FLORIDA POWER & LIGHT COMPANY
ST LUCIE UNIT NO. 1

EQUIPMENT QUALIFICATION RADIATION
DOSE MAP DEVELOPMENT

1. SOURCE TERMS

Several sets of source terms were developed for use in calculations of dose rates and doses for normal operating and design basis accident (DBA) conditions. Source terms used in calculations of the 40 year normal operations dose are, for the most part, those previously developed for plant shielding and personnel protection purposes, and are based on the type of radionuclide inventory information represented by the current revisions of Sections 11.1, 11.2 and 12.2 of the St. Lucie Unit No. 1 FSAR, as well as plant process information (flow rates, volumes, etc.) presented elsewhere. Although the introduction of Section 1.4 of NUREG-0588 speaks of "the normally expected radiation environment", the source terms actually used are maximum values, ultimately predicated on reactor coolant water containing a radionuclide inventory corresponding to an assumption of 1% failed fuel.

DBA source term sets were developed from the St. Lucie Unit No. 1 core inventory of radionuclides, which was derived from Table 4.3-1 of the Combustion Engineering System 80 "Radiation Design Guide".* The St. Lucie Unit No. 1 core inventory, separated for convenience into noble gases, halogens and other nuclides is shown in Table 1.

Positions 1.4(1)-(5) of NUREG-0588 provide guidance for the preparation of source terms used in an equipment dose assessment. The St. Lucie Unit No. 1 calculations followed these recommendations in developing mechanistically rational models for radionuclide transport as a function of time through the containment and other parts of the plant. The major division of DBA source term sets are for containment atmosphere, plateout, and containment sump water. These are discussed in the following sections.

1.1 CONTAINMENT ATMOSPHERE SOURCE

In accordance with Position 1.4(1) of NUREG-0588 (also, for example, Section II.B.2 of NUREG-0737 (10/80), Section 2.1.6.b of NUREG-0578 (7/79), Table 1 of Regulatory Guide 1.7 (R2, 11/78), and Position C.2 of Regulatory Guide 1.89 (R0, 11/74)), it was assumed that the following percentages of the core inventory of radionuclides are instantaneously released from the fuel to the containment atmosphere at the start of a LOCA:

100% noble gases
50% halogens
1% other nuclides

*Combustion Engineering, "Radiation Design Guide", Rev. 4, SYS80-PE-PG (7/12/79).

This release is assumed to be uniformly distributed in the containment atmosphere, a good assumption in a PWR containment which lacks the many compartments found in a BWR containment. This is also a conservative assumption for calculation of time-dependent dose rates, as stated in Section 4 of Appendix D of NUREG-0588, and is a "rational assumption", in accordance with the instructions of Position 1.4(3) of the NUREG. The St. Lucie Unit No. 1 containment free volume was taken as 2.60×10^6 ft^3 (7.36×10^{10} cm^3). Specific radionuclide activities corresponding to this model appear in Table 2. The above stated assumptions for St Lucie Unit No. 1 are the same as for St Lucie Unit No. 2.

1.1.1 Containment Spray Washout Model

The action of the containment spray system will remove certain radionuclides from the containment atmosphere as a function of time. The St. Lucie calculations model this effect, as suggested in Position 1:4(4) of NUREG-0588. The noble gas inventory is not affected by the spray, and would only be removed through radioactive decay. The St. Lucie Unit No. 1 model assumes that 90% of the airborne halogens (elemental) are removed at a rate characterized by a removal coefficient of 10.0 hr^{-1}, and that the remaining 10% are not removed at all. This is conservative compared to the NUREG-0588 Appendix D model of removal coefficients of: 27.2 hr^{-1} applied to 91% elemental halogens (iodine); and 0 hr^{-1} applied to 4% organic halogens (iodine) (note that the NUREG-0588 apportionment of the iodines among elemental, particulate and organic forms follows the guidance of Regulatory Guide 1.4 (R2, 6/74)). It was further assumed for St. Lucie Unit No. 1 that the solid fission products would be removed with a coefficient of 0.43 hr^{-1}. The removal coefficients applied to different species of nuclides are summarized in Table 3. The removal rates for the St Lucie Unit No. 1 Spray Washout Model are the same as those for St Lucie Unit No. 2.

