Florida Power & Light Company, 6501 South Ocean Drive, Jensen Beach, FL 34957

FPL

April 9, 2001

L-2001-082
10 CFR 50 Appendix E

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

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Emergency Plan Implementing Procedures

In accordance with 10 CFR 50 Appendix E, enclosed is a copy of the revised procedures that implement the Emergency Plan as listed below.

Number	Title	Revision	Implementation Date
EPIP-11	Core Damage Assessment	2	March 27, 2001
HP-200	Health Physics Emergency Organization	16	March 27, 2001
HP-207	Monitoring Evacuated Personnel During Emergencies	11	March 27, 2001

EPIP-11 Revision 2 removed Y2K caution statements. HP-200 Revision 16 added step to Technical Support Center (TSC) Health Physics Supervisor checklist to assist the Emergency Coordinator (EC) with radiological conditions and evaluations of Protective Action Recommendations (PAR). HP-207 Revision 11 revised the name of the Off-site Assembly Area at Jensen Beach and changed social security number (SSN) to thermoluminescent dosimeter (TLD) number on the Frisking Log. All three procedure revisions included editorial/administrative changes.

Please contact us if there are any questions regarding these procedures.

Very truly yours,

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/spt

Enclosures

cc: Regional Administrator, USNRC, Region II (2 copies)
Senior Resident Inspector, USNRC, St. Lucie Plant w/o

A045

FPL	ST. LUCIE PLANT EMERGENCY PLAN IMPLEMENTING PROCEDURE SAFETY RELATED	Procedure No. **EPIP-11** Current Rev. No. **2** Effective Date: **03/27/01**

Title:

CORE DAMAGE ASSESSMENT

Responsible Department: **EMERGENCY PLANNING**

Revision Summary

Revision 2 - Removed Y2K caution statements. Made editorial/administrative changes. (J.R. Walker, 03/22/01)

Revision 1 - Added caution statement to ensure proper use of the core damage assessment program, cord and make it Y2K ready. (R. Walker, 06/30/99)

PSL CONTROL COPY 818 PROCEDURE PRODUCTION

Revision	FRG Review Date	Approved By	Approval Date
0	12/17/97	J. Scarola Plant General Manager	12/17/97
Revision	FRG Review Date	Approved By	Approval Date
2	03/22/01	R. G. West Plant General Manager	03/22/01
		N/A Designated Approver	
		N/A Designated Approver (Minor Correction)	

S___OPS
DATE
DOCT PROCEDURE
DOCN EPIP-11
SYS
COMP COMPLETED
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1.0 PURPOSE

1.1 This procedure identifies the responsibility and methodology to perform core damage assessment for St. Lucie Units 1 and 2. Methods for estimating core damage assessment are based upon post-accident radionuclide concentrations within the Reactor Coolant System (RCS) and containment, and other plant indicators, including core exit thermocouple temperatures, hydrogen in the RCS and in containment, and Containment High Range Radiation Monitor (CHRRM) readings.

1.2 An estimate of core damage may be used to assist in validating Protective Action Recommendations (PARs), severity of plant conditions, and/or recovery operations.

1.3 This procedure incorporates instructions for hand calculations and/or for the use of computer software in the analysis of relevant plant data following an accident.

1.4 This procedure is only used to obtain an estimate of core damage within a major fuel damage category as identified by the NRC in NUREG-0737. The categories are defined in Attachment 1 to this procedure.

1.5 A detailed discussion of the basis for the core damage assessment methodology is included in reference 2.1.2.

2.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS

NOTE

One or more of the following symbols may be used in this procedure:

§ Indicates a Regulatory commitment made by Technical Specifications, Condition of License, Audit, LER, Bulletin, etc., and shall NOT be revised without Facility Review Group review and Plant General Manager approval.

¶ Indicates a management directive, vendor recommendation, plant practice or other non-regulatory commitment that should NOT be revised without consultation with the plant staff.

2.1 References

1. St. Lucie Plant Radiological Emergency Plan.

2. Development of the comprehensive procedure guideline for core damage assessment. CE Owners Group Task 467, July 1983. (Included in Reference 2.1.5).

3. "CORD Version 1A - Core Damage Assessment Computer Program for St. Lucie Units 1 and 2," IMPELL/FPL-85-116, June 3, 1985.

4. JPN Calculation No. PSL-BFJF-91-008, "Determination of Fission Product Source Inventories for PSL for Core Damage Assessment," Rev. 0, Approved 3/11/91.

5. FPL Letter, M. Jimenez to R.D. Mothena, "Core Damage Assessment Procedure, EPIP-1302, Revision 3 Documentation," May 17, 1995, NF-95-330.

6. US-NRC NUREG/BR-0150, Vol. 1, Rev. 3, "Response Technical Manual, RTM-93," November 1993, Page B-16 (included in Reference 2.1.5).

2.0 REFERENCES/RECORDS REQUIRED/COMMITMENT DOCUMENTS (continued)

2.2 Records Required

1. During an actual emergency, information used to estimate core damage, including appropriate worksheets, will be maintained by the Emergency Technical Manager or his staff designee at the Emergency Operations Facility (EOF), or by the Reactor Engineer in the Technical Support Center (TSC).

2. All written information will be forwarded to the Emergency Preparedness representative at the TSC or EOF.

2.3 Commitment Documents

1. Clarification of TMI Action Plan Requirements. NUREG 0737, Item II.B.3.

3.0 RESPONSIBILITIES

3.1 The Emergency Technical Manager ensures the performance of core damage assessment using the methodology in this procedure.

3.2 The EOF Nuclear Fuels Engineer performs core damage assessment using the guidelines in this procedure and engineering judgement.

4.0 DEFINITIONS

4.1 No Core Damage refers to a core state in which the integrity of the fuel rod cladding is intact and the only release of fission products to the Reactor Coolant System is that due to pre-existing fuel rod defects and iodine spiking.

Fuel Rod Cladding Failure refers to a core state in which the fuel rod cladding of some fraction of the fuel rods in the core has failed, resulting in the release of the fission products in the fuel rod gap space of the failed fuel rods to the Reactor Coolant System.

Fuel Overtemperature Damage refers to a core state in which the fuel pellets have reached a temperature where there is a rapid movement of fission products from the fuel pellet matrix to the Reactor Coolant System.

100% Fuel Rod Clad Damage refers to the rupture of the fuel rod cladding in 100% of the fuel rods in the core and the resultant release to the Reactor Coolant System of all fission products contained in the fuel rod gap space.

100% Fuel Overtemperature Damage refers to high temperatures in the fuel pellets in 100% of the fuel rods in the core and the resultant release to the Reactor Coolant System of fission products contained in the fuel pellet matrix.

Emergency Response Data Acquisition and Display System (ERDADS) also known as the Safety Assessment System (SAS) and includes the Safety Parameter Display System (SPDS) serves as a concentrated data source that permits EOF personnel to obtain desired information (plant parameter, radiological, meteorological, etc.) in a rapid, accurate, and convenient manner.

5.0 INSTRUCTIONS

NOTE

Available pertinent plant data needed to perform the core damage assessment should be provided through the ERDADS and/or communications with the TSC.

5.1 The EOF Nuclear Fuels Engineer will perform the core damage estimate using the methodology described in this procedure.

NOTE

- Computer generated estimate is the preferred option for assessing core damage, since the hand calculations are lengthy and complex.
- The hand calculation methods, Attachments 4, 5, 6, 7 and 8, are provided for backup purposes.

5.2 Core damage assessment will be performed using Attachment 2.

1. Attachment 2 provides instructions for the execution of computer programs to determine assessment of core damage.
2. The computer software test case is provided in Attachment 3.
3. When needed, the TSC staff may perform a core damage estimate using the indicators discussed in Attachments 4, 5, 6 and 7.

5.3 All pertinent data available should be used in estimating core damage, including the following:

1. Radionuclide data
2. Auxiliary indicators
   - **A.** Core Exit Thermocouple (CET) temperature
   - **B.** Hydrogen in the RCS and containment
   - **C.** Containment High Range Radiation Monitor (CHRRM) readings

5.0 INSTRUCTIONS (continued)

CAUTION

- The assessment of core damage obtained by using the attached methodology is only an estimate. The techniques employed are only accurate to locate the core condition within one or more of the 10 categories of core damage described in Table 1 in Attachment 1.
- Core damage assessment using indicators that are readily available (e.g., CHRRM) represents only preliminary estimates. Other plant indicators (e.g., radionuclide concentrations) should be obtained to improve upon estimation of core damage.
- Measurements obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage. If deemed necessary, these pertinent indicators should be measured within a minimum time period, particularly during rapidly changing conditions. It is recommended that measurements be made, if possible, when plant conditions stabilize.

5.4 Results in terms of fuel condition should be provided to the Emergency Technical Manager (ETM), the Recovery Manager (RM), and the Emergency Coordinator (EC) as timely as possible.

1. The type of core damage is described in terms of the 10 NRC categories defined in Table 1 in Attachment 1.

2. In the case of radionuclide analysis, the degree of core damage is described as the percent of the fission products in the source inventory at the time of the accident which is now in the sampled fluid and therefore available for release to the environment.

5.0 INSTRUCTIONS (continued)

5.5 Updated estimates of core damage may be requested periodically by the ETM, the RM or the EC as plant conditions change and/or stabilize.

1. These updates should be performed using the most recent available data.

2. Results shall continue to be reported to the ETM, the RM and EC.

END OF SECTION 5.0

ATTACHMENT 1
CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE
TABLE 1. CHARACTERISTIC ISOTOPES

GENERAL CATEGORY	NRC CATEGORY OF FUEL DAMAGE	MECHANISM OF RELEASE	SOURCE OF RELEASE	CHARACTERISTIC ISOTOPE	RELEASE OF CHARACTERISTIC ISOTOPE EXPRESSED AS PERCENT OF SOURCE INVENTORY
Normal Operation	1. No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I-131, Cs-137 Rb 88	Less than 1
Core Damage	2. Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Xe-131m, Xe-133, I-131, I-133	Less than 10
	3. Intermediate Cladding Failure		Gas Gap		10 to 50
	4. Major Cladding Failure		Gas Gap		Greater than 50
Severe Core Damage	5. Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Cs-134, Rb-88, Te-129, Te-132	Less than 10
	6. Intermediate Fuel Pellet Overheating		Fuel Pellet		10 to 50
	7. Major Fuel Pellet Overheating	Diffusional Release Fuel UO_2 Grains	Fuel Pellet		Greater than 50
	8. Fuel Pellet Melt	Escape from Molten Fuel	Fuel Pellet	Ba-140, La-140, La-142, Pr-144	Less than 10
	9. Intermediate Fuel Pellet Melt		Fuel Pellet		10 to 50
	10. Major Fuel Pellet Melt		Fuel Pellet		Greater than 50

ATTACHMENT 1
CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE
TABLE 2. CLADDING DAMAGE CHARACTERISTICS

NRC Category of Fuel Damage	Temperature Range (F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damage Rods
1. No Fuel Damage	approximately 750	None	N/A	N/A	less than 1
2. Initial Cladding Failure	1200 to 1800	Rupture Due to Gas Gap Over-pressurization	Maximum Core Exit Thermocouple Temperature	less than 1550 F *	less than 10
3. Intermediate Cladding Failure				less than 1700 F *	10 to 50
4. Major Cladding Failure				less than 2300 F less than 2 percent Oxidation	greater than 50
5. Initial Fuel Pellet Overheating	1800 to 3350	Loss of Structural Integrity Due to Fuel Clad Oxidation	Amount of Hydrogen Gas Produced (Equivalent to Percent Oxidation of Core)	Equivalent Core Oxidation less than 3 percent	less than 10
6. Intermediate Fuel Pellet Overheating				less than 18 percent	10 to 50
7. Major Fuel Pellet Overheating				less than 65 percent	greater than 50

* Depends on Reactor Pressure and Fuel Burnup Values Given for Pressure less than or equal to 1200 psia and Burnup greater than or equal to 0.

ATTACHMENT 1

CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

TABLE 3. PERCENT OF SOURCE INVENTORY RELEASED TO CONTAINMENT

NRC CATEGORY OF FUEL DAMAGE	MECHANISM OF RELEASE FROM CORE	SOURCE OF RELEASE	PERCENT OF SOURCE INVENTORY RELEASED TO CONTAINMENT	DISTRIBUTION OF FISSION PRODUCTS IN CONTAINMENT
1. No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	Less than 1	Airborne
2. Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release	Gas Gap	Less than 10	Airborne
3. Intermediate Cladding Failure		Gas Gap	10 to 50	Airborne
4. Major Cladding Failure		Gas Gap	Greater than 50	Airborne
5. Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Less than 10	Airborne: 100 percent Noble Gas 25 percent Halogen Plated Out: 25 percent Halogen 1 percent Solids
6. Intermediate Fuel Pellet Overheating		Fuel Pellet	10 to 50	
7. Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains	Fuel Pellet	Greater than 50	

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 1 of 10)

1. Purpose

This section provides the instructions for the use of the computer code CORD in performing core damage assessment (Reference 2.1.3). This code automates the functions described in Attachments 4 through 8.

2. Precautions and Limitations

A. Assigned engineers are responsible to follow the instructions of this procedure whenever performing core damage assessment for St. Lucie Units 1 and 2.

B. Prior to use of the code, validation must be performed by running the benchmark cases provided in Attachment 3.

/R2

3. Specific Instructions

Read and become familiar with the detailed user instructions provided in paragraph 1D of this attachment. These user instructions are generic in nature and will provide the user with a general understanding of how CORD works and description of the input types and editing keys. The instructions are designed to complement the user instructions and minimize the need for familiarity in the event of an actual emergency. Consequently, these instructions are more specific to the hardware equipment designated for core damage assessment use.

A. Set up the computer and printer.

B. Execute the computer program CORD (or later revision name).

C. Perform program validation by running the benchmark cases provided in Attachment 3.

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 2 of 10)

3. Specific Instructions (continued)

   D. Obtain from ERDADS and/or other available data source the following information:

      1. Unit, date and time of reactor shutdown

      2. Power history prior to accident

      3. Core exit temperatures

      4. Containment radiation dose rates, and

      5. PASS (Post-Accident Sampling System) sample and whether it is corrected to standard temperature and pressure (STP).

   E. Begin core damage assessment by choosing Option 7 to select the appropriate unit. Proceed to execute Options 1 through 4 as data becomes available. Based on typical accessibility of data, the most likely sequence is as follows:

      1. Option 3 - "Core Exit Temperature"

      2. Option 4 - "Radiation Dose Rate"

      3. Option 1 - "Radiological Analysis"

      4. Option 2 - "Hydrogen"

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 3 of 10)

3. Specific Instructions (continued)

F. Running Option 3 (Core Damage Assessment Using Core Exit Temperatures)

1. Enter maximum core thermocouple temperature (°F). Note that if this temperature is significantly higher than the average, it may indicate a faulty thermocouple. In this case, disregard the abnormally high reading and use the average of the rest of core exit thermocouple temperatures.

2. Enter RCS pressure (psia) corresponding to the time of the temperature reading.

3. Review the calculated percent of ruptured clad against those included on Table 1 in Attachment 1 to determine the appropriate NRC damage category. Note the caution and note included in the CORD output page for this option.

G. Running Option 4 (Core Damage Assessment Using Radiation Dose Rate)

/R2

1. Choose "1" to retrieve previous input data. Revise the input data with new information. Enter date of reactor shutdown (mm-dd-yr) and time in military time (00:00).

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 4 of 10)

3. Specific Instructions (continued)

G. Running Option 4 (Core Damage Assessment Using Radiation Dose Rate)

2. Enter representative power level in percent using engineering judgement. Note that the most recent power levels should be weighted more than the past levels.

3. Enter the higher of the two measured containment dose rates (Rad/Hr) with corresponding dates and times.

4. Print screen and review the calculated results against the correlations included on Figure 5-1, Containment High Range Monitor Dose Rate vs. Time After Trip, to confirm the appropriate NRC damage category.

5. Continue to execute this option as more data becomes available by adding new sets of data as in Step 3.G.2.

H. Running Option 1 (Radiological Analysis of Samples) /R2

1. Choose "1" to retrieve previous input data. Revise the input data with new information. Enter date of reactor shutdown (mm-dd-yr) and time in military time (00:00).

2. Enter power history, including power level in percent and number of days at each level, ending with the most recent power level.

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 5 of 10)

3. Specific Instructions (continued)

H. Running Option 1 (Radiological Analysis of Samples) (continued)

3. Enter PASS sample data as available for: RCS Hot Leg, Containment Atmosphere or Containment Sump. This data consists of measured activity in microCuries per gram (μCi/g).

4. Enter proper response for correction to STP in accordance with information provided with the PASS sample data.

5. Perform decay correction as appropriate by entering "yes."

6. Press the F1 key to continue through the "RECORD OF DECAY CORRECTION ACTIVITY RATIOS."

7. Print screen the "RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION" and determine the appropriate source (gas gap or fuel pellet) by comparing the calculated ratios to those in Data Sheet 8-3, Record of Fission Product Release Source Identification.

8. Press the F1 to continue and enter the following information as prompted by the program:

   - reactor water level (full, void, or below recorder)

   - Safety Injection Tank (SIT) volume injected (gallons)

   - Boric Acid Make-up Tank (BAMT) volume injected (gallons)

   - change in Refueling Water Tank (RWT) volume (gallons)

   This information is obtained from Mechanical Engineering at the EOF.

9. Press return to obtain the "RECORD OF RELEASE QUANTITY." Print screen and press F1 to obtain the "RELEASE (percent) OF GAS GAP AND FUEL PELLET INVENTORY."

