TECHNICAL EVALUATION REPORT
OF
PROPOSED SOIL AND CONCRETE SR-90 DCGLS
FOR
ST. ALBANS EXTENDED CARE CENTER, QUEENS, NEW YORK

ATTACHMENT

TECHNICAL EVALUATION REPORT

LICENSEE NAME: St. Albans Extended Care Center
LICENSE NUMBER: 31-02892-06 **DOCKET NUMBER:** 030-34751
FACILITY: VA Medical Center - Queens, New York
TYPE OF REQUEST EVALUATED: Approval of DCGLs for Strontium-90 in Soil and Concrete
DATE OF REQUEST: 2/14/00 **DATE EVALUATION COMPLETED:** Acceptance Review: 2/24/00, Final Review: 4/11/00
EVALUATED BY: Dominick Orlando, Richard Clement, (concrete), Mark Thaggard (soil)
SUPPORT: Rateb (Boby) Abu-Eid (resuspension factor), Ralph Cady (DandD Screen code)
RESPONSIBLE STAFF MEMBER: Dominick Orlando

SUMMARY:

Division of Waste Management staff (DWM) evaluated the licensee's request to use 35 pCi/g as the Derived Concentration Guideline Level (DCGL) for Strontium-90 Sr-90 in soil and concrete at its Queens New York facility. Based on the staff's evaluation of the licensee's request, the staff recommends that the DCGL for concrete not be approved until the licensee can adequately demonstrate to Region I that it is highly unlikely that an individual would occupy the Ejector Pit for more than about 1100 hours per year. In addition, the staff recommends that the DCGL for soil not be approved until the licensee has adequately justified either the use of the default distribution coefficients for Sr-90 used in its dose modeling or its use of the nondispersion modeling approach. The staff also recommends that Region I forward the licensee's justification for using the Sr-90 distribution coefficients or the nondispersion modeling approach to DWM for review.

SUPPLEMENTARY INFORMATION:

On February 14, 2000, staff of the Division of Industrial and Medical Nuclear Safety forwarded a Technical Assistance Request (TAR) from the Region I Division of Nuclear Materials Safety to DWM to review and approve the St Alban's Extended Care Center's DCGL of 35 picocuries per gram (pCi/g) of Strontium-90 (Sr-90) in soil and concrete that will remain at the facility at the completion of decommissioning.

To support this request, Region I staff provided the following information:

1. TAR dated February 2, 2000, summarizing the licensee's request and Region I's activities to validate the licensee's proposed DCGLs. Note that this TAR also included several ADAMS Accession Numbers for additional information supplied by the licensee to support the request;

2. Pages 3-3, 3-4, 5-7, and 5-8 of Volume I of the U.S. ARMY Corps of Engineers report entitled "Radiological Characterization Survey Report" describing the

concrete Ejector Pit and the method used to obtain concrete cores from the ejector pit floors and walls;

3. A letter from Michael T. Van Der Karr (Roy F. Weston, Inc.) to Randy Godfrey (U.S. Army, Corps of Engineers) dated March 3, 2000, summarizing the volumes of concrete that could remain if the proposed DCGLs were approved and core sampling results for the walls and floors of the Ejector Pit;

4. Pages 5-1 through 5-8 of Volume I of the U.S. ARMY Corps of Engineers report entitled "Radiological Characterization Survey Report" describing the radiological condition of the Ejector Pit;

5. Figure 5-4 from the report entitled "Draft Final Workplan - Volume 1 for the Radiological Characterization Survey of the St. Albans Veterans Administration Extended Care Center, Queens, New York and accompanying fax cover sheet discussing the dimensions of the Ejector Pit.

The DWM staff's evaluation of the request is summarized below.

CONCRETE

DWM staff evaluated the potential doses from residual radioactive material in the concrete at the proposed DCGL under building occupancy and concrete recycle/disposal scenarios. DWM staff believes that these scenarios encompass the most reasonable potential future uses of the concrete.

DWM staff evaluated the potential doses (Total Effective Dose Equivalent) to individuals occupying the Ejector Pit using Version 1 of the DandD Screen dose modeling code. The surface activity used in the code (18,648 dpm/100cm^2) was developed by assuming that the activity in the first cubic centimeter of concrete is at the proposed DCGL (35 pCi/g) and that all of this activity is present on the surface of the concrete. This is expressed in the following equation:

$$35 \text{ pCi/g} \times 2.4 \text{ g/cm}^3 \times 2.22 \text{ dpm/pCi} \times 1 \text{ cm} \times 100\text{cm}^2/(100\text{cm}^2) = 18{,}648 \text{ dpm/100cm}^2$$

Note that 2.4 g/cm^3 is the density of the concrete, based on information supplied by the licensee (density of 150 lbs/ft^3), as well as information in Draft NUREG 1640 "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities" and the Radiological Health Handbook (1984 edition).

Based on this evaluation, the potential dose from the residual radioactive material in the Ejector Pit is approximately 54 mrem/yr. However, approximately 51 mrem/yr of this dose is received via the inhalation pathway and requires that an individual is present in the Ejector Pit for approximately 2339 hours per year.

The staff evaluated the impact that modifying the resuspension factor had on the potential dose. The default resuspension factor used in the DandD Screen code (1.42E-5/m) has raised concerns from the staff in the past, as it is felt to be extremely conservative, and is based on a limited number of referenced values. Currently, the staff is re-evaluating the resuspension factor in the DandD Screen code and expects to revise the value in the future. Staff involved in this effort stated that the default resuspension factor could be reduced by as much as a factor of 20. DWM staff decreased the resuspension factor by a factor of 10 from 1.42E-5/m to 1.42E-6/m and determined that the potential dose from the residual radioactive material in the Ejector Pit to an individual occupying the Ejector Pit is approximately 8.5 mrem/yr.

The staff also determined that, as long as an individual did not occupy the Ejector Pit for more than approximately 1080 hours per year, (i.e. , 45, 24 hour periods) the dose to the individual would be less than 25 mrem/yr. Note that this estimate was developed using the default resuspension factor. Enclosure 1 contains the DandD Screen code printouts

DWM staff also modeled potential total effective dose equivalent (TEDE) rates to an individual occupying the Ejector Pit using the RESRAD-BUILD Version 2.37 computer code. The RESRAD-BUILD code is primarily intended for site specific analysis since it provides greater flexibility than the DandD Version 1.0 code used for screening purposes. However, the parameter values in the RESRAD-BUILD code have not been validated by NRC staff. Because the licensee did not provide justification for site specific values used in the code, NRC staff used the RESRAD-BUILD default parameters (with the exception of the indoor fraction parameter). Use of the default parameters would normally be expected to result in a conservative estimate of the TEDE rates.

The source term in the conceptual model considered secular equilibrium conditions (Sr-90+D) of the daughter progeny yttrium-90 (Y-90) with the parent Sr-90 of unit activity concentration. Enclosure 2A describes nuclear data for Sr-90 and Y-90. Calculated TEDE rates based on unity activity concentrations were scaled to 35 pCi/g. The model further assumed the following:

- Sr/Y-90 contamination is homogenous and uniformly distributed in concrete of 2.4 g/cm^3 density;
- total surface area of the contaminated walls is 117 ft^2 (10.9 m^2) (therefore each wall area represents 10.9 m^2 ÷ 4 = 2.7 m^2);
- total surface area of the contaminated floor is 200 ft^2 (18.6 m^2);
- depth of contamination in the walls and floor is 15 cm;
- room height is 6 ft (1.8 m);
- an individual occupies the center of the room with the receptor located at 1 meter above the floor; and,
- an exposure period of 2000 hours in one year.

The calculation was repeated for wall and floor thickness of 1 cm using the same parameter values and assumptions given in Enclosure 2B. Enclosures 2C and 2D include RESRAD-BUILD code printouts for 15 cm- and 1 cm-thick contaminated concrete. Using the RESBUILD-BUILD code, NRC staff calculated potential TEDE rates ranging from 0.15 mrem/y to 0.05 mrem/y, depending on the assumed thickness of the concrete (15 cm and 1 cm, respectively).

DWM staff also evaluated the potential dose to workers renovating the Ejector Pit using Version 1 of the DandD Screen code. Based on this evaluation, the potential doses to workers renovating the Ejector Pit is approximately 1.4 mrem/yr. Note that the use of the DandD Screen code for evaluating the renovation scenario has not been supported by the staff, as the parameters in the code have not been fully validated. In addition, in revising the DandD Screen code, staff has focused their efforts on the residential farmer and building occupancy scenarios, as these scenarios are believed to bound any potential doses from residual radioactivity at licensed facilities. As such, the DWM staff performed an evaluation of the potential doses under a building renovation scenario for reference only.

DWM staff also evaluated the potential doses from recycling or disposing of the concrete at the proposed DCGL using the dose conversion factors in Table 7.2 of Draft NUREG-1640 "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities ." At the 95 percent confidence interval, the dose conversion factor for Sr-90 is 2.8E-2 mrem/yr per pCi/g. Use of this factor yields a potential dose from recycling or disposing of the concrete of approximately 0.98 mrem/yr. Enclosure 3 contains Table 7.2 of Draft NUREG-1640 "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities ."

Based on the evaluations summarized above, it appears that recycling or disposing of the concrete does not result in a significant threat to the public health and safety. However, the potential dose to an individual occupying the Ejector Pit could be in excess of NRC's limits for unrestricted use, if the individual occupies the Ejector Pit as envisioned in the DandD Screen code (i.e., if the default parameters for the occupancy time and the resuspension factor are used).

As stated above the staff believes that these two factors are extremely conservative. In addition, the estimated doses to individuals occupying the Ejector Pit developed using the RESBUILD code indicate that potential doses are well below the NRC's criteria for decommissioning licensed nuclear facilities.

The staff believes that there are four options available to the NRC with respect to the use of the proposed DCGL for concrete:

- Staff could approve the proposed DCGL, because: a) it is unlikely that an individual would be present in the Ejector Pit for 2,337 hours per year; and b) the estimated dose using the RESRAD-BUILD code indicates that doses are well below the NRC's decommissioning criteria of 25 mrem/yr;

- Staff could deny the request because the estimated dose to an individual occupying the Ejector Pit may be in excess of the NRC's limit for unrestricted use, as determined using the DandD Screen code and all of the code's default parameters;

- Staff could delay a decision on the concrete until the staff's re-evaluation of the resuspension factor is completed and, assuming the revised resuspension factor is adopted, approve the request at that time. Note that staff involved in revising the resuspension factor cannot provide an estimate of when the re-evaluation of the resuspension factor will be completed; or,

- Staff could request additional information from the licensee to support a site-specific estimate of the potential doses from an individual occupying the Ejector Pit and base the final staff recommendation on the results of this evaluation.

RECOMMENDATION

Staff should only approve the DCGL for concrete if the licensee can adequately demonstrate to Region I that it is highly unlikely that an individual would occupy the Ejector Pit for more than about 1100 hours per year. Given the debate surrounding the resuspension factor in the DandD Screen code, the staff does not believe that it is appropriate to deny the licensee's request based on the estimated doses using the DandD Screen codes default parameters. However, there is no guarantee that the resuspension factor will be revised in the near future or in a manner such that the resulting estimate of the potential dose to an individual occupying the Ejector Pit would be acceptable. This could result in the staff requiring the licensee to remove the concrete at a later date. Relying on a demonstration that the occupancy time in the Ejector Pit is less than the default occupancy time in the DandD Screen code would permit the staff to approve the proposed DCGL using a more readily justifiable modification to the DandD Screen code. Relying on the reduced occupancy time in the Ejector Pit would also obviate any concerns that staff may have on the validity of the parameter values in the RESRAD-BUILD code.

