PILGRIM NUCLEAR POWER STATION

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CHANGE INSTRUCTION NOTICE (CIN)

Transmittal No.: 01-459 Date: 12-10-01

Please update your copy of ______
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REMOVE AND DISCARD	REV. NO.	INSERT	REV. NO.
T of C	73	T of C	74
EP-1P-100	15	EP-1P-100	16
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PILGRIM NUCLEAR POWER STATION

Procedure No. EP-IP-100

EMERGENCY CLASSIFICATION AND NOTIFICATION

STAR

Stop
Think
Act
Review

SAFETY RELATED

REVISION LOG

REVISION 16	**Date Originated 11/01**
Pages Affected	Description
27	Revise EAL 1.4.1.4 and EAL 1.4.1.3 for Torus High Range Area Radiation Monitors due to relocation of detectors.

REVISION 15	**Date Originated 5/01**
Pages Affected	Description
14	Change the term from "current" to "prior" classification level for instructions when lowering a classification.
27	Revise EAL 1.4.1.4 and EAL 1.4.1.3 for Drywell and Torus High Range Area Radiation Monitors.
31	Correct Seismic Recorder Operating annunciator panel location for EAL 7.4.3.1.

REVISION 14	**Date Originated 3/01**
Pages Affected	Description
7,32,35,40	Change the terms “Emergency Release” and “Release” to “Emergency Radioactive Release.”
7	Clarify definition of Emergency Radioactive Release above/below PAGs.
12	In Step 5.3[3], delete sentence regarding update periodicity.
15	Rewrite Section 5.6 to provide enhanced detail.
21,22,24,26	In the Alert announcement, Section A, change “handicapped personnel” to “persons with disabilities.” For the Alert, Site Area Emergency, and General Emergency announcements, eliminate Section D and rewrite the “Action Verification” section.
34,37	Change the term “auto-dialer” to “speed dialer” on the instructions for Attachments 6 and 7.

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1.0 PURPOSE

This Procedure provides instruction of the entry conditions at which specific emergency classifications must be declared, guidelines for the implementation of the Emergency Plan, and the process used to notify response personnel and organizations.

2.0 REFERENCES

[1] EP-AD-600, *"Emergency Action Level Technical Bases"*

[2] EP-PP-01, *"PNPS Emergency Plan"*

[3] PNPS Technical Specifications

3.0 DEFINITIONS

[1] Adequate Core Cooling - Heat removal from the Reactor sufficient to prevent rupturing the fuel clad. Three viable mechanisms for establishing adequate core cooling are defined: core submergence, spray cooling, and steam cooling.

(a) Submergence is the preferred method for cooling the core. The core is adequately cooled by submergence when it can be determined that RPV water level is at or above the top of the active fuel. All fuel nodes are then assumed to be covered with water and heat is removed by boiling heat transfer.

(b) Adequate Spray Cooling is provided, assuming a bounding axial power shape, when at least one Core Spray subsystem is injecting into the RPV at the design flow rate (3,600 GPM) and RPV water level is at or above the elevation of the jet pump suctions (-175 in.). The covered portion of the core is then cooled by submergence while the uncovered portion is cooled by the spray flow. The required spray flow must be supplied by a single subsystem to ensure adequate spray distribution to all fuel bundles.

(c) Steam Cooling is relied upon only if RPV water level cannot be restored and maintained above the top of the active fuel, cannot be determined, or must be intentionally lowered below the top of the active fuel. The core is adequately cooled by steam if the steam flow across the uncovered length of each fuel bundle is sufficient to maintain the hottest peak clad temperature below the appropriate limiting value; 1500°F if makeup can be injected, 1800°F if makeup cannot be injected. The covered portion of the core remains cooled by boiling heat transfer and generates the steam which cools the uncovered portion.

Steam cooling with makeup capability is established by maintaining RPV water level above the Minimum Steam Cooling RPV Water Level (-150 in.) or RPV pressure above the Minimum Alternate RPV Flooding Pressure (a function of the number of open SRVs). In either case, the peak clad temperature is limited to 1500°F, the threshold for fuel rod perforation.

Steam cooling without makeup capability is established as long as RPV water level remains above the Minimum Zero-Injection RPV Water Level (-160 in.). The peak clad temperature is permitted to rise to 1800°F, the threshold for significant metal-water reaction, to maximize the heat transfer to steam and to delay the need for RPV depressurization as long as possible. The minimum RPV water level at which adequate steam flow exists is higher when makeup capability exists because:

- The limiting fuel temperature is lower (1500°F). The higher limit of 1800°F is used only when cladding perforation cannot be avoided.
- With injection, water at the core inlet is subcooled. Some of the energy produced by the core must then be expended in raising the temperature of the liquid to saturation and less steam will be produced to cool the uncovered portions of the core.

[2] Alert - Events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety at PNPS. Any releases are expected to be limited to small fractions of EPA Protective Action Guideline exposure levels.

[3] BECONS - The PNPS community offsite notification system.

[4] Cannot Be Determined - The current value or status of an identified parameter relative to that specified in the Emergency Action Level (EAL) cannot be ascertained using all available indications (direct and indirect, singly or in combination).

[5] Cannot Be Maintained Above/Below - The value of the identified parameter(s) is not able to be kept above/below specified limits. This determination includes making an evaluation that considers both current and future system performance in relation to the current value and trend of the parameter(s). It does not imply that the actual value of the parameter must first pass the specified limit.

[6] Cannot Be Restored Above/Below - The value of the identified parameter(s) is not able to be returned to above/below specified limits after having passed those limits. This determination includes making an evaluation that considers both current and future system performance in relation to the current value and trend of the parameter(s). It does not imply any specific time interval, but does not permit prolonged operation beyond the limit without declaring the specified emergency classification. (May be used in combination with Definition 3.0[5].)

[7] Computerized Automated Notification System (CANS) - A computer-assisted system that, when activated, has the following capabilities:

(a) Activating the emergency pager system.

(b) Accepting calls from authorized responders to inform them of an abnormal or emergency condition at PNPS.

(c) Calling response personnel at their home or work phone to inform them of an abnormal or emergency condition at PNPS.

(d) Maintaining an updated list of personnel responding and their estimated time of arrival at their facility.

[8] Controlled Process - A preplanned activity for which the conditions specified in an EAL are anticipated to be or are intentionally exceeded as part of an approved Procedure.

[9] DNN - Digital Notification Network.

[10] Emergency Radioactive Release is/is not in Progress - For purposes of offsite notification, any release of radioactivity is considered 'an emergency radioactive release in progress which:

- Meets any EAL of Classification Subsection 5.1, Effluent Monitors

OR

- Involves an actual or suspected Turbine Building or unmonitored release which is associated with the emergency event.

[11] Emergency Radioactive Release is above/below Protective Action Guides (PAGs) - PAGs are defined by the EPA as dose in excess of 1 rem TEDE or 5 rem CDE Thyroid. For purposes of offsite notification, an emergency radioactive release is considered above PAGs when:

- General Emergency EAL 5.2.1.4 or EAL 5.2.2.4 is exceeded based on having dose projections or offsite measurements.

OR

- In the absence of having dose projections or offsite measurements. General Emergency EAL 5.1.1.4 is exceeded.

[12] ENS - NRC Emergency Notification System.

[13] Essential Information Checklist - The form used by oncoming Emergency Director when relieving present Emergency Director of the responsibilities outlined in this Procedure. This form may also be used to provide information to Media Center and Corporate personnel.

[14] Essential Personnel - Those individuals assigned specific emergency response duties as identified in the PNPS Emergency Plan and Implementing Procedures.

[15] Follow-Up Information Form - The form used to initiate and document periodic emergency classification updates to offsite agencies.

[16] General Emergency - Events are in progress or have occurred which involve actual or imminent substantial core degradation or melting with the potential for loss of containment integrity. Releases can be expected to exceed EPA Protective Action Guideline exposure levels for more than the area near the site boundary.

[17] Initial Notification Form - The form used to initiate and document initial emergency classification notifications to offsite agencies.

[18] Non-Essential Personnel - Personnel who are not assigned specific emergency response duties.

[19] Primary System - The pipes, valves, and other equipment which connect directly to the Reactor Pressure Vessel (RPV) such that a reduction in RPV pressure will effect a decrease in the steam or water being discharged through an unisolated break in the system.

[20] Radiation Monitor Channel A and B Above/Below EAL Threshold Readings - The value of both channels statements above or below a threshold reading may not apply when one channel is known to be inoperable or out of service. The determination of a valid reading on either channel that exceeds the EAL threshold constitutes that the intent of the EAL has been met and the associated classification should be declared.

[21] Shutdown - As regards Reactor status, the Reactor is shutdown if the Reactor is subcritical (power decreasing) and below the heating range (IRM range 7).

[22] Site Area Emergency - Events are in progress or have occurred which involve actual or likely major failures of PNPS functions needed for the protection of the public. Any releases are not expected to exceed EPA Protective Action Guidelines exposure levels except near the site boundary.

[23] Sustained Loss - Loss of power supply, system or component operability for which the return to service has not been determined to be imminent. It does not imply any specific time interval, but prolonged operation is not permitted without declaring the specified emergency condition based on the potential for degraded plant safety.

[24] Termination - The point at which the event is no longer considered to be an emergency. Termination of the emergency is formally identified by an Initial Notification message transmission and entry into Recovery.

[25] Transitory Event - An event in which PNPS exceeded an Emergency Action Level (EAL) but conditions improved prior to classification.

[26] Unusual Event - Events are in progress or have occurred that indicate a potential degradation of the level of safety at PNPS. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

4.0 RESPONSIBILITIES

[1] The Operations Shift Superintendent, or the Control Room Supervisor if the Operations Shift Superintendent is incapacitated or away from the Control Room, shall be responsible for the initial emergency classification declaration and implementation of this Procedure.

[2] The Operations Shift Superintendent, or Control Room Supervisor if the Operations Shift Superintendent is unavailable, shall assume the role of Emergency Director upon initial declaration of any emergency classification and shall continue to function as the Emergency Director until relieved of those duties by the on-call Emergency Director or other qualified individual (i.e., Emergency Plant Operations Supervisor or off-shift Operations Shift Superintendent).

[3] The Emergency Director is the final authority for determining the emergency classification level (initial classification, downgrading, or terminating to recovery). This authority may not be delegated.

[4] The Emergency Director is responsible for directing and overseeing notification of the on-call PNPS Emergency Response Organization.

[5] The Control Room Operations Assistant or a designated alternate is responsible for the following:

(a) When directed, notifying the on-call PNPS Emergency Response Organization.

(b) Periodically checking on the status of personnel responding to the notification process.

(c) If CANS and BEEPS fail, Security is responsible for notifying on-call personnel using commercial telephone lines.

5.0 PROCEDURE

5.1 RECOGNIZING AN EMERGENCY

[1] When indications of abnormal conditions are received, personnel will verify the symptoms/indications and then compare them with the Emergency Action Levels (Attachment 5).

[2] Identify the highest emergency classification level (if multiple EALs are exceeded) for which an EAL has been met or exceeded considering the following:

(a) If conditions warrant the issuance of Protective Action Recommendations (PARs), the classification of General Emergency will be made.

(b) If plant conditions indicate a possible radiological release or a release is in progress or suspected, evaluate the applicability of offsite dose-based EALs (category 5.2).

(c) If a classification level was met or exceeded but the classifiable condition no longer exists (a lesser classification level may or may not still be appropriate), refer to Section 5.4, Transitory Events.

5.2 INITIAL DECLARATION OF AN EMERGENCY FROM THE CONTROL ROOM

[1] The Operations Shift Superintendent, or Control Room Supervisor if the Operations Shift Superintendent is unavailable, shall announce to the Control Room operating staff:

(a) That an emergency has been declared.

(b) The emergency classification level.

(c) That the Operations Shift Superintendent (Control Room Supervisor) has assumed the role of Emergency Director.

[2] Conduct initial emergency notifications as follows:

NOTE

In the event the public address system is inoperative during notifications to Station personnel at any time, determine alternate means to disseminate information to plant personnel.

(a) For events which are classified as a General Emergency, complete the General Emergency Notifications Checklist (Attachment 4).

NOTE

If during the declaration process it becomes necessary to upgrade the emergency classification level before the actual initial notification transmittal, do not send simultaneous notifications. Generate and send a new notification form with the appropriate classification level.

(b) For events which are classified as a Site Area Emergency, complete the Site Area Emergency Notifications Checklist (Attachment 3).

(c) For events which are classified as an Alert, complete the Alert Notifications Checklist (Attachment 2).

(d) For events which are classified as an Unusual Event, complete the Unusual Event Notifications Checklist (Attachment 1).

[3] Contact (or direct an assistant to contact) the on-shift Radiation Protection Supervisor/Technician and direct them to review EP-IP-231 and assume the responsibilities of the Onsite Radiological Supervisor for emergency exposure controls until relieved by the on-call Onsite Radiological Supervisor.

[4] Contact (or direct an assistant to contact) on-shift Maintenance personnel not already involved in the emergency and inform them to report to the Control Room or the OSC to stand by for repair and corrective action activities if necessary.

[5] The Emergency Plant Operations Supervisor (EPOS) may relieve the Operations Shift Superintendent as Emergency Director until the on-call Emergency Director assumes responsibility for the position in the EOF.

[6] The on-shift Emergency Director will remain in the Control Room until relieved by the on-call Emergency Director.

5.3 WHILE IN A CLASSIFIED EMERGENCY

[1] As soon as possible, but no later than 1 hour after event classification, ensure that the ENS or a commercial telephone line is continuously staffed with a knowledgeable individual to provide additional event notification and plant information to the NRC.

[2] Emergency response personnel will continuously review the Emergency Action Levels (Attachment 5) to ensure proper and appropriate event classification.

(a) If the declaration of a higher classification is warranted, conduct initial emergency notifications as follows:

(1) For events which are classified as a General Emergency complete the General Emergency Notifications Checklist (Attachment 4).

> **NOTE**
>
> If during the declaration process it becomes necessary to upgrade the emergency classification level before the actual initial notification transmittal, do not send simultaneous notifications. Generate and send a new notification form with the appropriate classification level.

(2) For events which are classified as a Site Area Emergency, complete the Site Area Emergency Notifications Checklist (Attachment 3).

(3) For events which are classified as an Alert, complete the Alert Notifications Checklist (Attachment 2).

(b) If a higher classification level was met but that classifiable condition no longer exists (a lesser classification level may or may not still be appropriate), refer to Section 5.4, Transitory Events.

(c) If the emergency conditions support downgrading the classification level, refer to Section 5.5, Downgrading Emergency Classifications.

(d) If the situation has been controlled and a state of emergency is no longer necessary, refer to Section 5.6, Transition to Recovery.

[3] Provide periodic updates (hourly or whenever conditions change) to the Commonwealth and local communities using Follow-Up Information Forms (Attachment 7).

[4] As conditions improve and additional personnel and resources become available, certain recovery activities (described in EP-IP-520) may be initiated prior to termination of the emergency.

[5] Turnover of the Emergency Director Position

(a) The offgoing Emergency Director will provide the oncoming Emergency Director with a briefing of the emergency conditions and the status of offsite notifications.

(b) Items contained on the Essential Information Checklist (Attachment 9) will be used to facilitate the turnover briefing as follows:

(1) The offgoing Emergency Director may complete an essential Information Checklist and provide a copy to the oncoming Emergency Director to be used for the turnover briefing.

(2) The oncoming Emergency Director may complete an Essential Information Checklist while covering each item during conduct of the turnover briefing.

(c) Discuss any Protective Action Recommendations issued to offsite agencies as applicable.

5.4 TRANSITORY EVENTS

[1] For situations which begin under non-emergency conditions then experience events which ultimately result in a classifiable emergency, enter and execute the following Attachment which corresponds to the appropriate current classification level:

(a) Attachment 1, Unusual Event Notifications Checklist.

(b) Attachment 2, Alert Notifications Checklist.

(c) Attachment 3, Site Area Emergency Notifications Checklist.

[2] For situations which begin under non-emergency conditions, experience events which qualify as a classifiable emergency, and result in conditions which no longer meet a classification level:

(a) Consider the following items prior to entering Recovery:

(1) Conditions no longer meet an Emergency Action Level and it appears unlikely that conditions will deteriorate.

(2) Plant releases of radioactive materials to the environment are under control (within Technical Specifications) or have ceased and the potential for an uncontrolled radioactive release is acceptably low.

(3) The Reactor is in a stable condition and long-term core cooling is available.

(4) Offsite conditions do not unreasonably limit access of outside support to the Station and qualified personnel and support services are available.

(b) Complete and distribute an Initial Notification Form to specify the transitory event classification level and signify entry into Recovery (EAL number 0.0.0.0 is used to signify entry into Recovery).

(c) Exit this Procedure and enter EP-IP-520, *"Transition and Recovery."*

[3] For situations which begin in a classified emergency then experience events which ultimately result in:

(a) A return to the current classification level

Complete and distribute an Initial Notification Form to specify the transitory event classification level and signify return to the current classification level.

NOTE

Transitory events which occur during an emergency cannot directly result in a downgraded classification level or entry into Recovery. These actions must be performed separately.

(b) A lower classification level or no longer meet a classification level

Complete and distribute an Initial Notification Form to specify the transitory event classification level and signify return to the prior classification level.

[4] Return to Section 5.3.

5.5 DOWNGRADING EMERGENCY CLASSIFICATIONS

[1] Once in an Alert or higher classification level, the decision to downgrade below the Alert level shall only be made after the TSC, OSC, and EOF have been activated.

[2] Have the Control Room make the following announcement over the public address system, TWICE:

"Attention all personnel, attention all personnel: The emergency classification level has been downgraded to a/an (applicable classification level)."

[3] Complete and distribute an Initial Notification Form to signify entry into the lower emergency classification level.

[4] Return to Section 5.3.

5.6 TRANSITION TO RECOVERY

[1] If a state of emergency is no longer necessary based on plant status conditions, complete the Termination Checklist (Attachment 8) to determine whether conditions allow termination of the event and entry into Recovery.

[2] If conditions allow for termination of the emergency and entry into Recovery, exit this Procedure and enter EP-IP-520, *"Transition and Recovery"*.

[3] If conditions do not support termination of the emergency and entry into Recovery, continue following the guidance provided in Section 5.3 of this Procedure.

5.7 INITIAL NOTIFICATION FORM DESCRIPTION

<u>NOTE</u>

Upon activation of the EOF, the Control Room and EOF must coordinate the numbering sequence of notifications to ensure consecutive numbers are assigned to notification forms.

[1] <u>Form Number:</u> Notification form numbers are assigned sequentially from the start of the emergency. The sequence includes the Initial Notification Form as well as the Follow-Up Information Form.

[2] <u>Block 1:</u> Designation for notifications conducted as part of a drill or exercise versus an actual event.

[3] <u>Block 2:</u> Specifies the time, circumstance, and classification level applicable to the notification form as follows:

(a) The time and date denote the point at which the Emergency Director (or Operations Shift Superintendent) recognized and formally declared the new event classification level.

(b) The "entered" check box is used when the first Initial Notification Form is completed for an emergency.

(c) The "transitory" check box is used to provide a location to enter the highest classification level which was met during a transitory event.

(d) The "upgraded to" check box is used to provide initial notification of emergencies which require the declaration of a higher classification.

(e) The "downgraded to" check box is used to provide initial notification of emergencies which warrant lowering the classification level.

(f) The classification level boxes are used to indicate the classification applicable to the notification.

[4] Block 3: Provides the applicable EAL number and description for the specified classification level as follows:

(a) For events with more than one EAL in the highest classification level, provide the EAL number for which the event was classified on. Additional EAL numbers may be provided in the description section if desired.

(b) EAL number 0.0.0.0 is used to signify entry into Recovery.

(c) A brief non-technical description (avoiding abbreviations and acronyms) is provided in enough detail to allow an understanding of the nature of the EAL number (and the transitory event if applicable).

[5] Block 4: Provides indication of an emergency radioactive release and the relative magnitude. If necessary, refer to EP-IP-400, *"Protective Actions"*, to determine whether an emergency release is in progress, whether an emergency release is above protective action guides, and what protective action recommendations are necessary.

[6] Block 5: Provides the most recent meteorological data obtained from any of the available sources.

[7] Block 6: Provides any applicable protective action recommendations made by PNPS ERO personnel as follows

(a) Protective Action Recommendations are only issued by PNPS in a General Emergency. If the classification level is not a General Emergency, the "No Protective Actions Necessary" check box shall be selected.

(b) If MEMA or MDPH representatives are present in the EOF and the classification level is General Emergency, the "Provided to MEMA/MDPH" check box is selected.

(c) If MEMA or MDPH representatives are not present in the EOF and the classification level is General Emergency, the appropriate protective action recommendations check boxes and affected subareas are selected.

[8] Block 7: Indicates the time and date when transmission of the form was begun and the name of the individual performing the transmission.

[9] Block 8: Indicated the time and date when transmission of the form was received and the name of the individual receiving the transmission.

[10] Block 9: Provides the protective action recommendations made by PNPS ERO personnel when MEMA or MDPH representatives are present in the EOF (only applicable for General Emergency classifications).

[11] Block 10: Emergency Director approval signature.

5.8 FOLLOW-UP INFORMATION FORM DESCRIPTION

[1] Form Number: Notification form numbers are assigned sequentially from the start of the emergency. The sequence includes the Initial Notification Form as well as the Follow-Up Information Form.

[2] Block 1: Designation for notifications conducted as part of a drill or exercise versus an actual event.

[3] Block 2: Specifies the time and classification level applicable to the Follow-Up Information Form.

[4] Block 3: Indicates the prevailing trend for conditions at the Station.

[5] Block 4: Provides the applicable EAL number and description for the specified classification level as follows:

(a) For events with more than one EAL in the highest classification level, provide the EAL number for which the event was classified on. Additional EAL numbers may be provided in the description section if desired.

(b) A brief non-technical description (avoiding abbreviations and acronyms) is provided in enough detail to allow an understanding of the nature of the EAL number.

[6] Block 5: Indicates whether offsite assistance has been requested, the type of assistance, and the reason for the request.

[7] Block 6: Provides indication of an emergency radioactive release and the relative magnitude. If necessary, refer to EP-IP-400, *"Protective Actions"*, to determine whether an emergency release is in progress, whether an emergency release is above protective action guides, and what protective action recommendations are necessary.

[8] Block 7: Provides the most recent meteorological data obtained from any of the available sources.

[9] Block 8: Provides any applicable protective action recommendations made by PNPS ERO personnel as follows:

(a) Protective Action Recommendations are only issued by PNPS in a General Emergency. If the classification level is not a General Emergency, the "No Protective Actions Required" check box shall be selected.

(b) If MEMA or MDPH representatives are present in the EOF and the classification level is General Emergency, the "Provided to MEMA/MDPH" check box is selected.

(c) If MEMA or MDPH representatives are not present in the EOF and the classification level is General Emergency, the appropriate Protective Action Recommendations check boxes and affected subareas are selected.

[10] Block 9: Indicates the time and date when transmission of the form was begun and the name of the individual performing the transmission.

[11] Block 10: Indicates the time and date when transmission of the form was received and the name of the individual receiving the transmission.

[12] Block 11: Provides the protective action recommendations and their bases made by PNPS ERO personnel when MEMA or MDPH representatives are present in the EOF (only applicable for General Emergency classifications).

[13] Block 12: Emergency Director approval signature.

6.0 RECORDS

[1] The following documents are generated as a result of the implementation of this Procedure:

(a) Unusual Event Notifications Checklist

(b) Alert Notifications Checklist

(c) Site Area Emergency Notifications Checklist

(d) General Emergency Notifications Checklist

(e) Essential Information Checklist

(f) Initial Notification Form

(g) Follow-Up Information Form

(h) Termination Checklist

[2] All records shall be forwarded to Emergency Preparedness for final disposition following the termination of the emergency.

