February 17, 1988

Docket No. 50-286

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Mr. John C. Brons
Executive Vice President, Nuclear
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of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

At a meeting with the NRC staff on July 30, 1987, concerning the replacement of the Indian Point Nuclear Generating Unit No. 3 steam generators, you stated your conclusion that the replacement did not require Technical Specification changes and did not involve an unreviewed safety question as defined in 10 CFR 50.59. As a result, we understand that you have elected to replace the steam generators under the provisions of 10 CFR 50.59.

The safety evaluation which must be prepared in accordance with 10 CFR 50.59 supporting your determination that the Steam Generator Replacement Program does not involve an unreviewed safety question or a change to the Technical Specifications should include, but not necessarily be limited to, consideration of the following:

1. The identification of safety related equipment, systems, and components which may be affected by the replacement project;

2. The integrity of safety related systems and/or components which would need to be removed during the project, consideration of lay down loads and reinstallation of equipment following the steam generator replacement;

3. The load path to be used for the transport of the steam generator during removal and reinstallation including underpinning and shoring of existing floors;

4. The adequacy of post cut-out reinforced concrete steam generator compartments and temporary structures to be erected either within the containment or onsite;

5. The effect of the changes in mass and center of gravity of the new steam generator on the existing seismic analysis for the containment structure and the NSSS;

6. The effect of changes to the component design and tube flow area on the steam generator transients and accident analysis;

7. The location and methods to be used in cutting steam generator;

8. The effect of reattachment of the steam generator to the reactor on the steam line break accident analysis and reactor pressure boundary integrity;

9. The effect of the component design changes on the steam generator tube rupture analysis; and

10. Radiological and health physics considerations both on site and within the containment structure during and following the project.

Please inform us when your safety analysis is complete and documented.

Sincerely,

Steven A. Varga, Director
Division of Reactor Projects, I/II

cc: See next page

PDI-1	PDI-1	PDI-1	AD:DRP	D:DRP
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Mr. John C. Brons
Power Authority of the State
of New York

Indian Point Nuclear Generating
Unit No. 3

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New York Power Authority

TO: NRC Document Control CONTROL COPY NO.: 25

FROM: TERRY RYAN DATE: 02/02/88

SUBJECT: DISTRIBUTION OF THE INDIAN POINT #3 EMERGENCY PLAN REVISIONS

The enclosed sheets are to be incorporated into your controlled copy of the IP-3 Emergency Plan. Please discard the old sheets, insert the attached sheets, initial/date this transmittal sheet, and return it to Ms. Terry Ryan, IP-3 Documents Supervisor.
Thank you.

VOLUME I - EMERGENCY PLAN

OLD:	NEW:
VOLUME 1 - EMERGENCY PLAN REV. 18 APPENDIX B Pages 1 through 11	VOLUME 1 - EMERGENCY PLAN REV. 18 APPENDIX B Pages 1 through 11

Please do not be confused by this re-distribution of "REVISION 18 OF APPENDIX B", of the Emergency Plan.

The reason for this re-distribution is that pages 5 through 8, Appendix B were not included in the recent distribution of REVISION 18, Volume 1 of the Emergency Plan.

I acknowledge the receipt of these revisions to the IP-3 Emergency Plan.

_______________________ / _______________________
(Signature) (Date)

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APPENDIX B

TIME - DOSE - DISTANCE PLOTS AND ASSOCIATED INFORMATION

APPENDIX B

DISCUSSION:

ESTIMATED OFFSITE DOSES DURING THE DESIGN BASIS ACCIDENT

The dose assessment of the most serious design basis accident (Loss of Coolant Accident, LOCA) on the offsite population is illustrated in Figures I, II, and III. Figures I and II provide isodose graphs for thyroid and whole body doses respectively for time vs. distance. Figure III illustrates thyroid dose vs. distance for the specific case of two hours. For comparative purposes, two curve sets are shown on each figure to illustrate two different meteorological conditions:

(1) Meteorological condition which is exceeded only 5% of the time (5% worst condition).

(2) Meteorological condition which is exceeded 50% of the time (median condition).