1.1.2 Plateout Model

In addition to containment spray removal, the competing process of plateout of radionuclides on the exposed surfaces inside the containment also removes radionuclides from the containment atmosphere. Position 1.4(5) of NUREG-0588 suggests that a mechanistic model be used for plateout, rather than the nonmechanistic assumption of 50% instantaneous plateout of the halogens released from the core. The St. Lucie Unit No. 1 model assumes a plateout removal factor from the containment atmosphere of 1.0 day^{-1} for all radionuclides except noble gases, distributed uniformly on a surface area of 2.5×10^5 ft^2. This compares to a NUREG-0588 Appendix D assumption of a coefficient of 1.23 hr^{-1} for elemental iodine only, and a surface area of 5.0×10^5 ft^2.

1.2 CONTAINMENT PLATEOUT SOURCE

The plateout model for removal of radionuclides from the containment atmosphere through deposition on the exposed surfaces in the containment is discussed in Subsection 1.1.2. Naturally, the deposition rate on surfaces, which determines the time-dependent plateout source term set, is simply equal to the removal rate from the atmosphere. The St Lucie Unit No. 1 plateout removal factor of 1.0 day^{-1} in contrast to the St Lucie Unit No. 2 plateout removal factor of 1.0 hr^{-1} tends to enhance the conservatism for submersion γ cloud dose by increasing the source terms.

1.3 CONTAINMENT SUMP SOURCE

A mechanistic model of the time-dependence of the radioactive source in the containment sump water would begin with a zero source at the start of the DBA, building up with time as halogens wash out of the containment atmosphere due to containment spray removal. However, as recognized in Appendix D (Section 6) of NUREG-0588, since most of the halogens would be transferred from the containment atmosphere to the sump in a few minutes time, there would be little difference in calculation of dose whether a mechanistic model or the older (Regulatory Guide 1.7, TID-14844) model, which assumed instantaneous release of 50% of the core's inventory of halogens and 1% of the other nuclides to the sump, were used. Additionally the conservative assumption was made that 50% of the noble gases were dissolved immediately in the sump water without detracting from the 100% noble gas core inventory source terms assumed for air.

The dilution volume for the containment sump (recirculated) water was created by using the combined volumes of the reactor coolant, the Safety Injection Tanks, and the minimum volume of the Refueling Water Tank. This total volume is approximately 427,000 gallons. However, a more conservative (greater nuclide concentration) value of 400,000 gallons (1.52×10^9 cm^3) was actually used in the calculations. The resulting initial sump nuclide inventory is shown in Table 5. The inventories at later times can be derived by applying the appropriate radioactive decay factor to each nuclide. The St Lucie Unit No. 1 sump sources differed from St Lucie Unit No. 2 sump sources in that the latter did not assume noble gas source terms.

2. DOSE CALCULATIONS

Dose values were determined for normal and post-accident situations at many locations throughout the plant where safety-related equipment are located. Source terms were created using the models described in Section 1. Doses for 40 years normal operation are based on calculations done in the course of the St. Lucie Unit No. 1 design work, as described in Section 12 of the FSAR. These doses are quite conservative since the calculations assumed worst case conditions of operation, shielding, radionuclide inventories (e.g., 1% failed fuel in reactor coolant), etc. The normal dose values are recorded on the Equipment Radiation Dose Maps.

For most pieces of equipment, the dose received during and following a DBA event far exceeds the normal operating dose, and is, thus, the governing contribution. Calculations were done both inside and outside containment, for both gamma and beta radiation (as applicable), and values for four different times following the start of the DBA were recorded on the dose maps: 1 day, 30 days, 6 months, and 1 year. This differs from the 2 hour, 30 day, and 1 year times for St Lucie Unit No. 2.

Accident doses are discussed in the following section for inside and outside containment. Doses were generally calculated using the point-kernel technique integrated over time.

2.1 DOSE INSIDE CONTAINMENT

Three overall sources locations of radiation are considered inside the containment and modeled separately: (1) airborne; (2) plateout; and, (3) sump water. The airborne gamma and beta doses are based on the time-dependent source terms of Section 1.1, which take into account removal of activity from the containment atmosphere by radioactive decay, plateout, and containment spray washout. Shield walls and finite cloud sizes are factored into the calculations which were carried out at a number of point inside the containment. The relative contributions of these three sources to a particular piece of equipment depend on the exact location of the equipment in relation to each source, and to internal shield walls.

The final containment dose model treats radiation exposure due to containment sump water, and was used to produce gamma and beta dose contributions to equipment located in the vicinity of the sump, or immersed in the sump water following a LOCA. The primary source of activity in the sump water is the iodine which is rapidly washed out of the containment atmosphere by the containment spray system.

Doses from the three sources are combined at a number of locations in the containment, and are listed separately for gamma with instruction to multiply by a factor of six for beta dose. As described in the instructions for the use of the Equipment Radiation Dose Maps, the free-air beta dose given on the maps can be reduced through consideration of the attenuation effects of any covering that may be present between the beta source and the sensitive component of the equipment. These dose reduction factors, shown in Table 6, were determined as a function of time and covering thickness for an elastometric material of density 1.0 gm/cm^3 and a metal of density 8 gm/cm^3.

This treatment of beta dose, giving both unshielded abd shielded doses, is consistent with the guidelines of Positions 1.4(6)-(10) of NUREG-0588. Its guidance of 70 mils of elastrometric material thickness reducing the beta dose by a factor of 25 is somewhat more conservative than the Bulletin 79-01B guidelines of 100 reduction factor for a 70 mil thickness.

2.2 DOSES OUTSIDE CONTAINMENT

Doses to equipment located outside containment are comprised of contributions from direct gamma radiation shine through the containment and shield building walls, direct gamma exposure from systems containing recirculated (sump) water, internal gamma and beta exposure if the equipment itself contains radioactive fluid, and airborne gamma and beta cloud dose exposure from leakage of containment atmosphere out of the containment building. The last contribution conservatively assumes 0.5% leakage per day from the containment; all going to the Reactor Auxiliary Building.

The actual source terms used are discussed in Section 1. Calculational methods for determining doses to equipment located outside the containment are essentially the same as those used for equipment inside the containment; i.e., point-kernel integration. The factors considered and the methods used are consistent with the guidelines of NUREG-0588, in particular, Position 1.4(11).

2.3 EXAMPLES OF DOSE DETERMINATION OF PARTICULAR POINTS

This section presents two examples which depict the contributory factors and calculational methods used in generating doses for two actual points appearing on the Equipment Dose Maps. One example is for a dose point located inside containment; the other is for a dose point outside containment in the Reactor Auxiliary Building. In both cases, both normal and accident doses are computed.

The basic assumptions underlying the generation of the Equipment Dose Maps are given elsewhere in this Section. In particular, compliance with all NRC guidelines given in Position 1.4 of NUREG-0588, by adopting either the NRC or more conservative models, is demonstrated point-by-point. In brief, the Equipment Dose Map values were determined on the basis of time-dependent, mechanistic models of radioactive decay, containment spray washout, plateout, containment leakage, and filtration where applicable (e.g., the Shield Building Ventilation System Filters, and the Control Room Emergency Filters). Source terms ae discussed in Section 1.

Fourteen contributing factors to the dose were identified and considered. These are enumerated and discussed below. Gamma doses were determined using a point-kernel methodology of the Rockwell type. Beta doses were determined using the ratio of 6:1 for beta dose to gamma dose as per Lloyd Bonzon in his ANS paper*. Dose contributions from the factors comprise the matrix values for each of the two sample points as shown in Tables 7 and 8. The Unit No. 1 beta dose calculation maintained a fixed β/γ dose ratio of 6 independent of time, while the Unit No. 2 calculation determined β dose at a function of time after an accident.

<u>Dose Contributors</u>

(1) <u>Normal Operation External (γ,β)</u> - Direct external dose due to gamma and beta irradiation over a 40 year period. The source terms are based on the assumption of 1% failed fuel in the reactor coolant. No credit is taken for the plant capacity factor, however, credit is taken for conservatively assumed equipment capacity factor or use factor.

(2) <u>Normal Operations Internal (β)</u> - Similar to (1), except that this contributor refers to immersion of internal components in radioactive fluids (gaseous or liquid) containing the beta sources.

(3) <u>DBA Direct Radiation From Sump Water. (γ)</u> - Contribution from the direct external "shine" of gammas from safety or shutdown systems containing sump water following an accident.

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*L Bonzon, "In-Containment Radiation Environments Following the Hypothetical LOCA", Transactions of the 1976 ANS Annual Meeting, Toronto, Canada.

(4) DBA Direct Radiation From Sump Water (β) - Similar to (2), except that the source here is DBA sump water rather than water containing a radionuclide inventory resulting from 1% failed fuel during normal operation.

(5) DBA Direct Radiation From Containment (γ) - The conatinment atmosphere is treated as a spherical cloud source for dose calculations at points outside containment.

(6) DBA Direct Radiation From Ambient Sources (γ) - Similar to (3), except that filtration sources, such as the Shield Building Ventilation System filters, are considered rather than sump water containing sources.

(7) DBA Submersion Cloud (γ) - Direct shine due to gamma radiation from airborne sources.

(8) DBA Direct Radiation From Sump (γ) - Direct gamma radiation to points external to the containment sump from sump sources.

(9) DBA Submersion Sump Radiation (γ) - Direct gamma radiation to equipment submerged in the containment sump from sump sources.

(10) DBA Submersion Sump Radiation (β) - Similar to (9), except that beta rather than gamma radiation is treated.

(11) DBA Plateout (γ) - Gamma radiation from sources plated-out on exposed surfaces in the containment.

(12) DBA Plateout (β) - Similar to (11), except that beta rather than gamma radiation is treated.

(13) DBA Submersion Cloud (β) - Similar to (7), except that beta rather than gamma radiation is treated.

(14) DBA Direct Radiation From Sump (β) - Similar to (8), except that beta rather than gamma radiation is treated.

TABLE 1

CORE INVENTORY

Noble Gases

Nuclide	Ci	Nuclide	Ci	Nuclide	Ci	Nuclide	Ci
Kr-85m	2.10+7[a]	Kr-89	6.73+7	Xe-133	1.68+8	Xe-138	1.34+8
Kr-85	6.65+5	Kr-90	6.65+7	Xe-135m	3.39+7	Xe-140	6.87+7
Kr-87	3.85+7	Kr-91	4.91+7	Xe-135	3.02+7	Xe-143	1.64+6
Kr-88	5.49+7	Xe-131m	5.86+5	Xe-137	1.48+8	Xe-144	3.66+5

Halogens

Nuclide	Ci	Nuclide	Ci	Nuclide	Ci	Nuclide	Ci
Br-84	1.61+7	Br-89	2.42+7	I-131	8.37+7	I-135	1.56+8
Br-85	2.07+7	Br-90	1.53+7	I-132	1.22+8	I-137	7.00+8
Br-87	3.33+7	I-127	1.32+25[b]	I-133	1.68+8	I-138	3.51+7
Br-88	3.51+7	I-129	2.09+0	I-134	1.81+8		

Other Nuclides

Nuclide	Ci	Nuclide	Ci	Nuclide	Ci	Nuclide	Ci
Se-84	1.54+7	Nb-95	1.41+8	Te-131m	1.26+7	Cs-137	7.25+6
As-85	2.67+6	Zr-99	1.38+8	Te-131	7.25+7	Ba-137m	6.87+6
Se-85	9.52+6	Nb-99	1.45+8	Sn-132	1.43+7	Cs-138	1.43+8
Se-87	5.53+7	Mo-99	1.53+8	Sb-132	4.01+7	Cs-140	1.29+8
Rb-88	5.58+7	Tc-99m	1.32+8	Te-132	1.19+8	Ba-140	1.46+8
Sr-89	7.75+7	Mo-103	1.34+8	Sn-133	4.96+6	La-140	1.50+8
Rb-90	6.84+7	Tc-103	1.36+8	Sb-133	4.50+7	Cs-143	2.79+7
Sr-90	5.40+6	Ru-103	1.37+8	Te-133m	6.02+7	Ba-143	1.11+8
Y-90	5.67+6	Tc-106	5.65+7	Te-133	9.59+7	La-143	1.25+8
Rb-91	8.81+7	Ru-106	3.87+7	Cs-134	1.58+7	Ce-143	1.26+8
Sr-91	9.52+7	Sn-129	8.89+6	Sb-134	8.02+6	Pr-143	1.24+8
Y-91m	5.48+7	Sb-129	2.76+7	Te-134	1.27+8	Cs-144	8.52+6
Y-91	1.01+8	Te-129m	7.17+6	Sb-135	5.03+6	Ba-144	8.25+7
Sr-95	1.02+8	Te-129	2.62+7	Te-135	6.62+7	La-144	1.09+8
Y-95	1.34+8	Sn-131	2.45+7	Cs-135	2.10+1	Ce-144	9.95+7
Zr-95	1.40+8	Sb-131	6.75+7	Cs-136	4.42+6	Pr-144	1.00+8

Notes: (a) Read as 2.10×10^7 curies
(b) I-127 is stable.. Number given is total atoms.

TABLE 2

INITIAL CONTAINMENT ATMOSPHERE
SPECIFIC ACTIVITIES
FOLLOWING A LOCA (a)

Noble Gases

Nuclide	Ci/Cm^3	Nuclide	Ci/Cm^3	Nuclide	Ci/Cm^3	Nuclide	Ci/Cm^3
Kr-85m	2.86-4 (b)	Kr-89	9.15-4	Xe-133	2.29-3	Xe-138	1.81-3
Kr-85	9.04-6	Kr-90	9.04-4	Xe-135m	4.61-4	Xe-140	9.34-4
Kr-87	5.22-4	Kr-91	6.67-4	Xe-135	4.10-4	Xe-143	2.15-5
Kr-88	7.47-4	Ke-131m	7.96-6	Xe-137	2.02-3	Xe-144	4.97-6

Halogens

Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3
Br-84	1.10-4	Br-89	1.64-4	I-131	5.68-4	I-135	1.06-3
Br-85	1.41-4	Br-90	1.03-4	I-132	8.26-4	I-137	4.76-4
Br-87	2.26-7	I-129	1.42-11	I-133	1.14-3	I-138	2.39-4
Br-88	2.39-4			I-134	1.23-3		

Other Nuclides

Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3
Se-84	2.08-6	Nb-95	1.92-5	Te-131m	1.71-6	Cs-137	9.84-7
As-85	3.64-7	Zr-99	1.88-5	Te-131	9.84-6	Ba-137m	9.34-7
Se-85	1.29-6	Nb-99	1.97-5	Sn-132	1.94-6	Cs-138	1.94-5
Se-87	2.07-6	Mo-99	2.07-5	Sb-132	5.45-6	Cs-140	1.76-5
Rb-88	7.58-6	Tc-99m	1.78-5	Te-132	1.62-5	Ba-140	1.98-5
Sr-89	1.05-5	Mo-103	1.81-5	Sn-133	6.74-7	La-140	2.05-5
Rb-90	9.30-6	Tc-103	1.84-5	Sb-133	6.12-6	Cs-143	3.78-6
Sr-90	7.34-7	Ru-103	1.86-5	Te-133m	8.18-6	Ba-143	1.50-5
Y-90	7.71-7	Tc-106	7.67-6	Te-133	1.30-5	La-143	1.70-5
Rb-91	1.20-5	Ru-106	5.26-6	Cs-134	2.15-6	Ce-143	1.71-5
Sr-91	1.29-5	Sn-129	1.21-6	Sb-134	1.09-6	Pr-143	1.68-5
Y-91m	7.45-6	Sb-129	3.75-6	Te-134	1.73-5	Cs-144	1.16-6
Y-91	1.37-5	Te-129m	9.74-7	Sb-135	6.83-7	Ba-144	1.12-5
Sr-95	1.38-5	Te-129	3.56-6	Te-135	8.99-6	La-144	1.48-5
Y-95	1.81-5	Sn-131	3.38-6	Cs-135	2.86-12	Ce-144	1.35-5
Zr-95	1.91-5	Sb-131	9.17-6	Cs-136	6.01-7	Pr-144	1.36-5

Notes: (a) 100% core inventory noble gases
50% core inventory halogens
1% core inventory other nuclides

Diluted in $2.6x10^6\ ft^3$ ($7.36x10^{10}\ cm^3$)

(b) read as $2.86x10^{-4}$ curies/cm^3

TABLE 3

CONTAINMENT SPRAY
WASHOUT MODEL

Nuclide Species	Percentage of Core Inventory Released to Cont. Atm.	Percentage by Chemical Form SL1	Percentage by Chemical Form NRC[a]	Removal Coefficient hr^{-1} SL1	Removal Coefficient hr^{-1} NRC
Noble Gases	100	-	-	0.0	0.0
Halogens	50				
Elemental		90	91	10.0	27.2
Particulate		10	5	0.0	0.43
Organic			4		0.0
Other Nuclides	1	-	-	0.43	(b)

Notes: (a) NUREG-0588, Appendix D (R0, 12/79)

(b) Not stated in NUREG-0588

TABLE 4

PLATEOUT MODEL (a)

Nuclide Species	Percentage of Core Inventory Released to Cont. Atm.	Removal Coefficient SL1	Removal Coefficient NRC(b)
Noble Gases	100	0.0	0.0
Halogens	50		
Elemental		1.0 day^{-1}(c)	1.23 hr^{-1}
Particulate		1.0 day^{-1}(c)	0.0
Organic		1.0 day^{-1}(c)	0.0
Other Nuclides	1	1.0 day^{-1}	-

Notes: (a) Containment surface area

SL1- $2.5x10^5$ ft^2
NRC- $5.0x10^5$ ft^2

(b) NUREG-0588, Appendix D (RO, 12/79)

(c) The use of a small plateout removal factor tends to increase the source terms for the submersion γ cloud dose.

TABLE 5

INITIAL CONTAINMENT SUMP WATER SPECIFIC ACTIVITIES FOLLOWING A LOCA[a]

Noble Gases Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3
Kr-85m	6.93-3	Kr-89	2.22-2	Xe-133	5.55-2	Xe-138	4.43-2
Kr-85	2.20-4	Kr-90	2.20-2	Xe-135m	1.12-2	Xe-140	2.27-2
Kr-87	1.27-2	Kr-91	1.62-2	Xe-135	9.97-3	Xe-143	5.42-4
Kr-88	1.81-2	Xe-131m	1.93-4	Xe-137	4.89-2	Xe-144	1.21-4

Halogens Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3
Br-84	5.32-3[b]	Br-89	7.94-3	I-131	2.75-2	I-135	5.14-2
Br-85	6.82-3	Br-90	5.00-3	I-132	4.00-2	I-137	2.30-2
Br-87	1.09-2	I-129	6.87-10	I-133	5.51-2	I-138	1.16-2
Br-88	1.16-2			I-134	5.98-2		

Other Nuclides Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3	Nuclide	Ci/cm^3
Se-84	1.01-4	Nb-95	9.30-4	Te-131m	8.27-5	Cs-137	4.76-5
As-85	1.76-5	Zr-99	9.11-4	Te-131	4.76-4	Ba-137m	4.52-5
Se-85	6.26-5	Nb-99	9.53-4	Sn-132	9.39-5	Cs-138	9.39-4
Se-87	1.00-4	Mo-99	1.00-3	Sb-132	2.64-4	Cs-140	8.50-4
Rb-88	3.67-4	Tc-99m	8.64-4	Te-132	7.85-4	Ba-140	9.58-4
Sr-89	5.09-4	Mo-103	8.78-4	Sn-133	3.27-5	La-140	9.90-4
Rb-90	4.50-4	Tc-103	8.92-4	Sb-133	2.96-4	Cs-143	1.83-4
Sr-90	3.55-5	Ru-103	9.02-4	Te-133m	3.96-4	Ba-143	7.29-4
Y-90	3.73-5	Tc-106	3.71-4	Te-133	6.31-4	La-143	8.22-4
Rb-91	5.79-4	Ru-106	2.55-4	Cs-134	1.04-4	Ce-143	8.27-4
Sr-91	6.26-4	Sn-129	5.84-5	Sb-134	5.28-5	Pr-143	8.13-4
Y-91m	3.61-4	Sb-129	1.82-4	Te-134	8.36-4	Cs-144	5.61-5
Y-91	6.63-4	Te-129m	4.72-5	Sb-135	3.31-5	Ba-144	5.42-4
Sr-95	6.68-4	Te-129	1.72-4	Te-135	4.35-4	La-144	7.15-4
Y-95	8.78-4	Sn-131	1.62-4	Cs-135	1.38-10	Ce-144	6.54-4
Zr-95	9.25-4	Sb-131	4.44-4	Cs-136	2.91-5	Pr-144	6.59-4

Notes: (a) 50% core inventory noble gas
50% core inventory halogens
1% core inventory other nuclides
diluted in 400,000 gal ($1.52 \times 10^9 cm^3$)

(b) read as 5.32×10^{-3} curies/cm^3

TABLE 6

BETA DOSE REDUCTION FACTORS

Metal Thickness mils (ρ=8 gm/cm^3)	St Lucie 1 Beta Dose Reduction Factor	Non-Metal Thickness Elastrometric Coating mils (ρ=1 gm/cm^3)	St Lucie 1 Beta Dose Reduction Factor	NRC Beta Dose Reduction Factor[a]
1	.69	1	.96	
2	.48	5	.80	
3	.33	10	.63	
4	.23	15	.50	
5	.16	20	.40	
6	.11	25	.32	
7	.078	30	.25	.10
		35	.20	
8	.054	40	.16	
9	.037	45	.13	
		50	.10	
10	.026	60	.064	
15	.004	70	.04	.01
20	.00067	80	.026	
25	.0001	90	.016	
30	.000017	100	.01	

Notes: (a) From Bulletin IE 79-01B

TABLE 7

CONTRIBUTORS TO DOSE OF POINT INSIDE CONTAINMENT

The contributions from the fourteen factors previously listed are shown for a particular point inside containment, chosen at Elevation 111.00 on the vertical axis of the containment. Only significant contributions are listed. Note that the beta doses do not consider equipment coverings.

Dose Factor	Description	40 year Normal Op Dose	Dose at Various Times Following DBA: After 1 day γ	After 1 day β	After 30 Days γ	After 30 Days β	After 6 months γ	After 6 months β	After 1 year γ	After 1 year β
1	Normal Op External (γ,β)	1.0+4*	-	-	-	-	-	-	-	-
2	Normal Op Internal (β)	-	-	-	-	-	-	-	-	-
3	DBA Direct Radiation from Sump water (γ)	-	1.0+4	-	5.0+4	-	6.5+4	-	7.0+4	-
4	DBA Direct Radiation from Sump water (β)	-	-	-	-	-	-	-	-	-
5	DBA Direct Radiation from Containment (γ)	-	-	-	-	-	-	-	-	-
6	DBA Direct Radiation from Ambient Sources (γ)	-	-	-	-	-	-	-	-	-
7	DBA Submersion Cloud (γ)	-	4.2+6	-	1.7+7	-	1.8+7	-	2.0+7	-
8	DBA Direct Radiation from Sump (γ)	-	-	-	-	-	-	-	-	-

*Read as 1.0×10^4 rads

TABLE 7 (Cont'd)

Dose Factor	Description	40 year Normal Op Dose	Dose at Various Times Following DBA: After 1 day γ	After 1 day β	After 30 Days γ	After 30 Days β	After 6 months γ	After 6 months β	After 1 year γ	After 1 year β
9	DBA Submersion Sump Radiation (γ)	-	-	-	-	-	-	-	-	-
10	DBA Submersion Sump Radiation (β)	-	-	-	-	-	-	-	-	-
11	DBA Plateout (γ)	-	3.5+4	-	1.1+5	-	1.7+5	-	2.0+5	-
12	DBA Plateout (β)	-	-	-	-	-	-	-	-	-
13	DBA Submersion Cloud (β)	-	-	2.5+7	-	1.0+8	-	1.1+8	-	1.2+8
14	DBA Direct Radiation from Sump (β)	-	-	-	-	-	-	-	-	-
	SUBTOTAL	1.0+4	4.2+6	2.5+7	1.7+7	1.0+8	1.8+7	1.1+8	2.0+7	1.2+8
	TOTAL (Includes 40 year Normal Op Dose)		4.2+6	2.5+7	1.7+7	1.0+8	1.8+7	1.1+8	2.0+7	1.2+8

TABLE 8

CONTRIBUTORS TO DOSE OF POINT OUTSIDE CONTAINMENT

For the purpose of this example, a point located in the vicinity of the Control Room Emergency Filters, on Elevation 62.00' of the Reactor Auxiliary Building, was chosen.

Dose Factor	Description	40 year Normal Op Dose	Dose at Various Times Following DBA: After 1 day γ	After 1 day β	After 30 Days γ	After 30 Days β	After 6 months γ	After 6 months β	After 1 year γ	After 1 year β
1	Normal Op External (γ,β)	1.0+3*	-	-	-	-	-	-	-	-
2	Normal Op Internal (β)	-	-	-	-	-	-	-	-	-
3	DBA Direct Radiation from Sump water (γ)	-	-	-	-	-	-	-	-	-
4	DBA Direct Radiation from Sump water (β)	-	-	-	-	-	-	-	-	-
5	DBA Direct Radiation from Containment (γ)	-	<1.0+2	-	<1.0+2	-	<1.0+2	-	<1.0+2	-
6	DBA Direct Radiation from Ambient Sources (γ)	-	1.4+4	-	7.0+4	-	9.1+4	-	1.0+5	-
7	DBA Submersion Cloud (γ)	-	5.2+2	-	4.0+4	-	4.4+4	-	4.4+5	-
8	DBA Direct Radiation from Sump (γ)	-	-	-	-	-	-	-	-	-

*Read as 1.0 X 10^3 rads

TABLE 8 (Cont'd)

Dose Factor	Description	40 year Normal Op Dose	Dose at Various Times Following DBA: After 1 day γ	After 1 day β	After 30 Days γ	After 30 Days β	After 6 months γ	After 6 months β	After 1 year γ	After 1 year β
9	DBA Submersion Sump Radiation (γ)	-	-	-	-	-	-	-	-	-
10	DBA Submersion Sump Radiation (β)	-	-	-	-	-	-	-	-	-
11	DBA Plateout (γ)	-	-	-	-	-	-	-	-	-
12	DBA Plateout (β)	-	-	-	-	-	-	-	-	-
13.	DBA Submersion Cloud (β)	-	-	3.1+3	-	2.4+5	-	2.6+5	-	2.6+5
14	DBA Direct Radiation from Sump (β)	-	-	-	-	-	-	-	-	-
	SUBTOTAL	1.0+3	1.5+4	3.1+3	1.1+5	2.4+5	1.4+5	2.6+5	1.5+5	2.6+5
	TOTAL (Includes 40 year Normal Op Dose)		1.5+4	3.1+3	1.1+5	2.4+5	1.4+5	2.6+5	1.5+5	2.6+5