It is important to note that the release of volumetrically contaminated material is a policy issue currently being debated within NRC and is the subject of several Commission papers and initiatives (e.g. clearance). Until the Commission decides on whether or not it is appropriate to allow the release of volumetrically contaminated material, it may not be appropriate to approve the DCGL for concrete.

SOIL

Staff evaluated the licensee's derivation of the DCGL for soil and concluded that it was generally acceptable, as long as the non-irrigated plant pathway is the dominant exposure pathway. However, the licensee used distribution coefficient (K_d) of 30 cm^3/g. The default K_d in RESRAD for Sr-90, for the contaminant, unsaturated, and saturated zones is 30 cm^3/g. The default K_d value (and higher values) is sufficient to allow the activity to be essentially retained in the contaminant zone. Without a cover over the contaminated soil (as was assumed by the licensee), the predominant exposure pathway remains the non-irrigated plant ingestion pathway. However, when a lower K_d value such as 3 cm^3/g is assumed for the three zones, the Sr-90 readily migrates to the ground water resulting in the drinking water pathway becoming the predominant pathway.

The licensee justified using the default K_d (i.e., 30 cm^3/g) because the actual K_d value is believed to be smaller based upon a measurement at another site, and the smaller K_d results in a lower dose estimate (i.e., higher DCGL). No information was provided on why the K_d value is considered representative (e.g., similar soil texture, pH and redox conditions, etc.) for the licensee's site. Use of the smaller K_d value of 3 cm^3/g results in a smaller dose (i.e., larger DCGL) only because the licensee used the nondispersion ground-water model in RESRAD. Although the nondispersion model is the default modeling approach, it requires that certain assumptions be made with regards to the hypothetical well location in relationship to the contamination. Generally, the nondispersion modeling approach results in a less conservative dose estimate than the alternative mass balance modeling approach. Because it is generally less conservative than the mass balance approach and because the mass balance approach is more consistent with the modeling approach used in developing the screening DCGLs, the nondispersion approach should not be used without justification.

Using the mass balance approach, the derived DCGL is greatly reduced. A key parameter is the fraction of water obtained onsite. As with the fraction of onsite plant food, it is probably not reasonable to assume that the hypothetical resident will, in the near future, get all of its water from an onsite well given that the site is located in an urban setting. Based upon staff calculations, the peak ground water concentration occurs roughly 15 years after the onset of the analysis. Although the contaminants have been present at the site for more than 25 years (i.e., assuming as reported that the licensed activities ended on December 31, 1973), significant leaching from the soils are not likely to occur until the building is removed. Enclosure 4 contains the staff's review of the licensee's derivation of the soil DCGLs.

RECOMMENDATION

Staff believes that the COE has developed a conservative DCGL for Sr-90 for cleaning up the soils at the VAECC site, assuming that the Sr-90 is retained in the soil. However, staff does not believe that the COE has sufficiently justified the assumption that the Sr-90 will be retained in the soils and use of a smaller distribution coefficient could result in a higher dose estimate (i.e., lower DCGL) with the use of the less conservative mass balance modeling approach. Accordingly, additional information should be provided to either support the use of the default distribution coefficients for Sr-90 or use of the nondispersion modeling approach.

Enclosures:

1. DandD Screen code results for building occupancy and renovation scenarios

2. Memo from Richard Clement to Nick Orlando summarizing the RESRAD-BUILT evaluations of the licensee's proposed DCGLs

3. Table 7.2 of Draft NUREG 1640 "Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities"

4. Report entitled "Review of Proposed Soil Sr-90 DCGL for St. Albans" by Mark Thaggard

ENCLOSURE 1 - DandD SCREEN CODE RESULTS FOR BUILDING OCCUPANCY AND RENOVATION SCENARIOS

Program : DandD Version 1.0 Build 1.00.02
Session : St albans with daughters
Description :
st albans with yttrium duaghters

Executed : 03/29/00 at 06:07:30

NRC Report

dose using default parameters

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
90Sr	18648.00
90Y	18648.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
90Sr	1.8648E+004
90Y	0.0000E+000
90Y	1.8648E+004

Variable Parameters

No parameters have been changed.

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 5.39E+001 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.45E-001	0.27
Inhalation	5.05E+001	93.60
Ingestion	3.30E+000	6.13
Total	5.39E+001	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
90Sr	5.32E+001	98.71
90Y	6.96E-001	1.29
Total	5.39E+001	100.00

Output From Program 'OCCUPA'
Building Occupancy Scenario
Run Date: 03/29/2000
Run Time: 06:07:26

INPUT DATA:
Title: St albans with daughters
Notes:
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

PARAMETER DATA:
Time In Building : 9.7460E+01 d/y
Occupancy Period : 3.6525E+02 d
Breathing Rate : 1.4000E+00 m**3/h
Resuspension Factor : 1.4200E-05 /m
Secondary Ingestion Transfer Rate: 1.1100E-05 m**2/h

TIME DATA:
Start Time : 0.0000E+00 d
End Time : 3.6525E+02 d
DT Size : 3.6525E+02 d
Time Step Size : 3.6525E+02 d
Write results every : 1 calculation times

INITIAL ACTIVITIES:

Number of chains: 2

Chain Number	Chain Name	Initial Activity dpm/ 100 cm^2
1	90Sr	1.8648E+04
2	90Y	1.8648E+04

Chain No. 1: 90Sr

Nuclide	Chain Position	Half Life (d)	Initial Inventory (dpm/ 100 cm^2)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv
Nuclides in chain : 2									
90Sr	1	1.0600E+04	1.8648E+04	0	.00000	0	.00000	3.85E-08	3.5
90Y	2	2.6700E+00	0.0000E+00	1	1.00000	0	.00000	2.91E-09	2.2

Chain No. 2: 90Y

Nuclide	Chain Position	Half Life (d)	Initial Inventory (dpm/ 100 cm^2)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv
Nuclides in chain : 1									
90Y	1	2.6700E+00	1.8648E+04	0	.00000	0	.00000	2.91E-09	2.2

TIMESTEP DATA:
Number of nuclides : 2
Number of time steps : 1
Number of print steps: 1
Timestep of maximum : 1
Day of maximum : .3652500000E+03
Year of maximum : .1000000000E+01

For Period 1: 0.0000000000E+00 days to 3.6525000000E+02 days
0.0000000000E+00 years to 1.0000000000E+00 years

Dose Components of Maximum TEDE

Nuclide External Inhalation Ingestion Total Average

	Dose (mrem/y)	Dose (mrem/y)	Dose (mrem/y)	Dose (mrem/y)	Activity (dpm/100 cm^2)

Dose Components

90Sr	7.36273E-03	5.01319E+01	3.07025E+00	5.32095E+01	1.84271E+04
90Y	1.37711E-01	3.25724E-01	2.32121E-01	6.95556E-01	1.84317E+04
Totals	1.45074E-01	5.04576E+01	3.30237E+00	5.39050E+01	

Component Maximums and Time of Occurrence

Type	Time Step	Maximum Value	Time of Occurrence (days)	(years)	
External	1	1.45074E-01	0.00000E+00	0.00000E+00	
Inhalation	1	5.04576E+01	0.00000E+00	0.00000E+00	
Ingestion	1	3.30237E+00	0.00000E+00	0.00000E+00	
TEDE	1	5.39050E+01	0.00000E+00	0.00000E+00	
90Sr	1	7.36273E-03	0.00000E+00	0.00000E+00	Maximum External
90Sr	1	5.01319E+01	0.00000E+00	0.00000E+00	Maximum Inhalation
90Sr	1	3.07025E+00	0.00000E+00	0.00000E+00	Maximum Ingestion
90Sr	1	5.32095E+01	0.00000E+00	0.00000E+00	Maximum Nuclide
90Y	1	1.37711E-01	0.00000E+00	0.00000E+00	Maximum External
90Y	1	3.25724E-01	0.00000E+00	0.00000E+00	Maximum Inhalation
90Y	1	2.32121E-01	0.00000E+00	0.00000E+00	Maximum Ingestion
90Y	1	6.95556E-01	0.00000E+00	0.00000E+00	Maximum Nuclide

Program   : DandD Version 1.0 Build 1.00.02
Session   : St albans with daughters
Description :
    st albans with yttrium duaghters

Executed   : 03/29/00 at 06:21:27

NRC Report

dose with reduced resuspension factor

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
90Sr	18648.00
90Y	18648.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
90Sr	1.8648E+004
90Y	0.0000E+000
90Y	1.8648E+004

Basic Parameters

Name	Value	Units	Default
'Resuspension Factor'	1.4200E-006	m-1	1.4200E-005

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 8.49E+000 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.45E-001	1.71
Inhalation	5.05E+000	59.41
Ingestion	3.30E+000	38.88

Total	8.49E+000	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
90Sr	8.09E+000	95.26
90Y	4.02E-001	4.74
Total	8.49E+000	100.00

Output From Program 'OCCUPA'
Building Occupancy Scenario
Run Date: 03/29/2000
Run Time: 06:21:24

INPUT DATA:
Title: St albans with daughters
Notes:
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

PARAMETER DATA:

Time In Building	: 9.7460E+01 d/y
Occupancy Period	: 3.6525E+02 d
Breathing Rate	: 1.4000E+00 m**3/h
Resuspension Factor	: 1.4200E-06 /m
Secondary Ingestion Transfer Rate	: 1.1100E-05 m**2/h

TIME DATA:

Start Time	: 0.0000E+00 d
End Time	: 3.6525E+02 d
DT Size	: 3.6525E+02 d
Time Step Size	: 3.6525E+02 d
Write results every	: 1 calculation times

INITIAL ACTIVITIES:

Number of chains: 2

Chain Number	Chain Name	Initial Activity dpm/ 100 cm^2
1	90Sr	1.8648E+04
2	90Y	1.8648E+04

Chain No. 1: 90Sr

Nuclide	Chain Position	Half Life (d)	Initial Inventory (dpm/ 100 cm^2)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv
Nuclides in chain : 2									
90Sr	1	1.0600E+04	1.8648E+04	0	.00000	0	.00000	3.85E-08	3.5
90Y	2	2.6700E+00	0.0000E+00	1	1.00000	0	.00000	2.91E-09	2.2

Chain No. 2: 90Y

Nuclide	Chain Position	Half Life (d)	Initial Inventory (dpm/ 100 cm^2)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv
Nuclides in chain : 1									
90Y	1	2.6700E+00	1.8648E+04	0	.00000	0	.00000	2.91E-09	2.2

TIMESTEP DATA:

Number of nuclides	: 2
Number of time steps	: 1
Number of print steps	: 1
Timestep of maximum	: 1
Day of maximum	: .3652500000E+03
Year of maximum	: .1000000000E+01

For Period 1: 0.0000000000E+00 days to 3.6525000000E+02 days
0.0000000000E+00 years to 1.0000000000E+00 years

Dose Components of Maximum TEDE

Nuclide External Inhalation Ingestion Total Average

	Dose (mrem/y)	Dose (mrem/y)	Dose (mrem/y)	Dose (mrem/y)	Activity (dpm/100 cm^2)

