APPENDIX 15B

CONTAINMENT LEAKAGE AND DOSE CALCULATION MODELS

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APPENDIX 15B

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15B-1	CONTAINMENT LEAKAGE DOSE MODEL

APPENDIX 15B

15B.1 CONTAINMENT LEAKAGE AND DOSE CALCULATION MODEL

→(DRN 04-704, R14)

The RADTRAD computer code, as documented in NUREG/CR-6604, was developed for the US NRC in support of the Alternative Source Term project. The code calculates the time dependent concentration of airborne radioactivity in the containment atmosphere, integrated release of activity to the environment, and the Total Effective Dose Equivalent (TEDE), to individuals at specified locations. Only TEDE doses are required to be calculated to demonstrate conformance to 10CFR50.67, "Accident Source Terms."

The dose conversion factors used to determine TEDE doses are based on Federal Guidance Reports FGR-11 and FGR-12.

15B.1.1 Governing Equations

The radiological consequences of an accident in a nuclear reactor depend upon the quantity of the radioactive material that escapes to the environment or enters the control room. RADTRAD is designed to calculate doses at offsite locations, such as the low population zone, and in the control room. As material is transported through the containment and other buildings, credit is given for several natural and engineered removal mechanisms. Containment sprays remove iodine. The flow between buildings or rooms may be through HEPA filters or charcoal filters. Leakage to the environment may occur. Aerosols can deposit on surfaces within rooms and also in connection paths.

→(DRN 05-1551, R14)

The basic equation for radionuclide transport and removal is the same for all compartments. The governing equation for the number of nuclide *n*, in compartment *i*, during time step *m* is provided with all source and sink terms shown for clarity.

←(DRN 05-1551, R14)

$$\frac{d}{dt}N_{n,i}^{m} = \sum_{i} {}_{v}^{n-l}\beta_{n,v} N_{v,i}^{m}\lambda_{v} + S_{n,i}^{m}$$

$$- \left[{}_{j(j\neq u)}^{\;L} \left| F_{i,j(conv)}^{m} \right| + \lambda_{n} + \lambda_{spr,n}^{m}(t) + \lambda_{dep,n}^{m}(t) + \frac{\eta_{wu,j}^{m}}{100} F_{i,j(forced)}^{m} \right] N_{n,i}^{m}$$

$$+ {}_{j(j\neq i)}^{\;L} \left[\left(1 - \frac{\eta_{wi,j}^{m}}{100} \right) F_{i,j(forced)}^{m} + F_{i,j(conv)}^{m} \right] N_{n,j}^{m}$$

$$\lambda_{n} = \frac{ln(2)}{T_{n}^{1/2}}$$

where,

$N_{n,j}^{m}$ = number of atoms of nuclide *n* in compartment *i* during time step *m*,

$N_{v,i}^{m}$ = number of atoms of nuclide *v* in compartment *i* during time step *m*,

${}_{v}^{n-1}\beta_{n,v}$ = fraction of nuclide *v* that decays to nuclide *n* (dimensionless),

←(DRN 04-704, R14)

→(DRN 04-704, R14)

λ_n = radiological decay constant for nuclide *n* (s^{-1}),

λ_v = radiological decay constant for nuclide *v* (s^{-1}),

$T_n^{1/2}$ = half-life of nuclide *n* (s),

$F_{i,j(conv)}^m$ = volume normalized convection (leakage) air flow rate from compartment *j* to *i*, (s^{-1}),

$F_{i,j(forced)}^m$ = volume normalized forced (filtered) air flow rate from compartment *j* to *i*, (s^{-1}),

L = number of compartments in the plant model,

Vol_j = volume of compartment *j* (m^3),

$\lambda_{spr,n}^m(t)$ = time dependent spray removal coefficient for nuclide *n* (s^{-1}),

$\lambda_{dep,n}^m(t)$ = time dependent natural deposition removal coefficient for nuclide *n* (s^{-1}),

$S_{n,i}^m$ = source injection rate of nuclide *n* to compartment *i* during time step *m* (atoms/s), and

$\eta_{wi,j}^m$ = filter efficiency associated with nuclide *n* and pathway from *j* to *i* (%).

