CP&L

Carolina Power & Light Company

SERIAL: LAP-83-579

DEC 21 1983

Director of Nuclear Reactor Regulation
Attention: Mr. D. B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing
United States Nuclear Regulatory Commission
Washington, DC 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
NUREG-0737 ITEM II.B.3, POST ACCIDENT SAMPLING SYSTEM
ADDITIONAL INFORMATION SUBMITTAL

Dear Mr. Vassallo:

In your letter dated October 29, 1983, Carolina Power & Light Company (CP&L) was asked to provide some additional information concerning NUREG-0737, Item II.B.3, Post Accident Sampling System.

Please find enclosed the actions that were taken to satisfy criteria numbers 1, 2, and 10 of NUREG-0737, Item II.B.3 for Brunswick Unit Nos. 1 and 2. Also please find enclosed a revision to our previous response to criterion number 4 related to dissolved gas sampling capabilities. This revision is necessary due to vendor equipment design problems discovered during system pre-operational testing. The Commission has been informed earlier of this generic system deficiency by General Electric and BWR Owners' Group.

Should you have any question concerning this letter, please do not hesitate to contact a member of our licensing staff.

Yours very truly,

P. W. Howe

P. W. Howe
Vice President
Brunswick Nuclear Project

PPC/ccc (9101PPC)
Enclosure

cc: Mr. D. O. Myers (NRC-BSEP)
Mr. J. P. O'Reilly (NRC-RII)
Mr. M. Grotenhuis (NRC)

A046
1/1

8312290356 831221
PDR ADOCK 05000324
P PDR

SUPPLEMENTAL RESPONSE TO NUREG-0737, ITEM II.B.3
POST ACCIDENT SAMPLING CAPABILITY
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
CAROLINA POWER & LIGHT COMPANY

REFERENCE:

Letter, Domenic B. Vassallo to Mr. E. E. Utley, October 29, 1983, "Request for Additional Information, Post Accident Sampling System (NUREG-0737, II.B.3)"

The following presentation addresses NRC comments regarding CP&L's initial response to these criteria as previously submitted via our letter of January 28, 1983.

CRITERION 1

NRC COMMENT:

"Provide information regarding provisions for sampling in the event of loss of offsite power during an accident which requires post-accident sampling."

CP&L RESPONSE:

All equipment necessary (isolation valves, sample station power, gas line heat trace) to obtain the required post-accident sample is powered from the emergency AC distribution system which is fed by the emergency diesel generators. The systems is, therefore, reliably energized in the event of loss of offsite power.

CRITERION 2

NRC COMMENT:

Provide a core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage.

CP&L RESPONSE:

Please find enclosed a draft copy of Plant Emergency Procedure PEP-03.6.3 entitled, "Estimate of the Extent of Core Damage Under Accident Conditions," (Attachment 1). The procedure will be fully implemented by January 31, 1984.

CRITERION 10

NRC COMMENT:

Provide information demonstrating applicability of procedures and instrumentation in the post-accident water chemistry and radiation environment, and retraining of operators on semi-annual basis.

CP&L RESPONSE:

All equipment and procedures which are used for post-accident sampling and analysis are calibrated or tested at a frequency which will ensure, to a high degree of reliability, that this equipment and procedures will be available if required.

The following plant procedures control the operation and maintenance of the primary instruments required for post-accident sample analysis:

E&RC - 1199, "Cation and Anion Analysis by Ion Chromatography"
E&RC - 1222, "Operation of Gas Chromatograph"
E&RC - 1300, "Operation and Linearity Verification of pH Meters"
E&RC - 1504, "Post-Accident Chloride Analysis by Ion Chromatography"
E&RC - 2201, "Calibration/Operation of ND 6600 Multichannel Analyzer"

All of this equipment is being utilized on a routine basis for normal plant chemistry and counting functions, thereby assuring equipment operability, calibration and personnel qualification.

Utilization of the P.A.S.S. is included in semi-annual radiation protection drills as specified in Plant Emergency Procedure PEP-04.3, "Performance of Training, Exercises and Drills." Additionally, the system is tested semi-annually by withdrawing at least one liquid sample and one gas sample.

Approximately 75% of the E&C staff has received initial training and is presently qualified to operate the sample station. The balance of the E&C staff is scheduled to complete training by December 31, 1983. Refresher training for the operators will be provided at a six month interval.

CRITERION 4

The following presentation represents a revision to CP&L's initial response to Criterion 4 as previously submitted via our letter of January 28, 1983.

ORIGINAL REQUIREMENT:

Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or Hydrogen gas in reactor coolant samples is considered adequate. Measuring the Oxygen concentration is recommended, but is not mandatory.

ORIGINAL CP&L RESPONSE: (Via Letter, S. R. Zimmerman to D. B. Vassallo, January 29, 1983)

No capability to grab sample pressurized reactor coolant has been installed.

Alternatively, equipment for stripping dissolved gases from a fixed-volume pressurized liquid sample is provided in the General Electric PASS station. A fifteen milliliter vial gas sample is thus obtained for hydrogen/oxygen analysis via gas chromatography and for quantitative analysis for total dissolved gases.

REVISED CP&L RESPONSE:

Subsequent to our previous response, as stated above, we have encountered some problems with the dissolved gas sampling function of the General Electric system installed at Brunswick Units 1 and 2.

During pre-operational acceptance testing we discovered problems related to sensitivity, system dynamics, leaking valves and instrumentation. In addition to the functional and analytical difficulties observed, this also created the potential for excessive radiation doses to system operators since, on several occasions during our testing, liquid was actually collected in the bottle utilized for dissolved gas sampling.

In order to eliminate the above defined hazards, the dissolved gas sampling function of the two Brunswick systems was completely disabled. Adequate mechanical and electrical modifications were made to the General Electric equipment to prevent its inadvertent operation.

As a result of the problems observed by CP&L and other utilities installing the General Electric system, the BWR Owners' Group and General Electric have been active in identifying the scope of the problems and have initiated a strong effort to complete a satisfactory redesign of this portion of the system.

The nature of this problem and the efforts to remedy it have been previously identified to the Commission on a generic basis by the BWR Owners' Group and General Electric in a meeting held on July 20, 1983. Since that meeting, General Electric's projected date for provision of the necessary FDI's and modification hardware has slipped from the original target of October-November 1983 until the spring of 1984. Upon receipt of these materials, CP&L will install the modifications in order to satisfy the requirements of Criterion 4.

ATTACHMENT #1

CAROLINA POWER & LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT

UNIT NO. 0

ESTIMATE OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

PLANT EMERGENCY PROCEDURE: PEP-03.6.3

VOLUME XIII

Rev. 002

Recommended By: ______________________________ Date: ______________
Manager - E&RC

Approved By: ______________________________ Date: ______________
General Manager

LIST OF EFFECTIVE PAGES

PEP-03.6.3

Page(s)	Revision
1 - 22	2

PEP-03.6.3 ESTIMATE OF THE EXTENT OF CORE DAMAGE UNDER ACCIDENT CONDITIONS

1.0 Responsible Individual and Objectives

The Radiological Control Director is responsible to the Site Emergency Coordinator for determining the magnitude and rate of potential radioactive releases to the environment. The Radiological Control Director may delegate the calculational aspects to the Dose Projection Coordinator.

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 as interruption of adequate core cooling. Significant periods of core uncovery, 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

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

3.1 Limitations

3.1.1 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.

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

3.1.3 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

3.1.4 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.

3.1.5 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.

3.2 Evaluation of Liquid and Gaseous Samples

3.2.1 Have the plant monitoring team leader request samples from the PASS.

3.2.2 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.

3.2.3 Obtain the samples from the PASS and the concentration of the fission product i (C_{wi} in water or C_{gi} in gas is determined). See Appendix A.

3.2.4 Correct the measured concentration for decay to the time of reactor shutdown.

3.2.5 Ensure that the measured gaseous activity concentration has been corrected for temperature and pressure difference in the sample vial and the contaminated (torus) gas phase.

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

3.2.6 Calculate the fission product inventory correction factor F_{Ii} per Appendix B.

3.2.7 Calculate the C_{wi} and C_{gi} using the information obtained in Step 3.2.3 and the methods in Appendix A.

3.2.8 Obtain the plant parameter correction factor.

3.2.9 Using the correction factors, calculate the normalized concentration, C_{wi}^{Ref} or C_{gi}^{Ref}, per Appendix C.

3.2.10 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.

3.2.11 Primary coolant and containment gas.

Isotope	% Cladding Failure	% Fuel Meltdown
I-131		
Cs-137		
Xe-133		
Kr-85		

3.3 Evaluation of Metal-Water Reaction and Inventory Release

3.3.1 Use Appendix D to determine the percent metal-water reaction.

Percent metal-water reaction ____________.

3.3.2 Use Appendix E to determine the fuel inventory release to the containment.

Fuel inventory release to containment ________________.

3.4 Application of Other Significant Parameters to Core Damage Estimate

Section 3.2 provides an estimate of core damage based on radionuclide measurements. Based on 3.2.11, 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 10 possible damage assessment categories. For example, Category 3 would be descriptive of the condition where between 10 and 50 percent 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-pl nt condition The other parameters should then be evaluated, as identifed 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 Appendix D is used to verify that extensive clad damage had occurred. Use of the containment radiation monitor reading along with the correlation provided in Appendix E would verify that a significant fission product release to the contaiment 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.

4.0 References

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

EXHIBIT 3.6.3-1

SEQUENCE OF ANALYSIS FOR
ESTIMATION OF CORE DAMAGE

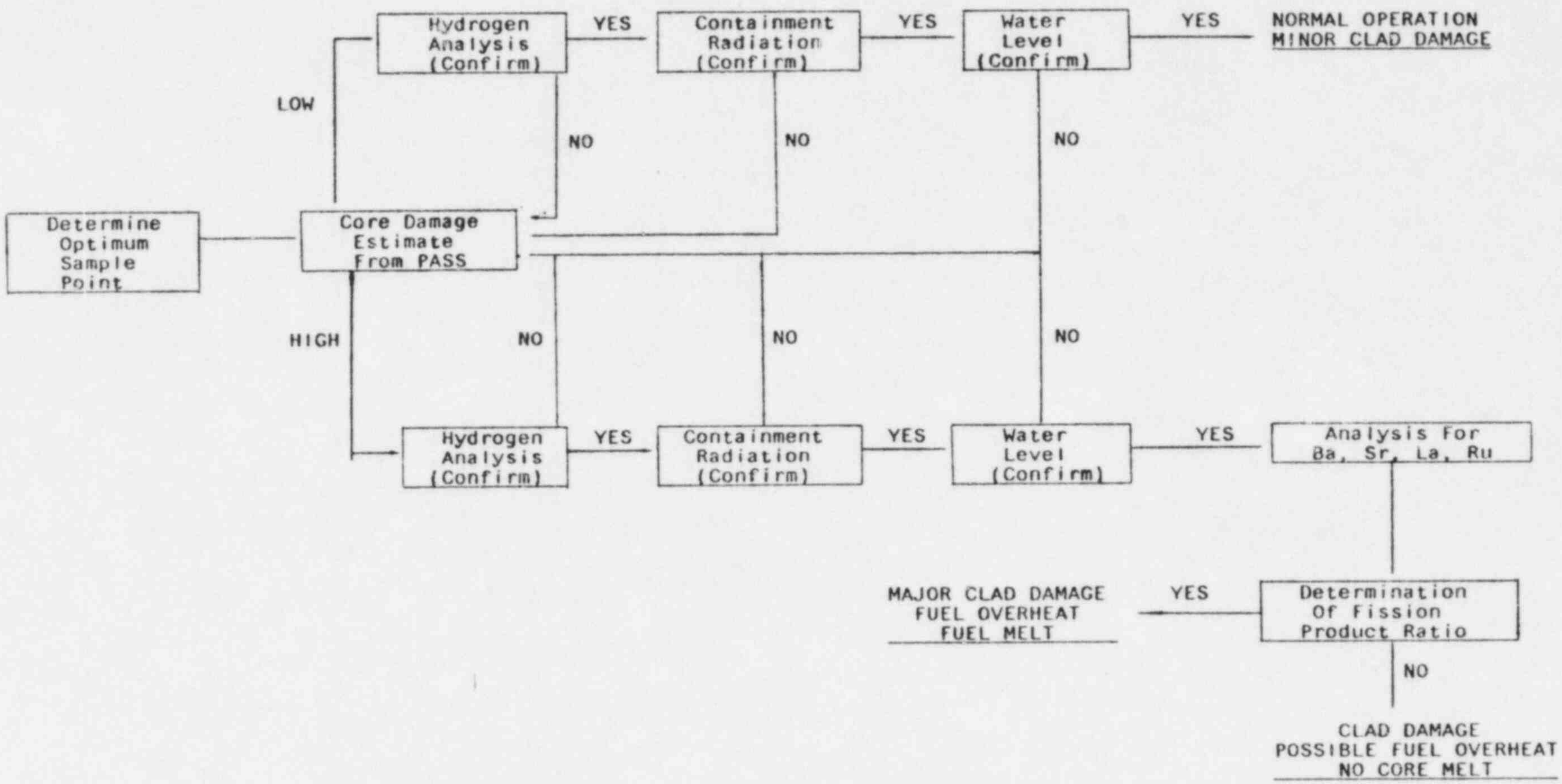
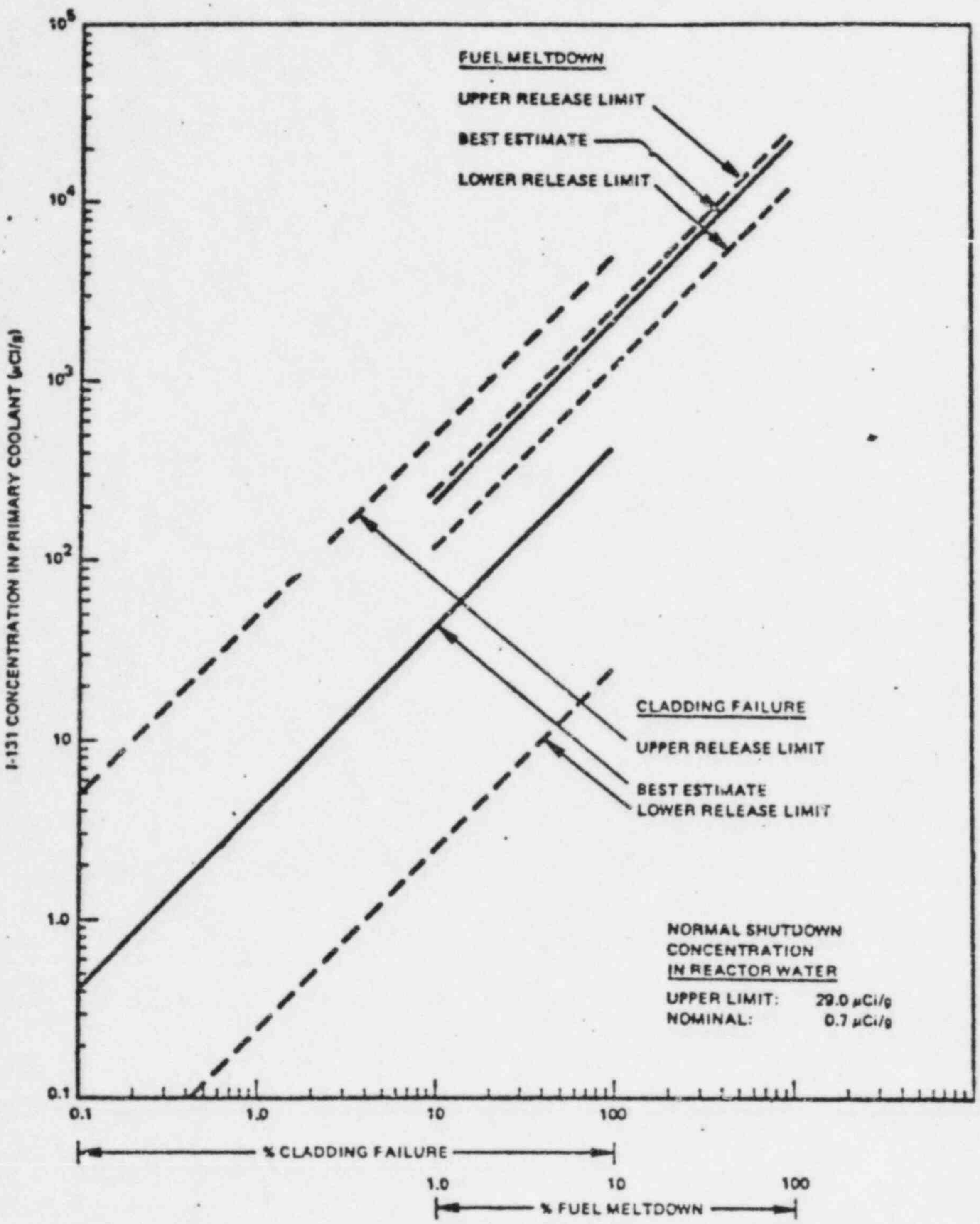
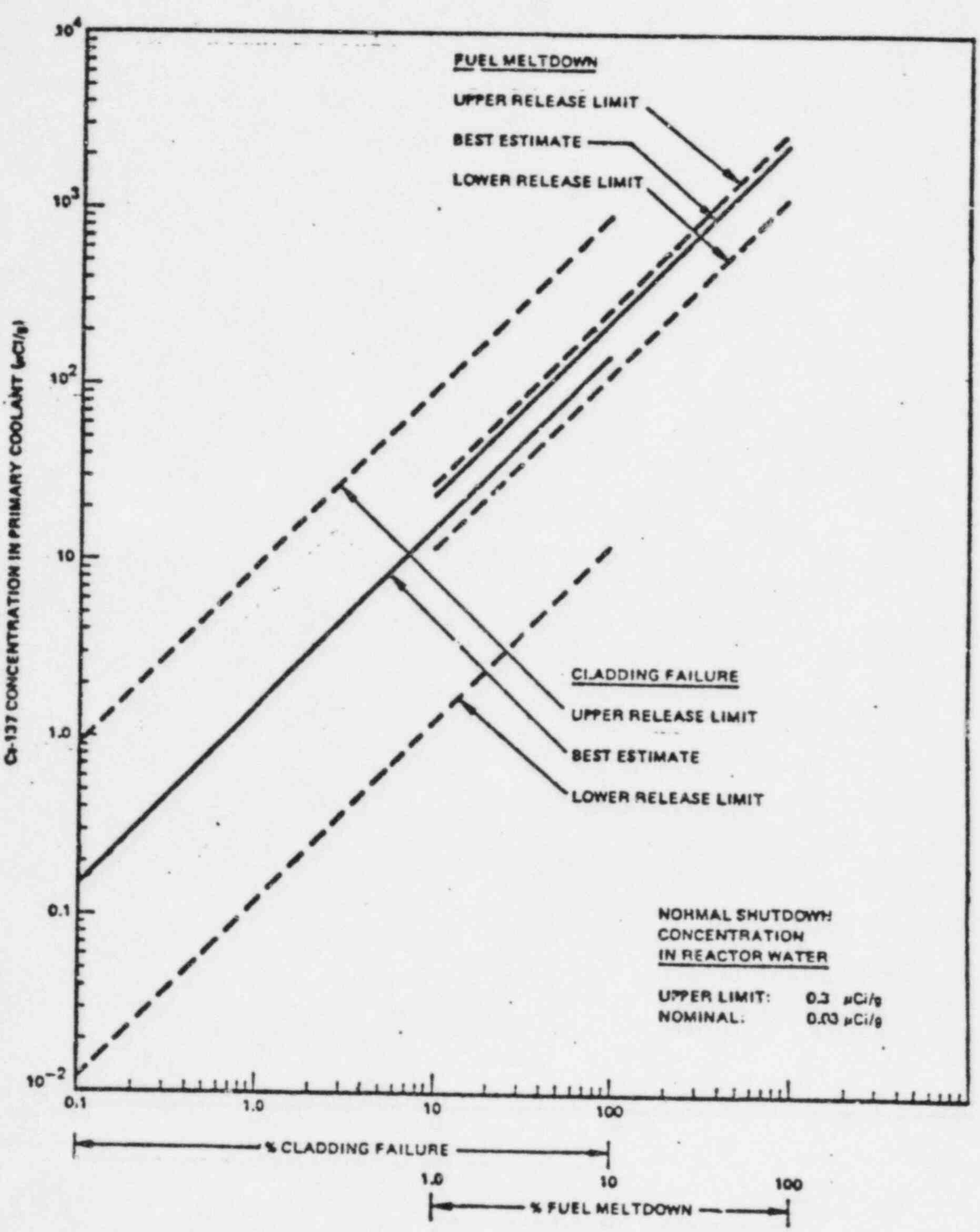


EXHIBIT 3.6.3-2



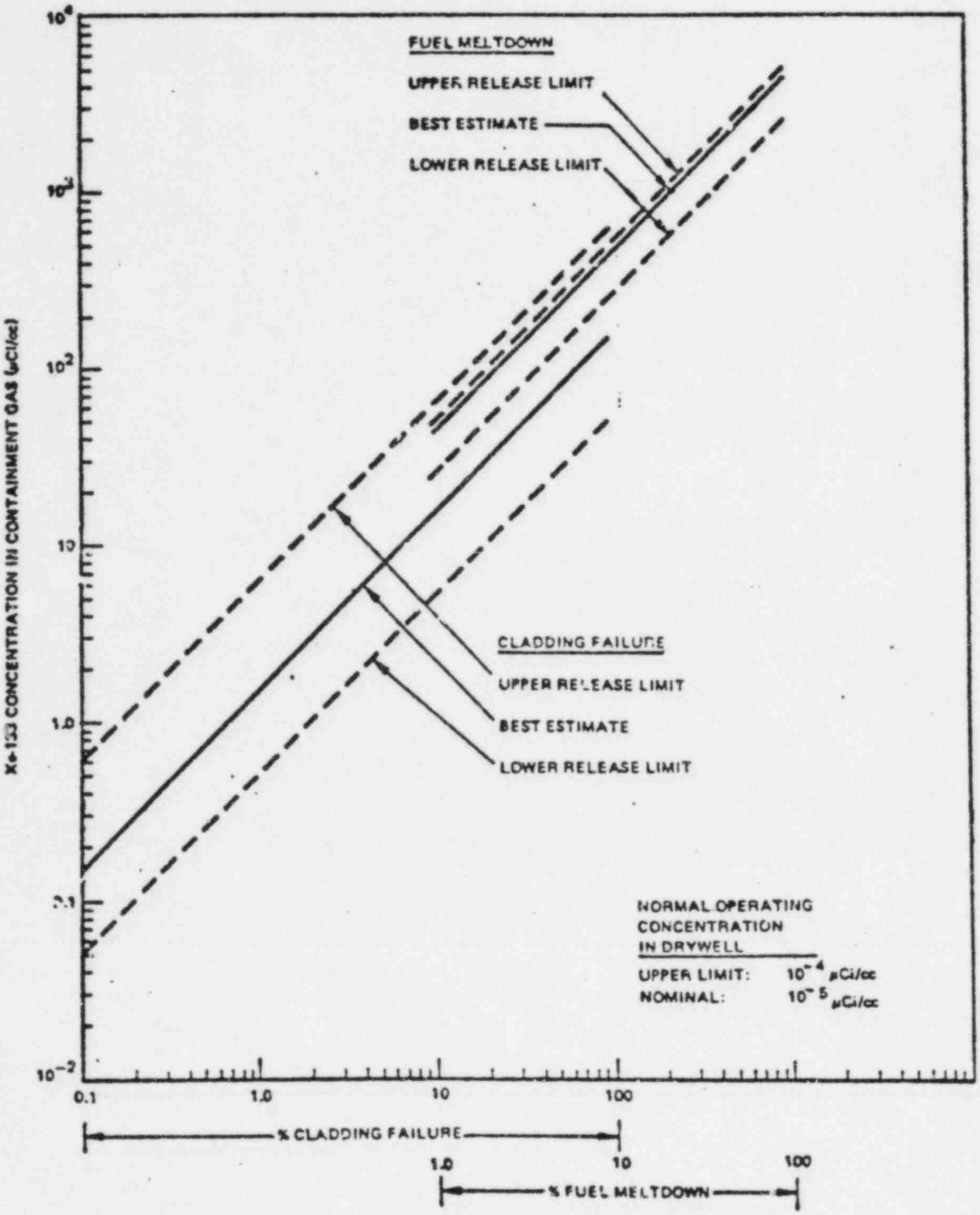
Relationship Between I-131 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant

EXHIBIT 3.6.3-2 (Cont'd)

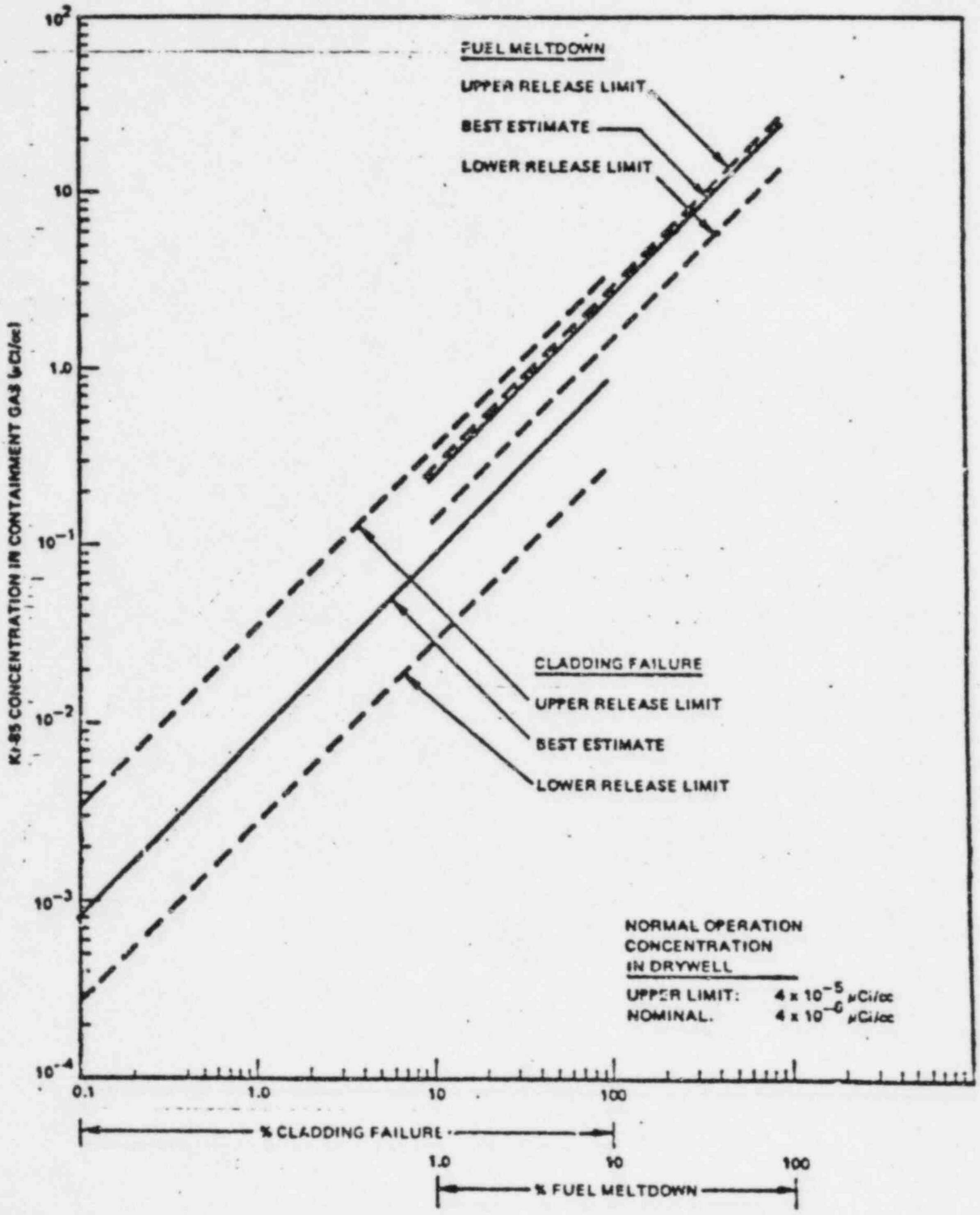


Relationship Between Cs-137 Concentration in the Primary Coolant (Reactor Water + Pool Water) and the Extent of Core Damage in Reference Plant.

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



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

APPENDIX 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.

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

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

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

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

APPENDIX B

Inventory Correction Factor

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

$$= \frac{3579\left(1-e^{-1095\lambda_i}\right)}{\sum_j \left[P_j\left(1-e^{-\lambda_i T_j}\right)\left(e^{-\lambda_i T_j^o}\right)\right]}$$

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 time of the last reactor shutdown (day).

3579 = reference plant MWt.

APPENDIX 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 at time (μCi/g). See Appendix A.

Cgi = Measured concentration of isotope i in BSEP's containment gas at time, tΔ(μCi/cc). See Appendix 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 Appendix B.

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

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

EXHIBIT 3.6.3-3

	Reference Plant	BSEP
Rated Reactor Thermal Power	3579 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 3351 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	9.09 h	26.1	250 (0.899)
Halogens	I-131	8.04 d	96.0	364 (0.812)
	I-132	2.29 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	2990.0	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	58.60 d	118.0	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	46.00 d	161.0	724 (0.437), 757 (0.553)
	Zr-97	16.80 h	166.0	743 (0.928)

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

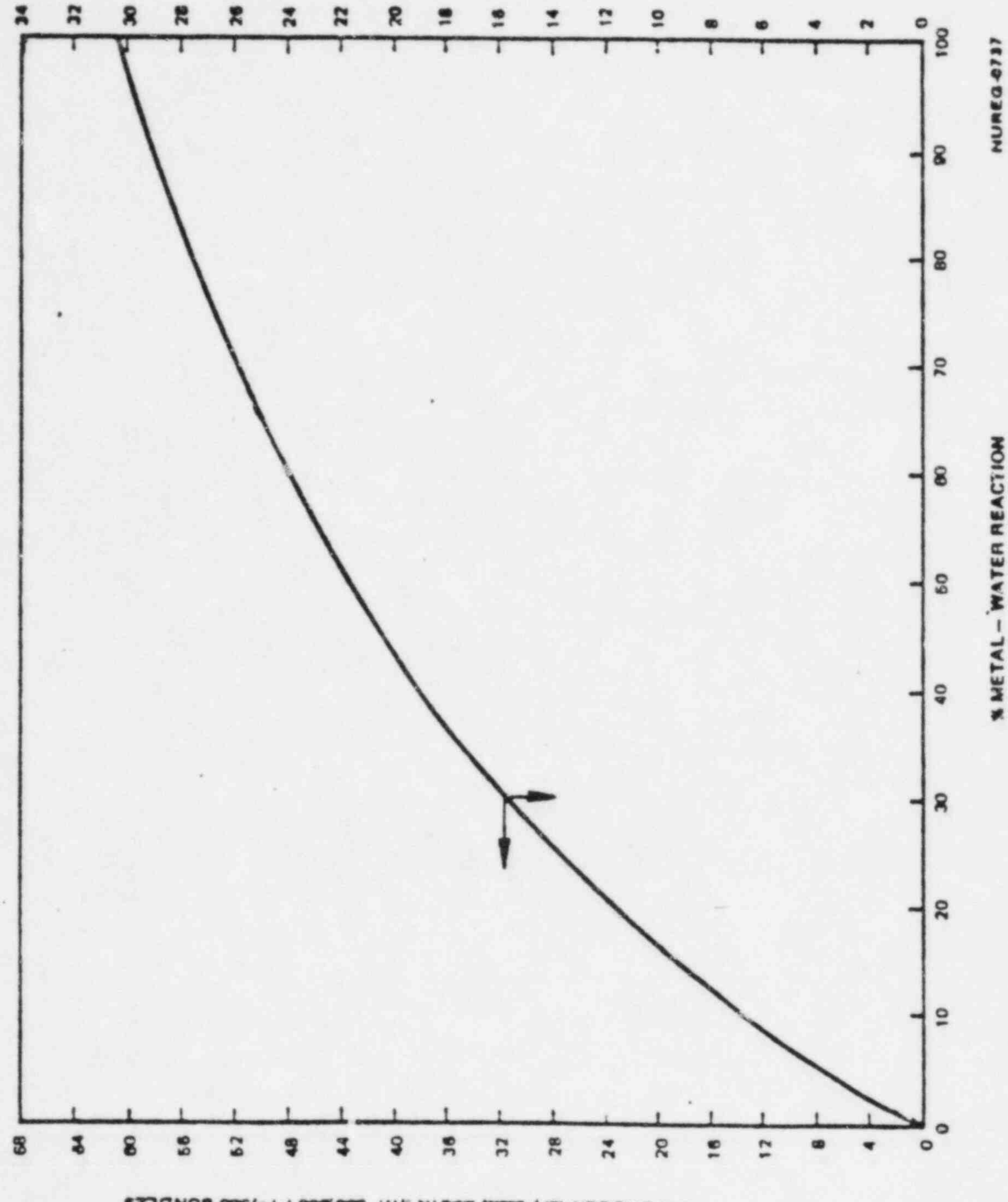
APPENDIX D

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 Mark 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 Mark I and II Containments as a Function of Metal-Water Reaction

APPENDIX D (Cont'd)

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

1. Containment Volume = 350,000 ft^3

2. Number of Bundles = 500

3. Fuel Type = 8x8 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.

APPENDIX 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) is 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, $ft^3 = 2.86 \times 10^5$

APPENDIX 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 appendix contains a correlation of the containment radiation monitor dose rate to the percent of fuel inventory airborne in the containment. The purpose of this appendix is to present that correlation and provide a method to use that correlation to determine the degree of core damage.

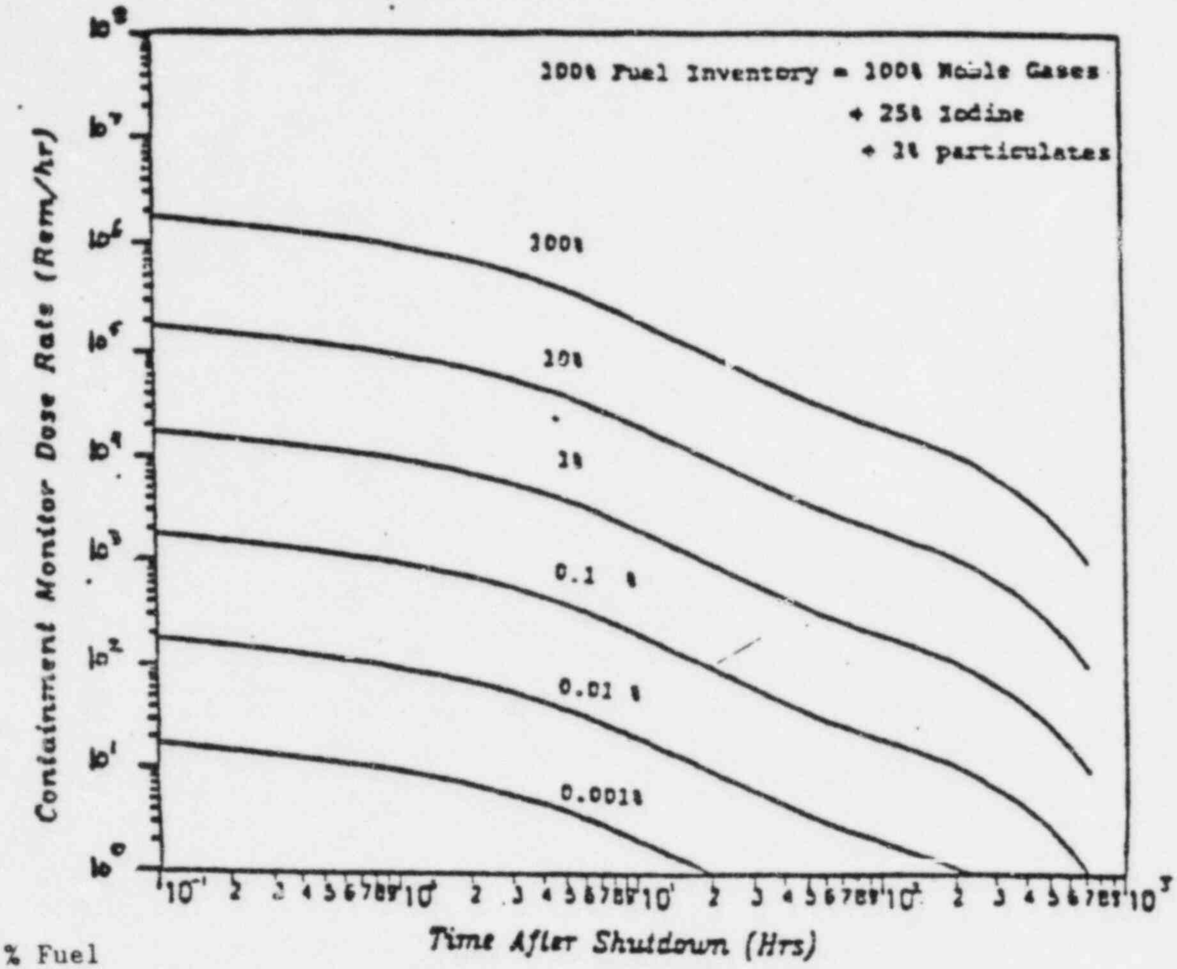
Exhibit 3.6.3-6 provides the results of a correlation performed for the Monticello plant. The key parameters which impact the containment dose rate are reactor power, containment volume, and monitor location within the containment.

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

APPENDIX E (Cont'd)

APPENDIX 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.

APPENDIX 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:

$$[I] = [I]_{ref} \left(\frac{1670}{P}\right) \left(\frac{V}{237,450}\right) (6/D)$$

where:

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

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

D = distance of detector from reactor biological shield wall, ft.