The dose received during the median condition will be less than the dose received during a 5% worst meteorological condition since greater effluent dilution is experienced under the median condition. These figures illustrate that the dose received by the offsite population is influenced by the specific meteorology existing at the time of an accident. In the unlikely event of an actual accident, dose estimates will be made using existing site meteorological conditions and will be continuously updated with information received from monitoring teams.

The assumptions used to develop these figures are tabulated in Table I. References justifying these assumptions are also provided. A review of the assumptions used to develop these figures will show that these figures represent conservative estimates of offsite doses.

The assumed containment leak rate is 0.075% of containment free volume/day; previous leak tests on Containment have shown the actual leak rate to be less than this assumed value. Also, these previous tests were performed without use of two engineered safeguards: the Isolation Valve Seal Water System (IVSWS) and the Weld Channel and Containment Penetration Pressurization System (WCCPPS). If these two systems were operating during this test, the actual leak rate would approach zero. The IVSWS and WCCPPS were designed and constructed to rigorous specifications and are required to be operational under Unit 3 Technical Specifications. Thus, it is extremely conservative not to take credit for these two safeguards in the analysis although they are designed to operate during the LOCA to substantially reduce, if not effectively eliminate, the Containment leakage.

Appendix A to Operating License DPR 64, Technical Specification and Bases, Section 3.3, 4.4.

The Containment leak rate value depends on the pressure differential across the Containment boundary; thus, the lower the pressure inside Containment, the smaller the leak rate. This analysis, in order to be conservative, assumes a constant 0.075%/day leak rate, whereas physically, the leak rate will decrease with time as the pressure inside Containment drops. Internal pressure would decrease due to steam condensing inside Containment through continuous heat removal by the Containment Spray and/or Containment Fan Cooler Recirculation Systems.

The percentages of core fission products inventory available for release from Containment during a LOCA are defined by values given in Regulatory Guide 1.4. Accordingly, 25% of the core's iodine inventory is immediately available for leakage from Containment. It is expected that with operation of the Minimum Safeguards Injection System the fuel clad temperature will be maintained well below the melting point of Zircaloy-4 and limit the zirconium-water reaction to be an insignificant amount, although some cladding failure may result in the hotter regions of the core. Clad failure results in the release of the volatile fission products in the pellet-cladding gap. It would be therefore conservative to assume that all gap activity is available for release since not all the cladding is expected to fail. The iodine isotope gap activities are listed in Table 14.3.5-1 of the FSAR as a function of percentage of core inventory. Iodine-131 has the highest percentage of those listed, therefore, it is conservative to equate other iodine isotope core inventory percentages to that of Iodine-131. This highest percentage (2.3%) is still more than factor of ten less than the iodine percentage (25%) given by Regulatory Guide 1.4 as available for release from Containment. Regulatory Guide 1.4 defines the percentage of noble gas core inventory immediately available for leakage from Containment as 100%. The actual amount which escapes from the pellet-clad gap and is available for leakage will be significantly less. Therefore, we have clearly been over conservative in using the percentages defined in Regulatory Guide 1.4 for noble gas and iodine releases to derive these figures.

The organic iodine charcoal filter efficiency assumed for this analysis is only 30%, an actual efficiency of 70% or greater is expected.

The meteorological conditions assumed in the development of Figures I, II, and III are wind speed of 0.73 meters/sec. and the Pasquill Stability Class F. These parameters are derived from the atmospheric dilution factor (X/Q) which is exceeded only 5% of the time as computed by the NRC Staff for the Indian Point Site. A wind speed of 0.73 meters/sec. was computed using this 5% X/Q value, the methodology presented in ERL-ARL-4, and assuming a Pasquill F stability class.

NRC Regulatory Guide 1.4, Rev.2, "Assumptions used in Evaluating the Potential Radiological Consequences of a Loss of Coolant for Pressurized Water Reactors".

Final Facility Description and Safety Analysis Report, Indian Point Unit No. 3, Table 14.3.5-1.

Final Facility Description and Safety Analysis Report, Indian Point Unit No. 3, Section 14.3.5.

