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 February 28, 1985

ØCANØ28513

Director of Nuclear Reactor Regulation
 ATTN: Mr. J. F. Stolz, Chief
 Operating Reactors Branch #4
 Division of Licensing
 U. S. Nuclear Regulatory Commission
 Washington, DC 20555

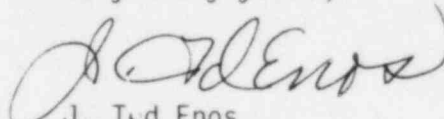
Director of Nuclear Reactor Regulation
 ATTN: Mr. James R. Miller, Chief
 Operating Reactors Branch #3
 Division of Licensing
 U. S. Nuclear Regulatory Commission
 Washington, DC 20555

SUBJECT: Arkansas Nuclear One - Units 1 & 2
 Docket Nos. 50-313 and 50-368
 License Nos. DPR-51 and NPF-6
 NUREG-0737 Item II.B.3, Methodology
for Estimating Core Damage

Gentlemen:

In our letter of January 24, 1984 (ØCANØ18413) AP&L committed to develop a methodology to estimate core damage based upon post accident sampling system (PASS) readings. This estimate will be used for informational purposes only and will not be used as input to emergency planning or other procedures. In your letter of August 27, 1984 (ØCANØ88425) you requested that AP&L submit this methodology by February 28, 1985 for your review. Therefore, our methodology for estimating core damage is attached.

Very truly yours,


 J. Ted Enos
 Manager, Licensing

JTE:MCS:ds

Attachment

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A. PURPOSE

This method is to be followed in order to estimate the type and degree of reactor core damage at Arkansas Nuclear One for plant conditions in which fuel failure is suspected.

B. REFERENCES

- 1) U.S. Nuclear Regulatory Commission - "Post-Accident Sampling Guide for Preparation of a Procedure to Estimate Core Damage"
- 2) U.S. Nuclear Regulatory Commission - NUREG-0737, "Clarification of TMI Action Plan Requirements", Item II.B.3, "Post-Accident Sampling Capability", November 1980
- 3) U.S. Nuclear Regulatory Commission - NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents", June 1981
- 4) U.S. Nuclear Regulatory Commission - NUREG/CR-1237, "Best-Estimate LOCA Radiation Signature", January 1980
- 5) AP&L Letter ØCANØ584Ø8 to J. Stolz, "Interim Guidelines for Assessing Core Damage at ANO", May 1984
- 6) Fort Calhoun Station Unit-1 Operating Instructions - OI-PAP-6, "Procedure to Estimate Core Damage"
- 7) Babcock and Wilcox Company - "Operator training - Degraded Core Recognition and Mitigation", June 1981

C. DEFINITIONS

Core damage or fuel damage is defined as the progressive failure of the two material boundaries which normally prevent the release of radioactive fission products into the RCS - the fuel pellet matrix and the fuel rod cladding. This method provides for the differentiation between the types and degrees of core damage shown in Table C.1.

Table C.1. Type and Degree of Core Damage

<u>Type</u>	<u>Degree</u>	<u>% Failure</u>
No Damage	NA	NA
Cladding Failure	Initial	<10%
	Intermediate	10% - 50%
	Major	>50%
Fuel Pellet Overheat	Intermediate	<50%
	Major	>50%
Fuel Pellet Melt	Initial	<10%
	Intermediate	10% - 50%
	Major	>50%

Source inventory is defined as the total quantity of each fission product isotope present in either source -- the fuel pellet or the fuel rod gas gap. Equilibrium source inventories for both ANO-1 and ANO-2 were calculated using the ORIGEN-2 computer code assuming 100% full power operation and a burnup of 26,000 MWD/MTU. For the case in which steady-state, full-power operation has not been adequately maintained, corrections to the equilibrium source inventory will be applied.

Cladding failure is defined as the rupture of the fuel rod zircaloy cladding when the internal rod gas pressure exceeds RCS pressure and the cladding yield strength is reduced due to elevated temperatures. Cladding failure temperatures depend upon several factors; rate of temperature rise, internal gas pressure and cladding physical and mechanical properties. For the purpose of this method, cladding failure temperatures range from 1100 to 2000°F.

Fuel pellet overheat is defined as the condition in which fission products trapped within the fuel pellet are released at an accelerated rate due to increasing temperature. The release mechanism is driven by the formation, swelling, and coalescence of bubbles of fission gases. At the temperature involved (>1600°F), the bubbles probably include vaporized cesium and iodine species as well as the noble gases. The expanded bubbles work to mechanically separate the UO₂ grains allowing the escape of gases and vapor.³ Fuel overheating temperatures typically range from 1600 to 3600°F cladding temperature⁶.

Fuel pellet melt is defined as the condition where the fuel has become molten - releasing a large fraction of the remaining core noble gases, iodines and cesiums and detectable amounts of Te-132, Ba-140, and Ru-103. For the purpose of this method, the temperature range for fuel pellet melting is 3600 to 5400°F.

D. RESPONSIBILITIES

- 1) Nuclear Fuel or Nuclear Support Supervisor: The Nuclear Fuel or Nuclear Support Supervisor is responsible for initiating and directing the performance of this method during accident (or other) conditions which indicate the possibility of core damage.
- 2) Radiochemistry: Radiochemistry personnel are responsible for operating the Post Accident Sampling System to obtain the results necessary for performance of this method.
- 3) Nuclear Fuel and Nuclear Support: Nuclear Fuel and Nuclear Support are responsible for maintaining this method and for aiding in collecting data and evaluating the results of this method.

E. PREREQUISITES

- 1) Post Accident Sampling System is operable - isotopic activities are available from the following locations (if applicable):
 - a) Reactor Coolant (RC) System
 - b) Containment Atmosphere (CA)
 - c) Containment Sump (CS)
- 2) Hydrogen concentration in the primary coolant sample and containment atmosphere is available.
- 3) Core exit thermocouple readings are available from plant computer.
- 4) Containment pressure and temperature are known.
- 5) Sample pressure and temperature are known.
- 6) Level in reactor building sump is known: Unit 1 - The containment building sump contains 26.6 gallons per % level indication. The sump level detection system indicates water levels in %. Unit 2 - The containment building sump contains 39 gallons per % level indication. The sump level detection system indicates water levels in %.
- 7) Previous Power history is known.
- 8) Equilibrium Fission Product (FP) inventories are known.

F. PRECAUTIONS AND LIMITATIONS

General

The assessment of core damage obtained using this method is only an estimate. The techniques employed in this method are accurate only to locate the core condition within one or more of the categories of core damage given in Table C.1.

- 1) The following precautions apply to the analysis of measured F.P. specific activities:

For the purpose of this method, it is assumed that samples taken from the RCS hot leg are representative of a homogeneous system. No provisions are made for the sampling of the pressurizer steam space where fission gases may collect disproportionately during normal operation and under accident conditions.

The method relies upon samples taken from multiple locations inside the Containment Building to determine the total quantity of fission products available for release to the environment. The amount of fission products present at each sample location may be changing rapidly due to transient plant conditions. Therefore, it is required that the samples be obtained within a minimum time period and if possible under stabilized plant conditions. Samples obtained during rapidly changing plant conditions should not be weighed heavily into the assessment of core damage.

A number of factors influence the reliability of the chemistry samples, such as the inability to obtain representative samples due to incomplete mixing of the fluids, and equipment limitations. The accuracy achieved in the radiological analyses is also influenced by a number of factors. For example, samples may not result from a uniform distribution of the sample fluid and cooling or reactions may take place in the long sample lines. Hence, the results obtained may not be representative of plant conditions. To minimize these effects, multiple samples should be obtained over an extended time period from each location.

- 2) The following precautions apply to the analysis of hydrogen concentrations:

Hydrogen samples are taken from the containment atmosphere and the Reactor Coolant System Hot Leg. Those samples may contain a mixture of hydrogen generated by clad oxidation, radiolytic dissociation of water and oxidation of metals in the containment. Therefore, a hydrogen measurement is not a direct indicator of the amount of core clad oxidation.

Since measured hydrogen does not directly indicate the amount of core clad oxidation (core damage), the interpretation of hydrogen data employed in this method is based on the presence or absence of excess levels of hydrogen.

- 3) The following precautions apply to the analysis of Core Exit Thermocouple (CET) temperatures:

Severe core damage cannot be quantified by CET analysis alone. This method is designed to provide a weighted estimate of core damage based on all available indicators.

The relationship between the CET temperature and the cladding temperature varies with the core uncover scenario. Due to the lack of more detailed data, the correlation of clad temperature to CET temperature and system pressure developed by Babcock and Wilcox (Attachment A) is used in this estimate.

G. INITIAL CONDITIONS

This method is to be employed for analysis of radiochemistry, hydrogen, and core exit thermocouple data when it is determined that a plant accident with the potential for core damage has occurred. The following is a list of plant symptoms to assist in this determination. One or more of these symptoms may exist at or before the time the sample is obtained.

- 1) High alarm on the Containment Radiation Monitor
- 2) Reactor coolant sample indicates high activity level

- 3) Reactor Coolant System sub-cooling low or zero
- 4) Any other condition exists in which the operator suspects fuel failure

H. SUMMARY

The core damage estimates obtained using this method will be based on the comparison of measured fission product concentrations from various sample locations to the known fission product inventory of the core. The four types of core damage (no damage, cladding failure, fuel pellet overheat and fuel pellet melt) can be differentiated by the presence, absence or abundance of certain fission products and by other plant indicators (i.e., core exit thermocouples and containment or primary system hydrogen concentration).

METHOD

- I. Select appropriate isotopes for estimate
- II. Determine appropriate sample locations
- III. Obtain necessary data
 - A. Plant data
 - B. Radiochemistry data
- IV. Correct equilibrium Fission Product (F.P.) inventories to recent power history
- V. Correct measured sample activities to standard temperature and pressure (STP)
- VI. Correct measured sample activities at STP for decay since shutdown
- VII. Calculate key isotopic ratios to determine source of release
- VIII. Calculate total activity of isotopes available for release
- IX. Calculate total activity of isotopes released
- X. Interpret results

The following is a statement of the objectives and brief summary of each step in the method:

STEP I: To select the appropriate isotopes for use in the estimate. The isotopes, their source of release and selection criteria are given in Table I.A. The fundamental selection criteria for the fourteen isotopes given in the table were; their detectability in early and later points in transients, their radiological significance in regard to off-site doses, the sensitivity of detector instrumentation to them

and the experience in analyzing samples for them. Additionally, empirical data concerning the fractional release of these isotopes as well as procedures to estimate core damage utilizing them have been published.

STEP II: To select the appropriate sample locations for the estimate based on the accident scenario. The suggested sample locations for three major accident scenarios are given in Table II.A. These scenarios are adequate to define core damage within the types and degrees defined in Table C.1 and are representative of three fundamental scenarios in which core damage may occur.

STEP III: To record pertinent plant data as closely as possible to the time at which radiological samples are obtained from the PASS. The data include reactor shutdown time, previous power history, RC pressure and temperature, CET temperature, CA pressure and temperature, etc., which are necessary in accurately locating core damage as defined in Table C.1. A worksheet is provided in Step III of the method.

STEP IV: To correct the computer-generated equilibrium source inventories for recent reactor power history. Equations for the calculation of corrected inventories and a worksheet for recording them are provided in Step IV of the method. There are two types of isotopes provided for the estimate of core damage; Group 1 isotopes, which require approximately 30 days of reactor operation at power levels $\pm 10\%$ of some level to reach equilibrium, and group 2 isotopes, which require approximately 4 days of operation as above. It should be noted that severe core uncover scenarios cannot be quantified by Group 2 isotopes alone. In the event power has not been maintained as indicated above, a general power correction factor is provided.

STEP V: To obtain and correct the measured samples specific activities for standard temperature and pressure. Although the PASS automatically makes this correction, the calculational method is provided in Step V of the method for the case in which samples have not been corrected for STP. Table V.A is provided as a record of these specific activities.

STEP VI: To correct the specific activities obtained from Step V for decay since shutdown occurred. The PASS will automatically make this correction, if Radiochemistry personnel are advised to program the reactor shutdown time. In the event samples have not been corrected for decay since shutdown, the calculational method is provided. An additional calculation is necessary in order to account for the parent-daughter relationship of Iodine and Xenon.

STEP VII: To calculate isotopic ratios in order to determine whether the source of inventory release is the fuel pellet or the gas gap. Table VII.A is provided to record the indicated release source based on the ratios provided in Step VII. Additional ratios for normal coolant can be calculated using data obtained from samples taken under normal operating conditions and additional ratios for releases can be deduced from the release fractions of Attachment B.

STEP VIII: To calculate the total activities of the isotopes available for release from the fuel pellets and gas gaps. The activities available in the fuel rod gas gap are obtained by multiplying the nominal gap release fractions of Attachment B by the power corrected equilibrium source inventories from Step IV. The activities available in the fuel pellets are obtained by multiplying the nominal meltdown release fractions of Attachment B by the power corrected equilibrium source inventories from Step IV. A table is provided in Step VIII for available activities.

STEP IX: To calculate the activities which were released based on PASS samples. The released activities, in curies, are the sums of the activities released to each sample location. Equations for this calculation are provided in Step IX.

STEP X: To interpret the data and calculate an estimate of the type and degree of core damage. Guidelines for making the estimate of core damage are provided in Step X. Attachment C provides a flow-chart for the interpretation of data and the calculation of a core damage estimate.

I. APPROPRIATE ISOTOPES FOR CORE DAMAGE ESTIMATE

Table I.A. Isotopes, Source of Release and Selection Criteria

<u>Isotope</u>	<u>Source of Release</u>	<u>Selection Criteria</u>
Kr-85m	Gas Gap	-Indicators of all degrees of cladding failure. -When seen in quantities greater than source inventory times the upper limit gap release fractions of Attachment B Indicators of overheating and the source of release is the fuel pellet.
Kr-87	Gas Gap	
Kr-88	Gas Gap	
Xe-131m	Gas Gap	
Xe-133	Gas Gap	
I-131	Gas Gap	
I-133	Gas Gap	
I-135	Gas Gap	
Te-132	Fuel Pellet	
Ba-140	Fuel Pellet	
Ru-103	Fuel Pellet	
Cs-136	N/A	Used for calculation of key ratios.
Cs-137	N/A	
Cs-138	N/A	

II. APPROPRIATE SAMPLE LOCATIONS FOR CORE DAMAGE ESTIMATE

Table II.A. Sample Locations

<u>Accident Scenario</u>	<u>Conditions</u>	<u>Sample Locations</u>		
		<u>RC</u>	<u>CS</u>	<u>CA</u>
Small Break LOCA	Sub-cooling margin <u>not</u> maintained	X	X*	X*
Large Break LOCA	N/A	X	X	X
Steam Line Break in Containment	Sub-cooling margin <u>not</u> maintained	X		

*Required only if alarm on containment building dome monitor.

III. RECORD OF PLANT DATA

Record the following plant conditions as closely as possible to the time at which radiological samples are obtained from the PASS.

- A. Time of reactor shutdown Date _____ Time _____
- B. Prior 30-day power history Power _____ % Duration _____ Days
_____ % _____ Days
_____ % _____ Days
_____ % _____ Days

C. Reactor Coolant System (RC):

- 1) Pressure _____ psia
- 2) Temperature _____ °F
- 3) CET Temperature _____ °F
- 4) Reactor Vessel Level _____ %
- 5) Pressurizer Level _____ %
- 6) RC Volume (V_{RC}) ANO-1 2.29×10^8 cc ANO-2 2.10×10^8 cc
- 7) Time of Measurement Date _____ Time _____

D. Containment Building:

- 1) Containment Atmosphere Pressure _____ psia
- 2) Containment Atmosphere Temperature _____ °F
- 3) Radiation Dose Rate _____ rads/hr

4) Containment Sump Level ANO-1 _____% ANO-2 _____%

5) Containment Sump Volume* ANO-1 _____cc ANO-2 _____cc

6) Containment Free Volume ANO-1 5.19x10¹⁰cc ANO-2 5.15x10¹⁰cc

7) Time of Measurement Date _____ Time _____

*Containment Sump Volume is calculated as follows:

$$\text{ANO-1: } (\text{Level } \%) \times \frac{25.6 \text{ gal}}{\%} \times 3785.4 \frac{\text{cc}}{\text{gal}}$$

$$\text{ANO-2: } (\text{Level } \%) \times \frac{39 \text{ gal}}{\%} \times 3785.4 \frac{\text{cc}}{\text{gal}}$$