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 6 of 10)

3. Specific Instructions (continued)

   H. Running Option 1 (Radiological Analysis of Samples) (continued)

      10. Print screen and use these results in conjunction with the isotope ratio evaluation of Step 3.H.7 to determine the category of core damage in accordance with Table 1, Characteristic Isotopes, in Attachment 1.

   I. Running Option 2 (Core Damage Assessment Using Hydrogen)

      1. Choose "1" to retrieve previous input data. Revise the input data with new information. Enter percent volume of Hydrogen in containment and temperature and pressure at sampling.

      2. Enter post-accident containment temperature history as available.

      3. Enter RCS sample information as prompted. Note that the input requires an estimate of core damage based on the evaluation of other parameters (Options 1, 3 and 4).

      4. Enter data on reactor vessel head void, including estimate of void volume.

      5. Continue by pressing the F1 key to obtain a summary of the Hydrogen analysis. Use these results along with Table 2, Cladding Damage Characteristics, in Attachment 1 to determine the category of core damage.

4. Generic CORD User Instructions

   A. Introduction

      CORD is a computer program which performs the calculations for the St. Lucie Units 1 and 2 in accordance with this procedure. The program is compiled using IBM compiler BASIC and can be run using the IBM BASIC interpreter. The CORD diskette contains the following files:

CORD.BAS	The CORD program source BASIC source code
CORD.EXE	The CORD executable file
CORDPSL1.DAT	The St. Lucie Unit 1 data file
CORDPSL2.DAT	The St. Lucie Unit 2 data file

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 7 of 10)

4. Generic CORD User Instructions (continued)

B. Getting Started

To use the CORD program, take the following steps:

1. Boot up the computer using DOS 2.0 or higher version.

2. Insert the CORD diskette in a PC drive.

3. Proceed to load the program by typing CORD, the main menu should appear.

C. Program Options

The main menu for CORD contains the following options:

1. RADIOLOGICAL ANALYSIS OF SAMPLES
2. CORE DAMAGE ASSESSMENT USING HYDROGEN
3. CORE DAMAGE ASSESSMENT USING CORE EXIT TEMPERATURES
4. CORE DAMAGE ASSESSMENT USING RADIATION DOSE RATE
5. UPDATE EQUILIBRIUM SOURCE INVENTORY
6. EXIT PROGRAM
7. TOGGLE FOR APPLICABLE UNIT

The first four options correspond to the four types of core damage assessment calculations outlined in this procedure. The inputs and calculations will not be discussed here, but are described elsewhere in this procedure.

The fifth option allows the user to change the equilibrium RCS sources used by Option 1. Once changed, the old data is discarded and all future execution of the program will use the latest equilibrium source data entered. Note that the old data can be preserved by copying the data file to another file name before executing the program.

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 8 of 10)

4. Generic CORD User Instructions (continued)

C. Program Options (continued)

The user selects Option 6 to exit the program and return to the DOS operating system.

The calculations are identical for St. Lucie Units 1 and 2, but each unit will have different input data. The user selects Option 7 to specify the unit for the current run.

D. Data Files

The two data files "CORDPSL1.DAT" and "CORDPSL2.DAT" store the most recently entered equilibrium source data and program input data for Units 1 and 2, respectively. Most Options of the program will ask the user if the calculations are to use the last data set or whether a new data set is to be entered. If the last data set option is selected, the data is recalled from the appropriate data file for the selected unit and is used as the default entry for all inputs. When a new data set is entered, it will be written over the data currently in the data file.

E. Input Types

The CORD program inputs are of four basic types: numeric, data, time, and yes/no responses.

numeric data — Numbers can be entered as integers, floating point numbers or in scientific notation. Examples of acceptable formats for numeric entries are: -123, 1.23, .123, 1.2E-4, and -1.23E-4. The letter "E" means "times 10 to the power of." Numbers will be right justified in the input field if accepted by the program.

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 9 of 10)

4. Generic CORD User Instructions (continued)

E. Input Types (continued) /R2

dates — All date entries in CORD are in the MM-DD-YY format, where MM = two digit month, DD = two digit day, and YY = two digit year.

The "-" are optional and can be replaced by a "/" or a space. Examples of acceptable date inputs using April 2, 1985 are: 4/02/85, 40285, 4-02-85, and 4 2 85.

times — Time entries are assumed to be military time ranging from 0:00 to 23:59. Acceptable entries are: 100, 1:00, 14:23 and 1630.

yes/no — Answers to "yes / no" questions are either "Y" or "y" for "yes," or "N" or "n" for "no."

ATTACHMENT 2

CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD

(Page 10 of 10)

4. Generic CORD User Instructions (continued)

F. Editing Keys

Most entries to the CORD program are made on input screens filled with data entry fields. These fields are the white background areas of the screen. The program limits the user to typing within the field areas, but also provides special editing keys for the user to move from field to field.

Key	Function
ESC	Clears the input field and places the cursor in the left most location within the field
BACKSPACE	Deletes the character to the left of the cursor
DEL	Deletes the character at the current cursor location
RETURN	Concludes the current entry and moves the cursor to the next field
HOME	Moves the cursor to the first field on the screen
END	Moves the cursor to the last field on the screen
UP ARROW	Moves the cursor to the previous field
DOWN ARROW	Moves the cursor to the next field (performs the same as a RETURN)
LEFT ARROW	Moves the cursor one space left
RIGHT ARROW	Moves the cursor one space right
FUNCTION KEYS	The function keys (F1 through F10) have special uses identified at the bottom of the input screen

ATTACHMENT 3
SOFTWARE TEST CASE
FOR
ST. LUCIE UNITS 1 & 2

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 1 of 14)

PROGRAM INPUT FOR OPTION 1
(Page 1 of 2)
GENERAL INFORMATION

ENTER DATE AND TIME OF REACTOR SHUTDOWN

DATE: 7/18/84
TIME: 1:00

% POWER	NO. OF DAYS
75	22
50	17
100	2

These entries should be in chronological order. The last entry is the interval prior to reactor shutdown.

RECORD OF SAMPLE SPECIFIC ACTIVITY

	RCS HOT LEG	CONT. ATMOS.	CONT. SUMP
Sample Number:	001	002	003
Date of Analysis:	7/18/84	7/18/84	7/18/84
Time of Analysis:	4:00	4:00	4:00
Temperature, Deg F:	300	150	150
Pressure, PSIG:	1600	.5	.5

SAMPLE ACTIVITIES (Ci/cc)				
	KR87	1	.01	.1
	XE131M	1	.01	.1
	XE133	100	.1	.00001
	I131	10000	.1	100
	I132	1	.01	.1
	I133	100	.001	.1
	I135	1	.01	.1
	CS134	1	.01	.1
	RB88	1	.01	.1
	TE129	1000	.01	10
	TE132	1	.01	.1
	SR89	1	.01	.1
	BA140	1	.01	.1
	LA140	1	.01	.1
	LA142	10	.01	.1
	PR144	1	.01	.1

F1 = DONE F3 = PREV SCREEN F10 = QUIT

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 2 of 14)

PROGRAM INPUT FOR OPTION 1
(Page 2 of 2)
GENERAL INFORMATION

ENTER REACTOR LEVEL CONDITION:

1. FULL
2. VOID
3. BELOW RECORDER

ENTER 1, 2, OR 3) 1

ENTER SAFETY INJECTION TANK VOLUME INJECTED IN GALLONS) 0

ENTER BORIC ACID MAKEUP TANK VOLUME INJECTED IN GALLONS) 0

ENTER CHANGE IN VOLUME OF THE REFUELING WATER TANK IN GALLONS) 0

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 3 of 14)

OPTION 1 (OUTPUT)
(Page 1 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:20

ENCLOSURE A4 - UNIT: PSL1
RECORD OF DECAY CORRECTION

TIME OF REACTOR SHUTDOWN: 7/18/84 1:00
(*) - indicates that decay time is too long to back calculate concentration

Isotope	Decay Const (1/SEC)	RCS		CONT ATMOS		CONT SUMP	
		@STP (µCi/cc)	CORRECTED (µCi/cc)	@STP (µCi/cc)	CORRECTED (µCi/cc)	@STP (µCi/cc)	CORRECTED (µCi/cc)
KR87	1.5E-04	1.04E+00	5.24E+00	1.20E-02	6.06E-02	1.00E-01	5.05E-01
XE131M	6.7E-07	1.04E+00	1.04E+00	1.20E-02	1.21E-02	1.00E-01	1.01E-01
XE133	1.5E-06	1.04E+02	1.05E+02	1.20E-01	1.22E-01	1.00E-05	1.02E-05
I131	9.9E-07	1.04E+04	1.05E+04	1.20E-01	1.21E-01	1.00E+02	1.01E+02
I132	8.4E-05	1.04E+00	2.57E+00	1.20E-02	2.97E-02	1.00E-01	2.48E-01
I133	9.3E-06	1.04E+02	1.15E+02	1.20E-03	1.33E-03	1.00E-01	1.11E-01
I135	2.9E-05	1.04E+00	1.42E+00	1.20E-02	1.64E-02	1.00E-01	1.37E-01
CS134	1.1E-08	1.04E+00	1.04E+00	1.20E-02	1.20E-02	1.00E-01	1.00E-01
RB88	6.5E-04	1.04E+00	1.16E+03	1.20E-02	1.34E-01	1.00E-01	1.12E+02
TE129	1.7E-04	1.04E+03	6.51E+03	1.20E-02	7.52E-02	1.00E+01	6.27E+01
TE132	2.5E-06	1.04E+00	1.07E+00	1.20E-02	1.23E-02	1.00E-01	1.03E-01
SR89	1.6E-07	1.04E+00	1.04E+00	1.20E-02	1.20E-02	1.00E-01	1.00E-01
BA140	6.3E-07	1.04E+00	1.04E+00	1.20E-02	1.21E-02	1.00E-01	1.01E-01
LA140	4.8E-06	1.04E+00	1.09E+00	1.20E-02	1.26E-02	1.00E-01	1.05E-01
LA142	1.2E-04	1.04E+01	3.79E+01	1.20E-02	4.38E-02	1.00E-01	3.65E-01
PR144	6.7E-04	1.04E+00	1.44E+03	1.20E-02	1.66E-01	1.00E-01	1.39E+02

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 4 of 14)

OPTION 1 (OUTPUT)
(Page 2 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:24

UNIT: PSL1
USE THESE RATIOS TO DETERMINE SOURCE OF RELEASE BY COMPARING THE RESULTS TO THE PREDICTED RATIOS IN ENCLOSURE A5.

NOBLE GAS RATIOS:

	RCS SAMPLE	CONT ATMOS	SUMP
KR87	0.0497	0.4972	%49718.9020
XE131M	0.0099	0.0991	%9910.7617
XE133	1.0000	1.0000	1.0000

IODINES:

I131	1.0000	1.0000	1.000
I132	0.0002	0.2451	0.0025
I133	0.0109	0.0109	0.0011
I135	0.0001	0.1353	0.0014

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 5 of 14)

OPTION 1 (OUTPUT)
(Page 3 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:28

ENCLOSURE A5 - UNIT: PSL1
RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

SAMPLE NUMBER: 001
LOCATION: RCS HOT LEG

Isotope	Decay Corr Spec Activity (Encl A4) µCi/cc	Calculated Isot Ratio	Fuel Pellet Inventory	ACT Ratio in Gas Gap	Identified Source
KR87	5.24E+00	4.97E-02	0.2	0.001	
XE131M	1.04E+00	9.91E-03	0.003	0.001 - 0.003	
XE133	1.05E+02	1.00E+00	1.0	1.0	
I131	1.05E+04	1.00E+00	1.0	1.0	
I132	2.57E+00	2.45E-04	1.4	0.01 - 0.05	
I133	1.15E+02	1.09E-02	2.0	0.5 - 1.0	
I135	1.42E+00	1.35E-04	1.8	0.1 - 0.5	

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 6 of 14)

OPTION 1 (OUTPUT)
(Page 4 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:32

ENCLOSURE A5 - UNIT: PSL1
RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

SAMPLE NUMBER: 002
LOCATION: CONTAINMENT ATMOSPHERE

Isotope	Decay Corr Spec Activity (Encl A4) µCi/cc	Calculated Isot Ratio	Fuel Pellet Inventory	ACT Ratio in Gas Gap	Identified Source
KR87	6.06E-02	4.97E-01	0.2	0.001	
XE131M	1.21E-02	9.91E-02	0.003	0.001 - 0.003	
XE133	1.22E-01	1.00E+00	1.0	1.0	
I131	1.21E-01	1.00E+00	1.0	1.0	
I132	2.97E-02	2.45E-01	1.4	0.01 - 0.05	
I133	1.33E-03	1.09E-02	2.0	0.5 - 1.0	
I135	1.64E-02	1.35E-01	1.8	0.1 - 0.5	

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 7 of 14)

OPTION 1 (OUTPUT)
(Page 5 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:35

ENCLOSURE A5 - UNIT: PSL1
RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION

SAMPLE NUMBER: 003
LOCATION: CONTAINMENT SUMP

Isotope	Decay Corr Spec Activity (Encl A4) µCi/cc	Calculated Isot Ratio	Fuel Pellet Inventory	ACT Ratio in Gas Gap	Identified Source
KR87	5.05E-01	4.97E+04	0.2	0.001	
XE131M	1.01E-01	9.91E+03	0.003	0.001 - 0.003	
XE133	1.02E-05	1.00E+00	1.0	1.0	
I131	1.01E+02	1.00E+00	1.0	1.0	
I132	2.48E-01	2.45E-03	1.4	0.01 - 0.05	
I133	1.11E-01	1.09E-03	2.0	0.5 - 1.0	
I135	1.37E-01	1.35E-03	1.8	0.1 - 0.5	

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 8 of 14)

OPTION 1 (OUTPUT)
(Page 6 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:46

ENCLOSURE A7 - UNIT: PSL1
RECORD OF RELEASE QUANTITY

Isotope	Reactor Coolant Sample Number, 1	Containment Sump Sample Number, 2	Contain Atmosphere Sample Number, 3	Total Quantity
KR87	1.49E+03	1.43E+02	3.58E+03	5.21E+03
XE131M	2.97E+02	2.86E+01	7.13E+02	1.04E+03
XE133	2.99E+04	2.89E-03	7.19E+03	3.71E+04
I131	2.98E+06	2.87E+04	7.16E+03	3.01E+06
I132	7.30E+02	7.03E+01	1.75E+03	2.55E+03
I133	3.26E+04	3.14E+01	7.83E+01	3.27E+04
I135	4.03E+02	3.88E+01	9.68E+02	1.41E+03
CS134	2.95E+02	2.84E+01	7.08E+02	1.03E+03
RB88	3.29E+05	3.18E+04	7.92E+05	1.15E+06
TE129	1.85E+06	1.78E+04	4.44E+03	1.87E+06
TE132	3.03E+02	2.92E+01	7.27E+02	1.06E+03
SR89	2.95E+02	2.84E+01	7.09E+02	1.03E+03
BA140	2.97E+02	2.86E+01	7.13E+02	1.04E+03
LA140	3.10E+02	2.99E+01	7.46E+02	1.09E+03
LA142	1.08E+04	1.04E+02	2.59E+03	1.35E+04
PR144	4.09E+05	3.94E+04	9.83E+05	1.43E+06

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 9 of 14)

OPTION 1 (OUTPUT)
(Page 7 of 7)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 01-29-1993
13:09:50

UNIT: PSL1
RELEASE OF GAS GAP AND FUEL PELLET INVENTORY

Isotope	GAS GAP		FUEL PELLET	
	Corrected Source Inv	% Rel	Corrected Source Inv	% Rel
KR87	1.48E+05	3.52	3.67E+07	0.01
XE131M	2.35E+04	4.43	4.03E+05	0.26
XE133	3.20E+06	1.16	8.09E+07	0.05
I131	4.23E+06	71.30	3.64E+07	8.28
I132	1.36E+06	0.19	9.61E+07	0.00
I133	5.02E+06	0.65	1.21E+08	0.03
I135	3.12E+06	0.05	1.27E+08	0.00
CS134			1.94E+05	0.53
RB88			5.28E+07	2.18
TE129			2.09E+07	8.94
TE132			6.52E+07	0.00
SR89			1.96E+07	0.01
BA140			6.78E+07	0.00
LA140			1.01E+08	0.00
LA142			1.11E+08	0.01
PR144			8.46E+07	1.69

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 10 of 14)

OPTION 2 (INPUT)
(Page 1 of 1)

CONTAINMENT SAMPLE INFORMATION

PERCENT VOLUME OF H2: .424 %
CONTAINMENT TEMP AT SAMPLING: 220 F
CONTAINMENT PRES AT SAMPLING: .5 PSIG
IS SAMPLE CORRECTED TO STP?: Y (Y=YES/N=NO)

TIME (HR)	TEMP (DEG F)
1:00	250
1:30	350
2:00	260
3:00	240
4:00	220

F1=DONE F10=QUIT

RCS SAMPLE INFORMATION

QUANTITY OF HYDROGEN: 1200 cc/kg
RCS TEMP AT SAMPLING: 300 F
RCS PRES AT SAMPLING: 1600 PSIG
IS SAMPLE CORRECTED TO STP: Y (Y=YES/N=NO)
REPRESENTATIVE POWER LEVEL: 50%
RCS PRES DURING UNCOVERY: 1000 PSIA
ESTIMATE OF FUEL OVERHEAT: 1 (1=INITIAL, 2=INTERMEDIATE, 3=MAJOR)

HYDROGEN IN REACTOR VOID

ESTIMATE OF VOID VOLUME: 0 cuft
TEMPERATURE OF LIQUID AT COOLANT SURFACES: 0 deg F
RCS PRESSURE: 0 psia
IS SAMPLE CORRECTED TO STP?: N (Y=YES/N=NO)

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 11 of 14)

OPTION 2 (OUTPUT)
(Page 1 of 1)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 02-10-1993
VERSION 1A (5/31/85) 14:09:59

SUMMARY OF HYDROGEN ANALYSIS - UNIT: PSL1

HYDROGEN IN CONTAINMENT ATMOSPHERE = 10599 cuft H2
HYDROGEN IN REACTOR COOLANT = 12480 cuft H2
HYDROGEN IN REACTOR VOID SPACE = 0 cuft H2
TOTAL HYDROGEN RELEASED = 23079 cuft H2

TOTAL H2 BY CONTAINMENT MATERIAL OXIDATION = 12520 cuft H2
UPPER LIMIT BY HYDROGEN MAJOR OVERHEAT = 1952 cuft H2
LOWER LIMIT BY H2 INITIAL OVERHEAT = 732 cuft H2
VALUE USED FOR RADIOLYSIS OF WATER = 732 cuft H2

TOTAL ESTIMATE OF CORE CLAD OXIDATION = 9826.803 cuft H2 2.33%

EST PERCENT OF FUEL WITH RUPTURED CLAD = 100.00%

UPPER EST % FUEL WITH EMBRITTLED CLAD = 21.05%
LOWER EST % FUEL WITH EMBRITTLED CLAD = 9.05%

USE THESE RESULTS FOR % RUPTURED CLAD AND % EMBRITTLED CLAD ALONG WITH ATTACHMENT 1 TO DETERMINE EXTENT OF CLAD DAMAGE.

