FPL.

February 29, 2016

L-2016-042
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, is a copy of the revised procedures that implement the Emergency Plan listed below.

Number	Title	Revision	Implementation Date
EPIP-03	Emergency Response Organization Notification / Staff Augmentation	27	February 18, 2016
EPIP-08	Off-Site Notifications And Protective Action Recommendations	40	February 18, 2016
EPIP-11	Core Damage Assessment	7	February 18, 2016

A revision summary is provided on the cover of each enclosed procedure. Contact us if there are questions regarding these procedures.

Sincerely,

SCISCENTE FOR ESK

Eric S. Katzman
Licensing Manager
St. Lucie Plant

ESK/tlt

Enclosures

AX45
NRR

Florida Power & Light Company

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Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED ____ INITIAL ____

FPL	ST. LUCIE PLANT EMERGENCY PLAN IMPLEMENTING PROCEDURE SAFETY RELATED **REFERENCE USE**	Procedure No. **EPIP-03** Current Revision No. **27**

Title:

EMERGENCY RESPONSE ORGANIZATION NOTIFICATION / STAFF AUGMENTATION

Responsible Department: **EMERGENCY PREPAREDNESS**

PSL CONTROL COPY 818 NUCLEAR RECORDS

REVISION SUMMARY:

Revision 27 - Incorporated PCR 2105870 to add instruction for desktop activation of the autodialer. (Author: J. Moody)

Revision 26 - Incorporated PCR 2076775 to remove the manual call-out process in Step 5.2 for ERO augmentation. (Author: J. Moody)

Revision 25 - Incorporated PCR 2014731 to remove the use of the Alphamate terminals for activation of the pagers. (Author: J. Moody)

Revision 24 - Incorporated PCR 1943239 to combine the EP Manager manual callout responsibilities into the TSC EP Coordinator's responsibilities. (Author: F. Baker)

Revision 23 - Incorporated PCR 1894809 to update procedure number reference. (Author: E. Young)

Revision 22 -Incorporated PCR 1854616 to update Shift Communicator actions and update Attachment 3 pager activation. (Author: R. Sandford)

Revision 21 - Incorporated PCR 1846204 show Nuclear Engineer removed from Emergency Staffing call tree. (Author: R. Young)

Revision 20 - Incorporated PCR 1834253 to replace the Duty Call Supervisor with Shift Communicator. (Author: Robert Sanford)

Implementation Date 2/18/16

Initials

Revision	Approved By	Approval Date
0	J. Scarola	01/30/98
27	D. Taylor	02/15/16

UNIT #	
DATE	
DOCT	PROCEDURE
DOCN	EPIP-03
SYS	
STATUS	COMPLETED
REV	27
# OF PGS	

TABLE OF CONTENTS

1.0 PURPOSE

> **NOTE**
>
> ¶1 The Staff Augmentation process is an essential part of the Emergency Plan in that it puts in place the resources necessary to mitigate an accident and protect the health and safety of the public.

This procedure provides instructions to:

1.1 Activate the St. Lucie Plant Emergency Response Organization (ERO) for staff augmentation in response to an emergency declaration.

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, Operating Experience, License Renewal, etc. and shall NOT be revised without the required Focus review and appropriate 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.
>
> Ψ Indicates a step that requires a sign off on an attachment.

2.1 References

1. §1 St. Lucie Plant Radiological Emergency Plan (E-Plan)
2. RP-SL-100-1005, Radiation Protection Emergency Organization
3. OPS-521, Emergency Operating Procedure Implementation
4. SY-AA-100-1010, For Cause Drug and Alcohol Testing Post Accident / Event or Observed Behavior
5. SY-AA-100-1013, Call In Testing
6. St. Lucie Plant Emergency Response Directory (ERD)
7. RM-AA-100-1000, Processing Quality Assurance Records

2.2 Records Required

1. None

2.3 Commitment Documents

1. ¶1 Condition Report CR 00-0544 - QA Audit QSL-EP-00-02: Discrepancies with Primary and Backup ERO Callout Processes

2. ¶2 Condition Report CR 02-0333 (Role of Duty Call Supervisor)

3. ¶3 Condition Report CR 03-0246 (NRC Recommended Procedure Improvements Regarding Security Events)

3.0 RESPONSIBILITIES

3.1 The Emergency Coordinator (EC) has the overall responsibility for the notification and call-out of the ERO as provided for in EPIP-02, Duties and Responsibilities of the Emergency Coordinator.

3.2 The Shift Communicator (SC)

1. The Shift Communicator reports to the affected Unit Control Room upon declaration of the emergency.

2. Complete the following as directed by the SM / EC:

   A. ¶2 Transmittal of off-site notifications (EPIP-08).

   B. Staff augmentation (per this procedure).

   C. Assists the SM / EC in making emergency off-site notifications and performing other activities, as directed.

   D. Conducts a turnover with the TSC OPS Coordinator (SM Communicator in the Control Room) regarding the status of communications and other tasks underway.

3.3 Members of the Emergency Response Organization (ERO):

1. §1 Advise the Emergency Preparedness Manager when his / her duties are changed such that he / she can no longer participate in the ERO.

2. When notified, report to the assigned Emergency Response Facility (ERF).

3.4 Emergency Preparedness Manager

1. §1 Ensure verification of the following for ERO personnel quarterly:

   A. Personnel phone / pager numbers

   B. Training qualifications in accordance with EPIP-12, Maintaining Emergency Preparedness, Radiological Emergency Plan Training.

3.5 ¶1 The Emergency Preparedness Manager is responsible to ensure that both primary and backup staff augmentation methodologies are adequately maintained. The requirements for maintaining the augmentation methodologies are detailed in EPIP-13, Maintaining Emergency Preparedness - Emergency Exercises, Drills, Tests and Evaluations.

4.0 DEFINITIONS

4.1 Autodialer

See FPL Emergency Recall System below.

4.2 §1 Shift Communicator (SC)

Shift Communicator - a specific shiftly designated individual trained and qualified to assist the Shift Manager / Emergency Coordinator in the control room in making emergency off-site notifications, activating the Emergency Response Organization, and performing other activities as directed.

4.3 Emergency Response Organization (ERO)

A trained group of personnel that are designated to perform specific duties during emergencies.

4.4 St. Lucie Plant Emergency Response Directory (ERD)

A directory which contains the names, positions, home phone numbers, and pager numbers for the members of the ERO.

4.5 FPL Emergency Recall System (ERS)

A computer-based automated call-out system used to activate the ERO. This system is also referred to as the "autodialer".

5.0 INSTRUCTIONS

5.1 Emergency Coordinator (EC)

1. Instructions for the EC are located in EPIP-02, Duties and Responsibilities of the Emergency Coordinator.

END OF SECTION 5.1

5.2 Shift Communicator (SC)

NOTE

- ERO shall be activated by one of the following methods within 5 minutes of the Emergency Declaration:
  - Emergency Recall System (Autodialer) (Primary)
  - Microsoft Outlook (Backup)
- Instructions for activation of the autodialer are located in the Shift Communicator Notebook which is maintained in accordance with OPS-521, Emergency Operating Procedure Implementation.
- ¶3 The Emergency Coordinator (EC) may request that special instructions be provided to the Emergency Response Organization (ERO) due to radiological conditions or a Security Event at the plant.
- Use Attachment 2 to activate the ERO during a Security Event.

1. Activate the Autodialer using the CommunicatorNXT link on a Control Room workstation, instructions in Attachment 1 of this procedure.

2. If the Autodialer fails, Then activate the ERO using Microsoft Outlook, instructions in Attachment 2 of this procedure, and activate the autodialer in accordance with Attachment 3.

3. If Step 5.2.1, Step 5.2.2 and Step 5.2.3 do not work, Then contact the EP Coordinator on duty.

END OF SECTION 5.2

5.3 ERO Members with no call-out duties

Report at once to your assigned emergency response facility.

<u>If</u> consumed alcohol in the past 5 hours, <u>Then</u> report to Security prior to entering the site or EOF.

END OF SECTION 5.3

ATTACHMENT 1

EMERGENCY RECALL SYSTEM (ERS) DESKTOP ACTIVATION INSTRUCTIONS

(Page 1 of 1)

1. Login to Communicator!NXT using the Log-In information found in the Shift Communicator Notebook:
   - Login Name
   - Password
   - Company Name
2. Select the appropriate ERO activation scenario based on the event occurring (Check the appropriate scenario box):
   - PSL 30 - 911 HAB
   - PSL 50 - 911 Emergency
3. Click on **Proceed to Activation** under the Activation Options section along the left side of the screen.
4. Click the **Message Options** Tab.
5. Modify the messages as needed (with approval from the Emergency Coordinator):
   - Voice Message (Click **Render Speech** if this message is updated)
   - Alpha Pager Message
6. Click the green **Activation** button at the top of the page.
7. Log Out of the system.

END OF ATTACHMENT 1

ATTACHMENT 2
MICROSOFT OUTLOOK PAGING INSTRUCTIONS
(Page 1 of 1)

1. Open MICROSOFT OUTLOOK.
2. Select "New" to open a new email.
3. Select "To" to obtain address.
4. Select "All Pagers" in Address Book.
5. Scroll to the bottom of the pager list.
6. Select ZZZPSLEROEplan Pager, Activation.
7. Select "To->".
8. Select "OK".
9. In the "**Subject:**" section type your message (125 characters available on a single page).

   A. Examples of pager messages:
   - ALERT EMERGENCY! PSL Site Responders to facility now 911
   - SITE AREA EMERGENCY! PSL Site Responders to facility now 911
   - GENERAL EMERGENCY! PSL Site Responders to facility now 911

   B. Examples of pager messages when a release is occurring or expected during staffing:
   - GENERAL EMERGENCY! PSL Site Responders to facility now 911, approach plant from *north* to avoid release
   - GENERAL EMERGENCY! PSL Site Responders to facility now 911, approach plant from *south* to avoid release

   C. Examples of pager messages when a security event is occurring:
   - PSL ERO: Security Event. ERO members currently off-site report to the EOF now. 911
   - PSL ERO: Security Event. ERO members currently on-site take cover and do not move. 911

10. Click on "Send".
11. Go to section 5.2, step 2, of this procedure and continue.

END OF ATTACHMENT 2

ATTACHMENT 3

FPL EMERGENCY RECALL SYSTEM (ERS) ACTIVATION CHECKLIST

(Page 1 of 2)

SYSTEM DIALOGUE AND RESPONSE

WHEN THE SYSTEM STATES....	YOU SHOULD RESPOND....
"This is the remote activation module. Please enter your **User ID** followed by the pound sign."	On the telephone keypad, **enter the five-digit User ID and the # sign.**
"Please enter your security PIN code followed by the pound sign."	On the telephone keypad, enter your five-digit security PIN code and the # sign.
"To start a scenario, enter the scenario ID followed by the # sign or press # alone for more options."	On the telephone keypad, enter the appropriate scenario to be used. **30#** - Security Event **50#** - phone calls with paging (ERO off site)
"If the scenario has been assigned a scenario PIN, then the system will prompt you for the assigned scenario PIN.	Enter the scenario PIN.
"To listen to the current scenario message press 1. To re-record the scenario message press 2. To start the scenario press 3. To return to the main menu press #."	On the telephone keypad, **enter 1.**
Message Plays	Listen for appropriateness (see note below).
"To listen to the current scenario message press 1. To re-record the scenario message press 2. To Start the scenario press 3. To return to the main menu press #."	If the message is correct, Then **enter 3**. If the message is not correct or requires additional information, Then enter 2, record a new message and press the # sign. When no change / further change is necessary, enter 3.
"The scenario is building. To start a scenario press 1. To stop a scenario press 2. To check scenario information press 3. To log on as a different user press 4. To end this call press #."	On the telephone keypad, **enter #**.
"Thank you. Goodbye."	Hang-up phone and monitor email for confirmation of autodialer activation.

Note *regarding changing the scenario message:*
If you choose you could give specific instructions to plant personnel. e.g., "Personnel should approach the plant from the south, as northern route is closed because of barge damage to Ft. Pierce bridge."

ATTACHMENT 3

FPL EMERGENCY RECALL SYSTEM (ERS) ACTIVATION CHECKLIST

(Page 2 of 2)

Name: ______________________________ Unit: ________

Date: ____/____/____ Time: ________

1. Obtain the Emergency Recall System (ERS) five-digit User ID, Security PIN, and Scenario (refer to the SC Notebook).

2. Prior to making the call, determine the appropriate Scenario ID to activate.

A. If the site is in a Security Event, Then use Scenario ID 30.

B. If the site is in a non-Security Event, Then use Scenario ID 50.

Scenario ID to be used:______

3. Call the Emergency Recall System at **866-775-4182**.

4. When the system answers, follow the on-line instructions and provide the requested information (system dialogue and response detailed on next page).

When finished, go to section 5.2, step 3, of this procedure and continue.

END OF ATTACHMENT 3

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FPL

ST. LUCIE PLANT

EMERGENCY PLAN IMPLEMENTING PROCEDURE

SAFETY RELATED

REFERENCE USE

Procedure No.

EPIP-08

Current Revision No.

40

Title:

OFF-SITE NOTIFICATIONS AND PROTECTIVE ACTION RECOMMENDATIONS

CONTROL COPY 018 NUCLEAR RECORDS

Responsible Department: **EMERGENCY PLANNING**

REVISION SUMMARY:

Revision 40 - Incorporated PCR 2083644 in support of ERO Teams and ERO realignment. (Author: C. Couture)

Revision 39 - Incorporated PCR 2070204 Section 1.1.8.B Substeps 1 and 3 are reversed and conflict with attachment 2 flow chart with regards to PAR P1 and PAR P2. (Author: R. Sandford)

Revision 38 - Incorporated PCR 02056788 to improve the process of determining Protection Action Recommendations with respect to dose assessment and inserted a page break between Step 2.B and 2.C such that ALL of step 2.C is on one page. (Author: R. Sandford)

Revision 37 - Incorporated PCR 1943000 to add information as to the classification of emergency to be transmitted at the start of notification to the State Watch Office. (Author: J. Moody)

Revision 36 - Incorporated PCR 1989087 to revise the PAR flowchart based on NUREG 0654, Supplement 3 and the St Lucie Evacuation Time Estimate Study. (Author: J. Moody)

AND

Incorporated PCR 2005194 to update the Dose Assessment process for determining PARs. (Author: J. Moody)

Revision 35 - Incorporated PCR 1921686 to make corrections pertaining to SNF Form and section references. (Author: J. Moody)

Revision	Approved By	Approval Date
0	R. G. West	05/03/00
40	D. Taylor	02/17/16

UNIT #	
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Implementation Date 2/18/16
Initials

TABLE OF CONTENTS

1.0 PURPOSE

1.1 Discussion

1. This procedure provides information and instructions for undertaking notifications of the State Watch Office (SWO) and the Nuclear Regulatory Commission (NRC) and for determination of Protective Action Recommendations (PARS).

2. This procedure is for use in the Control Room, Technical Support Center (TSC) and Emergency Operations Facility (EOF).

3. Upon declaration of an emergency classification the Shift Manager (SM) assumes the duties of the Emergency Coordinator (EC). The EC has initial responsibility for off-site notifications and PARs.

4. Once the EOF is operational and proper turnover has been conducted, the Recovery Manager (RM) assumes responsibility for off-site notifications and PARs from the EC.

5. At an Alert or higher level emergency, communications with the NRC transition to an open phone line from the TSC and the EOF (at a Site Area Emergency or higher level emergency).

6. The following table illustrates which facility has a responsibility for Classification, Notification or PARs.

	Control Room (**X** until EC function transfers to the TSC)	**TSC** (**X** when operational)	**EOF** (**X** when operational)
Classifications	**X** transfers →	**X**	
Notifications	**X** transfers →	**X** transfers →	**X**
PARs	**X** transfers →	**X** transfers →	**X**

7. Off-site Notification

   A. Purpose of Off-Site Notifications

   FPL is required to notify off-site agencies in the event of any emergency that could threaten the health and safety of the public. These notifications provide an early warning to agencies responsible for public protection.

1.1 Discussion (continued)

7. (continued)

> **NOTE**
> The State Department of Health (Bureau of Radiation Control) may not have their office staffed on a 24-hour basis. In the event that they do not answer the Hot Ring Down (HRD) telephone, the State Watch Office (SWO) assumes responsibility for notifying their duty officer. However, the EC/RM shall verify that the Bureau of Radiation Control has been notified.

B. Who Shall Be Notified

- State Division of Emergency Management
- State Department of Health (Bureau of Radiation Control)
- St. Lucie County Emergency Operations Center
- Martin County Emergency Operations Center
- NRC

1. State and County Notification

a. State and local agencies are notified by using the Hot Ring Down (HRD) telephone. The HRD rings the State Watch Office (SWO). The SWO puts the other agencies on line and reduces the need for individual calls.

2. NRC Notification

a. The NRC is notified using the Emergency Notification System (ENS) telephone.

b. NRC notifications occur through an open line of communication in the TSC and, when operational, the EOF.

1.1 Discussion (continued)

7. (continued)

C. Emergency Follow-up Information Requests from State and local agencies.

1. Incoming calls should come via the SWO over the HRD phone. If the HRD is inoperable, the SWO may use commercial telephone or EMNET (emergency satellite phone). If an off-site authority contacts the plant without going through the SWO, request that they contact the SWO. SWO shall verify that the agency calling is a risk county or the Department of Health (DOH) and shall notify other county and state agencies of the updated information, thus reducing the number of calls that may be directed to the plant.

2. If the State or one of the Counties provides either the TSC or EOF with new or pertinent information, Then bring that information to the attention of the EC or EC Assistant / Logkeeper in the TSC or the RM or the RM OPS Advisor / Logkeeper in the EOF.

8. Protective Action Recommendations

A. Protective actions for the general public are ordinarily NOT required prior to declaration of a General Emergency. It is possible however, that due to unusually stable and constant meteorological conditions, protective actions could be recommended at a Site Area Emergency based on projected doses. This is the exception rather than the rule.

Protective actions for the general public are required to be recommended if a General Emergency is declared. Initial Protective Action Recommendations (PARs) are normally based on plant conditions. This would NOT be true if the General Emergency was declared based on off-site dose (either measured or projected) or a Security Emergency (per the Security Plan). The predetermined minimum PARs (based on plant conditions) are as given below.

1.1 Discussion (continued)

8. (continued)

B. General Emergency - Minimum PARs

1. In any case where a GENERAL EMERGENCY has been declared, the minimum PAR shall be:

Evacuate all people within a 2-mile radius and out to 5 miles in the sectors affected. The sectors affected are at least three, the downwind sector plus the two adjacent sectors. Monitor and prepare all people in the remaining sectors from 2 to 5 miles and all sectors from 5 to 10 miles from the plant.

2. If a GENERAL EMERGENCY has been declared due to actual or projected severe core damage, the minimum PAR shall be:

Evacuate all people within a 2-mile radius from the plant and out to 5 miles in the sectors affected. Shelter all people in the affected sectors from 5 to 10 miles from the plant. Monitor and prepare all remaining sectors.

3. If a GENERAL EMERGENCY has been declared due to loss of physical control of the plant to intruders, including the Control Room or any other area(s) vital to the operation of the reactor system (as defined in the Security Plan), the minimum PAR shall be:

Shelter all people within a 2-mile radius from the plant and out to 5 miles in the sectors affected. Monitor and prepare all people in the remaining sectors from 2 to 5 miles and all sectors from 5 to 10 miles from the plant.

C. Once a release of radioactive material occurs, dose assessment should be utilized when evaluating PARs. The final determination of the PAR should consider all available information including off-site dose projections, plant conditions and field monitoring data. The most conservative recommendation shall be made.

D. If it is anticipated that a PAR threshold will be exceeded, DO NOT wait until the threshold is exceeded to make that PAR.

E. ¶12 Conditions (plant information, dose projections and field monitoring results) are to be continually assessed and PARs expanded, as necessary, to ensure that adequate (most conservative) PARs are issued.

1.1 Discussion (continued)

8. (continued)

F. ¶12 Previously issued PARs, unless found to be less conservative, are to remain in effect until the threat is fully under control and the event is being de-escalated.

G. ¶12 Only State and County officials can implement, change and / or terminate off-site protective actions.

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, Operating Experience, License Renewal, etc. and shall NOT be revised without the required Focus review and appropriate 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.
>
> Ψ Indicates a step that requires a sign off on an attachment.

2.1 References

1. St. Lucie Plant Updated Final Safety Analysis Report (UFSAR) Unit 1 and Unit 2
2. St. Lucie Plant Technical Specifications Unit 1 and Unit 2
3. §1 St. Lucie Plant Radiological Emergency Plan (E-Plan)
4. E-Plan Implementing Procedures (EPIP 00 – 13)
5. St. Lucie Plant Emergency Response Directory (ERD)
6. RM-AA-100-1000, Processing Quality Assurance Records
7. §2 NRC Bulletin 2005-02, Emergency Preparedness and Response Actions for Security – Based Events
8. ¶22 NRC Regulatory Issue Summary 2007-02: Clarification of NRC Guidance for Emergency Notifications During Quickly Changing Events.
9. NUREG 0654, Supplement III, Criteria for Preparation and Evacuation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, Guidance for Protective Action Strategies.

2.2 Records Required

1. All PAR worksheets and notifications forms (all attachments) shall be maintained in plant files in accordance with RM-AA-100-1000, Processing Quality Assurance Records.

2.3 Commitment Documents

1. ¶1 PMAI PM96-04-165, "ITR 96-006" (Unusual Event Declared Due to Dropped Rod)
2. ¶2 PMAI PM96-09-185, Condition Report CR-96-1750 (Off-site Notification Using Commercial Phone)
3. ¶3 NRC Inspection Report 91-01, Closure of IFIs 89-31-03 and 89-31-01
4. ¶6 PMAI PM96-05-233 (Off-site Notification Process)
5. ¶7 PMAI PM99-09-016 (PARs Based on FMT Data, Completion of NRC Notification Form)
6. ¶8 NUREG-1022, Event Reporting Guidelines 10 CFR 50.72 and 50.73, Section 4.2.4, ENS Event Notification Worksheet (NRC Form 361).
7. ¶9 Condition Reports CR-01-0726 and CR-01-0742 (NOUEs Associated with SDC During SL1-17 Outage)
8. ¶10 Condition Report CR-01-0389 (Alternate Met Data Source)
9. ¶11 Condition Report AR 1835816 (Role of Shift Communicator)
10. ¶12 Condition Report CR-03-2568 (Response to RIS 2003-12 Regarding PARs)
11. ¶13 Condition Report CR-04-0414 (Caution Regarding Determination of Affected Sectors)
12. ¶14 Condition Report CR-04-2700 (Criteria to Determine Core Status)
13. ¶15 Condition Report 2005-23499-CR (E-Plan Activation due to Hurricane Katrina)
14. ¶16 Condition Report 2005-27401-CR (NRC Accelerated Notification error)
15. ¶17 Condition Report AR 1872054 (Commitment Supplement – ORO Notifications)
16. ¶18 Condition Report 2005-26932-CR (SNF Process Improvement Opportunity regarding KI)

2.3 Commitment Documents (continued)

17.¶20 Condition Report 2006-3529-CR, 2006-4536-CR, 2006-4609-CR (60 Minute Update Notifications)

18.¶21 Condition Report 2006-22507-CR, (PARs in Response to Wind Shift)

3.0 RESPONSIBILITIES

3.1 Emergency Coordinator – Responsible for classifications, notifications and PARs.

3.2 Recovery Manager – Responsible for notifications and PARs.

3.3 TSC EC Assistant / Logkeeper – Prepares notification forms (Attachment 1, Florida Nuclear Plant Emergency Notification Form, and if necessary, Attachment 3, NRC Reactor Plant Event Notification Worksheets) for EC approval when the TSC is operational.

3.4 EOF RM OPS Advisor / Logkeeper – Prepares notification forms (Attachment 1 and if necessary, Attachment 3) for RM approval when the EOF is operational.

3.5 TSC HRD Communicator – Assists the TSC EC Assistant / Logkeeper or TSC OPS Coordinator with notification form preparation and makes calls to complete notifications to the SWO.

3.6 EOF HRD Communicator – Assists the EOF RM OPS Advisor with form preparation and makes calls to complete notifications to the SWO.

3.7 TSC Chemistry Supervisor (in his absence, TSC Dose Assessor) – Assists the EC with radiological dose assessment data and PARS.

3.8 EOF HP Manager (in his absence, EOF Dose Assessor) – Assists the RM with radiological dose assessment data and PARS.

3.9 TSC Supervisor – Oversees communications performed by the TSC Communicators (HRD, ENS, Health Physics Network (HPN), Sound-Powered Phonetalker, EOF and Field Monitoring Team).

3.10 EOF Nuclear Licensing Manager – Oversees EOF communications performed by the EOF Communicators (HRD, ENS, HPN and TSC).

3.11 Information Services – Maintains user copies, in the Unit 1 and Unit 2 Control Rooms, of the following checklist and supporting attachments for making notifications and developing Protective Action Recommendations:

- Appendix A, Notifications from the Affected Control Room
- Attachment 1 – Florida Nuclear Plant Emergency Notification Form - **Use Form: EPIP-08-F01**
- Attachment 1A – Directions for Completing the Florida Nuclear Plant Emergency Notification Form
- Attachment 2 – Determination of Protective Action Recommendations (PARs) **- Use Form: EPIP-08-F02**
- Attachment 3 – NRC Reactor Plant Event Notification Worksheet - **Use Form: EPIP-08-F03**
- Attachment 3A – Directions for Completing the NRC Reactor Plant Event Notification Worksheet
- Attachment 4 – Additional State Notification Information

3.12 ¶[11] Shift Communicator – Assists the Shift Manager/Emergency Coordinator in making emergency off-site notifications and performing other activities, as directed.

4.0 DEFINITIONS

4.1 **Airborne Threat** - There are a number of synonyms, such as Airborne Attack Threat, Airliner Attack Threat, Airliner Attack and Aircraft Threat; a threat posed by an aircraft, particularly an "airliner" or large aircraft with a potential for causing significant damage to the plant.

4.2 **Conservative** – Means more extensive or comprehensive action under a given set of circumstances to provide a greater measure of safety. For example, evacuation is more conservative than sheltering.

4.3 **Emergency** – Any off-normal event or condition which is classified into one of the four emergency classes (Unusual Event, Alert, Site Area Emergency, or General Emergency) by the SM in accordance with EPIP-01, Classification of Emergencies.

4.4 **Emergency Coordinator (EC)** – The title initially assumed by the SM, until relieved by plant management through proper turnover, in the event of plant conditions that trigger implementation of the Emergency Plan. The EC is responsible for notifying off-site authorities, emergency responders both inside and outside the company and has full authority and responsibility for on-site emergency response actions. The EC is also responsible for Protective Action Recommendations during the initial stages of an emergency.

4.5 **Florida Nuclear Plant Emergency Notification Form** – A predetermined format used by nuclear power plants throughout the State for notification and local authorities.

4.6 **Monitor and Prepare** - The instruction to monitor and prepare is intended to engage the population within the plume exposure pathway emergency planning zone, inform them of the emergency, and advise them that they should monitor the situation and prepare for the possibility of evacuation, SIP, or other protective actions. If an evacuation is underway, officials should ask members of the public who are not directed to evacuate to remain off the roadways to allow the evacuation to proceed.

4.7 **Operational** (status for an emergency facility) – The mandatory minimum staff is present and the facility has taken responsibility for its procedurally assigned functions.

4.8 **Protective Action Recommendations** (PARs) – Recommendations, for action instructions to protect the public, made by the Emergency Coordinator or Recovery Manager to State and County officials. FPL may recommend Monitor and Prepare, Sheltering or Evacuation.

4.9 **Recovery Manager** (RM) – A designated company officer or senior manager, who will have responsibility for the direction and control of the EOF. He / she has the authority to establish policy and to expend funds necessary to cope with emergency situations that trigger the implementation of the Emergency Plan.

4.10 Release

1. If, as a result of **accident conditions**, ANY of the following is true:

   - A rise of (approximately) 10 times or one decade above pre-transient values is seen on one of the following monitors used for accident assessment:
     - A/B Main Steam Line: RE 26-62/63 (RIM 26-71/72) - ONLY if a steam release is in progress
     - A/B ECCS: RSC 26-2/3 (RS 26-69/70)
     - Plant Vent: RSC 26-1 (RS 26-90)
     - Fuel Building: RSC 26-4 (RS 26-12)
   - Health Physics detects unplanned elevated airborne radiation levels outside of plant buildings due to accident conditions
   - A Steam Generator, with primary-to-secondary leakage, due to a tube leak or rupture, is vented to atmosphere
   - Operating the C AFW pump from an affected Steam Generator

   Then a RELEASE is in progress.

4.11 Shift Communicator – A specific shiftly designated individual trained and qualified to assist the Shift Manager/Emergency Coordinator in the control room in making emergency off-site notifications, and performing other activities as directed.

4.12 State Notification Form (SNF) – Less formal, more concise expression used in lieu of Florida Nuclear plant Emergency Notification Form.

5.0 INSTRUCTIONS

5.1 State and County Notification

1. Time Limits

> **NOTE**
> Following an accelerated notification to the NRC due to entry into a Security Event, notification of the off-site agencies may be required if a change in emergency classification did not occur.

A. Notification shall be initiated within 15 minutes of any of the following:

1. Recognition of entry into the Emergency Plan.
2. Escalation in Emergency Class.
3. De-escalation of the Emergency Class.
4. Protective Action Recommendation.
5. Change in Protective Action Recommendation.

> **CAUTION**
> A regulatory-based 15 minute notification takes precedence and is to supersede a 60 minute update notification.

B. ¶20 Update notifications should occur about every 60 minutes and should be initiated for any of the following:

1. At an Alert or higher Emergency Class, the time of the last update (unless a different frequency has been agreed to by the off-site agencies as during a hurricane).
2. A radiological release has been initiated.
3. A radiological release has been terminated.
4. A significant change in plant conditions has occurred (e.g., loss or restoration of off-site power or major plant equipment).
5. Termination of the emergency.

5.1 State and County Notification (continued)

2. Florida Nuclear Plant Emergency Notification Form

CAUTION

Notifications require the use of a form similar to Attachment 1, Florida Nuclear Plant Emergency Notification Form.

A. Notifications with 15 minute or 60 minute time limit shall be made using a form similar to Attachment 1, Florida Nuclear Plant Emergency Notification Form.

1. In the Control Room, lines 1 through 11 of the Florida Nuclear Plant Emergency Notification Form shall be completed and the form shall be approved by the Emergency Coordinator prior to transmittal.

2. In the Technical Support Center, lines 1 through 15 of the Florida Nuclear Plant Emergency Notification Form shall be completed and the form shall be approved by the Emergency Coordinator prior to transmittal.

3. In the Emergency Operations Facility, lines 1 through 15 of the Florida Nuclear Plant Emergency Notification Form shall be completed and the form shall be approved by the Recovery Manager prior to transmittal.

5.1 State and County Notification (continued)

3. Extraordinary Circumstances

NOTE

- The NRC expects that once a condition is recognized as having the potential for entry into the Emergency Plan, the Shift Manager/EC has 15 minutes to assess and validate the condition and to determine at what level of emergency the condition places the plant. Off-site agencies are to be provided timely and accurate information.
- If the Emergency Coordinator (EC) declares an emergency, the State MUST be notified within 15 minutes even if further analysis determines the emergency did not exist. Clarification of this situation may be provided as part of the notification.

A. Escalation of the Emergency Class due to rapidly degrading conditions **prior** to contacting State Watch Office:

1. ¶22 Complete a new notification form for the upgraded classification and transmit it to the State Watch Office within 15 minutes of the lesser emergency declaration. Information regarding the earlier classification (for which notification did not occur due to changing conditions) should be included in Section 7, Additional Information or Update on the State Notification Form.

OR

If the new State Notification Form cannot be completed and the notification initiated within 15 minutes of the lesser emergency declaration, Then make the notification for the lesser emergency within 15 minutes of its declaration. In parallel, prepare the State Notification Form for the higher emergency classification and make an additional notification within 15 minutes of its declaration.

5.1 State and County Notification (continued)

4. Concurrent Classification

CAUTION

There can not be two concurrent declared emergency classes under the St. Lucie Plant Radiological Emergency Plan.

A. If one Unit is in a classified event and the same or the other Unit enters into an event where the same or lesser Emergency Class would apply, Then a new classification should NOT be declared. The event should be documented on a SNF as "Additional Information or Update" and issued as soon as practicable, but no later than next required notification to the State and local agencies and the NRC.

B. If one Unit is in a classified event and the other Unit enters into a more severe event in which a higher Emergency Class would apply, Then a new classification shall be declared and promptly, within the regulatory time limits, issued to the State, Counties and the NRC.

5. Abbreviated Notification Due to an Airborne Threat

NOTE

Concurrence for use of an abbreviated notification in response to an on-site airborne threat situation has been obtained from State and county officials.

A. If the site is under a potential or actual airborne threat, Then complete lines 1 through 6 and line 11 of the State Notification Form, obtain approval from the Emergency Coordinator or Recovery Manager and transmit the abbreviated notification to the State Watch Office.

END OF SECTION 5.1

5.2 Nuclear Regulatory Commission (NRC) Notification

1. Time Limits

A. §2 Accelerated Notification – in accordance with NRC Bulletin 2005-02, "The staff finds that a notification time period of approximately 15 minutes from discovery of a security-based event to the NRC notification would allow the NRC to more quickly notify other licensees and Federal Agencies".

> **NOTE**
> (Non-Security Event)
>
> Notification of the NRC is expected immediately after notification of State and local agencies. The one-hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

B. The licensee shall notify the NRC immediately after notification of the appropriate State or local agencies and not later than one hour after the time the licensee declares one of the Emergency Classes (10 CFR 50.72 (a)(3)).