A number of these terms are not applicable to all volumes. For example, fission product removal due to containment spray and/or natural deposition is only credited for reactor containment volumes.

15B.1.1.1 Containment Spray Equations

Sprays may remove aerosols and iodine. It is assumed that noble gases are not affected by containment sprays. The impact of containment sprays to organic iodine is also neglected. The containment spray removal coefficient is modeled in accordance with NUREG-0800, the U.S. N.R.C. Standard Review Plan, Section 6.5.2 guidance.

Removal of particulate (aerosol) iodine due to containment spray is modeled using the following formulas for the spray removal coefficient:

$$\lambda_{spr(part),n}^m(t) = \frac{3hF\left(E/D\right)}{2Vol}$$

where

F = spray flow rate (1750 gpm),
Vol = containment volume (2.677E+06 ft^3),
h = fall height of the drop (150 feet), and
E/D = ratio of a dimensionless collection efficiency to the average drop diameter.

←(DRN 04-704, R14)

→(DRN 04-704, R14)
Since the removal of particulate materials depends markedly upon the relative sizes of the particles and the spray drops, it is convenient to combine parameters that cannot be known. It is conservative to assume (E/D) to be 10 per meter initially (i.e., 1% efficiency for spray drop of 1 millimeter in diameter), changing abruptly to one per meter after the aerosol mass has been depleted by a factor of 50. This results is a particulate spray removal coefficient of 3.596 hr^{-1} until the decontamination factor of 50 is reached, and 0.3596 hr^{-1} thereafter.
→(EC-5000081470, R301)
The Waterford 3 Large Break LOCA offsite and control room intake dose calculations conservatively neglect the removal of elemental iodine from the containment atmosphere via the containment sprays. However, for the isolated cases (e.g., CVAS filter shine) where it is credited Standard Review Plan guidance is used. The first-order removal coefficient by spray for elemental iodine is:
←(EC-5000081470, R301)

$$\lambda^{m}_{spr(elem),n}(t) = \frac{6K_gTF}{(V)D}$$

where,

K_g =	gas-phase mass-transfer coefficient,
T =	time of the fall of the drop,
F =	containment spray flow rate,
V =	containment volume, and
D =	mass-mean diameter of the spray drops.

The SRP states that maximum decontamination factor which may be applied is 200. The removal coefficient for Waterford 3 for elemental iodine was determined to be 20/hr.

15B.1.1.2 Natural Deposition Equations

Reduction of airborne radioactivity by natural deposition within the containment may be credited. The methodology used differs between elemental and particulate iodine. As with containment sprays, deposition of organic iodine is neglected.

During injection, the removal of elemental iodine by wall deposition may be estimated by the first-order estimate from the Standard Review Plan

$$\lambda_{w,elem} = \frac{K_w A}{Vol}$$

where,

$\lambda_{w,elem}$ =	the first-order removal coefficient by wall deposition,
A =	the wetted surface area for the primary containment,
Vol =	the containment building net free volume, and
K_w =	the mass-transfer coefficient.

All available experimental data is enveloped if K_w is taken to be 4.9 meters per second. If this is assumed then the resulting removal coefficient is 0.4 hr^{-1}.

For particulate/aerosol iodine, the Powers 10% aerosol deposition model is specified. This model is described in section 2.2.2.1 of NUREG/CR-6604. The lower bound of this deposition model (10^{th} percentile) is specified. The Powers 10% model is normalized to reactor thermal power as presented in Table 15B-4.
←(DRN 04-704, R14)

→(DRN 04-704, R14)

15B.1.2 PRIMARY CONTAINMENT LEAKAGE PATHWAYS

Post accident containment leakage is limited by Technical Specifications to 0.5 percent of the containment volume per day for the first 24 hours, and at 50 percent of this value for the duration of the accident. As documented in FSAR Section 6.2, containment pressure is reduced to less than half of its peak value by 24 hours into the event. From a dose evaluation standpoint, this leakage can take any of the three following pathways:

a) Leakage to the Shield Building annulus which will be treated by the Shield Building Ventilation System.

b) Leakage to controlled ventilation area of the Reactor Auxiliary Building which will be treated by the Controlled Ventilation Area System (once through filtration).

c) Bypass Leakage.

