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Exelon Nuclear

10 CFR 50, Appendix E

RS-03-208

November 5, 2003

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

Braidwood Station, Units 1 and 2
Facility Operating License NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Clinton Power Station, Unit 1
Facility Operating License NPF-62
NRC Docket Nos. STN 50-461

Dresden Nuclear Power Station, Units 2 and 3
Facility Operating License DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

LaSalle County Station, Units 1 and 2
Facility Operating License NPF-11 and NPF-18
NRC Docket Nos. 50-373 and 50-374

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Revisions to the Exelon Nuclear Standardized Radiological Emergency Plan Implementing Procedure

In accordance with 10 CFR 50, Appendix E, Section V, "Implementing Procedures," Exelon Generation Company, LLC (EGC) and AmerGen Energy Company (AmerGen) are submitting changes to several Emergency Plan procedures.

A045

Specifically, the following Exelon procedures have been revised:

- EP-AA-112-401, "Nuclear Duty Officer"
- EP-AA-110-301 "Core Damage Assessment (BWR)"
- EP-MW-110-200 "Dose Assessment."

Procedure EP-AA-112-401 has been revised to identify that the Nuclear Duty Officer (NDO) may be located in either the MidWest or MidAtlantic, but will respond to an event at any Exelon or AmerGen nuclear station. The activities of the NDO have been modified such an NDO in either region is capable of effectively responding to an event.

Procedure EP-MW-110-200 has been revised to reflect the adoption of the DAPAR dose assessment software. Corresponding changes to reflect the deletion of the previous dose assessment methodology have been incorporated. The adoption of this software allows a common dose assessment methodology and process between Exelon and the State of Illinois.

Procedure EP-AA-110-301 has been revised to reflect the implementation of the CDAM core damage assessment software at the Limerick and Peach Bottom stations. In addition, several administrative and editorial revisions were incorporated in response to user feedback.

The revised procedures are included in the attachments to this letter. These procedure changes were implemented between October 10 and October 22, 2003, and therefore must be submitted prior to November 9, 2003.

Should you have any questions concerning this letter, please contact Mr. T.W. Simpkin at 630-657-2821.

Respectfully,

Keith R. Jury

Keith R. Jury
Director – Licensing and Regulatory Affairs
Exelon Generation Company, LLC
AmerGen Energy Company, LLC

cc: Regional Administrator – NRC Region III (two copies)
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
NRC Senior Resident Inspector – Clinton Power Station
NRC Senior Resident Inspector – Dresden Nuclear Power Station
NRC Senior Resident Inspector – LaSalle County Station
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

Attachments:
Attachment A – Exelon Nuclear Procedure EP-AA-112-401, "Nuclear Duty Officer"
Attachment B – Exelon nuclear Procedure EP-MW-110-200, "Dose Assessment"
Attachment C – Exelon Nuclear Procedure EP-AA-110-301, "core Damage Assessment (BWR)"

ATTACHMENT A

EP-AA-112-401 NUCLEAR DUTY OFFICER

NUCLEAR DUTY OFFICER (NDO)

1. PURPOSE

1.1. This procedure describes the responsibilities and actions of the Exelon Nuclear Duty Officer (NDO), which is a designated "24 / 7" duty position in either the Mid-West or Mid-Atlantic Region.

When the Shift Manager decides that a situation warrants activation of the Emergency Response Organization (ERO) under the Emergency Plan, this procedure becomes applicable.

2. TERMS AND DEFINITIONS

None

3. RESPONSIBILITIES

3.1. The *Nuclear Duty Officer (NDO)* is responsible for functioning as the initial Exelon Nuclear Corporate Management contact when an emergency event is classified at an Exelon Nuclear station. The NDO shall decide the appropriate response for events not classified under the emergency plan.

3.2. The *NDO* is also responsible for interface with the State Duty Officers (or designated points of contact) regarding event information until the Corporate Emergency Director position is staffed.

4. MAIN BODY

4.1. **INITIATE** the appropriate Emergency Plan activities using the position checklist contained in Attachment 1.

5. DOCUMENTATION

None

6. REFERENCES

None

7. ATTACHMENTS

7.1. Attachment 1, Nuclear Duty Officer (NDO) Checklist

ATTACHMENT 1
NUCLEAR DUTY OFFICER (NDO) CHECKLIST
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Section 1, Event Classification

Section 2, Notification of Transportation Accident

Section 3, Hazardous Materials Emergency

Section 4, Activation of Electric Operations Emergency Load Conservation Program

1. **EVENT CLASSIFICATION**

NOTE: The NDO will receive an alpha page message from the Emergency Response Organization (ERO) Callout System identifying the affected station, event classification, and facilities being activated.

1.1. ____ **CONTACT** the affected Station Duty Manager to **VERIFY** and **OBTAIN** updated information concerning emergency response actions and event status.

1.2. ____ **NOTIFY** the Nuclear Duty Executive (NDE) and the Chief Nuclear Officer (CNO).

1.3. ____ **NOTIFY** the Exelon Nuclear Regional Communications Duty Officer of the event.

1.4. ____ **REVIEW** news releases created by Exelon Generation Communications & Public Affairs for accuracy prior to activation of the Emergency Public Information Organization.

1.5. ____ **RESPOND** to requests for information concerning the event from the State Duty Officer(s), if contacted.

1.6. ____ **UPDATE** the Exelon Nuclear Management, using the appropriate method listed below, until the EOF is in Command and Control or the event has been terminated:

1.6.1. ____ **MAINTAIN** the NDO Message voice mailbox up to date.

1.6.2. ____ **If** the event involves a Mid-Atlantic station, **then UPDATE** the Employee Emergency Notification Line for the affected station, using the instructions found in EP-MA-110-100.

1.7. ____ **MAINTAIN** a record of activities.

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NUCLEAR DUTY OFFICER (NDO) CHECKLIST
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1.8. For an Alert or higher classification, **PERFORM** the following:

1.8.1. ____ **NOTIFY** industry support organizations per the Reportability Manual, using the contact numbers listed in the ERF Telephone Directory:

- Institute of Nuclear Power Operations (INPO)
- American Nuclear Insurers (ANI)

1.8.2. ____ **TRANSFER** responsibility for notification of INPO and ANI to the EOF Logistics Manager once the EOF is activated.

Mid-West Region

Cantera NDO may relocate to the EOF to continue to assist with notifications to ANI, INPO and other support / governmental agencies.

2. **NOTIFICATION OF TRANSPORTATION ACCIDENT**

NOTE: A Transportation Accident is defined in 49 CFR 171.15 and 49 CFR 171.16.

2.1. ____ **REVIEW** OP-AA-106-102, "Accidents or Incidents Involving the Transportation of Rad Material."

2.2. ____ **CONTACT** the affected Station Duty Manager to get information on the event.

2.3. ____ **CONTACT** Exelon Nuclear Communication Services and coordinate press releases if necessary.

2.4. ____ **CONTACT** INPO to initiate the Voluntary Assistance Program, if necessary.

2.5. ____ **CONTACT** American Nuclear Insurers (ANI) to provide them with status of the event.

2.6. ____ **MAINTAIN** a record of activities.

ATTACHMENT 1
NUCLEAR DUTY OFFICER (NDO) CHECKLIST
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MID-WEST Region

3. ACTIVATION OF ELECTRIC OPS EMERGENCY LOAD CONSERVATION PROGRAM

3.1 ____ **NOTIFY** Distribution Dispatch Center (DDC)/Load Dispatcher that a station is in an emergency condition. DDC will maintain power to that station's Emergency Planning Zone (EPZ) for the Alert and Notification System (ANS) sirens.

3.2 ____ **NOTIFY** the ANS siren maintenance vendor of possible rolling power outages, and **INSTRUCT** the vendor to monitor siren performance.

3.3 ____ **TRANSFER** the ELCP Bridge phone call to the Corporate Operations Center.

3.4 ____ **UPDATE** the NDO Message voice mailbox. **MAINTAIN** the NDO Message voice mailbox up to date until the EOF is activated or the event has been terminated.

3.5 ____ **If** support from the Site Restoration Management Team at a nuclear station is necessary, **then COORDINATE** plant access for the Team.

3.6 ____ **If** the ANS siren maintenance vendor was notified of rolling power outages and the outages have been terminated, **then NOTIFY** them that the outages have been terminated.

3.7 ____ **If** DDC/Load Dispatch was notified of an emergency condition, and the emergency is terminated, **then NOTIFY** the DDC/Load Dispatch that there is no longer an emergency condition at the affected station.

ATTACHMENT B

EP-MW-110-200 DOSE ASSESSMENT

DOSE ASSESSMENT

1. PURPOSE

This procedure provides the methods and instructions for performing offsite dose assessment and projection by the Emergency Response Organization.

2. TERMS AND DEFINITIONS

2.1 **Centerline (plume):** An imaginary line drawn in the middle of the plume along its downwind travel direction. The plume concentrations and deposition are assumed to be the highest along the centerline

2.2 **Cloud Shine:** Gamma radiation from radioactive materials in the air (plume)

2.3 **Committed Dose Equivalent (CDE):** The internal dose equivalent to parts of the body that will be received from an intake of radioactive material by an individual over a 50-year period of time.

2.4 **Committed Effective Dose Equivalent (CEDE):** The sum of the internal dose equivalent for 50 years following intake (inhalation or ingestion) of a radionuclide to each organ multiplied by a weighting factor.

2.5 **Core Damage:** Damage to the components that comprise the reactor core. Core damage typically refers to the failure of fuel cladding and/or fuel melting as a result of overheating.

2.6 **Curie (Ci):** A unit of radioactivity equal to 3.7E+10 disintegrations per second.

2.7 **DAPAR:** Exelon Dose Assessment and Protective Action Recommendations (DAPAR) software provides two major functions (Quick Assessment and Full Assessment) in order to perform dose assessment.

A. **Quick Assessment** is used by the Control Room to arrive at offsite dose projections and PARs, or to verify classifications in as quick a time as possible during fast breaking events without taking too much time away from their event mitigating actions. A monitored release is the only method used in the quick assessment. Some assumptions and standard numbers are used to limit the amount of data Control Room personnel must enter prior to calculating a PAR.

B. **Full Assessment** is used by the called-in ERO Staff in the TSC/EOF and allows for more detailed assessment of a release. The following methods may be used to project offsite doses:

- **Monitored Release**: Offsite radiological assessment related to a monitored value taken at one of several release locations (Plant Vent Stack, Waste Processing Vent Stacks and Turbine Building Vent Stack) within the plant.

- **Containment Leakage/Failure**: Offsite radiological assessment related to a default, known, or predicted level of containment leakage or failure.

- **Field Team Survey and Sample Analysis**: Offsite radiological assessment related to comparisons of field team radiological survey and isotopic sample concentrations with predicted plume dispersion.

- **Release Point Sample Analysis**: Offsite radiological assessment related to a measured isotopic concentration taken at the point of release to the environment.

- **GEMS Analysis**: Allows for offsite dose assessment using isotopic release rates (uCi/sec of each isotope) provided by the State of Illinois (Division of Nuclear Safety) Gaseous Effluent Monitoring System.

2.8 **Delta T:** The difference in temperature from the lower temperature sensor and the upper temperature sensor on the Exelon meteorological tower. Delta-T is used to calculate stability class.

2.9 **Deposition:** The contamination found on the surface of the ground.

2.10 **Dose Commitment:** The dose that will be accumulated by a specific organ over a specified period following uptake.

2.11 **Dose Conversion Factor (DCF):** The dose equivalent per unit intake of a radionuclide (mrem/uCi) or the effects of exposure to a given concentration of an isotope in a plume. R/hr per uCi/cc.

2.12 **Dose Projection:** The calculation of individual radiation exposure at a given location at some time in the future. Dose projections are performed in response to an actual or anticipated release of radioactive material to the environment.

2.13 **Effective Dose Equivalent (EDE):** The sum of the dose equivalent from external exposure to each organ multiplied by a weighting factor. EDE is used to estimate the risk of delayed health effects.

2.14 **Emergency Planning Zone (EPZ):** An area around a nuclear power plant in which plans are in place for an emergency at the plant. Plans are in place to take immediate protective actions for individuals located within 10 miles of the Nuclear Plants. This area is called the Plume Exposure Emergency Planning Zone. In addition, longer-term plans are in place for the Ingestion Pathway Emergency Planning Zone which is within 50 miles of the plant.

2.15 **Evacuation Exposure Period:** The period during which those being evacuated are exposed to the radioactive plume.

2.16 **Millirem (mR):** One one-thousandth of a Rem. The Rem is a unit of measure that defines the extent of biological injury that results to the body when it is exposed to radiation.

2.17 **Pilot Operated Relief Valve (PORV):** A valve which serves to reduce pressure in the reactor coolant system or main steam system by allowing steam to escape from the Pressurizer or the steam generators. The PORVs can be operated remotely by Plant Operators or automatically by high pressure.

2.18 **Plant Parameter Display System (PPDS):** Electronic graphical display of plant, meteorological and radiological data needed for accident and dose assessment.

2.19 **Protective Action Guidelines (PAGs):** Radiation exposure guidelines established by the Environmental Protection Agency which are used to determine the appropriate protective actions to be taken on the part of emergency workers and the general public. These actions include sheltering and evacuation.

2.20 **Protective Action Recommendations (PARs):** A recommendation made by Exelon personnel to the offsite authorities on the appropriate protective actions to be taken on the part of the general public. The PARs are based on plant conditions or dose projections using the PAGs for guidance.

2.21 **Safety Relief Valve:** A valve that serves to reduce pressure in a fluid system should the pressure become too high. Both the reactor coolant system (located on pressurizer) and the main steam system (located on steam generators) have safety and relief valves to protect them from being damaged by excessive pressure.

2.22 **Site Boundary:** Defined as a circle with a radius of ¼ or ½ mile (depending on the site specific ODCM) and the containment building as its center.

2.23 **Station Vent:** That part of the plant's ventilation system through which the containment building and auxiliary building air may be processed to the outside atmosphere. The discharge of the station vent is continuously monitored for abnormal amounts of radiation and would be isolated long before radiation levels approach federal limits.

2.24 **Subareas:** Pre-designated areas offsite in which Protective Actions such as evacuation of sheltering will be performed.

2.25 **Total Effective Dose Equivalent (TEDE):** A method of converting exposure to radiation to the biological effects that it will cause to the human body. It combines the external and internal ionizing radiation exposure. The TEDE is the sum of Deep Dose Equivalent and Committed Effective Dose Equivalent.

3. RESPONSIBILITIES

3.1.1 The **Shift Manager**, or designated on-shift individual, shall perform required dose assessments prior to responsibility being transferred to either the Technical Support Center (TSC) or Emergency Operations Facility (EOF).

3.1.2 The **TSC Radiological Controls Coordinator** shall relieve the Control Room and perform required assessments if the transfer of PAR / dose assessment responsibilities to the EOF is delayed.

3.1.3 The **EOF Dose Assessor** shall relieve the TSC Radiological Controls Coordinator when directed by the EOF Dose Assessment Coordinator, and perform required dose assessments. Responsibility for dose assessments can be assumed directly from the Control Room.

4. MAIN BODY

4.1 Dose Assessment and Protective Action Recommendation (DAPAR)

- **REFER** to Attachment 1 for user guidelines.

4.2 Obtaining IDNS Monitor Data from the EOF

- **REFER** to Attachment 2 for user guidelines.

5. DOCUMENTATION

None

6. REFERENCES

6.1 EP-MW-123-1002, "Dose Assessment and Protective Action Recommendation (DAPAR) Program Technical Basis".

7. ATTACHMENTS

7.1 Attachment 1, DAPAR Users Guide

7.2 Attachment 2, Obtaining IDNS Monitor Data from the EOF

ATTACHMENT 1

DAPAR USERS GUIDE
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1. OVERVIEW

1.1 As a Windows based application designed in Access, DAPAR, uses many standard user interfaces. Instructions are not provided in basic computer operations in the Windows® environment. The user must be familiar with these to efficiently operate the program.

1.2 It is also assumed user is familiar with health physics fundamentals. Emergency Response Organization training will provide an overview of dose assessment methodologies.

1.3 DAPAR Program Use: The program is to be used to estimate the offsite consequences of a release or potential release of radioactive materials from an Exelon Station during an emergency. The primary purpose of these dose projections is to arrive at a Protective Action Recommendation given by Exelon management to offsite authorities. These PARs will be used by those authorities in their decision making process to take actions to protect the general public.

1.4 Limitations and Pre-Conditions for use DAPAR:

1.4.1 The program should not be used to calculate the actual dose received by populations exposed to radioactive materials from a release. Results may be used as part of the post accident investigations, but a much more in-depth analysis is needed to actually assign doses received by members of the public.

1.4.2 DAPAR should be used only when an emergency has been. The program makes many conservative assumptions to ensure proper actions are taken offsite prior to exposing the general public to any release of radioactive materials.

1.4.3 DAPAR release paths are based on the generalized BWR/PWR gaseous effluent pathways described in NUREG-1228. The Process Reduction Factor (PRF) of any additional release path(s) should be approximated by using one of the existing DAPAR pathways.

NOTE: Use of the DAPAR program to project doses based on routine plant readings would indicate offsite doses many magnitudes higher than actual offsite doses. Care should be taken in making a Protective Action Recommendation or notification based on DAPAR output if there are no indications of Core Damage (ODCM calculations should be used to calculate offsite doses when no core damage is expected).

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NOTE: There is a different version of DAPAR for each Station. Each version takes into account different release points and gaseous effluent radiation monitor conversion factors. Always verify that the correct station version of the program is being used.

2. START UP

2.1 Start the computer.

2.2 The application is accessed by one of the folowing:

2.2.1 Open the DAPAR folder desktop icon on applicable dose assessment computers.

1. Start the appropriate DAPAR program for the plant that has declared an emergency.
2. Programs are labeled <*Station Name*> - DAPAR v2.0.

2.2.2 Select RUN from the the 'Start Bar' and type in the file path and name as follows:

- C:\DAPAR\Braidwood DAPAR v2_0.mdb
- C:\DAPAR\Byron DAPAR v2_0.mdb
- C:\DAPAR\Clinton DAPAR v2_0.mdb
- C:\DAPAR\Dresden DAPAR v2_0.mdb
- C:\DAPAR\LaSalle DAPAR v2_0.mdb
- C:\DAPAR\Quad Cities DAPAR v2_0.mdb

2.3 **IF** the assigned Dose Assessment Computer cannot access the application or the DAPAR program will not run, **THEN** Install DAPAR on any computer from CDs or Disks located in the Control Room, TSC or the EOF Library. DAPAR is installed by copying appropriate file to computer's hard drive.

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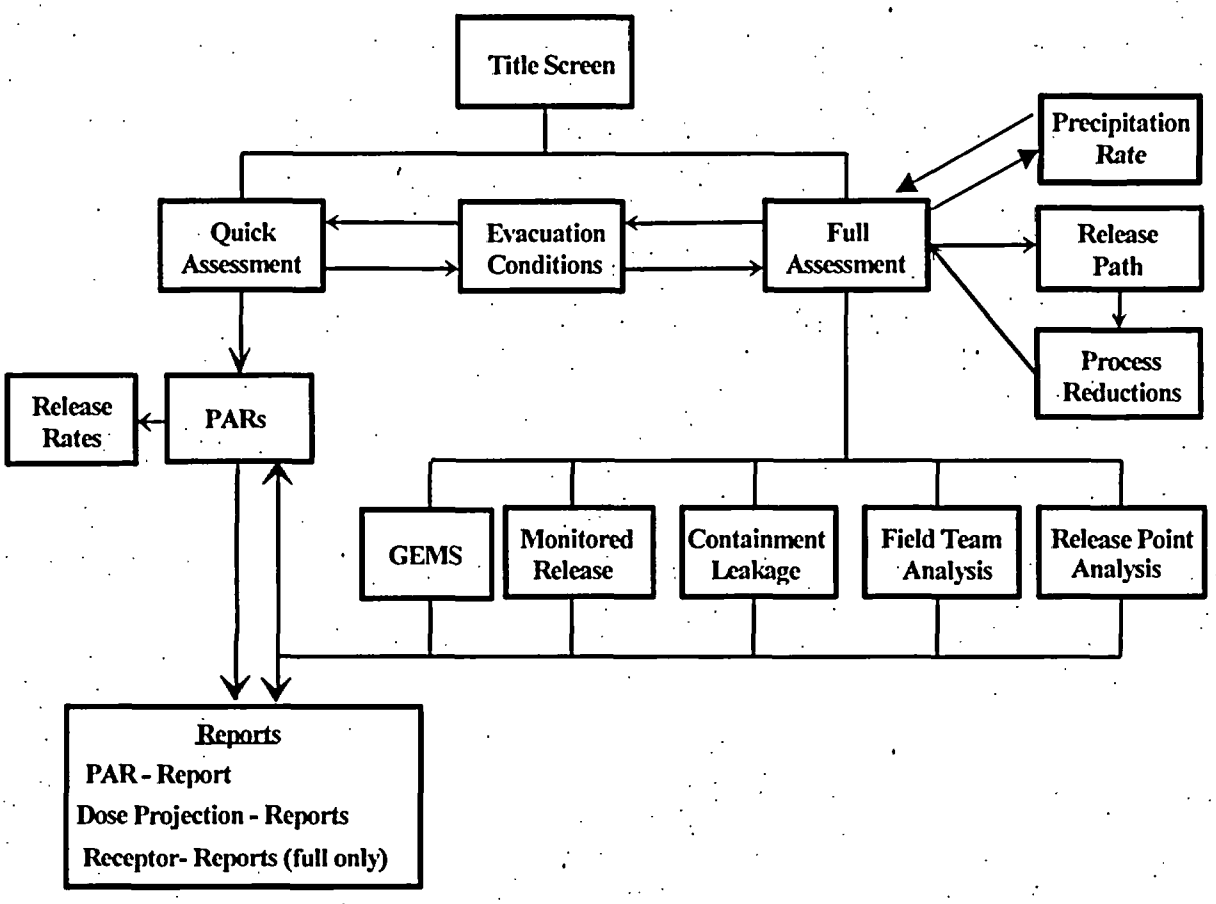
3. BASIC PROGRAM FLOW DIAGRAM



The above diagram shows basic tasks that can be performed by the DAPAR program and how a user would navigate between them.

NOTE: Values for specific plant data is obtained from the Plant Parameter Display System (PPDS) or the affected unit's Control Room.

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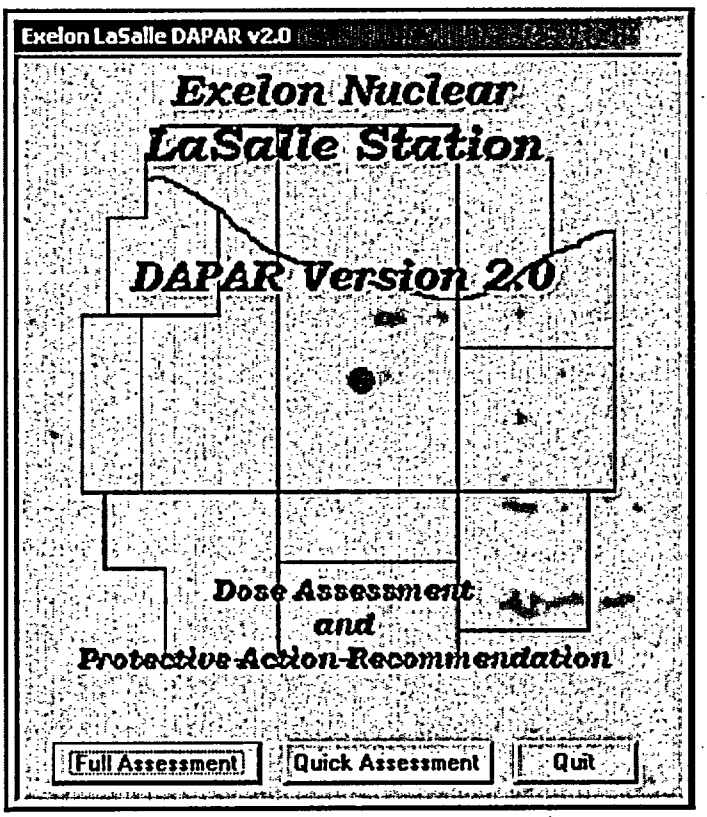
4. TITLE SCREEN
The title screen shows the application version and offers the user three options to direct program flow.

4.1 **Quick Assessment** - This option is designed to be used by the Control Room and TSC. It performs assessments based on design basis default source terms.

4.2 **Full Assessment** - This option is designed to be used by the Dose Assessment Staff in the TSC and EOF. It allows user more options in performing calculations.

4.3 **Quit** - Exits the Program.



NOTE: Once the User selects "Quick Assessment" or "Full Assessment," returning to the title screen will reset all program values.

4.4 **SELECT** either "Full Assessment" or "Quick Assessment" and then **GO TO** either:

4.4.1 Section 5 of this procedure for Quick Assessment.

4.4.2 Section 7 of this procedure for Full Assessment.

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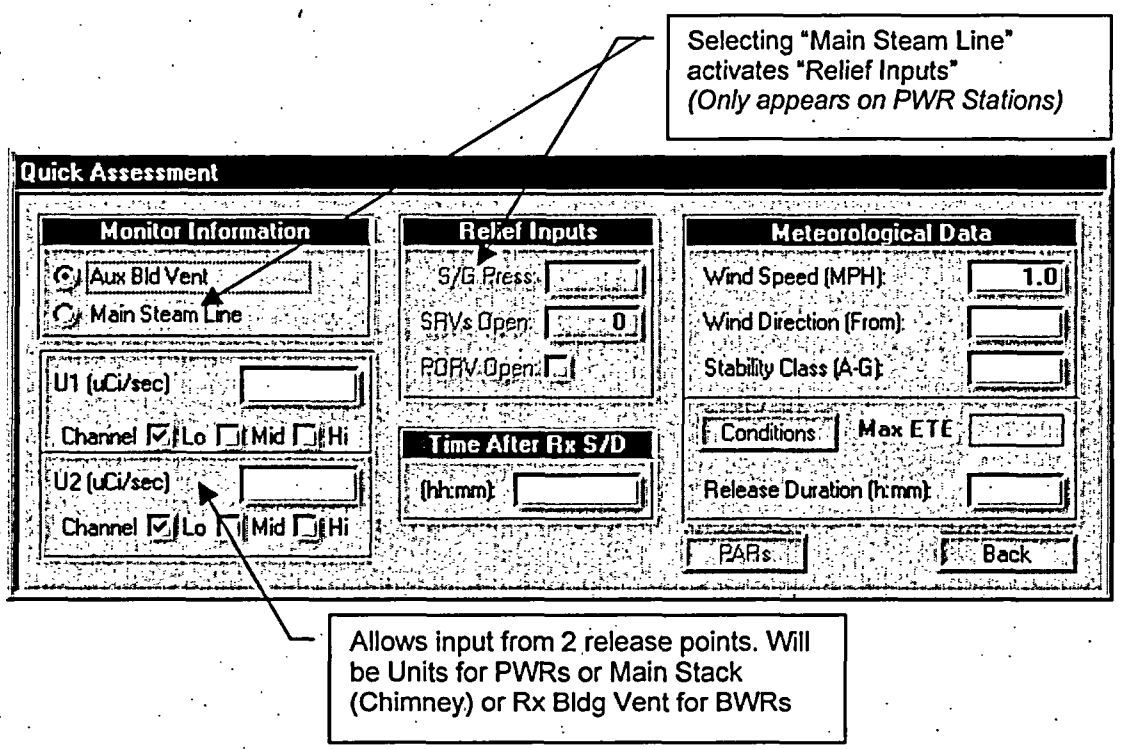
5. QUICK ASSESSMENT

The Quick Assessment operations and calculations are identical to the Full Assessment method for a monitored release, but utilizes default release path and core damage assumptions for the determination of offsite doses. This allows for a rapid assessment from the Main Control Room.



NOTE: User must choose the appropriate Channel, "Lo", "Mid", or "Hi". The same µCi/sec reading for the different channels can make a significant difference in projected doses offsite. Channel selection information is obtained from the Process Book Plant Data Displays or the Control Room.

5.1 *Monitor Information* – User chooses the appropriate monitor from the listed effluent monitors.

5.1.1 For **PWRs** the following choices are available:

1. Aux Bldg. Vent – **SELECT** this option for releases from the Auxiliary Building Vent.

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2. Main Steam Line – **SELECT** this option for releases from the Main Steam Line.

   – The Main Steam Line radiation monitors read out in mr/hr, therefore a flow has to be determined to calculate the uCi/sec release rate.

   – This is accomplished by entering the steam pressure and the number of relief valves that are open in the section labeled *Relief Inputs*.

5.1.2 For **BWRs**, normally there will be no choice available or needed here. Dresden Station DAPAR allows choice for a Unit 1 release point.

5.2 **ENTER** *Reading* Information – Enter the appropriate monitor reading in µCi/sec or mr/hr. Choose the appropriate channel, "Lo", "Mid" or "Hi". Both µCi/sec readings should be entered, if one reading is ≥ 1000 times the other, 0 may be entered for smaller number.

   – **IF** using Main Steam Line monitor release path **THEN** enter Steam Generator Pressure, Number of Relief Valves Open and PORV open or shut.

5.3 **ENTER** *Time After Shutdown* Information – Enter the time since the reactor was shutdown in hours and minutes (hh:mm).

5.4 **ENTER** *Meteorological Data* – Enter the appropriate data from plant instruments as follows:

NOTE: Met Data is available from the Plant Parameter Display System (PPDS) screens or the Control Room. If Met Tower data is unavailable from these locations, another source of meteorological data may be used such as the Meteorological Vendor, National Weather Service or a local TV or Radio broadcast stations. The stability class can be estimated from the Table 1-1 or Table 1-2 if Meteorological Vendor provides $\Delta T/\Delta z$ or σ_θ:

5.4.1 Wind Speed (MPH) – Obtain and enter wind speed in Miles per Hour (MPH)

5.4.2 Wind Direction (From) – Obtain and enter the direction the wind is coming FROM in degrees. (0°-360°)

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5.4.3 Stability Class (A-G) – Obtain and enter the stability class.

- A, Extremely unstable conditions
- B, Moderately unstable conditions
- C, Slightly unstable conditions
- D, Neutral conditions
- E, Slightly stable conditions
- F, Moderately stable conditions
- G, Extremely stable conditions

If the stability class is not available use the following tables to choose appropriated value:

Table 1-1
Stability Class Determination

Surface Wind Speed (mph)	Daytime Solar Radiation (For moderate cloud cover move one column to the right)			Nighttime Conditions			
	Summer Clear Sky	Spring/Fall Clear Sky	Heavy Overcast Rain	Winter	Thin overcast (>1/2 cloud cover)	< 3/8 cloud cover	Heavy Overcast Rain
< 9.0	A	A-B	D	B			D
to 9.0	A-B	B	D	C	E	F	D
to 13.5	B	B-C	D	C	D	E	D
> 13.5	C	C-D	D	D	D	D	D

Table 1-2
Stability Class Determination

Class	$\Delta T/\Delta z$ (°C/100m)	σ_θ (degrees)
A	≤ -1.9	≥22.5
B	> -1.9 to ≤ -1.7	<22.5 to ≥17.5
C	> -1.7 to ≤ -1.5	<17.5 to ≥12.5
D	> -1.5 to ≤ -0.5	<12.5 to ≥7.5
E	> -0.5 to ≤ +1.5	<7.5 to ≥3.8
F	> +1.5 to ≤ +4.0	<3.8 to ≥2.1
G	> +4.0	0.0 to <2.1

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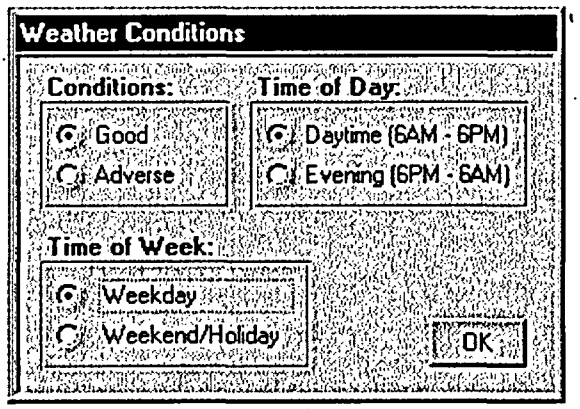
NOTE: The conditions Good or Adverse in the following step are in relation to the weather. Adverse is heavy rain or any other condition that would hinder the flow of traffic.

5.5 **Set Evacuation Conditions** - Click on the *Conditions* button to open the Weather Conditions Window.



5.6 **SELECT** the appropriate conditions for the program to calculate the Maximum Evacuation Time Estimate (ETE). Once the user sets the evacuation conditions, the program will place the Max ETE value in the *Release Duration* and *Max ETE* text boxes.

- Not all choices are available for every station or condition. The Program only allows appropriate choices.

5.7 **IF** the exact release duration is known, **THEN** change the displayed time to the known release duration. If a good estimate of the release duration cannot be determined, use the default ETE value entered by the program.

5.8 **SELECT** the *PARs* button – The program will calculate the downwind doses based on user inputs and display Protective Action Recommendation Window.

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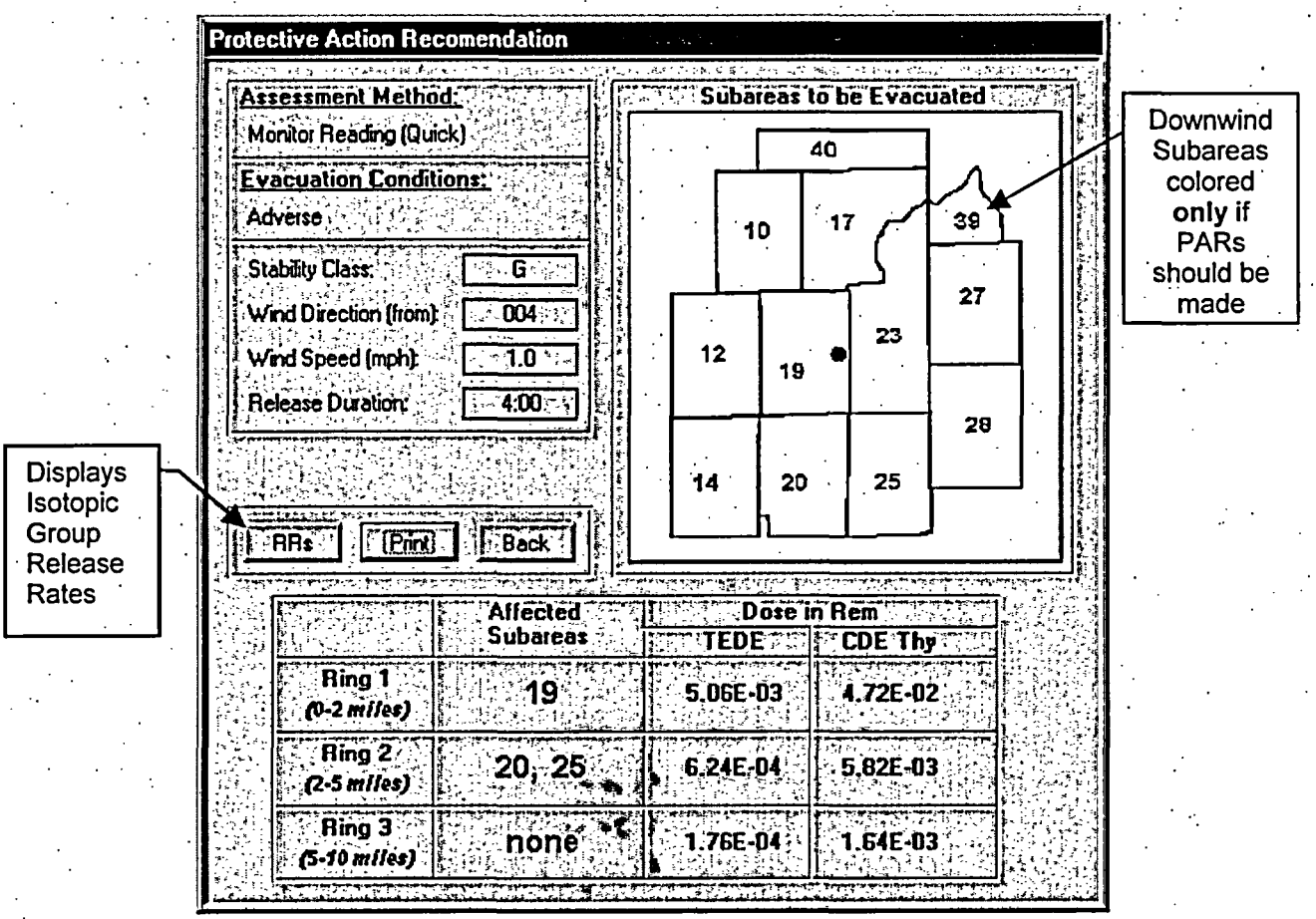
6. PROTECTIVE ACTION RECOMMENDATIONS

6.1 The Protective Action Recommendation (PAR) form displays a summary of the downwind dose projections with a map showing which Subareas (colored areas) where Protective Actions Recommendations should be made.



6.2 Explanation of displayed data:

6.2.1 Assessment Method – Method used to calculate downwind doses.

6.2.2 Evacuation Conditions – Entered evacuation conditions and meteorological data, along with Release Duration (form displays the meteorological data use to determine PAR).

6.2.3 Subareas to be Evacuated - Form displays an EPZ map. Subareas where the population may receive doses exceeding a PAG are colored.

6.2.4 Affected Areas – This is the downwind Subareas that are affected by the release.

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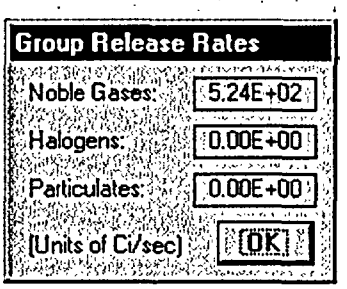
6.2.5 TEDE and CDE Thy – Shows the highest doses (no protection and sheltered) in Rem for each Ring.

6.3 The *RR* button will display the total release rates for isotopic groups in Ci/sec.

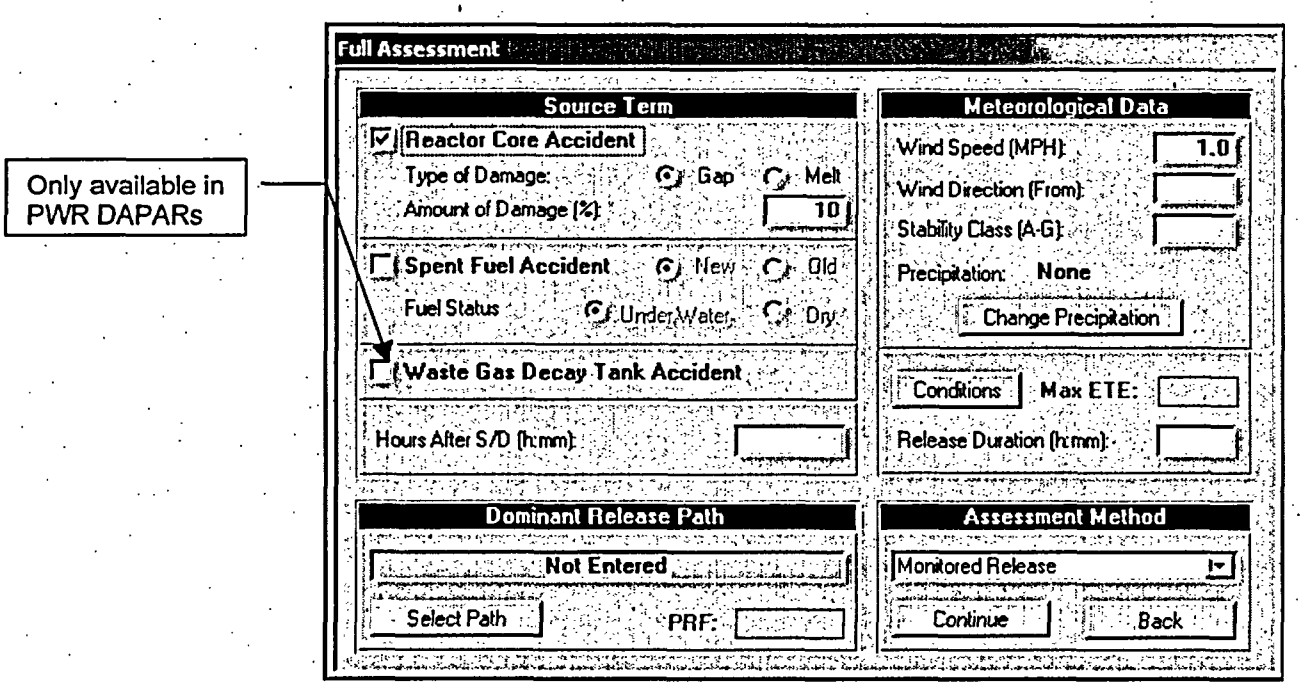


6.4 User can **SELECT** *Print* to print the PAR report or **SELECT** the *Go Back* button and modify inputs. This will return user to either Quick Assessment Form or one of the Assessment Method forms available in the Full Assessment mode.

7. FULL ASSESSMENT

The Full Assessment operations and calculations are identical to the Quick Assessment method for a monitored release, but it allows the user to make more choices in performing dose projection calculations.

Choosing the Full Assessment option directs the program to a baseline data entry window. The window is divided into four input areas.



7.1 **Source Term** – This allows user to choose the appropriate source term depending on plant conditions and the type of accident that has occurred.

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7.1.1 **SELECT** *Reactor Core Accident* if the source of the release is from the reactor core.

- **SELECT** *Gap* or *Melt* and **ENTER** the *% Damage* based on core damage estimates or known conditions in the plant.

1. **SELECT** *Fuel Handling Accident* if the release is caused by damage to the spent fuel.

   - **CHOOSE** between *New Fuel* or *Old Fuel* based on the type of fuel that has been damaged. *New Fuel* refers to fuel freshly transferred from the reactor to the fuel pool (i.e., program uses an estimate of the minimum time after shutdown allowed by the station before beginning transfer of fuel from the reactor). *Old Fuel* refers to fuel transferred to the fuel pool at an earlier outage (i.e., program uses a time after shutdown corresponding to the estimated minimum time between refuel outages).The program uses a gap release scenario and defaults to a reactor Time After Shutdown based on this choice.

NOTE: The choice of Waste Gas Decay Tank Accident is only available for PWRs.

2. **SELECT** *Waste Gas Decay Tank Accident* if the release is caused by damage to a failure of a waste gas decay tank. The program sets source to one failed Waste Gas Decay Tank inventory.

7.2 **Dominant Release Path** – This allows user to choose the most appropriate release path:

NOTE: DAPAR release paths are based on the generalized BWR/PWR gaseous effluent pathways described in NUREG-1228. The Process Reduction Factor (PRF) of any additional release path(s) should be approximated by using one of the existing DAPAR pathways.

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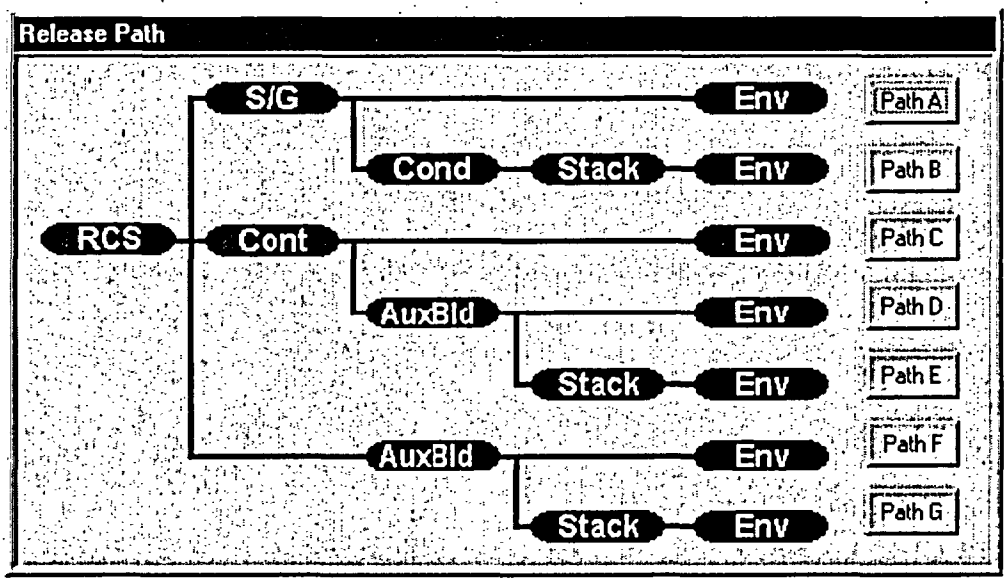
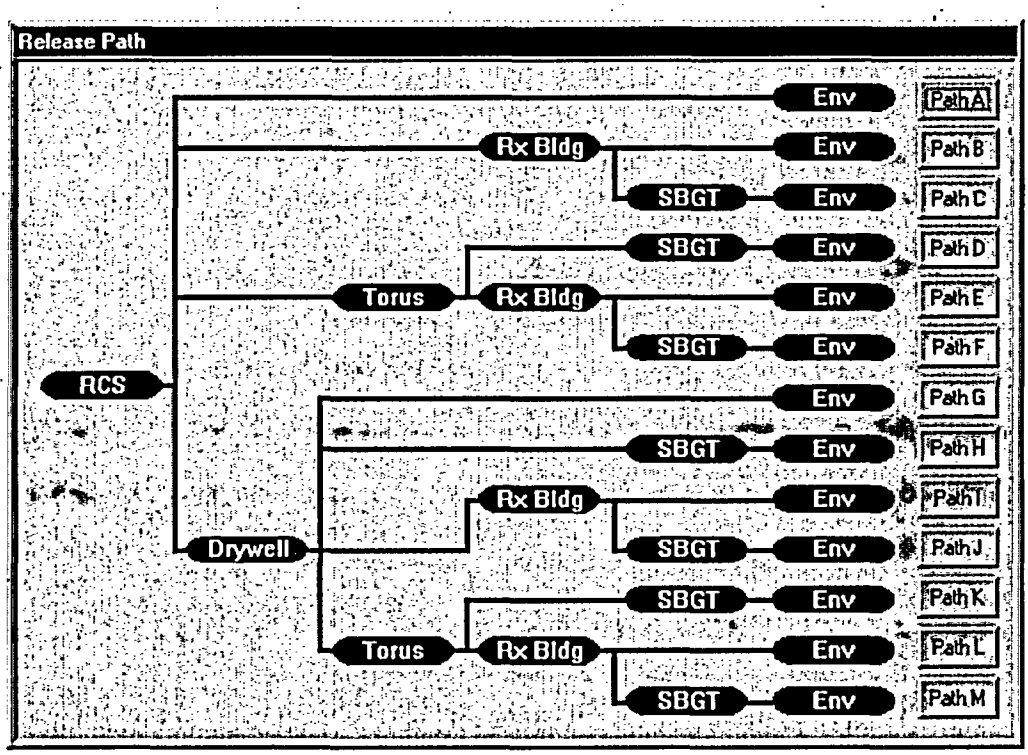
7.2.1 PWR Paths:



7.2.2 BWR Paths:

NOTE: Dresden Station DAPAR has one additional path (N) through the Isolation Condenser.



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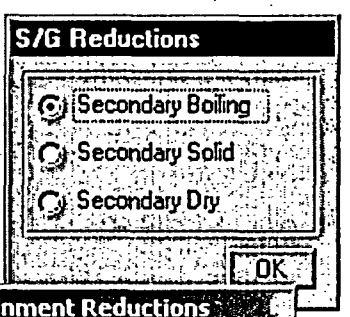
7.2.3 Depending on the path chosen, the user will be presented with more choices to pick the appropriate Process Reductions for the release.

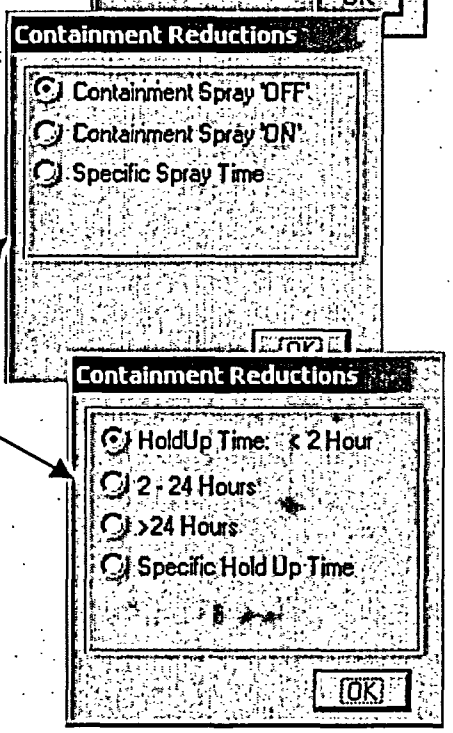
1. If a release through the Steam Generators is chosen:

   – Determine the status of the secondary side of the steam generator and select the appropriate condition.



2. If a release through containment or drywell is chosen:

   – First determine if containment sprays have been used since release of fission product materials to Containment.

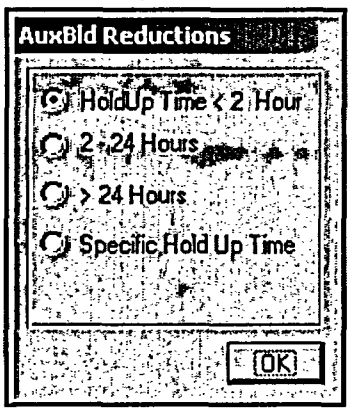


Representative Screens similar for PWR or BWR

   – Then determine Containment / Drywell holdup time.

3. If a release through the Aux Bldg or Rx Bldg.

   – Determine the RAB holdup time.



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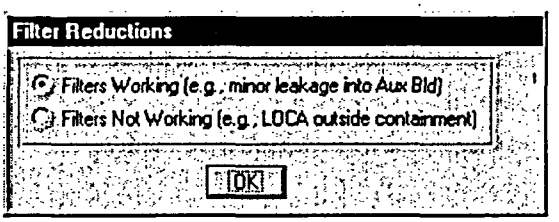
4. If a release through a filtered Vent or SBGT.

   – Choose if the filters are working or not.



   – If the release has been ongoing for a long time or contains a large amount of liquids filters may not be working.

NOTE: BWR DAPARs also have a choice for Torus or Suppression Pool status, "Subcooled", "Saturated" or "Bypassed".

7.3 **ENTER** *Meteorological Data* – Enter the appropriate data from plant instruments.

NOTE: Met Data is available from the Plant Parameter Display System (PPDS) screens or the Control Room. If Met Tower data is unavailable from these locations, another source of meteorological data may be used such as the Meteorological Vendor, National Weather Service or a local TV or Radio broadcast stations. The stability class can be estimated from the Table 1-1 or Table 1-2 if Meteorological Vendor provides $\Delta T/\Delta z$ or σ_θ:

7.3.1 Wind Speed (MPH) – Obtain and enter wind speed in Miles per Hour (MPH).

7.3.2 Wind Direction (From) – Obtain and enter the direction the wind is coming FROM in degrees. (0°-360°)

7.3.3 Stability Class (A-G) – Obtain and enter the stability class.

- A, Extremely unstable conditions
- B, Moderately unstable conditions
- C, Slightly unstable conditions
- D, Neutral conditions
- E, Slightly stable conditions
- F, Moderately stable conditions
- G, Extremely stable conditions

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If the stability class is not available use the following tables to choose appropriated value:

Table 1-1
Stability Class Determination

Surface Wind Speed (mph)	Daytime Solar Radiation (For moderate cloud cover move one column to the right)			Nighttime Conditions			
	Summer Clear Sky	Spring/Fall Clear Sky	Heavy Overcast Rain	Winter	Thin overcast (>1/2 cloud cover)	< 3/8 cloud cover	Heavy Overcast Rain
< 9.0	A	A-B	D	B			D
to 9.0	A-B	B	D	C	E	F	D
to 13.5	B	B-C	D	C	D	E	D
> 13.5	C	C-D	D	D	D	D	D

Table 1-2
Stability Class Determination

Class	$\Delta T/\Delta z$ (°C/100m)	σ_θ (degrees)
A	≤ -1.9	≥22.5
B	> -1.9 to ≤ -1.7	<22.5 to ≥17.5
C	> -1.7 to ≤ -1.5	<17.5 to ≥12.5
D	> -1.5 to ≤ -0.5	<12.5 to ≥7.5
E	> -0.5 to ≤ +1.5	<7.5 to ≥3.8
F	> +1.5 to ≤ +4.0	<3.8 to ≥2.1
G	> +4.0	0.0 to <2.1

7.4 Set Precipitation Conditions - Click on the *Change Precipitation* button to open Precipitation Window.

None – No rain or snow

Light Rain – Light Drizzle < 0.1 inchs per hour

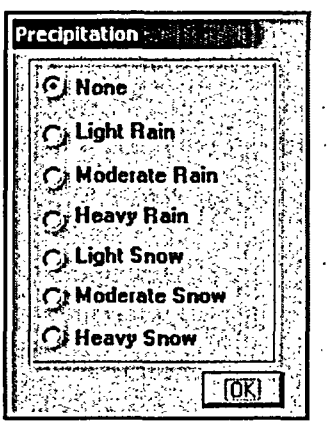
Moderate Rain – Heavy Drizzle 0.1 to 0.3 inches per hour

Heavy Rain – Greater than 0.3 inches per hour

Light Snow – Visibility 0.63 miles or greater

Moderate Snow – Visibility 0.31 to 0.63 miles

Heavy Snow – Visibility < 0.31 miles



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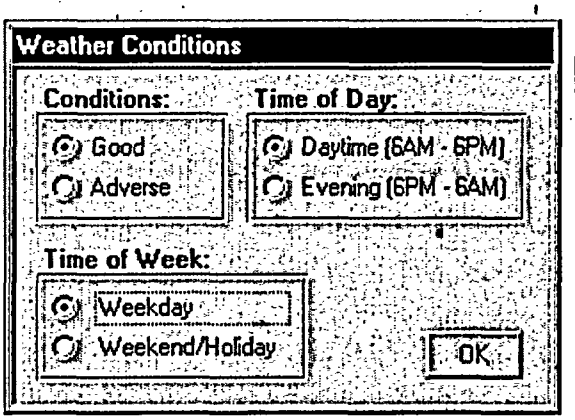
NOTE: The conditions Good or Adverse in the following step are in relation to the weather. Adverse is heavy rain or any other condition that would hinder the flow of traffic.

7.5 **Set Evacuation Conditions** - Click on the *Conditions* button to open the Weather Conditions Window.



7.6 **SELECT** the appropriate conditions for the program to calculate the Maximum Evacuation Time Estimate (ETE). Once the user sets the evacuation conditions, the program will place the Max ETE value in the *Release Duration* and *Max ETE* text boxes.

– Not all choices are available for every station or condition. The Program only allows appropriate choices.

7.7 **IF** a good estimate of the release duration (i.e., how long release is expected to last) cannot be determined, use the default ETE value entered by the program. The release duration is then assumed to be the time it takes for the population to be evacuated from the affected area. **IF** the exact release duration is known, **THEN** change the displayed time to the known release duration.

NOTE: User may switch back and forth between assessment methods as more information becomes available or conditions change. With the exception of the Time After Shutdown, which updates each time user returns to main form, the data on the Full Assessment form will not change unless user changes it.

7.8 *Assessment Methods* – **CHOOSE** the appropriate assessment method based on available inputs. The Assessment methods are:

7.8.1 *Monitored Release* – **SELECT** this method for a release through a plant vent or through the Main Steam Relief Valves. **GO TO** Section 8.

7.8.2 *Containment Leakage* – **SELECT** this method for containment failure scenarios. **GO TO** Section 9.

7.8.3 *Field Team Data* – **SELECT** this method if field team survey or sample data is available. **GO TO** Section 10.

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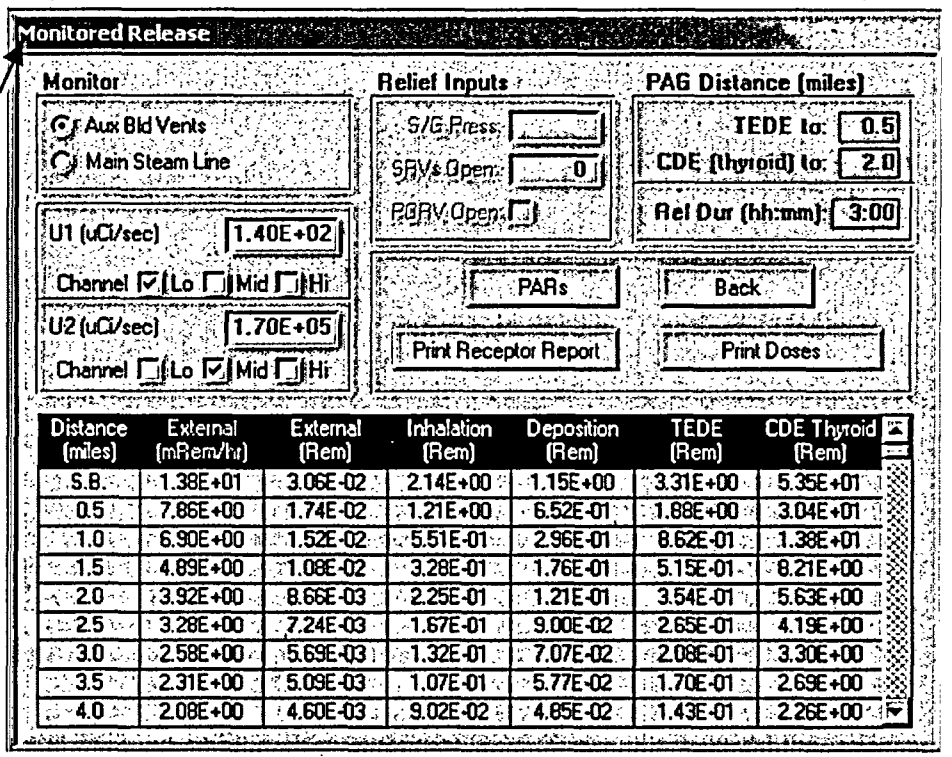
7.8.4 *Release Path* – **SELECT** this method for a sample of a release has been obtained and a release flow rate can be estimated. **GO TO** Section 11.

7.8.5 *GEMS Analysis*– **SELECT** this method for when entering release rate from the GEMS monitors. **GO TO** Section 12.

8. MONITORED RELEASE

Sample of Monitor Release Screen. Each Station's DAPAR has slight variations



Distance (miles)	External (mRem/hr)	External (Rem)	Inhalation (Rem)	Deposition (Rem)	TEDE (Rem)	CDE Thyroid (Rem)
S.B.	1.38E+01	3.06E-02	2.14E+00	1.15E+00	3.31E+00	5.35E+01
0.5	7.86E+00	1.74E-02	1.21E+00	6.52E-01	1.88E+00	3.04E+01
1.0	6.90E+00	1.52E-02	5.51E-01	2.96E-01	8.62E-01	1.38E+01
1.5	4.89E+00	1.08E-02	3.28E-01	1.76E-01	5.15E-01	8.21E+00
2.0	3.92E+00	8.66E-03	2.25E-01	1.21E-01	3.54E-01	5.63E+00
2.5	3.28E+00	7.24E-03	1.67E-01	9.00E-02	2.65E-01	4.19E+00
3.0	2.58E+00	5.69E-03	1.32E-01	7.07E-02	2.08E-01	3.30E+00
3.5	2.31E+00	5.09E-03	1.07E-01	5.77E-02	1.70E-01	2.69E+00
4.0	2.08E+00	4.60E-03	9.02E-02	4.85E-02	1.43E-01	2.26E+00

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8.1 **CHOOSE** the appropriate monitor:

8.1.1 For **PWRs**

1. Aux Bldg. Vent – **SELECT** this option for releases from the Auxiliary Building Vent.

2. Main Steam Line – **SELECT** this option for releases from the Main Steam Line.

   - The Main Steam Line radiation monitors read out in mr/hr, therefore a flow has to be determined to calculate the uCi/sec release rate.

   - This is accomplished by entering the steam pressure and the number of relief valves that are open in the section labeled *Relief Inputs*.

8.1.2 For **BWRs**, normally there will be no choice available or needed here. Dresden Station DAPAR allows choice for Unit 1 release point.

NOTE: User must choose the appropriate Channel, "Lo", "Mid", or "Hi". The same µCi/sec reading for the different channels can make a significant difference in projected doses offsite. Channel selection information is obtained from the Process Book Plant Data Displays or the Control Room.

8.2 Input **Reading** Information – User enters the appropriate monitor reading in uCi/sec or mr/hr. **IF** two (2) units or vent entries are shown a value must be entered in each before program calculates offsite dose (enter 0 if no release).

8.3 After User enters data the program calculates offsite doses. The user can now perform one of the following items:

8.3.1 **SELECT** the *Print Receptor Report* button to print a report of projected dose rates, TEDE and CED Doses and Ground Deposition at pre-designated receptor points.

8.3.2 **SELECT** the *Print* button to print a report of offsite dose projections based on the monitored release.

8.3.3 **SELECT** the *Back* button to change input data on the Full Assessment Form.

8.3.4 **SELECT** the *PARs* button to view PAR form – **GO TO** section 3.6

8.3.5 **SELECT** a different monitor and/or change readings to recalculate doses and update PAR.

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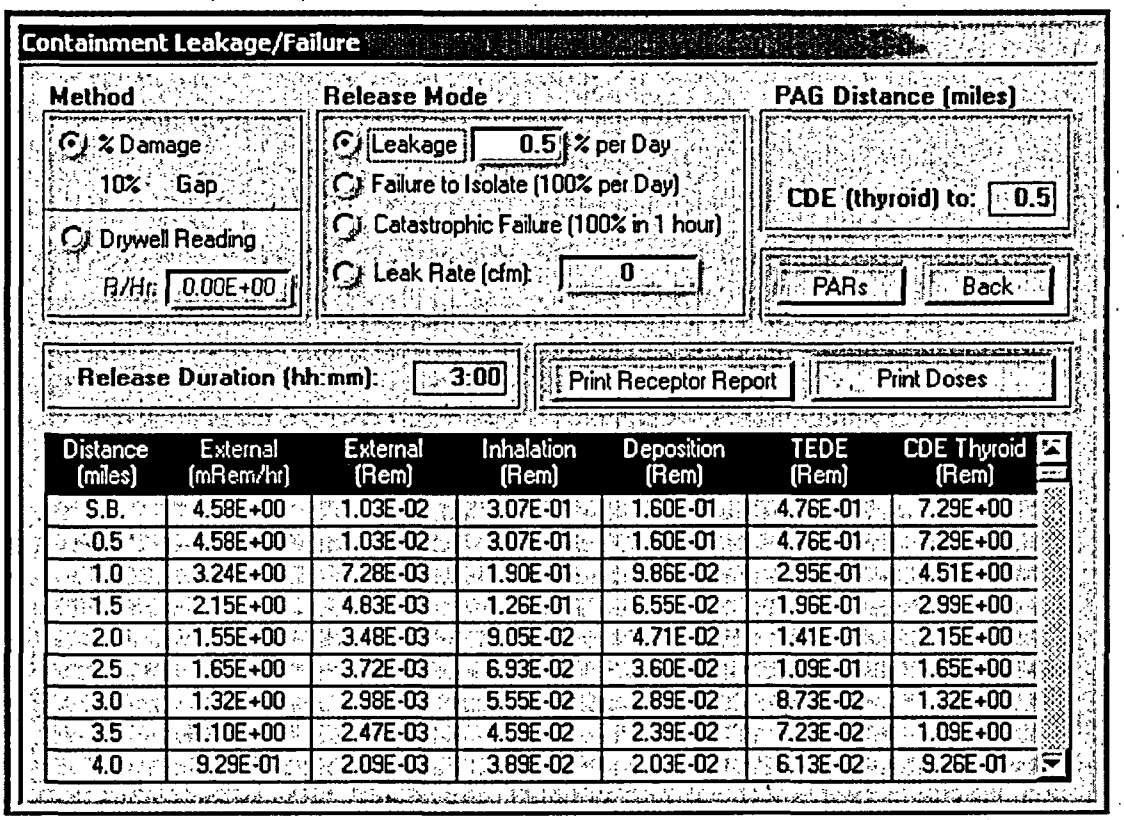
9. CONTAINMENT LEAKAGE/FAILURE



Distance (miles)	External (mRem/hr)	External (Rem)	Inhalation (Rem)	Deposition (Rem)	TEDE (Rem)	CDE Thyroid (Rem)
S.B.	4.58E+00	1.03E-02	3.07E-01	1.60E-01	4.76E-01	7.29E+00
0.5	4.58E+00	1.03E-02	3.07E-01	1.60E-01	4.76E-01	7.29E+00
1.0	3.24E+00	7.28E-03	1.90E-01	9.86E-02	2.95E-01	4.51E+00
1.5	2.15E+00	4.83E-03	1.26E-01	6.55E-02	1.96E-01	2.99E+00
2.0	1.55E+00	3.48E-03	9.05E-02	4.71E-02	1.41E-01	2.15E+00
2.5	1.65E+00	3.72E-03	6.93E-02	3.60E-02	1.09E-01	1.65E+00
3.0	1.32E+00	2.98E-03	5.55E-02	2.89E-02	8.73E-02	1.32E+00
3.5	1.10E+00	2.47E-03	4.59E-02	2.39E-02	7.23E-02	1.09E+00
4.0	9.29E-01	2.09E-03	3.89E-02	2.03E-02	6.13E-02	9.26E-01

9.1 **SELECT** the appropriate containment methods:

9.1.1 *% Damage* calculates doses based on release of total available fission products based on type and amount of core damage.

9.1.2 *Drywell Reading* calculates doses based on the drywell radiation monitor readings.

9.2 **SELECT** the appropriate containment release mode:

9.2.1 *Leakage* – Program defaults to 0.1% per day which is the Design Leakage rate per the UFSAR. If a different percentage of leak rate has been calculated by TSC engineers enter that value in the % per day text box.

9.2.2 *Failure to Isolate* – Assumes 100% of the isotopes available for release are released in a 24 hour time period.

9.2.3 *Catastrophic Failure* -- Assumes 100% of the isotopes available for release are released in a 1 hour time period.

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9.2.4 *Leak Rate (cfm)* – calculates release rate base on estimated leak rate from the containment.

9.3 After User enters data the program calculates offsite doses. The user can now have the program perform the following items:

9.3.1 **SELECT** the *Print Receptor Report* button to print a report of projected dose rates, TEDE and CED Doses and Ground Deposition at pre-designated receptor points.

9.3.2 **SELECT** the *Print* button to print offsite dose projections based on containment failure.

9.3.3 **SELECT** the *Back* button to change input data on the Full Assessment Form.

9.3.4 **SELECT** the *PARs* button to view PAR form – **GO TO** section 6.

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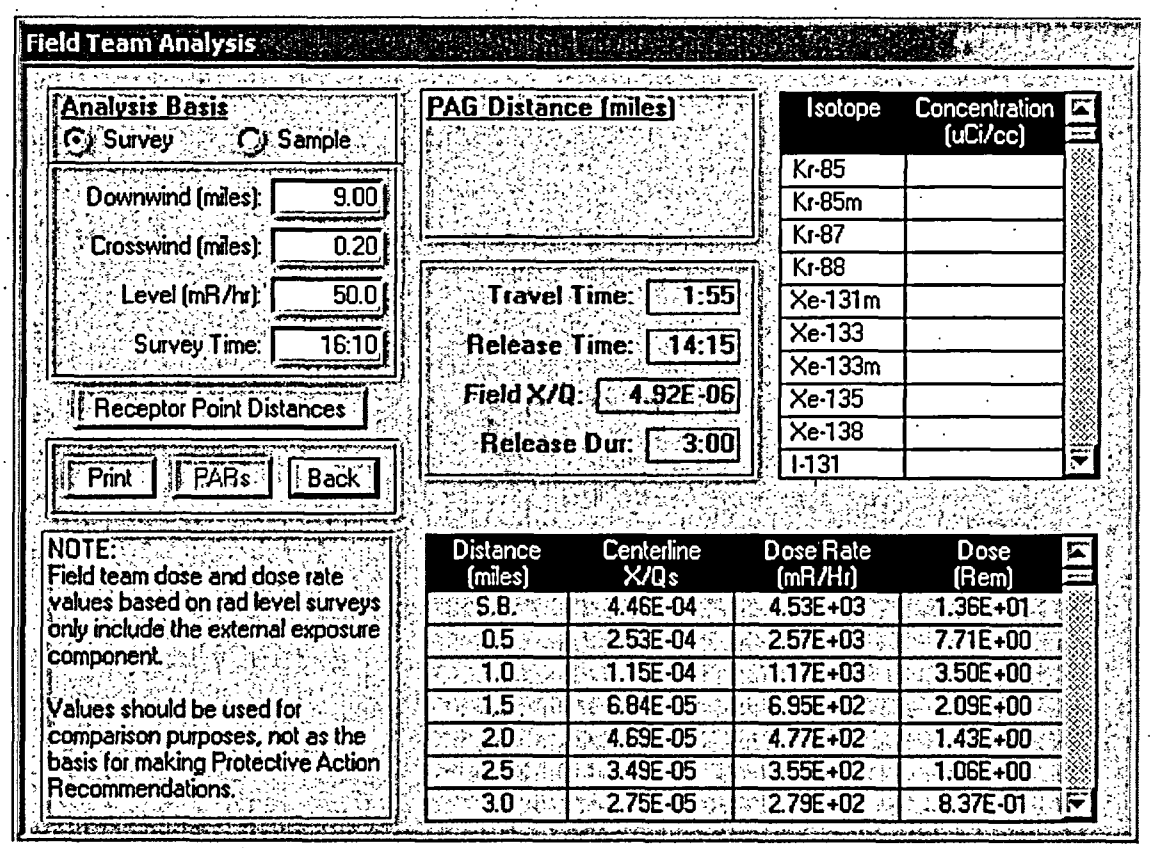
10. **FIELD TEAM ANALYSIS**

NOTE: The program calculates the plume Travel Time and Release Time to allow Dose Assessment personnel to compare previous dose assessment reports with data measured in the field.

10.1 *Dose Rate Survey* – **SELECT** this method if Field Team Survey Data is available.



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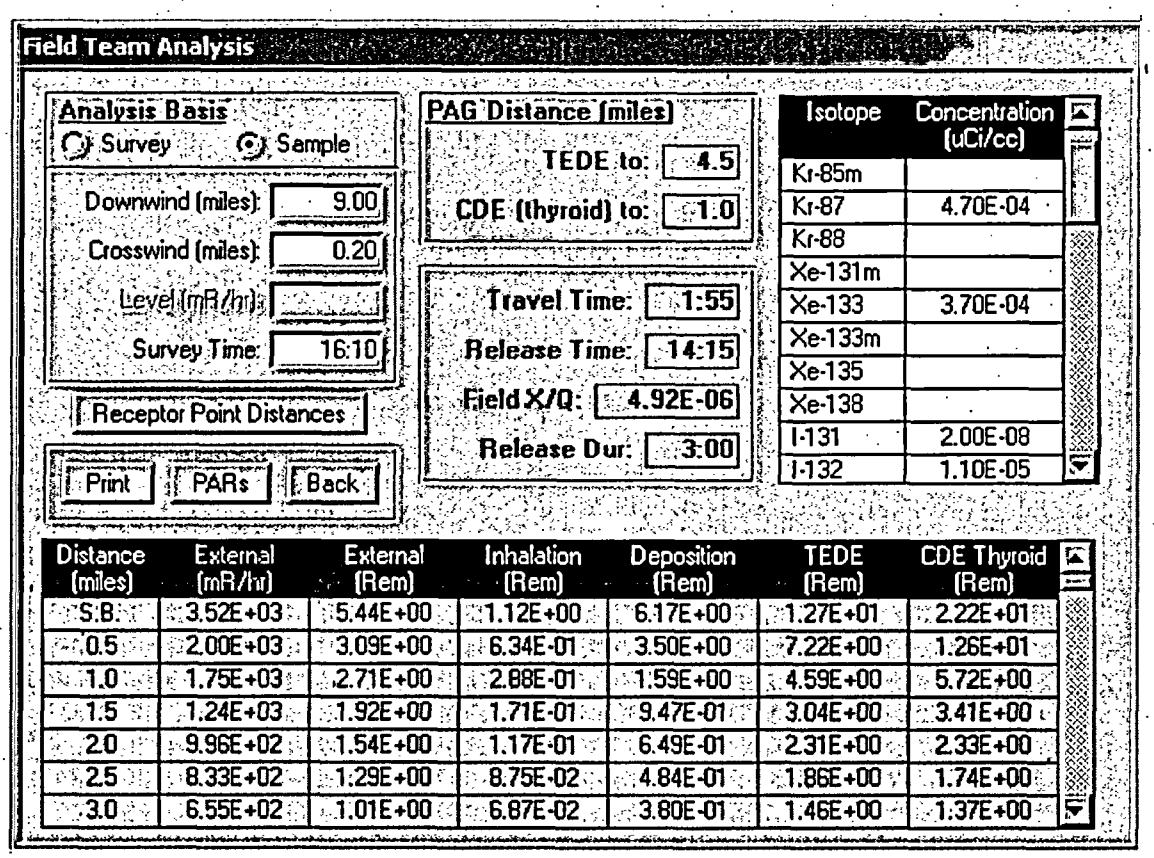
10.2 *Air Sample Results* – **SELECT** this method if Field Team Air Sample Data is available.



10.3 **ENTER** the Field Team information as follows:

NOTE: The program will not allow mr/hr readings for sample data or isotopic results for survey data.

10.3.1 **ENTER** *Downwind (miles)* – straight-line distance from release point to sample location.

10.3.2 **ENTER** *Crosswind (miles)* – the distance the team was away from the centerline when the sample was taken. The program will warn user if reported sample location is wider than expected plume width. The maximum width of any plume for the most unstable stability class is 2.96 miles 10 miles downwind.

10.3.3 **IF** the analysis basis is *Dose Rate Survey*– **ENTER** the Field Team Survey reading in the box labeled *Level.*

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10.3.4 **IF** Analysis basis is *Air Sample Results* – **ENTER** the uCi/cc values for each known isotope in the table at the upper right section of the form.

10.3.5 **ENTER** *Survey Time* – Enter the time the survey or sample was taken.

10.4 After User enters data the program calculates offsite doses. The user can now have the program perform the following items:

10.4.1 **SELECT** the *Print* button to print offsite dose projection reports based on Field Team Analysis.

10.4.2 **SELECT** the *Receptor Point Distance* button to view/print a report on the downwind and crosswind distance to pre-designated receptor points. This report can help in selection of appropriate sampling locations by categorizing each receptor point as being within 1, 2, or 3 Sigma Y of centerline. Sampling should be performed as close to centerline as possible.

10.4.3 **SELECT** the *Back* button to change input data on the Full Assessment Form.

10.4.4 **IF** Field Team samples were the selected basis, **SELECT** the *PARs* button to view PAR form – **GO TO** section 6.

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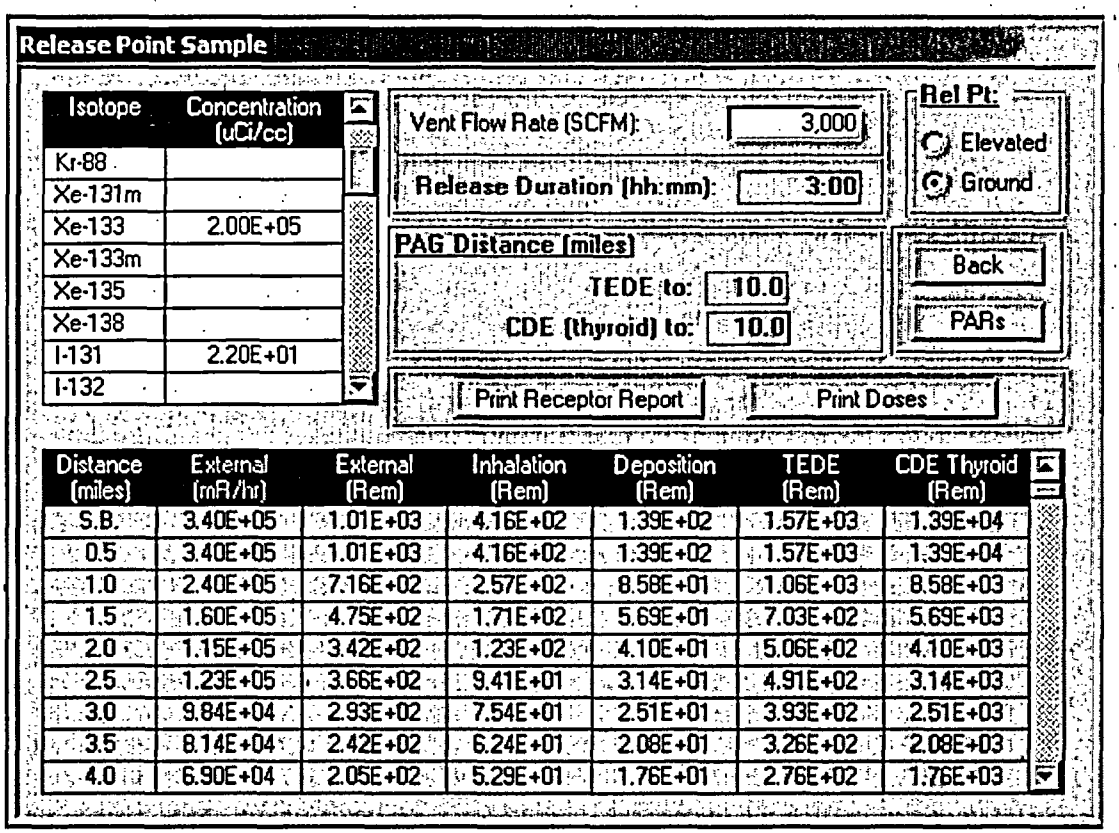
11. RELEASE POINT ANALYSIS



11.1 **ENTER** the known *Isotopic Concentration* for each isotope (if unknown leave blank).

11.2 **ENTER** *Vent Flow Rate* (or estimate flow rate for other releases) in SCFM.

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11.3 **SELECT** the proper release point (*Elevated* or *Ground*) to provide the proper atmospheric dispersion calculations. The table below provides guidance:

STATION	PATH	Rel Pt:
Dresden/Quad	Chimney	Elevated
	Reactor Bldg. Vent.	Ground
	Isolation Condenser (Dresden only)	Ground
	Hole in Wall / Other	Ground
Braidwood/Byron	Ventilation Stack	Ground
	Steam Generator Safety Valves/PORV	Ground
	Hole in Wall / Other	Ground
LaSalle	Station Vent Stack	Elevated
	Standby Gas Treatment Stack	Elevated
	Hole in Wall / Other	Ground
Clinton	HVAC Stack	Ground
	Standby Gas Treatment Stack	Ground
	Hole in Wall / Other	Ground

11.4 After User enters data the program calculates offsite doses. The user can now have the program perform the following items:

11.4.1 **SELECT** the *Print Doses* button to print offsite dose projections based on Release Point Analysis.

11.4.2 **SELECT** the *Print Receptor Report* button to print a report of projected dose rates, TEDE and CED Doses and Ground Deposition at pre-designated receptor points.

11.4.3 **SELECT** the *Back* button to change input data on the Full Assessment Form.

11.4.4 **SELECT** the *PARs* button to view PAR form – **GO TO** section 6.

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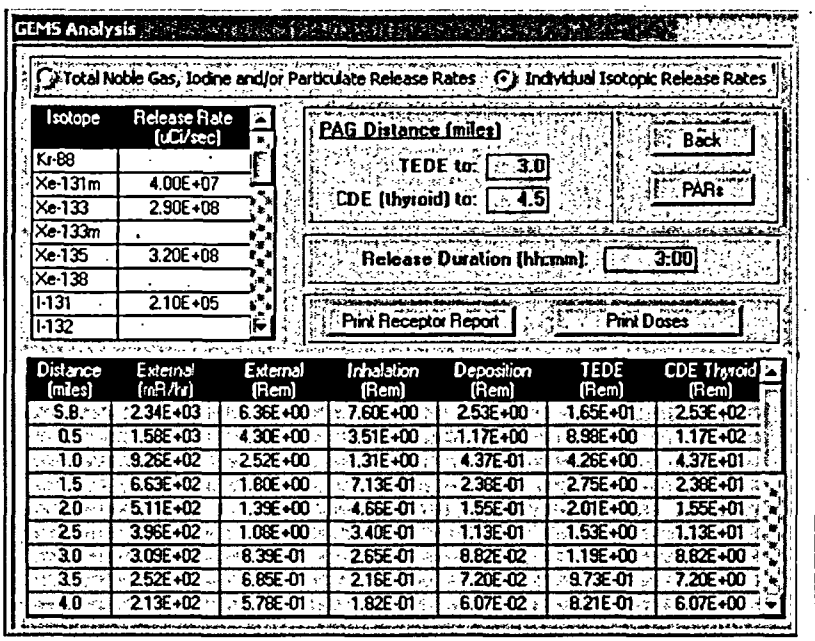
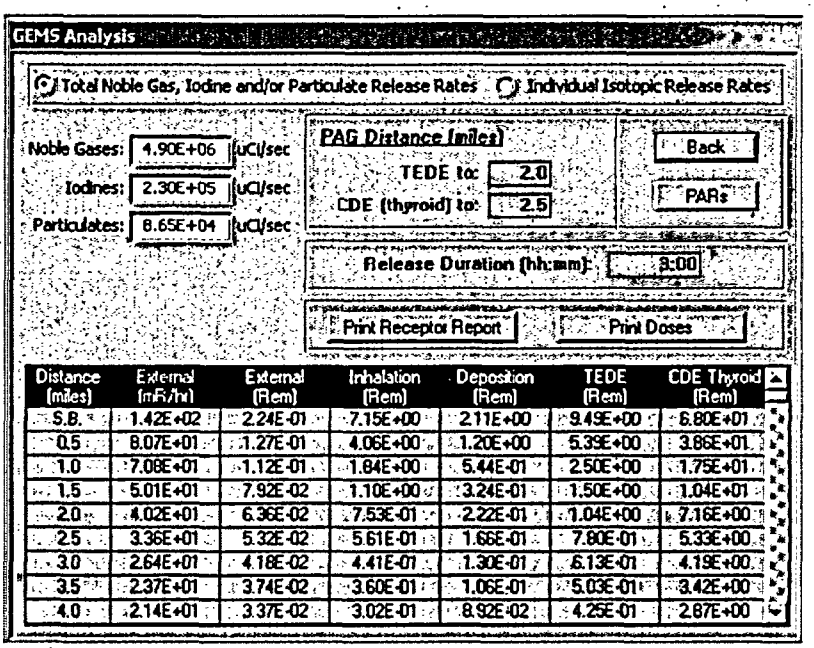
12. **GEMS ANALYSIS**

12.1 **SELECT** *Total, Noble Gas, Iodine and/or Particulate Release Rates* if Total values are available from GEMS,

or.



12.2 **SELECT** *Individual Isotopic Release Rates*



12.3 **ENTER** the known *Isotopic Concentration* for each group or each isotope (if unknown leave blank).

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12.4 After User enters data the program calculates offsite doses. The user can now have the program perform the following items:

12.4.1 **SELECT** the *Print Doses* button to print offsite dose projections based on Release Point Analysis.

12.4.2 **SELECT** the *Print Receptor Report* button to print a report of projected dose rates, TEDE and CED Doses and Ground Deposition at pre-designated receptor points.

12.4.3 **SELECT** the *Back* button to change input data on the Full Assessment Form.

12.4.4 **SELECT** the *PARs* button to view PAR form – **GO TO** section 6.

ATTACHMENT 2

OBTAINING IDNS MONITOR DATA FROM THE EOF

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This procedure provides instructions for obtaining Illinois Department of Nuclear Safety (IDNS) effluent radioactivity and offsite radiation data via a dedicated computer in the EOF.

For each Exelon nuclear power plant in Illinois, the following data is available from IDNS:

- GEMS (Gaseous Effluent Monitoring System) data on radioactivity in airborne effluents;
- Reuter-Stokes detector offsite radiation data

1. **PRECAUTIONS**

   None

2. **LIMITATIONS AND ACTIONS**

   None

3. **PROCEDURE**

3.1 **TURN ON** the Reverse State Link computer. This is a non-networked computer located in the Environmental Area of the EOF.

3.2 **CLICK** OK on the messages and dialog boxes that appear (including error messages) until you get to the Windows desktop. No password is required to log onto this computer.

3.3 **DOUBLE-CLICK** the icon labeled "pw" to obtain the connection password. This will be needed for Step 3.7, substep 6.

3.4 **CLOSE** the file which provides the connection password.

3.5 **DOUBLE-CLICK** the icon labeled "IDNS."

3.6 **WAIT** for the modem to establish a connection with the IDNS computer network.

ATTACHMENT 2

OBTAINING IDNS MONITOR DATA FROM THE EOF

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3.7 **INTERACT** with the IDNS network per the following dialog:

	Prompt	User Response
1	Blinking Yellow Rectangle	<enter>
2	#	reac <enter>
3	Enter username>	exelon <enter>
4	Local>	c eagle <enter>
5	Username:	exelon <enter>
6	Password:	[Use password determined in Step 3.2] <enter>
7	Eagle$	onfs <enter>
8	Enter Option (1-24), ...	env <enter> *[displays Environmental Menu]*
9	Enter Option (1-24), ...	*One of the following:* metra <enter> *[for meteorology, Reuter-Stokes data, and GEMS gross noble gas, iodine and particulate release rates]* gems <enter> *[for GEMS isotopic release rates]* ex <enter> *[to begin process of exiting from the IDNS network]* *When using the* metra *or* gems *option, follow the on-screen instructions for display choices or to exit to the Environmental Menu. If you are using the* metra ALL *display option, use* Control-c *to exit to the Environmental Menu.*
After entering ex *from the Environmental Menu, continue as follows:*		
10	Eagle$	lo <enter> *(letters "el" and "oh" for "logout")*
11	Local>	Alt (cd) *(While holding Alt key, type cd)*
12	Blinking Yellow Rectangle	Alt (fx) *(While holding Alt key, type fx) [This returns the computer to the Windows desktop.]*

3.8 When access to the IDNS network is no longer needed, **SHUTDOWN** the computer per standard Windows procedures.

ATTACHMENT C

EP-AA-110-301 CORE DAMAGE ASSESSMENT (BWR)

CORE DAMAGE ASSESSMENT (BWR)

1. PURPOSE

1.1. This procedure provides emergency response personnel with the methodology to estimate the degree of possible core damage at Exelon Nuclear's Boiling Water Reactor (BWR) stations, with the exception of Oyster Creek Generating Station (OCGS). Refer to EP-AA-110-302 for methodology to estimate potential core damage for a Pressurized Water Reactor (PWR).

1.2. This Core Damage Assessment process is designed to assist in estimating core damage after an accident with potential clad or core damage conditions, and is intended to provide an acceptable alternative to existing station core damage assessment models and methods utilized by Reactor Engineering to assist in the following:

- Determining if the fuel barriers are breached to evaluate the appropriate Emergency Action Level (EAL) classification.
- Providing input on core configuration (coolable or uncoolable) for prioritization of mitigating activities.
- Determining the potential quantity and isotopic mix of a radiological release to project offsite doses.
- Predicting the radiation protection actions that should be considered for long term recovery activities.
- Satisfying inquiries from local and federal government agencies and provide evidence that the utility knows the plant conditions.

1.3. Core damage may be assessed by:

- Evaluating the drywell radiation levels (and confirmed by evaluating the extent of time the core was uncovered),
- Concentration of certain isotopes in a reactor coolant analysis, or
- Concentration of hydrogen in the primary containment.
- History of Core Cooling

2. TERMS AND DEFINITIONS

2.1. **BWR** – Boiling Water Reactor

2.2. **Cladding** – The outer coating (usually zirconium alloy), which covers the nuclear fuel elements to prevent corrosion of the fuel and the release of fission products into the coolant.

2.3. **Containment Type** –

- Clinton (Mark III)
- Dresden (Mark I)
- LaSalle (Mark II)
- Limerick (Mark II): 764 assemblies
  Cont. Volume (384,570 ft^3) = Suppression Pool (149,380 ft^3) + Drywell (235, 190 ft^3)
- Peach Bottom (Mark I): 764 assemblies
  Cont. Volume (303,600 ft^3) = Suppression Pool (127,800 ft^3) + Drywell (175, 800 ft^3)
- Quad Cities (Mark I)

2.4. **Core Release Fraction** – The fraction of each isotope in the core inventory that is assumed to be released from the core under given core conditions.

2.5. **Core Uncovery Time** – For BWRs this is the period of time when reactor water level is less than that required for minimum steam cooling, or about ≥ 20% of the core active fuel is uncovered.

2.6. **Cladding Failure**

1. Also referred to as "Cladding Oxidation", "Gap Release" or "Clad Rupture" in other documents.

2. 100% clad failure refers to the rupture of 100% of the fuel rods in the core. This would result in all fission products contained in the gap space being released to the reactor coolant system.

2.7. **Equilibrium** – Conditions associated with evaluation of different volumes of liquid or gas that contain concentrations of radioactive materials or hydrogen, when these concentrations are approximately the same. This is normally an extended period of time following accident initiation.

2.8. **Fission Products** – The nuclei (fission fragments) formed by the fission of heavy elements or by subsequent radioactive decay of the fission fragments.

2.9. **Fuel Melt**

1. Referred to as "Core Melt," "In-Vessel Melt" or "Over-temperature" damage in reference documents.

2. 100% fuel melt refers to high temperatures in the fuel pellets in 100% of the fuel rods in the core. This would result in all the fission products contained in the fuel pellet matrix being released to the reactor coolant system.

2.10. **Gap** – The space inside a reactor fuel rod that exists between the fuel pellet and the fuel rod cladding.

2.11. **Gap Release** – The release into containment of fission products in the fuel pin gap.

2.12. **In-Vessel Core Melt** – A condition during a reactor accident in which some of the cladding or reactor fuel melts as a result of overheating the fuel and remains inside the reactor vessel.

2.13. **In-Vessel Core Melt Release** – A release into containment from the reactor vessel, which assumes the entire core has melted, releasing a representative mixture of radioisotopes.

2.14. **Minimum Steam Cooling RPV Water Level (MSCRWL)** – The lowest RP water level at which the covered portion of the reactor core will generate sufficient steam to maintain the hottest clad temperature below 1500oF.

2.15. **Minimum Zero-Injection RPV Water Level (MZIRWL)** – The lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to maintain the hottest clad temperature below 1800oF, assuming no injection into the RPV.

2.16. **Shutdown** – As defined by station emergency operating procedures.

2.17. **Slump** – Relocation of molten reactor core during an accident.

2.18. **Source Term** – The amount and isotopic composition of material released or the release rate, used in modeling releases of material to the environment.

2.19. **Spiked Coolant** – Reactor coolant containing increased concentrations of non-noble isotopes, sometimes seen with rapid shutdown or depressurization of primary system.

2.20. **Spiked Coolant Release** – The release into containment of 100 times the non-noble gas fission products found in the coolant.

2.21. **Subcritical** – The reactor condition when the number of neutrons released by the fission is not sufficient to achieve a self-sustaining nuclear chain reaction. Defined under station emergency operating procedures.

2.22. **Suppression Chamber** – May also be referred to as Wetwell or Torus. The Large steel pressure vessel containing a large volume of water that acts as a heat sink for the Drywell.

2.23. **TID** – Total Isotopic Distribution

2.24. **Vessel Melt-Through**

1. Referred to as "Ex-Vessel Melt" or "Melt Release" in reference documents.

2. Core debris is relocated to the primary containment building after the reactor pressure vessel has failed.

3. RESPONSIBILITIES

3.1. The TSC Core/Thermal Hydraulic Engineer shall serve as the Core Damage Assessment Methodology (CDAM) Evaluator.

3.2. The TSC Radiation Controls Engineer shall coordinate radiological and chemistry information with the Core/Thermal Hydraulic Engineer in support of core damage assessment.

3.3. The TSC Technical Manager shall coordinate core damage assessment activities.

4. MAIN BODY

4.1. **REFER** to Attachment 1, BWR CDAM User Guide for instructions on use of the Core Damage Assessment Methodology (CDAM) Software Program.

5. DOCUMENTATION

5.1. A Summary Form and method specific reports are generated by the BWR CDAM Software for use in documenting the results of the assessment.

6. REFERENCES

6.1. NEDO-22214, Procedures for the Determination of the Extent of Core Damage Under Accident Conditions

6.2. NEDC-33045P, Rev 0 (July 2001), Methods of Estimating Core Damage in BWRs

6.3. WCAP-14696 (July 1996) Westinghouse Owners Group Core Damage Assessment Guidance.

6.4. WCAP-14696-A (November 1999), Westinghouse Owners Group Core Damage Assessment Guidance.

6.5. NUREG-1228, "Source Term Estimation During Incident Response to Severe Nuclear Power Accidents"

6.6. Station Commitments

6.6.1. Peach Bottom

CM-1 T04511 (Attachment 1, 5.6)

6.6.2. Limerick Bottom

CM-2 T04512 (Attachment 1, 5.6)

7. **ATTACHMENTS**

7.1. Attachment 1, BWR CDAM User Guide

Attachment 1
BWR CDAM User Guide
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1. OVERVIEW

1.1. As a windows based application designed in Microsoft Access, BWR CDAM, uses many standard user interfaces. Instructions are not provided in basic computer operations in the windows® environment. The user must be familiar with these to efficiently operate the program.

1.2. It is also assumed user is familiar with basic reactor physics and core damage fundamentals. Emergency Response Organization training will provide an overview of core damage assessment methodologies.

1.3. The program should be used by qualified personnel as a tool to estimate the type and amount of core damage.

2. DETERMINE APPROPRIATE AND AVAILABLE ASSESSMENT METHODS

Mid-West Region Stations

REFER to EP-MW-110-1001 for a listing of appropriate plant parameter points to be used following a LOCA.

2.1. The magnitude and type of event, transport mechanism and time after shutdown will be influencing factors on the method(s) utilized to determine the extent of core damage. Damage estimates can be developed using one or more methods as they become available or applicable.

2.1.1. Indications Of Core Damage

1. The primary indicators of core damage that are available during the early phases of an event:
   - Drywell/Containment Radiation Monitor Readings
   - Drywell/Containment Hydrogen Readings

2. Auxiliary indicators that are used to confirm and better define the possible type of damage are:
   - Reactor Pressure Vessel Level Indication System readings
   - Estimation of maximum temperature reached within the core
   - Estimated core uncovery time
   - Abnormal Source Range Monitor readings

3. Long Term Indicators (once liquid or gaseous samples can be safely obtained) are:
   - Isotopic Ratios
   - Presence of high levels of rare isotopes
   - Quantity of isotopes present in samples

2.1.2. **SELECT** the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

Method	Use	Comment
Containment Radiation Monitor	Early Indication of Core Damage	Uncertainties due to variables in release of fission products from RCS and effects of containment sprays.
Core Conditions	Indication of onset of Core Damage	May not be reliable during later phases of core overheating due to changes in core geometry.
RPV Level	Indication of Core Uncovery	Indicates possible damage not useful in estimating the quantity of damage.
Source Range Monitor	Indication of Core Uncovery	Loss of water level leads to increase in gamma detection.
Containment Hydrogen Monitor	Early Indication of Core Damage	Significant uncertainties due to variable Hydrogen generation in core and in release of Hydrogen from RCS and effects of containment sprays.
RCS Samples and Containment Sump and Atmosphere Samples	Late Indication of Core Damage —Suppression Pool Samples provide indication of Rx Vessel Failure	Very large uncertainties until all systems have reached equilibrium. Useful in planning long term recovery.

3. START UP THE CDAM PROGRAM

3.1. **ACCESS** the application by one of the following:

3.1.1. **OPEN** the BWR CDAM desktop icon on applicable computers.

1. **START** the BWR CDAM program for the plant that has declared an emergency. Programs are labeled BWR CDAM.

2. **SELECT** the appropriate icon or run from the 'start bar' and type in the file path and name as follows **C:\CDAM\BWR CDAM.MDB**

3.1.2. **If** the assigned Core Damage Assessment Computer cannot access the application or the CDAM program will not run, **then** install BWR CDAM on any computer from CDs or Disks located in the TSC or the EOF Library.

1. **INSTALL** CDAM by copying appropriate file to computer's hard drive.

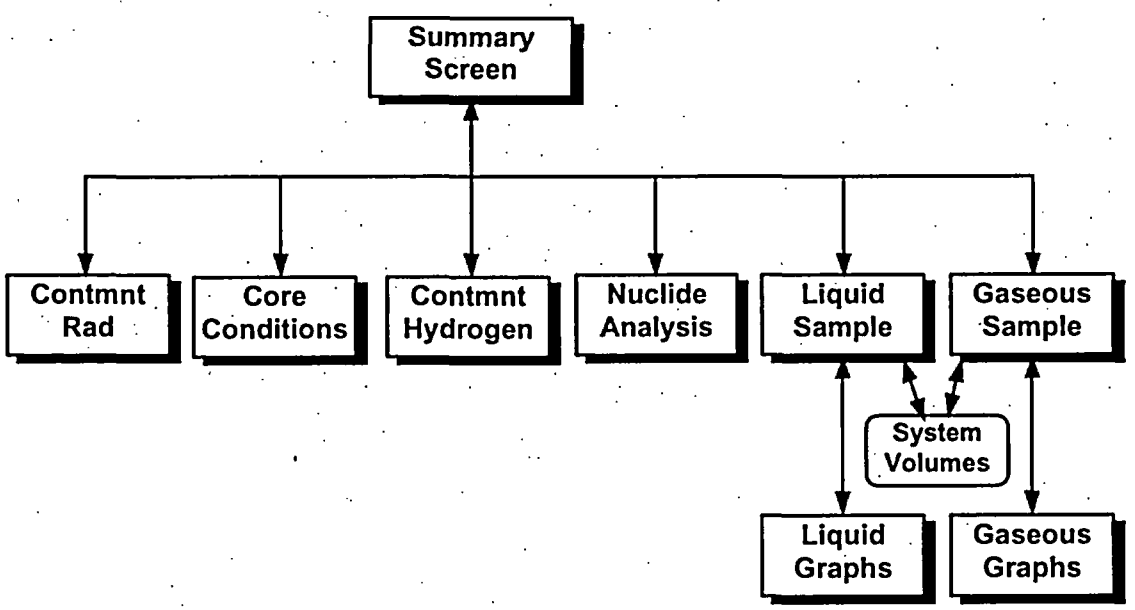
2. **UPDATE** the "properties" of the file by deselecting write protection.

4. SELECTION AND PERFORMANCE OF ASSESSMENT

4.1. **SELECT** the assessment method(s) most appropriate for the existing conditions. Methods available for assisting in the determination of the extent of core damage include the following:

- Containment Radiation Analysis - (Section 5.2)
- Core Conditions Analysis (Cooling History) - (Section 5.3)
- Containment Hydrogen Analysis - (Section 5.4)
- Nuclide Analyses (Ratios and Abnormal Isotopes) - (Section 5.5)
- Liquid Samples Analysis - (Section 5.6)
- Gaseous Samples Analysis - (Section 5.7)

Basic Program Flow Diagram



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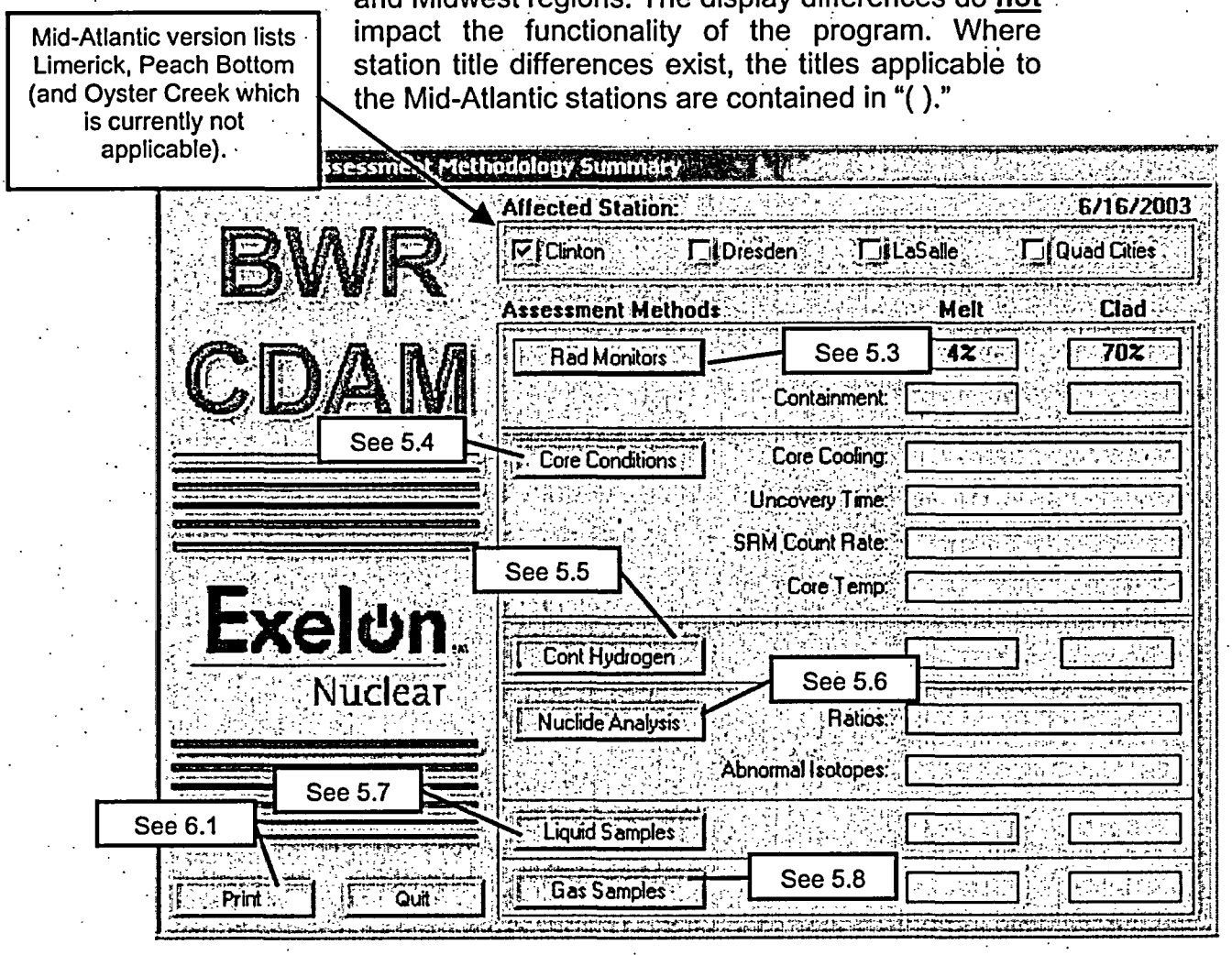
5. PROGRAM SCREENS AND INPUTS

5.1. When the program is started the following screen appears:

NOTE: The value boxes are empty when the program is originally launched. The examples below may deviate from the CDAM displays during use due to different software versions in use in the Mid-Atlantic and Midwest regions. The display differences do **not** impact the functionality of the program. Where station title differences exist, the titles applicable to the Mid-Atlantic stations are contained in "( )."



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CAUTION

Selecting an "Affected Station" resets all inputs to default values.

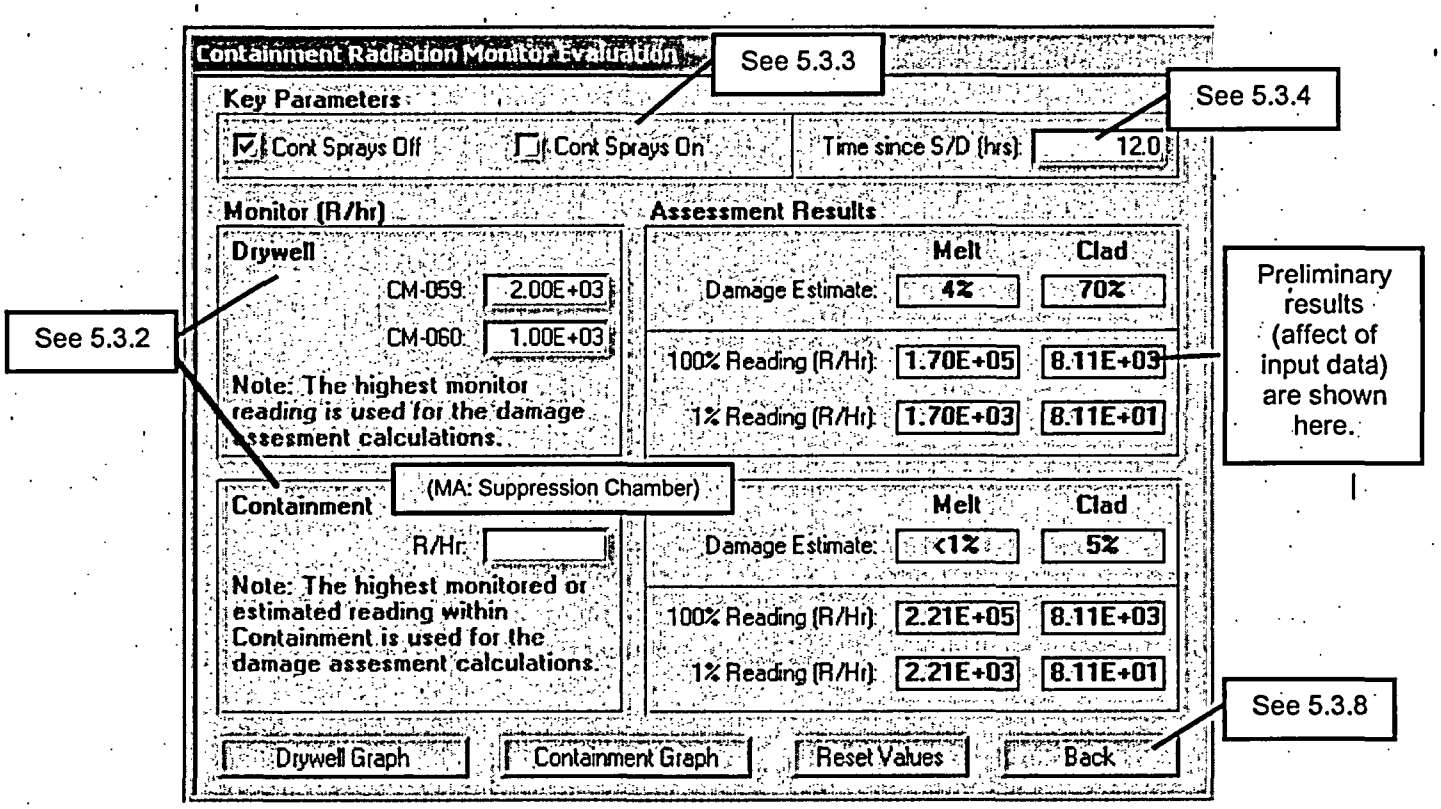
5.2. **SELECT** the Affected Station before other "Assessment Methods."

CAUTION

Pressing the **"Quit"** button exits the program. When the program is closed all data is reset. Program saves no information to disk; printed reports serve as record of core damage assessments.

5.3. Drywell/Containment Radiation Monitor Method

5.3.1. **PRESS** the "Cont Rad Monitors" button on the Summary Screen to open the following form:



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NOTE: Program allows entry from 2 high range monitors for Drywell location and 1 for Torus or Containment / Suppression Chamber, however a reading may be entered from any monitor or measurement taken external to suppression chamber, which accurately indicated containment radiation levels. If two entries are made only the highest is used.

5.3.2. **ENTER** the highest Drywell radiation monitor reading that occurred in these boxes

1. **If** Drywell radiation monitor readings are not available, **then** enter the containment / Suppression Chamber radiation monitor reading.

5.3.3. **SELECT** Drywell/Containment Spray status:

1. **If** the Drywell/Containment Spray system was operated for the majority of the time since the estimated time of the onset of core damage **then** choose "Drywell Spray On."

2. **If** the Drywell/Containment Spray system was **not** operated or only operated briefly (e.g., <10% of time since the estimated time of the onset of core damage) **then** choose "Drywell Spray Off."

5.3.4. **ENTER** the time after reactor shutdown, which corresponds the time the containment radiation reading was taken. Value must be between 1 hour and 24 hours after shutdown, which corresponds to the time period in which this method is considered effective.

NOTE: Pressing "Reset" button resets values on this form only.

5.3.5. **PRESS** "Containment Graph" or "Supp Chamber Graph" button to display a screen similar to the following:

(See example display on next page.)

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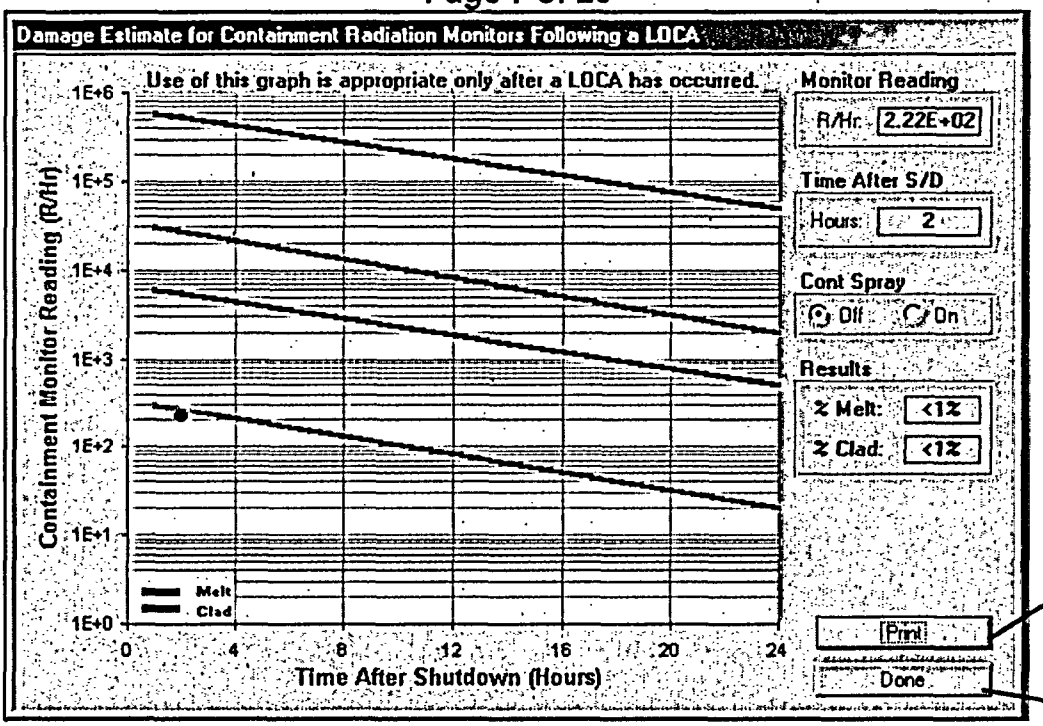

NOTE: Graph shows high and low containment radiation levels which correspond to 100% Melt or Clad or 1% Melt or Clad damage. A dot shows the last containment radiation level entered into the program for assessment.