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 12 of 14)

OPTION 3 (INPUT AND OUTPUT)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 02-10-1993
14:13:43

UNIT: PSL1

Input Parameters:
Temperature (max) = 2000 deg F
Pressure @ T-max = 900 psia
ESTIMATE OF PERCENT RUPTURED CLADDING BASED ON CETs = 95.68%

CAUTION
Estimates predicted by the methodology in this procedure are good if T-max remains below 1800°F during core uncovery and if the core remains uncovered for 20 minutes or longer. Estimates could be LOW if pressure during period of T-max drops to less than 100 psia within less than 2 minutes of accident initiation, a large break is indicated.

NOTE
This procedure yields damage estimates in NRC Categories 2, 3 and 4.

Prepared by: _____ Date: ____/____/____

Checked by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 13 of 14)

OPTION 4 (INPUT)

DOSE RATE INFORMATION

ENTER DATE AND TIME OF REACTOR SHUTDOWN:

DATE: 7/18/84
TIME: 1:00

ENTER REPRESENTATIVE POWER LEVEL: 50%

Measured Dose Rate RAD/HR	Time of Measurement	
	Date	Time
100000	7/18/84	3:00
50000	7/18/84	6:00
15000	7/19/84	1:00
4000	7/24/84	1:00

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 14 of 14)

OPTION 4 (OUTPUT)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 1A (5/31/85)

RUNTIME: 02-10-1993
14:14:13

#	EDR (R/HR)	TPA (HRS)	CURVE A	CURVE B	CURVE C	CURVE D	
1	0.20E+06	2.0	5.9E+03	6.1E+04	1.6E+05	1.0E+06	CATEGORY 6
2	0.10E+06	5.0	2.5E+03	2.2E+04	7.6E+04	4.4E+05	CATEGORY 6
3	0.30E+05	24.0	5.6E+02	4.1E+03	2.0E+04	1.1E+05	CATEGORY 6
4	0.80E+04	144.0	1.0E+02	6.0E+02	4.5E+03	2.1E+04	CATEGORY 6

NRC CATEGORY DEFINITIONS:

1 - NO FUEL DAMAGE

2 - INITIAL CLADDING FAILURE

3 - INTERMEDIATE CLADDING FAILURE

4 - MAJOR CLADDING FAILURE

5 - INITIAL FUEL PELLET OVERHEATING

6 - INTERMEDIATE FUEL PELLET OVERHEATING

7 - MAJOR FUEL PELLET OVERHEATING

ATTACHMENT 4

PRELIMINARY ESTIMATE OF CORE DAMAGE USING CORE EXIT THERMOCOUPLE (CET) TEMPERATURES

(Page 1 of 5)

1. Purpose

The purpose of this section is to estimate core damage based on core exit thermocouple temperatures up to about the time when the peak core temperature reaches about 2300°F. Core damage using this indicator is described by categories 2 through 4 of the seven NRC categories in Table 2, Cladding Damage Characteristics, in Attachment 1.

2. Definitions

A. Cladding Failure

Cladding failure is defined as a break in the fuel rod clad at least sufficient to release the internal gas pressure.

3. Precautions and Limitations

A. The assessment of core damage obtained by using this method is only an estimate. The techniques employed in this section are only accurate to locate the core condition within the first four of the seven categories of core damage described in Table 2, Cladding Damage Characteristics, in Attachment 1. The methodology is based on core exit temperature data. Other plant indications may be available which can improve upon the estimation of core damage.

B. The relationship between the core exit thermocouple temperature and the clad temperature varies with the core uncovery scenario. This procedure applies to slow core uncovery by boiloff of the coolant. For other more rapid uncovery scenarios, this procedure could yield a very low estimate of the number of ruptured rods. In general, for core uncovery at pressures below about 1200 psia, there is high confidence that at least the predicted estimate of rods are actually ruptured.

ATTACHMENT 4
PRELIMINARY ESTIMATE OF CORE DAMAGE USING CORE EXIT THERMOCOUPLE (CET) TEMPERATURES
(Page 2 of 5)

4. Instructions

A. Obtain the following from the instrument recordings:

From the recording of maximum core exit thermocouple temperature as a function of time, obtain and record on Data Sheet 4-1, Record of Temperature, Pressure and Damage Estimate, the maximum temperature and the time it occurs. As many thermocouples as possible should be used, in this way equipment malfunction may be detected if a thermocouple reads greater than 1650°F or varies considerably from its neighboring thermocouples.

From the recording of Reactor Coolant System pressure as a function of time, obtain and record on Data Sheet 4-1, Record of Temperature, Pressure and Damage Estimate the pressure during the period of maximum thermocouple temperature.

B. Select the temperature labeled curve on Figure 4-1, Percent of Fuel Rads with Ruptured Clad vs. Max Core Exit Thermocouple Temperature, which corresponds to a pressure approximately equal to or greater than the RCS pressure. Enter the abscissa (x-value) at the maximum CET temperature and read on the ordinate (y-value) the percent of the fuel rods which have ruptured clad. Record on Data Sheet 4-1, Record of Temperature, Pressure and Damage Estimate.

C. This is probably a lower limit estimate of damage. Some judgement on the bias is available in Reference 2.1.2.

5. Conclusions

Use the percent of rods ruptured from Data Sheet 4-1, Record of Temperature, Pressure and Damage Estimate, and the clad damage characteristics of Table 2 in Attachment 1 to determine the NRC category of cladding failure. This procedure yields damage estimates in NRC Categories 2, 3 and 4.

ATTACHMENT 4

PRELIMINARY ESTIMATE OF CORE DAMAGE USING CORE EXIT THERMOCOUPLE (CET) TEMPERATURES

(Page 3 of 5)

DATA SHEET 1. RECORD OF TEMPERATURE PRESSURE AND DAMAGE ESTIMATE

(Page 1 of 2)

Step 1 Record the following data:

> **NOTE**
> As many thermocouple readings as possible should be recorded. In this way, equipment malfunction may be detected if a thermocouple reads greater than 1650°F or varies considerably from its neighboring thermocouples.

Maximum Core Exit Thermocouple Temperature _____°F
(See Instruction 4.A in the text for guidelines)

Time of Maximum Temperature _____

Reactor Coolant System Pressure at Above Time _____ psia

Step 2 From Figure 4-1, Percent of Fuel Rods with Ruptured Clad vs. Max Core Exit Thermocouple Temperature, at maximum thermocouple temperature and at appropriate temperature based on pressure, read percent of ruptured rods. _____ %

Step 3 Comment on probable bias of results in Step 2. (Reference 2.1.2, Page E-5). For example:

a) A smooth core exit thermocouple recording and an uncovery duration of 20 minutes or longer are indicators for a good prediction.

b) For a large break LOCA, the thermocouple temperature may rise rapidly then quench when the core is covered. This procedure could yield a low estimate for that situation.

ATTACHMENT 4

PRELIMINARY ESTIMATE OF CORE DAMAGE USING CORE EXIT THERMOCOUPLE (CET) TEMPERATURES

(Page 4 of 5)

DATA SHEET 1. RECORD OF TEMPERATURE PRESSURE AND DAMAGE ESTIMATE

(Page 2 of 2)

Step 4 NRC Category of cladding failure from Table 2, Cladding Damage Characteristics, in Attachment 1. _____

Step 5 Enter summary information into Data Sheet 6-2, Summary Worksheet.

Maximum Core Exit Thermocouple Temperature

_____°F
_____°F
_____°F
_____°F
_____°F

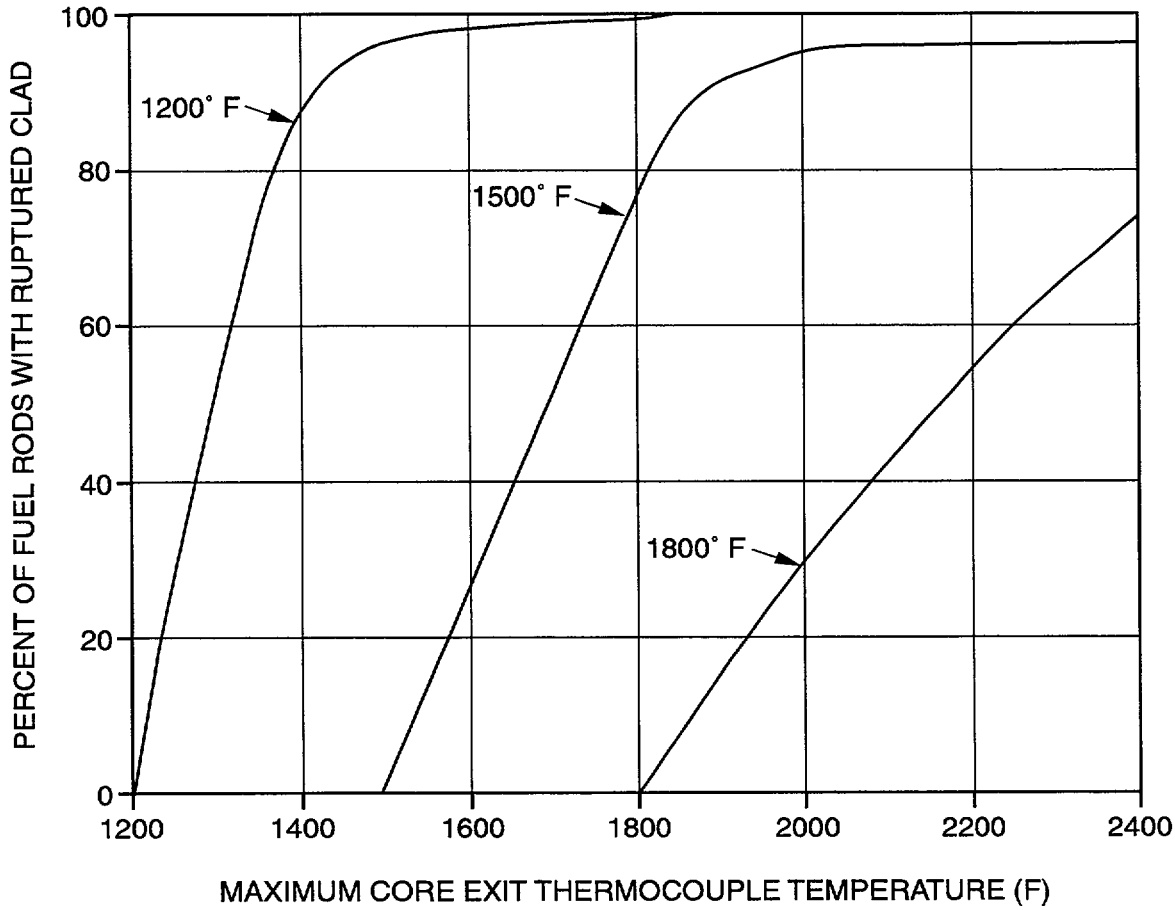
ATTACHMENT 4

PRELIMINARY ESTIMATE OF CORE DAMAGE USING CORE EXIT THERMOCOUPLE (CET) TEMPERATURES

(Page 5 of 5)

FIGURE 4-1. PERCENT OF FUEL RODS WITH RUPTURED CLAD VS MAX CORE EXIT THERMOCOUPLE TEMPERATURE

When The Pressure Is:	Use The Curves Labeled:
P less than or equal to 100 psia	1200° F
P is between 100 and 1200 psia	1500° F
P is between 1200 and 1650 psia	1800° F



(P/EP/EPIP-11/Ff-R0)

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 1 of 7)

1. Purpose

This section provides the methodology for use under post-accident plant conditions to determine the type and degree of core damage which may have occurred by using radiation dose rates measured inside the containment building using the Containment High Range Radiation Monitor (CHRRM). The radiation dose rate is related to the quantitative release of fission products from the core expressed as the percent of the source inventory at the time of the accident. The resulting observation of core damage is described by one or more of the seven categories of core damage in Table 3 in Attachment 1.

2. Definitions

A. Fuel Damage

For the purpose of this section, fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the Reactor Coolant, starting with a penetration in the zircaloy cladding.

B. Source Inventory

The source inventory is the total quantity of fission products expressed in Curies of each isotope present in either source; the fuel pellets or the fuel rod gas gap.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 2 of 7)

3. Precautions and Limitations

A. The assessment of core damage obtained by using the methodology in this section is only an estimate. The techniques employed in this section are only accurate to locate the core condition within one or more of the seven categories of core damage described in Table 3 in Attachment 1. The procedure is based on radiation dose rate. Other plant indications may be available which can improve upon the estimation of core damage. These include sample radiological analysis, incore temperature indicators, and the total quantity of hydrogen released from zirconium degradation. Whenever possible, these additional indicators should be factored into the assessment.

B. This section relies upon radiation dose rate measurements taken from the highest readings of two high range radiation monitors located inside the containment building to determine the total quantity of fission products released from the core and therefore available for release to the environment. The amount of fission products present at the location of the monitors may be changing rapidly due to transient plant conditions. Therefore, multiple measurements should be obtained within a minimum time period and when possible, under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

C. The methodology in this section is limited to the upper bound condition of fission product release from the core due to fuel overheat. Simultaneous with fuel overheat, there may be localized fuel pellet melting within the core. The transport of the non-volatile fission products released due to melting is not known. The dose rates measured under conditions of fuel pellet melting are anticipated to exceed those shown in Figure 5-1, Containment High Radiation Monitor Dose Rate vs. Time After Trip, for major fuel overheat. However, this procedure does not attempt to identify the extent of any potential fuel melting.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 3 of 7)

3. Precautions and Limitations (continued)

   D. This section is limited to the interpretation of the dose rate measurement resulting from a mix of fission products. The methodology cannot accurately distinguish between the conditions of fuel cladding failure and fuel overheat when the resulting dose rates are the same. The methodology does provide an upper limit estimate of the progressive core damage. Concurrent conditions of cladding failure and overheat should be anticipated due to the radial distribution of heat generation within the core. Distinction between the type of core damage requires the identification of the characteristic fission products. The procedure for core damage assessment using radiological analysis of fluid samples is required to explicitly distinguish between the categories.

   E. This methodology is limited in applicability to those conditions in which the fission product inventory in the core has had sufficient time to reach equilibrium. Equilibrium fission product inventory is a function of reactor power and burnup. Based upon the fission products of concern, equilibrium conditions are achieved after thirty days of operation at constant power. Constant power is considered to include changes of no greater than ± 10 percent. The methodology may be used following non-constant periods of operation by using engineering judgement to select the most representative power level during the period. This method may also be used if the reactor has produced power for less than thirty days, however, the resulting assessment of core damage would be an under-prediction of the actual conditions.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 4 of 7)

4. Instructions

A. Record the plant indications required in Data Sheet 5-1, Containment High Radiation Monitor vs. Time After Trip.

B. Plant Power Correction

The measured radiation dose rate inside the containment building is to be corrected for the plant power history. A correction factor is used to adjust the measured dose rate to the corresponding value had the plant been operating at 100 percent power.

To correct the radiation dose rate for the case in which plant power level has remained constant for a period greater than 30 days, a simple ratio of the power may be employed. The reactor power is considered to be constant if it has not changed by ± 10 percent within the last thirty days prior to the reactor trip.

To correct the radiation dose rate for the case in which reactor power level has not remained constant during the 30 days prior to the reactor shutdown, engineering judgement is used to determine the most representative power level. The following guidelines should be considered in the determination.

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 the earlier levels.

Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

In the case in which reactor has produced power for less than 30 days, this procedure may be employed. However, the estimate of core damage obtained under this condition may be an under-prediction of the actual condition.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 5 of 7)

4. Instructions (continued)

   C. The decay correction for the radiation dose rate requires the determination of the time duration between the reactor trip and the measurement of the dose rate. This is done simply using the time of reactor shutdown (trip) recorded in Data Sheet 5-1, Containment High Radiation Monitor vs. Time After Trip.

5. Conclusions

   The conclusion on the extent of core damage is made using the equilibrium dose rate, the duration of reactor shutdown (hours since reactor trip), and the analytically determined dose rates provided in Figure 5-1, Containment High Radiation Monitor vs. Time After Trip. The equilibrium dose rate is plotted as a function of time following reactor shutdown. Engineering judgement is used to determine which category of core damage shown on Figure 5-1, Containment High Radiation Monitor vs. Time After Trip, is most representative of the particular value that has been plotted. The following criteria should be considered in the determination.

   A. Dose rate measurements may have been recorded during periods of transient conditions within the plant. Measurements made during stable plant conditions should weigh more heavily in the assessment of core damage.

