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Attachment

NUCLEAR SAFETY RELATED

DAVIS-BESSE NUCLEAR POWER STATION

Emergency Plan Implementing Procedure

EP-2320

EMERGENCY TECHNICAL ASSESSMENT

Revision 0

FOR INFORMATION ONLY

Concurrence: J Scott-Wasilk Date: 8/18/86
Environmental & Emergency Preparedness Manager

Approval: [signature] Date: 8-20-86
Plant Manager

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LIST OF EFFECTIVE PAGES

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Table of Contents

1. PURPOSE

1.1 To describe necessary technical and engineering assessment activities during an emergency response to a major off-normal event involving the reactor systems.

1.2 To provide the methodology for estimating reactor core damage using the radiochemistry sample analysis, Emergency Containment Radiation Plots and other available parameters such as incore thermocouple readings and containment hydrogen concentration.

2. REFERENCES

2.1 Davis-Besse Nuclear Power Station Emergency Plan

2.2 NUREG-0654/FEMA-REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

2.3 EP-1500, "Emergency Classification"

2.4 EP-2310, "TSC Activation and Response"

2.5 EP-2610, "Emergency C&HP Organization Activation and Response"

2.6 SP 1104.55, "Containment Hydrogen Dilution and Purge System"

2.7 Bechtel to TED Letter dated October 22, 1984, Log No. BT 15004

3. DEFINITION

Post Accident Sampling System (PASS) - This is a system designed to obtain high radioactivity samples from containment building atmosphere, pressurizer liquid space, letdown system, decay heat loops one and two, and one reactor coolant system cold leg 2-1.

4. RESPONSIBILITY

The TSC Engineering Manager is responsible for implementing this procedure.

5. INITIATING CONDITIONS

5.1 Emergency classification of "Alert" or greater has been declared per EP-1500, "Emergency Classification" and the TSC has been activated.

5.2 This procedure shall become effective August 10, 1986 and replaces AD 1850.08.

6. PROCEDURE

6.1 Activation

NOTE 6.1.1 A plant status checklist is provided in EP-2310, "Technical Support Center Activation and Response," to assist in the initial transfer of information between the Control Room and the Technical Support Center. If not previously accomplished, refer to EP-2310 to complete this checklist.

All analyses, evaluations, and recommendations shall be documented on the forms provided, logs or other appropriate documents by the TSC Engineering staff.

6.1.1 When the TSC is activated, the TSC Engineering Staff shall, at the direction of the TSC Engineering Manager, determine the status of the plant equipment and systems.

1. The Mechanical Engineer shall determine the status of the plant process systems including:

   a. NSSS
   b. ECCS
   c. Primary plant support systems
   d. Containment systems
   e. Secondary plant systems

2. The Electrical Engineer shall determine the status of the plant electrical systems including:

   a. Essential AC Distribution System
   b. Emergency Diesel Generators
   c. Essential DC Distribution System
   d. Nonessential electrical distribution systems

3. The I&C Engineer shall determine the status of the plant I&C systems including:

   a. Reactor Protection System
   b. Safety Feature Actuation System
   c. Other essential plant instrumentation and control systems

4. The Core Thermal Hydraulics Engineer shall determine if event history or current plant conditions indicate the potential for core damage exists. Conditions which may indicate the potential for core damage include:

a. Loss of reactor coolant flow
b. Pressurizer level out of range low
c. Significant loss of reactor coolant
d. High containment sump levels
e. Loss of subcooled margin
f. Reactivity excursions
g. Failure of the RPS to shutdown the reactor
h. Failure of ECCS systems
i. High incore thermocouple temperatures
j. High Containment radiation or airborne radioactivity
k. High activity levels in the coolant
l. Any other condition which could cause inadequate core cooling
m. Higher than normal post trip source range counts on the excore detectors (NI 1&2)

The Core Thermal Hydraulics Engineer shall inform the TSC Engineering Manager and TSC Supervisor/Systems of any indications of the potential for core damage

5. The Operations Engineers, under the direction of TSC Supervisor/Operations, shall determine the status of the plant operations including:

a. Essential plant parameters and parameter trends
b. System and equipment status
c. Automatic and manual actions being taken to establish and maintain safe shutdown

6.1.2 The TSC Engineering staff shall evaluate the condition of the reactor, essential safety-related systems, and any significant problem areas identified in Step 6.1.1 above using, as necessary:

1. Plant parameters as obtained from the Data Acquisition and Display System (DADS) or from the Control Room

2. Safety Parameter Display System

3. Reports from the OSC

4. Piping and Instrumentation Drawings, piping system isometrics, general arrangement drawings electrical schematics, and systems descriptions

5. Applicable system operating and/or maintenance procedures

6. The DBNPS Updated Safety Analysis Report

7. Applicable equipment Technical Manuals

6.1.3 The TSC Engineering Manager shall inform the Emergency

Plant Manager of any immediate recommendations for controlling the emergency situation.

6.2 Operation

6.2.1 As directed by the TSC Engineering Manager, the TSC Engineering staff shall provide assistance to the Control Room staff by relieving the plant operators of peripheral duties such as:

1. Plotting cooldown curves, radiation levels or other key parameters to backup or assist in trend analysis

2. Verifying and logging instrument indications where continuous technical assessment and monitoring is required to provide notice of when core degradation is or may be occurring

3. Evaluating the adequacy of natural circulation flow, heat sink efficiency or other system operation

4. Continuing to monitor the status of plant systems and equipment, and plant operations as outlined in 6.1.1

5. Providing information and recommendations as required

6.2.2 If there are indications of the potential for core damage, the Core Thermal Hydraulic Engineer, under the direction of TSC Supervisor/Systems, shall perform an initial assessment of the extent of the core damage as follows:

1. Determine the SFAS radiation monitor readings (RE-2004, RE-2005, RE-2006, and RE-2007) Computer points R311, R312, R313, R314

2. If radiation levels in the containment are normal, go to Step 5. If radiation levels in the containment are significantly above normal, continue with Step 3

3. Evaluate the containment high range radiation dome monitor readings [RE-4596A (Valdyne

Computer Point R294) and RE-4596B] using Attachment 1, Record the results on the form provided in Attachment 2

NOTE 6.2.2.4. 1. For hydrogen monitor to monitor the containment hydrogen concentration containment isolation valves for these monitors must be kept open. Since the containment would be isolated on SFAS, prior to using hydrogen monitor readings verify that isolation valves for these monitors are opened

NOTE 6.2.2.4. 2. Attachment 3 does not include hydrogen generated by radioalysis or corrosion of materials which may effect the long term hydrogen concentration.

4. Determine the containment hydrogen concentration (computer point A302 and A303). If there is measurable hydrogen concentration, use Attachment 3 to determine the approximate percent metal-water reaction. Record results on the form provided on Attachment 2

5. Evaluate RCS pressure and incore thermocouple temperature for indications of possible cladding damage using Attachment 4. Record results on form provided in Attachment 2

6. Continue to monitor plant conditions for indications of degrading core conditions

6.2.3 The TSC Engineering Manager shall inform the Emergency Plant Manager of the results of any estimates of core damage

NOTE 6.2.4 The decision to obtain PASS samples will be made by the Emergency Plant Manager upon recommendation from the TSC Engineering Manager and the TSC C&HP Advisor.

6.2.4 If there are indications of the potential for core damage, and no previous sample data has been obtained, the TSC Engineering Manager shall consider recommending to the Emergency Plant Manager that reactor coolant and containment samples be obtained. Samples are used to more precisely quantify the extent of core damage. The feasibility of obtaining accurate samples will depend on plant conditions and radiation levels. Accurate samples may not be possible if plant conditions are unstable

1. If use of the PASS is required to obtain samples, have the TSC C&HP Advisor investigate the radiological and other hazards and conditions associated with obtaining samples in accordance with EP-2610

2. The following are guidelines for selecting which samples to collect:

   a. If there has been no significant loss of reactor coolant, recommend a sample of the RCS only

   b. If there has been a significant loss of reactor coolant, recommend samples of the RCS, containment sump and containment atmosphere

   For further information on sample locations see Attachment 5

6.2.5 The Core Thermal Hydraulics Engineer shall receive sample data directly from the Health Physics Monitoring Room Coordinator or via the TSC C&HP Advisor. The sample data shall be recorded on the form provided in Attachment 6.

