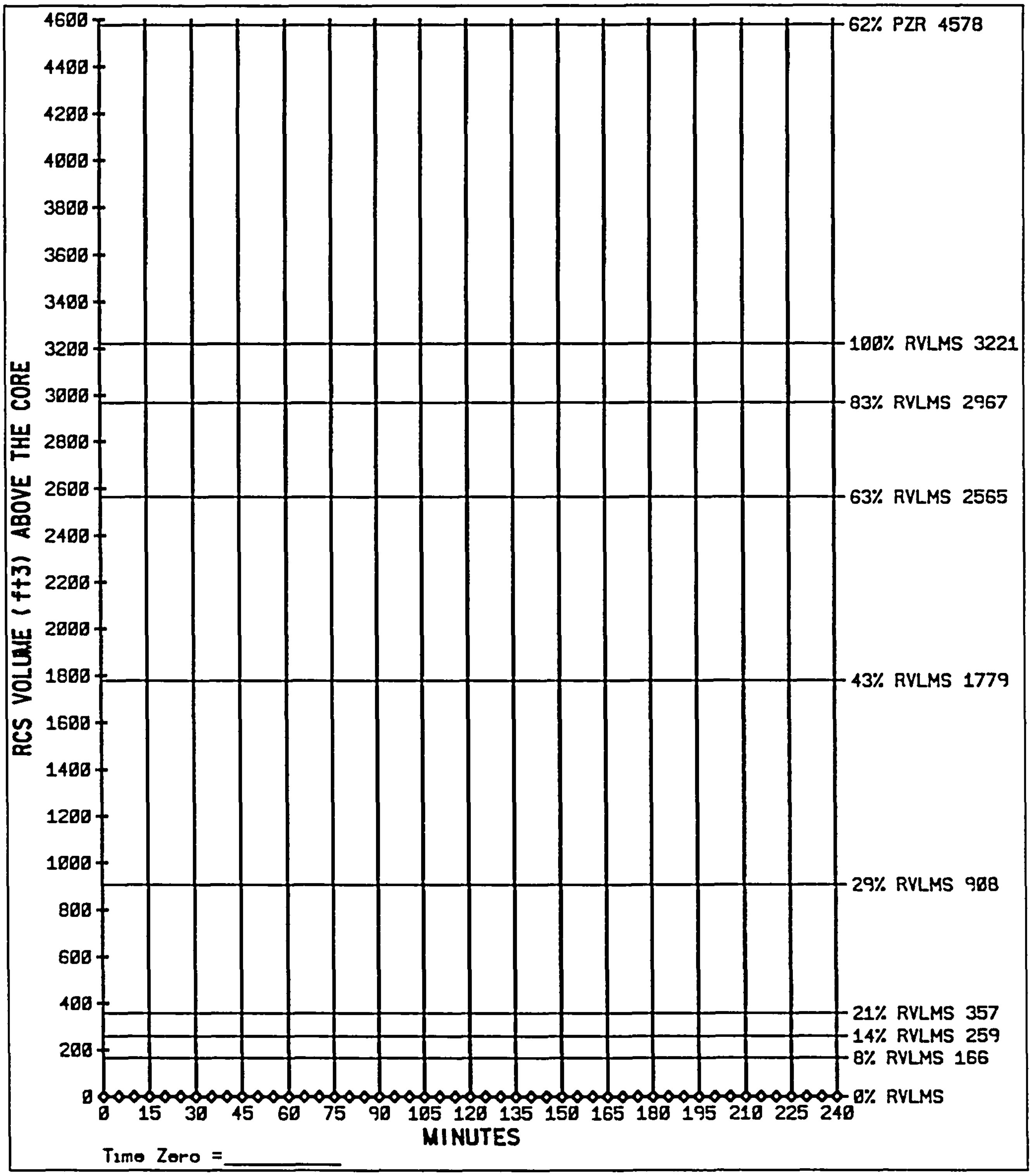
Attachment 6.1 - Trending RCS Inventory above the Core (Page 3 of 4)

Figure 1 - RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level

PZR/RVLMS (% Level)	RCS INV. ABOVE TOP OF CORE (ft^3)	LI-119, LI-197 (Feet)	NOTES
100 PZR	4962		
90 PZR	4821		
80 PZR	4760		
70 PZR	4665		
65 PZR	4608		
60 PZR	4533		
55 PZR	4445		
50 PZR	4356		
45 PZR	4267		
40 PZR	4179		
35 PZR	4090		
30 PZR	4001		
25 PZR	3913		
20 PZR	3824		
15 PZR	3735		
10 PZR	3646		
5 PZR	3558		
0 PZR	3469		
100 RVLMS	3221	1018.31	Top of Upper Head
83 RVLMS	2967	1017.174	
63 RVLMS	2565	1014.018	
43 RVLMS	1779	1010.867	
29 RVLMS	908	1007.708	Top of Hot Leg
21 RVLMS	357	1006.375	Center of Hot Leg
14 RVLMS	259	1005.042	Bottom of Hot Leg
8 RVLMS	166	N/A	
0 RVLMS	0	N/A	Top of Active Core

Attachment 6.1 - Trending RCS Inventory above the Core (Page 4 of 4)

Worksheet 2 - Plot of RCS Volume above the Core vs. Time



Attachment 6.2 - Using the Change in RCS Volume vs. Time to Estimate the Time to Core Uncovery

NOTE: When a bubble is in the Reactor Vessel use RVLMS to determine the RCS volume (i.e., Pressurizer Level at 40% and RVLMS at 83%, a volume of 2967 ft^3 based on 83% RVLMS should be used).

1. Record Time, Pressurizer Level, RVLMS Level and the RCS Volume above the Core from Attachment 6.1, Figure 1 - RCS Inventory above the Core vs. Pressurizer/RVLMS Level.

   T_1 (Time)__________ Pressurizer Level (%)__________ RVLMS Level (%)__________

   V_1 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) __________

2. Wait for drop in Pressurizer/RVLMS Level and record the Time, Pressurizer Level, RVLMS Level, and RCS Volume below:

   T_2 (Time)__________ Pressurizer Level (%)__________ RVLMS Level (%)__________

   V_2 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) __________

3. Determine the Time in Minutes (T_M) to Core Uncovery using the equation below.

$$T_M = \frac{V_2 \_\_\_\_\_\_ ft^3}{(V_1 \_\_\_\_\_\_ ft^3 - V_2 \_\_\_\_\_\_ ft^3) / (T_3 \_\_\_\_\_\_ \text{ minutes})} = \_\_\_\_\_\_ \text{Minutes}$$

   Where:

   T_3 = elapsed time in minutes ($T_2 - T_1$)

4. Add the time T_M to the time T_2 to determine the Projected Time of Core Uncovery.

   Projected Time of Core Uncovery = T_2__________ + T_M__________ = __________(hh:mm)

5. Remarks:

   ____
   ____
   ____

Done by:____________________

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 1 of 3)

1. Record Time, Pressurizer Level, RVLMS Level, RCS Volume from Attachment 6.1, Figure 1, Time of Reactor Trip/Shutdown, Steaming Rate from Attachment 6.3, Figure 1, Highest CET Temperature below and RCS Density from Attachment 6.3, Figure 2.