Dose Components

90Sr	7.36273E-03	5.01319E+00	3.07025E+00	8.09080E+00	1.84271E+04
90Y	1.37711E-01	3.25724E-02	2.32121E-01	4.02405E-01	1.84317E+04
Totals	1.45074E-01	5.04576E+00	3.30237E+00	8.49321E+00	

Component Maximums and Time of Occurrence

Type	Time Step	Maximum Value	Time of Occurrence (days)	(years)	
External	1	1.45074E-01	0.00000E+00	0.00000E+00	
Inhalation	1	5.04576E+00	0.00000E+00	0.00000E+00	
Ingestion	1	3.30237E+00	0.00000E+00	0.00000E+00	
TEDE	1	8.49321E+00	0.00000E+00	0.00000E+00	
90Sr	1	7.36273E-03	0.00000E+00	0.00000E+00	Maximum External
90Sr	1	5.01319E+00	0.00000E+00	0.00000E+00	Maximum Inhalation
90Sr	1	3.07025E+00	0.00000E+00	0.00000E+00	Maximum Ingestion
90Sr	1	8.09080E+00	0.00000E+00	0.00000E+00	Maximum Nuclide
90Y	1	1.37711E-01	0.00000E+00	0.00000E+00	Maximum External
90Y	1	3.25724E-02	0.00000E+00	0.00000E+00	Maximum Inhalation
90Y	1	2.32121E-01	0.00000E+00	0.00000E+00	Maximum Ingestion
90Y	1	4.02405E-01	0.00000E+00	0.00000E+00	Maximum Nuclide

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Program      : DandD Version 1.0 Build 1.00.02
Session      : St albans with daughters
Description :
     st albans with yttrium duaghters

Executed     : 03/29/00 at 06:52:19
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NRC Report

dose at 45 days

Occupancy Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	dpm/100cm^2
90Sr	18648.00
90Y	18648.00

Code-Generated Radionuclide Activities

Chain	dpm/100cm^2
90Sr	1.8648E+004
90Y	0.0000E+000
90Y	1.8648E+004

Basic Parameters

Name	Value	Units	Default
'Time In Building'	45.0000	days/year	97.4600

Occupancy Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 2.49E+001 TEDE (mrem) occurred 1.00 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	6.70E-002	0.27
Inhalation	2.33E+001	93.60
Ingestion	1.52E+000	6.13

Total	2.49E+001	100.00

Radionuclide Component of
Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
90Sr	2.46E+001	98.71
90Y	3.21E-001	1.29
Total	2.49E+001	100.00

Output From Program 'OCCUPA'
Building Occupancy Scenario
Run Date: 03/29/2000
Run Time: 06:23:13

INPUT DATA:
Title: St albans with daughters
Notes:
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

PARAMETER DATA:

Time In Building	: 4.5000E+01 d/y
Occupancy Period	: 3.6525E+02 d
Breathing Rate	: 1.4000E+00 m**3/h
Resuspension Factor	: 1.4200E-05 /m
Secondary Ingestion Transfer Rate	: 1.1100E-05 m**2/h

TIME DATA:

Start Time	: 0.0000E+00 d
End Time	: 3.6525E+02 d
DT Size	: 3.6525E+02 d
Time Step Size	: 3.6525E+02 d
Write results every	: 1 calculation times

INITIAL ACTIVITIES:

Number of chains: 2

Chain Number	Chain Name	Initial Activity dpm/ 100 cm^2
1	90Sr	1.8648E+04
2	90Y	1.8648E+04

Chain No. 1: 90Sr

Nuclides in chain : 2

Nuclide	Chain Position	Half Life (d)	Initial Inventory (dpm/ 100 cm^2)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv
90Sr	1	1.0600E+04	1.8648E+04	0	.00000	0	.00000	3.85E-08	3.5
90Y	2	2.6700E+00	0.0000E+00	1	1.00000	0	.00000	2.91E-09	2.2

Chain No. 2: 90Y

Nuclides in chain : 1

Nuclide	Chain Position	Half Life (d)	Initial Inventory (dpm/ 100 cm^2)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv
90Y	1	2.6700E+00	1.8648E+04	0	.00000	0	.00000	2.91E-09	2.2

TIMESTEP DATA:

Number of nuclides	: 2
Number of time steps	: 1
Number of print steps	: 1
Timestep of maximum	: 1
Day of maximum	: .3652500000E+03
Year of maximum	: .1000000000E+01

For Period 1: 0.0000000000E+00 days to 3.6525000000E+02 days
0.0000000000E+00 years to 1.0000000000E+00 years

Dose Components of Maximum TEDE

Nuclide	External	Inhalation	Ingestion	Total	Average

	Dose (mrem/y)	Dose (mrem/y)	Dose (mrem/y)	Dose (mrem/y)	Activity (dpm/100 cm^2)

Dose Components

90Sr	3.39958E-03	2.31473E+01	1.41762E+00	2.45683E+01	1.84271E+04
90Y	6.35852E-02	1.50396E-01	1.07177E-01	3.21158E-01	1.84317E+04
Totals	6.69847E-02	2.32977E+01	1.52480E+00	2.48895E+01	

Component Maximums and Time of Occurrence

Type	Time Step	Maximum Value	Time of Occurrence (days)	(years)	
External	1	6.69847E-02	0.00000E+00	0.00000E+00	
Inhalation	1	2.32977E+01	0.00000E+00	0.00000E+00	
Ingestion	1	1.52480E+00	0.00000E+00	0.00000E+00	
TEDE	1	2.48895E+01	0.00000E+00	0.00000E+00	
90Sr	1	3.39958E-03	0.00000E+00	0.00000E+00	Maximum External
90Sr	1	2.31473E+01	0.00000E+00	0.00000E+00	Maximum Inhalation
90Sr	1	1.41762E+00	0.00000E+00	0.00000E+00	Maximum Ingestion
90Sr	1	2.45683E+01	0.00000E+00	0.00000E+00	Maximum Nuclide
90Y	1	6.35852E-02	0.00000E+00	0.00000E+00	Maximum External
90Y	1	1.50396E-01	0.00000E+00	0.00000E+00	Maximum Inhalation
90Y	1	1.07177E-01	0.00000E+00	0.00000E+00	Maximum Ingestion
90Y	1	3.21158E-01	0.00000E+00	0.00000E+00	Maximum Nuclide

```
Program      : DandD Version 1.0 Build 1.00.02
Session      : St albans with daughters
Description  :
     st albans with yttrium duaghters

Executed     : 03/29/00 at 06:15:37
```

NRC Report

Renovation Input Section

Execution Options

History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

Initial Radionuclide Activities

Chain	pCi/gram
90Sr	35.00
90Y	35.00

Code-Generated Radionuclide Activities

Chain	pCi/gram
90Sr	3.5000E+001
90Y	0.0000E+000
90Y	3.5000E+001

Variable Parameters

No parameters have been changed.

Renovation Output Section

Maximum Annual TEDE

This scenario started 0.00 year(s) from now
and ran for 1.00 year(s).

The peak dose of 1.35E+000 TEDE (mrem) occurred 0.25 year(s) after
license termination.

Pathway Component of Maximum Annual Dose

Pathway	TEDE (mrem)	Percentage
External	1.40E-001	10.34
Inhalation	7.93E-001	58.70
Ingestion	4.18E-001	30.96
Total	1.35E+000	100.00

Radionuclide Component of Maximum Annual Dose

Radionuclide	TEDE (mrem)	Percentage
90Sr	1.18E+000	87.42
90Y	1.70E-001	12.58
Total	1.35E+000	100.00

Output From Program 'RENOVA'
Buiilding Renovation Scenario
Run Date: 03/29/2000
Run Time: 06:15:33
INPUT DATA:
Title: St albans with daughters
Notes:
History file will be generated.
Implicit progeny doses will not be included with explicit parent.
Concentration data will be calculated.

PARAMETER DATA:

```
Time on Job               : 6.2830E+01 d/y
Renovation Period         : 1.7900E+02 d
Breathing Rate            : 1.2700E+00 m**3/h
Air Dust Loading          : 9.1000E-03 g/m**3
Loose Dust Transfer Rate: 5.2000E-02 g/h
TIME DATA:
Start Time                : 0.0000E+00 d
End Time                  : 3.6525E+02 d
DT Size                   : 9.1313E+01 d
Time Step Size            : 1.7900E+02 d
Write results every       :      1 calculation times
```

INITIAL ACTIVITIES:

Number of chains: 2

Chain Number	Chain Name	Initial Activity (pCi/g)
1	90Sr	3.5000E+01
2	90Y	3.5000E+01

Chain No. 1: 90Sr

Nuclide	Chain Position	Half Life (d)	Initial Inventory (pCi/g)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv/

Nuclides in chain : 2

90Sr	1	1.0600E+04	3.5000E+01	0	.00000	0	.00000	3.85E-08	3.5
90Y	2	2.6700E+00	0.0000E+00	1	1.00000	0	.00000	2.91E-09	2.2

Chain No. 2: 90Y

Nuclide	Chain Position	Half Life (d)	Initial Inventory (pCi/g)	First Parent	Fractional Yield	Second Parent	Fractional Yield	Ingestion CEDE Factor (Sv/Bq)	Inhal CEDE F (Sv/

Nuclides in chain : 1

90Y	1	2.6700E+00	3.5000E+01	0	.00000	0	.00000	2.91E-09	2.2

TIMESTEP DATA:

```
Number of nuclides    :          2
Number of time steps :          4
Number of print steps:          4
Timestep of maximum   :          1
Day of maximum        :   .1790000000E+03
Year of maximum       :   .4900752909E+00
```

For Period 1: 0.0000000000E+00 days to 1.7900000000E+02 days
0.0000000000E+00 years to 4.9007529090E-01 years

Dose Components of Maximum TEDE

Nuclide	External Dose (mrem/y)	Inhalation Dose (mrem/y)	Ingestion Dose (mrem/y)	Total Dose (mrem/y)	Average Activity (pCi/g)

Dose Components

90Sr	4.21910E-03	7.87519E-01	3.88663E-01	1.18040E+00	3.47960E+01
90Y	1.35413E-01	5.11677E-03	2.93841E-02	1.69913E-01	3.48045E+01
Totals	1.39632E-01	7.92636E-01	4.18047E-01	1.35031E+00	

Component Maximums and Time of Occurrence

Type	Time Step	Maximum Value	Time of Occurrence (days)	(years)	
External	1	1.39632E-01	0.00000E+00	0.00000E+00	
Inhalation	1	7.92636E-01	0.00000E+00	0.00000E+00	
Ingestion	1	4.18047E-01	0.00000E+00	0.00000E+00	
TEDE	1	1.35031E+00	0.00000E+00	0.00000E+00	
90Sr	1	4.21910E-03	0.00000E+00	0.00000E+00	Maximum External
90Sr	1	7.87519E-01	0.00000E+00	0.00000E+00	Maximum Inhalation
90Sr	1	3.88663E-01	0.00000E+00	0.00000E+00	Maximum Ingestion
90Sr	1	1.18040E+00	0.00000E+00	0.00000E+00	Maximum Nuclide
90Y	1	1.35413E-01	0.00000E+00	0.00000E+00	Maximum External
90Y	1	5.11677E-03	0.00000E+00	0.00000E+00	Maximum Inhalation
90Y	1	2.93841E-02	0.00000E+00	0.00000E+00	Maximum Ingestion
90Y	1	1.69913E-01	0.00000E+00	0.00000E+00	Maximum Nuclide

ENCLOSURE 2- MEMO FROM RICHARD CLEMENTS TO NICK ORLANDO SUMMARIZING THE RESRAD-BUILT EVALUATIONS OF THE LICENSEE'S PROPOSED DCGLS

NOTE TO: Nick Orlando
FROM: Rich Clement
SUBJECT: RESRAD-BUILD CALCULATIONS FOR SR-90 CONTAMINATED CONCRETE
DATE: April 4, 2000

Please see the attached evaluation and supporting RESRAD-BUILD calculations for Sr-90 contaminated concrete, as a feeder to your Technical Assistance Request for Region I, in regards to the St. Albans Extended Care Center, Queens, NY. If you should have any questions, please see me.