Credit for plume meander is accounted for by equation (3) of Draft Regulatory Guide 1.XXX; it is applied when wind speeds are less than 6 meters/sec. coincident with Pasquill neutral (D) or stable (E, F, or G) conditions. The Pasquill F stability class and 0.73 meters/sec wind speed satisfy the conditions for using this plume meander equation. Accordingly, centerline X/Q values per distance were computed using equation (3) of Draft Regulatory Guide 1.XXX.

This methodology is conservative in that the dose received using these computed X/Q values occurs only along the plume centerline. Under stable conditions, the plume crosswind dimension of the one percent isopleth (line of constant concentration which is one percent of the plume's centerline value) is relatively small. Specifically, the distances from the plume centerline to the one percent isopleth at 1100 meters are 170 meters and 120 meters for Pasquill E and F, respectively. Therefore, an individual located on the plume centerline at 1100 meters need only move in a direction perpendicular to the centerline, 170 meters during a Pasquill E or 120 meters during a Pasquill F in order to reduce the dose rate by a factor of 100. Figure IV illustrates the distance to the one percent isopleth from the plume centerline (plume half-width) versus downwind distance for the Pasquill E and F stability classes.

For any time interval of interest as provided on the vertical axis of Figures I and II, the isodose curves assumed meteorological condition (5% or 50%) must prevail for that time interval in order for an individual to receive that curve's particular isodose. It is conservative to assume as the time interval increases, that the meteorological condition, and therefore wind direction, remain constant. Wind persistent analysis has illustrated higher wind speeds are correlated to a persistent wind direction, while during low wind speeds, the probability of constant direction diminishes and the relative concentrations are dispersed over a larger area thereby decreasing the peak value. As the time period increases, the probability of the wind direction remaining steady decreases. The valley wind system at Indian Point illustrates this wind direction variation effect. Analysis of 1977 Indian Point Site meteorological data shows that the persistence of winds diminishes with stable conditions and low speed. Successive overlapping eight-hour intervals were analyzed; under stable conditions and wind speeds less than 6 meters/sec., the wind persisted from a single 22.5^{o} sector for a period of eight hours less than one percent of the total annual intervals analyzed.

Safety Evaluation of the Indian Point Nuclear Generating Unit No. 3, Docket No. 50-286, US AEC, Director of Licensing, Sept. 21, 1973, pg. 244.

NOAA Technical Memorandum ERL-ARL-42, "A Program for Evaluating Atmospheric Dispersion from a Nuclear Power Station", Jerrold F. Sagendorf, May 1974.

NRC Draft Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", June 12, 1978.

Analysis of 1977 hourly meteorological data for the Indian Point Site shows that the Pasquill F stability occurs almost entirely (92.5%) during non-working hours (1800 to 0700). The breathing rate approximated during non-working (non-active) hours is the standard man's average. Since this analysis assumes that the accident occurs during a Pasquill F stability condition and therefore most probably during non-working hours, the breathing rate assumed in computation of the 5% curves of Figures I, II, and III is the standard man's average.

For comparison, dose curves assuming the median (50% value) meteorological condition are also provided on Figures I, II, and III. The median meteorological condition was again derived from methodology presented in ERL-ARL-42 using site hourly meteorological data for the entire year of 1977. This resulted in determining that the median X/Q value was not exceeded more than 50% of the time. Assuming a Pasquill E stability (occurrence exceeded 50% of the hourly cases during 1977) and the above median X/Q, a wind speed of 1.67 meters/sec. was derived. Therefore, the median meteorological conditions used are a wind speed of 1.67 meters/sec. and the Pasquill E stability class.

The breathing rates of individuals vary over different periods of the day. Since the Pasquill E stability, based on 1977 data randomly occurs over a twenty-four hour period, the maximum breathing rate (347 cc/sec.) is conservatively assumed for computation of thyroid dose when the accident is assumed to occur during the median meteorological condition.

Therefore, it can be concluded that the 5% dose curves of Figures I, II, and III represent highly conservative estimates.

Figure I

Thyroid Dose

Time - Dose - Distance Curves

Loss of Coolant Accident

Indian Point No. 3

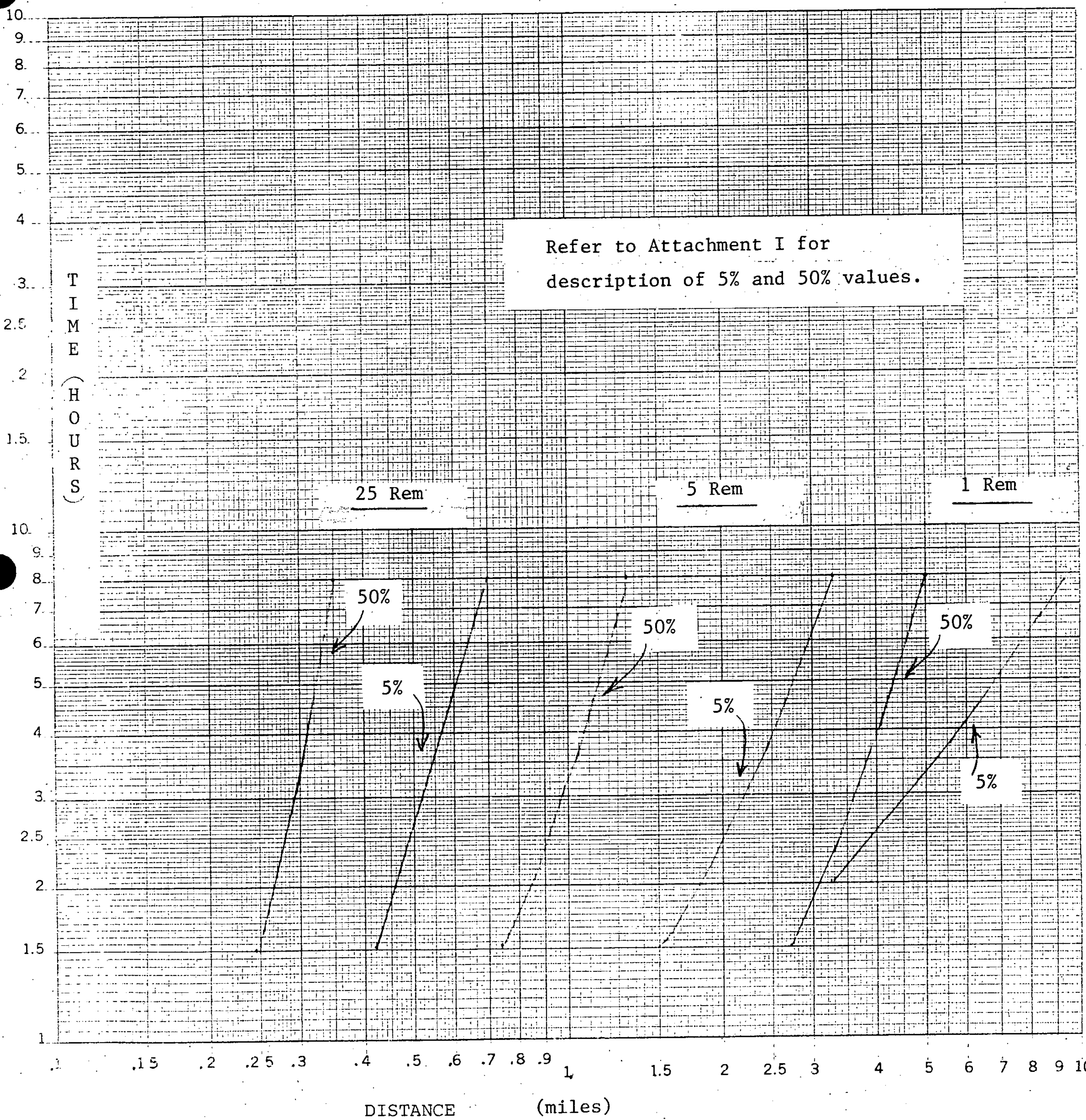

Figure II

Whole Body Dose
Time - Dose - Distance Curves
Loss of Coolant Accident
Indian Point No. 3

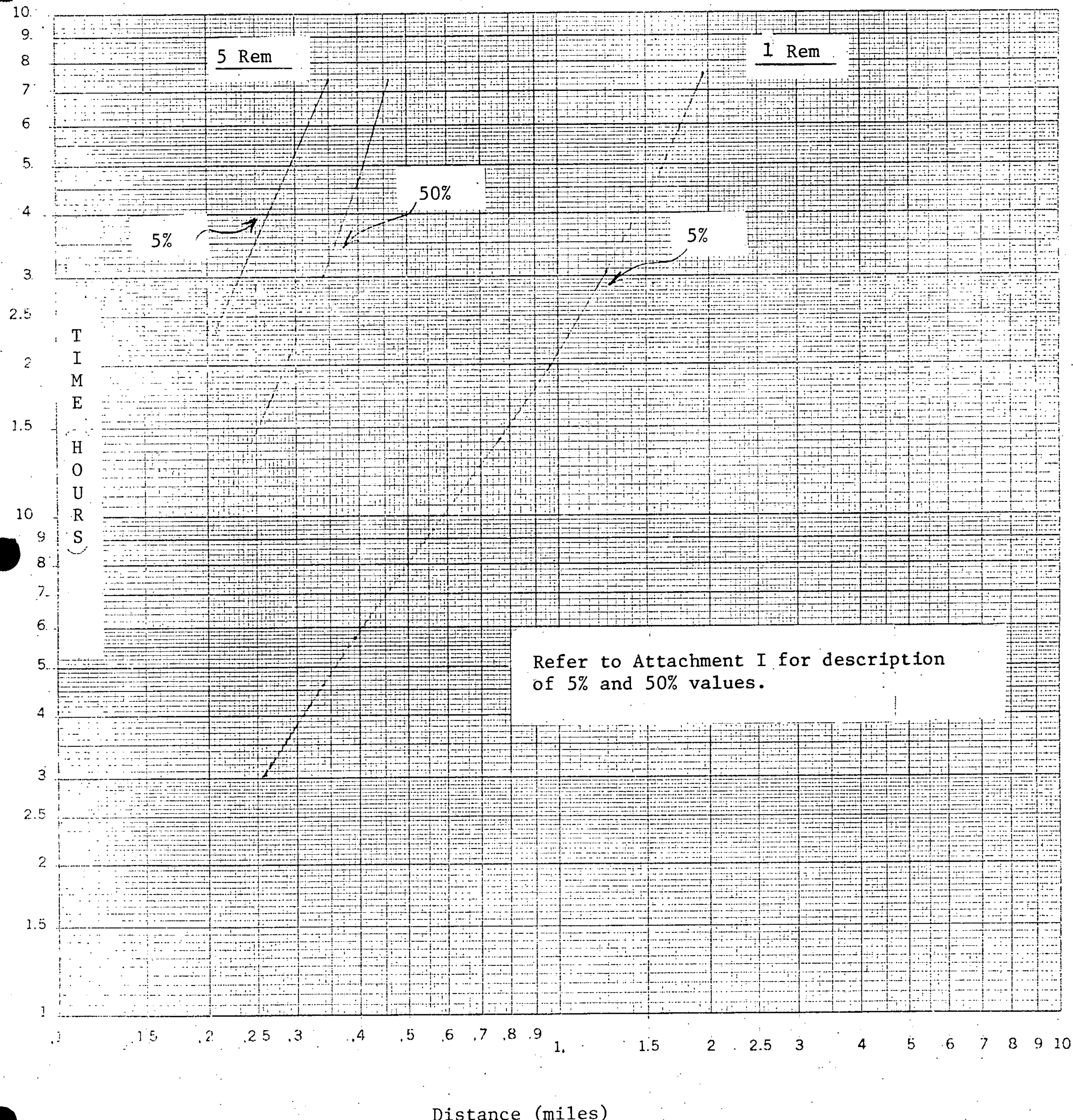

Figure III

Thyroid Dose
Two Hours - Distance vs. Dose
Loss of Coolant Accident
Indian Point No. 3

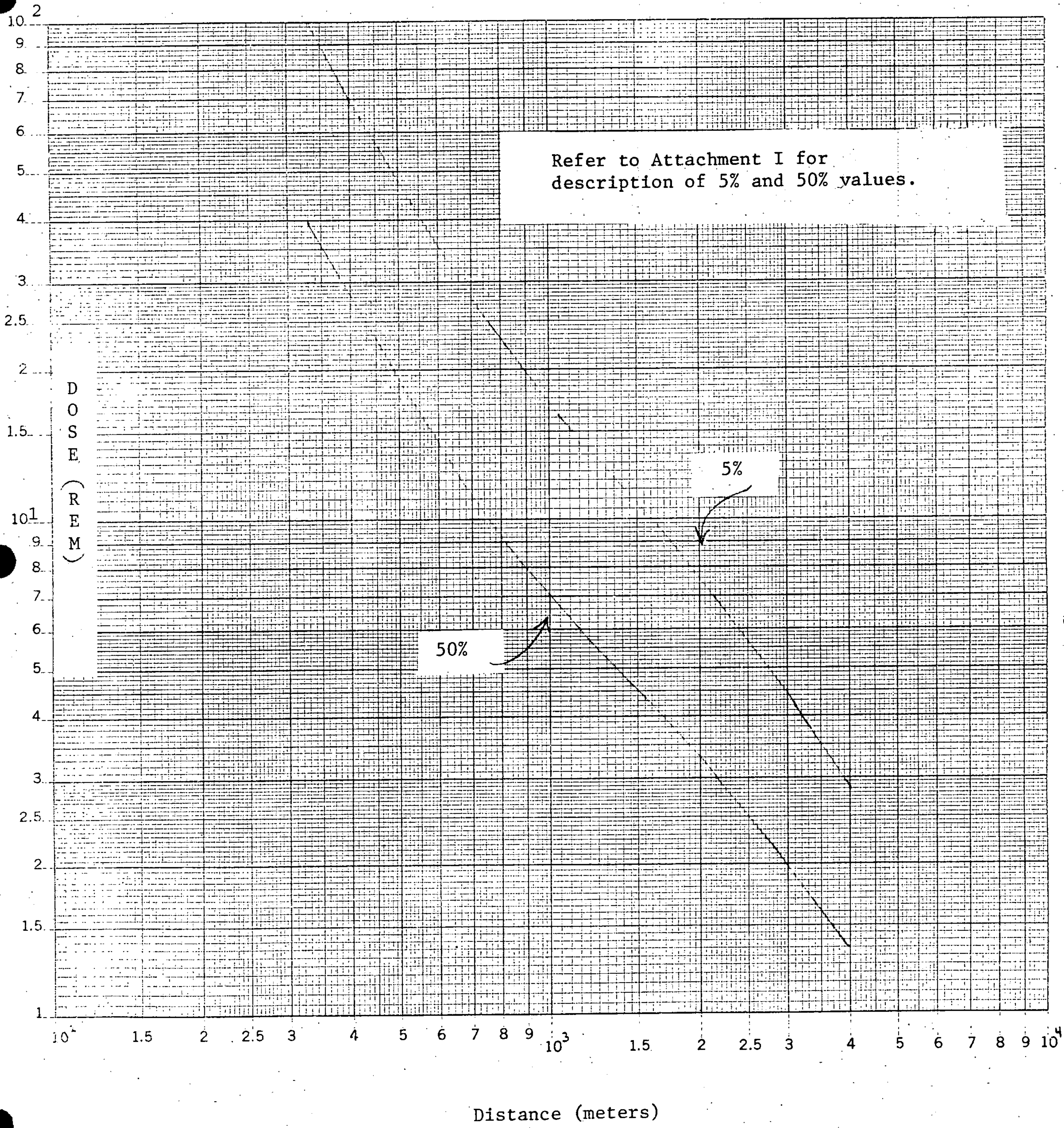

Figure IV

Plume Half - Width

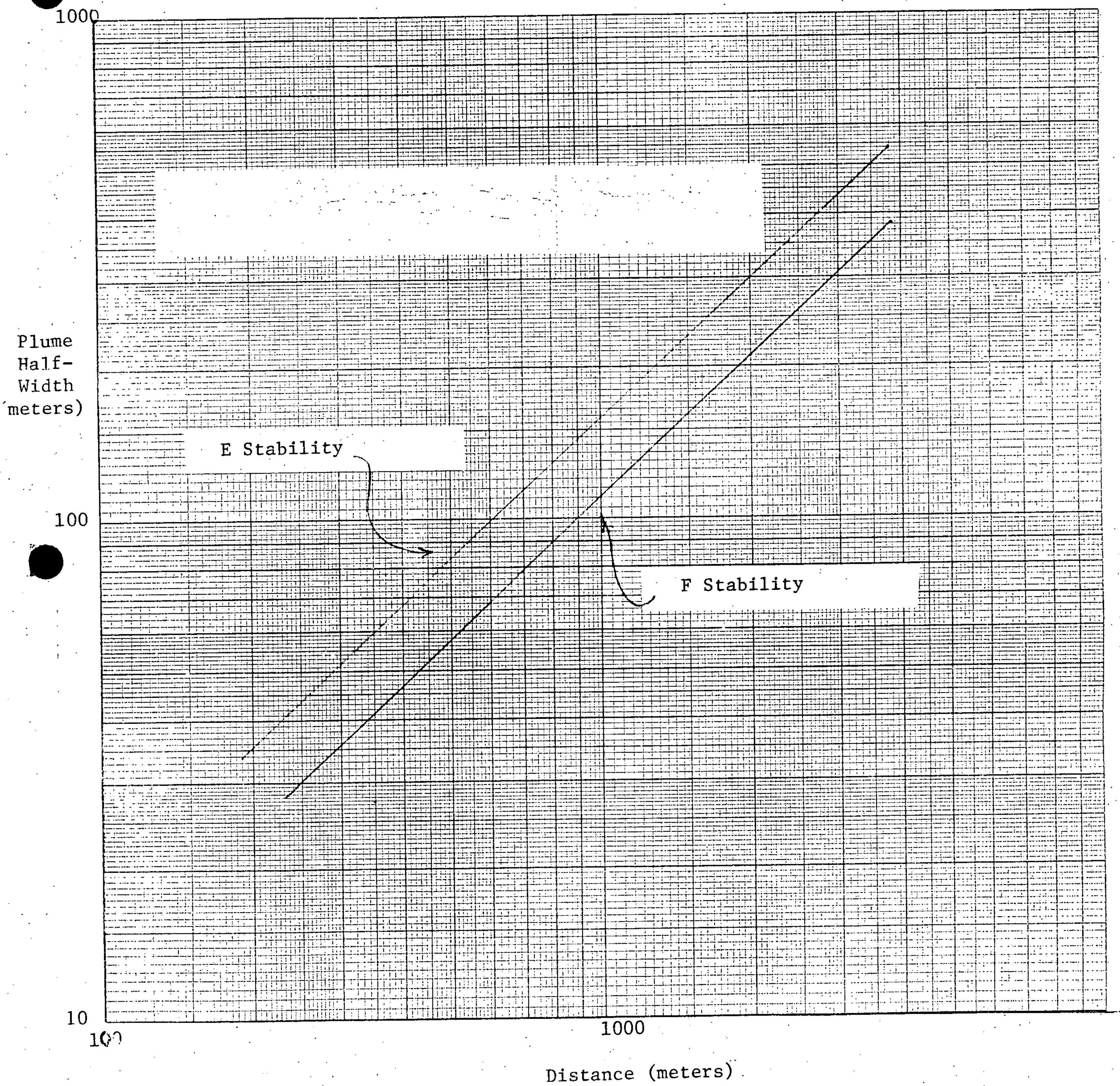

Table I

Assumptions for Thyroid and Whole Body Doses
Indian Point Unit No. 3 Loss of Coolant Accident

No.	Assumptions	
1)	Isotope Inventory	-
2)	Containment Leak Rate	.075% of Containment free volume/day
3)	Charcoal Filter Efficiency:	
	Organic Iodine	30%
	Inorganic Iodine	90%
	Particulate Iodine	90%
4)	Fraction of Iodine Released**	25%
5)	Fraction of Iodine Forms:	
	Inorganic Form	91%
	Organic Form	4%
	Particulate Form	5%
6)	Radiological decay considered during holdup on Containment	-
7)	Plant Power Level	3025 MWt
	Spray Removal Coefficient:	
	Inorganic Form	9.83 $hr.^{-1}$
	Methyl Form	0 $hr.^{-1}$
	Particulate Form	.45 $hr.^{-1}$
8)	Ventilation/Filtration System:	
	Ventilation flow rate per unit	8000 cfm
	Number of units assumed operating	3 of 5
9)	Breathing Rate*	231 cc/sec.
10)	Meteorological Conditions Assumed*	Pasquill Class F, .73 m/sec.wind speed
11)	Containment Free Volume	2.61×10^6 $ft.^3$
12)	Credit for Plume transit time and radiological decay during transit	-
13)	Use of reduction factors expressing deviation between a finite and infinite cloud when computing whole body dose	-
14)	Duration of spray effectiveness	1 day

*See Attachment I.
** Fraction of iodine available for release from Containment.

Table I - References

(1) Technical Information Document 14844, March 23, 1962, U.S. Atomic Energy Commission, Table I, pg. 20.

(2) Appendix J, 10CFR50.

(3) Final Facility Description and Safety Analysis Report, Indian Point Unit No. 3, Table Q14.11-1.

(4) NRC Regulatory Guide 1.4, Rev. 2, June 1974, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors".

(5) Indian Point Unit No. 3 Licensed Power Level.

(6) NRC Regulatory Guide 1.70.14, December 1974, "Information for Safety Analysis Report Emergency Planning".

(7) Meteorology and Atomic Energy, Atomic Energy Commission, 1968, 7-5.2.3.

Attachment I

(A) 5 and 50% Meteorological Conditions:

For the Indian Point Site, the 5% probability atmospheric dilutionfactor (X/Q) is 1.8×10^{-3} sec./m^3 at the site boundary (330 meters). Using this X/Q and assuming a Pasquill Class F, a corresponding wind speed of 0.73 m/sec. has been computed. Hence, the 5% meteorological condition is Pasquill Class F and wind speed of 0.73 m/sec.

An X/Q cumulative frequency was computed using one years's (1977)meteorological data for the Indian Point Site, the 50% X/Q for this data at 330 meters is 3.4×10^{-5} sec./m^3. Assuming this X/Q and a Pasquill Class E, a wind speed of 1.67 m/sec. was computed. Hence, the 50% meteorological condition is Pasquill Class E and a wind speed of 1.67 m/sec.

Credit for plume meander at low wind speed and stable conditions is given by Equation (3) of Draft Regulatory Guide 1.XXX. For the two sets of curves given in Figures I and II, the 5% curves utilize Equation (3) and the 5% condition and the 50% curves utilize Equation (3) and the 50% condition.

(B) Breathing rate for 5% and 50% Meteorological Conditions:

92.5% of all Pasquill F (or worse) hourly conditions (considering 1977 data) occur between 1800 and 0700. Thus, the breathing rate of 347 cc/sec. is not applicable since it assumes activity in the working day. For non-working hours, the rate is more closely approximated by the daily average rate of 231 cc/sec. and is used in computation of the 5% thyroid dose curves. Pasquill E occurs almost 50% of all hourly conditions for the 1977 data. For this reason 347 cc/sec. was assumed as the breathing rate for the 50% thyroid curves.

Safety Evaluation of the Indian Point Nuclear Generating Unit No. 3, Docket No. 50-286, US AEC, Directorate of Licensing, Sept. 21, 1973, pg. 2-14.

June, 1978, Draft Regulatory Guide 1.XXX, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants".