DEVELOPMENT OF BUILDING DCGLS

REVISION 0

CE WINDSOR SITE
WINDSOR, CONNECTICUT

US NRC LICENSE NUMBER 06-00217-06
DOCKET NUMBER 030-03754

DECEMBER 2008

ABB

MACTEC

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ABBREVIATIONS AND ACRONYMS

ABB .... ABB, Inc.
AEC .... United States Atomic Energy Commission

CE .... Combustion Engineering
CFR........................ Code of Federal Regulations
Co-60 .... cobalt 60
cm^2 .... square centimeter

DCF .... dose conversion factor
DCGL .... derived concentration guideline level
$DCGL_W$ .... derived concentration guideline level for the average (or median) concentration in the survey unit
DOE .... United States Department of Energy
DP .... Decommissioning Plan
dpm .... disintegration per minute

EPA........................ United States Environmental Protection Agency

HSA .... Historical Site Assessment

m .... meter
m^2 .... square meter
mrem .... millirem

NRC .... United States Nuclear Regulatory Commission

Pa-234 .... protactinium 234
pCi/g .... picocuries per gram

Ra-226 .... radium 226
RESRAD .... Residual Radioactivity (computer modeling code)
RME....................... reasonable maximum exposure

TEDE .... total effective dose equivalent
Th-230 .... thorium 230
Th-234 .... thorium 234

U-234 .... uranium 234
U-235 .... uranium 235
U-238 .... uranium 238

WWTP .... waste water treatment plant

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EXECUTIVE SUMMARY

This report presents building Derived Concentration Guidance Levels (DCGLs) for the Combustion Engineering, Inc. (CE) Site located at 2000 Day Hill Road in Windsor, Connecticut. The site is owned by ABB, Inc. (ABB) and ABB is currently licensed by the U.S. Nuclear Regulatory Commission (NRC) to possess radioactive materials at the site and to engage in the decontamination and decommissioning of the site (NRC 2007).

A derived concentration guideline level (DCGL) is a site-specific concentration determined to be protective of the health of individuals that might be exposed in the future to the residual radioactivity that might be left in place on the Site. The DCGLs have been calculated to meet requirements set by the NRC and Connecticut Department of Environmental Protection (CTDEP).

The Site was formerly used to perform design, engineering support, and manufacturing of uranium fuel components for both commercial and government reactors. Lesser functions supported at this Site included testing of nuclear reactor plant components, testing of materials, and servicing of radiologically contaminated reactor plant components. Because of these past activities, building surfaces of radiologically impacted buildings at the site may have residual surface radioactivity concentrations from enriched uranium (principally) and reactor byproduct materials (minimally).

The objective of ABB is to decontaminate and decommission the facilities and lands that supported these missions in accordance with applicable federal and state requirements and regulations such that the radioactive materials license held by ABB can be terminated and the land and buildings at the Site can be returned to unrestricted use.

It is anticipated that future uses of remaining buildings at the Site will be roughly consistent with its current use (commercial, light industrial uses). DCGLs were, therefore, calculated for two different potential future exposure scenarios: Commercial Occupancy and Building Remodel / Demolition.

Due to the range of uranium enrichments processed during operations in the impacted buildings at the Site, DCGLs were evaluated for two different enrichment levels. This was performed to determine if there was any significant difference between low enriched uranium (3%) and high enriched uranium (90%) with respect to potential future dose in the considered exposure scenarios.

The building DCGLs for the CE Windsor Site have been calculated using the *RESRAD-BUILD* Version 3.4 modeling code (Yu 2007). Each of the scenarios modeled results in a concentration corresponding to a 19 mrem per year dose limit (CTDEP 2002, CTAG 2003). Considering the potential future use scenarios, the limiting scenario (the one that results in the smallest concentration yielding 19 mrem per year) for uranium is the building renovation / demolition scenario with high enriched uranium (90%) and for reactor byproduct is the occupancy scenario. Based on these results, the proposed uranium building surfaces DCGL is total ($\alpha + \beta$) surface activity concentration of 20,148 dpm per 100 cm^2 and the proposed reactor byproduct building surfaces DCGL is Co-60 β surface concentration of 6,980 dpm per 100 cm^2.

Conservatism has been built into the prospective dose modeling (and thus the proposed DCGL) by conscientiously selecting exposure factor values that err on the side of safety when confronted with uncertainty in the selection of input parameter values.

The DCGL proposed has been derived using appropriate techniques in accordance with governing guidance, standards, and regulations. In addition, stakeholders including the CTDEP, have provided input into the parameter selection and scenario derivation. It is recommended that these building surface concentration values be approved and adopted as the Site-specific permissible building surface concentrations for the entire CE Windsor Site, as defined and as applicable.

1.0 INTRODUCTION

1.1 BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) regulates sites where radioactive materials are possessed or stored. The NRC manages its authority to regulate users of radioactive materials via a licensing system whereby it grants limited authority to a licensee and imposes certain restrictions and responsibilities on the licensee. In order to terminate such a license, the licensee must demonstrate that residual radioactivity remaining at the site following cessation of licensed activities is within applicable dose-based limits defined in regulation (NRC 1997).

From the mid-1950's, the Combustion Engineering (CE) Site in Windsor Connecticut has been involved in research, development, engineering, production, and servicing of nuclear fuels, systems and services (Harding 2002). ABB, Inc. (ABB) has prepared a site-wide Decommissioning Plan (DP) Revision 1, to allow for decommissioning that will lead to license termination and unrestricted release in accordance with the requirements of the License Termination Rule at 10 CFR Part 20, Subpart E (NRC 1997). The approved DP Revision 0 (ABB 2003), was written to specifically address the commercial operations areas of the site exclusively. DP Revision 1 addresses the remainder of the Site under license.

Under DP Revision 0 for the commercial areas, buildings were decontaminated and dismantled such that there are no remaining structures at the time of license termination. However, in the remaining areas to be addressed under DP Revision 1, the south end of Building 3 (High Bay) houses unique fossil fuel research facilities and will require release for unconditional use since it will remain operational at the time of license termination. This report documents and supports the development and derivation of dose-based, permissible concentrations of residual radioactivity, derived concentration guideline levels (DCGLs), on building surfaces at the Site.

1.2 BUILDING DCGL REPORT ROAD MAP

Section 1 of this report provides a brief introduction and identifies the relevant regulatory and site description. Section 2 provides a brief Site history to help orient the reader. Section 3 outlines the setup of the dose model and discusses the exposure scenarios evaluated. The results of the dose modeling are summarized in Section 4. Section 5 summarizes the development of the building surfaces DCGLs. A list of acronyms and a list of references are provided in Sections 6 & 7, respectively. Appendices are included to provide additional detail where appropriate.

1.3 DECOMMISSIONING OBJECTIVE

ABB's objective is to decommission the entire Site such that it will meet the criteria for unrestricted use as specified by the License Termination Rule at Title 10, Code of Federal Regulations (CFR) Part 20 (NRC 1997), and to terminate NRC license No. 06-00217-06 (NRC 2007). In support of this objective, ABB has derived dose-based DCGLs for radiologically impacted buildings that will remain at the time of license termination in accordance with applicable requirements and regulations. In particular, the High Bay of Building 3 will be surveyed to assess the concentrations of residual surface radioactivity remaining in the building and to verify that the residual surface radioactivity on the building's impacted surfaces has been reduced to concentrations less than the DCGLs. This report documents the derivation of dose-based DCGLs to be used in demonstrating that the criteria for unrestricted use have been met, and serves to support the regulatory decision to terminate the license.

1.4 SITE AND LICENSEE INFORMATION

The CE Windsor Site (Site) is located at 2000 Day Hill Rd., in Windsor, Connecticut and is subject to NRC Radioactive Materials License No. 06-00217-06 (NRC 2007) due to historical use involving licensable quantities of radioactive materials. The licensee for the Site is ABB. The address where licensed material will be used or possessed is, and where all correspondence concerning this license should be sent:

ABB Inc.
C/o John Conant
CEP 880-1911
2000 Day Hill Road
Windsor, CT 06095-0500

1.5 SITE DESCRIPTION

Between 1956 and 2001, the CE Windsor Site was used (at various times) to conduct and support research, development, engineering, production, and servicing of nuclear fuels and systems. These activities make the Site subject to regulatory requirements governing the use and termination of such use of radioactive materials (NRC 1997).

The CE Windsor property is located in the Town of Windsor, eight miles north of Hartford, Connecticut (Figure 1-1). The entire property consists of approximately 600 acres and is located at 2000 Day Hill Road, in Windsor, Connecticut. An overview of the site layout is shown on Figure 1-2.

The Site is industrially zoned by the Town of Windsor, and is located in a Mixed Land Use area of Hartford County. Nearby land uses are primarily commercial, commercial agricultural, industrial, and residential. Much of the northern and western portions of the property are wooded.

The Site is bordered by Day Hill Road to the south; commercial development and a sand and gravel quarry to the west; the Windsor/Bloomfield Sanitary Landfill and Recycling Center (Landfill) and the Rainbow Reservoir portion of the Farmington River to the north; and forested land with some residential and commercial development to the east.

1.6 BUILDING DCGLS REPORT

With the exception of the Building 3 High Bay, the radiologically impacted buildings at the site have been (or will be) demolished and disposed of off site in accordance with the applicable rules and regulations as described in the DP. The Building 3 High Bay structure is slated to remain in place up to and through the time of license termination. Consequently, the Building 3 High Bay is subject to the decommissioning rule's dose criteria and dose-based DCGLs must be developed in order to demonstrate that residual radioactivity present on the building's surfaces at the time of license termination meet the rule's dose criteria.

This report documents the derivation of building DCGLs that are protective of the decommissioning rule's annual dose criteria. It establishes a measureable, concentration-based, residual surface radioactivity criterion which forms the basis of the design and implementation of final status surveys that will be performed as part of the decommissioning activities at the Site.

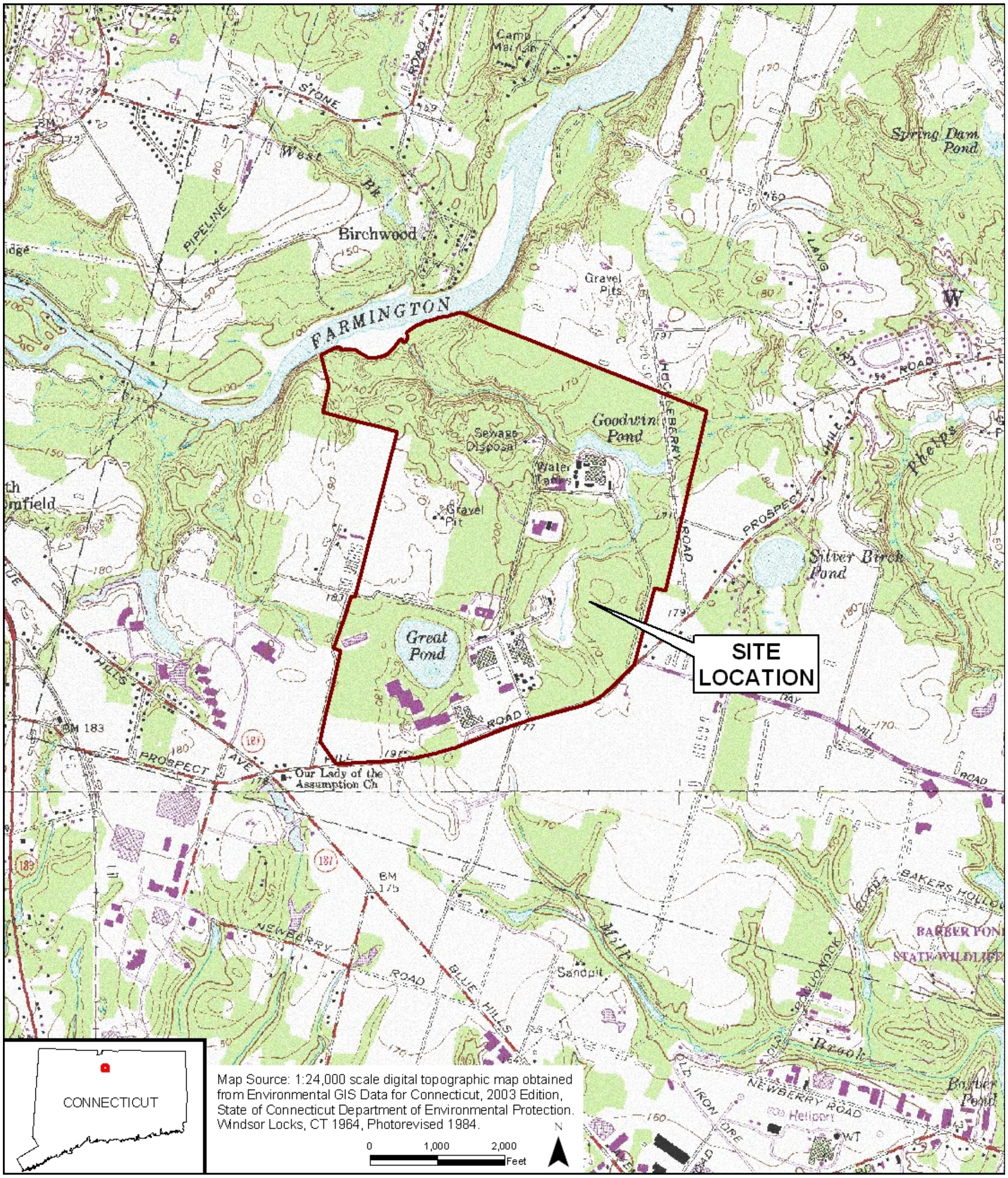

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Figure 1-1. Site Location Map

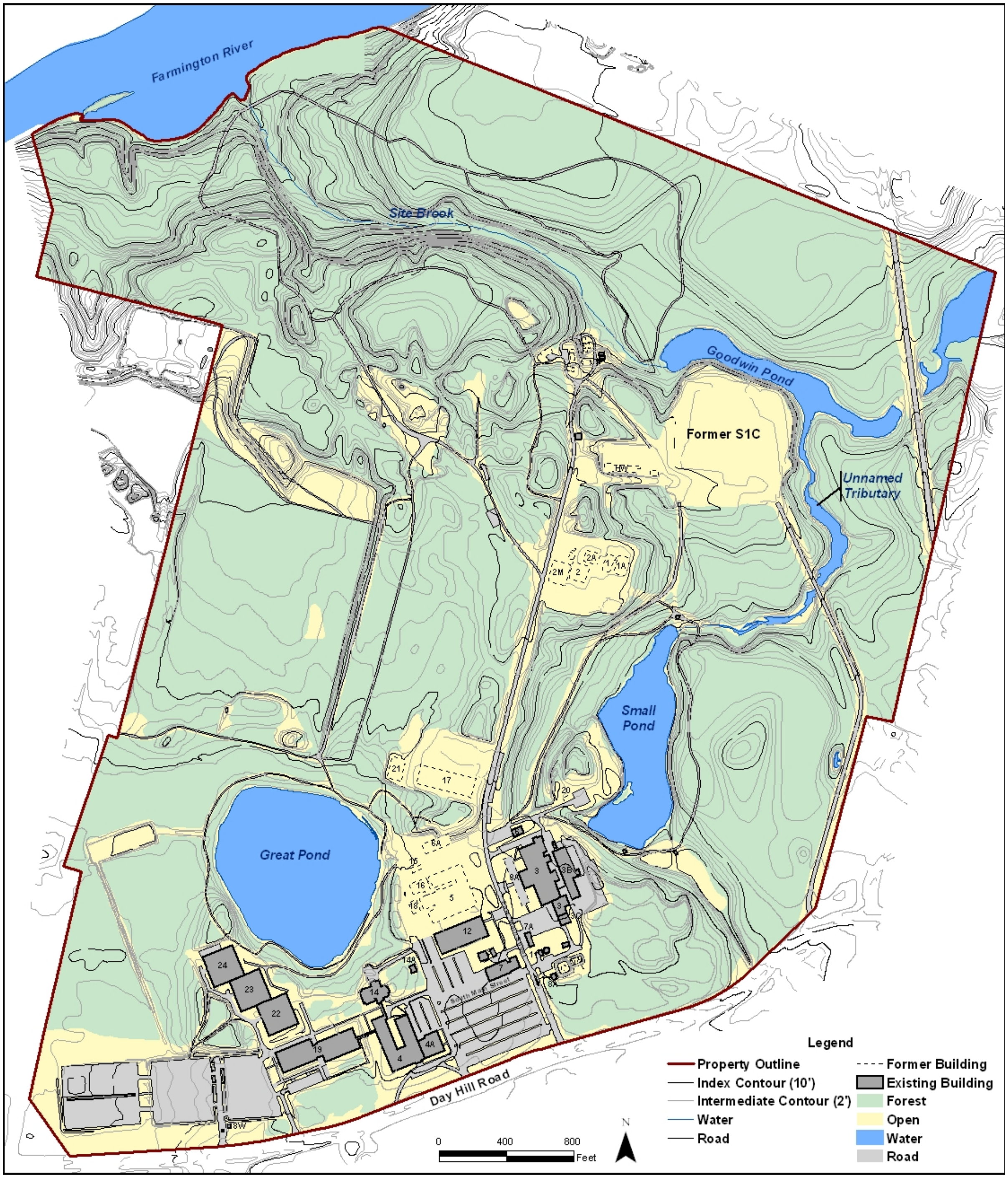

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Figure 1-2. Site Overview

2.0 SITE HISTORY

2.1 HISTORICAL SITE ACTIVITIES

ABB's activities at the Site started in 1955 with an Atomic Energy Commission (AEC) contract to begin research, development, and manufacturing of nuclear fuels for the United States Navy. Activities also included the construction, testing, and operation of the S1C facility, a U.S. Naval test reactor. Contracts with the AEC led to the construction of facilities in 1956 for the development, design, and fabrication of fuel element subassemblies for U.S. Navy submarine reactors. The sanitary wastewater treatment plant (WWTP), power plant, and support buildings were also constructed at that time to support AEC activities. AEC manufacturing and research and development activities were terminated by AEC by 1962.

From 1956 to 2000, ABB was involved in the research, development, engineering, production, and servicing of nuclear and fossil fuel systems. These activities were performed under both commercial and federal contracts. Projects included nuclear and combustion research for commercial use, as well as large-scale boiler test facilities and coal gasification. Nuclear fuel research and development and reactor plant outage servicing was conducted in Buildings 2, 5, and 17. Fuel components were manufactured in Building 17. Buildings 3 and 6 were designed and built (and initially used) for Naval nuclear fuel manufacturing at the Site. Subsequently, large-scale fossil fuel boiler tests were conducted in Building 3. Wastewater pumping and dilution was conducted in Building 6.

2.2 EXISTING KNOWLEDGE OF BUILDING'S RESIDUAL RADIOACTIVITY PROFILE

Based on the review of historical records, process knowledge, and the results of radiological surveys at the Site, the residual radioactivity potential for the buildings at the Site can be isolated to two credible source terms. The first is uranium series radionuclides associated with nuclear fuel manufacturing and research. The second potential source term is that associated with nuclear power plant outage support services (reactor byproduct series). Radionuclides in this category consist almost exclusively of the longer-lived isotopes of reactor activation products dominated by the radioactivity associated with cobalt 60 (Co-60). Based upon the results of sampling and analysis, it is evident that isotopes associated with enriched uranium (EU) varying in enrichment from approximately 3% to 90% are the predominant radioisotopes found at the Site in general.

2.2.1 Radiological Characteristics of Enriched Uranium

The suite of radionuclides found in EU is fixed by the physical and chemical processes used to produce the EU and by the laws of physics describing radioactive decay. The same physical laws govern the relative concentrations of these radionuclides, making their proportions at a given U-235 mass enrichment known with a high degree of certainty. Isotopically, EU does not vary substantially by batch for a given percent enrichment, assuming that the EU was produced using the same enrichment (isotope separation) process. The EU fuel stock used at the Site is known to have come from gaseous diffusion enrichment processes. There is no indication that EU fuel stock derived from other enrichment processes (e.g., centrifuge, laser) was ever used at the Site.

The uranium enrichment percentage does impact the relative concentrations of the uranium isotopes in the fuel mixture. In gaseous diffusion, the smaller U-234 atoms are more readily enriched than either U-235 or U-238 atoms. Likewise, U-235 atoms are more readily enriched than U-238 atoms. Uranium 234, having a radioactive half-life more than three orders of magnitude shorter than U-235, dominates the total uranium radioactivity for enrichments greater than naturally occurring abundances. The relationship between the uranium isotopes for enriched nuclear fuels created by gaseous diffusion can be described by Equation 2–1 (NRC 1974) and is illustrated for a wide array of enrichments in Figure 2-1 (DOE 2000).

$$SA = (0.4 + 0.38E + 0.0034E^2)10^{-6} \qquad (2\text{-}1)$$

Where: SA = specific activity of enriched uranium in Ci per gram

E = % U-235 by weight

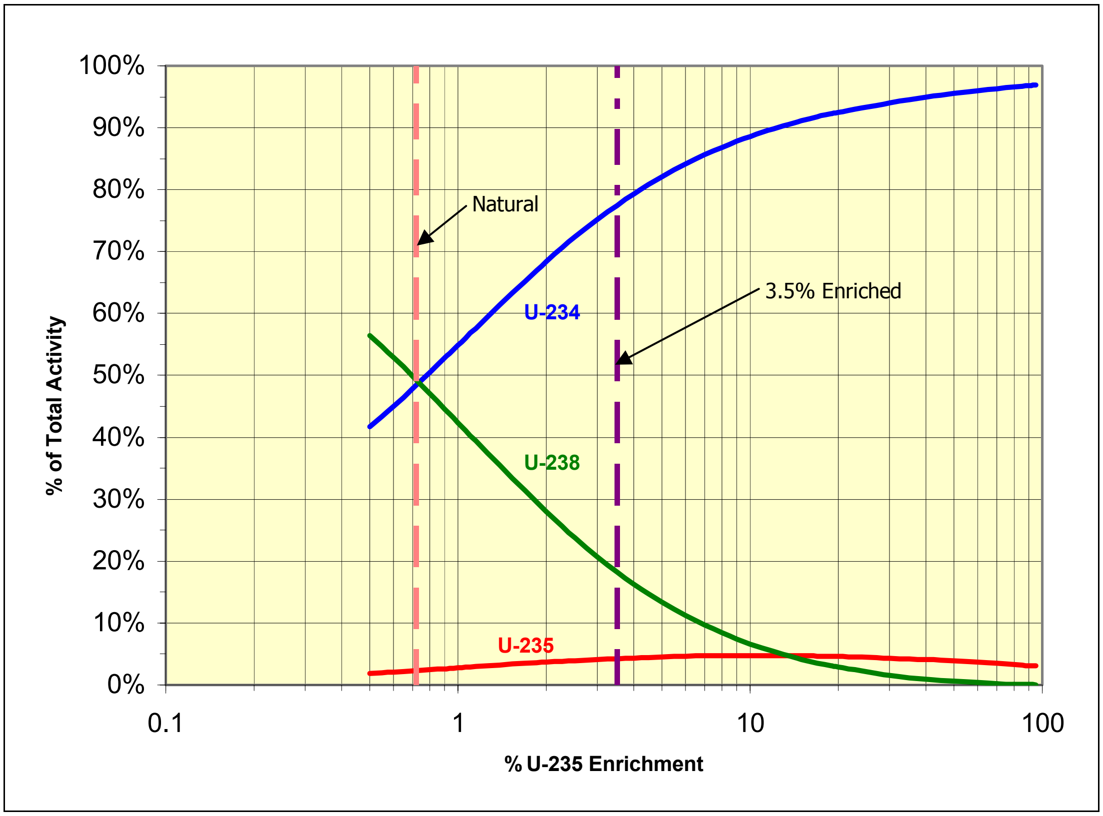

Figure 2-1. Relative Uranium Isotopic Concentration vs. Enrichment

The Historical Site Assessment (HSA) (Harding 2002) indicates that some depleted and natural uranium materials were present and used for research and development on Site. However, quantities of these materials used on Site are overwhelmed by the amount of EU used. Typical commercial grade, low enrichment uranium (LEU) fuel stock was supplied to CE at about 3.5% enrichment. Highly enriched uranium (HEU) with enrichments greater than 90% was used to manufacture special nuclear fuels for the Federal government. Thus, a range of potential uranium isotopic ratios might occur on the Site and could vary from one location to another depending upon enrichment of the material.

2.2.2 Radiological Characteristics of Byproduct Materials

Radionuclides produced in the operation of a nuclear reactor are classified as byproduct materials, as they are the "byproduct" of a nuclear reaction. There are two subcategories of isotopes collectively classed as byproduct materials. They are described by their production mechanisms: 1) fission products, and 2) activation products. The nuclear fuel services work performed by CE and later by Westinghouse at the Site involved the repair, maintenance, and testing of reactor plant components. Since nuclear fuel itself is clad, or jacketed, to prevent a significant release of fission products, the principle radionuclides associated with plant components handled at the Site are activation products.

Isotopes found in byproduct materials are generally characterized by short half-lives and beta decay mechanisms. The shortest-lived isotopes rapidly decay away and are essentially gone before components can be removed from a reactor plant for service. After one year, only a small number of the longest-lived radionuclides remain in potentially significant quantities.

The presence of byproduct radioactivity in the remaining radiologically impacted buildings at the Site is limited to primarily Building 6. Characterization data from the soil areas and remaining buildings along with data and experience from previous decommissioning activities associated with commercial operations facilities at the Site have clearly indicated that the predominate byproduct radionuclide associated with buildings is Co-60. In addition, evaluation of the waste stream profiles associated with nuclear facility support work identifies only five radionuclides that contribute 1% or more to the total byproduct radioactivity as shown in Table 2-1.

Table 2-1. Byproduct Material Radionuclide Profile (>1%)

Radionuclide	***% of Total Activity***
Co-60	73.2%
Cs-137	1.1%
Fe-55	2.9%
Mn-54	2.7%
Sb-125	5.0%

The radioactive decay of Co-60 results in the emission of both beta and gamma radiation. The mean beta energy is 96.41 keV with emission intensity of 100% and gamma radiation of 1173 keV along with 1332 keV, both with emission intensity of 100%. For the development of building DCGLs the byproduct material source term will consist of 100% beta and gamma radiation emissions.

3.0 DEVELOPMENT OF THE DERIVED CONCENTRATION GUIDELINE LEVEL

3.1 SELECTION OF THE ANNUAL PUBLIC DOSE LIMIT

The NRC's governing decommissioning and license termination regulation, 10 CFR 20.1402 (NRC 1997), limits radiation dose contribution to the average member of the critical exposure group (members of the exposed public) to no more than 25 millirem (mrem) in any single year following license termination. This criterion requires that all non-background sources of radiation and all credible and complete exposure pathways be considered in demonstrating compliance with the decommissioning dose limit.

While the State of Connecticut does not currently have a statute in place specifically addressing the decommissioning of a site that has operated under a nuclear materials license (or other authority as described in the Atomic Energy Act 1954, as amended by the Energy Reorganization Act 1974), CTDEP has communicated its intent to regulate sites having residual radioactive material to concentrations that would be protective of a total annual dose of 19 mrem (CTDEP 2002). Additionally, the State's Attorney General's Office has determined that the provisions of the Connecticut Transfer Act (CTAG 2003) together with the standards adopted by the State in regulation (CTDEP 1996) provide a legally enforceable basis for the State's regulatory authority to approve DCGLs for residual radioactivity in soils following decommissioning. The State of Connecticut has stipulated that post-decommissioning annual radiation doses should be limited to 19 millirem per year.

In deference to the more conservative Connecticut limit, ABB has selected an annual public dose limit of 19 millirem total effective dose equivalent (TEDE) as the dose basis for deriving the Building DCGLs. By such, both the NRC and State dose limits are satisfied with a single set of building surface DCGLs.

3.2 DOSE-CONCENTRATION RELATIONSHIP

The process to correlate a radioactivity concentration to dose can proceed after the annual public dose limit has been established. As in any health risk assessment, the process involves defining the source(s), the pathways for potential exposure to each source, and the availability of a receptor to receive a dose. The relationships between factors involved in defining the mechanisms for exposure are complex and often mutually dependent. The aid of a computer program to model the plausible exposure scenarios and to perform complex sets of computations is warranted.

3.2.1 RESRAD-BUILD Computer Modeling Code

The model selected to evaluate the exposure and dose potential to individuals from residual radioactive materials in the buildings at the Site was selected on the basis of its representation of the conditions being evaluated and its acceptance within the regulatory and health physics communities as an effective and suitable model. The temptation in using models of any description to predict the potential future exposure conditions associated with an actual site is to ascribe or imply some measure of "accuracy" to the results it provides. In reality, it is difficult to effectively measure the accuracy of any model. It is principally because accurate direct measurement within a reasonable time frame cannot be made that a model is used to make a prediction in the first place. It is the selection of the model that most closely approximates the scenario to be evaluated and the use of realistic and plausible input parameters describing the exposure scenario that determines the confidence one has about the modeled results. For these reasons, ABB chose the computer modeling code *RESRAD-BUILD*, Version 3.4[1] (Yu 2007) to perform the calculations relating annual dose to residual radioactivity concentrations on building surfaces.

The *RESRAD-BUILD* code has been in use for several years and has been extensively employed for applications such as the ABB Site building DCGL derivation. *RESRAD-BUILD* is unique as a tool for evaluating exposure from building contamination in that it is the only model available that allows for consideration of specific source geometry. The NRC has approved the use of *RESRAD-BUILD* for demonstrating compliance with the license termination criteria for its licensees.

The *RESRAD-BUILD* computer modeling code is a pathway analysis tool designed to evaluate the potential radiological dose incurred by an individual who works or lives in a building having residual radioactive material. The code was developed by Argonne National Laboratory for the U.S. Department of Energy (DOE).

1. Version 3.4 is most current available version of the RESRAD-BUILD code. Version 3.4 was released on December 20, 2007, and incorporates the most up-to-date dose conversion factors and exposure parameters, as accepted and approved by the International Council on Radiation Protection (ICRP 1990, ICRP 1994).

The transport of radioactive material inside a building from one compartment (or room) to another is calculated with an indoor air quality model that is integral to the overall model. The air quality model considers the transport of radioactive dust particles and radon progeny due to air exchange, deposition and resuspension, and radioactive decay and ingrowth. Seven exposure classes corresponding to the seven exposure pathways are calculated in the *RESRAD-BUILD* code (Figure 3-1):

External exposure directly from the source.

External exposure to radioactive materials deposited on the floor.

External exposure due to air submersion.

Inhalation of airborne radioactive particulates.

Inhalation of aerosol indoor radon progeny.

Ingestion of radioactive material directly from the source surface.

Ingestion of radioactive material from materials redeposited on surfaces within the building.[2]

2. The two ingestion sub-pathways are mathematically combined in the RESRAD-BUILD modeling program, yielding a single dose contribution from the combined ingestion pathways.

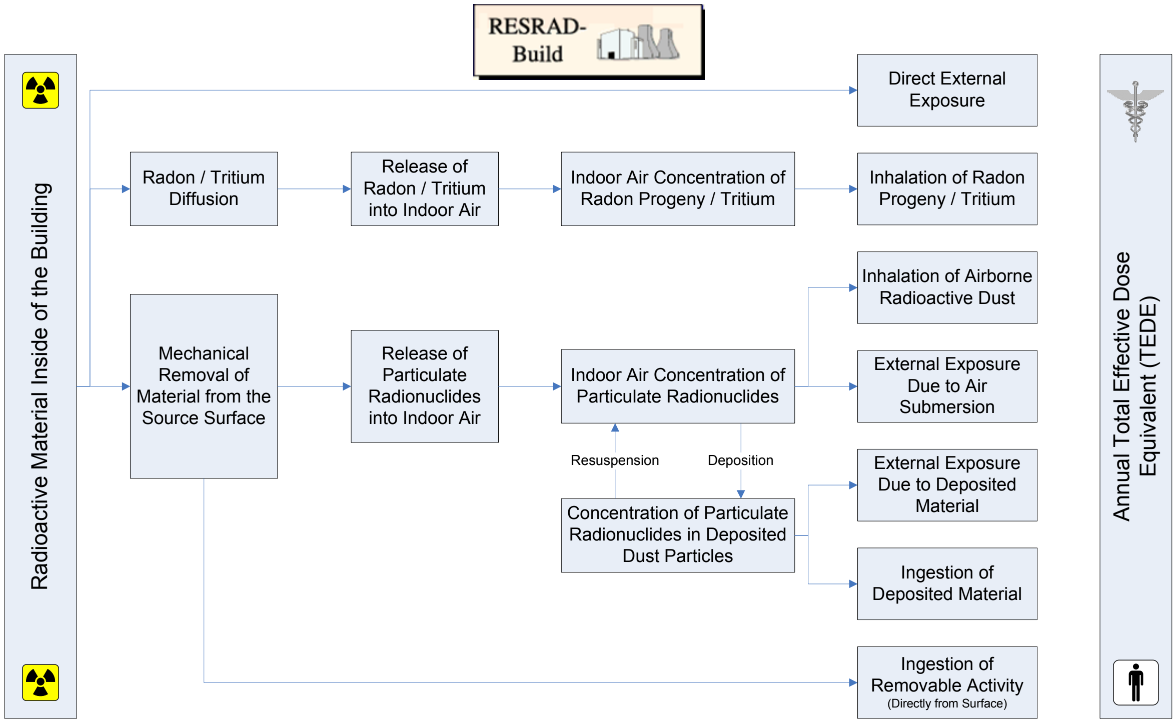

Figure 3-1. RESRAD-BUILD Conceptual Exposure Pathway Model

The *RESRAD-BUILD* code was used to evaluate the potential future doses from exposure during occupancy and demolition scenarios. Detailed modeling code input/output reports and related modeling technical information can be found in Appendices (described in greater detail in subsequent sections of this report). These exposure scenarios were "constructed" by adjusting the input parameters of the model to match or conservatively approximate the expected or site-specific conditions for the scenario. For each scenario, the concentration input parameter was iteratively varied to arrive at a concentration below which the potential radiological dose (TEDE) to a member of the public is less than 19 mrem in 1 year.

The exposure models portrayed in the computer code are quite conservative, but sufficiently represent the potential future exposure cases. The conservatism embodied in the exposure models result in conservative correlations between dose and concentration. Factors affecting the mechanisms for, and intensity of, exposure were identified, and appropriate values were assigned. The following section outlines the potential future exposure scenarios identified as plausible and used to develop the Building DCGLs.

3.3 RADIONUCLIDE SOURCE TERM

To appropriately calculate the dose resulting from residual radioactivity on the building surfaces, two different source terms were evaluated:

> Enriched Uranium - owing to the site's long and predominant history as a nuclear fuel manufacturing facility; and
>
> Co-60 - owing to the potential, albeit small, for the presence of trace amounts of reactor byproduct isotopes.

3.3.1 Enriched Uranium Source Term

Because reactor fuel with a variety of enrichments has been manufactured at the site, it was necessary to evaluate the variance in dose consequences from exposure to residual radioactivity in the buildings having variable enrichments of U-235. The dose-versus-enrichment relationships demonstrated in the report documenting the derivation of the soils DCGLs does not apply to exposures to radioactivity on building surfaces because the exposure pathways dominating the dose response relationship for exposures in the building are different. For the Building DCGL calculations, a uranium source term with 3% enrichment (Figure 3-2) and another with 90% enrichment (Figure 3-3) were considered.

The dose conversion factors (DCF) in the isotope library used in *RESRAD-BUILD* assumes that progeny isotopes with radioactive half-lives less than 180 days are in secular equilibrium with their parent (an isotope with a half-life greater than 180 days). Consequently, the dose contributions from the short-lived progeny of the long-lived uranium nuclides (e.g., Pa-234, Th-234, etc.) are automatically included in the calculations. In addition, *RESRAD-BUILD* automatically calculates the ingrowth concentrations of the longer-lived progeny (e.g., Th-230, Ra-226, etc.) and accounts for the dose contributions from these nuclides.

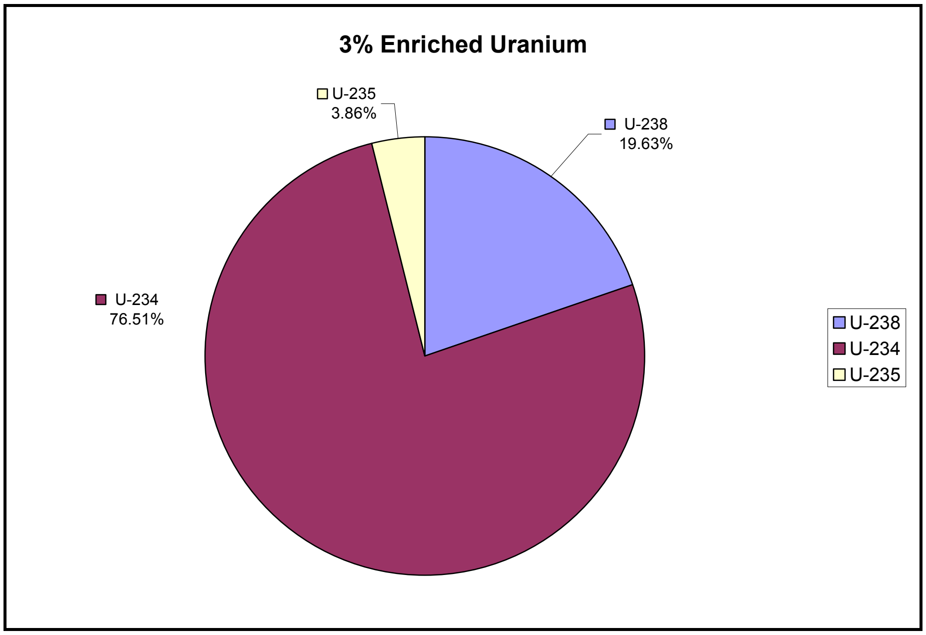

Figure 3-2. Uranium Source Term (3% Enriched)

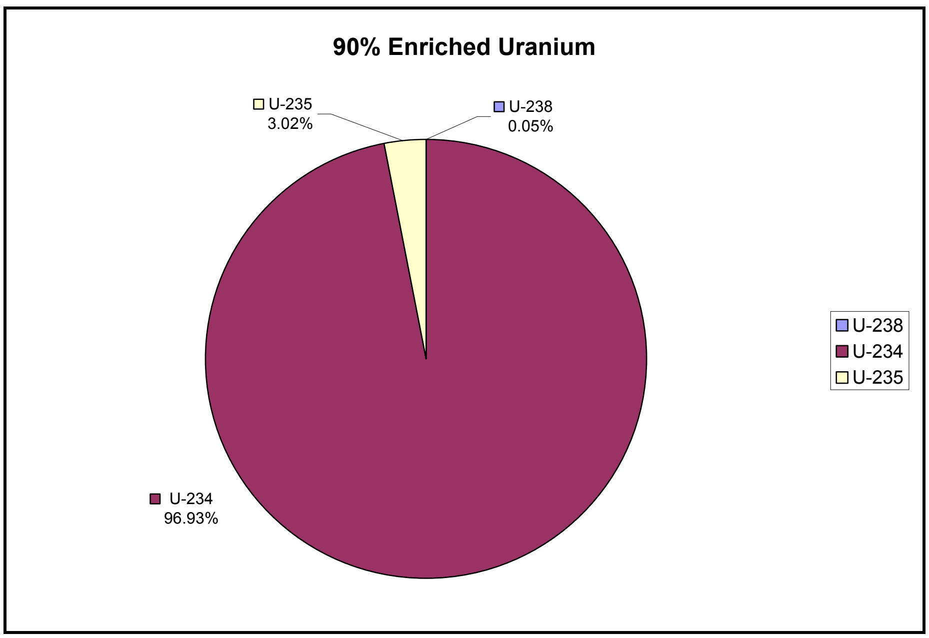

Figure 3-3. Uranium Source Term (90% Enriched)

3.3.2 Reactor Byproduct Source Term

Reactor byproduct sources are composed of a large number of radionuclides. However, most of these are either:

very short-lived and have already decayed away such that they are not present in discernable concentrations, or
they emit very weak radiation signals (so-called "hard-to-detect" isotopes) and, consequently, do not contribute measurably to radiation dose even if they are present in discernable concentrations.

This is the case with the byproduct source term (present in a few areas) at the site. In fact, after considering the array of radionuclides potentially present in the reactor byproduct source term at the Site, it was concluded that Co-60 is the only nuclide with a measurable dose contribution. (Co-60 produces more than 99% of the potential dose from exposure to the byproduct radionuclide mix present at the Site.) As a result, the derivation of the DCGL for the byproduct source term is based on the dose contribution from Co-60.

3.4 POTENTIAL FUTURE USE EXPOSURE SCENARIOS

The potential future use scenarios considered for the industrial buildings at the site are:

Re-use as an industrial / warehousing facility
Re-use as a commercial office building
Construction workers exposed during a remodel
Construction workers exposed during building demolition

In considering these scenarios, it is apparent that radiation exposure resulting from occupancy of the building in a commercial setting would be higher for the receptor with the longest exposure duration. Office workers would be expected to occupy the building for longer periods of time over the course of a year (on average) than would workers engaged in industrial or warehousing activities. Therefore, the first two scenarios were combined and exposure factors describing the office worker receptor were used.

Likewise, radiation exposure to a construction worker resulting from the demolition of the building would likely exceed that received by workers engaged in remodeling activities since demolition is likely to be far more invasive than remodeling. Therefore, the last two scenarios were combined and exposure factors describing the demolition construction worker receptor were used.

Two discrete scenarios emerged as likely future exposure scenarios to be evaluated for their dose-producing potential: Occupancy of the building for commercial use, and demolition of the structure at the conclusion of its useful lifetime (Figure 3-4). Ultimately, the scenario yielding the smallest activity concentration (i.e., the smallest dose-based concentration level) is selected as the limiting scenario, and the DCGL is selected to be protective of the limiting scenario. (Note: The limiting scenario for uranium contamination is the demolition scenario, while the limiting scenario for Co-60 is the occupancy for commercial use scenario.)

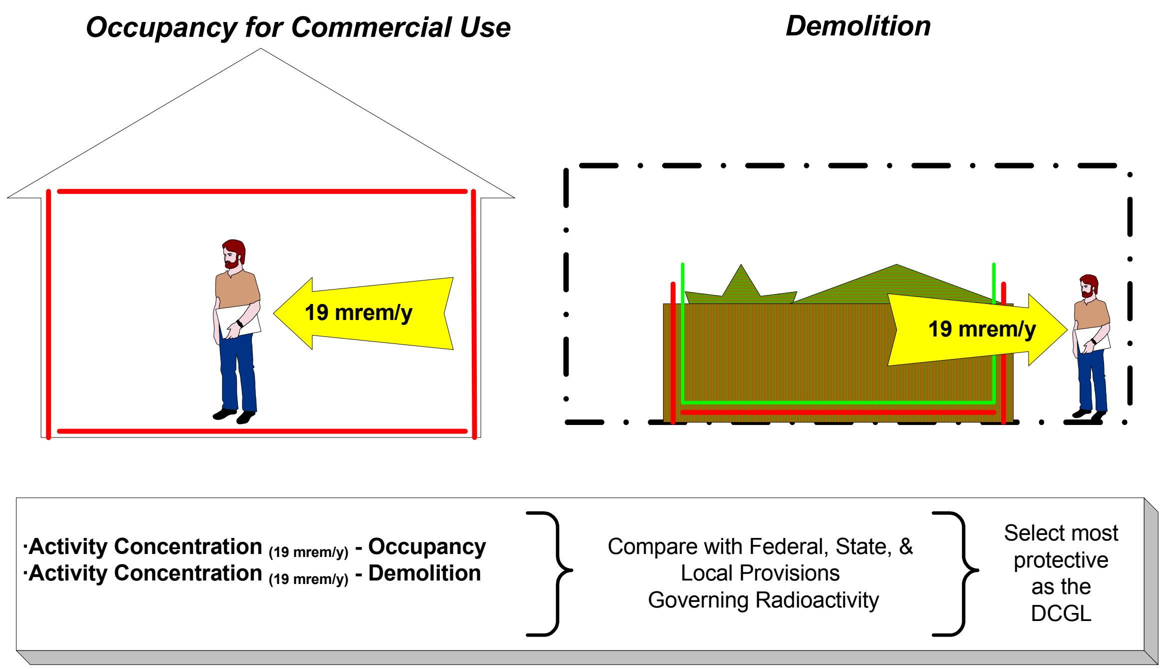

Figure 3-4. Conceptual Three Exposures for Building 20

3.5 COMMERCIAL OCCUPANCY OF THE BUILDING SCENARIO

3.5.1 Exposure Pathways

Typical computer dose modeling software allows the user to select (turn "on" or "off") the exposure pathways that are judged to be credible or complete. Table 3-1 identifies the pathways that have been retained for the analysis of the Commercial Occupancy scenario. *RESRAD-BUILD* does not provide operator control over the selection of pathways to be included in the analysis, but it does itemize the dose contributed by pathway, thereby allowing the dose modeler to exclude or discount pathways that are judged to be incomplete. For this analysis, ABB has retained all of the pathways and the dose produced in each pathway.

Table 3-1. Evaluation of Pathways for the Commercial Occupancy Scenario

Pathway	Retained	Remark
Direct External Exposure	✔	The source term found in the building produces gamma radiation. Exposure from direct penetrating radiation is expected to be a significant contributor to the overall dose, particularly from the Co-60 source term.
Inhalation of Radon Progeny / Tritium	✔	While radon is specifically excluded from consideration in the decommissioning dose criteria, the radon pathway was assumed to be complete and viable as a progeny from the long decay of uranium.
Inhalation of Airborne Radioactive Dust	✔	Allowance is made for some fraction of the source being liberated and suspended in the breathing air of potential occupants. Exposure from the inhalation of airborne radioactive dust is expected to be a significant contributor to the overall dose, particularly from the uranium source term.
External Exposure Due to Air Submersion	✔	This pathway accounts for external penetrating dose delivered by radioactive material in suspension in the air.
External Exposure Due to Deposited Material	✔	This pathway accounts for external penetrating dose delivered by radioactive material that has been deposited on surfaces after having been suspended in air.
Ingestion of Deposited Material	✔	This pathway accounts for internal dose delivered by radioactive material that has been ingested after having been suspended in air and subsequently deposited on surfaces.
Direct Ingestion of Removable Activity	✔	This pathway accounts for internal dose delivered by radioactive material that has been ingested after having been transferred directly from the source surfaces.

3.5.2 Source and Receptor Geometry

Figure 3-5 and Figure 3-6 schematically illustrate the modeled layout of the sources for the wall and floor surface source set (six sources) and the positions of two potential receptors within the building. This conceptualization, although simplified from the actual configuration, is considered sufficient (conservatively) to evaluate dose-based concentration levels. The sizes, positions, and orientations of the sources and receptors modeled are provided in Appendix A.

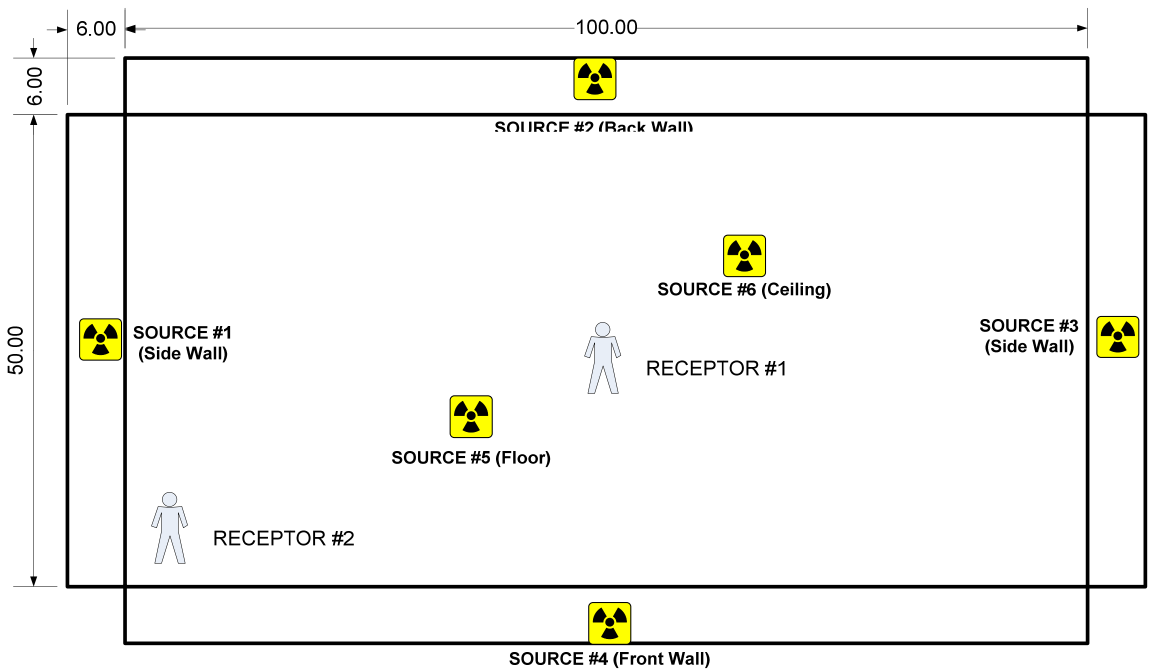

Figure 3-5. Conceptual Diagram of Source and Receptor Geometry (Unfolded View)

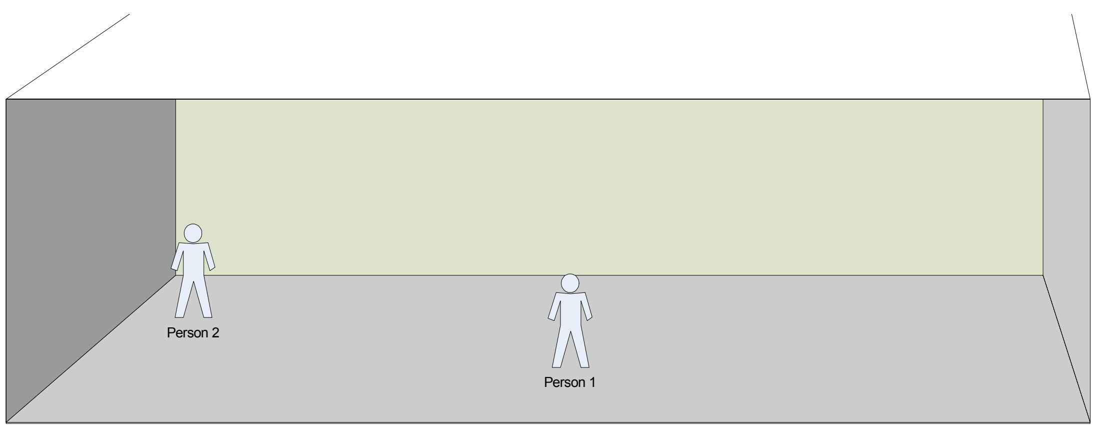

Figure 3-6. Conceptual Diagram of Source and Receptor Geometry (Visual Perspective)

3.5.3 Exposure Factors

The Commercial Occupancy scenario involves typical workday and work place exposure factors. The specific parameters used to define the scenario are almost exclusively the default parameters for the code which represents a very conservative case. Where appropriate, some of these parameters have been assigned values that are more conservative than the default values while others have been set with more realistic, albeit less conservative, values than the default values. Overall, the exposure factors used to define the Commercial Occupancy scenario are judged to be quite conservative. A table detailing each of *RESRAD-BUILD* input parameters and the value assigned in each of the exposure scenarios evaluated is presented in Appendix B.

3.6 BUILDING REMODEL/DEMOLITION SCENARIO

Inevitably, the building will serve its useful life and be demolished. Additionally, it is credible that remodeling and repairs may occur during the remaining useful life. A host of possible scenarios within this phase of the building life cycle were entertained. These included simple tasks like replacing flooring material, moving interior partitions, and ultimate demolition of the building. A number of construction managers and engineers were consulted to gain an understanding of the demolition methods that might be employed in razing the building. Each projected that the disposal method would almost certainly involve using heavy equipment to collapse the building and then crush the debris (e.g., a track hoe "walking" over the collapsed structure). The use of explosives to initially collapse the building was ruled out on the grounds that the structure could be easily razed without explosives and the cost to collapse the structure using explosives would be prohibitive.

Other plausible demolition methods were far less intrusive than that described above and more closely approximated an extensive remodel. To evaluate exposures associated with these events, a single conservative exposure scenario will be considered. While a remodel may occur over a longer period of time than the actual demolition is expected to take, the amount of source term that might be accessed during a typical remodel is substantially less than that accessed during the demolition envisioned. Therefore, any credible remodel scenario will produce a smaller dose to a receptor than the demolition scenario evaluated herein.

3.6.1 Exposure Pathways

Table 3-2 identifies the pathways that have been retained for the analysis of the Building Remodel/Demolition scenario. *RESRAD-BUILD* does not provide operator control over the selection of pathways to be included in the analysis, but it does itemize the dose contributed by pathway, thereby allowing the dose modeler to exclude or discount pathways that are judged to be incomplete. For this analysis, ABB has retained all of the pathways and the dose produced in each pathway.

Table 3-2. Evaluation of Pathways for the Building Remodel/Demolition Scenario

Pathway	Retained	Remark
Direct External Exposure	✔	The source term found in the building produces gamma radiation. Exposure from direct penetrating radiation is expected to be a significant contributor to the overall dose, particularly from the Co-60 source term.
Inhalation of Radon Progeny / Tritium	✔	While radon is specifically excluded from consideration in the decommissioning dose criteria, the radon pathway was assumed to be complete and viable as a progeny from the long decay of uranium.
Inhalation of Airborne Radioactive Dust	✔	Allowance is made for some fraction of the source being liberated and suspended in the breathing air of potential occupants. Exposure from the inhalation of airborne radioactive dust is expected to be a significant contributor to the overall dose, particularly from the uranium source term.
External Exposure Due to Air Submersion	✔	This pathway accounts for external penetrating dose delivered by radioactive material in suspension in the air.
External Exposure Due to Deposited Material	✔	This pathway accounts for external penetrating dose delivered by radioactive material that has been deposited on surfaces after having been suspended in air.
Ingestion of Deposited Material	✔	This pathway accounts for internal dose delivered by radioactive material that has been ingested after having been suspended in air and subsequently deposited on surfaces.
Direct Ingestion of Removable Activity	✔	This pathway accounts for internal dose delivered by radioactive material that has been ingested after having been transferred directly from the source surfaces.

3.6.2 Source and Receptor Geometry

Figure 3-7 and Figure 3-8 schematically illustrate the modeled layout of the sources for the wall and floor surface source set (six sources) and the positions of two potential receptors within the building. This conceptualization, although simplified from the actual configuration, is considered sufficient (conservatively) to evaluate dose-based concentration levels. The sizes, positions, and orientations of the sources and receptors modeled are identical to that modeled for the occupancy scenario. This represents a significant conservatism in the geometry of the modeled demolition scenario.

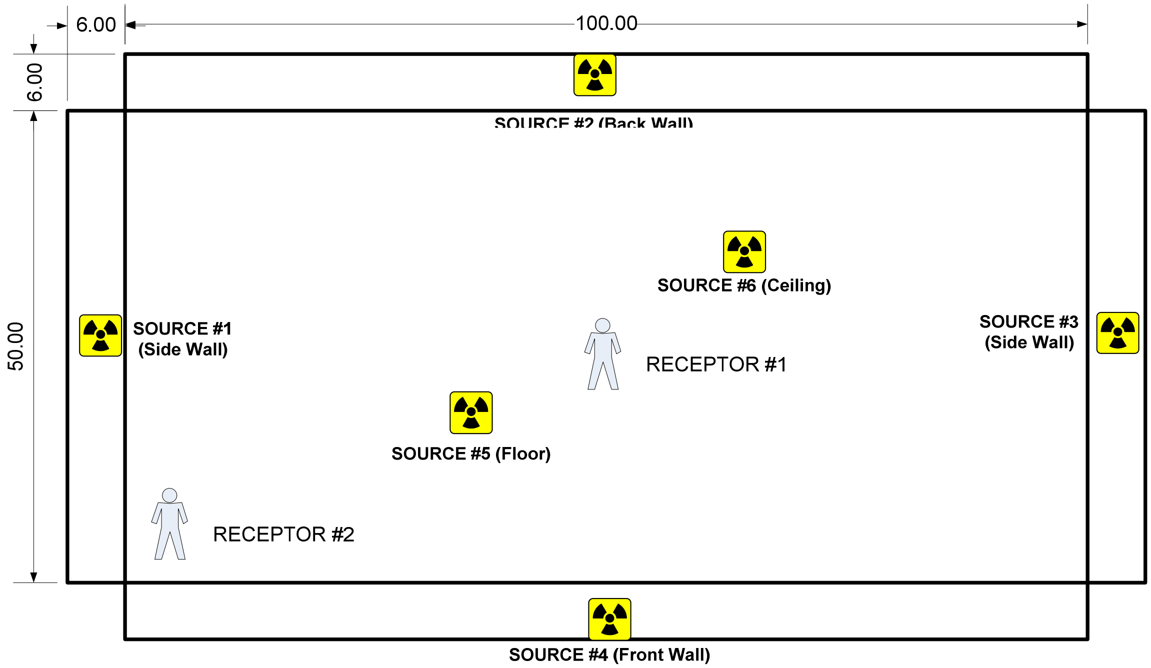

Figure 3-7. Conceptual Diagram of Source and Receptor Geometry (Unfolded View)

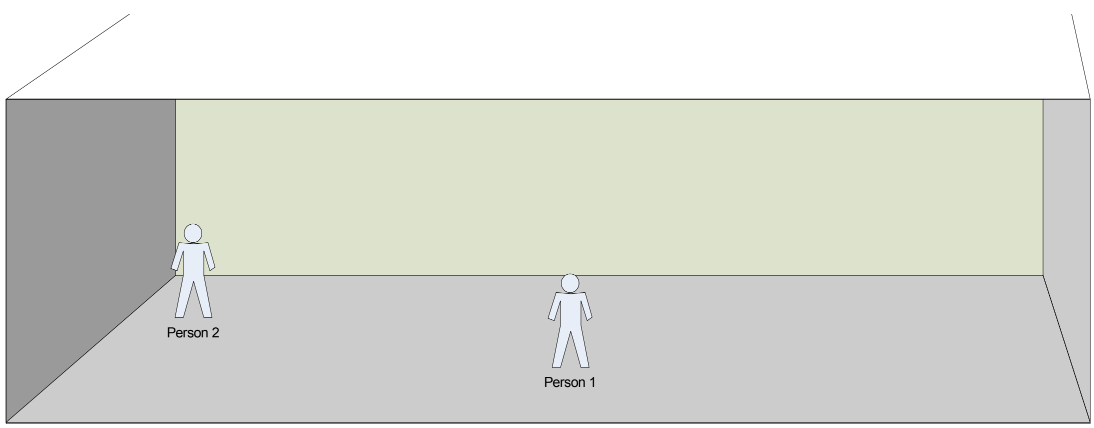

Figure 3-8. Conceptual Diagram of Source and Receptor Geometry (Visual Perspective)

3.6.3 Exposure Factors

The Building Remodel/Demolition scenario involves typical construction workday and work place exposure factors. The specific parameters used to define the scenario are almost exclusively the default parameters for the code which represents a very conservative case. Where appropriate, some of these parameters have been assigned values that are substantially more conservative than the default values in *RESRAD-BUILD*. In general, the demolition scenario is designed to reflect conditions that are significantly dustier and dirtier than the occupancy scenario. Increasing the factors influencing mechanisms responsible for generating and sustaining dust and limiting atmospheric dilution simulates these conditions. Constraining workers to within the limited dilution volume tends to intensify the exposure and increase potential doses. Overall, the exposure factors used to define the Building Remodel/Demolition scenario are judged to be quite conservative. A table detailing each of *RESRAD-BUILD* input parameters and the value assigned in each of the exposure scenarios evaluated is presented in Appendix B.

3.7 CONSERVATISM IN THE DCGL MODELING

In configuring the models used in the derivation of the building surfaces DCGLs, ABB strove to use conservative values for input parameters that affect the concentration-to-dose relationships. This was done with the understanding that it would likely result in the overestimation of receptor doses and, ultimately, a lower but more certainly protective DCGL. In this manner, the inherent uncertainty associated with the use of a model is regulated and directed by the biased selection of conservative input parameters so that it is unlikely that receptor dose will be greater than the model estimates. The major bounding assumptions responsible for producing conservative bias are identified in Table 3-3 with a qualitative remark indicating the affect on receptor dose estimates.

An important factor to consider in assessing the conservative bias in the selection of the DCGL is the assumption that the source concentration is uniformly distributed over the entirety of the interior surfaces of the building. This is an extremely unlikely scenario used to bound the dose potential and derives a maximum permissible concentration or DCGL. It is expected that far less than 10 percent of the building surfaces have measurable radioactivity and that the average concentrations over the surfaces that have residual concentrations will be substantially less than the source concentrations necessary to result in 19 mrem of dose annually.

A review of Table 3-3 indicates that the major assumptions affecting receptor dose estimates (source strength, pathway migration, and receptor exposure characteristics) have been overstated to ensure a conservative bias. Risk management decisions based upon the results of these modeling estimates will reflect this conservative bias and likely err on the side of safety.

Table 3-3. Major Sources of Conservatism in the Derivation of the DCGLs

Source of Conservatism	Remark
RESRAD-BUILD: Assumes entire indoor building surfaces contaminated at dose-based concentration level.	Preliminary investigation data collected in the building suggests that the source concentration (amount of radioactivity) used in the model is substantially higher than existing concentrations. This potential overestimation of the source concentration is in part the result of using the model to derive a maximum allowable surface concentration level rather than to model the actual exposure conditions in the building. An overestimated source concentration overstates receptor doses from all pathways.
RESRAD-BUILD: Continued use air exchange rate assumed to be 1 change per hour is well below engineering estimates for a high bay building and well below the ASHREA requirements for occupancy of a commercial building.	Underestimated ventilation rates overstate receptor doses from inhalation, ingestion of dust, radon and submersion.
RESRAD-BUILD: Removable Fraction 10% (0.1) over 20 years in the occupancy scenario and 2.5% (0.025) in 60 days in the demolition scenario.	Conservative estimates of the “removable” fraction will tend to overestimate the inhalation dose. Inspection of the building and extensive wipe sampling indicate that the residual radioactive material that is present in the building is firmly fixed in place and not readily removable.
RESRAD-BUILD: Air Release Fraction (0.1) in the occupancy scenario and (0.25) in the demolition scenario.	Conservative estimates of the fraction of erodible material that is liberated to the air will tend to overestimate the inhalation doses. The building is not dusty and there is no evidence of appreciable aerosol generation or suspension. The default value (a substantially conservative value) of 0.1 was chosen for the occupancy scenario. A value two and one half time more conservative than the default value (0.25) was chosen for the demolition scenario accounting for the greater likelihood that surface contaminants might be liberated directly into the air space.
RESRAD-BUILD: The Demolition scenario conservatively assumes a mixing volume for atmospheric dispersion equal to the volume of the building itself and no dust emission abatement.	Constraining the mixing volume to the volume of the building (the most conservative assumption that can be made) significantly limits the actual atmospheric dilution that would likely occur in a much larger volume. Additionally, no credit from standard demolition wet dust control methods is assumed.
RESRAD-BUILD: Assumes reasonable maximum exposure (RME) factors.	Assuming RME receptor characteristics is conservative and consistent with EPA guidance (EPA 1989). Coupling RME receptor characteristics with the potentially overstated source term and conservative exposure pathway assumptions results in highly conservative dose estimates.

4.0 RESULTS OF COMPUTER MODELING

In order to evaluate the average derived dose-based concentration, the computer modeling codes were run iteratively for each of the selected scenarios to arrive at the maximum uniform (average) concentrations that yields a total dose of 19 mrem to a single receptor (representing the average member of the critical exposure group) in 1 year. The computer code was set up to model each scenario with the input parameters identified and explained in Section 3.0. The following sections present the results of the computer modeling relating source concentrations with potential future doses in each of the two scenarios evaluated.

4.1 CONNERCIAL OCCUPANCY SCENARIO

Three different source terms - 3% enriched uranium, 90% enriched uranium, and reactor byproduct (Co-60) - were evaluated and modeled in the context of the commercial occupancy scenario. Each of these source terms yields a different concentration-to-dose relationship from which the DCGLs are derived. Detailed dose modeling reports are provided in Appendix C.

Table 4-1 summarizes the results of computer modeling for each of these source terms in the commercial occupancy scenario and yields the total source concentration input to the modeling code that corresponds with a TEDE of 19 mrem per year.

Table 4-1. Summary Results of Dose Modeling—Commercial Building Occupancy Scenario

Source Term	Isotope	Concentration Modeled (dpm/m^2)	Equivalent Activity (dpm/100cm^2)	Peak Annual Dose
3% Enriched Uranium	U-234	5.71E+06	124,429 (Total α/β)	19 mrem
	U-235	2.88E+05		
	U-238	1.47E+06		
90% Enriched Uranium	U-234	7.38E+06	80,865 (Total α/β)	19 mrem
	U-235	2.30E+05		
	U-238	3.57E+03		
Reactor Byproduct	Co-60	6.98E+05	6,980 (Total β)	19 mrem

As indicated by the results of dose modeling summarized in Table 4-1, 90% enriched uranium is somewhat more potent as a dose producer in the commercial occupancy scenario than is 3% enriched uranium. A review of the modeling results reveals that inhalation is by far the principal exposure pathway for the uranium source term (illustrated in Figure 4-1). For the reactor byproduct source term, the annual TEDE is dominated by the external gamma dose pathway (Figure 4-2).

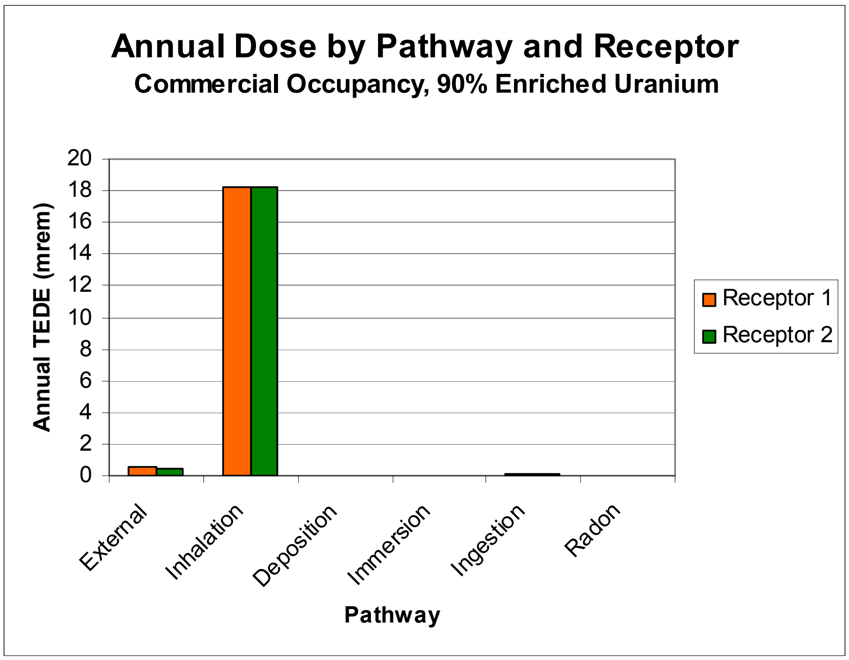

Figure 4-1. Pathway Contributions to Annual Dose in the Occupancy Scenario (90% Enriched Uranium)

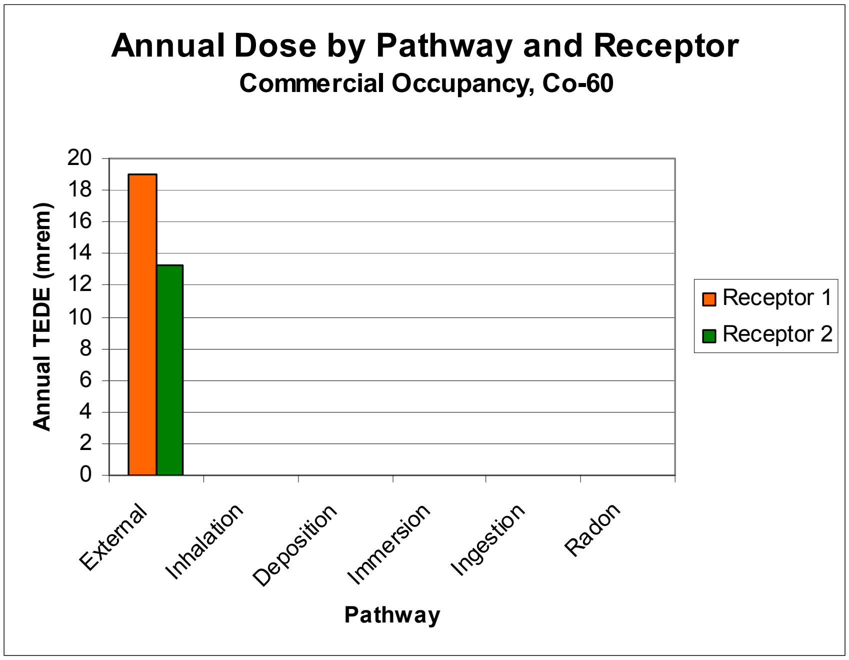

Figure 4-2. Pathway Contributions to Annual Dose in the Occupancy Scenario (Reactor Byproduct, Co-60)

4.2 BUILDING REMODEL / DEMOLITION SCENARIO

As with the commercial occupancy scenario, each of the three different source terms - 3% enriched uranium, 90% enriched uranium, and reactor byproduct (Co-60) - were evaluated and modeled in the context of the building remodel/demolition scenario. Each of these source terms yields a different concentration-to-dose relationship from which the DCGLs are derived. Detailed dose modeling reports are provided in Appendix C.

Table 4-2 summarizes the results of computer modeling for each of these source terms in the commercial occupancy scenario and yields the total source concentration input to the modeling code that corresponds with a TEDE of 19 mrem per year.

Table 4-2. Summary Results of Dose Modeling—Building Remodel / Demolition Scenario

Source Term	Isotope	Concentration Modeled (dpm/m^2)	Equivalent Activity (dpm/100cm^2)	Peak Annual Dose
3% Enriched Uranium	U-234	1.48E+06	32,229 (Total α/β)	19 mrem
	U-235	7.47E+04		
	U-238	3.80E+05		
90% Enriched Uranium	U-234	1.84E+06	20,148 (Total α/β)	19 mrem
	U-235	5.74E+04		
	U-238	8.88E+02		
Reactor Byproduct	Co-60	2.12E+06	21,205 (Total β)	19 mrem

Again, 90% enriched uranium is somewhat more potent as a dose producer in the building remodel/demolition scenario than is 3% enriched uranium. A review of the modeling results reveals that inhalation is by far the principal exposure pathway for the uranium source term (illustrated in Figure 4-1). For the reactor byproduct source term, the annual TEDE is dominated by the external gamma dose pathway (Figure 4-2).

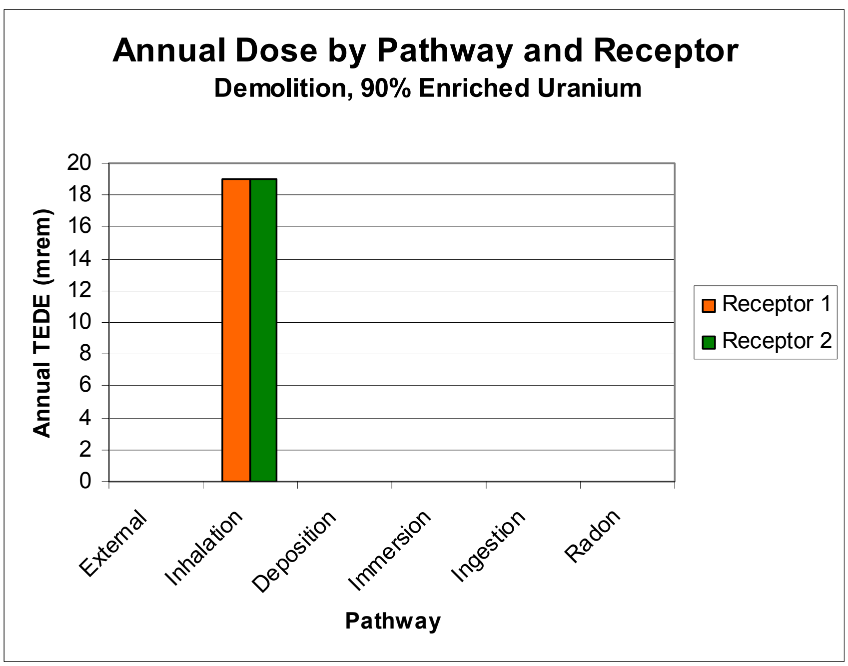

Figure 4-3. Pathway Contributions to Annual Dose in the Demolition Scenario (90% Enriched Uranium)

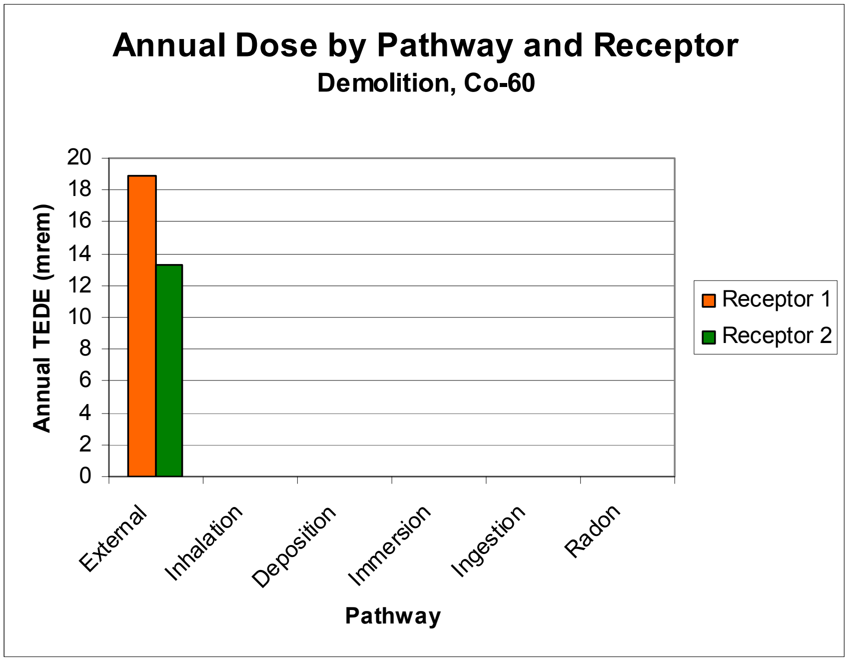

Figure4-4. Pathway Contributions to Annual Dose in the Demolition Scenario (Reactor Byproduct, Co-60)

4.3 SELECTION OF THE DCGLS FOR BUILDING SURFACES

Two distinct DCGLs are warranted; one for uranium and another for Co-60. As previously described, the building DCGLs are selected by comparing the concentration-to-dose relationships derived using the *RESRAD-BUILD* computer modeling tool.

4.3.1 Selection of the Uranium DCGL for Building Surfaces

The first comparison is *between* 3% enriched and 90% enriched uranium *within* a single scenario. This comparison is necessary, because it is difficult to actually measure enrichment in the field and because enrichment can vary from one location to the next depending upon a variety of factors. Comparing the dose producing potential of 3% and 90% uranium within the scenario allows the selection of a uranium DCGL that is protective of the annual dose limit, regardless of the enrichment percentage actually present in the field at the selected sample location. For both the commercial occupancy and building demolition scenarios, it has been shown that, for a given total uranium concentration, exposure rates increase as the enrichment increases. This is an artifact of the overwhelming dominance of the inhalation pathway for producing dose to receptors exposed inside the building. From this comparison, it is concluded that the DCGL for the uranium source term should be selected based on the models using 90% enriched uranium as their source term.

The second comparison is *between* exposure scenarios, each having been modeled with the 90% enriched uranium source term composition. This comparison allows the selection of the 90% enriched uranium source term concentration that is protective of the annual dose limit, regardless of the future use scenario a receptor might be exposed in. For the uranium source term, the demolition scenario proves to be the most limiting scenario. Both of these comparisons are presented in Table 4-3.

From this table, it is evident that a uranium DCGL based upon the building demolition scenario and the assumption that uranium activity present on the building surfaces is 90% enriched will be protective for each of the possible combinations evaluated. Therefore, the appropriate DCGL for building surfaces with residual uranium radioactivity is 19,547 dpm/100cm^2 α and 601 dpm/100cm^2 β for a total of 20,148 dpm/100cm^2, total α plus β activity.

Table 4-3. Comparison of Prospective DCGLs for the Uranium Source Term

Source Term (dpm/100cm^2, α + β)	**Scenario**	
	Commercial Occupancy	**Building Demolition**
3% Enriched Uranium	124,429	32,229
90% Enriched Uranium	80,865	20,148
Prospective DCGL values are expressed as total α plus β activity in units of dpm/100cm^2. Actual values used in the field will depend upon the type of measurement made but must be the radiological equivalent of these values.		

4.3.2 Selection of the Reactor Byproduct DCGL for Building Surfaces

A similar comparison *between* exposure scenarios, each having been modeled with the reactor byproduct (Co-60) source term was also performed. This comparison allows the selection of the Co-60 source term concentration that is protective of the annual dose limit, regardless of the future use scenario a receptor might be exposed in. This comparison is presented in Table 4-4. For the Co-60 source term, the commercial occupancy scenario proves to be the most limiting. Therefore, the appropriate DCGL for building surfaces with residual reactor byproduct radioactivity is 6,980 dpm/100cm^2, total β activity.

Table 4-4. Comparison of Prospective DCGLs for the Reactor Byproduct Source Term

Source Term (dpm/100cm^2, β)	Scenario	
	Commercial Occupancy	**Building Demolition**
Co-60	6,980	20,148
Prospective DCGL values are expressed as total β activity in units of dpm/100cm^2.		

5.0 SUMMARY AND RECOMMENDATIONS

The proposed $DCGL_W$s (average concentration uniformly distributed over all surfaces in the building) derived for the radiologically impacted building surfaces at the Site are:

Uranium: 20,148 dpm/100cm^2, total α plus β activity
Co-60: 6,980 dpm/100cm^2, total β activity

The Building DCGL evaluation described in this report provides risk managers and decision-makers with the substantive basis necessary to set and approve the building surfaces DCGLs, which are derived from the decommissioning dose limits and in consideration of the State of Connecticut's stipulated requirements.

The evaluation establishes a constrained post-decommissioning public dose limit of 19 mrem per year, from which the $DCGL_W$s are derived, satisfying the requirement that guidelines for release of real property from radiological controls be consistent with applicable Federal, State, and local requirements. Constraining the allowable dose to 19 mrem per year provides the risk managers with a built-in margin to safety, acknowledging that the NRC decommissioning dose limit is 25 mrem per year and the basic public dose limit is nominally set at 100 mrem per year. Additionally, the proposed DCGLs have been derived with a level of conservatism commensurate with the extent of the hazard and uncertainty in the estimation tools. Therefore, the use of a constrained dose limit coupled with the use of conservative techniques in deriving the DCGLs provides the risk managers with a very conservatively derived set of surface activity concentration guidelines that ensures protection of human health.