The total amount of all penetration leakage is limited by a further Technical Specification applied to those lines which are potential sources of bypass leakage.

→(EC-5000081470, R301)

Several relaxations of analysis assumptions in the calculations for offsite dose and for main control room personnel dose (due to both inhalation and shine) have not been relaxed for EQ dose calculations. These include:

* Not crediting SBVS charcoal filtration
* Assuming a 6 minute positive pressure period for the Shield Building
* Allowing Shield Building Maintenance Hatch leakage at 2 days.

These relaxations have been included in the calculations for offsite dose and for main control room personnel dose (due to both inhalation and shine), conservatively increasing these calculated doses. These assumptions are not included in EQ dose calculations.

←(EC-5000081470, R301)

15B.1.3 SECONDARY STEAMING RELEASE MODEL

→(EC-5000081470, R301)

For events that involve fuel failure and steaming releases, the source term of FSAR Table 12.2-12A is utilized. This is the source term developed to determine the gap fission product activities in peak power fuel rods.

←(EC-5000081470, R301)

A maximum allowable peak linear heat rate of 6.3 kw/ft rod average power is assumed for burnups exceeding 54 GWD/MTU, per RG 1.183 footnote 11.

Most sequences for which the source term dose (offsite and Main Control Room) is dominated by the secondary side steaming release pathway (e.g., CEA Ejection, SBLOCA, Inside Containment MSLB) assume a constant primary-to-secondary side leak rate. The mass release rate is conservatively evaluated at 350°F conditions, corresponding to Shutdown Cooling entry conditions.

For events where the steam generator (SG) is not in a faulted condition, the pressure differential between the RCS and the secondary side across the SG tubes is about the same or less than that under normal operating conditions. Thus, for non-faulted SG's when there is no large primary-to-secondary pressure differential, the operational leakage value specified in Technical Specifications is the appropriate primary-to-secondary leakage value to assume in radiological analyses. The actual leak rate during these transients would be the same or, because of reduced differential pressure between primary and secondary sides, less than that for normal operating conditions. Under certain conditions, secondary pressure could exceed primary pressure, resulting in backflow from the SG to the RCS. Each event scenario with non-faulted SG's has slightly different assumptions pertaining to the primary-to-secondary

←(DRN 04-704, R14)

→(DRN 04-704, R14)

leak rate, however at nominal RCS conditions a value of 150 gal/day is assumed, except for Small Break LOCA for which the Technical Specification value of 75 gal/day is assumed. For the MSLB or FWLB cases where a significant pressure drop exists across the SG tubes, a larger (540 gal/day) leakage rate is assumed.

Except for SGTR, the dose analyses assuming primary-to-secondary leakage assume the following RCS masses:

34,260	Pressurizer Liquid Mass (lbm)
4,016	Pressurizer Steam Mass (lbm)
395,502	Non-Pressurizer Liquid Mass (lbm)

For purposes of determining activity concentration, only the non-pressurizer liquid mass is considered. For purposes of determining the steaming rate due to cooldown, the liquid and steam masses of the pressurizer are also considered.

For SGTR, a CENTS analysis is performed for the cooldown. A total initial RCS mass of 467,000 lb. is assumed for that analysis.

Hot Zero Power steam generator inventories (241,450 lbm per SG) are assumed for most events. The HZP inventory is larger than the HFP inventory of 153,700 lbm and is considered more representative of the SG mass present in an intact SG that is being cooled to shutdown cooling entry conditions, and maximizes the initial activity present for the case of releases from a faulted SG. For the SGTR, a low initial SG level (corresponding to 106,300 lbm per SG) was conservatively assumed since this results in earlier uncovery of the top of the SG tubes, which results in increased flashing and a lower decontamination factor. This is a very conservative value that corresponds approximately to the reactor trip setpoint on Steam Generator Level Low.

→(EC-3277, R301)

For events where ADV's from both SG's are credited for steaming (e.g., CEA Ejection), it is assumed that there is an equal release from each ADV.