7.0 ATTACHMENTS

ATTACHMENT 1 - UNUSUAL EVENT NOTIFICATIONS CHECKLIST

ATTACHMENT 2 - ALERT NOTIFICATIONS CHECKLIST

ATTACHMENT 3 - SITE AREA EMERGENCY NOTIFICATIONS CHECKLIST

ATTACHMENT 4 - GENERAL EMERGENCY NOTIFICATIONS CHECKLIST

ATTACHMENT 5 - EMERGENCY ACTION LEVELS

ATTACHMENT 6 - SAMPLE - INITIAL NOTIFICATION FORM

ATTACHMENT 7 - SAMPLE - FOLLOW-UP INFORMATION FORM

ATTACHMENT 8 - SAMPLE - TERMINATION CHECKLIST

ATTACHMENT 9 - SAMPLE - ESSENTIAL INFORMATION CHECKLIST

ATTACHMENT 10 - ACTIVATION OF THE EMERGENCY RESPONSE ORGANIZATION

ATTACHMENT 11 - DOCUMENT CROSS-REFERENCES

ATTACHMENT 12 - IDENTIFICATION OF COMMITMENTS

UNUSUAL EVENT

NOTIFICATION OF STATION PERSONNEL - STANDBY STATUS

CAUTION

During a security threat it may be advisable **NOT** to sound an alarm.

Sound/have the Control Room sound the Operator Recall Alarm and make the following announcement over the public-address system, TWICE:

A. "Attention all personnel, attention all personnel: An Unusual Event has been declared due to (*brief description of initiating event*). All on-call members of the Emergency Response Organization stand by for further instructions. All other personnel continue with your present duties until additional instruction is given."

B. If there is a localized emergency (for example, high radiation, fire), announce its type and location and instruct personnel to stand clear of this area.

Time Completed: ____________

NOTIFICATION OF THE ERO - STANDBY STATUS

Notify or direct notification of the ERO in accordance with Attachment 10, Activation of the Emergency Response Organization.

Time Completed: ____________

NOTIFICATION OF STATE AND LOCAL AGENCIES

Within 15 minutes of the event classification transmit an Initial Notification Form (Attachment 6) to the Commonwealth and local authorities.

Time Completed: ____________

NOTIFICATION OF THE NRC

Inform the NRC of the event classification using the ENS or a commercial telephone (Attachment 6 Step 4).

Time Completed: ____________

RETURN TO THE PROCEDURE **(EITHER STEP 5.2[2] OR 5.4[1])**

ALERT

SITE & PUBLIC AREA NOTIFICATION - RELEASE OF NONESSENTIAL PERSONNEL

CAUTION

During a security threat it may be advisable **NOT** to sound an alarm.

Sound/have the Control Room sound the Operator Recall Alarm and make the following announcement over the public-address system, TWICE:

A. "Attention all personnel, attention all personnel: An Alert has been declared due to (*brief description of initiating event*). All on-call members of the Emergency Response Organization report to your designated emergency response facility. All Pilgrim Station personnel assemble in your normal office or shop area, report to your supervisor, and await instructions. All visitors, all nonessential contractor personnel, all declared pregnant females, and all persons with disabilities please leave the site at this time."

B. If there is a localized emergency (for example, high radiation, fire), announce its type and location and instruct personnel to stand clear of this area.

C. If there is a potential for an airborne radiological release, consider announcing that there will be no eating, drinking, or smoking until further notice.

Time Completed: ___________

NOTIFICATION OF THE ERO - EMERGENCY FACILITY ACTIVATION

Notify or direct notification of the ERO in accordance with Attachment 10, Activation of the Emergency Response Organization.

Time Completed: ___________

NOTIFICATION OF STATE AND LOCAL AGENCIES

Within 15 minutes of the event classification transmit an Initial Notification Form (Attachment 6) to the Commonwealth and local authorities.

Time Completed: ___________

ALERT (Continued)

NOTIFICATION OF THE NRC
Inform the NRC of the event classification using the ENS or a commercial telephone (Attachment 6 Step 4).
Time Completed: ____________

ACTION VERIFICATION
Ensure Security has evacuated and closed public access areas.
Time Completed: ____________

RETURN TO THE PROCEDURE	**(EITHER STEP 5.2[2], 5.3[2](a), OR 5.4[1])**

SITE AREA EMERGENCY

ASSEMBLY AREA DESIGNATION

Determine the Assembly Area based on meteorological conditions as follows:

Assembly Area	Wind Direction (° from)
☐ Support Building Cafeteria	000°-289° or 324°-360°
☐ Chiltonville Training Center	290°-323°

NOTIFICATION OF SECURITY (IF NOT PREVIOUSLY DONE)

1. Inform Security of the location of the designated Assembly Area and the official declaration time of the Site Area Emergency.
2. Direct Security to ensure that personnel in the Support Building are sent to their assembly area.
3. Direct Security to initiate personnel accountability procedures.
4. Direct Security to verify public access areas are being/have been evacuated.

Time Notified: ____________

SITE & PUBLIC AREA NOTIFICATION - PROTECTED AREA EVACUATION

CAUTION

During a security threat it may be advisable **NOT** to sound an alarm.

Consider radiological conditions when preparing to evacuate personnel. If high dose rates will be encountered, it may be better to shelter non-essential personnel on-site.

Sound/have the Control Room sound the Emergency Site Evacuation Alarm and make the following announcement over the public-address system, TWICE:

A. If entering from no event or an Unusual Event:
"Attention all personnel, attention all personnel: A Site Area Emergency has been declared due to (*brief description of event*). All on-call Emergency Response Organization members report to your designated emergency response facility. All other personnel evacuate to (*Assembly Area*)."

If upgrading from an Alert:
"Attention all personnel, attention all personnel: A Site Area Emergency has been declared due to (*brief description of event*). All personnel who are not part of the on-call Emergency Response Organization evacuate to (*Assembly Area*)."

Continued on next page.

SITE AREA EMERGENCY (Continued)

SITE & PUBLIC AREA NOTIFICATION - PROTECTED AREA EVACUATION (CONTINUED)

B. If there is a localized emergency (for example, high radiation, fire), announce its type and location and instruct personnel to stand clear of this area.

C. If there is a potential for an airborne radiological release, consider announcing that there will be no eating, drinking, or smoking until further notice.

Time Completed: ____________

NOTIFICATION OF THE ERO - EMERGENCY FACILITY ACTIVATION

If not already notified, notify or direct notification of the ERO in accordance with Attachment 10, Activation of the Emergency Response Organization.

Time Completed: ____________

NOTIFICATION OF STATE AND LOCAL AGENCIES

Within 15 minutes of the event classification transmit an Initial Notification Form (Attachment 6) to the Commonwealth and local authorities.

Time Completed: ____________

NOTIFICATION OF THE NRC

Inform the NRC of the event classification using the ENS or a commercial telephone (Attachment 6 Step 4).

Time Completed: ____________

VERIFY ACCOUNTABILITY

Security should report within 30 minutes of declaration of a Site Area Emergency that accountability is complete and provide the names of missing persons, if any. Log the time that accountability was completed.

Time Completed: ____________

ACTION VERIFICATION

Ensure Security has evacuated and closed public access areas.

Time Completed: ____________

RETURN TO THE PROCEDURE **(EITHER STEP 5.2[2], 5.3[2](a), OR 5.4[1])**

GENERAL EMERGENCY

ASSEMBLY AREA DESIGNATION

Determine the Assembly Area based on meteorological conditions as follows:

Assembly Area	Wind Direction (° from)
☐ Support Building Cafeteria	000°-289° or 324°-360°
☐ Chiltonville Training Center	290°-323°

NOTIFICATION OF SECURITY (IF NOT PREVIOUSLY DONE)

1. Inform Security of the location of the designated Assembly Area and the official declaration time of the General Emergency
2. Direct Security to ensure that personnel in the Support Building are sent to their assembly area.
3. Direct Security to initiate personnel accountability procedures.
4. Direct Security to verify public access areas are being/have been evacuated.

Time Notified: ___________

SITE & PUBLIC AREA NOTIFICATION - PROTECTED AREA EVACUATION

CAUTION

During a security threat it may be advisable **NOT** to sound an alarm.

Consider radiological conditions when preparing to evacuate personnel. If high dose rates will be encountered, it may be better to shelter nonessential personnel on-site.

Sound/have the Control Room sound the Emergency Site Evacuation Alarm and make the following announcement over the public-address system, TWICE:

A. If entering into a General Emergency from an Alert or lower: "Attention all personnel, attention all personnel: A General Emergency has been declared due to (*brief description of event*). All on-call members of the Emergency Response Organization report to your designated emergency response facility. All other personnel evacuate to (*Assembly Area*). There will be no eating, drinking, or smoking until further notice."

   If upgrading from an Site Area Emergency: "Attention all personnel, attention all personnel: A General Emergency has been declared due to (*brief description of event*). There will be no eating, drinking, or smoking until further notice."

Continued on next page.

GENERAL EMERGENCY (CONTINUED)

SITE & PUBLIC AREA NOTIFICATION - PROTECTED AREA EVACUATION (CONTINUED)

B. If there is a localized emergency (for example, high radiation, fire), announce its type and location and instruct personnel to stand clear of this area.

Time Completed: ____________

NOTIFICATION OF THE ERO - EMERGENCY FACILITY ACTIVATION

If not already notified, notify or direct notification of the ERO in accordance with Attachment 10, Activation of the Emergency Response Organization.

Time Completed: ____________

NOTIFICATION OF STATE AND LOCAL AGENCIES

Within 15 minutes of the event classification transmit an Initial Notification Form (Attachment 6) to the Commonwealth and local authorities. Protective Action Recommendations issued in accordance with EP-IP-400 are mandatory for a General Emergency classification.

Time Completed: ____________

NOTIFICATION OF THE NRC

Inform the NRC of the event classification using the ENS or a commercial telephone (Attachment 6 Step 4).

Time Completed: ____________

VERIFY ACCOUNTABILITY

If not previously done, Security should report within 30 minutes of declaration of a General Emergency that accountability is complete and provide the names of missing persons, if any. Log the time that accountability was completed.

Time Completed: ____________

ACTION VERIFICATION

Ensure Security has evacuated and closed public access areas.

Time Completed: ____________

RETURN TO THE PROCEDURE **(EITHER STEP 5.2[2] OR 5.3[2](a))**

1.0 REACTOR FUEL

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
1.1 Coolant Activity	**1.1.1.4** Reactor coolant system sample activity which indicates a core melt condition as determined by EP-IP-330, "Core Damage."		**1.1.1.2** Reactor coolant system sample activity > 200 microcuries/ml total iodine.	**1.1.1.1** Reactor coolant system sample activity > 20 microcuries/ml total iodine.
1.2 Off-gas Activity			**1.2.1.2** Air ejector off-gas radiation monitors 1705-3A and B Panel 910 reading > 20,000 mR/hr.	**1.2.1.1** Air ejector off-gas radiation levels approaching Technical Specification release limits as indicated by Air Ejector Off-gas Radiation Monitors 1705-3A and B Panel 910 High-High alarm which does not clear within 13 minutes.
1.3 Thermal Limits				**1.3.1.1** MCPR < Technical Specification fuel cladding integrity safety limit.
1.4 Radiation Monitors	**1.4.1.4** Valid Drywell High Range Area Radiation Monitor reading > 3400 R/hr (RIT-1001-606A and B) or Valid Torus High Range Area Radiation Monitor reading > 100 R/hr (RIT-1001-607A and B) due to degraded reactor fuel integrity conditions.	**1.4.1.3** Valid Drywell High Range Area Radiation Monitor reading > 650 R/hr (RIT-1001-606A and B) or Valid Torus High Range Area Radiation Monitor reading > 4 R/hr (RIT-1001-607A and B) due to degraded reactor fuel integrity conditions.	**1.4.1.2** Refuel Floor Ventilation Exhaust Radiation (RFVE) Monitor 1705-8 Channel A(A/C) and Channel B(B/D) High trip Panel 910. <u>AND</u> Upscale alarm on two or more of the following Area Radiation Monitors Panel 911 which cannot be attributed to a controlled process: • New fuel storage area (#11) • Refueling floor spent fuel pool room (#12) • Refuel floor reactor basin separator (#13) • Refuel floor reactor basin spent fuel area (#14)	

2.0 REACTOR PRESSURE VESSEL

2.1 Reactor Water Level	**2.1.1.4** Sustained RPV water level < -160 inches. <u>AND</u> No sources of RPV injection available. **2.1.2.4** RPV is depressurized and no mechanism of adequate core cooling can be established by injection into the RPV from any source (Primary Containment Flooding is required)	**2.1.1.3** RPV water level cannot be maintained above -125 inches (TAF). <u>AND</u> No sources (EOP 1 Table C Injection Subsystems) of adequate capacity are available to restore water level.	**2.1.1.2** RPV water level cannot be determined (RPV Flooding is required).	**2.1.1.1** RPV water level cannot be restored and maintained above +12 inches.
2.2 Reactor Pressure			**2.2.1.2** Sustained reactor vessel dome pressure > 1325 psig.	**2.2.1.1** Reactor vessel dome pressure cannot be maintained < 1115 psig (except during RPV hydrostatic testing).
2.3 Reactor Power	**2.3.1.4** Reactor power > 3% and torus temperature above the "Boron Injection Initiation Temperature" (BIIT) EOP Figure 4. <u>AND EITHER</u> • An SRV is open. <u>OR</u> • Drywell pressure > 2.2 psig.	**2.3.1.3** Boron injection into the RPV intentionally initiated (boron injection required) with either: • Standby Liquid Control System (SBLC). <u>OR</u> • One or more of the methods detailed in PNPS Procedure 5.3.20, "Alternate Borate Injection."	**2.3.1.2** A reactor Scram has been initiated. <u>AND</u> The reactor is not shutdown.	

3.0 PRIMARY CONTAINMENT

3.1 Drywell Temperature		**3.1.1.3** Bulk drywell temperature cannot be maintained < 280°F as determined by PNPS Procedure 2.1.27, "Drywell Temperature Indication."		**3.1.1.1** Bulk drywell temperature cannot be maintained < 150°F as determined by PNPS Procedure 2.1.27, "Drywell Temperature Indication." <u>AND</u> Primary containment integrity is required.
3.2 Torus water Temperature		**3.2.1.3** Torus water temperature cannot be maintained below the "Heat Capacity Temperature Limit" (HCTL) Figure 3.		**3.2.1.1** Torus water temperature > 110°F.
3.3 Containment Water Level	**3.3.1.4** Primary containment water level cannot be maintained < 77 feet.	**3.3.1.3** Both torus water level and RPV pressure cannot be maintained below the "SRV Tail Pipe Level Limit" (SRVTPLL) EOP Figure 2. **3.3.2.3** Torus water level cannot be maintained > 90 inches.	**3.3.1.2** Torus water level cannot be maintained < 180 inches.	**3.3.1.1** Reactor coolant system to drywell unidentified leakage > 5 gpm with reactor coolant temperature > 212°F. <u>OR</u> Reactor coolant system to drywell total leakage > 25 gpm with reactor coolant temperature > 212°F. **3.3.2.1** Torus water level cannot be maintained: • < 132 inches. (-1 inches. narrow range). <u>OR</u> • > 127 inches (-6 inches. narrow range). <u>AND</u> Primary containment integrity is required.
3.4 Primary Containment Pressure	**3.4.1.4** Torus pressure approaching "The Primary Containment Pressure Limit" (PCPL) EOP Figure 7 (prior to initiation of containment venting).	**3.4.1.3** Torus bottom pressure cannot be maintained below the "Pressure Suppression Pressure" (PSP) EOP Figure 6 (except during testng such as ILRT, etc.).	**3.4.1.2** Primary containment pressure cannot be maintained < 2.2 psig (except during testing such as ILRT, etc.)	
3.5 Containment H_2 & O_2 Concentration	**3.5.1.4** Drywell or torus hydrogen concentration ≥ 6%. <u>AND</u> Drywell or torus oxygen concentration ≥ 5% (Prior to initiation of containment venting). **3.5.2.4** Drywell or torus hydrogen concentration cannot be determined to be < 6%. <u>AND</u> Drywell or torus oxygen concentration cannot be determined to be < 5% (Prior to initiation of containment venting).	**3.5.1.3** Drywell or torus hydrogen concentration > 1%. <u>AND</u> Drywell or torus oxygen concentration > 4%.	**3.5.1.2** Drywell or torus hydrogen concentration > 1%.	

4.0 SECONDARY CONTAINMENT

4.1 Secondary Containment Area Water Levels		4.1.1.3 Secondary containment area water levels exceed the "Maximum Safe Operating Value" in two or more areas EOP Table L. AND A primary system is discharging into the area. AND Reactor coolant temperature > 212°F with irradiated fuel in the vessel.		4.1.1.1 Secondary containment area water levels exceed the "Maximum Safe Operating Value" in two or more areas EOP Table L. AND Reactor coolant temperature > 212°F with irradiated fuel in the vessel.
4.2 Secondary Containment Area Temperatures		4.2.1.3 Secondary containment area temperatures exceed the "Maximum Safe Operating Value" in two or more areas EOP Table L. AND A primary system is discharging into the area. AND Reactor coolant temperature > 212°F with irradiated fuel in the vessel		4.2.1.1 Secondary containment area temperatures exceed the "Maximum Safe Operating Value" in two or more areas EOP Table L. AND Reactor coolant temperature > 212°F with irradiated fuel in the vessel.
4.3 Secondary Containment Area Radiation Levels		4.3.1.3 Secondary containment radiation levels exceed the "Maximum Safe Operating Value" in two or more areas EOP Table L. AND A primary system is discharging into the area.	4.3.1.2 General area radiation levels measured in-plant which increase by a factor of 1000 times normal as a result of airborne radioactivity which cannot be attributed to a controlled process.	

5.0 RADIOACTIVITY RELEASE

5.1 Effluent Monitors	5.1.1.4 Valid Main Stack Process Radiation Monitor 1705-18 A and B Panel 910 reading > 2.4E5 cps.	5.1.1.3 Valid Main Stack Process Radiation Monitor 1705-18 A and B Panel 910 reading > 2.4E4 cps.	5.1.1.2 Valid Reactor Building Ventilation Exhaust (RBVE) monitor 1705-32A and B Panel 910 high-high alarm which does not clear within 15 minutes from the time action is taken to isolate the source. 5.1.2.2 Valid Main Stack Process Radiation Monitor 1705-18A and B Panel 910 reading > 10,000 cps for greater than 15 minutes.	5.1.1.1 Valid Reactor Building Ventilation Exhaust (RBVE) monitor 1705-32A and B Panel 910 high alarm which does not clear within 15 minutes from the time action is taken to isolate the source. 5.1.2.1 Main stack radioactivity effluents in excess of the high-high alarm setpoint for greater than 15 minutes as indicated by the Main Stack Process Radiation Monitors 1705-18A and B Panel 910. 5.1.3.1 Radwaste Effluent Radiation Monitor (1705-30) high alarm Panel 910. AND Radwaste discharge is not isolated (FV-7214-A and B).
5.2 Dose Projection & Environmental Measurements	5.2.1.4 Dose projection based on effluent monitors or actual environmental measurements which indicate doses in excess of 1000 mrem whole body or 5000 mrem thyroid at the site boundary or beyond. 5.2.2.4 Dose projection based on effluent monitors or actual environmental measurements which indicate dose rates in excess of 1000 mrem/hr whole body or 5000 mrem/hr thyroid at the site boundary or beyond.	5.2.1.3 Dose projection based on effluent monitors or actual environmental measurements which indicate doses in excess of 100 mrem whole body or 500 mrem thyroid at the site boundary or beyond.		
5.3 Contaminated Injury				5.3.1.1 Transportation of a contaminated injured person to an offsite medical facility.

6.0 INTERNAL EVENTS

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	UNUSUAL EVENT
6.1 Technical Specifications				**6.1.1.1** The plant is not brought to the required operating mode within Technical Specifications LCO Action Statement time.
6.2 Safety Systems		**6.2.1.3** Evacuation of Control Room without establishment of plant control from remote shutdown stations within 15 minutes. **6.2.2.3** Steam line break outside primary containment without isolation. AND Reactor coolant temperature > 212°F. **6.2.3.3** Inability to immediately isolate any Main Steam Line following a valid PCIS signal (Group I). AND Reactor coolant temperature > 212°F.	**6.2.1.2** Loss of Control Room habitability (e.g., fire, smoke, radiological hazards, etc.). **6.2.2.2** Inability to establish and maintain cold shutdown conditions as indicated by: • Reactor Mode Switch in "Shutdown." AND • Reactor coolant temperature cannot be maintained ≤ 212°F.	**6.2.1.1** Failure of a reactor Safety Relief Valve (RV-203-3A thru D) to close following reduction of applicable pressure as indicated by either: • Accoustic monitor ZI-203 Panel C171. OR • SRV tailpipe temperature recorder TR-260-20 Panel 921. AND Reactor coolant temperature > 212°F with irradiated fuel in the vessel.
6.3 Electrical System Failures		**6.3.1.3** Sustained loss of all AC power capability as indicated by the inability to power any 4kV bus (A1 thru A6) from any source (all 4kV buses de-energized). **6.3.2.3** Sustained loss of all 125 VDC power capability as indicated by: Voltage < 105 VDC Panel D-16. AND Voltage < 105 VDC Panel D-17.	**6.3.1.2** Loss of all AC power capability as indicated by the inability to power any 4kV bus (A1 thru A6) from any source (all 4kV buses de-energized). **6.3.2.2** Loss of all 125 VDC power capability as indicated by: Voltage < 105 VCD Panel D-16. AND Voltage < 105 VDC Panel D-17.	**6.3.1.1** Loss of all vital onsite AC power capability as indicated by: Inability to power Bus A5 from any onsite generator*. AND Inability to power Bus A6 from any onsite generator*. * Onsite generators are: • Main Generator via Unit Auxiliary Transformer • Emergency Diesel Generator A • Emergency Diesel Generator B • Station Blackout Diesel Generator **6.3.2.1** Loss of all vital offsite AC power capability as indicated by: Inability to immediately supply AC power to Bus A5 from any offsite power supply transformer*. AND Inability to to immediately supply AC power to Bus A6 from any offsite power supply transformer*. * Offsite power supply transformers are: • Startup Transformer (X4) • Shutdown Transformer (X13) • Main Transformer with Main Generator phase bus links removed (backscuttle)
6.4 Loss of Indication, Alarm, or Comm Capability			**6.4.1.2** Complete loss of plant process computer alarm and indications. AND Loss of most or all Control Room annunciators. AND Reactor coolant temperature is > 212°F.	**6.4.1.1** Loss of indications and/or alarms which cause a significant loss of assessment capabilities such as: Loss of indication or annunciation on safety-related equipment to an extent requiring shutdown by Technical Specifications. AND Reactor coolant temperature is > 212°F. **6.4.2.1** Loss of all ability to communicate with or adequately activate the Emergency Response Organization as indicated by the complete loss of the onsite telephone systems.

7.0 EXTERNAL EVENTS

7.1 Security Threats	**7.1.1.4** An ongoing security compromise which in the judgment of the Operations Shift Superintendent has led to the loss of physical control of the plant.	**7.1.1.3** An ongoing security compromise which in the judgment of the Operations Shift Superintendent may lead to the loss of physical control of the plant.	**7.1.1.2** Any ongoing security compromise (> 10 minutes) as determined by the Station Security Force.	**7.1.1.1** Any attempted unauthorized entry into the Protected Area as determined by the Station Security Force. **7.1.2.1** Any indication of attempted sabotage. **7.1.3.1** Receipt of a credible bomb threat as determined by the Station Security Force.
7.2 Fire		**7.2.1.3** Any fire which has affected the ability of two or more safety systems (Table 7-1) to perform their intended function and poses a significant potential for release of radioactivity. AND Reactor coolant temperature > 212°F.	**7.2.1.2** Fire burning out of control in a plant vital area.	**7.2.1.1** Fire within the Protected Area lasting > 10 minutes from the time firefighting efforts begin (Fire Brigade fire fighting efforts have begun when the Fire Brigade first applies fire fighting agents on the fire.). OR Any fire onsite for which offsite fire fighting assistance is requested (the required notification of the Plymouth Fire Department for any onsite fire does not constitute a request for offsite fire fighting assistance.).
7.3 Man-made Events		**7.3.1.3** Any of the following which has affected the ability of two or more safety systems (Table 7-1) to perform their intended function and poses a significant potential for release of radioactivity: • Aircraft crash on facility. • Missile impact from any source on facility. • Entry of toxic or flammable gas into a plant process building atmosphere. • Explosion (Unplanned). AND Reactor coolant temperature > 212°F.	**7.3.1.2** Any of the following events occurring which affect plant operation: • Aircraft crash on facility. • Missile impact from any source on facility. • Entry of toxic or flammable gas into a plant process building atmosphere (includes significant Main Generator hydrogen leaks). • Explosion (unplanned).	**7.3.1.1** Any of the following events occurring onsite: • Aircraft crash. • Explosion (unplanned). • Toxic or flammable gas release.
7.4 Natural Events		**7.4.1.3** Any of the following which has affected the ability of two or more safety systems (Table 7-1) to perform their intended function and poses a significant potential for release of radioactivity: • Earthquake • Tornado • Hurricane • Other Natural Phenomena AND Reactor coolant temperature is > 212°F. **7.4.2.3** Any earthquake onsite which has been determined to be greater than Safe Shutdown Earthquake levels (0.15 g).	**7.4.1.2** Any of the following which causes damage to permanent plant structures or equipment which affect plant operation: • Earthquake • Tornado • Hurricane • Other Natural Phenomena **7.4.2.2** Any earthquake onsite which has been determined to be greater than Operating Basis Earthquake levels (0.08 g).	**7.4.1.1** Sustained winds (> 5 minutes) in excess of 75 mph as indicated on wind speed recorder Panel MT1. **7.4.2.1** Report of a tornado onsite. **7.4.3.1** Any earthquake detected by seismic instrumentation as indicated by: • Seismic monitor event alarm Panel C911. OR • "SEISMIC RECORDER OPERATING" annunciator Panel C903 Right B1. AND Ground motion is felt by one or more plant operations personnel.

Table 7-1 Safety Systems
Residual Heat Removal
Core Spray
Emergency Diesel Generators
High Pressure Coolant Injection
Automatic Depressurization System
Reactor Protection System
ATWS/ARI
Primary Containment Isolation System
Standby Gas Treatment
Standby Liquid Control
Reactor Building Closed Cooling Water
Service Water

8.0 OTHER

8.1 Other	**8.1.1.4** In the opinion of the Operations Shift Superintendent or Emergency Director events are in progress which indicate actual or imminent core damage and the potential for a large release of radioactive material outside the site boundary.	**8.1.1.3** In the opinion of the Operations Shift Superintendent or Emergency Director events are in progress which indicate actual or likely failures of plant systems needed to protect the public and pose a significant radioactivity release potential.	**8.1.1.2** Any event which in the opinion of the Operations Shift Superintendent or Emergency Director could or has caused actual substantial degradation of the level of plant safety.	**8.1.1.1** Any event which in the opinion of the Operations Shift Superintendent or Emergency Director could or has led to a potential degradation of the level of safety of the plant. **8.1.2.1** Any event which in the opinion of the Operations Shift Superintendent or Emergency Director warrants the prompt notification of Commonwealth and local authorities and precautionary notification of Emergency Response Organization personnel.

PILGRIM NUCLEAR POWER STATION

INITIAL NOTIFICATION FORM

No. ______

1 THIS IS: ☐ A DRILL ☐ **AN ACTUAL EVENT**

2 AS OF: ______ ______ PILGRIM NUCLEAR POWER STATION HAS:
time date

☐ ENTERED:
☐ HAD A TRANSITORY ______________ THEN ENTERED:
☐ UPGRADED TO:
☐ DOWNGRADED TO:

☐ AN UNUSUAL EVENT
☐ AN ALERT
☐ A SITE AREA EMERGENCY
☐ A GENERAL EMERGENCY
☐ RECOVERY

3 EAL No. ___.___.___.___ BRIEF DESCRIPTION OF EVENT:

____
____
____
____

4 EMERGENCY RADIOACTIVE RELEASE:
☐ **IS** ☐ **IS NOT** ....IN PROGRESS
☐ **IS ABOVE** ☐ **IS BELOW** ....PROTECTIVE ACTION GUIDES

5 METEOROLOGICAL DATA AS OF ________ :
WIND DIRECTION FROM _______ ° TO _______ ° AT _______ mph

6 PNPS's PROTECTIVE ACTION RECOMMENDATIONS (PARs):
☐ NO PROTECTIVE ACTIONS NECESSARY

GENERAL EMERGENCY AND MEMA/MDPH ARE PRESENT IN THE EOF:
☐ PROVIDED TO MEMA/MDPH

GENERAL EMERGENCY AND MEMA/MDPH ARE NOT PRESENT IN THE EOF:
☐ SHELTER SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11
☐ EVACUATE SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11 12 (circle the affected subareas)

7 NOTIFICATION INITIATED ______ ______ BY ______________
time date name

8 NOTIFICATION RECEIVED ______ ______ BY ______________
time date name

INITIAL NOTIFICATION FORM (Cont.)

No.

NOTE

DO NOT TRANSMIT THIS SHEET OVER DNN

9 **APPLICABLE ONLY IN A GENERAL EMERGENCY**
PILGRIM STATION'S PROTECTIVE ACTION RECOMMENDATIONS GIVEN TO MEMA/MDPH REPRESENTATIVES IN THE EMERGENCY OPERATIONS FACILITY (circle the affected subareas):

- ☐ SHELTER SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11
- ☐ EVACUATE SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11 12

10 EMERGENCY DIRECTOR REVIEW AND APPROVAL:

THE EMERGENCY DIRECTOR SHALL REVIEW AND SIGN THESE FORMS INDICATING VERIFICATION OF AND APPROVAL FOR INFORMATION RELEASE AND PROTECTIVE ACTION RECOMMENDATION (if given).

APPROVED FOR RELEASE: ______________________
Emergency Director's Signature

INITIAL NOTIFICATION FORM TRANSMISSION INSTRUCTIONS **NO:** _____

(Check off the boxes as steps are completed.)

Step 1: TRANSMIT THE FORM

- ☐ Verify Blocks 1 - 6 are complete, release is approved by signature, and then fill out Block 7 of the form. *For guidance on individual Block descriptions, refer to EP-IP-100.*
- ☐ Place only the first page of the completed form face down in the DNN fax machine.
- ☐ Follow the posted instructions at the DNN fax machine. *Proceed immediately to Step 2.*

Step 2: CONTACT THE COMMONWEALTH AND LOCAL OFFICIALS

If notification is made on anything other than dedicated communication links (DNN or BECONS), request a verification callback.

- ☐ Pick up the DNN ring-down phone and read the following message twice:
  "Attention, attention. Please standby for a roll call."
- ☐ Perform a roll call and check off the responding locations as they are identified.

☐ Middleboro State Police	☐ Carver	☐ Marshfield
☐ Framingham MEMA	☐ Duxbury	☐ Plymouth
☐ Bridgewater	☐ Kingston	☐ Taunton

- ☐ Read the following message twice:
  "This is the Pilgrim Nuclear Power Station. An Initial Notification Form is being transmitted. Obtain the transmitted form from your telecopy machine or obtain a blank form and standby for initial notification data."

Step 3: TRANSMIT THE DATA

- ☐ Read the information in Block 1 through Block 6.
- ☐ Tell those parties who have complete information to hang up, then provide missing information to remaining parties.
- ☐ Contact locations not answering roll call via BECONS or commercial telephone (speed-dialer) and read the information in Block 1 through Block 6.

Step 4: NOTIFY THE NRC

- ☐ Using the ENS or commercial phone, read the information in Blocks 1 through 6.

  Name of Contact: ______________________________ Time:__________

Step 5: ORGANIZE THE REPORT

- ☐ Obtain the printed report from the DNN fax machine and staple it to this form.
- ☐ Inform the Emergency Director that transmission was completed at time:__________

COMPLETED BY: ______________________________ Time:__________
Signature of person making notifications

PILGRIM NUCLEAR POWER STATION

FOLLOW-UP INFORMATION FORM

No. ☐

1 THIS IS: ☐ A DRILL ☐ **AN ACTUAL EVENT**

2 AS OF: ______ ______ PILGRIM NUCLEAR POWER STATION IS STILL AT A:
time date

☐ UNUSUAL EVENT ☐ ALERT ☐ SITE AREA EMERGENCY ☐ GENERAL EMERGENCY

3 STATION CONDITIONS ARE: ☐ IMPROVING ☐ STABLE ☐ DEGRADING

4 EAL No. ___.___.___.___ BRIEF DESCRIPTION OF EVENT:

5 OUTSIDE ASSISTANCE: ☐ **HAS** ☐ **HAS NOT** BEEN REQUESTED

☐ AMBULANCE

☐ FIRE DEPARTMENT

☐ POLICE

☐ OTHER (Specify)

REASON FOR OUTSIDE ASSISTANCE:

6 EMERGENCY RADIOACTIVE RELEASE:

☐ **IS** ☐ **IS NOT** .... IN PROGRESS

☐ **IS ABOVE** ☐ **IS BELOW** .... PROTECTIVE ACTION GUIDES

7 METEOROLOGICAL DATA AS OF ________:

STABILITY CLASS ____ WIND DIRECTION FROM ______° TO ______° AT ______ mph

8 PNPS's PROTECTIVE ACTION RECOMMENDATIONS (PARs):

☐ NO PROTECTIVE ACTIONS REQUIRED

GENERAL EMERGENCY AND MEMA/MDPH ARE PRESENT IN THE EOF:

☐ PROVIDED TO MEMA/MDPH

GENERAL EMERGENCY AND MEMA/MDPH ARE NOT PRESENT IN THE EOF:

☐ SHELTER SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11

☐ EVACUATE SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11 12 (circle the affected subareas)

9 NOTIFICATION INITIATED ______ ______ BY ____________
time date name

10 NOTIFICATION RECEIVED ______ ______ BY ____________
time date name

FOLLOW-UP INFORMATION FORM (Cont.)

No.

NOTE

DO NOT TRANSMIT THIS SHEET OVER DNN

11 **APPLICABLE ONLY IN A GENERAL EMERGENCY**
PILGRIM STATION'S PROTECTIVE ACTION RECOMMENDATION GIVEN TO MEMA/MDPH REPRESENTATIVES IN THE EMERGENCY OPERATIONS FACILITY (circle the affected subareas):

❑ SHELTER SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11

❑ EVACUATE SUBAREA(s) 1 2 3 4 5 6 7 8 9 10 11 12

SINCE PREVIOUS NOTIFICATION, THIS RECOMMENDATION HAS:
❑ CHANGED ❑ REMAINED THE SAME

THIS RECOMMENDATION IS BASED ON:
❑ PLANT CONDITIONS ❑ PROJECTED RELEASE ❑ ACTUAL RELEASE

12 EMERGENCY DIRECTOR REVIEW AND APPROVAL:

THE EMERGENCY DIRECTOR SHALL REVIEW AND SIGN THIS FORM INDICATING VERIFICATION OF AND APPROVAL FOR INFORMATION RELEASE AND PROTECTIVE ACTION RECOMMENDATION (if given).

APPROVED FOR RELEASE: ______________________
Emergency Director's Signature

FOLLOW-UP INFORMATION FORM TRANSMISSION INSTRUCTIONS

No:______

(Check off the boxes as steps are completed.)

Step 1: TRANSMIT THE FORM

- ☐ Verify Blocks 1 - 8 are complete, release is approved by signature, and then fill out Block 9 of the form. *For guidance on individual Block descriptions, refer to EP-IP-100.*
- ☐ Place only the first page of the completed form face down in the DNN fax machine.
- ☐ Follow the posted instructions at the DNN fax machine. *Proceed immediately to Step 2.*

Step 2: CONTACT THE COMMONWEALTH AND LOCAL OFFICIALS

If notification is made on anything other than dedicated communication links (DNN or BECONS), request a verification callback.

- ☐ Pick up the DNN ring-down phone and read the following message twice:
  "Attention, attention. Please standby for a roll call."
- ☐ Perform a roll call and check off the responding locations as they are identified.

☐ Middleboro State Police	☐ Carver	☐ Marshfield
☐ Framingham MEMA	☐ Duxbury	☐ Plymouth
☐ Bridgewater	☐ Kingston	☐ Taunton

- ☐ Read the following message twice:

"This is the Pilgrim Nuclear Power Station. A Follow-Up Information Form is being transmitted. Obtain the transmitted form from your telecopy machine or obtain a blank form and standby for follow-up information data."

Step 3: TRANSMIT THE DATA

- ☐ Read the information in Block 1 through Block 8.
- ☐ Tell those parties who have complete information to hang up, then provide missing information to remaining parties.
- ☐ Contact locations not answering roll call via BECONS or commercial telephone (speed-dialer) and read the information in Block 1 through Block 8.

Step 4: NOTIFY THE NRC

- ☐ Using the ENS or commercial phone, read the information in Blocks 1 through 8.

Name of Contact: ______________________________ Time:__________

Step 5: ORGANIZE THE REPORT

- ☐ Obtain the printed report from the DNN fax machine and staple it to this form.
- ☐ Inform the Emergency Director that transmission was completed at time:__________

COMPLETED BY: ______________________________ Time:__________
Signature of person making notifications

TERMINATION CHECKLIST

		True	False
1.	Conditions no longer meet an Emergency Action Level and it appears unlikely that conditions will deteriorate. List any EAL(s) which is/are still exceeded and a justification as to why a state of emergency is no longer applicable: EAL Justification __________ __________ __________ __________ __________ __________	☐	☐
2.	Plant releases of radioactive materials to the environment are under control (within Technical Specifications) or have ceased and the potential for an uncontrolled radioactive release is acceptably low.	☐	☐
3.	The radioactive plume has dissipated and plume tracking is no longer required. The only environmental assessment activities in progress are those necessary to determine the extent of deposition resulting from passage of the plume.	☐	☐
4.	In-plant radiation levels are stable or decreasing, and acceptable given the plant conditions.	☐	☐
5.	The reactor is in a stable shutdown condition and long-term core cooling is available.	☐	☐
6.	The integrity of the Reactor Containment Building is within Technical Specifications limits.	☐	☐
7.	The operability and integrity of radioactive waste systems, decontamination facilities, power supplies, electrical equipment, and plant instrumentation including radiation monitoring equipment are acceptable.	☐	☐
8.	Any fire, flood, earthquake, or similar emergency condition or threat to security no longer exists.	☐	☐
9.	Any contaminated, injured person has been treated and/or transported to a medical care facility.	☐	☐

TERMINATION CHECKLIST (CONTINUED)

		True	False
10.	All notifications required by the event procedures have been made.	❑	❑
11.	Offsite conditions do not unreasonably limit access of outside support to the Station and qualified personnel and support services are available.	❑	❑
12.	Discussions have been held with Federal, Commonwealth, and local agencies and agreement has been reached and coordination established to terminate the emergency.	❑	❑

It is not necessary that all responses listed above be "TRUE"; however, all items must be considered prior to event termination and entry into Recovery. For example, it is possible that some conditions remain which exceed an Emergency Action Level following a severe accident but entry into Recovery is appropriate. Additionally, other significant items not included on this list may warrant consideration such as severe weather.

Comments: ______________________________

____

____

____

____

____

____

____

____

____

____

Approved: ______________________________ Date/Time: ______________
Emergency Director

ESSENTIAL INFORMATION CHECKLIST

❑ **Initial Message** ❑ **Update**

Date: ________ Time: ________ am/pm

Person Providing Information:

Name: ________________________

Location: ________________________

Phone: ________________________

1) Classification or Status

Time Declared: ________ am/pm

❑ Off-Normal Event
❑ Unusual Event
❑ Alert
❑ Site Area Emergency
❑ General Emergency
❑ Recovery

2) Brief Summary of Event and Mitigating Actions in Progress

____
____
____
____
____
____
____
____
____

3) Plant Conditions

❑ On-Line ❑ Off-Line
❑ Stable ❑ Unstable
❑ At Power ❑ Cooling Down
❑ Cold Shutdown

Time of Reactor Shutdown: ________ am/pm

❑ Improving ❑ Same ❑ Deteriorating

Describe equipment, instrument, or other problems:

____
____
____
____
____
____

4) Radiological Conditions

Emergency Radioactive Release Status:

❑ None ❑ Imminent ❑ In Progress
❑ Controlled ❑ Uncontrolled
❑ Below PAGs ❑ Above PAGs

Projected Release Duration: ________ hr(s)

Offsite Protective Actions:

❑ None Issued
❑ Provided to Commonwealth

Onsite Protective Actions:

❑ None
❑ Potassium Iodide
❑ Evacuation of Nonessential Personnel
❑ Other: ________________________

5) Site Personnel

Injuries: How Many_____ ❑ No ❑ Yes
Contamination: ❑ No ❑ Yes
Overexposure: ❑ No ❑ Yes
❑ Minor ❑ Major

Nature of injuries or contamination:

____
____
____
____

6) Offsite Agency Notification

NRC	Time: ________	am/pm
MEMA	Time: ________	am/pm
MDPH	Time: ________	am/pm
Local (last town)	Time: ________	am/pm

7) Offsite Assistance Requested

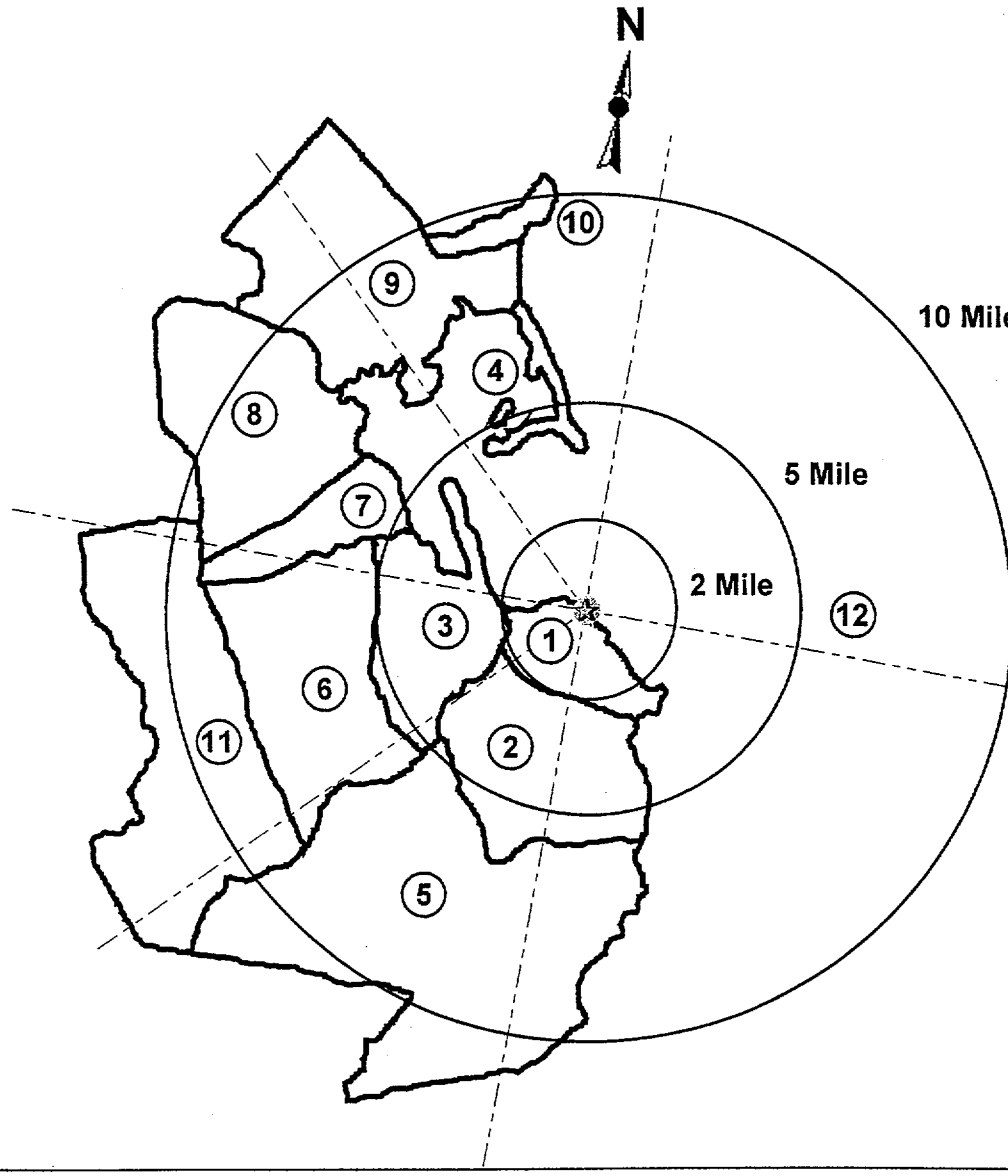
❑ None ❑ Fire Department
❑ Police ❑ Ambulance
❑ Other: ________________________

8) News Media Notification

❑ None
❑ Draft News Release
❑ News Release Approved
❑ Press Conference

9) Signature and Title

ESSENTIAL INFORMATION CHECKLIST



Additional Information:

ACTIVATION OF THE EMERGENCY RESPONSE ORGANIZATION

I. Emergency Response Organization Notification Using CANS

NOTE

If at any time CANS cannot be contacted or does not respond as indicated, go to Section II (Notification Using Group Pages if CANS Fails) in this Attachment.

If the individual activating CANS is not the Operations Shift Superintendent, Control Room Supervisor, Shift Control Room Engineer, or the Operations Assistant, it is necessary for the individual to obtain a security code (Social Security number) from one of these individuals.

[1] Ascertain the current emergency classification from the Emergency Director.

[2] Obtain the phone number for CANS from the Immediate Notification section of the PNPS Emergency Telephone Directory.

[3] Attempt to contact CANS using any touch-tone phone line.

[4] Listen for the introductory message and enter your security code (Social Security number) upon request followed by the # sign.

NOTE

For activation of the ERO for drills and exercise, precede the following classification codes with DRILL (37455 on telephone keypad). For example, for a drill classification of Alert, enter DRILL 2222 (374552222) as the emergency classification code.

[5] Listen for verbal prompt and enter the proper code for present emergency classification. The codes are:

- 1111 Unusual Event
- 2222 Alert
- 3333 Site Area Emergency
- 4444 General Emergency
- 0000 Termination of classifiable event/Recovery

[6] If you are NOT satisfied with the emergency classification, press the # sign to re-enter classification.

OR

If you are satisfied, hang up the phone to start the notification process.

[7] Verify CANS operability by checking the emergency pager located in the locker in the Operations Shift Superintendent's office. Ensure that the pager is activated and indicates the proper emergency classification. Allow approximately 5 minutes for CANS to activate the pager.

NOTE

If communication is established with the EOF prior to initiation of Step [8], Step [8] may be omitted.

[8] If the Emergency Operations Facility has not contacted the Control Room after approximately 35 to 40 minutes from the time the Control Room pager has activated, check on-call personnel response in the following manner:

(a) Obtain the CANS phone number from the Immediate Notification section of the PNPS Emergency Telephone Directory.

(b) Contact CANS using any available touch-tone phone. If no contact is made, pause and try again.

(c) Listen for the introductory message and enter your security code (Social Security number) when requested followed by the # sign.

(d) When prompted, enter '3' for the status of the classification scenario.