   T_1 (Time)__________ Pressurizer Level______________ RVLMS Level_______________

   V_1 (RCS Volume in ft^3 from Attachment 6.1, Figure 1) ______________

   Time of Reactor Trip/Shutdown ______________hh:mm

   SR (Steaming Rate from Figure 1) ____________lb/sec

   Highest CET Temperature __________________°F

   D (RCS Density from Figure 2) _______________lb/ft^3

2. Determine the T_M (Minutes to Core Uncovery) using the equation below:

$$\text{Minutes to Core Uncovery} = \frac{(V_1\_\_\_\_\_\_\_\_ ft^3) * (D\_\_\_\_\_\_\_\_ lb/ft^3)}{(SR\_\_\_\_\_\_\_\_ lb/sec) * (60)} = \_\_\_\_\_\_\_\_\_\_ \text{Minutes}$$

3. Determine the Time of Core Uncovery using the equation below:

   Time of Core Uncovery = (T_1__________) + (T_M____________) = ___________(hh:mm)

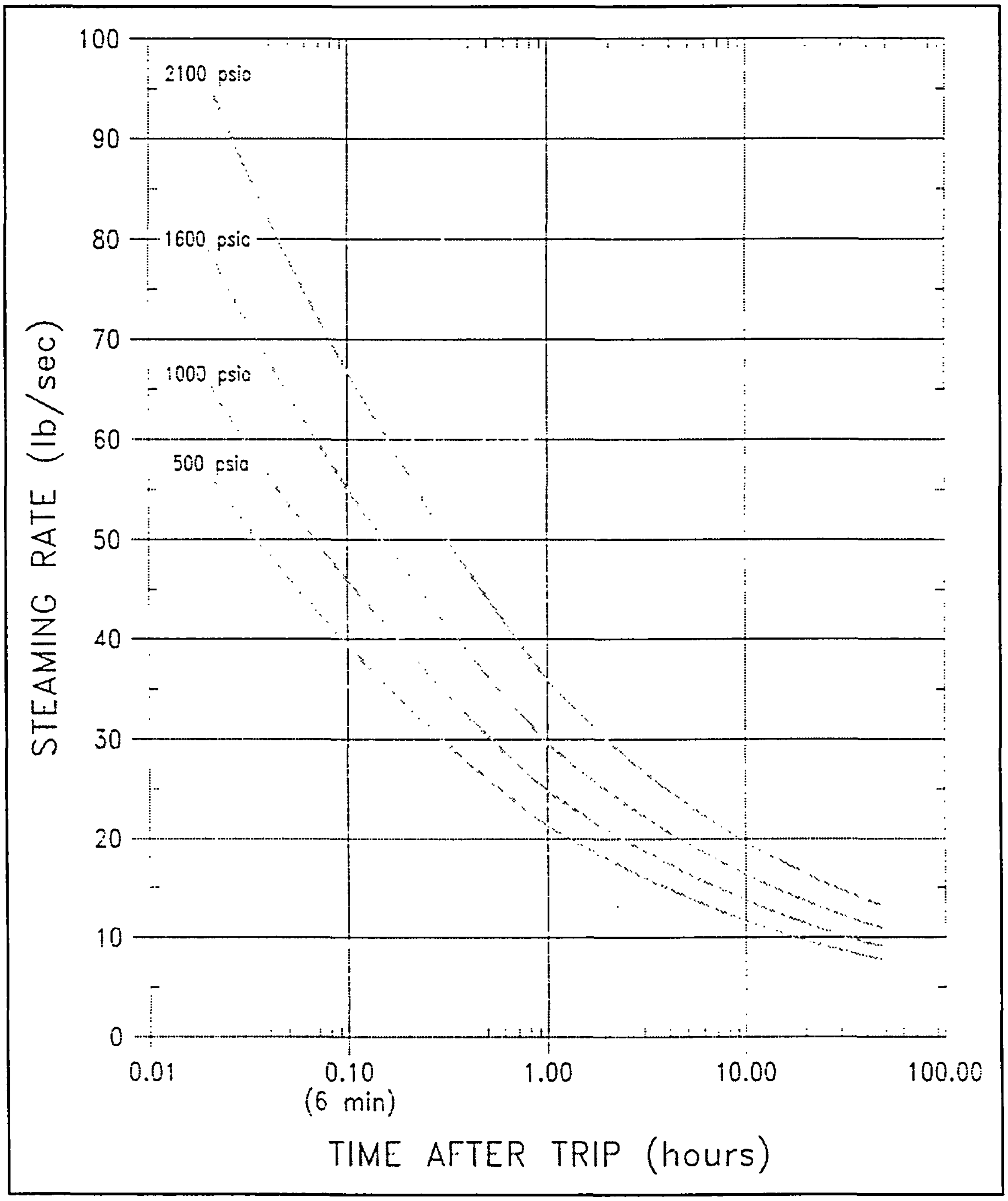
4. Remarks:

   ____
   ____
   ____
   ____

5. Done by:_______________________

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 2 of 3)

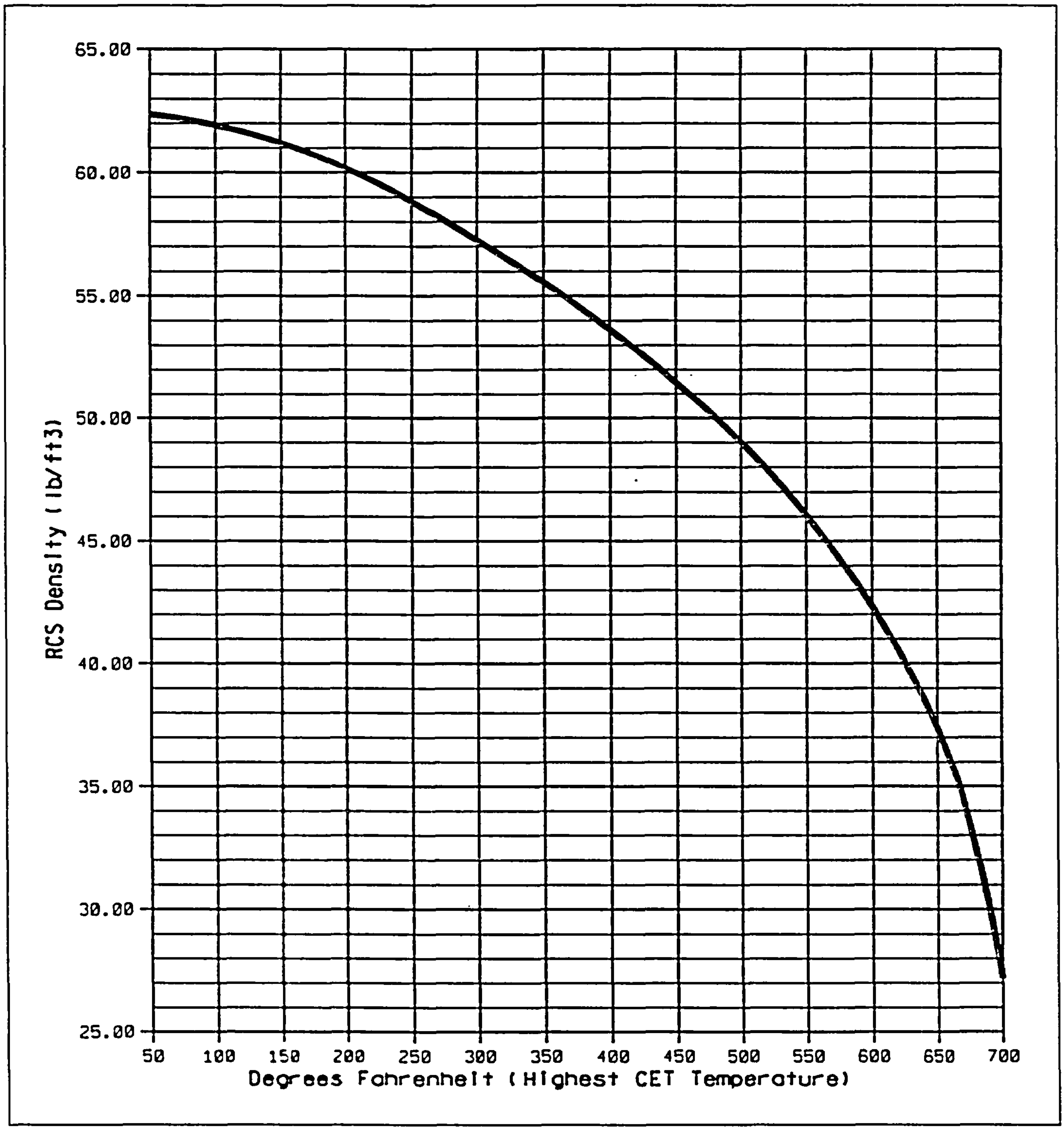
Figure 1, RCS Steaming Rate vs. Time After Trip



NOTE 1: If pressure is >2100 psia use the 2100 psia curve.
NOTE 2: If pressure =<2100 psia =>500 psia interpolate.
NOTE 3: If pressure <500 psia use the 500 psia curve.