←(EC-3277, R301)

→(EC-5000081470, R301)

The events explicitly analyzed that involve assumed releases from the ADV's include Small Break LOCA, SGTR, CEA Ejection, Inside Containment Main Steam Line Break, and the combined scenario of Feedwater Line Break / Outside Containment Main Steam Line Break. For secondary steaming release scenarios other than SGTR and FWLB/OC-MSLB, it is assumed that operators act at 30 minutes into the event to begin plant cooldown to shutdown cooling entry conditions of 350°F. A maximum cooldown rate of 100°F is assumed. This is the maximum allowed cooldown rate for forced circulation conditions and conservatively bounds the plant procedural limits of 50°F/hour for natural circulation conditions. A rapid cooldown increases the releases early in the event, thus increasing the releases subject to the higher X/Q's assumed for the initial two hours of the event.

←(EC-5000081470, R301)

The event duration for these secondary steaming cases is assumed to be 7.5 hours. This would bound the time required for shutdown cooling initiation (and thus for terminating steaming releases from the SG for decay heat removal) based on a nominal cooldown rate of 40°F/hour for natural circulation conditions. Thus, for most events, it is assumed that the plant is rapidly cooled to shutdown cooling entry conditions, but that plant conditions are maintained by steaming at those conditions for up to 7.5 hours into the event.

For the FWLB/OC-MSLB, described in FSAR section 15.2.3.1, it is assumed that operator action to cool the plant starts at 4.0 hours and shutdown cooling initiation is assumed at 8.0 hours into the event.

Steam releases for individual secondary steaming events are presented in the input table for the specific events. Note that because similar cooldown scenarios are used, results depend on whether or not a Loss of Offsite Power (LOOP) is assumed or on how many reactor coolant pumps are assumed operating (and providing energy input to the RCS) during the cooldown.

←(DRN 04-704, R14)

→(DRN 04-704, R14)
The cooldown scenario for SGTR is described specifically in FSAR Section 15.6.3.2.
←(DRN 04-704, R14)

→(DRN 05-1551, R14)
15B.1.4 ENGINEERING SAFETY FEATURES (ESF) LIQUID LEAKAGE
→(DRN 05-645, R14; EC-5000081470, R301; EC-3277, R301)
The large break LOCA dose model also accounts for liquid leakage from ESF injection systems as required by Regulatory Guide 1.183. Specifically, a 0.5 gpm leak of safety injection sump water is assumed to occur within the CVAS boundary, therefore this leakage represents a filtered release. This leakage is assumed to begin roughly 23.4 minutes into the event. For offsite and control room submersion/inhalation dose calculations, a ten percent (10%) flashing fraction is assumed, i.e., ten percent of the iodine in the leakage is assumed to go airborne. The ten percent flashing fraction is also assumed for determination of EQ doses. The ten percent flashing fraction is also assumed for the first 24 hours for the calculation of filter shine doses for Main Control Room personnel. After 24 hours a flashing fraction of two percent (2%) is assumed for the filter shine dose calculations for Main Control Room personnel.* Note that the 2 percent value is roughly a factor of ten greater than the maximum that would be expected from the post-LOCA sump temperature profile. For off-site and control room submersion/inhalation dose calculations the ESF leakage is directed to the environment with no credit being taken for mixing or dilution in the reactor auxiliary building. For filter shine doses, mixing in the reactor auxiliary building is credited. Dose rates are determined based on a deterministic model of radionuclide buildup on the CVAS filter train, and the total dose to control room operators is then determined using simple numerical integration techniques.
←(EC-5000081470, R301; EC-3277, R301)

* In the Safety Evaluation Report for Amendment 198, the NRC did not find the assumption of two percent airborne iodine acceptable but approved the filter shine analysis results due to compensating conservatisms. The NRC recommended that the flashing fraction be increased to ten percent or the lower value of two percent be justified when the large break LOCA filter shine dose analyses are revised in the future.

←(DRN 05-645, R14; 05-1551, R14)

→(DRN 04-704, R14)
15B.2 TEDE DOSE CALCULATIONS

The Alternative Source Term dose methodology regulated by 10CFR50.67 has dose acceptance criteria in Total Effective Dose Equivalent (TEDE). The dose acceptance criteria is as follows:

(i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).

(iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

TEDE doses are the sum of the “Deep Dose Equivalent” (DDE) for external exposures and the “Committed Effective Dose Equivalent” (CEDE) for internal exposures.

$$TEDE = DDE + CEDE$$

←(DRN 04-704, R14)

→(DRN 04-704, R14)

The DDE is calculated using a “semi-infinite” model and is determined as follows:

$$D_{c,n} = C_n(t)dt(DCF),$$

where

$C_n(t)$ =	the instantaneous concentration of nuclide n,
DCF =	is the dose conversion factor for nuclide n, and
dt =	the time interval of interest.

The concentration is assumed to be constant in all directions (above the ground). The DCFs are based on Federal Guidance Report (FGR12).

The CEDE doses are the sum of the products of the weighting factors (w_T) applicable to each of the body organs or tissues that are irradiated, and the committed dose equivalent to these organs or tissues.

$$H_E = \sum_T w_T H_T$$

where H_T is the committed dose equivalent, or the dose equivalent to tissues of reference (T) that will be received from an intake or radioactive material by an individual during the 50-year period following the intake. The DCFs to determine these doses are based on FGR 11, which is in turn based on ICRP Publication 30, “Limits for Intakes of Radionuclides by Workers.”

15B.3 MAIN CONTROL ROOM CALCULATIONS

15B.3.1 CONTROL ROOM VENTILATION MODEL

The Control Room Air Conditioning System has a single normal air intake and two widely separated air inlets for emergency operation. The system is a zone isolation system with filtered recirculated air and provisions for maintaining positive pressure in the control room envelope under accident conditions. Intake air for pressurization is filtered before entering the control room envelope.

During normal operation, air is drawn into the system through the single normal air intake. Air handling units condition the air for distribution into the control room envelope, and exhaust fans discharge air from the control room envelope to the environment.

The Control Room Air Conditioning System is automatically isolated on a Safety Injection Actuation Signal (SIAS) or by high radiation detected outside the normal air intake. Automatic isolation involves shutting the normal outside air intake and exhaust isolation valves, stopping the normally operating exhaust fans, opening recirculation dampers, and starting both emergency filtration units in the recirculation mode. Operation of only one emergency filtration unit is required. The control room operators may then initiate the pressurization mode of operation by opening a valve in one of the emergency air intakes. The intake air is ducted to the operating filtration unit before being discharged to the control room envelope. The emergency filtration unit continues to recirculate air within the control room envelope while providing filtered pressurization air flow.

→(EC-5000081470, R301)

Various control room and Control Room Air Conditioning System parameters are input to the control room dose analyses. Dependent on the particular scenario, a minimum (168,500 ft^3) or a maximum (220,000 ft^3) control room envelope volume is assumed in the control room dose evaluation. Recirculation flow through the filter of the operating emergency filtration unit is assumed to be 3,800 Cubic Feet per Minute (CFM). Consistent with Technical Specification limits, filter efficiency of 99% is assumed for aerosol, elemental, and organic iodine. No filtration of noble gases is credited. The pressurization flow rate is assumed to be 225 CFM.

←(DRN 04-704, R14; EC-5000081470, R301)

→(DRN 04-704, R14; EC-5000081470, R301)
Unfiltered in-leakage to the control room envelope distributes through the volume and is processed through the emergency filtration unit with filter efficiencies assumed as above. Control room out-leakage is assumed to be equal to the unfiltered in-leakage.

Unfiltered inleakage applies the most conservative atmospheric dispersion factor for determining the dose consequences from radionuclide leakage. The dose evaluations are performed such that the control room envelope can be either in the recirculation mode or pressurization mode from the beginning of the event. For the pressurization flow, it is assumed that the control room operator selects the more favorable air intake, therefore the lowest atmospheric dispersion factor is reduced by a factor of four as allowed by S.R.P. Section 6.4 and Regulatory Guide 1.183.
←(EC-5000081470, R301)

15B.3.2 CONTROL ROOM DOSE CALCULATION

The dose to a hypothetical individual in the control room is calculated based on the time-integrated concentration in the control room compartment. The air immersion dose in the control room is adjusted to a “finite” model using a “geometry factor”:

$$D_{c,n}^{CR} = C_n(t)dt\left(\frac{DCF}{GF}\right),$$

where

GF = the geometry factor = $\frac{1173}{V_{CR}^{0.338}}$

Inhalation doses in the control room are calculated as follows:

$$D_{t,n}^{CR} = C(t)xBRxOFxDCF_{i,n}$$

where

BR = breathing rate for control room operators,
V = control room volume, and
OF = control room occupancy factor.