   B. Dose rates significantly above the lower bound for the category of major fuel overheat may indicate concurrent fuel pellet melting. The methodology in this section may not be employed to estimate the degree of fuel pellet melting.

   C. Dose rates within any category of fuel overheating may be anticipated to include concurrent fuel cladding failure. The methodology in this section may not be used to distinguish the relative contributions of the two categories to the total dose rate. The methodology does give the estimate of the highest category of damage.

   D. Dose rates corresponding to the two categories of major cladding failure and initial fuel overheat are observed to overlap on Figure 5-1, Containment High Radiation Monitor vs. Time After Trip. The evaluation of other plant parameters may be required to distinguish between them. However, concurrent conditions may be anticipated.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 6 of 7)

DATA SHEET 5-1. CONTAINMENT HIGH RANGE RADIATION MONITOR (CORE DAMAGE ASSESSMENT) WORKSHEET

Highest Radiation Dose Rate (CHRRM) _____ Rad/Hr

Time of Measurement: Date: ____/____/____ Time: _____

Prior 30 Days Power History:

Power, Percent	Duration, Days
_____	_____
_____	_____
_____	_____
_____	_____
_____	_____

Time of Reactor Trip: Date: ____/____/____ Time: _____

$$\textit{Equilibrium Dose Rate (Rad/Hr)} = \textit{Measured Dose Rate (Rad/Hr)} \times \frac{100}{\textit{Reactor Power Level (\%)}} = _____ \textit{(Rad/Hr)}$$

Refer to Table 3, Percent of Source Inventory Released to Containment, in Attachment 1 and Figure 5-1, Containment High Radiation Monitor Dose Rate vs. Time After Trip, to obtain category of core damage.

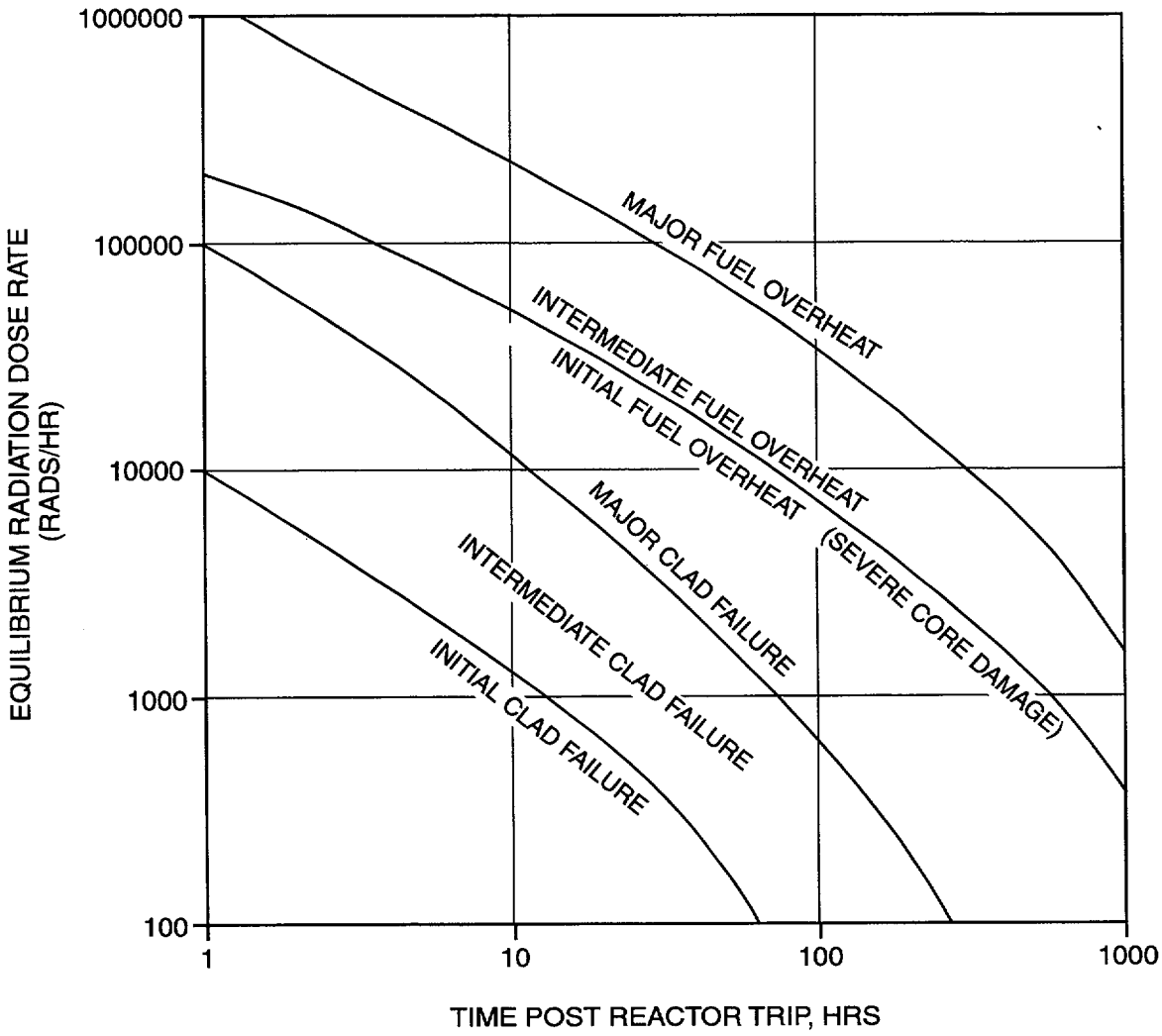
See Step 5 for guidance in formulating conclusions.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 7 of 7)

FIGURE 5-1. CONTAINMENT HIGH RADIATION MONITOR DOSE RATE VS TIME AFTER TRIP



(P/EP/EPIP-11/Fg-R0)

NOTE

Categories of core damage are indicated in Attachment I, Tables 1, 2, and 3. Determination of core damage should not be based solely from this graph.

ATTACHMENT 6

PRELIMINARY ESTIMATE OF CORE DAMAGE USING PRELIMINARY RADIOISOTOPIC DATA

(Page 1 of 3)

CAUTION

Core damage assessment using the readily available radioisotopic information should be used only to obtain a general estimate of the extent of core degradation. Analysis of radionuclide samples is needed to improve upon estimate of core damage.

1. Obtain available plant radioisotopic data and complete Data Sheet 6-1, Preliminary Radioisotopic data.

ATTACHMENT 6

PRELIMINARY ESTIMATE OF CORE DAMAGE USING PRELIMINARY RADIOISOTOPIC DATA

(Page 2 of 3)

DATA SHEET 6-1. PRELIMINARY RADIOISOTOPIC DATA

CAUTION

The concentrations assume uniform mix, no dilution due to injection, and 1/2 hour after shutdown. In the presence of dilution, this assessment will underestimate core damage.

STEP 1: Obtain preliminary radioisotopic data for the following isotopes as available:

	Activity (µCi/gm)
I-131	_____
I-133	_____
I-135	_____
Cs-134	_____
Cs-137	_____
Sr-90	_____

STEP 2: Determine the crude core damage category from the Table below.

Core Damage Category _____
[Core Damage (Gap Release) or Severe Core Damage (Fuel Pellet Release)]

PWR Baseline Coolant Concentrations Vs. Core Damage (from Reference 2.1.6)

Nuclide	Normal Concentration (µCi/gm)	Concentration After Gap Release (µCi/gm)	Concentration After Melt Release (µCi/gm)
I-131	4.5 E-02	6.8 E+03	3.4 E+05
I-133	1.4 E-01	1.4 E+04	6.8 E+05
I-135	2.6 E-01	1.2 E+04	6.0 E+05
Cs-134	7.1 E-03	1.5 E+03	3.0 E+04
Cs-137	9.4 E-03	9.4 E+02	1.9 E+04
Sr-90	1.2 E-05	Not Avail.	1.0 E+03

ATTACHMENT 6

PRELIMINARY ESTIMATE OF CORE DAMAGE USING PRELIMINARY RADIOISOTOPIC DATA

(Page 3 of 3)

DATA SHEET 6-2. SUMMARY WORKSHEET

RESULTS OF DETAILED RADIOISOTOPIC ANALYSIS (if available) FROM ATTACHMENT 8:

_____ Percent Cladding Failure _____ Percent Fuel Overheat _____ Percent Fuel Melt

RESULTS OF AUXILIARY INDICATORS (Attachments 4, 5, 6, 7):

	METHOD	NRC CATEGORY
CHRRM	_____ (R/Hr)	
ELAPSED TIME	_____ (Hrs)	_____
H_2 Analysis	_____ (Percent Embrittled)	_____
CET (Maximum)	_____ (°F)	_____
Characteristic Fission Product Concentration	I-131 _____ (μCi/gm) Cs-134 _____ (μCi/gm)	_____

IS RX VESSEL LEVEL BELOW ZERO? ____ YES ____ NO

HAS LEVEL DROPPED BELOW ZERO? ____ YES ____ NO

SUMMARY OF RESULTS: _____

_____

_____

NOTE

Compare percent cladding failure, percent fuel overheat, and percent fuel melt results obtained from the radionuclide analysis to those obtained from the auxiliary indicators analyses.

If results are in agreement, the core damage assessment is complete. If the results are not in agreement, a recheck of both analyses may be performed or certain indications may be discounted based on engineering judgement.

Prepared by: _____ Date: ____/____/____

Reviewed by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 1 of 14)

1. Purpose

This section provides the methodology for use under post-accident plant conditions to determine the extent of fuel clad damage which may have occurred. It utilizes hydrogen measured in samples obtained with the Post-Accident Sampling System (PASS) and containment hydrogen analyzers. The measured hydrogen is related to the amount of fuel clad oxidation. Clad oxidation is in turn related to cladding failure which is expressed in terms of the percent of fuel rods which are ruptured and the percent which are embrittled. The resulting observation of damage is described by one or more of the seven categories of core damage in Table 2, Cladding Damage Characteristics, in Attachment 1.

2. Definitions

A. Clad Rupture

Clad rupture is defined as a break in the fuel rod clad at least sufficient to release the internal gas pressure.

B. Clad Embrittlement

At temperatures above the rupture temperature, significant oxidation of the clad occurs. If the oxidation exceeds the embrittlement threshold, fragmentation of embrittled clad may subsequently occur from thermal shock or hydraulic pressure forces such that the structure of the fuel assembly is destroyed and substantial fuel pellet fragments are released to the coolant.

3. Precautions and Limitations

A. The assessment of core damage obtained by using this methodology is only an estimate. The techniques employed in this section are only accurate to locate the core condition within one or more of the seven categories of core damage in Table 2, Cladding Damage Characteristics, in Attachment 1.

B. The methodology in this section is applicable under conditions for which there are no voids measurable by the Reactor Vessel Level Monitoring System (RVLMS).

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 2 of 14)

4. Instructions

A. Core Uncovery Conditions: Record the core conditions over the time period of core uncovery on Data Sheet 7-1, Core Uncovery Conditions.

B. Sampling Conditions and Measured Hydrogen

1. Record the conditions in containment and the RCS at the time the hydrogen samples are obtained.

2. Enter on the worksheet of Data Sheet 7-2, Sampling Conditions and Measured Hydrogen.

3. Record the results of hydrogen sampling and analysis on the worksheet of Data Sheet 7-2, Sampling Conditions and Measured Hydrogen.

4. Follow the instructions to obtain the total amount of hydrogen measured in units of cubic feet of hydrogen at standard temperature and pressure.

C. Hydrogen Generated in Containment

NOTE

Data Sheet 7-3, Hydrogen Generated in Containment, utilizes measured data for the containment temperature as a function of time up to the sampling time and a plant specific curve of the rate of production as a function of containment temperature in Figure 7-2, Hydrogen Production Rate from Aluminum and Zinc vs. Temperature.

1. Data Sheet 7-3, Hydrogen Generated in Containment, is a worksheet for calculating the amount of hydrogen generated by oxidation of materials within the containment.

2. Record the data required on Data Sheet 7-3, Hydrogen Generated in Containment.

3. Complete the indicated calculations to obtain the cubic feet of hydrogen at STP generated by containment materials oxidation.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 3 of 14)

4. Instructions (continued)

D. Hydrogen Generated by Radiolysis

NOTE

1. The hydrogen generated by radiolysis is a function of operating power and decay time.
2. For the case in which the operating power is constant or has not changed by more than ±10 percent for a period greater than 30 days, that power is used.
3. For the case in which the power has not remained constant during the 30 days prior to the reactor trip, Engineering judgement is used to determine the most representative power level.

1. The following guidelines should be considered in the determination:

   a. The average power during the 30 day time period is NOT necessarily the most representative value for determining radiolysis by fission products.

   b. The last power levels at which the reactor operated should weigh more heavily in the judgement than the earlier levels.

   c. Continued operation for an extended period should weigh more heavily in the judgement than brief transient levels.

   d. For the case in which the reactor has produced power for less than 30 days, this methodology may be employed. However, the estimate of hydrogen from radiolysis will be too high and the calculated hydrogen by core oxidation will be too low. Hence, an under-prediction of core damage may result.

2. Record the data required on the worksheet of Data Sheet 7-4, Hydrogen Generated by Radiolysis.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 4 of 14)

4. Instructions (continued)

E. Core Damage Assessment, Hydrogen

1. Enter the amounts of hydrogen from Steps 4.B, C and D on the worksheet of Data Sheet 7-5, Core Damage Assessment from Hydrogen Measurement.

2. Subtract the amounts in Steps 4.C and D from 4.B as indicated on the worksheet to yield the cubic feet of hydrogen generated by core clad oxidation.

3. Complete the instructions of Data Sheet 7-5, Core Damage Assessment from Hydrogen Measurement, to determine the percentage of fuel rods with ruptured clad and the percentage of fuel rods with embrittled clad.

F. Conclusion

1. The conclusion on core damage is made using the two results from above. These are:

   a. Percentage of fuel rods with ruptured clad.

   b. Percentage of fuel rods with embrittled or structurally failed cladding.

2. Knowledgeable judgement is used to compare the above two results to the definitions of the seven NRC categories of fuel damage found in Table 2, Cladding Damage Characteristics, in Attachment 1. Core damage does NOT take place uniformly. Therefore, when evaluating damage using these results, Table 2, Cladding Damage Characteristics, in Attachment 1 may yield a combination of categories of damage which exist simultaneously.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 5 of 14)

DATA SHEET 7-1. CORE UNCOVERY CONDITIONS

Time period of core uncovery. Complete the following table using recorded instrument data.

Instrument	Estimated Core Uncovery Time	Estimated Core Recovery Time
Reactor Vessel Level Monitoring System	Lower Limit Elevation Uncovers (core uncovery) Time _____	Lower Limit Elevation Recovers Time _____
Core Exit Thermocouple Temperature	Start of Continuous Rise or Exceed 660°F Time _____ Temperature _____	Rapid Temperature Drop to Saturation Time _____ Temperature _____
Core Exit Thermocouple Saturation Margin	Start of Superheat Time _____	Return to Saturation Time _____

Interpret above data to obtain best estimate for time period of core uncovery and obtain pressurizer pressure range during that period. The superheat derived from the thermocouple temperature and corresponding system pressure is considered as the best indicator for core uncovery during boiloff and should be used, but should be compared with the other indicators to help identify possible anomalies.

	Core Uncovery	Core Recovery
Time	_____	_____
Pressure	_____	_____

Estimate vessel inlet flow rates during core uncovery heatup period, up to approximately the time of peak core exit thermocouple temperature. Net inlet flow indicates that the methodology may have additional bias which under-predicts clad damage.

Charging Flow Rate _____
Letdown Flow Rate _____
HPSI Flow Rate _____
LPSI Flow Rate _____
Other Inlet Flows _____

Net inlet flow = Charging Flow + High Pressure Safety Injection (HPSI) and Low Pressure Safety Injection (LPSI) flow + other inlet flow - Letdown Flow.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 6 of 14)

DATA SHEET 7-2. SAMPLING CONDITIONS AND MEASURED HYDROGEN

Obtain the RCS and containment conditions at the time of sampling for hydrogen.

Reactor Coolant System		Containment	
Sampling Time	_____	Sampling Time	_____
Pressure	_____psig	Atmospheric Pressure	_____psig
Temperature, T_{avg}	_____°F	Atmospheric Temperature	_____°F
Reactor Vessel Coolant Level	_____percent	Has Hydrogen Recombiner Operated?	Yes / No
Pressurizer Level	_____percent	Does Pressure or Temperature History Indicate a Hydrogen Burn?	Yes / No

Hydrogen Sample Data Reduction

Cont. Sample (Vol. percent/100) x Cont. Vol. ($ft.^3$) x (32 + 460) / (Normal Temp. + 460) = ft^3 H_2 at STP

_____ x 2.5 E6 x 492 / _____ = _____ft^3

RCS Sample (cc/kg at STP) x RCS Vol.* (ft^3) x Density Ratio ρ_{act}/ρ_{stp} (Figure C-2.A.1) / 1000 = ft^3 H_2 at STP

_____ x _____ x _____ / 1000 = _____ ft^3

Total = Cont. Sample (ft^3) + RCS Sample = _____ + _____ = _____ ft^3

Also record total on Data Sheet 7-5, Core Assessment from Hydrogen Measurement.

* RCS volume is: PSL1 = 10,401 ft^3
PSL2 = 10,198 ft^3

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 7 of 14)

DATA SHEET 7-3. HYDROGEN GENERATED IN CONTAINMENT

STEP 2.A.4.C

Record the containment temperature at selected time intervals and calculate the hydrogen generated by oxidation of containment materials utilizing the plant-specific production rates from Figure 7-2.

