APPENDIX B
METHODOLOGIES

1. Dose Calculation - Atmospheric Release Pathway

Dispersion of airborne radioactive material was calculated for each of the 16 compass sectors using the CAP88 dose model. CY 1996 site meteorology and 10 year wind averages were used to calculate annual dispersions for the midpoint of a given sector and distance. Facility-specific radionuclide release rates (in Ci per year) were also used. All annual site boundary and collective dose values were generated using the CAP88 computer code, which calculates the total dose due to contributions from the immersion, inhalation, and ingestion pathways.

2. Tritium Dose Calculation - Potable Water Pathway

The method used to calculate the maximum individual committed effective dose equivalent and the collective dose equivalent are shown along with the basic assumptions used in the calculation. For the maximum individual, the highest annual average tritium concentration, measured from a single potable well was used to calculate the total quantity of tritium ingested via the drinking-water pathway. For calculating the collective dose equivalent, the annual average tritium concentration was obtained by averaging all positive results from potable wells, which were in the demographic region adjacent to the Laboratory. The annual intake of tritium via the drinking water pathway was calculated from the following equation:

$$AI = 1 \times 10^{-6}\ C \cdot IR \cdot T$$

Where: AI = Activity Intake, µCi

C = annual average water concentration, pCi/L

IR = Ingestion Rate (2) L/d

T = Time, 365 d

The committed effective-dose equivalent was calculated from the following equation:

$$H = AI \times DCF \times P$$

where: H = committed effective dose-equivalent, rem

AI = Activity Intake, µCi

DCF = Dose Conversion Factor, rem/µCi (6.3E-5 rem/µCi)

P = Exposed population

To determine the maximum individual dose, the population parameter was set to unity. For the collective dose calculation, the exposed population was assumed to be approximately 500.

3. Dose Calculation - Fish Ingestion Pathway

To estimate the collective-dose equivalent from the fish consumption pathway, the following procedure was used:

a. Radionuclide data for fish samples were all converted to pCi/g wet weight; this is the form in which the fish is caught and consumed.
b. The average fish consumption for an individual engaged in recreational fishing in the Peconic River was based on a study done by the NYSDEC which suggests that the consumption rate is approximately 7 kg/yr (NYSDEC, 1985).
c. DOE Order 5400.5 50-year Committed Dose Equivalent factors (in rem per µCi intake) based on the ICRP 26 model were applied. They are as follows:

Tritum: 6.3E-05 rem/µCi
Strontium-90: 1.3E-01 rem/µCi
Cesium-137: 5.0E-02 rem/µCi

d. Calculation: Intake (7 kg/yr) × Activity in flesh µCi/kg × Factor rem/µCi = rem

4. Dose Calculation - Deer Meat Consumption

This calculation is performed in exactly the same way as shown in the previous section. The same DOE Order 5400.5 dose conversion factors are used. The only change is the estimate of total kilograms ingested in the course of a year. For deer meat, the consumption rate of 30 kg/yr is based on the upper range of venison consumption estimates supplied by the NYSDEC Wildlife Branch.

5. Radiological Data Processing

Radiation events occur in a random fashion such that if a radioactive sample is counted multiple times a distribution of results will be obtained. This spread, known as a Poisson distribution, will be centered about a mean value. If counted multiple times, the background activity of the instrument (the number of radiation events observed when no sample is present) will also be seen to have a distribution of values centered about a mean. The goal of a radiological analysis is to determine whether the sample in question contains activity in excess of the instrument or method blank background. Since the activity of the sample and the background are both Poisson distributed, subtraction of background activity from the measured sample activity results in a value which may vary slightly from one analysis to the next. Therefore, the concept of a minimum detection limit (MDL) is established to determine the statistical likelihood that the sample contains activity that is truly greater than the instrument background.

Identifying a sample as containing activity greater than background when it actually is not is known as a Type I error. As with most laboratories, the BNL Analytical Laboratory sets its acceptance of a Type I error at 5% when calculating the minimum detection limit for a given analysis. That is, for any value which is greater than or equal to the MDL there is 95% confidence that it represents the detection of true activity. Values, which are less than the MDL may be valid, but they have a reduced confidence associated with them. Therefore, all data is reported regardless of its value.

At very low sample activity levels, close to the instrument background, it is possible to obtain a sample result, which is less than the background. When the background activity is subtracted from the sample activity to obtain a net value, a negative value results. In such a situation, a single radiation event observed during a counting period could have a significant effect on the result. Subsequent analysis may produce a net result that is positive. Therefore, all negative values are retained for reporting as well. This data handling practice is consistent with the guidance provided in NCRP Report No. 58, *Handbook of Radioactivity Measurements Procedures* and DOE/EH-0173T, *Environmental Regulatory Guide for Radiological Effluent Monitoring and Environmental Surveillance.*