2. Special Instructions

A. Initial notification to the NRC using the Emergency Notification System (ENS) (usually done from the Control Room) should use Attachment 3, NRC Reactor Plant Event Notification Worksheet.

B. At an Alert or higher emergency class, the NRC will want to establish an open line of communication with the Control Room, utilizing an ENS conference bridge tying in the licensee with NRC Headquarters and region personnel. Once the Technical Support Center (TSC) is operational, the Control Room should transfer responsibility for NRC communications to the TSC.

C. The Emergency Operations Facility (EOF) should join the TSC on the ENS conference bridge and take the lead for NRC communications.

D. The TSC and EOF should also utilize the Health Physics Network (HPN) line in a manner similar to the ENS (i.e., establish a conference bridge with the NRC).

E. Both the ENS and HPN Communicators in both facilities should keep logs of information transmitted and received from the NRC in accordance with procedures.

END OF SECTION 5.2

5.3 ¶1 Erroneous Information

1. If erroneous information is transmitted to off-site agencies and the error is discovered prior to event termination, a correction should be provided in an update. The need for and urgency of providing the update is dependent upon the importance of the error.

2. If erroneous information is transmitted to off-site agencies and the error is discovered after event termination, the Licensing Department should be consulted to determine the need and method for contacting the off-site agencies with corrected information.

END OF SECTION 5.3

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 1 of 6)

INITIAL

NOTE

¶9 1. Completion of this checklist requires the following Attachments (all from EPIP-08):

Attachment 1 – Florida Nuclear Plant Emergency Notification Form

Attachment 1A – Directions for Completing the Florida Nuclear Plant Emergency Notification Form

Attachment 2 – Determination of Protective Action Recommendations (PARs)

Attachment 3 – NRC Reactor Plant Event Notification Worksheet

Attachment 3A – Directions for Completing the NRC Reactor Plant Event Notification Worksheet

2. Checklist Part 1 is for State Watch Office notification.
3. Checklist Part 2 is for NRC notification.
4. Completion of this Appendix (A) may be delegated to the Shift Communicator.

CAUTION

- §1 Notification of State and local agencies shall be made as soon as practicable within 15 minutes of declaration of an Emergency Class.
- Notification of State and local agencies (through Line 4 of the State Notification Form) of the level of emergency shall be made within 15 minutes of declaration of an Emergency Classification.
- ¶3 A new Florida Nuclear Plant Emergency Notification Form shall be completed for all updates.

1. If Security Event, Perform the Accelerated NRC Notification or N/A. ______

This is the St. Lucie Plant. My name is _________. My title is _______. The validation code is __________. I am providing notification of a Security Event (brief description) ________________________________
____.
Unit 1 is in Mode _____, _______% Power.
Unit 2 is in Mode _____, _______% Power. Additional information will be provided as soon as practical. ______

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 2 of 6)

INITIAL

2. State Watch Office Notification

A. Prepare the Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1).

1. Airborne Threat – Abbreviated State Notification – Prepare State Notification Form by filling out the following:

- Lines 1 through 6 ______

- Line 11 ______

OR

2. All other Security and Non-Security Events - Prepare the State Notification Form in accordance with Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form. ______

NOTE
1. Primary notification method to the State Watch Office (SWO) is to use the Hot Ring Down (HRD) phone.
2. If the HRD is out-of-service, alternate notification methods are provided in Section D, below.

B. Using the State HOT RING DOWN (HRD) Phone, dial 100. ______

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 3 of 6)

INITIAL

2. (continued)

C. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce:

"This is St. Lucie Nuclear Plant [as applicable (Unit 1, 2)] with an emergency message. **Contact Time is ________________ (official Control Room time)**. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy."

Allow the duty Officer to contact St. Lucie County, Martin County, and the Bureau of Radiation Control prior to transmitting any information. When the parties are on the line, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down.

1. ¶15,17 All four off-site agencies have been notified ______

D. Alternate Notification Methods (in order of priority)

> **NOTE**
> Use of the commercial telephone as an alternate notification method requires callback verification from the State Watch Office. Use of EMNET as an alternate notification method should include a callback verification number if available (e.g., cellular phone).

1. Alternate 1 – Commercial Phone

a. Call the State Watch Office using the number on the phone decal or in the St. Lucie Plant Emergency Response Directory (ERD). Announce:

"This is St. Lucie Nuclear Plant [as applicable (Unit 1 / 2)] with an emergency declaration. **Contact Time is _________________(official Control Room time)**. My callback number is ________."

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 4 of 6)

2. D. 1. (continued) INITIAL

b. Hang up the phone and standby for the callback. When the State Watch Office gives the go-ahead, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down. ______

c. ¶2 Request callback from the State Watch Office to verify that they notified St. Lucie County, Martin County and the Bureau of Radiation Control. ______

d. ¶15, 17 All four off-site agencies have been notified ______

NOTE
Use EMNET only if Alternate 1 – commercial phone is not available.

2. Alternate 2 - EMNET

a. Press the SWP speed dial button. ______

b. When the SWO answers announce "State Watch Office, this is St. Lucie Nuclear Plant [as applicable (Unit 1 / 2)] with an emergency declaration." ______

c. When the State Watch Office acknowledges, announce:

"State Watch Office, this is St. Lucie Nuclear Plant [as applicable (Unit 1 / 2)] declaring a / an (classification), repeat (classification). **Contact Time is ________________(official Control Room time)**. I am standing by to transmit Florida Nuclear Plant Emergency Notification Form information when you are ready to copy."

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 5 of 6)

2. D. 2. c. (continued) INITIAL

When the State Watch Office gives the go-ahead, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an _______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down.

d. Announce "St. Lucie clear" at the end of the conversation. _______

e. ¶15, 17 All four off-site agencies have been notified _______

E. Upon completion of reading the form, enter: _______

- Name of the State Watch Office Duty Officer or the individual that receives the notification.
- Date of the phone call

NOTE
Time of call termination is the start of the one (1) hour update clock.

- Time at the State Watch Office (request it from the Duty Officer)

F. Inform the EC if any off-site agencies were NOT on the call. _______

APPENDIX A
NOTIFICATIONS FROM THE AFFECTED CONTROL ROOM
(Page 6 of 6)

INITIAL

CAUTION
Notification of the NRC is expected immediately after notification of State and local agencies. The one hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

3. §1 NRC Notification

NOTE
The NRC will likely request that an open line be maintained.

A. Prepare the NRC Reactor Plant Event Notification Worksheet (Attachment 3) in accordance with Attachment 3A, Directions for Completing the NRC Reactor Plant Event Notification Worksheet. ______

B. EC approval. ______

NOTE
1. Primary notification method to the NRC is to use the Emergency Notification System (ENS) phone.
2. If the ENS is out-of-service an alternate notification method is provided in Section D, below.

C. Transmit the form by dialing one of the numbers shown on the phone or in the Emergency Response Directory (ERD). ______

D. Alternate Notification Method

1. If the ENS is out-of-service, Then use a commercial phone to accomplish the above. ______

END OF APPENDIX A

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 1 of 6)

INITIAL

NOTE
- Checklist Part 1 is for HRD Communications.
- Checklist Part 2 is for ENS Communications.

CAUTION
- §1 Notification of State and local agencies shall be made as soon as practicable within 15 minutes of declaration of an Emergency Class.
- Notification of State and local agencies (through Line 4 of the State Notification Form) of the level of emergency shall be made within 15 minutes of declaration of an Emergency Classification.
- ¶3 A new Florida Nuclear Plant Emergency Notification Form shall be completed for all updates.

1. State Watch Office Notification

   A. EC / TSC EC Assistant / Logkeeper prepare the Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1).

      1. Airborne Threat - Abbreviated State Notification - Prepare State Notification Form by filling out the following:

         - Lines 1 through 6 ______
         - Line 11 ______

      OR

      2. All other Security and Non-Security Events - Prepare the State Notification Form in accordance with Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form. ______

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 2 of 6)

1. (continued) INITIAL

> **NOTE**
> 1. Primary notification method to the State Watch Office (SWO) is to use the Hot Ring Down (HRD) phone.
> 2. If the HRD is out-of-service, alternate notification methods are provided in Section D, below.

B. Using the State HOT RING DOWN (HRD) Phone, dial 100. ______

C. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce:

"This is St. Lucie Nuclear Plant Technical Support Center with an emergency message. **Contact Time is** ________________ . I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy."

Allow the duty Officer to contact St. Lucie County, Martin County, and the Bureau of Radiation Control prior to transmitting any information. When the parties are on the line, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down.

1. ¶15,17 All four off-site agencies have been notified ______

APPENDIX B

NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)

(Page 3 of 6)

1. (continued) INITIAL

D. Alternate Notification Methods (in order of priority)

> **NOTE**
>
> Use of the commercial telephone as an alternate notification method requires callback verification from the State Watch Office. Use of EMNET as an alternate notification method should include a callback verification number if available (e.g., cellular phone).

1. Alternate 1 – Commercial Phone

a. Call the State Watch Office using the number on the phone decal or in the St. Lucie Plant Emergency Response Directory (ERD). Announce:

"This is St. Lucie Nuclear Plant Technical Support Center with an emergency declaration. **Contact Time is** ________________. My callback number is ________."

b. Hang up the phone and standby for the callback. When the State Watch Office gives the go-ahead, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down. ______

c. ¶2 Request callback from the State Watch Office to verify that they notified St. Lucie County, Martin County and the Bureau of Radiation Control. ______

d. ¶15, 17 All four off-site agencies have been notified ______

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 4 of 6)

1. D. (continued) INITIAL

2. Alternate 2 - EMNET

> **NOTE**
> - Use EMNET only if Alternate 1 – commercial phone is not available.
> - EMNET may not be available in a Loss of Off-site Power (LOOP) or Station Blackout (SBO) condition.

a. Press the SWO speed dial button. ______

b. When the SWO answers - announce "State Watch Office, this is St. Lucie Nuclear Plant Technical Support Center with an emergency declaration." ______

c. When the State Watch Office acknowledges, announce:

"State Watch Office, this is St. Lucie Nuclear Plant Technical Support Center declaring a / an (classification), repeat (classification). **Contact Time is** ________________. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy."

When the State Watch Office gives the go-ahead, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down.

d. Announce "St. Lucie clear" at the end of the conversation. ______

APPENDIX B

NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)

(Page 5 of 6)

1. **D.** **2.** (continued) INITIAL

e. ¶15, 17 All four off-site agencies have been notified ______

NOTE

1. Primary notification method to the NRC is to use the Emergency Notification System (ENS) phone.
2. If the ENS is out-of-service, an alternate notification method is provided in Section B, below.

CAUTION

Notification of the NRC is expected immediately after notification of State and local agencies. The one-hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

2. §1 NRC Notification

A. Choose and complete the appropriate steps, below:

1. Perform one of the following:

a. Security Event - Accelerated NRC Notification - as follows:

"This is the St. Lucie Plant. My name and title are ______________________. I am providing initial notification of a Security Event ________ (brief description) ___________. The status of the plant is: Unit 1 is __________ Unit 2 is ___________. Additional information will be provided as practical". ______

APPENDIX B
NOTIFICATIONS FROM THE TECHNICAL SUPPORT CENTER (TSC)
(Page 6 of 6)

2. **A.** **1.** (continued) INITIAL

OR

b. ¶17 Non-Security Event - If the NRC Reactor Plant Event Notification Worksheet (Attachment 3) has NOT previously been transmitted from the Control Room, Then request that the TSC EC Assistant / Logkeeper prepare the form. ______

2. Verify EC approval. ______

3. Transmit the form by dialing one of the numbers shown on the phone or in the Emergency Response Directory (ERD), then GO TO the next step to establish an open line of communication with the NRC. ______

OR

4. ¶17 If the NRC Reactor Plant Event Notification Worksheet (Attachment 3) has previously been transmitted by the Control Room, Then initiate an open line of communication with the NRC by dialing one of the numbers shown on the phone or in the ERD and request to be placed on the Conference Bridge with the NRC. ______

5. As requested, provide information to the NRC. ______

B. Alternate Notification Method

1. If the ENS is out-of-service, Then use a commercial phone to accomplish the above. ______

END OF APPENDIX B

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 1 of 6)

INITIAL

NOTE

- Checklist Part 1 is for HRD Communications.
- Checklist Part 2 is for ENS Communications.

CAUTION

- §1 Notification of State and local agencies shall be made as soon as practicable within 15 minutes of declaration of Emergency Class or change in Protective Action Recommendation (PAR).
- Notification of State and local agencies (through Line 4 of the State Notification Form) of the level of emergency shall be made within 15 minutes of declaration of an Emergency Classification.
- ¶3 A new Florida Nuclear Plant Emergency Notification Form shall be completed for all updates.

1. State Watch Office Notification

   A. RM / RM OPS Advisor prepare the Florida Nuclear Plant Emergency Notification Form (form similar to Attachment 1).

      1. Airborne Threat - Abbreviated State Notification - Prepare State Notification Form by filling out the following:

      - Lines 1 through 6 ______
      - Line 11 ______

      OR

      2. All other Security and Non-Security Events - Prepare the State Notification Form in accordance with Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form. ______

APPENDIX C

NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)

(Page 2 of 6)

1. (continued) INITIAL

NOTE

1. Primary notification method to the State Watch Office (SWO) is to use the Hot Ring Down (HRD) phone.
2. If the HRD is out-of-service, alternate notification methods are provided in Section D, below.
3. State and County representatives means Florida Division of Emergency Management (DEM), Florida Department of Health (DOH), St. Lucie County Department of Public Safety (DPS) and Martin County Department of Emergency Services (DES).
4. Notification forms means the Florida Nuclear Plant Emergency Notification Form.

B. Using the State Hot Ring Down (HRD) phone, dial 100. _______

C. Hold down the button on the handset while talking. This must be done each time you talk. Release the button in order to listen. When the State Duty Officer answers, announce:

"This is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency message. **Contact Time is ____________.** I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy."

Allow the duty Officer to contact St. Lucie County, Martin County, and the Bureau of Radiation Control prior to transmitting any information. When the parties are on the line, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down.

1. ¶15, 17 All four off-site agencies have been notified _______

APPENDIX C

NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)

(Page 3 of 6)

1. (continued) INITIAL

D. Alternate Notification Methods (in order of priority)

> **NOTE**
>
> Use of the commercial telephone as an alternate notification method requires callback verification from the State Watch Office. Use of EMNET as an alternate notification method should include a callback verification number if available (e.g., cellular phone).

1. Alternate 1 – Commercial Phone

a. Call the State Watch Office using the number on the phone decal or in the St. Lucie Plant Emergency Response Directory (ERD). Announce:

"This is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency declaration. **Contact Time is** ________________. My callback number is ________."

b. Hang up the phone and standby for the callback. When the State Watch Office gives the go-ahead, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down. ______

c. ¶2 Request callback from the State Watch Office to verify that they notified St. Lucie County, Martin County and the Bureau of Radiation Control. ______

d. ¶15, 17 All four off-site agencies have been notified ______

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 4 of 6)

1. **D.** (continued) INITIAL

2. Alternate 2 - EMNET

> **NOTE**
> Use EMNET only if Alternate 1 – commercial phone is not available.

a. Press the SWO speed dial. ______

b. When the SWO answers - announce "State Watch Office, this is St. Lucie Nuclear Plant Emergency Operations Facility with an emergency declaration." Then release the "push-to-talk" button in order to listen. ______

c. When the State Watch Office acknowledges, announce:

"State Watch Office, this is St. Lucie Nuclear Plant Emergency Operations Facility declaring a / an (classification), repeat (classification). **Contact Time is** ________________. I am standing by to transmit the Florida Nuclear Plant Emergency Notification Form when you are ready to copy."

When the State Watch Office gives the go-ahead, announce:

"St Lucie Nuclear Plant [as applicable (Unit 1, 2)] has declared a/an ______________ (Emergency Classification Level)."

Begin transmitting the Florida Nuclear Plant Emergency Notification Form slowly (in three word intervals) and deliberately, providing time for the information to be written down.

d. Announce "St. Lucie clear" at the end of the conversation. ______

e. ¶15, 17 All four off-site agencies have been notified ______

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 5 of 6)

INITIAL

CAUTION
Notification of the NRC is expected immediately after notification of State and local agencies. The one-hour time limit in 10 CFR 50.72 (a)(3) is to ensure timely NRC notification in cases where notification of State and local agencies is delayed or prolonged.

2. §1 NRC Notification

NOTE
1. Primary notification method to the NRC is to use the Emergency Notification System (ENS) phone.
2. If the ENS is out-of-service, an alternate notification method is provided in Section B, below.