5.3.6. **PRESS** the "Print" button to print a report of containment radiation method inputs and best estimate of damage.

5.3.7. **PRESS** the "Done" button to return to the Containment Radiation Monitor Evaluation Screen.

5.3.8. **PRESS** the "BACK" button to return to the Summary Screen.

5.4. Core Conditions Methods

NOTE: Each of these four methods is an independent assessment method.

5.4.1. **PRESS** the "Core Conditions" button on the Summary Screen to open the following form:

(See example form on next page.)

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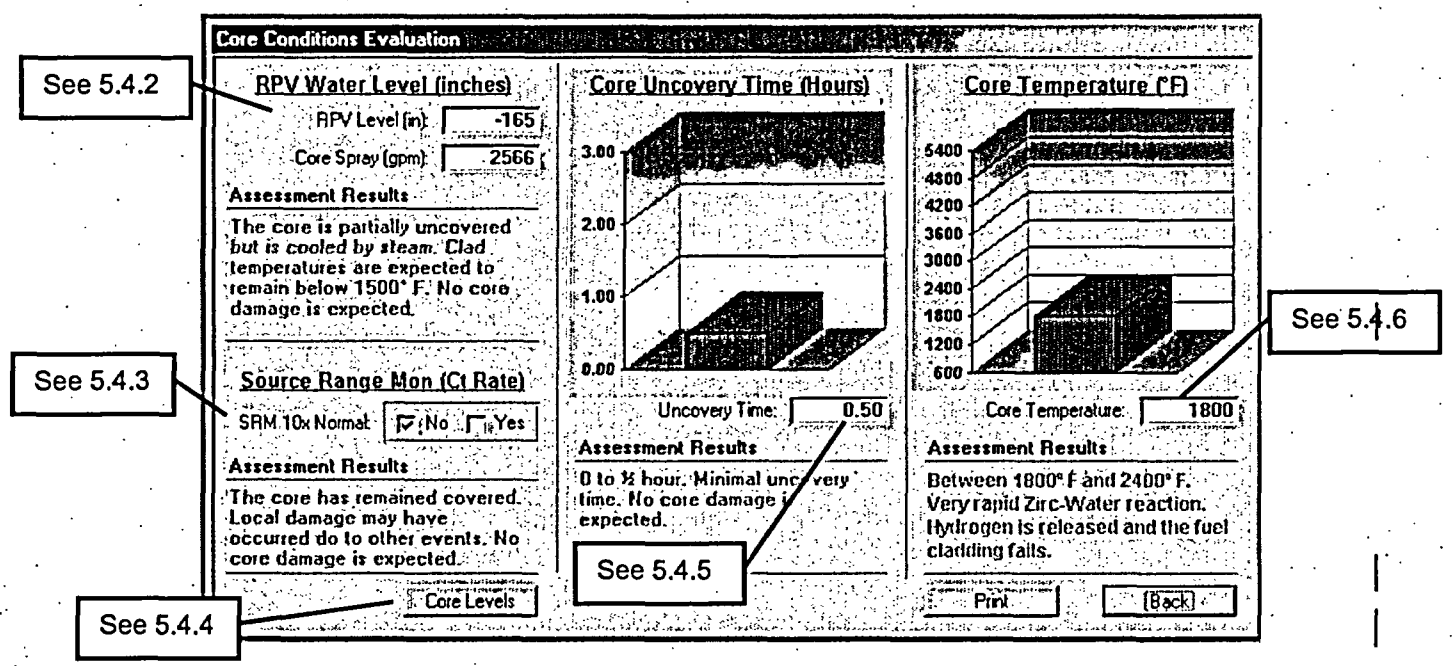

5.4.2. Under Reactor Pressure Vessel (RPV) Water Level **ENTER** the lowest recorded (or estimated) RPV level (range 0 to –350 inches) and core spray flow at time of lowest reading

NOTE: Steps 5.4.3 through 5.4.6 are based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as General Electric personnel).

5.4.3. Under Source Range Monitor **REVIEW** plant parameter history and if the SRM (Wide Range Monitor at Peach Bottom) had indications of a reading 10 times those expected check "Yes."

5.4.4. **PRESS** the "Core Levels" button to view information regarding water levels associated with the Station reactor and vessel level indications.

(See example form on next page.)

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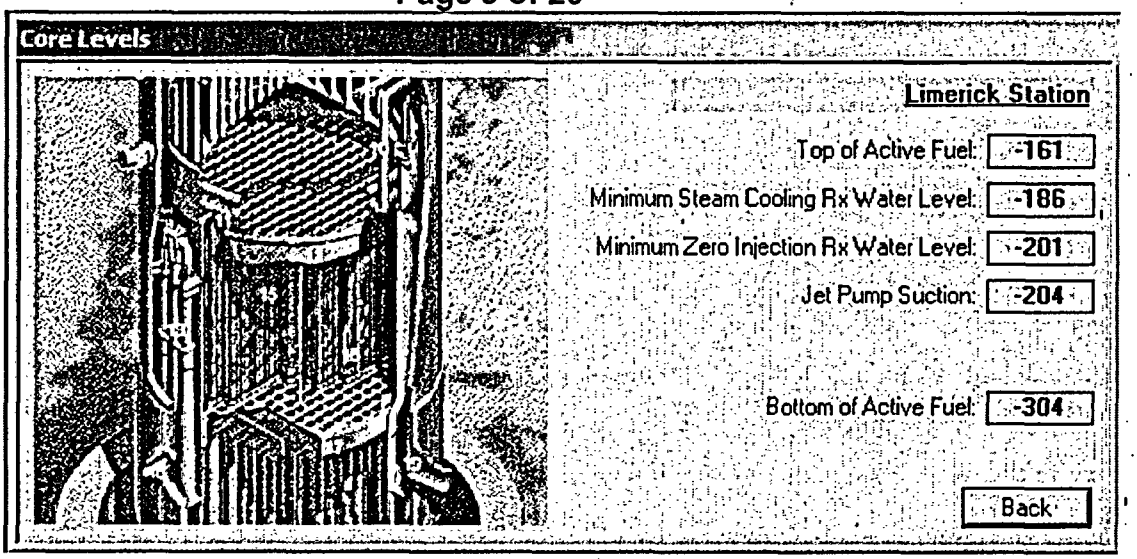

5.4.5. **ENTER** the estimated time the reactor core (20% of top of active core) was uncovered without steam (level below the Minimum Steam Cooling Rx Water Level) or spray cooling reactor core.

5.4.6. **ENTER** the estimated highest temperature reached in the reactor core.

5.4.7. **PRESS** the "Print" button to print a report of inputs and results of core temperature methods of core damage assessment.

5.4.8. **PRESS** the "Back" button to return to the Summary Screen.

5.5. Containment Hydrogen Evaluations

CAUTION

This CDAM assumes no ignitor operation. Ignitor use limits containment hydrogen concentration affecting the reliability of this method.

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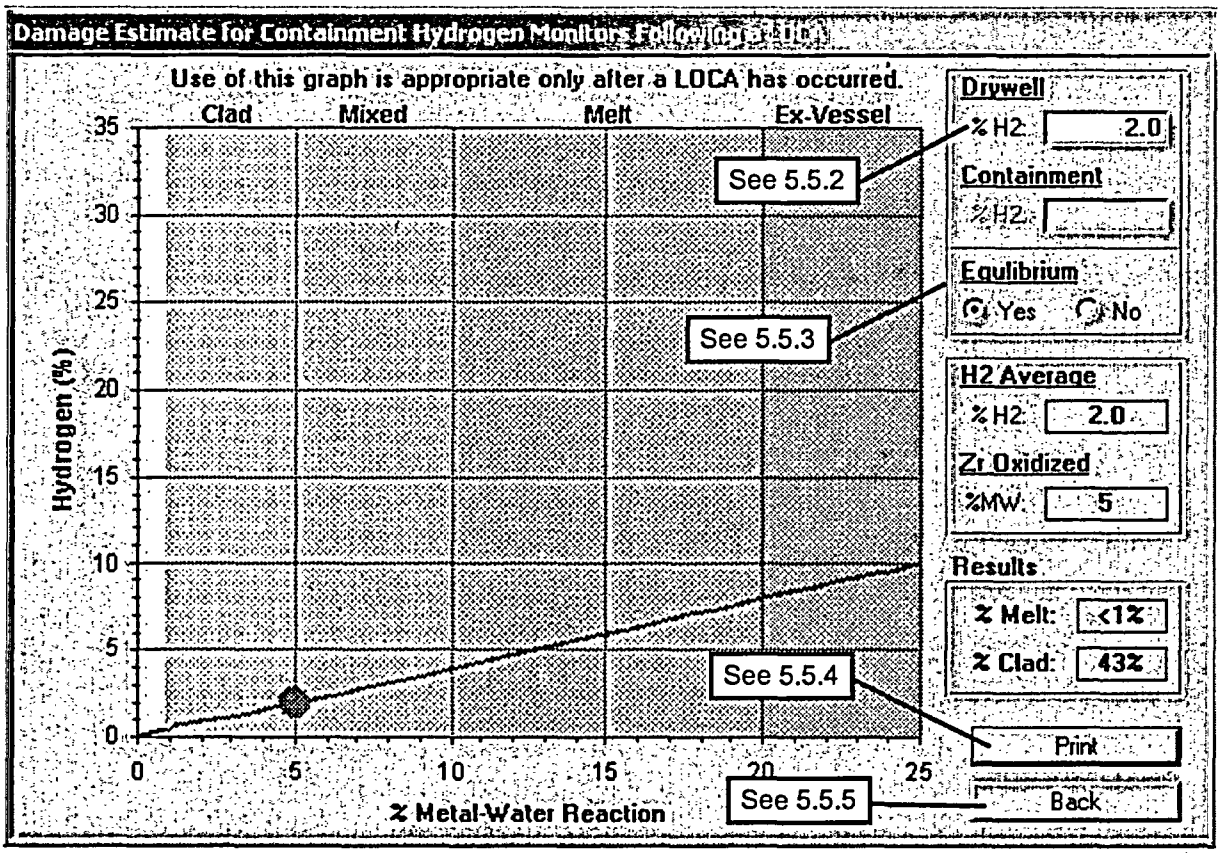
5.5.1. **PRESS** the "Cont Hydrogen" button on the Summary Screen to open the following form:



5.5.2. **ENTER** highest Drywell and/or Suppression Chamber hydrogen level measured.

NOTE: Suppression Chamber reading can only be entered if user selects "no" under Equilibrium in step 5.5.3 below.

5.5.3. **SELECT** the applicable System Equilibrium status based on the following:

1. **If** Containment and Suppression Chamber monitors read the same or only atmospheres are assumed equalized, **then SELECT** "Yes" for equilibrium.

2. **If** containment and suppression chamber atmospheres are not in equilibrium or only containment H2 reading is available, **then SELECT** "No" for equilibrium.

5.5.4. **PRESS** the "Print" button to print a report of inputs and results of core level methods of core damage assessment.

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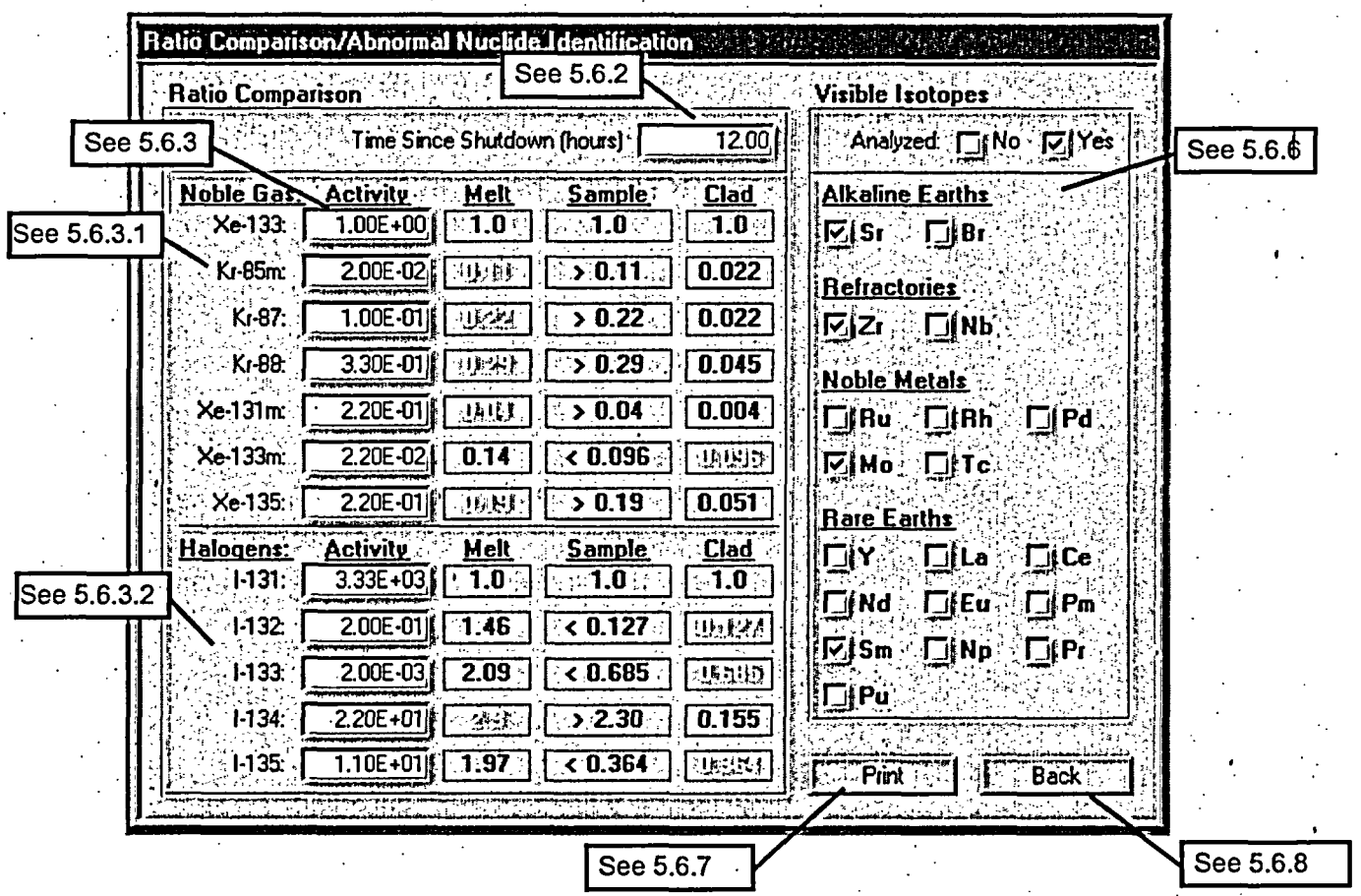
5.5.5. **PRESS** the "Back" button to return to the Summary Screen.

5.6. Nuclide Analysis **(CM-1, CM-2)**

5.6.1. **PRESS** the "Nuclide Analysis" button on the Summary Screen to open the following form:



5.6.2. **ENTER** the time since reactor shutdown (time between shutdown and sample being drawn).

5.6.3. **ENTER** isotopic sample results in uCi/cc. Sample results are to be decay corrected back to time after shutdown that the sample was drawn.

1. Noble Gases are ratioed to Xe-133

2. Halogens are ratioed to I-131

5.6.4. **If** the ratios evaluated above are greater than predicted melt ratio, **then** melt damage is predicted

5.6.5. **If** the ratios evaluated above are less than clad ratio, **then** clad damage is predicted.

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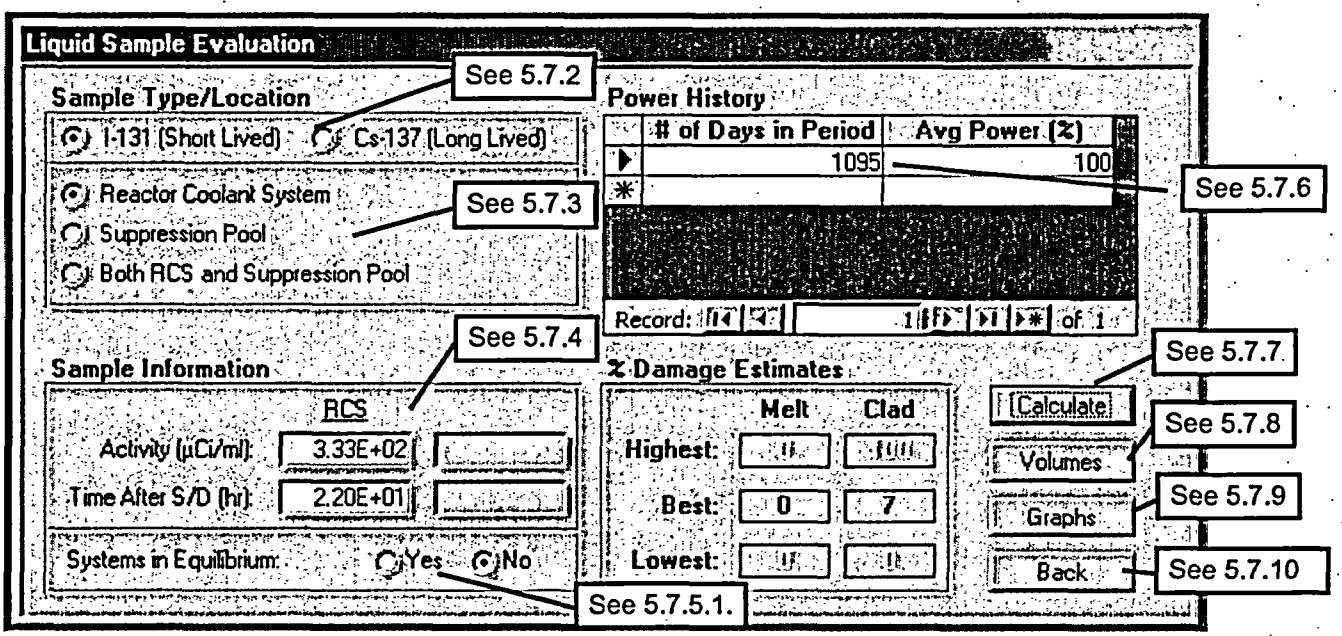
5.6.6. **If** abnormal levels of rare isotopes are present **then** check "Yes" **and** check which isotopes are present.

5.6.7. **PRESS** the "Print" button to print a report of inputs and results of core level methods of core damage assessment.

5.6.8. **PRESS** the "Back" button to return to the Summary Screen.

5.7. Liquid Samples

5.7.1. **PRESS** the "Liquid Samples" button on the Summary Screen to open the following form:



5.7.2. **SELECT** appropriate isotope.

5.7.3. **SELECT** sample location.

1. **If** samples are available from both locations, **then** select both.

5.7.4. **ENTER** Sample Information:

1. Activity is isotopic sample results in uCi/cc (uCi/ml). Sample results are to be decay corrected back to time after shutdown that the sample was drawn

2. Time After S/D (reactor shutdown) is the time between shutdown and sample being drawn.

5.7.5. **SELECT** the appropriate System Equilibrium status:

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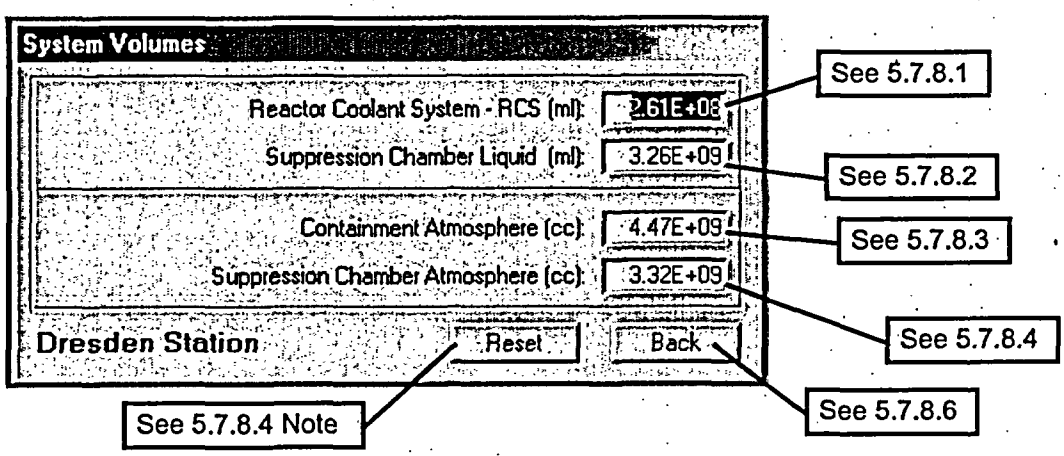
1. **If** sample was taken from only one location and systems are in equilibrium, **then** check "yes" for "Systems in Equilibrium," **otherwise** check "no."

5.7.6. **ENTER** power history (past to present, i.e. oldest steady state history as record number) of core since last refueling. Shutdown times are entered as the number of days with Ave Power (%) set at 0.

1. For short-lived isotopes, **EXTEND** Power History at least 30 days.
2. For long-lived isotopes, **EXTEND** power history at least 100 days, however the power history for the extent of the cycle is preferred.
3. **LIMIT** variations in steady state power to ± 20% within each operational period entered.

5.7.7. Once all data has been entered, **PRESS** the "Calculate" button to display the % Damage Estimates.

5.7.8. **PRESS** the "Volumes" button to display the follow screen:



1. Program enters default RCS volume, which the user may change based on RPV Level Readings at time of sample.
2. Program enters default Suppression Chamber volume, which the user may change based on readings at time of sample.
3. Program enters default Containment free air volume which user may change based on conditions at time of sample. Unless there has been significant flooding of drywell this value will not change.

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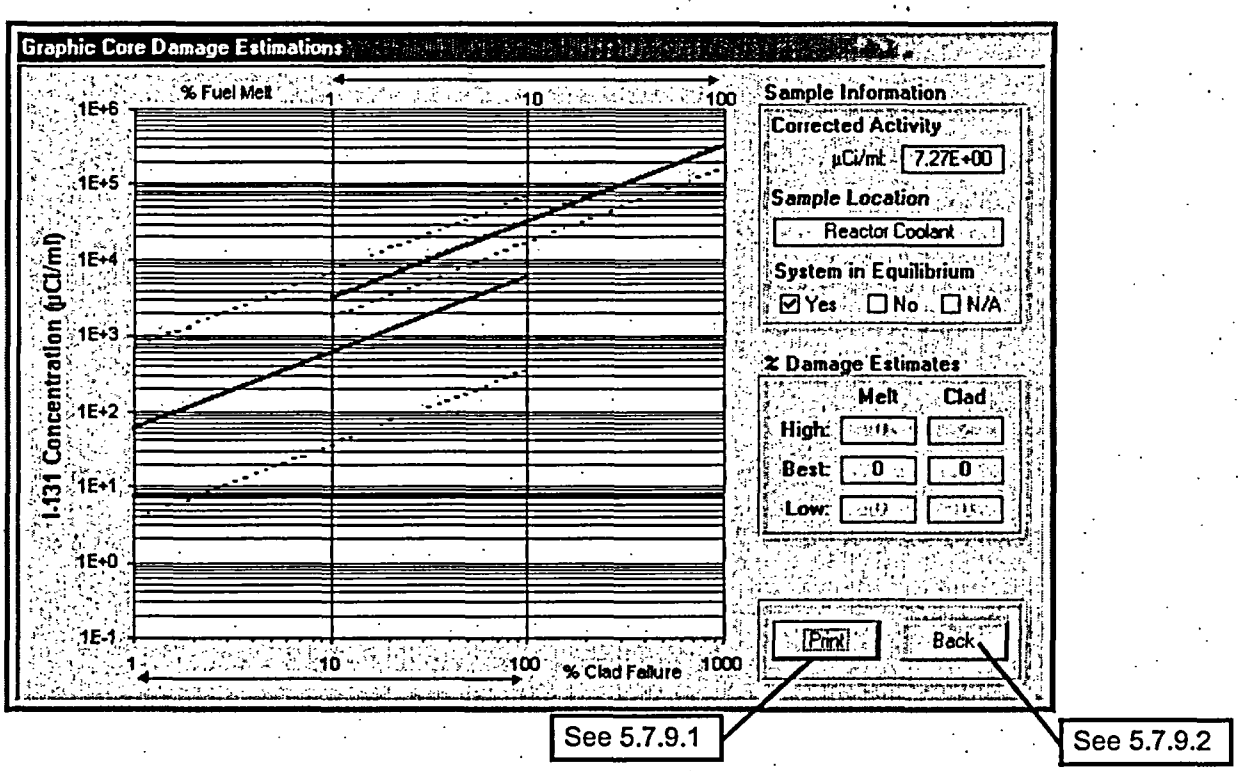
4. Program enters default Suppression Chamber free air volume which user may change based on conditions at time of sample. If there has been a significant increase or decrease in the water level in the Suppression Pool or Torus then the free air volume will change.

NOTE: Pressing the "Reset" button will reset all volumes to default values.

5. **PRESS** the "Back" button to return to the Liquid or Gaseous screen, which user used to call volume form.

5.7.9. **PRESS** the "Graph" button to display the following screen:



(See Note on next page.)

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NOTE: Graph on previous page shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating entered corrected sample activity.

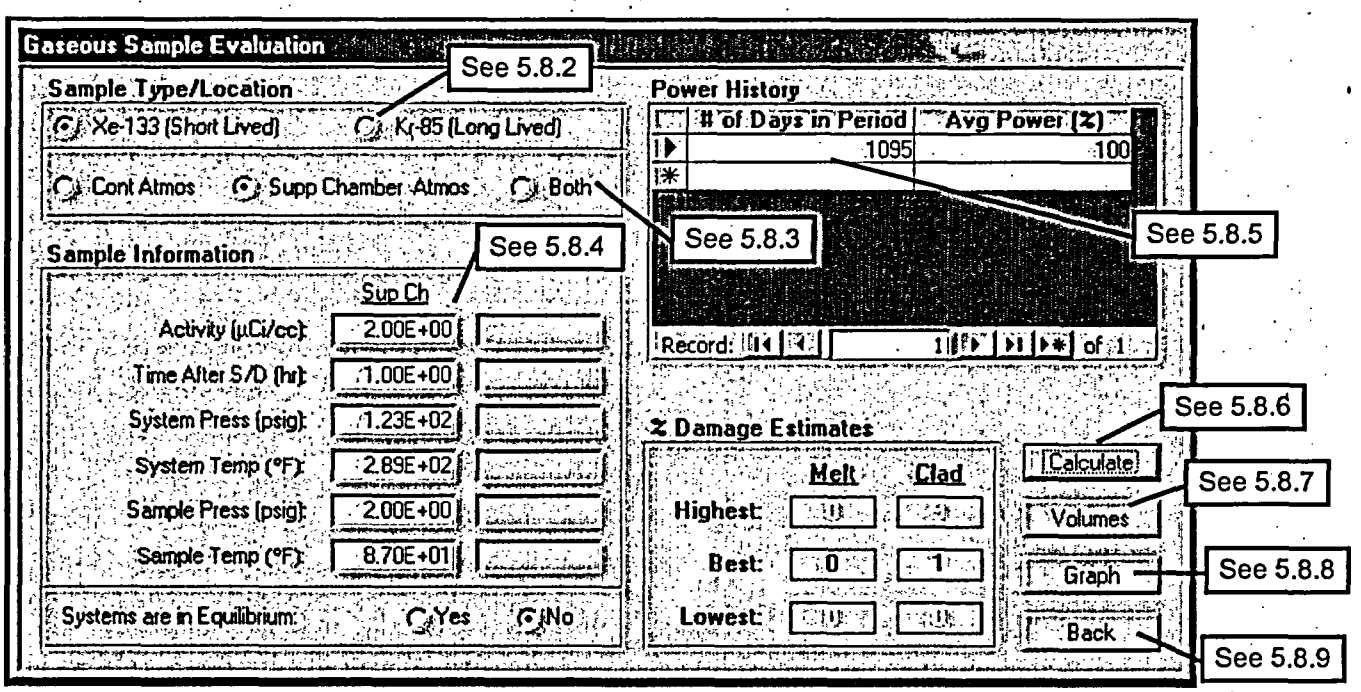
1. **PRESS** the "Print" button to print a graph and summary of inputs.

2. **PRESS** the "Back" button to go back to liquid or gaseous form which called this form.

5.7.10. **PRESS** the "Back" button to return to the Summary Screen.

5.8. Gaseous Samples

5.8.1. **PRESS** the "Gas Samples" button on the Summary Screen to open the following form:



5.8.2. **SELECT** appropriate isotope.

5.8.3. **SELECT** and sample location.

1. **If** samples are available form both locations, **then SELECT** "Both" option.

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5.8.4. **ENTER** Sample Information:

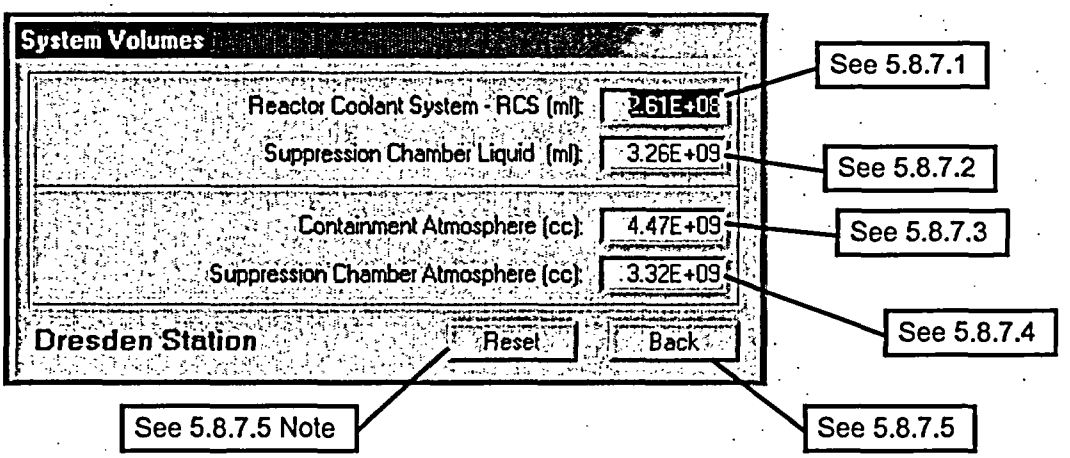
1. **ENTER** sample activity for selected isotope in uCi/cc (uCi/ml). Sample results are to be decay corrected back to time after shutdown that the sample was drawn
2. **ENTER** Time After S/D that sample was taken.
3. **ENTER** the pressure and temperature of the system sampled
4. **ENTER** the end pressure and temperature of sample.

5.8.5. **ENTER** power history (past to present, i.e. oldest steady state history as record number 1) of core since last refueling. Shutdown times are entered as the number of days with Avg Power (%) set at 0.

1. For short-lived isotopes, **EXTEND** Power History at least 30 days.
2. For long-lived isotopes, **EXTEND** power history at least 100 days, however the power history for the extent of the cycle is preferred.
3. **LIMIT** variations in steady state power to ± 20% within each operational period entered.

5.8.6. Once all data has been entered **PRESS** the "Calculate" button to display the % Damage Estimates.

5.8.7. **PRESS** the "Volumes" button to display the following screen (same as 5.7.8):



1. Program enters default RCS volume, which the user may change based on RPV Level Readings at time of sample.

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2. Program enters default Suppression Chamber volume, which the user may change based on readings at time of sample.

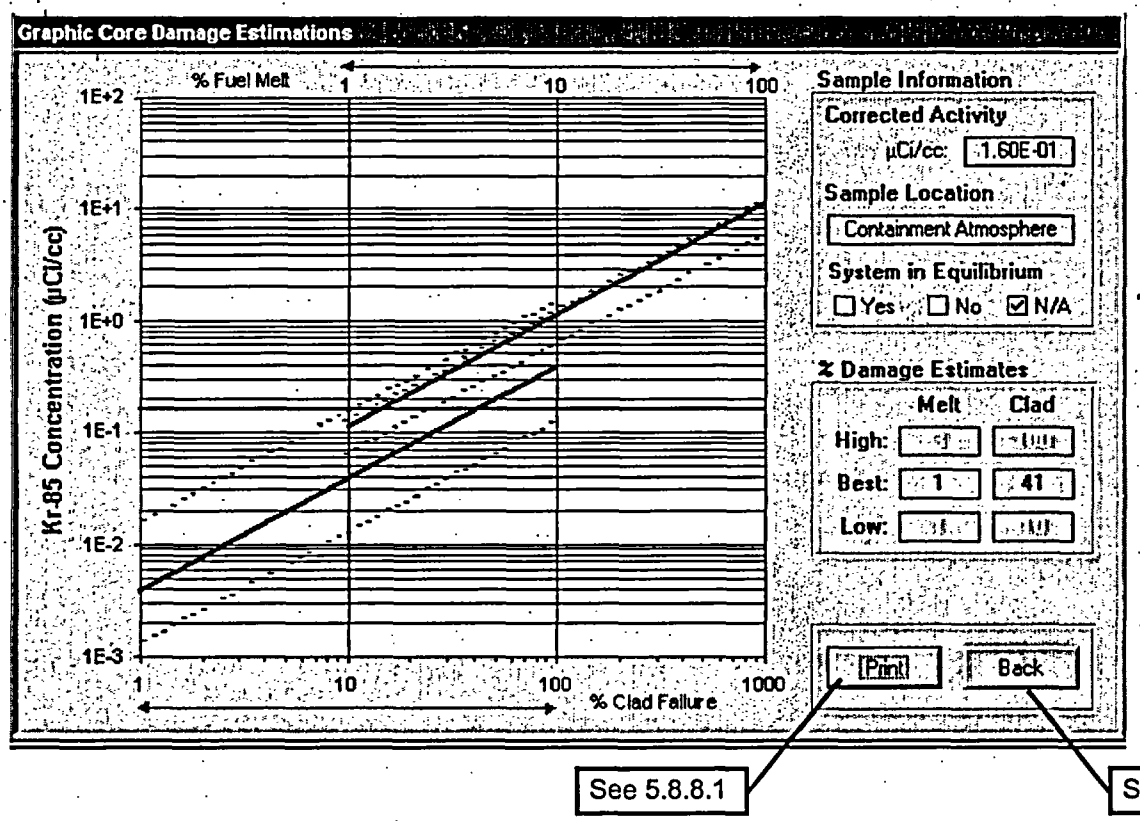
3. Program enters default Containment free air volume which user may change based on conditions at time of sample. Unless there has been significant flooding of drywell this value will not change.

4. Program enters default Suppression Chamber free air volume which user may change based on conditions at time of sample. If there has been a significant increase or decrease in the water level in the Suppression Pool or Torus then the free air volume will change.

NOTE: Pressing the "Reset" button will reset all volumes to default values.

5. **PRESS** the "Back" button to return to the Liquid or Gaseous screen, which user used to call volume form.

5.8.8. **PRESS** the "Graph" button to display the following screen:



NOTE: Graph shows High, Low, and Best melt curves; High, Low, and Best clad damage curves, and a red line across graph indicating entered.

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1. **PRESS** the "Print" button to print a graph and summary of inputs.

2. **PRESS** the "Back" button to go back to liquid or gaseous form which called this form.

5.8.9. **PRESS** the "Back" button to return to the Summary Screen.

6. **CORE DAMAGE SUMMARY REPORT**

6.1. Once the program user enters data for all available assessment methods and the program calculates damage based on inputs, **SELECT** the "Print" button to print a summary of all methods used.

6.2. The values presented in the Assessment Methods section of the summary report show that they are in percent (%). Containment Hydrogen values are also in percent (but do not show the % symbol)..

(Sample report on next page.)

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CDAM Method: **Core Damage Summary**

Station: ☐ Clinton ☐ Dresden ☐ LaSalle ☑ Quad Cities

Assessment Methods:		Melt	Clad
Containment Radiation Monitors*	Containment:	29%	79%
	Supperssion Chamber:	<1%	23%
Core Conditions	Core Cooling:	Clad Damage	
	Core Uncovery Time:	No Core Damage	
	SRM Count Rate:	No Core Damage	
	Core Temp:	Clad Failure	
Containment Hydrogen*		<1	20.8
Sample Analysis	Ratios:	Fuel Melt	
	Abnormal Isotopes:	6 of 19 Present	
	RCS: Liquid Samples:	0%	0%
	Chamber: Gas Samples:	23%	100%

* These methods should NOT be used for qualitative or quantitative assessment except in the case of a LOCA

Analyst's Estimate:

☐ No Core Damage ☐ Cladding Failure ☐ Fuel Melt Amount:

NRC Core Condition Category:

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

Generated By:

Name: _____ Date: 12/05/02 Time: 8:29 AM

Core Damage Summary **Exelon BWR CDAM v1.0**

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6.3. The Individual tasked with assessing core damage shall then **ANALYZE** the report to determine best estimate of type and amount of damage.

NOTE: The CDAM program does not use the Fuel Overheat Condition Category

6.4. Based on estimated type and amount of damage and following table (table also printed on summary report) **ASSIGN** NRC Core Condition Category (1-4 or 8 -10).

NRC Core Condition Categories

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

7. **QUITING, OR EXITING, THE PROGRAM**

NOTE: When the program is closed all data is reset.

CAUTION

Program saves no information to disk; printed reports serve as record of core damage assessments.

7.1. **PRESS** the "Quit" button on the Summary Screen exits the program.