IV. TABLE OF POWER HISTORY CORRECTED EQUILIBRIUM SOURCE INVENTORY

Isotope	Decay Constant (day ⁻¹)	Power History Grouping	Equilibrium Source Inventory (Curies)		PCF	Power Corrected Source Inventory (Curies)	
			ANO-1	ANO-2		ANO-1	ANO-2
Kr-85M	3.713	2	1.8x10 ⁷	2.2x10 ⁷			
Kr-87	1.280x10 ¹	2	3.5x10 ⁷	4.2x10 ⁷			
Kr-88	5.941	2	4.9x10 ⁷	5.9x10 ⁷			
Xe-131M	5.776x10 ⁻²	1	7.7x10 ⁵	8.3x10 ⁵			
Xe-133	1.315x10 ⁻¹	1	1.4x10 ⁸	1.5x10 ⁸			
I-131	8.611x10 ⁻²	1	6.9x10 ⁷	7.5x10 ⁷			
I-133	7.998x10 ⁻¹	2	1.4x10 ⁸	1.5x10 ⁸			
I-135	2.483	2	1.3x10 ⁸	1.4x10 ⁸			
Te-132	2.160x10 ⁻¹	1	9.8x10 ⁷	1.1x10 ⁸			
Ba-140	5.415x10 ⁻²	1	1.2x10 ⁸	1.4x10 ⁸			
Ru-103	1.750x10 ⁻²	1	1.1x10 ⁸	1.1x10 ⁸			
Cs-136	5.332x10 ⁻²	1	2.9x10 ⁶	2.7x10 ⁶			
Cs-137	6.548x10 ⁻⁵	N/A	6.7x10 ⁶	6.2x10 ⁶			
Cs-138	3.100x10 ¹	2	1.3x10 ⁸	1.4x10 ⁸			

Group 1 Power Correction Factor (PCF) = $\frac{\text{Steady-State Power Level in Prior 30 Days}}{100}$

*Group 2 PCF = $\frac{\text{Steady-State Power Level in prior 4 Days}}{100}$

**General PCF =

Where: P_j = steady reactor power in period j, %

t_j = duration of period j, days

t_j^0 = time from end of period j to reactor shutdown, days

λ_i = decay constant of isotope i, day⁻¹

*For use when steady-state power level ($\pm 10\%$) for has not been sustained for 30 days, but has been sustained for at least 4 days. Only Group 2 isotopes should be used in estimate.

**For use when steady-state power level has not been sustained between $\pm 10\%$ for 4 days. Entire 30-day power history should be used in applying the General Power Correction Factor.

V. CALCULATION OF STP CORRECTED SPECIFIC ACTIVITIES*

Table V.A. STP Corrected Specific Activities

Isotope	Measured Specific Activities ($\mu\text{Ci/cc}$)			Correction Factor**			STP Corrected Specific Activities ($\mu\text{Ci/cc}$)		
	<u>RC</u>	<u>CS</u>	<u>CA</u>	<u>RC</u>	<u>CS</u>	<u>CA</u>	<u>RC</u>	<u>CS</u>	<u>CA</u>
Kr-85m									
Kr-87									
Kr-88									
Xe-131m									
Xe-133									
I-131									
I-133									
I-135									
Te-132									
Ba-140									
Ru-103									
Cs-136									
Cs-137									
Cs-138									

*PASS automatically corrects sample to standard temperature and pressure -
Verify with Radiochemistry personnel that samples were corrected for STP.

**If applicable

The following calculational method is provided for the case in which samples have not been corrected for STP:

RC samples are corrected for RCS temperature and pressure using the factor for water density in Table V.B. and the following equation:

$$S_{RC} \text{ (corrected)} = S_{RC} \text{ (measured)} \times F_D$$

CS samples do not require correction for temperature and pressure within the accuracy of this method.

CA samples are corrected using the following equation.

$$S_{CA} \text{ (corrected)} = S_{CA} \text{ (measured)} \times \frac{P_2(T_1+460)}{P_1(T_2+460)}$$

where: P_1, T_1 = containment atmosphere sample line pressure and temperature ($^{\circ}\text{F}$).

P_2, T_2 = containment atmosphere pressure and temperature ($^{\circ}\text{F}$).

$S_{CA, RC}$ = specific activity ($\mu\text{Ci/cc}$) in CA, RC.

Table V.B. RC Density Correction Factor, F_D

F_D = RC density correction factor

<u>RCS Temperature at Time of Sample (°F)</u>	<u>RCS Density Correction Factor, F_D</u>
100	1
150	0.987
200	0.970
250	0.949
300	0.924
350	0.897
400	0.865
450	0.830
500	0.790
550	0.741
560	0.731
580	0.709
600	0.683
620	0.653
640	0.620
660	0.582
680	0.531
700	0.441

The values of F_D above were calculated assuming a 100°F sample temperature.