CC: Bob Nelson

DCB staff also modeled potential total effective dose equivalent (TEDE) rates to an individual occupying the Ejector Pit (concrete room), composed of four walls and a floor (no ceiling), with residual stronium-90 (Sr-90) contamination at the proposed DCGL of 35 pCi/g using the RESRAD-BUILD Version 2.37 computer code. The RESRAD-BUILD code (no users manual for this version is available) models both building occupation and remediation scenarios. RESRAD-BUILD is primarily intended for site specific analysis since it provides greater flexibility than the DandD Version 1.0 code used for screening purposes. However, the parameter values in the RESRAD-BUILD code have not been validated by NRC staff. Because the licensee did not provide justification for site specific values used in the code, NRC staff used the RESRAD-BUILD default parameters (with the exception of the indoor fraction parameter). Use of the default parameters would normally be expected to result in a conservative estimate of the TEDE rates. Additionally, NRC staff are planning to evaluate a probabilistic beta version of RESRAD-BUILD in the near future.

The source term in the conceptual model considered secular equilibrium conditions (Sr-90+D) of the daughter progeny yttrium-90 (Y-90) with the parent Sr-90 of unit activity concentration. Attachment 2A describes nuclear data for Sr-90 and Y-90. Calculated TEDE rates based on unity activity concentrations were scaled to 35 pCi/g. The model further assumed the following: Sr/Y-90 contamination is homogenous and uniformly distributed in concrete of 2.4 g/cm^3 density, total surface area of the contaminated walls is 117 ft^2 (10.9 m^2) (therefore each wall area represents 10.9 m^2 $\div$ 4 = 2.7 m^2), total surface area of the contaminated floor is 200 ft^2 (18.6 m^2), depth of contamination in the walls and floor is 15 cm, room height is 6 ft (1.8 m), an individual occupies the center of the room with the receptor located at 1 meter above the floor, and an exposure period of 2000 hours in one year. The calculation was repeated for wall and floor thickness of 1 cm using the same parameter values and assumptions given in Attachment 2B. Attachments 2C and 2D include RESRAD-BUILD code printouts for 15 cm- and 1 cm-thick contaminated concrete. Using the RESRAD-BUILD code, NRC staff calculated potential TEDE rates ranging from 0.15 mrem/y to 0.05 mrem/y, depending on the assumed thickness of the concrete (15 cm and 1 cm, respectively).

A calculation was also performed using an empirical equation[1] to describe the energy-range relationship for an electron (beta particle) in a low-Z (atomic number) material/absorber (i.e., concrete, water, aluminum, Lucite, air). The calculation assumed the most energetic beta energy of 2.28 MeV emitted by Y-90:

$$R_\beta = 0.412\ T_{\beta,max}^{\ 1.27 - 0.0954 \ln T_{\beta,max}} \quad (\text{for } 0.01 \le T_{\beta,max} \le 2.5 \text{ MeV})\ ,$$

where R_β (g/cm^2) is the range of the beta particle in the material/absorber; and $T_{\beta,max}$ (MeV) is the maximum energy of the beta particle. The distance d (cm) that the beta particle travels, or estimated amount of shielding required, in the material/absorber of density ρ (g/cm^3) is given as

$$d = R_\beta\, \rho^{-1}\ .$$

For a maximum beta energy $T_{\beta,max}$ = 2.28 MeV, and concrete density ρ_{conc} = 2.4 g/cm^3, the range R_β is calculated as 1.1 g/cm^2. The distance d_{conc} that the Y-90 beta particle travels in concrete is 1.1 cm^2 $\div$ 2.4 g/cm^3 = 0.5 cm. Therefore, concrete provides an effective shield material/absorber to attenuate beta particles emitted by Sr/Y-90, and also indicates that concrete contaminated with Sr/Y-90 presents more of a potential dose from the internal (inhalation) exposure pathway than the external (direct) exposure pathway. In comparison, the distance that a Y-90 beta particle travels in air was determined. Using R_β = 1.1 g/cm^2, determined from above, and the density of air ρ_{air} = 1.3E-3 g/cm^3, the estimated distance d_{air} that the Y-90 beta particle travels is: 1.1 cm^2 ÷ 1.3E-3 g/cm^3 = 846 cm or about 28 ft. As a rule of thumb (ROT), a beta particle travels about 12 feet in air per MeV of energy emitted. Using the ROT, a Y-90 beta particle travels: 2.28 MeV x 12 ft/MeV = 27 ft, which is in good agreement with the empirical calculation.

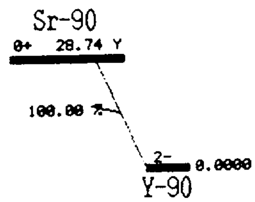
---

[1]Turner, J.E. Atoms, Radiation, and Radiation Protection. New York: Wiley-Interscience, 1995.

Attachment 2A

90SR B- DECAY

Parent state: G.S.
Half life: 28.74 Y(4)
Q(gs): 546.0(14) keV
Branch ratio: 1.0

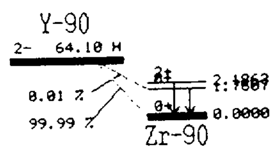


Beta ray:

Max.E(keV)	Avg.E(keV)	Intensity(rel)
546.0(16)	195.8(8)	100

90Y B- DECAY (64.10 H)

Parent state: G.S.
Half life: 64.10 H(8)
Q(gs): 2280.1(16) keV
Branch ratio: 1.0



Beta ray:

Max.E(keV)	Avg.E(keV)	Intensity(rel)
2280.1(-)	933.7(12)	99.9885(14)
519.4(-)	185.6(10)	0.0115(14)
93.8(-)	25.0(7)	1.4E-6(3)

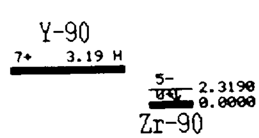
Gamma ray:

Energy(keV)	Intensity(rel)
1760.70(20)	-
2186.242(25)	1.4E-6(3)

E. BROWNE, *Nuclear Data Sheet* 82, 379 (1997)

90Y B- DECAY (3.19 H)

Parent state: 682.04(2) keV
Half life: 3.19 H(1)
Q(gs): 2280.1(16) keV
Branch ratio: 0.000018(2)



Beta ray: for absolute intensity multiply by 5.555E-4

Max.E(keV)	Avg.E(keV)	Intensity(rel)
643.2(-)	232.5(10)	0.0018(2)

Gamma ray:

Energy(keV)	Intensity(rel)
2318.968(10)	100

RESRAD-BUILD VERSION 2.37 CODE PARAMETER VALUES AND ASSUMPTIONS

- Sr/Y-90 concentration (assumed to be in secular equilibrium, homogenous and uniformly distributed in concrete) = 35 pCi/g
- Total contaminated concrete floor area = 200 ft^2 (18.6 m^2)
- Total contaminated concrete wall area = 117 ft^2 (10.9 m^2) (four wall areas assumed to be 2.7 m^2 each)
- Depth of contamination in concrete walls and floor = 15 cm, 1 cm
- Room height (no ceiling) = 6 ft (1.8 m)
- Receptor = 1 m above floor surface
- Exposure duration = 365 days (default parameter)
- Indoor fraction = 0.23 (changed from default parameter of 0.5)
- Calculation time = 1 year (default parameter)
- Building exchange rate = 0.8 /h (default parameter)
- Deposition velocity = 1E-2 m/s (default parameter)
- Resuspension rate = 5E-7 /s (default parameter)
- Breathing rate = 18 m^3/d (default parameter)
- Ingestion rate = 1E-4 m^2/d (default parameter)
- Direct ingestion = 0.1 g/d (default parameter)
- Concrete density = 2.4 g/cm^3 (default parameter)
- Erosion rate = 2.4E-8 cm/d (default parameter)
- Air fraction = 0.1 (default parameter)
- Internal (ingestion, inhalation) and external (surface, volume, air submersion) dose conversion factors from FGR 11 (EPA, 1988) and FGR 12 (EPA, 1993)

NOTE: Parameter values in RESRAD-BUILD Version 2.37 code have not been validated by NRC. Additionally, site-specific values were not provided by the licensee.

RESRAD-BUILD Program Output, Version 2.36 03/28/00 18:51 Page: 0- 0 : 1 **
tle : Sr-90 Dose Rate from Concrete
put File : C:\WINBLD\SR90A.INP

RESRAD-BUILD Table of Contents

RESRAD-BUILD Table of Contents

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP

RESRAD-BUILD Input Parameters

Number of Sources : 5
Number of Receptors: 1
Total Time : 3.650000E+02 days
Fraction Inside : 2.300000E-01

Receptor Information

Receptor	Room	x [m]	y [m]	z [m]	FracTime	Inhalation [m3/day]	Ingestion(Dust) [m2/hr]
1	1	2.400	2.400	1.000	1.000	1.80E+01	1.00E-04

Receptor-Source Shielding Relationship

Receptor	Source	Density [g/cm3]	Thickness [cm]	Material
1	1	2.40E+00	0.00E+00	Concrete
1	2	2.40E+00	0.00E+00	Concrete
1	3	2.40E+00	0.00E+00	Concrete
1	4	2.40E+00	0.00E+00	Concrete
1	5	2.40E+00	0.00E+00	Concrete

Building Information

Building Air Exchange Rate: 8.00E-01 1/hr

```
 Height[m]           Air Exchanges [m3/hr]
 Area [m2]
                    ******************************
                    *                            *
                    *                            *
                    *                          <=Q01: 2.68E+01
 H1:    1.800       *           Room  1          *       Q10 :  2.68E+01
                    *    LAMBDA: 8.00E-01        *
Area   18.600       *                            *
                    *                            *
                    ******************************
```

Deposition velocity: 1.00E-02 [m/s]   Resuspension Rate: 5.00E-06 [1/s]

```
Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP
```

======== Source Information ========

```
Source:  1
      Location::  Room :  1  x:   2.40 y:   2.40  z:   0.00[m]
      Geometry::  Type: Volume     Area:1.86E+01 [m2]  Direction: z
      Pathway ::
           Direct Ingestion Rate:     0.000E+00 [gm/hr]
           Fraction released to air: 1.000E-01


      Containment ::   Number of Regions:  1 Contaminated Region:  1
   Region                :    1
   Thickness [cm]        :1.50E+01
   Density [g/cm3]       :2.40E+00
   Material              :Concrete
   Erosion Rate [cm/day] :2.40E-08
```

Contamination::