Attachment 6.3 - Using Steaming Rate to Estimate the Time to Core Uncovery (Page 3 of 3)

Figure 2 - RCS Density vs. CET Temperature



Attachment 6.4 - Prior 30 Day Power History

Date of Reactor Shutdown:	Time of Reactor Shutdown:
If reactor power has been steady state (+/- 10%) in the four days before the Reactor Shutdown, record the steady state power level:	
If reactor power has been steady state (+/- 10%) in the 30 days before the Reactor Shutdown, record the steady state power level:	
If the plant's power history has not been steady state for the last 30 days, enter data below for each steady state power period during the last 30 days (record up to eight power periods where power has change more than +/- 10% power):	

Power Period (j)	Steady State Power for Period (P_j)	Duration of Power Period (d)		Time From the End of Operating Period (P_j) To Reactor Shutdown (t)	
Number	Percent	Days	Seconds Days X 8.64E4	Days	Seconds Days X 8.64E4
1					
2					
3					
4					
5					
6					
7					
8					

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 1 of 3)

1. Record Date and Time of Reactor Shutdown: Date:__________Time:__________(T_1)

2. Date and Time of RM-091A and RM-091B Reading: Date:__________Time:__________(T_2)

3. Record: RM-091A Reading: ______________ RM-091B Reading:__________________

4. Determine the Time Post Accident, Hours by subtracting the Date and Time of RM-091A and RM-091B Reading from the Date and Time of Reactor Shutdown.

   Time Post Accident, Hours = (T_2 - T_1) = ____________________________________ = Hours

NOTE: Use the following information to estimate the 30-Day Average Power Level:

- The average power during the 30-day time period is not necessarily the most representative value for correction to equilibrium conditions.
- The last power levels at which the reactor operated should weigh more heavily in the judgement than earlier levels.
- Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.
- In the case in which the reactor has produced power for less than 30 days the procedure may be employed. However, the estimate of core damage obtained under this condition may be an under estimate of the actual condition.

5. Estimate the 30-Day Average Power Level, using the plants power history recorded on Attachment 6.4, engineering judgement, and the information in the note above.

6. Record the 30 Day Average Power Level based on engineering judgement = __________%.

7. Determine the Equilibrium Dose Rate as follows:

$$\text{Equilibrium Dose Rate} = (\text{Higher reading of RM-091 A or B}) \times \frac{100}{(\text{30 Day Average Power})}$$

$$\text{Equilibrium Dose Rate} = (\_\_\_\_\_\_\_\ \text{R/hr}) \times \frac{100}{(\_\_\_\_\_\_\ \text{30 Day Average Power})} = \_\_\_\_\_\_\_\_\_\ \text{R/hr}$$

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 2 of 3)

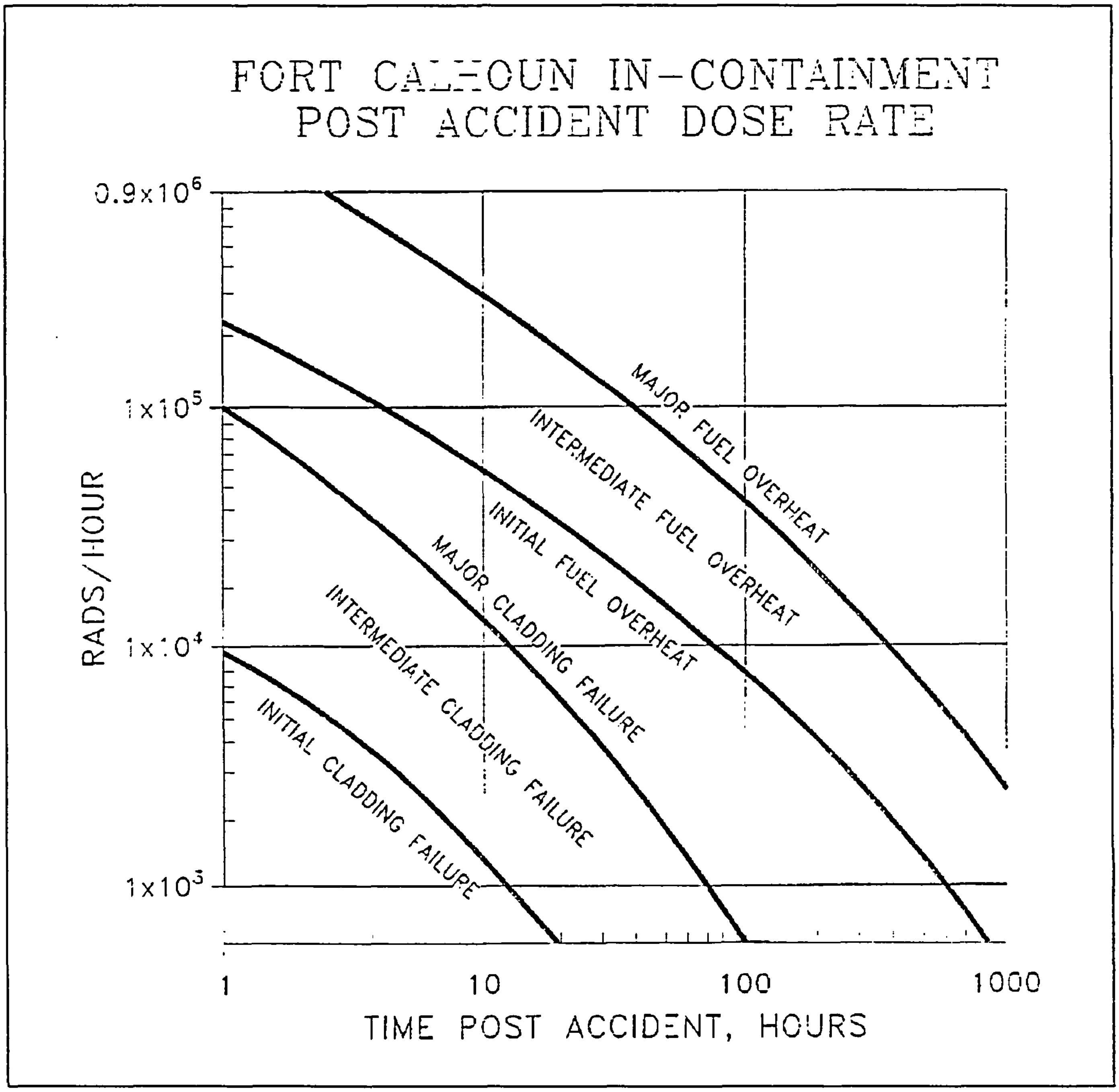
NOTE: *Use the following information to consider which category of core damage is most representative of the particular value:*

- Dose rate measurements made during stable plant conditions should weigh more heavily in the assessment of core damage.
- This attachment may not be employed to estimate the degree of Fuel Pellet Melting.
- Dose rates significantly above the lower bound for the category of Major Fuel Over Heat may indicate concurrent Fuel Pellet Melting.
- This attachment may not be used to distinguish the relative contributions of the two categories (Cladding Failure and Fuel Pellet Overheat) to the total dose rate. This procedure does give the estimate of the highest category of damage.
- Dose rates within any category of fuel overheating may be anticipated to include concurrent Fuel Cladding Failure.
- Dose rates corresponding to the two categories of major cladding failure and initial fuel overheat are observed to overlap on Figure 1 - Categories of Core Damage Based on Containment Post Accident Dose Rates. The evaluation of other parameters may be required to distinguish between them. However, concurrent conditions may be anticipated.