1 Time at Start of Intervals	2 Containment Temperature (°F)	3 Interval Duration (Hr)	4 Avg. Containment Temp. During Interval (°F)	5 H_2 Prod. Rate (ft^3/hr) Fig. 7-2	6 H_2 Produced (Col. 3) X (Col. 5)
Accident Starts					
Sample Time					

Long Term Hydrogen Production in Containment
Total (Summation of Column 6) _____ SCF

Short term rapid hydrogen production by containment aluminum,
2.277 ft^3 for PSL1 and 5,235 ft^3 for PSL2 (Reference 2.1.2, Table 4.3) +_____ SCF

Total Hydrogen Production in Containment = _____ SCF

Record total on Data Sheet 7-5, Core Damage Assessment from Hydrogen Measurement, also.

Items in Columns 1 and 2 are input plant data.
Interval Duration is the line difference between consecutive temperature readings.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 8 of 14)

DATA SHEET 7-4. HYDROGEN GENERATED BY RADIOLYSIS

Record the following data and utilize the curves of Figure 7-3, Specific Radiolytic Hydrogen Production vs. Decay Time, to determine the hydrogen generated by radiolysis.

Prior 30 days power history	Power, Percent	Duration, Days
Note: No calculation is required to determine power level, guidance on judgement is provided in Step 4.D.	_____	_____
	_____	_____
	_____	_____

Estimated Power Level based on a power history: _____

Operating Power (Mwt):

Power to use in evaluating long term hydrogen production by radiolysis =

$$(\textit{Full Power, Mwt}) \times \frac{\textit{Power Level}}{100}$$

(Full Power: PSL1, PSL2 = 2700 Mwt)

T_o = Time of Reactor Trip Time _____

T_i = Time Sample Taken _____

Decay Time (Time Interval, T_i - T_o) _____ Hours

Enter abscissa (x-value) on Figure 7-3, Specific Radiolytic Hydrogen Production vs. Decay Time, with above decay and read two values of hydrogen produced by radiolysis, one from each curve, in cubic feet of hydrogen at STP per Mwt operating power. Multiply by above power and record as follows:

Limit Curve	Hydrogen Produced (SCF/Mwt. Figure 7-3)	x	Operating Power (Mwt.)	=	Total Hydrogen Produced (SCF)
Upper	_____	x	_____	=	_____
Lower	_____	x	_____	=	_____

Using results from Radiological Analysis of Samples, estimate which results should be used; upper limit for major fuel overheat, lower limit for initial fuel overheat, or appropriate estimate between the two curves for intermediate fuel overheat. Circle corresponding value of hydrogen above and also record on Data Sheet 7-5, Core Damage Assessment from Hydrogen Measurement.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 9 of 14)

DATA SHEET 7-5. CORE DAMAGE ASSESSMENT FROM HYDROGEN MEASUREMENT (SUMMARY)

A. Hydrogen Measured, from Data Sheet 7-2, Sampling Conditions and Measured Hydrogen. _____ SCF

B. Hydrogen Produced in Containment, from Data Sheet 7-3, Hydrogen Generated in Containment. _____ SCF

C. Hydrogen Produced by Radiolysis, from Data Sheet 7-4, Hydrogen Generated by Radiolysis. _____ SCF

Subtract B and C from A to get Hydrogen Produced by Core Clad Oxidation _____ SCF

Divide by (4210 for PSL1) or (4640 for PSL2). These values represent the quantity in SCF of hydrogen produced per percent of Zirconium oxidized for St. Lucie Unit 1 and Unit 2, respectively. (Reference 2.1.2, Table 4.2). = _____ %

= % Core Clad Oxidized

Enter abscissa (x-value) on Figure 7-4, Percent of Fuel Rods with Ruptured Clad vs. Percentage of Core Clad Oxidation, with "Percent Oxidation of Core Clad" and read ordinate from temperature labeled curve corresponding to the pressure during core uncovery as given on Data Sheet 7-1, Core Uncovery Conditions. Record here Percent of Fuel Rods with Ruptured Clad.

_____%.

Enter abscissa (x-value) on Figure 7-5, Oxidation Embrittlement vs. Total Core Oxidation, with above "Percent Oxidation of Core Clad" and read range of values on ordinate (y-value). Record here.

Percent of Fuel Rods Embrittled:

Range - Upper _____ %

- Lower _____ %

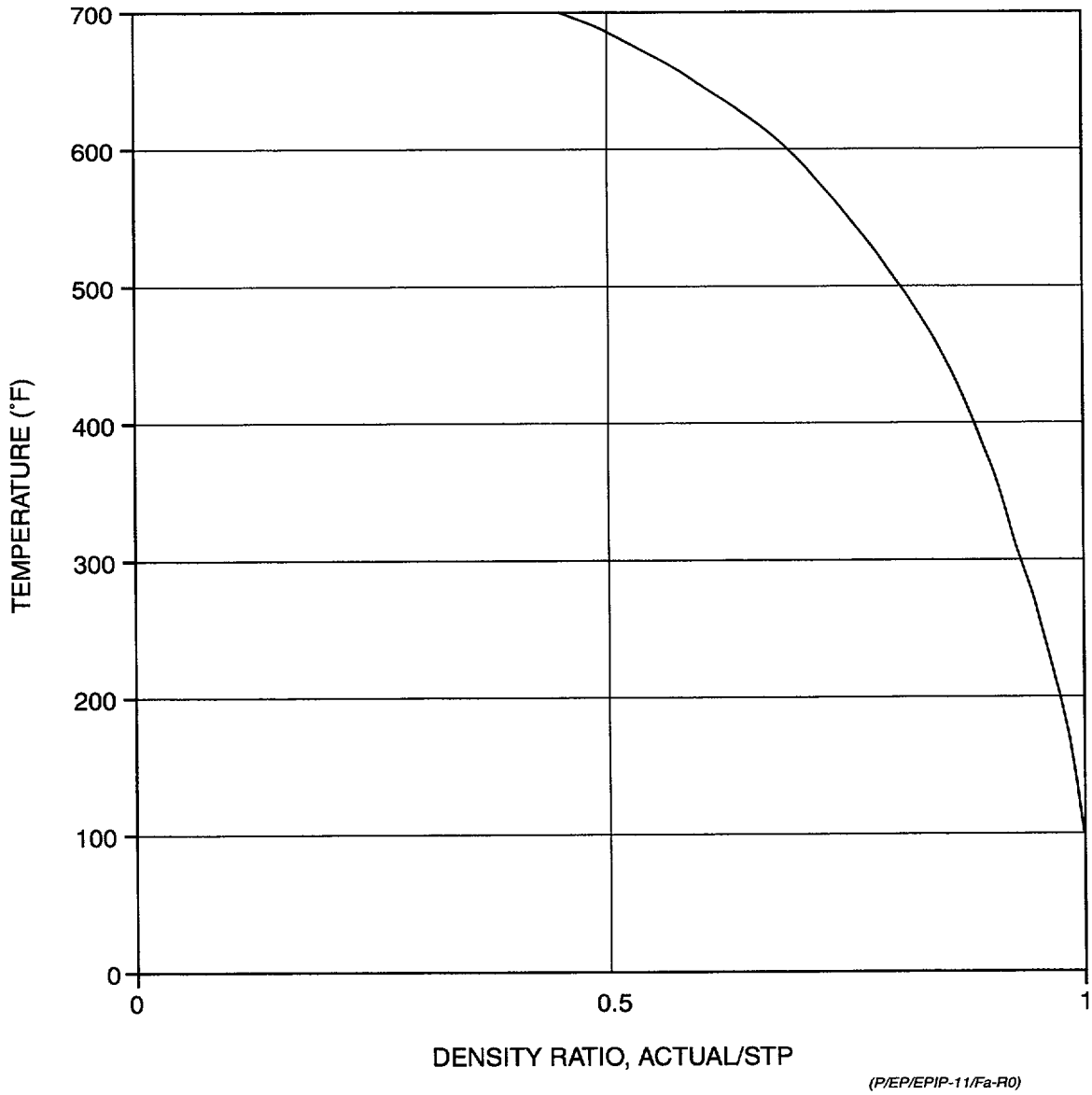
From Table 2, Cladding Damage Characteristics, in Attachment 1, select the core clad damage categories based on the above percentages of rods embrittled (damaged) and enter in Data Sheet 6-2, Summary Worksheet. Note that this assessment will under-predict fuel damage if hydrogen recombiners have operated or Hydrogen burn has occurred.

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 10 of 14)

FIGURE 7-1. RATIO OF H2O DENSITY TO H2O DENSITY AT STP vs TEMPERATURE



(P/EP/EPIP-11/Fa-R0)

ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

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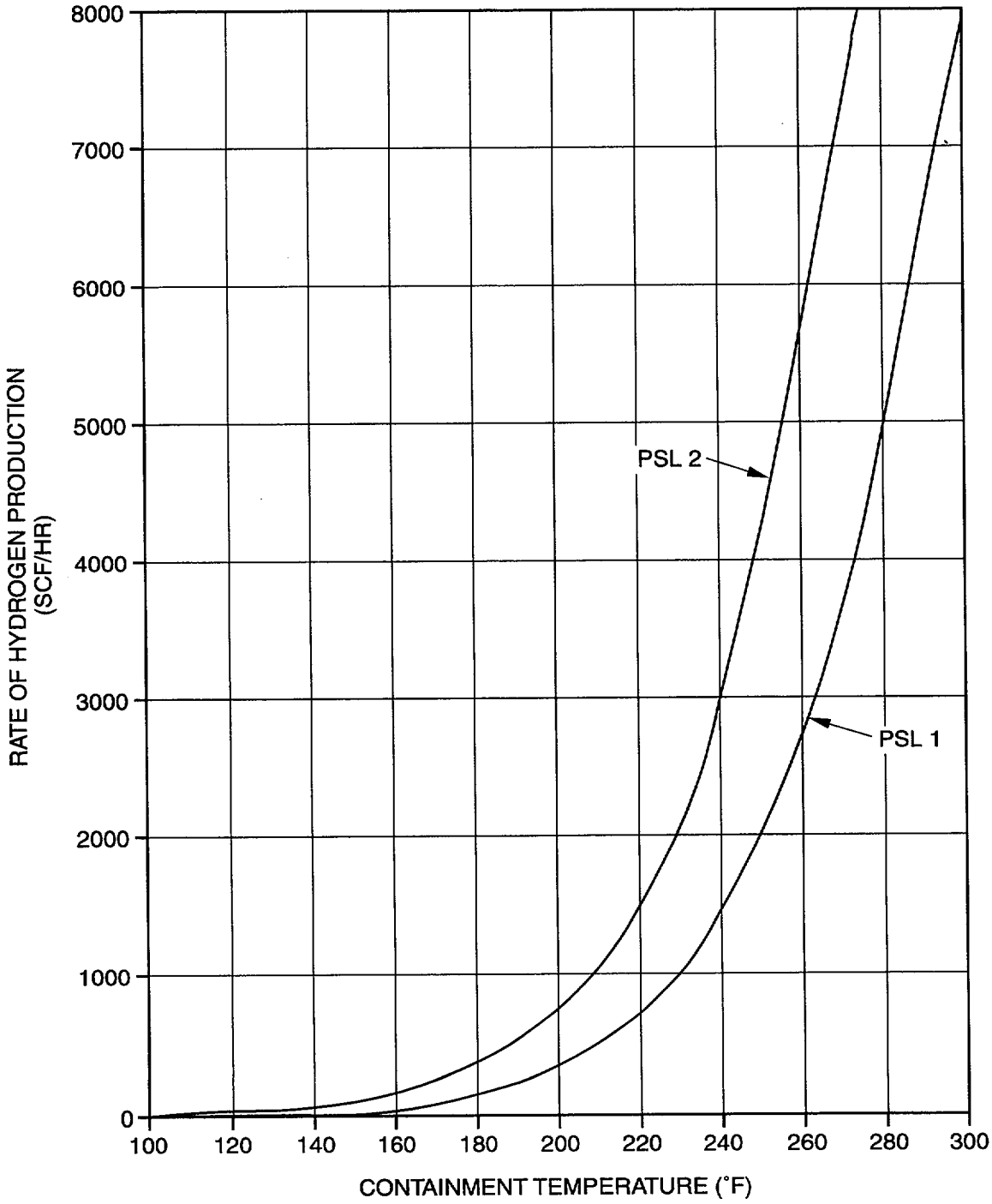
FIGURE 7-2. HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC vs TEMPERATURE



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ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

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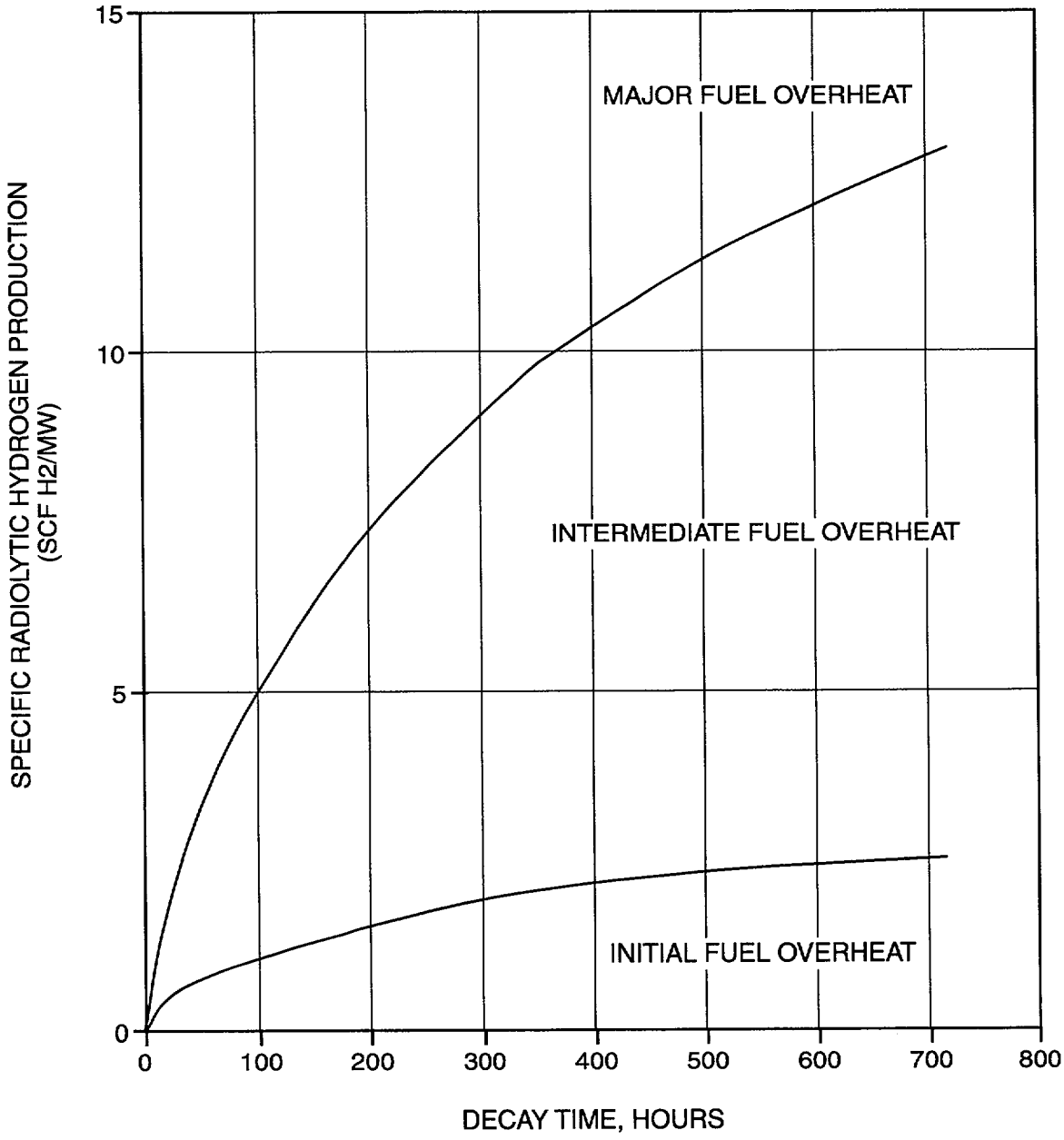
FIGURE 7-3. SPECIFIC RADIOLYTIC HYDROGEN PRODUCTION VS DECAY TIME



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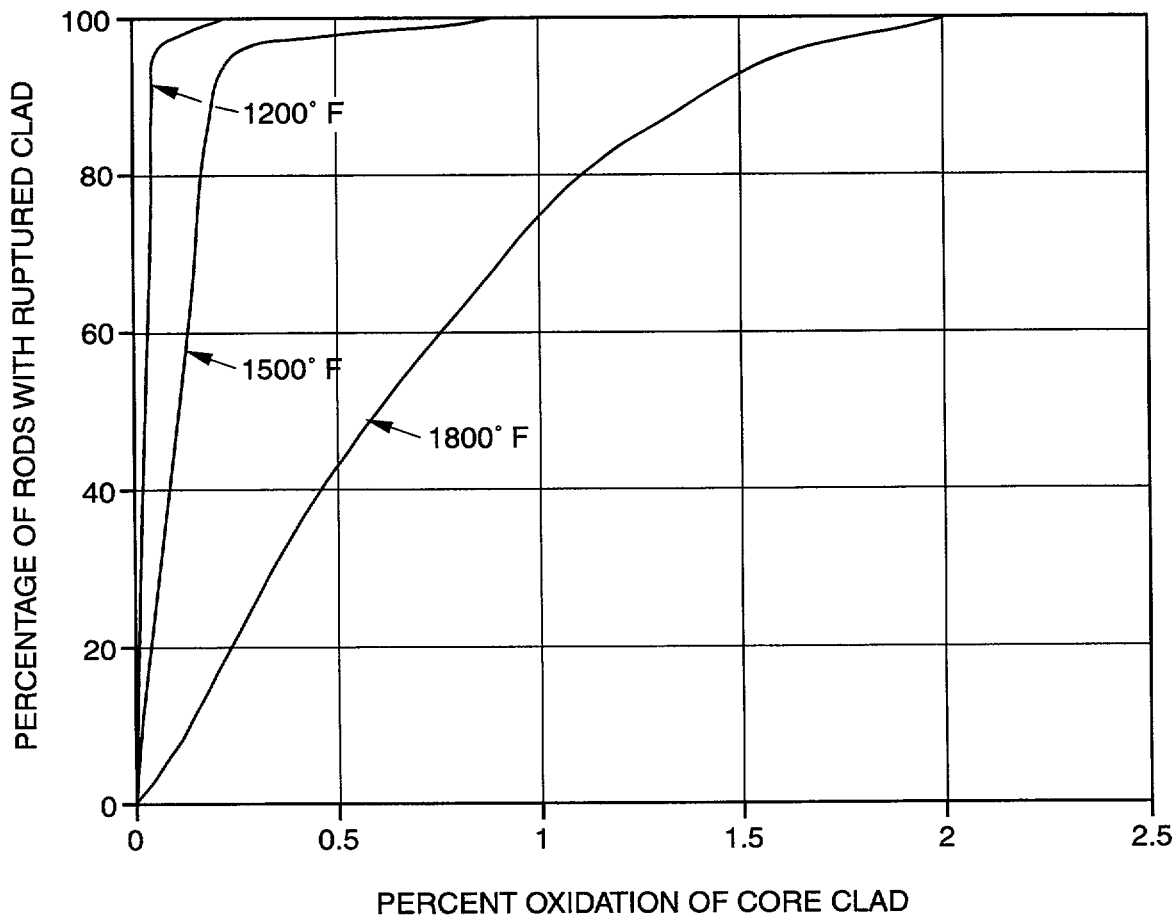
ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 13 of 14)

FIGURE 7-4. PERCENT OF FUEL RODS WITH RUPTURED CLAD VS. PERCENTAGE OF CORE CLAD OXIDATION

When The Pressure Is:	Use The Curves Labeled:
P less than or equal to 100 psia	1200° F
P is between 100 and 1200 psia	1500° F
P is between 1200 and 1650 psia	1800° F



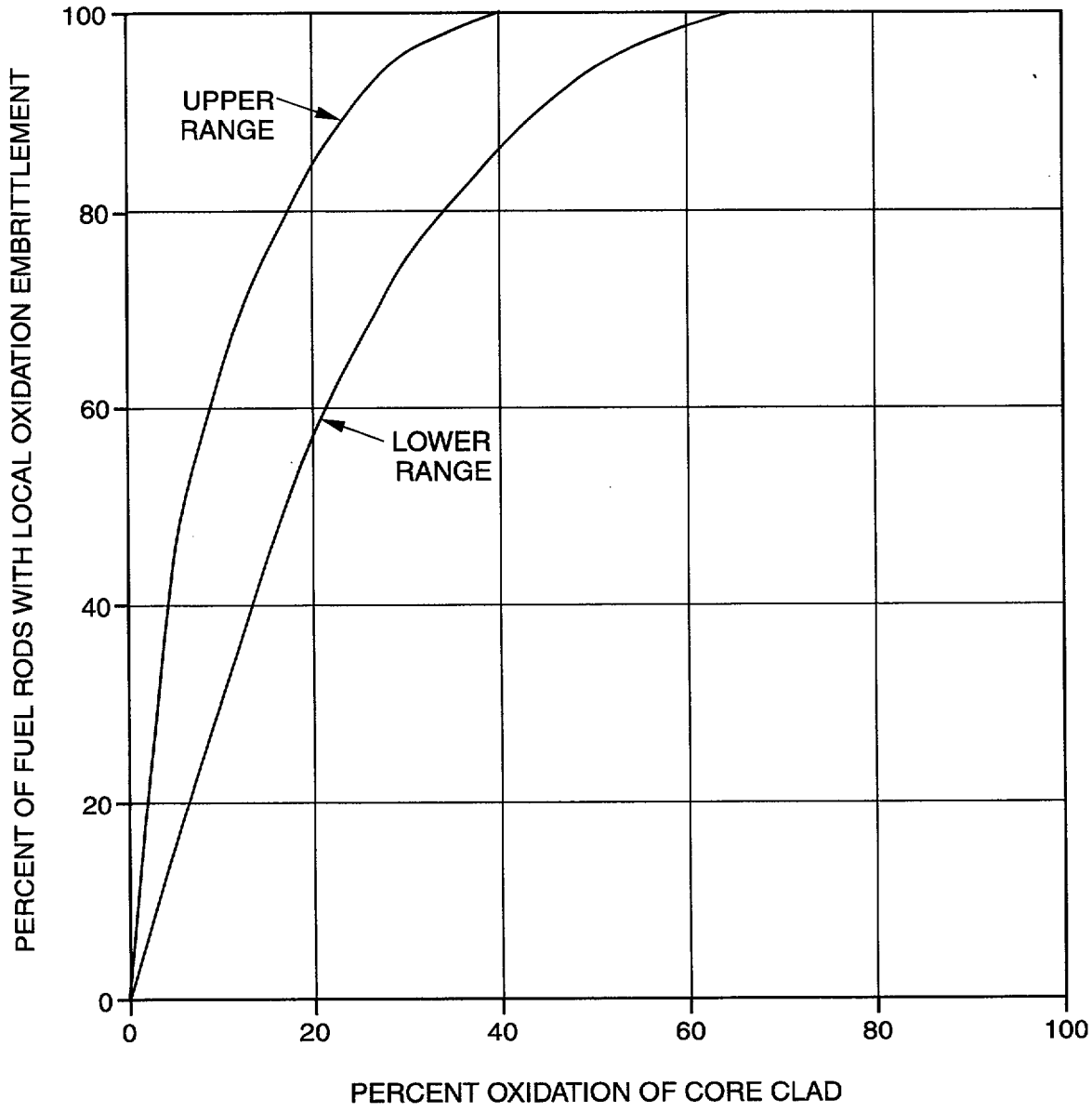
ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 14 of 14)

FIGURE 7-5. OXIDATION EMBRITTLEMENT VS TOTAL CORE OXIDATION

ST. LUCIE UNITS 1 & 2



(P/EP/EPIP-11/Fe-R0)

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 1 of 17)

1. Purpose

This section provides a method under post-accident plant conditions to determine the type and degree of reactor core damage which may have occurred by using fission product isotopes measured in samples obtained from the Post-Accident Sampling System (PASS). There are three factors considered in this section which are related to the specific activity of the samples. These are (1) the identity of those isotopes which are released from the core, (2) the respective ratios of the specific activity of those isotopes, and (3) the percent of the source inventory at the time of the accident which is observed to be present in the samples. The resulting observation of core damage is described by one or more of the ten categories of fuel damage in Table 1 in Attachment 1.

2. Definitions

A. Fuel Damage

For the purpose of this methodology, fuel damage is defined as a progressive failure of the material boundary to prevent the release of radioactive fission products into the Reactor Coolant, starting with a penetration in the zircaloy cladding.

B. Source Inventory

The source inventory is the total quantity of fission products expressed in Curies of each isotope present in either source, the fuel pellets or the fuel rod gas gap.

3. Precautions and Limitations

A. The methodology in this section relies upon samples taken from multiple locations inside the containment building to determine the total quantity of fission products available for release to the environment. The amount of fission products present at each sample location may be changing rapidly due to transient plant conditions. Therefore, it is recommended that the samples should be obtained within a minimum time period and if possible, under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 2 of 17)

3. Precautions and Limitations (continued)

   B. A number of factors influence the reliability of the chemistry samples upon which this section is based. Reliability is influenced by the ability to obtain representative samples due to incomplete mixing of the fluids, and equipment limitations.

   The accuracy achieved in the radiological analyses are also influenced by a number of factors. The equipment employed in the analysis may be subjected to high levels of radiation exposure over extended periods of time. Chemists are recommended to exercise considerable caution to minimize the spread of radioactive materials. Samples have the potential of being contaminated by numerous sources. Cooling or reactions may take place in the long sample lines. Therefore, the results obtained may not be representative of plant conditions. To minimize these effects, multiple samples should be obtained over an extended time period from each location.

4. Instructions

   A. Obtain and record the plant indications and source of indication requested on Data Sheet 8-1, Input Parameters. Because of transient conditions, the values should be recorded as close as possible to the time at which the radiological samples are obtained.

      1. Request sampling at the locations recommended for core damage assessment using the guidelines provided in Table 8-1, Sample Locations Recommended for Core Damage Assessment.

      2. Obtain results of sampling and analysis and record the required sample data, corrected to Standard Temperature and Pressure (STP), and time of sample collection on Data Sheet 8-1, Input Parameters. All of the isotopes listed in Data Sheet 8-1, Input Parameters, may not be observed in the sample.

   B. Correct the sample specific activity at STP for decay back to the time of reactor trip following the instruction on Data Sheet 8-2, Record of Measured Specific Activity (Decay Corrected).

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 3 of 17)

4. Instructions (continued)

C. Identification of the Fission Product Release Source

1. Calculate the ratios for each noble gas and iodine isotope using the specific activities obtained in Step 4. Record these ratios on Data Sheet 8-3, Record of Fission Product Release Source Identification.

2. Determine the source of release (gas gap or fuel pellet) by comparing the results obtained in Step 4.C.1 to the predicted ratios provided in Data Sheet 8-3, Record of Fission Product Release Source Identification. An accurate comparison is not anticipated. Within the accuracy of this methodology, it is appropriate to select as the source of release, that ratio which is closest to the value obtained in Step 4.C.1.

D. Quantitative Release Assessment

1. Calculate the total quantity of fission products found in the RCS per the instructions on Data Sheet 8-4, Quantitative Release Assessment Worksheet.

2. Calculate the quantity of fission products found in the containment building sump per the instructions on Data Sheet 8-4, Quantitative Release Assessment Worksheet.

3. Calculate the quantity of fission products found in the containment building atmosphere per the instructions on Data Sheet 8-4, Quantitative Release Assessment Worksheet.

4. The total quantity of fission products available for release to the environment is equal to the sum of the values obtained from each sample location (liquid and gas) as recorded on Data Sheet 8-5, Record of Core Release Inventory.

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 4 of 17)

4. Instructions (continued)

E. Plant Power Correction

The quantitative release of the fission products is expressed as the percent of the source inventory at the time of the accident. The equilibrium source inventories are to be corrected for plant power history.

1. Steady State Power Correction

To correct the source inventory for the case in which plant power level has remained constant for a period greater than four radioactive half-lives, complete Data Sheet 8-6, Record of Transient Power Correction. Half-lives are included in Data Sheet 8-2, Record of Measured Specific Activity (Decay Corrected).

2. Transient Power Correction

To correct the source inventory for the case in which plant power level has not remained constant prior to reactor trip, follow the instructions of Data Sheet 8-7, Record of Transient Power Correction, where the transient Power Correction Factor is defined as:

$$PCF = \frac{1}{100} \sum P_j \ (1 - e^{-\lambda t_j}) \ e^{-\lambda t_j^{\circ}}$$

Where P_j = Steady reactor power in time period j

t_j = duration of time period j (sec)

t_j° = time from reactor trip to end of time period j (sec)

λ = isotope decay constant from Data Sheet 8-2, Record of Measured Specific Activity (Decay Corrected)

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 5 of 17)

4. Instructions (continued)

F. Comparison of Measured Data with Source Inventory

The total quantity of fission products available for release to the environment obtained in Step 4.D.4, Data Sheet 8-5, Record of Core Release Inventory, is compared to the source inventory corrected for plant power history obtained in Step 4.E, Data Sheet 8-6, Record of Steady State Power Correction, or 8-7, Record of Transient Power Correction. This comparison is made by dividing the total quantity available for release by the power corrected source inventory. Record this percentage on Data Sheet 8-8, Record of Percent Release.

G. Conclusion

The conclusion on core damage is made using the three parameters developed above. These are:

1. Identification of the fission product isotopes which most characterize a given sample, Step 4.A, Data Sheet 8-1, Input Parameters.

2. Identification of the source of the release, Step 4.C, Data Sheet 8-3, Record of Fission Product Release Source Identification.

3. Quantity of fission product available for release to the environment expressed as a percent of source inventory, Step 4.F, Data Sheet 8-8, Record of Percent Release.

Knowledgeable judgement is used to compare the above three parameters to the definitions of the ten NRC Categories of Fuel Damage found in Table 1, Characteristic Isotopes, in Attachment 1. Core damage is not anticipated to take place uniformly. Therefore, when evaluating the three parameters listed above, the methodology in this section is anticipated to yield a combination of one or more of the ten categories defined in Table 1, Characteristic Isotopes, in Attachment 1. These categories will exist simultaneously.

ATTACHMENT 8

DETAILED RADIOLOGICAL ANALYSIS

(Page 6 of 17)

TABLE 8-1. SAMPLE LOCATIONS RECOMMENDED FOR CORE DAMAGE ASSESSMENT

(Reference Step 4.1.A)

Accident Scenario Known	RCS Hot Leg	RCS Pressurizer	Containment Sump (*)	Containment Atmosphere	Shutdown Cooling System	Steam Generator Secondary
Small Break LOCA, Reactor Power greater than 1 percent	Yes	Yes	---	Yes	Yes	---
Small Break LOCA, Reactor Power less than 1 percent	Yes	Yes	---	---	Yes	---
Small Steam Line Break	Yes	Yes	---	---	---	---
Large Break LOCA, Reactor Power greater than 1 percent	Yes	---	Yes	Yes	Yes	---
Large Break LOCA, Reactor Power less than 1 percent	---	---	Yes	Yes	Yes	---
Large Steam Line Break	Yes	---	---	Yes	---	---
Steam Generator Tube Rupture	Yes	---	---	---	---	Yes

* Available only on recirculation

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 7 of 17)
DATA SHEET 8-1. INPUT PARAMETERS
(Page 1 of 2)

Unit: _____

Reactor Coolant System:

Pressure _____ PSIG

Temperature (T_{avg}) _____ °F

Reactor Vessel Level Shows: Full Void Below Recorder
(Circle One)

Pressurizer Level _____Percent _____

Containment Building:

Atmosphere Pressure _____ PSIG _____

Atmosphere Temperature _____ °F _____

Prior 30 Days Power History:

Power, Percent	Duration, Days
_____	_____
_____	_____
_____	_____
_____	_____
_____	_____

Estimated Average Power Level During Last 30 Days _____ Percent

Estimated Average Power Level During Last 4 Days _____ Percent

Time of Reactor Trip: Date: ____/____/____ Time: _____

Change in volume of RWT: _____ gal. Time: _____

Change in volume of BAMT: _____ gal. Time: _____

SIT injected (yes / no): _____

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 8 of 17)
DATA SHEET 8-1. INPUT PARAMETERS
(Page 2 of 2)

RADIONUCLIDE DATA
(Reference Step 4)

Unit: _____ Sample Number: _____

Sample Location (RCS, Sump, Containment): _____

Time of Sample Collection: _____

Isotope	Measured Specific Activity at STP A(µCi/cc)
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	

NOTE: N/I if not identified.

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
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DATA SHEET 8-2. RECORD OF MEASURED SPECIFIC ACTIVITY (DECAY CORRECTED)
(Reference Step 4.B)

Unit: _____ Time of Reactor (Rx) Trip, Data Sheet 8-1, Input Parameters (Page 1 of 2): _____

Sample Number: _____

Sample Location (RCS, Sump, Containment): _____

Time of Sample Collection: _____

Elapsed Time, t (Rx Trip to Sample): _____ sec.

Isotope	Half Life	Decay Constant λ (1/sec)	Measured Specific Activity @ STP A (μCi/cc)	Decay Corrected Specific Activity, Ao (μCi/cc)
Kr 87	76m	1.5 E-4		
Xe-131m	12d	6.7 E-7		
Xe 133	5.4d	1.5 E-6		
I-131	8d	9.9 E-7		
I-132	2h	8.4 E-5		
I-133	21h	9.3 E-6		
I-135	6.8h	2.9 E-5		
Cs-134	2yr	1.1 E-8		
Rb-88	2m	6.5 E-4		
Te-129	70m	1.7 E-4		
Te-132	78h	2.5 E-6		
Sr-89	52.7d	1.6 E-7		
Ba-140	12.8d	6.3 E-7		
La-140	40h	4.8 E-6		
La-142	90m	1.2 E-4		
Pr-144	17.4m	6.7 E-4		

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

Where: A and λ are as above, and t = time period in seconds from reactor trip to sample collected.

NOTE: N/I if not identified.

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
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DATA SHEET 8-3. RECORD OF FISSION PRODUCT RELEASE SOURCE IDENTIFICATION
(Reference Step 4.C.1)

Unit: _____ Sample Number: _____

Location: _____

Isotope	Decay Corrected Specific Activity Data Sheet 8-2, µCi/cc	Calculated Isotope Ratio*	Activity Ratio in Fuel Pellet Inventory**	Activity Ratio in Gas Gap Inventory**	Identified Source (Gas Gap or Fuel Pellet)
Kr 87			0.2	less than 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	

* *Noble Gas Ratio* – $\dfrac{\textit{Decay Corrected Noble Gas Specific Activity}}{\textit{Decay Corrected Xe-133 Specific Activity}}$

Iodine Ratio – $\dfrac{\textit{Decay Corrected Iodine Isotope Specific Activity}}{\textit{Decay Corrected I-131 Specific Activity}}$

** Table 3.3 of Reference 2.1.2

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
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DATA SHEET 8-4. QUANTITATIVE RELEASE ASSESSMENT WORKSHEET
(Reference Step 4.D)
(Page 1 of 2)

RCS ACTIVITY ($A_{T,RCS}$)

RCS T_{avg} _____°F (At or Near Time of Sample)

Vessel Level Indication (Full, Void, Below Recorder): _____

IF FULL OR VOID, perform the following calculation for each isotope measured. IF BELOW RECORDER, use the Containment Sump calculation below instead.

($A_{T,RCS}$) (C_i) = Ao (μCi/cc) x RCS Volume x 1.0 E-06 (Ci/μCi)

Where: Ao = decay corrected specific activity of RCS sample (Data Sheet 8-2, Record of Measured Specific Activity (Decay Corrected))

RCS volume = Water Volume x Density Ratio at RCS Tavg (Figure 7-1, Ratio of H_2O Density at STP vs. Temperature). PSL1 water volume is 2.945 E+08 cc and PSL2 water volume is 2.889 E+08 cc.

Enter results in Data Sheet 8-5, Record of Core Release Inventory ($A_{T,RCS}$)

SUMP ACTIVITY ($A_{T,sump}$)

Determine sump water volume by adding the following:

			PSL 1	PSL 2
RCS Volume	=	_____ gal	58,300	57,400
SIT Injected Volume	= +	_____ gal	34,049	46,564
BAMT Injected Volume	= +	_____ gal	(Data Sheet C-3.A)	
RWT Volume Change	= +	_____ gal	(Data Sheet C-3.A)	

V_s = Total Sump Volume = _____ gal x 3785 cc/gal = _____ cc

($A_{T,sump}$) = A_o (μCi/cc) x V_s x 1.0 E-06 (Ci/μCi)

Where A_o = decay corrected specific activity of SUMP sample (Data Sheet 8-2, Record of Measured Specific Activity (Decay Corrected))

Enter results in Data Sheet 8-5, Record of Core Release Inventory ($A_{T,sump}$).

ATTACHMENT 8

DETAILED RADIOLOGICAL ANALYSIS

(Page 12 of 17)

DATA SHEET 8-4. QUANTITATIVE RELEASE ASSESSMENT WORKSHEET

(Reference Step 3.D.4)

(Page 2 of 2)

CONTAINMENT ACTIVITY ($A_{T,cont}$)

Calculate Containment Volume in cc, including pressure and temperature corrections.

$$V_c = Containment\ Volume\ (cc) = 7.096\ E10 \times \frac{14.7}{(P1 + 14.7)} \times \frac{(T1 + 460)}{(32 + 460)}$$

Where: P1 = Containment pressure in psig (Data Sheet 8-1, Input Parameters)
T1 = Containment temperature in °F (Data Sheet 8-1, Input Parameters)

$(A_{T,cont})$ = Ao (μCi/cc) x V_c x 1.0 E-6 (Ci/μCi)

Where: Ao = Decay corrected specific activity for containment sample (Data Sheet 8-2, Record of Measured Specific Activity (Decay Corrected))

Enter results in Data Sheet 8-5, Record of Core Release Inventory ($A_{T,cont}$).

ATTACHMENT 8

DETAILED RADIOLOGICAL ANALYSIS

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DATA SHEET 8-5. RECORD OF CORE RELEASE INVENTORY

(Reference Step 4.D.4)

Unit: _____

Isotope	Reactor Coolant Sample $A_{T,RCS}$ (Ci)	Containment Sump Sample + $A_{T,sump}$ (Ci)	Containment Atmosphere Sample + $A_{T,cont}$ (Ci)	= Total Quantity (Ci)
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				

$$Total\ Quantity\ (Ci) = A_{T,RCS} + A_{T,sump} + A_{T,cont}$$

ATTACHMENT 8

DETAILED RADIOLOGICAL ANALYSIS

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DATA SHEET 8-6. RECORD OF STEADY STATE POWER CORRECTION

(Reference Step 4.E.1)

Unit: _____ Average 30 Days Power Level: _____

Average 4 Days Power Level: _____

Isotope	Fuel History Grouping	Power Correction Factor x	Equilibrium Source Inventory*	= Power Corrected Source Inventory
Gas Gap Inventory				
Kr 87	2		1.48 E+05	
Xe 131m	1		4.13 E+04	
Xe 133	1		5.06 E+06	
I 131	1		6.98 E+06	
I 132	2		1.36 E+06	
I 133	2		5.58 E+06	
I 135	2		3.13 E+06	
Fuel Pellet Inventory				
Kr 87	2		3.67 E+07	
Xe 131m	1		7.09 E+05	
Xe 133	1		1.28 E+08	
I 131	1		6.01 E+07	
I 132	2		9.61 E+07	
I 133	2		1.34 E+08	
I 135	2		1.27 E+08	
Cs 134	1		7.73 E+06	
Rb 88	2		5.28 E+07	
Te 129	2		2.09 E+07	
Te 132	1		9.60 E+07	
Sr 89	1		6.98 E+07	
Ba 140	1		1.21 E+08	
La 140	1		1.29 E+08	
La 142	2		1.11 E+08	
Pr 144	2		8.46 E+07	

Corrected Source Inventory = Power Correction Factor x Equilibrium Source Inventory.

* Values from Reference 2.1.4.

Group 1 Power Correction Factor = Average Level for Prior 30 Days / 100.

Group 2 Power Correction Factor = Average Level for Prior 4 Days / 100.

ATTACHMENT 8

DETAILED RADIOLOGICAL ANALYSIS

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DATA SHEET 8-7. RECORD OF TRANSIENT POWER CORRECTION

(Reference Step 3.D.5.B)

Unit: _____

Prior 30 Days Power History:

Power (Percent)	Duration (Days)	Time to Trip (Days)
_____	_____	_____
_____	_____	_____
_____	_____	_____

Isotope	Equilibrium Source Inventory* x	Power Correction Factor =	Power Corrected Source Inventory
Gas Gap Inventory			
Kr 87	1.48 E+05		
Xe 131m	4.13 E+04		
Xe 133	5.06 E+06		
I 131	6.98 E+06		
I 132	1.36 E+06		
I 133	5.58 E+06		
I 135	3.13 E+06		
Fuel Pellet Inventory			
Kr 87	3.67 E+07		
Xe 131m	7.09 E+05		
Xe 133	1.28 E+08		
I 131	6.01 E+07		
I 132	9.61 E+07		
I 133	1.34 E+08		
I 135	1.27 E+08		
Cs 134	7.73 E+06		
Rb 88	5.28 E+07		
Te 129	2.09 E+07		
Te 132	9.60 E+07		
Sr 89	6.98 E+07		
Ba 140	1.21 E+08		
La 140	1.29 E+08		
La 142	1.11 E+08		
Pr 144	8.46 E+07		

Corrected Source Inventory = Power Correction Factor x Equilibrium Source Inventory.

* Values from Reference 2.1.4

ATTACHMENT 8

DETAILED RADIOLOGICAL ANALYSIS

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DATA SHEET 8-8. RECORD OF PERCENT RELEASE

(Reference Step 4.F)

(Page 1 of 2)

Unit: _____

Isotope	Total Quantity Available for Release (Ci) (Data Sheet 8-5)	Power Corrected Source Inventory (Ci) (Data Sheet 8-6 or 8-7)	Percent*
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			

* Percent = (Total Quantity Available for Release ÷ Power Corrected Source Inventory) x 100

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 17 of 17)
DATA SHEET 8-8. RECORD OF PERCENT RELEASE
(Reference Step 4.F)
(Page 2 of 2)

Summary of Results:

_____
_____
_____
_____
_____
_____
_____

NOTE

Compare percent clad damage, percent fuel overheat, and percent fuel melt results obtained from the radionuclide analysis to those obtained from the auxiliary indicators analyses.

If results are in agreement, the core damage assessment is complete. If the results are not in agreement, a re-check of both analyses may be performed or certain indications may be discounted based on engineering judgement.

Prepared by: _____ Date: ____/____/____

Reviewed by: _____ Date: ____/____/____

Approved by: _____ Date: ____/____/____

FPL	ST. LUCIE PLANT HEALTH PHYSICS PROCEDURE SAFETY RELATED	Procedure No. **HP-200** Current Rev. No. **16** Effective Date: **03/27/01**

Title:

HEALTH PHYSICS EMERGENCY ORGANIZATION

Responsible Department: **HEALTH PHYSICS**

Revision Summary

Revision 16 - Added step to TSC HP Supervisor checklist to assist EC with radiological conditions and evaluations of PARs. Made editorial/administrative changes. (J.R. Walker, 03/22/01)

Revision 15 - Deleted low vol air sample from OSC checklist and included checklist for HPN Communicator. (Don Reisinger, 11/30/99)

Revision 14 - Clarified radiation protection technologist and health physics technician positions. Added EAL triggers to TSCHPS Checklist, OSC relocation threshold dose rate information to HPOSC Checklist and editorial changes. (J. R. Walker, 3/2/99)

PSL CONTROL 818 COPY PROCEDURE PRODUCTION

Revision	FRG Review Date	Approved By	Approval Date
0	02/01/82	J. H. Barrow (for) Plant General Manager	02/04/82
16	03/22/01	R. G. West Plant General Manager	03/22/01
		N/A Designated Approver	
		N/A Designated Approver (Minor Correction)	

S___OPS
DATE
DOCT PROCEDURE
DOCN HP-200
SYS
COMP COMPLETED
ITM 16

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

1.0 TITLE:

HEALTH PHYSICS EMERGENCY ORGANIZATION

2.0 REVIEW AND APPROVAL:

See cover page

3.0 PURPOSE:

This procedure defines areas of responsibility and provides general guidelines for action to be taken by Health Physics Department personnel upon implementation of the St. Lucie Plant Radiological Emergency Plan (E-Plan). It also references those Health Physics (HP) procedures necessary to carry out specific HP activities during a declared radiological emergency (Alert, Site Area Emergency and General Emergency).

4.0 PRECAUTIONS AND LIMITATIONS:

4.1 The Health Physics Department is responsible for protecting all personnel from excessive radiological exposures during accident conditions. In order to effectively carry out this responsibility, it is necessary that all HP personnel quickly man their emergency stations and assemble and check their equipment and await directions from the Technical Support Center Health Physics Supervisor (TSCHPS) or his designee.

4.2 The TSCHPS is responsible for the procedures to be implemented and when implementation is to be effected.

4.3 Complete all procedural steps if applicable or indicate as non-applicable by writing N/A in the provided blank.

4.4 When Health Physics normal operating procedures and emergency procedures differ, the emergency procedures take precedence.

4.5 Delegation of duties and watch reliefs shall be authorized only by the TSCHPS or his designee with approval of the Emergency Coordinator.

4.6 It is the responsibility of all personnel to limit their own exposure and to assist others in limiting their exposures.

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

5.0 RELATED SYSTEM STATUS:

None

6.0 REFERENCES:

6.1 St. Lucie Plant Radiological Emergency Plan (E-Plan)

6.2 E-Plan Implementing Procedures (EPIP 00-13)

6.3 HP-2, "FP&L Health Physics Manual"

6.4 NRC I&E Information Notice No. 86-97: Emergency Communications System

6.5 HP-201, "Emergency Personnel Exposure Control"

6.6 HP-202, "Environmental Monitoring During Emergencies"

6.7 HP-203, "Personnel Access Control During Emergencies"

6.8 HP-204, "In-Plant Radiation and Contamination Surveys During Emergencies"

6.9 HP-205, "Emergency In-Plant Air Sampling"

6.10 HP-206, "Analysis of Emergency In-Plant Air Samples"

6.11 HP-207, "Monitoring Evacuated Personnel During Emergencies"

6.12 HP-208, "Personnel Decontamination During Emergencies"

¶1 6.13 PMAI PM97-04-148, OSC Merlins

7.0 RECORDS REQUIRED:

7.1 Completed copies of the below listed documents shall be maintained in the plant files in accordance with QI-17-PSL-1, "Quality Assurance Records."

Forms similar to:

1. Form HP200.1, Technical Support Center Health Physics Supervisor Checklist

2. Form HP200.2, Health Physics OSC Supervisor (HPOSC) Checklist

3. Form HP200.3, HPN Communicator Checklist

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

8.0 INSTRUCTIONS:

8.1 Health Physics Emergency Organization

1. TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR (TSCHPS)

   The Health Physics (HP) Supervisor or his alternate (see the St. Lucie Plant Emergency Response Directory (ERD)) shall assume duties as the TSCHPS in the Technical Support Center (TSC). He is responsible for all HP activities and reports to the Emergency Coordinator (EC). A TSCHPS Checklist, Form HP-200.1 is provided in this procedure. See Section 8.2 for instructions to be followed prior to activation of the TSC and/or prior to arrival of the TSCHPS. /R16

2. HEALTH PHYSICS OSC SUPERVISOR (HPOSC)

   The senior HP Operations Supervisor shall assume duties as the HP Supervisor in the Operational Support Center (HPOSC). He reports to the TSCHPS in the TSC. He is responsible for coordinating all HP activities from the Operational Support Center (OSC). A HPOSC Supervisor Checklist, Form HP-200.2 is provided in this procedure. /R16

3. HEALTH PHYSICS TECHNICAL STAFF

   The Health Physics Technical Staff shall report immediately to the TSCHPS for assignment.

4. HEALTH PHYSICS TECHNICIAN (HPT)

   Radiation Protection Technologists (RPTs) assume the role of Health Physics Technicians (HPTs = OSCHP Tech) and shall immediately report to or be in contact with the OSC. They will be assigned duties by the HPOSC. /R16

8.2 ON-SHIFT HEALTH PHYSICS RESPONSE TO EMERGENCIES /R16

1. An Emergency Class declaration of an Alert or higher during off-normal working hours will require additional HP staffing. The senior HP representative on-site will implement the HP emergency procedures. It is expected that this initial period will last for about one hour. /R16

8.0 INSTRUCTIONS: (continued)

8.2 (continued)

2. The senior HP representative on-site shall notify the Emergency Coordinator and apprise him of HP assistance available on-site. He will take his orders directly from the Emergency Coordinator and should assume his duties in the plant or OSC and not in the TSC unless otherwise directed by the Emergency Coordinator. He should attempt to remain in a position to be reached by the Emergency Coordinator if necessary. /R16

3. Since there will be only limited health physics coverage available, it is very important for the senior HP representative on-site to discuss with the Emergency Coordinator (or his designee) the coverage which each feels is necessary and to prioritize that coverage. The following list may be used to assist in the decision of assigning priorities: (in order of preference) /R16

   A. Radiological coverage necessary to allow expedient entry to areas when required to place the plant in a safe condition

   B. Treatment of contaminated personnel

   C. Radiological coverage during high activity sampling

   D. Preparations for extensive in-plant monitoring and surveillance

4. When the additional HP support arrives, the initial period will have passed. In order to maintain continuity and to effect a smooth transfer from the interim to the fully staffed mode it is necessary that the HP command function not change hands more than is absolutely necessary. Therefore, even though the senior HP representative on-site can be relieved by a more senior Technologist or Supervisor, he should not be relieved by anyone except the HP Supervisor or his alternate. /R16

   HP personnel shall report to the OSC when they arrive on-site and should contact the senior HP representative on-site for assignments. /R16

5. The senior HP representative on-site shall initiate the TSCHPS's Checklist, Form HP-200.1. /R16

8.0 INSTRUCTIONS: (continued)

8.3 Health Physics Emergency Operations Areas

1. Operations HP personnel will assemble in and work out of the OSC. HP Technical Staff personnel will assemble and work primarily in the TSC.

2. The TSCHPS will determine if the affected unit Reactor Auxiliary Building (RAB) Control Point is to be manned and will inform the EC and the HPOSC.

3. If the affected unit RAB Control Point becomes untenable, the TSCHPS will direct the HPOSC to man the RAB Control Point of the unaffected unit or designate an alternate Control Point. The TSCHPS will inform the EC of the alternate location.

8.4 Logistics and Supplies

1. Records and logs specified in the specific HP emergency procedures shall be kept up to date and shall be reviewed by the TSCHPS.

2. Emergency radiation protection supplies are located for use in the following places:

    1. RAB Control Points (Unit 1 & Unit 2)

    2. Operational Support Center (OSC)

    3. Site Assembly Station (SAS)

    4. Unit 1 Control Room (for use by TSC and Unit 1 C.R. personnel)

    5. Unit 2 Control Room

3. The HPOSC will ensure that materials and equipment are provided to operating areas as needed.

8.0 INSTRUCTIONS: (continued)

8.5 Emergency Personnel Exposure Control (HP-201)

1. The TSCHPS shall ensure that all personnel on-site during emergency operations wear proper dosimetry. He shall determine when special dosimetry is required. /R16

2. The HPTs through the HPOSC will provide radiological surveys and/or coverage for all areas in which personnel access is required.

3. If personnel exposures are likely to exceed plant guidelines, the guidelines in Health Physics Procedure HP-201, "Emergency Personnel Exposure Control," shall be followed.

4. All personnel exposures during emergency operations will be maintained As Low As Reasonably Achievable - ALARA.

8.6 Off-site and On-Site Environmental Monitoring (HP-202)

1. The Emergency Coordinator is responsible for ensuring that the TSCHPS initiates off-site radiological monitoring, in accordance with the E-Plan, within a 10 mile radius of the plant. Off-site field monitoring activities will be coordinated with the State of Florida.

2. The TSCHPS will direct the HPOSC to dispatch the Field Monitoring Teams to the Site Assembly Station.

> NOTE
> If the Field Monitoring Team communicator/control has not been activated, the Field Monitoring Teams should make contact with the OSC and report their status.

3. The Field Monitoring Teams will assemble their equipment, check it for operability and establish contact with the TSC. The TSCHPS in the TSC provides supervision for the Field Monitoring Teams as per EPIP-10, "Off-site Radiological Monitoring," and HP-202, "Environmental Monitoring During Emergencies."

8.0 INSTRUCTIONS: (continued)

8.7 Personnel Access Control (HP-203)

1. No re-entry into areas affected by the emergency shall be made unless authorized by the Emergency Coordinator.

2. The initial entry of the Re-entry Team and all subsequent entries, until radiation areas have been properly marked, shall take place under the supervision of the TSCHPS as per EPIP-05, "Activation and Operation of the Operational Support Center."

3. Following re-entry procedures, the TSCHPS will direct the HPOSC to establish the access control point(s). The HPTs shall maintain access control to all affected areas of the plant for the purpose of controlling personnel exposures as per HP-203, "Personnel Access Control During Emergencies."

8.8 Radiation and Contamination Surveys (HP-204)

NOTE

In the event of a Steam Generator Tube Rupture (SGTR), the following areas should initially be posted as contaminated:

A. Steam Trestle
B. Condenser Air Ejector
C. Condensate Polisher

1. The Emergency Coordinator and TSCHPS will determine the extent of surveys required.

2. The TSCHPS will direct the HPOSC to establish survey teams utilizing the buddy system. The HPOSC will direct the conduct of all in-plant surveys, ensure data is properly recorded and posted and keep the TSCHPS informed of the results.

3. Surveillance for emergency situation shall include as a minimum:

    1. Radiation surveys

    2. Contamination surveys

8.0 INSTRUCTIONS: (continued)

8.8 (continued)

3. (continued)

3. Airborne activity surveys

4. Radiological monitoring of potentially high activity chemistry sample operations

5. Surveys as called for in the Emergency Operating Procedures (EOPs).

6. Special surveys as determined by the TSCHPS.

8.9 In-Plant Air Sampling and Counting (HP-205, HP-206)

1. During an emergency, higher than normal radiation levels and airborne concentrations can be expected. It is important that sampling be commenced as expeditiously as possible to support rapid re-entry if necessary.

2. All in-plant air sampling will be performed in such a manner as to ensure personnel exposures are ALARA.

3. The procedures HP-205, "Emergency In-Plant Air Sampling" and HP-206, "Analysis of Emergency In-Plant Air Samples," should be followed in sampling and analyzing samples.

8.10 Personnel Monitoring Following Evacuation (HP-207)

1. In the event it becomes necessary to evacuate personnel from the plant and a release has occurred or is in progress, check points will be established immediately to allow monitoring of these personnel. The check points will be at Jaycee Park, unless alternate routes and assembly locations are specified by the EC.

8.0 INSTRUCTIONS: (continued)

8.11 Personnel Decontamination (HP-208)

1. Personnel decontamination following an accident can pose special problems not encountered in everyday situations including extremely high levels of contamination and/or large numbers of personnel being contaminated at the same time.

2. Personnel decontamination at the Off-site Assembly Area will be under the cognizance of the TSCHPS and will be directed by his designee at that area. HP-208, "Personnel Decontamination During Emergencies," addresses off-site personnel decontamination. /R16

3. Personnel decontamination on-site will be under the direction of the HPOSC and should be conducted in the hot shower area of the unaffected unit or at a location specified by the TSCHPS.

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.1
TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR CHECKLIST

INITIAL

1. Contact the OSC and assess available HP resources on site: _____

   a. TSC

      HP Network Communicator _____

      Field Monitoring Team Communicator _____

   b. OSC

      Number of HP Techs _____

      Number of Dosimetry Techs _____

      Number of Utility Workers _____

      HPOSC Supervisor _____

   c. Number of HPTs assigned to Unit 1 Control Room _____

   d. Number of HPTs assigned to Unit 2 Control Room _____

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.1
TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR CHECKLIST
(continued)

INITIAL

CAUTION
Be aware of the following conditions. These Emergency Action Levels (EALs) are associated with Initiating Conditions (ICs) used in the classification of emergencies (EPIP-01, Classification of Emergencies) the Emergency Coordinator needs to know if any of these conditions exist.

1. Measured dose rates from off-site surveys at the site boundary (1 mile) exceed either of the following:

   a. 1000 mrem/hr (total dose rate)

   b. 5000 mrem/hr (thyroid dose rate)

2. Contact Emergency Coordinator (EC)

   a. Inform EC of HP Department status in the TSC and OSC. _____

   b. Determine and prioritize immediate HP coverage needs:

NOTE
Operations will require that radiation surveys be conducted to support the EOPs.

Priority	Job/Location	# HPs Required	
_____	_____	_____	
_____	_____	_____	
_____	_____	_____	
_____	_____	_____	_____

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HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.1
TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR CHECKLIST
(continued)

2. (continued) INITIAL

   c. Event Classification Date/Time

      a) UNUSUAL EVENT _____/_____
      b) ALERT _____/_____
      c) SITE AREA EMERGENCY _____/_____
      d) GENERAL EMERGENCY _____/_____

3. OSC Considerations

   a. Establish communications with the OSC (if activated). _____

   b. Appoint HPOSC Supervisor Name:_____

      Phone:_____ _____

   c. Inform HPOSC to complete HPOSC Checklist HP-200.2. _____

   d. Exchange information on plant status, event classification, available personnel and prioritize jobs requiring HP coverage. Also, discuss planning strategies and personnel allocations. _____

   e. Direct HPOSC to dispatch Field Monitoring Teams according to the following classification schedule:

      ALERT - On-site, out of plant - 1 Team (RED)

      SITE AREA OR GENERAL EMERGENCY - On-site, out of plant - 1 Team (RED) and off-site - 2 Teams (ORANGE,BLUE)

      _____

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.1
TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR CHECKLIST
(continued)

3. (continued) INITIAL

> NOTE
> It is possible for the TSC and OSC to be in operation for weeks following a plant accident. It is the TSCHPS responsibility to determine when to activate frisking stations for those facilities and the locations of those stations. Once frisking is initiated all personnel entering these facilities shall be monitored for contamination.

f. Direct HPOSC to establish RCA/RAB access control points if portions of the RCA were evacuated or RAB re-entry is planned.

CONTROL POINT LOCATIONS _____
_____ _____

g. If a Site Evacuation has been ordered and a release has occurred or is in progress, direct the HPOSC to dispatch two HPTs to the Off-Site Assembly Area to monitor the evacuees. _____

h. If additional HP personnel resources may be needed, consider requesting assistance through:

- PTN
- Industry (through INPO)
- Department of Energy (through NRC) _____

4. TSC Considerations

a. In conjunction with the TSC Chemistry Supervisor, advise the EC and/or TSC EC Assist/Logkeeper on radiological conditions and Protective Action Recommendations, as necessary. _____

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HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.1
TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR CHECKLIST
(continued)

4. (continued) INITIAL

b. Assign person to continuously man the Health Physics Network (HPN) phone.

(Ref. I&E Notice 86-97) _____

c. Contact the TSC Chemistry Supervisor to determine if accident samples are required (yes/no). _____

d. Assign person to direct field monitoring teams and evaluate the data as it becomes available. _____

e. Confer with chemistry and/or the EOF on dose projections and effected EPZ sectors. _____

f. Establish communications between the TSC Field Monitoring Team and the EOF Field Monitoring Coordinator (when activated).

1. Exchange names and phone numbers

Name(s)_____ Phone # _____

2. Relay field monitoring results as available. _____

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.1
TECHNICAL SUPPORT CENTER HEALTH PHYSICS SUPERVISOR CHECKLIST
(continued)

4. (continued) INITIAL

g. Direct that habitability surveys of the following areas are performed. _____

* 1. Control Room No. 1

* 2. Control Room No. 2

* 3. TSC (turn on CAM)

4. OSC

5. RAB Control Points (if inhabited)

* Surveys of TSC and Control Rooms should be completed by personnel assigned to those areas.

5. Additional Comments by TSCHPS:

Completed by:_____

Date Completed:____/____/____

NOTE
File this Checklist in accordance with QI-17-PSL-1, "Quality Assurance Records."

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.2
HPOSC SUPERVISOR CHECKLIST

INITIAL

1. Perform accountability of HPTs and provide to OSC Supervisor: _____

> NOTE
> Consult the Radiation Exposure Summary Report for selection of respirator qualified field monitoring team members.

2. Dispatch on-site Field Monitoring Team to Site Assembly Station when directed by TSCHPS. _____

   1 HPT, 1 driver and vehicle. _____

3. Dispatch off-site Field Monitoring Teams to Site Assembly Station when directed by TSCHPS. _____

   2 HPTs, 2 drivers, 2 vehicles.

4. If RCA is NOT EVACUATED start preparations for accumulating supplies and instruments in the event of RCA evacuation. _____

5. If RCA is NOT EVACUATED perform necessary job coverage for OPS or maintenance personnel attempting to mitigate the problem. _____

HP200.2
HPOSC SUPERVISOR CHECKLIST
(continued)

INITIAL

6. If RCA evacuation is necessary, perform the following:

   1. Dispatch HPT to each Control Room _____

   2. Remove Instruments to OSC _____

¶1 3. Take boxes of Electronic Dosimeters from RAB entrance stations to OSC. _____

   4. Determine H.P. Personnel available in OSC Report to the TSCHPS and OSC Supervisor _____

   5. Establish OSC access control points(s) _____

   6. Establish Dosimetry Section in OSC. _____

   7. Establish Contamination Control for OSC. _____

   8. Inform TSCHPS _____

   9. Setup a Continuous Monitoring Count Rate Meter In OSC _____

> CAUTION
> The OSC affords limited protection against a release of radioactive material. During the time that a radioactive release is occurring, the habitability of the OSC is to be monitored. A measured dose rate of 50 mrem/hr, in the facility, is established as the threshold for relocation of the OSC.

   10. Perform habitability surveys of the OSC and, if inhabited, RAB Control Point and provide survey results to TSCHPS. _____

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP200.2
HPOSC SUPERVISOR CHECKLIST
(continued)

INITIAL

7. If directed by the TSCHPS, dispatch two HPTs to the offsite assembly area:

   Name:_____ Dispatch Time:_____

   Name:_____ Dispatch Time:_____ _____

8. Ensure all Reentry Teams are adequately briefed in accordance with HP-203.1 prior to being dispatched from the OSC. _____

9. Advise and assist the OSC Supervisor as necessary in actions to mitigate accident. _____

10. HPOSC Supervisor Comments:

Completed by:_____

Date Completed:____/____/_____

NOTE
File this Checklist in accordance with QI-17-PSL-1, "Quality Assurance Records."

ST. LUCIE PLANT
HEALTH PHYSICS PROCEDURE NO. HP-200, REVISION 16
HEALTH PHYSICS EMERGENCY ORGANIZATION

HP 200.3
HPN COMMUNICATOR CHECKLIST

INITIAL

1. Report / Sign in / Obtain ID Badge. _____

2. Get status from TSC HPS. _____

3. Get HP supply case & procedures. _____

> NOTE
> Typical HPN Questions:
>
> A. Meteorological Data
> B. Release data
> C. Plant radiological inquiries
> D. Field team data
> E. Assist FMT Coordinator

4. Activate HPN phone: _____

   A. Call NRC - Identify self & activity. _____

   B. Request to be coupled to the HPN bridge network. _____

5. Review ERDADS (Plant Rad Monitor Data). _____

6. Review off-site Dose Rad Assessment Board. _____ /R16

7. Review off-site monitoring field team status board. _____ /R16

8. Provide data to TSC HPS. _____

9. Assist TSC HPS as needed (dispense dosimetry, set-up air monitor, TSC Rad Surveys, provide rad monitor data, etc.). _____

> NOTE
> File this Checklist in accordance with QI-17-PSL-1, Quality Assurance Records.

FPL	ST. LUCIE PLANT HEALTH PHYSICS PROCEDURE SAFETY RELATED	Procedure No. **HP-207** Current Rev. No. **11** Effective Date: **03/27/01**

Title:

MONITORING EVACUATED PERSONNEL DURING EMERGENCIES

Responsible Department: **HEALTH PHYSICS**

Revision Summary

Revision 11 - Revised name of the Off-site Assembly Area at Jensen Beach. Changed SSN to TLD on Frisking Log. Made editorial/administrative changes. (J.R. Walker, 03/22/01)

Revision 10 - Added minor administrative changes. (J. R. Walker, 03/18/99)

PSL CONTROL COPY 018 PROCEDURE PRODUCTION

Revision	FRG Review Date	Approved By	Approval Date
0	02/01/82	J. H. Barrow (for) Plant General Manager	02/04/82
Revision	FRG Review Date	Approved By	Approval Date
11	03/22/01	R. G. West Plant General Manager	03/22/01
		N/A Designated Approver	
		N/A Designated Approver (Minor Correction)	

S___OPS
DATE
DOCT PROCEDURE
DOCN HP-207
SYS
COMP COMPLETED
ITM 11

ST. LUCIE PLANT
HEALTH PHYSICS EMERGENCY PROCEDURE NO. HP-207, REVISION 11
MONITORING EVACUATED PERSONNEL DURING EMERGENCIES

1.0 TITLE:

MONITORING EVACUATED PERSONNEL DURING EMERGENCIES

2.0 REVIEW AND APPROVAL:

See cover page

3.0 PURPOSE:

This procedure provides guidelines for monitoring all plant personnel during emergencies.

4.0 PRECAUTIONS AND LIMITATIONS:

4.1 This procedure shall be used during site evacuations.

4.2 Every effort shall be made to minimize personnel contamination and radiation exposure.

4.3 Personnel monitoring check points should be established outside of the affected area. They should be in an area of low background radiation and contamination.

4.4 Caution should be exercised early in the event to verify the check point is sufficiently equipped and arranged to prevent the spread of contamination.

5.0 RELATED SYSTEM STATUS:

None

6.0 REFERENCES:

6.1 St. Lucie Plant Radiological Emergency Plan (E-Plan).

6.2 E-Plan Implementing Procedures (EPIP00-13). /R11

6.3 HP-2, FP&L Health Physics Manual.

6.4 HPP-30, Personnel Monitoring.

6.5 HPP-70, Personnel Contamination Monitoring.

6.6 HP-208, Personnel Decontamination During Emergencies.

7.0 RECORDS REQUIRED:

7.1 Completed copies of the below document shall be maintained in the plant files in accordance with QI-17-PSL-1 "Quality Assurance Records."

1. Form HP207.1, Personnel Monitoring/Frisking Log.

ST. LUCIE PLANT
HEALTH PHYSICS EMERGENCY PROCEDURE NO. HP-207, REVISION 11
MONITORING EVACUATED PERSONNEL DURING EMERGENCIES

8.0 INSTRUCTIONS:

8.1 Jaycee Park/Jensen Public Beach Parking Area: /R11

1. Following a site evacuation order, and if radiological conditions warrant action, the Technical Support Center Health Physics Supervisor (TSCHPS) will direct personnel to establish check points at Jaycee Park or Jensen Public Beach Parking Area as directed by the Emergency Coordinator (EC). A radio equipped vehicle should be used by HP personnel. /R11

2. Take additional copies of HP207.1, "Personnel Monitoring/Frisking Log" and HPP-70.1, "Personnel Skin and Clothing Contamination Report" for use at the Off-site Assembly Area.
3. Locate the check point in a convenient area to allow entry and exit without spread of contamination.

4. Personnel should be kept near the entrance until they can be monitored to prevent the spread of contamination.

5. If personnel are expected to be contaminated, they should be kept still and away from others until they have been monitored and declared clean.

6. All personnel shall be monitored using a Count Rate Meter and Beta Sensitive Probe and results recorded on form HP207.1.

7. Contaminated personnel shall be segregated and decontaminated in accordance with HP-208, 'Personnel Decontamination During Emergencies.'

8. Results of personnel monitoring shall be communicated to the TSCHPS or his designee.

9. Records of personnel monitoring shall be retained and forwarded to the TSCHPS upon his request.

10. If additional OSC HP Tech support is required, contact the TSCHPS.

8.2 Operational Support Center (OSC):

1. Following activation of the OSC, HP Supervisor in the OSC (HPOSC) will direct OSC HP Techs to establish a check point at the OSC.

2. The check point will be located in a convenient area to allow entry and exit without spread of contamination.

8.0 INSTRUCTIONS: (continued)

8.2 (continued)

3. Personnel entering the OSC from other plant areas will be kept near the entrance until they can be monitored to prevent the spread of contamination.

4. If personnel are expected to be contaminated, they should be kept still and away from the normal entrance until they can be monitored. Use Anti-C's and remote monitoring to prevent spread of contamination to the normal OSC entrance.

5. All personnel entering the OSC shall be monitored using a Count Rate Meter and Beta Sensitive Probe or Dose Rate Instrument and results recorded on form HP207.1.

6. Contaminated personnel shall be segregated and decontaminated in accordance with HP-208.

7. Results of personnel monitoring shall be kept in the OSC. The TSCHPS shall be informed of any contaminated individuals being found.

8.3 Technical Support Center (TSC):

1. Traffic in and out of the TSC shall be kept to a minimum.

2. If traffic is necessary, a frisking record shall be initiated.

3. Personnel attempting to enter the TSC and are found to be contaminated shall be denied entrance to the TSC. They should be sent to the OSC for decontamination processing. Notify OSC of situation.

ST. LUCIE PLANT
HEALTH PHYSICS EMERGENCY PROCEDURE NO. HP-207, REVISION 11
MONITORING EVACUATED PERSONNEL DURING EMERGENCIES

HP 207.1
PERSONNEL MONITORING/FRISKING LOG

/R11

NAME, TLD OR BADGE NO.	ASSEMBLY LOCATION	DATE/TIME	READING[1] DPM	INSTRUMENT USED MODEL/SER.NO.	SURVEYOR INITIALS
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[1]If reading is above background refer to HP-208, 'Personnel Decontamination During Emergencies'.