A. Choose and complete the appropriate steps, below:

1. Perform one of the following:

a. Security Event - Accelerated NRC Notification - as follows:

"This is the St. Lucie Plant. My name and title are ____________________. I am providing initial notification of a Security Event ________ (brief description) __________. The status of the plant is: Unit 1 is __________ Unit 2 is ___________. Additional information will be provided as practical". ______

OR

b. ¶17 Non-Security Event - If the NRC Reactor Plant Event Notification Worksheet (Attachment 3) has NOT previously been transmitted from the Control Room or Technical Support Center, Then request that the RM OPS Advisor prepare the form. ______

2. Verify RM approval. ______

APPENDIX C
NOTIFICATIONS FROM THE EMERGENCY OPERATIONS FACILITY (EOF)
(Page 6 of 6)

2. **A.** (continued) INITIAL

3. Transmit the form by dialing one of the numbers shown on the phone or in the Emergency Response Directory (ERD), then GO TO the next step to establish an open line of communication with the NRC. ______

OR

4. ¶17 If the NRC Reactor Plant Event Notification Worksheet (Attachment 3) has previously been transmitted by either the Control Room or the TSC, Then initiate an open line of communication with the NRC by dialing one of the numbers shown on the phone or in the ERD and request to be placed on the Conference Bridge with the NRC and the St. Lucie TSC. ______

5. Take the lead in providing information to the NRC. ______

B. Alternate Notification Method

1. If the ENS is out-of-service, Then use a commercial phone to accomplish the above. ______

END OF APPENDIX C

ATTACHMENT 1
FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 1 of 1)

On-line Verification: ☐ SWP/DEM ☐ DOH/BRC ☐ St. Lucie County/Ft. Pierce ☐ Martin County

1.* A. ☐ THIS IS A DRILL B. ☐ THIS IS AN EMERGENCY

2. A. Date: __/__/__ B.* Contact Time: ______ C. Reported by: (Name) ____________
   D. Message Number: ______ E. Reported from: ☐ Control Room ☐ TSC ☐ EOF
   F. ☐ Initial / New Classification OR ☐ Update Notification

3.* Site: A. ☐ Crystal River Unit 3 B. ☐ St. Lucie Unit 1 C. ☐ St. Lucie Unit 2
   D. ☐ Turkey Point Unit 3 E. ☐ Turkey Point Unit 4

4.* Emergency Classification: A. ☐ Notification of Unusual Event B. ☐ Alert
   C. ☐ Site Area Emergency D. ☐ General Emergency

5.* A. ☐ Emergency Declaration: B. ☐ Emergency Termination: Date: __/__/__ Time: ______

6.* Reason for Emergency Declaration: A. ☐ EAL Number ___/___/___ (Alpha / Alpha / Numeric) OR B. ☐ Description:
____

7. Additional Information or Update: A. ☐ None OR B. ☐ Description:
____
____

8.* Weather Data: A. Wind direction from ______ degrees B. Downwind Sectors Affected: ____________

9.* Release Status: A. ☐ None (Go to Item 11) B. ☐ In progress C. ☐ Has occurred, but stopped (Go to Item 11)

10. Release Significance Category at the Site Boundary:
   A. ☐ Under evaluation B. ☐ Release within normal operating limits
   C. ☐ Non-Significant (fraction of protective action guide range) D. ☐ Protective action guide range
   E. ☐ Liquid release (no actions required)

11.* UTILITY PROTECTIVE ACTION RECOMMENDATIONS FOR THE PUBLIC:
   A. ☐ No utility recommended actions at this time.
   B. ☐ The utility recommends the following protective actions:

				Evacuate Sectors	Shelter Sectors	Monitor & Prepare Sectors
Evacuate Zones:	______	OR	0 – 2	______	______	______
Shelter Zones:	______		2 – 5	______	______	______
			5 – 10	______	______	______

AND consider issuance of potassium iodide (KI)*

If form is completed in the Control Room, go to Item 15. If completed in the TSC or EOF, continue with item 12.

12. Plant Conditions: A. Reactor Shutdown? ☐ Yes ☐ No B. Core Adequately Cooled? ☐ Yes ☐ No
   C. Containment Intact? ☐ Yes ☐ No D. Core Condition: ☐ Stable ☐ Degrading

13. Weather Data: A. * Wind Speed ______ mph B. Stability Class ______

14. Additional Release Information: A. ☐ Not Applicable (Go to Item 15)

Distance	*Projected Thyroid Dose (CDE) for ___ hour(s)*	*Projected Total Dose (TEDE) for ___ hour(s)*
1 Mile (Site Boundary)	B. ____________ mrem	C. ____________ mrem
2 Miles	D. ____________ mrem	E. ____________ mrem
5 Miles	F. ____________ mrem	G. ____________ mrem
10 Miles	H. ____________ mrem	I. ____________ mrem

15. (Do Not Read) EC or RM Approval Signature: ____________ Date: __/__/__ Time: ______

   Message Received By: Name: ____________ Date: __/__/__ Time: ______

* Items are evaluated for NRC Performance Indicators (PI's)

EPIP-08-F01 Form Revision: 4

END OF ATTACHMENT 1

¶6 ATTACHMENT 1A

DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

(Page 1 of 6)

ITEM ENTRY

> **NOTE**
> On-line verification occurs when notification of the State Watch Office (SWO) is initiated.

On-line Verification - **Check the appropriate boxes** as the State Watch Office Florida Division of Emergency Management (DEM) requests that the Department of Health Bureau of Radiation Control, St. Lucie County Department of Public Safety and the Martin County Department of Emergency Management get on the line, prior to initiating the notification. **All four agencies must be notified (includes Florida DEM)** through the SWO or alternate means.

1. Check appropriate box for drill or actual emergency as the case may be. During exercises, drills, or tests, each message shall be checked **THIS IS A DRILL.**

2A. Date

2B. Contact Time

2C. Reported by (Name)

2D. Enter the message number beginning with #1 and following sequentially throughout the event.

2E. Reported from

2F. Initial/New Classification or Update Notification

3. Site
Check the affected site / unit.

4. Emergency Classification

5. Emergency Declaration or Emergency Termination
Enter the **date** and **time** when the current emergency classification was (A) declared OR (B) terminated.

6. Reason for Emergency Declaration

    A. Enter the Emergency Action Level (EAL) number as given in the EAL Tables.

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 2 of 6)

7. Additional Information or Update

Check "A" (None) if there is no information or "B" to provide additional information/clarification or reason for the update. This will be provided by SM/EC.

Examples of additional information include:

- Change in PARs
- Change in radiological conditions
- Change in equipment status
- Injuries
- Condition affects both units
- Condition that would have resulted in the declaration of the same or lesser emergency classification

8. Weather Data

NOTE
10 meter data should be used.

A. ¶10 Wind direction can be obtained from ERDADS (DCS).

1. Alternate sources of wind direction data:

a. Met Tower Indicator Panel in the Unit 1 Control Room

b. Contact NOAA / NWS Melbourne Office (321)-254-6083 (24-hr coverage - Ask to speak with a forecaster).

c. Method 3, Attachment 7, Meteorological Data in EPIP-09, Off-Site Dose Calculations.

B. If the wind direction is **greater than 360°** the wind direction is determined by **subtracting 360°** from the indicated number.

- Wind direction should be rounded to the nearest whole number.
- If the wind direction fraction is exactly one half (e.g., 123.5 degrees), Then round up UNLESS rounding down would cause the addition of another sector (e.g., round 123.5 degrees down to 123 degrees to pick up the additional sector).

ATTACHMENT 1A
DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM
(Page 3 of 6)

8. (continued)

C. If the wind is located on the edge of a sector (i.e., 11°, 33°, etc.) an additional (fourth) sector should be added.

D. Enter the wind direction (wind from) in degrees in item "A."

E. Enter the downwind sectors in item "B" using the following chart

Wind From	Sectors Affected	Wind From	Sectors Affected	Wind From	Sectors Affected
348-11	HJK	123-146	PQR	236-258	CDE
11-33	JKL	146-168	QRA	258-281	DEF
33-56	KLM	168-191	RAB	281-303	EFG
56-78	LMN	191-213	ABC	303-326	FGH
78-101	MNP	213-236	BCD	326-348	GHJ
101-123	NPQ	There is <u>no</u> "O" sector		There is <u>no</u> "I" sector	

9. <u>Release Status</u>

A. If there are no indications of a release of radioactive material, check box "A" and go to item 11.

B. If a release is in progress, check box "B" and go to item 10.

C. If a release has occurred, but stopped, check box "C" and go to item 11.

10. <u>Release Significance Category</u>
Check the appropriate box as given by Dose Assessment or EC

ATTACHMENT 1A

DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

(Page 4 of 6)

11. Utility Recommended Protective Actions

A. If there are no Protective Action Recommendations (PARs), check Box “A,” only.

B. If PARs are necessary, check Box "B". Use the "sector" format and determine appropriate PARs using the guidance in Attachment 2 to this procedure. Copy the PARs into item 11 “B.” Indicate PARs using only the words NONE, ALL, ALL REMAINING or by listing the letters of the sectors affected.

AND

¶18 (If PARs are issued, Then read this statement) “Consider issuance of Potassium Iodide (KI)”.

NOTE
If completing the State Notification Form in the Control Room, continue to Step 15.

12. Plant Conditions

NOTE
¶14 Safety Function Status is available from the affected Control Room.

Answer these questions by checking the appropriate box.

A. Is the reactor shut down?

- If the Reactivity Control Safety Function is being met, Then check the YES box.

B. Is the core adequately cooled?

- If the Core Heat Removal AND RCS Heat Removal Safety Functions are being met, Then check the YES box.

ATTACHMENT 1A

DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

(Page 5 of 6)

12. (continued)

C. Is the containment intact?

- If the Containment Isolation Safety Function is being met, Then check the YES box.

D. Is the condition of the core stable or degrading?

- If "B" above is YES, Then check the STABLE box.

OR

- If "B" above is NO, Then check the DEGRADING box.

13. Weather Data

NOTE

10 meter data should be used.

C. Stability Class - Enter the stability class as determined by using the figure below.

Delta-T	Stability Class
Less than or equal to -1.7	A
-1.6 to -1.5	B
-1.4	C
-1.3 to -0.5	D
-0.4 to +1.4	E
+1.5 to +3.6	F
Greater than +3.6	G

14. Additional Release Information

- Check box "A" Not Applicable (N/A) if no release is occurring and/or if dose information is not available.
- Otherwise, enter the information provided by dose assessor

ATTACHMENT 1A

DIRECTIONS FOR COMPLETING THE FLORIDA NUCLEAR PLANT EMERGENCY NOTIFICATION FORM

(Page 6 of 6)

> **NOTE**
> A peer check helps to avoid missing and/or incorrect information.

15. Message Received By

    Obtain EC or RM approval

    Transmit the form in accordance with EPIP-08, Appendix "A," step 2.B.

END OF ATTACHMENT 1A

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 1 of 8)

1. Guidelines for determination of Protective Action Recommendations (PARs)

A. Dose assessment and PAR development are expected to be made promptly. A goal of 15 minutes from data availability is a reasonable period of time to develop or expand a PAR.

B. Initial notification from the Control Room may utilize PARs based on plant conditions.

C. Once dose assessment begins, PARs should be made utilizing all available data including off-site dose projections, plant conditions and field monitoring data.

> **NOTE**
> - Both plant conditions and off-site dose should be considered for PARs.
> - The most conservative recommendations should be made.

D. ¶12 Conditions (plant information, dose projections and field monitoring results) are to be continually assessed and PARs expanded, as necessary, to ensure that adequate (most conservative) PARs are issued.

E. It should be expected that a change in meteorological and / or radiological conditions will require a change in PARs, particularly if current PARs are based on plant conditions.

1. ¶21 During a continuous wind shift situation:

a. When making a PAR associated with a General Emergency, the affected sectors should be "ALL" in the 2-5 mile distance. [This PAR will be based on Plant Conditions because doses can not be calculated / projected until the wind has stabilized].

F. ¶12 Previously issued PARs are to remain in effect, unless they provide less of a dose savings.

For example:

- Initial - Downwind sectors are K, L, M. Issue PAR for sectors K, L, M.
- Subsequent - Downwind sectors are J, K, L. Issue PAR for sectors J, K, L, M.
- Maintain the original sectors and add the new sector(s).

G. ¶12 Only State and County officials can implement, change and / or terminate off-site protective actions.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 2 of 8)

2. Determination of PARs

Responsibility for Determination of Protective Action Recommendations

Facility	Position(s)	Plant Conditions	Dose
Control Room	Emergency Coordinator	X	X[1]
Technical Support Center	EC Assistant / Logkeeper	X	
	Chemistry Supervisor		X
Emergency Operations Facility	RM OPS Advisor	X	
	HP Manager		X

[1] If a release is in progress, Then the on shift Chemistry Technician will perform dose assessment and provide dose information to the Control Room Emergency Coordinator.

A. Instructions for selecting PARs based on plant conditions

1. Refer to the PARs Based on Plant Conditions flowchart.

2. Begin in the upper left hand corner of the chart by answering the General Emergency (GE) question.

3. Correctly answer the questions until you reach one of the boxes that provides PAR information based on plant conditions.

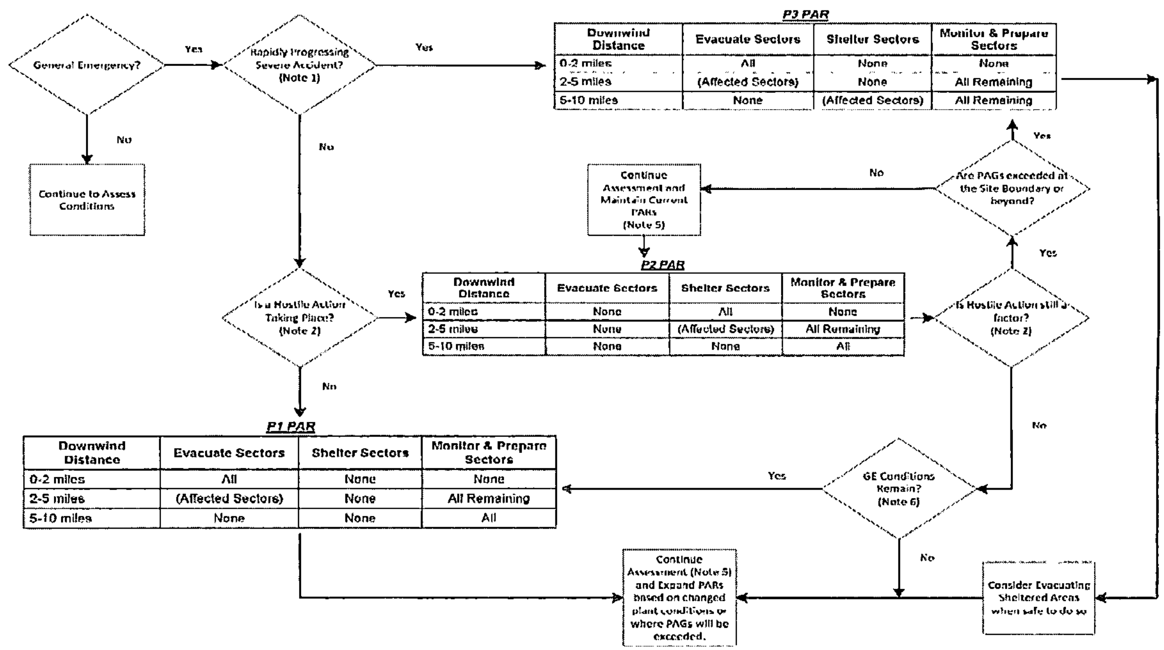
4. Circle the box with the appropriate PARs.

5. If there is no release, Then transfer the PARs to the State Notification Form, Section 11. The sectors affected can be determined by referring to number 8, Weather Data, in Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form.

6. If a release is involved, Then go to Section B, Instructions for PARs Based on Off-site Dose, below.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 3 of 8)

PARs Based on Plant Conditions



P3 PAR

Downwind Distance	Evacuate Sectors	Shelter Sectors	Monitor & Prepare Sectors
0-2 miles	All	None	None
2-5 miles	(Affected Sectors)	None	All Remaining
5-10 miles	None	(Affected Sectors)	All Remaining

P2 PAR

Downwind Distance	Evacuate Sectors	Shelter Sectors	Monitor & Prepare Sectors
0-2 miles	None	All	None
2-5 miles	None	(Affected Sectors)	All Remaining
5-10 miles	None	None	All

P1 PAR

Downwind Distance	Evacuate Sectors	Shelter Sectors	Monitor & Prepare Sectors
0-2 miles	All	None	None
2-5 miles	(Affected Sectors)	None	All Remaining
5-10 miles	None	None	All

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 4 of 8)

PARs Based on Plant Conditions

NOTE:

(1) A Rapidly Progressing Severe Accident involves a containment failure with >20% clad damage or PAG's exceeded at site boundary within 1 hour.

Loss of containment integrity = EALs indicate containment barrier loss. This path is used for scenarios in which containment integrity can be determined as bypassed or immediately lost during a GE with core damage.

20% Clad Damage is identified by ANY the following:
- Pressure ≤ 100 psia and 1250 CET Temp (F)
- Pressure between 100 and 1200 psia and 1550 CET Temp (F)
- Pressure between 1200 and 1650 psia and 1925 CET Tempt (F)

If this scenario cannot be immediately confirmed, assume it is not taking place and answer "no" to this decision block.