(e) Retrieve the CANS report from the DNN facsimile machine and review the list for unfilled positions.

(f) Repeat Steps (a) through (e) above as often as necessary to keep apprised of the notification process.

II. Notification Using Group Pages if CANS Fails

NOTE

If contact with the paging system cannot be made or the paging system does not respond as indicated, go to Section III (Notification Using Telephones if Both CANS and the Paging System Fail) of this Attachment.

[1] Obtain the ERO pager PIN and the access telephone number from the Immediate Notification section of the PNPS Emergency Telephone Directory.

[2] Contact the paging system by dialing the access telephone number.

[3] Upon contact, listen for the verbal prompt to enter the PIN number. When prompted, enter the ERO pager PIN followed by the # sign. If the verbal prompt is not heard, repeat Step [2].

[4] Listen for the verbal prompt to enter display message. When prompted, enter the proper code for present emergency classification. The codes are:

- 1111 Unusual Event
- 2222 Alert
- 3333 Site Area Emergency
- 4444 General Emergency
- 0000 Termination of classifiable event/Recovery

[5] Press the # sign to complete the entry.

III. Notification Using Telephones if Both CANS and the Paging System Fail

[1] Request the Emergency Director to inform Security at the Primary Access Control Point to notify the Emergency Response Organization using the PNPS Emergency Telephone Directory and commercial telephone lines in accordance with EP-IP-240.

[2] Call out one individual for each of the following emergency positions using the PNPS Emergency Telephone Directory. When contact is made, ask if any alcoholic beverage has been consumed within the last 5 hours. If the individual answers YES, inform them that an alternate shall be contacted. If the individual answers NO, inform them of the event and that CANS and the pager system have failed. Record the name and response time below.

Emergency Director (EOF)	
______________________	____________
Name	Response Time (min.)

Emergency Plant Manager (TSC/OSC)	
______________________	____________
Name	Response Time (min.)

Emergency Offsite Manager (EOF)	
______________________	____________
Name	Response Time (min.)

DOCUMENT CROSS-REFERENCES

This Attachment lists those documents, other than source documents, which may be affected by changes to this Procedure.

Document Number	Document Title
EP-IP-400	Protective Action Recommendations
EP-IP-520	Transition and Recovery
EP-AD-600*	PNPS EAL Technical Bases Document
PNPS 2.1.27	Drywell Temperature Indication
PNPS 5.3.20	Alternate Borate Injection
----	EAL Wall Chart

* Any revision to Attachment 5 of EP-IP-100 shall require a corresponding revision to EP-AD-600.

IDENTIFICATION OF COMMITMENTS

This Attachment lists those external commitments (i.e., NRC commitments, QAA audit findings, and INPO inspection items) implemented in this Procedure.

Reference Document	Commitment	Affected Section(s)/Step(s)
NRC Inspection Finding 81-15-34	Develop and implement a system for use by the Control Room staff to aid in promptly classifying events.	Att. 5, EAL Chart
NRC Inspection Finding 81-15-35	Provide EALs which include specific and observable Control Room instrument readings for each EAL corresponding to the respective initiating condition.	Att. 5
NRC Inspection Finding 81-15-35, 84-41-02, 86-33-01	Provide EALs which address and conform to all pertinent initiating conditions contained in Appendix 1 of NUREG-0654.	Att. 5
NRC Inspecting Finding 81-15-36	Revise the offsite notification procedures to specify protective action recommendations in the notification messages.	Att. 6
NRC Inspecting Finding 81-15-41	Revise the communication procedures to ensure (verify) correct transmission.	Att. 6 & 7
NRC Inspection Finding 84-05-04	Provide EALs based on field monitoring results and on the methods used if the effluent and containment monitors are inoperable or off-scale.	Att. 5 (Category 5.2)
NRC Inspection Finding 84-41-01	Revise the EAL on hurricane wind speed to reflect the National Weather Service definition for hurricanes.	Att. 5 (EAL 7.4.1.1)
NRC Inspecting Finding 84-41-06	Review the offsite notification procedure and make appropriate changes to provide a high probability of completing offsite notification.	Att 6 & 7
QAA Audit Report 87-48 DR 1723, Issue 1	Include in the EALs all initiating conditions in Appendix 1 of NUREG-0654.	Att. 5
QAA Audit Report 87-48 DR 1723, Item 4	Ensure public access areas are closed at an Alert or above.	Att. 2, 3, 4
NRC Inspection Report 50-293/88-28 Item 2.4 and 7.4.2.3	Provide further clarification and quantification for earthquake EALs.	Att. 5 (EAL 7.4.2.2)
	Modifications to transmitted portions of Initial Notification and Follow-Up Information Form shall be reviewed with the Commonwealth.	

Beginning
Of
Document

RTYPE H8.24

PILGRIM NUCLEAR POWER STATION

Procedure No. EP-IP-330

CORE DAMAGE

STAR

Stop
Think
Act
Review

SAFETY RELATED

REVISION LOG

REVISION 4	**Date Originated 11/01**
Pages Affected	Description
5	Update References.
50-52	Incorporate revised Torus curves for melt, gap, and spike conditions due to relocation of Torus CHRMS detectors.

REVISION 3	**Date Originated 2/01**
Pages Affected	Description
All	Reformat IAW PNPS 1.3.4-1. Revision bars are not shown for reformatting.
5,15,54	Update reference documents.
6,15	Update/revise organization titles.
7,10,17,24-26,28, 35,41,46	Revise terminology of "cladding failure" to be synonymous with "gap" activity.
12,13	Update source term scenarios for both Drywell and Torus CHRMS curves.
13,18,19,30-33	Revise 'Damage' computer window screens and instructions to separate Drywell and Torus curves for melt, gap and spike conditions.
14	Revise step to clarify nomogram usage.
37,38,43,44	Add legends for I-131, Cs137, Xe-133, and Kr-85 PASS graphs.
47-49	Incorporate revised Drywell CHRMs curves for melt, gap, and spike conditions.
50-52	Incorporate revised Torus curves for melt, gap, and spike conditions.

TABLE OF CONTENTS

TABLE OF CONTENTS

1.0 PURPOSE

The purpose of this core damage assessment Procedure is to provide guidance and direction for the methods used to determine the extent of core damage and to summarize all the available information into an overall best estimation which reflects the accident profile.

2.0 REFERENCES

[1] BWR Owners Group Letter BWROG-8324 dated June 17, 1983

[2] General Electric Source Document NEDO-22215, "Procedures for the Determination of Core Damage Under Accident Conditions"

[3] EP File 1.6.4 'DAMAGE' version 3.0 Computer Application Verification, Validation and Documentation"

[4] EP File 1.6.4 'DAMAGE' version 4.0 Computer Application Verification, Validation and Documentation

[5] Entergy Calculation # PNPS-1-ERHS-XII.E-11, Rev. 0, dated 12/21/00, entitled, "Post-Accident Torus CHRMS Response

[6] Entergy Calculation # PNPS-1-ERHS-XII.E-12, Rev. 0, dated 4/11/01, entitled, "Post-Accident Drywell CHRMS Response"

[7] Entergy Calculation # PNPS-1-ERHS-XIV.B-P103-EC20, Rev. 0, dated 3/1/82, entitled, "A Nomogram for Estimating Reactor Core Damage Using As A Basis The Radiation Levels From A Beaker Containing Diluted Primary Coolant"

[8] PNPS 7.11.4, *"Radioisotopic Analysis of Liquid Samples Under Accident Conditions"*

[9] PNPS 7.11.5, *"Radioisotopic Analysis of Gas Samples Under Accident Conditions"*

3.0 DEFINITIONS

None

4.0 DISCUSSION

None

5.0 RESPONSIBILITIES

[1] The Engineering Coordinator (Systems) is responsible for developing and submitting the summary core damage report.

[2] The Core Damage Engineer is responsible for completing the overall estimation of core damage in accordance with this Procedure.

[3] The Reactor Engineer is responsible for obtaining Reactor power historical information and providing it to the Core Damage Engineer and for aiding the Core Damage Engineer in completing core damage calculations in accordance with this Procedure.

[4] The Chemistry Coordinator is responsible for providing information for estimations based on sample analysis data.

[5] The Radiation Protection Engineer is responsible for aiding the Core Damage Engineer in completing core damage calculations in accordance with this Procedure.

6.0 PROCEDURE

6.1 ESTIMATIONS OF THE EXTENT OF CORE DAMAGE

An overall estimation of the extent of core damage can be made when information accumulated from all available sources and methods is evaluated. The NRC defines the overall condition of the core using a matrix of 10 categories as follows:

Degree of Degradation	Minor (< 10%)	Intermediate (10% - 50%)	Major (> 50%)
No Fuel Damage	1	1	1
Cladding Failure (Gap)	2	3	4
Fuel Overheat	5	6	7
Fuel Melt	8	9	10

NRC Damage Categories

The NRC recognizes four general classifications with three degrees of core damage within each (excepting the 'No Fuel Damage classification'). It is important to recognize that different methodologies may provide indications that point to several degrees if not several classifications simultaneously.

[1] There are several methods and indications which can be used to estimate the amount or type of core damage under accident conditions which include:

(a) PASS Sample Analysis: A direct method that can yield accurate numerical estimations. Applicable for all types of accidents. Requires the sampled system(s) be in a steady state that usually prevents its use until the plant is in a stable shutdown condition. (See Section 6.2.)

(b) PASS Isotopic Ratio Comparison: A direct method that is used to help establish the type of core damage (clad failure or fuel melt). Applicable under all types of accidents. Valid any time following an accident although accuracy will decrease over time due to the relatively short half-lives of the isotopes used. (See Section 6.3.)

(c) Presence of Abnormal Isotopes: A direct method that is used to indicate some degree of fuel melt by the presence of unusually high concentrations of any of the less volatile fission products. (See Section 6.4.)

(d) Containment Hydrogen Concentration: An indirect method that is used to determine the amount of clad failure (gap). Assumes all the hydrogen generated by the metal-water reaction is released into containment. (See Section 6.5.)

(e) Containment Radiation Levels: An indirect method that is used to determine the amount of core damage. Based upon an end-of-life source term and static nuclide ratio assumptions. (See Section 6.6.)

(f) Radiation Levels From a Beaker of Primary Coolant: An indirect method which is used to determine the amount of fuel melt by relating dose rate measured from a Reactor coolant sample to degree of core damage using a nomogram. (See Section 6.7.)

(g) Other Plant Parameters: An indirect method that is immediately available and is used to indicate the potential for core damage. Applicable for all types of accidents. Cannot provide tangible numerical estimations but rather can be used as a yes/no/maybe indicator or as confirmation for other methods. (See Section 6.8.)

[2] Precise damage estimates are based upon accounting for all of the radioactivity released from the core. Methods that provide a numerical estimation of the extent of core damage should be evaluated to ensure all activity has been accounted for.

[3] An overall estimation of the extent of core damage should be provided to the Emergency Director through the TSC and Emergency Plant Manager whenever information becomes available or is revised throughout the course of an accident. Attachment 1 is used to summarize the overall damage estimation.

6.2 CORE DAMAGE ESTIMATION USING PASS SAMPLE ANALYSIS

The information provided in Steps 6.2[1] through 6.2[3] should be reviewed prior to determining the PASS sample location.

[1] Sample Points

(a) Activity should be accounted for in each location within each containment system (liquid or gaseous) either directly or indirectly. The most accurate results are obtained when both locations are used for the damage estimation (such as both Reactor coolant and Torus samples for liquid system estimations).

(b) The initial sample should be taken from the plant location where a majority of the activity released from the core is expected to reside. It is extremely important to correctly choose the appropriate PASS location when estimations are made using only one sample point. For example, if an accident involving core damage occurs with the activity primarily contained in the Reactor coolant system, a sample of the containment atmosphere would greatly underestimate the amount of damage.

(c) The results from liquid and gaseous system analysis should indicate a similar amount of damage for accidents in which the source is distributed throughout all locations.

NOTE

Errors introduced by incorrect determination of equilibrium can be up to a factor of 10 for liquid systems and a factor of 2 for containment gases.

[2] System Equilibrium

(a) The determination of the status of system equilibrium must be made whenever sample data is available from only a single source.

(b) When systems are considered to be at equilibrium, the activity present in the sample is assumed to be present in each location. For example, the activity measured in a Reactor coolant sample would be assumed to be present, in the same concentration, in the Torus.

(c) When systems are not considered to be at equilibrium, the sampled location is assumed to contain the vast majority of activity released from the core. For example, in the case of an accident involving core damage but no loss of coolant or blowdown to the Torus, the activity measured in Reactor coolant would be assumed to be all the activity released with little, if any, activity expected in the Torus liquid.

[3] Power History Corrections

(a) Each system has two isotopes that can be used to determine damage estimates: I-131 and Cs-137 in liquid, and Xe-133 and Kr-85 in gaseous containment. The short-lived isotopes, I-131 and Xe-133, will tend to be predominant up to several weeks following an accident. Once the shorter-lived isotopes have decayed, Cs-137 and Kr-85 become the isotopes of interest.

(b) When short-lived isotopes are used in damage estimations, approximately six half-lives (approximately 50 days) worth of power history data should be provided to adequately account for buildup and decay.

(c) Power history for long-lived isotopes should account for the entire time the fuel has been in the core.

(d) Variation in power over any single operating period will have less effect on calculations using long-lived isotopes.

(e) Variations in steady-state power should be limited to $\pm$20% within each period.

[4] Performing Damage Estimations Using PASS Sample Analysis Results

(a) Once a PASS sample has been taken and analyzed, core damage can be estimated using the core damage computer application (the user guide is provided in Attachment 2) or by using the appropriate worksheets (Attachment 3 or 4).

(b) The worksheets are designed to estimate core damage using results from only one isotope. However, the worksheets do allow multiple liquid or multiple gas sample points to be used simultaneously, provided both sample points use the same isotope. The steps of the worksheet should be followed in numerical order.

(c) Three damage estimates are obtained when reading the graphs provided in Attachments 3 and 4 and in the DAMAGE computer code; an upper release estimate, a lower release estimate, and a best estimate of core damage. Other factors should be considered if the activity is such that there is an overlap between cladding failure (gap) and fuel melt, such as the presence of actual fuel melt isotopes and isotopic ratios of selected fission products.

6.3 CORE DAMAGE ESTIMATION USING PASS ISOTOPIC RATIO ANALYSIS

[1] Core damage estimations are based on the relationship between nuclides in a sample. Any sample location that is expected to contain a representative mix can be used. Sample locations that have not been affected by a removal process (such as filtration, ion exchange, washout, or plateout) should be chosen.

[2] The isotopes used in the ratio analysis are relatively short-lived. The estimate accuracy is reduced when samples are taken more than several days after the accident.

[3] Once a PASS sample has been taken and analyzed, the type of core damage can be estimated using the core damage computer application (the user guide is provided in Attachment 2) or by using the Attachment 5 worksheet.

6.4 CORE DAMAGE ESTIMATION USING THE PRESENCE OF ABNORMAL ISOTOPES

[1] Unusually high concentrations of any of the less volatile fission products are indicative of some degree of fuel melt. These fission products may include soluble or insoluble isotopes of the following elements:

- Alkaline Earths: Sr, Ba
- Noble Metals: Ru, Rh, Pd, Mo, Tc
- Rare Earths: Y, La, Ce, Nd, Pr, Eu, Pm, Sm, Np, Pu
- Refractories: Zr, Nb

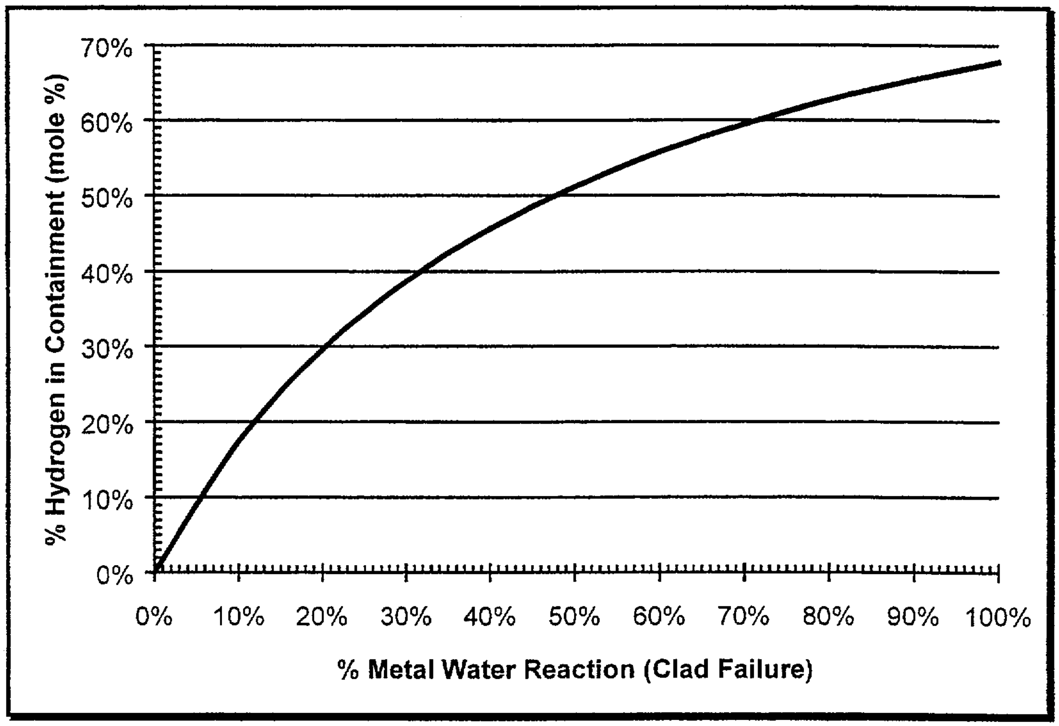
[2] Historical Reactor power operations and chemistry data should be used as a basis in determining the threshold that constitutes unusually high.

6.5 CORE DAMAGE ESTIMATION USING CONTAINMENT HYDROGEN CONCENTRATION

[1] Estimations using hydrogen concentration in containment are an indirect method of determining the extent of core damage. The fundamental assumption requires that the source released from the core be present within the containment. This method is not valid and should not be used for accidents that do not involve a LOCA.

[2] Obtain the highest credible containment hydrogen readings from both Drywell and Torus monitors.

[3] The amount of clad failure can be estimated using the core damage computer application (the user guide is provided in Attachment 2) or by using the graphic relationship below:



Containment Hydrogen Concentration

6.6 CORE DAMAGE ESTIMATION USING CONTAINMENT RADIATION LEVELS

[1] Estimations using high levels of radiation in containment are an indirect method of determining the extent of core damage.

[2] The drywell and torus CHRM curves can be used for any accident condition within PNPS design basis space (i.e., the curves are not applicable to ATWS/SAG beyond design basis events) as long as the monitors are responding within their established range. The curves also assume radioactive material is released to the drywell atmosphere, torus atmosphere, torus water, and drywell/torus piping that circulate primary coolant fluids. The purpose of these curves is to provide early estimates of core degradation <u>quantitatively</u> as well as <u>qualitatively</u> prior to taking PASS samples and analyzing them for more accurate results.

[3] The drywell and torus CHRM curves are independent of break size, amount and type of primary fluid released to the primary containment (i.e., drywell atmosphere, torus atmosphere, and torus water), or the mechanism by which the activity was transferred to the primary containment.

[4] The drywell and torus CHRM curves are not based on volume or mass of primary fluids released to the primary containment, but rather on amounts of radioactive material.

[5] The drywell and torus CHRM curves were generated by choosing various source terms (ranging from a full core melt releasing the TID-14844 source term to the release of radioactivity present in a spiked primary coolant with no core damage) and dispersing them in the drywell atmosphere and torus atmosphere/water volumes. It should be noted that the time of the accident equals the time of reactor Scram/shutdown (i.e., t = 0 hrs). The following source term scenarios were used to generate the drywell and torus CHRM curves.

(a) Full Core Melt - This would correspond to a typical worst case DBA LOCA scenario releasing conservative core damage estimates as specified in TID-14844 that assumes releasing 100% core noble gases, 50% core halogens, and 1% of the remaining core fission products (solids) from the core inventory. From this source term assumption, monitor response curves for both drywell and torus are plotted showing the detector dose rate response versus time after shutdown corresponding to 100%, 10%, and 1% of the assumed source term for fuel core melt conditions.

(b) Gap Activity Release - This would correspond to an accident that would cause a breach of 100% of the core fuel cladding from the core inventory. The gap space is assumed to contain, for design basis accident purposes, 10% of the core noble gases and 10% of the core halogens. From this source term assumption, monitor response curves for both drywell and torus are plotted showing the detector dose rate response versus time after shutdown corresponding to 100%, 10%, and 1% of the assumed source term for gap activity released conditions.

(c) Spiked Primary Coolant Release - This would correspond to releasing the entire inventory of primary coolant activity (present under full power normal operating conditions) with the total iodine levels spiked to Technical Specifications values (20 μCi/ml of total iodine). From this source term assumption, monitor response curves for both drywell and and torus are plotted showing the detector dose rate response versus time after shutdown corresponding to the total iodine Technical Specifications values and 10 times the total iodine Technical Specifications values (i.e., 200 μCi/ml of total iodine).

[6] The drywell/torus CHRM will respond not only to drywell/torus atmosphere airborne activity and torus water activity, but also to radiation fields being generated by piping in the vicinity of the monitors that may also be carrying core damage releases.

[7] For estimations using drywell radiation levels, obtain the highest credible monitor reading using the RIT-1001-606A and B monitors.

[8] For estimations using torus radiation levels, obtain the highest credible monitor reading using the RIT-1001-607A and B monitors.

NOTE

Core damage of 10% gap activity or greater may overlap in the fuel melt condition graph for core damage estimates. If the core was not uncovered and the fuel did not reach fuel melt temperature, then, in all likelihood, one would expect no more than fractions of gap releases rather than fractions of core melt.

[9] The amount of core damage can be estimated using the core damage computer application (the user guide is provided in Attachment 2) or by using the graphic relationships provided in the applicable Attachment (Attachments 6 and 7).

6.7 CORE DAMAGE ESTIMATION USING RADIATION LEVELS FROM A BEAKER OF PRIMARY COOLANT

[1] Estimates using the radiation levels from a beaker of primary coolant are an indirect method of determining the extent of core melt damage. This method uses a nomogram to estimate the degree of core melt damage based on radiation levels emitted by fission product nuclides assumed to be present in the coolant after an accident.

[2] This method requires a 200ml sample of Reactor coolant in a 250ml beaker. The coolant may be diluted by the factors listed on the nomogram to reduce dose rates to meet the nomogram limits and/or for ALARA considerations.

[3] The following information is required when using this method of core damage assessment:

(a) Survey instrument reading. The reading must be from 1 mR/hr to 1000 R/hr.

(b) Time since Reactor Scram. This is the time since the Reactor was placed in a shutdown condition (from 0 to 1000 hours).

(c) Distance from surface of beaker to survey meter. The distance should coincide with one given on the nomogram (0.5 inches to 36 inches).

(d) Initial Dilution Volume. This refers to the volume in which the fission products are dispersed (in cubic feet). For example, if the accident is such that all the fission products are contained in the Reactor Vessel and recirculation system piping (no leakage from the RCS), then the initial dilution volume would be 10,000 ft^3.

(e) Additional sample dilution. This is the dilution factor by which the actual sample was diluted before the measurement was obtained. A factor of 1 means no dilution was performed.

[4] Once the information listed in Step 6.7[3] has been obtained, use the nomogram (Attachment 8) and proceed as follows:

(a) Locate the survey instrument reading on the left side Y-axis.

(b) Move horizontally to the right until the line corresponding to the proper "TIME SINCE REACTOR SCRAM" is intercepted. (Interpolate if necessary.)

