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Waterford 3

W3F1-2000-0143
A4.05
PR

October 19, 2000

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

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
Emergency Plan Implementing Procedure

Gentlemen:

In accordance with Appendix E of 10CFR50 and 10CFR50.4(b)(5), Entergy is submitting the enclosed revision to one Waterford 3 Emergency Plan Implementing Procedure. This revision was reviewed in accordance with 10CFR50.54(q) requirements and was determined not to decrease the effectiveness of the emergency plan.

This letter does not contain any commitments.

Included in this submittal is the revision for the following procedure:

1. EP-002-090 (Revision 7), Core Damage Assessment. The changes include adding the capability to calculate core damage using a computer, updating attachments and reformatting the procedure.

A045

Should you have any questions concerning this procedure, please contact J.J. Lewis, Emergency Planning Manager, at (504) 739-6185.

Very truly yours,

E.P. Perkins, Jr.
Director,
Nuclear Safety Assurance

EPP/DCM/rtk
Enclosure: EP-002-090, Revision 7

cc: (w/Enclosure)
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ENCLOSURE 1 to
W3F1-2000-0143

EP-002-090, REVISION 7
(CORE DAMAGE ASSESSMENT)

REQUEST/APPROVAL PAGE

SAFETY RELATED

Required Review Level (check one)
- ☐ PORC
- ☒ QUALIFIED REVIEWER

PROCEDURE NUMBER: EP-002-090 REVISION: 7 CHANGE: 0

TITLE: Core Damage Assessment

EFFECTIVE DATE/MILESTONE: ~~N/A~~ 11/16/00 (N/A If Same as Approval Date)

PROCEDURE OWNER: Emergency Planning (Position Title)

PREPARER (Print Name / Initial): M.W. VanDerHorst / DATE: 09/28/00

ACTION:
- ☐ New Procedure
- ☐ Deletion
- ☒ Revision
- ☐ Change EC? ☐ N/A (Applicable LI-101 Step Numbers)
- ☐ Deviation Expiration Date/Milestone:
- ☐ Temporary Procedure Applicable Conditions:

DESCRIPTION AND JUSTIFICATION OF CHANGE:
Reformatted and designated procedure as "Safety Related" in accordance with W2.109. Added "Core Damage Assessment Using EP-002-090 Computer Code " as Attachment 7.5. The following attachments were revised with Human Factoring incorporated: Attachment 7.4, Core Damage Assessment Data Sheet, Attachment 7.6, Core Damage Assessment Using Core Exit Thermocouples, Attachment 7.7, Core Damage Assessment Using Containment Dose Rates, Attachment 7.8, Power Correction Factor, Attachment 7.9, Core Damage Assessment Using Group 1 Isotopes, Attachment 7.10, Core Damage Assessment Using Group 2 Isotopes, and Attachment 7.11, Core Damage Assessment Using Radioisotopic Analysis. Revision bars are not shown, in accordance with W2.109 guidance, because this is a significant revision. Attachment 7.6 was also revised to remove guidance to extrapolate values between graph lines and provide instruction to stay on the appropriate RCS pressure line in accordance with Combustion Engineering procedure guidance.

☐ Request/Approval Page Continuation Sheet(s) attached.

EC SUPERVISOR		APPROVAL:	N/A	DATE:
50.59 REVIEWER	Required? ☒	REVIEW:		DATE: 10-4-00
50.54 REVIEWER	Required? ☒	REVIEW:		DATE: 10/9/00
TECHNICAL REVIEWER		REVIEW:		DATE: 10-4-00

Change Notice (CN)? ☐

CHANGE NOTICE (CN) SUPERVISOR	APPROVAL:	N/A	DATE:
CHANGE NOTICE (CN) ON-SHIFT SS/CRS	APPROVAL:	N/A	DATE:

Final Approval Due By:

QUALIFIED REVIEWER	Required? ☒	REVIEW:		DATE: 10/9/00
GROUP/DEPT. HEAD	REVIEW ☐ or	APPROVAL ☒		DATE: 10-9-00
GM, PLANT OPERATIONS	REVIEW ☐ or	APPROVAL ☐	N/A	DATE:
VICE PRESIDENT, OPERATIONS		APPROVAL:	N/A	DATE:

CONTROLLED

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12,13	Revision 4

1.0 PURPOSE

1.1 The purpose of this procedure is to estimate the type and degree of reactor core damage which may exist in the plant, using a combination of the following:

1.1.1 Radioisotopic analysis of various samples obtained from the Post Accident Sampling System (PASS)

1.1.2 Core Exit Thermocouple (CET) Temperatures

1.1.3 Containment Dose Rates

2.0 REFERENCES

2.1 CE-NPSD-241, Development of the Comprehensive Procedure Guideline for Core Damage Assessment, CEOG Task 467, July 1983, RAN 459000340

2.2 NUREG-0737, Item II.B.3

2.3 Regulatory Guide 1.97

2.4 EP-002-050, Offsite Dose Assessment (Manual)

2.5 CE-003-400, Operation of the Plasma Emission Spectrophotometer

2.6 CE-003-402, Operation of the Spectrospan 7 DCP Spectrometer

2.7 CE-003-900, Operation of the Post Accident Sampling System

2.8 CE-003-901, Post Accident Sampling of Containment Atmosphere

2.9 CE-003-903, Post Accident Gamma Spectroscopy Analysis

2.10 CE-003-906, Post Accident Gaseous Effluent Sampling

3.0 RESPONSIBILITIES

3.1 The Technical Support Center (TSC) Supervisor is responsible for the implementation of this procedure under the direction of the Emergency Coordinator.

3.2 The TSC Nuclear Engineer is responsible for performing the calculations outlined in this procedure.

3.3 The TSC Chemistry Engineer is responsible for initiating Post Accident Sampling and Analysis and for providing radiochemistry isotopic analysis data to the TSC Nuclear Engineer.

4.0 INITIATING CONDITIONS

4.1 This procedure is initiated upon any of the following conditions:

4.1.1 At the direction of the Emergency Coordinator.

4.1.2 Declaration of any of the following emergency conditions:

4.1.2.1 Alert

4.1.2.2 Site Area Emergency

4.1.2.3 General Emergency

4.1.3 An event has occurred that has placed the reactor and/or its auxiliary systems:

4.1.3.1 In a degraded condition

4.1.3.2 In a situation with the potential for degradation.

5.0 PROCEDURE

5.1 Preparation for Calculations

5.1.1 Identify appropriate sample points (See Attachment 7.3).

5.1.2 Request the samples through the TSC Chemistry Engineer.

5.1.3 As the samples are being taken, complete Attachment 7.4.

5.1.4 Commence trending of CET Temperatures and RCS Pressure. The maximum CET Temperatures and RCS Pressure are required for step 5.2.1.1, and Attachment 7.6.

5.1.5 As soon as practical, as data becomes available, perform Calculations in accordance with Section 5.2.

5.2 Core Damage Assessment Calculation Methods

5.2.1 Consider the following calculation methods:

5.2.1.1 To perform Core Exit Thermocouple (CET) calculation, CET Temperature must be ≥ 1200°F AND the RCS Pressure at which it occurs must be ≤ 1650 PSI.

a. Use the EP-002-090 computer program (See Attachment 7.5).

b. If the EP-002-090 computer program is unavailable, then use Attachment 7.6.

5.2.1.2 To perform Containment Dose Rate core damage assessment calculations, use the EP-002-090 computer program Attachment 7.5.

5.2.1.2.1 If the EP-002-090 computer program is unavailable, then use Attachment 7.7.

5.2.1.3 To perform Radioisotopic Core Damage Assessment Calculations, use the EP-002-090 computer program Attachment 7.5.

5.2.1.3.1 If the EP-002-090 computer code program is unavailable then perform the following:

a. If steady state power (± 10% Full Power) did not occur during the previous 4 days, then determine Power Correction Factor for each of the radioisotopes included in the samples in Attachment 7.4 using the 30-day power history. (Use Attachment 7.8)

b. If steady state power (± 10% Full Power) existed for the 30 day period previous to the emergency, then perform core damage assessment based on isotopic analysis of GROUP 1 isotopes. (Use Attachment 7.9)

c. If the plant has not been at steady state power (± 10% Full Power) for 30 days, but has been at steady state power (±10% Full Power) for the 4 day period previous to the emergency, then perform core damage assessment based on isotopic analysis of GROUP 2 isotopes. (Use Attachment 7.10)

d. If the plant has not been at steady state power (± 10% Full Power) conditions, then perform core damage assessment based on 30 day power history using radioisotopic analysis. (See Attachment 7.11)

6.0 FINAL CONDITIONS

6.1 Core damage assessment is being performed with the results being forwarded to the TSC Lead Engineer.

7.0 ATTACHMENTS

7.1 Characteristics of NRC Categories of Fuel Damage
7.2 Target Isotope Data
7.3 Sample Locations Appropriate for Core Damage Assessment
7.4 Core Damage Assessment Data Sheet
7.5 Core Damage Assessment Using the EP-002-090 Computer Code
7.6 Core Damage Assessment Using Core Exit Thermocouples
7.7 Core Damage Assessment Using Containment Dose Rates
7.8 Power Correction Factor
7.9 Core Damage Assessment Using Group 1 Isotopes
7.10 Core Damage Assessment Using Group 2 Isotopes
7.11 Core Damage Assessment Using Radioisotopic Analysis