The breathing rate assumed is $3.47x10^{-4}$ m^3 per second for the main control room. The control room occupancy factor is assumed to be 1.0 for the first 24 hours, 0.6 from 1 to 4 days, and 0.4 for the remaining duration of the event.

←(DRN 04-704, R14)

→(DRN 04-704, R14)

15B.4 REFERENCES:

15B-1 ICRP, “Limits for Intakes of Radionuclides by Workers,” ICRP Publication 30, 1979.

15B-2 K.F. Eckerman et. al., “Limiting Values of Radionuclide Intake and Air Concentration and dose Conversion Factors for Inhalation, Submersion and Ingestion,” Federal Guidance Report 11, Environmental Protection Agency, 1988.

15B-3 K.F. Eckerman and J.C. Ryman, “External Exposure to Radionuclides in Air Water and Soil,” Federal Guidance Report 12, Environmental Protection Agency, 1993.

→(EC-5000081470, R301)

←(DRN 04-704, R14; EC-5000081470, R301)

TABLE 15B-1 Revision 14 (12/05)

ISOTOPE PROPERTIES

→(DRN 04-704, R14)

Isotope	Half-life (s)	$\bar{E}_\gamma$ (MeV/dis)	$\bar{E}_\beta$ (MeV/dis)
I-131	6.9466E+05	0.381	0.194
I-132	8.2800E+04	2.283	0.496
I-133	7.4880E+04	0.608	0.410
I-134	3.1560E+03	2.624	0.623
I-135	2.3796E+04	1.580	0.367
Kr-83m	6.5880E+03	0.002	0.037
Kr-85m	1.6127E+04	0.159	0.258
Kr-85	3.3830E+08	0.002	0.251
Kr-87	4.5780E+03	0.793	1.324
Kr-88	1.0224E+04	1.957	0.364
Xe-131m	1.0282E+06	0.020	0.143
Xe-133m	1.8904E+05	0.042	0.190
Xe-133	4.5317E+05	0.045	0.135
Xe-135m	9.1740E+02	0.432	0.098
Xe-135	3.2724E+04	0.247	0.316
Xe-138	1.0199E+03	1.128	0.672

←(DRN 04-704, R14)

TABLE 15B-2 Revision 14 (12/05)

ICRP30 ISOTOPE PROPERTIES

→(DRN 04-704, R14)