(c) At the interception point in Step [4](b) above, reflect upward until the line corresponding to the proper "DISTANCE FROM SURFACE OF BEAKER TO SURVEY METER" is intercepted.

(d) Now, reflect horizontally to the right to the correct "INITIAL DILUTION VOLUME" until the line is intercepted.

(e) At this point, reflect downward to the correct "ADDITIONAL SAMPLE DILUTION" line.

(f) Finally, reflect horizontally to the right and read percent core damage.

6.8 CORE DAMAGE ESTIMATION USING OTHER PLANT PARAMETERS

[1] Core damage estimations from plant parameters include indications such as core uncovery time, Reactor water level, peak power, temperature and pressure, and process and area radiation monitors.

[2] Plant operating parameters are usually the first type of information available for core damage evaluation. Generally, they are unable to provide a tangible numerical value but can help determine whether the severity of the accident involves actual core damage.

7.0 RECORDS

[1] This Procedure generates the following documents:

(a) Core Damage Assessment Summary Report

(b) Core Damage Worksheet: Pass Liquid Samples

(c) Core Damage Worksheet: Pass Gaseous Samples

(d) Core Damage Worksheet: Pass Isotopic Ratio Analysis

(e) Core Damage Estimate Using Drywell/Torus CHRMS Curves

[2] All completed documents shall be submitted to the Emergency Plant Manager who will review and submit them to the Nuclear Assessment Director.

8.0 ATTACHMENTS

ATTACHMENT 1 - CORE DAMAGE ASSESSMENT SUMMARY REPORT

ATTACHMENT 2 - 'DAMAGE' COMPUTER APPLICATION USER GUIDE

ATTACHMENT 3 - CORE DAMAGE WORKSHEET: PASS LIQUID SAMPLES

ATTACHMENT 4 - CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES

ATTACHMENT 5 - CORE DAMAGE WORKSHEET: PASS ISOTOPIC RATIOS

ATTACHMENT 6 - CORE DAMAGE CURVES: DRYWELL CHRMS

ATTACHMENT 7 - CORE DAMAGE CURVES: TORUS CHRMS

ATTACHMENT 8 - CORE DAMAGE NOMOGRAM: PRIMARY COOLANT SAMPLE

ATTACHMENT 9 - DOCUMENT CROSS-REFERENCE

ATTACHMENT 10 - IDENTIFICATION OF COMMITMENTS

CORE DAMAGE ASSESSMENT SUMMARY REPORT

Method	**Best Estimate Clad (Gap)**	**Melt**
Containment Atmosphere PASS Sample Analysis Location: ❑ Drywell ❑ Torus ❑ Both		
Liquid System PASS Sample Analysis Location: ❑ Reactor Coolant ❑ Torus ❑ Both		
PASS Isotopic Ratio Analysis		
Presence of Abnormal Isotopes		
Containment Hydrogen Concentration Analysis*		
Drywell Radiation Level Analysis		
Torus Radiation Level Analysis*		
Reactor Coolant Sample Radiation Level Analysis		
Indications From Other Plant Parameters		

Overall Assessment of the Extent of Core Damage

Damage Type: ❑ No Damage ❑ Clad Failure (Gap) ❑ Fuel Melt Amount: ___________

NRC Core Condition Category: ______________________ (See base document Section 6.1)

Comments:

____

____

____

____

* **Use of this method during accidents other than LOCA gives qualitative information only. It should NOT be used for quantitative assessment except in the case of a LOCA.**

Analyst: ________________________________ Date: _________ Time: _________

Report Results to:

❑ Engineering Coordinator-Systems
❑ Emergency Plant Manager
❑ Emergency Director
❑ TSC Supervisor
❑ Offsite Rad Supervisor
❑ Operations Advisor

'DAMAGE' COMPUTER APPLICATION USER GUIDE

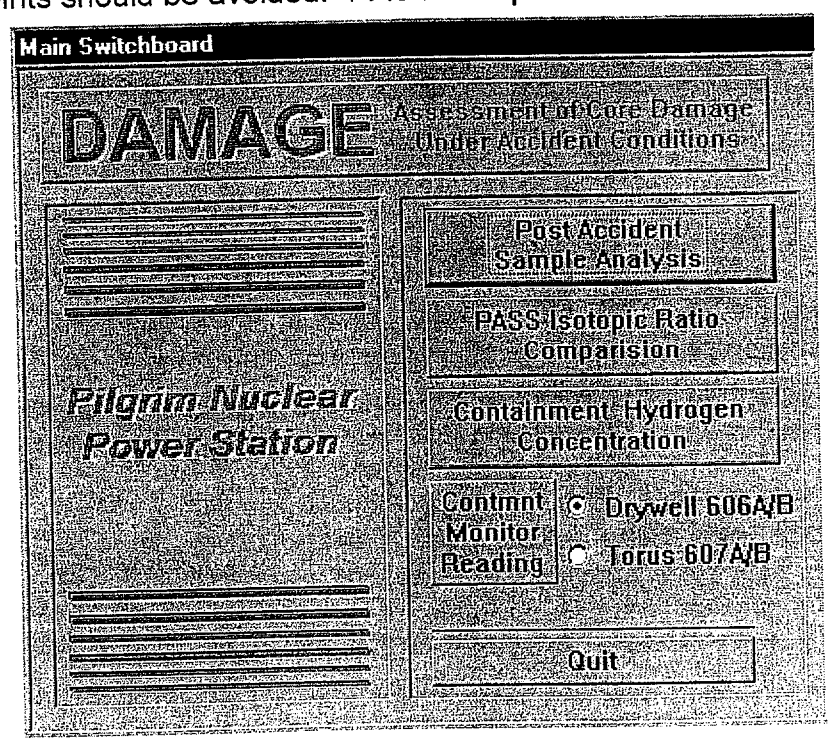
This user guide contains detailed information for operating and understanding the DAMAGE computer application. It is not required nor intended to be used as a step-by-step instruction when using DAMAGE; however, the user must be familiar with basic computer operations within the Microsoft Windows environment in order to operate the application.

A brief description of the purpose and an illustration of each of the application's windows are provided by this user guide. Since users are not constrained to operate the application in any step order, a table is used to describe information and functions for all the objects (such as buttons, fields and graphs) contained in DAMAGE. Information and functions include descriptions, options, units, and limits as applicable.

A menu bar is provided to allow the user to quit the program from any point. Additionally, the application version and serial number can be obtained from the menu bar. A description of the function for each control is provided at the bottom of the application window (within the status bar) whenever the control has the focus.

The Main Switchboard Window

Upon startup, DAMAGE opens to the main switchboard window. DAMAGE offers four separate methods to aid in the overall determination of the extent of core damage. PASS sample analysis will normally provide the most accurate assessment but requires a stable, steady state condition to be valid. Taking samples while the activity released from the core is still circulating between the sample points should be avoided. PASS isotopic ratio comparisons are valid as soon as a sample can be taken following an accident but can only provide an indication of the type of core damage (clad failure or fuel melt). Isotopes utilized by the ratio comparison method are relatively short lived. This method will lose accuracy over time and should be avoided for periods later than about seven days following the accident. Containment radiation or hydrogen concentrations can also provide the early assessments of the amount of core damage but assume a loss of coolant accident has occurred. No single method should be relied upon for a definitive damage estimation. All available data and sound engineering principles should be used to compile the best overall estimation.



'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

The main switchboard window allows for the selection of and switching between the desired evaluation methods. A description of the information provided within the window and for control options and inputs is as follows:

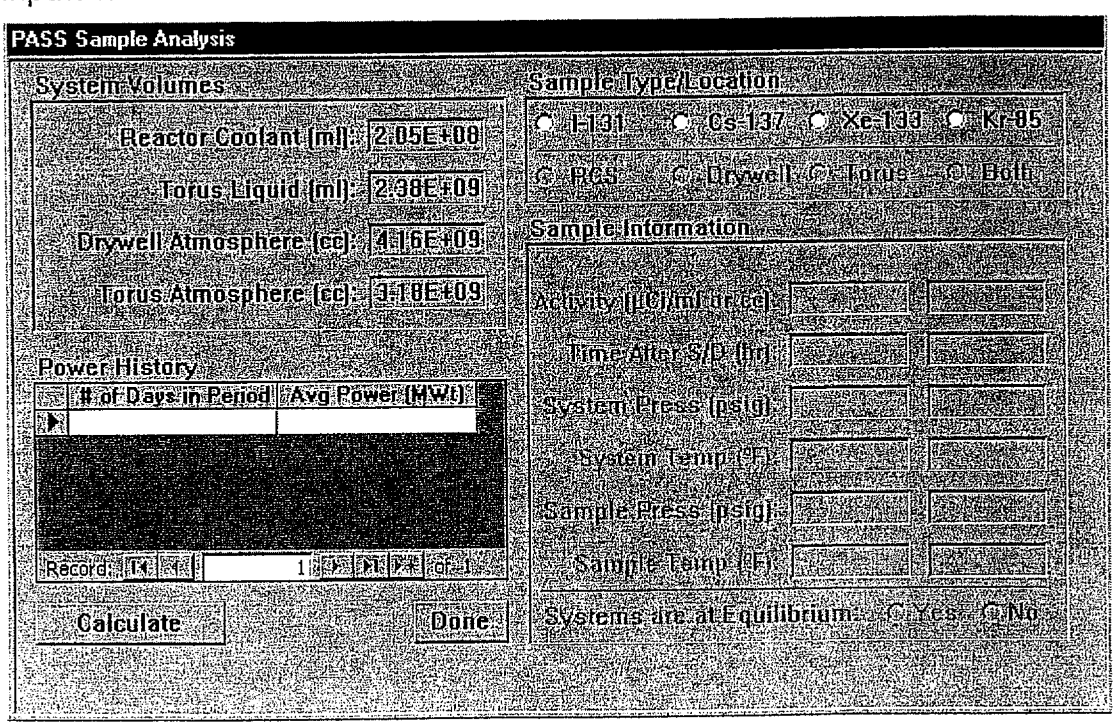
Main Switchboard Information and Functions

Object	Type	Information or Functional Description
PASS Analysis	Command Button	Closes the main switchboard window and opens the PASS sample analysis window. Applicable after the plant has been placed in a stable, steady-state condition following the accident.
PASS Ratio	Command Button	Closes the main switchboard window and opens the PASS isotopic ratio analysis window. Applicable any time following an accident although accuracy will decrease with time.
Containment Hydrogen	Command Button	Closes the main switchboard window and opens the containment hydrogen concentration window. Assumes all of the generated hydrogen gas is released into containment.
CHRM Readings	Command Button	Closes the main switchboard window and opens the radiation monitor window for the selected monitor.
Containment Monitors	Option Buttons	Available selections are: • Drywell 606A/B Monitors • Torus 607A/B Monitors Determines the monitor window opened when the CHRM Reading command button is selected (defaults to fuel melt curve for the selected monitor).
Quit	Command Button	Closes the DAMAGE application and returns to the Windows desktop.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

The PASS Sample Analysis Window

Upon selection of the 'PASS Sample Analysis' button, a window is displayed which allows the user to enter power history and sample information. PASS sample analysis evaluations will provide the most accurate core damage estimations. The assessment involves the analysis of samples taken from a plant location where a majority of the activity released from the core is expected to reside. Estimations will be able to account for all of the contained activity released from the core when information is available from either the liquid or gaseous containment system. It is extremely important to correctly choose the appropriate PASS location when assessments are made using only one sample point. For example, if an accident involving core damage occurs with the activity primarily contained in the Reactor coolant system, a sample of the containment atmosphere would greatly underestimate the amount of damage. Additionally (following the previous scenario), if the liquid systems were considered to be at equilibrium, a gross overestimation of core damage would result. Each system, liquid and gaseous, has two isotopes that can be used for damage estimates. The short-lived isotopes, I-131 and Xe-133, will tend to be predominant up to several weeks following an accident. Once the shorter-lived isotopes have decayed, Cs-137 and Kr-85 should become visible to analysis. When short-lived isotopes are used in damage estimations, approximately six half-lives (approximately 50 days) worth of power history data should be provided. Power history for long-lived isotopes should account for the entire time the fuel has been in the core. Variation in power over any single operating period will have less effect on calculations using long-lived isotopes. Variations in steady-state power should be limited to ±20% within each period. A description of the information provided within the window and for control options and inputs is as follows:



PASS Sample Analysis Window

PASS Sample Analysis Information and Functions

Object	Type	Information or Functional Description
System Volumes	Text Boxes	System volumes in ml or cc at time of sample. Defaults are set to normal PNPS volumes as follows: • RCS ....2.05E+08 ml • Torus Liquid ....2.38E+09 ml • Drywell Atmosphere.....4.16E+09 cc • Torus Atmosphere ....3.18E+09 cc Sets values for system volumes used in damage estimation calculations. Values must be greater than 0.
Power History Entry Area	N/A	Provides an area within the PASS sample analysis window where historic Reactor power data can be entered. Controls common to Windows-based applications within the table include: • Record Selector: A right facing arrowhead which selects the entire record, or several records when dragged. • Navigation Buttons: Allows navigation among records within the table. They include 'Go To First', 'Go To Previous', Go To Next', and 'Go To Last'. • Record #: A record number can be entered directly to go to the desired record. • Vertical Scroll Bar: Allows scrolling through the records when the number of records extends beyond the length of the table.
# Days in Period	Text Box	Number of days in the historical period, either operational or shutdown. Entered from **past to present** when more than one period is used in calculations. Value must be entered as greater than 0.
Average Power	Text Box	Average power during the historical period in MWt. Value must be from 0 to 3651 MWt.
Sample Type	Option Buttons	Available selections are: • I-131 • Cs-137 • Xe-133 • Kr-85 Specifies the particulate or gaseous isotope used in the damage estimation.

PASS Sample Analysis Information and Functions

Object	Type	Information or Functional Description
Sample Location	Option Buttons	Available selections are: • RCS • Drywell • Torus • Both Specifies the sample location to be used in the damage estimation. RCS is unavailable whenever Xe-133 or Kr-85 is selected. Drywell is unavailable whenever I-131 or Cs-137 is selected. The location option is reset whenever a new isotope is selected.
Sample Information	N/A	Provides an area for entry of sample data. Column headings will be appropriately titled after the sample type and location have been selected. Enabled fields are indicated by a solid title text. Disabled fields are indicated by a shaded title text.
Activity	Text Box(es)	Sample activity in µCi/ml or µCi/cc. First text box is activated following the selection of a single sample location. Both text boxes are activated following the selection of 'Both' sample locations. Value(s) must be greater than or equal to 0.
Time After Shutdown	Text Box(es)	Time interval between sample and Reactor shutdown in hours. First text box is activated following the selection of a single sample location. Both text boxes are activated following the selection of 'Both' sample locations. Value(s) must be greater than or equal to 0 and less than 1.0E+04.
System Pressure	Text Box(es)	Pressure of the sampled system in psig. First text box is activated following the selection of a single gaseous sample location. Both text boxes are activated following the selection of 'Both' gaseous sample locations. Value(s) must be greater than or equal to 0.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

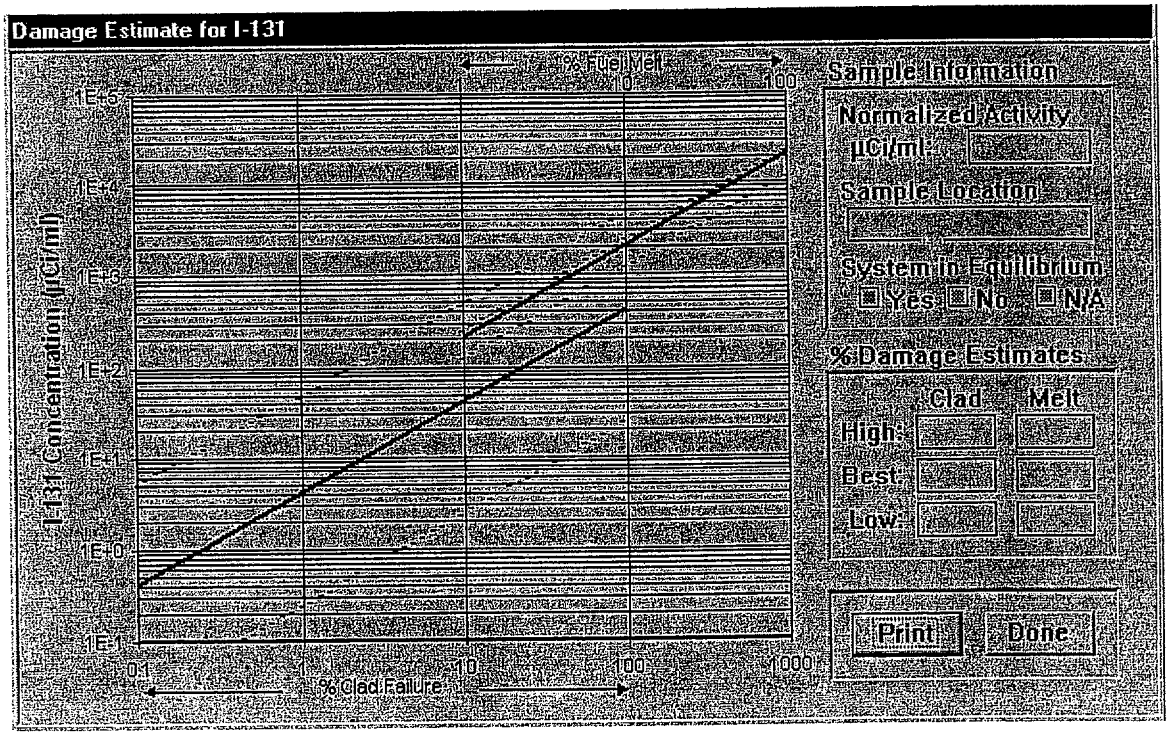
PASS Sample Analysis Information and Functions

Object	Type	Information or Functional Description
System Temperature	Text Box(es)	Temperature of the sampled system in °F. First text box is activated following the selection of a single gaseous sample location. Both text boxes are activated following the selection of 'Both' gaseous sample locations. Value(s) must be greater than or equal to 0.
Sample Pressure	Text Box(es)	Pressure of the sample vial in psig. First text box is activated following the selection of a single gaseous sample location. Both text boxes are activated following the selection of 'Both' gaseous sample locations. Value(s) must be greater than or equal to 0.
Sample Temperature	Text Box(es)	Temperature of the sample vial in °F. First text box is activated following the selection of a single gaseous sample location. Both text boxes are activated following the selection of 'Both' gaseous sample locations. Value(s) must be greater than or equal to 0.
Equilibrium	Option Buttons	Selections include 'Yes' or 'No'. Determines whether the activity is assumed to be equally distributed throughout the entire liquid or gaseous volumes or primarily contained within the sampled system. Disabled when 'Both sample locations are selected.
Calculate	Command Button	Calculates the normalized activity. Closes the PASS sample analysis window and opens the appropriate damage estimate window.
Done	Command Button	Closes the PASS sample analysis window and returns to the main switchboard.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

The PASS Damage Estimation Windows

Upon selection of the 'Calculate' button, a window is presented which numerically describes and graphically illustrates the core damage estimation for the analyzed isotope. A description of the information provided within the window and for control options and inputs is as follows:



PASS Damage Estimation Window

PASS Damage Estimation Information and Functions

Object	Type	Information or Functional Description
Isotopic Damage Curves	Graph	Information only (noneditable). Provides an illustration of the % Clad Failure (Gap) and % Fuel Melt curves. Normalized activity is plotted as a red horizontal line.
Normalized Activity	Text Box	Information only (noneditable). Displays the sample activity normalized for the reference plant.
Sample Location	Text Box	Information only (noneditable). Displays the selected sample location(s) from the PASS sample analysis window.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

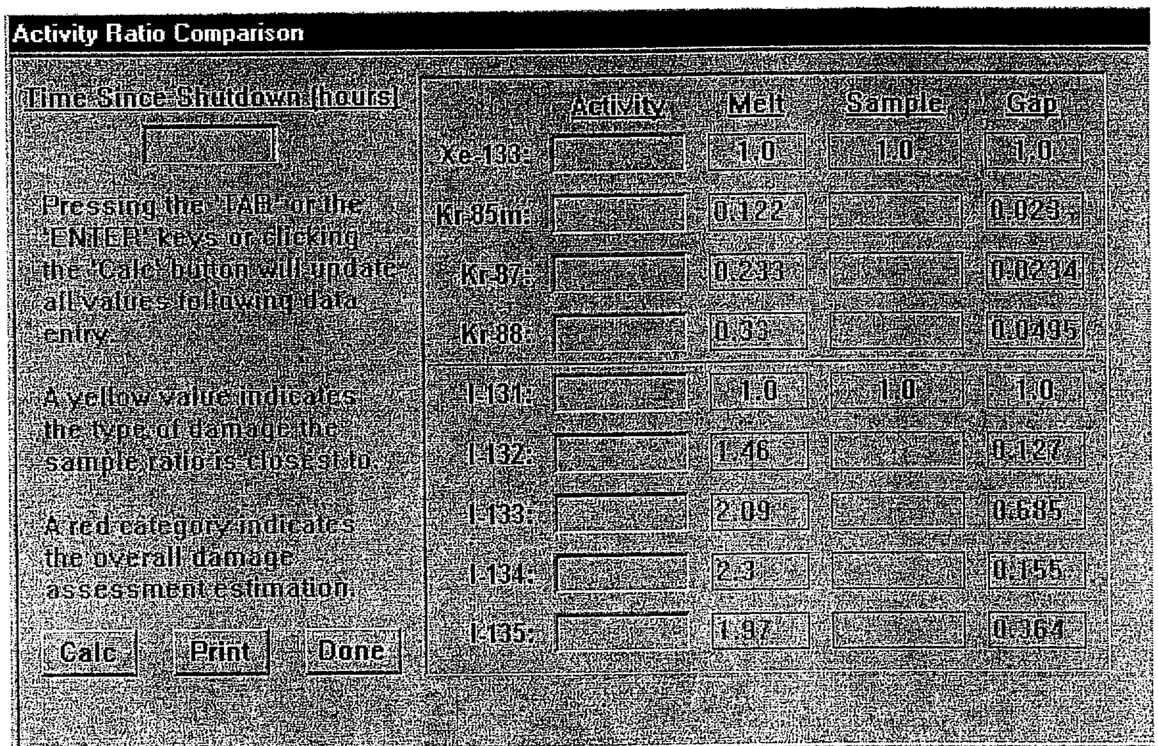
PASS Damage Estimation Information and Functions (Continued)

Object	Type	Information or Functional Description
% Damage Estimations	Text Boxes	Information only (noneditable). Displays the numerical values for the damage estimations for both Clad Failure (Gap) and Fuel Melt curves as follows: • High ....Yellow values • Low.........Yellow values • Best ....Red values Clad Failure (Gap) estimations are limited to values greater than 0.1%. Fuel Melt estimations are limited to values greater than 1%.
Print	Command Button	Sends a PASS isotopic damage estimation report to the default printer. [1]
Done	Command Button	Closes the PASS damage estimation window and returns to the PASS sample analysis window.