8. On Figure 1, Category of Core Damage Based on Containment Post Accident Dose Rates, **plot** the Equilibrium Dose Rate as a function of the Time Post Accident, Hours.

Attachment 6.5 - Assessment of Core Damage using Containment Radiation Dose Rates
(Page 3 of 3)

Figure 1- Category of Core Damage Based on Containment Post Accident Dose Rates



Attachment 6.6 - Assessing Core Damage Using CETs (Page 1 of 3)

1. Record the following:

    1.1 Maximum CET Temperature: ________________________________F.

    1.2 Pressurizer Pressure at the time of Max CET Temperature: __________psia

    1.3 Time of the maximum CET Temperature: ________________________

2. Select the curve on Figure 1, Percent of Fuel Rods with Rupture Clad vs. Maximum CET Temperature and Pressure that represents a pressure approximately equal to or greater than the Pressurizer Pressure at the time of the Maximum CET Temperature. Read across the curve and record the Percent of Fuel Rods With Ruptured Clad Below:

    Percent of Fuel Rods With Ruptured Clad ________________________%

NOTE: The Percent of Fuel Rods With Ruptured Clad obtained above is probably a lower limit of the estimate of damage. Some judgement on the bias is available as follows:

- This procedure applies most directly for slow core uncovery with a maximum temperature below the rapid oxidation temperature of 1800° F. A smooth rise in CET temperature and an uncovery duration of 20 minutes or longer are indicators for good prediction of clad ruptures.

- If pressure dropped to less than 100 psia within two minutes of the accident initiation, a large break is suggested. This causes undetected core heat up followed by flashing during a refill. Depending on the rate of refill, the CET temperature may rise rapidly then quench when the core is recovered. This procedure would then yield a very low estimate for the percent of fuel rods ruptured.

- If pressure was above 1650 psia, it could exceed the fuel rod internal gas pressure depending on rod burn up, causing clad collapse onto the fuel pellet instead of outward clad ballooning. The clad rupture criteria are less well defined for such conditions, but at temperatures above 1800° F where the highest pressure curve applies on Figure 1, clad failure sufficient to release fission gas is likely and this procedure may be used to obtain estimates of damage.

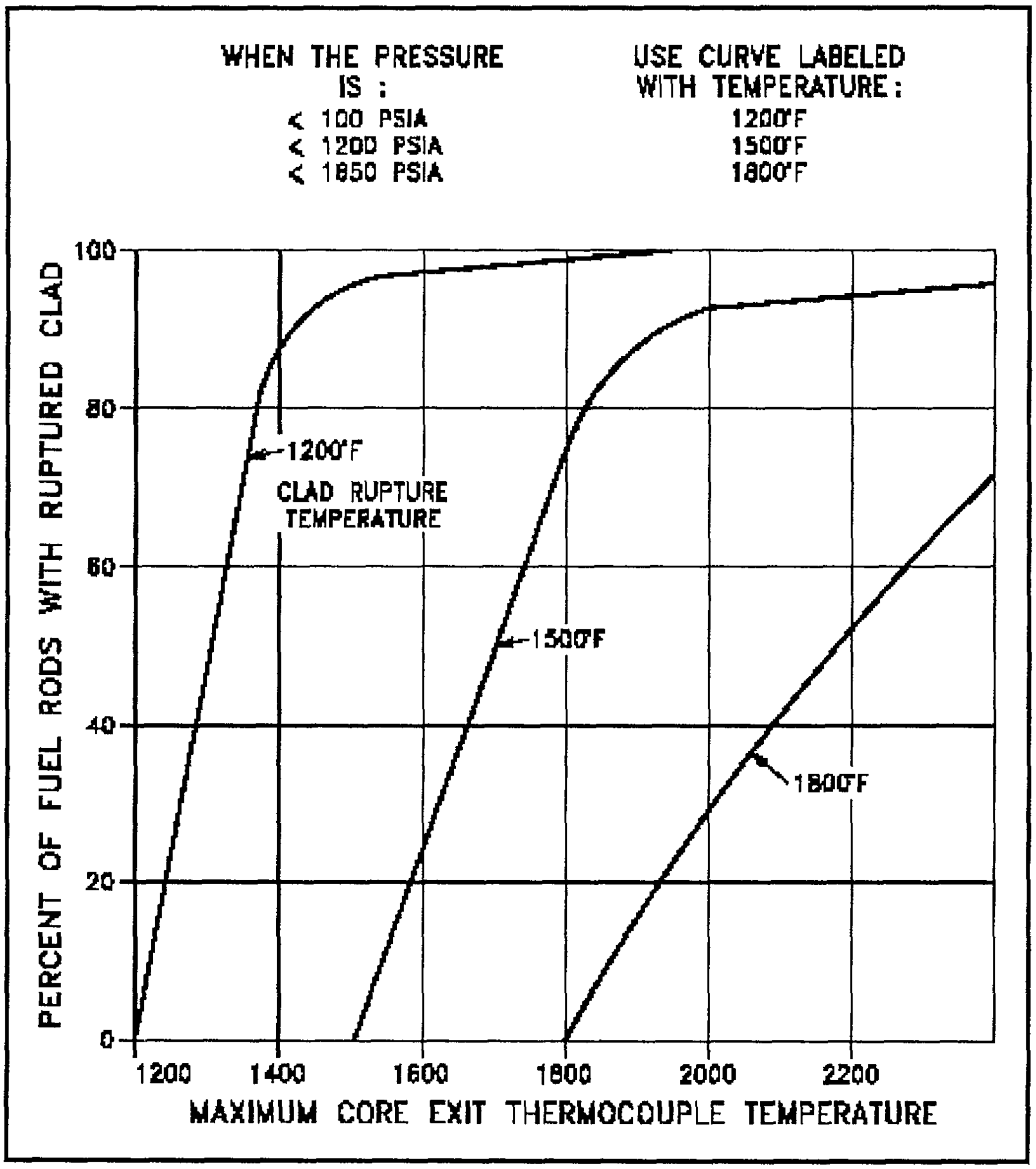
3. Comment on the probable bias in the result of Percent of Fuel Rods With Rupture Clad:

____
____
____
____
____
____
____

4. On Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage, determine and underline the NRC Category of Fuel Damage and its Characteristics.

Attachment 6.6 - Assessing Core Damage Using CETs (Page 3 of 3)

Figure 1, Percent of Fuel Rods with Rupture Clad vs. Maximum CET Temperature and Pressure



1. On Worksheet #1, Section 1 complete the following:

   1.1 At the time of the Reactor Coolant Sample / LPSI Pump sample isolation. record the Date/Time, RCS Pressure, RCS Temperature, Pressurizer Level, RVLMS, and Containment Sump Level.

   1.2 Determine and record the time in seconds from the time of reactor shutdown to the time of the Reactor Coolant Sample / LPSI Pump sample isolation.

   1.3 Record the Sample Specific Activities from the Reactor Coolant / Discharge of the LPSI Pump sample.

   1.4 Record the Temperature Correction Factor as follows:

      1.4.1 For RCS samples obtain the temperature correction factor from Figure #1- Ratio of H_20 Density to H_20 Density at STP vs. Temperature.