VI. CALCULATION OF DECAY CORRECTED SPECIFIC ACTIVITIES*

Table VI.A. Decay Corrected Specific Activities

Isotope	Decay Constant (day ⁻¹)	Power History Grouping	STP Corrected Specific Activity			Correction Factor	Decay Corrected Specific Activities (μCi/cc)		
			RC	CS	CA		RC	CS	CA
Kr-85m	3.713	2							
Kr-87	1.280x10 ¹	2							
Kr-88	5.941	2							
Xe-131m	5.776x10 ⁻²	1							
Xe-133	1.315x10 ⁻¹	1							
I-131	8.611x10 ⁻²	1							
I-133	7.998x10 ⁻¹	2							
I-135	2.483	2							
Te-132	2.160x10 ⁻¹	1							
Ba-140	5.415x10 ⁻²	1							
Ru-103	1.750x10 ⁻²	1							
Cs-136	5.332x10 ⁻²	1							
Cs-137	6.548x10 ⁻⁵	N/A							
Cs-138	3.100x10 ¹	2							

*PASS will automatically decay correct specific activities if Radiochemistry personnel are advised to program the shutdown time. The equation used is:

$$S_0 = \frac{S}{e^{-\lambda_i t}}$$

where:

S_0 = the specific activity of the STP corrected sample corrected back to the time of reactor shutdown, μCi/cc

S = the STP corrected specific activity, μCi/cc

λ_i = the decay constant for isotope i , day⁻¹

t = the time period from reactor shutdown to sample analysis, days.

For Xe-131m and Xe-133, an additional calculation must be made in order to account for the increase in Xe inventory due to I decay since shutdown.

$$\text{Xe-131m(real)} = \text{Xe-131m(measured)} e^{\lambda_x t} - 0.008 \frac{(\lambda_I) \text{I-131(measured)}}{\lambda_x} (e^{\lambda_x t} - 1)$$

$$\text{Xe-133(real)} = \text{Xe-133(measured)} e^{\lambda_x t} - 0.976 \frac{\lambda_{II-133} \text{I-133(measured)}}{\lambda_x} (e^{\lambda_x t} - 1)$$

where t = number of days between shutdown and sample analysis.

λ_x = decay constant of Xenon isotope, days⁻¹

λ_i = decay constant of Iodine isotope, days⁻¹

VII. KEY ISOTOPIC RATIOS IN REACTOR COOLANT AS AN INDICATION OF FUEL DAMAGE

<u>#</u>	<u>Key Ratios</u>	<u>Normal Coolant</u>	<u>Transient Spiking</u>	<u>Gap Release</u>	<u>Fuel Matrix Release</u>
1	I-131:I-133	1	~2	5	0.6
2	I-131:I-135	2	~4	16	0.6
3	Cs-137:Cs-136	5	~10	90	6
4	Cs-137:Cs-138	0.6	~4	110	0.04

Table VII.A. Release Source Determination

<u>Ratio #</u>	<u>Calculated Ratio</u>	<u>Indicated Release Source</u>
1		
2		
3		
4		

VIII. CALCULATION OF ACTIVITIES AVAILABLE FOR RELEASE

TABLE VIII.A. Total Activities Available for Release*

<u>Isotope</u>	<u>Pellet (Ci)</u>	<u>Gap (Ci)</u>
Kr-85m		
Kr-87		
Kr-88		
Xe-131m		
Xe-133		
I-131		
I-133		
I-135		
Te-132		
Ba-140		
Ru-103		
Cs-136		
Cs-137		
Cs-138		

*Total activities available for release are obtained by multiplying the release fractions for the source or release tabulated on Attachment B by the power history corrected source inventories of Step IV.

IX. CALCULATION OF TOTAL ACTIVITIES RELEASED (A)

TABLE IX.A. Total Activities Released

Isotope	Power History Grouping	STP and Decay Corrected Specific Activities ($\mu\text{Ci/cc}$)			Total Corrected Activity Released (Ci)			Sum of Activity Released (Ci)
		<u>RC</u>	<u>CS</u>	<u>CA</u>	<u>RC</u>	<u>CS</u>	<u>CA</u>	
		Kr-85m	2					
Kr-87	2							
Kr-88	2							
Xe-131m	1							
Xe-133	1							
I-131	1							
I-133	2							
I-135	2							
Te-129	2							
Te-132	1							
Ba-140	1							
Ru-103	1							
Cs-136	1							
Cs-137	N/A							
Cs-138	2							

Total corrected activities released are calculated as follows:

- 1) RC - If the water level in the reactor vessel recorded in step III indicates that the vessel is full:

$$A(\text{Ci}) = \text{STP \& Decay Corrected Specific Activity} \times V_{\text{RC}} \times 10^{-6}$$

If the water level in the reactor vessel and pressurizer recorded in step III indicate that a steam void is present, then the activity released is calculated as above. However, it must be recognized that the value obtained will overestimate the actual quantity released. Therefore, this sample should be repeated at such time when the operators have removed the reactor vessel void.

If the water level in the reactor vessel recorded in Step III is below the low end capability of the indicator, it is not possible to determine the activity released from this sample because the volume of water in the reactor coolant system is unknown. In this case, assessment of core damage is obtained using the containment sump sample.

- 2) CS -

$$A(\text{Ci}) = \text{STP \& Decay Corrected Specific Activity} \times V_{\text{CS}} \times 10^{-6}$$
- 3) CA -

$$A(\text{Ci}) = \text{STP \& Decay Corrected Specific Activity} \times V_{\text{CA}} \times 10^{-6}$$

The total activity released is the sum of the released activity in each sample location.

X.A. INTERPRETATION OF CORE DAMAGE

The core damage estimate will start by determining which of the four fuel conditions (no damage, cladding failure, fuel overheat or fuel melt) best describes the core environment. A flow chart has been prepared to show the logic for estimating core damage (Attachment C).

A. No Damage

The core conditions which pertain to no damage or normal operation are as follows:

- 1) The reactor was shutdown with no abnormal conditions and other plant indicators verify that there has been adequate core cooling such as:
 - All of the core exit thermocouple (CET) readings indicate that core damage is unlikely per Attachment A.
 - No excess amount of hydrogen is found in the primary system and no detectable hydrogen is found in the containment atmosphere.
- 2) If the above best describes the core environment then divide the total released activity A by the upper limit normal operating activities given on Attachment D.

Isotope

Calculated Fraction

Kr-85m
Kr-87
Kr-88
Xe-131m
Xe-133
I-131
I-133
I-135
Te-132
Ba-140
Ru-103
Cs-136
Cs-137
Cs-138

4) If $\frac{A}{\text{total upper limit activity}} > 1$

for any fission products, then it is possible that some cladding failure has occurred. In this case recheck the CET readings to see if any one of them indicated possible cladding failures and determine the percent cladding failure.

B. Cladding Failure

The core conditions which pertain to cladding failure are as follows:

- 1) The reactor was shutdown with adequate core cooling and other plant indicators verify that there has been a possibility for cladding failures, such as:

- one or more of the CET readings indicate the possibility of cladding failures per Attachment A.

- no excess amount of hydrogen is found in the primary system and no detectable hydrogen is found in the containment atmosphere.

- $\frac{A}{\text{total upper limit activity}} > 1$

- 2) If the above best describes the core environment, then divide the total released activity, A, by the total available gap release activity.*

Isotope

Calculated Fraction

Kr-85m
 Kr-87
 Kr-88
 Xe-131m
 Xe-133
 I-131
 I-133
 I-135
 Te-132
 Ba-140
 Ru-103
 Cs-136
 Cs-137
 Cs-138

*Gap release activities are obtained from Step VIII.

Then, $\frac{A \times 100}{\text{total gap release activity}} = \text{percent cladding failure}$

- 3) If $\frac{A \times 100}{\text{total gap release activity}} > 100$ for any of the above

fission products, then there is a possibility that some fuel overheating or fuel melting has occurred. Recheck the following plant indicators to determine if fuel overheating or fuel melting has occurred:

- one or more core exit thermocouple readings indicate the possibility of fuel pellet overheat per Attachment A followed by detection of excess hydrogen in the containment atmosphere or the primary system.
- the presence of any low-volatile fission products (i.e. Te, Ba, or Ru) indicates that some fuel melting may have occurred.

NOTE: Absence of low-volatiles (i.e., Te, Ba, Ru) is a good indication that no fuel melting has occurred.

C. Fuel Overheating

The core conditions which pertain to fuel overheating are as follows:

- 1) There has been an abnormal shutdown and there is a possibility that the fuel has been partially uncovered for a period of time greater than a few minutes. Other plant indicators verify that there has been fuel overheating such as:

- one or more core exit thermocouple readings indicate the possibility of fuel pellet overheat per Attachment A.
- excess hydrogen is found in the primary system or a detectable amount of hydrogen is found in the containment atmosphere.

- $\frac{A}{\text{total available gap release activity}} > 1$

- 2) If the above best describes the core environment, then divide the total released activity, A, by the total available core inventory.

<u>Isotope</u>	<u>Calculated Fraction</u>
Kr-85m	
Kr-87	
Kr-88	
Xe-131m	
Xe-133	
I-131	
I-133	
I-135	
Te-132	
Ba-140	
Ru-103	
Cs-136	
Cs-137	
Cs-138	

- 3) If $\frac{A}{\text{total core inventory}} < 0.2$ for all of the noble gases

then it could be concluded that there has been <50% fuel overheating. This conclusion can be further verified by indication that some of the core exit thermocouple readings indicate the possibility of fuel pellet overheat per Attachment A followed by detection of excess hydrogen in primary system or containment. In case low volatiles (i.e. Te-132, Ru-103 or Ba-140) were detected, determine the percent fuel melting.

- 4) If $\frac{A}{\text{total core inventory}} > 0.2$ for any of the noble gases

then it could be concluded that there has been >50% fuel overheating. This conclusion can be further verified by indication that >50% of core exit thermocouple readings indicate the possibility of fuel pellet overheat per Attachment A. In case low-volatiles (i.e. Te-132, Ru-103 or Ba-140) were detected, determine the percent fuel melting.

D. Fuel Melting

The core conditions which pertain to fuel melting are as follows:

- 1) There has been a severe accident and the core has been uncovered for a long period of time. The following plant indicators verify that there has been fuel melting.

- more than one core exit thermocouple reading indicates the possibility of fuel pellet overheat per Attachment A.

- one or more core exit thermocouple readings indicate the possibility of fuel pellet melt per Attachment A.

- excess amount of hydrogen is found in the primary system or a detectable amount of hydrogen is found in the containment atmosphere.

- $\frac{A}{\text{total core inventory}} < 0.2$ with presence of low-volatiles (i.e. Te-132, Ru-103, or Ba-140)

- $\frac{A}{\text{total core inventory}} > 0.2$ with presence of low-volatiles

- $\frac{A}{\text{total gap release activity}} > 1$ with presence of low volatiles

2) If these conditions best describe the core environment, then divide the total released activity, A, by the total available meltdown release activity.*

<u>Isotope</u>	<u>Calculated Fraction</u>
Kr-85m	
Kr-87	
Kr-88	
Xe-131m	
Xe-133	
I-131	
I-133	
I-135	
Te-132	
Ba-140	
Ru-103	
Cs-136	
Cs-137	
Cs-138	

*Nominal meltdown release activities are obtained in Step VIII

then $\frac{A \times 100}{\text{total available meltdown release activity}} = \text{percent fuel melted}$

NOTE: The fuel melting estimates will be primarily based on the data from low-volatiles (i.e. Te, Ba, or Ru) since the data from other fission products (i.e. Xe, Kr, I, Cs) will not provide for a good estimate of fuel melting in the lower sub-group (i.e. lower than 10% fuel melting). This is due to the fact that Xe, Kr, I and Cs fission products are released in significant quantities during fuel overheating. Therefore, it would be

difficult to determine whether the releases were due to fuel overheating or fuel melting. However, estimates based on Xe, Kr and I can provide verification of the estimates which are based on Te, Ba, or Ru.

- E. The following notes should be considered in interpretation of the core damage estimates:
- 1) If the cladding failure or fuel overheating estimates based on Xe and Kr are significantly different (i.e. Xe and Kr estimates are in different sub-groups), recheck the estimates and if necessary obtain new samples to determine new concentrations.
 - 2) If the core has been uncovered for a long time and the estimates based on noble gases and iodines indicate significantly higher percentage of fuel melting than the estimates based on Te, Ru, or Ba (i.e. the estimates are in a different sub-group), the low-volatile estimates must be rechecked and if necessary new samples obtained to determine new concentrations. If the results remain the same as before, the fuel melting estimates should be based on the low-volatiles.
 - 3) The fuel melting estimates based on Xe, Kr, or I will not provide for a good estimate in the <10% sub-group. This is due to the fact that these fission products are released in significant quantities during fuel overheating and it would be difficult to determine whether the releases were due to fuel overheating or fuel melting.
 - 4) The samples taken during the later stages of the accident may provide for a more accurate core damage estimate, than those taken in the early stages of the accident.
 - 5) If the accident did not involve a breach of the primary system, CS and CA samples will provide no useful information.

X.B. TABLE OF PERCENT RELEASED DATA

<u>Isotope</u>	<u>Total Activity Released(A)</u>	<u>Percent Release</u>
Kr-85m		
Kr-87		
Kr-88		
Xe-131m		
Xe-133		
I-131		
I-132		
I-133		
I-135		
Te-132		
Ba-140		
Ru-103		
Cs-136		
Cs-137		
Cs-138		