Nuclide	Concentration [pCi/g]	Ingestion [mrem/pCi]	Inhalation [mrem/pCi]	External (Surface) [mrem/yr/(pCi/m2)]	External (Volume) [mrem/yr/(pCi/m3)]	Submersion [mrem/yr/(pCi/m3)]
		Dose Conversion Factors				
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP

Source: 2
Location:: Room : 1 x: 4.80 y: 2.40 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: x
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.50E+01
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP

Source: 3
Location:: Room : 1 x: 0.00 y: 2.40 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: x
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.50E+01
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP

Source: 4
Location:: Room : 1 x: 2.40 y: 0.00 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: y
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.50E+01
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP

Source: 5
Location:: Room : 1 x: 2.40 y: 4.80 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: y
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.50E+01
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 0.000000 years

Assessment for Time: 1
Time =0.00E+00 yr

Source Information

```
Source:  1
      Location::  Room :  1  x:   2.40 y:   2.40  z:   0.00 [m]
      Geometry::  Type: Volume     Area:1.86E+01 [m2]  Direction: z
      Pathway ::
           Direct Ingestion Rate    :0.000E+00 [gm/hr]
           Fraction released to air: 1.000E-01

      Containment ::   Number of Regions:  1 Contaminated Region:  1
   Region                  :    1
   Thickness [cm]          :1.50E+01
   Fraction Contaminated   :1.00E+00
   Density [g/cm3]         :2.40E+00

      Contamination::     Nuclide    Concentration
                                       [pCi/g]
                           SR-90      1.000E+00
```

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 0.000000 years

```
Source:  2
        Location::  Room :  1  x:   4.80 y:   2.40  z:   0.00 [m]
        Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: x
        Pathway ::
             Direct Ingestion Rate   :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01

        Containment ::  Number of Regions:  1 Contaminated Region:  1
    Region                  :    1
    Thickness [cm]          :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]         :2.40E+00

        Contamination::      Nuclide    Concentration
                                          [pCi/g]
                              SR-90      1.000E+00

Source:  3
        Location::  Room :  1  x:   0.00 y:   2.40  z:   0.00 [m]
        Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: x
        Pathway ::
             Direct Ingestion Rate   :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01

        Containment ::  Number of Regions:  1 Contaminated Region:  1
    Region                  :    1
    Thickness [cm]          :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]         :2.40E+00

        Contamination::      Nuclide    Concentration
                                          [pCi/g]
                              SR-90      1.000E+00
```

```
Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time:    0.000000     years


 Source:  4
        Location::  Room :  1  x:   2.40 y:   0.00  z:   0.00 [m]
        Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: y
        Pathway ::
             Direct Ingestion Rate   :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01


        Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


        Contamination::     Nuclide   Concentration
                                        [pCi/g]
                             SR-90     1.000E+00

 Source:  5
        Location::  Room :  1  x:   2.40 y:   4.80  z:   0.00 [m]
        Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: y
        Pathway ::
             Direct Ingestion Rate   :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01


        Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


        Contamination::     Nuclide   Concentration
                                        [pCi/g]
                             SR-90     1.000E+00
```

RESRAD-BUILD Dose Tables

Source Contributions to Receptor Doses

[mrem]

	Source 1	Source 2	Source 3	Source 4	Source 5	Total
Receptor 1	3.3E-03	2.5E-04	2.5E-04	2.5E-04	2.5E-04	4.3E-03
Total	3.3E-03	2.5E-04	2.5E-04	2.5E-04	2.5E-04	4.3E-03

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 0.000000 years

Pathway Detail of Doses

[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	3.33E-03	2.90E-10	8.83E-12	3.29E-06	0.00E+00	1.02E-07
Total	3.33E-03	2.90E-10	8.83E-12	3.29E-06	0.00E+00	1.02E-07

Source: 2

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Source: 3

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Source: 4

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Source: 5

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	2.46E-04	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 0.000000 years

Nuclide Detail of Doses

[mrem]

Source: 1

Nuclide	Receptor 1	Total
SR-90		
SR-90	3.33E-03	3.33E-03

Source: 2

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.46E-04	2.46E-04

Source: 3

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.46E-04	2.46E-04

Source: 4

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.46E-04	2.46E-04

Source: 5

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.46E-04	2.46E-04

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 1.00000 years

Assessment for Time: 2
Time =1.00E+00 yr

Source Information

Source: 1
Location:: Room : 1 x: 2.40 y: 2.40 z: 0.00 [m]
Geometry:: Type: Volume Area:1.86E+01 [m2] Direction: z
Pathway ::
Direct Ingestion Rate :0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.50E+01
Fraction Contaminated	:1.00E+00
Density [g/cm3]	:2.40E+00

Contamination::	Nuclide	Concentration [pCi/g]
	SR-90	9.765E-01

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 1.00000 years

```
 Source:  2
       Location::  Room :  1  x:   4.80 y:   2.40  z:   0.00 [m]
       Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: x
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01


       Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


       Contamination::      Nuclide    Concentration
                                         [pCi/g]
                             SR-90      9.765E-01

 Source:  3
       Location::  Room :  1  x:   0.00 y:   2.40  z:   0.00 [m]
       Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: x
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01


       Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


       Contamination::      Nuclide    Concentration
                                         [pCi/g]
                             SR-90      9.765E-01
```

```
 Source:  4
        Location::  Room :  1  x:    2.40 y:    0.00  z:    0.00 [m]
        Geometry::  Type: Volume      Area:2.70E+00 [m2]  Direction: y
        Pathway ::
             Direct Ingestion Rate    :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01


        Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


        Contamination::      Nuclide     Concentration
                                            [pCi/g]
                              SR-90        9.765E-01

 Source:  5
        Location::  Room :  1  x:    2.40 y:    4.80  z:    0.00 [m]
        Geometry::  Type: Volume      Area:2.70E+00 [m2]  Direction: y
        Pathway ::
             Direct Ingestion Rate    :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01


        Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.50E+01
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


        Contamination::      Nuclide     Concentration
                                            [pCi/g]
                              SR-90        9.765E-01
```

RESRAD-BUILD Dose Tables

Source Contributions to Receptor Doses

[mrem]

	Source 1	Source 2	Source 3	Source 4	Source 5	Total
Receptor 1	3.3E-03	2.4E-04	2.4E-04	2.4E-04	2.4E-04	4.2E-03
Total	3.3E-03	2.4E-04	2.4E-04	2.4E-04	2.4E-04	4.2E-03

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90A.INP Evaluation Time: 1.00000 years

Pathway Detail of Doses

[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	3.25E-03	2.83E-10	8.62E-12	3.21E-06	0.00E+00	9.99E-08
Total	3.25E-03	2.83E-10	8.62E-12	3.21E-06	0.00E+00	9.99E-08

Source: 2

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Source: 3

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Source: 4

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Source: 5

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	2.40E-04	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Nuclide Detail of Doses

[mrem]

Source: 1

Nuclide	Receptor 1	Total
SR-90		
SR-90	3.25E-03	3.25E-03

Source: 2

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.41E-04	2.41E-04

Source: 3

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.41E-04	2.41E-04

Source: 4

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.41E-04	2.41E-04

Source: 5

Nuclide	Receptor 1	Total
SR-90		
SR-90	2.41E-04	2.41E-04

RESRAD-BUILD Dose (Time) Tables

Receptor Doses Received for the Exposure Duration

(mrem)

Evaluation Time [yr]

	0.00E+00	1.00E+00
1	4.32E-03	4.22E-03

Receptor Dose/Yr Averaged Over Exposure Duration

(mrem/yr)

Evaluation Time [yr]

	0.00E+00	1.00E+00
1	4.32E-03	4.22E-03

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

RESRAD-BUILD Table of Contents

RESRAD-BUILD Table of Contents

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

RESRAD-BUILD Input Parameters

Number of Sources : 5
Number of Receptors: 1
Total Time : 3.650000E+02 days
Fraction Inside : 2.300000E-01

Receptor Information

Receptor	Room	x [m]	y [m]	z [m]	FracTime	Inhalation [m3/day]	Ingestion(Dust) [m2/hr]
1	1	2.400	2.400	1.000	1.000	1.80E+01	1.00E-04

Receptor-Source Shielding Relationship

Receptor	Source	Density [g/cm3]	Thickness [cm]	Material
1	1	2.40E+00	0.00E+00	Concrete
1	2	2.40E+00	0.00E+00	Concrete
1	3	2.40E+00	0.00E+00	Concrete
1	4	2.40E+00	0.00E+00	Concrete
1	5	2.40E+00	0.00E+00	Concrete

Building Information

Building Air Exchange Rate: 8.00E-01 1/hr

```
 Height[m]          Air Exchanges [m3/hr]
 Area [m2]
                   ******************************
                   *                            *
                   *                            *
                   *                            <=Q01: 2.68E+01
 H1:   1.800       *          Room  1           *      Q10 :  2.68E+01
                   *   LAMBDA: 8.00E-01         *
Area  18.600       *                            *
                   *                            *
                   ******************************
```

Deposition velocity: 1.00E-02 [m/s]   Resuspension Rate: 5.00E-06 [1/s]

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

Source Information

Source: 1

Location:: Room : 1 x: 2.40 y: 2.40 z: 0.00[m]
Geometry:: Type: Volume Area:1.86E+01 [m2] Direction: z
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.00E+00
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

Source: 2
Location:: Room : 1 x: 4.80 y: 2.40 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: x
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.00E+00
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

Source: 3
Location:: Room : 1 x: 0.00 y: 2.40 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: x
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.00E+00
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

Source: 4
Location:: Room : 1 x: 2.40 y: 0.00 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: y
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.00E+00
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP

Source: 5
Location:: Room : 1 x: 2.40 y: 4.80 z: 0.00[m]
Geometry:: Type: Volume Area:2.70E+00 [m2] Direction: y
Pathway ::
Direct Ingestion Rate: 0.000E+00 [gm/hr]
Fraction released to air: 1.000E-01

Containment :: Number of Regions: 1 Contaminated Region: 1

Region	: 1
Thickness [cm]	:1.00E+00
Density [g/cm3]	:2.40E+00
Material	:Concrete
Erosion Rate [cm/day]	:2.40E-08

Contamination::

Nuclide	Concentration	Dose Conversion Factors				
		Ingestion	Inhalation	External (Surface)	External (Volume)	Submersion
	[pCi/g]	[mrem/pCi]	[mrem/pCi]	[mrem/yr/ (pCi/m2)]	[mrem/yr/ (pCi/m3)]	[mrem/yr/ (pCi/m3)]
SR-90	1.000E+00	1.530E-04	1.310E-03	6.560E-07	1.540E-08	2.310E-05

Title : Sr-90 Dose Rate from Concrete

Input File : C:\WINBLD\SR90B.INP Evaluation Time: 0.000000 years

```
==========================================================
==========================================================
=          Assessment for Time:  1                       =
=                Time =0.00E+00 yr                       =
==========================================================
==========================================================

          ======== Source Information ========

Source:  1
        Location::  Room :  1  x:   2.40 y:   2.40  z:   0.00 [m]
        Geometry::  Type: Volume     Area:1.86E+01 [m2]  Direction: z
        Pathway ::
             Direct Ingestion Rate    :0.000E+00 [gm/hr]
             Fraction released to air: 1.000E-01

        Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00

        Contamination::      Nuclide    Concentration
                                          [pCi/g]
                              SR-90      1.000E+00
```

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 0.000000 years

```
Source:  2
       Location::  Room :  1  x:   4.80 y:   2.40  z:   0.00 [m]
       Geometry::  Type: Volume      Area:2.70E+00 [m2]  Direction: x
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01

       Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00

       Contamination::      Nuclide    Concentration
                                          [pCi/g]
                             SR-90       1.000E+00

Source:  3
       Location::  Room :  1  x:   0.00 y:   2.40  z:   0.00 [m]
       Geometry::  Type: Volume      Area:2.70E+00 [m2]  Direction: x
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01

       Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00

       Contamination::      Nuclide    Concentration
                                          [pCi/g]
                             SR-90       1.000E+00
```

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 0.000000 years

```
Source:  4
       Location::  Room :  1  x:   2.40 y:   0.00  z:   0.00 [m]
       Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: y
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01

       Containment ::  Number of Regions:  1 Contaminated Region:  1
   Region                :    1
   Thickness [cm]        :1.00E+00
   Fraction Contaminated :1.00E+00
   Density [g/cm3]       :2.40E+00

       Contamination::     Nuclide   Concentration
                                       [pCi/g]
                           SR-90      1.000E+00

Source:  5
       Location::  Room :  1  x:   2.40 y:   4.80  z:   0.00 [m]
       Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: y
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01

       Containment ::  Number of Regions:  1 Contaminated Region:  1
   Region                :    1
   Thickness [cm]        :1.00E+00
   Fraction Contaminated :1.00E+00
   Density [g/cm3]       :2.40E+00

       Contamination::     Nuclide   Concentration
                                       [pCi/g]
                           SR-90      1.000E+00
```

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 0.000000 years

RESRAD-BUILD Dose Tables

Source Contributions to Receptor Doses

[mrem]

	Source 1	Source 2	Source 3	Source 4	Source 5	Total
Receptor 1	1.1E-03	8.0E-05	8.0E-05	8.0E-05	8.0E-05	1.4E-03
Total	1.1E-03	8.0E-05	8.0E-05	8.0E-05	8.0E-05	1.4E-03

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 0.000000 years

Pathway Detail of Doses

[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	1.07E-03	2.90E-10	8.83E-12	3.29E-06	0.00E+00	1.02E-07
Total	1.07E-03	2.90E-10	8.83E-12	3.29E-06	0.00E+00	1.02E-07

Source: 2

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Source: 3

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Source: 4

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Source: 5

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08
Total	7.93E-05	4.20E-11	1.28E-12	4.77E-07	0.00E+00	1.49E-08

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 0.000000 years

Nuclide Detail of Doses

[mrem]

Source: 1

Nuclide	Receptor 1	Total
SR-90		
SR-90	1.08E-03	1.08E-03

Source: 2

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.98E-05	7.98E-05

Source: 3

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.98E-05	7.98E-05

Source: 4

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.98E-05	7.98E-05

Source: 5

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.98E-05	7.98E-05

Assessment for Time: 2
Time =1.00E+00 yr

Source Information

```
Source:  1
      Location::  Room :  1  x:   2.40 y:   2.40  z:   0.00 [m]
      Geometry::  Type: Volume    Area:1.86E+01 [m2]  Direction: z
      Pathway ::
           Direct Ingestion Rate   :0.000E+00 [gm/hr]
           Fraction released to air: 1.000E-01


      Containment ::   Number of Regions:  1 Contaminated Region:  1
   Region                 :    1
   Thickness [cm]         :1.00E+00
   Fraction Contaminated  :1.00E+00
   Density [g/cm3]        :2.40E+00


      Contamination::      Nuclide    Concentration
                                        [pCi/g]
                            SR-90      9.765E-01
```

```
Source:  2
      Location::  Room :  1  x:   4.80 y:   2.40  z:   0.00 [m]
      Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: x
      Pathway ::
           Direct Ingestion Rate   :0.000E+00 [gm/hr]
           Fraction released to air: 1.000E-01


      Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


      Contamination::      Nuclide    Concentration
                                        [pCi/g]
                            SR-90       9.765E-01

Source:  3
      Location::  Room :  1  x:   0.00 y:   2.40  z:   0.00 [m]
      Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: x
      Pathway ::
           Direct Ingestion Rate   :0.000E+00 [gm/hr]
           Fraction released to air: 1.000E-01


      Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated :1.00E+00
    Density [g/cm3]        :2.40E+00


      Contamination::      Nuclide    Concentration
                                        [pCi/g]
                            SR-90       9.765E-01
```

```
Source:  4
       Location::  Room :  1  x:   2.40 y:   0.00  z:   0.00 [m]
       Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: y
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01

       Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated  :1.00E+00
    Density [g/cm3]        :2.40E+00

       Contamination::      Nuclide    Concentration
                                          [pCi/g]
                             SR-90      9.765E-01

Source:  5
       Location::  Room :  1  x:   2.40 y:   4.80  z:   0.00 [m]
       Geometry::  Type: Volume     Area:2.70E+00 [m2]  Direction: y
       Pathway ::
            Direct Ingestion Rate   :0.000E+00 [gm/hr]
            Fraction released to air: 1.000E-01

       Containment ::   Number of Regions:  1 Contaminated Region:  1
    Region                 :    1
    Thickness [cm]         :1.00E+00
    Fraction Contaminated  :1.00E+00
    Density [g/cm3]        :2.40E+00

       Contamination::      Nuclide    Concentration
                                          [pCi/g]
                             SR-90      9.765E-01
```

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 1.00000 years

RESRAD-BUILD Dose Tables

Source Contributions to Receptor Doses

[mrem]

	Source 1	Source 2	Source 3	Source 4	Source 5	Total
Receptor 1	1.1E-03	7.8E-05	7.8E-05	7.8E-05	7.8E-05	1.4E-03
Total	1.1E-03	7.8E-05	7.8E-05	7.8E-05	7.8E-05	1.4E-03

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 1.00000 years

Pathway Detail of Doses

[mrem]

Source: 1

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	1.05E-03	2.83E-10	8.62E-12	3.21E-06	0.00E+00	9.99E-08
Total	1.05E-03	2.83E-10	8.62E-12	3.21E-06	0.00E+00	9.99E-08

Source: 2

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Source: 3

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Source: 4

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Source: 5

Receptor	External	Deposition	Immersion	Inhalation	Radon	Ingestion
1	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08
Total	7.74E-05	4.10E-11	1.25E-12	4.66E-07	0.00E+00	1.45E-08

Title : Sr-90 Dose Rate from Concrete
Input File : C:\WINBLD\SR90B.INP Evaluation Time: 1.00000 years

Nuclide Detail of Doses

[mrem]

Source: 1

Nuclide	Receptor 1	Total
SR-90		
SR-90	1.05E-03	1.05E-03

Source: 2

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.79E-05	7.79E-05

Source: 3

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.79E-05	7.79E-05

Source: 4

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.79E-05	7.79E-05

Source: 5

Nuclide	Receptor 1	Total
SR-90		
SR-90	7.79E-05	7.79E-05

RESRAD-BUILD Dose (Time) Tables

Receptor Doses Received for the Exposure Duration

(mrem)

Evaluation Time [yr]

	0.00E+00	1.00E+00
1	1.40E-03	1.36E-03

Receptor Dose/Yr Averaged Over Exposure Duration

(mrem/yr)

Evaluation Time [yr]

	0.00E+00	1.00E+00
1	1.40E-03	1.36E-03

ENCLOSURE 3 - TABLE 7.2 OF DRAFT NUREG 1640 "RADIOLOGICAL ASSESSMENTS FOR CLEARANCE OF EQUIPMENT AND MATERIALS FROM NUCLEAR FACILITIES"

[redacted]

Table 7.2 Concrete recycle critical-group dose factors—mass

	(µSv/y per Bq/g)				(mrem/y per pCi/g)				
	Critical-group dose factor	90% Confidence interval			Critical-group dose factor	90% Confidence interval			
Radionuclide		5th (lower bound)	50th (median)	95th (upper bound)		5th (lower bound)	50th (median)	95th (upper bound)	Scenario
H-3	8.8E-03	9.9E-06	5.0E-04	3.9E-02	3.3E-05	3.7E-08	1.9E-06	1.5E-04	CN-SCRP-LANDFIL-N
C-14	1.6E+01	2.7E-02	1.1E+00	5.7E+01	5.9E-02	1.0E-04	4.1E-03	2.1E-01	CN-SCRP-LANDFIL-N
Na-22	2.2E+02	1.4E+02	2.2E+02	3.0E+02	8.1E-01	5.3E-01	8.1E-01	1.1E+00	CN-SCRP-TRANSPO-W
P-32	1.4E-01	8.6E-02	1.4E-01	1.9E-01	5.1E-04	3.2E-04	5.0E-04	7.2E-04	CN-SCRP-TRANSPO-W
S-35	**3.5E-03**	**1.5E-03**	**3.3E-03**	**6.2E-03**	**1.3E-05**	**5.4E-06**	**1.2E-05**	**2.3E-05**	**CN-SCRP-HANDLIN-W**
Cl-36	1.6E+02	1.0E+00	1.9E+01	5.4E+02	5.9E-01	3.9E-03	6.9E-02	2.0E+00	CN-SCRP-LANDFIL-N
K-40	4.7E+01	1.9E-01	5.7E+00	2.2E+02	1.7E-01	6.9E-04	2.1E-02	8.2E-01	CN-SCRP-LANDFIL-N
Ca-41	4.8E+00	1.4E-02	4.6E-01	1.9E+01	1.8E-02	5.2E-05	1.7E-03	7.0E-02	CN-SCRP-LANDFIL-N
Ca-45	1.8E-02	6.4E-03	1.7E-02	3.4E-02	6.7E-05	2.4E-05	6.2E-05	1.3E-04	CN-SCRP-HANDLIN-W
Cr-51	**2.5E+00**	**1.6E+00**	**2.5E+00**	**3.5E+00**	**9.4E-03**	**5.9E-03**	**9.2E-03**	**1.3E-02**	**CN-SCRP-TRANSPO-W**
Mn-54	8.5E+01	5.6E+01	8.4E+01	1.2E+02	3.2E-01	2.1E-01	3.1E-01	4.4E-01	CN-SCRP-TRANSPO-W
Fe-55	3.4E-03	1.1E-03	3.2E-03	6.5E-03	1.3E-05	4.2E-06	1.2E-05	2.4E-05	CN-SCRP-HANDLIN-W
Co-57	6.6E+00	1.4E+00	5.9E+00	1.3E+01	2.4E-02	5.3E-03	2.2E-02	5.0E-02	CN-SCRP-ROADBED-W
Co-58	9.6E+01	6.2E+01	9.5E+01	1.3E+02	3.5E-01	2.3E-01	3.5E-01	4.9E-01	CN-SCRP-TRANSPO-W
Fe-59	**1.2E+02**	**7.4E+01**	**1.2E+02**	**1.6E+02**	**4.3E-01**	**2.8E-01**	**4.3E-01**	**6.0E-01**	**CN-SCRP-TRANSPO-W**
Ni-59	7.4E-03	1.8E-05	8.1E-04	3.0E-02	2.7E-05	6.8E-08	3.0E-06	1.1E-04	CN-SCRP-LANDFIL-N
Co-60	2.5E+02	1.7E+02	2.5E+02	3.5E+02	9.4E-01	6.1E-01	9.3E-01	1.3E+00	CN-SCRP-TRANSPO-W
Ni-63	1.0E-02	2.7E-05	1.0E-03	3.8E-02	3.7E-05	1.0E-07	3.7E-06	1.4E-04	CN-SCRP-LANDFIL-N
Zn-65	6.0E+01	3.9E+01	5.9E+01	8.3E+01	2.2E-01	1.4E-01	2.2E-01	3.1E-01	CN-SCRP-TRANSPO-W
Cu-67	**2.9E+00**	**1.1E+00**	**2.5E+00**	**5.6E+00**	**1.1E-02**	**3.9E-03**	**9.2E-03**	**2.1E-02**	**CN-SCRP-TRANSPO-W**
Se-75	2.5E+01	5.5E+00	2.3E+01	5.2E+01	9.3E-02	2.0E-02	8.4E-02	1.9E-01	CN-SCRP-ROADBED-W
Sr-85	4.7E+01	3.0E+01	4.7E+01	6.6E+01	1.8E-01	1.1E-01	1.7E-01	2.4E-01	CN-SCRP-TRANSPO-W
Sr-89	1.4E-01	5.3E-02	1.3E-01	2.7E-01	5.3E-04	1.9E-04	4.8E-04	9.8E-04	CN-SCRP-ROADBED-W
Sr-90	2.2E+00	3.0E-03	1.3E-01	7.6E+00	8.3E-03	1.1E-05	4.8E-04	2.8E-02	CN-SCRP-LANDFIL-N
Y-91	**4.3E-01**	**1.2E-01**	**3.9E-01**	**8.7E-01**	**1.6E-03**	**4.6E-04**	**1.4E-03**	[illegible]	**CN-SCRP-ROADBED-W**
Mo-93	3.8E+01	9.9E-02	3.5E+00	1.6E+02	1.4E-01	3.7E-04	1.3E-02	6.1E-01	CN-SCRP-LANDFIL-N
Nb-93m	2.0E-02	7.8E-03	1.8E-02	4.2E-02	7.5E-05	2.9E-05	6.7E-05	1.6E-04	CN-SCRP-HANDLIN-W
Nb-94	2.9E+02	4.4E-01	2.1E+01	1.1E+03	1.1E+00	1.6E-03	7.9E-02	3.9E+00	CN-SCRP-LANDFIL-N
Nb-95	7.4E+01	4.7E+01	7.3E+01	1.0E+02	2.7E-01	1.7E-01	2.7E-01	3.8E-01	CN-SCRP-TRANSPO-W
Zr-95	**7.3E+01**	**4.7E+01**	**7.2E+01**	**1.0E+02**	**2.7E-01**	**1.7E-01**	**2.7E-01**	**3.7E-01**	**CN-SCRP-TRANSPO-W**
Tc-99	1.2E+02	8.8E-01	1.8E+01	4.6E+02	4.4E-01	3.3E-03	6.7E-02	1.7E+00	CN-SCRP-LANDFIL-N
Ru-103	4.7E+01	3.0E+01	4.7E+01	6.6E+01	1.7E-01	1.1E-01	1.7E-01	2.4E-01	CN-SCRP-TRANSPO-W
Ru-106	2.1E+01	1.3E+01	2.0E+01	2.9E+01	7.6E-02	5.0E-02	7.6E-02	1.1E-01	CN-SCRP-TRANSPO-W
Ag-108m	1.7E+02	2.3E-01	1.2E+01	7.4E+02	6.4E-01	8.6E-04	4.4E-02	2.7E+00	CN-SCRP-LANDFIL-N
Cd-109	**2.5E-01**	**8.8E-02**	**2.3E-01**	**4.7E-01**	**9.2E-04**	**3.2E-04**	**8.5E-04**	**1.7E-03**	**CN-SCRP-ROADBED-W**

7 EVALUATION OF RECYCLE AND DISPOSAL OF CONCRETE

Seven potential exposure scenarios for cleared concrete debris that are realistically based on current American industries are evaluated in this section. As in the evaluations for metals, concrete scenario categories include handling and processing, transportation, product use, and disposal. Unique to the concrete evaluation is the inclusion of a scenario involving a resident on the site of a closed landfill. A radionuclide-specific, probabilistic dose factor distribution was calculated for the members of all exposed groups. Dose factors for an average member of each critical group was calculated in the same manner as for metals recycle. The mean dose factor for each critical group is reported in normalized units of μSv/y per Bq/g scrap (mrem/y per pCi/g) and μSv/y per Bq/cm^2 (mrem/y per pCi/cm^2) for each radionuclide. Dose factors at the 5th, 50th, and 95th percentiles are also reported.

Similar to the metals analyses, the most common critical group consists of commercial truck drivers carrying cleared concrete debris, accounting for critical group designation for almost half (33) of the 85 radionuclides in the analysis. Unlike the metals analyses, the scenario involving a resident on a closed landfill containing cleared concrete debris was analyzed and included in critical group determination. This scenario accounts for critical group designation for one-third (28) of the radionuclides in the analysis. Scenarios involving handling concrete debris accounted for most of the remaining critical groups.

Mean critical-group dose factors range from a high of 7E+04 μSv/y per Bq/g (3E+02 mrem/y per pCi/g) for Np-237 to a low of 3E-03 μSv/y per Bq/g (1.3E-05 mrem/y per pCi/g) for Fe-55. The surficial mean critical-group dose factors range from 1.4E+03 μSv/y per Bq/cm^2 (Np-237) down to 5E-05 μSv/y per Bq/cm^2 (S-35). The Np-237 highest mean dose factors are a factor of 30 higher than the next highest mean dose factor. The high mean dose factor of Np-237 is the highest calculated for all materials (metals and concrete) and is attributed to the drinking-water exposure pathway in a residential scenario on the site of a closed landfill. Seventy-four of the radionuclide-specific, mean dose factors for concrete critical groups are 270 μSv/y per Bq/g or lower (1 mrem/y per pCi/g).

This section describes the technical evaluation of the recycle and disposal of concrete debris that could be cleared by Nuclear Regulatory Commission (NRC) licensed facilities. Similar to the analyses of metal scrap (Sections 4, 5, and 6), the likely flow of cleared concrete was probabilistically modeled, distributions of radionuclide concentrations in recycled products and disposed materials were calculated, and potential doses to average members of potential critical groups were estimated.

7.1 Introduction to Analysis

The analysis process for calculating potential exposures from concrete recycle and disposal is relatively simple when compared to the analyses conducted for recycled and disposed metals. A simple material flow model specific to concrete recycle and disposal was developed, radioactivity concentrations in reprocessed concrete were calculated, and scenario evaluations were conducted. In contrast to the metal analyses, the processing of concrete does not involve redistribution of radionuclides, and there is only one resulting end product. The exposure scenario evaluations for concrete are similar to those for the metals, with only several minor scenario parameter differences, and only one fundamentally different scenario is used in the determination of critical groups (resident on a closed landfill site).

7.2 Flow of Recycled and Disposed Concrete

The material flow model for concrete represents the general processes that concrete cleared by NRC licensed facilities would go through from the time of clearance to the time of final disposal. Decontamination is assumed to have already taken place and is not included in the model. The material flow model comprises two types of models: conceptual and mathematical. The conceptual model describes the recycling process and defines the limits of this analysis, while the mathematical model presents the mathematical equations that implement the information presented in the conceptual model.

The material flow model is based largely on information gathered from personal communications with individuals in the concrete recycling industry. It is intended to provide a defensible basis for developing appropriate exposure scenarios for recycled concrete. In addition, the output of the material flow model provides key input to the dose calculations for each scenario. This input consists of radionuclide-specific concentrations in recycle products containing recycled concrete, in effluents, and in waste products of the recycling process. Because of the number of scenarios being analyzed and the differing times at which they occur, the mathematical model presented in this section does not include radioactive decay. Like other materials, this is accounted for in the scenario evaluations.

This section presents the conceptual model that follows the concrete through a typical recycling process, beginning with cleared concrete debris, through the recycling process, product use, and ending with disposal. Figure 7.1 presents a schematic diagram of the conceptual model of the overall flow of recycled and disposed concrete. The modeling of both mass and radioactivity in recycled concrete follow this flow model. As seen in Figure 7.1, there are several distinct steps in the recycling process. Each of these are discussed separately in the following subsections.

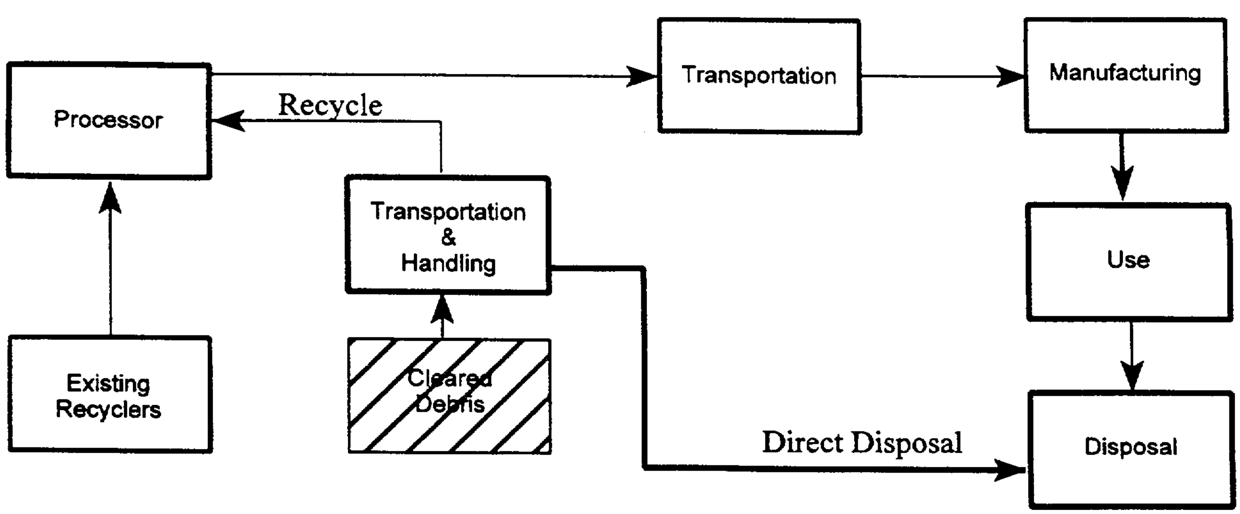


Figure 7.1 Conceptual model of the overall flow of recycled and disposed concrete

7.2.1 Source of Material

Concrete for recycling comes mainly from commercial sources, which contributes concrete from demolished buildings or parking lots.

The main producer of concrete debris (scrap) for this analysis is the nuclear industry, which consists mainly of commercial power plants, test and research reactors, and industrial nuclear facilities. Other producers of concrete debris include the Department of Energy (DOE) weapons complex and the Department of Defense (DOD), which could contribute concrete debris primarily from conventional weapons testing and army and navy test reactors.

Unlike the metals analyzed, concrete is more likely to be recycled only during the decontamination and decommissioning (D&D) of a nuclear facility. This would result in a large amount of concrete available for recycling during a very short period of time. The D&D of a typical pressurized water reactor would result in about 180,000 t (198,000 tons), whereas approximately 348,000 t (383,000 tons) would be available from a typical boiling water reactor (NUREG/CR-5610[1]). Large masses of concrete could also be available during the D&D of non-reactor facilities.

7.2.2 Transportation and Processing

Concrete debris cleared from a nuclear facility may be transported, usually by a commercial company, to the processor. Processing could also occur at the origination site of the concrete debris. The operations at both places are similar. Because not all concrete debris is likely to be suitable for recycle, some would be transported directly to a disposal site. In both cases, transportation would occur by truck. This transportation could occur either before or after processing occurs.

Prior to use, concrete debris must be crushed and sorted. Any steel rebar in the concrete would be removed with a magnet prior to crushing and sent to an iron and steel refinery. The concrete must be in blocks approximately 1.5 x 1.5 x 0.8 m to be crushed. During crushing, water is sprayed on the dust to contain it. Although not required by regulation, workers near the site typically wear dust respirators. One major recycler of concrete, Dykes Paving & Construction Co., Inc., recycles about 230,000 t (250,000 tons) of concrete per year.[2]

As soon as the concrete debris is needed after processing and crushing, it is transported by truck to the site of the final use.

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[1]Recycle/Reuse Literature Search Report, SAIC, 1994, scheduled to be published as NUREG/CR-5610.

[2]Personal communication between M. Anderson, SAIC, and J. Dykes, Dykes Paving and Construction Co., Inc., January 13, 1997.

7.2.3 Manufacturing

In order to prepare recycled, crushed concrete debris for final use, it is mixed with a small amount of additives. The additives are added to promote adhesion. According to Dykes,[1] it is very common for the final concrete end product to be made of entirely recycled concrete and additives, although occasionally other aggregate is added to the recycled concrete. If the recycled concrete is used just as aggregate, such as for the base of roadways, then there is no additional processing after the crushing and sizing at the scrap dealer/processor.

7.2.4 Product Use

Typical end products for recycled concrete include the base for roads, a stabilizer for asphalt, and an aggregate for non-structural materials. Recycled concrete is not used as aggregate in structural concrete, such as that used in houses or buildings.[1]

7.2.5 Disposal

The ultimate endpoint for some concrete is a public sanitary landfill (i.e., Resource Conservation and Recovery Act (RCRA) Subtitle D landfill). Concrete debris that is not suitable for recycling is sent directly to a landfill. Also, end products containing recycled concrete can also be sent to a landfill after use.

7.3 Mathematical Model for the Flow of Recycled Concrete Debris

This section presents the equations that represent the information discussed previously in Section 7.2 and are used to formulate the mathematical model

According to the material flow conceptual model, there is only one point at which the cleared concrete debris radionuclide concentration is potentially changed: mixing with other concrete or aggregate before or during recycling. Dilution with other concrete is radionuclide-independent. The following equations calculate the concentrations at points of interest (i.e., where scenarios would occur).

The total mass of material entering the recycling facility in a year, including cleared concrete debris and other concrete/aggregate was calculated using Equation 7.1.

$$M_T = M_C + M_{CA} \tag{7.1}$$

where

M_T = total mass of material entering the recycling facility in a year (g/y)

[1] Personal communication between M. Anderson, SAIC, and J. Dykes, Dykes Paving and Construction Co., Inc., January 13, 1997.

and

M_C = mass of cleared concrete debris entering the recycling facility per year (g/y)
M_{CA} = mass of other concrete debris/aggregate entering the recycling facility per year (g/y)

The radionuclide concentration in the recycled concrete entering the manufacturing process was calculated using Equation 7.2. The following equations are intended to be evaluated separately for each radionuclide in the analysis. Therefore, parameters such as radionuclide concentration, decay factors, decay constants, and dose conversion factors are not explicitly subscripted for each radionuclide.

$$C_T = \frac{M_C}{M_T} * C_C \qquad 7.2$$

where

C_T = radionuclide concentration in recycled concrete entering the manufacturing process (pCi/g)

and

M_C = mass of cleared concrete debris entering the recycling facility per year (g/y)
M_T = total mass of material entering the recycling facility in a year (g/y)
C_C = original radionuclide concentration of cleared concrete debris (pCi/g)

7.4 Radioactivity Concentrations in Concrete Recycle Products

Radioactivity concentrations in concrete recycle products are calculated using the models described in the previous sections. The evaluation for concrete recycle and disposal assumes some mixing of cleared concrete debris with other concrete debris. No mass or elemental partitioning occurs during processing. Therefore, for a unit radionuclide concentration in cleared concrete, the concentration in concrete recycle products would be the same for all radionuclides and all scenarios except immediate handling or transport of cleared concrete.

The radioactivity concentrations that characterize the distribution calculated in the concrete material flow model are listed in Table 7.1.

Table 7.1 Results of concrete material flow model

	Radionuclide concentrations in recycled concrete (Bq/g per Bq/g concrete debris)			
Scenario Category	Mean	5^{th}	50^{th}	95^{th}
Large Object[a]	8.7E-01	7.82E-01	8.83E-01	9.40E-01
Other Scenarios	9.71E-01	8.61E-01	9.93E-01	1.00E+00

a. The only concrete scenario using this set of radionuclide concentrations is CN-SCRP-LGMASS-N.

7.5 Dose Assessment for Concrete Recycle and Disposal

There are no scenarios in addition to those discussed for steel that are used for the evaluation of concrete recycling and disposal. However, because landfill disposal of large masses of concrete is more likely than for metals, a landfill resident scenario was included in the determination of critical groups for concrete. For other scenarios, the analyses are the same as described for steel recycle in Section 4, except for several parameter value differences. These are discussed in the next section.

7.6 Parameter Discussion

Calculation of potential impacts from recycle of cleared concrete depends on many parameter values. Parameter values are needed for both material flow and dose assessment models. The material flow parameter values rely on information from the concrete recycling industry. The parameter values for the dose modeling are a function of receptor behavior, such as time on a specific job or breathing rate, as well as how the radionuclides are transported in the environment. The dose modeling parameter values used for evaluation of concrete recycling and disposal are generally the same as those used in the iron and steel modeling. Only those parameter values that are different than those used in the iron and steel modeling are discussed in this section.

7.6.1 Scenario Timing

As described previously, all exposure scenarios were derived from the material flow conceptual modeling, and scenarios were assumed to occur at specific points in time following clearance from the NRC licensed facility. In order to incorporate radioactive decay between clearance and scenario dose modeling, the specific times were quantified as described in this section. Where a reference was used as a basis for the assumed time period, it is cited. If no citation is given, general knowledge of practices gained from talking with industry personnel combined with professional judgement was used as the basis. The following is a list of basic assumptions that were used in developing the timing of the concrete recycle scenarios:

Scrap Handling:

- All scrap concrete is assumed to be initially taken to a scrap processor. It takes approximately 4 days for the scrap concrete to reach the scrap dealer.

Refining and Processing:

- The scrap concrete remains at the scrap processor for a period of approximately 7 days prior to use or disposal.

Use and Disposal:

- A recycled concrete product is used for 30 years, based on an assumed use in the construction industry.

Post-Disposal:

- A sanitary landfill operates for 40 years.
- Disposed items are placed in the landfill at the midpoint of the landfill operating period.
- Post-closure monitoring of landfills is done for 30 years.
- A disposal site is released for potential residential use after post-closure monitoring ends.

Specific scenario timing values are listed in Appendix B, Table B.15.

7.6.2 Dose Evaluation

This section presents a discussion of parameter values used for the analysis of concrete recycle scenarios. The analysis of seven concrete scenarios is based on the scenario descriptions provided in the steel analysis (section 4). Similar scenarios are analyzed using the same models as for the steel recycle analysis. Values used for many of the parameters are also the same as for the steel recycle analysis. For some parameters, however, values specific to the recycle of concrete were used in order to more appropriately model the scenarios as they would likely occur for concrete recycle. The following discussion is limited to these specific parameter values. If a parameter is not discussed below and listed in Table B.15, the same value as used for the steel evaluation (Table B.7) is used for concrete.

7.6.2.1 Input Concentrations from the Material Flow

The undecayed radionuclide concentration in the source material, C_x, is used as the source material input concentration for scenario evaluations. For concrete recycling, this parameter is radionuclide-independent and is calculated in the material flow modeling. Representative values from the input concentration are listed in Table 7.1.

7.6.2.2 External Exposure Parameters

The only unique parameter value used for evaluation of the external pathway for concrete recycle is the density of concrete, for which a fixed value of 2.33E+6 g/m^3 was used.

7.6.3 Concrete Surface-to-Mass Ratio

The surface-to-mass ratio for concrete is representative of concrete slabs (e.g., floors and walls) with residual radioactivity on only one side. The surface-to-mass ratio is not sensitive to the size of the pieces made from these slabs but is sensitive to the original thickness of the slabs. The surface-to-mass ratio is also dependent on the density of concrete.

ENCLOSURE 4 - REPORT ENTITLED "REVIEW OF PROPOSED SOIL SR-90 DCGL FOR ST. ALBANS" BY MARK THAGGARD

Review of Proposed Soil Sr-90 DCGL for St. Albans

by: Mark Thaggard, Sr. Systems Performance Analyst

1.0 Introduction

1.1 Purpose

The purpose of this report is to respond to a technical assistance request to review the dose modeling provided by the NRC Region I and the licensee to determine if the soil derived concentration guideline level (DCGL) for Sr-90 was developed in an acceptable manner and results in a dose that is in compliance with the license termination rule (LTR). The U.S. Army Corps of Engineers (COE) on behalf of the Veteran Administration Medical Center in Brooklyn, New York (license number 31-02892-06) has proposed a DCGL for Sr-90 for decommissioning contaminated soil at the St. Albans Veterans Administration Extended Care Center (VAECC).

1.2 Site Description

The VAECC is located on 55 acres in Queens, New York. The VAECC consists of 15 buildings encompassing approximately 65,000 m^2 (700,000 ft^2). Radioactivity was primarily used in Building 90, which formerly housed a nuclear medicine laboratory. Sampling results indicate Sr-90 contamination in the upper soil surface in the vicinity of the clean out/ floor drain system serving the nuclear medicine laboratory. Soil concentrations as high as 2,054 pCi/g were reported. Building 90 covers a footprint of roughly 900 m^2 (10,000 ft^2); therefore, the maximum area of soil contamination is estimated to be 900 m^2 (10,000 ft^2).

No information is provided on the specific radionuclides that were used at the site or whether other radionuclides beside Sr-90 were considered in the soil sampling. Elevated tritium activity was found in the building. Therefore, it cannot be concluded that Sr-90 is the only radionuclide of potential concern within the soil. However, because the technical assistance request specifically requested an evaluation of the DCGL for Sr-90, for the purposes of this review, it is assumed that Sr-90 is the only radionuclide present within the soils. If tritium is determined to be present within the soil, a separate DCGL, specific for tritium will need to be derived, and the "sum of fraction" rule applied to determine compliance with the LTR.

2.0 Licensee's Derivation of the Soil DCGL

The COE used RESRAD 5.91 to develop a site-specific soil DCGL for Sr-90 of 35 pCi/g. This site-specific DCGL is significantly higher than the screening DCGL of 1.7 pCi/g published by the NRC (64 FR 68395).

The licensee assumed a modified residential farmer scenario in the assessment. A combination of RESRAD default parameters and default parameter values recommended by the NRC in the "Preliminary Guidelines for Evaluating Dose Assessments in Support of Decommissioning" were used, with the exception of the following parameters designed to modify the scenario: