February 28, 1992

Subject: Offsite Dose Calculation Manual
Revision 35

The General Office Radiation Protection Staff is transmitting to you this date, Revision 35, of the Offsite Dose Calculation Manual. As this revision affects the manual's generic section, the approval of each station manager has been obtained. Please update your copy No. 51, and discard the affected pages.

REMOVE THESE PAGES		INSERT THESE PAGES	
iii	Rev. 28	iii	Rev. 35
1-1	Rev. 5	1-1	Rev. 35
1-2	Rev. 5	1-2	Rev. 35
Table 1.2-1	No Rev. #	Table 1.2-1	Rev. 35
Table 1.2-2	Rev. 5	Table 1.2-2	Rev. 35
2-1	No Rev. #	2-1	Rev. 35
3-2	Rev. 5	3-2	Rev. 35
3-8	Rev. 28	3-8,3-8a	Rev. 35
3-9	Rev. 28	3-9	Rev. 35
3-10	Rev. 28	3-10	Rev. 35

NOTE: As this letter, with it's attachments, contains "LOEP" information please insert this letter in front of the December 26, 1991.

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Effective Date: 3/01/92

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Enclosure

JUSTIFICATIONS FOR REVISION 35

PAGE #	
iii	Added the words "Selected License Commitments".
1-1	Same as Page iii
1-2	Added additional information to definition of Mi
Table 1.2-1	Added the residential shielding attenuation factor of 0.7 to ki values.
Table 1.2-2	Corrected the inhalation values for ZN-65 and TE-127M.
2-1	Same as Page iii
3-2	Corrected typo in formulas.
3-8	Added/Enhanced information provided in Sections 3.1.3 and 3.2.
3-9	Same as Page iii
3-10	Added reference to Ladtap and Gaspar to Section 3.3.5.

INTRODUCTION

The Offsite Dose Calculation Manual provides the methodology and parameters to be used in the calculation of off-site doses due to radioactive liquid and gaseous effluents to assure compliance with the dose limitations of the Selected Licensee Commitments and/or Technical Specifications. These dose limitations assure that:

1) the concentration of radioactive liquid effluents from the site to the unrestricted area will be limited to the concentration levels of 10CFR20, Appendix B, Table II;

2) the exposures to any individual from radioactive liquid effluents will not result in doses greater than the design objectives of 10CFR50, Appendix I;

3) the dose rate at any time at the site boundary from radioactive gaseous effluents will be limited to the annual dose limits of 10CFR20 for unrestricted areas; and

4) the exposure to any individual from radioactive gaseous effluents will not result in doses greater than the design objectives of 10CFR50, Appendix I.

The methodology used to assure compliance with the dose limitations described above shall also be used to prepare the radioactive liquid and gaseous effluent reports required by the Selected Licensee Commitments and/or Technical Specifications. To assure compliance with 40CFR190 when twice the design objectives of 10CFR50, Appendix I are exceeded, the methodology and parameters to be used in calculating the off-site dose to any individual resulting from the entire fuel cycle except mining and waste management facilities are provided in this Manual.

The Manual also provides the methodology and parameters to be used in the calculation of radioactive liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints to assure compliance with the concentration and dose rate limitations of the Selected Licensee Commitments and/or Technical Specifications. Changes to the methodology and parameters used in this Manual shall be reviewed by a qualified reviewer(s) and approved by the Station Manager and the System Radiation Protection Manager prior to implementation and shall be audited by the Nuclear Safety Review Board. Changes to this Manual shall be submitted to the Nuclear Regulatory Commission in accordance with plant Selected Licensee Commitments and/or Technical Specifications.

Normally GASPAR and LADTAP are used for the calculation of offsite doses but this document also describes a method for the calculation of offsite doses when GASPAR and/or LADTAP are not available.

This Manual does not replace any station implementing procedures.

1.0 RELEASE RATE CALCULATIONS

The release rate calculations presented in the following sections are site release limits. Sites containing two or more units shall administratively control releases to assure that the release rate calculations limit releases as stated in the Selected Licensee Commitments and/or Technical Specifications. Administrative controls could limit the number of releases occurring at one time and/or apportion the release rate between the units.

1.1 LIQUID EFFLUENTS

To comply with Selected Licensee Commitments and/or Technical Specifications and to assure that the concentration of radioactive liquid effluents from the site to the unrestricted area is limited to the concentrations of 10CFR20, Appendix B, Table II, Column 2, the following release rate calculation shall be performed:

$$f \leq F \div \left(\sigma \sum_{i=1}^{n} \frac{C_i}{MPC_i}\right)$$

where:

C_i = The concentration of radionuclide, 'i', in undiluted liquid effluent, in µCi/ml.

MPC_i = the concentration of radionuclide, 'i', from 10CFR20, Appendix B, Table II, Column 2, in µCi/ml.

f = the undiluted effluent flow from the tank, in gpm.

F = the dilution flow from the site discharge structure to unrestricted area receiving waters, in gpm.

σ = recirculation factor at equilibrium; this factor accounts for the fraction of discharged water reused by the station; this factor is one for stations on rivers or lakes where discharged water cannot be reused, and varies for sites where water is recirculated and is specified in the appropriate Appendix.

1.2 GASEOUS EFFLUENTS

In order to comply with the Selected Licensee Commitments and/or Technical Specifications and to assure that the dose rate, at any time, in the unrestricted area due to radioactive materials released in gaseous effluents from the site is limited to ≤500 mrem/yr to the total body and ≤3000 mrem/yr to the skin for the noble gases and is limited to ≤1500 mrem/yr to any organ for all radioiodine and for all radioactive materials in particulates form and radionuclides other than noble gases with half lives greater than 8 days, the following release rate calculations shall be performed. These calculations, when solved for 'f', i.e. flowrate, are the release rates for noble gases and for radioiodines, particulates and other radionuclides with half-lives greater than 8 days. The most conservative of release rates calculated shall control the release rate.

1.2.1 Noble Gases

$$\sum_i K_i \times [(\overline{X/Q})\tilde{Q}_i] < 500 \text{ mrem/yr, and}$$

$$\sum_i (L_i + 1.1\, M_i)\, [(\overline{X/Q})\tilde{Q}_i] < 3000 \text{ mrem/yr}$$

where:

K_i = The total body dose factor due to gamma emissions for each identified noble gas radionuclide, in mrem/yr per µCi/m³ from Table 1.2-1.

L_i = The skin dose factor due to beta emissions for each identified noble gas radionuclide, in mrem/yr per µCi/m³ from Table 1.2-1.

M_i = The air dose factor due to gamma emissions for each identified noble gas radionuclide, in mrad/yr per µCi/m³ from Table 1.2-1 (unit conversion constant of 1.1 mrem/mrad converts air dose to skin dose; no correction for structural shielding is assumed - GASPAR uses a factor of 0.7 reduction in gamma contributions to the skin dose consistant with Regulatory Guide 1.109, equation B-9).

P_i = The dose parameter for radionuclides other than noble gases for the inhalation pathway, in mrem/yr per µCi/m³ and for the food and ground plane pathways in m².(mrem/yr per µCi/sec) from Table 1.2-2. The dose factors are based on the critical individual organ and most restrictive age group (child or infant).

$\tilde{Q}_i$ = The release rate of radionuclides, 'i', in gaseous effluent from all release points at the site, in µCi/sec.

$\overline{X/Q}$ = The highest calculated annual average dispersion parameter for any area at or beyond the unrestricted area boundary.

W = The highest calculated annual average dispersion parameter for estimating the dose to an individual at the controlling location.

$$\tilde{Q}_i = k_1 C_i f \div k_2 = 4.72E{+}2\ C_i f$$

where:

C_i = the concentration of radionuclide, 'i', in undiluted gaseous effluent, in µCi/ml.

f = the undiluted effluent flow, in cfm.

k_1 = conversion factor, 2.83E+04 ml/ft³.

k_2 = conversion factor, 6.0E+01 sec/min.

TABLE 1.2-1
(1 of 1)
DOSE FACTORS FOR NOBLE GASES AND DAUGHTERS*

Radionuclide	Total Body Dose Factor K_i** (mrem/yr per µCi/m³)	Skin Dose Factor L_i (mrem/yr per µCi/m³)	Gamma Air Dose Factor M_i (mrad/yr per µCi/m³)	Beta Air Dose Factor N_i (mrad/yr per µCi/m³)
Kr-83m	5.29E-02***	---	1.93E+01	2.88E+02
Kr-85m	8.19E+02	1.46E+03	1.23E+03	1.97E+03
Kr-85	1.13E+01	1.34E+03	1.72E+01	1.95E+03
Kr-87	4.14E+03	9.73E+03	6.17E+03	1.03E+04
Kr-88	1.03E+04	2.37E+03	1.52E+04	2.93E+03
Kr-89	1.16E+04	1.01E+04	1.73E+04	1.06E+04
Kr-90	1.09E+04	7.29E+03	1.63E+04	7.83E+03
Xe-131m	6.40E+01	4.76E+02	1.56E+02	1.11E+03
Xe-133m	1.76E+02	9.94E+02	3.27E+02	1.48E+03
Xe-133	2.06E+02	3.06E+02	3.53E+02	1.05E+03
Xe-135m	2.18E+03	7.11E+02	3.36E+03	7.39E+02
Xe-135	1.27E+03	1.86E+03	1.92E+03	2.46E+03
Xe-137	9.94E+02	1.22E+04	1.51E+03	1.27E+04
Xe-138	6.18E+03	4.13E+03	9.21E+03	4.75E+03
Ar-41	6.19E+03	2.69E+03	9.30E+03	3.28E+03

*The listed dose factors obtained from Regulatory Guide 1.109, Table B-1 are for radionuclides that may be detected in gaseous effluents.
**Includes a residential structure shielding attenuation factor of 0.7.
***7.56E-02 = 7.56×10^{-2}
5.29E-02 = = 5.29×10^{-2}

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TABLE 1.2-2

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DOSE PARAMETERS FOR RADIOIODINES AND RADIOACTIVE PARTICULATE, GASEOUS EFFLUENTS*

P(I), DOSE PARAMETERS FOR RADIOIODINES AND RADIOACTIVE PARTICULATES IN GASEOUS EFFLUENTS

Radionuclide	Pathways Inhalation (mrem/yr per µCi/m³)	Pathways Food and Ground (m².mrem/yr per µCi/sec)	Radionuclide	Pathways Inhalation (mrem/yr per µCi/m³)	Pathways Food and Ground (m².mrem/yr per µCi/sec)
H 3	1.125 E+03	2.4E+03	RU 103	6.625 E+05	1.6E+08
Alpha Act	1.100 E+08	1.8E+10	RU 106	1.432 E+07	2.0E+08
CR 51	1.698 E+04	1.1E+07	AG 110M	5.476 E+06	1.5E+10
MN 54	1.576 E+06	1.1E+09	CD 115M	2.920 E+05	5.2E+07
FE 55	1.110 E+05	1.1E+08	SN 123	3.550 E+06	3.7E+09
FE 59	1.269 E+06	7.2E+08	SN 126	1.120 E+07	1.1E+10
CO 58	1.106 E+06	5.8E+08	SB 124	3.240 E+06	1.4E+09
CO 60	7.067 E+06	4.6E+09	SB 125	2.320 E+06	9.1E+08
NI 63	8.214 E+05	3.0E+10	TE 127M	1.480 E+06	1.0E+09
ZN 65	9.953 E+05	1.8E+10	TE 129M	1.761 E+06	1.3E+09
RB 86	1.983 E+05	2.1E+10	CS 134	1.014 E+06	5.6E+10
SR 89	2.157 E+06	1.1E+10	CS 136	1.709 E+05	5.7E+09
SR 90	1.010 E+08	1.0E+11	CS 137	9.065 E+05	5.0E+10
Y 91	2.627 E+06	5.9E+06	BA 140	1.743 E+06	2.6E+08
ZR 95	2.231 E+06	3.5E+08	CE 141	5.439 E+05	3.2E+07
NB 95	6.142 E+05	3.8E+08	CE 144	1.195 E+07	1.6E+08
MO 99	1.354 E+05	3.2E+08	I 131	1.624 E+07	1.0E+12
			I 133	3.848 E+06	9.6E+09

*If SR-90 analysis is performed, use P(I) given in I-131 for unidentified components. If SR-90 and I-131 analyses are performed, use P(I) given in CS-137 for unidentified components. If SR-90, I-131, and CS-137 analyses are performed, use P(I) given in Zn-65 for unidentified components.

2.0 RADIATION MONITORING SETPOINTS

Effluent radiation monitor alarm/trip setpoints shall be determined using the calculations presented in the following sections. The calculations define the relationships between the measured effluent activity, the maximum allowable effluent activity, the effluent flowrate, and the dilution available in the restricted area (as defined for effluent releases in the Selected Licensee Commitments and/or Technical Specifications) which must be controlled to assure that the instantaneous release rate is not exceeded.

The setpoints shall be determined for those monitors listed in the appropriate tables of the Selected Licensee Commitments and/or Technical Specifications.

2.1 LIQUID MONITORS

The following equation shall be used to calculate liquid radiation monitor setpoints:

$$\frac{Cf}{F + f} \leq MPC$$

where:

MPC = the effluent concentration limit implementing 10CFR20 for the site, in µCi/ml.

C = the radioactivity concentration in µCi/ml, in the effluent line prior to dilution and subsequent release, which may be the setpoint and, if so, represents a value which, if exceeded, would result in concentrations exceeding the limits of 10CFR20 in the unrestricted area.

f = the flow measured at the radiation monitor location in gpm.

F = the dilution water flow as measured prior to the release point in gpm.

(Note that if no dilution is provided, $C \leq MPC$. Also, note that when (F) is large compared to (f), then $F + f \cong F$.)

2.2 GAS MONITORS

The following equation shall be used to calculate noble gas radiation monitor setpoints based on Xe-133:

$$K_i(\overline{X/Q})\tilde{Q}_i < 500$$

$$\tilde{Q}_i = 4.72E{+}2\ C\ f \text{ (See Section 1.2.1)}$$

where:

C = the gross activity in undiluted effluent, in µCi/ml.

f = the flow from the tank or building and varies for various release sources, in cfm.

where:

1.14E5 = 10^6pCi/μCi x 10^3ml/kg ÷ 8760 hr/yr.

U_{aw} = Water consumption by age group, ℓ/yr.

infant	330
child	510
teen	510
adult	730

D_w = Dilution factor from the near field area to the potable water intake.

U_{af} = fish consumption by age group, kg/yr.

infant	--
child	6.9
teen	16
adult	21

BF_i = Bioaccumulation factor for radionuclide, 'i', in fish, pCi/kg per pCi/l, from Table 3.1-1.

$DF_{ai\tau}$ = Dose conversion factor for radionuclide, 'i', by age group in pre-selected organ, τ, in mrem/pCi, from Tables 3.1-2, 3.1-3, 3.1-4, and 3.1-5, respectively.

Using the above information, $A_{ai\tau}$ values for the adult have been calculated for each site. This information is provided in the Table "X" 4.0-3 where "X" is the appendix for the site in question.

3.1.2 Gaseous Effluents

The dose contributions from measured quantities of radioactive materials identified in gaseous effluent released to unrestricted areas shall be calculated for the maximum exposed individual using the following equations:

3.1.2.1 Noble Gases

For gamma radiation:

$$D_\gamma = 3.17\ E\text{-}8 \sum_i M_i\ [(\overline{X/Q})\ \tilde{Q}_i]$$

For beta radiation:

$$D_\beta = 3.17\ E\text{-}8 \sum_i N_i\ [(\overline{X/Q})\ \tilde{Q}_i]$$

where:

3.17E-08 = The inverse of the number of seconds in a year.

M_i = The air dose factor due to gamma emissions for each identified noble gas radionuclide, in mrad/yr per μCi/m³ from Table 1.2-1.

f_L = the fraction of the annual intake of fresh leafy vegetation grown locally, (1.0).

f_g = the fraction of the annual intake of stored vegetation grown locally, (0.76).

t_L = the average time between harvest of leafy vegetation and its consumption, in seconds, 8.6×10^4 (1 day).

t_h = the average time between harvest of stored vegetation and its consumption, in seconds, 5.18×10^6 (60 days).

Y_v = the vegetation area density, 2.0 kg/m^2.

and all other factors are previously defined.

The concentration of tritium in vegetation is based on the airborne concentration rather than the deposition. Therefore, the R_i^V is based on [X/Q]:

$$R_i^V[X/Q] = K'K''' \quad U_a^L f_L + U_a^S f_g \ (DFL_i)_a \quad [0.75(0.5/H)] \text{ (mrem/yr per } \mu\text{Ci/m}^3\text{)}.$$

All terms defined previously.

3.1.3 Direct Radiation

Direct radiation is that radiation from confined sources and does not include any external component from radioactive effluents. The point kernel method has been used to calculate offsite dose rates from radioactive materials stored in the refueling water storage tanks, reactor makeup water storage tanks, and temporary on-site radwaste storage tanks. Dose calculations using this method performed for Duke Nuclear Stations indicate direct radiation doses are much less than 0.01 mrem/yr. and, therefore, makes a negligible contribution to individual dose. Likewise, direct and air-scatter radiation dose contributions from the onsite Independent Spent Fuel Storage Installation (ISFSI) at Oconee have been calculated. The maximum dose rate to the nearest potential resident from the Oconee ISFSI is <0.1 mrem/yr. Direct radiation doses will not be calculated routinely.

3.2 SIMPLIFIED DOSE PROJECTIONS

Station long-term historical and dose projection calculations are periodically performed to determine the station's status with respect to meeting annual ALARA goals specified in the Selected Licensing Commitment Manual and/or Technical Specifications. Such calculations are used to verify that adequate margin remains during a report period to allow normal station and radwaste system operation, including anticipated operational occurrences, for the remainder of the report period without exceeding applicable goals. Station long-term historical and dose projection calculations can be performed using generic methodology, LADTAP and/or GASPAR, or simplified dose calculation methods.

Simplified liquid and gaseous dose calculation methods yield assumed maximum individual total body and maximum organ dose results for historical critical age group(s) and organ(s) using only data for the historical critical pathway(s) and radionuclide(s). Offsite doses calculated for regulatory

reporting or compliance purposes as well as individual pre-release dose calculations require B, the Selected Licensing Commitment Manual and/or Technical Specifications and/or abnormal release offsite dose contributions are not calculated using simplified dose calculation methods due to the extensive use of historical based assumptions in developing the simplified methodology.

Simplified dose calculation procedures are presented in site specific sections where they apply.

3.3 FUEL CYCLE CALCULATIONS

In accordance with the requirements of 40CFR190, the annual dose commitment to any member of the general public shall be calculated to assure that doses are limited to 25 millirems to the total body or any organ with the exception of the thyroid which is limited to 75 millirems. In accordance with the requirements of the Selected Licensee Commitments and/or Technical Specifications, the annual dose commitment shall also be calculated any time that one of the quarterly dose limits of the Selected Licensee Commitments and/or Technical Specifications is exceeded; these annual dose commitments may not just be calculated for the calendar year.

The "Uranium fuel cycle" is defined in 40CFR Part 190.02(b) as:

> "Uranium fuel cycle means the operations of milling or uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle."

Based on this definition of the fuel cycle and the information in 10CFR51 Table S-3 and WASH-1248, the radiological impact of the following operations has been assessed for Duke Nuclear Stations:

3.3.1 Milling

No milling operations occur within fifty miles of any Duke Nuclear Station. The increment of dose from milling operations to any individual within fifty miles of any Duke Nuclear Station is negligible.

3.3.2 Conversion

No uranium hexafluoride production occurs within fifty miles of any Duke Nuclear Station. The increment of dose from UF_6 production to any individual within fifty miles of any Duke Nuclear Station is negligible.

3.3.3 Enrichment

No uranium enrichment operations occur within fifty miles of any Duke Nuclear Station. The increment of dose from enrichment operations to any individual within fifty miles of any Duke Nuclear Station is negligible.

3.3.4 Fuel Fabrication

No fuel fabrication operations occur within fifty miles of any Duke Nuclear Station. The increment of dose from fabrication operations to any individual within fifty miles of any Duke Nuclear Station is negligible.

3.3.5 Nuclear Power Production

The production of electricity for public use using light-water-cooled nuclear power stations results in increments of dose to individuals within fifty miles of any station due to liquid and gaseous effluent releases and direct radiation or skyshine. The increments of dose resulting from liquid and gaseous effluent releases will be calculated using the methodology presented in Sections 3.1.1 and 3.1.2 or the LADTAP and GASPAR computer programs. The dose from direct radiation, skyshine, and radiation from the station storage facilities has been estimated using conservative assumptions (see Section 3.1.3).

In certain situations more than one nuclear power station site may contribute to the doses to be considered in making fuel cycle dose assessments in accordance with 40CFR190. Situations involving more than one station will be presented in the section on site specific information.

3.3.6 Fuel Reprocessing

No fuel reprocessing operations occur within fifty miles of any Duke Nuclear Station. The increment of dose from reprocessing operations to any individual within fifty miles of any Duke Nuclear Station is negligible.

To summarize, only dose increments from nuclear power production operations (Section 3.3.5) need be considered in calculations to demonstrate compliance with the requirements of 40CFR190.