ATTACHMENT A

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

CORE DAMAGE ASSESSMENT (BWR)

1. PURPOSE

1.1. The purpose of this attachment is to provide emergency response personnel with the methodology to estimate the degree of possible core damage at Exelon Nuclear's Boiling Water Reactor (BWR) stations. 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. This is done to assist in:

1.2.1. Determining if the fuel barriers are breached to evaluate the appropriate Emergency Action Level (EAL) classification.

1.2.2. Providing input on core configuration (coolable or uncoolable) for prioritization of mitigating activities.

1.2.3. Determining the potential quantity and isotopic mix of a radiological release to project offsite doses.

1.2.4. Predicting the radiation protection actions that should be considered for long term recovery activities.

1.2.5. 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

This procedure is intended to provide an acceptable alternative to existing station core damage assessment models and methods utilized by Reactor Engineering.

2. TERMS AND DEFINITIONS

2.1. BWR – Boiling Water Reactor

2.2. Cladding – The outer coating (usually zirconium alloy, aluminum or stainless steel), 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
  Containment Volume (384,570 ft^3) = Suppression Pool (149,380 ft^3) + Drywell (235, 190 ft^3)
- Peach Bottom (Mark I): 764 assemblies
  Containment 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. 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.

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

2.7. 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.

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

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

2.10. 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.

2.11. 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.

2.12. 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 1500°F.

2.13. 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 1800°F, assuming no injection into the RPV.

2.14. Shutdown – As defined by station emergency operating procedures, normally assumed to be less than 4% for a BWR.

2.15. Slump – Relocation of molten reactor core during an accident.

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

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

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

2.19. 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.

2.20. TID – Total Isotopic Distribution

2.21. 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 Manager* shall coordinate core damage assessment activities.

4. MAIN BODY

4.1. Use the appropriate Attachment to assess status of the affected unit's reactor core:

4.1.1. For **Mid Alantic ROG** BWR Stations:

1. **ASSESS** the status of critical BWR safety functions and **MONITOR** for indications that the core may already been uncovered or may soon become uncovered.

2. **PROJECT** core damage if uncovered using the following methods, as applicable:
   - Attachment 3, "Core Uncovery Time Method"
   - Attachment 4, "Containment Hydrogen Concentration Method"

3. **MONITOR** radiation levels in containment to confirm and assess core damage.

4. **PROJECT** core damage using Containment Radiation Level Method (Attachment 1).

5. **CONTINUE** to assess core damage a reactor coolant sample to confirm extent of core damage using the Attachment 2 (Reactor Coolant Sampling Method).

6. **CONTINUE** to assess core damage and **INFORM** those assessing consequences, classification and protective actions.

4.1.2. For **Mid West ROG** BWR Stations refer to Attachment 5, BWR CDAM User Guide, to use the Core Damage Assessment Software Program.

5. DOCUMENTATION

5.1. Reactor core damage assessment methods derived from U.S. NRC RTM-96, Chapter A (Reactor Core Damage Assessment). Figures included in RTM-96, Chapter A were generated using CONDOS II (NUREG/CR-2068).

5.2. 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. U.S. Nuclear Regulatory Commission (NRC) Response Technical Manual (RTM) 96, March 1996

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

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

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

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

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

6.7. Exelon Nuclear Radiological Emergency Plan Annex for Limerick Generating Station

6.8. Exelon Nuclear Radiological Emergency Plan Annex for Peach Bottom Atomic Power Station EP-MW-123-1003, Rev.0, Core Damage Assessment Methodology (CDAM) Program Technical Basis

7. ATTACHMENTS

7.1. Attachment 1, Containment Radiation Level Method

7.2. Attachment 2, Reactor Coolant Sampling Method

7.3. Attachment 3, Core Uncovery Time Method

7.4. Attachment 4, Containment Hydrogen Concentration Method

7.5. Attachment 5, BWR CDAM User's Guide

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD
Page 1 of 8

Per RTM-96, Chapter A (Method A.4), these calculations should provide the maximum reading expected under the conditions stated. The calculations assume: (1) a prompt release to containment of all fission products in the coolant, spike, gap or from in-vessel core melt; (2) uniform mixing in the containment; and (3) an unshielded monitor that can see most of the containment volume (minus torus / suppression pool volume). Because the mix is most likely different from that assumed in the calibration of the monitor, the actual reading at the upper end of the scale could differ by a factor of 10-100 if a shielded detector is used for higher radiation measurements.

The levels of damage indicated should be considered minimum levels unless there are inconsistent monitor readings. Inconsistent readings may be caused by the uneven mixing in containment (e.g., steam rising to top of dome, not enough time for uniform mixing to occur).

Figures have been prepared for each station, which represent plots of percent core damage versus containment radiation readings and percent clad damage versus containment radiation readings. Uncorrected containment radiation readings shall be used to provide a preliminary estimate of percent core and clad damage during the first ten (10) hours following a reactor shutdown.

CAUTION

Containment radiation levels cannot be used to assess core damage in all cases. The release may bypass the containment, be retained in the coolant, be released over a long period of time, or not be uniformly mixed. Therefore, a low containment radiation level reading does not guarantee a lack of core damage.

This method should not be used if containment radiation monitors do not "see" more than 50% of Drywell volume.

1. **MONITOR** containment radiation levels to assess the degree of clad and/or core damage.

   NOTE: The readings taken following a Loss of Coolant Accident (LOCA) into the primary containment will be most accurate using the containment radiation monitors.

2. **DETERMINE** the time following the event and the uncorrected containment radiation reading and compare to the following figures for assessing the degree of core or clad damage.

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD
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Limerick

REFER to Figure 1-1 for PERCENT OF FUEL INVENTORY AIRBORNE IN THE CONTAINMENT VERSUS APPROXIMATE SOURCE AND DAMAGE ESTIMATE

Peach Bottom

REFER to Figure 1-2 for PERCENT OF FUEL INVENTORY AIRBORNE IN THE CONTAINMENT VERSUS APPROXIMATE SOURCE AND DAMAGE ESTIMATE

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD

FIGURE 1-1
LIMERICK GENERATING STATION
Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate

Curve No.	% Fuel* Inventory Released	Approximate Source and Damage Estimate
1	100.	100% TID-14844, 100% fuel damage, potential core melt.
	50.	50% TID noble gases, TMI source.
2	10.	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
	3.	3% TID, 100% WASH-1400 gap activity, major clad failure.
3	1.	1% TID, 10% NRC gap, Max. 10% clad failure.
4	.1	.1% TID, 10% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
5	.01	.01% TID, .1% NRC gap, clad failure of 3/4 of a fuel assembly (36 rods).
6	10^{-3}	.01% NRC gap, clad failure of a few rods
	10^{-4}	100% coolant release with spiking.
7	$5X10^{-6}$	100% coolant inventory release.
	10^{-6}	Upper range of normal airborne noble gas activity in containment.

* 100% Fuel Inventory = 100% Noble Gases +25% Iodine + 1% particulates.

NOTES (1) These curves account for the finite containment volume and shield wall seen by the detector but do not account for any monitor physical or shielding characteristics or calibration uncertainties.
(2) The curves assume that both airborne noble gases and iodines are significant. Sprays (if used) would make the iodine and particulate contribution (presently about 50%) insignificant. However, particulate plateout on the monitor casing and direct shine doses from components may make the readings unreliable.
(3) Curve uncertainties are on the order of a factor of 5 to 10.

LIMERICK STATION

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD
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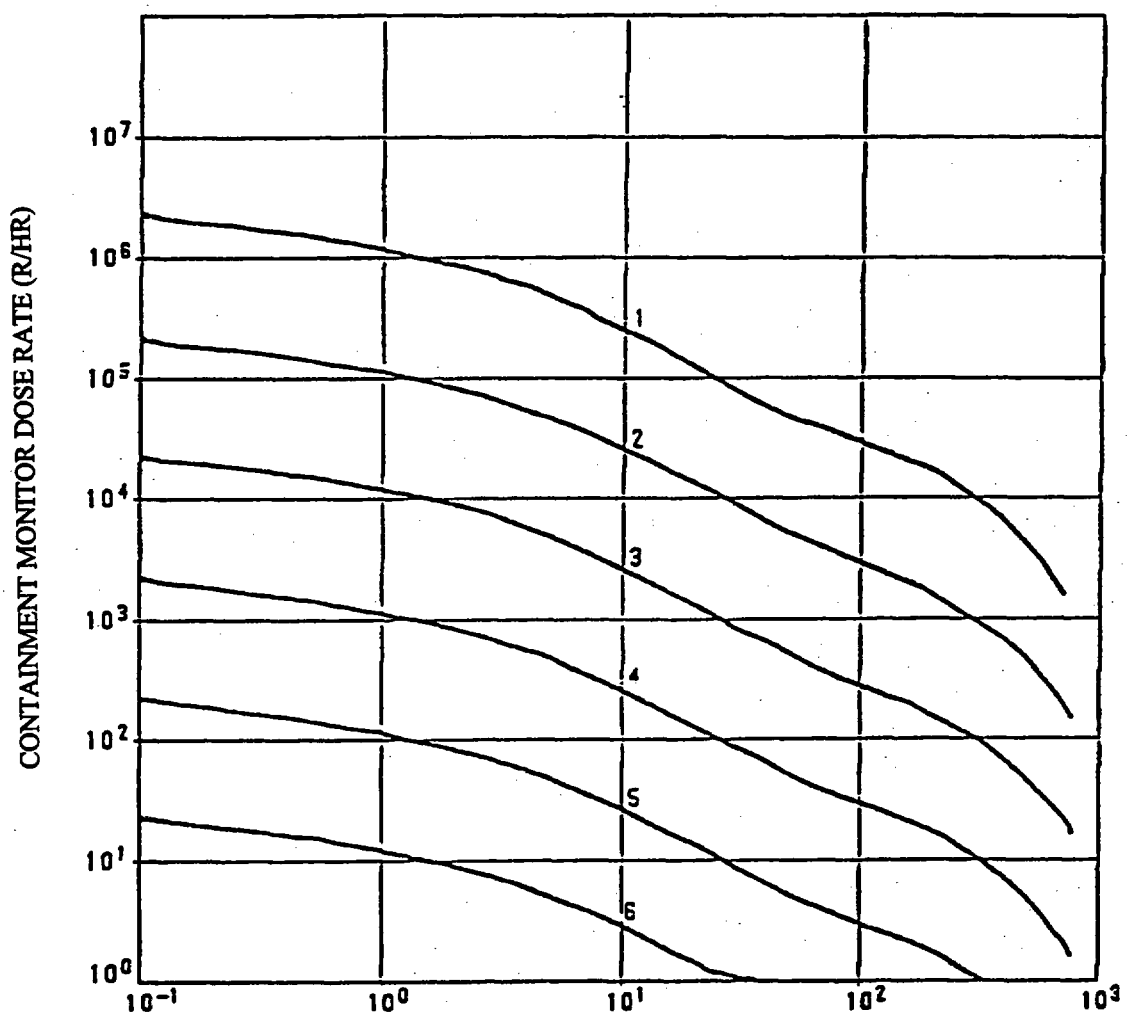
FIGURE 1-1
LIMERICK GENERATING STATION
Percent of Fuel Inventory Airborne in the Containment
vs. Approximate Source and Damage Estimate
Page 2 of 3

LGS CONTAINMENT RADIATION MONITOR CURVES



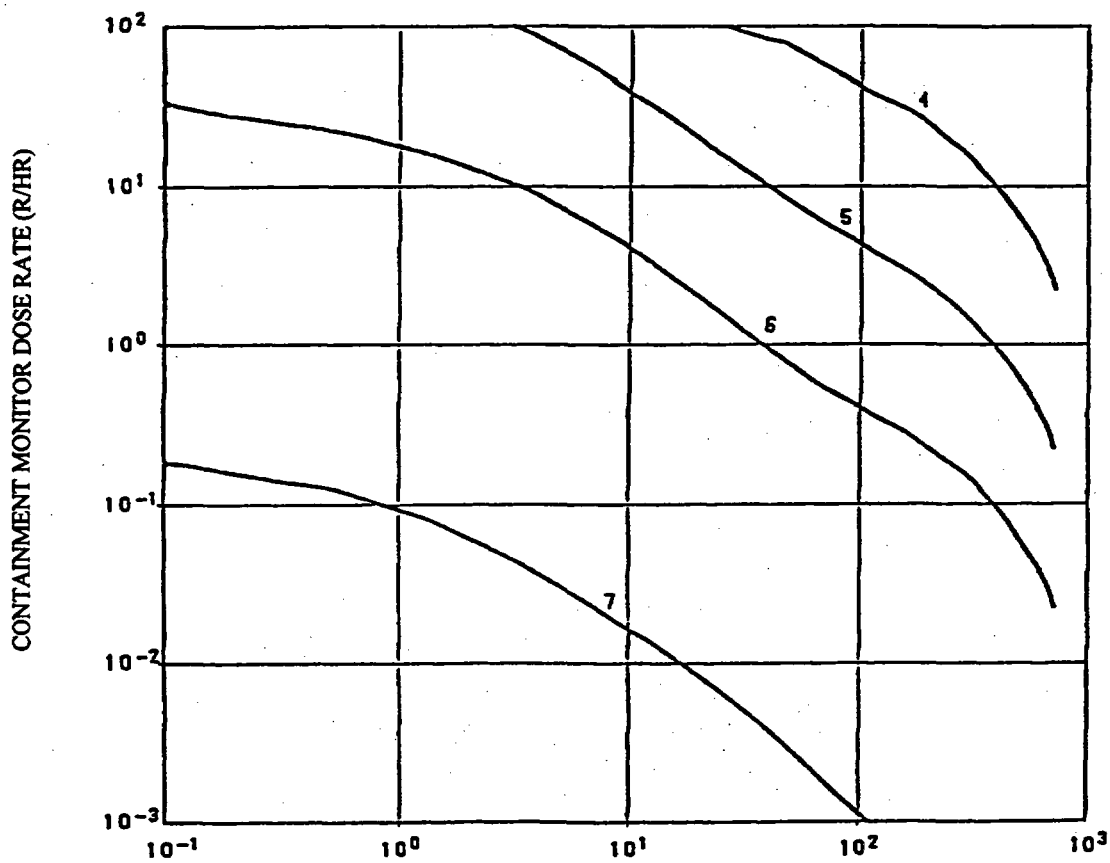
1 - 100% TID
2 - 10% TID
3 - 1% TID
4 - 0.1% TID
5 - 0.01% TID
6 - 0.001% TID

LIMERICK STATION

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD
Page 5 of 8

FIGURE 1-1
LIMERICK GENERATING STATION
Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate
Page 3 of 3

LGS CONTAINMENT RADIATION MONITOR CURVES



TIME AFTER SHUTDOWN (HRS)

4 - 0.1% TID
5 - 0.01% TID
6 - 0.001% TID
7 - 100% REACTOR COOLANT

LIMERICK STATION

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD

FIGURE 1-2
PEACH BOTTOM ATOMIC POWER STATION
Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate

Curve No.	% Fuel* Inventory Released	Approximate Source and Damage Estimate
1	100.	100% TID-14844, 100% fuel damage, potential core melt.
	50.	50% TID noble gases, TMI source.
2	10.	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
	3.	3% TID, 100% WASH-1400 gap activity, major clad failure.
3	1.	1% TID, 10% NRC gap, Max. 10% clad failure.
4	.1	.1% TID, 10% NRC gap, 1% clad failure, local heating of 5-10 fuel assemblies.
5	.01	.01% TID, .1% NRC gap, clad failure of 3/4 of a fuel assembly (36 rods).
6	10^{-3}	.01% NRC gap, clad failure of a few rods
	10^{-4}	100% coolant release with spiking.
7	$5X10^{-6}$	100% coolant inventory release.
	10^{-6}	Upper range of normal airborne noble gas activity in containment.

* 100% Fuel Inventory = 100% Noble Gases +25% Iodine + 1% particulates.

NOTES (1) These curves account for the finite containment volume and shield wall seen by the detector but do not account for any monitor physical or shielding characteristics or calibration uncertainties.
(2) The curves assume that both airborne noble gases and iodines are significant. Sprays (if used) would make the iodine and particulate contribution (presently about 50%) insignificant. However, particulate plateout on the monitor casing and direct shine doses from components may make the readings unreliable.
(3) Curve uncertainties are on the order of a factor of 5 to 10.

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD
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FIGURE 1-2
PEACH BOTTOM ATOMIC POWER STATION
Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate
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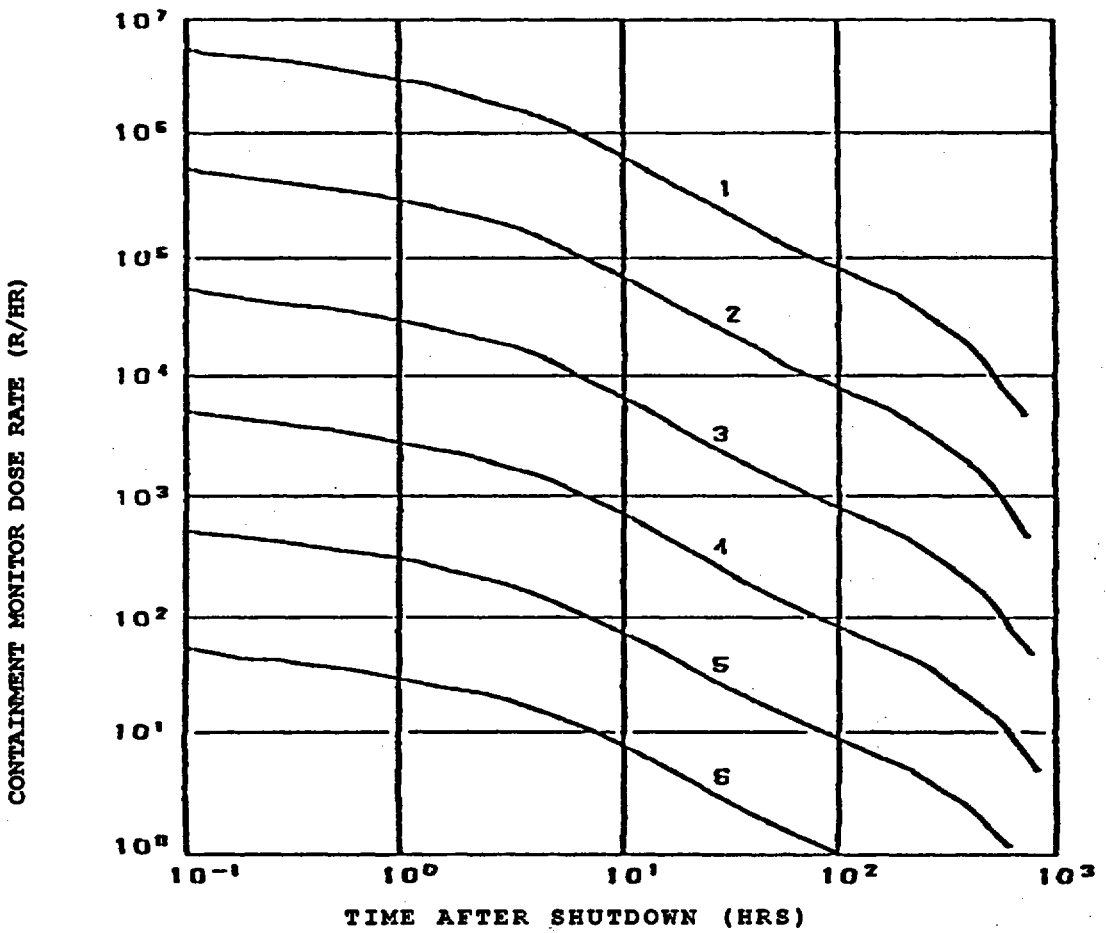
PEACH BOTTOM CONTAINMENT RADIATION MONITOR CURVES