1 The default printer is determined by the Windows Print Manager.

The PASS Isotopic Ratio Comparison Window

Upon selection of the 'PASS Isotopic Ratio' button, a window is displayed which allows the user to enter and evaluate the relationship between fission product nuclides from a PASS sample. This method does not provide a numerical estimation of core damage. It is useful in the determination of the type of core damage when other methods indicate the possibility for either clad failure (gap) or fuel melt. The isotopes used in the calculations are relatively short-lived, therefore the accuracy of the results will diminish with time. The use of this method beyond about seven days after shutdown should be avoided. A description of the information provided within the window and for control options and inputs is as follows:



PASS Isotopic Ratio Comparison Window

PASS Isotopic Ratio Comparison Information and Functions

Object	Type	Information or Functional Description
Time After Shutdown	Text Box	Interval between the sample time and the time of Reactor shutdown in hours. Value must be greater or equal to 0 and less than 10,000 hours.
Activity	Text Boxes	Isotopic sample activities for ratio comparison given in µCi/ml or µCi/cc. Reference Isotopes are Xe-133 and I-131. Values must be entered as greater than 0 or left blank.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

PASS Isotopic Ratio Comparison Information and Functions (Continued)

Object	Type	Information or Functional Description
Sample Column	Text Boxes	Information only (noneditable). Indicates the results of the ratio comparison provided the reference isotope has been entered. Ratios falling below the estimated gap values are indicated as less than. Ratios falling above the estimated melt values are indicated as greater than.
Ratio Columns	Text Boxes	Information only (noneditable). Indicates the results of the expected ratio for the applicable type of damage. Ratio comparison is made on a logarithmic vice linear scale. The ratio most closely related to the sample ratio is highlighted with yellow text. The column that contains the greater number of ratios closest to the sample ratios is highlighted with a red heading.
Calculate	Command Button	Forces a recalculation of all possible ratios. All possible ratio calculations are also run each time an editable field is updated (input/edited followed by pressing the ENTER or TAB keys or otherwise leaving the field).
Print	Command Button	Sends a PASS isotopic ratio comparison report to the default printer. [2]
Done	Command Button	Closes the PASS isotopic ratio comparison window and returns to the main switchboard.

2 The default printer is determined by the Windows Print Manager.

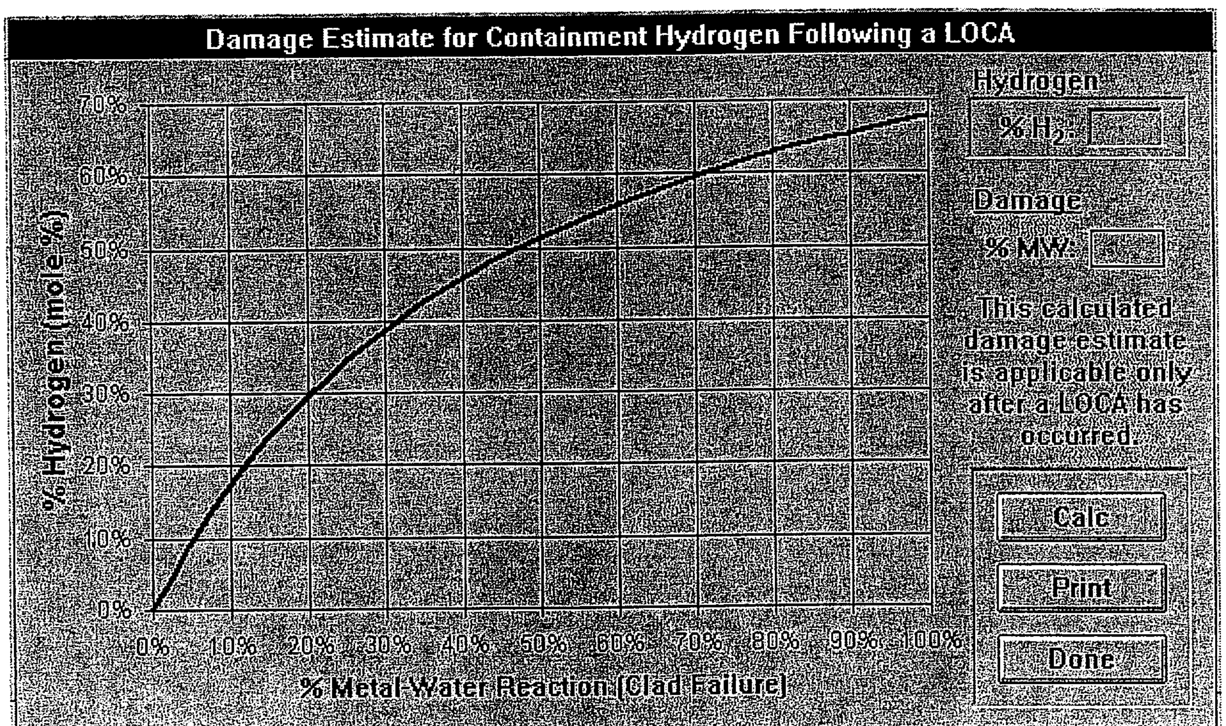
'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

The Containment Hydrogen Concentration Window

Upon selection of the 'Containment Hydrogen Concentration' button, a window is displayed which allows the user to enter and evaluate the relationship between containment hydrogen concentration and clad failure (gap). This method is based on the following assumptions:

1. All hydrogen generated by the reaction is released to containment.
2. Perfect mixing conditions exist in containment.
3. No depletion of hydrogen occurs (such as containment leakage).
4. Ideal gas behavior in containment.

A description of the information provided within the window and for control options and inputs is as follows:



Containment Hydrogen Concentration Window

Containment Hydrogen Concentration Information and Functions

Object	Type	Information or Functional Description
Containment Hydrogen Damage Curves	Graph	Information only (noneditable). Provides an illustration of the % Clad Failure (Gap) for a given hydrogen concentration in containment. Hydrogen concentration is plotted as a red dot on the % metal-water reaction curve.

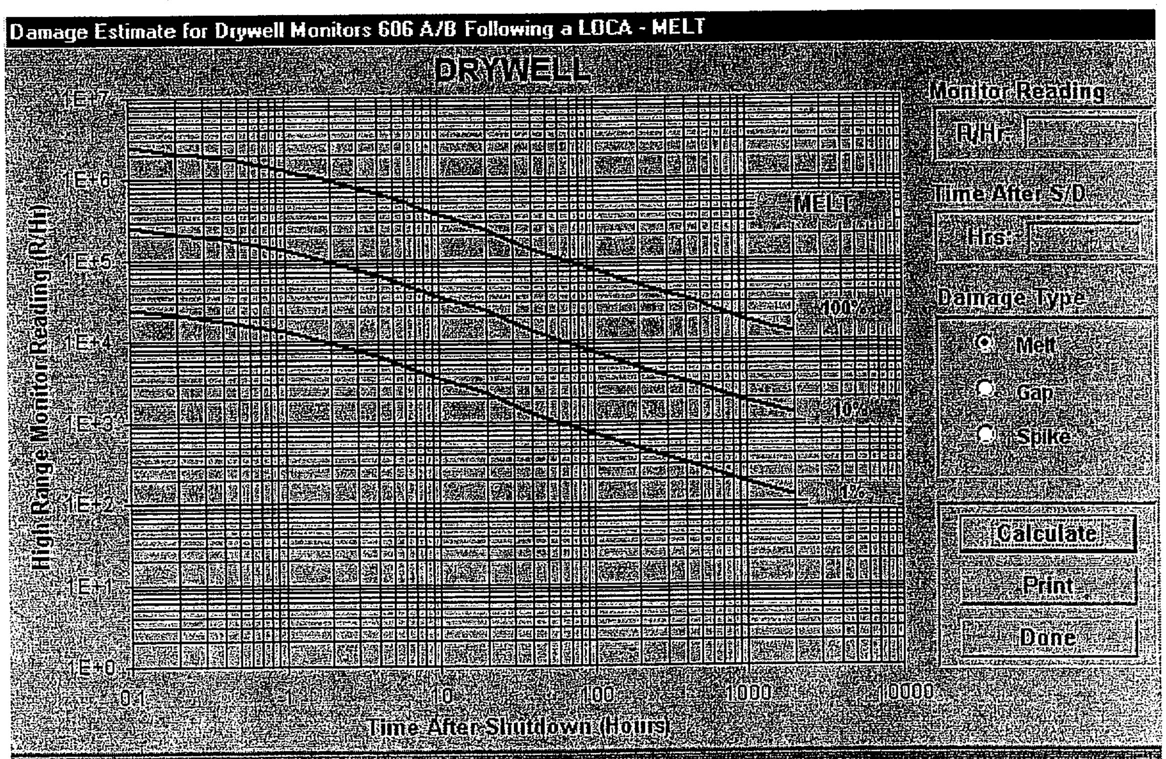
Containment Hydrogen Concentration Information and Functions (Continued)

Object	Type	Information or Functional Description
% Hydrogen	Text Box	Containment hydrogen entered in mole %. Value must be greater than 0 and less than or equal to 67.8 or left blank.
% Metal-Water Reaction	Text Box	Information only (noneditable). Provides a value of the % clad failure for a given hydrogen concentration in containment.
Calculate	Command Button	Forces a recalculation of % clad failure. Calculation is also performed each time the hydrogen field is updated (input/edited followed by pressing the ENTER or TAB keys or otherwise leaving the field).
Print	Command Button	Sends a containment hydrogen concentration damage estimation report to the default printer. [3]
Done	Command Button	Closes the containment hydrogen concentration window and returns to the main switchboard.

3 The default printer is determined by the Windows Print Manager.

The Drywell High Radiation Levels Window

Upon selection of the 'CHRM Readings' button with the Drywell monitor selected, a window is opened to allow the user to plot radiological data on a graph containing predicted core damage curves. The highest Drywell monitor reading and the time after Reactor shutdown are used to describe a point that can be compared to curves describing an amount of core damage. The curves, which relate a monitor reading to a specific amount of damage, are based on an end-of-life core inventory and a static isotopic mix ratio. These assumptions will usually result in an overestimation of the amount of damage and should be used with caution. A description of the information provided within the window and for control options and inputs is as follows:



Drywell High Radiation Levels Window

Drywell Radiation Levels Information and Functions

Object	Type	Information or Functional Description
CHRM Damage Curves	Graph	Information only (noneditable). Provides separate illustration of fuel melt, clad failure (gap), and spike condition curves for a given monitor value as a function of time after shutdown. The graph screen defaults to fuel melt conditions - refer to Damage Type option button to select other fuel type conditions.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

Drywell Radiation Levels Information and Functions (Continued)

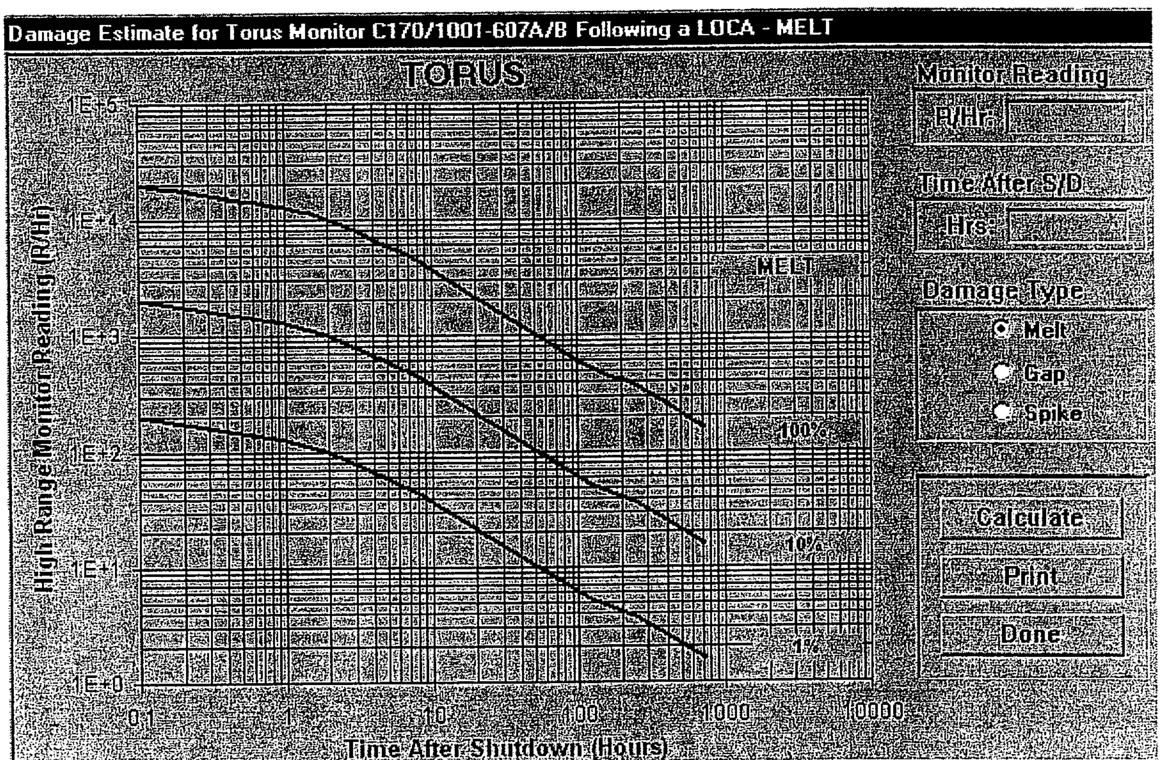
Object	Type	Information or Functional Description
Damage Type	Option Button	Available selections are: • Melt • Gap • Spike Determines the graph window opened for the damage type when option button is selected. NOTE Core damage of 10% gap activity or greater may overlap in the fuel melt condition graph for core damage estimates.
Monitor Reading	Text Box	Highest credible Drywell monitor 606 A/B reading in R/hr as applicable to the chosen window. Value must be greater than 1 and less than 1.0E+07 R/hr.
Time After Shutdown	Text Box	Interval between the monitor reading and the time of Reactor shutdown in hours. Time of accident equals the time of reactor Scram/shutdown (i.e., t = 0 hrs). Value must be greater than 0.1 and less than 1.0E+04 hours.
Calculate	Command Button	Forces plotting of the monitor reading data point. Plots Drywell high range monitor reading as a function of time after shutdown with a red dot on the graph. Replotting is also performed each time an editable field is updated (input/edited followed by pressing the ENTER or TAB keys or otherwise leaving the field).
Print	Command Button	Sends a Drywell high radiation analysis report to the default printer. [4]
Done	Command Button	Closes the Drywell high radiation analysis window and returns to the main switchboard.

4 The default printer is determined by the Windows Print Manager.

'DAMAGE' COMPUTER APPLICATION USER GUIDE (Continued)

The Torus High Radiation Levels Windows

Upon selection of the 'CHRM Readings' button with either of the Torus monitors selected, a window is opened to allow the user to plot radiological data on a graph containing predicted core damage curves. The Torus monitor reading and the time after Reactor shutdown are used to describe a point that can be compared to curves describing an amount of core damage. The curves, which relate the monitor reading to a specific amount of damage, are based on an end-of-life core inventory and a static isotopic mix ratio. These assumptions will usually result in an overestimation of the amount of damage and should be used with caution. A description of the information provided within the window and for control options and inputs is as follows:



Torus High Radiation Levels Windows

Torus High Radiation Levels Information and Functions

Object	Type	Information or Functional Description
CHRM Damage Curves	Graph	Information only (noneditable). Provides separate illustration of fuel melt, clad failure (gap), and spike condition curves for a given monitor value as a function of time after shutdown. The graph screen defaults to fuel melt conditions - refer to Damage Type option button to select other fuel type conditions.

Torus High Radiation Levels Information and Functions (Continued)

Object	Type	Information or Functional Description
Damage Type	Option Button	Available selections are: • Melt • Gap • Spike Determines the graph window opened for the damage type when option button is selected. NOTE Core damage of 10% gap activity or greater may overlap in the fuel melt condition graph for core damage estimates.
Monitor Reading	Text Box	Highest credible Torus monitor 607A or B reading in R/hr as applicable to the chosen window. Value must be greater than 1 and less than 1.0E+05 R/hr.
Time After Shutdown	Text Box	Interval between the monitor reading and the time of Reactor shutdown in hours. Time of accident equals the time of reactor Scram/shutdown (i.e., t = 0 hrs). Value must be greater than 0.1 and less than 1.0E+04 hours.
Calculate	Command Button	Forces plotting of the monitor reading data point. Plots Torus high range monitor reading as a function of time after shutdown with a red dot on the graph. Replotting is also performed each time an editable field is updated (input/edited followed by pressing the ENTER or TAB keys or otherwise leaving the field).
Print	Command Button	Sends a Torus high radiation analysis report to the default printer. [5]
Done	Command Button	Closes the Torus high radiation analysis window and returns to the main switchboard.

5 The default printer is determined by the Windows Print Manager.

CORE DAMAGE WORKSHEET: PASS LIQUID SAMPLES

1. Obtain PASS liquid sample information from the Chemistry Coordinator:

   Sample Time: ___________ Sample Date: ___________

   Isotope Analyzed: ☐ I-131 ☐ Cs-137

Parameter	Reactor Coolant	Torus Liquid
Sample Activity (μCi/ml)		
Time interval between Rx S/D and sample measurement (hours)		

2. Correct the sample activity for decay from time of Reactor shutdown to time of measurement:

Calculation	Reactor Coolant	Torus Liquid
$A_{CORR} = (A_{INITIAL})(e^{\lambda t})$		

Where:

$A_{INITIAL}$ = Measured concentration in μCi/ml (from Step 1).

λ = ☐ 3.59E-03 (hrs^{-1}) for I-131.
☐ 2.62E-06 (hrs^{-1}) for Cs-137.

t = Time interval between Reactor shutdown and sample measurement (from Step 1).

> **NOTE**
>
> If both Reactor and Torus water were sampled, then decay corrected activities must be averaged using Step 3. If only one sample point is being used, go directly to Step 4.

3. Weighted average of the activities when information from both sample locations is available:

Calculation	Reactor Coolant	Torus Liquid
$A_{AVG} = (A_{RCS})(0.0793) + (A_{TOR})(0.9207)$		

Where:

A_{RCS} = Decay corrected Reactor coolant sample activity (from Step 2).

0.0793 = Reactor coolant relative volume correction.

A_{TOR} = Decay corrected Torus liquid sample activity (from Step 2).

0.9207 = Torus liquid relative volume correction.

CORE DAMAGE WORKSHEET: PASS LIQUID SAMPLES (Continued)

4. Obtain the historical plant operations data from the Reactor Engineer and calculate the plant inventory correction factor using the worksheet on Sheet 3 of Attachment 3.

Calculation	Result
F_{PNPS} (from worksheet)	

5. Determine the inventory normalization factor as follows:

Calculation	Result
$Inv_{NORM} = F_{REF} / F_{PNPS}$	

Where:

F_{REF} = ☐ 3651 for I-131.
☐ 243.4 for Cs-137.

F_{PNPS} = PNPS inventory correction factor (from Step 4).

6. Determine the normalized activity concentration as follows:

Calculation	Result
A_{NRM} = Activity x Inv_{NRM} x Vol_{NRM}	

Where:

Activity = ☐ A_{CORR} for a single sample location (from Step 2).
☐ A_{AVG} for both sample locations (from Step 3).

Inv_{NRM} = Inventory normalization factor (from Step 5).

Vol_{NRM} = ☐ 5.23E-2 for Reactor coolant samples.
☐ 6.08E-1 for Torus liquid samples.
☐ 6.60E-1 for both sample locations.

7. Estimate the amount of core damage, in percent, using the appropriate graphs on Sheet 4 and Sheet 5 of Attachment 3 and normalized activity concentrations from Step 6.

Damage Estimate	% Clad Failure (Gap)	% Fuel Melt
High		
Best		
Low		

Analyst: ______________________ Date:__________ Time: __________

CORE DAMAGE WORKSHEET: PASS LIQUID SAMPLES (Continued)

Period	Power (P)	Buildup (B)	Decay (D)	$(P)(1-e^{-\lambda B})(e^{-\lambda D})$
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
Total of the inventory corrections for each period (F_{PNPS}):				

Where:

P = Steady state power level during the operational period in MWt.

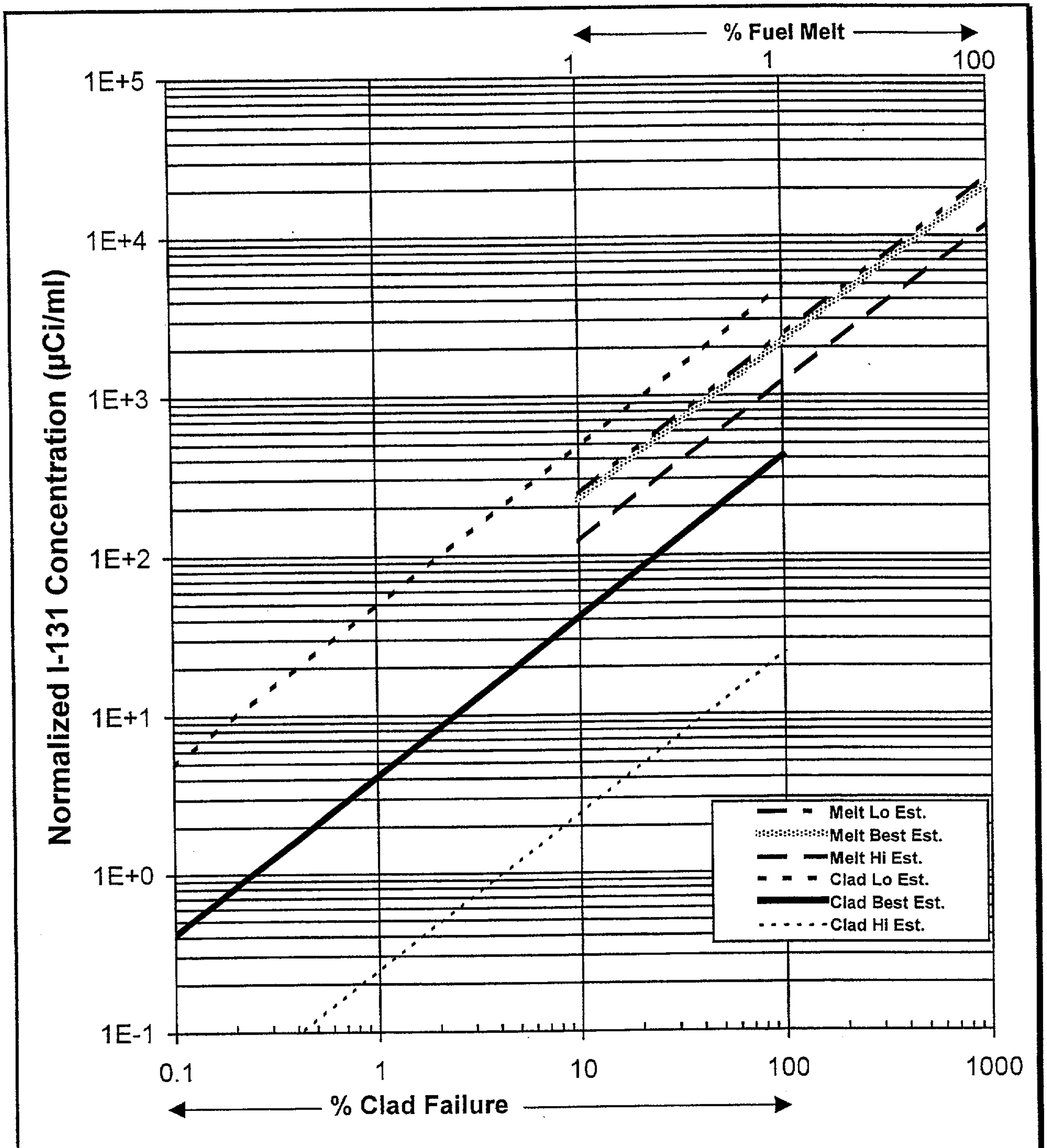
B = Duration of the operational period in days.

D = Duration from the end of the operational period to the time of damage shutdown in days.

λ = ☐ 8.62E-02 ($days^{-1}$) for I-131.

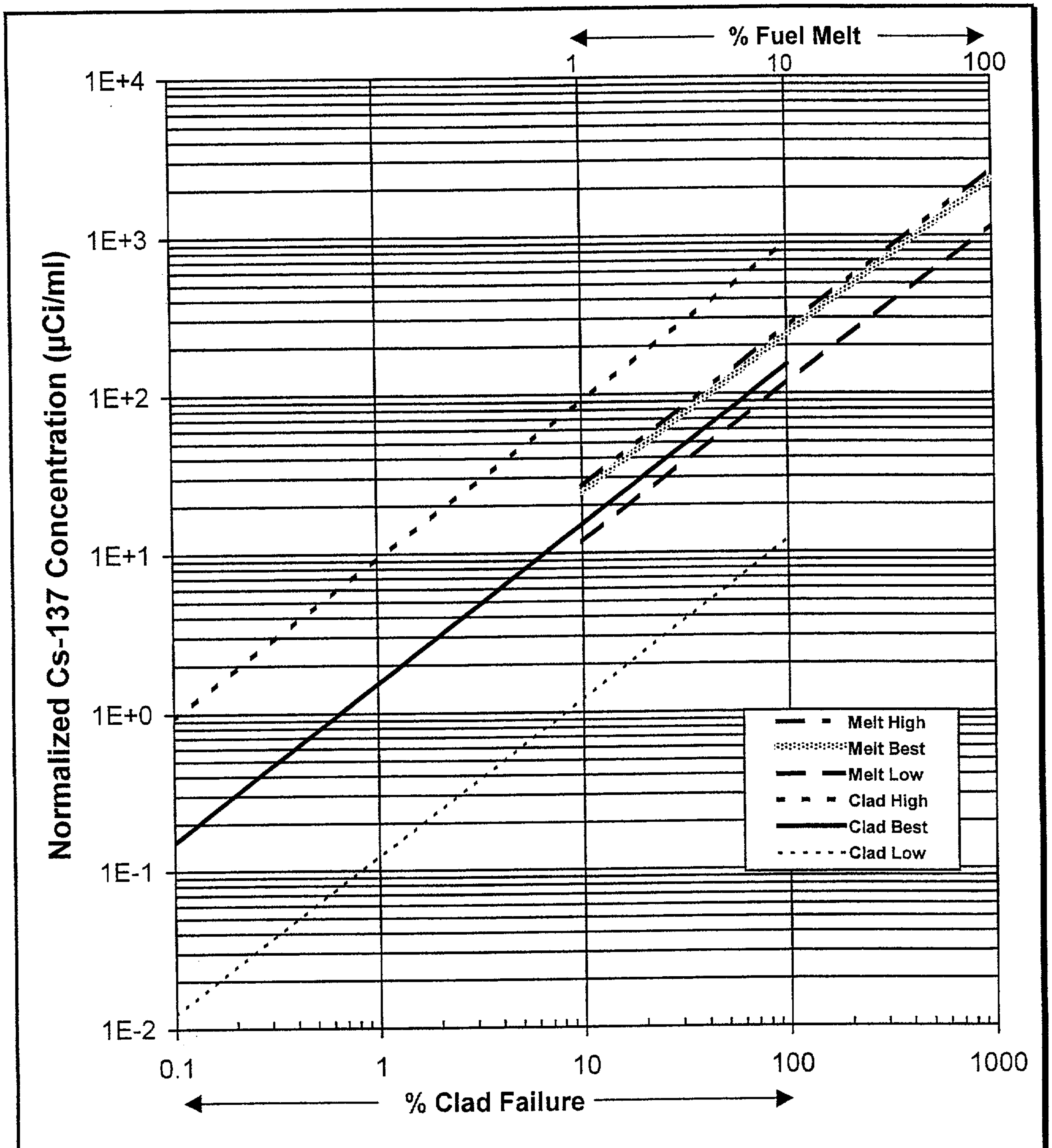
☐ 6.30E-05 ($days^{-1}$) for Cs-137.

CORE DAMAGE WORKSHEET: PASS LIQUID SAMPLES (Continued)



The Relationship Between I-131 in Coolant and the Extent of Core Damage

CORE DAMAGE WORKSHEET: PASS LIQUID SAMPLES (Continued)



The Relationship Between Cs-137 in Coolant and the Extent of Core Damage

CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES

1. Obtain PASS gaseous sample information from the Chemistry Coordinator:

Sample Time: ____________ Sample Date: ______________

Isotope Analyzed: ❑ Xe-133 ❑ Kr-85

Parameter	Drywell Atmos.	Torus Atmos.
Sample Activity (µCi/cc)		
Sample Vial Pressure (psig)		
Sample Vial Temperature (°F)		
Sample Point Pressure (psig)		
Sample Point Temperature (°F)		
Time interval between Rx S/D and sample measurement (hours)		

2. Correct the sample for decay from time of Rx S/D to time of measurement:

Calculation	Drywell Atmos.	Torus Atmos.
$A_{DC} = A_{INITIAL} \times e^{\lambda t}$		

Where:

$A_{INITIAL}$ = Measured concentration in µCi/cc (from Step 1).

λ = ❑ 5.50E-03 (hrs^{-1}) for Xe-133.

❑ 7.38E-06 (hrs^{-1}) for Kr-85.

t = Time interval between Reactor shutdown and sample measurement (from Step 1).

3. Correct the sample activity for temperature and pressure as follows:

Calculation	Drywell Atmos.	Torus Atmos.
$A_{CORR} = A_{DC} \times \frac{(P_2 + 14.7)(T_1 + 460)}{(P_1 + 14.7)(T_2 + 460)}$		

Where:

A_{DC} = Decay corrected activity in µCi/cc (from Step 2).

P_1 & T_1 = sample vial pressure in psig and temperature in °F.

P_2 & T_2 = sample point pressure in psig and temperature in °F.

NOTE

If both the Drywell and Torus were sampled, the decay corrected activities must be averaged using Step 4. If only one sample point is being used, go directly to Step 5.

CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES (Continued)

4. Calculate the weighted average of the activities when information from both sample locations is available:

Calculation	Drywell Atmos.	Torus Atmos.
$A_{AVG} = (A_{DW}) (0.5668) + (A_{TOR}) (0.4332)$		

Where:

A_{DW} = Decay corrected Drywell sample activity (from Step 3).

0.5668 = Drywell atmosphere relative volume correction.

A_{TOR} = Decay corrected Torus atmosphere sample activity (from Step 3).

0.4332 = Torus atmosphere relative volume correction.

5. Obtain the historical plant operations data from the Reactor Engineer and calculate the plant inventory correction factor using the worksheet on Sheet 3:

Calculation	Result
F_{PNPS} (from worksheet)	

6. Determine the inventory normalization factor as follows:

Calculation	Result
$Inv_{NORM} = F_{REF} / F_{PNPS}$	

Where:

F_{REF} = ❑ 3651 for Xe-133.
❑ 643.3 for Kr-85.

F_{PNPS} = PNPS inventory correction factor (from Step 5).

7. Determine the normalized activity concentration as follows:

Calculation	Result
$A_{NORM} = (Activity) (Inv_{NORM}) (Vol_{NORM})$	

Where:

Activity = ❑ A_{CORR} for a single sample location (from Step 3).
❑ A_{AVG} for both sample locations (from Step 4).

Inv_{NORM} = Inventory normalization factor (from Step 6).

Vol_{NORM} = ❑ 1.03E-1 for Drywell atmosphere samples.

❑ 7.90E-2 for Torus atmosphere samples.

❑ 1.82E-1 for both sample locations.

CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES (Continued)

8. Estimate the amount of core damage, in percent, using the appropriate graph on Sheet 5 or Sheet 6 of Attachment 4.

Damage Estimate	% Clad Failure (Gap)	% Fuel Melt
High		
Best		
Low		

Analyst: ____________________ Date:__________ Time: __________

CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES (Continued)

Period	Power (P)	Buildup (B)	Decay (D)	$(P)(1-e^{-\lambda B})(e^{-\lambda D})$
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
Total of the inventory corrections for each period (F_{PNPS}):				

Where:

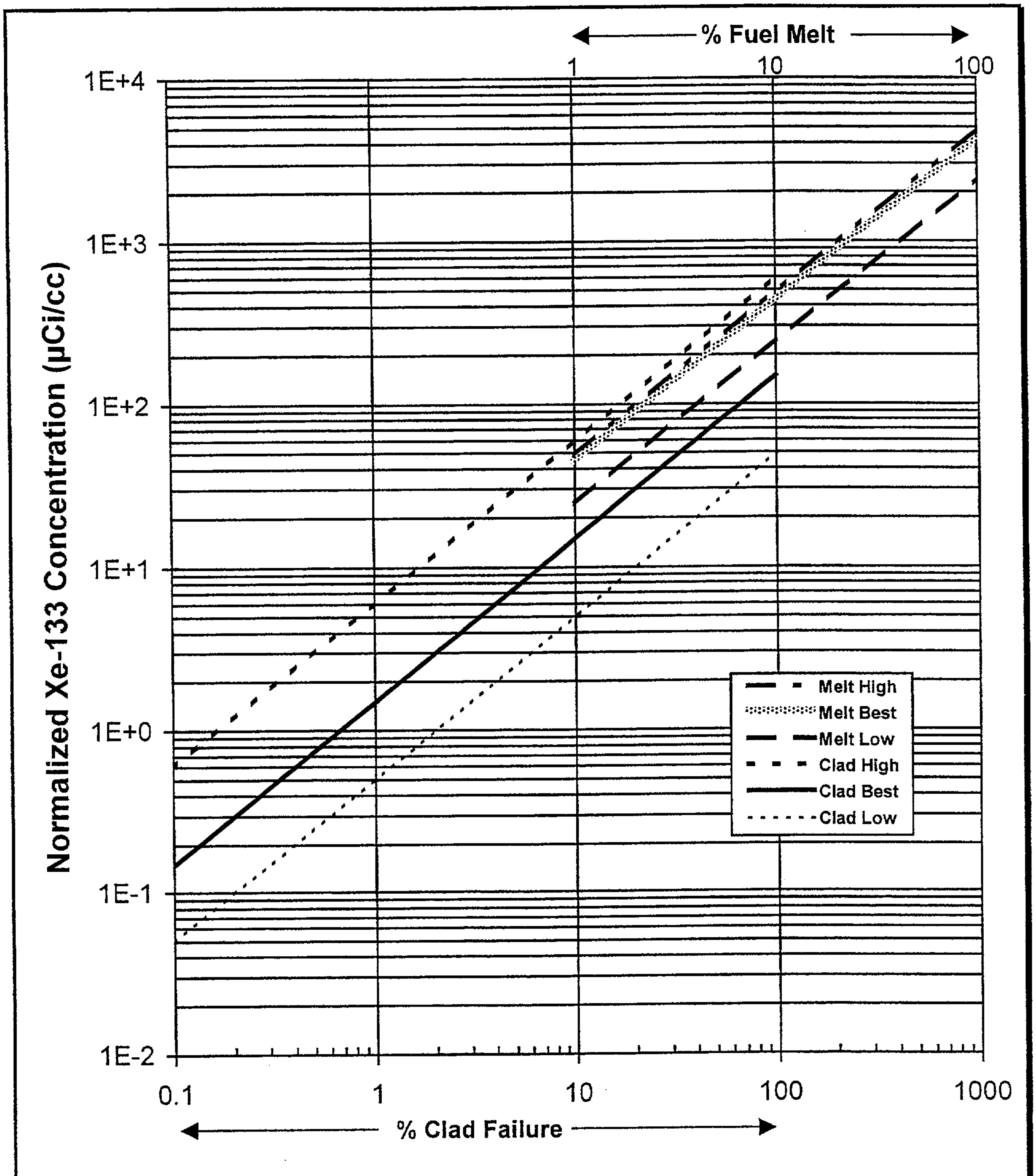
P = Steady-state power level during the operational period in MWt.

B = Duration of the operational period in days.

D = Duration from the end of the operational period to the time of damage shutdown in days.

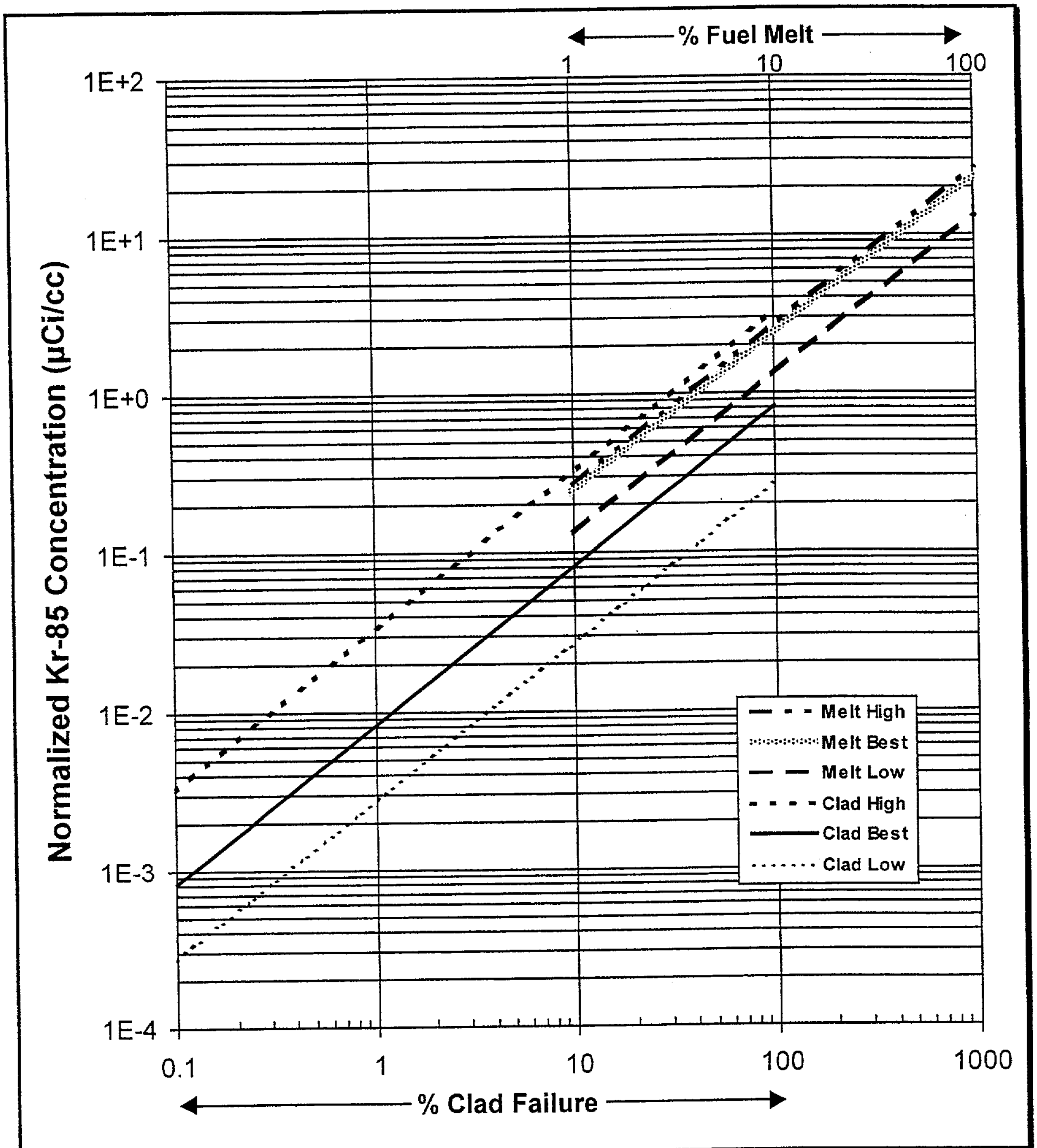
λ = ☐ 1.32E-01 (days^{-1}) for Xe-133.
☐ 1.77E-04 (days^{-1}) for Kr-85.

CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES (Continued)



The Relationship Between Xe-133 in Containment and the Extent of Core Damage

CORE DAMAGE WORKSHEET: PASS GASEOUS SAMPLES (Continued)



The Relationship Between Kr-85 in Containment and the Extent of Core Damage

CORE DAMAGE WORKSHEET: PASS ISOTOPIC RATIOS

Sample Time: __________ Sample Date: __________

1. Obtain PASS sample information from the Chemistry Coordinator and decay correct the sample activity from time of Reactor shutdown to time of measurement as follows:

$$A_{DC} = A_{INITIAL} \times e^{\lambda t}$$

Where:

A_{DC} = Decay corrected activity of the sample in µCi/ml or cc.

$A_{INITIAL}$ = Measured sample concentration in µCi/ml or cc.

λ = See table below.

t = Time interval between Reactor shutdown and sample measurement.

Time Interval (t): ______________

Isotope	Activity (Sampled)	λ (Hours-1)	Activity (Corrected)
Xe-133		5.50E-03	
Kr-85m		1.55E-01	
Kr-87		5.45E-01	
Kr-88		2.79E-01	
I-131		3.59E-03	
I-132		3.01E-01	
I-133		3.33E-02	
I-134		7.90E-01	
I-135		1.05E-01	

2. Calculate the ratios for the nuclides and evaluate the results as follows:

$$Ratio_{NG} = A_{ISOTOPE} / A_{Xe\text{-}133}$$

and

$$Ratio_{IODINES} = A_{ISOTOPE} / A_{I\text{-}131}$$

a. If the ratio of the sample is greater than the ratio given for fuel melt, check the fuel melt box.

b. If the ratio of the sample is less than the ratio given for a gap release, check the clad failure box.

CORE DAMAGE WORKSHEET: PASS ISOTOPIC RATIOS (Continued)

c. If the ratio is between the values, evaluate the ratio as follows:

$$\mathbf{Ratio_{MELT} / Ratio_{SAMPLE} < Ratio_{SAMPLE} / Ratio_{GAP}}$$

- If the above relationship is true, then check the fuel melt box.
- If the above relationship is not true, then check the clad failure box.

Isotope	Fuel Ratio	Sample Ratio	Gap Ratio
Xe-133	1.0	1.0	1.0
Kr-85m	❑ 0.122		❑ 0.023
Kr-87	❑ 0.233		❑ 0.0234
Kr-88	❑ 0.33		❑ 0.0495
I-131	1.0	1.0	1.0
I-132	❑ 1.46		❑ 0.127
I-133	❑ 2.09		❑ 0.685
I-134	❑ 2.3		❑ 0.155
I-135	❑ 1.97		❑ 0.364

3. The overall damage indication is determined by the column which contains the most checked boxes.

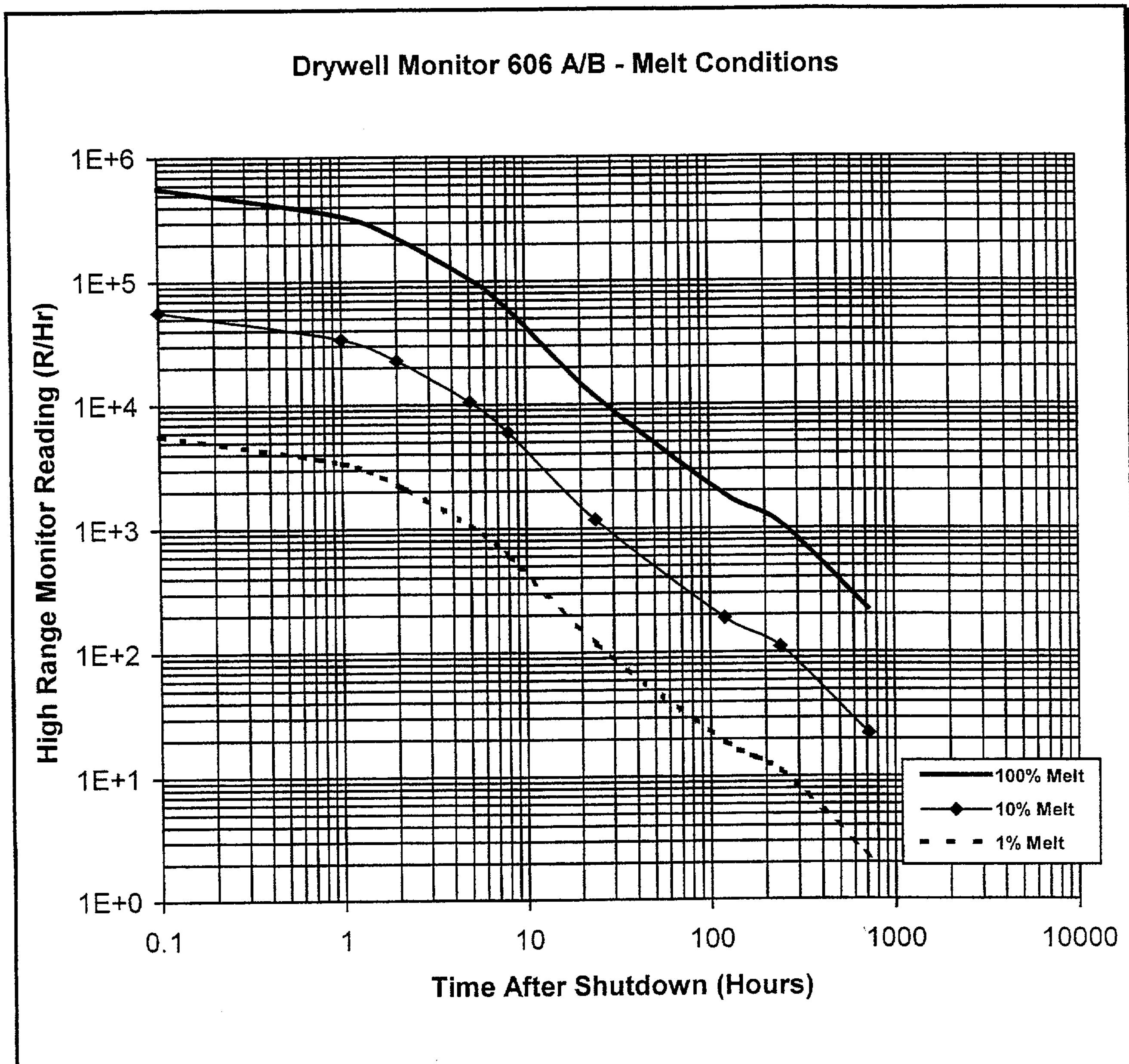
   Unusually high concentrations of any of the less volatile fission products are indicative of some degree of fuel melt. These fission products may include soluble or insoluble isotopes of the following elements:

   - Alkaline Earths: Sr, Ba
   - Noble Metals: Ru, Rh, Pd, Mo, Tc
   - Rare Earths: Y, La, Ce, Nd, Pr, Eu, Pm, Sm, Np, Pu
   - Refractories: Zr, Nb

Overall Assessment: ❑ Fuel Melt ❑ Clad Failure (Gap) ❑ Undetermined

Analyst: ______________________ Date: __________ Time: __________

CORE DAMAGE CURVES: DRYWELL CHRMS



Record appropriate information below:

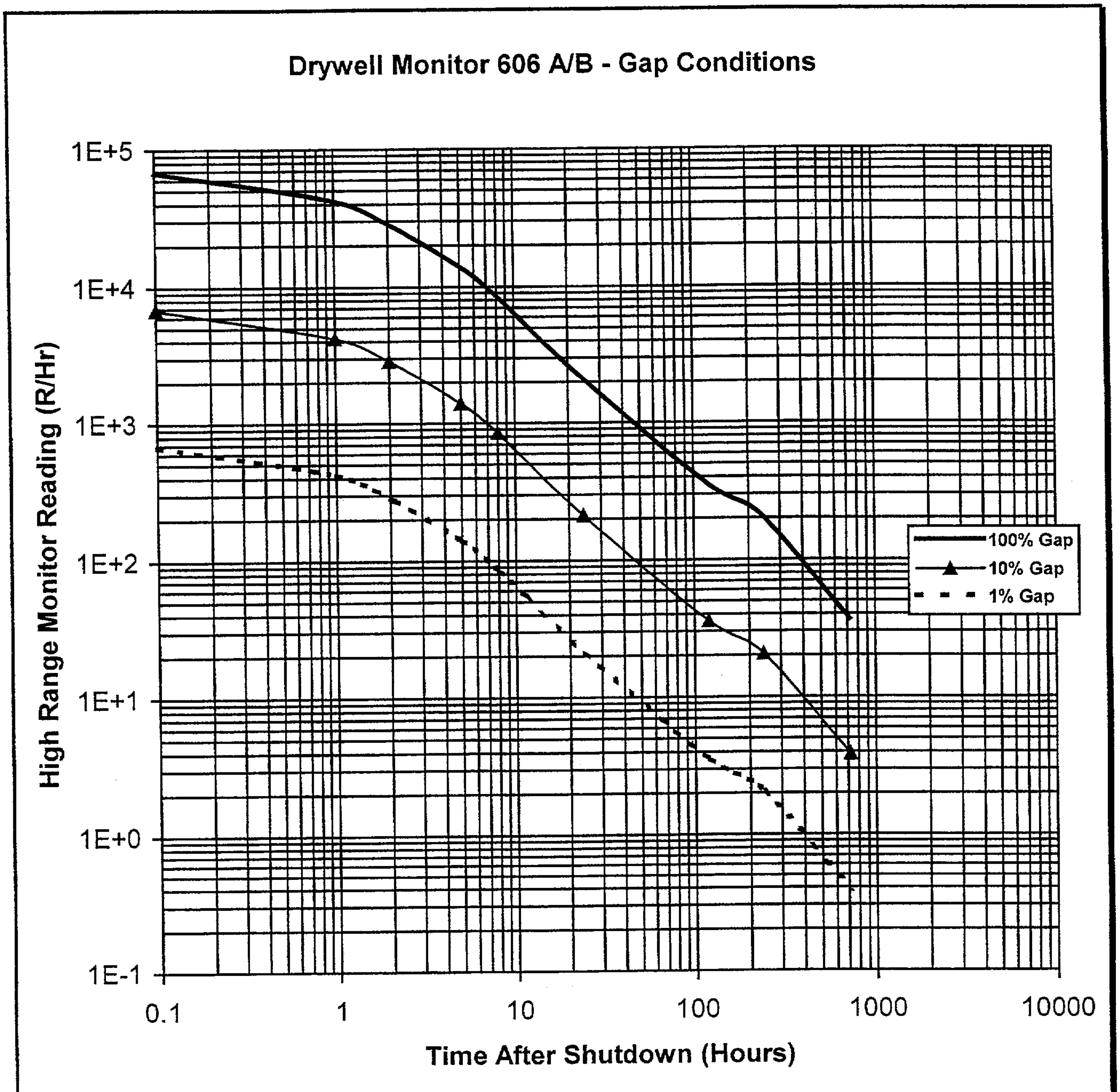
Monitor Reading (R/hr): ____________	Time After Shutdown (Hrs): ____________

Assessment Data:

% Melt Estimate: ____________________

Name: ______________________________________ Date: ______________ Time: ______________

CORE DAMAGE CURVES: DRYWELL CHRMS (Continued)



Record appropriate information below:

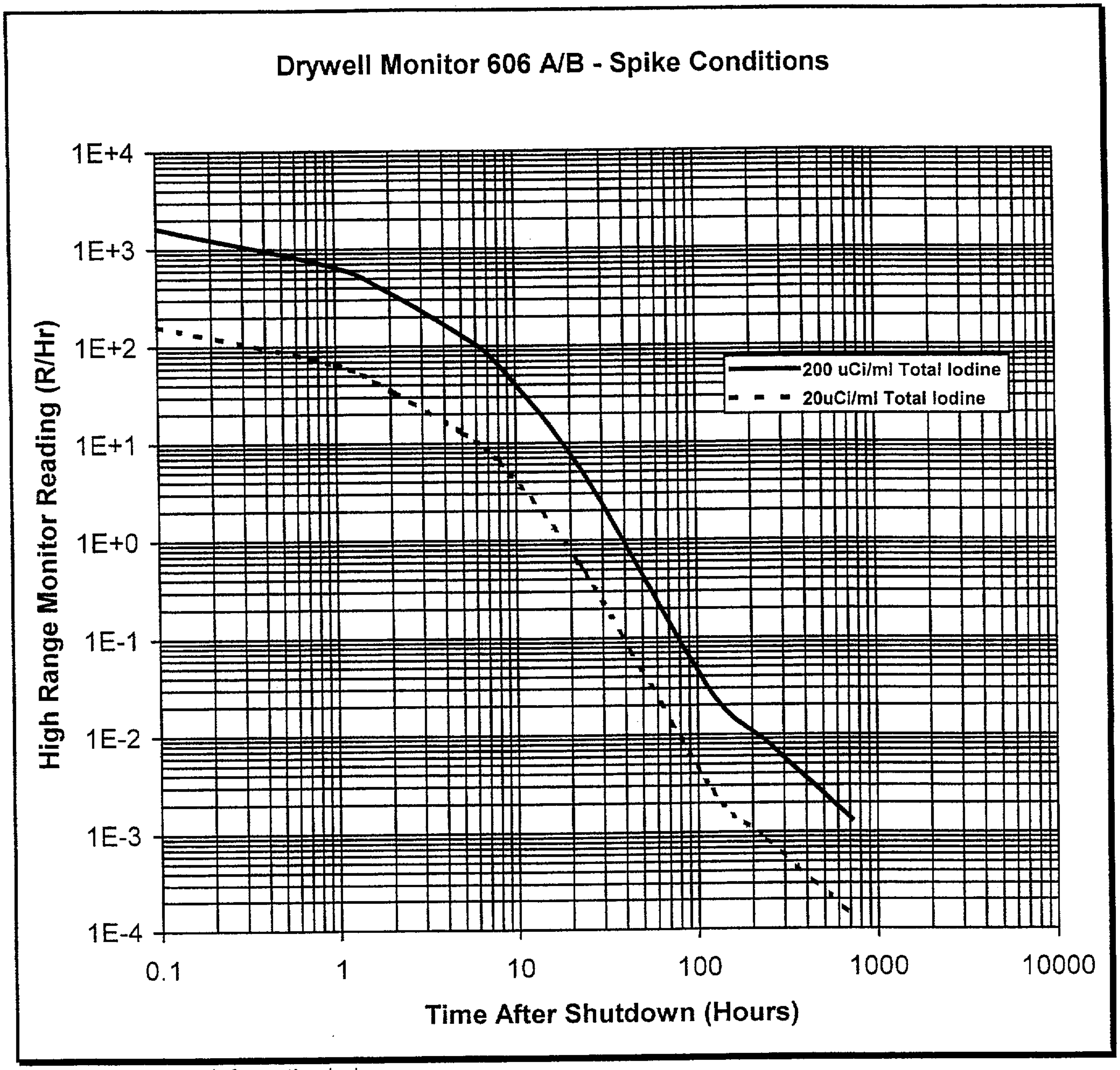
Monitor Reading (R/hr): ____________ Time After Shutdown (Hrs): ____________

Assessment Data:

% Gap Estimate: ____________________

Name: ______________________________________ Date: ______________ Time: ______________

CORE DAMAGE CURVES: DRYWELL CHRMS (Continued)



Record appropriate information below:

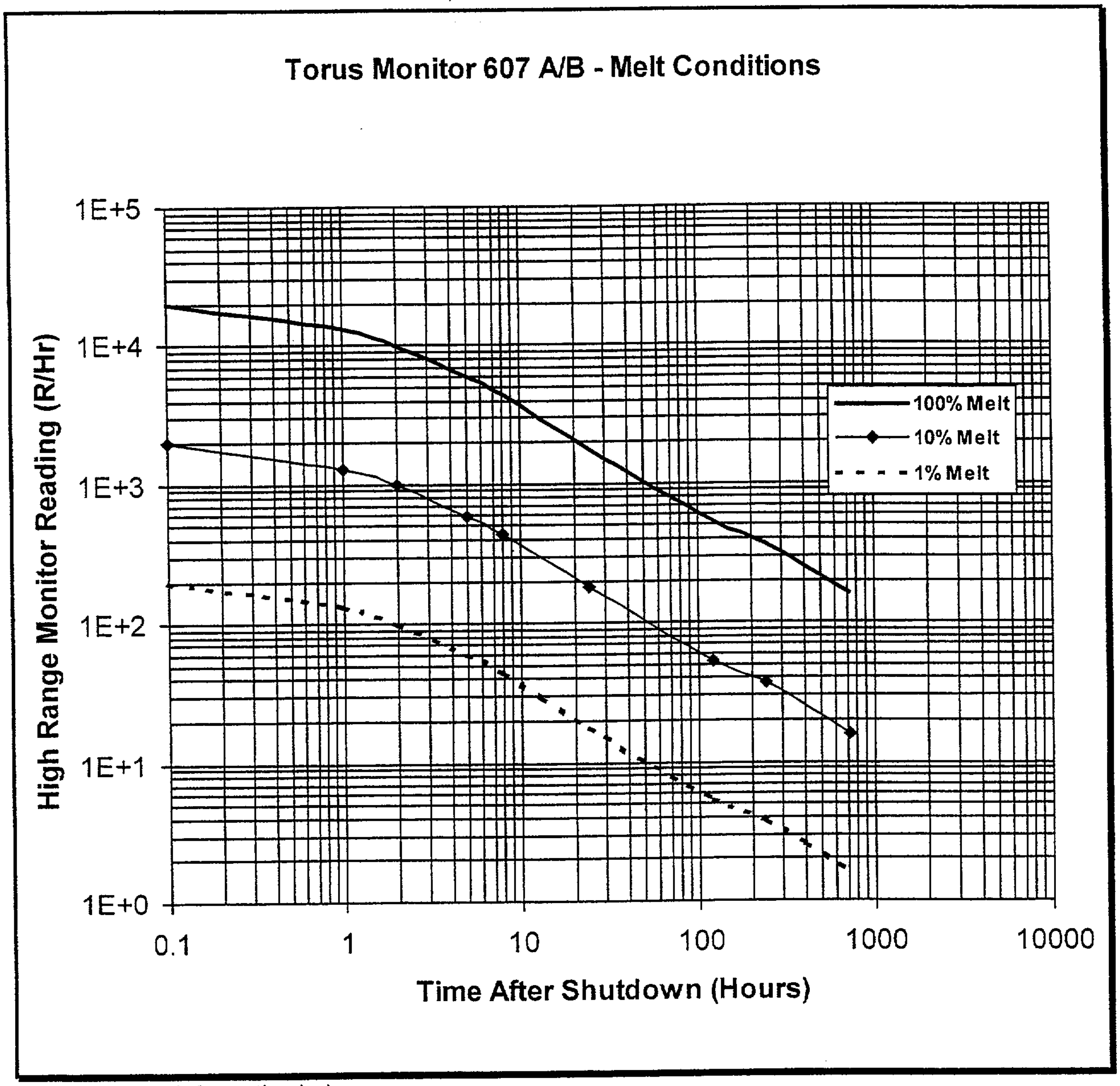
Monitor Reading (R/hr): ____________ Time After Shutdown (Hrs): ____________

Assessment Data:

Total Iodine Is: ☐ Above ☐ Equal to ☐ Below Technical Specifications

Name: ______________________________ Date: ____________ Time: ____________

CORE DAMAGE CURVES: TORUS CHRMS



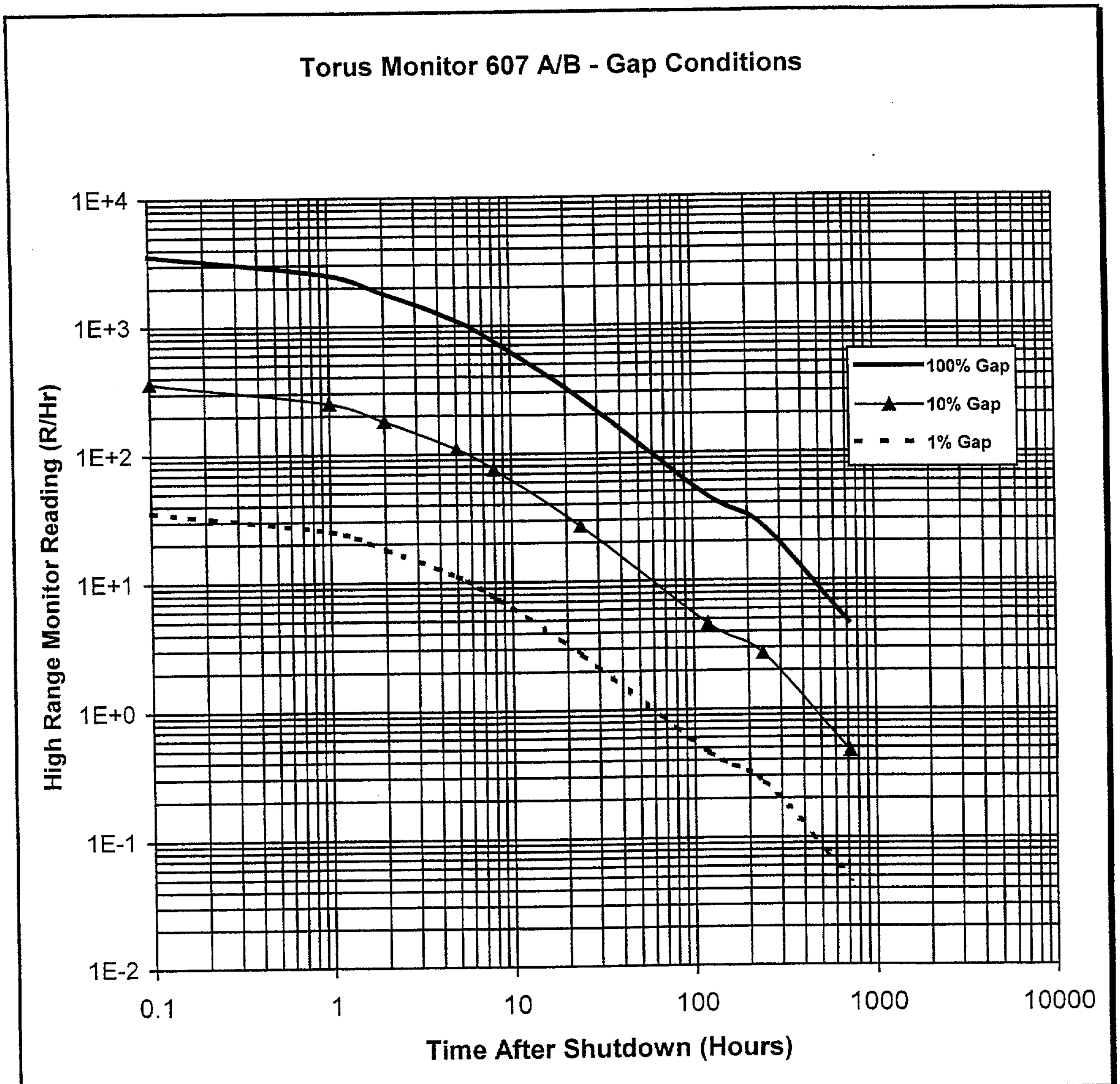
Record appropriate information below:

Monitor Reading (R/hr): ____________	Time After Shutdown (Hrs): ___________

Assessment Data:

% Melt Estimate: ____________________

Name: ____ Date: ______________ Time: ______________



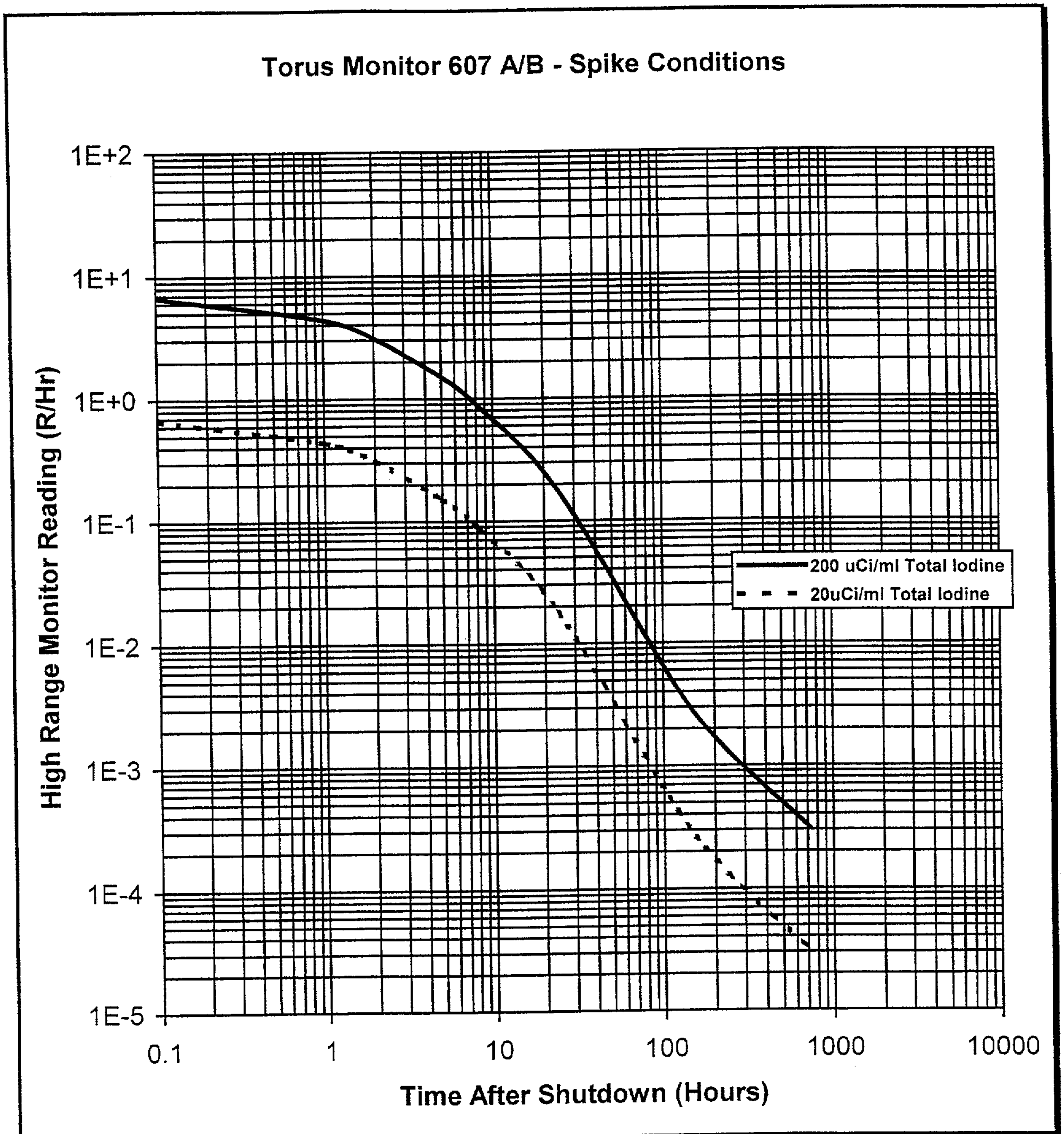
Record appropriate information below:

Monitor Reading (R/hr): ____________ Time After Shutdown (Hrs): ____________

Assessment Data:

% Gap Estimate: ____________________

Name: ____ Date: ______________ Time: ______________



Record appropriate information below:

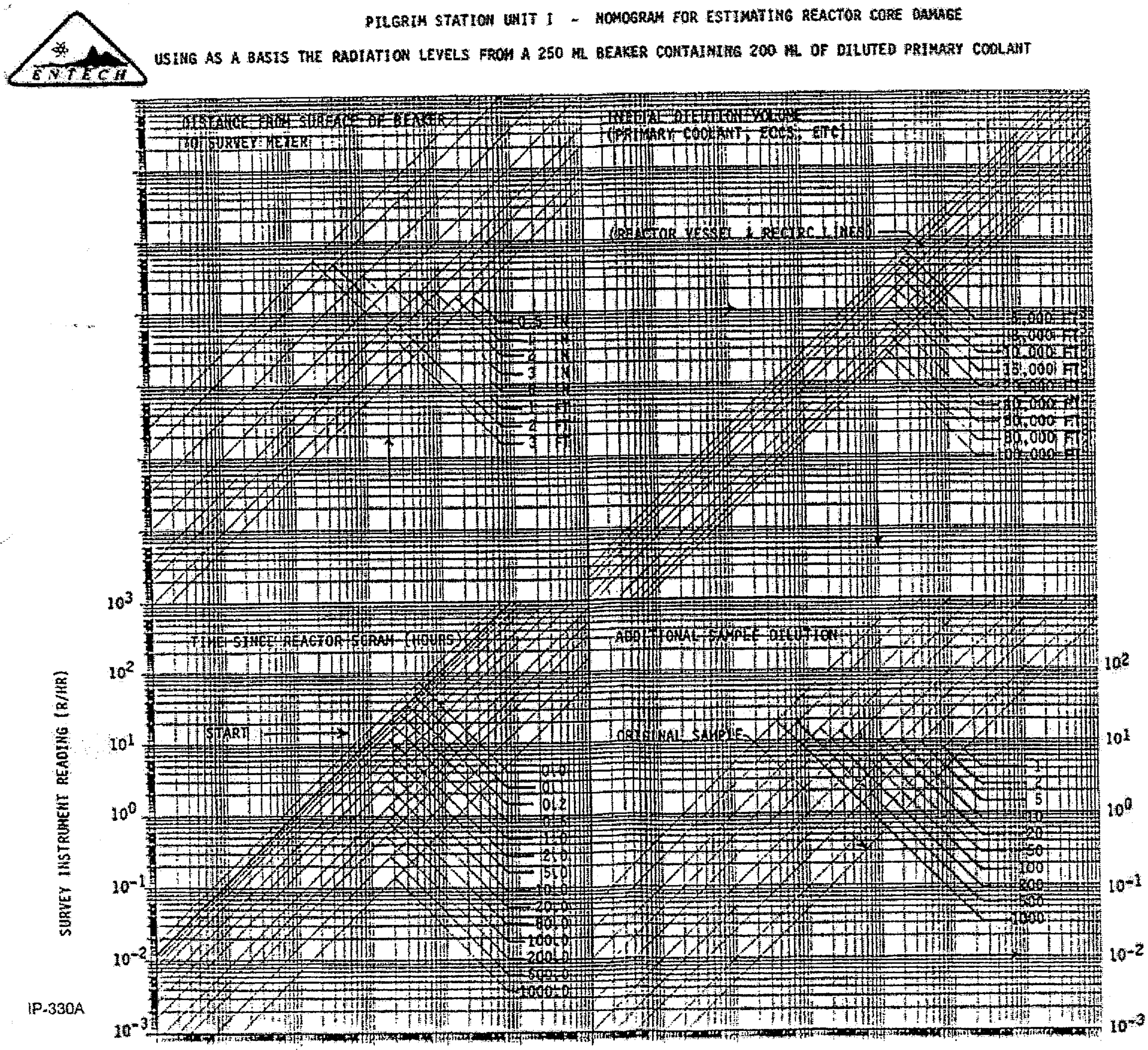
Monitor Reading (R/hr): ____________	Time After Shutdown (Hrs): ____________

Assessment Data:

Total Iodine Is: ☐ Above ☐ Equal to ☐ Below Technical Specifications

Name: ______________________________ Date: ____________ Time: ____________

CORE DAMAGE NOMOGRAM: PRIMARY COOLANT SAMPLE



IP-330A

DOCUMENT CROSS-REFERENCE

This Attachment lists those documents, other than source documents, which may be affected by changes to this Procedure.

Document Number	**Document Title**
EP-IP-100	Emergency Classification & Notification
EP-IP-400	Protective Action Recommendations
EP-AD-600	EAL Technical Bases Document
EP File 1.6.4	'DAMAGE' version 2.0 Computer Application Verification, Validation and Documentation" 11/22/95

IDENTIFICATION OF COMMITMENTS

This Attachment lists those external commitments (i.e., NRC commitments, QA audit findings, and INPO inspection items) implemented in this Procedure.

Reference Document	Commitment	Affected Section(s)/Step(s)
None		