CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

ESTIMATE OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

PLANT EMERGENCY PROCEDURE: PEP-03.6.3

VOLUME XIII

Rev. 004

Approved By: L. E. Boyer Date: 1/29/85
General Manager/
Director - Administrative Support

LIST OF EFFECTIVE PAGES

PEP-03.6.3

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1-35	4

1.0 Responsible Individual and Objectives

The Radiological Control Director is responsible to the Site Emergency Coordinator for determining the magnitude of potential radioactive releases to the environment. The Radiological Control Director may delegate the calculational aspects to the Plant Sampling and Analysis Team Leader.

The Dose Projection Coordinator and the Accident Assessment Team Leader should be familiar with this procedure and available for consultation as requested by the Plant Sampling and Analysis Team Leader.

2.0 Scope and Applicability

This procedure is to be implemented by the Site Emergency Coordinator or the Radiological Control Director whenever the potential for core damage exists and/or there exists a potential or actual radiological release to the environment (e.g., site or general emergency).

This procedure provides information on inventories of reactor full-power radioisotopes in curies and gives methods for comparing actual radioactive liquid and gaseous samples with expected activity levels after a reactor accident based on cesium, noble gases, and iodines. There are several other plant parameters which are measured in the BWR which can provide sufficient information to confirm the initial core damage estimate based on radionuclide measurements.

Containment radiation level provides a measure of core damage, because it is an indication of the inventory of airborne fission products (i.e., noble gases, a fraction of the halogens, and a much smaller fraction of the particulates) released from the fuel to the containment. Containment hydrogen levels, which are measurable by the PASS or the containment gas analyzers, provide a measure of the extent of metal water reaction which, in turn, can be used to estimate the degree of clad damage.

Another significant parameter for the estimation of core damage is reactor vessel water level. This parameter is used to establish if there has been an interruption of adequate core cooling. Significant periods with the core uncovered, as evidenced by reactor vessel water level readings, would be an indicator of a situation where core damage is likely. Water level measurement would be particularly useful in distinguishing between bulk core damage situations caused by loss of adequate cooling to the entire core and localized core damage situations caused by a flow blockage in some portion of the core.

There are other parameters which may provide an indication that a core damage event has occurred. These are main steam line radiation level and reactor vessel pressure. The usefulness of main steam line radiation measurement is limited because the main steam line radiation monitors are downstream of the main steam isolation valves (MSIVs) and would be unavailable following vessel isolation. Reactor vessel pressure measurement would provide an ambiguous indication of core damage, because, although a high reactor vessel pressure may be indicative of a core damage event, there are many nondegraded core events which could also result in high reactor vessel pressure.

There are other measurements besides radionuclide measurements which are obtainable using the PASS which would further aid in estimating core damage. Detection of such elements in the reactor coolant as Sr, Ba, La, and Ru is evidence of fuel melting. These indications could be factored into the final core damage estimate.

3.0 Actions and Limitations

3.1 Summary of Method

Liquid and gaseous samples will be obtained from the Postaccident Sampling System (PASS)--Liquid from the reactor coolant and/or suppression pool and gaseous samples from the primary and/or secondary containment. The samples will be quantitatively analyzed on the appropriate equipment. The results of the above analysis, in addition to containment radiation level, hydrogen analysis, and the core water level history, will be used in the estimation. This procedure follows the General Electric procedure NEDO-22215, August 1982.

List of Exhibits

List of Attachments

List of Worksheets

3.2 Limitations

3.2.1 Analysis of PASS samples for concentrations of Ba, Sr, La, and Ru and consideration of the relative amounts of fission products would indicate if any fuel melt has occurred.

3.2.2 The selection of a sample location should account for the type of event which will determine where the fission products will concentrate.

3.2.3 The recommended sampling locations are as follows:

Gaseous

Event Type	Sample Location
Nonbreaks (e.g., MSIV)	Suppression pool atmosphere
Small breaks	Drywell (before depressurization); suppression pool atmosphere (after depressurization)
Large breaks (liquid or steam) in primary containment	Drywell
Large breaks outside primary containment	Suppression pool atmosphere

3.2.4 The recommended sampling location for liquid for all events is the jet pumps as long as there is sufficient reactor pressure (normally > 50 psig) to provide a sample from that location. If there is not sufficient reactor pressure to allow a sample to be taken from the jet pumps, the sample should be taken from the sample points on the RHR System.

3.2.5 If a jet pump liquid sample is requested at low (< 1%) power conditions for a small break or nonbreak event, recommend to Operations that the reactor water level be raised to the level of the moisture separators. This will fully flood the moisture separators and will provide a thermally induced recirculation flow path for mixing.

3.3 Actions

3.3.1 Evaluations of Liquid and Gaseous Samples

NOTE: The extent of core damage can be determined by comparing the measured concentrations of major fission products in either the gas or water samples, after appropriate normalization, with the reference plant data.

3.3.1.1 The Plant Sampling and Analysis Team Leader should request samples from the PASS.

NOTE: Steps 3.3.1.2 through 3.3.1.7 can be accomplished using PASS, a computer program developed for use on the Dose Projection Team's IBM Personal Computer. To use the program, the Plant Sampling and Analysis Team Leader should complete Exhibit 3.6.3-7, Computer Inputs for the PASS Program, and give the completed exhibit to the Dose Projection Coordinator who will run the program and return the results. Exhibit 3.6.3-8 provides example test cases which can be used to verify that the computer program PASS is working properly. Expected results for known computer inputs are given. These test cases should be used to demonstrate the validity of PASS each time the program is initially used.

3.3.1.2 Obtain the samples from the PASS and determine the concentration of the fission product i (C_{wi} in water or C_{gi} in gas as determined in Attachment A using data provided in Exhibit 3.6.3-3).

3.3.1.3 Correct the measured concentration for decay to the time of reactor shutdown. Ensure that the measured gaseous activity concentration has been corrected for temperature and pressure difference in the sample vial and the containment (torus) gas phase.

NOTE: This is normally included in the quantitative analysis results.

3.3.1.4 Calculate the fission product inventory correction factor F_{Ii} per Attachment B and record on Worksheet A2.

3.3.1.5 Calculate the C_{wi} and C_{gi} using the information obtained in Step 3.3.1.2 and the methods in Attachment A and record on Worksheet A1.

3.3.1.6 Using the correction factors, determined in Attachments A and B, calculate the normalized concentration, C_{wi}^{Ref} or C_{gi}^{Ref}, per Attachment C and record on Worksheet A3.

3.3.1.7 Use Exhibit 3.6.3-2 to estimate the extent of fuel or cladding damage using C_{wi}^{Ref} for Cs-137 and I-131 and C_{gi}^{Ref} for Xe-133 and Kr-85. Record data on Worksheet A4.

3.3.2 <u>Evaluation of Metal-Water Reaction and Inventory Release</u>

3.3.2.1 Use Attachment D to determine the percent metal-water reaction. Record data on Worksheet B1.

3.3.2.2 Use Attachment E to determine the fuel inventory release to the containment. Record data on Worksheet B2.

3.3.3 <u>Application of Other Significant Parameters to Core Damage Estimate</u>

Section 3.3.1 provides an estimate of core damage based on radionuclide measurements. Based on Step 3.3.1.7, an initial assessment of core damage is made. Based on a clarification provided by the NRC, that assessment would appear in a matrix as follows:

Degree of Degradation	Minor (< 10%)	Intermediate (10% - 50%)	Major (> 50%)
No fuel damage	←	1	→
Cladding failure	2	3	4
Fuel overheat	5	6	7
Fuel melt	8	9	10

As recommended by the NRC, there are four general classes of damage and three degrees of damage within each of the classes except for the "no fuel damage" class.

Consequently, there are a total of ten possible damage assessment categories. For example, Category 3 would be descriptive of the condition where between 10% and 50% of the fuel cladding has failed. Note that the conditions of more than one category could exist simultaneously. The objective of the final core damage assessment procedure is to narrow down, to the maximum extent possible, those categories which apply to the actual in-plant situation.

The initial core damage assessment based on radionuclide measurement will provide one or several candidate categories which most likely represent the actual in-plant condition. The other parameters should then be evaluated (as identified in Section 3.3) to corroborate and further refine the initial estimate.

For example, fission product measurement using PASS may indicate Category 4 core damage and, additionally, the potential for fuel overheat and fuel melt (i.e., Categories 5 through 10). Measurement of hydrogen in containment and use of the hydrogen correlation provided in Attachment D is used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading along with the correlation provided in Attachment E would verify that a significant fission product release to the containment had occurred, further verifying the initial assessment.

Further analysis of the PASS samples for concentrations of Ba, Sr, La, and Ru and consideration of the relative amounts of fission products released would indicate if any fuel melt had occurred.

Exhibit 3.6.3-1 indicates how the analysis of the other significant parameters relates to the estimation of core damage based on radionuclide measurements.

3.3.4 Consult with the Dose Projection Coordinator and the Radiological Control Director when results of this procedure are determined and repeat this procedure as necessary.

4.0 References

Lin, C. C., "Procedure for the Determination of the Extent of Core Damage Under Accident Conditions," NEDO-22215, 1982.

Letter and Attachment from Mr. D. K. Smith, Service Supervisor - Nuclear, General Electric to Mr. A. C. Tollison, Jr., General Manager, Brunswick Steam Electric Plant, dated November 9, 1979, Subject: Radiation Source Term Information.

Letter and Attachments form Mr. T. J. Dente, Chairman - BWR Owner's Group to Mr. D. G. Eisenhut, Licensing Director - USNRC, dated June 17, 1983, Subject: Transmittal of Generic Procedures for Estimation of Core Damage Using Postaccident Sampling System.

EXHIBIT 3.6.3-1

SEQUENCE OF ANALYSIS FOR
ESTIMATION OF CORE DAMAGE

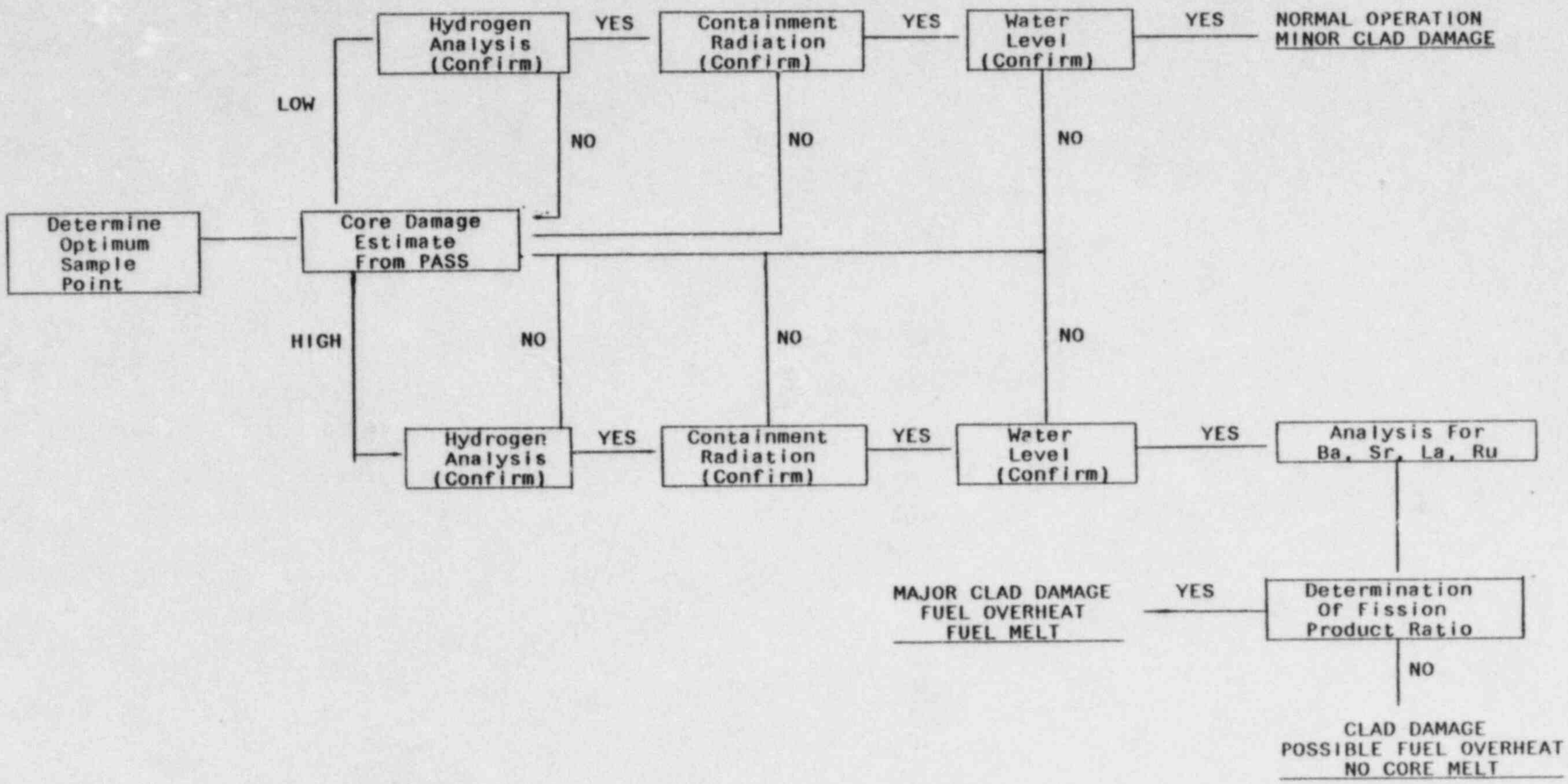
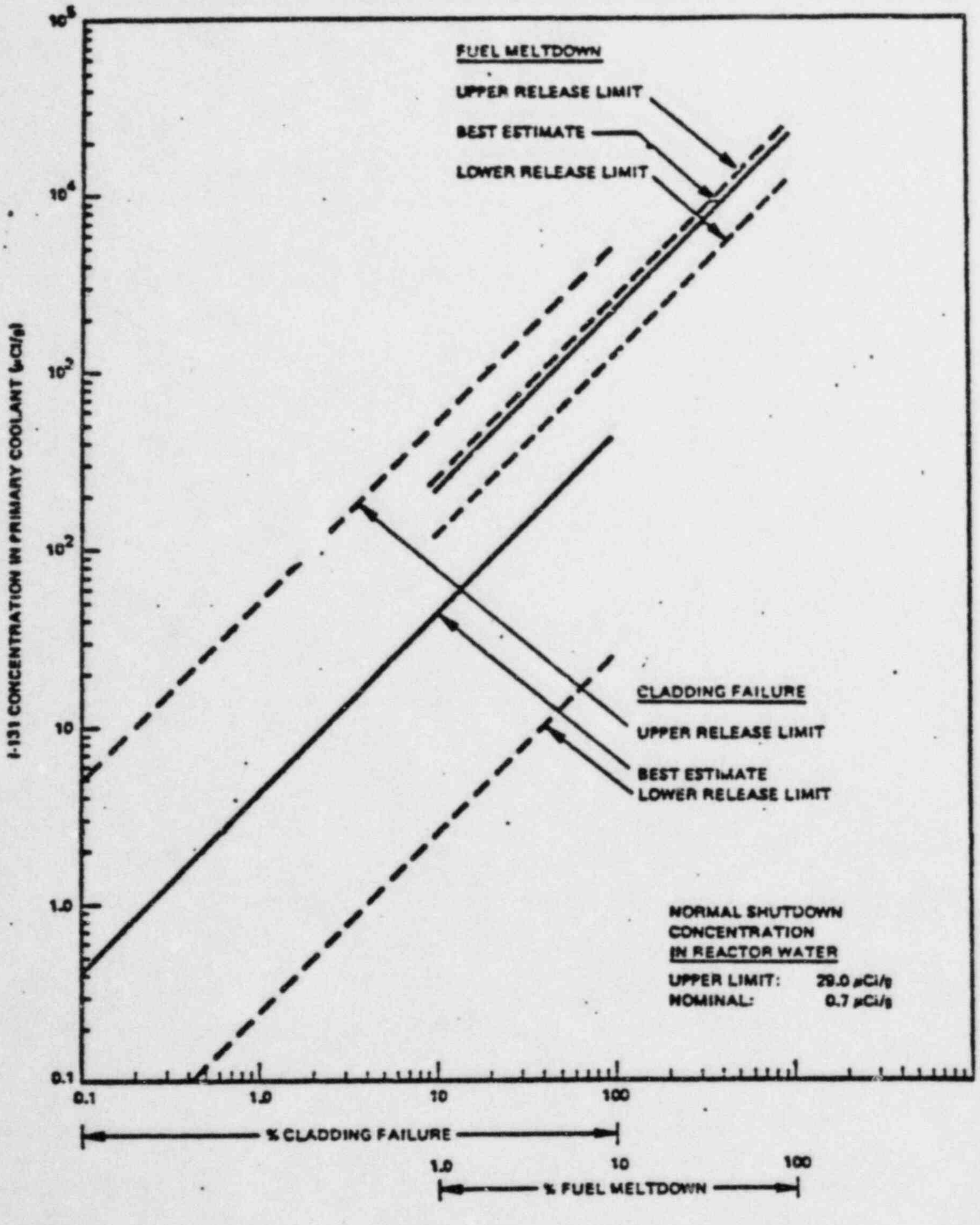
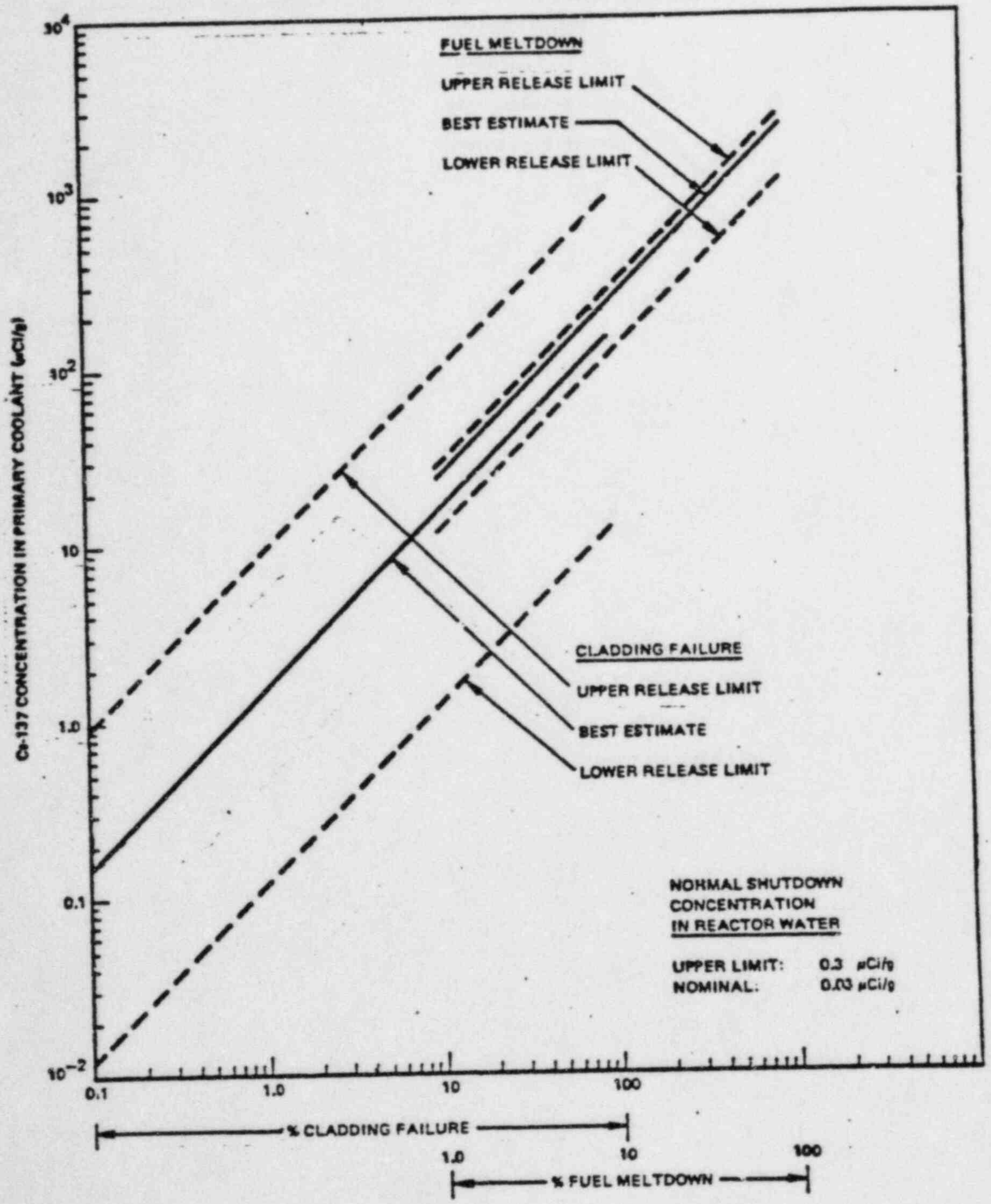


EXHIBIT 3.6.3-2

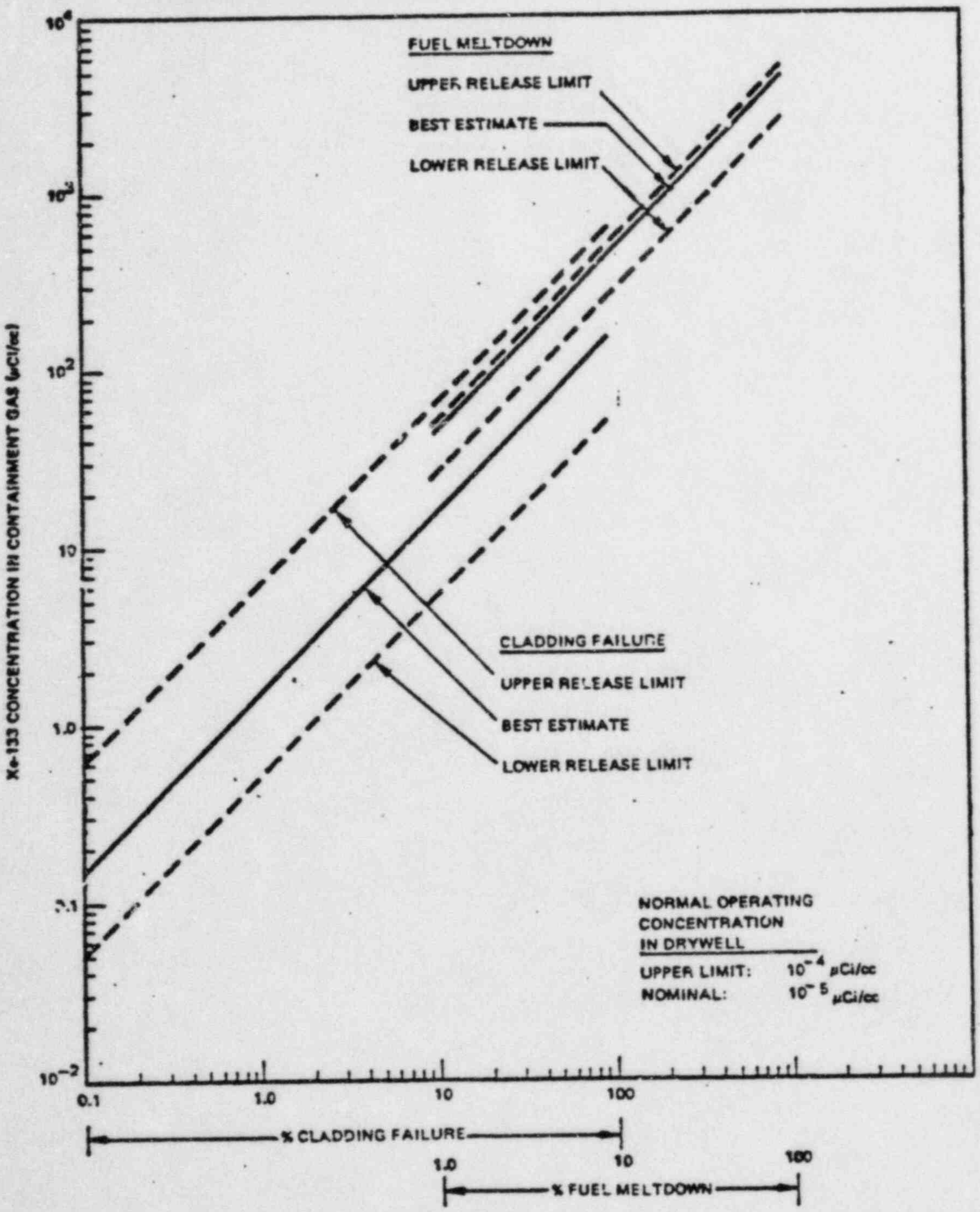


Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)

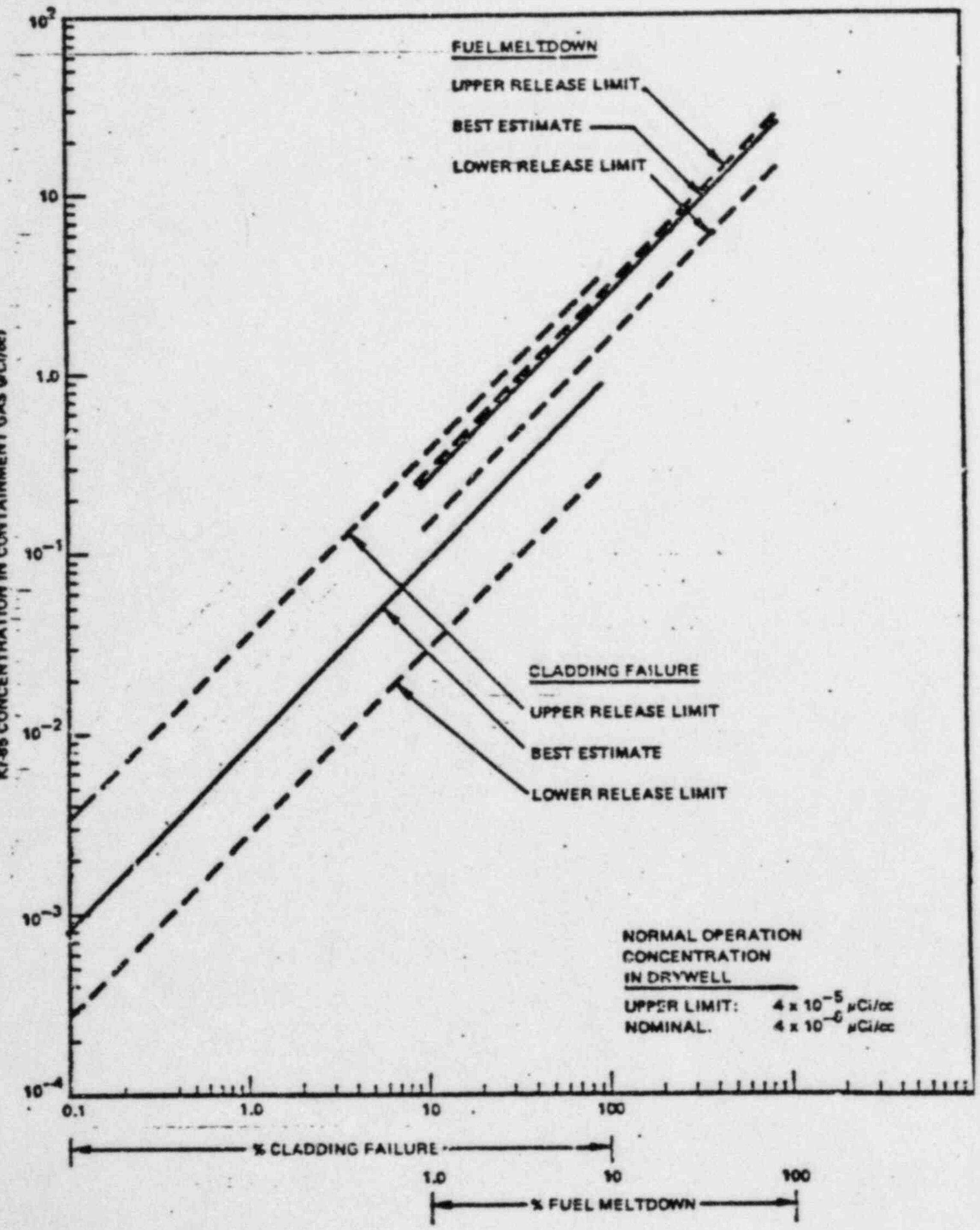


Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant



Relationship Between Xe-133 Concentration in the Containment Gas (Drywell Torus Gas) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)



Relationship Between Kr-85 Concentration in the Containment Gas (Drywell Torus Gas) and the Extent of Core Damage in Reference Plant

ATTACHMENT A

Plant Parameter Correction Factors

Fission products measured together for reactor water and suppression pool water or drywell gas and torus gas.

$$F_w = \frac{\text{BSEP total coolant mass } (2.69 \times 10^9 \text{ g})}{\text{reference plant coolant mass } (3.92 \times 10^9 \text{ g})}$$

$$= 0.68622$$

$$F_g = \frac{\text{BSEP total containment gas volume } (8.11 \times 10^9 \text{ cc})}{\text{reference plant containment gas volume } (4 \times 10^{10} \text{ cc})}$$

$$= 0.20275$$

Fission products measured separately for reactor water and suppression pool water or drywell gas and torus gas.

$$C_{wi} = \frac{\text{(conc. in Rx wtr) (Rx water mass) + (conc. in pool) (pool wtr mass)}}{\text{reactor water mass + pool water}}$$

$$= \frac{\text{(conc. in Rx water)}(2.14 \times 10^8 \text{ g}) + \text{(conc. in pool)}(2.48 \times 10^9 \text{ g})}{2.69 \times 10^9 \text{ g}}$$

$$C_{gi} = \frac{\text{(conc. in drywell) (drywell gas vol.) + (conc. in torus) (torus gas vol.)}}{\text{drywell gas volume + torus gas volume}}$$

$$= \frac{\text{(conc. in drywell)}(4.65 \times 10^9 \text{ cc}) + \text{(conc. in torus)}(3.46 \times 10^9 \text{ cc})}{8.11 \times 10^9 \text{ cc}}$$

ATTACHMENT B

Inventory Correction Factor

$$F_{Ii} = \frac{\text{inventory in reference plant}}{\text{inventory in operation plant}}$$

$$= \frac{3651 \left(1-e^{-1095\lambda_i}\right)}{\sum_j \left[P_j \left(1-e^{-\lambda_i T_j}\right) e^{-\lambda_i T_j^{o}}\right]}$$

where:

P_j = average steady reactor power operated in period j (MWt).

T_j = duration of operating period j (day).

T_j^{o} = time between the end of operating period j and the time of the last reactor shutdown (day).

3651 = reference plant MWt.

If the unit operating history is not readily available, use the following F_I values (based upon Brunswick plant operations under the same operational constraints):

Nuclide	Conservative F_I	λ (day^{-1})
I-131	1.34	0.0862
Cs-137	1.39	6.29×10^{-5}
Xe-133	1.46	0.1320
Kr-85	1.51	1.77×10^{-4}

ATTACHMENT C

Comparison With Reference Plant Data

The extent of core damage can be estimated from the measured fission product concentrations in either the gas or water samples, as described for the reference plant. However, the measured concentration must be corrected for the differences in operation power level, time of operation, primary coolant mass, and containment gas volume.

$$C_{wi}^{Ref} = C_{wi} e^{\lambda_i t} \times F_{Ii} \times F_w$$

OR

$$C_{gi}^{Ref} = C_{gi} e^{\lambda_i t} \times F_{Ii} \times F_g$$

C_{wi}^{Ref} = Concentration of isotope i in the reference plant coolant (µCi/g).

C_{gi}^{Ref} = Concentration of isotope i in the reference plant containment gas (µCi/cc).

C_{wi} = Measured concentration of isotope i in BSEP's coolant (µCi/g). See Attachment A.

C_{gi} = Measured concentration of isotope i in BSEP's containment gas (µCi/cc). See Attachment A.

$e^{\lambda_i t}$ = Decay correction to the time of reactor shutdown.

λ_i = Decay constant of isotope i (day^{-1}).

t = Time between the reactor shutdown and the sample time (days).

F_{Ii} = Inventory correction factor for isotope i. See Attachment B.

F_g = Containment gas volume correction factor. See Attachment A.

F_w = Primary coolant mass correction factor. See Attachment A.

EXHIBIT 3.6.3-3

BSEP TO REFERENCE PLANT PARAMETERS

	Reference Plant	BSEP
Reactor Thermal Power	3651 MWt	2436 MWt
Number of Fuel Bundles	748 bundles	560 bundles
Total Primary Coolant Mass (reactor water plus suppression pool water)	3.92×10^{9} g	2.69×10^{9} g
Total Drywell and Torus Gas Space Volume	4.0×10^{10} cc	8.11×10^{9} cc
Reactor Water	2.46×10^{8} g	2.14×10^{8} g
Suppression Pool	3.67×10^{9} g	2.48×10^{9} g
Drywell Gas Volume	7.77×10^{9} cc	4.65×10^{9} cc
Torus Gas Volume	3.25×10^{10} cc	3.46×10^{9} cc

EXHIBIT 3.6.3-4

Core Inventory of Major Fission Products in a Reference Plant Operated at 3651 MWt for Three Years

Chemical Group	Isotope	Half-Life*	Inventory 10^6Ci	Major Gamma Ray Energy-Intensity - keV(γ /d)
Noble Gases	Kr-85m	4.48 h	24.6	151 (0.753)
	Kr-85	10.72 y	1.1	514 (0.0044)
	Kr-87	76.00 m	47.1	403 (0.495)
	Kr-88	2.84 h	66.8	196 (0.26), 1530 (0.109)
	Xe-133	5.25 d	202.0	81 (0.365)
	Xe-135	9.11 h	26.1	250 (0.899)
Halogens	I-131	8.04 d	96.0	364 (0.812)
	I-132	2.30 h	140.0	668 (0.99), 773 (0.762)
	I-133	20.80 h	201.0	530 (0.86)
	I-134	52.60 m	221.0	847 (0.954), 884 (0.653)
	I-135	6.59 h	189.0	1132 (0.225), 1260 (0.286)
Alkali Metals	Cs-134	2.06 y	19.6	605 (0.98), 796 (0.85)
	Cs-137	30.17 y	12.1	662 (0.85)
	Cs-138	32.20 m	178.0	463(0.307), 1436 (0.76)
Tellurium Group	Te-132	78.00 h	138.0	228 (0.88)
Noble Metals	Mo-99	66.02 h	183.0	740 (0.128)
	Ru-103	39.40 d	155.0	497 (0.89)
Alkaline Earths	Sr-91	9.52 h	115.0	750 (0.23), 1024 (0.325)
	Sr-92	2.71 h	123.0	1384 (0.9)
	Ba-140	12.80 d	173.0	537 (0.254)
Rare Earth	Y-92	3.54 h	124.0	934 (0.139)
	La-140	40.20 h	184.0	487 (0.455), 1597 (0.955)
	Ce-141	32.50 d	161.0	145 (0.48)
	Ce-144	284.40 d	129.0	134 (0.108)
Refractories	Zr-95	64.00 d	161.0	724 (0.437), 757 (0.553)
	Zr-97	16.90 h	166.0	743 (0.928)

* h = hour
d = day
m = month
y = year

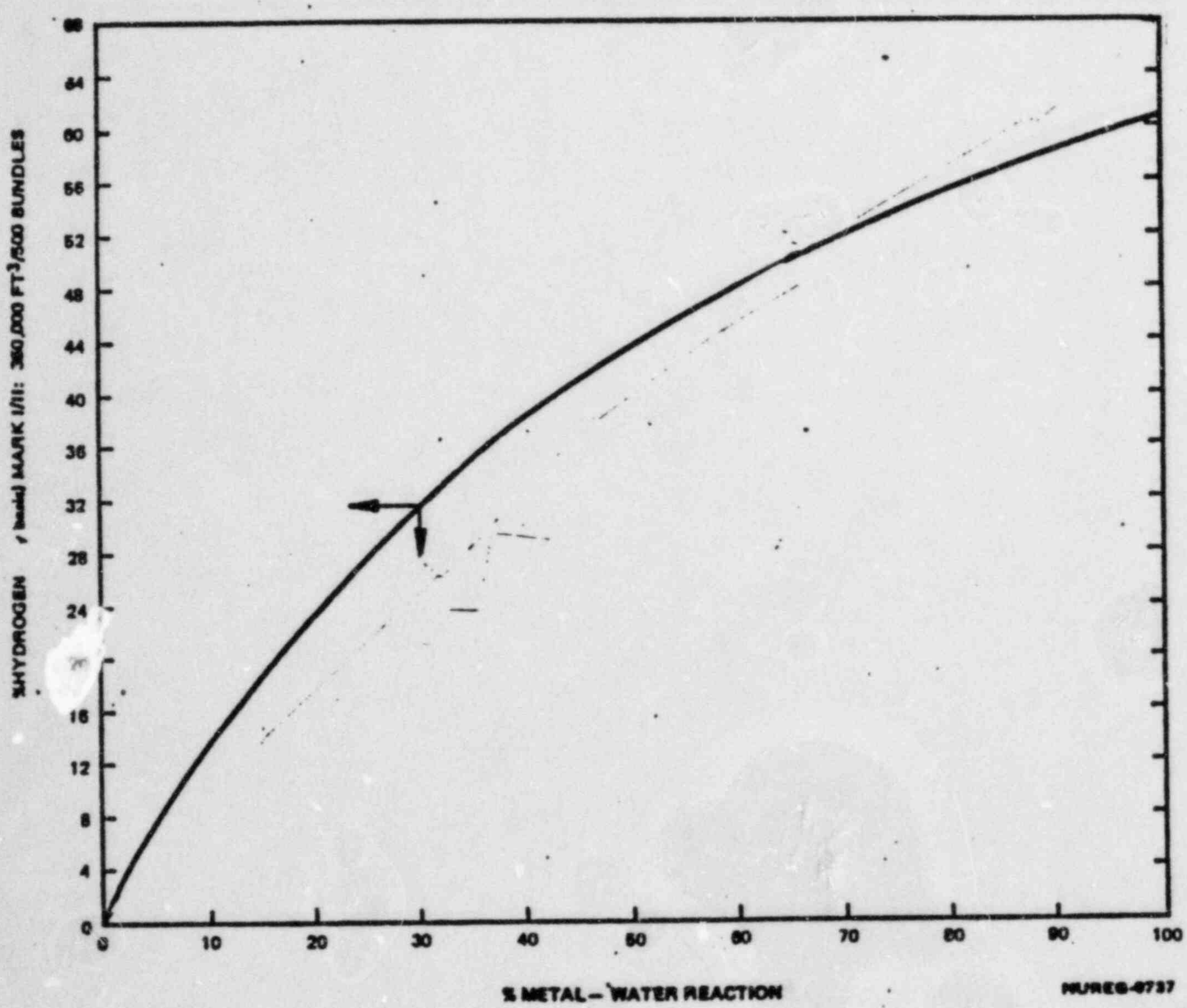
ATTACHMENT D

Integration of Containment Atmosphere Hydrogen Measurement Into Core Damage Estimate

The extent of fuel clad damage as evidenced by the extent of metal-water reaction can be estimated by determination of the hydrogen concentration in the containment. That concentration is measurable by either the containment hydrogen monitor or by the Postaccident Sampling System.

A correlation has been developed which relates containment hydrogen concentration to the percent metal-water reaction for Marks I and II type containments. That correlation is shown in Exhibit 3.6.3-5. Note A to that exhibit indicates the major assumptions used in developing the correlation. Note B indicates the method by which Brunswick plant can use the correlation to determine the extent of clad damage.

EXHIBIT 3.6.3-5



Hydrogen Concentration for Marks I and II Containments as a Function of Metal-Water Reaction

ATTACHMENT D (Cont'd)

Note A to Exhibit 3.6.3-5
Analytical Assumptions
(For Marks I and II Containments)

1. Containment Volume = 350,000 ft^3

2. Number of Bundles = 500

3. Fuel Type = 8 x 8 R

4. All hydrogen from metal-water reaction released to containment.

5. Perfect mixing in containment.

6. No depletion of hydrogen (e.g., containment leakage).

7. Ideal gas behavior in containment.

ATTACHMENT D (Cont'd)

Note B to Exhibit 3.6.3-5

Determination of Clad Damage From Hydrogen Monitor Reading

Step 1. Obtain containment hydrogen monitor reading in percent.

Step 2. Using the curve in Exhibit 3.6.3-5, determine the metal-water reaction for the reference plant, MWR_{ref}.

Step 3. The metal-water reaction from the actual in-plant conditions (MWR) in determined from the following equation:

$$\% \text{ MWR} = (MWR_{ref}) \times \frac{500}{N} \times \frac{V}{350{,}000}$$

where:

N = Number of Bundles = 560
V = Total Containment Free Volume, ft^3 = 2.86×10^5

ATTACHMENT E

Integration of Containment Atmosphere Radiation Measurement
Into Core Damage Estimate

An indication of the extent of core damage is the containment radiation level which is a measure of the inventory of fission products released to the containment. This attachment contains a correlation of the containment radiation monitor dose rate to the percent of fuel inventory airborne in the containment. The purpose of this attachment is to present that correlation and provide a method to use that correlation to determine the degree of core damage.

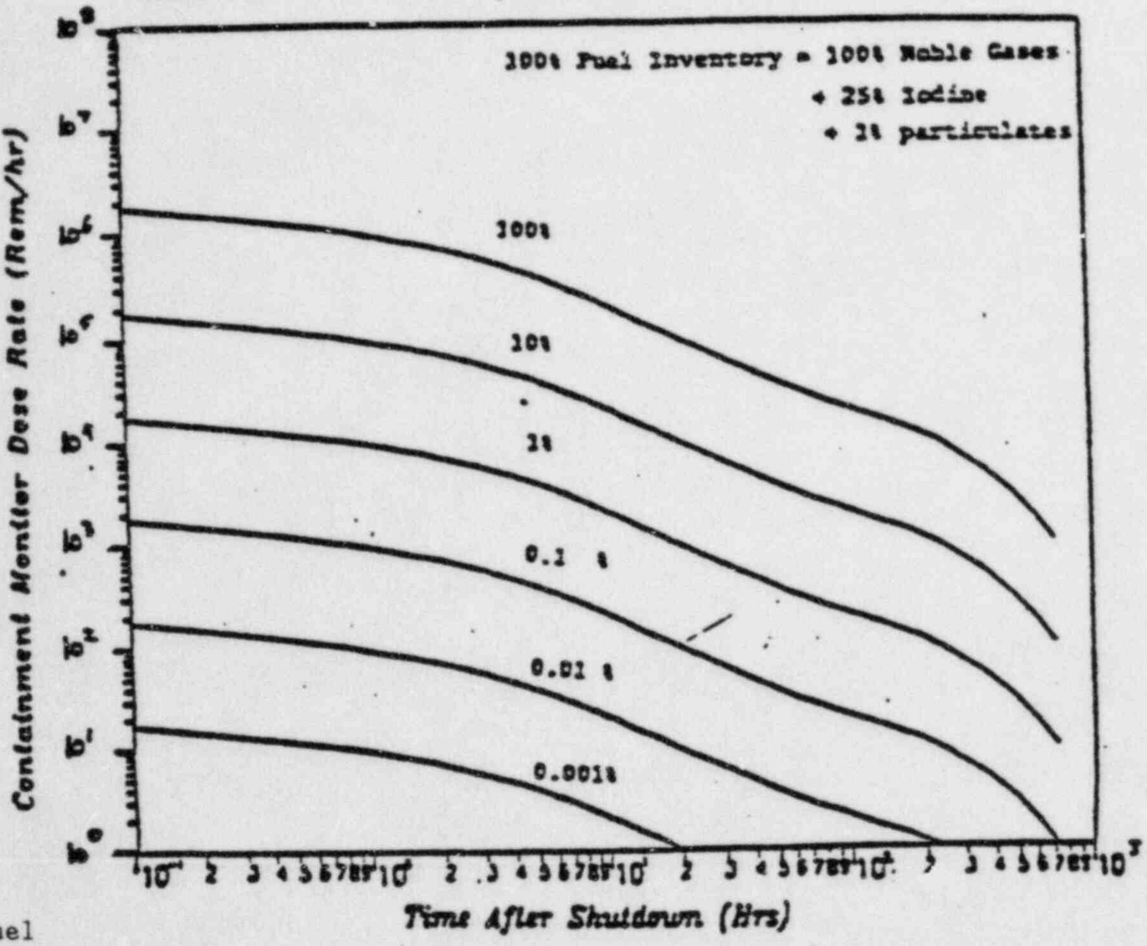
Exhibit 3.5.3-6 provides the results of a correlation performed for the Monticello plant. The key parameters which impact the containment dose rate are reactor power and containment volume.

The method whereby individual plants can apply this correlation is provided in Note A to Exhibit 3.6.3-6.

ATTACHMENT E (Cont'd)

EXHIBIT 3.6.3-6

Percent of Fuel Inventory Airborne in the Containment



% Fuel Inventory Released	Approximate Source and Damage Estimate
100.00	100% TID-14844, 100% fuel damage, potential core melt.
50.00	50% TID noble gases, TMI source.
10.00	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
3.00	3% TID, 100% WASH-1400 gap activity, major clad failure.
1.00	1% TID, 10% NRC gap, maximum 10% clad failure.
0.10	0.1% TID, 1% NRC gap, 1% clad failure, local beating of 5-10 fuel assemblies.
0.01	0.01% TID, 0.1% NRC gap, clad failure of 3/4 fuel element (36 rods).
10^{-3}	0.01% NRC gap clad failure of a few rods.
10^{-4}	100% coolant release with spiking.
5×10^{-6}	100% coolant inventory release.
10^{-6}	Upper range of normal airborne noble gas activity in containment.

ATTACHMENT E (Cont'd)

NOTE A to Exhibit 3.6.3-6

Determination of Clad Damage From Containment
Radiation Monitor Reading

The procedure for determination of fraction of fuel inventory released to the containment is as follows:

Step 1: Obtain containment radiation monitor reading, [R] in rem/hr.

Step 2: Determine elapsed time from plant shutdown to the containment radiation monitor reading [t] in hours.

Step 3: Using Exhibit 3.6.3-6, determine the fuel inventory release for the reference plant $[I]_{ref}$ in percent.

Step 4: Determine the inventory release to the containment [I] using the following formula:

$$[I] = [I]_{ref}\left(\frac{1670}{P}\right)\left(\frac{V}{237,450}\right)$$

where:

P = reactor power level MW_{th} (BSEP = 2436 MW_{th}).

V = total containment free volume, ft^3 (BSEP = 286, 370 ft^3).

NOTE: Monitor location within the containment is assumed to have an insignificant impact on dose rate due to fuel inventory airborne in containment.

EXHIBIT 3.6.3-7

Computer Inputs for the PASS Program

Concentration of I-131 in Reactor Water (μCi/ml)	________
Concentration of I-131 in Suppression Pool (μCi/ml)*	________
Concentration of Cs-137 in Reactor Water (μCi/ml)	________
Concentration of Cs-137 in Suppression Pool (μCi/ml)*	________
Concentration of Xe-133 in Drywell (μCi/cc)	________
Concentration of Xe-133 in Torus (μCi/cc)**	________
Concentration of Kr-85 in Drywell (μCi/cc)	________
Concentration of Kr-85 in Torus (μCi/cc)**	________
Time between Reactor Shutdown and Sample Time (days)	________

If time and availability permits, attach information necessary for the calculation of Inventory Correction Factors (see Attachment B); otherwise, conservative default correction factors will be used.

Plant Sampling and Analysis Team Leader: Give completed exhibit to Dose Projection Coordinator.

Dose Projection Coordinator: Enter data into PASS computer program and provide results to Plant Sampling and Analysis Team Leader.

*If unavailable, assume suppression pool activity = 0 μCi/ml.
**If unavailable, assume torus concentration equal to drywell in μCi/cc.

EXHIBIT 3.6.3-8

VERIFICATION OF PASS
(A Computer program for estimating core damage
based on Postaccident Sampling System results)

This exhibit is intended to provide a means to ensure that PASS, a core damage estimate program designed for the IBM Personal Computer, is working properly. This is demonstrated by duplicating expected results of known computer inputs. These results can be validated by comparison to manual calculations for the same input.

Two different test cases are presented so that a number of alternate paths within the program can be tested. The test cases with their expected results follow.

TEST CASE 1

Computer Prompt	Expected Input
Enter The Concentration of the Fission Products	
Concentration of I-131 in Reactor Water (μCi/ml)	1.72E + 3
Concentration of I-131 in Suppression Pool (μCi/ml)	1.49E + 2
Concentration of Cs-137 in Reactor Water (μCi/ml)	6.55E + 2
Concentration of Cs-137 in Suppression Pool (μCi/ml)	5.70E + 1
Concentration of Xe-133 in Drywell (μCi/cc)	1.82E + 2
Concentration of Xe-133 in Torus (μCi/cc)	2.41E + 2
Concentration of Kr-85 in Drywell (μCi/cc)	1.43E + 0
Concentration of Kr-85 in Torus (μCi/cc)	1.90E + 0

For the inventory correction factor do you want to use the conservative default values which are bases upon BSEP's operations under the same operational constraints (YES or NO)? YES

Enter time between the reactor shutdown and the Sample Time (Days) 2

The results should resemble the printout on the following page. If they do not, carefully check your inputs and try the test again. If the results still are not similar, try a backup copy of the program. If that fails, then seek programming help.

EXHIBIT 3.6.3-8 (Cont'd)
ESTIMATE THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

DATE: 03 28-1984
TIME: 1 :21:27

The concentration of the fission products are:

I-131 in Reactor Water	1.72E + 3 µCi/ml
I-131 in Suppression Pool	1.49E + 2 µCi/ml
Cs-137 in Reactor Water	6.55E + 2 µCi/ml
Cs-137 in Suppression Pool	5.70E + 1 µCi/ml
Xe-133 in Drywell Air	1.82E + 2 µCi/cc
Xe-133 in Torus Air	2.41E + 2 µCi/cc
Kr-85 in Drywell Air	1.43E + 0 µCi/cc
Kr-85 in Torus Air	1.90E + 0 µCi/cc

Time between the reactor shutdown and the sample time is: 2 days

The Concervative Default values of the Inventory Correction Factors were used.

Estimate of fuel/cladding damage
Primary Coolant Analysis

Nuclide	CwREF (µCi/ml)	% Cladding Failure	% Fuel Meltdown
I-131	3.00E + 02	69.00	1.35
Cs-137	1.00E + 02	64.54	4.27

Containment Gas Analysis

Nuclide	CwREF (µCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe-133	7.99E + 01	53.26	1.84
Kr-85	5.00E - 01	56.35	1.92

EXHIBIT 3.6.3-8 (Cont'd)

TEST CASE 2

Computer Prompt	Expected Input
Enter The Concentration of the Fission Products	
Concentration of I-131 in Reactor Water (µCi/ml)	1.35E + 3
Concentration of I-131 in Suppression Pool (µCi/ml)	1.18E + 2
Concentration of Cs-137 in Reactor Water (µCi/ml)	1.17E + 2
Concentration of Cs-137 in Suppression Pool (µCi/ml)	1.02E + 1
Concentration of Xe-133 in Drywell (µCi/cc)	1.84E + 2
Concentration of Xe-133 in Torus (µCi/cc)	2.45E + 2
Concentration of Kr-85 in Drywell (µCi/cc)	2.91E - 1
Concentration of Kr-85 in Torus (µCi/cc)	3.86E - 1
For the inventory correction factor do you want to use the conservative default values which are bases upon BSEP's operations under the same operational constraints (YES or NO)?	NO
Enter time between the reactor shutdown and the Sample Time (Days)?	2
Enter number of Operating Periods from the unit operating history?	3
For period number (1) enter:	
Average steady reactor power operated in this period (MWT)?	1000
Duration of this operating period (days)?	60
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	254
For period number (2) enter:	
Average steady reactor power operated in this period (MWT)?	2000
Duration of this operating period (days)?	200
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	44
For period number (3) enter:	
Average steady reactor power operated in this period (MWT)?	3000
Duration of this operating period (days)?	14
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	0

The results should resemble the printout on the following page. If they do not, carefully check your inputs and try the test again. If the results still are not similar, try a backup copy of the program. If that fails, then seek programming help.

EXHIBIT 3.6.3-8 (Cont'd)
ESTIMATE THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS
DATE: 03-28-1984
TIME: 13:27:17

The concentration of the fission products are:

I-131 in Reactor Water	1.35E + 3 μCi/ml
I-131 in Suppression Pool	1.18E + 2 μCi/ml
Cs-137 in Reactor Water	1.17E + 2 μCi/ml
Cs-137 in Suppression Pool	1.02E + 1 μCi/ml
Xe-133 in Drywell Air	1.84E + 2 μCi/cc
Xe-133 in Torus Air	2.45E + 2 μCi/cc
Kr-85 in Drywell Air	2.91E - 1 μCi/cc
Kr-85 in Torus Air	3.86E - 1 μCi/cc

Time between the reactor shutdown and the sample time is: 2 days

The Inventory Correction Factors were calculated from the following:

Period No.	Operation Time (days)	Time Between Period & Last Shutdown (days)	Average Power (MWt)
1	60	254	1000
2	200	44	2000
3	14	0	3000

Estimate of Fuel/Cladding Damage
Primary Coolant Analysis

Nuclide	CwREF (μCi/ml)	% Cladding Failure	% Fuel Meltdown
I-131	3.00E + 02	69.02	1.35
Cs-137	9.99E + 01	64.49	4.27

Containment Gas Analysis

Nuclide	CwREF (μCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe-133	8.00E + 01	53.30	1.84
Kr-85	5.00E - 01	56.40	1.92

WORKSHEET A1

CALCULATION OF ISOTOPIC CONCENTRATIONS IN PRIMARY WATER AND SUPPRESSION POOL WATER (Cw_i) AND DRYWELL GAS AND TORUS GAS (Cg_i)

References

Section 3.3.1.2
Section 3.3.1.5
Attachment A
Exhibit 3.6.3-3

Cw_i (μCi/ml) = (Concentration Rx H_2O)$_i$ (0.08) +
(Cs^{137}, I^{131}) (Concentration Suppression Pool H_2O)$_i$ (0.92)

= [____________________ + ____________________] (μCi/ml)

= __________ $\mu Ci/ml_{Cs}{}^{137}$

and = __________ $\mu Ci/ml_{I}{}^{131}$

Cg_i (μCi/ml) = (Concentration Drywell)$_i$ (0.57) + (Concentration Torus)$_i$ (0.43)
(Xe^{133}, Kr^{85})

= [____________________ + ____________________] (μCi/cc)

= __________ $\mu Ci/cc_{Xe}{}^{133}$

and = __________ $\mu Ci/cc_{Kr}{}^{85}$

WORKSHEET A2

CALCULATION OF INVENTORY CORRECTION FACTOR (FI_i)

References

Section 3.3.1.4
Attachment B
Exhibit 3.6.3-4

P_j = ______ $MW_{thermal}$

T_{jo} = ______ Days

T_j = ______ Days

λ_i = ______ $Days^{-1}$

$$FI_i = \frac{3651\,(1 - e^{-1095\lambda_i})}{\Sigma_j \left[P_j\,(1 - e^{-\lambda_i T_j})(e^{-\lambda_i T_{jo}})\right]}$$

= ______ (Cs^{137})

______ (I^{131})

______ (Xe^{133})

______ (Kr^{85})

WORKSHEET A3

CALCULATION OF NORMALIZED ISOTOPIC CONCENTRATIONS IN PRIMARY WATER AND SUPPRESSION POOL WATER (Cw_i^{Ref}) AND DRYWELL GAS AND TORUS GAS (Cg_i^{Ref})

References

Section 3.3.1.6
Attachment C
Worksheet A1
Worksheet A2

NOTE: For BSEP,
$Fw = 0.68622$
$Fg = 0.20275$

$$Cw_i^{Ref} = Cw_i \, e^{\lambda it} \times FI_i \times Fw$$
(Cs^{137}, I^{131})

= ______________________________ $\mu Ci/ml_{Cs^{137}}$

______________________________ $\mu Ci/ml_{I^{131}}$

$$Cg_i^{Ref} = Cg_i \, e^{\lambda it} \times FI_i \times Fg$$
(Xe^{133}, Kr^{85})

= ______________________________ $\mu Ci/cc\ Xe^{133}$

______________________________ $\mu Ci/cc\ Kr^{85}$

WORKSHEET A4

ESTIMATE OF FUEL/CLADDING DAMAGE

References

Section 3.3.1.7
Exhibit 3.6.3-2
Worksheet A3

Primary Coolant Analysis

Isotope	Cw_i^{Ref} (μCi/ml)	% Cladding Failure	% Fuel Meltdown
I^{131}			
Cs^{137}			

Containment Gas Analysis

Isotope	Cg_i^{Ref} (μCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe^{133}			
Kr^{85}			

WORKSHEET B1

DETERMINATION OF CLAD DAMAGE FROM HYDROGEN MONITOR READING

References

Section 3.4.1
Attachment D
Exhibit 3.6.3-5

Containment Hydrogen Monitor Reading: ________________%

MWR ref : ________________%

Calculate % MWR:

% MWR = (MWR ref)(0.73)
= ______________

WORKSHEET B2

DETERMINATION OF FUEL INVENTORY RELEASE BASED ON CONTAINMENT RADIATION MONITOR READING

References

Section 3.4.2
Attachment E
Exhibit 3.6.3-6

Containment Radiation Monitor Reading: ____________________ rem/hr

Time from Shutdown to Monitor Reading: ____________________ hrs

[I] ref (Reference Fuel Inventory
Release, %, from Exhibit 3.6.3-6) : ____________________ %

I (Actual Fuel Inventory Release) = [I] ref * 0.827

= ____________ %

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

ESTIMATE OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

PLANT EMERGENCY PROCEDURE: PEP-03.6.3

VOLUME XIII

Rev. 004

Approved By: L. E. Boyer Date: 1/29/85
General Manager/
Director - Administrative Support

LIST OF EFFECTIVE PAGES

PEP-03.6.3

Page(s)	Revision
1-35	4

1.0 Responsible Individual and Objectives

The Radiological Control Director is responsible to the Site Emergency Coordinator for determining the magnitude of potential radioactive releases to the environment. The Radiological Control Director may delegate the calculational aspects to the Plant Sampling and Analysis Team Leader.

The Dose Projection Coordinator and the Accident Assessment Team Leader should be familiar with this procedure and available for consultation as requested by the Plant Sampling and Analysis Team Leader.

2.0 Scope and Applicability

This procedure is to be implemented by the Site Emergency Coordinator or the Radiological Control Director whenever the potential for core damage exists and/or there exists a potential or actual radiological release to the environment (e.g., site or general emergency).

This procedure provides information on inventories of reactor full-power radioisotopes in curies and gives methods for comparing actual radioactive liquid and gaseous samples with expected activity levels after a reactor accident based on cesium, noble gases, and iodines. There are several other plant parameters which are measured in the BWR which can provide sufficient information to confirm the initial core damage estimate based on radionuclide measurements.

Containment radiation level provides a measure of core damage, because it is an indication of the inventory of airborne fission products (i.e., noble gases, a fraction of the halogens, and a much smaller fraction of the particulates) released from the fuel to the containment. Containment hydrogen levels, which are measurable by the PASS or the containment gas analyzers, provide a measure of the extent of metal water reaction which, in turn, can be used to estimate the degree of clad damage.

Another significant parameter for the estimation of core damage is reactor vessel water level. This parameter is used to establish if there has been an interruption of adequate core cooling. Significant periods with the core uncovered, as evidenced by reactor vessel water level readings, would be an indicator of a situation where core damage is likely. Water level measurement would be particularly useful in distinguishing between bulk core damage situations caused by loss of adequate cooling to the entire core and localized core damage situations caused by a flow blockage in some portion of the core.

There are other parameters which may provide an indication that a core damage event has occurred. These are main steam line radiation level and reactor vessel pressure. The usefulness of main steam line radiation measurement is limited because the main steam line radiation monitors are downstream of the main steam isolation valves (MSIVs) and would be unavailable following vessel isolation. Reactor vessel pressure measurement would provide an ambiguous indication of core damage, because, although a high reactor vessel pressure may be indicative of a core damage event, there are many nondegraded core events which could also result in high reactor vessel pressure.

There are other measurements besides radionuclide measurements which are obtainable using the PASS which would further aid in estimating core damage. Detection of such elements in the reactor coolant as Sr, Ba, La, and Ru is evidence of fuel melting. These indications could be factored into the final core damage estimate.

3.0 Actions and Limitations

3.1 Summary of Method

Liquid and gaseous samples will be obtained from the Postaccident Sampling System (PASS)--Liquid from the reactor coolant and/or suppression pool and gaseous samples from the primary and/or secondary containment. The samples will be quantitatively analyzed on the appropriate equipment. The results of the above analysis, in addition to containment radiation level, hydrogen analysis, and the core water level history, will be used in the estimation. This procedure follows the General Electric procedure NEDO-22215, August 1982.

List of Exhibits

List of Attachments

List of Worksheets

3.2 Limitations

3.2.1 Analysis of PASS samples for concentrations of Ba, Sr, La, and Ru and consideration of the relative amounts of fission products would indicate if any fuel melt has occurred.

3.2.2 The selection of a sample location should account for the type of event which will determine where the fission products will concentrate.

3.2.3 The recommended sampling locations are as follows:

Gaseous

Event Type	Sample Location
Nonbreaks (e.g., MSIV)	Suppression pool atmosphere
Small breaks	Drywell (before depressurization); suppression pool atmosphere (after depressurization)
Large breaks (liquid or steam) in primary containment	Drywell
Large breaks outside primary containment	Suppression pool atmosphere

3.2.4 The recommended sampling location for liquid for all events is the jet pumps as long as there is sufficient reactor pressure (normally > 50 psig) to provide a sample from that location. If there is not sufficient reactor pressure to allow a sample to be taken from the jet pumps, the sample should be taken from the sample points on the RHR System.

3.2.5 If a jet pump liquid sample is requested at low (< 1%) power conditions for a small break or nonbreak event, recommend to Operations that the reactor water level be raised to the level of the moisture separators. This will fully flood the moisture separators and will provide a thermally induced recirculation flow path for mixing.

3.3 Actions

3.3.1 Evaluations of Liquid and Gaseous Samples

NOTE: The extent of core damage can be determined by comparing the measured concentrations of major fission products in either the gas or water samples, after appropriate normalization, with the reference plant data.

3.3.1.1 The Plant Sampling and Analysis Team Leader should request samples from the PASS.

NOTE: Steps 3.3.1.2 through 3.3.1.7 can be accomplished using PASS, a computer program developed for use on the Dose Projection Team's IBM Personal Computer. To use the program, the Plant Sampling and Analysis Team Leader should complete Exhibit 3.6.3-7, Computer Inputs for the PASS Program, and give the completed exhibit to the Dose Projection Coordinator who will run the program and return the results. Exhibit 3.6.3-8 provides example test cases which can be used to verify that the computer program PASS is working properly. Expected results for known computer inputs are given. These test cases should be used to demonstrate the validity of PASS each time the program is initially used.

3.3.1.2 Obtain the samples from the PASS and determine the concentration of the fission product i (C_{wi} in water or C_{gi} in gas as determined in Attachment A using data provided in Exhibit 3.6.3-3).

3.3.1.3 Correct the measured concentration for decay to the time of reactor shutdown. Ensure that the measured gaseous activity concentration has been corrected for temperature and pressure difference in the sample vial and the containment (torus) gas phase.

NOTE: This is normally included in the quantitative analysis results.

3.3.1.4 Calculate the fission product inventory correction factor F_{Ii} per Attachment B and record on Worksheet A2.

3.3.1.5 Calculate the C_{wi} and C_{gi} using the information obtained in Step 3.3.1.2 and the methods in Attachment A and record on Worksheet A1.

3.3.1.6 Using the correction factors, determined in Attachments A and B, calculate the normalized concentration, C_{wi}^{Ref} or C_{gi}^{Ref}, per Attachment C and record on Worksheet A3.

3.3.1.7 Use Exhibit 3.6.3-2 to estimate the extent of fuel or cladding damage using C_{wi}^{Ref} for Cs-137 and I-131 and C_{gi}^{Ref} for Xe-133 and Kr-85. Record data on Worksheet A4.

3.3.2 <u>Evaluation of Metal-Water Reaction and Inventory Release</u>

3.3.2.1 Use Attachment D to determine the percent metal-water reaction. Record data on Worksheet B1.

3.3.2.2 Use Attachment E to determine the fuel inventory release to the containment. Record data on Worksheet B2.

3.3.3 <u>Application of Other Significant Parameters to Core Damage Estimate</u>

Section 3.3.1 provides an estimate of core damage based on radionuclide measurements. Based on Step 3.3.1.7, an initial assessment of core damage is made. Based on a clarification provided by the NRC, that assessment would appear in a matrix as follows:

Degree of Degradation	Minor (< 10%)	Intermediate (10% - 50%)	Major (> 50%)
No fuel damage	←	1	→
Cladding failure	2	3	4
Fuel overheat	5	6	7
Fuel melt	8	9	10

As recommended by the NRC, there are four general classes of damage and three degrees of damage within each of the classes except for the "no fuel damage" class.

Consequently, there are a total of ten possible damage assessment categories. For example, Category 3 would be descriptive of the condition where between 10% and 50% of the fuel cladding has failed. Note that the conditions of more than one category could exist simultaneously. The objective of the final core damage assessment procedure is to narrow down, to the maximum extent possible, those categories which apply to the actual in-plant situation.

The initial core damage assessment based on radionuclide measurement will provide one or several candidate categories which most likely represent the actual in-plant condition. The other parameters should then be evaluated (as identified in Section 3.3) to corroborate and further refine the initial estimate.

For example, fission product measurement using PASS may indicate Category 4 core damage and, additionally, the potential for fuel overheat and fuel melt (i.e., Categories 5 through 10). Measurement of hydrogen in containment and use of the hydrogen correlation provided in Attachment D is used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading along with the correlation provided in Attachment E would verify that a significant fission product release to the containment had occurred, further verifying the initial assessment.

Further analysis of the PASS samples for concentrations of Ba, Sr, La, and Ru and consideration of the relative amounts of fission products released would indicate if any fuel melt had occurred.

Exhibit 3.6.3-1 indicates how the analysis of the other significant parameters relates to the estimation of core damage based on radionuclide measurements.

3.3.4 Consult with the Dose Projection Coordinator and the Radiological Control Director when results of this procedure are determined and repeat this procedure as necessary.

4.0 References

Lin, C. C., "Procedure for the Determination of the Extent of Core Damage Under Accident Conditions," NEDO-22215, 1982.

Letter and Attachment from Mr. D. K. Smith, Service Supervisor - Nuclear, General Electric to Mr. A. C. Tollison, Jr., General Manager, Brunswick Steam Electric Plant, dated November 9, 1979, Subject: Radiation Source Term Information.

Letter and Attachments form Mr. T. J. Dente, Cnairman - BWR Owner's Group to Mr. D. G. Eisenhut, Licensing Director - USNRC, dated June 17, 1983, Subject: Transmittal of Generic Procedures for Estimation of Core Damage Using Postaccident Sampling System.

EXHIBIT 3.6.3-1

SEQUENCE OF ANALYSIS FOR
ESTIMATION OF CORE DAMAGE

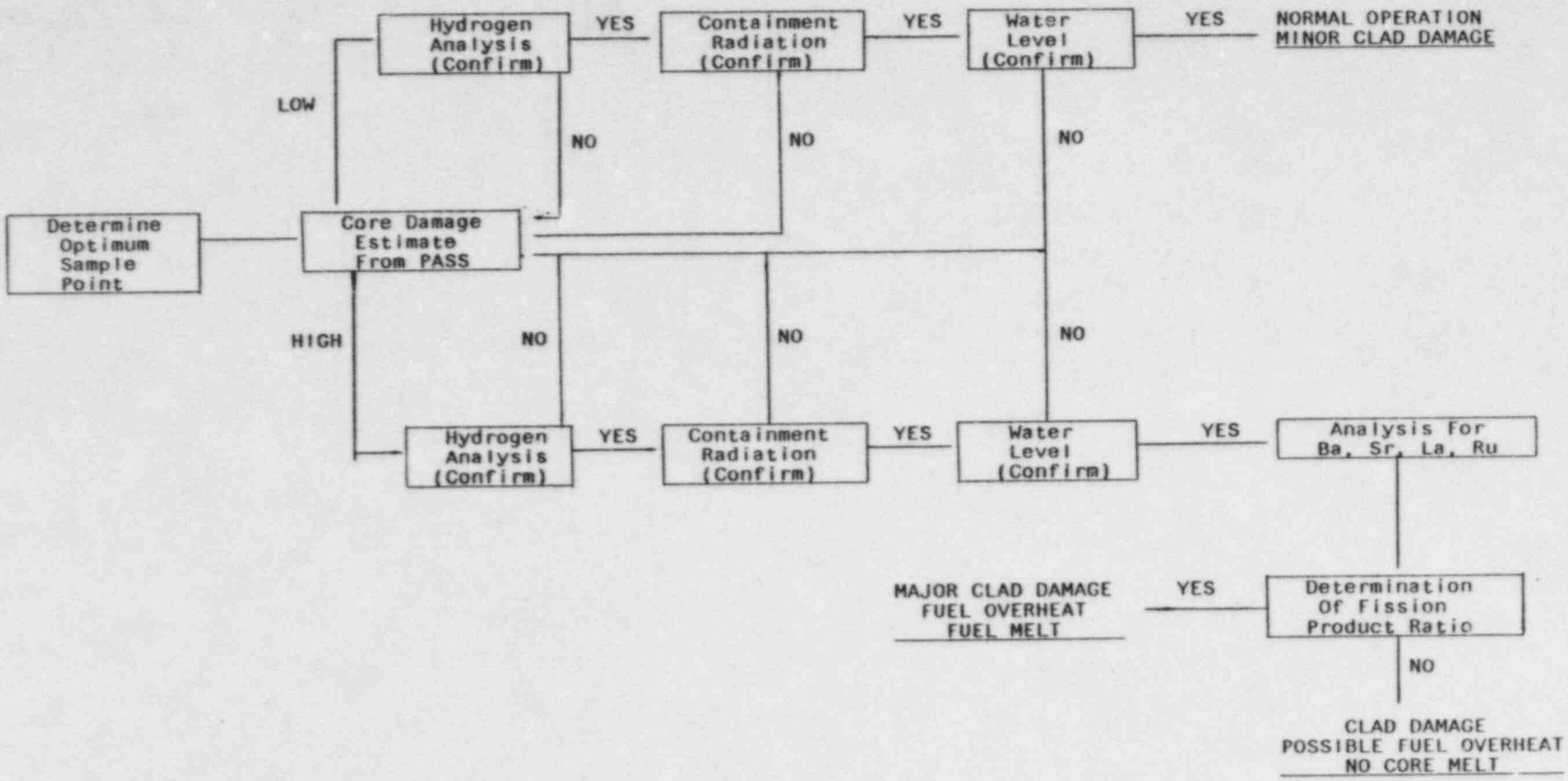
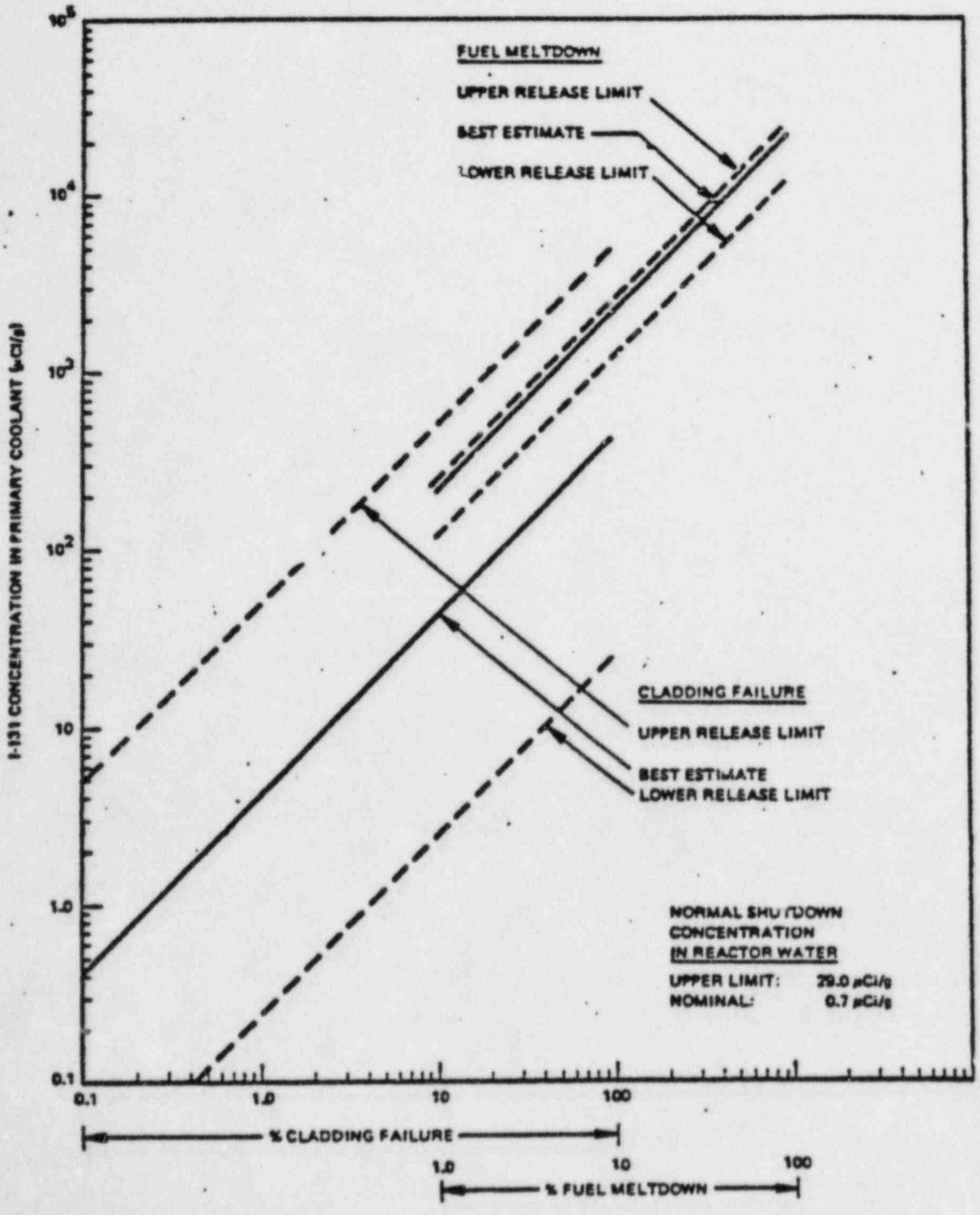
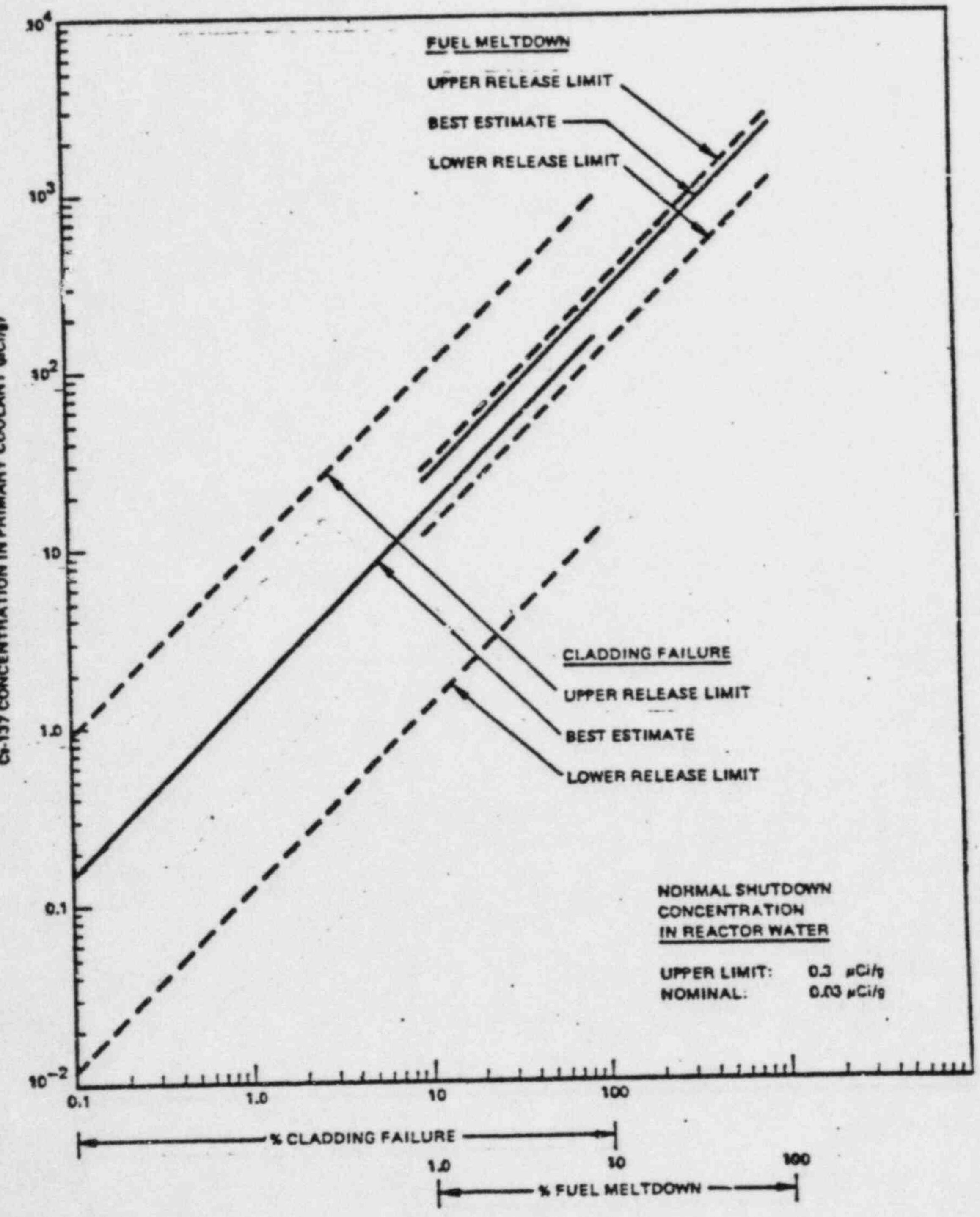


EXHIBIT 3.6.3-2



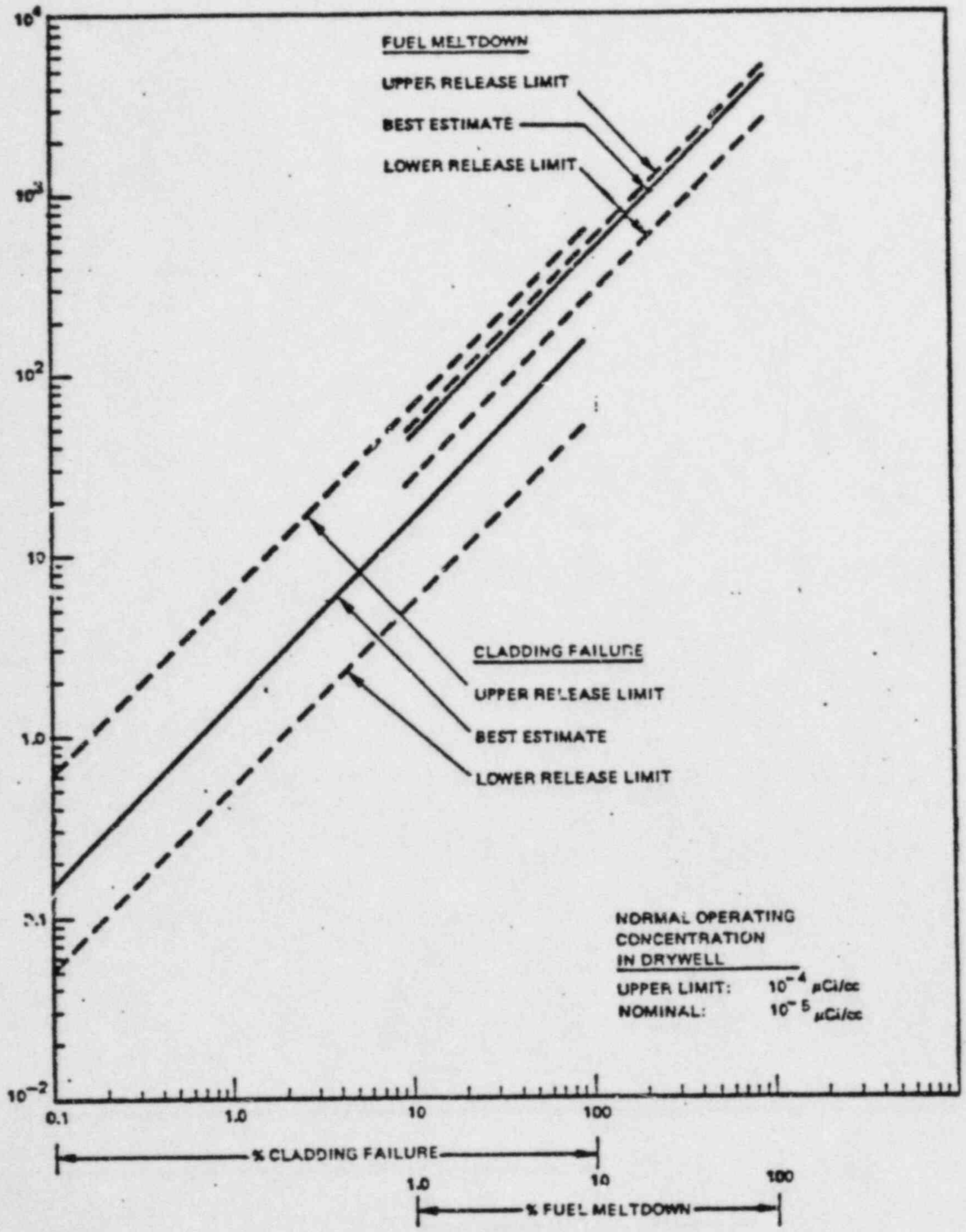
Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)



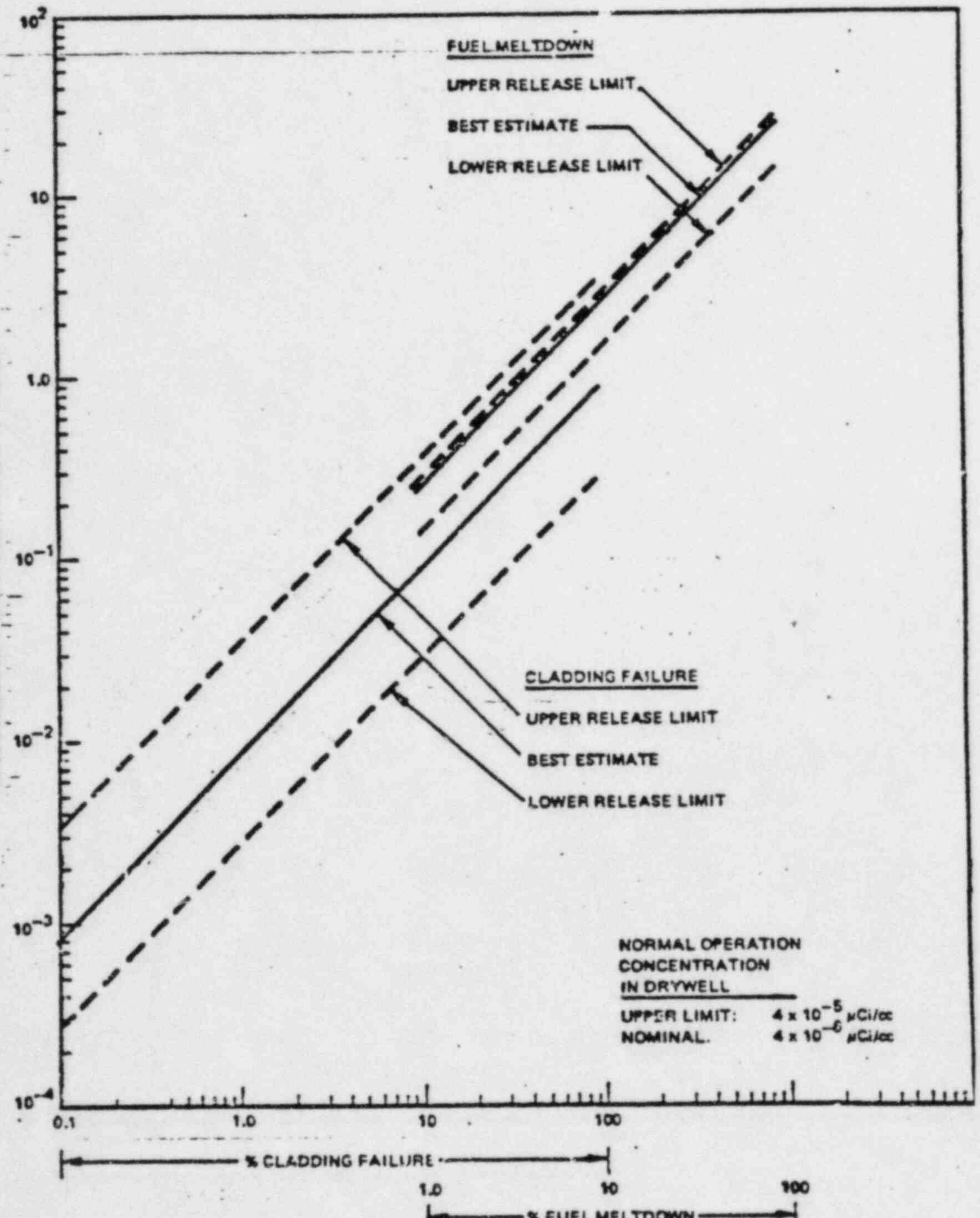
Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)



Relationship Between Xe-133 Concentration in the Containment Gas (Drywell Torus Gas) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)



Relationship Between Kr-85 Concentration in the Containment Gas (Drywell Torus Gas) and the Extent of Core Damage in Reference Plant

ATTACHMENT A

Plant Parameter Correction Factors

Fission products measured together for reactor water and suppression pool water or drywell gas and torus gas.

$$F_w = \frac{\text{BSEP total coolant mass } (2.69 \times 10^9 \text{ g})}{\text{reference plant coolant mass } (3.92 \times 10^9 \text{ g})}$$

$$= 0.68622$$

$$F_g = \frac{\text{BSEP total containment gas volume } (8.11 \times 10^9 \text{ cc})}{\text{reference plant containment gas volume } (4 \times 10^{10} \text{ cc})}$$

$$= 0.20275$$

Fission products measured separately for reactor water and suppression pool water or drywell gas and torus gas.

$$C_{wi} = \frac{\text{(conc. in Rx wtr) (Rx water mass) + (conc. in pool) (pool wtr mass)}}{\text{reactor water mass + pool water}}$$

$$= \frac{\text{(conc. in Rx water)}(2.14 \times 10^8 \text{ g}) + \text{(conc. in pool)}(2.48 \times 10^9 \text{ g})}{2.69 \times 10^9 \text{ g}}$$

$$C_{gi} = \frac{\text{(conc. in drywell) (drywell gas vol.) + (conc. in torus) (torus gas vol.)}}{\text{drywell gas volume + torus gas volume}}$$

$$= \frac{\text{(conc. in drywell)}(4.65 \times 10^9 \text{ cc}) + \text{(conc. in torus)}(3.46 \times 10^9 \text{ cc})}{8.11 \times 10^9 \text{ cc}}$$

ATTACHMENT B

Inventory Correction Factor

$$F_{Ii} = \frac{\text{inventory in reference plant}}{\text{inventory in operation plant}}$$

$$= \frac{3651 \left(1-e^{-1095\lambda_i}\right)}{\sum_j \left[P_j \left(1-e^{-\lambda_i T_j}\right) e^{-\lambda_i T_j^{o}}\right]}$$

where:

P_j = average steady reactor power operated in period j (MWt).

T_j = duration of operating period j (day).

T_j^{o} = time between the end of operating period j and the time of the last reactor shutdown (day).

3651 = reference plant MWt.

If the unit operating history is not readily available, use the following F_I values (based upon Brunswick plant operations under the same operational constraints):

Nuclide	Conservative F_I	λ (day $^{-1}$)
I-131	1.34	0.0862
Cs-137	1.39	6.29×10^{-5}
Xe-133	1.46	0.1320
Kr-85	1.51	1.77×10^{-4}

ATTACHMENT C

Comparison With Reference Plant Data

The extent of core damage can be estimated from the measured fission product concentrations in either the gas or water samples, as described for the reference plant. However, the measured concentration must be corrected for the differences in operation power level, time of operation, primary coolant mass, and containment gas volume.

$$C_{wi}^{Ref} = C_{wi} e^{\lambda_i t} \times F_{Ii} \times F_w$$

OR

$$C_{gi}^{Ref} = C_{gi} e^{\lambda_i t} \times F_{Ii} \times F_g$$

C_{wi}^{Ref} = Concentration of isotope i in the reference plant coolant (µCi/g).

C_{gi}^{Ref} = Concentration of isotope i in the reference plant containment gas (µCi/cc).

C_{wi} = Measured concentration of isotope i in BSEP's coolant (µCi/g). See Attachment A.

C_{gi} = Measured concentration of isotope i in BSEP's containment gas (µCi/cc). See Attachment A.

$e^{\lambda_i t}$ = Decay correction to the time of reactor shutdown.

λ_i = Decay constant of isotope i (day^{-1}).

t = Time between the reactor shutdown and the sample time (days).

F_{Ii} = Inventory correction factor for isotope i. See Attachment B.

F_g = Containment gas volume correction factor. See Attachment A.

F_w = Primary coolant mass correction factor. See Attachment A.

EXHIBIT 3.6.3-3

BSEP TO REFERENCE PLANT PARAMETERS

	Reference Plant	BSEP
Reactor Thermal Power	3651 MWt	2436 MWt
Number of Fuel Bundles	748 bundles	560 bundles
Total Primary Coolant Mass (reactor water plus suppression pool water)	3.92×10^{9} g	2.69×10^{9} g
Total Drywell and Torus Gas Space Volume	4.0×10^{10} cc	8.11×10^{9} cc
Reactor Water	2.46×10^{8} g	2.14×10^{8} g
Suppression Pool	3.67×10^{9} g	2.48×10^{9} g
Drywell Gas Volume	7.77×10^{9} cc	4.65×10^{9} cc
Torus Gas Volume	3.25×10^{10} cc	3.46×10^{9} cc

EXHIBIT 3.6.3-4

Core Inventory of Major Fission Products in a Reference Plant Operated at 3651 MWt for Three Years

Chemical Group	Isotope	Half-Life*	Inventory 10^6Ci	Major Gamma Ray Energy-Intensity - keV(γ/d)
Noble Gases	Kr-85m	4.48 h	24.6	151 (0.753)
	Kr-85	10.72 y	1.1	514 (0.0044)
	Kr-87	76.00 m	47.1	403 (0.495)
	Kr-88	2.84 h	66.8	196 (0.26), 1530 (0.109)
	Xe-133	5.25 d	202.0	81 (0.365)
	Xe-135	9.11 h	26.1	250 (0.899)
Halogens	I-131	8.04 d	96.0	364 (0.812)
	I-132	2.30 h	140.0	668 (0.99), 773 (0.762)
	I-133	20.80 h	201.0	530 (0.86)
	I-134	52.60 m	221.0	847 (0.954), 884 (0.653)
	I-135	6.59 h	189.0	1132 (0.225), 1260 (0.286)
Alkali Metals	Cs-134	2.06 y	19.6	605 (0.98), 796 (0.85)
	Cs-137	30.17 y	12.1	662 (0.85)
	Cs-138	32.20 m	178.0	463(0.307), 1436 (0.76)
Tellurium Group	Te-132	78.00 h	138.0	228 (0.88)
Noble Metals	Mo-99	66.02 h	183.0	740 (0.128)
	Ru-103	39.40 d	155.0	497 (0.89)
Alkaline Earths	Sr-91	9.52 h	115.0	750 (0.23), 1024 (0.325)
	Sr-92	2.71 h	123.0	1384 (0.9)
	Ba-140	12.80 d	173.0	537 (0.254)
Rare Earth	Y-92	3.54 h	124.0	934 (0.139)
	La-140	40.20 h	184.0	487 (0.455), 1597 (0.955)
	Ce-141	32.50 d	161.0	145 (0.48)
	Ce-144	284.40 d	129.0	134 (0.108)
Refractories	Zr-95	64.00 d	161.0	724 (0.437), 757 (0.553)
	Zr-97	16.90 h	166.0	743 (0.928)

* h = hour
d = day
m = month
y = year

ATTACHMENT D

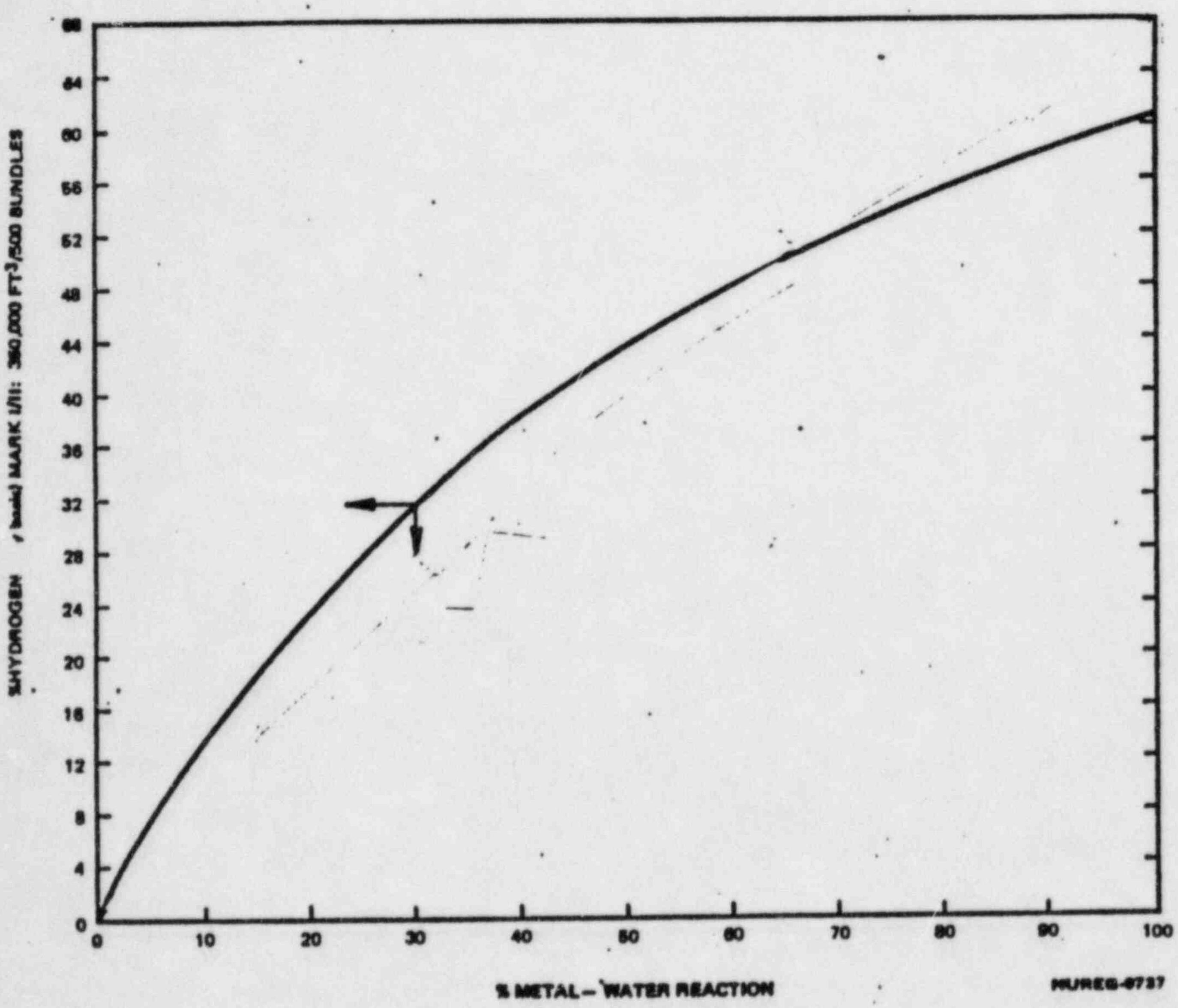
Integration of Containment Atmosphere Hydrogen Measurement Into Core Damage Estimate

The extent of fuel clad damage as evidenced by the extent of metal-water reaction can be estimated by determination of the hydrogen concentration in the containment. That concentration is measurable by either the containment hydrogen monitor or by the Postaccident Sampling System.

A correlation has been developed which relates containment hydrogen concentration to the percent metal-water reaction for Marks I and II type containments. That correlation is shown in Exhibit 3.6.3-5. Note A to that exhibit indicates the major assumptions used in developing the correlation. Note B indicates the method by which Brunswick plant can use the correlation to determine the extent of clad damage.

ATTACHMENT D (Cont'd)

EXHIBIT 3.6.3-5



Hydrogen Concentration for Marks I and II Containments as a Function of Metal-Water Reaction

ATTACHMENT D (Cont'd)

Note A to Exhibit 3.6.3-5
Analytical Assumptions
(For Marks I and II Containments)

1. Containment Volume = 350,000 ft^3
2. Number of Bundles = 500
3. Fuel Type = 8 x 8 R
4. All hydrogen from metal-water reaction released to containment.
5. Perfect mixing in containment.
6. No depletion of hydrogen (e.g., containment leakage).
7. Ideal gas behavior in containment.

ATTACHMENT D (Cont'd)

Note B to Exhibit 3.6.3-5

Determination of Clad Damage From Hydrogen Monitor Reading

Step 1. Obtain containment hydrogen monitor reading in percent.

Step 2. Using the curve in Exhibit 3.6.3-5, determine the metal-water reaction for the reference plant, MWR_{ref}.

Step 3. The metal-water reaction from the actual in-plant conditions (MWR) in determined from the following equation:

$$\% \; MWR = (MWR_{ref}) \times \frac{500}{N} \times \frac{V}{350{,}000}$$

where:

N = Number of Bundles = 560
V = Total Containment Free Volume, $ft^3 = 2.86 \times 10^5$

ATTACHMENT E

Integration of Containment Atmosphere Radiation Measurement Into Core Damage Estimate

An indication of the extent of core damage is the containment radiation level which is a measure of the inventory of fission products released to the containment. This attachment contains a correlation of the containment radiation monitor dose rate to the percent of fuel inventory airborne in the containment. The purpose of this attachment is to present that correlation and provide a method to use that correlation to determine the degree of core damage.

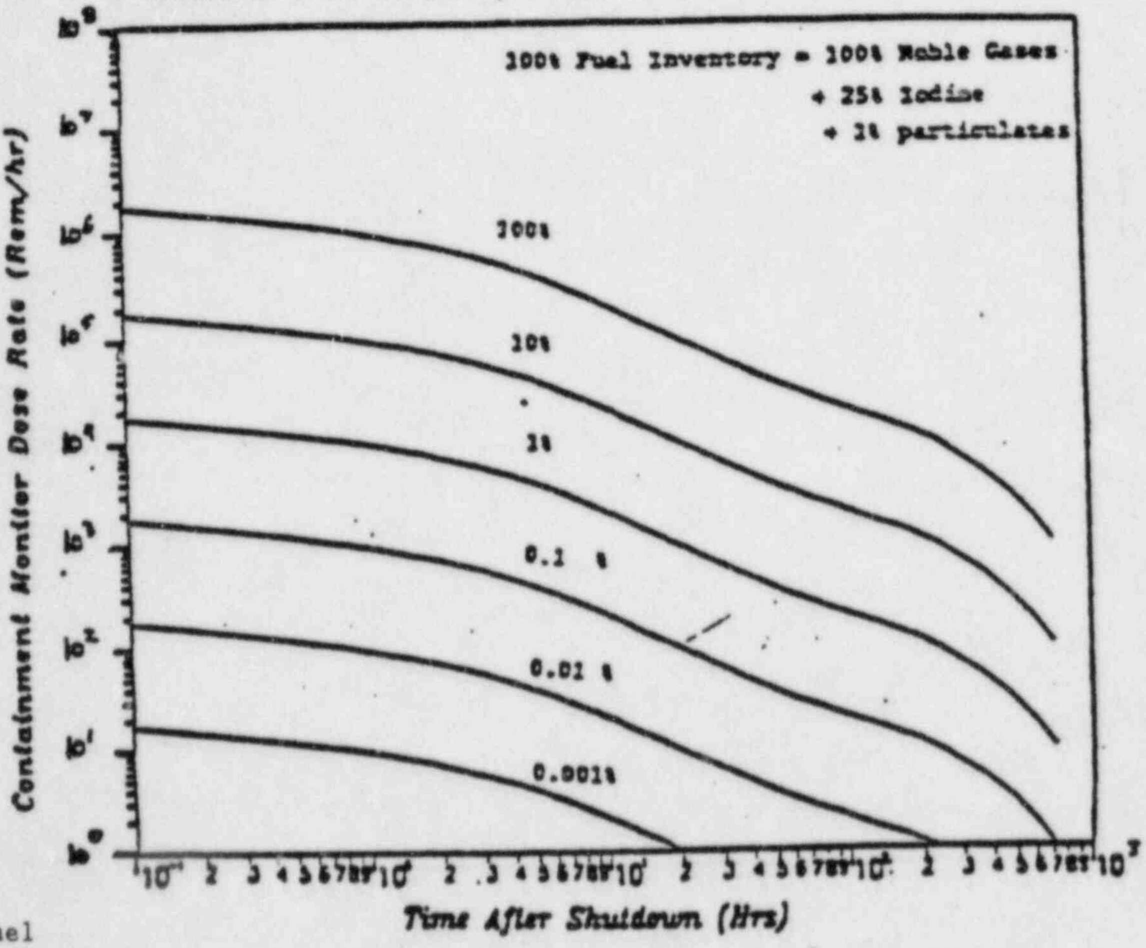
Exhibit 3.5.3-6 provides the results of a correlation performed for the Monticello plant. The key parameters which impact the containment dose rate are reactor power and containment volume.

The method whereby individual plants can apply this correlation is provided in Note A to Exhibit 3.6.3-6.

ATTACHMENT E (Cont'd)

EXHIBIT 3.6.3-6

Percent of Fuel Inventory Airborne in the Containment



% Fuel Inventory Released	Approximate Source and Damage Estimate
100.00	100% TID-14844, 100% fuel damage, potential core melt.
50.00	50% TID noble gases, TMI source.
10.00	10% TID, 100% NRC gap activity, total clad failure, partial core uncovered.
3.00	3% TID, 100% WASH-1400 gap activity, major clad failure.
1.00	1% TID, 10% NRC gap, maximum 10% clad failure.
0.10	0.1% TID, 1% NRC gap, 1% clad failure, local beating of 5-10 fuel assemblies.
0.01	0.01% TID, 0.1% NRC gap, clad failure of 3/4 fuel element (36 rods).
10^{-3}	0.01% NRC gap clad failure of a few rods.
10^{-4}	100% coolant release with spiking.
5×10^{-6}	100% coolant inventory release.
10^{-6}	Upper range of normal airborne noble gas activity in containment.

ATTACHMENT E (Cont'd)

NOTE A to Exhibit 3.6.3-6

Determination of Clad Damage From Containment Radiation Monitor Reading

The procedure for determination of fraction of fuel inventory released to the containment is as follows:

Step 1: Obtain containment radiation monitor reading, [R] in rem/hr.

Step 2: Determine elapsed time from plant shutdown to the containment radiation monitor reading [t] in hours.

Step 3: Using Exhibit 3.6.3-6, determine the fuel inventory release for the reference plant $[I]_{ref}$ in percent.

Step 4: Determine the inventory release to the containment [I] using the following formula:

$$[I] = [I]_{ref}\left(\frac{1670}{P}\right)\left(\frac{V}{237,450}\right)$$

where:

P = reactor power level MW_{th} (BSEP = 2436 MW_{th}).

V = total containment free volume, ft^3 (BSEP = 286, 370 ft^3).

NOTE: Monitor location within the containment is assumed to have an insignificant impact on dose rate due to fuel inventory airborne in containment.

EXHIBIT 3.6.3-7

Computer Inputs for the PASS Program

Concentration of I-131 in Reactor Water (µCi/ml)	________
Concentration of I-131 in Suppression Pool (µCi/ml)*	________
Concentration of Cs-137 in Reactor Water (µCi/ml)	________
Concentration of Cs-137 in Suppression Pool (µCi/ml)*	________
Concentration of Xe-133 in Drywell (µCi/cc)	________
Concentration of Xe-133 in Torus (µCi/cc)**	________
Concentration of Kr-85 in Drywell (µCi/cc)	________
Concentration of Kr-85 in Torus (µCi/cc)**	________
Time between Reactor Shutdown and Sample Time (days)	________

If time and availability permits, attach information necessary for the calculation of Inventory Correction Factors (see Attachment B); otherwise, conservative default correction factors will be used.

Plant Sampling and Analysis Team Leader: Give completed exhibit to Dose Projection Coordinator.

Dose Projection Coordinator: Enter data into PASS computer program and provide results to Plant Sampling and Analysis Team Leader.

*If unavailable, assume suppression pool activity = 0 µCi/ml.
**If unavailable, assume torus concentration equal to drywell in µCi/cc.

EXHIBIT 3.6.3-8

VERIFICATION OF PASS
(A Computer program for estimating core damage based on Postaccident Sampling System results)

This exhibit is intended to provide a means to ensure that PASS, a core damage estimate program designed for the IBM Personal Computer, is working properly. This is demonstrated by duplicating expected results of known computer inputs. These results can be validated by comparison to manual calculations for the same input.

Two different test cases are presented so that a number of alternate paths within the program can be tested. The test cases with their expected results follow.

TEST CASE 1

Computer Prompt	Expected Input
Enter The Concentration of the Fission Products	
Concentration of I-131 in Reactor Water (µCi/ml)	1.72E + 3
Concentration of I-131 in Suppression Pool (µCi/ml)	1.49E + 2
Concentration of Cs-137 in Reactor Water (µCi/ml)	6.55E + 2
Concentration of Cs-137 in Suppression Pool (µCi/ml)	5.70E + 1
Concentration of Xe-133 in Drywell (µCi/cc)	1.82E + 2
Concentration of Xe-133 in Torus (µCi/cc)	2.41E + 2
Concentration of Kr-85 in Drywell (µCi/cc)	1.43E + 0
Concentration of Kr-85 in Torus (µCi/cc)	1.90E + 0

For the inventory correction factor do you want to use the conservative default values which are bases upon BSEP's operations under the same operational constraints (YES or NO)? YES

Enter time between the reactor shutdown and the Sample Time (Days) 2

The results should resemble the printout on the following page. If they do not, carefully check your inputs and try the test again. If the results still are not similar, try a backup copy of the program. If that fails, then seek programming help.

EXHIBIT 3.6.3-8 (Cont'd)
ESTIMATE THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

DATE: 03-28-1984
TIME: 13:21:27

The concentration of the fission products are:

I-131 in Reactor Water	1.72E + 3 µCi/ml
I-131 in Suppression Pool	1.49E + 2 µCi/ml
Cs-137 in Reactor Water	6.55E + 2 µCi/ml
Cs-137 in Suppression Pool	5.70E + 1 µCi/ml
Xe-133 in Drywell Air	1.82E + 2 µCi/cc
Xe-133 in Torus Air	2.41E + 2 µCi/cc
Kr-85 in Drywell Air	1.43E + 0 µCi/cc
Kr-85 in Torus Air	1.90E + 0 µCi/cc

Time between the reactor shutdown and the sample time is: 2 days

The Concervative Default values of the Inventory Correction Factors were used.

Estimate of fuel/cladding damage
Primary Coolant Analysis

Nuclide	CwREF (µCi/ml)	% Cladding Failure	% Fuel Meltdown
I-131	3.00E + 02	69.00	1.35
Cs-137	1.00E + 02	64.54	4.27

Containment Gas Analysis

Nuclide	CwREF (µCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe-133	7.99E + 01	53.26	1.84
Kr-85	5.00E - 01	56.35	1.92

EXHIBIT 3.6.3-8 (Cont'd)

TEST CASE 2

Computer Prompt	Expected Input
Enter The Concentration of the Fission Products	
Concentration of I-131 in Reactor Water (µCi/ml)	1.35E + 3
Concentration of I-131 in Suppression Pool (µCi/ml)	1.18E + 2
Concentration of Cs-137 in Reactor Water (µCi/ml)	1.17E + 2
Concentration of Cs-137 in Suppression Pool (µCi/ml)	1.02E + 1
Concentration of Xe-133 in Drywell (µCi/cc)	1.84E + 2
Concentration of Xe-133 in Torus (µCi/cc)	2.45E + 2
Concentration of Kr-85 in Drywell (µCi/cc)	2.91E - 1
Concentration of Kr-85 in Torus (µCi/cc)	3.86E - 1
For the inventory correction factor do you want to use the conservative default values which are bases upon BSEP's operations under the same operational constraints (YES or NO)?	NO
Enter time between the reactor shutdown and the Sample Time (Days)?	2
Enter number of Operating Periods from the unit operating history?	3
For period number (1) enter:	
Average steady reactor power operated in this period (MWT)?	1000
Duration of this operating period (days)?	60
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	254
For period number (2) enter:	
Average steady reactor power operated in this period (MWT)?	2000
Duration of this operating period (days)?	200
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	44
For period number (3) enter:	
Average steady reactor power operated in this period (MWT)?	3000
Duration of this operating period (days)?	14
Time between the end of this operating period and the time of the most recent reactor shutdown (days)?	0

The results should resemble the printout on the following page. If they do not, carefully check your inputs and try the test again. If the results still are not similar, try a backup copy of the program. If that fails, then seek programming help.

EXHIBIT 3.6.3-8 (Cont'd)
ESTIMATE THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS
DATE: 03-28-1984
TIME: 13:27:17

The concentration of the fission products are:

I-131 in Reactor Water	1.35E + 3 μCi/ml
I-131 in Suppression Pool	1.18E + 2 μCi/ml
Cs-137 in Reactor Water	1.17E + 2 μCi/ml
Cs-137 in Suppression Pool	1.02E + 1 μCi/ml
Xe-133 in Drywell Air	1.84E + 2 μCi/cc
Xe-133 in Torus Air	2.45E + 2 μCi/cc
Kr-85 in Drywell Air	2.91E - 1 μCi/cc
Kr-85 in Torus Air	3.86E - 1 μCi/cc

Time between the reactor shutdown and the sample time is: 2 days

The Inventory Correction Factors were calculated from the following:

Period No.	Operation Time (days)	Time Between Period & Last Shutdown (days)	Average Power (MWt)
1	60	254	1000
2	200	44	2000
3	14	0	3000

Estimate of Fuel/Cladding Damage
Primary Coolant Analysis

Nuclide	CwREF (μCi/ml)	% Cladding Failure	% Fuel Meltdown
I-131	3.00E + 02	69.02	1.35
Cs-137	9.99E + 01	64.49	4.27

Containment Gas Analysis

Nuclide	CwREF (μCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe-133	8.00E + 01	53.30	1.84
Kr-85	5.00E - 01	56.40	1.92

WORKSHEET A1

CALCULATION OF ISOTOPIC CONCENTRATIONS IN PRIMARY WATER AND SUPPRESSION POOL WATER (Cw_i) AND DRYWELL GAS AND TORUS GAS (Cg_i)

References

Section 3.3.1.2
Section 3.3.1.5
Attachment A
Exhibit 3.6.3-3

Cw_i (µCi/ml) = (Concentration Rx H_2O)$_i$ (0.08) +
(Cs^{137}, I^{131}) (Concentration Suppression Pool H_2O)$_i$ (0.92)

= [____________ + ____________] (µCi/ml)

= ____________ µCi/ml$_{Cs}{}^{137}$

and = ____________ µCi/ml$_{I}{}^{131}$

Cg_i (µCi/ml) = (Concentration Drywell)$_i$ (0.57) + (Concentration Torus)$_i$ (0.43)
(Xe^{133}, Kr^{85})

= [____________ + ____________] (µCi/cc)

= ____________ µCi/cc$_{Xe}{}^{133}$

and = ____________ µCi/cc$_{Kr}{}^{85}$

WORKSHEET A2

CALCULATION OF INVENTORY CORRECTION FACTOR (FI_i)

References

Section 3.3.1.4
Attachment B
Exhibit 3.6.3-4

P_j = __________ $MW_{thermal}$

T_{j^o} = __________ Days

T_j = __________ Days

λ_i = __________ $Days^{-1}$

$$FI_i = \frac{3651\,(1 - e^{-1095\lambda_i})}{\Sigma_j \left[P_j\,(1 - e^{-\lambda_i T_j})(e^{-\lambda_i T_{j^o}})\right]}$$

= ______________________________ (Cs^{137})

______________________________ (I^{131})

______________________________ (Xe^{133})

______________________________ (Kr^{85})

WORKSHEET A3

CALCULATION OF NORMALIZED ISOTOPIC CONCENTRATIONS IN PRIMARY WATER AND SUPPRESSION POOL WATER (Cw_i^{Ref}) AND DRYWELL GAS AND TORUS GAS (Cg_i^{Ref})

References

Section 3.3.1.6
Attachment C
Worksheet A1
Worksheet A2

NOTE: For BSEP,
$Fw = 0.68622$
$Fg = 0.20275$

$Cw_i^{Ref} = Cw_i \, e^{\lambda it} \times FI_i \times Fw$
(Cs^{137}, I^{131})

= ______________________________ $\mu Ci/ml_{Cs^{137}}$

______________________________ $\mu Ci/ml_{I^{131}}$

$Cg_i^{Ref} = Cg_i \, e^{\lambda it} \times FI_i \times Fg$
(Xe^{133}, Kr^{85})

= ______________________________ $\mu Ci/cc$ Xe^{133}

______________________________ $\mu Ci/cc$ Kr^{85}

WORKSHEET A4

ESTIMATE OF FUEL/CLADDING DAMAGE

References

Section 3.3.1.7
Exhibit 3.6.3-2
Worksheet A3

Primary Coolant Analysis

Isotope	Cw_i^{Ref} (µCi/ml)	% Cladding Failure	% Fuel Meltdown
I^{131}			
Cs^{137}			

Containment Gas Analysis

Isotope	Cg_i^{Ref} (µCi/ml)	% Cladding Failure	% Fuel Meltdown
Xe^{133}			
Kr^{85}			

WORKSHEET B1

DETERMINATION OF CLAD DAMAGE FROM HYDROGEN MONITOR READING

References

Section 3.4.1
Attachment D
Exhibit 3.6.3-5

Containment Hydrogen Monitor Reading: _______________%

MWR ref : _______________%

Calculate % MWR:

% MWR = (MWR ref)(0.73)
= _______________

WORKSHEET B2

DETERMINATION OF FUEL INVENTORY RELEASE BASED ON CONTAINMENT RADIATION MONITOR READING

References

Section 3.4.2
Attachment E
Exhibit 3.6.3-6

Containment Radiation Monitor Reading: ____________________ rem/hr

Time from Shutdown to Monitor Reading: ____________________ hrs

[I] ref (Reference Fuel Inventory
Release, %, from Exhibit 3.6.3-6) : ____________________ %

I (Actual Fuel Inventory Release) = [I] ref * 0.827

= ___________ %