      1.4.2 For the Discharge of the LPSI Pump samples record a temperature correction factor of 1.0.

   1.5 Multiply the Sample Specific Activity by the Temperature Correction Factor and record the Temperature Corrected Specific Activity (A_T).

   1.6 Determine and record the Decay Corrected Specific Activity using the following equation:

   $$A_0 = \frac{A}{e^{-\lambda t}}$$

   Where:

   A_0 = the Decay Corrected Specific Activity in µCi/cc.

   A_T = the Temperature Corrected Specific Activity in µCi/cc.

   λ = the radioactive decay constant, 1/sec.

   t = the time from a reactor shutdown to sample isolation in seconds.

2. On Worksheet #2, complete the following:

   2.1 At the time of the Containment Atmosphere sample isolation, record the Date/Time. Containment Pressure and the Containment Temperature.

2.2 Determine and record the time in seconds from the time of reactor shutdown to the time of the Containment Atmosphere sample.

2.3 Determine the Temperature/Pressure Correction Factor for the Containment using the equation below and record on Worksheet #3, Section 1:

$$\text{Correction Factor} = \frac{14.2}{(P_1 + 14.2)} \times \frac{(T_1 + 460)}{492}$$

Where:

P_1 = Containment pressure (psig) at time of Containment Atmosphere sample isolation

T_1 = Containment Temperature at the time of the Containment Atmosphere sample isolation

2.4 Correct the Sample Specific Activities to standard Temperature and Pressure by multiplying the Sample Specific Activity by the Temperature/Pressure Correction Factor and record in the Column labeled Temperature/Pressure Corrected Specific Activity.

2.5 Correct and record the Decay Corrected Specific Activities for decay back to the time of reactor shutdown using the following equation:

$$A_0 = \frac{A_{PT}}{e^{-\lambda t}}$$

Where:

A_0 = the Decay Corrected Specific Activities in µCi/cc.
A_{PT} = the Temperature Pressure Corrected Specific Activity in µCi/cc.
λ = the radioactive decay constant, 1/sec.
t = the time from a reactor shutdown to sample isolation in seconds.

3. On Worksheet #3, Identify the Fission Product Release Source (Gas Gap or the Fuel Pellet) as follows:

3.1 Copy the Decay Corrected Specific Activities from Worksheet #1 for KR-87, Xe-131m. Xe-133, I-131, I-132. I-133, and I-135.

3.2 Copy the Decay Corrected Specific Activity from Worksheet #2 for KR-87, Xe-131m. Xe-133, I-131, I-132. I-133, and I-135.

3.3 Calculate the noble gas and iodine ratios using the equations below:

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Decay Corrected Specific Activity}}{\text{Xe-133 Decay Corrected Specific activity}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Decay Corrected Specific Activity}}{\text{I-131 Decay Corrected Specific Activity}}$$

NOTE: Select as the source of the release that ratio that is closest to the Sample Isotope Ratio. An accurate comparison is not anticipated.

3.4 Determine the Identified Source (Fuel Pellet or Gas Gap) of the release by comparing the Sample Isotope Ratio results with the predicted isotope ratios for the Fuel Pellet and the Gas Gap Inventories. Record the source in the column labeled Identified Source Fuel Pellet or Gas Gap.

4. On Worksheet #4, determine the Volumes of the Reactor Coolant System, Containment Sumps and Containment Atmosphere as follows:

4.1 Determine the volume (ft^3) of the RCS by using the Pressurizer Level and RVLMS level recorded on Worksheet #1 and Attachment 6.1, Figure #1 - RCS Inventory Above the Core (ft^3) vs. Pressurizer/RVLMS Level, record the volumes in Section 1.

4.2 Determine and record the Density Correction in Section 1 as follows:

4.2.1 If an RCS Sample was obtained, obtain density correction from Figure #1 - Ratio of H_20 Density to H_20 Density at STP vs. Temperature by using the RCS Temperature recorded on Worksheet #1, at the time of the RCS sample.

4.2.2 If a sample was obtained from the discharge of the LPSI Pump, enter a Density Correction of 1.0 in Section 1.

4.3 Complete the RCS Volume Calculation in Section 1.

4.4 Determine and record the volume in the Containment Sump by using the Containment Sump Level recorded on Worksheet #1 and Figure #2 - Containment Sump/Basement Volume vs. L-387 or L-388 in Section 2. Record the volume in gallons in Section 2 of Worksheet #4.

4.5 Complete the calculation for the Containment Sump Volume in Section 2 on Worksheet #4.

4.6 Correct the Containment Atmosphere Volume to STP in Section 3 using the Containment Pressure and Temperature on Worksheet #2 and the following equation:

$$\text{Containment Atmosphere Volume} = (2.97E+10) \times \left(\frac{14.2 + P_1}{14.2}\right) \times \left(\frac{492}{T_1 + 460}\right)$$

Where:

2.97E+10 = the free Volume of the Containment in cc

P_1 = Containment Pressure (psig)

T_1 = Containment Temperature (° F)

NOTE: The Specific activities in the Reactor Coolant System and the Containment Sump are assumed to be equal. The results of the RCS/Discharge of the LPSI pump sample may be used for both the Containment Sump and Reactor Coolant System calculations below.

5. On Worksheet #5, calculate the Total Quantity (curies) in Containment as follows:

5.1 Using the equation below calculate the Quantity (Curies) in the Containment Sump for each isotope and record in Column #2.

$$\text{Curies in the Containment Sump} = \frac{SA}{1.0E+6} \times \text{CS Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #1

CS Vol = the Containment Sump Volume From Worksheet #4

1.0E+6 = Constant to convert μCurie to Curies

5.2 Using the equation below calculate the Quantity (Curies) in the Reactor Coolant System for each isotope and record in Column #3:

$$\text{Curies in the Reactor Coolant System} = \frac{SA}{1.0E+6} \times \text{RCS Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #1

RCS Vol = the Reactor Coolant Volume From Worksheet #4

1.0E+6 = Constant to convert μCuries to Curies

5.3 Using the equation below calculate the Quantity (Curies) in the Containment Atmosphere for each isotope and record in Column #3:

$$\text{Curies in the Containment Atmosphere} = \frac{SA}{1.0E+6} \times \text{CA Vol}$$

Where:

SA = the Decay Corrected Specific Activity on Worksheet #2

CA Vol = the Containment Atmosphere Volume From Worksheet #4

1.0E+6 = Constant to convert μCurie to Curies

5.4 Add Columns 2, 3, and 4. Record the sum (Total Quantity in Containment (Curies)) in Column 5.

6. On Worksheet #6, determine the Power Correction Factor and correct the Equilibrium Source Inventory for power history as follows:

   6.1 Using the power history data recorded on Attachment 6.4, determine the Power Correction Factor as follows:

      6.1.1 If the reactor power has been steady state (+/- 10%) for the four (4) days before the Reactor Shutdown, calculate using the equation below and enter the Power Correction Factor for the Fuel History Group (2) isotopes in Column 2.

$$PCF = \frac{\text{Steady State Power Level for Prior 4 Days}}{100}$$

      6.1.2 If the reactor power has been steady state (+/- 10%) for the thirty (30) days before the Reactor Shutdown, calculate using the equation below and enter the Power Correction Factor for the Fuel History Group (1) isotopes in Column 2.

$$PCF = \frac{\text{Steady State Power Level for Prior 30 Days}}{100}$$

NOTE: The following step is used to figure out the Power Correction Faction for those isotopes in the Fuel History Groups not determined in steps 6.1.1 and 6.1.2 above.

   6.2 If the reactor power has not been steady state (+/- 10%), determine the Power Correction Factor for each of the applicable power periods listed on Attachment 6.4, and record on Pages 2 and 3 of Worksheet #6 using the equation below:

$$PCF = \frac{P_j (1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of period j to a reactor shutdown

   6.3 For nonsteady state power histories, Sum up the power correction factors for each isotope on Pages 2 and 3 of Worksheet #6 and record on Page 1, Column #2 of Worksheet #6.