PEACH BOTTOM MONITOR RESPONSE CURVES
FOR
PRIMARY CONTAINMENT LOW RANGE MONITORS

CURVE INDEX		
1.	100%	FUEL INVENTORY (100% TID-14844)
2.	10%	FUEL INVENTORY (100% GAP ACTIVITY/R.G. 1.25)
3.	1%	FUEL INVENTORY (10% NRC GAP-CLAD FAILURE)
4.	0.1%	FUEL INVENTORY
5.	0.01%	FUEL INVENTORY
6.	0.001%	FUEL INVENTORY

PEACH BOTTOM STATION

ATTACHMENT 1
CONTAINMENT RADIATION LEVEL METHOD
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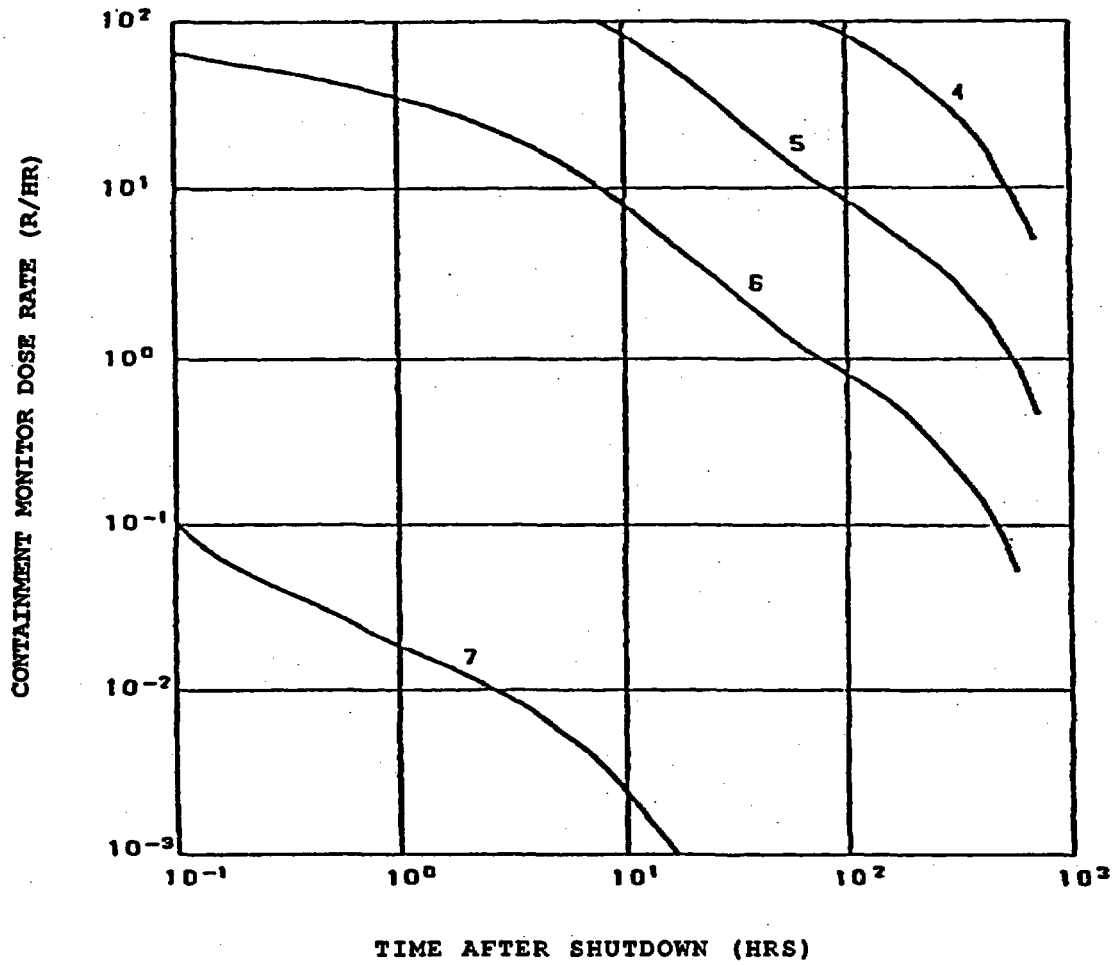
FIGURE 1-2
PEACH BOTTOM ATOMIC POWER STATION
Percent of Fuel Inventory Airborne in the Containment vs. Approximate Source and Damage Estimate
Page 3 of 3

PEACH BOTTOM CONTAINMENT RADIATION MONITOR CURVES



PEACH BOTTOM MONITOR RESPONSE CURVES FOR PRIMARY CONTAINMENT LOW RANGE MONITORS

CURVE INDEX
4 - 0.1% FUEL INVENTORY
5 - .01% FUEL INVENTORY
6 - .001% FUEL INVENTORY
7 - 100% COOLANT ACTIVITY

PEACH BOTTOM STATION

ATTACHMENT 2
REACTOR COOLANT SAMPLING METHOD
Page 1 of 1

Per RTM-96, Chapter A (Method A.5), for BWR this method of confirming core damage assumes that releases from the core are uniformly mixed in the coolant and torus / suppression pool with there is no dilution from injection. If most of the core release is confined to the reactor coolant system, the concentrations in the coolant could be up to 10 times higher. The baseline coolant concentrations are for 0.5 hours after shutdown of a core that has been through at least one refueling cycle. The half-life of the fission products should be considered in analyzing samples. The plant-specific coolant system volume does not have a major influence on coolant concentrations (< 20%).

CAUTION

Since samples may require hours to collect and analyze, and may not be representative of primary system concentrations (e.g., no flow through sample line), coolant concentrations should NOT be required to confirm core damage.

1. **COMPARE** coolant isotopic concentrations with the concentrations listed in Table 2-1 to estimate core damage for gap releases or in-vessel melting.

Table 2-1
BASELINE BWR COOLANT CONCENTRATION[1]

Nuclide	Normal Concentration[2] (uCi/g)	Concentration after gap release[3] (uCi/g)	Concentration after in-vessel melt[3] (uCi/g)
^{131}I	2×10^{-3}	1×10^{3}	1×10^{4}
^{133}I	1×10^{-2}	3×10^{3}	2×10^{4}
^{135}I	2×10^{-2}	2×10^{3}	2×10^{4}
^{134}Cs	3×10^{-5}	1×10^{2}	6×10^{2}
^{137}Cs	8×10^{-5}	8×10^{1}	4×10^{2}
^{140}Ba	NC	NC	2×10^{3}
^{90}Sr	7×10^{-6}	NC	1×10^{3}

NC – Not Calculated

[1] Table 2-1 is consistent with Table A-4 (BWR Baseline Coolant Concentration) from U.S. NRC Response Technical Manual (RTM) 96, March 1996
[2] Source for "normal coolant" is ANSI/ANS 18.1, 1984 (confirmed by NUREG/CR-4245, Table 3.2)
[3] Concentration in the reactor coolant system and torus / suppression pool

ATTACHMENT 3
CORE UNCOVERY TIME METHOD
Page 1 of 1

Per RTM-96, Chapter A (Method A.3), it can be assumed that the fuel in the core will heat up at 1-2°F per second within 5-10 minutes after the top of an active core isuncovered. These heatup estimates are reasonable within a factor of two if the core is uncovered within a few hours of shutdown (including failure to scram) for a boil-down case (without injection). If there is injection, core heatup may be stopped or slowed because of steam cooling. However, steam cooling may not prevent core damage under accident conditions.

1. **DETERMINE** the time that the core was or is projected to be uncovered.

2. **ESTIMATE** the possible core damage by comparing the time of core uncovery to the values in Table 3-1.

Table 3-1
CORE DAMAGE VS. TIME THAT REACTOR CORE IS UNCOVERED[4]

Time 20% of the core is uncovered (minutes)	Core Temperature (°F)	Core Temperature (°C)	Possible Core Damage
0	>600	>315	**None**
30-45	1800-2400	980-1300	**GAP Release** – Local fuel melting – Burning of cladding with steam production (exothermic Zr-$H_2$0 reaction with rapid H_2 generation) – Rapid fuel cladding failure (gap release core)
30-90	2400-4200	1300-2300	**In-Vessel Melt** – Rapid release of volatile fission products (in-vessel severe damage release from the core) – Possible relocation (slump) of molten core – Possible uncoolable core
120-180+	>4200	>2300	**Vessel Melt-Through** Melt-through of vessel with possible containment failure and release of additional less-volatile fission products

[4] Table 3-1 is consistent with Table A-2 (Core Damage vs. Time That Reactor Core Is Covered) from U.S. NRC Response Technical Manual (RTM) 96, March 1996

ATTACHMENT 4
CONTAINMENT HYDROGEN CONCENTRATION METHOD
Page 1 of 1

This method may be used to assess the core damage based on the hydrogen concentration in samples of the containment atmosphere.

