Attachment to IPN-84-34

Final Core Damage Assessment Procedure

New York Power Authority
Indian Point 3 Nuclear Power Plant
Docket No. 50-286

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CORE DAMAGE ASSESSMENT

1.0 PURPOSE

To provide a methodology to determine the extent of core damage following an accident. The assessment is based on radionuclide concentrations and other parameters.

2.0 DISCUSSION

2.1 The core damage procedure is based on quantitative and qualitative assessments of various plant parameters, some of which are interrelated.

2.1.1 Radiation monitors in the VC: This is a gross but immediately available measurement of noble gases released.

2.1.2 Radioactivity released from the core:

- o Measured in RCS, VC atmosphere, VC sump.
- o Correct by power history, decay, etc.
- o Evaluate versus expected radioactivity released for clad damage, fuel overheat, fuel melt.

2.1.3 Hydrogen in containment: A measure of the amount of the zirconium water reaction from the fuel cladding.

2.1.4 Core Exit Thermocouples and Reactor Vessel Level Instrumentation: When available, used to determine whether the core has been uncovered and what type of fuel damage may have occurred.

2.2 Qualitative Assessment of Core Damage

Attachment 1, Qualitative Assessment of Core Damage, should be used in conjunction with the quantitative assessment of core damage which follows in section 3.

2.3 Limitations of This Procedure

- o This procedure is an approximate method and may give some conflicting results. Engineering judgement must be used throughout.
- o Some areas for potential errors are:
  - plateout of samples in containment or in sample lines
  - gamma spectroscopy of highly radioactive samples
  - estimates of ECCS water volumes or sump volumes
  - calculations of core inventories
  - effect of multiple precursors in the parent-daughter decay chains and unequal release fractions.

- The uncertainties are such that core damage estimates using this methodology are sufficient only to establish major categories of fuel damage. This categorization, with confirmation will require extensive additional analysis for some several days past the accident date.

3.0 PROCEDURE: QUANTITATIVE CALCULATIONS OF CORE DAMAGE - can be performed by following the attachments as worksheets.

3.1 Data Collection

- Using Attachment 2, record all appropriate data concerning RCS, VC sump, and VC atmosphere sampling.

- Power History

  - Record EFPD and calendar days of operation on Attachment 4, part 1.
  - If reactor has been at steady state power (± 10% of average power level) for 4 days or more, record power level on Attachment 4, part 2.
  - If reactor has not been at steady state power for at least 30 days, use Attachment 4A to record Power History over the last 30 days.

- Record sample results from RCS, VC Sump, VC atmosphere on Attachments 5 and 6.

3.2 Power History Correction Factor (PCF)

The inventories of fission products shown in Attachment 3 are for end of core life 100% power steady state operating conditions. This must be corrected for actual power history.

3.2.1 Steady state power (± 10%) prior to shutdown:

- Long-lived nuclide correction is calculated in Attachment 4, Part 3.
- Short-lived and medium-lived nuclide corrections are calculated in Attachment 4, Part 4.

3.2.2 Transient power history:

- Long-lived nuclides: transient power history not applicable, use Attachment 4 to calculate power correction factor.
- Short and medium lived nuclides - correct each nuclide separately using Attachment 4A as a calculation sheet.

3.3 Chemistry Sample Corrections

Samples of RCS, VC Sump, and VC atmosphere must be corrected using Attachment 5 (Water Samples) and Attachment 6 (VC Atmosphere).

3.3.1 Back-Decay Correction CF(bd)

- This factor is used to correct the sample result back to the time of reactor shutdown.

- Nuclides that are daughters in a chain must be accounted for by following the calculations in Attachment 5A.

- The Chemistry computer has the capability to back-decay nuclides. Ensure that this correction is not applied twice. The daughters as discussed above should not be back-decayed by the Chemistry computer.

3.3.2 Temperature-Pressure Correction CF(tp)

- This factor is used to account for the differences in temperature and pressure between the sample and the sampled system (e.g., RCS, VC air).

- Water samples are corrected for temperature only.

- Air samples are corrected for both temperature and pressure.

3.4 Calculation of Percent Core Damage

The calculation of percent core damage involves 3 basic steps:

- Determination of activity released from the core
- Determination of the power corrected activity inventory
- Comparison of the actual activity released to the expected inventory.

This calculation is performed for clad damage, fuel overheat and fuel melt using Attachments 7, 7A, 7B, 7C, and 7D.

3.4.1 Calculate total activity released by radionuclide

- Using Attachment 7 as a calculation sheet, add the activity from RCS, Containment Sump, and Containment atmosphere to determine total activity released from the core.

- Values for activity concentrations are obtained from Attachments 5 and 6, and should have been previously corrected for decay, dilution, temperature, pressure, etc. in accordance with Attachment 5 and 6.

3.4.2 Calculate activity normally present in the RCS during operations

- Using Attachment 7A as a calculation sheet, determine the amount of each nuclide present during normal operations.

- This activity is subtracted from the total amount released from the core.

- This calculation is only used in assessing clad damage. For other types of fuel damage, it is an insignificant fraction of the activity.

3.4.3 Calculate Percent Fuel Damage

- o Use Attachments 7B (Clad Damage), 7C (Fuel Overheat), and 7D (Fuel Melt) as calculation sheets.
- o Correct the nuclide inventories from Attachment 3 using the previously developed Power Correction Factors.
- o Compare the activity released (Attachments 7, 7A) to the corrected inventories to obtain percent fuel damage.

3.5 Assessment of Core Damage using Activity Released

Assessment of core damage involves determining:

- o The type of core damage - clad damage
  - fuel overheat
  - fuel melt
- o The amount of core damage - 0 to 100% in each of the above categories.

3.5.1 Comparison with expected inventories released:

Attachment 3 lists the nuclides associated with the 3 types of fuel damage and the amount of activity expected to be released for 100% clad damage, 100% fuel overheat, and 100% fuel melt.

3.5.2 The nuclides released are characteristic of the type of damage, as are the ratios of nuclides.

3.5.3 Clad Damage

- o Nuclides associated with cladding damage are primarily the medium-lived and long-lived noble gases and iodines.
- o Attachment 7B contains the calculated percent clad damage.
- o The ratios of the noble gases to Xe-133 (and Iodines to I-131) in the gap differ from the ratios in the fuel itself. The ratios are shown in Attachment 3 and can help to ascertain whether the release was from the fuel (fuel overheat or melt) or from the gap (clad damage).
- o RCS pressure, temperature, and power transients may result in Idoine spiking, where the Iodine concentrations in the RCS increase sharply. This is not indicative of cladding failure, but should be considered so that it is not confused with clad damage. Attachment 8 provides an estimate of the total I-131 release that might be expected during an iodine spike.

- Clad rupture is dependent on fuel temperature and RCS pressure where higher RCS pressures will delay clad rupture.

3.5.4 Fuel Overheat

- Moderately volatile fission products are released during fuel overheat conditions, including cesium, ruthenium, and tellurium in addition to the more volatile noble gases and iodines. Lesser amounts of barium and strontium are also released.

- Attachment 7C provides the calculated percent fuel overheat.

- The use of the isotopic ratios listed in Attachment 3 can be used to determine the source of the noble gases and iodines.

3.5.6 Fuel Melt

- Fuel pellet melting leads to rapid release of noble gases, iodines, bromines, and cesiums remaining after fuel overheat.

- Significant release of the Strontium, barium-lanthanum chemical groups is the most distinguishing feature of fuel melt conditions.

- Attachment 7D provides the calculated percent fuel melt.

- The use of isotopic ratios listed in Attachment 3 can be used to determine the source of the noble gases and iodines.

3.5.7 Non-Uniform Core Damage

- The above evaluations address an assumed uniform distribution of core damage. The degree of damage may vary within the core, and this should be considered in explaining any conflicting data.

4.0 AUXILIARY INDICATORS

There are plant indicators monitored during an accident which can provide verification of the initial estimate of core damage based on the radionuclide analysis. The plant indicators include containment hydrogen concentration, core exit thermocouple temperatures, reactor vessel water level, and containment radiation level.

4.1 Containment Hydrogen Concentration

- An accident in which the core is uncovered and the fuel rods are exposed to steam may result in the reaction of the zirconium of the cladding with the steam which produces hydrogen. It is assumed that all of the hydrogen that is produced is released to the containment atmosphere.

- The hydrogen dissolved in the primary system during normal operation contributes an insignificant amount of the total hydrogen released to the containment. The hydrogen recombiners will not have a significant effect on a zirconium - steam reaction in the case of severe core degredation.

- The percentage of zirconium water reaction does not equal the percentage of clad damaged but it does provide a qualitative verification of the extent of clad damage estimated from the radionuclide analysis.

- Attachment 9 shows the relationship between the hydrogen concentration and the percentage of zirconium water reaction.

4.2 Core Exit Temperatures and Reactor Vessel Water Levels

Core exit thermocouples (CETC) and the Reactor Vessel Level Indication System (when available) (RVLIS) readings can be used for verification of core damage estimates in the following ways.

- Due to the heat transfer mechanisms between the fuel rods, steam, and thermocouples, the highest clad temperature will be higher than the CETC readings. Therefore, if thermocouples read greater than 1300°F, clad failure may have occurred. 1300°F is the lower limit for cladding failures.

- If any RCPs are running, the CETCs will be good indicators of clad temperatures and no core damage should occur since the forced flow of the steam-water mixture will adequately cool the core.

- No generalized core damage can occur if the core has not uncovered. So if RVLIS full range indicates that the collapsed liquid level has never been below the top of the core and no CETC has indicated temperatures corresponding to superheated steam at the corresponding RCS pressure, then no generalized core damage has occurred.

- Attachment 10 provides information on types of damage to fuel at increasing temperatures.

4.3 Containment Radiation Levels

- R-25 and R-26 are located just above the 95' VC and can be used as a gross indication of activity (primarily noble gases) in the containment atmosphere.

- R-25 and R-26 would be expected to read approximately the same value if there were noble gases dispersed in containment.

- Attachment 11 provides data on expected radiation levels for clad damage, fuel overheat and fuel melt conditions.

5.0 REFERENCES

References are listed in Attachment 12.

LIST OF ATTACHMENTS

ATTACHMENT	TITLE
1	Qualitative Assessment of Core Damage
2	Sampling Data for Core Damage Calculations
3	Core Release Inventories of Characteristic Fission Products
4	Power Correction for Core Inventories - Steady State
4A	Power Correction for Core Inventories - Transient Conditions
5	Water Sample Data and Calculations
5A	Parent-Daughter Decay Correction
6	VC Atmosphere Sample Data and Calculations
7	Calculation of Total Activity Released from Core
7A	Calculation of Activity Present During Normal Operations
7B	Calculation of Percent Clad Damage
7C	Calculation of Percent Fuel Overheat
7D	Calculation of Percent Fuel Melt
8	Expected Iodine Spike vs. Normal Iodine Activity
9	VC Hydrogen Concentration vs. Zirconium-Water Reaction
10	Expected Fuel Damage Correlation with Fuel Rod Temperature
11	Expected Containment Radiation Levels Post-Accident (R-25/R-26)
12	References

QUALITATIVE ASSESSMENT OF CORE DAMAGE

	No Damage	Clad Damage	Fuel Overheat	Fuel Melt
Radiation Levels in VC* (R-25 and R-26)	1 R/hr (Minimum Readirg)	Up to 9200 R/hr	Up to 1.4E6 R/hr	up to 2.7E6 R/hr
% Hydrogen in VC**	0%	←	Up to 12.6%	→
Core Exit Thermocouples	~ 600°F	>1300°F, and check temp vs. press. for superheated steam		
RVLIS (if available)***	Full	Used in conjunction with CETC's to determine core uncover		
Expected Nuclides	Kr, Xe, I	Kr, Xe, I	Cs, Te	Sr, Ba, La, Pr

* time dependent R-25/R-26 readings can be found in Attachment 11.

** presence of Hydrogen is indicative of reaction of the cladding, but does not indicate whether fuel overheat or melt has occurred.

*** no generalized core damage can occur if the core remains covered.

SAMPLING DATA FOR CORE DAMAGE CALCULATIONS

Calculation No. _____

Current Date _____ Reactor Shutdown: Date _____

Current Time _____ Time _____

Sample and Media Data

	RCS	VC Atmosphere	VC Sump	Other
Sample No.	_____	_____	_____	_____
Date Of Sample	_____	_____	_____	_____
Time of Sample	_____	_____	_____	_____
Sample Temp (^{o}F)	_____ ^{o}F	_____ ^{o}F	_____ ^{o}F	_____
Sample Press (psia)	_____ psia	_____ psia	_____ psia	_____
System Temp (^{o}F)	_____ ^{o}F	_____ ^{o}F	_____ ^{o}F	_____
System Press (psia) or level (ft)	_____ psia	_____ psia	_____ ft	_____

Volume of ECCS dilution water _____ gallons

CORE RELEASE INVENTORIES OF CHARACTERISTIC FISSION PRODUCTS

Nuclide	Half-Life	Decay Constant (day^{-1})	Gap Release (Ci)	Fuel Overheat Release (Ci)	Fuel Melt Release (Ci)	Fuel Pellet** Activity Ratio	Gap** Activit Ratio
Clad Failure Nuclides							
Kr-85*	10.72yr	1.77E-4	1.6E4	9.0E5	1.5E6	.01	.11
Kr-87	76.3m	1.31E+1	3.1E3	1.8E7	3.0E7	.22	.022
Kr-88	2.84h	5.86E0	6.7E3	2.5E7	4.2E7	.29	.045
Xe-131m	11.84d	5.85E-2	7.5E2	2.8E5	4.7E5	.004	.004
Xe-133	5.245d	1.32E-1	1.5E5	8.8E7	1.5E8	1.0	1.0
I-131	8.04d	8.62E-2	2.4E5	4.3E7	7.2E7	1.0	1.0
I-133	20.8h	8.00E-1	1.6E5	8.8E7	1.5E8	2.1	.71
I-135	6.61h	2.52E0	8.3E4	7.9E7	1.3E8	1.9	.39
Fuel Overheat Nuclides							
Cs-137*	30.17y	6.3E-5	N/A	4.9E6	8.1E6	-	-
Te-129	69.6m	1.4E+1	N/A	1.5E7	2.4E7	-	-
Te-132	78.2h	2.1E-1	N/A	6.2E7	1.0E8	-	-
Fuel Melt Nuclides							
Ba-140	12.79d	5.4E-2	N/A	2.2E5	3.5E7	-	-
La-140	40.22h	4.1E-1	N/A	2.5E5	3.7E7	-	-
La-142	95.4m	1.1E+1	N/A	1.9E5	3.1E7	-	-
Pr-144	17.28m	5.8E+1	N/A	1.5E5	1.4E6	-	-

* Long-Lived Nuclides

** Ratio for Noble Gases is to Xe-133 = NG Isotope/Xe-133
Ratio for Iodines is to I-131 = I Isotope/I-131

POWER HISTORY CORRECTION FOR STEADY STATE POWER HISTORY

1. Data For Long-Lived Nuclides Power Correction Factor

	EFPD		Calendar Days	
Current Cycle	_____		Start Date of the Oldest Fuel Cycle	_____ (A)
Previous Cycle	_____		Current Date	_____ (B)
2 Cycles Previous	_____		Days Between (A) & (B)	_____
$\sum =$	_____			

2. Data if Plant Has Been At Steady-State Power (Within + 10% of Average Power Level)

Steady State Power Level (last 4 days) = _____ %

Steady State Power Level (last 30 days) = _____ %

3. Calculation of Long-Lived Power Correction Factor

	Nuclide	Half-Life	EFPD/Calendar Days
Long-Lived Nuclides	Kr-85	10.72y	
	Cs-137	30.17y	_____

4. Calculation of Short and Medium-Lived Power Correction Factor - (Steady State Operation)

	Nuclide	Half-Life	Steady State Power Level (%) (Last 4 days): P(4)	P(4)/100%
Short Lived Nuclides	Kr-87	76.3m		
	Kr-88	2.84h		
	I-133	20.8h		
	I-135	6.61h		
	Te-129	69.6m	_____	_____
	La-142	95.4m		
	Pr-144	17.28m		

	Nuclide	Half-Life	Steady State Power Level (%) (last 30 days: P(30)	P(30)/100%
Medium-Lived Nuclides	Xe-131m	11.84d		
	Xe-133	5.245d		
	I-131	8.04d	_____	_____
	Te-132	78.2h		
	Ba-140	12.79d		
	La-140	40.22h		

Note: Short Lived Power Correction Factor (PCF) = P(4)/100%
Medium Lived Power Correction Factor (PCF) = P(30)/100%
Long Lived Power Correction Factor (PCF) = EFPD/Calendar days

POWER HISTORY CORRECTION FOR NUCLIDE i
(Transient Power History)

Nuclide: _____

Half Life: _____

λ = _____ day^{-1} (from Attachment 3)

Period	P_j Power Level (%)	T_j Duration (days)	t_j (Days) Decay Time	$(1-e^{-\lambda_i T_j})$	$(e^{-\lambda t_j})$	$P_j(1-e^{-\lambda_i T_j})(e^{-\lambda t_j})$
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						

$\Sigma P_j(1-e^{-\lambda_i T_j})\ (e^{-\lambda_i t_j})$ = _____

p_j = steady reactor power level (percent)
λ_i = decay constant for isotope i (day^{-1})
T_j = time at power level P_j (days)
tj = time since end of T_j to reactor shutdown (days)

$$PCF_i = \frac{\Sigma P_j(1-e^{-\lambda_i T_j})(e^{-\lambda t_j})}{100\%} = _____$$

NOTE: Power history should cover the last 30 days or more

WATER SAMPLE DATA AND CALCULATIONS

Water Sample Type ☐ RCS ☐ VC Sump ☐ Other _____

Sample No. _____

(A) Nuclide	(B) Half-Life	(C) Decay Constant λ (hr^{-1})	(D) Reported uCi/cc	(E) Back-Decay Corr. Factor	(F) Temp Corr. Factor	(G) Corrected uCi/gram
Kr-85	10.72y	7.38 E-6				
Kr-87	76.3m	5.45 E-1				
Kr-88	2.84h	2.44 E-1				
*Xe-131m	11.84d	2.44 E-3				
*Xe-133	5.245d	5.51 E-3				
I-131	8.40d	3.59 E-3				
I-133	20.8h	3.33 E-2				
I-135	6.61h	1.05 E-1				
Cs-137	30.17y	2.62 E-6				
*Te-129	69.6m	5.98 E-1				
Te-132	78.2h	8.86 E-3				
Ba-140	12.79d	2.26 E-3				
*La-140	40.22h	1.72 E-2				
*La-142	95.4m	4.36 E-1				
*Pr-144	17.28m	2.41 E0				

Column E: Back-Decay Correction Factor = CF(bd) = $\frac{1}{e^{-\lambda t}} = e^{\lambda t}$

* Note: Nuclides marked with * are daughters in a decay chain. This must be taken into account in order to back-decay correct. Attachment 5A should be followed for those nuclides.

Column F: Temperature Correction Factor CF(t)

This factor converts uCi/cc to uCi/g
If temperature of the water is $<200^{\circ}F$, CF(t) = 1, and uCi/cc = uCi/g
If temperature of the water is $\geq 200^{\circ}F$, use the Table below to determine CF(t).

Column G: Corrected uCi/g = reported uCi/cc x CF(bd) x CF(t)

(G) = (D) x (E) x (F)

RCS Water Temperature	Temp. Con. Factor CF(t)
$\leq 150^{\circ}F$	1.0
$200^{\circ}F$	.97
$300^{\circ}F$	.92
$400^{\circ}F$	.86
$500^{\circ}F$	.79
$600^{\circ}F$	.68
$700^{\circ}F$	.44

PARENT-DAUGHTER DECAY CORRECTION

The Table on page 2 of this attachment lists the significant parent-daughter relationships. The decay scheme of the parent-daughter is described as follows:

$$Q_B(t) = K \frac{\lambda_B}{\lambda_B - \lambda_A} Q_A^o \left(e^{-\lambda_A t} - e^{-\lambda_B t}\right) + Q_B^o e^{-\lambda_B t}$$

where:

Q_A^o = 100% fuel melt inventory (Ci) of parent*

Q_B^o = 100% fuel melt inventory (Ci) of daughter*

$Q_B(t)$ = hypothetical daughter activity (Ci) at sample time

K = branching factor*

λ_A = parent decay constant, (hr^{-1})*

λ_B = daughter decay constant (hr^{-1})*

t = time period from shutdown to time of sample (hr)

1. Calculate the hypothetical daughter concentration, $Q_B(t)$ at the time of sampling assuming 100% fuel melt release of both parent and daughter activity.

2. Determine the fraction (Fr) of the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time.

$$Fr = \frac{Q_B^o \left(e^{-\lambda_B t}\right)}{Q_B(t)}$$

3. Calculate the amount of the measured sample specific activity associated with the decay of the daughter that was released.

   M_B = Fr x measure specific activity (uCi/gm or uCi/cc)

   where M_B = Measured activity of B

4. Use this value of M_B as the reported uCi/cc in column D of attachment 5 or 6 and continue with further corrections as necessary on Attachment 5 or 6.

* See page 2 of this attachment for data on affected nuclides.

Parent Nuclide	$\lambda_A (hr^{-1})$	Q_A^o	K	Daughter Nuclide	$\lambda_B (hr^{-1})$	Q_B^o	QB(t)	Fr	M_B^*
I-131	3.59E-3	7.2E7	.008	Xe-131m	2.44E-3	4.7E5	_____	_____	_____
I-133	3.33E-2	1.5E8	.976	Xe-133	5.51E-3	1.5E8	_____	_____	_____
Xe-133m	1.28E-2	2.1E7	1.0	Xe-133	5.51E-3	1.5E8	_____	_____	_____
Sb-129	.161	2.3E7	.827	Te-129	.598	2.4E7	_____	_____	_____
Te-129m	8.47E-4	5.8E6	.68	Te-129	.598	2.4E7	_____	_____	_____
Ba-140	2.26E-3	3.5E7	1.0	La-140	1.72E-2	3.7E7	_____	_____	_____
Ba-142	3.78	3.3E7	1.0	La-142	.436	3.1E7	_____	_____	_____
Ce-144	1.02E-4	1.3E6	1.0	Pr-144	2.41	1.4E6	_____	_____	_____

* M_B should be transferred to Attachment 5 or 6 into Column D, reported uCi/cc

VC ATMOSPHERE SAMPLE DATA AND CALCULATIONS

Sample No. _____

A	B	C	D	E	F	G
Nuclide	Half-Life	Decay Constant: $\lambda(hr^{-1})$	Reported uCi/cc	Back-Decay Corr. Factor	Temp/Press Corr. Factor	Corrected uCi/cc
Kr-85	10.72 yr.	7.38 E-6				
Kr-87	76.3m	5.45 E-1				
Kr-88	2.84h	2.44 E-1				
*Xe-131m	11.84d	2.44 E-3				
*Xe-133	5.245d	5.51 E-3				
I-131	8.04d	3.59 E-3				
I-133	20.8h	3.33 E-2				
I-135	6.61h	1.05 E-1				
Cs-137	30.17y	2.62 E-6				
*Te-129	69.6m	5.98 E-1				
Te-132	78.2h	8.86 E-3				
Ba-140	12.79d	2.26 E-3				
*La-140	40.22h	1.72 E-2				
*La-142	95.4m	4.36 E-1				
*Pr-144	17.28m	2.41 E0				

Column E: Back-Decay Correction Factor = CF(bd) = $\frac{1}{e^{-\lambda t}} = e^{\lambda t}$

* NOTE: Nuclides marked with * are daughters in a decay chain. This must be taken into account in order to back-decay correct. Attachment 5A should be followed for those nuclides.

Column F: Temperature/Pressure Correction Factor = CF(tp) = $\frac{P(a)}{P(s)} \times \frac{(T(s) + 460)}{(T(a) + 460)}$

T(a), P(a) = VC atmosphere temperature °F and pressure (psia)
T(s), P(s) = VC sample temperature °F and pressure (psia)

Column G: Corrected uCi/cc = reported uCi/cc x CF(bd) x CF(tp)

G = D x E x F

CALCULATION OF ACTIVITY RELEASED FROM CORE

Nuclide	RCS Corrected (uCi/gram)	x	RCS (grams*)	x10^6 =	RCS (Ci)	Sump Corrected (uCi/gram)	x	Sump (Grams)**	x10^6 =	Sump (Ci)	VC Atmos. Corrected (uCi/cc)	x 7.39E4*** =	VC atmos. (Ci)	Total Activity Σ RCS & Sump & VC atmos. (Ci)
Kr-85														
Kr-87														
Kr-88														
Xe-131m														
Xe-133														
I-131														
I-133														
I-135														
Cs-137														
Te-129														
Te-132														
Ba-140														
La-140														
La-142														
Pr-144														

* Normally 91,600 gal x 3785 cc/gal. x 1 gram/cc = 3.47 E8 grams

** Sump gallons (from sump level) x 3785 cc/gal. x 1 gram/cc = sump grams
This value should be approximately equal to ECCS volume added,

*** uCi/cc x 2.61 E6 cu. ft. x 2.83 E4 cc/cu ft. x 10^{-6} Ci/uCi = 7.39 E4

CALCULATION OF ACTIVITY PRESENT DURING NORMAL OPERATIONS

(A) Nuclide	(B) Normal Ops RCS Conc (uCi/cc)* x 320**	(C) RCS (Ci) Normal Ops	(D) Normal Ops VC Conc (uCi/cc)*x7.4E4***	(E) VC (Ci) Norm Ops	(F) Act. Released from Core (Att. 7)	(G)= F-C-E Corrected Ci Released from Core
Kr-85						
Kr-87						
Kr-88						
Xe-131m						
Xe-133						
I-131						
I-133						
I-135						

* Obtain from recent pre-shutdown RCS Sample
- available from chemistry or site reactor engineer
- if unavailable, use the following approximate values as a sum of the operation activity:

	Ci
Kr-85	12
Kr-87	12
Kr-88	20
Xe-131m	40
Xe-133	200
I-131	8
I-133	10
I-135	10

** 320 = 3.2 E8 cc RCS x 1E-6 Ci/uCi

*** 7.4E4 = 7.4 E10 cc in VC x 1E-6 Ci/uCi

The results in column G to be used in Attachment 7B

NOTE: Account for iodine spiking in accordance with Section 3.5.3 and Attachment 8, if necessary.

CALCULATION OF PERCENT CLAD DAMAGE

(A) Nuclide	(B) Uncorrected Clad Damage Inventory	(C) Power Correction Factor (Att. 4 or 4A)	(D)=(B)x(C) PCF Corrected Clad Damage Inv.	(E) Activity Released from Core (Ci)*	(F) (E/D) x 100% Perc. Clad Damage	(G) NG or Iodine Ratios**
Kr-85	1.6E4					
Kr-87	3.1E3					
Kr-88	6.7E3					
Xe-131m	7.5E2					
Xe-133	1.5E5					
I-131	2.4E5					
I-133	1.6E5					
I-135	8.3E4					

* From Attachment 7A

** Noble Gas Isotope / Xe-133 or Iodine Isotope / I-131 (Compare to Ratios in Attachment 3)

NOTE: The percent fuel damage values can only be considered as approximations. If the actual age of the fuel assembly(s) damaged and the power region in the core is different from the core average, (core average was used to develop the inventories in Column B) then the actual inventories in the fuel damaged could differ by a factor of 2-3. The calculated percent damage must be considered along with the isotopic ratios (Column G), presence of other nuclides, and other parameters as discussed elsewhere in this procedure.

CALCULATION OF PERCENT FUEL OVERHEAT

(A) Nuclide	(B) Uncorrected Fuel Overheat Release Inv. (Ci)	(C) Power Correction Factor (Att. 4 or 4A)	(D)=(B)x(C) PCF Corrected Fuel Overheat Inventory (Ci)	(E) Activity Released from Core (Ci)*	(F) (E/D x 100%) Percent Fuel Overheat	(G) NG or I ratios**
Kr-85	9.0E5					
Kr-87	1.8E7					
Kr-88	2.5E7					
Xe-131m	2.8E5					
Xe-133	8.8E7					
I-131	4.3E7					
I-133	8.8E7					
I-135	7.9E7					
Cs-137	4.9E6					N/A
Te-129	1.5E7					N/A
Te-132	6.2E7					N/A
Ba-140	2.2E5					N/A
La-140	2.5E5					N/A
La-142	1.9E5					N/A
Pr-144	1.5E5					N/A

* From Attachment 7

** $\frac{\text{Noble Gas Isotope}}{\text{Xe-133}}$ or $\frac{\text{Iodine Isotope}}{\text{I-131}}$ (Compare to ratio in Attachment 3)

NOTE: The percent fuel damage values can only be considered as approximations. If the actual age of the fuel assembly(s) damaged and the power region in the core is different from the core average (core average was used to develop the inventories in Column B) then the actual inventories in the fuel damaged could differ by 20 - 30%. The calculated percent damage must be considered along with the isotopic ratios (Column G), presence of other nuclides, and other parameters as discussed elsewhere in this procedure.

CALCULATION OF PERCENT FUEL MELT

(A) Nuclide	(B) Uncorrected Fuel Melt Release Inv. (Ci)	(C) Power Correction Factor (Att. 4 or 4A)	(D)=(B)x(C) PCF Corrected Fuel Melt Inventory (Ci)	(E) Activity Released from Core (Ci)*	(F) (E/D x 100%) Percent Fuel Melt	(G) NG or I ratios**
Kr-85	1.5E6					
Kr-87	3.0E7					
Kr-88	4.2E7					
Xe-131m	4.7E5					
Xe-133	1.5E8					
I-131	7.2E7					
I-133	1.5E8					
I-135	1.3E8					
Cs-137	8.1E6					N/A
Te-129	2.4E7					N/A
Te-132	1.0E8					N/A
Ba-140	3.5E7					N/A
La-140	3.7E7					N/A
La-142	3.1E7					N/A
Pr-144	1.4E6					N/A

* From Attachment 7

** Noble Gas Isotope / Xe-133 or Iodine Isotope / I-131 (compare to ratios in Attachment 3)

NOTE: The percent fuel damage values can only be considered as approximations. If the actual age of the fuel assembly(s) damaged and the power region in the core is different from the core average (core average was used to develop the inventories in column B) then the actual inventories in the fuel damaged could differ by 30-40%. The calculated percent damage must be considered along with the isotopic ratios (column G), presence of other nuclides, and other parameters as discussed elsewhere in this procedure.

EXPECTED IODINE SPIKE VS. NORMAL IODINE ACTIVITY

I-131 uCi/gram*	Average I-131 Release (Curies)	Maximum I-131 Release (Curies)
0.5 - 1.0	3400	6500
0.1 - 0.5	380	950
.01 - 0.1	200	650
.001 - .01	100	300
$<$.001	2	10

* Normal operating I-131 specific activity in RCS

VC HYDROGEN CONCENTRATION VS. ZIRCONIUM-WATER REACTION

Percent Zirconium Water Reaction	Hydrogen Concentration in VC (volume %)
10	1.3%
20	2.5%
30	3.8%
40	5.0%
50	6.3%
60	7.5%
70	8.8%
80	10.0%
90	11.3%
100	12.6%

EXPECTED FUEL DAMAGE CORRELATION WITH FUEL ROD TEMPERATURE

Fuel Damage	Temperature °F*
No Damage	<1300
Clad Damage	1300 - 2000
Ballooning of zircaloy cladding	>1300
Burst of zircaloy cladding	1300 - 2000
Oxidation of cladding and hydrogen generation	>1600
Fuel Overtemperature	2000 - 3450
Fission product fuel lattice mobility	2000 - 2550
Grain boundary diffusion release of fission products	2450 - 3450
Fuel Melt	>3450**
Dissolution and liquefaction of UO_2 in the Zircaloy - ZrO_2 eutectic	>3450**
Melting of remaining UO_2	5100**

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* These temperatures are material property characteristics and are non-specific with respect to locations within the fuel and/or fuel cladding.

** Core Exit Thermocouple are not valid over 3000°F

NOTE: When narrow range thermocouple readings go off-scale (as indicated by an asterisk on the thermocouple map), use the wide range readings.

EXPECTED CONTAINMENT RADIATION LEVELS POST-ACCIDENT (R-25/R-26)

Time After Shutdown	R/hr For 100% Clad Damage	R/hr For 100% Fuel Overheat	R/hr for 100% Fuel Melt
0	9200	1.4 E6	2.7 E6
1 hr.	7500	1.0 E6	2.0 E6
2 hr.	6300	7.5 E5	1.5 E6
4 hr.	4800	4.8 E5	9.6 E5
8 hr.	3400	2.6 E5	5.1 E5
12 hr.	2700	1.7 E5	3.3 E5
24 hr.	1900	8.0 E4	1.6 E5
48 hr.	1400	4.2 E4	8.3 E4
4 days	960	2.4 E4	4.8 E4
7 days	680	1.6 E4	3.2 E4
14 days	340	8.0 E3	1.6 E4
30 days	83	1.7 E3	3.4 E3

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