8.0 RECORDS

8.1 The following records are generated as a result of this procedure:

- Attachment 7.4, Core Damage Assessment Data Sheet
- Attachment 7.5, Core Damage Assessment Using the EP-002-090 Computer Code
- Attachment 7.6, Core Damage Assessment Using Core Exit Thermocouples
- Attachment 7.7, Core Damage Using Containment Dose Rates
- Attachment 7.8, Power Correction Factor
- Attachment 7.9, Core Damage Assessment Using Group 1 Isotopes
- Attachment 7.10, Core Damage Assessment Using Group 2 Isotopes
- Attachment 7.11,Core Damage Assessment Using Radioisotopic Analysis

CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE

RADIOISOTOPIC ANALYSIS

TABLE 1

NRC Category of Fuel Damage	Mechanism of Release	Source of Release	Characteristic Isotope	Release of Characteristic Isotope Expressed as a Percent of Source Inventory
1.0 No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	I-131, Cs-137	Less than 1
2.0 Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release			Less than 10
3.0 Intermediate Cladding Failure			Xe-131m, Xe-133 I-131, I-133	10 to 50
4.0 Major Cladding Failure				Greater than 50
5.0 Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Cs-134, Rb-88 Te-129, Te-132	Less than 10
6.0 Intermediate Fuel Pellet Overheating				10 to 50
7.0 Major Fuel Pellet Overheating	Diffusional Release From UO_2 Grains			Greater than 50
8.0 Initial Fuel Pellet Melt	Escape from Molten Fuel		Ba-140, La-140 La-142	Less than 10
9.0 Intermediate Fuel Pellet				10 to 50
10.0 Major Fuel Pellet Melt				Greater than 50

CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE
CORE EXIT THERMOCOUPLE ANALYSIS

TABLE 2

NRC Category of Fuel Damage	Temperature Range (°F)	Mechanism of Damage	Characteristic Measurement	Measurement Range	Percent of Damaged Rods
1.0 No Fuel Damage	750	None	—	—	Less than 1
2.0 Initial Cladding Failure	1200-1800	Rupture Due to Gas Gap Over-pressurization	Maximum Core Exit Thermocouple Temperature	<1550°F*	Less than 10
3.0 Intermediate Cladding Failure				<1700°F*	10 to 50
4.0 Major Cladding Failure				~ <2300°F ~ <2% oxidation	Greater than 50

* Depends upon reactor pressure. Values given are for pressure ≤ 1200 psia.

CHARACTERISTICS OF NRC CATEGORIES OF FUEL DAMAGE
DOSE RATE ANALYSIS

TABLE 3

NRC Category of Fuel Damage	Mechanism of Release From Core	Source of Release	Percent of Source Inventory Release To Containment	Distribution of Fission Products in Containment
1.0 No Fuel Damage	Halogen Spiking Tramp Uranium	Gas Gap	Less than 1	Airborne
2.0 Initial Cladding Failure	Clad Burst and Gas Gap Diffusion Release		Less than 10	
3.0 Intermediate Cladding Failure			10 to 50	
4.0 Major Cladding Failure			Greater than 50	
5.0 Initial Fuel Pellet Overheating	Grain Boundary Diffusion	Fuel Pellet	Less than 10	Airborne: 100% Noble Gas 25% Halogens
6.0 Intermediate Fuel Pellet Overheating			10 to 50	
7.0 Major Fuel Pellet Overheating	Diffusional Release From UO2 Grains		Greater than 50	Plated Out: 25% Halogens 1% Solids

TARGET ISOTOPE DATA

Table 1

DECAY CONSTANTS OF

TARGET ISOTOPES

Isotope	Decay Constant (λ), 1/hr
Kr-87	5.472 E-01
Xe-131m	2.423 E-03
Xe-133	5.501 E-03
I-131	3.592 E-03
I-132	3.027 E-01
I-133	3.332 E-02
I-135	1.053 E-01
Cs-134	3.830 E-05
Rb-88	2.350 E+00
Te-129	5.941 E-01
Te-132	8.887 E-03
Ba-140	2.258 E-03
La-140	1.723 E-02
La-142	4.496 E-01

TARGET ISOTOPE DATA

Table 2

EQUILIBRIUM SOURCE INVENTORIES OF TARGET ISOTOPES

Isotope	Equilibrium Source Inventories, (Ci)
Gas Gap Inventory	
Kr-87	9.5 E+00
Xe-131m	6.6 E+04
Xe-133	1.8 E+07
I-131	9.0 E+06
I-132	9.9 E+03
I-133	8.9 E+06
I-135	1.6 E+06
Fuel Pellet Inventory	
Kr-87	4.7 E+07
Xe-131m	7.0 E+05
Xe-133	2.0 E+08
I-131	9.9 E+07
I-132	1.4 E+08
I-133	2.0 E+08
I-135	1.9 E+08
Cs-134	1.8 E+07
Rb-88	6.8 E+07
Te-129	3.1 E+07
Te-132	1.4 E+08
Ba-140	1.7 E+08
La-140	1.8 E+08
La-142	2.2 E+08

SAMPLE LOCATIONS APPROPRIATE FOR CORE DAMAGE ASSESSMENT

KNOWN ACCIDENT SCENARIO	RCS HOT LEG	RCS PRESSURIZER	CONTAINMENT SUMP	CONTAINMENT ATMOSPHERE	SHUTDOWN COOLING SYSTEM	STEAM GENERATOR SECONDARY
Small Break LOCA Reactor Power > 1%	YES	YES	___	YES	YES	___
Small Break LOCA Reactor Power < 1%	YES	YES	___	___	YES	
Small Steam Line Break	YES	YES	___	___	___	___
Large Break LOCA Reactor Power > 1%	YES	___	YES	YES	YES	___
Large Break LOCA Reactor Power < 1%	___	___	YES	YES	YES	___
Large Steam Line Break	YES	___	___	YES	___	___
Steam Generator Tube	YES	___	___	YES	___	YES

SAMPLE LOCATIONS APPROPRIATE FOR CORE DAMAGE ASSESSMENT

UNKNOWN ACCIDENT SCENARIO	RCS HOT LEG	RCS PRESSURIZER	CONTAINMENT SUMP	CONTAINMENT ATMOSPHERE	SHUTDOWN COOLING SYSTEM	STEAM GENERATOR SECONDARY
SIS Actuated	YES	YES	___	___	YES	___
Alarm on Containment Bldg. Radiation Monitor	___	___	YES	YES	___	___
Alarm on CVCS Letdown Radiation Monitor	YES	YES	___	___	___	___
Alarm on Containment Bldg. Sump Level	___	___	YES	YES	___	___

CORE DAMAGE ASSESSMENT DATA SHEET

Date_____ Time_____

Date/Time of Reactor Shutdown: _____

RCS DATA (Data from TSC1 mimic)

Temperature: T cold (avg.): _____°F T hot (avg.): _____°F
Pressurizer: Pressure: _____psia Level:_____%
Maximum CET Temperature: _____°F at an RCS Pressure of:_____psia
CET Saturation Margin: _____°F subcooled

CONTAINMENT DATA (Data from CNTMT2 mimic)

Pressure: _____psia Temperature: _____°F
Containment Sump Level:_____ft.

CONTAINMENT DATA (Data from CNTMT1 mimic)

Dose Rate:
High Range Monitor (CAS561 CA A): _____R/hr Time: _____
High Range Monitor (CAS561 CA B): _____R/hr Time: _____

POWER HISTORY DATA

Obtain the data required below for the past 30 days, or for 10 time periods within the past 30 days, whichever is less. (Period 1 is the oldest in time and Period 10 is the newest in time.)

	PERIODS									
	1	2	3	4	5	6	7	8	9	10
Average Power (in % Full Power), P										
Duration of Time period i (hrs), t_i										
Time since end of period i (hrs), t_a										

30 day Average: _____ Constant? (±10% Full Power) Y__ N __

Last 4 days Average: _____ Constant? (±10% Full Power) Y__ N __

CORE DAMAGE ASSESSMENT DATA SHEET

SAMPLE RESULTS

NOTE

Multiple copies of this sheet may be required depending upon the number and location of the samples which are to be collected.

Sample No.: _____ Sample Location: _____

Time of Analysis: _____

For Containment Atmosphere Samples Only:

Sample Temperature: _____°F Sample Pressure: _____psia

(Use 250°F and 14.7 psia, due to heat tracing, if actual data is not available)

Isotope	Fuel History Group	Sample Activity (uCi/cc)
Kr-87	2	_____
Xe-131m	1	_____
Xe-133	1	_____
I-131	1	_____
I-132	2	_____
I-133	2	_____
I-135	2	_____
Cs-134	1	_____
Rb-88	2	_____
Te-129	2	_____
Te-132	1	_____
Ba-140	1	_____
La-140	1	_____
La-142	2	_____

NOTE 1

Acceptable format for numerical input is relatively flexible. The only restriction is that no embedded blanks may be used. The Following are all acceptable:

1.0E+2	1E+2
1+2	100

1. At a PC with EP-002-090 installed, start the program by double clicking the icon.

CAUTION

In an accident situation with high temperature (> 250°F) in the containment, the high range radiation monitor may respond erroneously or show a negative response on the lower radiation decades (1 R/hr to 500 R/hr). The higher decades are not affected. If such a situation is present and it is necessary to perform core Damage assessment using containment dose rates, then request that the Health Physics Coordinator/Radiological Assessment Coordinator Calculate the containment dose rates in accordance with Attachment 7.7 of EP-002-050. Use the calculated value for the monitor reading.

2. Select the core damage assessment option from the drop down menu and if the printout is unavailable, then pages 2 and 3 of this attachment may be used to record the results of this program.

CORE DAMAGE ASSESSMENT USING EP-002-090 COMPUTER CODE

Results of Radioisotopic Analysis

For Group 1 Steady State or for Transient cases:

_____Percent of the measurements indicate the gas gap as the release source.

_____Percent of the measurements indicate the fuel pellets as the release source.

Percentages of equilibrium sources:

GAS GAP INVENTORY

Isotope	Percentage
Kr-87	_____
Xe-131m	_____
Xe-133	_____
I-131	_____
I-132	_____
I-133	_____
I-135	_____

FUEL PELLET INVENTORY

Isotope	Percentage
Kr-87	_____
Xe-131m	_____
Xe-133	_____
I-131	_____
I-132	_____
I-133	_____
I-135	_____
Cs-134	_____
Rb-88	_____
Te-129	_____
Te-132	_____
Ba-140	_____
La-140	_____
La-142	_____

Categories of core damage:

Indicated By this Procedure	Mechanism	Source

*G.G. = Gas Gap F.P. = Fuel Pellet

CORE DAMAGE ASSESSMENT USING EP-002-090 COMPUTER CODE

Results of Core Exit Thermocouple Analysis

_____Percent of fuel rods may have ruptured.

Categories of core damage indicated:

_____

_____

_____

Results of Containment Dose Rate Analysis

Categories of core damage indicated by this procedure	Mechanism	Source *	Percentage of equilibrium inventory available for release	Distribution of fission products
* G.G.= Gas Gap	F.P.= Fuel Pellet			

CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES

NOTE

This method of estimating core damage applies to scenarios where core uncover occurs slowly as a result of coolant boil-off. This method may yield a very low estimate of core damage during rapid uncover scenarios. In general, for core uncover at pressures below approximately 1200 psia, there is high confidence that at least the estimated percentage of fuel rods have sustained damage.

1. Using the Core Exit Thermocouple (CET) data recorded on Attachment 7.4, estimate the Percent of Fuel Rods with Ruptured Clad from Figure 1 of this attachment. When using figure 1,
   Use Curve 1 for RCS Pressure $\leq$ 100 psia.
   Use Curve 2 for 100 psia $<$ RCS Pressure $\leq$ 1200 psia.
   Use Curve 3 for 1200 psia $<$ RCS Pressure $\leq$ 1650 psia.

   Percent of Fuel Rods with Ruptured Clad:_____%

2. Using the value determined above, determine the NRC Category of Fuel Damage (categories 1 through 4) from Table 2 of Attachment 7.1, with respect to percent of damaged rods.

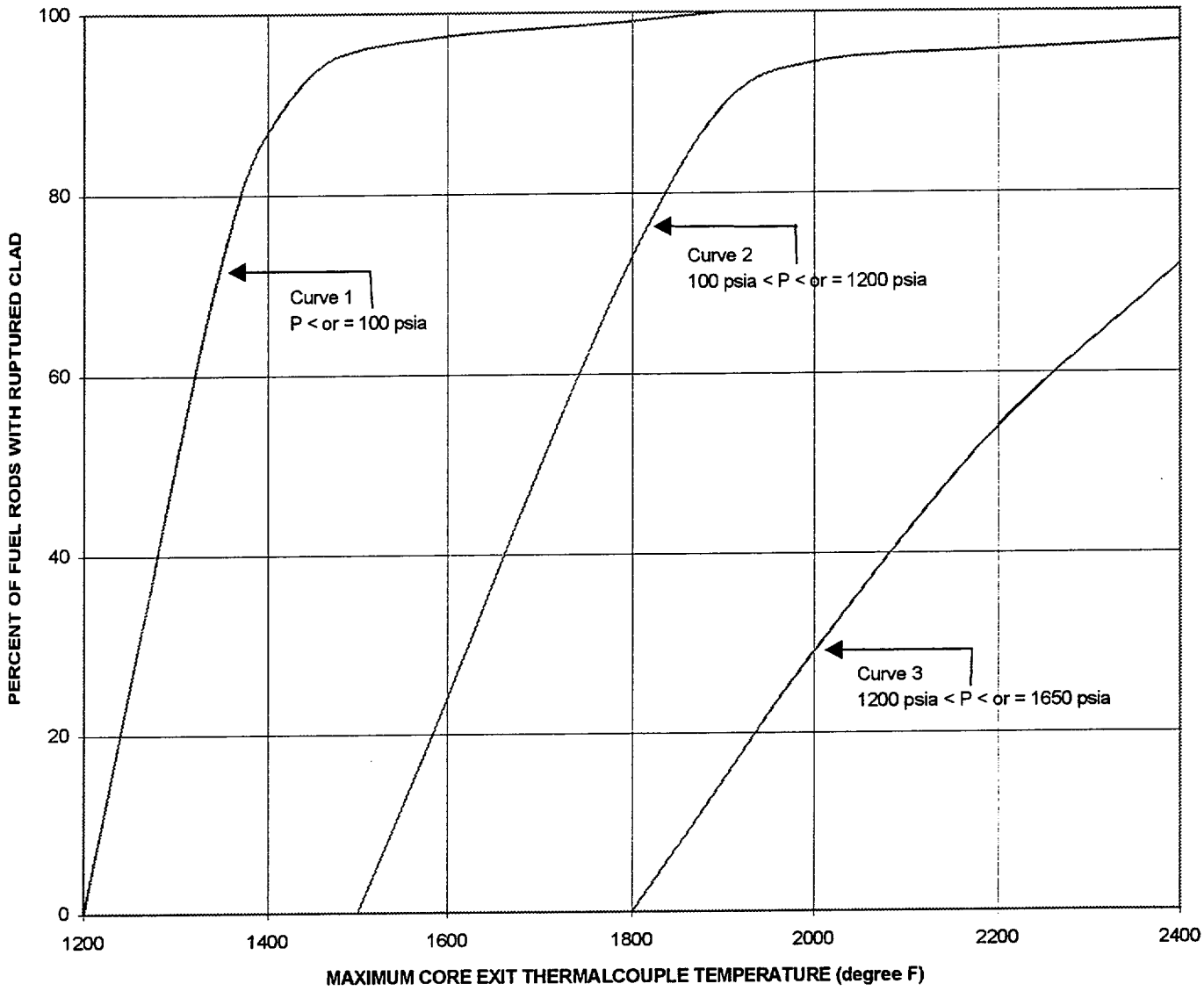
   NRC Category of Fuel Damage:_____

CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES

PERCENT OF FUEL RODS WITH RUPTURED CLAD Vs MAXMIUM

CORE EXIT THERMOCOUPLES TEMPERATURE

FIGURE 1



CORE DAMAGE ASSESSMENT USING CONTAINMENT DOSE RATES

NOTE

If the reactor has produced power for less than 30 days in this cycle, then the estimate of core damage produced by this method may under predict actual conditions.

NOTE

In an accident situation with high temperatures (>250°F) in the containment, the high range radiation monitor may respond erroneously or show a negative response on the lower radiation decades (1R/hr to 500 R/hr). The higher decades are not affected. If such a situation is present and it is necessary to perform core damage assessment using containment dose rates, then request that the Health Physics Coordinator/Radiological Assessment Coordinator calculate the containment dose rates in accordance with Attachment 7.7 of Reference 2.4. Use this calculated value for the monitor reading.

1. Representative Plant Power

   1.1 If the Power History Data of Attachment 7.4 indicates that plant power was constant for 30 days or more, then the representative power is the 30-day average power.

   1.2 If power was not constant, then calculate a representative plant power by completing Table 1 of this attachment using data recorded on Attachment 7.4.

2. Calculate the Dose Rate Correction Factor (DCF) as follows:

$$DCF = \frac{100}{\text{Representative Power}} = \frac{100}{(_____)}$$

DCF = _____

3. Calculate the Equilibrium Containment Dose Rate for each detector for which data was recorded on Attachment 7.4 as follows:

Equilibrium = Measured Dose x DCF (step 2)
Dose Rate Rate (Att. 7.4)

Equilibrium = _____ x _____
Dose Rate

Equilibrium = _____
Dose Rate

NOTE
This method gives an estimate of the highest category of core damage. For example, if this method predicts Intermediate Fuel Overheating, then Concurrent Major Cladding Failure should be anticipated.

4. The estimate of the category of core damage is read from Figure 1 of this attachment using the Equilibrium Dose Rate calculated above and the time from reactor trip/shutdown recorded on Attachment 7.4. The curves of Figure 1 define the boundaries between the failure and over heat regions.

5. Once the category of core damage has been determined from Figure 1, the characteristics of this damage category can be found in Table 3 of Attachment 7.1

NRC category of Damage:_____

POWER CORRECTION FACTOR

CALCULATION SHEET

Radioisotope:_____ Sample No.:_____

$$PCF = \sum_{i=1}^{n} PCF_i$$

PCF TABLE

Table 2

I	PCFi
1	
2	
3	
4	
5	
6	
7	
8	
9	
10	

PCF = _____

Transfer the value of PCF for this isotope into the appropriate block of Table 1, SUMMARY TABLE, of this attachment.

NOTE

Perform Sections 1 through 3 for each of the samples, which were collected.

1. RCS Samples

1.1 Complete the Sample Data block and the Sample Activity column of Table 1 using data recorded on Attachment 7.4 and the results of the RCS sample.

1.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 1.

1.3 Determine the Density Correction Factor (DCF) from Figure 1, of this attachment, using the RCS temperature at the time the sample was collected (from Attachment 7.4). Enter the DCF into the "DCF" column of Table 1.

1.4 Calculate the Decay Correction Factor on Table 1, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 1.

1.5 Calculate the Corrected Activity in the RCS by multiplying columns 2 through 5 of Table 1 together. Enter the result into the "Corrected Activity in the RCS" column of Table 1.

2. Containment Sump Samples

2.1 Complete the Sample Data block and the Sample Activity column of Table 2 using data recorded on Attachment 7.4 and the results of the Containment Sump Sample.

2.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 2 using data from Attachment 7.4.

2.3 Determine the volume of liquid in the containment sump from Figure 2 using the Sump Level recorded on Table 2. Enter the result into the "Sump Volume" column of Table 2.

2.4 Calculate the Decay Correction Factor on Table 2, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 2.

2.5 Calculate the Corrected Activity in the Containment Sump by multiplying columns 2 through 4 of Table 2 together. Enter the result into the "Corrected Activity in the Containment Sump" column of Table 2.

3. Containment Atmosphere Samples

3.1 Complete the Sample Data block and the Sample Activity column of Table 3 using data recorded on Attachment 7.4 and the results of the Containment Atmosphere Sample.

3.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 3.

3.3 Determine the Volume Correction Factor (VCF) on Table 3, using the Containment pressure and temperature at the time the sample was collected. Enter the VCF into the "VCF" column of Table 3.

3.4 Calculate the Decay Correction Factor on Table 3 for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 3.

3.5 Calculate the Corrected Activity in the Containment Atmosphere by multiplying columns 2 through 4 of Table 3 together. Enter the result into the "Correction" column of Table 3.

4. Transfer the corrected activities of each sample into the appropriate columns of Table 4.

5. Calculate the TOTAL ACTIVITY by adding columns 2 through 4 and then dividing by 1.0 E+6 to convert from uCi to Ci.

6. Calculate the Power Correction Factor (PCF) for Table 5 as follows:

$$PCF = \frac{\text{30-day average power}}{100}$$

$$PCF = \frac{\text{(Attachment 7.4)}}{100}$$

$$PCF = \frac{_____}{100}$$

PCF = _____

7. Complete table 5 using values from the following sources:

Total Activity (Ci)	-	Table 4, Column 6,
Power Correction Factor	-	Step 6, above,
Equilibrium Activity	-	Table 2 of Attachment 7.2.

8. The estimate of Core Damage can now be made by taking the Percent of Equilibrium Inventory Released (column 6) of Table 5 and comparing these numbers with the Release of Characteristic Isotope Expressed as a Percent of Source Inventory (column 5) of Table 1 of Attachment 7.1.

CORE DAMAGE ASSESSMENT USING GROUP 1 ISOTOPES

Table 1

RCS SAMPLES

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____

t_S = _____ hrs.

Average RCS Temperature: _____°F Density Correction Factor: _____

(DCF, Figure 1)

RCS Pressure: _____ psia

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Density Correction DCF	X	(4) Decay Correction	X	(5) RCS Volume (cc)	=	(6) Corrected Activity in RCS (uCi)
Xe-131m		X		X		X		=	
Xe-133		X		X		X		=	
I-131		X		X		X		=	
Cs-134		X	_____	X		X	3.189E+8	=	
Te-132		X		X		X		=	
Ba-140		X		X		X		=	
La-140		X		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

CORE DAMAGE ASSESSMENT USING GROUP 1 ISOTOPES

Table 2

CONTAINMENT SUMP SAMPLES

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____

t_S = _____ hrs.

Sump Level: _____ ft Sump Volume (Fig. 2): _____cc

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Sump Volume (cc)	X	(4) Decay Correction	=	(5) Corrected Activity In Containment Sump, (uCi)
Xe-131m		X	_____	X		=	
Xe-133		X		X		=	
I-131		X		X		=	
Cs-134		X		X		=	
Te-132		X		X		=	
Ba-140		X		X		=	
La-140		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

CORE DAMAGE ASSESSMENT USING GROUP 1 ISOTOPES

Table 3

CONTAINMENT ATMOSPHERE SAMPLES

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____ t_S = _____hrs.

Containment Pressure: _____psia Containment Temperature: _____°F

Sample Pressure: _____psia Sample Temperature: _____°F

(Use 14.7 psia and 250°F due to Heat Tracing, if actual sample data is unavailable)

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Volume Correction VCF, (cc)	X	(4) Decay Correction	=	(5) Corrected Activity In Containment Atmosphere, (uCi)
Xe-131m		X	_____	X		=	
Xe-133		X		X		=	
I-131		X		X		=	
Cs-134		X		X		=	
Te-132		X		X		=	
Ba-140		X		X		=	
La-140		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

$$VCF = 7.589 \times 10^{10} \text{ (cc)} \times \frac{P_{cont}\text{(psia)}}{P_{samp}\text{ (psia)}} \times \frac{(T_{samp} + 460)^{\circ}F(^{abs})}{(T_{cont} + 460)^{\circ}F(^{abs})}$$

CORE DAMAGE ASSESSMENT USING GROUP 1 ISOTOPES

Table 4

TOTAL QUANTITY OF RADIONUCLIDES IN SAMPLES

(1) Isotope	(2) Corrected Activity In RCS (uCi)	+	(3) Corrected Activity in Containment Sump, (uCi)	+	(4) Corrected Activity in Containment Atmosphere (uCi)	÷	(5) uCi-Ci Conversion Constant (uCi/Ci)	=	(6) TOTAL ACTIVITY (Ci)
Xe-131m		+		+		÷		=	
Xe-133		+		+		÷		=	
I-131		+		+		÷		=	
Cs-134		+		+		÷	1.0 E+6	=	
Te-132		+		+		÷		=	
Ba-140		+		+		÷		=	
La-140		+		+		÷		=	

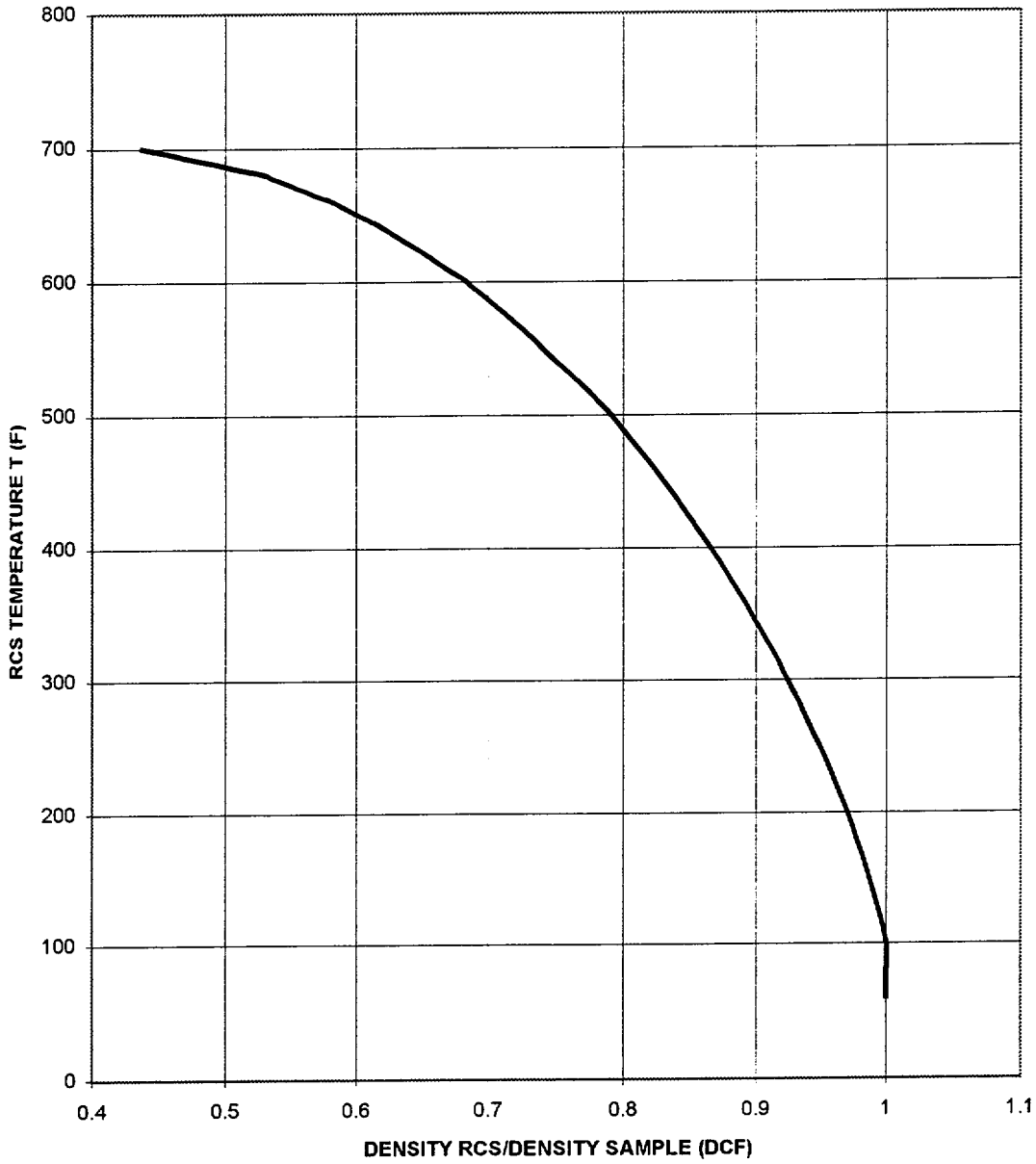
CORE DAMAGE ASSESSMENT USING GROUP 1 ISOTOPES

Table 5

(1) Isotope	(2) Total Activity (Ci)	÷	(3) Power Correction Factor	÷	(4) Equilibrium Inventory	X	(5) 100%	=	(6) Percent of Equilibrium Inventory Released
GAS GAP									
Xe-131m		÷		÷	6.6E+4	X	100%	=	
Xe-133		÷		÷	1.8E+7	X		=	
I-131		÷		÷	9.0E+6	X		=	
FUEL PELLET									
Xe-131m		÷		÷	7.0E+5	X	100%	=	
Xe-133		÷		÷	2.0E+8	X		=	
I-131		÷		÷	9.9E+7	X		=	
Cs-134		÷		÷	1.8E+7	X		=	
Te-132		÷		÷	1.4E+8	X		=	
Ba-140		÷		÷	1.7E+8	X		=	
La-140		÷		÷	1.8E+8	X		=	

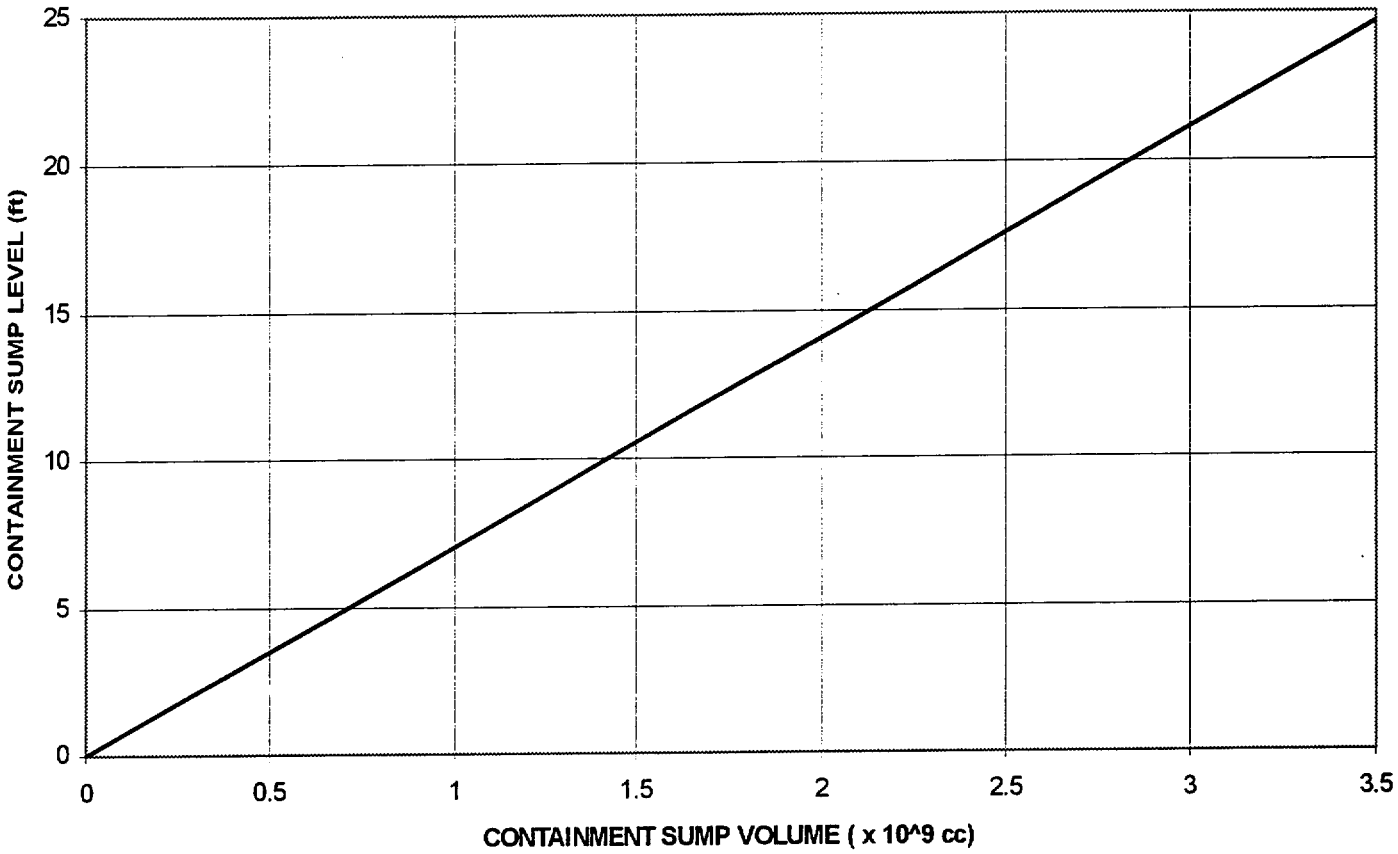
DENSITY CORRECTION FACTOR
FOR
REACTOR COOLANT SYSTEM TEMPERATURE

Figure 1



CONTAINMENT SUMP LEVEL
VS.
CONTAINMENT SUMP VOLUME

Figure 2



NOTE

Perform Sections 1 through 3 for each of the samples, which were collected.

1. RCS Samples

1.1 Complete the Sample Data block and the Sample Activity column of Table 1 using data recorded on Attachment 7.4 and the results of the RCS sample.

1.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 1.

1.3 Determine the Density Correction Factor (DCF) from Figure 1, of this attachment, using the RCS temperature at the time the sample was collected. Enter the DCF into the "DCF" column of Table 1.

1.4 Calculate the Decay Correction Factor on Table 1, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 1.

1.5 Calculate the Corrected Activity in the RCS by multiplying columns 2 through 5 of Table 1 together. Enter the result into the "Corrected Activity in the RCS" column of Table 1.

2. Containment Sump Samples

2.1 Complete the Sample Data block and the Sample Activity column of Table 2 using data recorded on Attachment 7.4 and the results of the Containment Sump Sample.

2.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 2.

2.3 Determine the volume of liquid in the containment sump from Figure 2 using the Sump Level recorded on Table 2. Enter the result into the "Sump Volume" column of Table 2.

2.4 Calculate the Decay Correction Factor Table 2, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 2.

2.5 Calculate the Corrected Activity in the Containment Sump by multiplying columns 2 through 4 of Table 2 together. Enter the result into the "Corrected Activity in the Containment Sump" column of Table 2.

3. Containment Atmosphere Samples

3.1 Complete the Sample Data block and the Sample Activity column of Table 3 using data recorded on Attachment 7.4 and the results of the Containment Atmosphere Sample.

3.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 3 using data recorded on Table 3.

3.3 Determine the Volume Correction Factor (VCF) on Table 3, using the Containment pressure and temperature at the time the sample was collected. Enter the VCF into the "VCF" column of Table 3.

3.4 Calculate the Decay Correction Factor on Table 3, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 3.

3.5 Calculate the Corrected Activity in the Containment Atmosphere by multiplying columns 2 through 4 of Table 3 together. Enter the result into the "Corrected Activity in the Containment Atmosphere" column of Table 3.

4. Transfer the corrected activities of each sample into the appropriate columns of Table 4.

5. Calculate the TOTAL ACTIVITY by adding columns 2 through 4 and then dividing by 1.0 E+6 to convert from uCi to Ci.

6. Calculate the Power Correction Factor (PCF) for Table 5 as follows:

$$PCF = \frac{\text{4-day average power}}{100}$$

$$PCF = \frac{\text{(Attachment 7.4)}___}{100}$$

$$PCF = \frac{_____}{100}$$

PCF = _____

7. Complete Table 5 using values from the following sources:

Total activity (Ci)	-	Table 4, Column 6,
Power Correction Factor	-	above,
Equilibrium Activity	-	Table 2 of Attachment 7.2.

8. The estimate of Core Damage can now be made by taking the Percent of Equilibrium Inventory Released (column 6) of Table 5 and comparing these numbers with the Release of Characteristic Isotope Expressed as a Percent of Source Inventory (column 5) of Table 1 of Attachment 7.1.

CORE DAMAGE ASSESSMENT USING GROUP 2 ISOTOPES

TABLE 1

RCS SAMPLES

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____ =

t_S = _____ hrs.

Average RCS Temperature:_____°F Density Correction Factor:_____

(DCF, Figure 1)

RCS Pressure:_____psia

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Density Correction DCF	X	(4) Decay Correction	X	(5) RCS Volume (cc)	=	(6) Corrected Activity in RCS (uCi)
Kr-87		X		X		X		=	
I-132		X		X		X		=	
I-133		X		X		X		=	
I-135		X	_____	X		X	3.189E+8	=	
Rb-88		X		X		X		=	
Te-129		X		X		X		=	
La-142		X		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

CORE DAMAGE ASSESSMENT USING GROUP 2 ISOTOPES

TABLE 2

Containment Sump Samples

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____=

t_S = _____ hrs.

Sump Level:_____ft Sump Volume (Fig. 2):_____ cc

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Sump Volume (cc)	X	(4) Decay Correction	=	(5) Corrected Activity In Containment Sump, (uCi)
Kr-87		X		X		=	
I-132		X		X		=	
I-133		X		X		=	
I-135		X	_____	X		=	
Rb-88		X		X		=	
Te-129		X		X		=	
La-142		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

CORE DAMAGE ASSESSMENT USING GROUP 2 ISOTOPES

TABLE 3

Containment Atmosphere Samples

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____ =

t_S = _____ hrs.

Containment Pressure:_____psia Sample Pressure:_____psia

Containment Temperature:_____°F Sample Temperature:_____°F

(Use 14.7 psia and 250°F, due to heat tracing, if actual data is not available)

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Volume Correction VCF, (cc)	X	(4) Decay Correction	=	(5) Corrected Activity In Containment Atmosphere, (uCi)
Kr-87		X		X		=	
I-132		X		X		=	
I-133		X		X		=	
I-135		X	_____	X		=	
Rb-88		X		X		=	
Te-129		X		X		=	
La-142		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

$$VCF = 7.589 \times 10^{10} \text{ (cc)} \times \frac{P_{cont}(\text{psia})}{P_{samp}(\text{psia})} \times \frac{(T_{samp} + 460)^{\circ}F(^{abs})}{(T_{cont} + 460)^{\circ}F(^{abs})}$$

CORE DAMAGE ASSESSMENT USING GROUP 2 ISOTOPES

TABLE 4

Total Quantity of Radionuclides in Samples

(1) Isotope	(2) Corrected Activity In RCS (uCi)	+	(3) Corrected Activity in Containment Sump, (uCi)	+	(4) Corrected Activity in Containment Atmosphere (uCi)	÷	(5) uCi-Ci Conversion Constant (uCi/Ci)	=	(6) TOTAL ACTIVITY (Ci)
Kr-87		+		+		÷	1.0 E+6	=	
I-132		+		+		÷		=	
I-133		+		+		÷		=	
I-135		+		+		÷		=	
Rb-88		+		+		÷		=	
Te-129		+		+		÷		=	
La-142		+		+		÷		=	

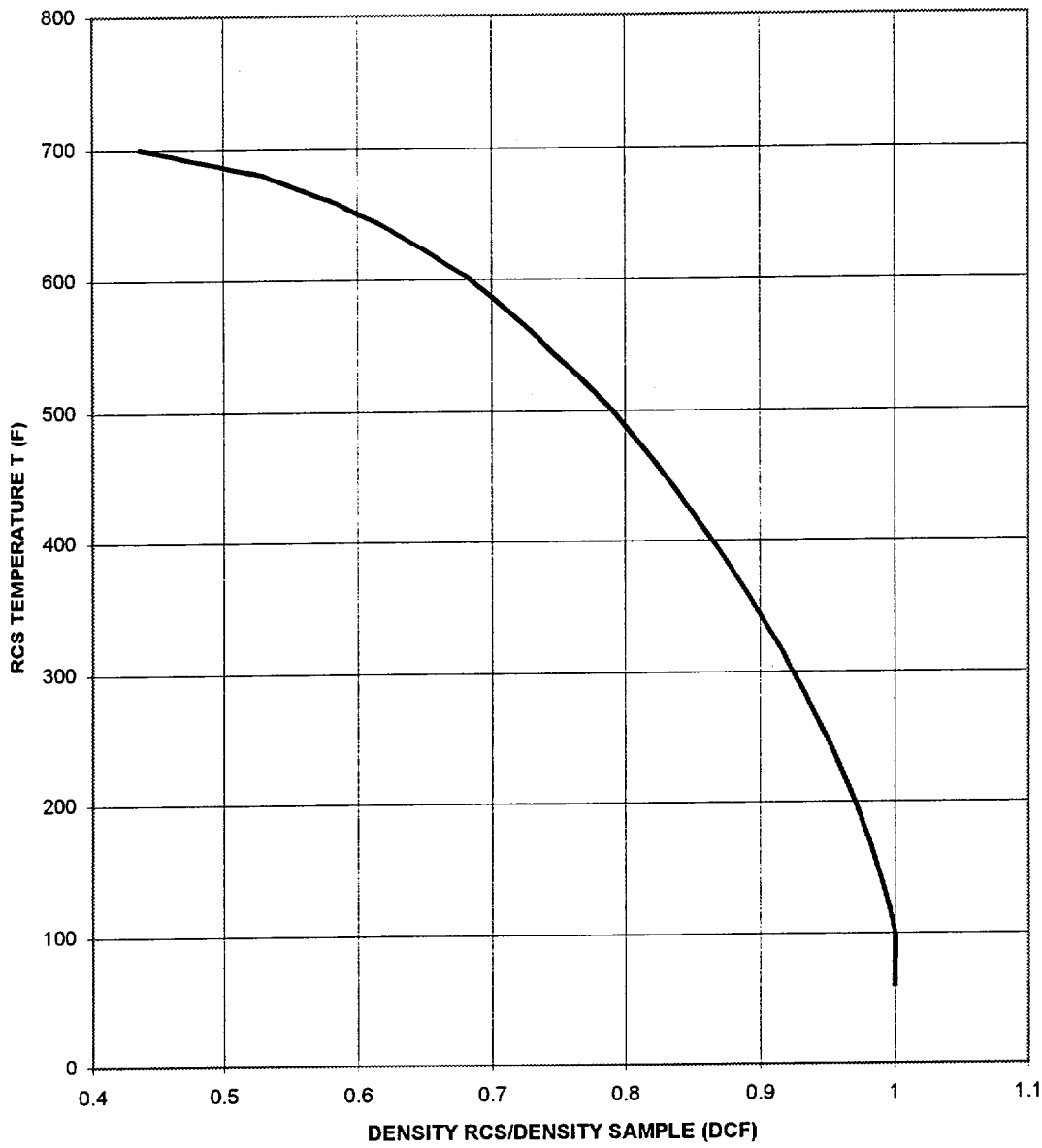
CORE DAMAGE ASSESSMENT USING GROUP 2 ISOTOPES

TABLE 5

(1) Isotope	(2) Total Activity (Ci)	÷	(3) Power Correction Factor	÷	(4) Equilibrium Inventory	X	(5) 100%	=	(6) Percent of Equilibrium Inventory Released
GAS GAP									
Kr-87		÷		÷	9.5E+0	X	100%	=	
I-132		÷		÷	9.9E+3	X		=	
I-133		÷		÷	8.9E+6	X		=	
I-135		÷		÷	1.6E+6	X		=	
FUEL PELLET									
Kr-87		÷		÷	4.7E+7	X	100%	=	
I-132		÷		÷	1.4E+8	X		=	
I-133		÷		÷	2.0E+8	X		=	
I-135		÷		÷	1.9E+8	X		=	
Rb-88		÷		÷	6.8E+7	X		=	
Te-129		÷		÷	3.1E+7	X		=	
La-142		÷		÷	2.2E+8	X		=	

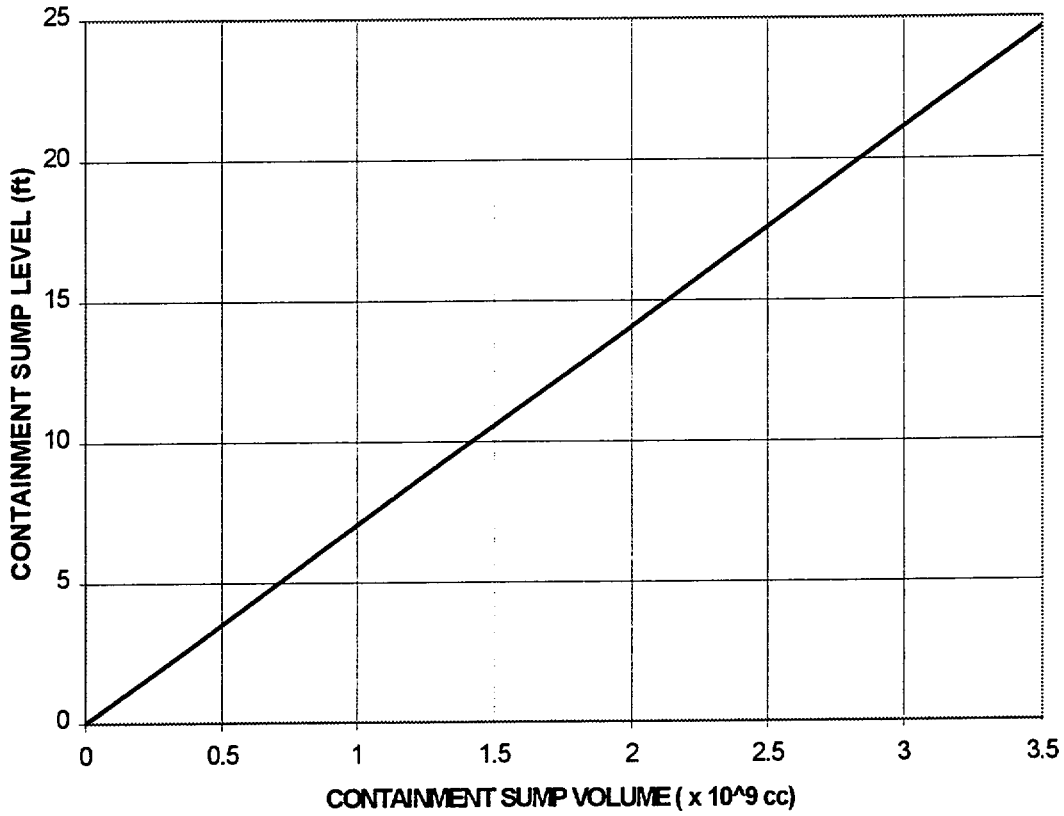
DENSITY CORRECTION FACTOR
FOR
REACTOR COOLANT SYSTEM TEMPERATURE

Figure 1



CONTAINMENT SUMP LEVEL
VS.
CONTAINMENT SUMP VOLUME

Figure 2



NOTE

Perform Sections 1 through 3 for each of the samples, which were collected.

1. RCS Samples

1.1 Complete the Sample Data block and the Sample Activity column of Table 1 using data recorded on Attachment 7.4 and the results of the RCS sample.

1.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 1.

1.3 Determine the Density Correction Factor (DCF) from Figure 1, of this attachment, using the RCS temperature at the time the sample was collected (from Attachment 7.4). Enter the DCF into the "DCF" column of Table 1.

1.4 Calculate the Decay Correction Factor on Table 1, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 1.

1.5 Calculate the Corrected Activity in the RCS by multiplying columns 2 through 5 of Table 1 together. Enter the result in the "Corrected Activity in the RCS" column of Table 1.

2. Containment Sump Samples

2.1 Complete the Sample Data block and the Sample Activity column of Table 2 using data recorded on Attachment 7.4 and the results of the Containment Sump Sample.

2.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 2.

2.3 Determine the volume of liquid in the containment sump from Figure 2 using the Sump Level recorded on Table 2. Enter the result into the "Sump Volume" column of Table 2.

2.4 Calculate the Decay Correction Factor on Table 2, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 2.

2.5 Calculate the Corrected Activity in the Containment Sump by multiplying columns 2 through 4 of Table 2 together. Enter the result into the "Corrected Activity in the Containment Sump" column of Table 2.

3. Containment Atmosphere Samples

3.1 Complete the Sample Data block and the Sample Activity column of Table 3 using data recorded on Attachment 7.4 and the results of the containment Atmosphere Sample.

3.2 Calculate the time between reactor shutdown and the time of sample analysis (t_S) on Table 3.

3.3 Determine the Volume Correction Factor (VCF) on Table 3, using the Containment pressure and temperature at the time the samples was collected. Enter the VCF into the "VCF" column of Table 3.

3.4 Calculate the Decay Correction Factor on Table 3, for each target isotope present in the sample. Enter the results into the "Decay Correction" column of Table 3.

3.5 Calculate the Corrected Activity in the Containment Atmosphere by multiplying columns 2 through 4 of Table 3 together. Enter the result into the "Corrected Activity in the Containment Atmosphere" Column of Table 3.

4. Transfer the corrected activities of each sample into the appropriate columns of Table 4.

5. Calculate the TOTAL ACTIVITY by adding columns 2 through 4 and then dividing by 1.0 E+6 to convert from uCi to Ci.

> **NOTE**
>
> It is not anticipated that the isotope ratios calculated in table 5 would exactly equal the values in columns 4 and 5 of that table. It is appropriate to select, as the probable source, that ratio which is closest to the value obtained in column 3 of Table 5.

6. Complete table 5 using Corrected Activity numbers from Table 1 to identify the probable source of release of the radionuclides from the fuel.

> **NOTE**
>
> For the radionuclides listed in Table 5 of this attachment, complete Table 6 based upon the results of the calculations of Table 5 (i.e., Complete Table 6 for either the Gas Gap or Fuel Pellet or Both as indicated by the Probable Source column of Table 5).

7. Complete Table 6 using values from the following sources:

Total Activity (Ci)	-	Table 4, Column 6,
Power Correction Factor	-	Attachment 7.8,
Equilibrium Activity	-	Table 2 of Attachment 7.2.

8. The estimate of Core Damage can now be made by taking the Percent of Equilibrium Inventory Released (column 6) of Table 6 and comparing these numbers with the Release of Characteristic Isotope Expressed as a Percent of Source Inventory (column 5) of Table 1 of Attachment 7.1.

CORE DAMAGE ASSESSMENT USING RADIOISOTOPIC ANALYSIS

TABLE 1

RCS SAMPLES

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____ =

t_S = _____ hrs.

Average RCS Temperature:_____°F Density Correction Factor:_____

(DCF, Figure 1)

RCS Pressure:_____psia

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Density Correction DCF	X	(4) Decay Correction	X	(5) RCS Volume (cc)	=	(6) Corrected Activity in RCS (uCi)
Kr-87		X		X		X		=	
Xe-131m		X		X		X		=	
Xe-133		X		X		X		=	
I-131		X		X		X		=	
I-132		X		X		X		=	
I-133		X		X		X		=	
I-135		X	_____	X		X	3.189E+8	=	
Cs-134		X		X		X		=	
Rb-88		X		X		X		=	
Te-129		X		X		X		=	
Te-132		X		X		X		=	
Ba-140		X		X		X		=	
La-140		X		X		X		=	
La-142		X		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

CORE DAMAGE ASSESSMENT USING RADIOISOTOPIC ANALYSIS

TABLE 2

Containment Sump Samples

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____ =

t_S = _____ hrs.

Sump Level:_____ft Sump Volume (Fig. 2):_____ cc

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Sump Volume (cc)	X	(4) Decay Correction	=	(5) Corrected Activity In Containment Sump, (uCi)
Kr-87		X		X		=	
Xe-131m		X		X		=	
Xe-133		X		X		=	
I-131		X		X		=	
I-132		X		X		=	
I-133		X		X		=	
I-135		X	_____	X		=	
Cs-134		X		X		=	
Rb-88		X		X		=	
Te-129		X		X		=	
Te-132		X		X		=	
Ba-140		X		X		=	
La-140		X		X		=	
La-142		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

CORE DAMAGE ASSESSMENT USING RADIOISOTOPIC ANALYSIS

TABLE 3

Containment Atmosphere Samples

Sample No.:_____ Sample Location:_____

t_S = Time of Analysis:_____ - Time of Reactor Shutdown:_____ =

t_S = _____ hrs.

Containment Pressure:_____psia Sample Pressure:_____psia

Containment Temperature:_____°F Sample Temperature:_____°F

(Use 14.7 psia and 250°F, due to heat tracing, if actual data is not available)

(1) Isotope	(2) Sample Activity (uCi/cc)	X	(3) Volume Correction VCF, (cc)	X	(4) Decay Correction	=	(5) Corrected Activity In Containment Atmosphere, (uCi)
Kr-87		X		X		=	
Xe-131m		X		X		=	
Xe-133		X		X		=	
I-131		X		X		=	
I-132		X		X		=	
I-133		X		X		=	
I-135		X	_____	X		=	
Cs-134		X		X		=	
Rb-88		X		X		=	
Te-129		X		X		=	
Te-132		X		X		=	
Ba-140		X		X		=	
La-140		X		X		=	
La-142		X		X		=	

Decay Correction Factor: $e^{+\lambda t_s}$

$$VCF = 7.589 \times 10^{10} \text{ (cc)} \times \frac{P_{cont}(\text{psia}) \times (T_{samp} + 460)^{\circ}F(^{abs})}{P_{samp}(\text{psia})\ (T_{cont} + 460)^{\circ}F(^{abs})}$$

CORE DAMAGE ASSESSMENT USING RADIOISOTOPIC ANALYSIS

TABLE 4

Total Quantity of Radionuclides in Samples

(1) Isotope	(2) Corrected Activity In RCS (uCi)	+	(3) Corrected Activity in Containment Sump, (uCi)	+	(4) Corrected Activity in Containment Atmosphere (uCi)	÷	(5) uCi-Ci Conversion Constant (uCi/Ci)	=	(6) TOTAL ACTIVITY (Ci)
Kr-87		+		+		÷		=	
Xe-131m		+		+		÷		=	
Xe-133		+		+		÷		=	
I-131		+		+		÷		=	
I-132		+		+		÷		=	
I-133		+		+		÷		=	
I-135		+		+		÷	1.0E+6	=	
Cs-134		+		+		÷		=	
Rb-88		+		+		÷		=	
Te-129		+		+		÷		=	
Te-132		+		+		÷		=	
Ba-140		+		+		÷		=	
La-140		+		+		÷		=	
La-142		+		+		÷		=	

CORE DAMAGE ASSESSMENT USING RADIOISOTOPIC ANALYSIS

TABLE 5

RCS SAMPLES

Sample No.:_____ Sample Location:_____

(1) Isotope	(2) Corrected Activity In RCS (uCi/CC)	(3) Calculated Isotope Ratio*	(4) Fuel Pellet Inventory Ratio	(5) Gas Gap Inventory Ratio	(6) Probable Source
Kr-87			0.2	0.001	
Xe-131m			0.003	0.001-0.003	
Xe-133		NA	1.0	1.0	Both
I-131		NA	1.0	1.0	Both
I-132			1.4	0.01-0.05	
I-133			2.0	0.5-1.0	
I-135			1.8	0.1-0.5	

* Equations:

$$\text{Noble Gas Ratio} = \frac{\text{Corrected Activity of Noble Gas Isotope}}{\text{Corrected Activity of Xe-133}}$$

$$\text{Iodine Ratio} = \frac{\text{Corrected Activity of Iodine Isotope}}{\text{Corrected Activity of I-131}}$$

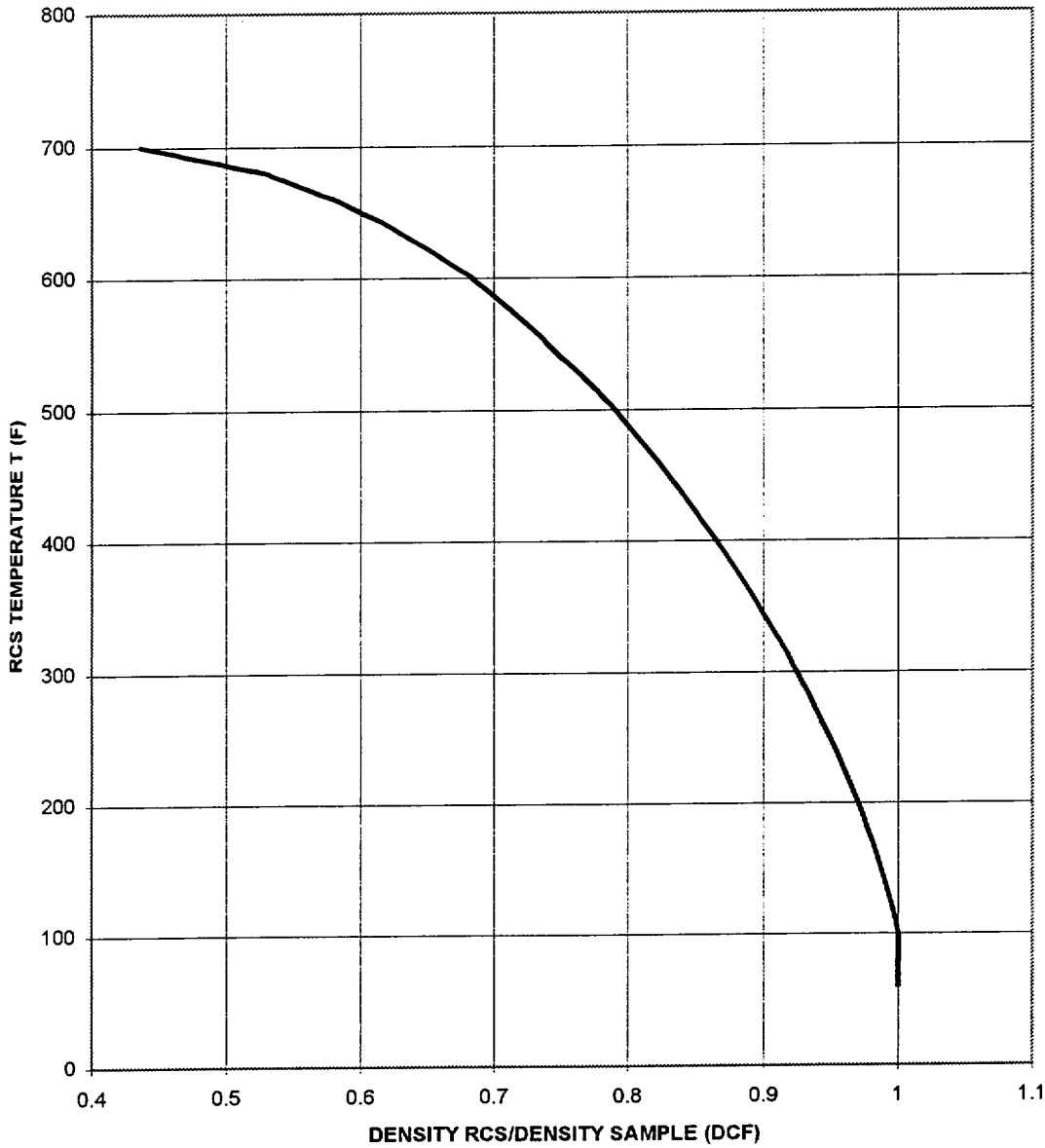
CORE DAMAGE ASSESSMENT USING RADIOISOTOPIC ANALYSIS

TABLE 6

(1) Isotope	(2) Total Activity (Ci)	÷	(3) Power Correction Factor	÷	(4) Equilibrium Inventory	X	(5) 100%	=	(6) Percent of Equilibrium Inventory Released
GAS GAP									
Kr-87		÷		÷	9.5E+0	X		=	
Xe-131m		÷		÷	6.6E+4	X		=	
Xe-133		÷		÷	1.8E+7	X		=	
I-131		÷		÷	9.0E+6	X	100%	=	
I-132		÷		÷	9.9E+3	X		=	
I-133		÷		÷	8.9E+6	X		=	
I-135		÷		÷	1.6E+6	X		=	
FUEL PELLET									
Kr-87		÷		÷	4.7E+7	X		=	
Xe-131m		÷		÷	7.0E+5	X		=	
Xe-133		÷		÷	2.0E+8	X		=	
I-131		÷		÷	9.9E+7	X		=	
I-132		÷		÷	1.4E+8	X		=	
I-133		÷		÷	2.0E+8	X		=	
I-135		÷		÷	1.9E+8	X	100%	=	
Cs-134		÷		÷	1.8E+7	X		=	
Rb-88		÷		÷	6.8E+7	X		=	
Te-129		÷		÷	3.1E+7	X		=	
Te-132		÷		÷	1.4E+8	X		=	
Ba-140		÷		÷	1.7E+8	X		=	
La-140		÷		÷	1.8E+8	X		=	
La-142		÷		÷	2.2E+8	X		=	

DENSITY CORRECTION FACTOR
FOR
REACTOR COOLANT SYSTEM TEMPERATURE

Figure 1



CONTAINMENT SUMP LEVEL
VS.
CONTAINMENT SUMP VOLUME

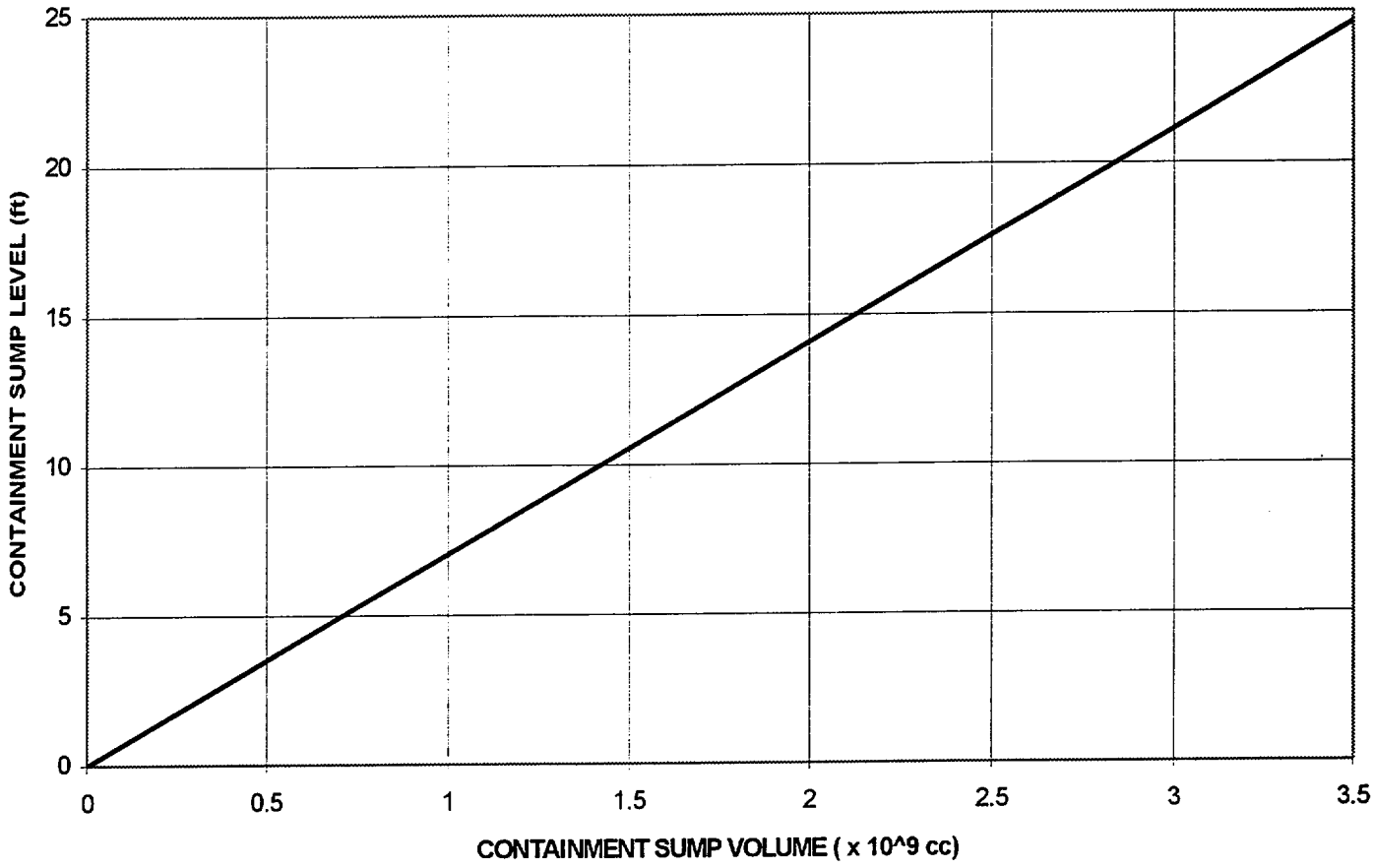



Figure 2