Per RTM-96, Chapter A (Method A.6), **the hydrogen concentrations using this method are for wet samples**; however, most hydrogen samples are dry (steam removed). If a dry sample concentration is used, one may overestimate considerably the level of core damage. This method assumes that all hydrogen is released to the containment and is completely mixed in the containment atmosphere.

> **CAUTION**
>
> Since containment samples may require hours to collect and analyze, and may not be representative of actual concentrations, hydrogen should NOT be relied upon to confirm core damage in all cases.

1. **COMPARE** the hydrogen percentage obtained to the graph in Figure 3 to estimate percentage of metal-water reaction and possible levels of core damage.

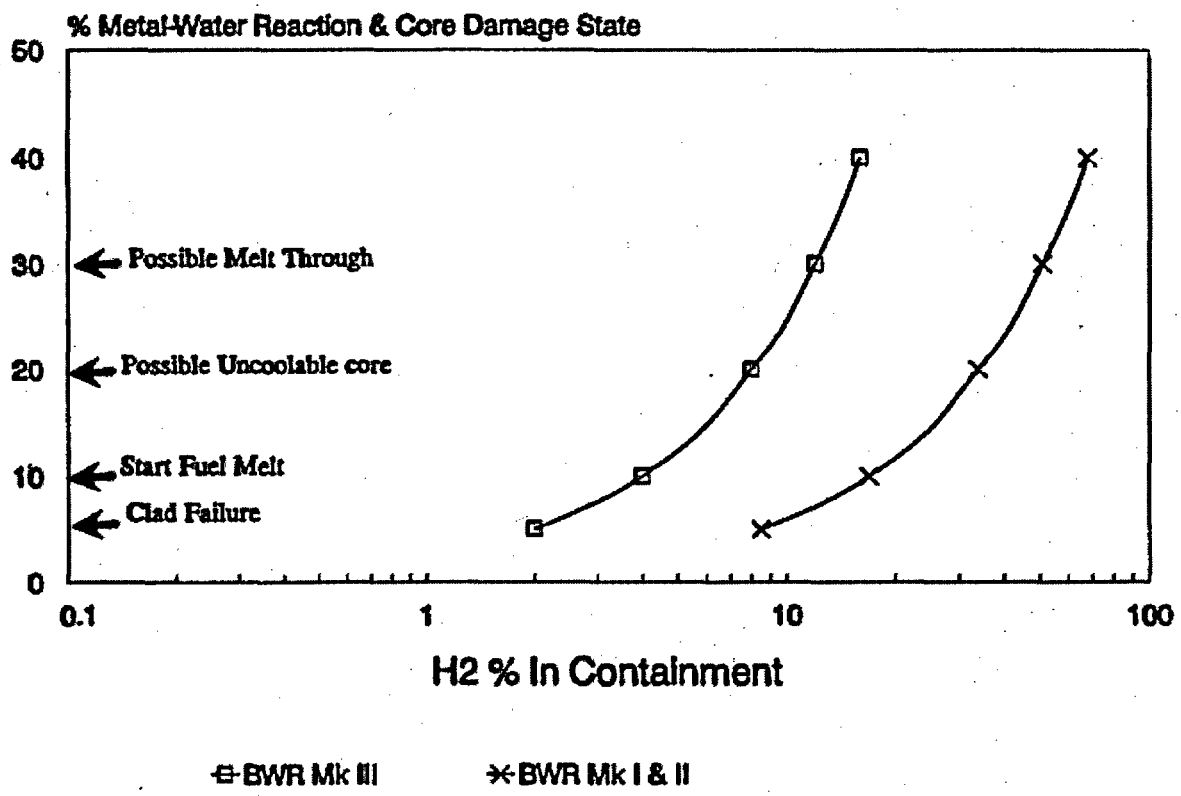


Figure 4-1[5]

[5] Figure 4-1 is consistent with Figure A-13 (Percentage of Hydrogen in Containment Relative to Core Damage) from U.S. NRC Response Technical Manual (RTM) 96, March 1996

ATTACHMENT 5
BWR CDAM User's Guide
PAGE 1 OF 18

1. OVERVIEW

1.1. As a windows based application designed in 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 ROG

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

A. The primary indicators of core damage that are available during the early phases of an event:

B. Drywell/Containment Radiation Monitor Readings

C. Drywell/Containment Hydrogen Readings

2.1.2. Auxiliary indicators that are used to confirm and better define the possible type of damage are:

A. Reactor Pressure Vessel Level Indication System readings

B. Estimation of maximum temperature reached within the core

C. Estimated core uncovery time

D. Abnormal Source Range Monitor readings

2.1.3. Long Term Indicators (once liquid or gaseous samples can be safely obtained) are:

A. Isotopic Ratios

B. Presence of high levels of rare isotopes

C. Quantity of isotopes present in samples

2.2. Choose 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

3.1. The application is accessed by one of the following:

3.1.1. Open the BWR CDAM desktop icon on applicable dose assessment computers.

A. Start the BWR CDAM program for the plant that has declared an emergency.

B. Programs are labeled BWR CDAM.

3.1.2. Select run from the 'start bar' and type in the file path and name as follows:

C:\CDAM\BWR CDAM.MDB

3.2. 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. CDAM is installed by copying appropriate file to computer's hard drive.

4. SELECTION AND PERFORMANCE OF ASSESSMENT

4.1. Choose 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:

4.1.1. Containment Radiation Analysis - (Section 5.2)

4.1.2. Core Conditions Analysis (Cooling History) - (Section 5.3)

4.1.3. Containment Hydrogen Analysis - (Section 5.4)

4.1.4. Nuclide Analyses (Ratios and Abnormal Isotopes) - (Section 5.5)

4.1.5. Liquid Samples Analysis - (Section 5.6)

4.1.6. Gaseous Samples Analysis - (Section 5.7)

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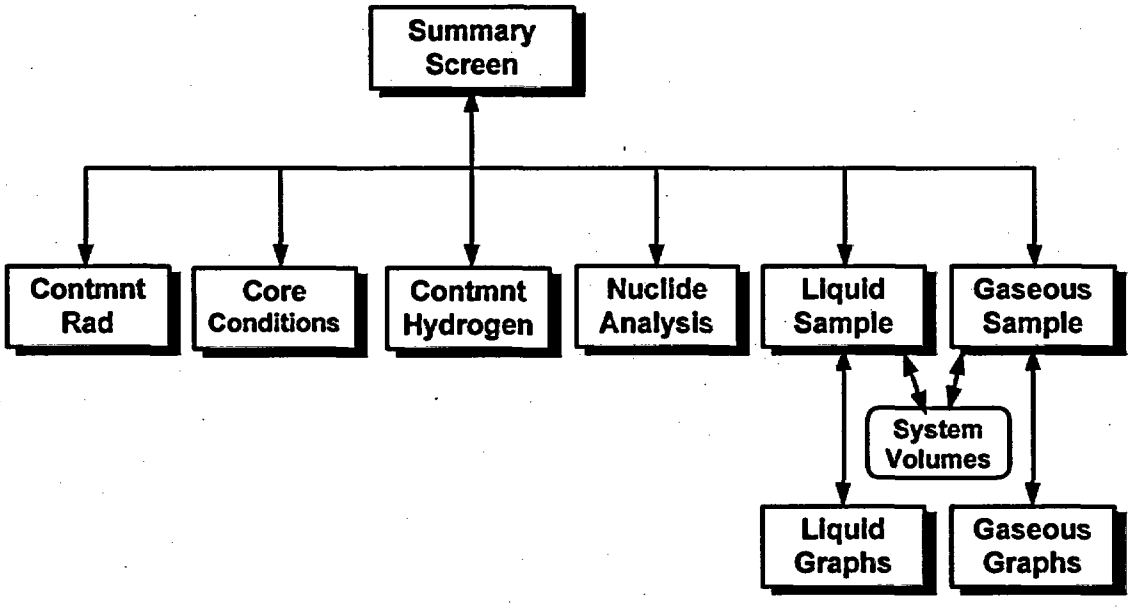
4.2. Basic Program Flow Diagram