6.4 Correct the Equilibrium Source Inventory for power history, by multiplying the Power Correction Factor by the Equilibrium Source Inventory and record in Column 4 of Worksheet #6.

7. Determine and record the Percent of Source Inventory in Column 6 of Worksheet #5 using the equation below:

$$PSI(\%) = \frac{TQC}{CSI} \times 100$$

Where:

PSI(%) = Percent of Source Inventory in %

TQC = Total Quantity in Containment in Column 5 on Worksheet #5

CSI = Corrected Source Inventory in Column 4 of Worksheet #6

100 = Constant used to change a result to a percent

Attachment 6.7 - Manual Core Damage Assessment (Page 8 of 18)

Worksheet #1, Measured Specific Activity (μCi/cc) and Sample Temperature Correction for RCS/LPSI Sample

At the time RCS / LPSI Pump sample isolation record the following:

Date/Time__________ RCS Pressure __________ RCS Temperature __________

Pressurizer Level __________ RVLMS __________ Containment Sump Level __________

Time in Seconds between reactor shutdown and RCS / LPSI Pump sample isolation: __________ (seconds)

Temperature Correction Factor from Step 1.4: __________

	Sample Specific Activity	Temperature Corrected Specific Activity (A_T)	Decay Constant, λ (1/sec)	Decay Corrected Specific Act. $A_0 = \frac{A_T}{e^{-\lambda t}}$
Kr-87			1.5E-4	
Xe-131m			6.7E-7	
Xe-133			1.5E-6	
I-131			9.9E-7	
I-132			8.4E-5	
I-133			9.3E-6	
I-135			2.9E-5	
Cs-134			1.1E-8	
Rb-88			6.5E-4	
Te-129			1.7E-4	
Te-132			2.5E-6	
Sr-89			1.6E-7	
Ba-140			6.3E-7	
La-140			4.8E-6	
La-142			1.2E-4	
Pr-144			6.7E-4	

Attachment 6.7 - Manual Core Damage Assessment (Page 9 of 18)

Worksheet #2, Measured Specific Activity (µCi/cc) and Sample Temperature Pressure Correction for Containment Atmosphere Sample

At the time of Containment Atmosphere sample isolation record the following: Date/Time__________ Containment Pressure ________ Containment Temperature _______
Time in seconds between reactor shutdown and Containment Sample isolation (t): _________
Determine Temperature / Pressure Correction Factor: $\text{Correction Factor} = \frac{14.2}{(P_1 + 14.2)} \times \frac{(T_1 + 460)}{492} = \frac{14.2}{(\quad +14.2)} \times \frac{(\quad +460)}{492} = \_\_\_\_\_\_$

Isotope	Containment Atmosphere Sample Specific Act.	Temperature Pressure Corrected Spec. Act. (A_{PT})	Decay Constant, λ (1/sec)	Decay Corrected Specific Act. $A_0 = \frac{A_{PT}}{e^{-\lambda t}}$
Kr-87			1.5E-4	
Xe-131m			6.7E-7	
Xe-133			1.5E-6	
I-131			9.9E-7	
I-132			8.4E-5	
I-133			9.3E-6	
I-135			2.9E-5	
Cs-134			1.1E-8	
Rb-88			6.5E-4	
Te-129			1.7E-4	
Te-132			2.5E-6	
Sr-89			1.6E-7	
Ba-140			6.3E-7	
La-140			4.8E-6	
La-142			1.2E-4	
Pr-144			6.7E-4	

Attachment 6.7 - Manual Core Damage Assessment (Page 10 of 18)

Worksheet #3 - Fission Product Release Source Identification

Fission Product Release Source Identification for RCS/LPSI Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #1	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Fission Product Release Source Identification for Containment Atmosphere Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #2	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Attachment 6.7 - Manual Core Damage Assessment (Page 10 of 18)

Worksheet #3 - Fission Product Release Source Identification

Fission Product Release Source Identification for RCS/LPSI Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #1	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Fission Product Release Source Identification for Containment Atmosphere Sample					
Isotope	Decay Corrected Spec. Act. Worksheet #2	Sample Isotope Ratio	Predicted Fuel Pellet Inventory Ratio	Predicted Gas Gap Inventory Ratio	Identified Source Fuel Pellet or Gas Gap
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		1.0	1.0	1.0	N/A
I-131		1.0	1.0	1.0	N/A
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

Attachment 6.7 - Manual Core Damage Assessment (Page 12 of 18)

Worksheet #4, Figure #1, Ratio of H_20 Density to H2O Density at STP vs. Temperature

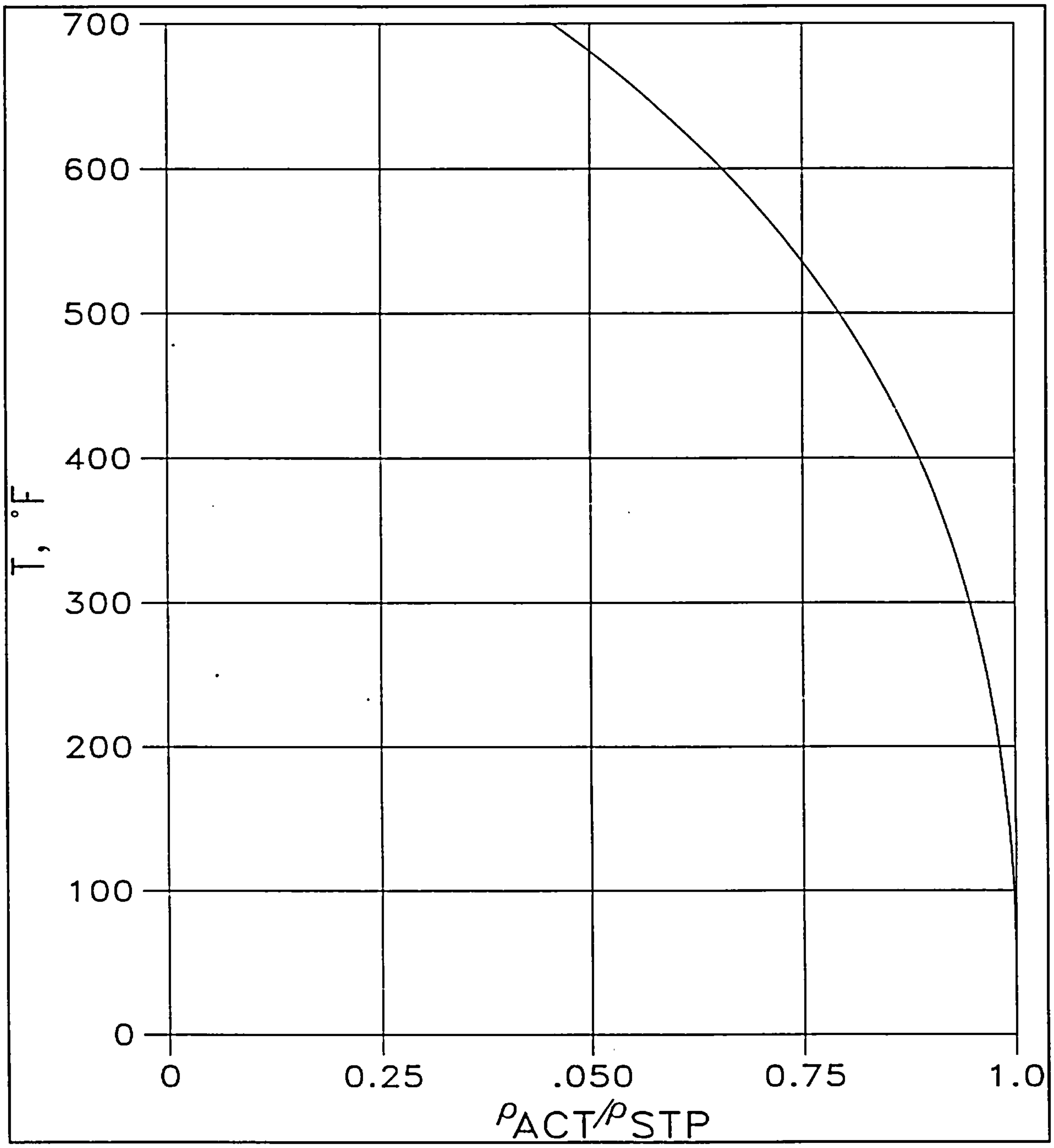
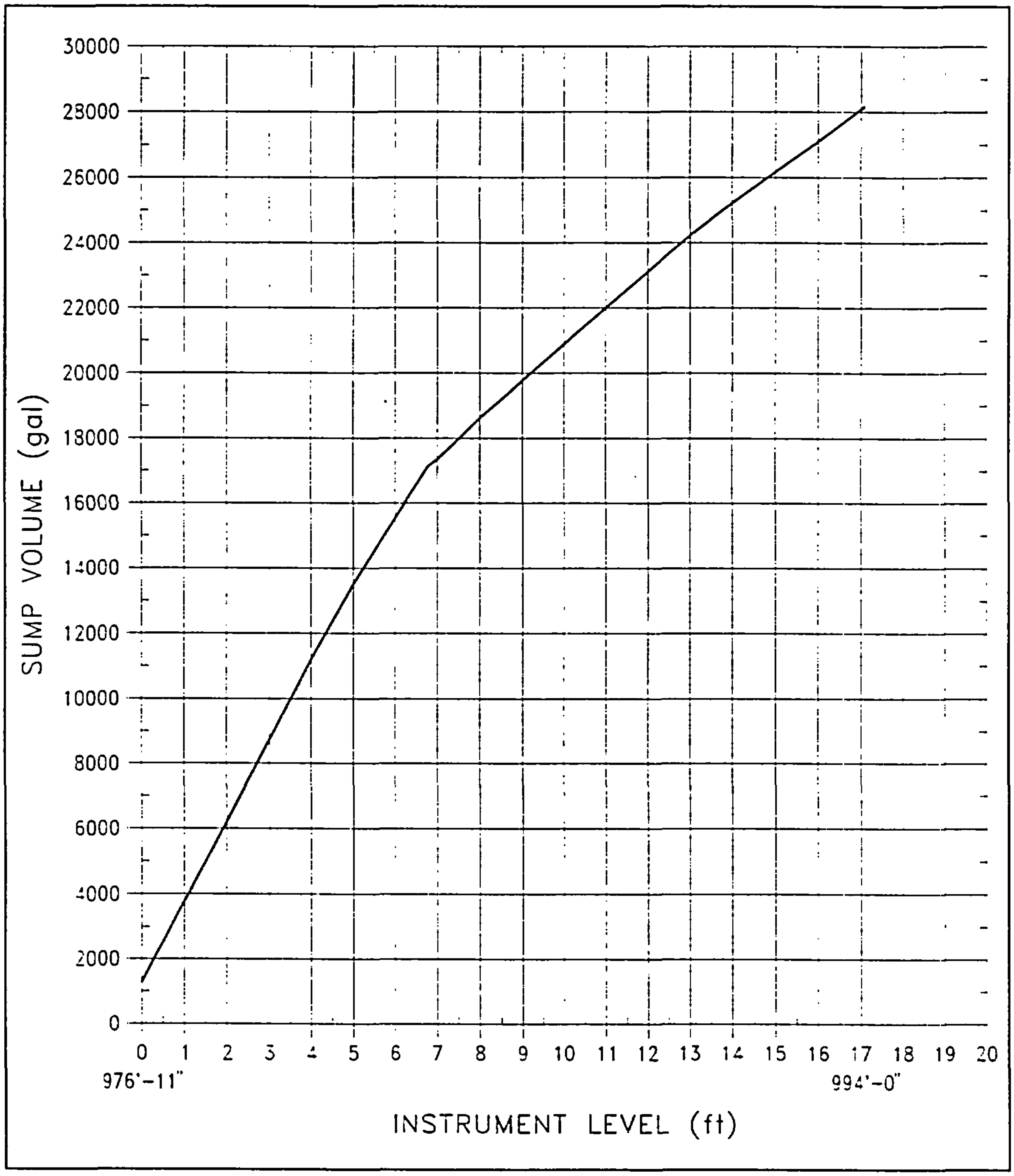
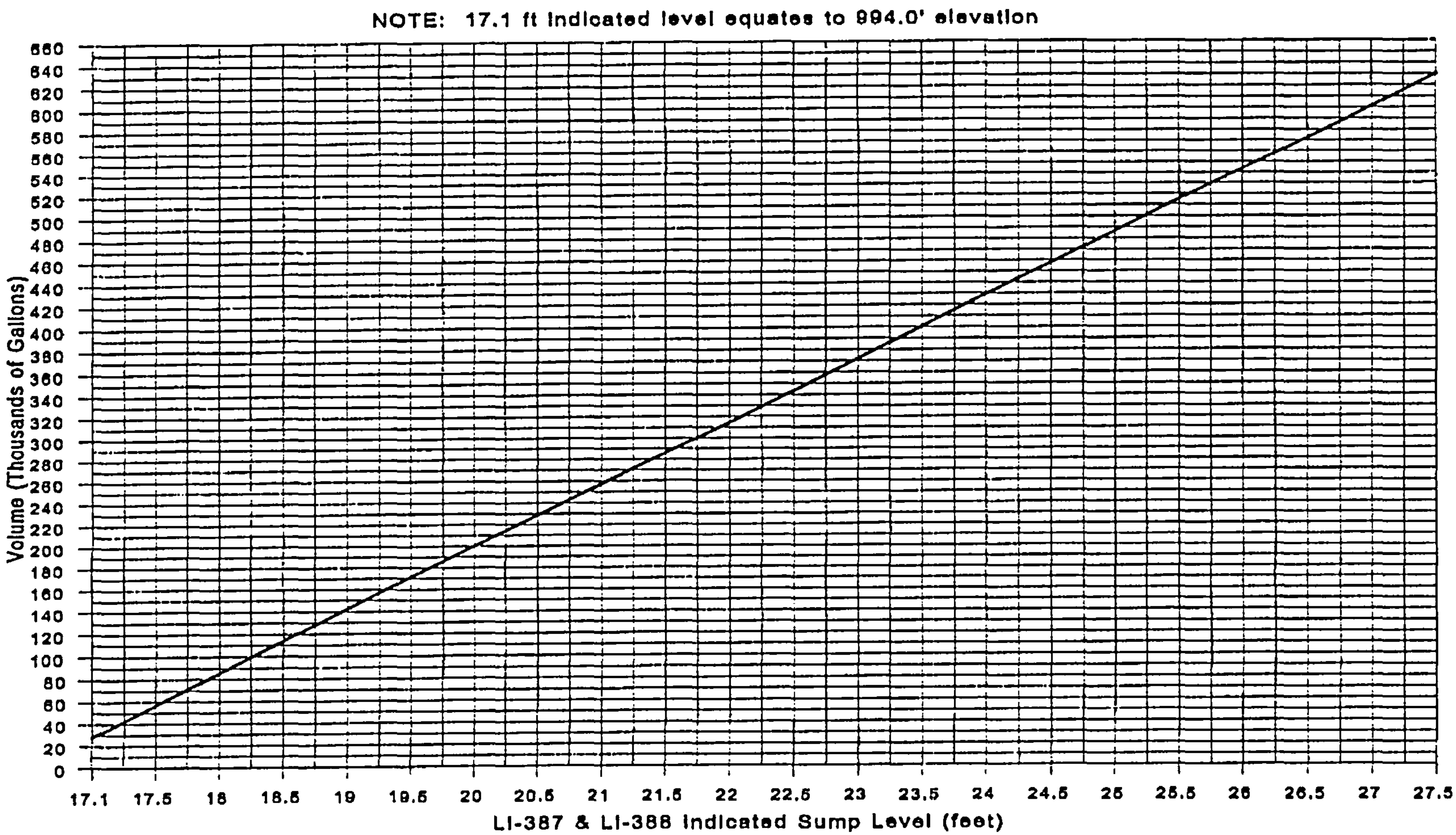


Figure 2, Containment Sump/Basement Volume vs. L-387 or L-388 (Page 1 of 2)



Attachment 6.7 - Manual Core Damage Assessment (Page 14 of 18)

Figure 2, Containment Sump/Basement Volume vs. L-387 or L-388 (Page 2 of 2)



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Worksheet #5, Record of Release Quantity in the Containment

Column 1	Column 2	Column 3	Column 4	Column 5	Column 6
Isotope	Quantity in the Containment Sump (Curies)	Quantity in the RCS (Curies)	Quantity in the Containment Atmosphere (Curies)	Total Quantity in Containment (Curies)	Percent of Source Inventory
Gas Gap Inventory					
Kr-87					
Xe-131m					
Xe-133					
I-131					
I-132					
I-133					
I-135					
Fuel Pellet Inventory					
Kr-87					
Xe-131m					
Xe-133					
I-131					
I-132					
I-133					
I-135					
Cs-134					
Rb-88					
Te-129					
Te-132					
Sr-89					
Ba-140					
La-140					
La-142					
Pr-144					

Attachment 6.7 - Manual Core Damage Assessment (Page 16 of 18)

Worksheet #6, Power Correction Factor (Page 1 of 3)

Column 1	Column 2	Column 3	Column 4
Isotopes (Fuel History Grouping)	Power Correction Factor	Equilibrium Source Inventory (Curies)[1]	Corrected Source Inventory
Gas Gap Inventory			
Kr-87 (2)		9.64E+05	
Xe-131m (1)		2.17E+04	
Xe-133 (1)		3.84E+06	
I-131 (1)		1.93E+06	
I-132 (2)		2.79E+06	
I-133 (2)		3.95E+06	
I-135 (2)		3.70E+06	
Fuel Pellet Inventory			
Kr-87 (2)		1.83E+07	
Xe-131m (1)		4.12E+05	
Xe-133 (1)		7.30E+07	
I-131 (1)		3.67E+07	
I-132 (2)		5.30E+07	
I-133 (2)		7.50E+07	
I-135 (2)		7.02E+07	
Cs-134 (1)		8.59E+07	
Rb-88 (2)		2.61E+07	
Te-129 (2)		1.16E+07	
Te-132 (1)		5.22E+07	
Sr-89 (1)		3.47E+07	
Ba-140 (1)		6.45E+07	
La-140 (1)		6.92E+07	
La-142 (2)		5.89E+07	
Pr-144 (2)		4.86E+07	

NOTE[1]: Source Inventory based on 80% of values in EA-FC-90-111. Gas Gap Inventory is 5% of the source inventory (NUREG 1465). Fuel Pellet inventory is 0.95% of the Source Inventory.

Attachment 6.7 - Manual Core Damage Assessment (Page 17 of 18)

Worksheet #6, Power Correction Factor (Page 2 of 3)

Isotope	Decay Constant (1/sec)	PCF for Period 1	PCF for Period 2	PCF for Period 3	PCF for Period 4
Kr-87 (2)	1.5E-4				
Xe-131m (1)	6.7E-7				
Xe-133 (1)	1.5E-6				
I-131 (1)	9.9E-7				
I-132 (2)	8.4E-5				
I-133 (2)	9.3E-6				
I-135 (2)	2.9E-5				
Cs-134 (1)	1.1E-8				
Rb-88 (2)	6.5E-4				
Te-129 (2)	1.7E-4				
Te-132 (1)	2.5E-6				
Sr-89 (1)	1.6E-7				
Ba-140 (1)	6.3E-7				
La-140 (1)	4.8E-6				
La-142 (2)	1.2E-4				
Pr-144 (2)	6.7E-4				

$$PCF = \frac{P_j (1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of Power Period j to reactor shutdown

Attachment 6.7 - Manual Core Damage Assessment (Page 18 of 18)

Worksheet #6, Power Correction Factor (Page 3 of 3)

Isotope	Decay Constant (1/sec)	PCF for Period 5	PCF for Period 6	PCF for Period 7	PCF for Period 8
Kr-87 (2)	1.5E-4				
Xe-131m (1)	6.7E-7				
Xe-133 (1)	1.5E-6				
I-131 (1)	9.9E-7				
I-132 (2)	8.4E-5				
I-133 (2)	9.3E-6				
I-135 (2)	2.9E-5				
Cs-134 (1)	1.1E-8				
Rb-88 (2)	6.5E-4				
Te-129 (2)	1.7E-4				
Te-132 (1)	2.5E-6				
Sr-89 (1)	1.6E-7				
Ba-140 (1)	6.3E-7				
La-140 (1)	4.8E-6				
La-142 (2)	1.2E-4				
Pr-144 (2)	6.7E-4				

$$PCF = \frac{P_j (1-e^{-\lambda d})e^{-\lambda t}}{100}$$

Where:

P_j = steady state power in period j
d = the duration of the period j in seconds
λ = the decay constant (1/sec)
t = the time in seconds from the end of period j to reactor shutdown

Attachment 6.8 - Clad Damage Characteristics of NRC Categories of Fuel Damage

NRC Category of Fuel Damage	Damage Mechanism	Release Mechanism	Release Source	Characteristic Isotopes	Characteristic Measurement	Measurement Range	Percentage Released From Source
① None	None	Halogen Spiking Tramp Uranium	Gas Gap	I-131, Cs-137, Rb-88			<1
② Initial Cladding Failure	Rupture due to Gas Gap	Clad Burst and Gas Gap	Gas Gap	Xe-131m Xe-133 I-131 I-133	Maximum Core Exit	<1500°F*	<10
③ Intermediate Cladding Failure	Over Pressurization		Gas Gap		Thermocouple Temperature	<1700°F*	10 to 50
④ Major Cladding Failure			Gas Gap			≈< 2300°F ≈ <2% Oxidation	> 50
⑤ Initial Fuel Pellet Overheating	Loss of Structural Integrity Due to Fuel Cladding Oxidation	Grain Boundary Diffusion	Fuel Pellet	Cs-134 Rb-88 Te-129 Te-132	Amount of H_2 Gas Produced (Equivalent to % of Core Oxidation)	Equivalent Core Oxidation < 3%	< 10
⑥ Intermediate Fuel Pellet Overheating			Fuel Pellet			Equivalent Core Oxidation < 18%	10 to 50
⑦ Major Fuel Pellet Overheating		Diffusional Release From UO_2 Grains	Fuel Pellet			Equivalent Core Oxidation [illegible]	>50
⑧ Fuel Pellet Melt	Reactions between UO_2 and solid metallic zircaloy melting of control Rod materials, and zirconium	Escape From Molten Fuel	Fuel Pellet	Sr-89 Ba-140 La-140 La-142 Pr-144			<10
⑨ Intermediate Fuel Pellet Melt			Fuel Pellet				10 to 50
⑩ Major Fuel Pellet Melt			Fuel Pellet				> 50

* Depends on Reactor Pressure and Fuel Burn up. Values given for Pressure _1200 psia and Burn up 0.