6.2.6 The Core Thermal Hydraulics Engineer shall evaluate sample data in accordance with the guidelines provided in Attachment 5 of this procedure.

1. Record the results of the required calculations on the form provided in Attachment 7.

2. Report the results of the evaluation to the TSC Engineering Manager.

6.2.7 The TSC Engineering Manager shall inform the Emergency Plant Manager of the results of any sample evaluations.

6.2.8 The TSC Engineering Staff shall continue to assess plant conditions, analyze core damage and assist the Control Room as outlined above. Evidence of further core degradation shall be immediately reported to the Emergency Plant Manager.

6.2.9 The TSC Engineering Manager shall coordinate the design and installation of short term instrument and control modifications, including preparation of the installation or abnormal operating procedures necessary to support the evolution.

6.2.10 The TSC Engineering Manager shall arrange for additional engineering support as necessary, such as for:

1. Supplemental transient or accident analysis which may include event tree analyses or computer calculations necessary to provide particular event possibilities that could occur and should be considered

2. Engineering development of system modifications necessary to ensure the immediate safe shutdown of the reactor and any system additions necessary to maintain long term shutdown capabilities

NOTE 6.2.11 For additional information on hydrogen control, refer to SP 1104.55, "Containment Hydrogen Dilution and Purge System".

6.2.11 If there is measurable primary containment atmosphere hydrogen concentration, the TSC Engineering Manager should recommend that the Emergency Plant Manager initiate steps to obtain a hydrogen recombiner

1. Request a hydrogen recombiner from Duquesne Light Company's Beaver Valley Nuclear Station.

2. The Institute of Nuclear Power Operations (INPO) can be utilized to locate other sources for a hydrogen recombiner. See the INPO Emergency Resource Manual

3. Monitor containment hydrogen concentration and recommend corrective action as necessary in accordance with SP 1104.55, "Containment Hydrogen Dilution and Purge System".

6.3 Deactivation

6.3.1 When the emergency has been terminated and the TSC has been deactivated, review and forward any records generated during the emergency to the Emergency Planning Supervisor

6.3.2 Document and report any deficiencies in equipment and procedures to the Emergency Planning Supervisor.

7. FINAL CONDITIONS

7.1 The TSC has been deactivated and the TSC staff duties associated with emergency operations of the plant are no longer required.

7.2 All records generated during the emergency are forwarded to the Emergency Planning Supervisor.

8. RECORDS

8.1 The calculations performed using the attachments to this procedure during an actual emergency or annual NRC exercise will be considered Quality Records.

8.2 All facility logs.

8.3 All records shall be forwarded to the Emergency Planning Supervisor who shall submit Quality Records and any other records deemed necessary to Nuclear Records Management.

9. LIST OF ATTACHMENTS

9.1 Core Damage Assessment Using Containment Radiation Levels, Attachment 1

9.2 Initial Core Damage Assessment Form, Attachment 2

9.3 Title Relationship Between Containment Hydrogen Concentration and Metal Water Reaction, Attachment 3

9.4 Core Damage Assessment Using Core Exit Thermocouples, Attachment 4

9.5 Core Damage Assessment Guidelines Using PASS Sample, Attachment 5

9.6 Sample Analysis Results Form, Attachment 6

9.7 PASS Sample Core Damage Assessment Form, Attachment 7

9.8 Identification of Commitments, Attachment 8

CORE DAMAGE ASSESSMENT USING CONTAINMENT RADIATION LEVELS

Evaluation

Evaluate the extent of core damage using containment radiation levels as follows:

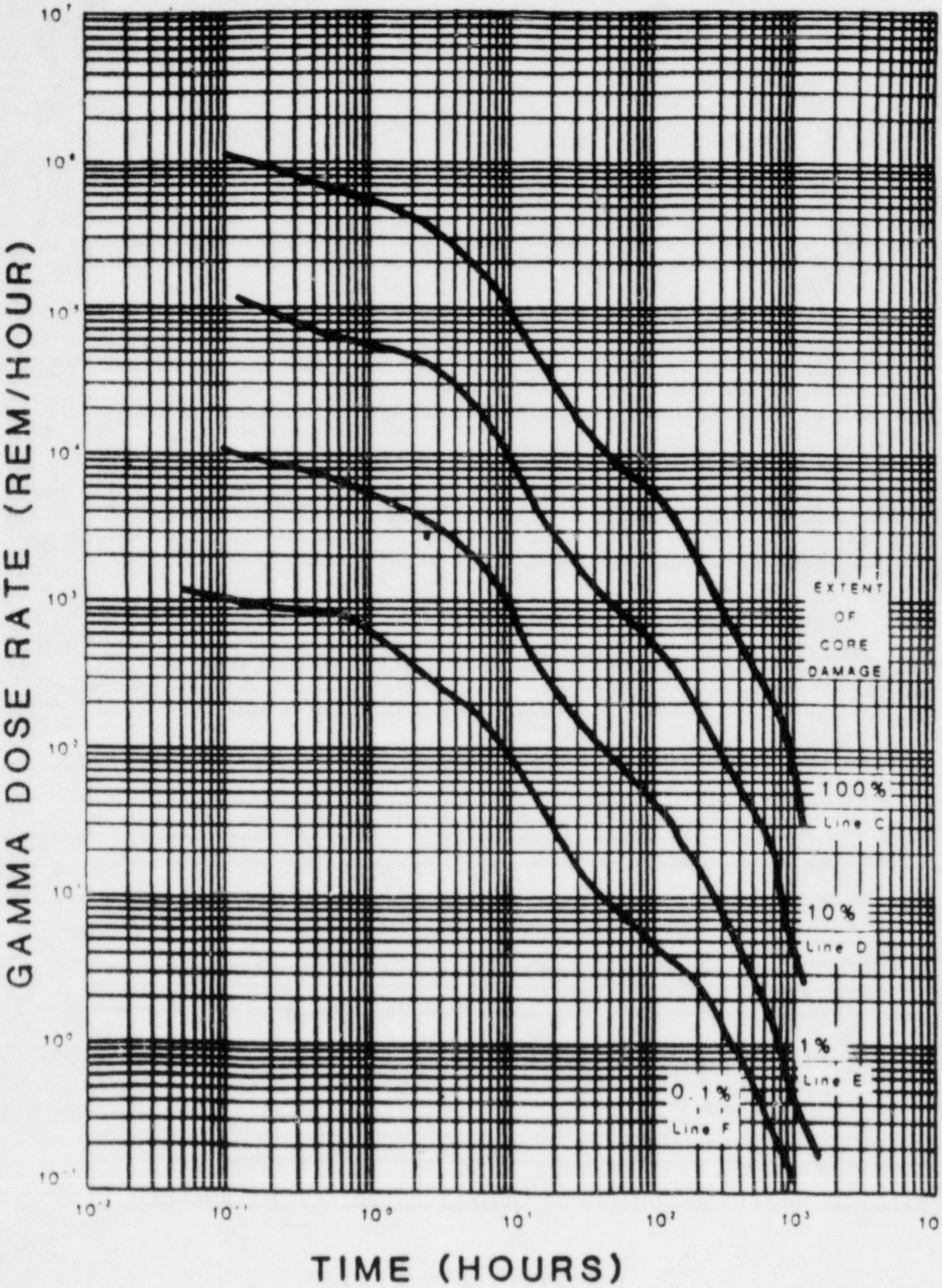
1. Locate a point on the Figure on page 2 of Attachment 1 based on the highest reading in R/hr from RE-4596A or B, and the time after the LOCA; this is Point "X"

2. Determine the region in which Point "X" is located, i.e., Region C-D, D-E, E-F, or Below F

3. The radiation level where the "hours" intersect the appropriate line (C, D, E or F) is used to determine Point C, D, E or F used in Step 4

4. Calculate % fuel inventory release, or % clad failure using the appropriate equations in Table 1.

CORE DAMAGE ASSESSMENT USING CONTAINMENT RADIATION LEVELS (Cont.)



CORE DAMAGE ASSESSMENT USING CONTAINMENT RADIATION LEVELS (Cont.)

TABLE 1

CORE DAMAGE EVALUATION USING HIGH RANGE CONTAINMENT MONITORS

Region	Equation	Approximate Source and Core Damage Estimate
C-D ($C \geq X \geq D$)	$Y = \frac{X}{C} \times 100\%$	Y = % fuel inventory released (based on RG 1.4)
		100% clad failure (based on RG 1.25)
		Potential core melt
D-E ($D > X \geq E$)	$Y = \frac{X}{D} \times 10\%$	Y = % fuel inventory released (based on RG 1.4)
	$Z = \frac{X}{D} \times 100\%$	Z = % clad failure (based on RG 1.25)
		Core partially uncovered
E-F ($E > X \geq F$)	$Y = \frac{X}{E} \times 1\%$	Y = % fuel inventory released (based on RG 1.4)
	$Z = \frac{X}{E} \times 10\%$	Z = % clad failure (based on RG 1.25)
Below F ($X < F$)	$Z = \frac{X}{F} \times 1\%$	Z = % clad failure (based on RG 1.25)
		NOTE: Z should be less than 1%.

INITIAL CORE DAMAGE ASSESSMENT FORM

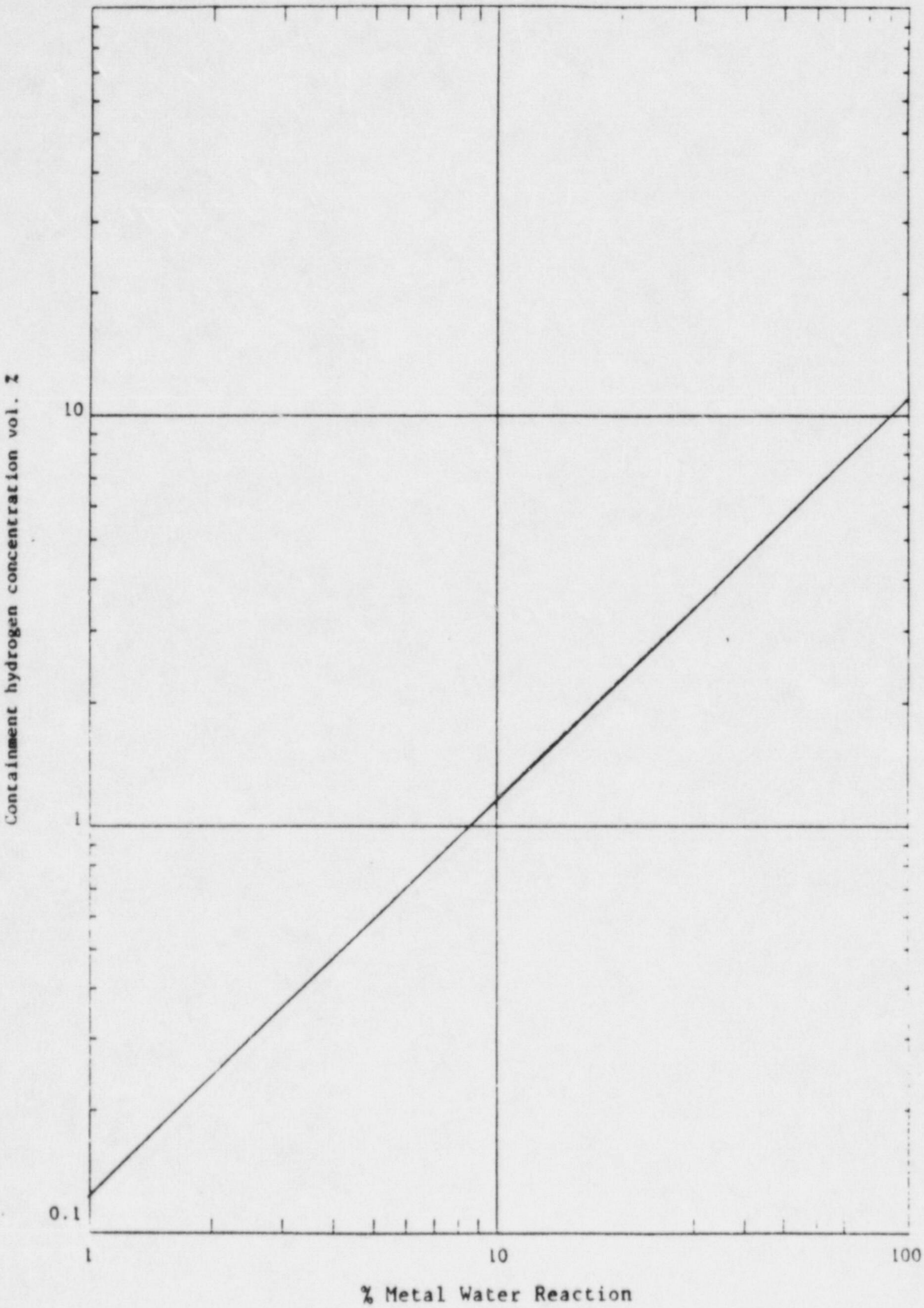
Prepared By: ____________________

Checked By: ____________________

Date: ____________

Containment Radiation					Containment Hydrogen			Core Exit Thermocouples						
Time	Cont. Rad. (R/hr) RE-4596A / RE-4596B	Region	C, D, E, or F	Inventory Released (%) / Clad Failure (%)	Time	Cont. H_2 (%) A302 / A303	Metal-Water Reaction (%)	Time	Core TCs Used	Temp. (°F)	Ave. Temp. (°F)	RCS Pressure (psig)	Clad Temp. (°F)	Fuel Damage State

RELATIONSHIP BETWEEN CONTAINMENT HYDROGEN CONCENTRATION AND METAL WATER REACTION



CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES

Evaluation

Evaluate the extent of core damage using core exit thermocouples as follows:

If core conditions are no longer deteriorating, use parameter history using DADS to determine worst case conditions during the transient.

1. If more than eight core exit thermocouples are available, calculate the average core exit thermocouple temperature by averaging no less than five highest reading thermocouples (computer points T511 to T562). If eight or less thermocouples are available, use the three highest thermocouples

2. Determine the RCS system pressure as measured by hot leg pressure sensors (P724, P725, P732, and P733)

3 Use the Figure on page 2 of Attachment 7 to determine approximate cladding temperature

4. Use Table 1 to determine possible cladding and fuel damage states.

Background

The core exit thermocouple readings are normally used in conjunction with the radiochemistry sample analysis to evaluate the core damage. However these readings provide significant information on the core conditions prior to taking PASS samples. The evaluation of temperature readings in various regions of the core will provide an estimate of damage in those regions.

If the core exit thermocouple readings indicate the reactor is subcooled throughout the transient (i.e., the temperature is to the left of the saturation curve in Figure 1) there should not be any core damage.

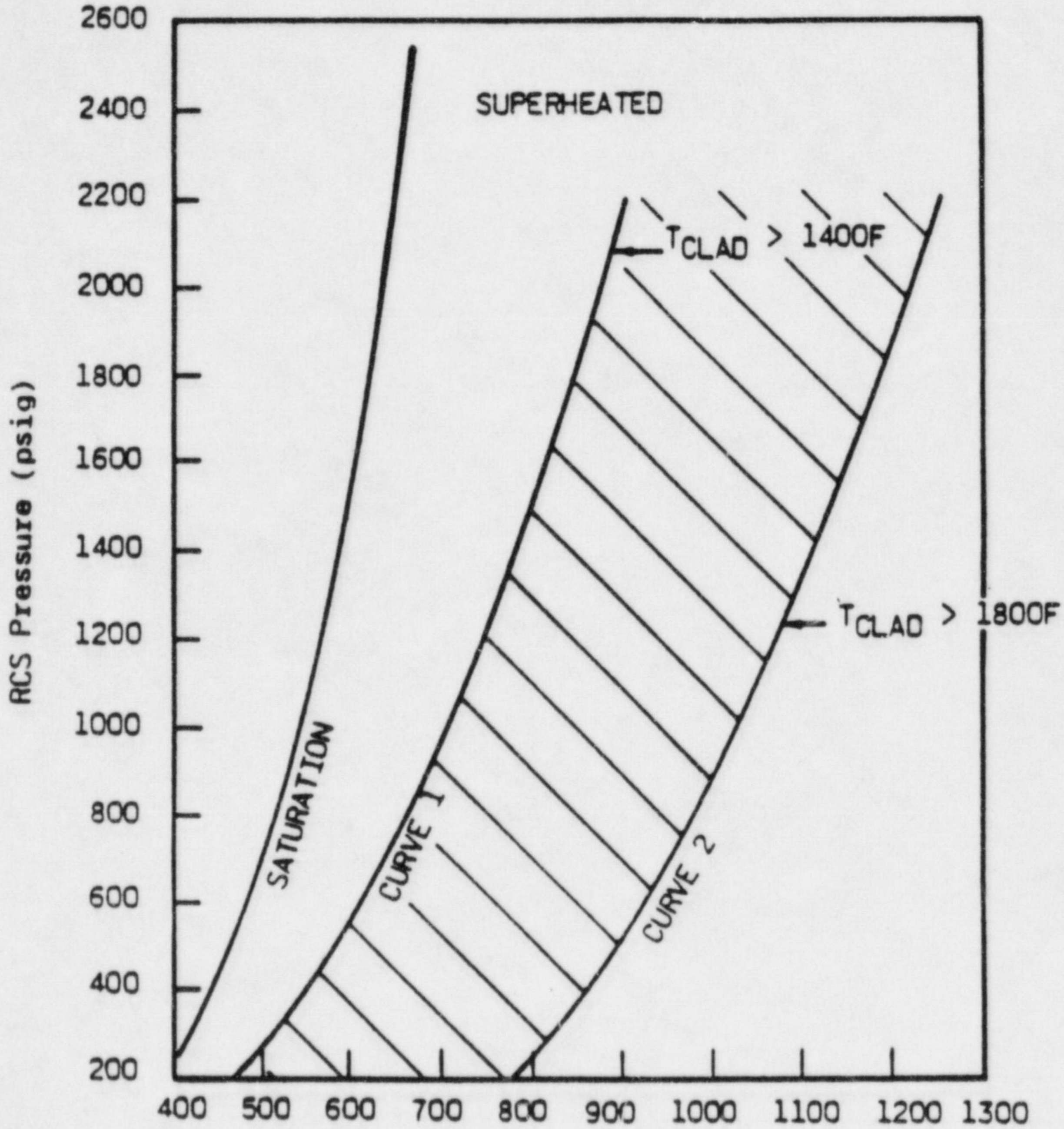
As the temperature increases beyond saturation and approaches curve 1 in the Figure on page 2 of Attachment 7, the cladding will experience ballooning and bursting. This failure will start at temperatures above 1300°F and start accelerating at 1400°F.

In the temperature region between curves 1 and 2 (i.e., 1400°F and 1800°F), the cladding will experience major damage. Around 1600°F oxidation of cladding due to Zr-metal water reaction will start generating significant quantities of hydrogen. If the temperature approaches curve 2, it can be assumed that there is a major cladding damage and possible fuel overheating.

If the cladding temperature is beyond 1800°F, it can be assumed that there is a significant fuel overheating and possible fuel melt. Since the Figure 1 is a correlation for the cladding temperature, it is not possible to determine the extent of fuel overheating or melting.

CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES (Cont.)

Correlation of Core Exit
and Cladding Temperatures



Core Exit Thermocouple Temperature (F)

CORE DAMAGE ASSESSMENT USING CORE EXIT THERMOCOUPLES (Cont.)

TABLE 1

RELATIONSHIP BETWEEN FUEL DAMAGE AND CLAD TEMPERATURE

Cladding Temperature °F	Fuel damage state
<1300	No damage
>1300	Possible cladding damage
>1400	Cladding damage accelerates
>1600	Oxidation of cladding and hydrogen generation. Possible release due to overheat when temperature approaches 1800°F
>1800	Fuel overheating and possible fuel melt when temperatures are significantly greater than 1800°F[1]

[1]The liquifaction temperature for the UO_2-Zr-ZrO_2 liquid eutectic is approximately 3500° F.

CORE DAMAGE ASSESSMENT GUIDELINES USING PASS SAMPLE

1. PURPOSE

The purpose of this attachment is to provide guidance on determining the extent of core damage during anticipated operational occurrences and under accident conditions using PASS samples. In addition, these guidelines utilize the available plant parameters such as core exit thermocouple readings, hydrogen concentrations, etc. which provide indications of core damage.

2. REFERENCES

2.1 Operator Training - Degraded core recognition and mitigation TRG-81-3, B&W, May 1981

2.2 NUREG 0956 (draft) Radionuclide release under specific LWR Accident Conditions January 1983

2.3 Rogovin Report. Three Mile Island, a report to the commissioners and to the public. Volume II, Part 2

2.4 Requirements for Post-Accident Sampling and Analysis - Frank Witt, USNRC. Presented at the 1983 ANS Winter Meeting

3. GUIDELINES

3.1 The post accident samples that are required to assess the core damage are dependent on the accident scenario. For accidents that do not involve the breach of reactor coolant system (RCS) the sample analysis results from the RCS are adequate to assess the core damage. For the loss of coolant accidents, samples from the RCS, Sump and Containment atmosphere are required. However if at the time of sampling the sump water has been recirculated through the core for some time, it can be assumed that the activity in the sump and the RCS are the same and only a sump sample is needed. The methodology used in this procedure utilizes RCS, Sump and Containment air sample results. For a more accurate assessment, samples from additional sample locations (eg. pressurizer) could be used. For accidents involving secondary system (eg. steam generator tube rupture, steam line break) samples from steam generator would provide additional input to the core damage assessment.

If the sample results are not already available, request C&HP to obtain the necessary samples and analyze the samples. See Table 1 to determine the required samples. The applicable sampling procedures are also listed in this table.

3.2 The criteria used in selecting the isotopes used in this methodology are as follows:

a. The selected isotopes should have sufficient core inventory and half life to permit the analysis of samples at some time following an accident

b. The number of isotopes selected should be sufficient to differentiate the four major fuel damage states: no damage, fuel cladding failure, fuel overheating and core melt and to quantify any damage as: <10%, 10-50%, or >50%. The volatility of the fission products during various stages of the accidents is used as the basis for the methodology.

During normal operation, the fission products diffuse through the fuel to the fuel cladding gap. All the activity that is accumulated in the fuel clad gap will be released to the RCS at the time of cladding failures. Since the diffusion rate for noble gases and iodines is higher than for the other elements, the activities of these isotopes are used to estimate the cladding failures. The draft NUREG-0956 assumes an average temperature of 900 C (1650 F) for cladding failures. Reference 1 states the rate of cladding failures increases due to hoop stress when cladding temperature rises to 1400°F. However, cladding failures may occur before this temperature

As the fuel temperature increases, the diffusion of noble gases and iodines from the fuel would increase and at the same time some other materials such as Cesium are volatized and can diffuse out. The release of Cesium is quite variable and could be caused by other factors such as compound formation (Ref. 3). For this reason only noble gases and iodines are used to estimate core overheating

At temperatures that can cause fuel liquification or melting some fraction of other fission products such as Tellurium, Ruthinium, Strontium, Barium could be released. But under certain conditions Tellurium and Ruthinium could be released before fuel melt. Thus the presence of Tellurium and Ruthinium does not necessarily indicate fuel melting (Ref. 2.3). For this reason Ba-140 is considered to estimate fuel melt.

Other noble gas and Iodine isotopes in addition to I-131, I-133, Xe-133 and Kr-88 could be used in the damage assessment for added accuracy. However, depending on the isotope selected, additional corrections for the decay of the parent isotope parent-daughter relationships should be applied. For example, if Xe-135 or I-132 are used, corrections for decay of I-135 or Te-132, respectively, should be considered.

3.3 The following step is needed only if the sample activities obtained from C&HP are not adjusted for decay to the time of reactor shutdown, accident or transient. Calculate Decay Correction Factor (DCF) to estimate the concentrations when the reactor was shutdown.

$$DCFi = e^{-\lambda_i \tau_s}$$

where λ_i = decay constant for isotope i, hr^{-1} (see Table 4)

τ_s = time elapsed between reactor shutdown and sampling or sample counting (hr).

Calculate the sample concentration at shutdown for each isotope i using the following formula:

$$Ci = \frac{\text{measured concentration}}{DCFi}$$

|CAUTION 3.4| The transients resulting in a power change or the RCS pressure or temperature changes may cause RCS iodine concentration to increase due to iodine spiking. This should not be interpreted as an increase in the failed fuel. Iodine spiking will be more pronounced for I-131 isotope. If large increase in I-131 concentration is noted, use the noble gas isotopes and other Iodine isotopes to determine whether additional fuel failures have occurred.

3.4 For the accidents in which RCS is not breached, compare the measured concentration with the average reactor coolant concentration prior to reactor shutdown. If there is no marked increase in the reactor coolant concentration it can be concluded no fuel failures have occurred.

3.5 If it is determined that there is an activity release during the transient, calculate the total activity released from the fuel during the transient. Since the pressure and temperatures of the RCS, sump and containment are different from the samples, the sample activities need to be corrected to the density differences.

Calculate total activity (Ti) released during the accident for each isotope.

$$Ti = Ai\ (RCS) + Ai\ (Sump) + Ai(CB) - Ni$$

3.5.1 Ai (RCS) is the activity contained in the reactor coolant system in curies for each isotope and is calculated using

$$Ai\ (RCS) = Ci\ (RCS)\ \mu Ci/cc \times \text{RCS volume gal} \times 3785\ \frac{cc}{gal} \times D \times 10^{-6}\ \frac{Ci}{\mu Ci}$$

Where Ci (RCS) is the RCS sample concentration for isotope i corrected for decay and D is the density correction factor for RCS temperature (Table 3)
RCS Volume = 83028 gal

NOTE 3.5.2 Sump activity is needed only for transients in which significant quantity of reactor coolant released to containment. Additional samples may be needed for some other transients eg. steam generator tube rupture. For those events the total activity contained in each system can be estimated using similar equations.

3.5.2 Ai (Sump) is the activity contained in the sump in curies for each isotope and is calculated using

Ai (Sump) = Ci (Sump) x water volume in sump in gal. x 3785 x D x 10^{-6}.

Where Ci (Sump) is the sump sample concentration (μCi/cc) for isotope i corrected for decay and D is the density correction factor for sump temperature

The water volume in the sump can be estimated using containment water level wide range instrumentation (computer points L-321 and L-322) and Table 2.

3.5.3 Ai (CB) is the activity contained in the containment in curies for each isotope and is calculated using

Ai (CB) = Ci (CB) x containment volume ft^3

$$\times 28320\ cc/ft^3 \times \frac{P_1 T_2}{P_2 T_1} \times 10^{-6}$$

Where Ci (CB) is the containment air sample (μCi/cc) for isotope i corrected for decay

Containment volume = 2.83 E6 ft^3

P_1, P_2 are containment and sample pressures (PSIA) respectively.

T_1, T_2 are containment and sample temperatures (°R) respectively.

3.5.4 Ni is the activity in curies contained in the RCS prior to transient. Ni can be ignored if post transient RCS concentrations are significantly larger than pretransient concentrations.

Ni is calculated using the same expression given for Ai (RCS).

3.6 The core inventories listed in Table 4 assume that the reactor was operating at 100% power to the end of core life. However in reality the reactor would be operating at various power levels depending on the power needs. The core inventory of each isotope varies depending on the half life, and the length of operation at a given power level. Normally the activity levels in the reactor are considered to be in equilibrium if the reactor is operated at a steady state power level for four half lives. For example, for isotopes with half life of one day, the activity will reach equilibrium after four days of operation at constant power. For operation at varying power levels, the calculated core release fractions should be adjusted to account for power history. It can be assumed that the reactor is at a constant power if the power level did not change by 10%. For the isotopes selected above, the use of time weighted average power level for 30 days or less would provide a reasonable estimate of power correction factor (PCF). However a more accurate estimate could be made by taking into account the buildup and decay of the isotope during operation at various power levels. Use of this method will involve more computation time. Both the methods are presented below. Select the method based on the accuracy desired.

3.6.1 The Power Correction Factor (PCF) for each isotope can be calculated using the following formula:

$$PCFi = \frac{\sum_j Pj\ (1-e^{-\lambda itj})\ e^{-\lambda itdj}}{2772\ (1-e^{-\lambda ito})}$$

where Pj = average power level (MWT) during operating time period tj

λi = decay constant for isotope i $days^{-1}$
(Table 4 provides decay constant in hr^{-1} it has to be converted to $days^{-1}$

tj = operating period in days during which power did not change ±10% (Pj)

tdj = time in days between shutdown and power change

to = total operating time in days

3.6.2 Alternately, to save computation time, the following correction factors can be used:

For I-133 and Kr-88

$$PCF = \frac{\text{Time weighted Average Power level for prior 4 days}}{2772}$$

For Ba-140, I-131, Xe-133

$$PCF = \frac{\text{Time weighted Average Power level for prior 30 days}}{2772}$$

3.7 Calculate the fraction of core inventory (Fi) released for each isotope.

$$Fi = \frac{Ti}{PCFi \times Corei}$$

where:

Ti = total activity (from Step 3.5)

PCFi = power correction factor for each isotope (Step 3.6)

Core i = the core inventory for full power operation for isotope i (See Table 4).

3.8 Using core release fractions calculated from Step 3.7 and Figures 1, 2 and 3 estimate.

a. maximum cladding failures, by arbitrarily attributing all activity to cladding failures (Figure 1).

b. maximum fuel overheated, by arbitrarily attributing all activity to fuel overheat (Figure 2).

c. maximum fuel melted, by arbitrarily attributing all activity to fuel melt (Figure 3).

3.9 Because of the overlap in the releases, and the release fractions would be dominated by a small amount of overheat or melt to apportion the releases to each type of failure mechanism, an engineering judgement based on all the available information should be employed. This information was previously recorded in Attachment 2.

a. Presence of Ba-140 indicates fuel overheating or melt.

b. Cladding temperatures indicate cladding failures, fuel overheating or fuel melt.

c. Core exit thermocouple temperature distribution will also provide an indication on the extent of damage.

d. The curves used in the estimation of core damage assume uniform distribution of activity in the core. However, in reality, the activity in various regions of the core would be different. The estimated core damage may be adjusted based on core exit thermocouple temperature distribution, and burnup or neutron flux information for various regions of the core.

e. Hydrogen concentration in the containment will provide an indication on maximum amount of zirconium reacted.

f. Based on the above, provide an estimate for each of the fuel damage state.

See the examples for further guidance.

TABLE 1

SAMPLING LOCATIONS FOR VARIOUS ACCIDENTS

Symptom	Principal Sample Location	Procedure	Other Sample Locations
Reactor coolant pressure boundary intact	RCS cold leg	SP 1103.00	Pressurizer
LOCA	RCS cold leg	SP 1103.00	
	Containment Sump (through decay heat loop)	SP 1103.00	
	Containment Atmos.	SP 1103.01	
Steam Line Break	RCS cold leg	SP 1103.00	Pressurizer
Steam Generator tube rupture	RCS cold leg	SP 1103.00	Affected Steam Generator

TABLE 2

CORRELATION BETWEEN CONTAINMENT WATER LEVEL AND WATER VOLUME

NOTE 1: Assume linear relationship between water level and volume in between the incremental steps.

Water Level Elevation, Ft.	Water Volume Gallons
544	3.43E4
546	3.91E4
548.5	4.73E4
550.5	5.53E5
553	6.96E4
557	9.42E4
561	1.16E5
563.5	1.33E5
565	1.48E5
567.5	3.13E5
568.5	3.84E5
572	6.44E5

TABLE 3

DENSITY CORRECTION FACTORS

NOTE: The temperature correction factors are applied based on the temperature of the sample source (e.g., RCS, Sump, etc.).