(2) Hostile Action: An act toward a Nuclear Power Plant (NPP) or its personnel that includes the use of violent force to destroy equipment, take hostages, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, projectiles, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. Hostile Action should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP.

(3) Shelter in Place (SIP) means that instructions are given to members of the public to remain indoors, turn off heating or air conditioning (as appropriate for the region and season), close windows, monitor communications channels, and prepare to evacuate.

(4) Monitor and Prepare: The instruction to monitor and prepare is intended to engage the population within the plume exposure pathway emergency planning zone, inform them of the emergency, and advise them that they should monitor the situation and prepare for the possibility of evacuation, SIP, or other protective actions. If an evacuation is underway, officials should ask members of the public who are not directed to evacuate to remain off the roadways to allow the evacuation to proceed.

(5) Continue Assessments: Radiological and meteorological assessments should be continued and evacuation considered for any areas where dose projections or field measurements indicate that PAGs may be exceeded. Communications with the public should be maintained while protective actions are in effect. **Additionally, changes in wind direction may indicate that if a release begins, it would affect different downwind sectors. If a licensee believes that containment may fail, it should pursue the expansion of PARs.**

(6) GE Conditions Remain : If the plant has mitigated the conditions that caused the GE declaration (i.e., core cooling is restored), expanding the PAR to evacuate downwind sectors upon completion of the initial staged evacuation may not be necessary. However, if GE emergency action levels are still met, expansion of the PAR to the downwind sectors may be appropriate. If the plant restores core cooling, it must still perform a radiological assessment to identify the extent of contamination, if any. If surveys or dose projections reveal areas under no protective action direction where protective action guidelines (PAGs) could be exceeded, the members of the public in those areas should be evacuated or sheltered, as appropriate.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 5 of 8)

NOTE

- Calculated off-site doses should be compared to field monitoring data, if available, when determining PARs.
- The sectors affected can be determined by referring to number 8, Weather Data, in Attachment 1A, Directions for Completing the Florida Nuclear Plant Emergency Notification Form.

B. Instructions for selecting PARs based on Dose Assessment

1. If a Manual Dose Calculations has been performed, obtain the Dose Calculation Worksheet.

a. On the Dose Calculation Worksheet, locate the calculated dose values listed on line 18 (Total Dose [TEDE]) and line 7 (Projected Thyroid Dose [CDE]).

b. Obtain the PARs Based on Dose Assessment Table

1. Begin at the top of the table; select PARs Based on Dose Assessment as instructed within the table.

2. Circle the selected PARs based on dose assessment.

c. Go to Section 3, Instructions for the PAR Worksheet.

2. If Unified RASCAL Interface (URI) Dose Assessment has been performed for a single pathway, obtain the URI Dose Assessment report. If Unified RASCAL Interface (URI) Dose Assessment has been performed for multiple pathways, obtain the Dose Assessment Summation report.

a. On the report, locate the TEDE (mrem) and CDE Thyroid (mrem) columns.

1. Begin at the top of the table; select PARs Based on Dose Assessment as instructed within the table.

2. Circle the selected PARs based on dose assessment.

b. Go to Section 3, Instructions for the PAR Worksheet.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 6 of 8)

PARs Based on Dose Assessment Table

NOTE

- If TEDE Dose or CDE Thyroid Dose is ≥ PAGs (≥ 1.00E+03 and ≥ 5.00E+03 respectively) at the Site Boundary (1 mile for Manual Dose Calculations), notify the Emergency Coordinator and/or Recovery Manager and verify current Emergency Classification is General Emergency.
- When using URI Dose Assessment report or Dose Assessment Summation report, ensure Affected Sectors includes all sectors indicated on the Evacuation Area map.

Proceed down the page until PARs based on Dose Assessment are selected. Once PARs based on Dose Assessment have been selected, go to step 2.

1. Is all calculated dose < 1.00E+3 TEDE and/or < 5.00 E + 3 CDE (Thyroid)? If so no PARs based on Dose Assessment are required. If not, proceed to step 2.

2. At greater than or equal to 10 miles is calculated dose ≥ 1.00E+03 TEDE and/or ≥ 5.00E+03 CDE (Thyroid)? If so, select the following PARs based on Dose Assessment and go to step 6. Additionally, perform a URI 50 mile assessment. If not, continue to step 3.

Miles	Evacuate	Shelter	Monitor & Prepare
0-2	All	None	None
2-5	(Affected Sectors)	None	All Remaining
5-10	(Affected Sectors)	None	All Remaining
> 10	(Affected Sectors)	None	All Remaining

3. At greater than 2 miles and less than 10 miles is any calculated dose ≥ 1.00E+03 TEDE and/or ≥ 5.00E+03 CDE (Thyroid)? If so, select the following PARs based on Dose Assessment and go to step 6. If not, continue to step 4.

Miles	Evacuate	Shelter	Monitor & Prepare
0-2	All	None	None
2-5	(Affected Sectors)	None	All Remaining
5-10	(Affected Sectors)	None	All Remaining
> 10	None	(Affected Sectors)	All Remaining

4. At less than or equal to 2 miles is any calculated dose ≥ 1.00E+03 TEDE and/or ≥ 5.00E+03 CDE (Thyroid)? If so, select the following PARs based on Dose Assessment and go to step 6. If not, continue to step 5.

Miles	Evacuate	Shelter	Monitor & Prepare
0-2	All	None	None
2-5	(Affected Sectors)	None	All Remaining
5-10	None	(Affected Sectors)	All Remaining
> 10	None	None	None

5. Compare the selected PARs based on Dose Assessment with Plant Condition PARs using the PAR Worksheet.

Once selection of PARs based on dose assessment is complete, dose assessors must continue to assess conditions and update dose assessment if:

- Release Point Information increases by more than 25 percent or additional release pathways are discovered;
- Wind speed decreases to less than one half of previous value;
- Atmospheric stability becomes more stable by more than one class (e.g., change from Stability Class D to F);
- Wind direction changes by more than 22.5 degrees (i.e., plume centerline is more than one sector away from prior location).

ATTACHMENT 2

DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)

(Page 7 of 8)

3. Instructions for the PAR Worksheet

A. Fill-in the time, date and emergency class.

B. In Part A, PAR Comparison determines the basis for the most conservative PAR by comparing the PARs based on plant conditions against those based on off-site dose. It is important to compare PARs at each distance (0-2, 2-5, 5-10) because the basis of the most conservative PAR could be different at different distances.

C. Enter the most conservative PARs into the table in Part B, Protective Actions to be recommended by FPL. Use the word(s) NONE, ALL, ALL REMAINING or list the individual affected sectors by letter.

D. Persons responsible for PAR determination are to sign the completed form. In the:

Control Room (CR) (if PARs are based on other than just plant conditions) - the Emergency Coordinator

Technical Support Center (TSC) - the EC Assistant / Logkeeper (for plant condition PARs) and the Chemistry Supervisor (for dose PARs)

Emergency Operations Facility (EOF) - the RM OPS Advisor (for plant condition PARs) and the HP Manager (for dose PARs)

E. Transfer the PARs to be recommended to the State Notification Form (SNF) in Section 11. The PARs will be approved by the Emergency Coordinator (CR, TSC) or Recovery Manager (EOF) as part of the SNF.

ATTACHMENT 2
DETERMINATION OF PROTECTIVE ACTION RECOMMENDATIONS (PARs)
(Page 8 of 8)

PAR Worksheet

Date:_____/_____/_____ **Emergency Class:** ☐ **SAE** ☐ **GE**

A. PAR Comparison
Compare protective actions based on plant conditions versus dose, determine the more conservative and check the applicable box in the table below. If no PAR is necessary, Then check "none".

Downwind Distance	PAR Basis (Check applicable basis)		
	Plant Conditions	Dose	None
0-2 miles			
2-5 miles			
5-10 miles			
10-TBD miles			

B. Protective Actions to be Recommended by FPL
Complete the table below with the protective actions to be recommended. Use these terms: **NONE, ALL, ALL REMAINING** (or fill in the letters of the sectors affected)

Downwind Distance	Evacuate Sectors	Shelter Sectors	Monitor and Prepare
0-2 miles			
2-5 miles			
5-10 miles			
10-TBD miles*			

*If necessary, add to State Notification Form.

C. Potassium Iodide (KI)
If PARs are issued, Then recommend: Consider issuance of potassium iodide (KI)

D. Sign-off by facility that determined the PARs

Control Room

_______________________ _______
Emergency Coordinator Time

Technical Support Center

_______________________ _______________________ _______
TSC EC Assistant / Logkeeper TSC Chemistry Supervisor Time

Emergency Operations Facility

_______________________ _______________________ _______
EOF RM OPS Advisor EOF HP Manager Time

E. Attach to the State Notification Form

EPIP-08-F02 Form Revision: 5

END OF ATTACHMENT 2

ATTACHMENT 3

¶8 NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET

(Page 1 of 2)

NRC FORM 361
(12-2000)

U.S. NUCLEAR REGULATORY COMMISSION
OPERATIONS CENTER

REACTOR PLANT EVENT NOTIFICATION WORKSHEET

EN #

NRC OPERATION TELEPHONE NUMBER: PRIMARY – 301-816-5100 or 800-532-3469*, BACKUPS – [1st] 301-951-0550 or 800-449-3694*, [2nd] 301-415-0550 and [3rd] 301-415-0553
* Licensees who maintain their own ETS are provided these telephone numbers.

NOTIFICATION TIME	FACILITY OR ORGANIZATION	UNIT	NAME OF CALLER	CALL BACK #
				U1/U2 (700) 821-0015

EVENT TIME & ZONE	EVENT DATE	POWERMODE BEFORE	POWERMODE AFTER

	EVENT CLASSIFICATIONS			1-Hr. Non-Emergency 10 CFR 50.72(b)(1)			(v)(A) Safe S/D Capability	AINA
	GENERAL EMERGENCY	GEN/AAEC		TS Deviation	ADEV		(v)(B) RHR Capability	AINB
	SITE AREA EMERGENCY	SIT/AAEC		4-Hr. Non-Emergency 10 CFR 50.72(b)(2)			(v)(C) Control of Rad Release	AINC
	ALERT	ALE/AAEC		(i) TS Required S/D	ASHU		(v)(D) Accident Mitigation	AIND
	UNUSUAL EVENT	UNU/AAEC		(iv)(A) ECCS Discharge to RCS	ACCS		(xii) Offsite Medical	AMED
	50.72 NON-EMERGENCY	(see next columns)		(iv)(B) RPS Actuation (scram)	ARPS		(xiii) Loss Comm/Asmt/Resp	ACOM
	PHYSICAL SECURITY (73.71)	DDDD		(xi) Offsite Notification	APRE		60-Day Optional 10 CFR 50.73(a)(1)	
	MATERIAL/EXPOSURE	B???		8-Hr. Non-Emergency 10 CFR 50.72(b)(3)			Invalid Specified System Actuation	AINA
	FITNESS FOR DUTY	HFIT		(ii)(A) Degraded Condition	ADEG		Other Unspecified Requirement (Identify)	
	OTHER UNSPECIFIED REQMT.	(see last column)		(ii)(B) Unanalyzed Condition	AUNA			NONR
	INFORMATION ONLY	NNF		(iv)(A) Specified System Actuation	AESF			NONR

DESCRIPTION

Include: Systems affected, actuations and their initiating signals, causes, effect of event on plant, actions taken or planned, etc. (Continue on back)

NOTIFICATIONS	YES	NO	WILL BE			
NRC RESIDENT				ANYTHING UNUSUAL OR NOT UNDERSTOOD?	☐ YES (Explain above)	☐ NO
STATE(s)				DID ALL SYSTEMS FUNCTION AS REQUIRED?	☐ YES	☐ NO (Explain above)
LOCAL						
OTHER GOV AGENCIES				MODE OF OPERATION UNTIL CORRECTED:	ESTIMATED RESTART DATE:	ADDITIONAL INFO ON BACK ☐ YES ☐ NO
MEDIA/PRESS RELEASE						

ATTACHMENT 3
¶8 NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET
(Page 2 of 2)

RADIOLOGICAL RELEASES: CHECK OR FILL IN APPLICABLE ITEMS (specific details/explanations should be covered in event description)

	LIQUID RELEASE		GASEOUS RELEASE		UNPLANNED RELEASE		PLANNED RELEASE		ONGOING		TERMINATED
	MONITORED		UNMONITORED		OFFSITE RELEASE		T.S. EXCEEDED		RM ALARMS		AREAS EVACUATED
	PERSONNEL EXPOSED OR CONTAMINATED				OFFSITE PROTECTIVE ACTIONS RECOMMENDED				*State release path in description		

	Release Rate (Ci/sec)	% T.S. LIMIT	HOO GUIDE	Total Activity (Ci)	% T.S. LIMIT	HOO GUIDE
Noble Gas			0.1 Ci/sec			1000 Ci
Iodine			10 uCi/sec			0.01 Ci
Particulate			1 uCi/sec			1 mCi
Liquid (excluding tritium and dissolved noble gases)			10 uCi/min			0.1 Ci
Liquid (tritium)			0.2 Ci/min			5 Ci
Total Activity						

	PLANT STACK	CONDENSER/AIR EJECTOR	MAIN STEAM LINE	SG BLOWDOWN	OTHER
RAD MONITOR READINGS					
ALARM SETPOINTS					
% T.S. LIMIT (if applicable)					

RCS OR SG TUBE LEAKS: CHECK OR FILL IN APPLICABLE ITEMS: (specific details/explanations should be covered in event description)

LOCATION OF THE LEAK (e.g., SG#, valve, pipe, etc.)				
LEAK RATE	UNITS: gpm/gpd	T.S. LIMITS	SUDDEN OR LONG-TERM DEVELOPMENT	
LEAK START DATE	TIME	COOLANT ACTIVITY AND UNITS:	PRIMARY	SECONDARY
LIST OF SAFETY RELATED EQUIPMENT NOT OPERATIONAL				

EVENT DESCRIPTION (Continued from front)

END OF ATTACHMENT 3

¶7 **ATTACHMENT 3A**

DIRECTIONS FOR COMPLETING THE NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET

(Page 1 of 2)

A. Contact information - to be completed following contact

1. Name of the person contacting the NRC or other designated FPL contact.
2. NRC Contacts Name - will be provided upon contact. Also obtain the event number and notification time as received from the HOO should be recorded on the top of the worksheet.

B. Reactor Plant Event Notification Worksheet, Page 1

> **NOTE**
> The "EN #" is provided by the NRC.

1. Notification Time - enter the time contact is made.
2. Facility or Organization - enter St. Lucie Plant
3. Unit - enter the appropriate unit number: Enter "0" for a classification common to both units.
4. Callers Name - enter the name of the person making the call.
5. Call back # - enter the number of the ENS phone that you are calling from and the commercial phone number at which you can be reached.
6. Event time and Zone - enter the military time, the zone will be "EST" for Eastern Standard Time or "EDT" for Eastern Daylight-savings Time.
7. Event Date - enter the date the event is occurring.
8. Power / Mode Before & Power / Mode After - enter the power in percent and the mode number (1-6) before and after the event.

> **NOTE**
> Abbreviations / acronyms (e.g., UNU / AAEC, SIT / AAEC, etc.) are for NRC use only.

9. Event Classifications - check one of the four blocks for General Emergency, Site Area Emergency, Alert, or Notification of Unusual Event.

¶7 ATTACHMENT 3A

DIRECTIONS FOR COMPLETING THE NRC REACTOR PLANT EVENT NOTIFICATION WORKSHEET

(Page 2 of 2)

B. (continued)

> **NOTE**
> No other blocks in the upper half of the form are required.

10. Description - provide a written description of the event.

> **NOTE**
> Check the blocks in the lower portion of the form based on current conditions.

11. Mode of operation until corrected - provided if known.

12. Estimate for restart date - enter "unknown".

13. Additional info on Page 2 - enter yes or no.

C. Reactor Plant Event Notification Worksheet, Page 2

1. Page 2 is used to collect information on:

- Radiological releases

  OR

- Reactor Coolant System (RCS) leakage

  OR

- Steam generator (SG) tube leaks

2. Fill in as much of the information on the form as is immediately available - do not create undue delay in making the notification. This information can be gained once the open line of communication is established.

D. Approval

1. Information entered on the worksheet shall be reviewed and approved by the EC or RM (if used in the EOF), prior to transmission.

2. The EC / RM may initial on the worksheet to indicate approval. There is no formal sign-off location on the worksheet.

END OF ATTACHMENT 3A

ATTACHMENT 4

ADDITIONAL STATE NOTIFICATION FORM INFORMATION

(Page 1 of 2)

1. SNF Line 3: Site
Do not check more than one unit. For dual unit events such as the approach of a hurricane or loss of off-site power, the fact that the other unit is affected by the condition should be stated in the "Additional Information or Update" section.

2. SNF Line 5: When opening and closing an event in one notification:

   - On line 5, both box A and box B apply
     Date(s) and times of declaration and termination are to be indicated.

3. SNF Line 9: Release Status

   A. If, as a result of **accident conditions**, ANY of the following is true:

   - A rise of (approximately) 10 times or one decade above pre-transient values is seen on one of the following monitors used for accident assessment:

     - A/B Main Steam Line: RE 26-62/63 (RIM 26-71/72) - ONLY if a steam release is in progress
     - A/B ECCS: RSC 26-2/3 (RS 26-69/70)
     - Plant Vent: RSC 26-1 (RS 26-90)
     - Fuel Building: RSC 26-4 (RS 26-12)
     - Health Physics detects unplanned elevated airborne radiation levels outside of plant buildings due to accident conditions
     - A Steam Generator, with primary-to-secondary leakage, due to a tube leak or rupture, is vented to atmosphere
     - Operating the C AFW pump from an affected Steam Generator
     - A release of radioactive material is in progress, even though it may be less than normal operating limits

     Then a RELEASE is in progress.

ATTACHMENT 4
ADDITIONAL STATE NOTIFICATION FORM INFORMATION
(Page 2 of 2)

4. SNF Line 10: Release Significance Category

A. If a release is in progress or has occurred and dose information is not available at the time of notification, check box "A" and follow up as soon as dose assessment is performed.

B. Check box "B" if both noble gas and iodine release rates are less than or equal to the following:

Noble Gas release $\leq$ 3.5 E+5 μCi/sec (3.5 E-1 Ci/sec)
Iodine release $\leq$ 4.6 E+1 μCi/sec (4.6 E-5 Ci/sec)

C. Check box "C" if either noble gas or iodine release rates exceed the values in "B" (above) but forecasted 1 mile doses are less than 1000 mrem TEDE and 5000 mrem Thyroid CDE. These doses are less than the state's Protective Action Guide (PAG) levels.

D. Check box "D" if forecasted 1 mile doses are greater than or equal to either 1000 mrem TEDE or 5000 mrem Thyroid CDE. These PAG levels require state and county action.

E. Check box "E" if liquid release exceeds ODCM limits per Chemistry analysis.

5. SNF Line 13: Weather Data

A. ¶10 Temperature, wind speed and wind direction can be obtained from ERDADS (DCS).

B. The Met Tower Indicator Panel in the Unit 1 Control Room is an alternate source.

C. If these two sources are not available, refer to Attachment 10, URI Meteorological Data, in EPIP-14, Dose Assessment Using the Unified Rascal Interface.

END OF ATTACHMENT 4

FOR INFORMATION ONLY
Before use, verify revision and change documentation (if applicable) with a controlled index or document.
DATE VERIFIED ______ INITIAL ______

FPL	**ST. LUCIE PLANT** **EMERGENCY PLAN IMPLEMENTING PROCEDURE** SAFETY RELATED **REFERENCE USE**	Procedure No. **EPIP-11** Current Revision No. **7**

Title:

CORE DAMAGE ASSESSMENT

PSL CONTROL 810 COPY
NUCLEAR RECORDS

Responsible Department: **EMERGENCY PLANNING**

REVISION SUMMARY:

Revision 7 - Incorporated PCR 2100704 to update Records Required and Responsibilities sections. (Author: J. Moody)

Revision 6 - Incorporated PCR 1792249 for Unit 2 Extended Power Uprate EC 249985 and minor enhancements. (Author: R. Sandford)

Revision 5 - Incorporated PCR 1781155 for EC 246569 EPU Power Uprate. (Author: R. Young)

Revision 4 - Incorporated PCR 563352 to address PC/M 09070, PSL Unit 2 Emergency Response Data Acquisition and Display System (ERDADS) Replacement. (Author: R. Young)

Revision 3A - Incorporated PCR 05-3000 for CR 2005-18614 to add Level of Use to procedure cover page. (Helga Baranowsky, 10/18/05)

Revision 3 - Deleted references to Post Accident Sample System (PASS). (J. R. Walker, 07/23/01)

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)

Revision	Approved By	Approval Date
0	J. Scarola	12/17/97
7	D. Taylor	02/15/16

UNIT #	
DATE	
DOCT	PROCEDURE
DOCN	EPIP-11
SYS	
STATUS	COMPLETED
REV	7
# OF PGS	

Implementation Date 2/18/16
Initials

TABLE OF CONTENTS

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, Operating Experience, License Renewal, etc. and shall NOT be revised without the required Focus review and appropriate 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.

Ψ Indicates a step that requires a sign off on an attachment.

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).

2.1 References (continued)

3. "CORD Version 1A - Core Damage Assessment Computer Program for St. Lucie Units 1 and 2," IMPELL/FPL-85-116, June 3, 1985. (Updated to Version 2.0 - "CORD-EPU-EXE" - for EPU by Numerical Applications Division.

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).

7. Numerical Applications Software Release - FPL Class A - St Lucie - EPU. NAI-1631-001 Revision 0.

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 designee, 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, through Juno Beach Fuels Engineers, as needed.

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

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 (DCS)) 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 (DCS) 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

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.

5.5 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
(Page 1 of 3)

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	Diffusionals 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

(Page 2 of 3)

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

(Page 3 of 3)

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	

END OF ATTACHMENT 1

ATTACHMENT 2
CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD
(Page 1 of 8)

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.

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 8)

3. Specific Instructions (continued)

D. Obtain from ERDADS (DCS) 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. For any post accident sample, 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"

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.

ATTACHMENT 2
CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD
(Page 3 of 8)

3. Specific Instructions (continued)

F. (continued)

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)

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 representative power level in percent using engineering judgment. 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)

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.

3. Enter 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).

ATTACHMENT 2
CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD
(Page 4 of 8)

3. Specific Instructions (continued)

H. (continued)

4. Enter proper response for correction to STP in accordance with information provided with the 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."

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.

ATTACHMENT 2
CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD
(Page 5 of 8)

3. Specific Instructions (continued)

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 2.0 and can be run using the IBM BASIC 2.0 interpreter. The CORD-EPU program/code contains the following files:

CORD-EPU.BAS	The CORD program source BASIC source code
CORD-EPU.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 6 of 8)

4. Generic CORD User Instructions (continued)

B. Getting Started

To use the CORD program, take the following steps:

1. Access CORD-EPU Version 2.0, 2012.

2. Run CORD-EPU.EXE; 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 7 of 8)

4. Generic CORD User Instructions (continued)

C. (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.

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.

ATTACHMENT 2
CORE DAMAGE ASSESSMENT USING THE COMPUTER CODE CORD
(Page 8 of 8)

4. Generic CORD User Instructions (continued)

E. (continued)

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."

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.

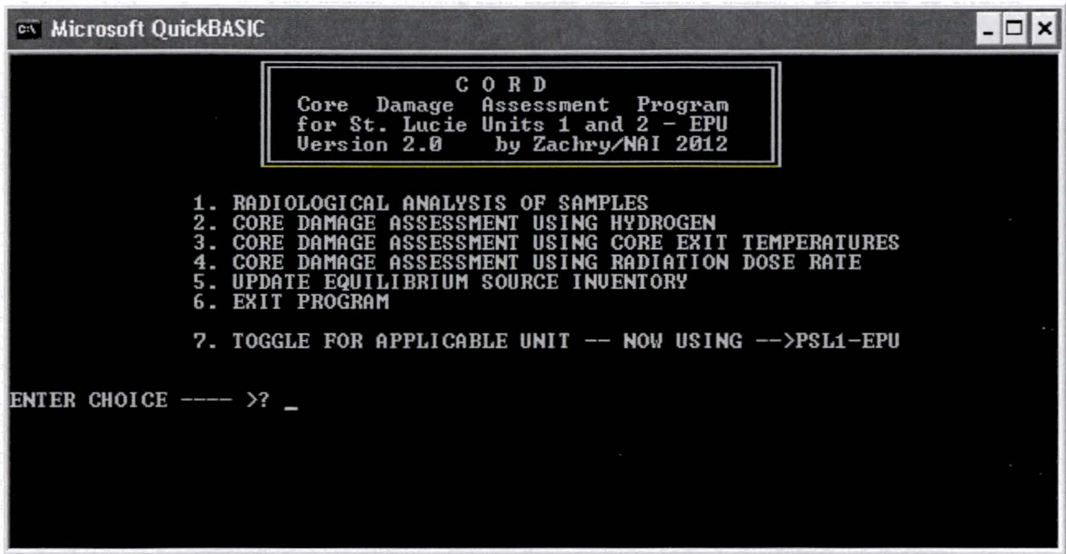
Key	Function
ESC	Clears the input field and places the cursor in the left most location within the field
BACKSPACE	Deletes the character to the left of the cursor
DEL	Deletes the character at the current cursor location
RETURN	Concludes the current entry and moves the cursor to the next field
HOME	Moves 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

END OF ATTACHMENT 2

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 1 of 16)

PROGRAM INPUT FOR OPTION 1
(Page 1 of 3)
GENERAL INFORMATION

> **NOTE**
> Although the test cases shown in this attachment use Unit 1 data, these cases provide acceptable validation of the CORD program for use for .both Units.



Allow the CORD-EPU code to default to St. Lucie Unit 1 for these test cases.

Select Option 1 for this test case, which will generate the following screen:

ENTER OPTION:

1. USE LAST DATA SET
2. ENTER NEW DATA SET

ENTER 1 OR 2 > ? 1

Note: Using the last data set will pre-populate the program with the contents of data files CORDPSL1.DAT or CORDPSL2.DAT. When running these following test cases, the user should verify that the pre-populated data matches the test case input described below.

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 2 of 16)

PROGRAM INPUT FOR OPTION 1
(Page 2 of 3)
GENERAL SAMPLE INFORMATION

ENTER DATE AND TIME OF REACTOR SHUTDOWN

DATE: 7/18/12
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:	7/18/12	7/18/12	7/18/12
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 3 of 16)

PROGRAM INPUT FOR OPTION 1
(Page 3 of 3)
GENERAL SAMPLE INFORMATION

Do you want to correct sample to STP (Y / N)? Y

Do you want to correct sample for radioactive DECAY (Y / N)? Y

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 4 of 16)

OPTION 1 (OUTPUT)
(Page 1 of 5)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:20

ENCLOSURE A4 - UNIT: PSL1-EPU
RECORD OF DECAY CORRECTION

TIME OF REACTOR SHUTDOWN: 7/18/12 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	5.36E+00	1.20E-02	6.19E-02	1.00E-01	5.16E-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	1.0E-06	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	2.58E+00	1.20E-02	2.98E-02	1.00E-01	2.49E-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	1.20E+03	1.20E-02	1.38E-01	1.00E-01	1.15E+02
TE129	1.7E-04	1.04E+03	6.38E+03	1.20E-02	7.37E-02	1.00E+01	6.15E+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	4.00E+01	1.20E-02	4.62E-02	1.00E-01	3.85E-01
PR144	6.7E-04	1.04E+00	1.42E+03	1.20E-02	1.65E-01	1.00E-01	1.37E+02

Prepared by: ______________________________ Date: ____/____/____

Checked by: ______________________________ Date: ____/____/____

Approved by: ______________________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 5 of 16)

OPTION 1 (OUTPUT)
(Page 2 of 5)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:24

UNIT: PSL1-EPU
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	0.0508	0.5079	%5079.1450
XE131M	0.0099	0.0991	%9908.0859
XE133	1.0000	1.0000	1.0000

IODINES:

I131	1.0000	1.0000	1.000
I132	0.0002	0.2463	0.0025
I133	0.0109	0.0109	0.0011
I135	0.0001	0.1357	0.0014

Prepared by: ______________________ Date: ____/____/____

Checked by: ______________________ Date: ____/____/____

Approved by: ______________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 6 of 16)

OPTION 1 (OUTPUT)
(Page 3 of 5)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:28

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

SAMPLE NUMBER: 001
LOCATION: RCS HOT LEG

Isotope	Decay Corr Spec Activity (Encl A4) µCi/cc	ACT Ratio Calculated For Sample*	ACR Ratio Fuel Pellet Inventory**	ACT Ratio Gas Gap Inventory**	Identified Source
KR87	5.36E+00	5.08E-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	N/A
I131	1.05E+04	1.00E+00	1.0	1.0	N/A
I132	2.58E+00	2.46E-04	1.4	0.01 - 0.05	
I133	1.15E+02	1.09E-02	2.0	0.5 - 1.0	
I135	1.42E+00	1.36E-04	1.8	0.1 - 0.5	

* N. G. Ratio of Sample Activity to Xe-133 Activity
Iodine Ratio of Sample Activity to I-131 Activity

** Typical Activity Ratios from EPIP-11, Reference 2.1.2, Table 3.3 of CEOG Task 467, July 1983

Prepared by: ______________________ Date: ____/____/____

Checked by: ______________________ Date: ____/____/____

Approved by: ______________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 7 of 16)

OPTION 1 (OUTPUT)
(Page 4 of 5)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:32

ENCLOSURE A5 - UNIT: PSL1-EPU
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	6.19E-02	5.08E-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	N/A
I131	1.21E-01	1.00E+00	1.0	1.0	N/A
I132	2.98E-02	2.46E-01	1.4	0.01 - 0.05	
I133	1.33E-03	1.09E-02	2.0	0.5 - 1.0	
I135	1.64E-02	1.36E-01	1.8	0.1 - 0.5	

* N. G. Ratio of Sample Activity to Xe-133 Activity
Iodine Ratio of Sample Activity to I-131 Activity

** Typical Activity Ratios from EPIP-11, Reference 2.1.2, Table 3.3 of CEOG Task 467, July 1983

Prepared by: ______________________ Date: ____/____/____

Checked by: ______________________ Date: ____/____/____

Approved by: ______________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 8 of 16)

OPTION 1 (OUTPUT)
(Page 5 of 5)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:35

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

SAMPLE NUMBER: 003
LOCATION: CONTAINMENT SUMP

Isotope	Decay Corr Spec Activity (Encl A4) µCi/cc	ACT Ratio Calculated For Sample*	ACT Ratio Fuel Pellet Inventory**	ACT Ratio Gas Gap Inventory**	Identified Source
KR87	5.16E-01	5.08E+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	N/A
I131	1.01E+02	1.00E+00	1.0	1.0	N/A
I132	2.49E-01	2.46E-03	1.4	0.01 - 0.05	
I133	1.11E-01	1.09E-03	2.0	0.5 - 1.0	
I135	1.37E-01	1.36E-03	1.8	0.1 - 0.5	

* N. G. Ratio of Sample Activity to Xe-133 Activity
Iodine Ratio of Sample Activity to I-131 Activity

** Typical Activity Ratios from EPIP-11, Reference 2.1.2, Table 3.3 of CEOG Task 467, July 1983

Prepared by: ______________________ Date: ____/____/____

Checked by: ______________________ Date: ____/____/____

Approved by: ______________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 9 of 16)

ADDITIONAL PROGRAM INPUT FOR OPTION 1
(Page 1 of 1)
RELEASE QUANTITY 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 10 of 16)

OPTION 1 (FINAL OUTPUT)
(Page 1 of 2)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:46

ENCLOSURE A7 - UNIT: PSL1-EPU
RECORD OF RELEASE QUANTITY

Isotope	Reactor Coolant Sample Number, 1	Containment Sump Sample Number, 2	Contain Atmosphere Sample Number, 3	Total Quantity
KR87	1.52E+03	1.47E+02	3.66E+03	5.33E+03
XE131M	2.97E+02	2.86E+01	7.15E+02	1.04E+03
XE133	2.99E+04	2.89E-03	7.21E+03	3.72E+04
I131	2.98E+06	2.87E+04	7.17E+03	3.01E+06
I132	7.33E+02	7.07E+01	1.77E+03	2.57E+03
I133	3.25E+04	3.14E+01	7.84E+01	3.27E+04
I135	4.04E+02	3.89E+01	9.73E+02	1.42E+03
CS134	2.95E+02	2.84E+01	7.10E+02	1.03E+03
RB88	3.40E+05	3.27E+04	8.18E+05	1.19E+06
TE129	1.81E+06	1.75E+04	4.36E+03	1.83E+06
TE132	3.02E+02	2.92E+01	7.29E+02	1.06E+03
SR89	2.95E+02	2.84E+01	7.11E+02	1.03E+03
BA140	2.97E+02	2.86E+01	7.14E+02	1.04E+03
LA140	3.10E+02	2.99E+01	7.47E+02	1.09E+03
LA142	1.13E+04	1.09E+02	2.73E+03	1.42E+04
PR144	4.04E+05	3.90E+04	9.74E+05	1.42E+06

Prepared by: ______________________________ Date: ____/____/____

Checked by: ______________________________ Date: ____/____/____

Approved by: ______________________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 11 of 16)

OPTION 1 (FINAL OUTPUT)
(Page 2 of 2)

CORD - CORE DAMAGE ASSESSMENT PROGRAM RUNTIME: 06-08-2012
VERSION 2.0 (5/25/12) 13:09:50

UNIT: PSL1-EPU
RELEASE OF GAS GAP AND FUEL PELLET INVENTORY

Isotope	GAS GAP		FUEL PELLET	
	Corrected Source Inv	% Rel	Corrected Source Inv	% Rel
KR87	1.65E+05	3.23	4.11E+07	0.01
XE131M	2.63E+04	3.96	4.54E+05	0.23
XE133	3.56E+06	1.04	9.05E+07	0.04
I131	4.73E+06	63.75	4.13E+07	7.30
I132	1.53E+06	0.17	1.08E+07	0.00
I133	5.60E+06	0.58	1.35E+08	0.02
I135	3.48E+06	0.04	1.42E+08	0.00
CS134			2.11E+05	0.49
RB88			5.90E+07	2.02
TE129			2.34E+07	7.83
TE132			7.25E+07	0.00
SR89			2.16E+07	0.00
BA140			7.56E+07	0.00
LA140			1.11E+08	0.00
LA142			1.25E+08	0.01
PR144			8.41E+07	1.68

Prepared by: ______________________ Date: ____/____/____

Checked by: ______________________ Date: ____/____/____

Approved by: ______________________ Date: ____/____/____

ATTACHMENT 3
CORD BENCHMARK RUNS
(Page 12 of 16)

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 13 of 16)

OPTION 2 (OUTPUT)
(Page 1 of 1)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 2.0 (5/25/12)

SUMMARY OF HYDROGEN ANALYSIS - UNIT: PSL1-EPU

HYDROGEN IN CONTAINMENT ATMOSPHERE	= 10625	cuft H2
HYDROGEN IN REACTOR COOLANT	= 12480	cuft H2
HYDROGEN IN REACTOR VOID SPACE	= 0	cuft H2
TOTAL HYDROGEN RELEASED	= 23105	cuft H2

TOTAL H2 BY CONTAINMENT MATERIAL OXIDATION	= 12520	cuft H2
UPPER LIMIT BY HYDROGEN MAJOR OVERHEAT	= 2184	cuft H2
LOWER LIMIT BY H2 INITIAL OVERHEAT	= 818	cuft H2
VALUE USED FOR RADIOLYSIS OF WATER	= 818	cuft H2

TOTAL ESTIMATE OF CORE CLAD OXIDATION = 9765.492 cuft H2 2.32%

EST PERCENT OF FUEL WITH RUPTURED CLAD = 100.00%

UPPER EST % FUEL WITH EMBRITTLED CLAD = 20.94%
LOWER EST % FUEL WITH EMBRITTLED CLAD = 9.00%

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 14 of 16)

OPTION 3 (INPUT AND OUTPUT)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 2.0 (5/25/12)

SUMMARY OF CET DAMAGE ANALYSIS - UNIT: PSL1-EPU

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 15 of 16)

OPTION 4 (INPUT)

DOSE RATE INFORMATION

ENTER DATE AND TIME OF REACTOR SHUTDOWN:

DATE: 7/18/12
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 16 of 16)

OPTION 4 (OUTPUT)

CORD - CORE DAMAGE ASSESSMENT PROGRAM
VERSION 2.0 (5/25/12)

UNIT: PSL1-EPU

EQUILIBRIUM DOSE RATE AND TIME POST ACCIDENT

#	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

END OF ATTACHMENT 3

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 judgment 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 (See Instruction 4.A in the text for guidelines) ______°F

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.

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

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

Maximum Core Exit Thermocouple Temperature

____________ °F

____________ °F

____________ °F

____________ °F

____________ °F

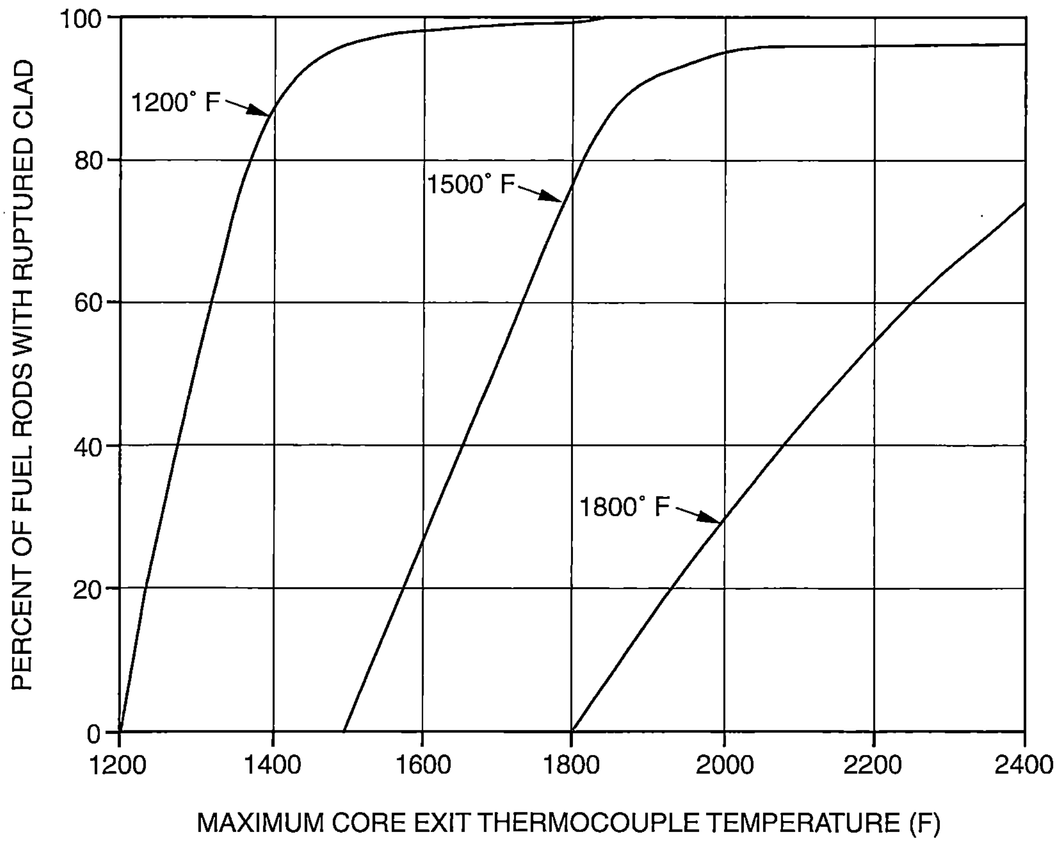
ATTACHMENT 4

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/Att 4-R0)

END OF ATTACHMENT 4

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.

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.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 2 of 7)

3. Precautions and Limitations (continued)

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.

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.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 3 of 7)

3. Precautions and Limitations (continued)

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 judgment 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.

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 judgment 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.

ATTACHMENT 5

PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES

(Page 4 of 7)

4. Instructions (continued)

   B. (continued)

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

      Continued operation for an extended period should weigh more heavily in the judgment 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.

   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 judgment 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.

ATTACHMENT 5
PRELIMINARY ESTIMATE OF CORE DAMAGE USING RADIATION DOSE RATES
(Page 5 of 7)

5. Conclusions (continued)

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: __________

$$\text{Equilibrium Dose Rate (Rad/Hr)} = \text{Measured Dose Rate (Rad/Hr)} \times \frac{100}{\text{Reactor Power Level (\%)}} = \underline{\hspace{2cm}} \text{ (Rad/Hr)}$$

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.

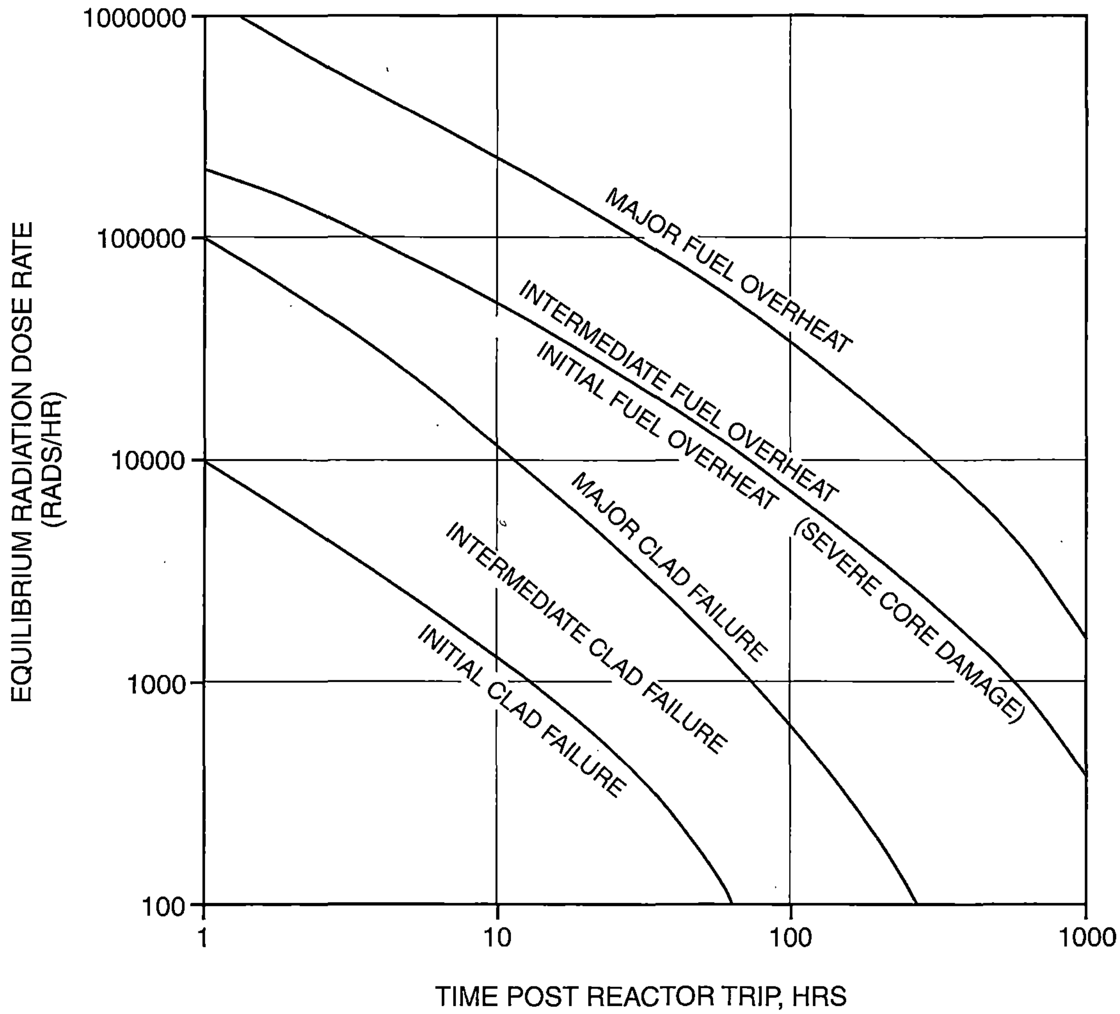
See Step 5 for guidance in formulating conclusions.

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



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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.

END OF ATTACHMENT 5

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 judgment.

Prepared by: ______________________________ Date: ____/____/____

Reviewed by: ______________________________ Date: ____/____/____

Approved by: ______________________________ Date: ____/____/____

END OF ATTACHMENT 6

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 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 judgment than the earlier levels.

   c. Continued operation for an extended period should weigh more heavily in the judgment 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 judgment 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 = 10,400 ft^3
PSL2 = 10,199 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 judgment 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 =

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

(Full Power: PSL1-EPU 3020 Mwt, PSL2 3020 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 ____________ %

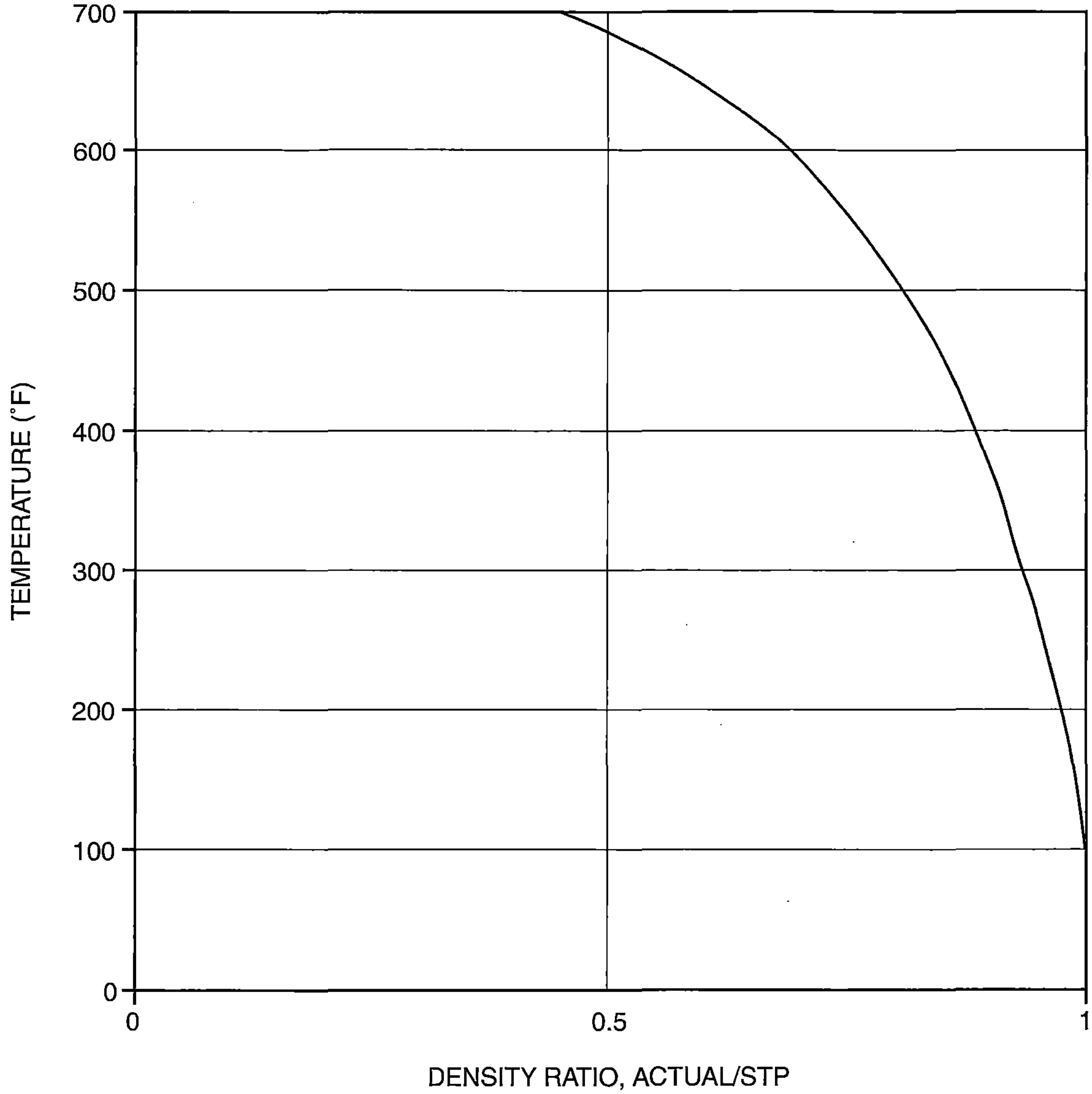
From Table 2, Cladding Damage Characteristics, in Attachment 1, select the core clad damage categories based on the above percentages of rods embrittled (damaged) and enter in Data Sheet 6-2, Summary Worksheet. Note that this assessment will under-predict fuel damage if hydrogen recombiners have operated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



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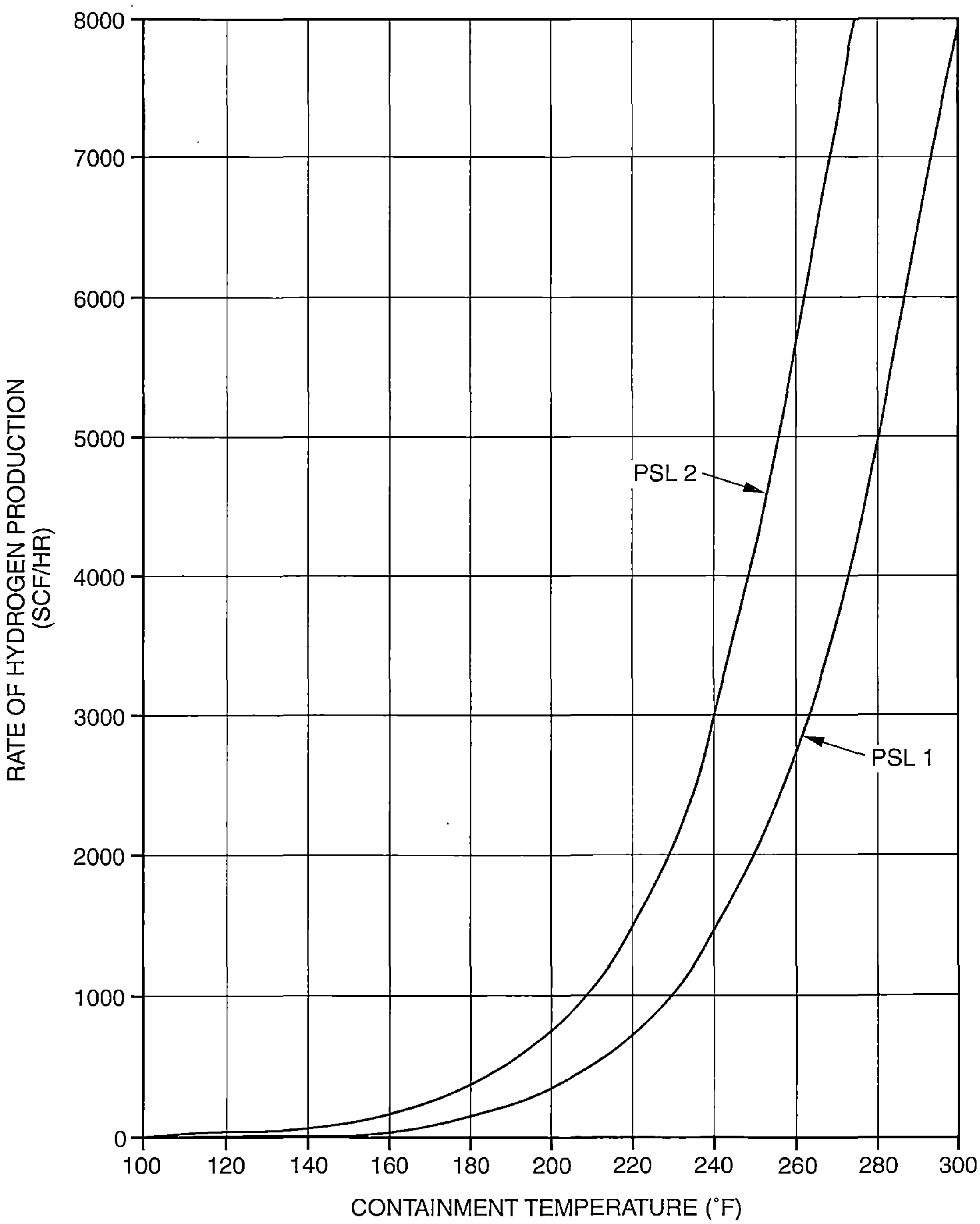
ATTACHMENT 7

PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN

(Page 11 of 14)

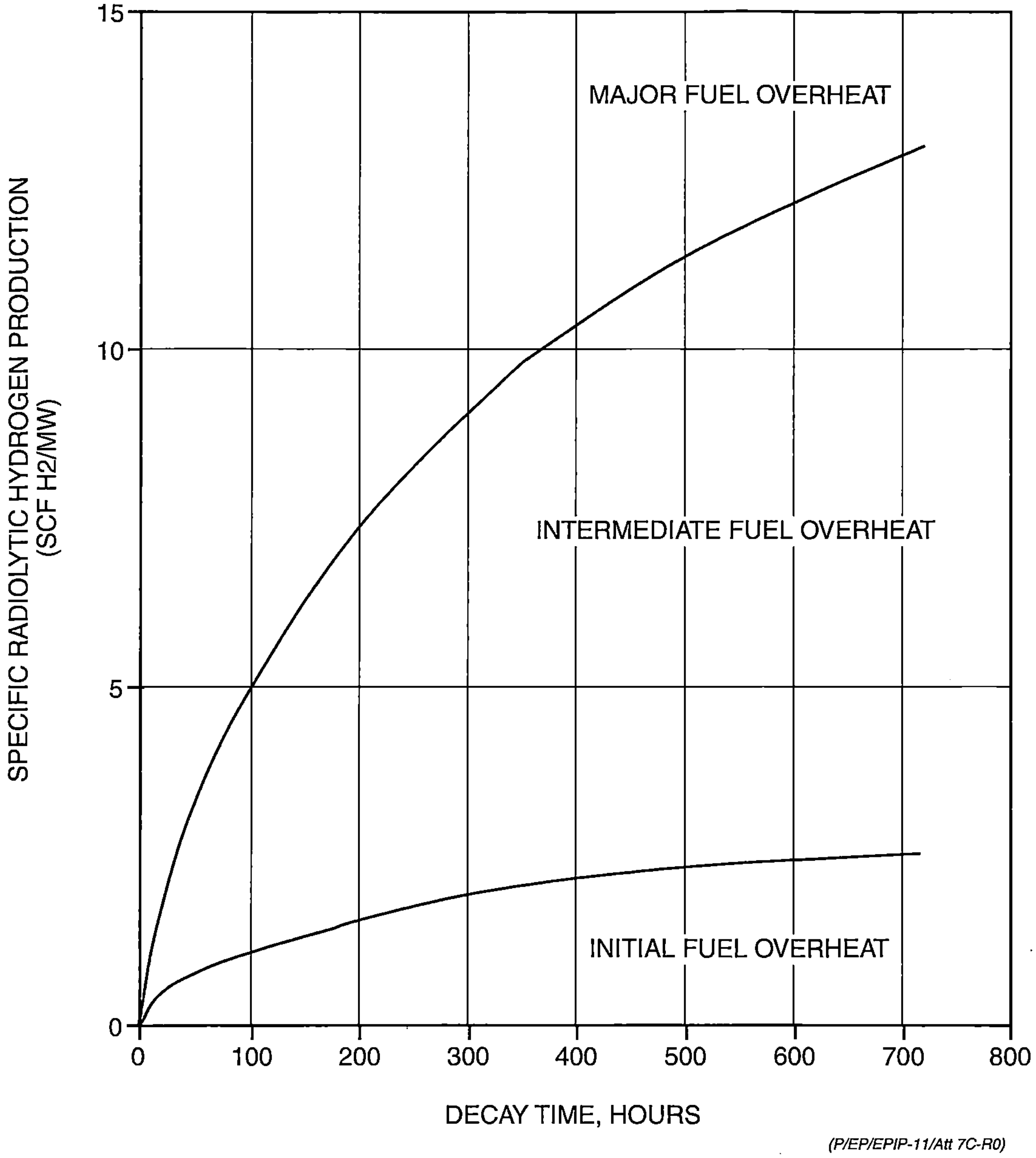
FIGURE 7-2. HYDROGEN PRODUCTION RATE FROM ALUMINUM AND ZINC vs TEMPERATURE



(P/EP/EPIP-11/Att 7B-R0)

ATTACHMENT 7
PRELIMINARY ESTIMATE OF CORE DAMAGE USING HYDROGEN
(Page 12 of 14)

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

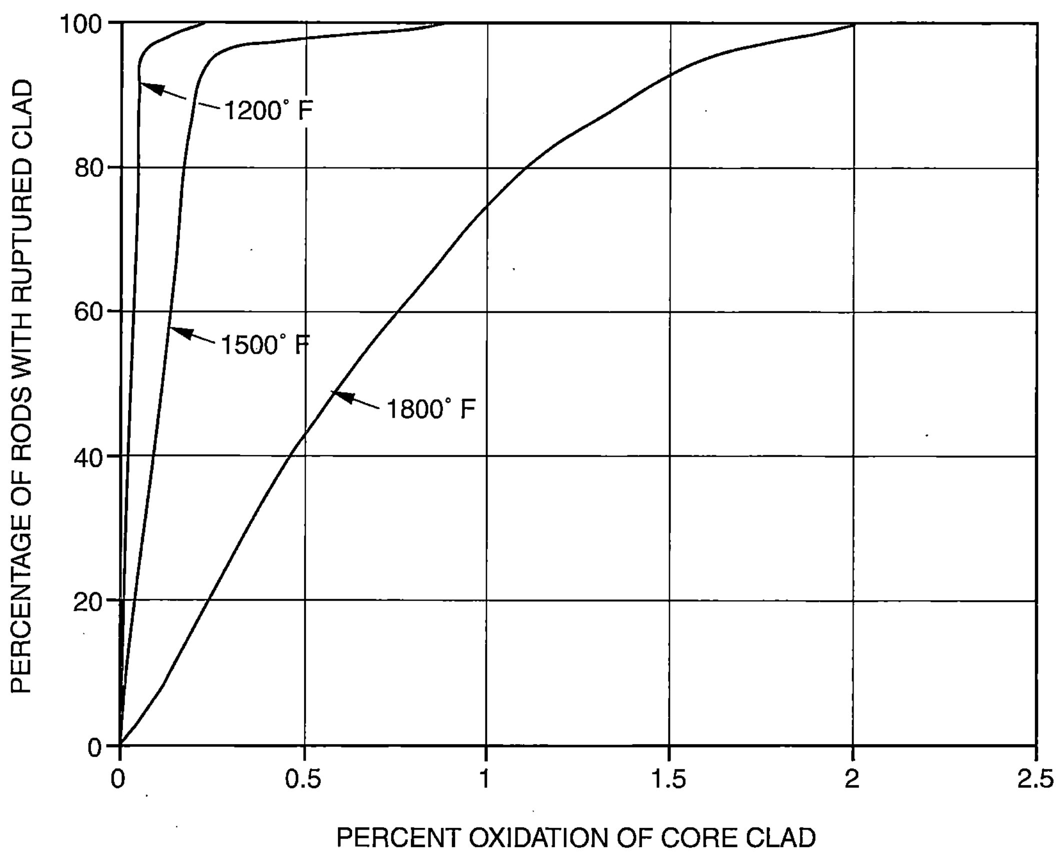


(P/EP/EPIP-11/Att 7C-R0)

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

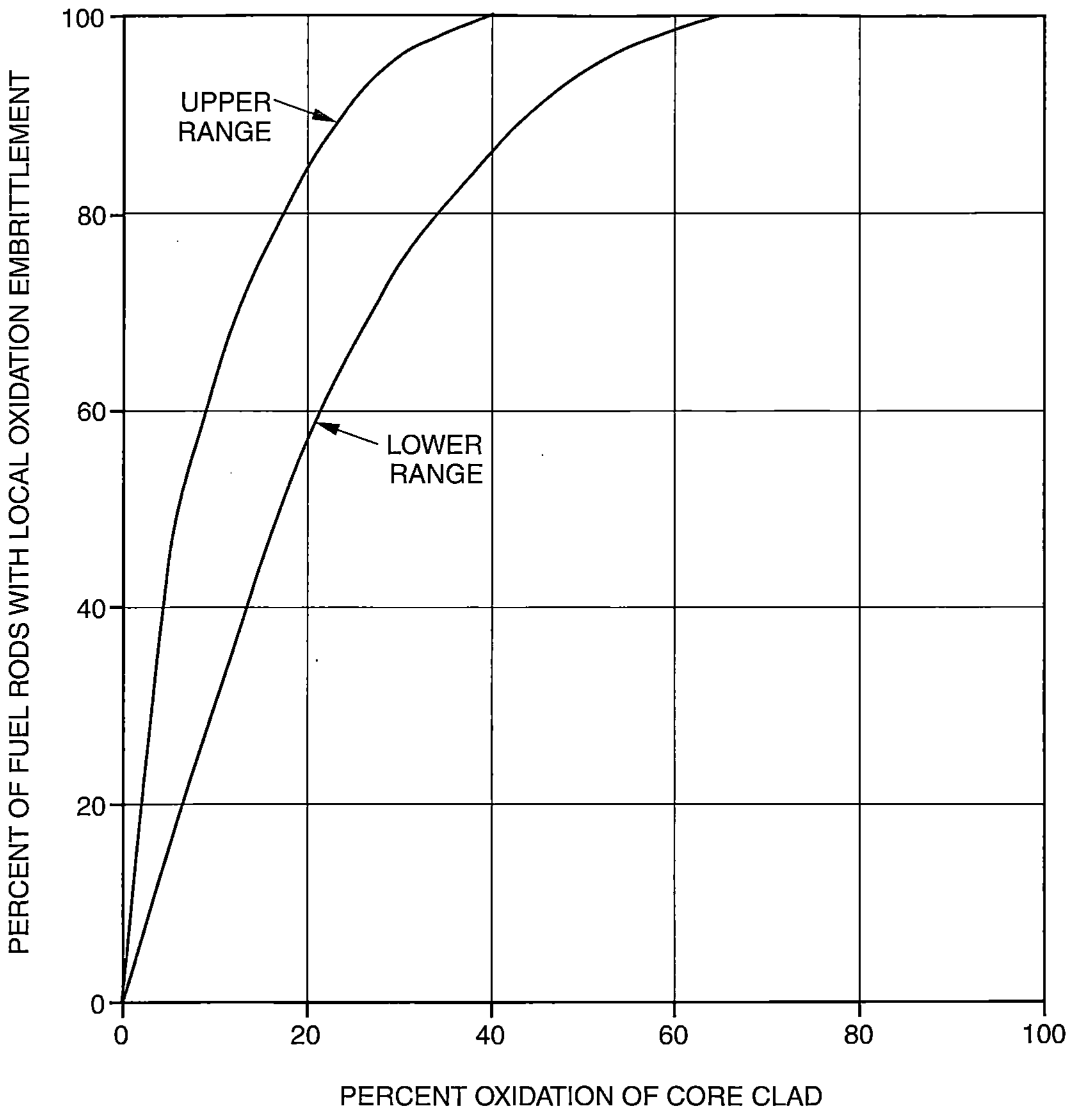


(P/EP/EPIP-11/Att 7D-R0)

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/Att 7E-R0)

END OF ATTACHMENT 7

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. 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:

$$PCF = \frac{1}{100} \Sigma P_j \left(1 - e^{-\lambda t j}\right) e^{-\lambda t^{\circ}_j}$$

Where P_j = Steady reactor power in time period j

t°_j = duration of time period j (sec)

t = 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 judgment 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
(Page 9 of 17)

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	1.52 E-4		
Xe-131m	12d	6.73 E-7		
Xe 133	5.4d	1.53 E-6		
I-131	8d	9.98 E-7		
I-132	2h	8.45 E-5		
I-133	21h	9.26 E-6		
I-135	6.8h	2.92 E-5		
Cs-134	2yr	1.06 E-8		
Rb-88	2m	6.53 E-4		
Te-129	70m	1.68 E-4		
Te-132	78h	2.46 E-6		
Sr-89	52.7d	1.57 E-7		
Ba-140	12.8d	6.29 E-7		
La-140	40h	4.78 E-6		
La-142	90m	1.25 E-4		
Pr-144	17.4m	6.69 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

(Page 10 of 17)

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	

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

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

** Table 3.3 of Reference 2.1.2

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 11 of 17)

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.

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

Where: P1 = Containment pressure in psig (Data Sheet 8-1, Input Parameters)
T1 = Containment temperature in °F (Data Sheet 8-1, Input Parameters)
7.096E10 = Nominal containment volume (cc)

($A_{T,cont}$) = 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
(Page 13 of 17)

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				

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

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 14 of 17)

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 EPU Inventory* =	Power Corrected Source Inventory
Gas Gap Inventory				
Kr 87	2		1.65E+05	
Xe 131m	1		4.62E+04	
Xe 133	1		5.62E+06	
I 131	1		7.80E+06	
I 132	2		1.53E+06	
I 133	2		6.23E+06	
I 135	2		3.49E+06	
Fuel Pellet Inventory				
Kr 87	2		4.11E+07	
Xe 131m	1		7.99E+05	
Xe 133	1		1.43E+08	
I 131	1		6.81E+07	
I 132	2		1.08E+08	
I 133	2		1.50E+08	
I 135	2		1.42E+08	
Cs 134	1		8.66E+06	
Rb 88	2		5.90E+07	
Te 129	2		2.34E+07	
Te 132	1		1.07E+08	
Sr 89	1		7.81E+07	
Ba 140	1		1.35E+08	
La 140	1		1.45E+08	
La 142	2		1.25E+08	
Pr 144	2		8.41E+07	

Corrected Source Inventory=Power Correction Factor x Equilibrium Source EPU 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
(Page 15 of 17)

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 EPU Inventory* x	Power Correction Factor =	Power Corrected Source Inventory
	Gas Gap Inventory		
Kr 87	1.65E+05		
Xe 131m	4.62E+04		
Xe 133	5.62E+06		
I 131	7.80E+06		
I 132	1.53E+06		
I 133	6.23E+06		
I 135	3.49E+06		
	Fuel Pellet Inventory		
Kr 87	4.11E+07		
Xe 131m	7.99E+05		
Xe 133	1.43E+08		
I 131	6.81E+07		
I 132	1.08E+08		
I 133	1.50E+08		
I 135	1.42E+08		
Cs 134	8.66E+06		
Rb 88	5.90E+07		
Te 129	2.34E+07		
Te 132	1.07E+08		
Sr 89	7.81E+07		
Ba 140	1.35E+08		
La 140	1.45E+08		
La 142	1.25E+08		
Pr 144	8.41E+07		

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

* Values from Reference 2.1.4

ATTACHMENT 8
DETAILED RADIOLOGICAL ANALYSIS
(Page 16 of 17)

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 judgment.

Prepared by: ______________________________ Date: ____/____/____

Reviewed by: ______________________________ Date: ____/____/____

Approved by: ______________________________ Date: ____/____/____

END OF ATTACHMENT 8