Isotope	Effective Cloudshine Sv-s/Bq-m^3	Effective Inhaled Sv-s/Bq-m^3	Isotope	Effective Cloudshine Sv-s/Bq-m^3	Effective Inhaled Sv-s/Bq-m^3
Co-58	4.760E-14	2.940E-09	Te-131m	7.463E-14	1.758E-09
Co-60	1.260E-13	5.910E-08	Te-132	1.030E-14	2.550E-09
Kr-85	1.190E-16	0.000E-00	I-131	1.820E-14	8.890E-09
Kr-85m	7.480E-15	0.000E-00	I-132	1.120E-13	1.030E-10
Kr-87	4.120E-14	0.000E-00	I-133	2.940E-14	1.580E-09
Kr-88	1.020E-13	0.000E-00	I-134	1.300E-13	3.550E-11
Rb-86	4.810E-15	1.790E-09	I-135	8.294E-14	3.320E-10
Sr-89	7.730E-17	1.120E-08	Xe-133	1.560E-15	0.000E-00
Sr-90	7.530E-18	3.510E-07	Xe-135	1.190E-14	0.000E-00
Sr-91	4.924E-14	4.547E-10	Cs-134	7.570E-14	1.250E-08
Sr-92	6.790E-14	2.180E-10	Cs-136	1.060E-13	1.980E-09
Y-90	1.900E-16	2.280E-09	Cs-137	2.725E-14	8.630E-09
Y-91	2.600E-16	1.320E-08	Ba-139	2.170E-15	4.640E-11
Y-92	1.300E-14	2.110E-10	Ba-140	8.580E-15	1.010E-09
Y-93	4.800E-15	5.820E-10	La-140	1.170E-13	1.310E-09
Zr-95	3.600E-14	6.390E-09	La-141	2.390E-15	1.570E-10
Zr-97	4.432E-14	1.171E-09	La-142	1.440E-13	6.840E-11
Nb-95	3.740E-14	1.570E-09	Ce-141	3.430E-15	2.420E-09
Mo-99	7.280E-15	1.070E-09	Ce-143	1.290E-14	9.160E-10
Tc-99m	5.890E-15	8.800E-12	Ce-144	2.773E-15	1.010E-07
Ru-103	2.251E-14	2.421E-09	Pr-143	2.100E-17	2.190E-09
Ru-105	3.810E-14	1.230E-10	Nd-147	6.190E-15	1.850E-09
Ru-106	1.040E-14	1.290E-07	Np-239	7.690E-15	6.780E-10
Rh-105	3.720E-15	2.580E-10	Pu-238	4.880E-18	7.790E-05
Sb-127	3.330E-14	1.630E-14	Pu-239	4.240E-18	8.330E-05
Sb-129	7.140E-14	1.740E-10	Pu-240	4.750E-18	8.330E-05
Te-127	2.420E-16	8.600E-11	Pu-241	7.250E-20	1.340E-06
Te-127m	1.470E-16	5.810E-09	Am-241	8.180E-16	1.200E-04
Te-129	2.750E-15	2.090E-11	Cm-242	5.690E-18	4.670E-06
Te-129m	3.337E-15	6.484E-09	Cm-244	4.910E-18	6.700E-05

←(DRN 04-704, R14)

TABLE 15B-3 Revision 14 (12/05)

→(DRN 04-704, R14)

EFFECTIVE DOSE EQUIVALENT WEIGHTING FACTORS (ICRP 30)

Organ/tissue	W_T
Gonads	0.25
Breast	0.15
Red Marrow	0.12
Lungs	0.12
Thyroid	0.03
Bone Surface	0.03
Remainder *	0.30

→(DRN 05-1551, R14)

* A value of WT = 0.06 is applicable to each of the five remaining organs or tissues (such as liver, kidneys, spleen, brain, small intestine, upper large intestine, lower large intestine, etc., but excluding skin, lens of the eye, and the extremities) receiving the highest doses

←(DRN 04-704, R14; 05-1551, R14)

TABLE 15B-4 Revision 14 (12/05)

→(DRN 04-704, R14)

POWERS 10% NATURAL DEPOSITION MODEL

Phase	Time Interval (s)	Correlation (hr^{-1})
gap	0 – 1800 (0 – 0.5 hrs)	$\lambda(10) = 0.0182 + 3.260x10^{-6} P(MWt)$
gap	1800 – 6480 (0.5 – 1.8 hrs)	$\lambda(10) = 0.0645\left[\ 1 - \exp\left(-0.938P(MWt)/1000\right)\ \right]$
early in-vessel	1800 – 6480 (0.5 – 1.8)	$\lambda(10) = 0.0326\left[\ 1 - \exp\left(-0.910P(MWt)/1000\right)\ \right]$
gap + early in-vessel	6480 – 13680 (1.8 – 3.8 hr)	$\lambda(10) = 0.094\left[\ 1 - \exp\left(-0.869P(MWt)/1000\right)\ \right]$
gap + early in-vessel	12680 – 49680 (3.8 - 13.8 hr)	$\lambda(10) = 0.0811 + 10.15x10^{-6} P(MWt)$
gap + early in-vessel	49680 – 80000 (13.8 – 22.22 hr)	$\lambda(10) = 0.094\left[\ 1 - \exp\left(-2.384P(MWt)/1000\right)\ \right]$

←(DRN 04-704, R14)