Temp.	Density correction factor (D)
100	1
150	0.987
200	0.97
250	0.949
300	0.924
350	0.897
400	0.865
450	0.830
500	0.790
540	0.752
560	0.731
580	0.708
600	0.682

TABLE 4

RADIONUCLIDE INVENTORIES AT FULL POWER AND DECAY CONSTANT

NOTE: The core inventories above are based on 277 EFPD equilibrium cycle. For the 18 month fuel cycle these numbers vary ±15% based on plutonium power fraction. For a more in depth analysis specific calculations for the fuel cycle of interest may be needed.

Isotope	Decay Constant $Hr.^{-1}$	Inventory at full power (Corei) curies
Kr-87	5.47E-1	4.52E7
Kr-88	2.48E-1	6.84E7
Xe-133	5.48E-3	1.43E8
I-131	3.59E-3	7.44E7
I-133	3.41E-2	1.44E8
I-135	1.04E-1	1.4E8
Ba-140	2.26E-3	1.44E8

Figure - 1 Relationship Between Cladding Failures and Core Release Fraction

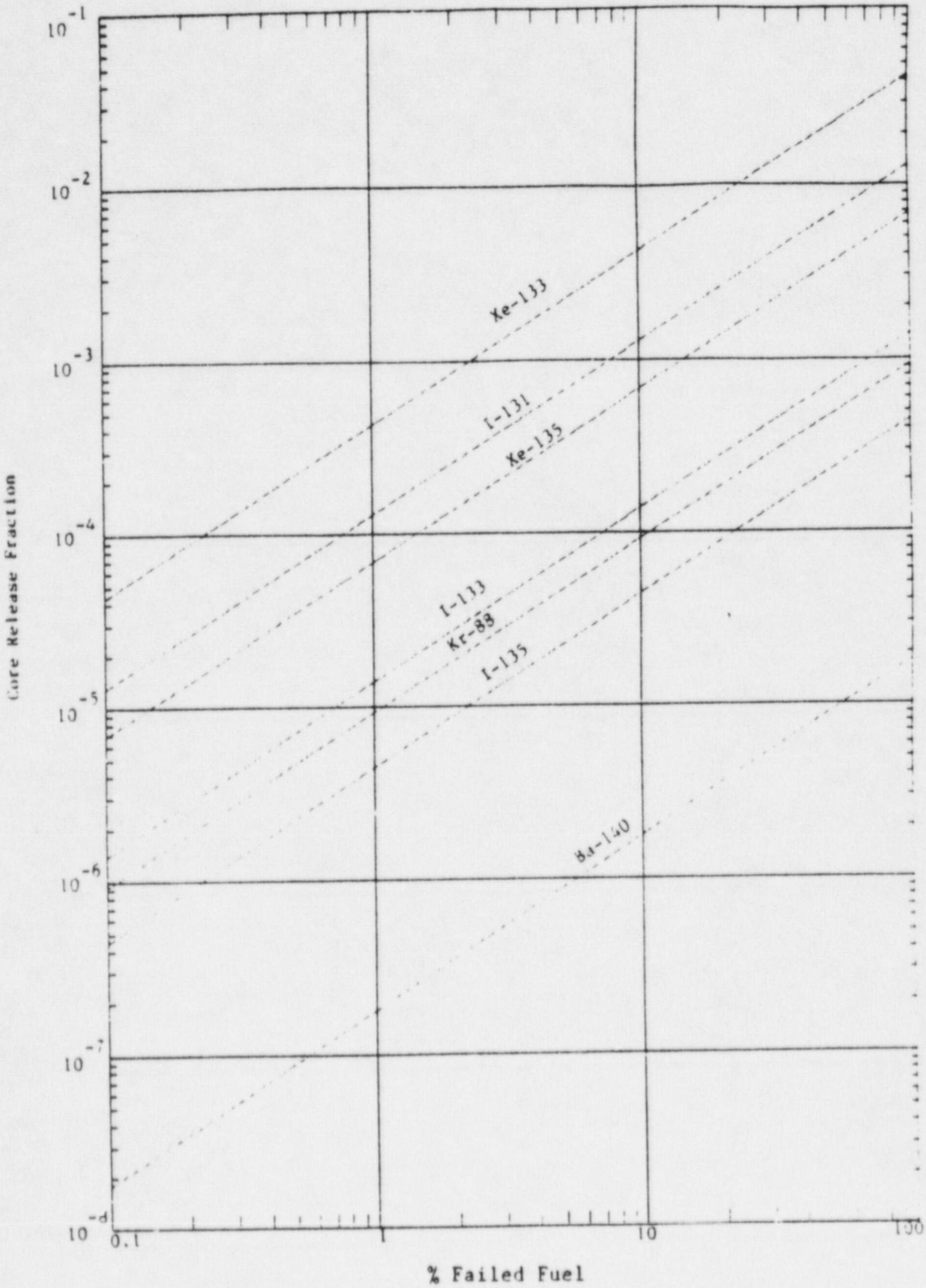


Figure - 2 Relationship Between Fuel Overheating and Core Release Fraction

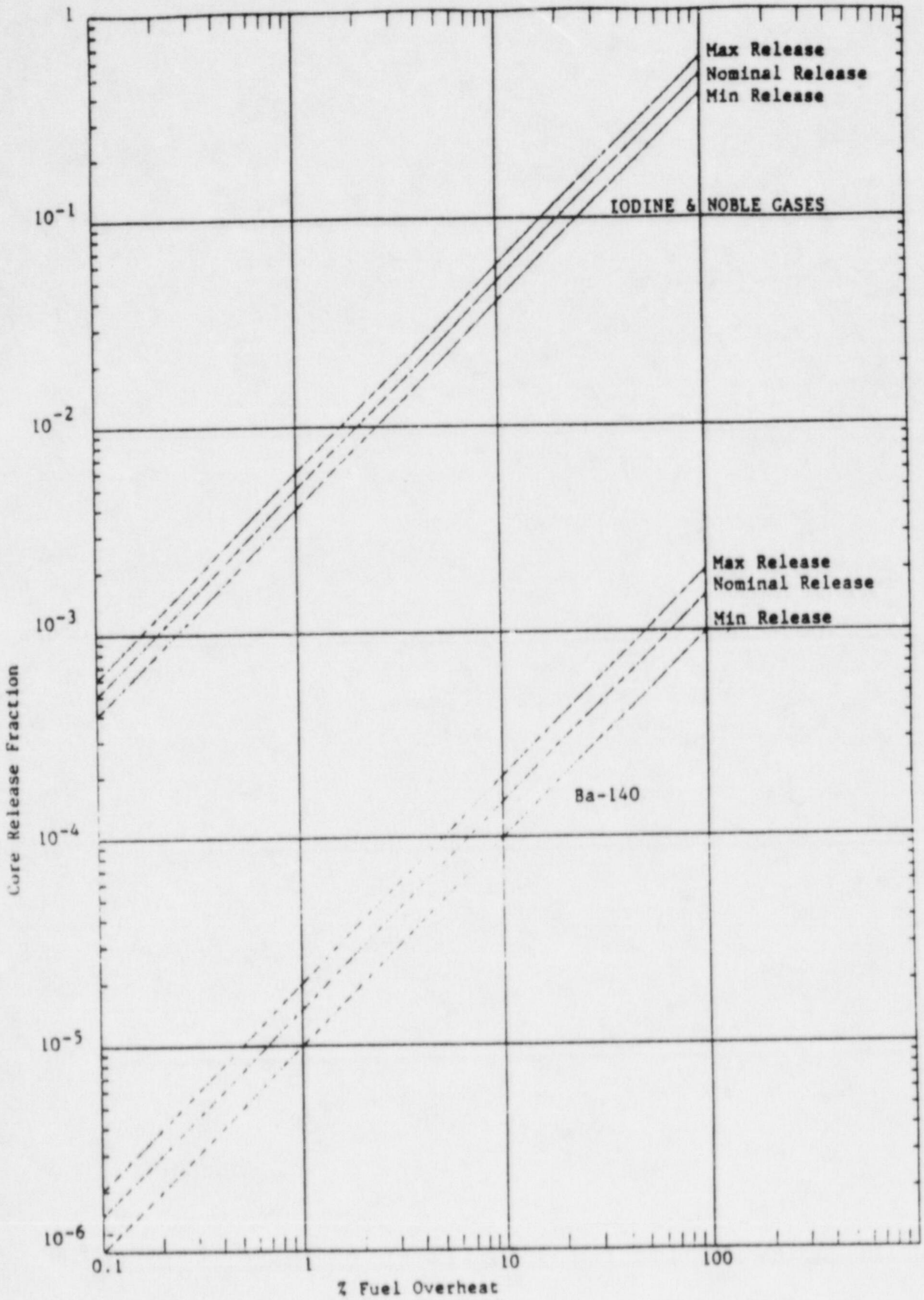
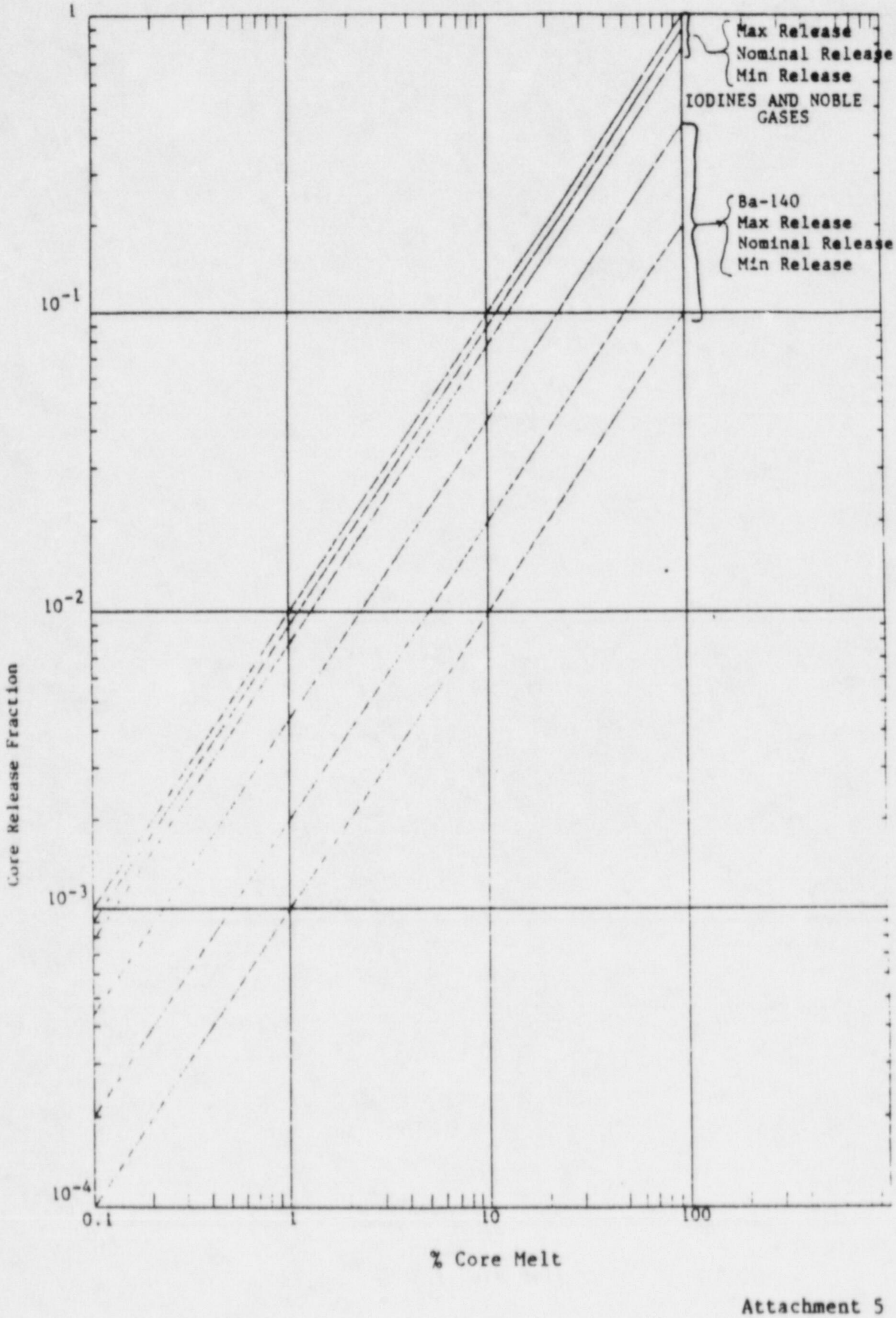


Figure - 3 Relationship Between Core Melt and Core Release Fraction



Examples

The following examples illustrate the use of the core damage assessment methodology. All these examples assume the following conditions for power history and radio chemistry prior to the transient.

Reactor Coolant Concentrations
μCi/cc

I-131	0.3
I-132	0.1
I-133	0.26
Xe-133	23
Kr-87	0.06
Kr-88	0.19
Ba-140	5.2E-4

Reactor is at power for more than 100 days.

Recent power history prior to shutdown. Shutdown is T = 0

T<4 days	1940 MWT
4d<T<10 days	2360 MWT
10d<T<30 days	2772 MWT
T>30	2360 MWT

Example 1

Reactor experienced a trip due to an instrument malfunction. RCS sample was taken 3 hours following the trip and was counted 1 hour after obtaining the sample. The following are the sample results.

RCS temperature	400°F
Sample temperature	100°F
I-131	1.3 μCi/cc
I-133	0.25μCi/cc
Xe-133	24 μCi/cc

Adjust RCS concentration for decay. (Step 3.3)

$$\text{I-131} = \frac{1.3}{e^{-(3.59\ \text{E-3})\times 4}} = 1.32\ \mu\text{Ci/cc}$$

$$\text{I-133} = \frac{0.25}{e^{-(3.41\ \text{E-2})\times 4}} = 0.29\ \mu\text{Ci/cc}$$

$$\text{Xe-133} = \frac{24}{e^{-(5.48\ \text{E-3})\times 4}} = 24.5\ \mu\text{Ci/cc}$$

Comparison of these concentrations with RCS concentrations show that only I-131 concentration is high and I-133 and Xe-133 concentrations are normal. This indicates iodine spiking. No apparent fuel damage. This is also confirmed by no abnormal thermocouple readings.

Example 2

Reactor experienced a trip due to loss of feedwater flow. During cooldown it was noticed that subcooling margin was lost. Few core exit thermocouples in the center region recorded temperatures above 700°F and the RCS pressure was 1900 PSIG. The transient was brought under control. The containment radiation monitors and sump level instrumentation indicate that there is no breach of RCS boundary. The RCS samples were taken 3 hours from the time subcooling margin was lost and counted in the following 1 hour. The following are the sample results.

I-131	15.2	μCi/cc
I-133	3.2	μCi/cc
Xe-133	97	μCi/cc
Kr-88	0.4	μCi/cc
Ba-140	0.04	μCi/cc

RCS temperature = 400°F Sample temperature = 100°F

REC concentration after correcting for decay. (Step 3.3)

$$\text{I-131} = \frac{15.2}{e^{-(3.59\ E\text{-}3)\times 4}} = 15.4\ \mu\text{Ci/cc}$$

I-133 = 3.7 μCi/cc

Xe-133 = 99 μCi/cc

Kr-88 = 1.1 μCi/cc

Ba-140 = 4.04E-2 μCi/cc

Comparison of the sample results with radiochemistry samples prior to transient indicate that the concentrations for all isotopes are higher than normal. This indicates some cladding damage.

The thermocouple readings and corresponding RCS pressure indicate the cladding temperatures are below 1400°F. The cladding failures may not be the result of clad overheating.

Calculation of Cladding Failures

Activity in the RCS (Step 3.5)

RCS density correction factor at 400°F = 0.865 (from Table 3)

Ai (RCS) for I-131 = $15.4 \times 83028 \times 3785 \times 0.865 \times 10^{-6} = 4.19E3$ curies.

Since the post transient concentration is significantly greater than the RCS concentration prior to the transient, calculation of Ni is not performed.

Isotope	Total RCS activity µCi/cc After transient	Prior to transient	Activity released during transient Ci Step 3.5
I-131	15.4	0.3	4.19E3
I-133	3.7	0.26	1.01E3
Xe-133	99	23	2.69E4
Kr-88	1.1	0.19	2.99E2
Ba-140	0.0404	5.2E-4	1.1E1

Estimation of Power Correction Factors (Step 3.6)

$$PCF\ (I\text{-}131) = (2360\ (1-e^{-0.086 \times 70})\ e^{-0.086 \times 30}$$
$$+2772\ (1-e^{-0.086 \times 20})\ e^{-0.086 \times 10}$$
$$+2360\ (1-e^{-0.086 \times 6})\ e^{-0.086 \times 4}$$
$$+1940\ (1-e^{-0.086 \times 4}))/2772\ (1-e^{-0.086 \times 100})$$
$$= 0.86$$

Using alternate formula

$$PCF(I\text{-}131) = \frac{2772 \times 20 + 2360 \times 6 + 1940 \times 4}{2772 \times 30} = 0.93$$

Because the difference is approximately 7% in order to save computation time, alternate formula could be used.

Isotope	PCF
I-131	0.93
I-133	0.70
Xe-133	0.93
Kr-88	0.70
Ba-140	0.93

Calculation of core fraction released (Step 3.7)

$$I\text{-}131 = \frac{4.19E3}{7.44E7 \times 0.93} = 6.1E\text{-}5$$

Isotope	Core inventory at full power (Table) Curies	Activity released to RCS Curies	Core fraction released
I-131	7.44E7	4.19E3	6.1E-5
I-133	1.44E8	1.01E3	1E-5
Xe-133	1.43E8	2.69E4	2E-4
Kr-88	6.84E7	2.99E2	6.2E-6
Ba-140	1.44E8	1.1E1	8.2E-8

Estimation Of Failed Fuel Fraction (from Figure 1) (Step 3.8)

I-131 - 0.45%

I-133 - 0.7%

Xe-133 - 0.5%

Kr-88 - 0.65%

Ba-140 - 0.45%

Based on the above, approximately 0.4 to 0.7% cladding failures have occured during the transient.

Example 3

In this example, it is assumed that plant experienced an accident in which significant quantity of water was accumulated in the sump and the containment high range radiation monitors indicate high radiation in the containment. Fifty percent of the core exit thermocouples indicated temperatures greater than 950°F at a corresponding RCS pressure of 1400 PSIG. The samples from RCS, sump and containment are taken at three hours following the accident and counted within one hour. The following are sample results.

	RCS μCi/cc	Sump μCi/cc	Containment μCi/cc
I-131	1.2E4	3.84E3	9E - 2
I-133	1.4E4	4.6E3	1E - 1
Xe-133	-	-	1.6E2
Kr-88	-	-	1.92E1
Ba-140	57	12	

The following are sample conditions.

Sample	Source Parameters	Sample Parameters
RCS	Temp 300°F	Temp 100°F
Sump	Temp 200°F	Temp 100°F
Containment	Press 20 psia Temp 200°F	Press 14.7 psia Temp 100°F

Containment radiation monitors indicate 5E4 R/hr at one hour following the accident.

The containment hydrogen monitors indicate hydrogen concentration of 5% by volume.

Estimated volume of water in the sump is 250,000 gallons.

Correct the sample for decay

$$\text{I-131} = \frac{1.2\text{E}4}{e^{-3.59\ \text{E-}3 \times 4}} = 1.22\text{E}4\ \mu\text{Ci/cc}$$

	RCS (μCi/cc)	Sump (μCi/cc)	Containment (μCi/cc)
I-131	1.22E4	3.9E3	9.14E-2
I-133	1.6E4	5.27E3	1.15E-1
Xe-133			1.63E2
Kr-88			5.19E1
Ba-140	5.75E1	1.21E1	

Presence of high Ba-140 concentrations indicate that some portion of the core might have overheated or melted.

Calculate total activity released during accident for each isotope (Step 3.5)

I-131 RCS = $1.22E4 \times 83028 \times 3785 \times 0.924 \times 10^{-6}$ = 3.54E6 Ci

I-131 Sump = $3.9E3 \times 250{,}000 \times 3785 \times 0.97 \times 10^{-6}$ = 3.58E6 Ci

I-131 CB = $9.14E\text{-}2 \times 8.01 \times E4 \quad \frac{20(100+460)}{14.7(200+460)}$ = 8.47E3 Ci

ISOTOPE	RCS	SUMP	CB	TOTAL
I-131	3.54E6	3.58E6	8.47E3	7.13E6
I-133	4.65E6	4.84E6	1.07E4	9.50E6
Xe133			1.51E7	1.51E7
Kr-88			4.8E6	4.8E6
Ba-140	1.67E4	1.11E4		2.78E4

Since post-accident concentrations are significantly larger than normal concentrations can be ignored.

Calculate core release fractions (Step 3.7) and % fuel failures (Step 3.8)

$$\text{I-131} = \frac{7.13E6}{0.93 \times 7.44E7} = 0.1$$

Isotope	Fraction Released	Cladding Failure %	Fuel Over Heat %	Fuel Melt %
I-131	0.10	100	16 to 25	10 to 15
I-133	0.09	100	14 to 22	9 to 13
Xe-133	0.12	100	18 to 30	12 to 20
Kr-88	0.10	100	16 to 25	10 to 15
Ba-140	2.1E-4	100	11 to 20	<0.21

Because of the overlap in the release fractions and small fraction of Ba-140, overheat or melt would dominate the release. Engineering judgement is used to apportion the releases to each type of failure mechanism. Based on Ba-140 release fraction, it can be seen that this release corresponds to less than 0.1% to 0.2% of fuel melting or less than 11% to 20% fuel overheating. Based on core exit thermocouple readings and Attachment 4, it can be seen that the clad temperature did not exceed 1800 °F. However, cladding temperatures exceeded cladding failure temperatures and approached temperatures that may result in fuel overheating. Based on the hydrogen concentration measurements, it can be determined that 44% of zirconium in the core has undergone metal water reaction. This indicates significant overheating of core. Based on this available information, a best estimate for core condition can be made.

1. Major fuel cladding damage (greater than 50%) occurred. (Indicators, release fraction, core exit thermocouple, containment hydrogen concentration).

2. Ten to 20% of fuel overheated (indicators Ba-140 release fraction, core exit thermocouple readings).

3. No melting of fuel (Ba-140 release fraction, core exit thermocouples).

<u>Core damage estimation from containment high range monitor</u>

The dose rate of 5E4 R/hr at one hour is in region between curves D and E.

at t = 1 hr. D = 6E4 R/hr

Fuel inventory released $\frac{5 \times 10^4}{6 \times 10^4} \times 10\% = 8.3\%$

clad failures $\frac{5 \times 10^4}{6 \times 10^4} \times 100\% = 83\%$

Use of Radiation Monitors will provide a quick estimate on fuel condition.

SAMPLE ANALYSIS RESULTS FORM

NOTE: Make sure the gaseous activity of RCS represents the results of liquid phase and gaseous phase of RCS sample.

		RCS	Sump	Containment	Other
Sample	Date				
	Time				
	Temperature F				
	Pressure PSIA				
System	Temperature F				
	Pressure PSIA				
Activity μCi/cc					
	Xe-133				
	Kr-88				
	I-131				
	I-133				
	Ba-140				
	Other isotopes				
Sample activities are adjusted for decay to					
	Date				
	Time				

PASS SAMPLE CORE DAMAGE ASSESSMENT FORM

Prepared By: ______________________

Checked By: ______________________

Date: ____________ Time: ____________

Isotope	RCS Activity Prior to Event (μCi/cc)	Sample Activity (μCi/cc)	Adjusted Sample Activity (μCi/cc)	Sample Activity > Normal? (Yes/No)	Activity Released During Event (Ci)	Power Correction Factor	Core Fraction Released (All Samples)	Core Damage Evaluation: Cladding Failures (%)	Core Damage Evaluation: Fuel Overheat (%)	Core Damage Evaluation: Fuel Melt (%)	Other Data and Comments

Summary of Core Damage Assessment Circle As Applicable

No Damage			
Cladding Damage	<10%	10-50%	>50%
Fuel Overheat	<10%	10-50%	>50%
Fuel Melt	<10%	10-50%	>50%