5. PROGRAM SCREENS AND INPUTS

5.1. Main Screen – Summary Page

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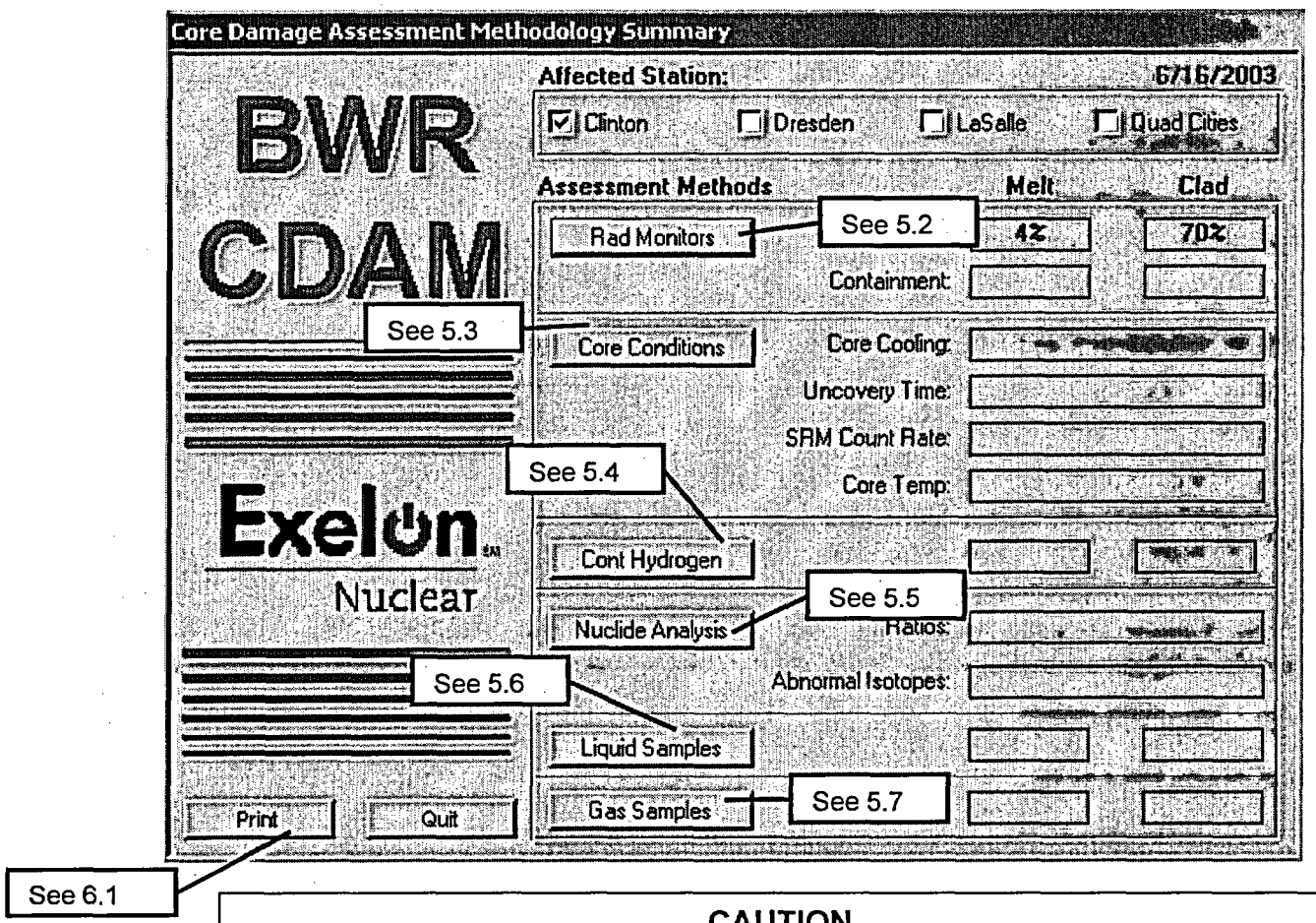
5.1.1. When program is started the following screen appears: (boxes are empty when program is originally launched.)



CAUTION

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

5.1.2. 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.

5.2. Containment Radiation Monitor Method

5.2.1. Pressing "Cont Rad Monitors" button opens the following form:

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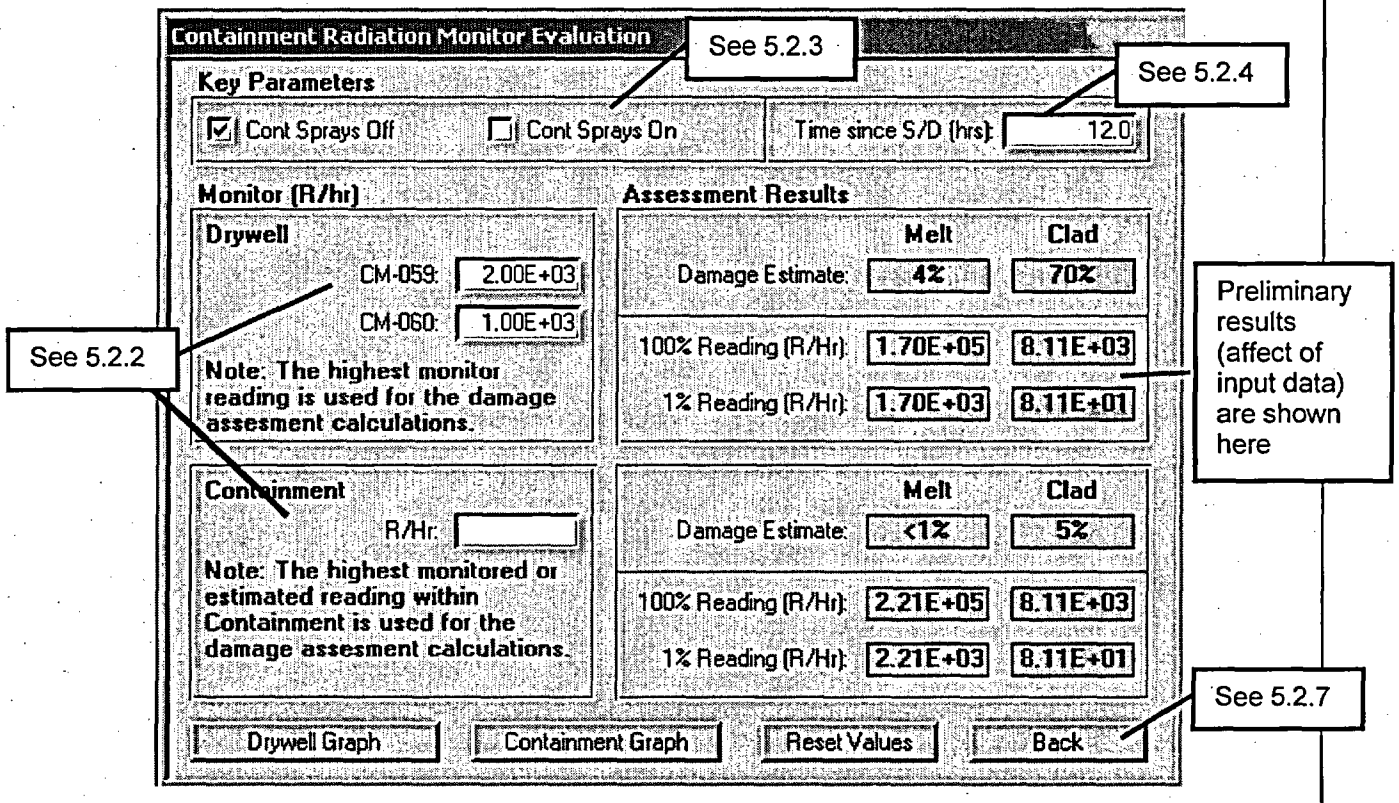

5.2.2. Highest drywell or containment / Suppression Chamber radiation monitor reading that occurred is entered in these boxes. Program allow entry from 2 high range monitors for drywell location and 1 for Torus or Suppression Chamber, however a reading may be entered from any monitor which accurately showed containment radiation levels. If two entries are made only the highest is used.

5.2.3. Drywell/Containment Spray

A. **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".

B. **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".

5.2.4. 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.

5.2.5. Pressing "Reset" button resets values on this form only.

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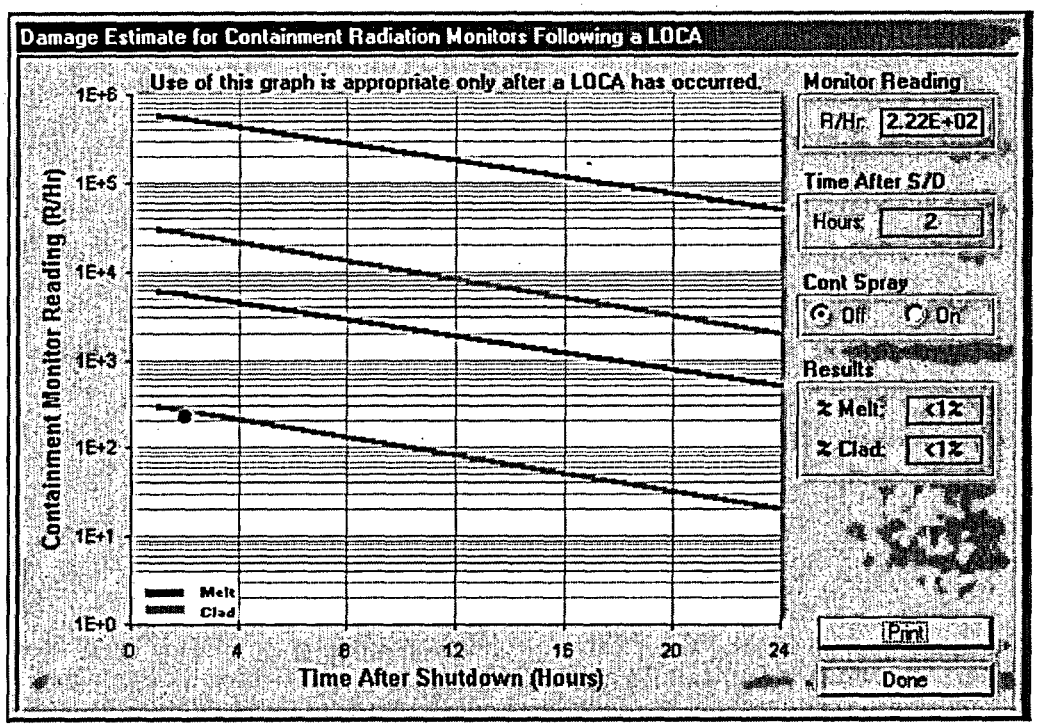
5.2.6. Pressing "Containment Graph" or "Supp Chamber Graph" button displays a screen similar to the following:



A. 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.

B. Pressing "**Print**" button will print report of containment radiation method inputs and best estimate of damage.

5.2.7. Pressing "**Back**" button takes the user back to the summary screen.

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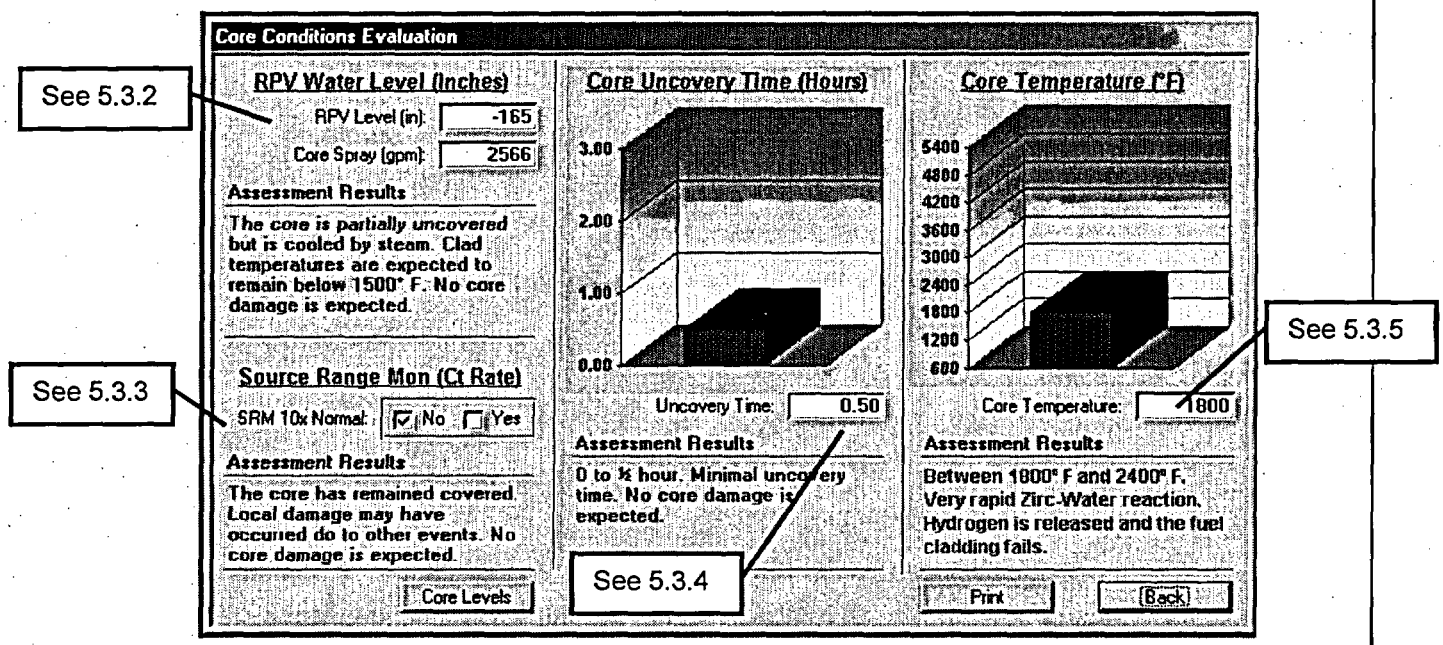
5.3. Core Conditions Methods

5.3.1. Pressing **"Core Conditions"** button opens the following form:



5.3.2. Reactor Pressure Vessel (RPV) Water Level

A. Enter the lowest recorded (or estimated) RPV Level (range 0 to –350) and core spray flow at time of lowest reading.

5.3.3. Source Range Monitor.

A. Review plant parameter history and if the SRM had indications of a reading 10 times those expected check yes.

5.3.4. Based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as General Electric personnel) enter the estimated time the reactor core (20% of top of active core) was uncovered without steam or spray cooling reactor core.

5.3.5. Based on inputs from Reactor Operators, TSC Staff and other engineering personnel (including outside sources such as General Electric personnel) enter the estimated highest temperature reached in the reactor core.

5.3.6. Pressing "**Reset**" button resets values on this form only.

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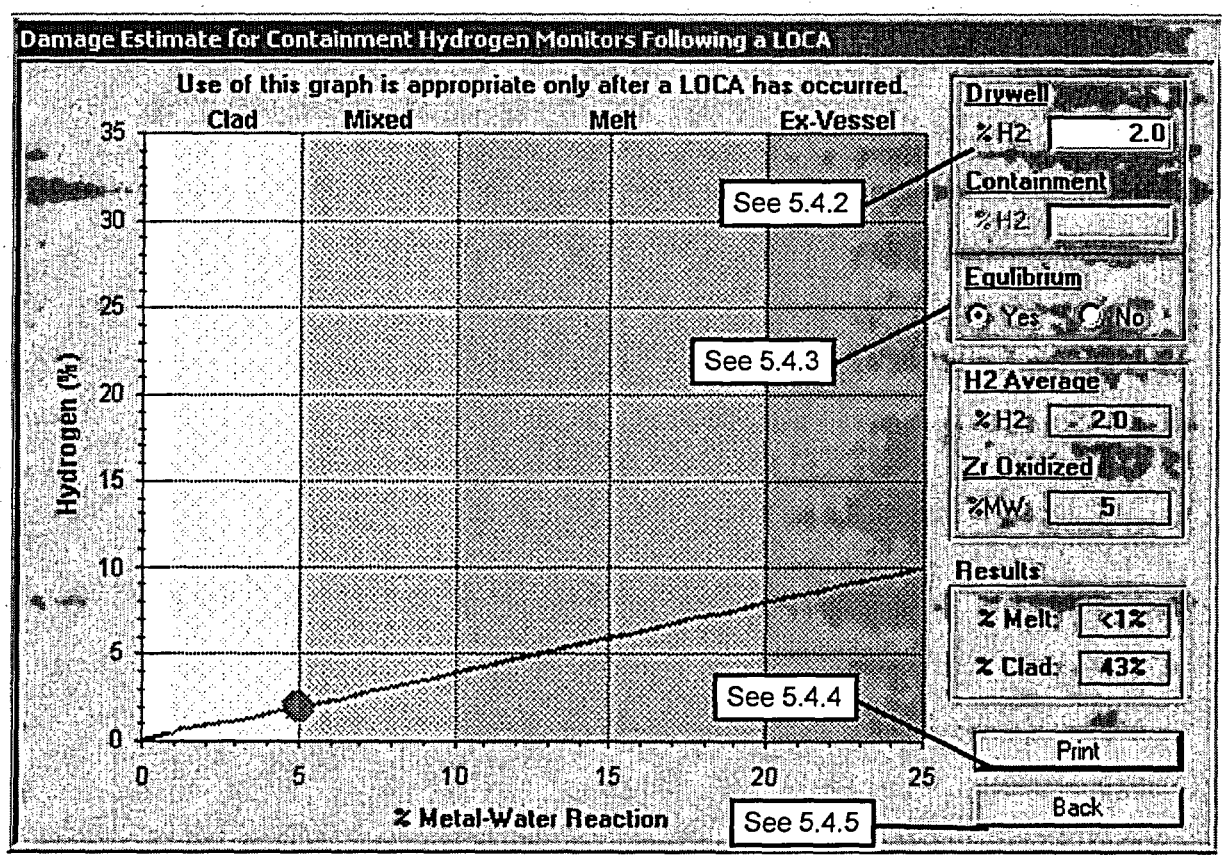
5.3.7. Pressing "**Print**" button prints report of inputs and results of core temperature methods of core damage assessment.

5.3.8. Pressing "**Back**" button takes the user back to the summary screen.

5.4. Containment Hydrogen Evaluations

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

5.4.1. Pressing **"Cont Hydrogen"** button opens the following form:



5.4.2. Enter highest Drywell and/or Suppression Chamber hydrogen level measured. Suppression chamber reading can only be entered if user selects "no" under Equilibrium.

5.4.3. System Equilibrium:

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A. Select "**Yes**" for equilbrium if containment and suppression chamber monitors read the same or only atmospheres are assumed equalized.

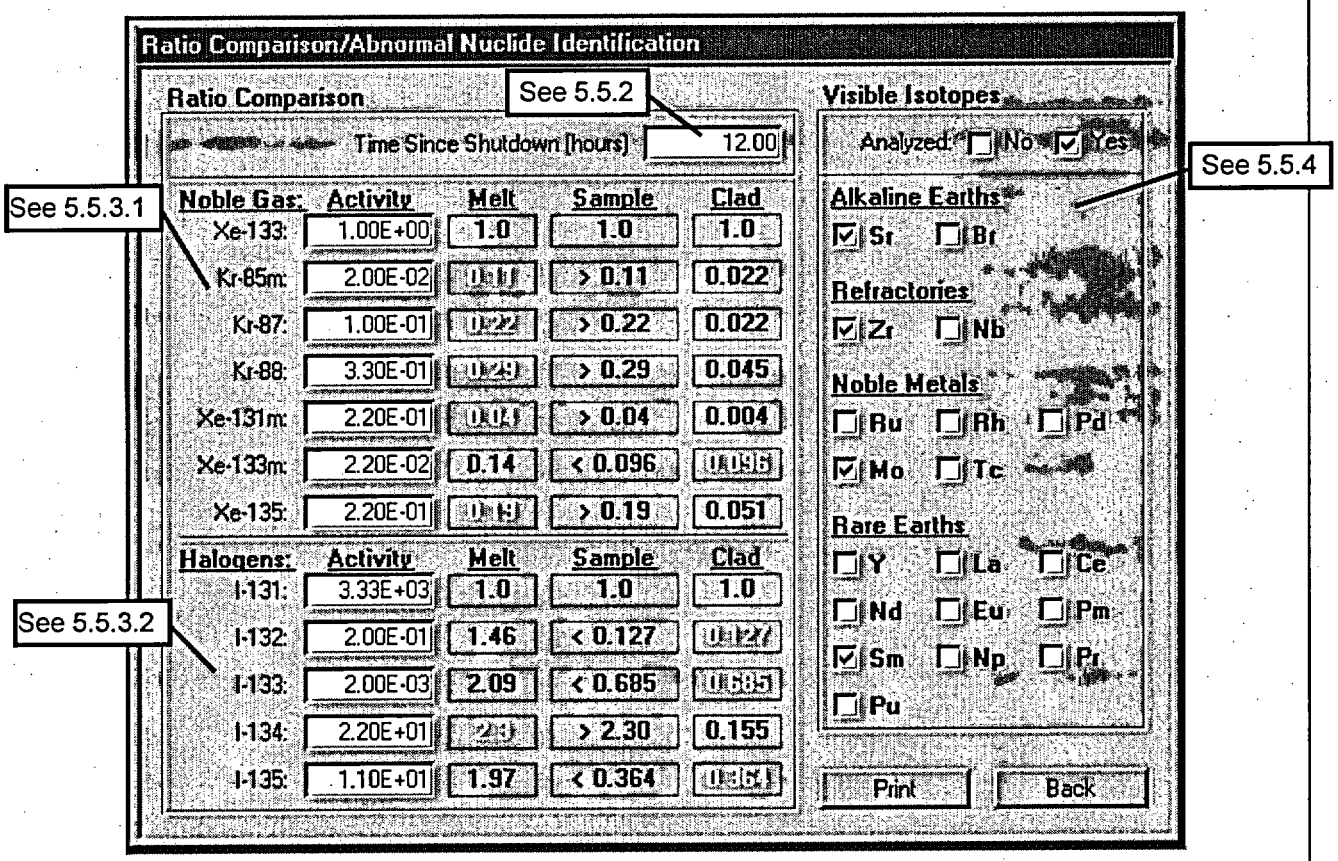
B. Select "**No**" for equilbrium if containment and suppression chamber atmospheres are not in equilibrium or only Containmen H2 reading is available.

5.4.4. Pressing "**Print**" button prints report of inputs and results of core level methods of core damage assessment.

5.4.5. Pressing "**Back**" button takes the user back to the summary screen.

5.5. Nuclide Analysis

5.5.1. Pressing **"Nuclide Analysis"** button opens the following form:



5.5.2. Enter the time since reactor shutdown when the sample was taken and sample results in uCi/cc.

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5.5.3. If the ratio is greater than predicted melt ratio, melt damage is predicted if less than clad ratio, clad damage is predicted.

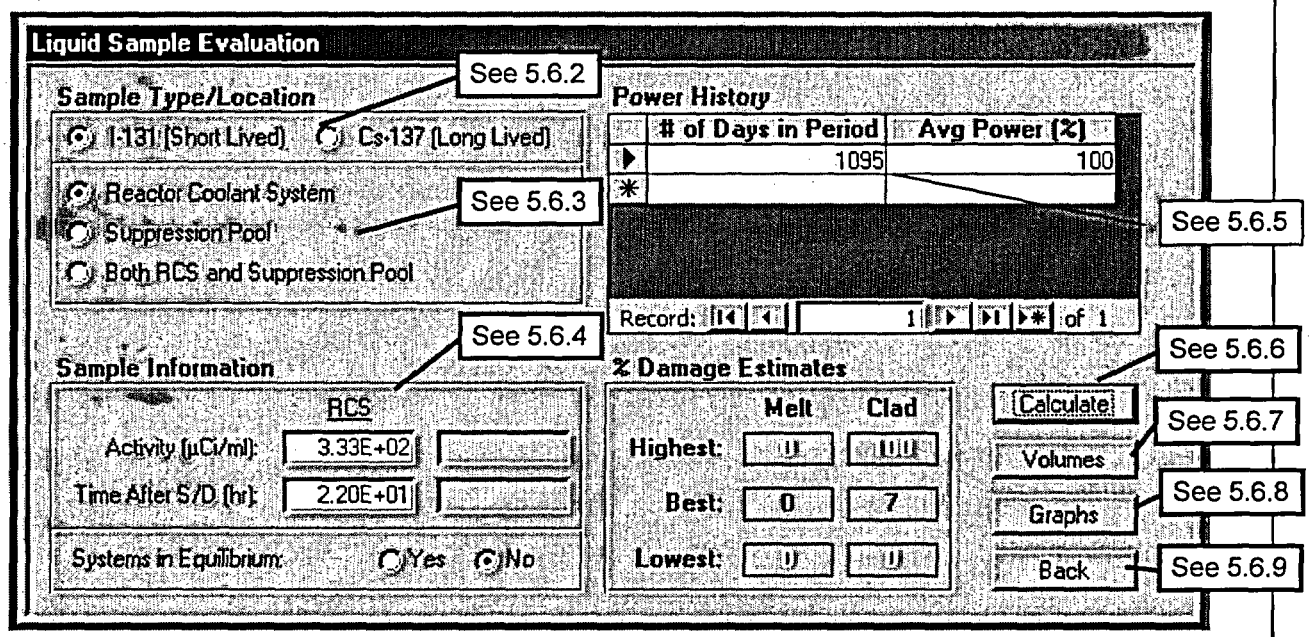
A. Noble Gases are ratioed to Xe-133

B. Halogens are ratioed to I-131

5.5.4. **IF** abnormal levels of rare isotopes are present **THEN** check yes **AND** check which isotopes are present.

5.6. Liquid Samples

5.6.1. Pressing "**Liquid Samples**" button opens the following form:



5.6.2. Select appropriate isotope.

5.6.3. Select sample location. If samples are available from both locations select both.

5.6.4. Enter sample activity(s) and Time After S/D that samples were taken. If sample was taken from only one location and systems are in equilibrium check yes for "Systems in Equilibrium".

5.6.5. 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.

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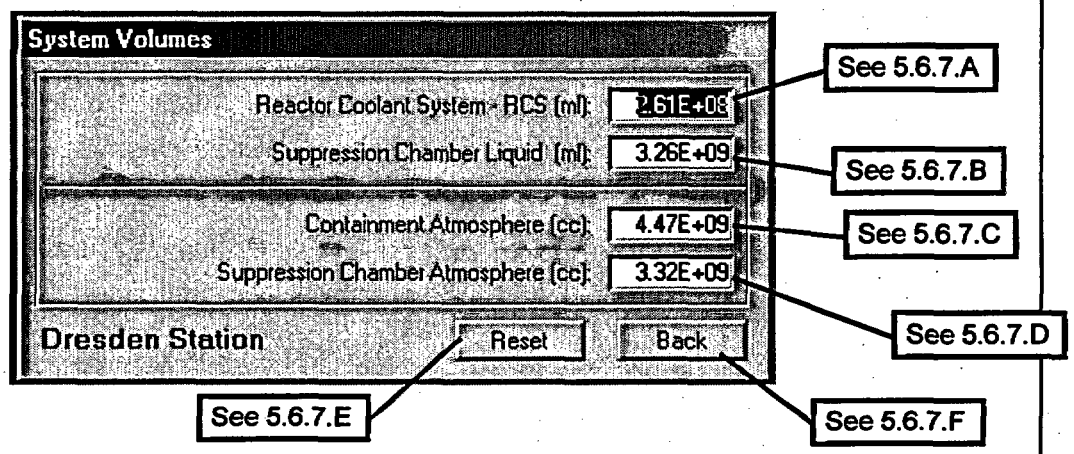
A. For short-lived isotopes power history should extend at least 30 days.

B. For long-lived isotopes power history should extend at least 100 days, however the power history for the extent of the cycle is preferred.

C. Variations in steady state power should be limited to ± 20% within each operational period entered.

5.6.6. Once all data has been entered pressing the "Calculate" button will display the % Damage Estimates.

5.6.7. Pressing "**Volumes**" button displays the follow screen:



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

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

C. 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.

D. 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.

E. Pressing "**Reset**" button resets all volumes to default values.

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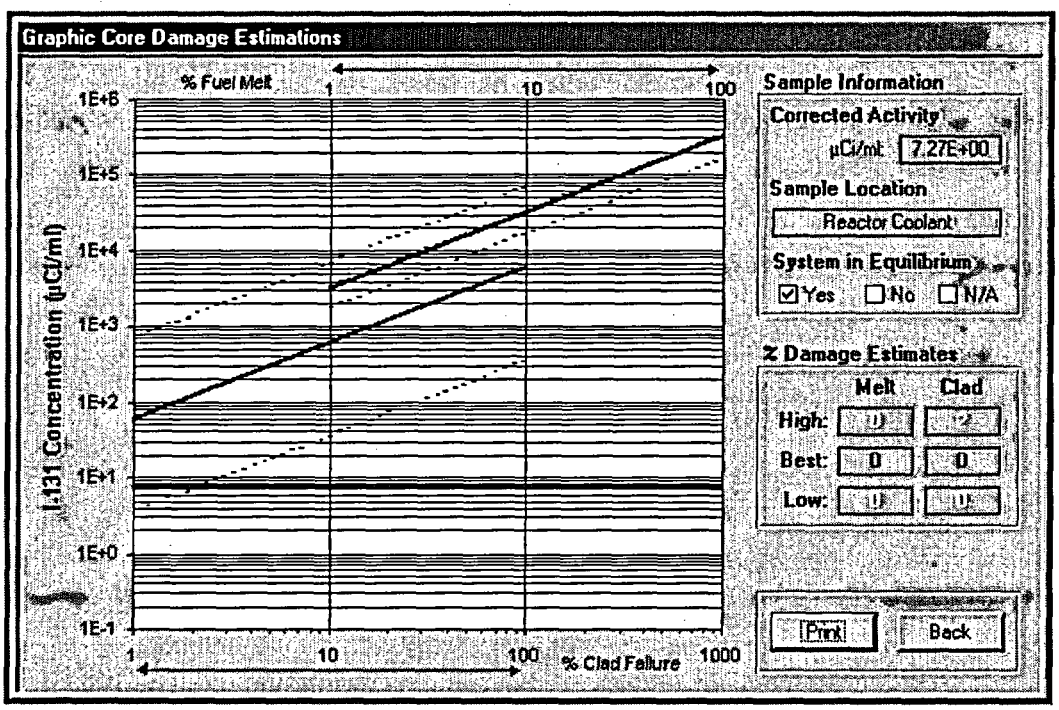
F. Pressing "**Back**" button takes the user back to the Liquid or Gaseous screen, which user used to call volume form.

5.6.8. Pressing "**Graph**" button displays the following screen:



A. Graph 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.

B. User can select "**Print**" button to print graph and summary of inputs or press "Back" button to go back to liquid or gaseous form which called this form.

5.6.9. Pressing "**Back**" button takes the user back to the summary screen.

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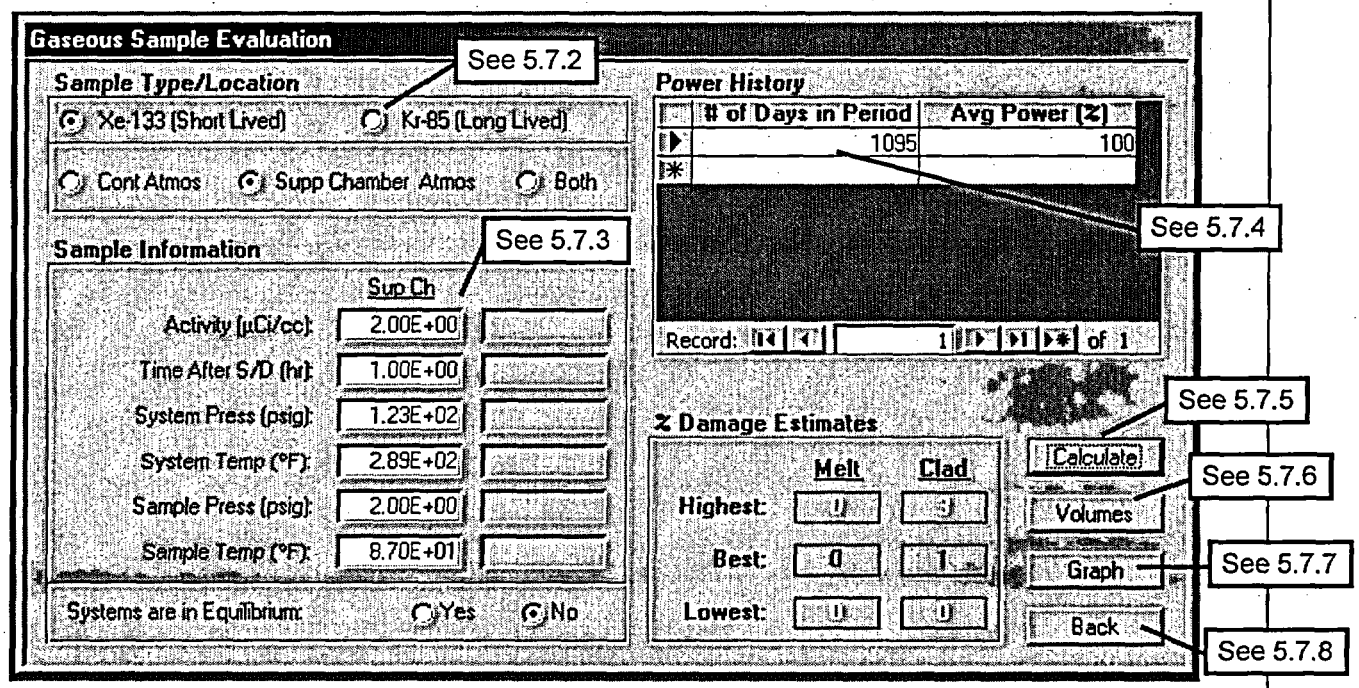
5.7. Gaseous Samples

5.7.1. Pressing "**Gas Samples**" button opens the following form:



5.7.2. Select appropriate isotope and sample location.

5.7.3. Enter Sample Information:

A. Enter sample activity for selected isotope.

B. Enter Time After S/D that sample was taken.

C. Enter the pressure and temperature of the system sampled

D. Enter the end pressure and temperature of sample.

5.7.4. Enter power history 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.

A. For short-lived isotopes power history should extend at least 30 days.

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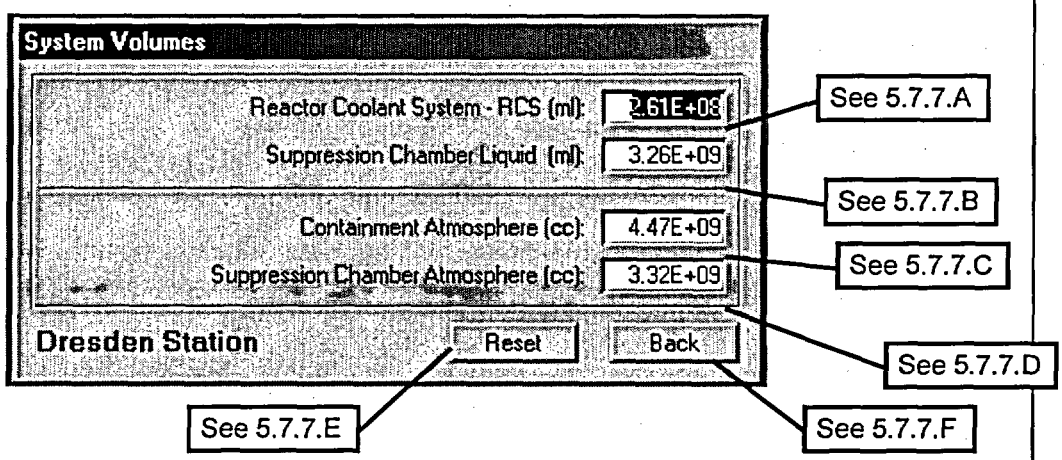
B. For long-lived isotopes power history should extend at least 100 days, however the power history for the extent of the cycle is preferred.

C. Variations in steady state power should be limited to ± 20% within each operational period entered.

5.7.5. Once all data has been entered pressing the "Calculate" button will display the % Damage Estimates.

5.7.6. Pressing "**Volumes**" button displays the follow screen (same as 5.6.7):



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

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

C. 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.

D. 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.

E. Pressing "**Reset**" button resets all volumes to default values.

F. Pressing "**Back**" button takes the user back to the Liquid or Gaseous screen, which user used to call volume form.

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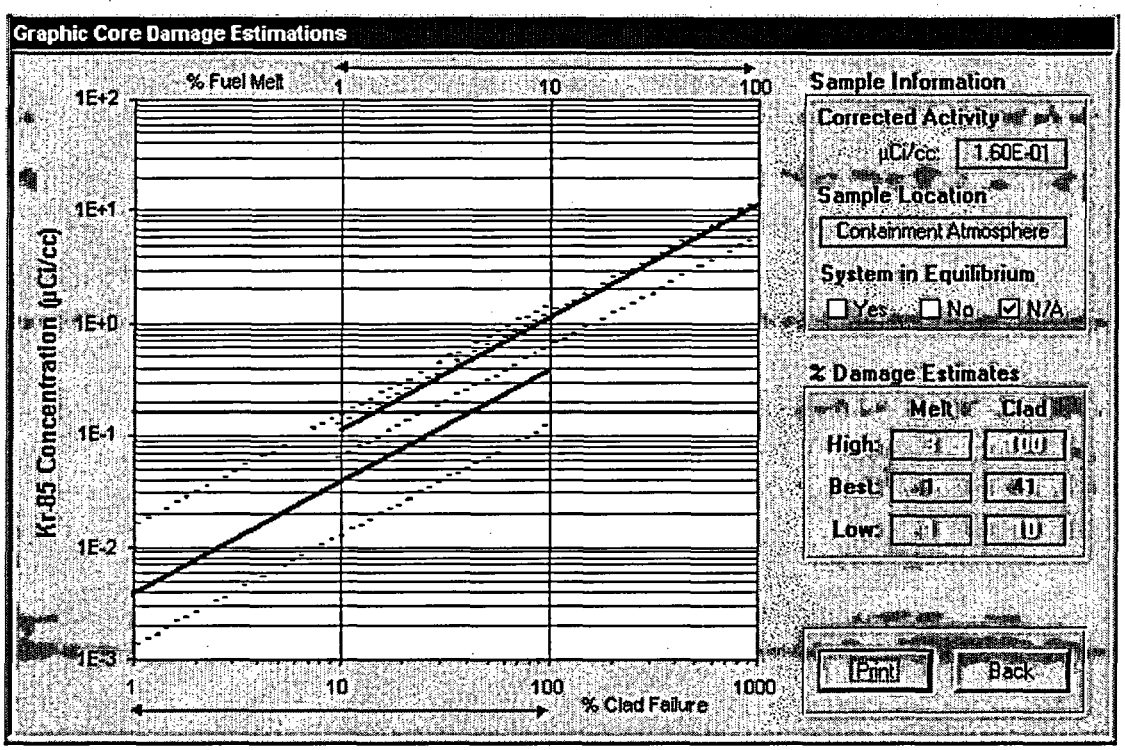
5.7.7. Pressing **"Graph"** button displays the following screen:



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

B. User can select **"Print"** button to print graph and summary of inputs or press "Back" button to go back to liquid or gaseous form which called this form.

5.7.8. Pressing **"Back"** button takes the user back 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. A sample report is shown on the next page.

6.3. Individual tasked with assessing core damage shall then analyze 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 (1-10).

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

7.1. Pressing the "Quit" button on the Summary Screen exits the program.

7.1.1. When the program is closed all data is reset.

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

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SAMPLE SUMMARY REPORT

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**