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3.0 IDENTIFICATION OF REMAINING SITE DISMANTLEMENT ACTIVITIES

3.1 INTRODUCTION

In accordance with 10 CFR 50.82 (a)(9)(ii)(B), the License Termination Plan (LTP) must identify the major dismantlement and decontamination activities. This chapter was written following the guidance of NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, and Regulatory Guide 1.179, *Standard Format and Content of License Termination Plans for Nuclear Power Reactors*, and will discuss those dismantlement activities as of December 31, 2002 [References 3-15 and 3-17]. Information is presented to demonstrate that these activities will be performed in accordance with10 CFR Part 50 and will not be inimical to the common defense and security or to the health and safety of the public pursuant to 10 CFR 50.82(a)(10). Information that demonstrates that these activities will not have a significant effect on the quality of the environment is provided in LTP Chapter 8, *Supplement to the Environmental Report*.

The information includes those areas and equipment that need further remediation and an estimate of radiological conditions that may be encountered. Included are estimates of associated occupational radiation dose and projected volumes of radioactive waste. These activities are undertaken pursuant to the current 10 CFR 50 license and are consistent with the Post Shutdown Decommissioning Activities Report (PSDAR).

Consumers Energy's primary goals are to decommission the Big Rock Point (BRP) Nuclear Plant safely and to maintain the continued safe storage of spent fuel in an Independent Spent Fuel Storage Installation (ISFSI). Consumers Energy will decontaminate and dismantle BRP in accordance with the DECON alternative, as described in NUREG-0586, Final Generic Environmental Impact Statement (FGEIS) [Reference 3-12]. Completion of the DECON option is contingent upon continued access to one or more low-level waste (LLW) disposal sites. Currently, BRP has access to LLW disposal facilities located in Barnwell, South Carolina, and in Clive, Utah. Chapter 1, *General Information*, of this LTP contains a description of the BRP Restoration Project Greenfield condition.

Consumers Energy is currently concluding decontamination and dismantlement (D&D) activities at the BRP site in accordance with the BRP PSDAR. Decommissioning activities are being coordinated with the appropriate Federal and State regulatory agencies in accordance with plant administrative procedures. In order to minimize the impact of ongoing decommissioning activities, a spent fuel pool island was established to separate spent fuel storage system safety functions from other decommissioning activities. By the end of the second quarter of 2003, it is expected that all special nuclear material and greater than Class C waste material will be located at the ISFSI.

Decommissioning activities at BRP shall be conducted in accordance with the BRP Updated Final Hazards Summary Report (UFHSR), Defueled Technical Specifications, Consumers Energy Quality Program Description (QPD) for

Nuclear Power Plants, existing 10 CFR Part 50 license, and the requirements of 10 CFR 50.82(a)(6) and (a)(7). If an activity requires prior NRC approval under 10 CFR 50.59(c)(2) or a change to the BRP Defueled Technical Specifications or license, a submittal shall be made to the NRC for review and approval before implementation of the activity in question.

Decommissioning activities are conducted under the BRP Radiation Protection Program, the Off-Site Dose Calculation Manual (ODCM), Safety Program, and Bulk Material Control Program. Such activities are and shall be conducted in accordance with these established programs that are frequently inspected by the NRC. Activities conducted during decommissioning do not pose any greater radiological or safety risk than those conducted during former plant operations. Decommissioning activity radiological risk is bounded by previously analyzed radiological risk for former operating activities that occurred during major maintenance and outage evolutions.

The activities described in Section 3.4, Future Decommissioning Activities, include activities up to the future release of the site, with the exception of facilities supporting ISFSI operations. This section provides an overview and describes the major remaining components of contaminated plant systems and, as appropriate, a description of specific equipment remediation considerations. Table 3-1 describes the decommissioning phases at BRP. Table 3-2 contains a list of major systems and components that have been or are to be removed. Table 3-5 contains a list of future decommissioning activities to be performed by Consumers Energy or Consumers Energy major component removal contractor.

Information related to the remaining D&D tasks is also provided. This information includes an estimate of the quantity of radioactive material to be released to unrestricted areas, a description of proposed control mechanisms to ensure areas are not recontaminated, estimates of occupational exposures, and characterization of radiological conditions to be encountered and the types and quantities of radioactive waste. This information supports the assessment of impacts considered in other sections of the LTP and provides sufficient detail to identify inspection or technical resources needed during the remaining dismantlement activities. Many of these dismantlement tasks require coordination with other federal, state or local regulatory agencies or groups. Big Rock Point's coordination with these agencies and groups is also described.

The dismantlement activities described in Section 3.4.3 provide the NRC the information to support site release and future license termination pursuant to 10 CFR 50.82(a)(11)(i). Therefore, this section was written to clearly indicate each dismantlement activity that remains to be completed prior to qualifying for license termination. The final state of the site will be a Greenfield (as defined in Chapter 1 of this LTP) with buried foundations and piping removed. Decommissioning activities performed will reduce residual radioactivity to a level that permits release of the property for unrestricted use.

3.2 DECOMMISSIONING UPDATE

Decommissioning activities were initiated following the decision to permanently cease BRP power operations on August 29, 1997. At that time, BRP performed evaluations of major plant structures, systems, and components (SSCs) to determine what function, if any, these SSCs would be expected to perform during the evolution to a decommissioned site. Each major plant SSC was evaluated to determine if the SSC, in its entirety or any portion thereof, was important for the safe storage of the spent fuel (ISSSF), was important for the monitoring and control of radiological hazards (IMCRH), or was needed to perform a function during the D&D of the plant.

Performance of decommissioning activities that have the possibility of affecting the safe storage of spent fuel or monitoring and control of radiological hazards are controlled by BRP's administrative processes. Big Rock Point administrative procedures specify the standard methods of accomplishing plant activities and processes. They are the documents used to implement the requirements of the QPD. This QPD ensures that BRP complies with the requirements of 10 CFR 50, Appendix B for quality assurance. Examples of administrative processes controlled by procedures include: ALARA (as low as reasonably achievable) reviews, radiation protection including airborne and contamination control, radioactive waste processing (including transportation and release requirements), safety programs, and control of design basis (modification and work package procedures).

System and equipment evaluations resulted in SSCs becoming available for decommissioning. These SSCs were drained, de-energized and deactivated as appropriate. A work control process is applied to SSCs available for decommissioning to document work performed and apply plant processes and controls to the activities. If an activity requires NRC review pursuant to 10 CFR 50.59, an amendment to the BRP Defueled Technical Specifications (DTS) or license will be submitted. Table 3-1 provides a general outline of the phases for BRP Decommissioning.

As of December 30, 2002, NRC review for BRP license amendments included the following:

- Amendment 120, December 24, 1998. Approval of the Defueled Technical Specifications (DTS).
- Amendment 121, January 13, 2000. Deletion of the definition of site boundary and removal of site map.
- Amendment 122, September 28, 2001. DTS reflected control of heavy loads, spent fuel handling considerations and installation of a single-failure proof crane.
- Amendment 123, July 18, 2002. License was revised to include the approved ISFSI Security Plan reference.
- Amendment 124, September 11, 2002. Addition of Spent Fuel Pool applicability statements.

Table 3-1. Decommissioning Periods

Decommissioning Plan and Preparation for Shutdown
- Decommissioning Team formed, October 10, 1993
- Original Decommissioning Plan Submitted February 27, 1995
- Plant Shutdown August 29, 1997

Hazard Reduction
- Core Off-Load complete 9/20/1997
- Chemical Decontamination of the Primary System complete 1/21/1998
- Phase 1 of Decommissioning Power complete 3/11/1998
- Phase 2 of Decommissioning Power complete 2/12/1999
- Spent Fuel Pool Cleanout (Non-SNM and GTCC materials) complete 3/10/2000
- SSC radiological and hazardous waste (asbestos) decontamination in progress

Relocation / Reconfiguration of SSCs (ISSSF or IMCRH)
- High Efficiency Particulate Air Filters complete 1/21/1998
- Monitoring Station relocated 2/12/1999
- Spent Fuel Pool Cooling Skid complete 5/5/1999
- Construction of the ISFSI complete 6/30/2001
- Containment Building Crane Installation complete 10/30/2001
- Relocation of fuel to the ISFSI complete 3/26/2003
- ISFSI Operation in progress

Decommissioning Work Projects – in progress

Decontamination of Structures prior to demolition – in progress

Removal of Major Systems and Components
- Spent Fuel Pool and Fuel Racks (projected 2nd quarter 2003)
- Reactor Vessel (projected 4th quarter 2003)
- Steam Drum (projected 4th quarter 2003)
- Spent Fuel Pool Liner (projected 1st quarter 2004)
- Irradiated Concrete (projected 3rd quarter 2004)

Demolition of Structures Not ISSF or IMCRH
- ASD Building Removal complete 4/26/2001
- Solid Radwaste Building (projected 1st quarter 2003)
- Administrative Building (projected 1st quarter 2004)
- Liquid Radwaste Vaults (projected 3rd quarter 2004)
- Turbine Building (projected 3rd quarter 2004)
- Screenhouse (projected 4th quarter 2004)
- Stack (projected 4th quarter 2004)
- Buried Piping and Building Foundations (projected 1st quarter 2005)
- Containment Building (projected 1st quarter 2005)

License Termination
- General Licensee under 10 CFR Part 72 (2/15/2001)
- Develop and Submit Plan (4/1/2003)
- Public Meetings on the LTP (projected 2nd quarter 2003)
- Plan Approval (projected 4th quarter 2004)
- Final Site Survey (Non-ISFSI area) (projected 4th quarter 2005)
- Regulatory Review of Final Site Survey (Non-ISFSI area) (projected 2005)
- Partial Release of 10 CFR 50 Site (Tentative) (projected 2005)
- Shipment of Fuel to Department of Energy (2012)
- Final Site Survey of the ISFSI (2012)
- Regulatory Final Site Survey Review (2012)
- Termination of Part 50 License (2012)

3.3 COMPLETED DECOMMISSIONING ACTIVITIES AND TASKS

3.3.1 Overview

The major accomplishments described in the following sections are included in this LTP because they are similar to, and indicative of, the complexity of future activities to be performed. The successful completions of these activities demonstrate the project team's ability to safely and effectively decommission the BRP site. Consumers Energy initiated decommissioning activities in 1997 with existing plant resources supplemented by contract employees. Consumers Energy personnel are performing contract oversight functions.

Table 3-2 lists completed decommissioning projects through 2002. These projects supported the safe storage of spent fuel, monitoring of radiological hazards, provided worker safety from radiological and industrial hazards, and supported the D&D process. This section describes activities performed to remove the facility from service from the time that BRP certified that the facility had permanently ceased operation until the fuel was moved from the spent fuel pool to the ISFSI. In some instances, the projects may overlap. Table 3-2 also lists the corresponding section in which it is summarized in this chapter of the LTP.

All projects and significant activities are subject to a detailed scheduling process to ensure efficient use of plant resources. To date, decommissioning activities have not required prior NRC approval or license amendment, as identified through the 10 CFR 50.59 review process. The list is not intended to be all-inclusive, but rather an overview of the decommissioning project from the date of plant shutdown. The completion dates shown may be the date the project or activity was declared operable, or the date the SSC became available for dismantlement. The dates do not necessarily reflect actual project closeout dates.

Table 3-2. Completed Decommissioning Projects

Project Title	*Completion Date*	*Section Discussed*
Chemically Decontaminate the Primary System	1/21/1998	3.3.4
Remove Acid Tank	2/4/1998	3.3.3
Remove Reactor Feed Pump and Motors	3/13/1998	3.3.3
Install Containment High Efficiency Particulate Absorber (HEPA) Filter	1/21/1998	3.3.4
Cut and Remove Reactor Shield Plug and Thermal Shield Plug	5/13/1998	3.3.3
Remove Condensate Pump Room Wall	5/27/1998	3.3.3
Cut and Cap 20" lines, Poison and Flow Transmitter Lines (from Reactor Vessel)	5/22/1998	3.3.3
Remove Feedwater & Condensate Piping on Turbine Deck	6/11/1998	3.3.3
Remove Reactor Feedpumps	6/19/1998	3.3.3
Clear Hydrogen Control Panel Area	6/25/1998	3.3.3
Drain and Remove Lube Oil	6/30/1998	3.3.3
Remove Recirculation Pumps	7/23/1998	3.3.3

Project Title	*Completion Date*	*Section Discussed*
Remove Pipe Tunnel Block Walls	8/11/1998	3.3.3
Remove Main Generator	9/16/1998	3.3.3
Remove Turbine	11/13/1998	3.3.3
Install Decommissioning Air	12/10/1998	3.3.2
Remove Asbestos	Continuing	3.3.4
Clear Fuel Pool Sock Area Rooms	1/7/1999	3.3.3
Install Liquid Radwaste Processing Skid	1/21/1999	3.3.2
Clear Condensate Pump Room	1/27/1999	3.3.3
Fuel Validation and Verification Program	1/29/1999	3.3.2
Install Decommissioning Power	2/12/1999	3.3.2
Relocate Control Room – New Monitoring Station	2/12/1999	3.3.2
Relocate Standby Diesel Generator	3/12/1999	3.3.2
Clear UPS Battery Room	3/12/1999	3.3.3
Label Plant Communications equipment	3/12/1999	3.3.4
Chemically Decontaminate Chromated Systems	3/31/1999	3.3.4
Clear the Control Room	4/9/1999	3.3.3
Clear the Alternate Shutdown Building Equipment	4/28/1999	3.3.3
Install Spent Fuel Pool Cooling Skid	5/5/1999	3.3.2
Clear the Substation	5/7/1999	3.3.3
Clear the Accumulator Rooms	5/18/1999	3.3.3
Clear the Air Ejector Platform	6/2/1999	3.3.3
Remove the Condenser and Turbine Low Pressure Shell	6/21/1999	3.3.3
Relocate Access Control	6/23/1999	3.3.2
Install Fiber Optics and Upgrade the Telephone System	6/24/1999	3.3.4
Isolate and Remove the Turbine Condenser	7/2/1999	3.3.3
Remove the Feedwater Heaters	7/2/1999	3.3.3
Clear the Pipe Tunnel	7/6/1999	3.3.3
Clear the Interior Cable Penetration Area	7/19/1999	3.3.3
Clear the Exterior Cable Penetration Area	8/3/1999	3.3.3
Clear the Core Spray Equipment Room 427	8/13/1999	3.3.3
Clear the Reactor Cooling Heat Exchanger Room 420	9/16/1999	3.3.3
Clear the Electrical Equipment Room	9/27/1999	3.3.3
Install Pipe Tunnel Door	11/4/1999	3.3.3
Clear Fuel Pit Pumps and Heat Exchanger Room	11/24/1999	3.3.3
Clear Filter Room 419	11/24/1999	3.3.3
Modify the Circulation Water System	12/23/1999	3.3.2
Clear the CRD Pump Room 403	1/5/2000	3.3.3
Remove the Recirculation Pumps	2/3/2000	3.3.3
Complete Spent Fuel Pool Cleanout Project	3/16/2000	3.3.2
Clear the Emergency Condenser Area Room 452	3/30/2000	3.3.3
Clear the Shutdown Heat Exchanger Room 417	4/20/2000	3.3.3
Relocate Chemistry Laboratory and Equipment	4/27/2000	3.3.2
Clear Containment Instrument Room 442	6/7/2000	3.3.3
Clear Storage Room 441	6/7/2000	3.3.3
Clear CRD Room & Access Room 401 & 407	6/22/2000	3.3.3
Remove Reactor Vessel Internals	6/27/2000	3.3.3
Remove Lube Oil Tank and Clear Lube Oil Tank Room	7/14/2000	3.3.3
Decontaminate Pipe Tunnel	7/26/2000	3.3.4
Cleanout of LRW Tank and Remove Resin	7/28/2000	3.3.3

Project Title	*Completion Date*	*Section Discussed*
Clear Old Chemistry Lab and Counting Rooms	8/11/2000	3.3.3
Clear Recirculation Pump Rooms 400, 426, and 429	8/16/2000	3.3.3
Prepare area for ISFSI Pad	9/8/2000	3.3.2
Clear Condensate Demineralizer Room 126	9/22/2000	3.3.3
Install Containment Construction Access	10/5/2000	3.3.2
Clear the Machine Shop	10/20/2000	3.3.3
Clear the Electrical and I&C Shop Rooms 122 and 123	10/20/2000	3.3.3
Clear Balance of Accumulator Rooms	10/26/2000	3.3.3
Remove Asphalt for the Heavy Haul Road (Transportation Path for Loaded Fuel Canisters from Containment to the ISFSI)	11/16/2000	3.3.2
Clear Balance of Recirculation Pump Room to Elev. 610'	12/1/2000	3.3.3
Remove Poison Tank and Emergency Condenser	12/13/2000	3.3.3
Clear Turbine Deck General Area	1/5/2001	3.3.3
Clear Core Spray Tank Rooms 421 through 424	1/12/2001	3.3.3
Install Loading Dock Extension	2/15/2001	3.3.2
Remove Steam Drum Blast Wall and Aggregate	3/1/2001	3.3.3
Remove Reactor Shield Tank	4/12/2001	3.3.3
Demolish the Alternate Shutdown Building	4/26/2001	3.3.3
Remove Reactor Shield Tank and Prepare Vessel for Shipment	6/19/2001	3.3.3
Remove Structure to Accommodate Reactor Vessel Removal	6/29/2001	3.3.3
Construct ISFSI Pad	6/30/2001	3.3.2
Modify Containment Building – Cover Fuel Pool, Install platform over Reactor Vessel	8/2/2001	3.3.4
Complete DFS Haul Road Construction	8/27/2001	3.3.2
Remove Condensate Demineralizer Tank	9/25/2001	3.3.3
Install and Test Containment Building Crane	10/30/2001	3.3.2
Delivery of Dry Fuel Storage Canister Prototype	10/31/2001	3.3.2
Install and Test ISFSI Security Systems	1/31/2002	3.3.2
Conduct Readiness Review for Dry Fuel Storage	1/31/2002	3.3.2
Clear Office Building Rooms	2/8/2002	3.3.3
Issue 10 CFR 72.212 Draft for Review	2/12/2002	3.3.2
Clear Cleanup Demineralizer Pump Room 440	2/15/2002	3.3.3
Complete Dry Runs for Dry Fuel Storage	10/23/2002	3.3.3
Load Seven DFS Canisters and Transport to ISFSI	In Progress	3.3.3

3.3.2 Activities Performed in Support of Continued Spent Fuel Storage

This portion of the process began immediately after the fuel was permanently removed from the reactor vessel. All SSCs formerly used to support the production of electricity were made obsolete. Only systems needed to support the spent fuel pool (where the fuel resides until transfer to an ISFSI) were maintained. In some cases, alternate means of functions to support decay heat removal from the pool were established (spent fuel pool cooling skid and liquid radwaste processing skid installation).

The following activities include facility modifications and major activities that were required to maintain SSCs important for safe storage of spent fuel or for monitoring and control of radiological hazards in support of D&D activities.

- Installation of Decommissioning Air
- Installation of the Liquid Radwaste Processing Skid
- Validation and Verification Program for Fuel
- Installation of Decommissioning Power
- Relocation of Control Room to a new Monitoring Station
- Relocation of the Standby Diesel Generator
- Installation of the Spent Fuel Pool Cooling Skid
- Relocation of Access Control
- Modification of the Circulation Water System
- Completion of the Spent Fuel Pool Cleanout Project
- Relocation of the Chemistry Laboratory and Equipment
- Installation of the Containment Construction Access
- Construction of Transportation Path from Containment to the ISFSI
- Construction of the ISFSI
- Installation of new Containment Building Crane
- Installation of ISFSI Security
- Completion of a Characterization Study (see Chapter 2, *Site Characterization*, of this LTP)
- Preparation for Dry Fuel Storage through Regulatory / Administrative processes

3.3.3 Activities Performed in Support of Radiological Decommissioning

This section describes radiological activities and resulting radiological impacts of activities performed to support the D&D process. Structures, systems, and components at BRP have systematically been removed from service. Structures, systems, and components were declared available for decommissioning once their functions were no longer needed for the safe storage, control, or maintenance of spent fuel. Structures, systems, and components removed were not relied upon for accident mitigation in the defueled condition.

Decontamination and dismantlement included reduction of area dose rates using chemical decontamination and accepted industry practices, and removal of plant mechanical, electrical, and heating, ventilating and air conditioning (HVAC) equipment in areas "cleared" for dismantlement. Portable HVAC equipment was provided, as necessary, for personnel radiological and occupational safety. Mechanical equipment removed includes such items as pumps, motors, valves, and piping. Electrical equipment includes instrumentation and conduit. Heating, ventilating and air conditioning equipment includes heaters, blowers, and ventilation ducts. Clearing an area includes cutting and capping piping, conduit, or ductwork into and out of the area. In cases where active systems run through a cleared area, the active system components are either re-routed (in the case of a makeup line to the spent fuel pool), or left in-place and removed later when the SSC is no longer needed (in the case of the recirculation pump room area, final clearing will be completed after the spent fuel pool is drained and no longer needed.)

In late 1997 and early 1998, primary system decontamination project was conducted. PN Services, under the direction of BRP plant staff, used skid-mounted equipment to perform the chemical decontamination of the Primary Coolant System (PCS), the Reactor Vessel, the Reactor Cleanup System (RCS), the Shutdown Cooling System (SCS), and the Steam Drum and Reactor Water level elements. The decontamination was performed in two phases. First, six high temperature Decontamination for Decommissioning (DfD) cycles were applied to the reactor, steam drum, PCS piping and pumps, tube side of the RCS regenerative and non-regenerative heat exchangers, the reactor water level Yarways, and the PCS pumps controlled seal leakoff lines. Solvent was continuously recirculated through the primary system using the PCS pumps.

At the completion of primary system decontamination, the average post-decon contact readings at pre-determined points were ten mRem/hr. General area dose levels were reduced by a factor of at least ten while certain areas were lower by a factor of 27. A total of 406 curies of gamma-emitting radionuclides were removed from solution by cation exchange. Of this total, 83 percent was Co-60. Table 3-3 provides a summary of solution activity removal and radionuclide distribution throughout the process. A noted benefit of the chemical decontamination was the apparent removal of transuranics, which has resulted in essentially no alpha contamination in the primary system piping.

Table 3-3. Primary System Decontamination Activity Removal Summary

High Temperature Phase Cycle Number	Radionuclide (Ci)						Total
	Cr-51	Mn-54	Co-58	Fe-59	Co-60	Zn-65	
1	7.4	15.3	2.1	0.4	164.5	4.6	195.3
2	2.3	2.5	0.7	0.2	41.5	0.3	47.5
3	3.1	1.2	0.7	0.2	27.2	0	32.3
4	6.2	2.2	1.0	0.2	45.6	3.3	58.5
5	3.1	0.8	0.4	0.1	22.0	0	26.4
6	4.4	1.0	0.5	0.1	27.6	0	33.6
Phase Total	26.5	23.0	5.3	1.2	328.4	8.2	392.6
Low Temperature Phase	3.8	1.1	0.2	0.0	8.5	0.1	13.7
Process Total	30.3	24.1	5.5	1.2	336.9	8.3	406.3
Percent	7.5	5.9	1.4	0.3	82.9	2.0	100

In March of 1999, PN Services decontaminated the chromate-treated systems under the direction of Consumers Energy staff. Three chromate-treated systems were decontaminated: the Ventilation Air System (Containment Heating and cooling system, VAS), the Post Incident System (PIS), and the Reactor Cooling Water (RCW) System. The purpose of these decontaminations was to reduce the radiation levels in the piping and components, and to remove the hazardous hexavalent chromium (chromate), thereby reducing the potential for personnel exposure to these hazards during subsequent decommissioning.

The chromated systems were decontaminated in three applications of the CITROX™ process. Dissolved metals, including radionuclides, and all solvent chemicals were removed on ion exchange resins at the completion of the decontamination, leaving only demineralized water in the systems. The final decontamination waste consisted of approximately 2.5 m^3 (90 ft^3) of ion exchange resin and two sets of spent filter cartridges [23 76.2cm (30-inch) long cartridges per set, approximately 0.4 m^3 (14 ft^3) total waste]. This waste was disposed of at Chem Nuclear's facility in Barnwell, South Carolina.

The final SSCs to be decontaminated and dismantled involve planning for demolition of the current Industrial Area. This Industrial Area includes the Containment Building, Ventilation Stack, Screenhouse, Turbine Building and attached Service Building, underground Liquid Radwaste Vaults west of the Turbine Building, the solid Radwaste Building south of the turbine building. Figure 3-1 is a site map illustrating the locations of these buildings.

3.3.4 Completed Dismantlement Activities

Listed in Table 3-4 are short descriptions of completed dismantlement activities. These activities have generated radiological waste. Potentially clean waste was sent to the radwaste-processing contractor for additional evaluation. Contaminated materials and equipment was disposed of as radiologically contaminated material. Hazardous and other regulated waste was also appropriately dispositioned in accordance with applicable State and Federal regulations.

Table 3-4. Descriptions of Completed Decommissioning Activities

Activity	Description
Remove Acid Tank	This activity removed the concentrated acid tank (T-42) from the site. The tank was made obsolete by the installation of the liquid radwaste-processing skid. The acid tank was no longer needed with an alternate means of processing wastewater.
Remove Reactor Feedpump and Motors	The reactor feedwater system (FWS) pumps and motors were removed from the turbine deck.
Cut and Remove Reactor Shield Plug and Thermal Shield Plug	This activity was performed to enable cutting and capping reactor vessel lines (see below).
Remove Condensate Pump Room Wall	The condensate pump room wall was removed to facilitate the removal of condenser components.
Cut and Cap lines from the Reactor Vessel	This activity was done to isolate the reactor vessel.
Clear Hydrogen Control Panel Area	Hydrogen equipment was removed from the rooms that contained components from the turbine generator system (TGS.)
Remove Feedwater and Condensate Piping on the Turbine Deck	FWS and condensate system (CDS) piping was removed from the turbine deck.
Remove the Reactor Feed Pumps	The reactor FWS pumps and motors were removed from the turbine deck.
Drain and Remove Lube Oil	Lubrication oil for the FWS pump motors was no longer needed once the feedpumps were removed.
Remove Pipe Tunnel Block Walls	Pipe tunnel block walls were removed to facilitate the removal of the turbine and condenser.
Remove Main Generator	The main turbine generator was removed.
Remove Turbine	The turbine was removed.
Clear the Fuel Pool Sock Area Rooms	Equipment was removed from the rooms that contained radwaste system (RWS) components made obsolete by the installation of the liquid radwaste-processing skid.

Activity	Description
Clear the Condensate Pump Room	Equipment was removed from the rooms that contained components from the air ejector system (AES - made obsolete with the installation of the HEPA filter), CAS (made obsolete with the installation of the decommissioning air system), CDS, demineralized water system, and the resin regeneration system (DMW/RGS-made obsolete by the installation of the liquid radwaste processing skid.)
Clear the UPS Battery Room	Equipment was removed from the rooms that contained components from the emergency power system (EPS - made obsolete with the installation of decommissioning power.)
Clear the Alternate Shutdown Building Equipment	Equipment from the Alternate Shutdown (ASD) Building including HVAC unit, control panels, battery chargers, emergency lights, smoke detectors, and card readers. All ASD Building interior components were removed. All that remained was bare concrete walls, main structural steel, and the exterior door.
Clear the Substation	This project dismantled and removed former substation. Decommissioning power new substation installation rendered the old substation obsolete.
Clear the Accumulator Rooms	Equipment was removed from the rooms which contained components from the control rod drive (CRD) system.
Clear the Air Ejector Rooms and Platform	Equipment was removed from the rooms that contained components from the air ejector and waste gas systems (AES / WGS - made obsolete with the installation of the HEPA filter), control air system (CAS made obsolete with the installation of the decommissioning air system), and the RWS (made obsolete by the installation of the liquid radwaste processing skid.)
Remove the Condenser and Turbine Low Pressure Shell	These components were removed.
Remove the Feedwater Heaters	These components were removed.
Clear the Interior and Exterior Cable Penetration Rooms	Equipment was removed from the rooms that contained obsolete components from the CIS, power system (replaced by decommissioning power supplies), and reactor protection system (RPS) components.

Activity	Description
Clear the Pipe Tunnel	Equipment was removed from the rooms that contained components from the CAS (made obsolete with the installation of the decommissioning air system), CDS, DWS, containment isolation system (CIS), CRD system, DMW / RWS (made obsolete by the installation of the liquid radwaste processing skid), FWS, heater extraction drain (HED) system, main steam system (MSS), reactor cooling water system (RCW), and the TGS.
Clear the Core Spray Equipment Room	Equipment was removed from the rooms that contained PIS/emergency core cooling components.
Clear the Reactor Cooling Water Heat Exchanger Room	Equipment was removed from the rooms that contained RCW components
Clear the Electrical Equipment Room	Equipment was removed from the rooms that contained components from the station power system (SPS - made obsolete with the installation of decommissioning power.)
Install the Pipe Tunnel Door	This modification installed an access door between containment recirculation pump room and the pipe tunnel in the turbine building. The door was installed to facilitate dismantlement. The design of the door met the containment closure definition from BRP's Defueled Technical Specifications (DTS). Installation of openings in containment meets the DTS requirements but render the containment building as a non-pressure vessel.
Clear the Fuel Pit Pumps and Heat Exchanger Room	Equipment was removed from the rooms which contained spent fuel pool (SFP) obsolete components - i.e., heat exchangers made obsolete with the installation of the SFP cooling skid.
Clear the CRD Pump Room	Equipment was removed from the rooms that contained CRD components.
Remove the Reactor Cooling Water Recirculation Pumps	Reactor recirculation pumps were part of the FWS for the reactor.
Clear the Emergency Condenser Area	Equipment was removed from the rooms that contained liquid poison system (LPS) component.
Clear the Shutdown Heat Exchanger Room	Equipment was removed from the rooms that contained SCS components.
Clear the Containment Instrument Room	Equipment was removed from the rooms that contained components from the neutron monitoring system (NMS).
Clear Storage Area in Containment	Equipment was removed from the rooms that contained CRD nitrogen charging bottles.

Activity	Description
Clear CRD Access Room	Equipment was removed from the rooms that contained PCS and PIS components.
Remove Reactor Vessel Internals	Reactor vessel grid bars were removed to prepare the vessel for shipment.
Remove Lube Oil Tank and Complete Clearing the Lube Oil Tank Room	The lube oil tank and room components were not needed since they were components from the MSS and turbine lube oil system (SLO).
Clean out Liquid Radwaste and Remove Resin	Resins from the liquid radwaste tank were no longer needed with the installation of the liquid radwaste skid.
Clear Chemistry Laboratory	Equipment was removed from the former lab areas after the lab was moved to the new access control portable building.
Clear Condensate Demineralizer Room	Equipment was removed from the rooms, which contained components from the AES (made obsolete with the installation of the HEPA filter), CAS (made obsolete with the installation of the decommissioning air system), CDS, demineralized water system, and the resin regeneration system (DMW / RGS- made obsolete by the installation of the liquid radwaste processing skid.)
Clear Machine Shop	Equipment was removed from the rooms that contained components from the SPS (made obsolete with the installation of decommissioning power.)
Clear Electrical and I&C Shop	Obsolete mechanical, electrical, HVAC equipment was removed from the electrical and I&C shop areas.
Clear Balance of Accumulator Room	Equipment was removed from the rooms that contained CRD components.
Clear Filter Room 419	Equipment was removed from the rooms that contained RCW / LPS components.
Clear Balance of Recirculation Pump Room	Equipment was removed from the rooms, which contained components from the CRD system, the emergency core cooling system (ECS), the FWS, the LPS, the MSS, the PCS, the RCW system, the reactor steam drum system (RSD), the RWS, and others.
Remove Poison Tank and Emergency Condenser	These components were removed.
Clear Turbine Deck General Area	The turbine deck was cleared of all components that were obsolete with decommissioning.

Activity	Description
Clear Core Spray Tank Rooms	Equipment was removed from the rooms which contained PIS / ECS components.
Remove Steam Drum Blast Wall and Aggregate	These components were removed.
Demolish the Alternate Shutdown Building	The ASD Building was demolished. It was a small structure located inside the protected area. It had previously been cleared of all components and consisted of the concrete structure and steel support beams. Prior to demolition, the structure was radiologically surveyed. Debris from the structure is currently being stored in an Impacted area. It will be shipped to a State of Michigan licensed landfill if it meets the landfill release specifications of the 10 CFR 20.2002 NRC approved alternate disposal procedure.
Remove Reactor Shield Tank and Prepare Reactor Vessel for Shipment	The reactor shield tank was removed and the reactor vessel prepared for shipment to provide for efficient decommissioning.
Remove Containment Structures to Accommodate Reactor Vessel Removal	A facility change was done to remove part of the containment structures to enable removal of the reactor vessel.
Modify Containment Building (SFP cover)	Containment Building modifications were performed prior to the installation of the new Containment Building crane. Other modifications were performed to enable more efficient dry fuel transfer from the SFP to the dry fuel storage canisters.
Remove Condensate Demineralizer Tanks	These components were removed.
Clear Office Building Rooms	Equipment was removed from the former office areas.
Clear Demineralizer Pump Room	Equipment was removed from the rooms that contained RCS system components.
Demolish Radwaste Building and Storage Vaults	The Radwaste Building and internal concrete storage vaults located south of the Protected Areas were demolished. This building was a steel structure with a concrete floor. Prior to demolition, the structure was decontaminated and radiologically surveyed. Debris from the building is currently being stored in an Impacted area. It will be shipped to a State of Michigan licensed landfill if it meets the landfill release specifications of the 10 CFR 20.2002 NRC approved alternate disposal process. The concrete vaults were disposed of a radioactive waste.

3.3.5 Activities Performed in Support of Worker Safety

This section describes activities performed to ensure worker safety from radiological and non-radiological hazards. Non-radiological hazards include occupational health hazards (asbestos, PCB, lead, silica, etc.), energized item protection (electrical), and worker safety measures. Immediately after cessation of reactor operation, several activities were performed to remove the facility from service and to reduce radiological and non-radiological hazards to workers performing decommissioning.

Below are descriptions of these activities:

- Asbestos Abatement
  The following areas were subjected to removal of asbestos: Turbine Deck, Pipe Tunnel, Emergency Condenser, Turbine Building, Steam Drum, Recirculation Pump Room, Screenhouse/MDG Room, Sphere, Air Compressor Room, Shutdown Heat Exchanger Room, RCW Heat Exchanger Room, Ventilation/HVAC Room, and other miscellaneous areas. The process of asbestos removal began June 11, 1997, and continued until Septembers 28, 2000, when the majority of asbestos abatement was completed for structures and components. Administrative processes were used to control the method of abatement. In the event that small quantities of asbestos are discovered that require abatement, the administrative process remains in place to remove asbestos. This is an ongoing activity.

- Chemical Decontamination of the Primary System
  The Electric Power Research Institute (EPRI) *DfD* chemical decontamination process, developed by Bradtec Ltd, was performed at BRP in December 1997. The EPRI *DfD* process uses dilute fluoroboric acid (HBF_4) under conditions of controlled oxidation potential to remove contamination from surfaces and collect the contamination on ion exchange resin. The process was used to remove loose and fixed contamination from the internal surfaces of the RCS, as well as the reactor vessel, steam drum, reactor coolant cleanup and shutdown cooling systems. Over 400 curies of radioactivity were removed, which accounted for 96% of the estimated internal contamination. A dose savings of at least 400 person-rem to plant workers is expected during dismantlement as result of performing the post shutdown chemical decontamination.

- HEPA Filter Installation
  A HEPA filter was installed to minimize radioactive particulate release through the stack during major dismantlement activities. The modification incorporated a HEPA filtration skid, which is valved into the existing containment exhaust flow path when activities are performed with significant potential for release of radioactive materials into containment.

- Plant Communications Systems Upgrade
  Plant communications systems were upgraded for two reasons:
  1. To eliminate electrical hazards, and
  2. To consolidate communications equipment.

- Chromated Systems Chemical Decontamination
  In March 1999, three chromated systems (VAS, PIS and RCW) were decontaminated using the CITROX process (see Section 3.3.3). The purpose of the decontamination was two-fold. It was performed to reduce radiation levels in the piping and system components and to remove the hazardous hexavalent chromium within the process fluid. The end result of the decontamination process was a reduction of the potential to radiation and chromium hazard to workers during decommissioning activities. Termination of the process left the three systems filled with demineralized water. All the chromium removed from the systems was deposited on resin and verified by testing to be non-hazardous.

- Pipe Tunnel Decontamination Project
  This project was performed to reduce general dose rates in the pipe tunnel to provide ALARA conditions for the workers in the area.

3.3.6 Non- Radiological Decommissioning Activities

Two major activities have been completed that are considered as non-radiological decommissioning activities; these are construction of the ISFSI facilities and expansion/relocation of offices and parking lots.

3.4 FUTURE DECOMMISSIONING ACTIVITIES

The focus of this section is on the decontamination and demolition of components and structures in the former Industrial Area. Table 3-5 is a list of future decommissioning activities planned at BRP. The activities listed are intended to provide an overview of the remaining decommissioning activities and an estimated time schedule for those activities. Major scheduled activities are outlined in the BRP PSDAR. Schedules provided in this section are for general guidance and illustrative purposes only. There is no intent to revise this LTP when schedule changes occur. Current schedules are available onsite for review.

Future decommissioning activities will continue to emphasize control and monitoring of radiological hazards, safe storage, control, and monitoring of spent fuel, and worker safety. Consumers Energy intends to perform decommissioning activities with plant resources supplemented by contract employees. Consumers Energy personnel are expected to continue the contract oversight function. This oversight includes ensuring the contractor is subject to and follows training and radiation protection standards as outlined in the QPD, plant programs and procedures, and all applicable State and Federal regulations.

The dismantlement process and associated radiation protection measures are controlled using BRP's procedure process. Procedure revisions require Site Review Committee review and evaluation using the 10 CFR 50.59 process. Only one activity, to date, required prior NRC approval. NRC Bulk Assay approval process is discussed in Section 3.4.2.1 of this chapter. It is anticipated that future decommissioning activities will be bounded by current UFHSR and PSDAR analyses and evaluations.

Clearing of SSCs consists of preparing areas for demolition. These activities include removal of all components in an area (including mechanical and electrical equipment, piping, conduits, HVAC equipment, etc.) and general decontamination in accordance with site procedures to allow demolition of a building. The assumptions and philosophy of BRP demolition activities are as follows:

- General demolition activities are consistent with clean-to-dirty and south-to-north groundwater flow approach.
- Excavated locations may remain open to support ongoing work (Turbine, Administrative, and Containment Building footprints).
- Groundwater and surface water control will be evaluated for all excavations and required if potential for migration of radioactive or hazardous materials is identified.
- Groundwater control will be required for excavations to remove subsurface structures.
- Groundwater discharge to Lake Michigan will require an NPDES Permit.
- Demolition activities in open areas will require radiological control and isolation from concurrent work. The work plan for the demolition will take into account any radiological controls needed to prevent cross- or re-contamination by air or water paths.
- Final Status Survey (FSS) on areas cannot be performed until physical demolition work in the area is complete (see Chapter 5, *Final Status Survey Plan*, of this LTP).
- Excavations will be backfilled with soil determined acceptable for backfilling (see Section 5.4.2).
- Final Status Survey of surface soils will occur after all demolition, remediation and backfill but prior to topsoil addition for Greenfield plantings. NOTE: Greenfield definition applies to the former Industrial Area and does not include the ISFSI facilities.
- License termination for the ISFSI area will occur once the ISFSI area is demolished and a remediation/FSS is performed.

Figure 3-2 provides a graphical illustration of tasks performed by different work groups on site. Currently, BNFL is the major structural demolition contractor for the former Industrial Area. The on-site labor contractor is currently PMC Constructors and Technical Services, LLC (PMC). PMC will perform the labor involved in moving soil for remediation or fill after final status surveys are performed in the former Industrial Area. In the future, a different contractor may perform ISFSI site demolition and remediation. Contracted personnel will be under the direct supervision of Consumers Energy personnel. Soil surveys will either be performed in-house or contracted with Consumers Energy personnel oversight. Soil survey methods are outlined in Chapter 5 of this LTP.

Table 3-5. Future Decommissioning Activities

Activity	*Projected Date*
Move Fuel Canisters to ISFSI	1st Quarter 2003
Clear Service Building First Floor	1st Quarter 2003
Declare Bulk Assay Equipment Operational	2nd Quarter 2003
Clear Non-Regen and Regen Heat Exchanger Room 439	2nd Quarter 2003
Remove Concentrated Waste, and Radwaste Tanks	2nd Quarter 2003
Clear Maintenance Building Room 901	2nd Quarter 2003
Remove Spent Fuel Racks	2nd Quarter 2003
Move Greater than Class C Canister to ISFSI	2nd Quarter 2003
Clear Heating Boiler Room	2nd Quarter 2003
Clear Ventilation Room	2nd Quarter 2003
Drain Spent Fuel Pool	3rd Quarter 2003
Perform Final clearing of Recirculation Pump Room	3rd Quarter 2003
Remove Turbine Building Stairs	3rd Quarter 2003
Clear Decontamination Washdown area – room 121	3rd Quarter 2003
Clear Turbine and Service Building Roofs	3rd Quarter 2003
Clear Waste Holdup Tank Room	3rd Quarter 2003
Clear Track Alley – Turbine Building	3rd Quarter 2003
Clear Decontamination and Washdown Rooms 444 and 451	3rd Quarter 2003
Remove Reactor Water Cleanup Heat Exchanger	3rd Quarter 2003
Clear Diesel Generator Room	3rd Quarter 2003
Remove Heating Boiler	3rd Quarter 2003
Remove Condensate and Demineralizer Tanks	3rd Quarter 2003
Clear Spent Fuel Storage Areas Rooms 437 and 448	4th Quarter 2003
Clear Reactor Building Remaining Systems	4th Quarter 2003
Remove Reactor Vessel and Support	4th Quarter 2003
Remove Steam Drum	4th Quarter 2003
Clear Liquid Radwaste Area	4th Quarter 2003
Transport Reactor Vessel to Disposal Site	4th Quarter 2003
Clear Liquid Radwaste Operating Gallery	4th Quarter 2003
Clear Surge Tank Room 448	1st Quarter 2004
Remove Turbine Building Crane	1st Quarter 2004
Clear Reactor Building Stairways	1st Quarter 2004
Clear Supply Air Shed	1st Quarter 2004
Demolish Admin and Services Building	1st Quarter 2004
Remove Spent Fuel Pool Liner	1st Quarter 2004
Clear Shield Laydown Area, Vent Panel, and Supply Fan Areas	1st Quarter 2004
Clear Stack	2nd Quarter 2004
Remove Containment Building Crane	2nd Quarter 2004

Activity	*Projected Date*
Clear Security Building Room 909	3rd Quarter 2004
Clear Screenwell and Pump House	3rd Quarter 2004
Clear Remaining Systems – Turbine Building	3rd Quarter 2004
Demolish Concrete and Shield Cooling Panels – Reactor	3rd Quarter 2004
Demolish Liquid Radwaste Vaults	3rd Quarter 2004
Demolish Turbine Building and Foundation	3rd Quarter 2004
Decontaminate Concrete – Containment Building Complete	3rd Quarter 2004
Demolish Screenwell and Pump House	4th Quarter 2004
Demolish Off-Gas Stack	4th Quarter 2004
Clear Personnel and Equipment Locks	4th Quarter 2004
Demolish Containment Building Sphere	1st Quarter 2005
Removal of Remaining Structures and Foundations from Former Industrial Area	1st Quarter 2005
Complete Final Restoration of Site (other than ISFSI area)	3rd Quarter 2005
Final Status Survey (other than ISFSI area)	4th Quarter 2005
Transport Fuel to DOE Repository	Projected 2012
Complete Final Restoration of Remaining Site	Projected 2012
Terminate License	Projected 2012

3.4.1 Discussion of Major Decommissioning Activities

Big Rock Point Administrative Processes and the Consumers Energy Quality Program control major decommissioning projects discussed below [References 3-1, 3-2, and 3-6]. The activities described are intended to be a general outline of the technical aspects of the activity along with some of the radiological controls developed for the activity. In some cases, the planning of activities has not proceeded to a point where specific tasks may be cited. Administrative processes include documentation of the activities in Decommissioning Work Packages (DWP), Milestone Work Packages (MWP), or other work order process. Consumers Energy or contractors under the supervision of Consumers Energy shall perform the activities.

3.4.1.1 Reactor Vessel Removal

Preparation of the reactor vessel includes removal of secondary supports and cutting and plugging all nozzles. Caps or plugs will provide a barrier to contain grout that will be injected into the RV once placed inside the shipping container. The nozzle work will be completed during ever decreasing RV water levels to ensure maximized shielding from the water as far as practical. Additional remote cutting techniques will be used to further minimize worker dose. A slot will be cut into the concrete on the west side to allow adequate clearance for removal of the vessel from its cavity.

The RV will be removed from its cavity without its head. (The RV head will have been previously removed and shipped to Envirocare in Utah.) In place of the head, the RV will have a Top Cover Plate (TCP) bolted to it. The TCP doubles as a lifting lug plate and an end cover for the shipping container. Once removed from its cavity, the RV will be placed into the vertical shipping container located in the containment sphere. The TCP will then be welded to the shipping container closing the shipping container.

The RV internals will then be grouted through the lifting/cover plate and the closed shipping container containing the RV will be down ended. Once down ended, the annulus (i.e., the space between the RV and the inside of the shipping container) will be grouted. When the grout is cured, all the injection ports will be plugged and welded shut.

The shipping container will then be transferred out of the containment sphere, placed onto the road transporter, transferred onto a rail car, and transported to Barnwell, SC, where it will be off loaded and buried.

Prior to and at specific times during the transportation evolution, dose checks will be taken to ensure dose rates are below 10 CFR 71 levels. If, prior to shipment, in the very unlikely event that the package dose rate exceeds the 10 CFR 71 levels, the design and licensing of the shipping container includes for the provision of of up to 1/2" of additional shielding, as necessary. However, no additional shielding is expected to be required.

3.4.1.2 Steam Drum Removal

A blowout panel bordered the North side of the Steam Drum. This panel had aggregate sandwiched between two walls of steel sheeting. The aggregate was removed prior to steam drum removal. Dust control and radiation control (HEPA filter) measures were used in the aggregate removal process.

The nozzles on the steam drum will be capped/plugged.

Holes will be cut into the south wall of the steam drum enclosure to support the steel slide beams. Prior to cutting, RP&ES will survey the concrete to ensure contamination levels are acceptable. If contamination levels are high, concrete scabbling may be required. Two steel beams and support saddles with heavy-duty rollers will be placed under the steam drum and extended out over the reactor cavity with adequate support in critical locations. These beams will be supported on the south end by cutting holes in the south wall and cribbing under the beams. Supports will be built up under the north end of the beams from the refueling floor. These steel beams will provide a solid surface to slide the steam drum out of the enclosure. Manual hydraulic jacks will be placed under the steel beams on the south end and lift systems hydraulic jacks will be placed under the steel beams on the north end.

The existing box section beams filled with concrete (former blowout panel support) will be removed. The steam drum will be lifted and the remaining steam drum supports and restraints will be removed.

The steam drum will be moved over the former refueling floor from the enclosure using adequate restraining devices. Once the steam drum is over the reactor cavity, it will be lifted by the Containment Building crane and then lowered to the 599' level. The Steam Drum will be moved out of containment and placed on a transporter in readiness shipping to a disposal facility.

3.4.1.3 Containment Building Crane Removal

In 2001, the Big Rock Point single-failure-proof crane was installed using the Facility Change Process. FC-706, "Containment Building Crane," removed the original 75-ton crane and replaced it with a 105-ton single-failure-proof crane that was designed to meet the applicable industry and regulatory criteria and guidelines [References 3-11, 3-13, and 3-14].

A contractor will dismantle the containment-building crane when it is no longer needed for major equipment removal. Prior to release of the crane, it will be surveyed (and decontaminated as necessary) in accordance with existing plant procedures and processes.

3.4.1.4 Containment Building Sphere Removal

The sphere will be cut into manageable pieces. Once at grade level, the pieces will be further size reduced (cut) to enable placement into shipping containers.

Prior to cutting for demolition, the sphere shall be abated to eliminate hazardous material and radiological concerns.

3.4.1.5 Off-Gas Stack Demolition

The proposed off-gas stack removal is provided below; however, detailed MWP packages have not been developed at the time of LTP submittal. The stack will be decontaminated prior to demolition to ensure NUREG-0586, "Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities," and its supplement bound the structure for accident considerations and assumptions in the UFHSR [References 3-12 and 3-13]. FGEIS accident analyses assumptions and methodology shall be reviewed, and the 10 CFR 50.59 and 10 CFR 50.82 review processes shall be used to ensure methods used to demolish the Off-Gas Stack, such as explosive demolition, are analyzed and no prior NRC approval is required.

3.4.2 Decontamination of Structures, Systems, and Components

Decontamination methods include wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. Selection of the preferred method is based on the specific situation. Other decontamination technologies will be considered and utilized, as appropriate. Approved administrative procedures and processes control decontamination. These controls ensure that wastewater is collected for processing by liquid waste processing systems. Airborne contamination control and waste processing systems are used, as necessary, to control and monitor releases.

3.4.2.1 Demolition Debris Disposal Program

By letters dated May 18, 2001, and June 20, 2001, Consumers Energy submitted a request for approval of proposed disposal procedures in accordance with 10 CFR 20.2002 to the NRC for review [References 3-7 and 3-8]. On December 3, 2001, the NRC provided an Environmental Assessment and Finding of No Significant Impact related to BRP's request for approval of proposed disposal procedures in accordance with 10 CFR 20.2002 [Reference 3-9]. The NRC approved the request by a letter dated February 5, 2002 [Reference 3-10].

The NRC approved for BRP to dispose, in a State of Michigan licensed Type II landfill, demolition debris, using a detection limit for potential unidentified total principal gamma emitter concentration of 5 pCi/g. To ensure that the 5 pCi/g principal gamma emitter detection limit is not exceeded, radiological surveys will be performed on structural surfaces prior to demolition to verify that radioactive surface contamination does not exceed 5000 disintegrations per minute per 100 square centimeters (dpm/100 cm^2), averaged over areas appropriate for the detection system used. All demolition debris is required to be monitored by a bulk assay radiation detection system with an alarm setpoint established at the 5 pCi/g principal gamma emitter limit prior to disposal. If the bulk assay system detects licensed radioactive material above the 5 pCi/g limit, the demolition debris is surveyed to identify the licensed radioactive material. The licensed radioactive material may be removed and the container re-assayed or the entire container shipped to the waste contractor as radioactive waste. Radioactive material so removed will be transported to an offsite contractor for secondary waste processing or disposal as radioactive waste. Figure 3-3 is a process flowchart for demolition debris evaluation and disposal.

Big Rock Point demonstrated in the request for approval of proposed disposal procedures, that assuming all non-detectable activity was at the 5 pCi/g detection limit for principal gamma emitters, an estimated 19.2 million kilograms (42.25 million pounds) of demolition debris being transported and disposed of in a State of Michigan-licensed Type II landfill will not result in a Total Effective Dose Equivalent (TEDE) dose exceeding 1 mrem/year to individuals who have the potential to receive the maximum dose. Three scenarios were considered which included a resident/farmer scenario after closure and unrestricted release of the landfill, a landfill worker exposure scenario during demolition debris disposition at the landfill, and a transportation scenario for potential driver exposure during transportation of the demolition debris to the landfill.

It is estimated that 38.3 million kilograms (84.5 million pounds) of demolition debris will originate from this decommissioning project. Approximately one-half of this debris is non-impacted (i.e., has never had the potential for neutron activation or has not been exposed to licensed radioactive material). The demolition debris that will originate from demolition and removal of structures at BRP consist of concrete debris including rebar, structural steel, sheet metal, roofing materials, foundation concrete, asphalt and minor amounts of soil associated with digging up foundations. Demolition debris does not include plant systems or components. Demolition debris will not be processed for any other use or recycled.

3.4.3 Dismantlement of Structures, Systems, and Components

Big Rock Point intends to remove all below-grade foundations and buried piping in its decommissioning effort. In general, the following sequence has been and will continue to be followed in the dismantlement and demolition process. No SSCs are dismantled or demolished until they are declared available for decommissioning, usually on a system basis or building basis. In some instances, Facility Changes or Minor Alterations (design processes) are used to isolate portions of systems to make them available for decommissioning. When all the spent fuel is on the ISFSI, the former Industrial Area will be available for decommissioning. These processes are controlled by procedure and are part of an approved quality assurance program.

The decommissioning and demolition process is as follows:

1. Remediate, strip, package, and ship components from the buildings or individual rooms or areas to be decommissioned. Components that are not radiologically clean are sent to a radioactive LLW disposal facility via the waste contractor. Remediation may also include removal of hazardous or other regulated wastes (e.g., asbestos and lead) by approved, controlled methods.

2. Decontaminate structural components, if required or feasible, to meet the established release criteria.[1]

3. Perform pre-demolition and confirmatory surveys of the structural components.

4. Demolish the structure.

[1] Structural components may require demolition via wire saw cutting, softening, or other process to facilitate radiation surveys. Any liquid or airborne wastes from these activities will be controlled and monitored by approved procedure or process.

5. Perform bulk assay evaluation of demolition debris to allow material release (per BRP Demolition Debris Disposal Program – see Section 3.4.1.1)[2].

Multi-disciplined review of work packages and procedures for D&D activities include radiation protection reviews, design basis review (availability for decommissioning), construction review for feasibility of demolition method, and review for safety and environmental concerns.

3.4.4 Underground Piping

It is BRP's intent that no underground piping will be left in place within the Industrial Area. As provided in the Chapter 1 Greenfield definition, the septic field west of the Protected Area and intake pipe and structure in Lake Michigan will remain in place. Buried piping in the Industrial Area includes fire protection, well water, sanitary, compressed air, service water, circulating water, and radwaste system piping in addition to electrical cable and conduit and stormwater piping. Removal of this piping will be conducted through the BRP work planning process utilizing plant drawings to identify buried piping and equipment.

Excavation for removal of the Turbine and Containment Building foundations and excavation slopes required for soil stability will uncover much of the piping that originated from operating plant systems. As the dismantlement process progresses, any pipe "end-points" will be marked for future removal.

Prior to FSS, a ground-penetrating radar survey will be performed, as necessary, to verify that all underground piping has been removed. Appendix 2-E of Chapter 2 contains detailed descriptions of buried piping and equipment for the different BRP survey units.

3.4.5 Control Mechanisms to Ensure No Recontamination

Recontamination may occur through transmission of contaminants through groundwater, air, or surface water. Control mechanisms to prevent the above scenarios include engineering controls and evaluation of demolition methods to minimize impacts on cross or re-contamination. Engineering controls will include provisions for minimizing groundwater and rain/snow seepage into open excavations. Methods such as temporary covering and limiting length of time an excavation is open may be used.

Big Rock Point's demolition schedule is from radiologically cleaner to dirtier areas/structures. Activities will generally be performed in a south-to-north direction, to minimize the impact of groundwater contamination. Water pumped from excavations will be monitored and disposed of in accordance with State of Michigan discharge permits. Methods used for demolition will be examined to reduce the potential for cross- or re-contamination verses the efficiency of

[2] If radiation surveys do not allow release or shipment of demolition debris to a local landfill, they are sent to a radioactive LLW facility.

demolition. All processes that affect control and monitoring of radiological hazards are included in our Quality Program, and require procedures and administrative review.

License Termination Plan Chapter 2 contains detailed information on the method and procedures used in the site characterization. License Termination Chapter 4, *Site Remediation Plan*, describes the various methods to be used during the BRP decommissioning to reduce the levels of radioactivity to that which meets the NRC radiological release criteria in 10 CFR 20.1402. License Termination Plan Chapter 5 contains detailed information on isolation and control of survey units that have been remediated, surveyed, and found to meet the site release criteria.

3.5 CURRENT RADIOLOGICAL STATUS AND EXPOSURE ESTIMATES

3.5.1 Occupational Exposure

Figure 3-4 provides BRP cumulative site dose and goals for the decommissioning project. This estimate was developed to provide site management ALARA goals. The goals are verified by summation of actual site dose, as read by personnel dosimetry via thermoluminescent dosimeters (TLDs). ALARA estimates are a compilation of work plan (radiation work permit) estimates for the period. This information is in addition to information gathered for reporting of yearly site dose in accordance with BRP Defueled Technical Specification 6.7.1. The annual report of occupational dose meets the guidance of NRC Regulatory Guide 1.16, *Reporting of Operating Information*, Appendix A [Reference 3-16]. The estimated total nuclear worker exposure during decommissioning is estimated to be 700 person-rem. This estimate is in the 700-1600 person-rem range of the FGEIS [Reference 3-12].

3.5.2 Public Exposure

Continued application of BRP's Radiation Safety Program, Demolition Debris Disposal Plan, Radiological Effluent Technical Specification Program and Radiological Environmental Monitoring Program assures public protection in accordance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I. Section 8.5.1 contains an evaluation of estimated public exposure as a result of decommissioning activities including the transportation of radioactive waste as compared to the FGEIS [Reference 3-12].

3.5.3 *Estimate of Quantity of Radioactive Material to be Shipped for Disposal or Processing*

Total volume of waste projected for BRP decommissioning is 17,333 cubic meters (612,100 cubic feet), which includes 2,042 cubic meters (72,100 cubic feet) of radioactive waste and 15,291 cubic meters (540,000 cubic feet) of demolition debris with no detectable radioactivity. In 1997, TLG Services, Inc. completed this estimate for the BRP PSDAR. This volume of waste is bounded by FGEIS volume for the reference boiling water reactor of 18,760 cubic meters (662,500 cubic feet) (which includes disposable containers) [Reference 3-12].

3.5.4 Solid Waste Activity and Volume

Big Rock Point also reports, in accordance with Defueled Technical Specification 6.7.2 and 6.7.3, Annual Radioactive Environmental and Radioactive Effluent Release including data on solid waste activity and volumes. The set of data provided in Table 3-6 provides a compilation of this information.

Big Rock Point's Annual Radioactive Effluent Release Report, submitted in accordance with DTS and 10 CFR 50.36(a), includes a report on solid waste. This report is submitted before May 1st each year. A summary of solid waste effluent releases for 1997 through 2002 is provided in Table 3-6. Future updates may be obtained from BRP for inspection.

Consumers Energy contracted with Duratek to provide waste processing/disposal services for the BRP decommissioning project. To support contractual arrangements and administration, waste quantities are estimated in units of weight (kg/pounds) as opposed to volume. This is consistent with TLG Services, Inc., estimates of waste quantity units. In addition, BRP has secured fixed-price contracts for disposal of large components and significant waste streams. Table 8-3, Total Estimated Waste (Section 8.5.1) to be shipped from BRP, summarizes the estimated waste types by waste class pursuant to 10 CFR 61.55.

Table 3-6. Solid Waste Effluent Release Report Summary

Waste Class	Source	Year	Volume (kg)	Volume (lbs)	Total Curies	Principle Radionuclides
AU*	DAW; compacted waste, from plant demolition, and incineration ash	1997 1998 1999 2000 2001 2002	59.1 1101 1175 25506 20107 9263.3	130.4 2427.0 2589.5 56,230.8 44,329.4 20,422.0	0.22 1.32 2.40 5.47 7.20 1.51	Co-60, Mn-54, Cs-137, Zn-65, Fe-55, Ni-63, H-3,
AS**	DAW and Irradiated Metals; CRD system	1998 2000 2002	88.5 95.7 284	195.0 211.0 626.0	0.32 1.28 10.50	Co-60, Mn-54, Cs-137, Zn-65, Fe-55, Ni-63
AS	Resin from processing water	2000	91.7	202.1	7.86	Co-60, Mn-54, Fe-55, Ni-63, Cs-137
B	Dewatered Resin	1997 2000	537.5 218.2	1,184.9 481.0	184.24 255.00	Co-60, Mn-54, Cs-137, Zn-65, Fe-55, Ni-63
C	Dewatered Resins	1998	692.5	1,526.8	1,748.00	Co-60, Mn-54, Cs-137, Zn-65, Fe-55
C	Dewatered Filters	1998 1999 2000 2001	60.1 120.1 223.8 40.6	132.4 264.8 493.3 90.3	26.60 18.90 29.30 3.93	Co-60, Mn-54, Cs-137, Zn-65, Fe-55, Ni-63
C	DAW	1999 2000	61.2 60.8	134.9 134.0	2.81 0.92	Co-60, Mn-54, Cs-137, Zn-65, Fe-55, Fe-55, Ni-63
C	Irradiated Components	1999	331.9	731.8	65,100.00	Co-60, Mn-54, Cr-51, H-3, Fe-55, C-14
C	Irradiated Metal; Fuel channels and reactor internals	2000	125.6	277.0	5,390.00	Co-60, Mn-54, Fe-55, Ni-63, Cs-137

*Class A, Unstable: Waste, which may leave voids when buried at the LLW facility.
**Class A, Stable: Waste, which will leave no voids when buried at the LLW facility.

3.5.5 Liquid Waste Activity and Volume

Big Rock Point also reports, in accordance with Defueled Technical Specification 6.7.2 and 6.7.3, Annual Radioactive Environmental and Radioactive Effluent Release, including data on liquid waste. The set of data provided in Table 3-7 provides a compilation of this information. A summary of the liquid waste effluent release report for 1997 through 2001 is provided below. Liquid effluent release data was not available for 2002 at the time of this LTP submission; future updates may be obtained from BRP for inspection.

Table 3-7. Liquid Waste Effluent Releases

Year	Tritium Release Ci	Dissolved and Entrained Gas Release Ci	Alpha Release Ci	Other Fission and Activation Gas Release Ci	Volume Liters	Volume of Dilution Water Liters	Max. Dose Commitment Whole Body mrem	Max Dose Commitment Organ mrm
1997	1.36E -01	0.00	4.34 E-06	2.42 E-02	8.72 E+04	6.11 E+10	0.042	0.073
1998	7.30E -01	0.00	4.75E-06	8.76E-02	3.26E+05	3.15 E+10	0.057	0.085
1999	1.07 E-02	0.00	1.08E-06	1.19E-02	5.36 E+04	1.80 E+09	0.312	0.527
2000	7.00 E-02	0.00	1.27 E-05	2.99 E-02	2.57 E+05	3.63 E+10	0.184	0.203
2001	2.38 E-02	0.00	2.13 E-07	6.50 E-03	6.74 E+04	4.80 E+10	0.017	0.025

3.5.6 Gaseous Waste Activity and Volume

Big Rock Point also reports, in accordance with Defueled Technical Specification 6.7.2 and 6.7.3, Annual Radioactive Environmental and Radioactive Effluent Release, including data on gaseous waste. The set of data provided in Table 3-8 provides a compilation of this information. A summary of the liquid waste effluent release report for 1997 through 2001 is provided below. Gaseous effluent release data was not available for 2002 at the time of this LTP submission; future updates may be obtained from BRP for inspection.

Table 3-8. Gaseous Waste Effluent Releases

Year	Fission and Activation Gas Release Ci	Iodines Ci	Particulates Ci	Tritium Ci	Whole Body Dose, β mrads	Whole Body Dose, γ mrads	Organ Dose mrem
1997	2.21 E +01	3.50 E –03	3.68 E –03	2.31 E 00	1.04 E –02	1.81 E –02	1.88 E -02
1998	0.00	0.00	2.24 E –04	1.19 E 00	0.00	0.00	2.05 E-04
1999	0.00	0.00	1.64 E-04	1.19 E00	0.00	0.00	2.05 E -03
2000	0.00	0.00	9.77E-06	1.19 E00	0.00	0.00	1.66 E-03
2001	0.00	0.00	3.71E-06	1.19 E00	0.00	0.00	1.66 E-03

3.6 COORDINATION WITH OUTSIDE ENTITIES

The decommissioning and termination of BRP 10 CFR Part 50 license involves, among others, the US NRC, the US State of Michigan Occupational Safety and Health Administration, the Environmental Protection Agency, the Michigan Department of Environmental Quality, State Historic Preservation Office, the State Fire Marshall, the US Coast Guard, the US Department of Energy, and local law enforcement agencies. Chapter 8 of this LTP discusses Federal, State, and local requirements.

The BRP Citizen's Advisory Board (CAB) was established in 1995 to enhance opportunities for public involvement in the BRP decommissioning process. Citizen's Advisory Board was established for the purpose of providing a formal channel of communication and feedback to BRP and Consumers Energy management on issues relevant to the plant in the area of operations and decommissioning. Citizen's Advisory Board members include Petoskey and Charlevoix City Managers, Charlevoix and Emmet County Commissioners, Petoskey and Charlevoix Chamber of Commerce representatives, a local environmentalist, local township representatives, and local business representatives. Members or member affiliations of the BRP CAB may change over the decommissioning project; membership changes will not alter the effectiveness of this group.

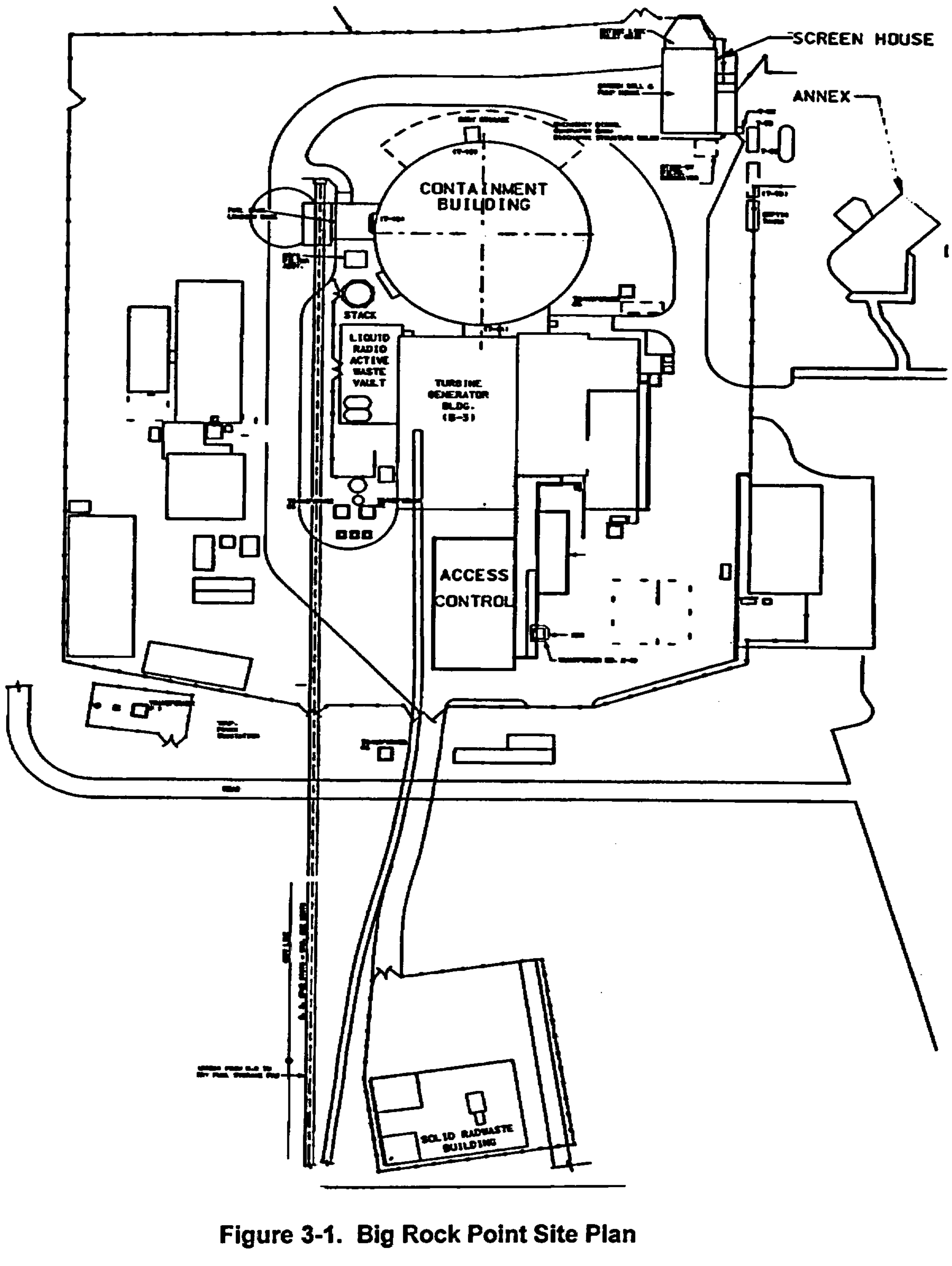
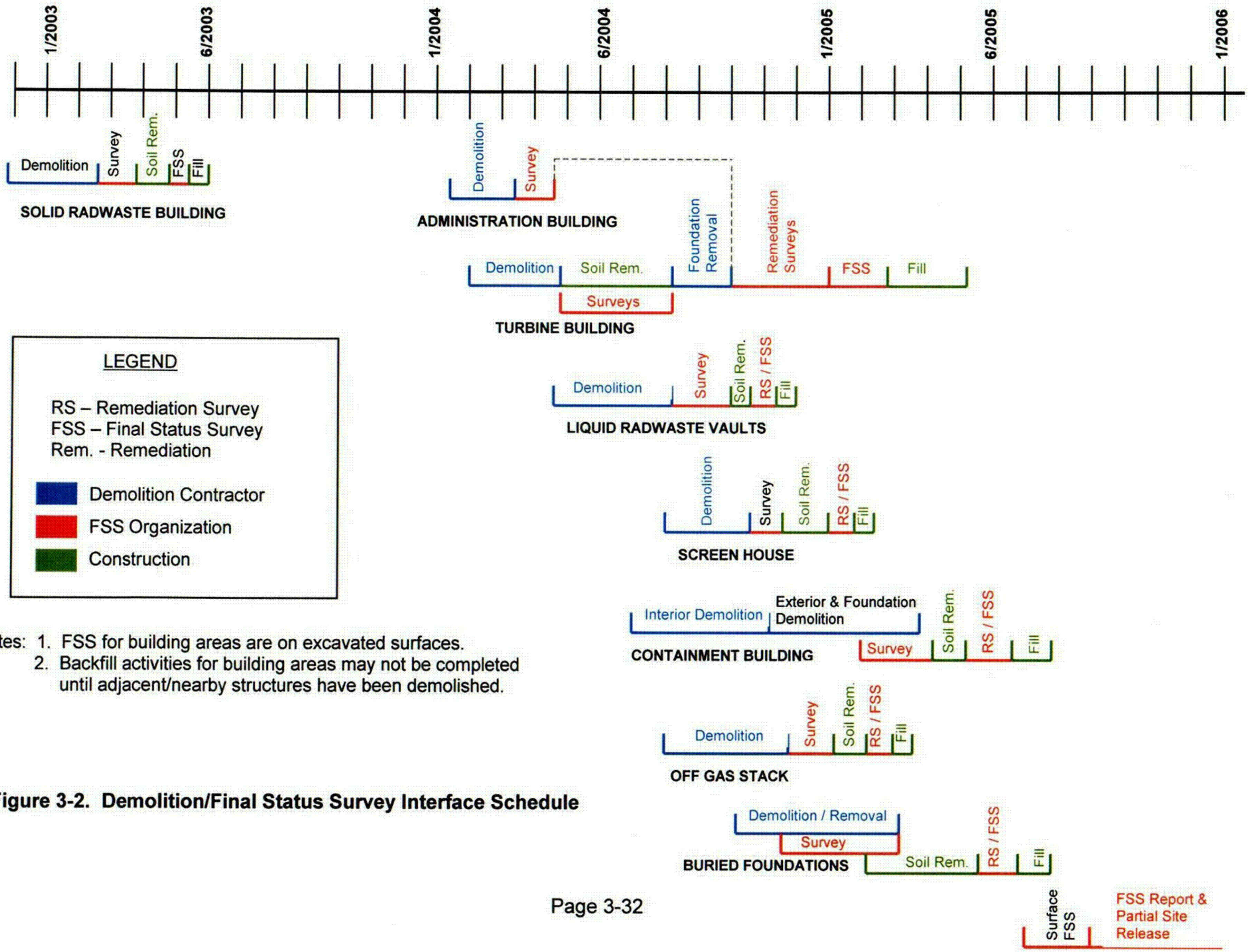


Figure 3-1. Big Rock Point Site Plan



Notes: 1. FSS for building areas are on excavated surfaces.
2. Backfill activities for building areas may not be completed until adjacent/nearby structures have been demolished.

Figure 3-2. Demolition/Final Status Survey Interface Schedule

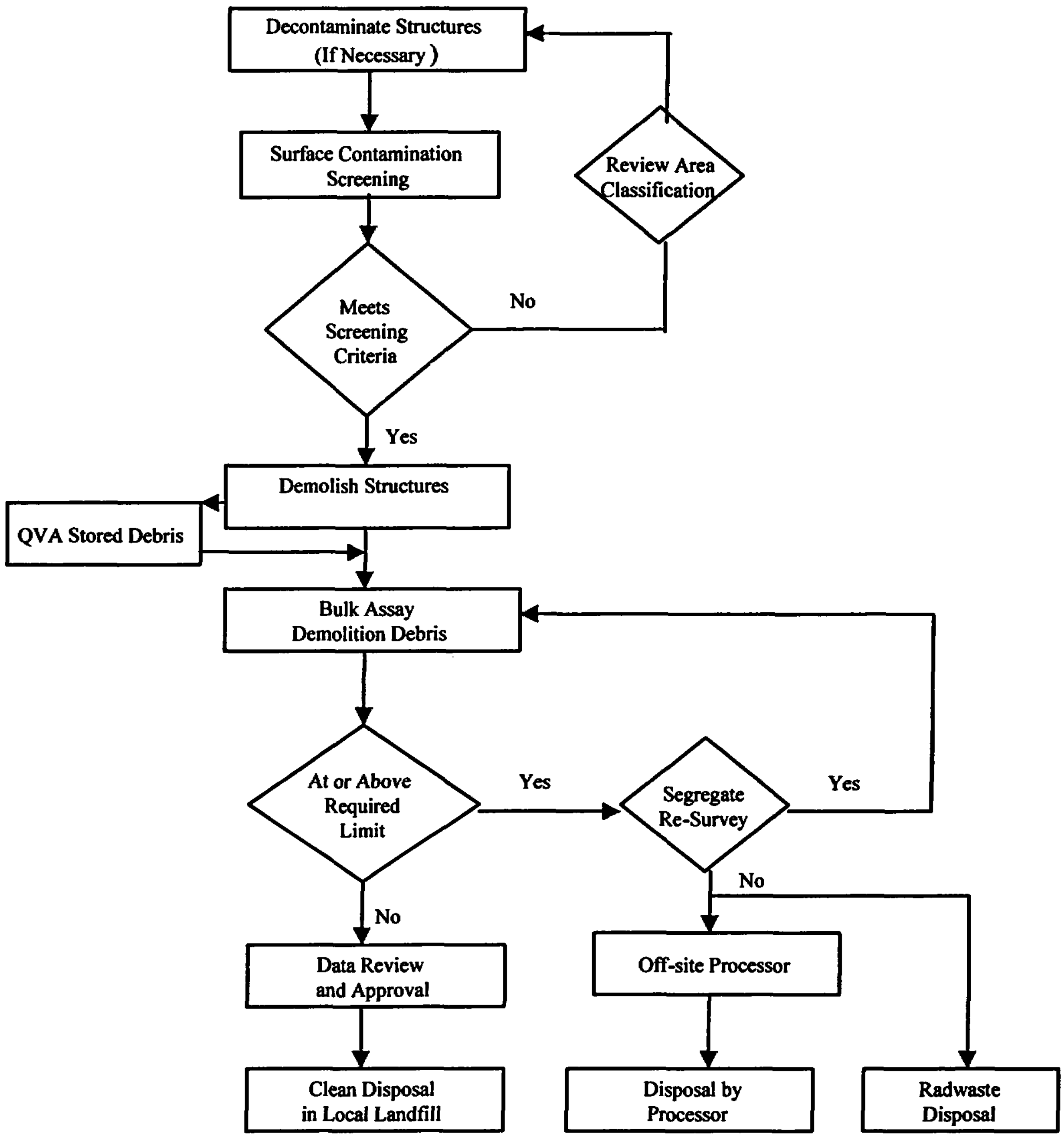
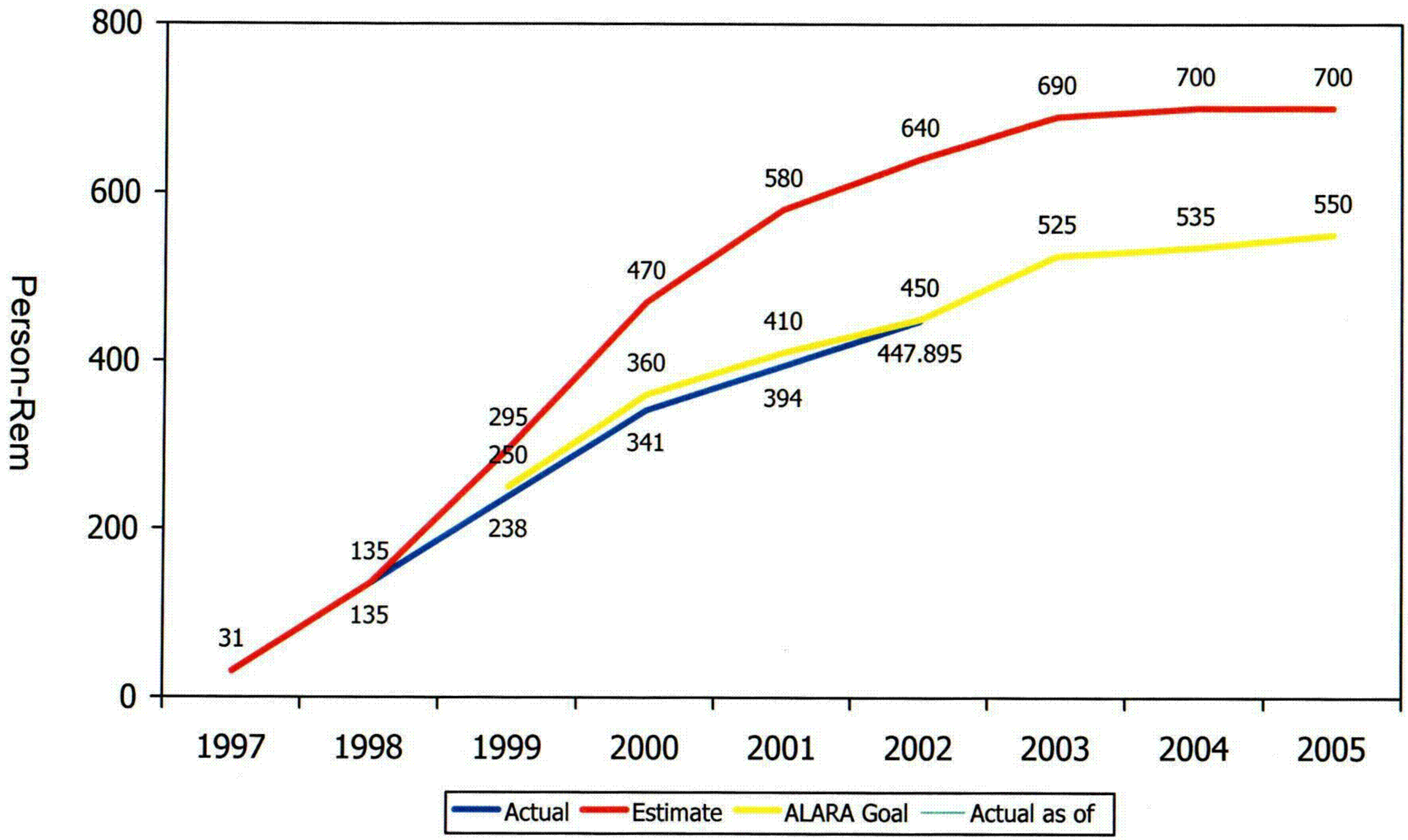


Figure 3-3. Flow Diagram of the Demolition Debris Disposal Process



C-02

Figure 3-4. Dose-Goal, Estimate and Actual

3.7 REFERENCES

3-1 Big Rock Point Decommissioning Administrative Procedure, Volume 1, D3.1, Decommissioning Work Packages

3-2 Big Rock Point Decommissioning Administrative Procedure, Volume 1, D3.1.3, Milestone Work Packages

3-3 Big Rock Point Defueled Technical Specifications, Volume 2

3-4 Big Rock Point Post Shutdown Decommissioning Activities Report (PSDAR), Revision 2, March 26, 1998

3-5 Big Rock Point Updated Final Hazards Summary Report (UFHSR), Revision 10, September 18, 2002

3-6 Consumers Energy Quality Program Description for Nuclear Power Plants, (CPC-2A) (Part 1) – Big Rock Point

3-7 Letter from Consumers Energy, Big Rock Point to U.S. Nuclear Regulatory Commission, *Request for Approval of Proposed Disposal Procedures in Accordance with 10 CFR 20.2002*, May 18, 2001

3-8 Letter from Consumers Energy, Big Rock Point to U.S. Nuclear Regulatory Commission, *Request for Approval of Proposed Disposal Procedures in Accordance with 10 CFR 20.2002*, June 20, 2001

3-9 Letter from U.S. Nuclear Regulatory Commission to Consumers Energy, Big Rock Point, *Environmental Assessment and Finding of No Significant Impact Related to Request for Approval of Proposed Disposal Procedures in Accordance with 10 CFR 20.2002*, December 3, 2001

3-10 Letter from U.S. Nuclear Regulatory Commission to Consumers Energy, Big Rock Point, *Proposed Disposal Procedures in Accordance with 10 CFR 20.2002 (TAC NO. MB1463)*, February 5, 2002

3-11 U.S. Nuclear Regulatory Commission NUREG-0554, *Single-Failure-Proof Cranes for Nuclear Power Plants*, January 1979

3-12 U.S. Nuclear Regulatory Commission NUREG-0586, *Final Generic Environmental Impact Statement (FGEIS) on Decommissioning of Nuclear Facilities*, August 1998

3-13 U.S. Nuclear Regulatory Commission NUREG-0586, Supplement 1, *Generic Environmental Impact Statement (GEIS) on Decommissioning of Nuclear Facilities*, October 2002

3-14 U.S. Nuclear Regulatory Commission NUREG-0612, *Control of Heavy Loads at Nuclear Plants*, July 1980

3-15 U.S. Nuclear Regulatory Commission NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, April 2000

3-16 U.S. Nuclear Regulatory Commission Regulatory Guide 1.16, *Reporting of Operating Information – Appendix A, Technical Specifications*, August 1975

3-17 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, *Standard Format and Content of License Termination Plans for Power Reactors*, January 1999

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4.0 SITE REMEDIATION PLAN

4.1 INTRODUCTION

This section of the License Termination Plan (LTP) describes remediation actions proposed for use in the decommissioning of Big Rock Point (BRP). In accordance with 10 CFR 20, Appendix 4-B, remediation will assure that dose less than or equal to the annual dose limit of 25 mrem/year, reduced to as low as reasonably achievable (ALARA), is met for the critical population group. In addition, the portion of this dose attributable to drinking water sources has been subtracted from the dose available to radionuclides in soil in the calculation of site-specific Derived Concentration Guideline Level (DCGL). LTP Section 6.8.2 provides a complete discussion of site-specific DCGL derivation. Remediation will be performed under the Radiation Protection Program in a manner that will assure that both worker doses and environmental doses will be ALARA.

4.1.1 Purpose

This remediation plan describes the use of characterization data to define the requirements for remediation, projected remediation volumes, and methods to be utilized to reduce site soils to levels that will allow the Greenfield site to meet unrestricted release criteria.

4.1.2 Scope

This remediation plan addresses current and planned remediation actions for soil and water at BRP, as well as remediation that may be performed on sediments in portions of the plant discharge canal. Because the site is intended to be released after the removal of all aboveground and most belowground structures, (see LTP Section 4.2), this remediation plan addresses only briefly the activities involved with structures, systems and components (SSCs). Further description of previous and future decommissioning activities involving SSCs may be found in LTP Chapter 3, *Identification of Remaining Site Dismantlement Activities.*

Remediation of soils has occurred as a normal course of business at BRP, both during and following the plant operational phase. Early remediation has served to limit both lateral and vertical spread of residual radioactivity with time. However, residual radioactivities in areas of high dose rate, and in areas inaccessible to workers during power operations, have remained for future remediation. This plan focuses on current and future remediation actions. Past remediations are described in Appendix 2-B of LTP Chapter 2, *Site Characterization.*

This section also discusses BRP's commitment to health physics procedures and approved work practices applicable to remediation, in order that worker health and safety is assured, and releases of radioactive materials to the environment are minimized. It also addresses ALARA for dose to the public following unrestricted site release.

4.2 REMEDIATION ACTIONS

The BRP site will be decommissioned to the Greenfield definition provided in LTP Section 1.4. All equipment, buildings and structures, including subsurface foundations are expected to be removed, either as material meeting unrestricted release criteria, material meeting the requirements of the BRP 10 CFR 20.2002 license requirements for disposal of demolition debris, or material that requires disposal as radioactive waste [Reference 4-5]. Contaminated soil and groundwater, as the only materials remaining at the time of site release, are the only materials that are subject to remediation.

NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, guidance for the content of LTP Chapter 4 calls for discussion of activities involving remediation of all structures with residual radioactivity levels in excess of unrestricted use limits [Reference 4-8]. Due to the Greenfield definition, no structures will be remediated for release with the site. Consequently, specific discussion is limited to remediation considerations for surface and subsurface soil, and surface and subsurface ground water at this site. However, general discussion of structures, systems and components (SSCs) dismantlement activities is provided in LTP Section 4.2.1, and in LTP Chapter 3.

Soil remediation may be initiated as residual radioactivity is identified in areas accessible for safe and effective remediation. If an area is not immediately available, remediation may be initiated when the area becomes available due to structure removal, when dose rates from surrounding activities and facilities are ALARA, and when the area can be adequately protected from recontamination. The approximate schedule for completion of soil remediation in various areas is linked to the schedule for final status surveys. This schedule is provided in LTP Section 5.8.6.2 and summarized in Table 5-11. Figure 3-2 of this LTP illustrates the relationships among demolition and remediation activities, remediation surveys and the Final Status Survey (FSS). These schedules are meant to illustrate the general order and duration of activities, and are not to be taken as licensing commitments.

4.2.1 Structures, Systems and Components

All equipment, components and structures, including subsurface foundations, but excluding an on-site septic drainfield and the plant intake water pipe, will be removed rather than undergo remediation[1]. The drainfield is being retained in place, with the concurrence from local and state health officials. The drainfield is in an impacted area, but characterization studies show that it will meet site release criteria without remediation. The plant intake water pipe will be plugged and retained in place, in order to minimize environmental impact. The intake water pipe is a non-impacted structure, and exists predominately in public waters.

[1] Facilities supporting the ISFSI will remain on site until after spent fuel has been transported to a permanent federal repository.

Following system and component removal, contaminated structures are removed in a general sequence of perimeter buildings first, followed last by the Containment Building and its critical support facilities such as liquid radwaste processing, heating, ventilating and air conditioning (HVAC), and plumbing. The Containment Building's steel sphere remains in place until all fuel is removed to dry fuel storage, and the major sources of contamination, including fuel pool structure, reactor vessel and reactor bioshield, either have been removed, or have been packaged for shipping. This sequence assures that radioactive materials are appropriately protected against uncontrolled release to the environment during the dismantlement process. Table 3-5 of this LTP provides projected dates for future decommissioning activities.

Aboveground portions of structures are decontaminated to safe levels prior to demolition. This serves to limit the possibility of environmental residual radioactivity, and also aids in maximizing the amount of material that may be disposed of by unrestricted release or as industrial demolition debris under the provisions of the NRC approved10 CFR 20.2002 alternate disposal procedure request [Reference 4-5]. Belowground portions of structures subsequently are removed, evaluated for appropriate disposal, and shipped to the appropriate disposal facility. Soils uncovered by these activities are surveyed and remediated, as necessary, prior to *backfill. Soils with radionuclide concentrations higher than acceptable for retention* onsite (higher than DCGL for the mixture) will be packaged and shipped for disposal as radioactive waste.

4.2.2 Soil and Water

Considering the 35-year operating history of BRP, relatively little soil contamination has occurred. Factors that have mitigated residual radioactivity external to the containment sphere are discussed in LTP Section 2.2.4.2. However, one significant event of groundwater contamination occurred due to leakage of approximately 20,000 gallons of steam condensate from the Condensate Storage Tank in 1984. Details of this event are provided in LTP Section 2.2.5.3.d.3. Other liquid spills also occurred as identified in LTP Chapter 2, but all were much less significant. Analyses of water samples from groundwater monitoring wells (1994-present), numerous samplings of groundwater obtained by core borings through the Turbine Building and Containment floors, and water samples from excavations during ongoing demolition activities, have identified only tritium to be present above environmental LLDs (lower limits of detection).

4.2.2.1 Soil

There are four areas where sampling has shown that limited residual radioactivity levels remain present in soils above site-specific DCGLs. To place the extent of residual radioactivity in perspective, as of midsummer, 2002, only seven of more than 1100 soil sample locations had indicated residual radioactivity levels higher than the site-specific DCGLs. The areas with sufficient residual radioactivity to warrant consideration as candidates for remediation are [Reference 4-2]:

1) West of and Beneath Turbine Building

This area has been affected by past leaks from the outside Condensate Storage Tank. Although some soil was removed following the incidents during the plant operational phase, the volume of soil still requiring remediation is estimated to be approximately 10 cubic meters. Ratio of Cs-137 to Co-60 from soil samples in this area is approximately 200.

2) Pipe Tunnel, South of Containment Building

Expansion joints and penetrations through the floor allowed condensed liquid from steam leaks to soak into soil beneath the pipe tunnel floor in the area adjacent to the Containment Building on its south side. There also were other recorded incidents of contamination in this area (see Appendix 2-B of Chapter 2). The volume of soil estimated to require removal is approximately 30 cubic meters. Ratio of Cs-137 to Co-60 from soil samples in this area is approximately 0.05.

3) North of Condensate Storage Tank

An area west of the Turbine Building and north of the Condensate Storage Tank location that was affected by Waste Hold Tank overflows and possible contamination from radioactive filters and resins during transfer from the Liquid Waste Vaults (the vaults are just east of the residual radioactivity area). This area has been partially remediated in the past, but is still estimated to require removal of approximately 40 cubic meters of contaminated soil. Ratio of Cs-137 to Co-60 from soil samples in this area is approximately 20.

4) Plant Discharge Canal

The area to be remediated is approximately 600 square meters. Residual radioactivity averages less than 15 cm deep into sediments that lie interspersed with and in some low flow areas, on top of, cobble. Remediation of contaminated sediment is expected to result in approximately 30 cubic meters of waste. This area has been impacted by permitted discharge into Lake Michigan, and currently is under water. However, due to the Greenfield objective of returning the lakeshore to original contours, the current deep area adjacent to the plant may be filled. The filled area would be accessible by the public at the current low water level of Lake Michigan. The purpose of remediation would be to treat the area as if it were part of the plant site. Ratio of Cs-137 to Co-60 from sediment samples in this area is approximately 0.6.

5) Other Potential Areas

An estimate for all additional potential, but yet unidentified residual radioactivities above site-specific DCGLs has been made based on a residual radioactivity depth of 15 cm for 200 square meters of subsurface foundation area (30 cubic meters). This volume includes foundation areas not yet available for characterization, including a suspect area beneath the Liquid

Radwaste Vault where high dose rates have not allowed core borings to be obtained directly through the floor to this point in time. Ratio of Cs-137 to Co-60 in these areas is assumed to be at the site-wide soil average of approximately 1.2.

Based on the known and potential residual radioactivity areas described above, the total soil volume that contains at least some residual radioactivity above site-specific DCGLs is expected to be approximately 150 cubic meters. Residual radioactivity in soil above the site-specific DCGLs will be removed from the above areas, (and any other locations which may later be found greater than site-specific DCGL concentrations), and will be disposed of as radioactive waste. Where appropriate, site excavation procedures, work permits and other appropriate site requirement documents will address dewatering, analysis and disposition of liquids accumulating within excavations, control of dust, control of contamination, safety requirements, and other constraints. As needed, additional investigations will be performed to ensure that any changing soil residual radioactivity profile during the remediation actions is adequately identified and addressed [Reference 4-6].

Soil remediation equipment will include, but not be limited to, back and track hoe excavators, front-end loaders and hand digging. As practical, when the remediation depth approaches the soil interface region for unacceptable and acceptable residual radioactivity levels, a squared edge excavator bucket design or similar technique may be used. This minimizes the mixing of contaminated soils with acceptable lower soil layers as would occur with a toothed bucket. However, due to the cobbly nature of many plant soils, toothed buckets or hand digging may be required where residual radioactivity occurs in the vicinity of imbedded cobble and rocks. In the plant discharge area, either hand digging or specialized underwater operations may be performed to remove pockets of sediment with concentrations above site-specific DCGLs for disposal as radioactive waste.

The site characterization process has established the location and extent of residual radioactivity in soil. Details may be found in Chapter 2 of this LTP. However, it should also be noted that soil volume estimates (or implied volumes of soil used in LTP Chapter 6, *Compliance with the Radiological Criteria for License Termination,* dose calculations, etc.) may vary slightly from section to section.

Depth profiles have shown that residual radioactivity in soils greater than site-specific DCGLs do not extend deeper than 0.30 meter except for one sample in vicinity of the waste hold tank leaks and a second sample at a floor penetration in the pipe tunnel. In both cases, nearby samples did not detect deep residual radioactivity. Thus, the extent of subsurface residual radioactivity is known to be very limited. However, local residual radioactivity not detected by core samples performed to date may be found when foundations or subsurface piping are removed. While these and other excavations are open, evaluations will be performed by use of sampling, scanning, in situ measurements, or other approved method defined by plant procedures, to assess the need for remediation. License Termination Plan Section 5.4 discusses soil sampling and survey methods.

4.2.2.2 Groundwater

Tritium exists in groundwater as described in Sections 2.4.5.3 and 6.8.2.1 of this LTP. However, at the current time, no tritium above the Environmental Protection Agency (EPA) drinking water guidelines has been detected in the aquifer suitable as a drinking water supply (maximum detected in the a potential drinking water aquifer is 1560 pCi/l, well below the EPA guideline of 20,000 pCi/l). Twenty monitoring wells are utilized to monitor for tritium and other potential radionuclides in groundwater. No radionuclide other than tritium has been detected in groundwater at greater than environmental LLD levels [Reference 4-4].

In the latter part of 2002, samples were taken of soil and water beneath the Turbine Building slab during the removal of an equipment drain. The water contained tritium at levels, corrected for decay, evaluated to be consistent with those originally present in the condensate tank at the time of the leak in 1984. The water appeared to have been trapped by subsurface structures (turbine pedestal structure, load-supporting wall footings, a concrete mud-mat poured beneath the footing elevations, and in at least one location, a subsurface concrete slab). The water occurred in a sand fill that exists between the Turbine Building floor slab (still in place) and the subsurface concrete. Removal efforts for the confined water, currently in progress, involve the installation of well points into this area. The water is being pumped at a rate of a few gallons per hour to holding tanks. Disposal will be by approved liquid discharge, or shipment by approved means to a licensed disposal site. Pumping is from the areas of highest tritium activity. Drawdown from surrounding areas should result in infiltration of uncontaminated groundwater as the pumping continues. The tritium concentration is expected to be below 20,000 pCi/l by mid-2003. A pumping process similar to this also may be used for tritium identified in water confined beneath the south side of the containment structure (see Table 2-8 of LTP Chapter 2).

The EPA guideline of 20,000 pCi/l does not apply to water confined in subsurface structures beneath the Turbine Building and Containment, which is unsuitable by health standards as a potable water supply. However, reduction to below this level serves to assure that this water source would not eventually reach the deeper potable supply at concentrations capable of elevating that aquifer to or above the EPA guideline value.

4.2.2.3 Surface Water

No radionuclides of plant origin have been detected in surface water at the plant site, other than those detected in samples taken at the discharge weir during permitted release of liquid batch discharges to the discharge canal. Consequently, no remediation of surface waters is required. Discharge canal sediments, the only place that radionuclides of plant origin remain from those permitted discharges may be remediated as discussed in LTP Section 4.2.2.1.

4.3 REMEDIATION ACTIVITY IMPACT ON THE RADIATION PROTECTION PROGRAM

The Radiation Protection Program approved for decommissioning is similar to the program in place during 35 years of commercial power operation. During power operations, contaminated SSCs were decontaminated in order to perform maintenance or repair actions, and contaminated soils were removed to prevent further spread of contamination into the environment. The techniques were similar to those being used for decommissioning. Many components were removed and replaced during operation. The techniques used for component removal did not differ significantly from those being used during decommissioning. However, chemical decontamination and radioactive decay both allow component removal at lower dose levels than observed during the plant operational period. Reactor system chemical decontamination at BRP was performed immediately after final shutdown. Subsequent component removal and dismantlement activities have been performed at much lower dose rates than would have been present otherwise.

The BRP Radiation Protection Program adequately controlled radiation and radioactive contamination during power operation, although plant age and design provided significant challenges to the limitation of worker dose. Since shutdown, dose rates have been lowered by chemical decontamination, and ALARA engineers have been able to plan major personnel dose evolutions such as reactor vessel and bioshield removal late in the decommissioning schedule. At these later dates, radioactive decay of activated materials (primarily Co-60) has lowered component dose rates another 50%. The combination of these factors has greatly improved the plant's ability to limit personnel exposure to radiation. Consequently, Radiation Protection Program effectiveness has improved for decommissioning in comparison to power operations.

With specific reference to soil remediation, application of the Greenfield definition for site release is seen as a major factor in reduction of worker dose. With removal of equipment and structures from an area prior to major soil remediation efforts, only the radioactive materials within the soil remain present as a source of radiation exposure to the remediation worker. Since average concentration of soil radioactivity is well below the DCGL, worker doses are minimal under such conditions.

4.3.1 Commitment to Radiation Protection Procedures

Big Rock Point intends to continue under its Part 50 license throughout the decommissioning process. Radiation Protection procedures are required under the Defueled Technical Specifications for implementation of the requirements of 10 CFR 20. The Defueled Technical Specifications, in conjunction with the requirement for maintaining an Offsite Dose Calculation Manual (containing relocated technical specifications), also specify requirements for effluent controls, environmental monitoring, reporting requirements and limitation of offsite dose within the guidelines of 10 CFR 50, Appendix I.

Radiation Protection procedures developed and utilized during plant operation continue to be utilized for the protection of plant and contract personnel, and for protection of the environment. Changes and additions to these procedures, to improve applicability to specific circumstances during the decommissioning process, are made under plant administrative controls that include appropriate reviews and approvals. Training and instructions provided to plant and contract workers follow the requirements of 10 CFR 19.12, and reports of personnel exposure are provided pursuant to the applicable sections of 10 CFR Parts 19 and 20. The requirements of these regulations are implemented by means of plant procedures.

4.3.1.1 Remediation Procedures

Procedures applicable to remediation of soils during plant operation have been found to be equally applicable to use during decommissioning, although the criteria for extent of remediation has changed. During plant operation, various ALARA and cost criteria were used to determine the extent of remediation. For decommissioning, criteria for release of an area specifically for the FSS are in accordance with criteria for survey area turnover (see LTP Section 5.1.3.2). In general, remediation will not be considered complete for decommissioning purposes until the sum of the DCGL fractions is shown to be less than 1.0 for Class 1 and Class 2 survey areas and less than 0.5 for Class 3 areas. Remediation surveys are discussed in LTP Chapter 5, *Final Status Survey Plan.*

The soil remediation process begins with a remediation area walkdown and subsequent evaluation of appropriate work processes necessary to assure worker safety, environmental protection and effective removal of residual radioactivity. Deep excavations require consideration of need for shoring or alternatively, requirements for angle of slope, dependent on soil type, to avoid cave-in. Potential need for personnel monitoring and engineering controls to limit inhalation and airborne environmental release is assessed, as is the need for controls to prohibit or limit contamination spread into or out of the excavation by other means, such as by surface water flow or transfer by equipment. Potential for the presence of subsurface installations such as piping or electrical conduit is evaluated. Based on these and other inputs, a work plan is developed, along with any radiation work permit(s) necessary for the work to be performed.

The processes of soil handling and analysis for excavated soils are described in LTP Section 5.4.2.4. LTP Section 5.4.1 provides descriptions of instruments and survey methods applicable both to remediation and FSS activities. Approved procedures are utilized for all such activities.

No unique safety or remediation issues have been identified associated with remediation of soil or groundwater at BRP. As described above, decommissioning remediation processes and procedures are similar to those utilized during plant operation, but differ in terms of endpoint of remediation. Consequently, decommissioning remediation work is coordinated closely with FSS personnel in order to ensure appropriate monitoring and control of excavated materials and the level of remediation required for the specific area (see LTP Sections 5.3.6.4 and 5.3.6.5). At the conclusion of the remediation process, FSS personnel perform a walkdown of the remediated area, and the area may be turned over for final survey, as appropriate.

4.4 ALARA EVALUATION

As described in LTP Chapter 6, dose assessment scenarios were evaluated for the residual radioactivity that could remain in soils. The ALARA analysis is conservatively based on a modified resident farmer scenario. The resident farmer critical group applies to existing open land areas and all site areas where standing buildings have been removed.

4.4.1 Dose Models

As discussed above, the critical group for BRP under the Greenfield definition is the resident farmer. Accordingly, the ALARA evaluations for remediation actions in this section use the parameters for population density, evaluation time interval, monetary discount rate and area that are applicable to the resident farmer scenario.

4.4.2 Methods for ALARA Evaluation

NUREG-1727, *Decommissioning Standard Review Plan*, Section 7.0, ALARA Analysis, states, "Licensees or responsible parties that remediate building surfaces or soil to the generic screening levels established by the NRC staff do not need to demonstrate that these levels are ALARA" [Reference 4-7]. The DCGLs for BRP soil are site-specific values. Approximately half of the site-specific DCGLs are below the generic screening levels, and half above. Appendix 4-C-1 includes screening levels, and Appendix 4-C-2 includes site-specific DCGLs for the radionuclides present in BRP soils. Chapter 6 of this LTP provides details on the BRP site-specific DCGL derivation. Because some of the site-specific DCGLs exceed screening level values, BRP is conservatively providing an ALARA evaluation of the remediation actions for soil.

The ALARA evaluations were performed in accordance with the guidance in NUREG-1727. The dose contribution of each radionuclide in the BRP mixture was evaluated, and the contributions summed. The principle equations used for the calculations are presented in Appendix 4-A. The evaluation determines if the benefit of the dose averted by the remediation is greater or less than the cost of the remediation. When the benefit is greater than the cost, additional remediation is required. Conversely, when the benefit is less than the cost, remediation is not required beyond the point that the average member of the critical population group would receive 25 mrem/year.

4.4.3 Remediation Methods and Cost

Remediation methods and costs for other than soil remediation are not applicable to BRP, due to the plant's commitment of restoration to a Greenfield condition. Additional information on the restoration process is provided in LTP Chapter 3. Groundwater remediation is discussed in LTP Section 4.2.2.2.

4.4.4 Remediation Cost Basis

Unit costs for soil excavation are established for 150 m^3 of contaminated soil excavated and shipped from the site, and also for a lower volume of 33 m^3. The larger volume of 150 m^3 represents soil at the average concentration of contaminated soils measured in the characterization analyses (average concentration for the total mixture of radionuclides is approximately 1.6 pCi/g, per Case 1 of Appendices 4-C-1 and 4-C-2). Description of component areas and volumes is provided in LTP Section 4.2.2. The lower volume of 33 m^3 is commensurate with removal of only the small volumes of residual radioactivity where the sum of the site-specific DCGL and screening level fractions of the individual nuclides equals unity for the shipped mixture, or approximately 7 pCi/g, per Case 2 of Appendices 4-C-1 and 4-C-2. The 150 m^3 and 33 m^3 soil volume cases are presented here, as Case 1 and Case 2, respectively (see Section 4.5), to illustrate the variations in costs and benefit as a function of volume. Remediation volumes from BRP are likely to be nearest 150 m^3, and are not expected to be less than 33 m^3. If volumes greater than 150 m^3 are generated, the cost/benefit ratio will remain above unity: the calculations show that the larger the volume, the greater the cost/benefit ratio.

Soil remediation unit costs (dollars per cubic meter) were determined for methods appropriate to the volumes removed. Power excavation equipment, hand shoveling, or specialized underwater sediment removal systems in the case of the discharge canal, may be employed in contaminated soils removal. Non-radioactive materials such as rocks and cobble may be separated out either during the excavation process, or later. For sediments, dewatering and/or drying may be required to meet shipping requirements. In each case, the costs are based on the net volume of soil shipped for disposal.

Certain activities which are wholly or partially attributable to other decommissioning activities, such as removal of asphalt paving, foundations and floors of buildings, and excavation of soils over subsurface pipes and drains, are not included in the cost of remediation, since these activities would be undertaken whether or not soils in the area were contaminated. Likewise, dewatering activities are excluded, since dewatering is routinely performed to assist in safe removal of subsurface structures, as well as to assist in radiological surveys and remediation, when required.
Dewatering small volumes of tritium trapped by underground structures has not been

Case 1 Soil at the average concentration of radionuclides identified by characterization studies to date, (150 m^3), and

Case 2 Soil at the generic screening and site-specific DCGL levels for the mixture of radionuclides identified by characterization. This total nuclide concentration is approximately a factor of 4.5 times as high as the mean identified for the 150 m^3 case, and thus represents a volume of 1/4.5 = 0.22 times 150 m^3, or 33 m^3.

For Case 1, the most significant cost factors are the soil excavation cost factor, at \$1808/m^3, and the waste disposal cost factor, at \$1349/m^3. Total cost factor is \$3192/m^3. These are provided in Appendix 4-B, along with associated component costs and appropriate reference bases.
For Case 2, unit excavation cost is lower (\$887/m^3) due to lower power equipment utilization, and disposal costs are greater (\$2398/m^3) because the higher radionuclide concentrations require a greater fraction of waste to be shipped to Envirocare in the state of Utah. Total unit cost for this case is \$3340/m^3. Further description of these costs follow, and a summary of data for both cases is provided in Appendix 4-B.

4.5.1 Waste Disposal Cost

Round trip truck transportation:
Clive, Utah (Envirocare site) round trip: 5140 km.
Oak Ridge, TN round trip: 2830 km
Average distance = 3985 km

Case 1 Disposal Cost:
The BRP contract for soil disposal is \$X per pound, if sent to Utah, and \$Y per pound, if meeting criteria for disposal at Oak Ridge.
1.6 kg/liter of soil, or $(f_1)(1{,}600\ kg/m^3)(0.454\ lb/kg)\ (\$X/lb) = (726.4)(f_1)(\$X)/m^3$
1.6 kg/liter of soil, or $(f_2)(1{,}600\ kg/m^3)(0.454\ lb/kg)\ (\$Y/lb) = \underline{(726.4)(f_2)(\$Y)/m^3}$
$\$1{,}349/m^3$
Note: X and Y are proprietary values defined by negotiated contract; f_1 and f_2 are fractions of waste sent to Utah and Tennessee, respectively.

Case 1 Accident Cost:

Volume of 13.6 m^3 of soil per truck shipment, per NUREG-1727, Table D2, with average distance of 3985 km/ round trip, total miles for 150 m^3 (11 trips) equals 43,800 km. Rail transport for soil is not anticipated due to the small volumes of soil involved. The distance and haul volume are used for determining transport accident cost in accordance with NUREG-1727, Appendix D, (Equations D6) and the fatal accident rate of 3.8E-08/km given in NUREG-1727, Table D2.

Transport accident cost totals \$5,000.

Case 2 Disposal and Accident Costs

Due to the higher concentrations, more waste is assumed sent to Clive, Utah. Unit cost, calculated as for Case 1 above = $2398/m^3.

Three truckloads, conservatively assuming a haul distance of 5140 km = 15420 km: Transport accident cost totals $1,758.

4.5.2 Worker Accident Costs

To determine worker accident cost in accordance with NUREG-1727 Appendix D, the accident rate of 4.2E-08/hour was applied to NUREG-1727 Equation (D5). The hours used for labor cost, times number of field workers on a crew, were used for worker accident cost. Accident cost of $60 for Case 1 and $30 for Case 2 are not significant contributors to total cost.

4.5.3 Worker Dose

Costs associated with worker dose are a function of the hours worked and the workers' radiation exposure for the task. Slightly elevated background dose rates, with an average of (0.5 mrem/hour) used for all soil remediation activities. This low value is due to application of the Greenfield definition: most soil remediation follows removal of structures and equipment and, therefore, dose rates are significantly reduced. At $2000 per person-rem, this cost totals $474 for Case 1 and $236 for Case 2.

4.5.4 Labor Costs

Manpower cost assumptions are based on contracts established with the principal site contractors and for Consumers Energy site personnel. Appendix 4-B utilizes an hourly cost for the remediation crew of $180/hour in Case 1 and $160/hour for Case 2. These crew rates assume a manual labor rate of $35/hour, and technician, supervisory and specialized equipment operator rates of $50/hour. Engineering and technical support costs are included as identified in Appendix 4-B. Unit labor costs are $237/m^3 for Case 1 (high equipment utilization), and $402/m^3 for Case 2 (predominantly manual labor with lower volume productivity).

4.5.5 Equipment Costs

Hand tools, power excavation equipment and underwater sediment removal equipment are among the tools that may be utilized for soil remediation. Total cost of $210,700 is calculated for this cost category in Case 1. A cost of $1000 is included for equipment mobilization in Case 1 and $500 is included for Case 2. The bulk of Case 1 equipment cost derives from the underwater sediment removal activities at the discharge canal. Case 1 unit cost is $1405/m^3. For Case 2, there is limited powered land excavation, but no underwater work. Backhoe and loader costs are $100/hour, and dump trucks $50/hour, excluding the $50/hour operator cost that is assigned to labor costs, above. Equipment utilization cost of $16,000 is assigned for Case 2 and equipment unit cost is calculated to be $485/m^3.

4.5.6 Schedule Delay Cost

A contract clause requires payment of fees for certain delays in schedule. A total delay cost of $25,000 is assumed for Case 1. The type of delay most likely to occur is assumed to be that of major equipment (specialized underwater sediment removal equipment) availability, either due to delivery problems, or operability problems during use. No delay charges are assumed in Case 2, since hand digging and small equipment operations are less likely to have major impact on schedule.

4.6 PRESENT WORTH OF FUTURE COLLECTIVE AVERTED DOSE

The remediation cost of $3192/m^3 derived for soil in Appendix 4-B for Case 1 and $3340 for Case 2 was compared to the benefit of the dose averted through the remediation action. As noted earlier, activity concentration in the lower volume (33 m^3) Case 2 is a factor of approximately 4.5 times as high as in Case 1. The benefits of averted dose were calculated using Equations D1 and D2 in NUREG-1727 as modified to account for multiple radionuclides as presented in Equation A-2 of this LTP Section. Appendices 4-C-1 and 4-C-2 provide spreadsheet tables of these calculations for both volumes against NRC screening levels and site-specific DCGLs, respectively. The averted dose, being proportional to excavated soil concentration, also is a factor of approximately 4.5 times higher for Case 2 than for Case 1. It is conservatively assumed that the soil is spread over a 10,000 m^2 area, 15 cm thick in both cases, even though there would not be enough soil to actually meet this condition.

The parameters used in the equations were taken from NUREG-1727, Table D2, as suitable for the resident farmer scenario. The calculation of present worth of the future collective averted dose was performed in Case 1 for the observed average total concentration equaling 1.6 pCi/g, with individual nuclides included as low as 0.0093 pCi/g (Sr-90) [Reference 4-1]. Concentration inputs are shown in the spreadsheets of Appendices 4-C-1 and 4-C-2. Results of all radionuclides are summed. Based on NRC default screening levels, a value of $415 is obtained for the total benefit from the collective averted dose in Case 1 and $1889 is obtained for Case 2. Based on site-specific DCGLs, a value of $398 is obtained for the total benefit from the collective averted dose in Case 1 and $1704 is obtained for Case 2.

4.7 ALARA CALCULATION RESULTS

The total remediation cost may be compared with the benefit from the collective averted dose using Equation D8 from NUREG-1727. This equation is presented in Appendix 4-A as Equation A-2. The calculational inputs and outputs are presented in Appendix 4-C-1 for screening levels, and in Appendix 4-C-2 for site-specific DCGLs.

4.7.1 ALARA Cost Benefit (Concentration/DCGL) for Soil Excavation

Due to high removal and shipping costs, excavation of the expected quantities of soil from the site show that the residual radioactivity is ALARA at both the site-specific DCGL and generic screening level for the mix without additional actions in either Case 1 or Case 2. A farm resident population of four individuals on areas of 10,000 m^2 is assumed for each of the cost/benefit calculations. However, at a contamination depth of 0.15 m, the 150 m^3 volume (Case 1) would cover only 10% of this area, and the 33 m^3 volume (Case 2) would cover only approximately 2.2% of this area. Use of these fractions would reduce dose, and each of the cost/benefit ratios given below would be increased. Such increased values would provide increased support that concentrations need not be reduced below either the site-specific DCGLs or the generic screening levels to be ALARA.

For Case 1, a Concentration/DCGL value of $478,800/$415 = 1178 is obtained with plant soil concentrations evaluated against the surface soil screening values from NUREG-1727 Table C2.3. When evaluated against site-specific DCGLs, the value is $478,800/$398 = 1203.

For the 33 m^3 soil excavation volume (Case 2), the screening level case gives $110,223/$1889 = 58.3, and the site-specific DCGL case gives $110,223/$1704 = 64.7.

Since the Concentration/DCGL values are greater than one for all cases of soil remediation, remediation below the 25 mrem/year dose limit is not justified. It is observed that the cost/benefit ratio is lower for smaller volumes of higher activity concentration, and increases with larger volumes of lower activity concentration.

It will be noted that all calculations in this LTP chapter equate both the generic screening levels and the site-specific DCGLs to 25 mrem/year. This conforms to the guidance in NUREG-1727. However, 25 mrem/year gives slightly conservative doses for specific concentrations of radionuclides in soil at BRP since, as described in Section 6.8.2 of this LTP, site-specific DCGLs for BRP soils are based on an annual dose of 24.219 mrem/year. This lower dose basis compensates for a conservatively calculated annual dose of 0.782 mrem/year from tritium in an aquifer that could be used as a future drinking water supply [Reference 4-3]. Validity of the ALARA cost/benefit analysis result is not affected by this minor difference.

4.8 REFERENCES

4-1 Big Rock Point Engineering Analysis EA-BRP-SC-02-06, *Surrogate Measurement of Hard-to-Detect Nuclides*

4-2 Big Rock Point Engineering Analysis EA-BRP-SC-02-04, *Radionuclides Present in Onsite Soil and Water*

4-3 Big Rock Point Engineering Analysis EA-BRP-SC-02-05, *Effect of Subsurface Soil and Water Contamination and the Development of Surface Soil DCGLs*

4-4 Big Rock Point Offsite Dose Calculation Manual, Volume 25A, Section I, *Procedural and Surveillance Requirements,* (Relocated Technical Specifications)

4-5 Letter from U.S. Nuclear Regulatory Commission letter to Big Rock Point dated February 5, 2002, *Big Rock Point Plant - Proposed Disposal Procedures in Accordance with 10 CFR 20.2002*

4-6 U.S. Nuclear Regulatory Commission NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, August 2000

4-7 U.S. Nuclear Regulatory Commission NUREG-1727, *Decommissioning Standard Review Plan*

4-8 U.S. Nuclear Regulatory Commission NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, April 2000

A.1 Present Worth of Future Collective Averted Dose

The present worth of future collective averted dose may be estimated by use of Equation D2 of NUREG-1727:

$$PW = (P_D)(A)(0.025)(F)\left[\frac{Conc}{DCGL}\right] * \left[\frac{1-e^{-(r+\lambda)N}}{r+\lambda}\right]$$

Equation A-1

Where:

Pd = Population Density for Resident Farmer ($0.0004/m^2$)

A = Area of residual radioactivity (10,000 m^2) for this resident farmer calculation

0.025 = Annual dose (rem/y) to the average resident farmer from residual radioactivity at the DCGL

F = Fraction of the residual radioactivity removed by the remedial action. Assumed equal to 0.95.

Conc = Average concentration of residual radioactivity being evaluated in units of pCi/g of soil

DCGL = Derived concentration guideline level that represents dose rate of 25mrem/year in pCi/g

r = Monetary discount rate ($0.030y^{-1}$)

λ = Radiological decay constant for a radionuclide in units of y^{-1}

N = Length of time collective dose is calculated (1,000y)

A.2 Concentration/DCGL

Concentration/DCGL is the ratio of soil concentration to the derived concentration guideline that is calculated as a cost to benefit ratio for the remediation process. If the ratio exceeds 1.0, further costs are not justified for remediation to below 25mrem/year. If the cost-benefit ratio is less than 1.0, it is cost effective to reduce levels to a level below 25 mrem/year that gives a ratio of 1.0 or greater.

Equation D8 of NUREG-1727 may be expressed for the summation of all nuclides as follows:

$$\frac{Conc}{DCGL} = \frac{Cost_T}{\sum_i^n (\$2000)(P_D)(0.025)(Df_i)(F)(A)\left[\frac{1-e^{-(r+\lambda_i)N}}{r+\lambda_i}\right]}$$

Equation A-2

Where:

$$Df_i = \text{Dose Fraction} = \frac{(Di)}{\sum_i^n (Di)} = \frac{(Conc/DCGL)_i}{\sum_i^n (Conc/DCGL)_i}$$

D_i = (25)(Conc/DCGL)$_i$ = annual dose contribution (mrem) from individual residual nuclide at site release This equals (25)(Df_i) when the mixture is at the site-specific DCGL or generic screening level for the mix.

$\sum_i^n (Di)$ = annual dose total from all residual nuclides in soil at time of site release (mrem).

And the remaining symbols are as defined for Equation A-1.

Df_i relates dose of a single radionuclide of a mixture to total dose from the mixture. A mixture that will provide 25 mrem/year may be termed the mixture DCGL. Setting Df_i equal to 1.0 in this equation (thus reducing it to the equivalent of Equation A-1, rearranged for expression of Conc/DCGL) will result in a value of Conc/DCGL at the nuclide's observed concentration, rather than at the mixture's DCGL.

Case 1. Soil Excavation Remediation –150m^3 at observed average concentration

Volume Evaluated For Unit Cost Determination: 150 m^3 , Total mix conc. = 1.6 pCi/g
Primary Crew Size: 3. Supervisor, 1; RP, 0.5; Laborers, 1; Equipment operator, 0.5
Support Personnel: 0.4. Resident and Schedule Engineers, 0.15; HP/ Environmental, 0.25 Hourly Cost: $180
Excavation Rate, including beneficiation: 1.9 m^3/h
Hours: 158 (150 m^3/1.9m^3/h)(2.0) [2.0 = contingency]
Labor Cost: $35,550 (Includes 25% for contingency and non-field support staff)
Equipment Costs: Land, $23,700; Underwater Equipment, $186,000 (consumables $3,100)
Mobilization Costs: $1000
Prime Contract Delay Cost: $25,000
Total Excavation Cost: $271,250 = $1808/ m^3
Waste Generation: 150 m^3 (5,300 ft^3/35.315 ft^3/m^3)
Waste Disposal Cost: $202,000 ($1349/m^3)
Worker Accident Cost @158x3 hours = 2.0E-5 fatalities: $60 Per NUREG-1727
Transportation Accident Cost @ 43,800 km (1.66E-3 fatalities by truck): $5000 Per NUREG-1727, Appendix D
Worker Dose, 158 hours for 3 in field @ 0.0005 rem/hour: 0.237 person-rem = $474
Total Costs: $478,800
Cost per m^3: $3,192

Case 2. Soil Excavation Remediation – 33 m^3 at default screening level or site-specific DCGL level for nuclide mix

Volume Evaluated For Unit Cost Determination: 33 m^3, Total mix Concentration = 6.87 pCi/g
Primary Crew Size: 3.8, Supervisor, 1; RP, 0.5; Laborers, 2; Equip Operator, 0.3
Support Personnel: 0.4, Resident and Schedule Engineers, 0.15; HP/ Environmental, 0.25 Hourly Cost: $160
Excavation Rate, with minor beneficiation: 0.8 m^3/h
Hours: 62 (33 m^3/0.8m^3/h)(1.5) [1.5 = contingency]
Labor Cost: $13,260 (Includes 25% for contingency and non-field support staff)
Equipment Costs: $15,500 (consumables $2,000)
Mobilization Costs: $500
Prime Contract Delay Cost: None
Total Excavation Cost: $29,260 = $887/ m^3
Waste Generation: 33 m^3 (1165 ft^3/35.315 ft^3/m^3)
Waste Disposal Cost: $79,134 ($2398/m^3)
Worker Accident Cost @ 62 hours x 3.8 workers = 9.9E-6 fatalities: $30 Per NUREG-1727
Transportation Accident Cost @ 15,420 km (5.86E-4 fatalities by truck): $1758 Per NUREG-1727, Appendix D
Worker Dose, 62 hours x 3.8 workers in field @ 0.0005 rem/hour: 0.118 person-rem = $236
Total Costs: $110,223
Cost per m^3: $3340

Case 1 PW @ Actual mean Soil Concentration vs Screening Level (150 cu. m)

Radionuclide	λ	Pd*A*F/40	Conc	Screen	N	r	PW
	(1/yr)	(unitless)	(pCi/g)	(pCi/g)	(years)	(1/y)	(unitless)
Cs-137	0.0231	0.095	0.6680	1.10E+01	1000	0.03	1.09E-01
Co-60	0.131749	0.095	0.5720	3.80E+00	1000	0.03	8.84E-02
Sr-90	0.0246619	0.095	0.0094	1.70E+00	1000	0.03	9.56E-03
Mn-54	0.5749069	0.095	0.0370	1.50E+01	1000	0.03	3.87E-04
Fe-55	0.2665385	0.095	0.2694	1.00E+04	1000	0.03	8.63E-06
H-3*	0.0563415	0.095	0.048	1.10E+02	1000	0.03	4.80E-04
Total			1.6038				0.207
Benefit of Averted Dose =		$2000*PW					$ 414.98

Case 2 PW @ Screening Level for Mix in 33 cubic m

Radionuclide	λ	Pd*A*F/40	Conc	Screen	N	r	PW
	(1/yr)	(unitless)	(pCi/g)	(pCi/g)	(years)	(1/y)	(unitless)
Cs-137	0.0231	0.095	3.04E+00	1.10E+01	1000	0.03	4.95E-01
Co-60	0.131749	0.095	2.60E+00	3.80E+00	1000	0.03	4.02E-01
Sr-90	0.0246619	0.095	4.26E-02	1.70E+00	1000	0.03	4.35E-02
Mn-54	0.5749069	0.095	1.68E-01	1.50E+01	1000	0.03	1.76E-03
Fe-55	0.2665385	0.095	1.23E+00	1.00E+04	1000	0.03	3.93E-05
H-3*	0.0563415	0.095	2.18E-01	1.10E+02	1000	0.03	2.19E-03
Total			7.30E+00				0.944
Benefit of Averted Dose =		$2000*PW					$ 1,888.99

Calculation of Gross Activity Screening Level for Mix[1]

Radio-nuclide	Original Conc	Screening Level	Conc/ Screen	Conc @ Mix Scrn	Dose Check
	(pCi/g)	(pCi/g)	-	(pCi/g)	(mrem/yr)
Cs-137	0.6680	1.10E+01	6.07E-02	3.04E+00	6.91E+00
Co-60	0.5720	3.80E+00	1.51E-01	2.60E+00	1.71E+01
Sr-90	0.0094	1.70E+00	5.50E-03	4.26E-02	6.26E-01
Mn-54	0.0370	1.50E+01	2.47E-03	1.68E-01	2.81E-01
Fe-55	0.2694	1.00E+04	2.69E-05	1.23E+00	3.07E-03
H-3*	0.048	1.10E+02	4.36E-04	2.18E-01	4.97E-02
Total	1.6038		2.20E-01	7.30E+00	2.50E+01

* Original Tritium Assumes 3% soil moisture content (by weight) and water at 1,000 pCi/l

[1] Equation 4-4 of NUREG-1575

Case 1 PW @ Actual mean Soil Concentration vs DCGL (150 cu. m)

Radionuclide	λ	Pd*A*F/40	Conc	DCGL	N	r	PW
	(1/yr)	(unitless)	(pCi/g)	(pCi/g)	(years)	(1/y)	(unitless)
Cs-137	0.0231	0.095	0.6680	1.32E+01	1000	0.03	8.72E-02
Co-60	0.131749	0.095	0.5720	3.21E+00	1000	0.03	1.05E-01
Sr-90	0.024662	0.095	0.0094	2.48E+00	1000	0.03	6.58E-03
Mn-54	0.574907	0.095	0.0370	1.37E+01	1000	0.03	4.24E-04
Fe-55	0.266538	0.095	0.2694	3.58E+05	1000	0.03	2.41E-07
H-3**	0.056341	0.095	0.048	3.28E+02	1000	0.03	1.63E-04
Total			1.6038				0.199
Benefit of Averted Dose =	$2000*PW						$ 398.12

Case 2 PW @ DCGL for Mix in 33 cubic meters of Soil

Radionuclide	λ	Pd*A*F/40	Conc	DCGL	N	r	PW
	(1/yr)	(unitless)	(pCi/g)	(pCi/g)	(years)	(1/y)	(unitless)
Cs-137	0.0231	0.095	2.86E+00	1.32E+01	1000	0.03	3.73E-01
Co-60	0.131749	0.095	2.45E+00	3.21E+00	1000	0.03	4.48E-01
Sr-90	0.024662	0.095	4.00E-02	2.48E+00	1000	0.03	2.82E-02
Mn-54	0.574907	0.095	1.58E-01	1.37E+01	1000	0.03	1.82E-03
Fe-55	0.266538	0.095	1.15E+00	3.58E+05	1000	0.03	1.03E-06
H-3**	0.056341	0.095	2.05E-01	3.28E+02	1000	0.03	6.96E-04
Total			6.87E+00				0.852
Benefit of Averted Dose =	$2000*PW						$1,704.36

Calculation of Gross Activity DCGL for Mix*

Radionuclide	Original Conc (pCi/g)	Site-Spec. DCGL (pCi/g)	Conc/ DCGL -	Conc @ Mix DCGL (pCi/g)	Dose Check (mrem/yr)
Cs-137	0.6680	1.32E+01	4.88E-02	2.86E+00	5.22E+00
Co-60	0.5720	3.21E+00	1.78E-01	2.45E+00	1.91E+01
Sr-90	0.0094	2.48E+00	3.79E-03	4.00E-02	4.05E-01
Mn-54	0.0370	1.37E+01	2.70E-03	1.58E-01	2.89E-01
Fe-55	0.2694	3.58E+05	7.53E-07	1.15E+00	8.05E-05
H-3**	0.048	3.28E+02	1.48E-04	2.05E-01	1.58E-02
Total	1.6038		2.34E-01	6.87E+00	2.50E+01

*Equation 4-4 of NUREG-1575
**Original Tritium Assumes 3% soil moisture content (by weight) and water at 1,000 pCi/l

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5.0 FINAL STATUS SURVEY PLAN

5.1 INTRODUCTION

The Big Rock Point (BRP) Final Status Survey (FSS) Plan has been prepared using the applicable regulatory and industry guidance. This plan will be used to develop site procedures and work instruction to perform the FSS of the BRP site.

5.1.1 Purpose

The FSS Plan describes the final survey process used to demonstrate that the BRP site complies with radiological criteria for unrestricted use specified in 10 CFR 20.1402, i.e., annual dose limit of 25 mrem plus ALARA for all dose pathways. Nuclear Regulatory Commission (NRC) regulations applicable to radiation surveys are found in 10 CFR 50.82(a)(9)(ii)(D) and 10 CFR 20.1501(a) and (b).

5.1.2 Scope

Big Rock Point intends to release site land using a phased approach. The first phase includes the majority of the site land (approximately 560 acres) scheduled for release after all demolition, remediation and FSS activities associated with plant operation are complete. The second and final phase of site release includes the Independent Spent Fuel Storage Installation (ISFSI) following spent fuel removal, facility dismantlement and any required remediation. Once both these phases are complete the BRP site license under 10 CFR Part 50 will be terminated. It is possible that release of non-impacted portions of the site could occur prior to completing demolition activities, should Consumers Energy management decide accordingly.

This FSS Plan addresses requirements applicable to the first phase of site release. This Plan address only land areas that are identified as contaminated or potentially contaminated (impacted) resulting from activities associated with commercial nuclear plant operation. All site buildings and subsurface structures and equipment, with the exception of the facilities supporting ISFSI, will be demolished and removed from the site prior to the FSS on surface spills. Surveys on excavated areas will be performed prior to backfilling (see Sections 5.2.2.3, 5.2.4.1, and 5.4.2.1 for detailed discussion on FSSs).

To the extent practical, excavation area surveys will be designed and conducted in accordance with NUREG-1575 guidance for surface surveys [Reference 5-10].

5.1.3 Final Status Survey Preparation and Implementation Overview

The FSS Plan contained in this chapter will be used as the basis for developing FSS procedures and applying existing procedures to the FSS process. Section 5.1.4 contains a list of regulatory documents used as guidance in preparing the FSS Plan. Figure 5-1 provides an overview of the FSS process. Quality Assurance requirements are outlined in Section 5.8 and apply to activities associated with decommissioning and FSS activities.

An FSS Package will be produced for each survey area; this survey package is a collection of documentation detailing survey design, survey implementation and data evaluation for a final status survey of an area. The sections below describe specific elements of the FSS organization, preparation and implementation. All processes associated with final status surveys will be conducted in accordance with approved site procedures.

5.1.3.1 FSS Organization

The general FSS organization will consist of supervision, technical specialists, work planning coordinators, field coordinators, data analysts, and technicians. Since the FSS organization has not been implemented at the time of LTP development, it is expected that specific job titles may vary over the period of project execution. These titles are used within this document to describe various functional areas of responsibility and do not necessarily refer to specific job titles. Refer to Section 5.8.2.1 for additional detail on the FSS organization.

5.1.3.2 Survey Preparation

Survey preparation is the first step in the final status survey process and occurs after any necessary remediation is completed. In areas where remediation is required, a remediation survey or equivalent evaluation will be performed to confirm that remediation was successful prior to initiating final status survey activities. Remediation surveys, turnover surveys, or equivalent evaluation, for areas not requiring remediation, will be performed using the same process and controls as a final status survey so that data from these surveys may be used as part of the final status survey data. In order for survey data to be used for final status survey, it must have been designed and collected in compliance with Sections 5.3 through 5.5 and the area controlled in accordance with Section 5.2.4. Following turnover/remediation surveys or post-remediation evaluation, the FSS is performed. Areas to be surveyed are isolated and/or controlled to ensure that radioactive material is not reintroduced into the area from ongoing demolition or remediation activities nearby and to maintain control of the area. Section 5.2 address specific survey preparation requirements and considerations.

- *Survey Package Initiation*
  Each survey unit and package is assigned a unique identification number. To allow continuity of area identification, the protocol used for identifying survey areas during the characterization survey is used, as appropriate. Survey unit identification numbers differing from those used for characterization survey may be necessary if survey boundaries are modified. Survey unit nomenclature is defined in Section 2.4.1.

- *Review of Historical Site Assessment and Characterization Survey*
  Historical data applicable to the survey area are reviewed; this information is used for survey design and is filed in the survey package. Sources of historical data include:

- Historical Site Assessment,
- Characterization Survey,
- Remediation Files,
- Background Study,
- Survey Records,
- Personnel Accounts, and
- 10 CFR 50.75(g) File

- *Survey Area Walkdown*
  A walkdown is performed to gather information about the physical characteristics of the survey area. The walkdown provides an opportunity to determine if any physical or safety related interferences are present that may affect survey design or survey implementation, and to determine any support activities necessary to implement surveys. Typical walkdown observation items include observation that demolition work is complete and all debris has been removed, grade is suitable for surveys, and the work area is safe for survey activities. The walkdown is documented and filed in the survey package. In conjunction with or following the walkdown, representative maps of the survey area are prepared.

- *Survey Area Readiness*
  Prior to performing final status surveys all decommissioning, remediation and housekeeping activities identified as having the potential to affect the area are completed. Radiation Protection personnel may perform surveys to verify that the area meets specific radiological criteria for performance of the FSS. These surveys include readiness surveys that are conducted to 1) support remediation activities, 2) determine when a site or survey unit is ready for the final status survey, and 3) provide updated estimates of site-specific parameters to use for planning the FSS (see NUREG-1575, Section 5.4.1). In order to differentiate among these three applications, the following terminologies are utilized:

  a. Remedial Action Support Survey

     An in-process survey performed to expedite the remediation process. This survey provides information to assist in cost effective remediation, but is not normally expected to provide information sufficient to demonstrate compliance with final release criteria. However, data from these surveys may provide valid information for use in evaluation of readiness for turnover in preparation for the FSS.

  b. Turnover Survey (a.k.a. Post-Remediation Survey)

     A survey, or data compilation from Remedial Action Support Surveys, that may be used to evaluate the completion status of remediation activities. If this is a formal survey designed to meet the objectives of a final status survey as described by NUREG-1575, it may serve as the FSS for the remediated area.

c. Characterization Survey (Remediation Area or Other)

This survey, if found necessary by virtue of the turnover survey or the evaluation of other applicable data, is performed in accordance with NUREG-1575, Section 5.3. As with the Turnover Survey, if the Characterization Survey is designed to meet the objectives of a final status survey as described by NUREG-1575, it may serve as the FSS.

The survey unit can then be posted to indicate that the area is controlled for the performance of final status surveys. Controls are implemented to prevent contamination of areas during and following final status surveys, as appropriate.

5.1.3.3 Survey Design

The impacted area is organized into survey units and classified by potential for residual radioactivity as Class 1, Class 2, or Class 3. The size of the survey unit is based on survey unit classification requirements in accordance with the guidance provided in NUREG-1575 (MARSSIM). The survey design process establishes the methods and performance criteria used to conduct the FSS and defines the sample point locations and type of measurements to be performed for each survey unit. Section 5.2 contains a detailed discussion of survey design requirements.

A survey map is prepared for each survey unit and a reference grid is superimposed on the map to allow use of an (x,y) coordinate system. Random numbers between 0 and 1 are generated, which are then multiplied by the maximum x and y axis values of the sample grid. This provides coordinates for each random sample location, or a random start location for systematic grid, as appropriate. The measurement/sample locations are plotted on the map. Each measurement/sample location is assigned a unique identification code, which identifies the measurement/sample by survey unit, and sequential number. The appropriate instruments and detectors, instrument operating modes and survey methods to be used to collect and analyze data are specified.

Written survey instructions that incorporate the requirements set forth in the survey design are completed. Direction is provided, as applicable to survey design, for selection of instruments, count times, instrument modes, survey methods, required documentation, alarm/investigation setpoints, alarm actions, background requirements and other appropriate instructions. In conjunction with the survey instructions, survey data forms may be prepared to assist in survey documentation. Alternatively, electronic data recording systems may be utilized. The survey design is reviewed and quality verification steps applied to ensure that appropriate instruments, survey methods and sample locations have been properly identified.

5.1.3.4 Survey Data Collection

After preparation of a survey package, the final survey data are collected. Trained and qualified personnel will perform the necessary measurements using calibrated instruments in accordance with approved procedures and instructions contained in the survey package. Section 5.5 addresses FSS data collection requirements.

Survey areas and/or locations are identified by gridding, markings, or flags as appropriate. An FSS field coordinator performs a pre-survey briefing with the survey technicians during which the survey instructions are reviewed and additional survey unit considerations are discussed (e.g., safety). The technicians gather instruments and equipment as indicated and perform surveys in accordance with the appropriate procedures and survey package specifications. Technicians are responsible for documenting survey results and maintaining custody of samples and instrumentation. At the completion of surveys, technicians return instruments and prepare samples for analysis. Survey instruments provided to the technicians are prepared in accordance with appropriate procedures and the survey instructions. Instrument calibration and performance checks are performed in accordance with applicable procedures. Data are reviewed to flag any measurements that exceed investigation criteria so that appropriate investigation surveys and remediation can be performed as necessary.

Following completion of a FSS, the need for Quality Control (QC) surveys (replicate surveys, sample recounts, etc.) is determined. If necessary, a QC survey package is developed. QC measurement results are compared to the original measurement results. If QC results do not agree with the original survey, an investigation is performed. Section 5.8 provides additional detail regarding QC survey requirements. Following investigation, the survey data validity is assessed (see below).

5.1.3.5 Data Evaluation

Survey data assessment is performed to verify that the data are sufficient to demonstrate that the survey unit meets the unrestricted use criterion. Statistical analyses are performed on the data and compared to pre-determined investigation levels (see Section 5.3.6.2). Depending on the results of the data assessment and any required investigation, the survey unit may either be released or require further remediation, reclassification, and/or resurvey. Assumptions and requirements in the survey package are reviewed for applicability and completeness; additional data needs are identified during this review. Specific data assessment requirements are contained in Section 5.6.

A review is performed of survey data and sample counting reports to verify completeness, legibility and compliance with survey design and associated instructions. As directed by FSS supervision, the following types of activities may be performed:

a. Convert data to reporting units,
b. Calculate mean, median and range of the data set,
c. Review the data for outliers,
d. Calculate the standard deviation of the data set,
e. Calculate minimum detectable concentration (MDC) for each survey type performed, and
f. Create posting, frequency or quantile plots for visual interpretation of data.

Computer programs may be utilized for these activities. FSS personnel include data quality verifications in their evaluations of statistical calculations; integrity and usefulness of the data set and the need for further data or investigation is also included. The data evaluation process is documented and filed in the survey package.

5.1.3.6 Final Status Survey Package Completion

Survey results are documented by survey unit in corresponding survey packages. Each FSS Package may contain the data from the several survey units that are contained in a given survey area. The data are reviewed, analyzed, and processed and the results documented in the FSS Package. This documentation file provides a record of the information necessary to support the decision to release the survey units for unrestricted use. An FSS Report will be prepared to provide the necessary data and analyses from survey packages for submittal to the NRC. Section 5.7 addresses reporting of survey results and conclusions.

5.1.4 Regulatory Requirements and Industry Guidance

This FSS Plan has been developed using the guidance contained in the following documents:

a. NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, (August 2000) [Reference 5-10].

b. NUREG-1505, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*, Revision 1 (June 1998 draft) [Reference 5-12].

c. NUREG-1507, *Minimum Detectable Concentrations With Typical Radiation Survey Instruments for Various Contaminants and Field Conditions* (June 1998) [Reference 5-12].

d. NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, (December 1998, draft) [Reference 5-13].

e. NUREG-1727, *NMSS Decommissioning Standard Review Plan*, (September 2000) [Reference 5-14]

f. Regulatory Guide 1.179, *Standard Format and Content of License Termination Plans for Nuclear Power Reactors*, (January 1999) [Reference 5-15].

Other documents used in the preparation of this plan are listed in the References section (see Section 5.9).

Big Rock Point anticipates both the NRC and the Michigan Department of Environmental Quality (MDEQ) – Radiological Protection Measurement and Standards Unit may choose to conduct confirmatory measurements during BRP FSS activities. The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the FSS and associated documentation demonstrate that the site is suitable for release in accordance with the criteria established in 10 CFR Part 20, subpart E.

5.2 DEVELOPMENT OF SURVEY PLAN

5.2.1 Radiological Status

The following sections provide a summary of site characterization and dose modeling results applicable to development of the BRP FSS Plan.

5.2.1.1 Identification of Radiological Contaminants

Big Rock Point plant conducted extensive radiological characterization of the site property between 1997 and 2000 to identify and document residual contamination resulting from nuclear plant operation. The effort included reviews of historical information as well as physical measurements of onsite soils and groundwater. LTP Chapter 2, *Site Characterization,* contains a detailed discussion of this effort [References 5-4 and 5-6].

5.2.1.2 Dose Modeling Summary

Dose models based on NUREG/CR-5512, Volume 1 and RESRAD Version 6.2 were used to calculate Derived Concentration Guideline Levels (DCGLs) for the BRP site. These dose models translate residual radioactivity levels into potential radiation doses to the public and are defined by three factors: (1) exposure scenario, (2) exposure pathways, and (3) exposed critical group. The scenarios presented in NUREG/CR-5512 address the major exposure pathways of direct exposure to penetrating radiation and inhalation and ingestion of radioactive materials. These scenarios also identify the critical group. The "critical group" is the group of individuals reasonably expected to receive the greatest exposure to residual radioactivity within the assumption of the particular land use scenario.

Since all buildings, above- and below-grade structures, and equipment within the industrial area will be demolished and removed from the site, a modified resident farmer scenario was selected to develop site-specific soil DCGLs for BRP. Due to site-specific environmental parameters, the modified residential farmer scenario is considered a very conservative dose model for the BRP site. Chapter 6, *Compliance with the Radiological Criteria for License Termination*, contains the basis and results of dose modeling performed for BRP and describes modifications of the standard resident farmer scenario to exclude meat and milk pathways. Table 5-1 provides a list of all potential radionuclides that may be present in onsite soils and the corresponding DCGLs.

Table 5-1. BRP Site-Specific Radionuclides and Soil DCGL Values

Radionuclide	25 mrem/yr Limit Open Land Areas (Surface and Subsurface Soils) (pCi/g)
H-3	3.28 E+02
Mn-54	1.37 E+01
Fe-55	3.58 E+05
Co-60	3.21 E+00
Sr-90	2.48 E+00
Cs-137	1.32 E+01
Eu-152*	7.36 E+00
Eu-154*	6.79 E+00
Eu-155*	2.87 E+00

* Europium is included due to the potential to contaminate soil from concrete demolition activities.

5.2.1.3 Tritium Evaluation for Soils

The final site survey will address tritium in soils either by direct tritium analyses of soil samples or by means of assigning conservative tritium soil concentrations based on the highest observed tritium concentration in any of the three onsite aquifers and a soil moisture content of 8.75% which represents the moisture capacity for Alpena sandy gravelly loam present in the contaminated area. This latter method is the equivalent of assuming that the aquifer of highest tritium concentration is used for irrigation of crops. The water concentration to be utilized in this calculated soil concentration will be based on the highest concentration found in any of the onsite monitoring wells at the time of the FSS.

A calculation that utilizes a bounding tritium concentration of 20,000 pCi/l is presented in Appendix 5-A. At the 20,000 pCi/l concentration for water, soil concentration for tritium is 1.03 pCi/g. This represents an annual dose contribution of 0.076 mrem/year.

5.2.1.4 DCGLs for Surrogate Measurements

DCGLs for CS-137 and Co-60 presented above will be modified to account for the presence of hard-to-detect (HTD) nuclides, Sr-90 and Fe-55, respectively, using surrogate ratios developed from characterization data and in accordance with NUREG-1575, Section 4.3.2 [Reference 5-8]. The equation below will be used to adjust DCGL values for Cs-137 and Co-60 to demonstrate compliance with DCGLs for Sr-90 and Fe-55, respectively.

$$DCGL_{ADJ} = DCGL_M \ x \frac{DCGL_{HTD}}{[(C_{HTD} / C_M) x DCGL_M] + DCGL_{HTD}}$$

where:

$DCGL_{ADJ}$ - DCGL adjusted for surrogate measurements

$DCGL_M$ - DCGL for measured nuclide (e.g., Cs-137)

$DCGL_{HTD}$ - DCGL for HTD nuclide (e.g., Sr-90)

C_{HTD} / C_M - Concentration ratio for HTD nuclide to measured nuclide

5.2.2 Classification of Areas

Prior to the FSS, a thorough characterization of the radiological status and history of the site will be completed. Although more than 90% complete at this time, characterization of a few inaccessible areas await further dismantlement. The methods and results from site characterization are described in Chapter 2. Initial classification of site areas is based on historical information and site characterization data and was performed following the guidance in Section 4.4 of NUREG-1575 and Appendix E of NUREG-1727. Since all buildings and structures will be demolished and removed from the site for disposal prior to the FSS, classification of building areas was not performed. Area classification ensures that the number of samples and the scan coverage are commensurate with the potential for residual contamination to exceed the unrestricted use criteria. Reclassification of a survey unit will only occur to a more restrictive classification, e.g., from Class 2 to a Class 1 area; and would occur if future data indicate that the initial classification was incorrect. Any survey unit reclassification will include an evaluation of the basis for the initial classification and also the potential for survey unit classification programmatic deficiency. The basis for any reclassification will be documented, a redesign of the survey package completed, and the redesigned survey initiated. If during the conduct of a FSS, sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit, the survey may be terminated without completing the survey unit package.

5.2.2.1 Non-Impacted Areas

Non-Impacted areas are defined as areas that have no reasonable potential for residual contamination resulting from nuclear plant operations. Non-impacted areas are shown on Figure 5-2.

5.2.2.2 Impacted Areas

Impacted areas may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. The definitions provided below are from NUREG-1727, Pages E1 and E2.

a. Class 1 areas are impacted areas that, prior to remediation, are expected to contain residual contamination in excess of the $DCGL_w$[1].

b. Class 2 areas are impacted areas that are not likely to contain residual radioactivity in excess of the $DCGL_w$.

c. Class 3 areas are impacted areas that have a low probability of containing residual radioactivity.

5.2.2.3 Initial Classification

Based on more than 1100 measurements made during the site characterization process and from information gathered during the Historical Site Assessment (HSA), all land areas were assigned an initial classification in preparation for the FSS. In areas where data were limited, the initial classification is considered to be conservative to minimize future reclassification and additional sampling. The scope of the FSS includes all BRP impacted land areas. Table 5-2 provides a summary of initial survey unit classifications (see Chapter 2, Appendix E for a more detailed discussion of survey units). Figure 5-3 depicts these area classifications in a site map. The scope and boundaries of the FSS will be increased if survey data show significant levels of radioactivity above background in peripheral areas.

Characterization was performed and reported by initial survey unit. The area designations developed for the characterization process were used, for the most part, to delineate and classify areas for final status survey. This allows characterization data to be efficiently used for final survey area classification and for estimating the sigma value for sample size determination. For operational efficiency, each of the final survey areas listed in Table 5-2 may be subdivided into multiple areas. Smaller survey areas may be necessary to enhance the efficiency of data collection, processing, and review and serve to better support the decommissioning schedule. The classification of all subdivided survey areas will be the same as indicated in Table 5-2, unless reclassified in accordance with this LTP.

Areas beneath removed foundations will undergo final status survey prior to backfill. Survey units encompassing future excavations have been classified based on characterization data and the potential for subsurface soils to contain residual radioactivity from site operations. Excavated footprints of several structures may be combined into a single survey unit for final status survey. Following the satisfactory performance of FSS on the excavated foundation footprint surface, the excavation area will be backfilled. After backfill, all soil footprints, regardless of location will be combined with the surrounding land survey area and a surface survey performed commensurate with the area classification.

[1] The "w" in $DCGL_w$ refers to the Wilcoxon Rank Sum test per MARSSIM (NUREG-1575, page 2-3) and generally represents the uniform level of residual contamination that results in the dose limit, regardless of the statistical test used. Big Rock Point intends to use the Sign Test and will still use the term $DCGL_w$ to denote soil contamination limits, see Section 5.4.2.

5.2.2.4 Classification Changes

Initial classification of site areas is based on historical information and site characterization data. Data from operational surveys performed in support of decommissioning, routine surveillance and any other applicable survey data may be used to change the initial classification of an area up to the time of commencement of the FSS as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. Areas within initial survey units may be upgraded in classification due to future requirements for laydown and storage areas during demolition activities or incorrect initial classification.

Table 5-2. Big Rock Point Initial Land Area Survey Units

Survey Unit Designation*	Description**	Initial MARSSIM Classification	Physical Size (m^2)
1	Southwest corner of Protected Area (PA) contains various storage and warehouse buildings and subsurface piping/utilities.	1	1624
2	West–central section of PA contains Maintenance/ Construction Complex (MCC) and subsurface piping/utilities.	1	2002
3	Northwestern section of PA contains subsurface structures and piping/utilities.	1	2000
4	North-central section of PA contains subsurface foundations of Equipment Lock and Containment Sphere and various piping/utilities	1	1613
5(1)	Southwest of Containment encompasses northern section underground Radwaste Vaults/Liquid Radwaste Processing System and portions of Equipment Lock, Containment Sphere and northwest Turbine Building foundations	1	687
5(2)	West of Turbine Building encompasses the Liquid radwaste storage tanks; Stack, southwest Turbine Building and storage tanks foundations and various buried piping/utilities	1	837
6	South-central section of PA contains various subsurface piping/equipment including liquid radwaste discharge line	1	1512
7	South-east section of PA contains subsurface piping/equipment include liquid radwaste discharge line	1	2160
8PT#	Area beneath Pipe Tunnel, between Containment and Turbine Buildings, contains some subsurface piping	1	313
8Sphere	Area beneath Containment contains significant subsurface foundations.	1	851
8TB##	Area beneath Turbine/Service Building contains extensive subsurface structures/foundations and significant system piping and electrical conduit.	1	1347

Survey Unit Designation*	Description**	Initial MARSSIM Classification	Physical Size (m^2)
9	Northeast corner of PA includes below-grade structures including Screenhouse foundation, portions of Containment foundation and underground storage tanks	1	2884
10	East-central section of PA includes portions of Turbine/Service and Security Buildings	1	943
11	Radwaste compound due south of PA	1	2350
12	Lake Michigan beach north of PA	2	9778
13	Lake Michigan beach east of discharge canal to east property line	3	49,370
14	Lake Michigan beach west of industrial area	3	43,334
15(1)	Wooded property due west of PA bounded by drainage ditch to the east, includes septic drainfield	2	11,611
15(2)	Wooded property west of industrial area, south of beach property	2	10,285
15(2R)	Remediated section of survey unit 15(2)	1	18
15(3)	Wooded property west of industrial area, south of beach property	3	10,056
15(4)	Wooded property west of industrial area, south of beach property	3	10,955
16	Lake Michigan beach due east of discharge canal	2	8347
17	Wooded property east of industrial area	3	263,220
18	Wooded property west of industrial area	3	84,677
19	Wooded, wetland property south of PA surrounding Radwaste Compound (Survey Unit 11)	2	11,891
59	Site property located south of US Hwy 131 (plant railroad spur)	3	34
Canal	Plant discharge canal sediment, currently below water surface	1	610
Drainage Ditch	Seasonal stream and surface water conveyance located west of the Industrial Area flowing into Lake Michigan	2	370

*Figure 5-3 depicts locations of survey units listed on this table.
**Chapter 2, Appendix 2-E provides a detailed description of BRP survey units and associated operating history.
#PT – Pipe Tunnel
##TB –Turbine Building

5.2.3 Establishing Survey Units

Survey units are areas that have similar physical characteristics and contamination levels. Survey units are assigned only one classification. The site land areas will be surveyed, evaluated, and released on a survey unit basis.

5.2.3.1 Survey Unit Size

Survey unit sizes will be selected based on area classification, survey execution logistics, building demolition sequence and applicable regulatory guidance documents.

Typical survey unit sizes for soil are listed below in Table 5-3; these are consistent with NUREG-1575 guidance. Class 1 and 2 areas provided in Table 5-2 may be further subdivided into smaller areas to meet the guidelines present in Table 5-3. If larger survey unit areas are used, a technical evaluation will be presented in the FSS Package for the specific survey unit justifying the survey unit size.

Table 5-3. Survey Unit Sizes

Impacted Area Classification	Suggested Survey Unit Land Area
1	2000 m^2
2	>2000 to 10,000 m^2
3	No Limit

5.2.3.2 Site Reference Coordinate System (Reference Grid)

A reference coordinate system is used for impacted areas to facilitate the identification of survey units within the survey area. The reference coordinate system is an X-Y plot of the site area referenced to the North American Datum (NAD) – Michigan Georeference Coordinate System. Once the reference point is established, grids may be overlaid parallel to lines of latitude and longitude.

5.2.4 Access Control Measures

5.2.4.1 Turnover

Due to the large scope of demolition activities, it is anticipated that some surveys will be performed in parallel with dismantlement activities. This will require a systematic approach to turnover of areas be established. Prior to acceptance of a survey unit for final status survey, the following conditions must be satisfied in accordance with applicable procedures. These include:

a. Decommissioning activities having the potential to contaminate a survey unit shall be complete or measures taken to eliminate such potential.

b. Tools and equipment not required for the survey must be removed, and housekeeping and cleanup shall be complete.

c. Decontamination activities in the area shall be complete.

d. Access control or other measures to prevent recontamination must be implemented.

e. Turnover or remediation surveys may be performed and documented to the same standards as final status surveys so that data can be used for the FSS.

5.2.4.2 Walkdown

The principal objective of the walkdown is to assess the physical scope of the survey unit. Walkdowns of open land areas will be completed when the final configuration of the area is known, usually near or after completion of decommissioning activities for the area. The walkdown ensures that the area has been left in the necessary configuration for FSS or that any further work has been identified. The walkdown provides detailed physical information for survey design. Details such as structural interferences or sources needing special survey techniques can be determined. Specific requirements will be identified for accessing the survey area and obtaining support functions necessary to conduct final status surveys, such as excavation shoring, interference removal, dewatering, etc. Industrial safety and environmental concerns will also be identified during this walkdown.

5.2.4.3 Transfer of Control

Once a walkdown has been performed and the turnover requirements have been met, control of access to the area is transferred from the Radiation Protection - Operations group to the FSS group. Access control and isolation methods are described in the subsection below.

5.2.4.4 Isolation and Control Measures

Since all decommissioning activities will not be completed prior to the start of the FSS, measures will be implemented to protect survey areas from contamination during and subsequent to the FSS. Decommissioning activities creating a potential for the spread of contamination will be completed within each survey unit prior to the FSS. Additionally, decommissioning activities that create a potential for the spread of contamination to adjacent areas will be evaluated and controlled. Upon commencement of the FSS for survey areas where there is a potential for re-contamination, implementation of one or more of the following control measures will be required:

- Personnel training,
- Installation of barriers to control access to surveyed areas,
- Installation of barriers to prevent the migration of contamination from adjacent or overhead areas from water runoff, etc.,
- Installation of postings requiring contamination monitoring prior to surveyed area access,
- Locking entrances to surveyed areas of the facility,
- Installation of tamper-evident devices at entrance points, or
- Routine surveys to monitor and verify adequacy of isolation and control measures.

Routine surveys will not be required for open land areas that are not normally occupied and are unlikely to be impacted by decommissioning activities. Post-FSS survey locations will be judgmentally selected for survey, based on technical or site-specific knowledge and current conditions present in or near the survey area. These surveys are primarily designed to detect the potential migration of contaminants from decommissioning activities taking place in adjacent areas.

5.3 SURVEY DESIGN AND DATA QUALITY OBJECTIVES

This section describes the methods and data required to determine the number and location of measurements or samples in each survey unit and the coverage fraction for scan surveys. The design activities described in this section will be documented in a survey package for each survey unit. Survey design includes the following:

- Data Quality Objectives (DQOs)
- Scan Survey Coverage
- Sample Size Determination
- Reference Grid and Sample Location

5.3.1 Data Quality Objectives (DQOs)

The appropriate design for a given survey area is developed during the DQO process as outlined in NUREG-1575 (MARSSIM, Appendix D). These seven steps are:

- State the problem
- Identify the decision
- Identify inputs to the decision
- Define the study boundaries
- Develop a decision rule
- Specify limits on decision errors
- Optimize the design for obtaining data

The DQO process will be used for designing and conducting all final status surveys at BRP. Each survey package will contain the appropriate information, hypothesis, statistical parameters and contingencies to support the DQO process.

5.3.2 Scan Survey Coverage

The area covered by scan measurement is based on the survey unit classification as described in NUREG-1575 (MARSSIM) and as shown in Table 5-4 below. A 100% scan coverage of Class 1 survey units will be required. The emphasis will be placed on scanning the higher risk areas of Class 2 survey units. Scanning percentage of Class 3 survey units will be performed on potential areas of contamination based on the judgment of FSS technical personnel.

Table 5-4. Scan Measurement Coverage

Class	Coverage Percent
1	100%
2	10 – 100% *
3	Judgmental (<10%)

* For Class 2 Survey Units, the amount of scan coverage will be proportional to the potential for finding areas of elevated activity or areas close to the release criterion in accordance with NUREG-1575 (MARSSIM, Section 5.5.3). Accordingly, BRP personnel will use the results of individual measurements collected during characterization to correlate this activity potential to scan coverage levels.

5.3.3 Sample Size Determination

NUREG-1575 (MARSSIM) describes the process for determining the number of survey measurements necessary to ensure a data set sufficient for statistical analysis. Sample size is based on the relative shift, Type I and II errors, sigma, and the specific statistical test used to evaluate the data. Data point measurements or samples are used in the statistical analysis assume a random distribution. Alternate measurement processes or new technologies may be utilized provided they meet the applicable requirements of this plan for calibration, detection limit, area coverage, operator qualification, etc.

5.3.3.1 Statistical Tests

Appropriate tests will be used for the statistical evaluation of survey data as described in NUREG-1575 (MARSSIM). For BRP final status surveys, it has been determined that contaminants (i.e., Cs-137) present in background constitute only a small fraction of the DCGL; therefore, the Sign Test will be used for the majority of the survey unit data evaluations (see Section 2.3.3). Alternate statistical tests may be employed as appropriate or as specific situations are encountered.

5.3.3.2 Decision Errors

The probability of making decision errors is controlled by hypothesis testing. The survey results will be used to select between one condition of the environment (the null hypothesis) and an alternate condition (the alternative hypothesis). These hypotheses, chosen from NUREG-1575, are defined as follows:

- Null Hypothesis (H_0): The survey unit does not meet the release criteria
- Alternate Hypothesis (H_a): The survey unit does meet the release criteria

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criteria. This occurs when the null hypothesis is rejected when it is actually true. The probability of making a Type I error is designated as "α". A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criteria. This occurs when the Null Hypothesis is accepted when it is not true. The probability of making a

Type II error is designated as "β". Type I and Type II decision error probabilities are initially set at 0.05. These values may be modified to optimize survey designs following the guidance of NUREG-1575, Appendix D. Type I decision errors will only be increased in accordance with the requirements of Section I.

5.3.3.3 Relative Shift

The relative shift (δ / σ) is an expression of the resolution of the measurements in units of uncertainty. The relative shift is calculated as follows:

$$\delta/\sigma = (DCGL - LBGR)/\sigma$$

where:

δ – Shift or width of Gray Region equivalent to (DCGL-LBGR)
DCGL – Derived Concentration Guideline Level
LBGR – Lower Bound of the Gray Region
σ - Sigma, estimate of the standard deviation of the concentration of radioactivity in the survey unit

- *Lower Bound of the Gray Region*
The Lower Bound of the Gray Region (LBGR) is the point of acceptable Type II (β) error. The LBGR is initially set at 0.5 times the $DCGL_W$; however, this value may be adjusted to optimize the relative shift for the determination of sample size as described in NUREG-1575, Appendix D. Generally, Table 5-5 of NUREG-1575 will be used to determine the number of sample data points.

- *Sigma*
The sigma value is the estimate of the standard deviation of the concentration of radioactivity in the survey unit. Sigma values for survey design are developed from survey data that have utilized identical measurement techniques as those to be performed in the FSS. Sigma values may also be determined by estimation based on site-specific knowledge.

5.3.3.4 Sign Test Sample Size

The number of data points is determined from NUREG-1575 (MARSSIM), Table 5.5, for application of the Sign Test. This table includes the recommended 20% adjustment to ensure an adequate sample size and will be used to determine the appropriate sample size using the applicable parameters for Type I (α) and Type II (β) decision errors and relative shift (δ / σ) discussed previously.

5.3.3.5 Elevated Measurement Comparison (EMC) Sample Size Adjustment

If the actual scan MDC is greater than the required scan MDC (see Section 5.3.6.3), the sample size for the area of elevated activity will be determined using the equation provided below.

$N_{EMC} = A / A_{EMC}$

where:

N_{EMC} - Elevated Measurement Comparison sample size
A - Survey unit area
A_{EMC} - Area corresponding to the area factor calculated using the scan MDC Concentration (see Section 5.3.6.3)

5.3.4 Background Reference Area

Since it has been determined Cs-137 is present at less than 5% of the DCGL, background reference area measurements are not necessary (see Section 2.3.3). In the event that it is determined that background measurements are needed, they will be collected as described in Chapter 12 of NUREG-1505. If this occurs, the appropriate statistical test will be utilized in accordance with NUREG-1575 (MARSSIM).

5.3.5 Sample Locations and Reference Grid

Sample location is a function of the number of measurements required, the survey unit classification, and the contaminant variability.

5.3.5.1 Sample Locations

Measurement locations within the survey unit are clearly identified and documented for purposes of reproducibility. Actual measurement locations are identified by tags, labels, flags, stakes, paint marks, geopositioning units or photographic records. An identification code matches a survey location to a particular survey unit. Sample points for Class 1 and Class 2 survey units are positioned in a systematic pattern or grid throughout the survey unit by first randomly selecting a start point coordinate. A random number generator is used to determine the start point of the square grid pattern. The grid spacing, L, is a function of the area of the survey unit as shown below:

for a square grid *A* $$L = \sqrt{\frac{A}{n}}$$

where:

L = grid spacing length
A = the area of the survey unit,
n = the number of sample points in the survey unit (determined from NUREG-1575 Table 5.5)

Sample points are located, L, distance from the random start point in both the X and Y directions. Random sample point locations are used for Class 3 survey units. Sample location coordinates are specified using a random number generator. Measurement locations selected using a random selection process or a randomly-started systematic pattern that do not fall within the survey unit or that cannot be surveyed due to site conditions are replaced with additional random point locations as appropriate.

5.3.5.2 Reference Grid

The sample reference grid is illustrated on sample location maps. Grid reference points may also be physically marked in the field. An example reference grid and sample location map is shown in Figure 5-4. Global Positioning System (GPS) instruments may be used in open land areas to determine reference or sample grid locations within the survey area.

5.3.6 Investigation Levels and Elevated Areas Test

During survey unit measurements, levels of radioactivity may be identified that warrant investigation. Depending on the results of the investigation, the survey unit *may require no action, remediation, and/or reclassification and resurvey.* Investigation process and investigation levels are described below.

5.3.6.1 Investigation Process

During the survey process, locations with residual activity exceeding investigation levels are marked for further investigation. The elevated survey measurement is verified by resurvey. For Class 1 areas, size and average activity level in the elevated area is acceptable if it complies with the area factors and other criteria that may apply to evaluation of the DCGL for elevated measurements $DCGL_{EMC}$. As discussed in Section 5.3.6.3 below, the $DCGL_{EMC}$ is applicable only for Class 1 areas. If any location in a Class 2 area exceeds the $DCGL_{W}$, scanning coverage in the vicinity is increased in order to determine the extent and level of the elevated reading(s) and the area evaluated for reclassification. If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area evaluated for reclassification and resurvey under the criteria of the new classification. All survey unit investigations will be conducted in accordance with the applicable FSS data quality objectives (DQOs).

Investigations should address: (1) the assumptions made in the survey unit classification; (2) the most likely or known cause of the contamination; and (3) the effects of summing multiple land areas with elevated activity within the survey unit. Depending on the results of the investigation, a portion of the survey unit may be reclassified or combined with an adjacent area with similar characteristics if there is sufficient justification. Either action would result in resurvey of the (new) area(s). The results of the investigation process are documented in the Survey Package.

5.3.6.2 Investigation Levels

NUREG-1575 (Table 5.8) and NUREG-1727 (Table E.2) provide investigation levels for scan surveys. In addition to investigation levels for scan surveys, direct measurement survey investigation levels have also been developed. These additional investigation levels include a very conservative value for Class 3 survey units as shown in Table 5-5.

Table 5-5. Investigation Levels

Classification	Scan Measurement	Soil Sample Analyses
Class 1	> $DCGL_{EMC}$	> $DCGL_{EMC}$ or > $DCGL_w$ + (statistical based parameter value)
Class 2	> $DCGL_w$ or >MDC_{scan}*	> $DCGL_w$
Class 3	> $DCGL_w$ or >MDC_{scan}*	> 0.5$DCGL_w$

* The larger of MDC_{scan} or $DCGL_w$ will be used.

5.3.6.3 Elevated Measurement Comparison (EMC)

The elevated measurement comparison is used for Class 1 survey units when one or more scan or static measurement exceeds the investigation level. The EMC provides assurance unusually large measurements receive the proper attention and that any area having the potential for significant dose contribution is identified. As stated in NUREG-1575, the EMC is intended to flag potential failures in the remediation process and should not be considered the primary means to identify whether or not a survey unit meets the release criterion. Locations identified by scan methodology with levels of residual radioactivity which exceed the $DCGL_{EMC}$ or soil sample analyses measurements with levels of residual radioactivity which exceed the $DCGL_{EMC}$ or $DCGL_w$ + (mean + 3σ) (of survey unit) are subject to additional surveys to determine compliance with the elevated measurement criteria. The size of the area containing the elevated residual radioactivity and the average level of residual activity within the area are determined. The average level of activity is compared to the $DCGL_W$ based on the actual area of elevated activity. The initial $DCGL_{EMC}$ is established during the survey design and is calculated as follows:

$$DCGL_{EMC} = \text{Area Factor} \times DCGL_w$$

The area factor is the multiple of the $DCGL_w$ that is permitted in the area of elevated residual radioactivity without remediation. The area factor is related to the size of the area over which the elevated activity is distributed. That area is generally bordered by levels of residual radioactivity below the $DCGL_w$ and is determined by the investigation process. Area factors calculations are described in Section 6.10 and summarized in Table 5-6 [Reference 5-5]. Alternatively, Figures 6-2 and 6-3 provide a graphical method for selecting applicable area factors. The actual area of elevated activity is determined by investigation surveys and the area factor is adjusted for the actual area of elevated activity. The product of the adjusted area factor and the $DCGL_w$ determines the $DCGL_{EMC}$. If the $DCGL_{EMC}$ is exceeded, the area is remediated and resurveyed. The results of the elevated area investigations in a given survey unit that are below the $DCGL_{EMC}$ limit are evaluated using the equation below. If more than one elevated area is identified in a given survey unit, the unity rule can be used to determine compliance. If the formula value is less than unity, no further elevated area testing is required and the EMC test is satisfied.

Table 5-6. Area Factors for Open Land Areas*

Contaminated Area (m²)	Calculated Area Factors at Time of Peak Dose								
	H-3	Mn-54	Fe-55	Co-60	Sr-90	Cs-137	Eu-152	Eu-154	Eu-155
8094	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
4047	1.00	1.01	1.00	1.01	1.00	1.02	1.02	1.01	1.02
2024	1.00	1.03	1.00	1.03	1.00	1.03	1.03	1.03	1.03
1012	1.35	1.04	1.00	1.04	1.00	1.04	1.05	1.04	1.04
506	2.91	1.09	1.98	1.08	1.98	1.13	1.07	1.07	1.06
253	6.05	1.14	3.95	1.13	3.94	1.20	1.11	1.11	1.09
126	12.4	1.20	7.93	1.20	7.87	1.29	1.17	1.16	1.14
63	24.9	1.30	15.8	1.30	15.6	1.41	1.27	1.26	1.23
32	49.2	1.49	31.2	1.49	30.5	1.62	1.44	1.45	1.39
16	98.9	1.78	62.0	1.78	59.9	1.93	1.72	1.73	1.63
8	198	2.38	123	2.38	117	2.58	2.30	2.31	2.14
4	397	3.61	243	3.62	230	3.91	3.49	3.52	3.19
2	794	5.68	473	5.75	452	6.14	5.48	5.55	4.90
1	1590	9.57	905	9.73	887	10.3	9.24	9.39	7.88

* Table 5-6 is identical to Appendix 6-L of Chapter 6.

The following formula applies to a single radionuclide contaminant. When multiple radionuclides are present, the calculation is made for each radionuclide and the sum for all radionuclides must total less than one. Alternatively, if the mixture is known to be constant throughout the survey area, gross activity $DCGL_W$ for the mixture may be calculated in accordance with NUREG-1575, Equation 8-2.

$$\frac{\delta}{DCGL_W} + \frac{(Conc_{AVG} - \delta)}{(AreaFactor)(DCGL_W)} < 1$$

where:

$Conc_{AVG}$ - average concentration in elevated area
δ - Estimate of average concentration of residual radioactivity

If more than one elevated area in the survey unit, a separate term check will be included for each.

In practice, the ratio of Co-60 to Cs-137 in BRP soils is variable. Although this does not affect the ability to determine actual DCGL for the mixture from laboratory analyses or in-situ gamma spectrum analyses, it does affect ability to pre-define investigation levels and scan MDCs where ratios vary within a survey area. To access this issue, BRP will employ one of the following approaches:

1) Subdivide the survey unit into areas having low variability so that actual pre-determined nuclide ratios may be taken into account in the mixture DCGL and MDC calculations, or

2) Utilize conservative calculational techniques for variable ratio survey areas, which assume that the most limiting gamma-emitting radionuclide dominates the mixture (Co-60).

5.3.6.4 Remediation and Reclassification

As shown in Table 5-7, Class 1 or Class 2 areas of elevated residual activity above the $DCGL_{EMC}$ are remediated to reduce the residual radioactivity to acceptable levels. Based on survey data, it may be necessary to remediate an entire survey unit or only a portion of it. If an individual survey measurement (scan or direct) in a Class 2 survey unit exceeds the $DCGL_W$, the survey unit or a portion of it may be reclassified to a Class 1 survey unit and the survey redesigned and re-performed accordingly. If an individual survey measurement in a Class 3 survey unit exceeds 0.5 $DCGL_W$, the survey unit, or portion of a survey unit, will be evaluated, and if necessary, reclassified to a Class 2 and the survey redesigned and re-performed accordingly.

Table 5-7. Investigation Actions for Individual Survey Unit Measurements

Area Classification	Action If Investigation Results Exceed:		
	$DCGL_{EMC}$	$DCGL_W$	.5 $DCGL_W$
Class 1	Remediate or perform EMC evaluation	Acceptable*	N/A
Class 2	Reclassify and investigate**	Reclassify and investigate**	N/A
Class 3	Classify and investigate**	Reclassify and investigate**	Reclassify and resurvey, increase scan coverage

*For individual measurements above $DCGL_W$ plus the designated statistical-based parameter, the Sign Test will be conducted on the survey unit and an EMC evaluation performed.
**Requires an investigation of the initial classification process and a survey unit evaluation of sufficient intensity to satisfy the requirements of new classification status.

5.3.6.5 Resurvey

Following an investigation, if a survey unit is reclassified to a more restrictive classification or if remediation activities were performed, a resurvey is performed in accordance with approved procedures. If a Class 2 area had contamination greater than the $DCGL_W$ it should be reclassified to a Class 1 area. If the average value of Class 2 direct survey measurements was less than the $DCGL_W$, the scan MDC was sensitive enough to detect the $DCGL_{EMC}$ and there were no areas greater than the $DCGL_{EMC}$, the survey redesign may be limited to obtaining a 100% scan without having to re-perform the soil sample analyses. This condition assumes that the sample density meets the requirements for a Class 1 area.

5.4 SURVEY METHODS AND INSTRUMENTATION

5.4.1 Survey Measurement Methods

Survey measurements and sample collection are performed by personnel who have received training and are qualified to perform these activities. The techniques for performing survey measurements or collecting samples are specified in approved procedures. Final status survey measurements include surface scans and gamma spectroscopy of soil samples. In-situ gamma spectroscopy or other methods may be utilized provided that approved procedures are utilized and instrument sensitivity is sufficient to meet or exceed minimum required detection levels. Volumetric soils samples may be analyzed using on-site laboratory gamma spectroscopy, in-situ gamma spectroscopy, and sodium-iodide gamma detection for scanning in accordance with applicable procedures. Off-site laboratory facilities may also be used for QC as specified in applicable procedures. Analytical methods for both onsite and offsite laboratory facilities will be established to ensure minimum detection levels of 10% to 50% of the DCGL value. Other methods not specifically described may also be used for final status surveys provided that approved procedures for these methods are utilized.

Soil will receive scan surveys at the coverage level described in Table 5-4 and volumetric samples will be taken at designated locations. Surface soil samples will normally be taken at a depth of 0 to15 cm. Areas of sub-surface soil contamination will require sampling at a depth greater than 15 cm. The potential for sub-surface contamination will be addressed during the survey design process, and the associated survey package will contain requirements for sampling or in-situ measurements of soil below 15 cm. All activities will be performed in accordance with approved procedures.

5.4.1.1 Scans

Open land areas are scanned for gamma emitting nuclides. The gamma emitters are used as surrogates for the Hard-to-Detect (HTD) radionuclides. Sodium iodide detectors will be used for scanning open land areas at the BRP site.

5.4.1.2 Volumetric Samples

Laboratory gamma spectroscopy is used to analyze collected soil samples. Soil samples size is approximately 1600 grams. Surface samples are collected from the top 15 cm of soil and subsurface soil samples are collected at depths below 15 cm. Sample preparation includes removing extraneous material, homogenizing, and drying the soil for gamma isotopic analysis. Separate containers are used for each sample and each container is tracked through the analysis process using a chain-of-custody record. Samples are split when required by the applicable quality control procedures.

If contamination below 15 cm is suspected or known, samples will be collected using an auger or equivalent method. If a survey area has already been excavated and remediated to the soil DCGL, this area will be treated as a surface soil, and the FSS will be performed on the excavated area. Since all buildings and structures used during nuclear plant operations are scheduled for removal, it is expected that the majority of the survey units within the Industrial Area will include excavated areas. Soil samples will be collected to depths at which there is high confidence that deeper samples will not result in higher concentrations. Alternatively, a sodium-iodide detector or intrinsic germanium detector of sufficient sensitivity to detect DCGL concentrations may be utilized in a "down hole" configuration to identify the presence or absence of subsurface contamination, and the extent of such contamination. If the detector identifies the presence of contamination at a significant fraction of the DCGL, confirmatory laboratory analyses of soil samples of the suspect areas will be performed. Areas where subsurface samples may be collected include the Turbine Building, Containment Building, liquid and solid radioactive waste vaults, and effluent stack foundation areas. All subsurface sampling will be performed in accordance with the guidance in Section 11.1 of NUREG-1727. The sample size for subsurface samples will be determined using the same methods described for surface soil. Per NUREG-1727, scanning is not applicable to subsurface areas; however, BRP FSSs will employ scanning techniques commensurate with the survey unit classification. Scanning on subsurface soils, where accessible as an excavated surface, will demonstrate compliance with site release criteria.

5.4.2 Specific Survey Area Considerations

5.4.2.1 Septic Field

The only onsite subsurface equipment to remain in place after site release is the septic system drainfield located just west of the former plant protected area. The drain field is a ceramic tile system, approximately 1200 m^2. All piping leading to this drainfield and associated tanks will be removed. The drainfield is considered a Class 2 area and will be surveyed as a single survey unit. The FSS for this survey unit will include both an evaluation of surface and subsurface soils. Mechanical coring equipment may be utilized to provide access for down-hole measurements and to obtain necessary subsurface samples.

5.4.2.2 Pavement-Covered Areas

Survey of paved areas will be required along the roadways providing ingress and egress to the site. Evaluation has determined that paved roadways are Class 3 areas. All other pavement, including that in parking lots and asphalt within the protected area, will be removed for disposal prior to the FSS. The survey design of paved areas will be based on soil survey unit sizes since they are outdoor areas where the exposure scenario is most similar to direct radiation to surface soil. Scan and static gamma and beta-gamma surveys are made as determined by the survey unit design. If the potential exists for sub-surface contamination under pavement, either the pavement/asphalt will be removed prior to the FSS or samples obtained through the pavement. Pavement will remain only in Class 3 areas; scanning of paved areas will generally not be conducted, as Class 3 areas only require judgmental scanning of a small portion of the survey unit (see Section 5.3.2). Paved areas may be separate survey units or they may be incorporated into surveys of other adjacent open land areas of like classification. Surveys of paved areas may include road right-of-ways to check for radioactivity relocated from water runoff. Right-of-ways may also be separate survey units.

5.4.2.3 Discharge Canal Sediment

The discharge canal area will be evaluated as a separate survey unit. A coffer dam or equivalent method will be used to control lake water during survey activities in this area. The FSS will be performed prior to returning the shoreline to its original contour. The FSS of the discharge canal may be performed in conjunction with surveys of adjacent beach areas.

5.4.2.4 Stored Excavated Soil

It is expected that soil will be stored around the site resulting from various decommissioning activities. Soil volumes resulting from excavations to remove building foundations or buried piping may be relocated to support work activities. The locations from which stored soil originated will be tracked and storage areas controlled so that no contamination of soils will occur as a result of decommissioning activities. It is anticipated that much of this stored soil can be used for backfilling soil excavation areas after the FSS is completed. The following paragraphs provide the survey methodology for verifying that this soil is acceptable for backfill purposes.

For small excavations, a combination of laboratory analysis and scans will be performed to characterize the excavated soil. A sampling will be conducted to ensure that no activity above the DCGL action level (see Table 5-7) exists. If sample results are acceptable this soil may be returned to its original location or used elsewhere on site for fill. After all work activities are complete, the storage and fill areas would receive a FSS in accordance with approved procedures. This approach is expected to be utilized for shallow trenches facilitating removal of small diameter piping or conduit, removal of buried equipment, or demolition of isolated footings.

For larger excavation areas, it is expected stored soils will be evaluated using one of two methods. The soil may be placed into containers suitable for volumetric or in-situ analysis using germanium detectors. Alternatively, stored soils could be spread out to maximum depth of one meter, which would then be suitable for a MARSSIM survey commensurate with the area classification for the original soil location and in accordance with site procedures. Other methods may be employed for evaluation of stored soils provided that the proposed method receives a technical evaluation to ensure DCGLs are met prior to using the soil as fill material. Soil removed from various Class 3 areas may be combined prior to sampling and evaluation. In other than Class 3 areas, controls will be instituted to prevent mixing of excavated soils from different survey areas prior to evaluation.

5.4.2.5 Groundwater Surveys

Groundwater sampling and monitoring will be performed during excavation of building foundations and subsurface structures and during FSS of corresponding survey areas, as necessary. Groundwater sampling will consist of gamma spectrum analysis and tritium analysis since this is the only radionuclide identified in site groundwater. Additionally, groundwater and surface water flow control measures will be in place during demolition activities to minimize or eliminate the impact of water movement. Dewatering activities may require placement of temporary barriers to inhibit groundwater flow; groundwater flow is not expected to be influenced beyond the demolition interval.

5.4.3 Instrumentation

Radiation detection and measurement instrumentation for the FSS is selected to provide both reliable operation and adequate sensitivity to detect the radionuclides identified at the site at levels sufficiently below the DCGL. Site history and characterization efforts have identified Cs-137 and Co-60 as the predominant radionuclides present in BRP site soils. Other radionuclides of plant origin, including hard-to-detect nuclides are present at levels much lower that those of Cs-137 and Co-60. Table 5-1 provides a list of potential radionuclides for evaluation in BRP site soils. Soil sampling and analysis have demonstrated that direct measurements of Cs-137 and Co-60 can be used as surrogates for estimating levels of other contaminants that may be present in BRP soils.

Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field. Portable instruments, laboratory instruments and bulk assay equipment may be used to perform these basic survey measurements:

- surface scanning,
- laboratory gamma spectroscopy of soil samples, and
- gamma spectroscopy using the bulk assay monitor equipment
- direct surface contamination measurements (static or in-situ),

Radiation protection instrumentation procedures control the issuance, use, and calibration of instrumentation. Records supporting the instrumentation program are maintained in accordance with site document control procedures.

5.4.3.1 Selection

Radiation detection and measurement instrumentation is selected to meet the requirements of survey design. Gamma spectroscopy instruments used for soil sample analyses are capable of residual radioactivity detection at values less than 10% of the $DCGL_W$. Instruments used for surface scanning are capable of detecting radioactive material at levels below the $DCGL_{EMC}$ in Class 1 areas [Reference 5-3]. MDC values for scanning instruments used in Class 1 and 2 areas have the capability of detecting residual radioactivity below the $DCGL_W$. Instrumentation currently proposed for use in the FSS is listed in Table 5-8 found in Section 5.4.3.5. Instrument MDCs are discussed in Section 5.4.3.4 and nominal MDC values are also listed in Table 5-8.

Other measurement instruments or technologies, such as in-situ gamma spectroscopy or continuous data collection scan devices may be utilized if evaluation determines that alternate instrumentation is equally or more efficient than the survey instruments currently proposed in this plan. The acceptability of alternate instruments or technologies for use in the FSS Program would be justified in a technical basis evaluation document to ensure equivalent or better instrument sensitivity. An instrument technical analysis document would include among other things the following:

a. Description of the conditions under which the method would be used;
b. Description of the measurement method, instrumentation and criteria;
c. Justification that the technique would provide the required sensitivity for the given survey unit classification in accordance with Table 5-5; and
d. Demonstration that the instrument provides sufficient sensitivity for measurement below the release criteria with Type 1 error equivalent to 5% or less.

5.4.3.2 Calibration And Maintenance

Instruments and detectors are calibrated for the radiation types and energies of interest at the site. Gamma scintillation detectors are calibrated using Cs-137, but the energy response and MDC for Co-60 has also been determined since discrete areas of Co-60 contamination have been found by soil surface scans. Instrumentation for detecting alpha contamination is not expected to be required for BRP FSSs based on HSA information and site characterization data (see Sections 2.2.4 and 2.3.2).

Instrumentation used for the FSS will be calibrated and maintained in accordance with the BRP Radiation Protection & Environmental Services Department procedures. Radioactive sources used for calibration are traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. If vendor services are used, these will be obtained in accordance with purchasing requirements for quality related services, to ensure the same level of quality.

5.4.3.3 Response Checks

Instrumentation response checks are conducted to assure proper instrument response and operation. An acceptable response for field and laboratory instrumentation is an instrument reading within +/- 3 sigma as documented on a control chart. As a minimum, response checks are performed daily prior to instrument use. Source checks use source energies consistent with the nuclides encountered at the BRP site. If an instrument fails a response check, it is appropriately identified and withheld from use until the problem is corrected in accordance with applicable procedures. Measurements made from the time of the last acceptable check and the failed checks are evaluated to determine if they should remain in the data set.

5.4.3.4 Minimum Detectable Concentration (MDC)

An MDC is determined for each of the instruments used for final status surveys. The MDC is the concentration of radioactivity that an instrument can be expected to detect 95 percent of the time.

- *Laboratory Spectrometer Analysis*
  The onsite chemistry laboratory maintains two gamma isotopic spectrometers that are calibrated to various sample geometries, including a one-liter marinelli geometry for soil analysis. These systems are calibrated using a NIST mixed gamma source. Both detectors are manufactured by PGT and operate using the VMS Genie platform from Canberra Industries. Laboratory counting systems have software controlled count times which are set to meet a maximum MDC of 0.15 pCi/g for Cs-137 in soil; this is calculated as follows:

$$MDC = {L_d}/{K * V * T}$$

where:

L_d – limit of detection = $3 + 4.65 * \sqrt{bkg}$
K - proportionality constant relating detector response to activity level
V - mass of sample
T - count time

- *Land Area Scans*
  Evaluation of open land areas for unrestricted release must include a detection methodology of sufficient sensitivity to allow identification of small areas of potentially elevated activity within a survey unit. Surface scanning measurements for BRP land areas are performed by continuously passing a 2-inch x 2-inch sodium iodide (NaI) gamma scintillation detector over the subject land surface at a specified rate of speed. The scan technique uses an audible response to monitor for increase in count rate (see Chapter 2, Appendix D). An audible increase in instrument count rate requires further investigation to verify findings and define the level and extent of contamination.

  Alternatively, in-situ gamma spectroscopy of variable size areas may be utilized for areas with special considerations. In-situ measurements will cover up to one survey grid (10m x 10m) of the survey area and may also provide measurements of the same small areas (one square meter) of potentially elevated activity. The value of in-situ spectroscopy is its ability to distinguish between residual radioactive material contamination and natural radioactive materials in stone or rock. Such natural materials have been found in onsite characterization surveys and are difficult to analyze by laboratory means when large stones or rocks are responsible for the elevated activity.

- *Land Area Scan Instrument Sensitivity*
  A determination of scanning sensitivity is performed to ensure that the measurement system is able to detect concentrations of radioactivity at levels below the release limit. Expressed in terms of the scan MDC, this sensitivity is the lowest concentration of radioactivity for a given background that the measurements system is able to detect at a specified performance level. Survey planning for soil survey units will be based on a scan MDC associated with a one-meter wide strip for scanning. Rate of scan transverse will be such that the resulting scan MDC is sufficient to detect areas of elevated activity with a measurement accuracy confidence limit of 95%. This two-stage scanning technique is described in NUREG-1507 and has been evaluated for use on BRP open land areas (see Chapter 2, Appendix 2-D).

- *In-Situ Measurement Sensitivity*
  A Canberra ISOCS in-situ system, utilizing a 40% intrinsic germanium detector, has been used in site characterization [Reference 5-2]. Sensitivity for Cs-137 and Co-60 of less than 10% of initial DCGLs has been demonstrated. In-situ gamma spectroscopy for FSS will be conducted in association with the laboratory analysis of soil grab samples and/or other approved measurement techniques. The resulting analyses are then compared for the identification of potential non-homogeneous radioactivity concentrations within the survey unit. Survey units failing acceptance criteria will require further investigation.

5.4.3.5 Detection Sensitivity

The nominal detection sensitivity of detectors that may be used for final status surveys has been determined. Count times are instrument-specific and are selected to ensure that the measurements are sufficiently sensitive for the DCGL. For example, the count times associated with gamma spectroscopy of volumetric materials are administratively established to achieve MDCs less than the DCGL. The MDC scan values may not always be less than the $DCGL_W$, but will be less than $DCGL_{EMC}$ (see Section 5.3.6.3). Table 5-8 provides a summary of FSS instruments for BRP.

Table 5-8. Typical FSS Instrumentation Characteristics

Instrument and Detector	Measurement Type	Instrument Efficiency	MDA/MDC
2" x 2" NaI	Gamma*	1200 cpm/mR/hr (Cs-137)	Class I < $DCGL_{EMC}$** Class 2&3 < $DCGL_W$
Canberra Genie	Laboratory Gamma	9.7% & 44.1% (2 detectors)	< 10% of < $DCGL_W$
Canberra ISOCS	In-Situ Gamma	40%	< $DCGL_W$

* Scan for gamma emitting nuclides using a rate meter.
** MDC values for varying background values are provided in Chapter 2, Appendix 2-D.

5.5 DATA COLLECTION AND PROCESSING

This section describes data collection, review, validation and record keeping requirements for final status surveys.

5.5.1 Sample Handling and Record Keeping

A chain-of-custody record accompanies each sample from the point of collection through obtaining the final results to ensure the validity of the sample data. Sample tracking records are controlled and maintained in accordance with applicable procedures. Each survey unit has a document package associated with it, that covers the design and field implementation of the survey requirements. Survey unit records are considered quality records.

5.5.2 Data Management

Survey data are collected from several sources during the data life cycle and are evaluated for validity throughout the survey process. QC replicate measurements are not used as final status survey data. (See Section 5.8.4.2 for design and use of QC measurements.) Measurements performed during turnover and investigation surveys can be used as final status survey data if they were performed according to the same requirements as the final status survey data. These requirements are:

a. Survey data shall reflect the as-left survey unit condition, i.e., no further remediation required;

b. The application of isolation measures to the survey unit to prevent re-contamination and to maintain final configuration are in effect; and

c. The data collection and design were in accordance with FSS methods and procedures, e.g., scan MDC, investigation levels, survey data point number and location, statistical tests, and EMC tests.

Measurement results stored as final status survey data constitute the final survey of record and are included in the data set for each survey unit used for determining compliance with the site release criteria. Measurements are recorded in units appropriate for comparison to the applicable DCGL. Numerical values, even negative numbers, are recorded. Measurement records include, at a minimum, the surveyor's name, the location of the measurement, the instrument used, measurement results, the date and time of the measurement, any surveyor comments, and records of applicable reviews.

5.5.3 Data Verification and Validation

The final status survey data are reviewed before data assessment to ensure that they are complete, fully documented and technically acceptable. The review criteria for data acceptability will include at a minimum, the following items:

a. The instrumentation MDC for fixed or volumetric measurements was below the $DCGL_{EMC}$ for Class 1, below the $DCGL_{w}$ for Class 2 and below 0.5 $DCGL_{w}$ for Class 3 survey units;

b. The instrument calibration was current and traceable to NIST standards;

c. The field instruments were source checked with satisfactory results each day data was collected or data was evaluated if instruments did not pass a response check in accordance with Section 5.4.3.3;

d. The MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey;

e. The survey methods used to collect data were proper for the types of radiation involved and for the media being surveyed;

f. "Special methods" for data collection were properly applied for the survey unit under review;

g. The chain-of-custody was tracked from the point of sample collection to the point of obtaining results;

h. The data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflect the radiological status of the facility; and

i. The data have been properly recorded.

If the data review criteria are not met, the discrepancy will be evaluated and the decision to accept or reject the data will be documented in accordance with approved procedures. The BRP Corrective Action Program will be used to document and resolve discrepancies as applicable.

5.5.4 Graphical Data Review

Survey data may be graphed to identify patterns, relationships or possible anomalies that might not be evident using other methods of review. A posting plot or a frequency plot may be made. Other special graphical representations of the data will be made as the need dictates.

5.5.4.1 Posting Plots

Posting plots may be used to identify spatial patterns in the data. The posting plot consists of the survey unit map with the numerical data shown at the location from which it was obtained. Posting plots can reveal patches of elevated radioactivity or local areas in which the DCGL is exceeded. Incongruities in the background data may be the result of residual, undetected activity.

5.5.4.2 Frequency Plots

Frequency plots may be used to examine the general shape of the data distribution. Frequency plots are basically bar charts showing data points within a given range of values. Frequency plots reveal such things as skewness and bimodality (having two peaks). Skewness may be the result of a few areas of elevated activity. Multiple peaks in the data may indicate the presence of isolated areas of residual radioactivity due to variation of soil types. Variability may also indicate the need to subdivide the survey unit by soil type or by different nuclide distributions.

5.6 DATA ASSESSMENT AND COMPLIANCE

An assessment is performed on final status survey data to ensure that they are adequate to support the determination to release the survey unit. Simple assessment methods such as comparing the survey data to the DCGL or comparing the mean value to the DCGL are first performed. The statistical tests are then applied to the final data set, where required, and conclusions are made as to whether the survey unit meets the site release criterion.

5.6.1 Data Assessment Including Statistical Analysis

The results of the survey measurements are evaluated to determine whether the survey unit meets the release criterion. In some cases, the determination can be made without performing complex, statistical analyses.

5.6.1.1 Interpretation of Sample Measurement Results

An assessment of the measurement results is used to quickly determine whether the survey unit passes or fails the release criterion. Final status surveys for BRP land will utilize the Sign Test on the basis that contaminants originating from plant operation that may reside in soil (or groundwater) do not exist in general background in appreciable quantities. Evaluation criteria for the Sign Test is provided in Table 5-9.

Table 5-9. Interpretation of Sample Measurements for Sign Test

Measurement Results	Conclusion
All concentrations less than $DCGL_w$	Survey unit meets release criterion
Average concentration greater than $DCGL_w$	Survey unit fails
Any concentration greater than $DCGL_w$ and average concentration less than $DCGL_w$	Conduct Sign Test and elevated measurements test.

In addition, survey data are evaluated against the elevated measurement comparison criteria as previously described in Section 5.3.6.3. The statistical test is based on the null hypothesis (H_0) that the residual radioactivity in the survey unit exceeds the $DCGL_w$. There must be sufficient survey data at or below the $DCGL_w$ to reject the null hypothesis and conclude the survey unit meets the site release criterion for dose. Statistical analyses are performed using specially designed computer-based calculations or, if necessary, using hand calculations.

5.6.1.2 Sign Test

The Sign Test and Sign Test Unity Rule are one-sample statistical tests used for situations in which the radionuclide of concern is not present in background, or is present at acceptable low fractions compared to the $DCGL_W$. If contaminant is present in background, the gross measurement is assumed to be entirely from plant activities. This option is used when it can be reasonably expected that including the background concentration will not affect the outcome of the Sign Test. The advantage of using the Sign Test is a background reference area is not needed.[2]

The Sign Test is conducted as follows:

a. The survey unit measurements, X_i, i = 1, 2, 3, ...n; where n = the number of measurements, are listed.

b. X_i is subtracted from the $DCGL_W$ to obtain the difference ($DCGL_W$ - X_i, i = 1, 2, 3,...,n).

c. Differences where the value is exactly zero are discarded and n is reduced by the number of such zero measurements.

d. The number of positive differences are counted. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the site release criterion.

e. The value of S+ is compared to the critical value given in Table I.3 of NUREG-1575. The table contains critical values for given values of N and Alpha (α). The value of α is set at 0.05 during survey design. If S+ is greater than the critical value given in the table, the survey unit meets the site release criterion. If S+ is less than or equal to the critical value, the survey unit fails to meet the release criterion.

5.6.2 Data Conclusions

The results of the statistical tests, including application of the EMC, allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release criterion. The data provide statistical evidence that the level of residual radioactivity in the survey unit does not exceed the release criterion. The decision to release the survey unit is made with sufficient confidence and without further analysis. The second conclusion that can be made is that the survey unit fails to meet the release criterion. The data are not conclusive in showing that the residual radioactivity is less than the release criterion. The data are analyzed further to determine the reason for the failure. Possible reasons are that:

a. The average residual radioactivity exceeds the $DCGL_W$, or

b. The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

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[2] The Sign Test may be used with background reference areas in accordance with Chapter 12 of NUREG-1505.

The power of the statistical test is a function of the number of measurements made and the standard deviation in measurement data. The power is determined from 1-β where β is the value for Type II errors. A retrospective power analysis may be performed using the methods described in Appendices I.9 and I.10 of NUREG-1575. If the power of the test is insufficient due to the number of measurements, additional samples may be collected as directed by procedure. A greater number of measurements increases the probability of passing if the survey unit actually meets the release criterion. If failure was due to the presence of residual radioactivity in excess of the release criterion, the survey unit must be remediated and resurveyed.

5.6.3 Compliance

The FSS is designed to demonstrate licensed radioactive materials have been removed from BRP property to the extent remaining residual radioactivity is below the radiological criteria for unrestricted release. The site-specific radiological criteria presented in this plan demonstrate compliance with the criteria of 10 CFR 20.1402. If the measurement results pass the requirements of Table 5-5 and the elevated areas evaluated per Section 5.3.6.3 pass the elevated measurement comparison, the survey unit is suitable for unrestricted release. If survey measurements do not meet the criteria specified in Table 5-5, an investigation will be performed. Investigations will include an evaluation of survey design, instrumentation use and calculations, as necessary. All investigations of this nature will be documented using the corrective action process as discussed in Section 5.8.2 and 5.8.5.

5.7 REPORTING FORMAT

Survey results are documented in history files, survey unit release records, and in the FSS Report. Other reports may be generated as requested by the NRC.

5.7.1 History File

A history file of relevant operational and decommissioning data is compiled. The purpose of the history file is to provide a substantive basis for the survey unit classification, and hence, the level of intensity of the FSS. The history file contains the following items:

- Operating history which could affect radiological status,
- Summarized scoping and site characterization data, and
- Other relevant information, as deemed necessary.

5.7.2 Survey Unit Release Record

A separate release record is prepared for each final status survey unit. The survey unit release record is a stand-alone document containing the information necessary to demonstrate compliance with the site release criteria. This record includes:

- Description of the survey unit,
- Survey unit design information,
- Survey unit measurement locations and corresponding data,

- Survey unit investigations performed and their results, and
- Survey unit data assessment results.

When a survey unit release record is given final approval, it becomes a quality record.

5.7.3 Final Status Survey Report

Survey results will be described in a written report to the NRC. The FSS Report provides a summary of the survey results and the overall conclusions to demonstrate the BRP site meets the radiological criteria for unrestricted use. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results will be included in the report. The level of detail is sufficient to clearly describe the FSS Program and to certify the results. The FSS Report will contain the following topics:

- Overview of the Results
- Discussion of Changes to FSS
- Final Status Survey Methodology
  - Survey unit sample size
  - Justification for sample size
- Final Status Survey Results
  - Number of measurements taken
  - Survey maps
  - Sample concentrations
  - Statistical evaluations, including power curves
  - Judgmental and miscellaneous data sets
- Anomalous Data
- Conclusion for Survey Units
- Summary of Changes from Initial Assumptions on Residual Radioactivity.

5.7.4 Other Reports

Other reports relating to final status survey activities may be prepared and submitted as necessary.

5.8 FINAL STATUS SURVEY QUALITY PROGRAM

Quality is built in to each phase of the FSS Program and measures must be taken during the execution of the plan to determine whether the expected level of quality is being achieved. The FSS Program will ensure that the site will be surveyed, evaluated and determined to be acceptable for unrestricted release if the residual activity results in an annual Total Effective Dose Equivalent (TEDE) to the average member of the critical group of 25 mrem/year or less for all pathways. The following sections provide a description of applicable BRP quality programs and specific quality elements of the FSS Program.

5.8.1 Big Rock Point Quality Assurance Program

The BRP quality assurance program is applied to systems, structures, components and activities important to the safe storage, control and maintenance of spent nuclear fuel and to the monitoring and control of radiological hazards. The Consumers Energy Quality Program Description (QPD) for Nuclear Power Plants defines the responsibilities and requirements to ensure decommissioning and construction/operation of the ISFSI comply with licenses and applicable regulations (10 CFR 50 and 10 CFR 72). This QPD addresses organizational responsibilities, staff qualifications, procedure review and approval, design and modification controls, procurement, measurement and test equipment (M&TE) calibration and control, testing of installed equipment, document control, and other information pertinent to quality [Reference 5-9].

5.8.2 FSS Quality Assurance Project Plan (QAPP)

The objective of the FSS QAPP is to ensure the survey data collected are of the type and quality needed to demonstrate with sufficient confidence the site is suitable for unrestricted release. The objective is met through use of the DQO process for FSS design, analysis and evaluation. The plan ensures the following items are accomplished:

a. The elements of the FSS Plan are implemented in accordance with the approved procedures;
b. Surveys are conducted by trained personnel using calibrated instrumentation;
c. The quality of the data collected is adequate; and
d. Corrective actions, when identified, are implemented in a timely manner and are determined to be effective.

The following sections describe the basic elements of the FSS QAPP.

5.8.2.1 Project Management and Organization

An FSS organization will be established for the BRP site. This organization will be responsible for planning and implementation of final status surveys. Since the FSS organization has not been fully implemented at the time of LTP development, specific job titles may vary over the period of project execution. However, the following descriptions refer to various functional areas of responsibility and do not necessarily correspond to specific job titles. It is also important to note qualified individuals may assume the responsibilities of more than one of the functional positions described below. The FSS organization consists of the following functional areas:

- *Final Status Survey Supervisor*
  The FSS Supervisor has overall responsibility for program direction, technical content, and ensuring the program complies with applicable NRC regulations and guidance. This supervisor is responsible for preparation and implementation of the FSS procedures. Additional responsibility areas may include resolution of issues or concerns raised by the NRC, the Michigan Department of Environmental Quality (MDEQ) or other Stakeholders, as well as programmatic issues raised by BRP site

management. The FSS Supervisor provides overall FSS project coordination, which may include, but is not limited to, interfaces with site personnel in areas of nuclear licensing, demolition and waste disposal.

- *Final Status Survey Technical Specialists*
  Responsibilities of FSS Technical Specialists may include technical support and development of FSS procedures, design of final status surveys, preparation of survey execution instructions, development of specific technical analysis documents supporting FSS activities, and review of survey packages and data collected in support of the FSS.

- *Work Planning Coordinators*
  Work Planning Coordinators develop detailed, job-specific work instructions using the site work order process. These individuals are tasked with ensuring the appropriate interface between various site functional groups is specified in work order documents. These individuals possess specific knowledge regarding Radiation Protection, FSS, and Industrial Safety requirements.

- *Final Status Survey Field Coordinators*
  Final Status Survey Field Coordinators are responsible for control and implementation of survey packages during field activities. Specific responsibilities are likely to include:

  - Coordination of turnover surveys,
  - Survey area preparation (e.g., gridding),
  - Ensuring final status survey sampling is conducted in accordance with applicable procedures and work instructions,
  - Maintaining access controls over completed FSS survey areas,
  - Determining survey area accessibility requirements,
  - Coordination and scheduling of FSS Technicians to support the decommissioning schedule, and
  - Ensuring all necessary instrumentation and other equipment is available to support survey activities.

- *Final Status Survey Data Specialist*
  The FSS Data Specialist is responsible for maintaining the FSS data records in both electronic formats and hardcopy files, as applicable. This includes maintaining survey measurement data and supporting data files and generating reports of survey results. Responsibilities also include maintaining the integrity of the FSS database and implementing FSS Database QA requirements.

- *Final Status Survey Technician*
  Final Status Survey Technicians are responsible for performance of final status survey measurements and collection of final status survey samples in accordance with applicable site procedures and survey package instructions. An FSS Technician will be responsible for maintaining the pedigree of instrumentation used in the survey by implementing the procedural requirements for calibration, maintenance and daily checks. Final Status Survey Technicians will be trained and task-qualified for the performance of the final status activities assigned to them. Final Status Survey Technicians may also participate in survey area preparations.

5.8.2.2 Written Procedures

Sampling and survey tasks must be performed properly and consistently in order to assure the quality of final status survey results. The measurements will be performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for final status survey measurements. Table 5-10 provides a list of BRP site procedures applicable to final status surveys and their current status.

Table 5-10. BRP Procedures Applicable to FSSs

Procedure	Title	Status
D5.1	Radiation Protection and Environmental Services Policy and Program Description	Active
D5.3	Big Rock Point Radiological Environmental Program	Active
D5.24	ALARA Program	Active
D5.19	Radiation Detection Instrumentation Calibration Facility and Source Control	Active
D5.XX	FSS Organization and Responsibilities	Under development
RP-XX	FSS Design	Under development
RP-XY	FSS Sampling	Under development
RIP-59	Scan Measurements	Active
RIP-60	Calibration and Operation of the Canberra Genie 2000 (In-Situ Gamma Spectroscopy)	Active
RM-72	Sample Chain of Custody	Active
CIP-46	Operation of Canberra "Genie"	Active
CIP-50	Calibration, Functional Check and Use of Acculab V-4kg Balance	Active
RM-59	Sampling and Analysis of Bulk Material for Site Characterization or Free Release	Active
RP-29	Radiological Surveys	Active

Procedure	Title	Status
RM-70	Packaging Radioactive Waste for Burial	Active
D1.3	Corrective Action	Active
D1.2	Plant Documents	Active
D1.5	Personnel Safety	Active
D1.7	Master Training Plan	Active
D1.10	Computer Software Control	Active
D1.11	10 CFR 50.59 and 10 CFR 50.82 Evaluations	Active
D3.3	Work Management Process	Active
D3.1.2	Engineering Analysis and Sketches	Active
D4.2.4	Procurement General Requirements	Active
Volume 28, Section 1	Spill Prevention	Active
Volume 28, Section 2	Stormwater Plan	Active
Volume 25	BRP Offsite Dose Calculation Manual	Active
CPC-2A	Quality Program Description for Nuclear Power Plants (Part 1) – Big Rock Point	Active

5.8.2.3 Training and Qualification

Personnel performing final status survey measurements will be trained and qualified. Training will include the following topics:

- Procedures governing the conduct of the FSS,
- Operation of field and laboratory instrumentation used in the FSS, and
- Collection of final status survey measurements and samples.

Qualification is obtained upon satisfactory demonstration of proficiency in implementation of procedural requirements. The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity required to be performed by that individual. Records of training and qualification will be maintained in accordance with approved training procedures [References 5-1 and 5-7].

5.8.2.4 Measurement and Data Acquisitions

The FSS records have been designated as quality documents and will be governed by site quality programs and procedures. Generation, handling and storage of the original final status survey design and data packages will be controlled by site procedures. Each final status survey measurement will be identified by individual, date, instrument, location, type of measurement, and mode of operation.

a. Quality Control Surveys

The FSS Procedure has built-in quality control (QC) checks for the survey process, instrumentation, field and laboratory measurements. A minimum of 5% of final survey soil, water, and sediment samples will be evaluated through the QC program. Quality Control will consist of one or more of the following: in-house recounts, split samples, third party analysis, and/or statistical comparisons. Acceptance criterion will be based on NRC Inspection Procedure 84750 or a standard statistical test. Unacceptable QC comparisons will require a documented investigation and may result in reanalysis, resurvey, or resampling.

b. Instrumentation Selection, Calibration and Operation

Proper selection and use of instrumentation will ensure sensitivities are sufficient to detect radionuclides at the minimum detection capabilities as specified in Section 5.4.3 as well as assure the validity of the survey data. Instrument calibration will be performed with NIST traceable sources using approved procedures. Issuance, control and operation of the survey instruments will be conducted in accordance with the instrumentation procedures.

5.8.2.5 Assessment and Oversight

Assessments and project oversight will include the following:

a. Self-assessments will be conducted in accordance with approved procedures and programs. As applicable, actions will be tracked in accordance with these documents.

b. Independent review of randomly selected survey packages (approximately 5%) from selected survey units will be performed by onsite quality assurance programs to ensure the survey measurements have been taken and documented in accordance with approved procedures.

c. The BRP Corrective Action Program will be applied to the FSS Program in accordance with site procedures. Applicable procedures describe the methods used to initiate condition reports (CRs) and resolve associated corrective actions.

d. Assessment and oversight by independent organizations will be conducted on a periodic basis, as deemed appropriate by site management.

5.8.2.6 Data Validation and Verification

Survey data will be reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels will be made and measurements exceeding the investigation levels will be evaluated. Procedurally verified data will be subjected to the Sign Test as discussed in Section 5.8.1.2. Technical evaluations or calculations used to support the development of DCGLs will be independently verified to ensure correctness of the method and the quality of data.

a. Confirmatory Measurements

Consumers Energy Co. anticipates that both the NRC and the MDEQ – Radioactive Material & Standards Unit may choose to conduct confirmatory measurements. The NRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the FSS and associated documentation demonstrate the site is suitable for release in accordance with the criteria for decommissioning in 10 CFR Part 20, subpart E. Confirmatory measurements may include collecting radiological measurements for the purpose of verifying compliance with applicable state laws and confirming and verifying compliance with NRC standards for unrestricted license termination. Timely and frequent communications with these agencies will ensure they are afforded sufficient opportunity for these confirmatory measurements prior to implementing any irreversible decommissioning actions.

b. Project Schedule

Portions of the FSS may be performed following demolition activities as areas become available for survey. Table 5-11 provides an estimate of FSS schedule based on dismantlement and demolition of specific structures and buildings. Table 5-11 is presented at a target schedule summary only; specific timeframes may be adjusted by site management or due to weather conditions, as necessary, during the demolition phase of decommissioning.

Table 5-11. Target Schedule for FSS of Various Areas

Survey Unit Description	FSS Target Start Date	Estimated FSS Duration
Radwaste Building Vault Area	March 2003	1 month
Liquid Radwaste Vault Area	September 2004	1 month
Screenhouse/Discharge Canal Area	November 2004	1 months
Administrative/Service Building Area	December 2004	2 months
Turbine Building Area	January 2005	3 months
Stack Area	March 2005	1 month
Containment Building Area	September 2005	2 months
Yard Area – remaining foundations	October 2005	2 months
All Remaining soil surfaces	January 2006	3 months

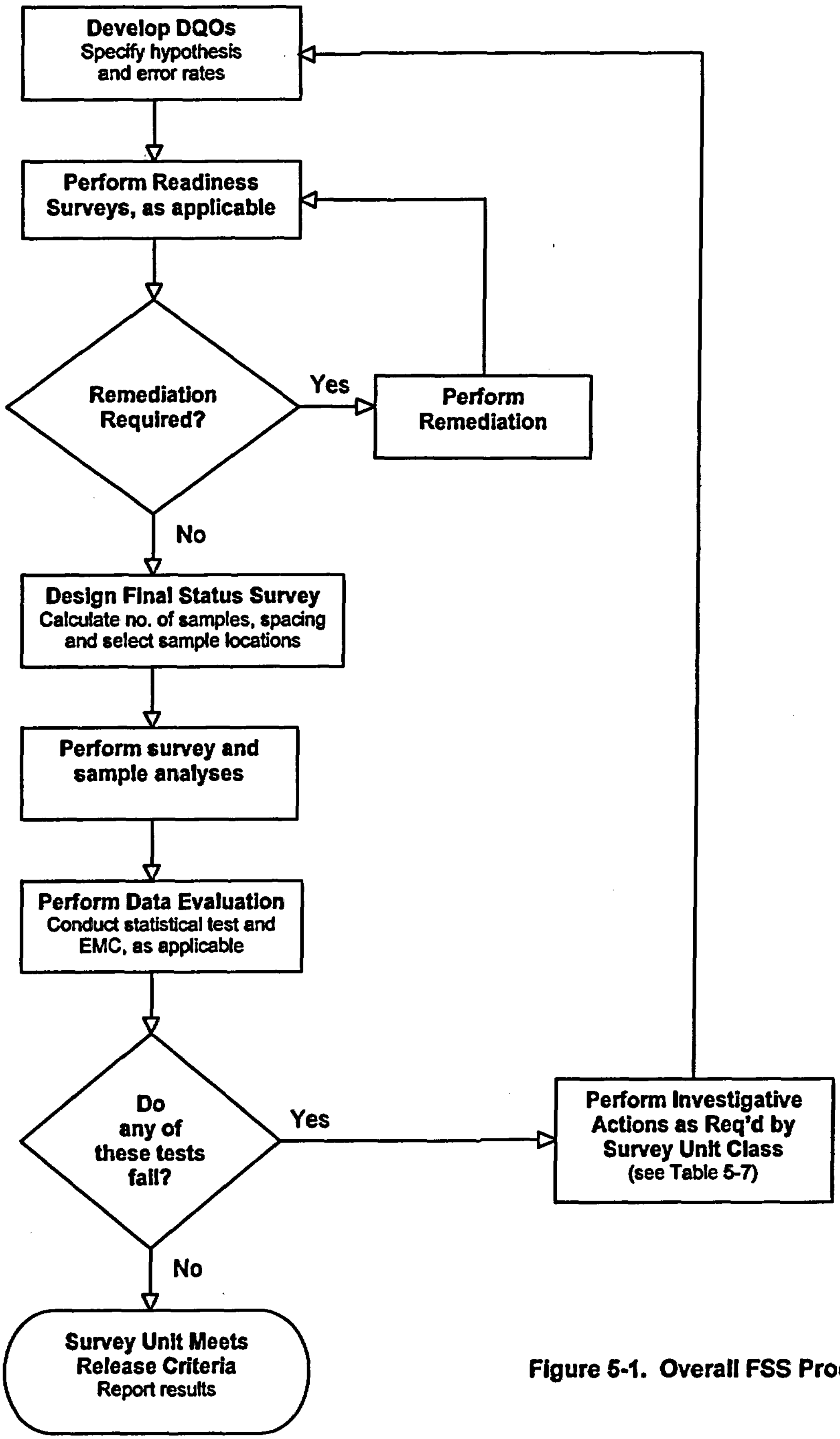

Figure 5-1. Overall FSS Process

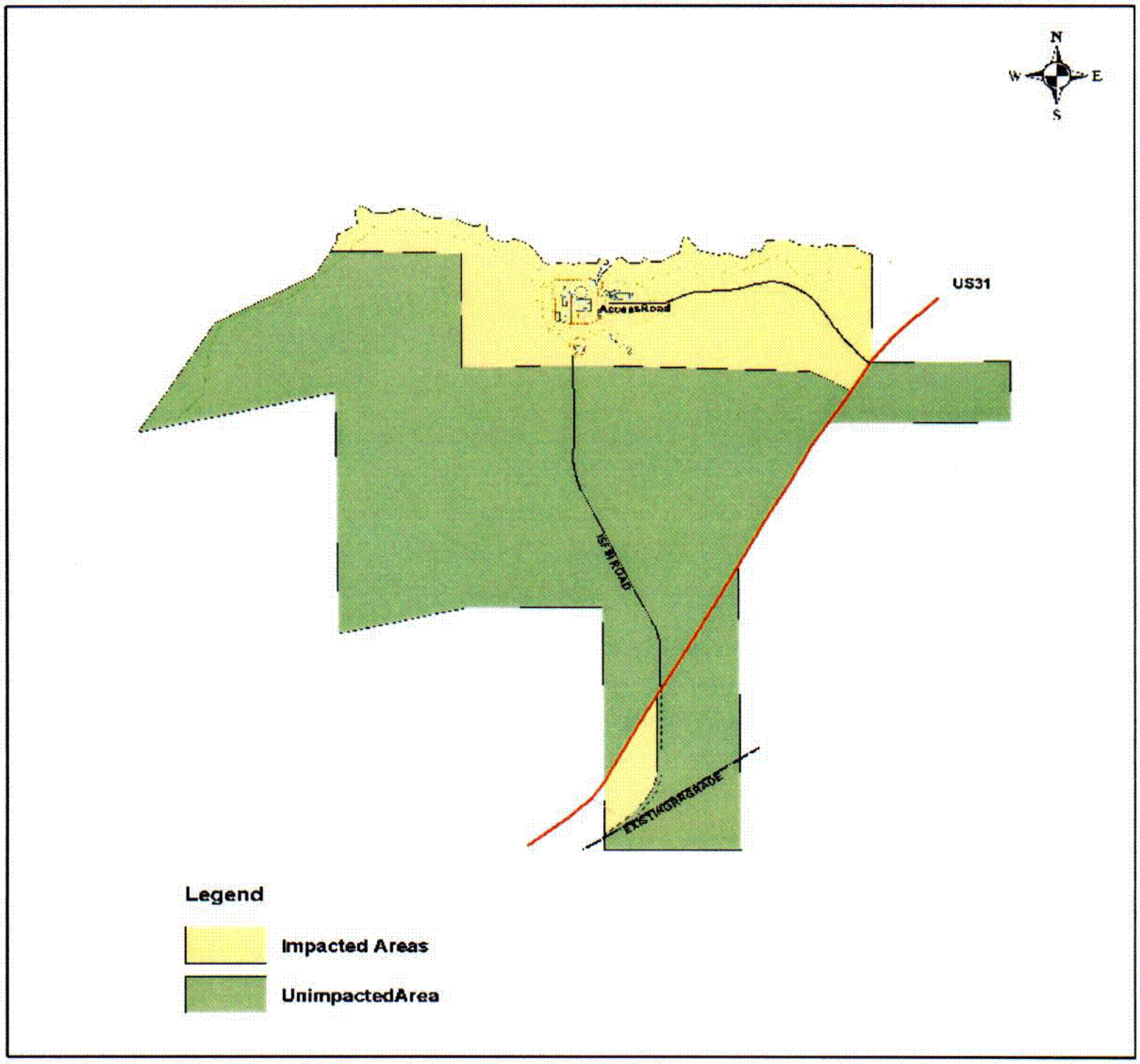

Figure 5-2: Big Rock Point Owner Controlled Area

C-03

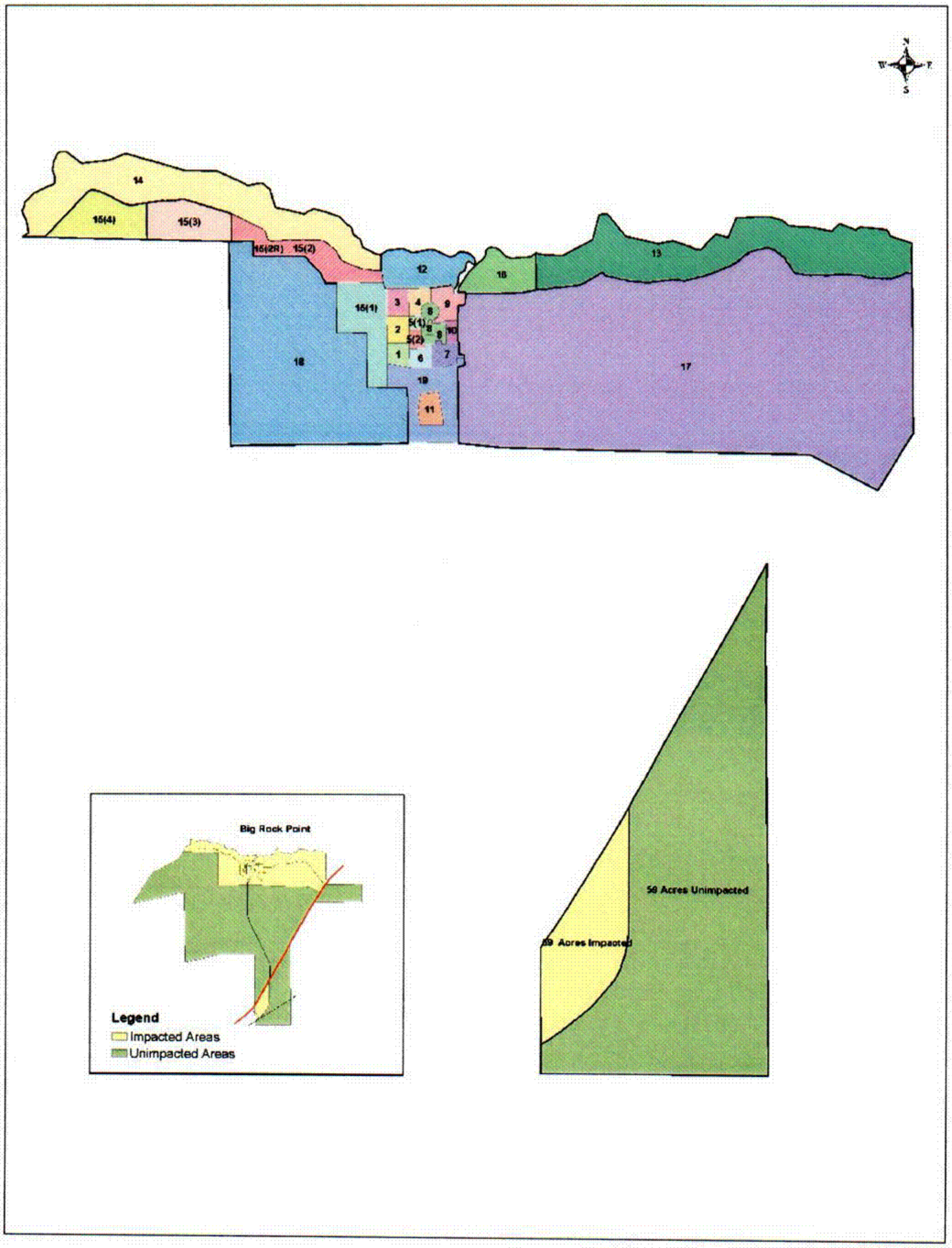

Figure 5-3. Initial Land Area Survey Units

C-04

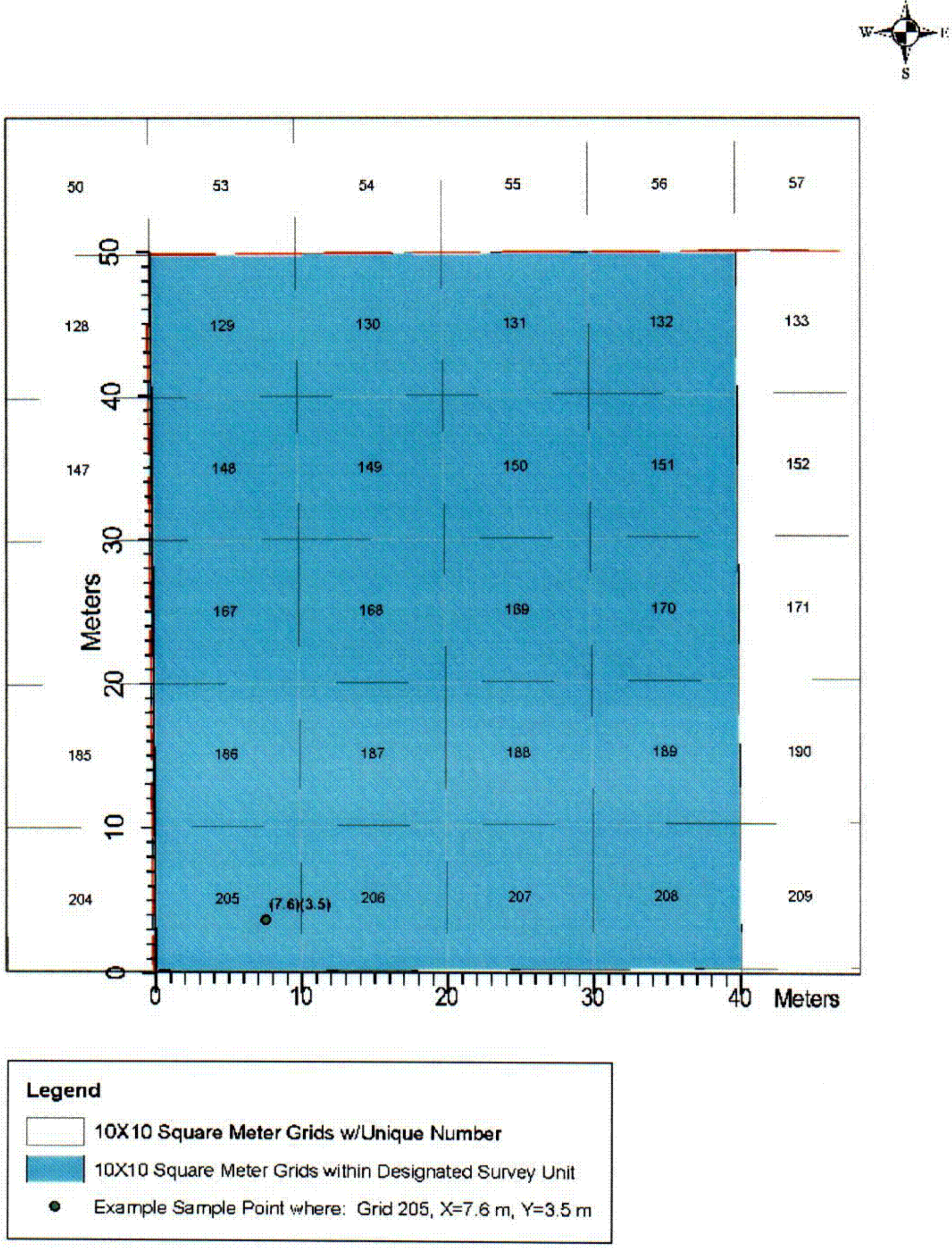

Figure 5-4. Example of Survey Unit Local Coordinate Grid

C-05

5.9 REFERENCES

5-1 Big Rock Point Administrative and Working Level Procedures

5-2 Big Rock Point Engineering Analysis EA-BRP-DW-98-01, *Efficiency Calibration of the Portable Gamma Spectrometer*

5-3 Big Rock Point Engineering Analysis EA-BRP-SC-01-01, *Deriviation of Scaling Factors for Hard-to-Detect Nuclides in Soil"*

5-4 Big Rock Point Engineering Analysis EA-BRP-SC-02-01, *NaI Scanning Sensitivity for Open Land Survey*

5-5 Big Rock Point Engineering Analysis EA-BRP-SC-02-04, *Radionuclides Present in Onsite Soil and Water*

5-6 Big Rock Point Engineering Analysis EA-BRP-SC-03-03, *Area Factors for Use in BRP Final Status Surveys*

5-7 Big Rock Point Historical Site Assessment, 1962-1997, May 2001

5-8 Big Rock Point Initial and Continuing Training for Decommissioning Course Descriptions

5-9 Consumers Energy Quality Program Description for Nuclear Power Plants, (CPC-2A) (Part 1) – Big Rock Point

5-10 U.S. Nuclear Regulatory Commission NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, August 2000

5-11 U.S. Nuclear Regulatory Commission Draft NUREG-1505, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys,* Revision 1, June 1998

5-12 U.S. Nuclear Regulatory Commission Draft NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Field Conditions*, June 1998

5-13 U.S. Nuclear Regulatory Commission Draft NUREG-1700, *Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans*, April 2000

5-14 U.S. Nuclear Regulatory Commission NUREG-1727, *NMSS Decommissioning Standard Review Plan*, September 15, 2000

5-15 U.S. Nuclear Regulatory Commission Regulatory Guide 1.179, *Standard Format and Content of License Termination Plans for Power Reactors*, January 1999

5-16 U.S. Nuclear Regulatory Commission Inspection Manual Procedure 84750, March 1994

5-17 USDA (vs. Department of Agriculture) Soil Survey, Charlevoix County, 1978

Calculation of Tritium in Soils

Concentration of tritium in soil is calculated from the maximum value of tritium in onsite monitoring wells accordance with Equation 5x.1. A bounding case is presented for tritium in water at a concentration of 20,000 pCi/l.

$S_t = [(W_t)(0.0875)/(\rho_s)]$

Where:

S_t = Soil concentration of tritium (pCi/g)

W_t = Maximum monitoring well water concentration of tritium (20,000 pCi/l for bounding case)

0.0875 = Maximum water saturation (by weight) for Algoma sandy loam soil in contaminated area [Reference 5-16]

ρ_s = Soil density (1,600 g/l) [Reference 5-2]

And

$D_t = (S_t)(24.218 \text{ mrem})/(DCGL_t)$

Where:

D_t = Annual dose from tritium in soil (mrem/year)

$DCGL_t$ = Tritium DCGL (328 pCi/g)

24.218 mrem = Annual dose associated with soil tritium DCGL (adjusted from 25 mrem for contribution from tritium in the three groundwater zones)

Thus,

S_t = (0.0825)(20,000)/1,600 = 1.03 pCi/g

D_t = (S_t)(24.218)/(328) = (1.03)(0.0738) = 0.076 mrem/year

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6.0 COMPLIANCE WITH THE RADIOLOGICAL DOSE CRITERIA

6.1 INTRODUCTION

The goal of the Big Rock Point (BRP) Restoration Project is to release the site, after removal of all structures, for unrestricted use in compliance with the Nuclear Regulatory Commission's (NRC) annual dose limit of 25 mrem/year plus ALARA (as low as reasonably achievable). The NRC dose limit applies to residual radioactivity that is distinguishable from background. This chapter of the License Termination Plan provides the methods for calculating the annual dose from residual radioactivity that may remain when the site is released for unrestricted use and the methods used to demonstrate compliance with the unrestricted use criteria.

Structures, foundations, paved surfaces and buried piping and utilities will have been removed prior to performance of the Final Status Survey (FSS) for each specified survey area.[1] Therefore, the scope of this chapter is limited to calculating annual dose resulting from surface and subsurface soil and groundwater contamination. This chapter will also provide a description of and the justification for:

- Source term assumptions,
- Exposure scenarios considering the site environment,
- The mathematical model/computational method used, and
- The parameter values and a measure of their uncertainty.

The Historical Site Assessment (HAS) and the Site Characterization have shown that the majority of the site area contains no residual radioactivity that is distinguishable from background radiation. Therefore, compliance with the unrestricted use criteria for these areas will be demonstrated by comparison of the FSS results with published unrestricted release screening criteria in accordance with the acceptance criteria contained in Section 5.0 of NUREG-1727, NMSS Decommissioning Standard Review Plan [Reference 6-19]. However, residual radioactivity has been identified in surface and subsurface soil and groundwater within the Industrial Area of the site. Therefore, since these areas do not meet the criteria for the use of unrestricted release screening criteria, site-specific Derived Concentration Guideline Levels (DCGLs) have been established in accordance with the description and the justification provided in this chapter for use in these areas.

[1] In accordance with the Greenfield description in Chapter 1, the septic drainfield and the intake structure and pipe will remain in place after all other restoration activities have been completed.

6.2 SITE RELEASE CRITERIA

6.2.1 Radiological Criteria for Unrestricted Use

The site release criteria for the BRP site will correspond to the radiological criteria for unrestricted use given in 10 CFR 20.1402, or:

- Dose Criterion: The residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/year, including that from groundwater sources; and

- ALARA Criterion: The residual radioactivity has been reduced to levels that are ALARA.

6.3 SITE CONDITIONS

6.3.1 General Description

A detailed description of applicable BRP environmental conditions and parameters is provided in Chapter 8 *Supplement to the Environmental Report*. These parameters include site and surrounding area physical descriptions including population, topography, vegetation, soil types, surface water quality, climate and meteorology, hydrology, and geology. Additionally Chapter 2, *Site Characterization,* (Section 2.4.3) contains a detailed description of the BRP site Hydrogeological Assessment results completed in 2002. The information contained in Chapters 2 and 8 form the basis for determining many of the site-specific dose modeling inputs (see Section 6.7.1). Site-specific environmental conditions and parameters were also compared to the basis for development of generic screening DCGL values discussed in Section 6.8.1.

6.4 SOURCE TERM ASSUMPTIONS

6.4.1 Potential Radionuclides of Concern

As part of the source-term abstraction process, an analysis was performed to identify a suite of radionuclides that could potentially be present in site soils and groundwater following completion of decommissioning activities and structure demolition [Reference 6-1]. This analysis considered radionuclides identified in two different companion NUREGs; NUREG/CR-3474, Long-Lived Activation Products in Reactor Materials, and NUREG/CR-4289, Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants [References 6-21 and 6-22].

NUREG/CR-3474 identified radionuclides and an activity inventory that would be present in BWR internals at the time of reactor shutdown. This inventory included trace elements that would not likely be found in site area soils due to their low abundance. An evaluation of radionuclides that may be discounted as being of potential importance for BRP was performed. Based on this evaluation, it was determined that individual radionuclides which contributed less than 0.1 percent of the total activity and with half-lives less than 244 days could be discounted from the list of identified radionuclides providing that dose contributed by the sum of the radionuclides discounted does not exceed one percent of the total calculated dose. The total percentage of activity attributed to radionuclides that meet these criteria amounts to 0.065 percent. The radionuclides that were discounted on this basis included Cl-36, Ar-39, Ca-41, Sc-46, Cr-51, Mn-53, Co-58, Fe-59, Se-79, Kr-81, Kr-85, Nb-92m, Zr-93, Mo-93, Ag-108m, Sn-121m, Sb-124, Ba-133, Cs-135, Ce-141, Pm-145, Sm-146, Sm-151, Tb-158, Ho-166m, Hf-178m, Pb-205, and U-233.

The dose contribution from discounted radionuclides was evaluated using the RESRAD computer code (version 6.0) in the deterministic mode using the resident farmer scenario to represent post demolition Greenfield conditions. An activity concentration was input for each radionuclide based on percent of total activity attributed to it. One hundred percent corresponded to 100 pCi/g. All RESRAD parameters were allowed to remain at their default values except for the thickness of the contaminated zone, which was set to 15 cm to correspond with the NRC's definition of thickness of contaminated surface soil. This evaluation indicated that the radionuclides discounted under the above criteria contributed less than one percent of the calculated dose at the year of maximum dose following site release.

NUREG/CR-4289 investigated residual radionuclide concentrations, distributions and inventories at seven nuclear power plants (four shutdown and three operating) to provide a database for use in formulating policies, strategies and guidelines for the eventual decommissioning of retired nuclear power plants. This study addressed radionuclides (both activation and fission products) transported from the reactor pressure vessel and deposited in all other contaminated systems of each nuclear plant. Emphasis was placed on measuring the long-lived radionuclides that are of special concern from a low-level waste management standpoint. The additional radionuclides identified in NUREG/CR-4289 were evaluated under the same criteria for discounting used for the radionuclides identified by NUREG/CR-3474.

Radionuclides identified by NUREG/CR-3474 and NUREG/CR-4289 were combined to form a single list of radionuclides. However, not all of these radionuclides were applicable for a site-specific suite of radionuclides for BRP because historical, site-specific radionuclide data exists from 10 CFR Part 61 waste stream characterization analyses. Review of this historical data shows that Nb-94, Ru-106, Sb-125, Ce-144 and Np-237 have not been observed in BRP waste stream analyses at concentrations above their reported minimum detectable activity (MDA) concentrations. Therefore, these radionuclides contained in NRC regulatory guidance may be discounted for purposes of the site-specific suite of radionuclides. In addition to identifying the absence of the radionuclides identified above, review of the historical data also indicated that Cm-243 has been observed in waste stream characterization analyses. Therefore, Cm-243 was added to the BRP site-specific suite of radionuclides.

An evaluation was performed to evaluate the validity of discounting Nb-94, Ru-106, Sb-125, Ce-144 and Np-237. This consisted of performing a RESRAD calculation for all of the radionuclides included in the site-specific suite of radionuclides. This calculation was again performed using the resident farmer scenario to represent post demolition Greenfield conditions. Reported MDA concentration values were used as input concentration values for Nb-94, Ru-106, Sb-125, Ce-144 and Np-237 as well as other radionuclides reported to have concentrations at less than MDA. This evaluation indicated that the discounted radionuclides contributed less than one percent of the calculated dose per year at the year of maximum dose following site release. This validates the assumption that Nb-94, Ru-106, Sb-125, Ce-144 and Np-237 may be discounted from the site-specific suite of radionuclides for use at BRP.

The above analysis resulted in a BRP site-specific suite of radionuclides that includes the 24 radionuclides listed in Table 6-1.

Table 6-1. BRP Site-Specific Suite of Radionuclides

H-3	Ni-63	Cs-134	Pu-239
C-14	Zn-65	Cs-137	Pu-240
Mn-54	Sr-90	*Eu-152	Pu-241
Fe-55	Tc-99	*Eu-154	Am-241
Ni-59	Ag-110m	*Eu-155	Cm-243
Co-60	I-129	Pu-238	Cm-244

* The europium isotopes are applicable specifically to the analysis of activated concrete and other materials, including soils, only after performance of activated concrete demolition work that could lead to contamination by these radionuclides.

6.4.2 Observed Radionuclides in the Big Rock Point Environs

During the site characterization process, soil and groundwater characterization data were judgmentally selected to be representative of the highest radioactivity levels at BRP. Additionally, soil samples were analyzed for combined gamma emitting and hard-to-detect (HTD) radionuclides (13 samples obtained for representative high contamination areas), subsurface soil samples analyzed for combined gamma emitting and HTD radionuclides (seven samples obtained in two areas where contamination exceeded 15 cm depth), and from groundwater samples analyzed for gamma and HTD nuclides (seven samples, including representative samples from each groundwater zone).

The samples analyzed for HTD radionuclides were sent to offsite contract laboratories for radiochemical analysis to determine the presence of 23 out of the 24 radionuclides identified in Table 6-1. The soil samples were not analyzed for tritium because the samples had been dried; however, tritium is known to be present in groundwater onsite and is, therefore, potentially present in soil. These soil samples represented the highest specific activity samples identified by onsite gamma spectroscopy during site characterization studies and are considered to be representative of site soils potentially contaminated with radionuclides. Water samples were selected from both the highest tritium areas (four samples) and from each representative subsurface groundwater aquifer (three samples). This sampling and analysis has been evaluated in an Engineering Analysis, EA-BRP-SC-0204, Radionuclides Present in Onsite Soil and Water [Reference 6-4].

The radionuclides Mn-54, Co-60 and Cs-137 have been identified in soil samples by onsite gamma spectroscopy. These same gamma-emitting radionuclides plus the HTD radionuclides Fe-55 and Sr-90 were identified in soil samples sent to the contract laboratories. Tritium has been identified in samples in the three subsurface groundwater aquifers. No gamma emitting or HTD radionuclides other than tritium have been identified in the eight water samples analyzed for HTD nuclides. It is concluded that this nuclide list includes all radionuclides present at measurable levels in onsite soils and groundwater.

Based on the above, the radionuclides listed in Table 6-2 are known to be potentially present in BRP surface and subsurface soils. Dose modeling was based on these radionuclides. The three europium isotopes were also included in the dose modeling because of the potential for soil contamination from activated concrete demolition debris.

Table 6-2. Radionuclides for Dose Modeling

H-3	Fe-55	Sr-90	Eu-152
Mn-54	Co-60	Cs-137	Eu-154
			Eu-155

6.5 EXPOSURE SCENARIOS CONSIDERING THE SITE ENVIRONMENT

6.5.1 Identification of the Critical Group

In the case of using screening DCGLs, the decisions involved in identifying the appropriate critical group have already been made. Critical group descriptions acceptable to the NRC for use in generic screening are developed and contained in NUREG/CR-5512, Volume 1 [Reference 6-23]. NUREG/CR-5512 and NUREG-1549 provide the rationale for applicability of the critical group [Reference 6-18]. The critical group described is a resident farmer, which is the most conservative critical group for generic screening. A farmer moves onto the site and grows some of his or her diet and uses water tapped from the onsite bedrock aquifer.

The critical group for site-specific analysis of the Industrial Area is considered to be a modified resident farmer; this individual lives on the site and grows some of his or her diet in an adjacent garden and uses water tapped from the onsite bedrock aquifer. However, as discussed in Sections 6.5.2 and 6.6.3, this resident would not consume animal products grown onsite.

6.5.2 Exposure Pathways for Scenario Consideration

6.5.2.1 Screening DCGL Exposure Pathways

The generic screening scenario accounts for exposure involving residual radioactivity that is initially in the surficial soil. The twelve pathways for exposure include:

- External exposure to penetrating radiation from volume soil sources while outdoors,
- External exposure to penetrating radiation from volume sources while indoors,
- Inhalation exposure to resuspended soil while outdoors,
- Inhalation exposure to resuspended soil while indoors,
- Inhalation exposure to resuspended surface sources of soil tracked indoors,
- Direct ingestion of soil,
- Inadvertent ingestion of soil tracked indoors,
- Ingestion of drinking water from a contaminated groundwater source,
- Ingestion of plant products grown in contaminated soil,
- Ingestion of plant products irrigated with contaminated groundwater,
- Ingestion of animal products (meat and milk) grown on the site, and
- Ingestion of fish from a contaminated surface water source.

6.5.2.2 Site-Specific Industrial Area Exposure Pathways

The modified resident farmer scenario for the Industrial Area also accounts for exposure involving residual radioactivity that is initially in the surficial soil. However, the scenario must also account for exposure involving subsurface contaminated soil and exposure from the three groundwater zones that contain low concentrations of tritium.

The exposure pathways for the site-specific Industrial Area dose modeling are the same as those involved with the generic screening scenario; however, the pathways involving exposure from the ingestion of animal products (meat and milk) grown on the site are suppressed. Because of the Industrial Area's Lake Michigan shoreline location, it is highly unlikely that the site would ever be used for subsistence farming (see Section 6.6.3 for additional discussion of site soil and demographic considerations).

6.6 MATHEMATICAL MODEL/COMPUTATIONAL METHOD USED

6.6.1 Mathematical Model for Dose Calculations

6.6.1.1 Screening DCGL Mathematical Model

As discussed in Section 6.8.1, screening DCGLs published by the NRC may be used to demonstrate compliance with the site release criteria presented in Section 6.2 for Impacted Areas of the site that do not have the potential for subsurface or groundwater contamination. Areas of the site classified as non-impacted, by definition, do not require demonstration of compliance with the site release criteria.

The screening DCGLs were derived from a mathematical model related to surface soils and a scenario analysis that combined exposure pathways for inhalation, external exposure, ingestion of contaminated drinking water, and ingestion of soil and agricultural food products, including fish from a pond. The residential scenario was intentionally developed in a conservative manner to account for potential residential and light agricultural activities.

The purpose of the generic modeling was to derive a groundwater concentration from residual radioactive materials in soil in a conservative manner to permit screening and to indicate when additional site data or modeling sophistication is warranted.

6.6.1.2 Site-Specific DCGL Mathematical Model

The hydrogeology of the site Industrial Area leads to modeling the Industrial Area as two distinct regions, the Southern Industrial Area and the Northern Industrial Area. The Southern Industrial Area is upgradient of all the site's significantly contaminated areas. It consists of six distinct unconsolidated geological units and spans approximately 80 feet from the surface to fractured bedrock. The Northern Industrial Area also consists of six distinct unconsolidated geological units, but spans only approximately 60 feet from the surface to fractured bedrock [Reference 6-12]. The RESRAD mathematical dose models for both the Southern Industrial Area and the Northern Industrial Area are depicted in Figure 6-1.

a. Southern Industrial Area Subsurface Description

Unit 6a, the uppermost unit consists of 0.91 to 1.8 meters (3.0 to 6.0 feet) of loosely compacted and well-rounded gravel or sand. Fine-grained clay or silt materials are noticeably absent within this unit. For purposes of the RESRAD mathematical model, a mean thickness of 1.37 meters (4.5 feet) is used. Physical parameters for Unit 6a are included in Appendix 6-B. Both the RESRAD dose model contaminated zone and the first unsaturated zone below it consist of Unit 6a soil.

The layer below Unit 6a in the Southern Industrial Area is Unit 5 which consists of approximately 7.3 meters (24 feet) of a very dense, unsorted mixture of roughly equal parts sand, fine grained clay and silt, and fine to coarse gravel (up to 7.6 centimeter (3.0 inch) diameter). For purposes of the RESRAD mathematical model, a thickness of 7.32 meters (24 feet) is used. Physical parameters for Unit 5 are included in Appendix 6-B. Unit 5 defines the second unsaturated zone in the RESRAD dose model.

Unit 4 is below Unit 5 in the Southern Industrial Area. Unit 4 consists of a moderately well sorted medium to coarse sand with some fine gravel. The thickness of Unit 4 in the Southern Industrial Area is approximately 1.5 meters (five feet). For purposes of the RESRAD mathematical model, a thickness of 1.52 meters (five feet) is used. Physical parameters for Unit 4 are included in Appendix 6-B.

Unit 4 defines the third unsaturated zone in the RESRAD dose model. Unit 4 is a groundwater zone under artesian conditions and is depicted in the RESRAD dose model as an unsaturated zone with physical parameters measured from the groundwater zone.

Unit 3 is below Unit 4 in the Southern Industrial Area. Unit 3 is a very dense deposit containing abundant fine-grained clay and silt, but also including significant sand and fine gravel fractions. This unit is similar to Unit 5, in that the clay rich portions appear to be unsorted, but it also contains frequent thin sand seams 0.30 to 1.5 centimeters (0.01 to 0.05 feet) thick. The sand seam frequency varies, but the average is estimated to be three to four seams per 30 centimeters (foot). The thickness of Unit 3 in the Southern Industrial Area is approximately 9.1 meters (30 feet). For purposes of the RESRAD mathematical model, a thickness of 9.14 meters (30 feet) is used. Physical parameters for Unit 3 are included in Appendix 6-B. Unit 3 defines the fourth unsaturated zone in the RESRAD dose model.

The layer below Unit 3 in the Southern Industrial Area is Unit 2, which consists of approximately 6.1 meters (20 feet) of a very dense to hard dry deposit, composed of significant fractions of both clay and silt with only minor sand. This layer is absent in the Northern Industrial Area. Physical parameters for Unit 2 are included in Appendix 6-B. Unit 2 defines the fifth unsaturated zone in the RESRAD dose model.

Unit 1 comprises the bottom zone in the RESRAD dose model for the Southern Industrial Area. It consists of gray-brown, fine to medium grained limestone with occasional fossil rich zones. The bedrock exhibited frequent natural fractures in both high-angle and bedding parallel orientations. Other features present included numerous thin, irregular zones containing dark colored, carbonaceous material. Physical parameters for Unit 1 are included in Appendix 6-B. Unit 1 defines the saturated zone in the RESRAD dose model and is the source of potable water supply for an onsite aquifer.

b. Northern Industrial Area Subsurface Description

Unit 6a, the uppermost unit consists of 0.91 to 1.8 meters (3.0 to 6.0 feet) of loosely compacted and well-rounded gravel or sand. Fine grained clay or silt materials are noticeably absent within this unit. For purposes of the RESRAD mathematical model, a mean thickness of 1.37 meters (4.5 feet) is used. Physical parameters for Unit 6a are included in Appendix 6-C. Both the RESRAD dose model contaminated zone and the first unsaturated zone below it consist of Unit 6a soil.

The layer below Unit 6a in the Northern Industrial Area is Unit 6b, which consists of 0.91 to 1.8 meters (three to six feet) of a slightly more compact sand and gravel unit with intermixed clay-rich layers. The gravel in this zone is generally finer grained (less than 1.3 centimeter (0.5 inch) diameter), and all of the granular particles tended to be subrounded to slightly angular; the individual clay-rich layers (clay, clayey sand, clayey gravel) ranged in thickness from 3.0 to 15 centimeters (0.1 to 0.5 feet). For purposes of the RESRAD mathematical model, a mean thickness of 1.37 meters (4.5 feet) is used. Physical parameters for Unit 6b are included in Appendix 6-C. Unit 6b defines the second unsaturated zone in the RESRAD dose model. Unit 6b is an unconfined groundwater zone and it is depicted in the RESRAD dose model as an unsaturated zone with physical parameters measured from the groundwater zone.

The layer below Unit 6b in the Northern Industrial Area is Unit 5 which consists of approximately 2.1 to 6.1 meters (seven to 20 feet) of a very dense, unsorted mixture of roughly equal parts sand, fine grained clay and silt, and fine to coarse gravel (up to 7.6 centimeter [3.0 inch] diameter). For purposes of the RESRAD mathematical model, a thickness of 4.11 meters (13.5 feet) is used. Physical parameters for Unit 5 are included in Appendix 6-C. Unit 5 defines the third unsaturated zone in the RESRAD dose model.

Unit 4 is below Unit 5 in the Northern Industrial Area. Unit 4 consists of a moderately well sorted medium to coarse sand with some fine gravel. The thickness of Unit 4 in the Northern Industrial Area is approximately 0.61 meters (two feet). For purposes of the RESRAD mathematical model, a thickness of 0.61 meters (two feet) is used. Physical parameters for Unit 4 are included in Appendix 6-C. Unit 4 defines the fourth unsaturated zone in the RESRAD dose model. Unit 4 is a groundwater zone under artesian conditions and is depicted in the RESRAD dose model as an unsaturated zone with physical parameters measured from the groundwater zone.

Unit 3 is below Unit 4 in the Northern Industrial Area. Unit 3 is a very dense deposit containing abundant fine-grained clay and silt, but also including significant sand and fine gravel fractions. This unit is similar to Unit 5, in that the clay rich portions appear to be unsorted, but it also contains frequent thin sand seams (0.30 to 1.5 centimeters [0.01 to 0.05 feet] thick). The sand seam frequency varies, but the average is estimated to be three to four seams per 0.30 meter (foot). The thickness of Unit 3 in the Northern Industrial Area is approximately 7.3 meters (24 feet). For purposes of the RESRAD mathematical model, a thickness of 7.32 meters (24 feet) is used. Physical parameters for Unit 3 are included in Appendix 6-C. Unit 3 defines the fifth unsaturated zone in the RESRAD dose model.

Unit 1 comprises the bottom zone in the RESRAD dose model for the Northern Industrial Area. It consists of gray-brown, fine to medium grained limestone with occasional fossil rich zones. The bedrock exhibited frequent natural fractures in both high-angle and bedding parallel orientations. Other features present included numerous thin, irregular zones containing dark colored, carbonaceous material. Physical parameters for Unit 1 are included in Appendix 6-C. Unit 1 defines the saturated zone in the RESRAD dose model and is the source of potable water supply for an onsite aquifer.

6.6.2 Computational Method Used for Dose Calculations

RESRAD v6.21 was selected as the computer code to perform site-specific dose modeling within the site Industrial Areas because of the ability to model subsurface soil contamination and groundwater contamination contained within the code. Argonne National Laboratory (ANL) developed the RESRAD computer code under the sponsorship of the U.S. Department of Energy (DOE). The code has been used widely by DOE and its contractors, the U.S. NRC, U.S. Environmental Protection Agency (EPA), U.S. Army Corps of Engineers, industrial firms, universities, and foreign government agencies and institutions. This code is a pathway analysis model designed to evaluate potential radiological doses to an average member of the specific critical group.

The NRC adopted a risk-informed approach in assessing impacts on the health and safety of the public from radioactive contamination. Therefore, the NRC tasked ANL to develop parameter distribution functions and parametric analysis for RESRAD for conducting probabilistic dose analysis. As part of this effort, external modules equipped with probabilistic sampling and analytical capabilities were developed for the RESRAD code. The modules are also equipped with user-friendly input/output interface features to accommodate numerous parameter distribution functions and to fulfill results display requirements.

The RESRAD database includes inhalation and ingestion dose conversion factors from the EPA's Federal Guidance Report (FGR) No. 11, direct external exposure dose conversion factors from FGR-12 and radionuclide half-lives from International Commission on Radiological Protection Publication 38 [References 6-15, 6-16 and 6-10].

6.6.3 *Identification of the Critical Group*

The critical group for site-specific analysis of the Industrial Area is considered to be a modified resident farmer who moves onto the site and grows some of his or her diet in a garden and uses water tapped from the bedrock aquifer beneath the site. However, this resident farmer would not consume animal products grown onsite. Since the Industrial Area is located on the Lake Michigan shoreline, it is highly unlikely the site would ever be used for subsistence farming. The lakeshore of Little Traverse Bay in Lake Michigan is highly developed for summer residence and recreational uses. In addition, there currently are no Lake Michigan shoreline farms within 20 miles of Charlevoix. Only 10.1 percent of Charlevoix County land is used for agricultural purposes and the county has an established declining trend in land use for agricultural purposes. Also, lakeshore soils in the area are poorly suited for subsistence farming in that the soil is a gravelly-sandy loam containing low natural fertility and having a moderately low organic content. Finally, lakeshore property values would effectively prohibit use of the site for subsistence farming and it is likely that the future use of the site would be resort or recreational use. For these reasons, it is justifiable to suppress the meat and milk pathways in the RESRAD dose model.

Suppression of the meat and milk pathways has minimal impact on dose modeling. Turning the meat and milk pathways on would result in less than a one percent increase in potential discounted dose discussed in Section 6.7.2. The impact on DCGL values discussed in Section 6.8.2 would range from no impact on DCGL values for the europium isotopes to a 16.5 percent reduction in the DCGL value for Sr-90. The DCGL value for Fe-55 would have a 77.5 percent reduction, but still have a large value of 8.07 E+04 pCi/g.

6.7 PARAMETER VALUES AND A MEASURE OF THEIR UNCERTAINTY

6.7.1 Selection of Parameters for Use with RESRAD

6.7.1.1 Parameter Classification

Dose assessment input parameters may be generally classified as physical, behavioral, or metabolic as defined below.

Physical Parameter: Any parameter whose value would not change if a different group of receptors were considered is classified as a physical parameter. Physical parameters would be determined by the source, its location, and geological characteristics of the site (i.e., these parameters are source- and site-specific). These include the hydrogeological, geochemical, and meteorological characteristics of the site. The characteristics of atmospheric and biospheric transport up to, but not including, uptake by, or exposure of, the dose receptor, would also be considered physical input parameters.

Behavioral Parameter: Any parameter whose value would depend on the receptor's behavior and the scenario definition is classified as a behavioral parameter. For the same group of receptors, a parameter value could change if the scenario changed (e.g., parameters for recreational use could be different from those for residential use).

Metabolic Parameter: If a parameter represents the metabolic characteristics of the potential receptor and is independent of scenario, it is classified as a metabolic parameter. The parameter values may vary in different population age groups. According to the recommendations of the International Commission on Radiological Protection, Report 43 (ICRP 1985), parameters representing metabolic characteristics are defined by average values for the general population [Reference 6-11]. These values are not expected to be modified for a site-specific analysis because the parameter values would not depend on site conditions.

NUREG-1727, *NMSS Decommissioning Standard Review Plan*, states the licensee may use default values for the behavioral and metabolic parameters, with limited justification, if the values are consistent with the generic definition of the average member of the critical group, and the screening group is reflective of the scenario. Therefore, RESRAD v6.21 default values will be used for the behavioral and metabolic parameters. RESRAD v6.2 input parameters that impact dose, along with their classification and description are provided in Appendix 6-A. These parameters and their classification and description are based on Table 2.1 of Attachment A to NUREG/CR-6697, *Development of Probabilistic RESRAD 6.0* and RESRAD-BUILD 3.0 Computer Codes [Reference 6-26].

6.7.1.2 Default Parameters

As discussed in Section 6.5, BRP dose calculations and derivation of site-specific DCGLs was based on a modified resident farmer scenario. The modified resident farmer scenario accounts for exposure involving residual radioactivity that is initially in the surficial soil, contained in subsurface soils at known maximum concentration levels and contained in the three Industrial Area groundwater zones at known maximum concentration levels. A resident moves onto the site and grows default amounts of his or her diet in a garden and uses water tapped from the bedrock aquifer under the site. This scenario is based on the following assumptions:

- Radioactive contamination occurs in a surface soil layer, subsurface soil and the three Industrial Area groundwater zones.
- The property will not be used for livestock and dairy animal production.
- Residency can occur immediately after release of the property.
- Radioactive dose results from exposure via external exposure, inhalation, and ingestion.

The model includes eleven exposure pathways created by the activities considered in the scenario as follows:

- External exposure to penetrating radiation from volume soil sources while outdoors,
- External exposure to penetrating radiation from volume sources while indoors,
- Inhalation exposure to resuspended soil while outdoors,
- Inhalation exposure to resuspended soil while indoors,
- Inhalation exposure to resuspended surface sources of soil tracked indoors,

- Direct ingestion of soil,
- Inadvertent ingestion of soil tracked indoors,
- Ingestion of drinking water from a contaminated groundwater source,
- Ingestion of plant products grown in contaminated soil,
- Ingestion of plant products irrigated with contaminated groundwater, and
- Ingestion of fish from a contaminated surface water source.

In accordance with NUREG-1727, default values for the behavioral and metabolic parameters were used in the model.

6.7.1.3 Big Rock Point Site-Specific Measured Physical Parameters

To the extent possible, site-specific values were used for physical parameters and mixed physical (parameters classed as physical and behavioral or metabolic parameters). The site-specific values were derived from information contained in BRP Plant Manual, Volume 32, *Environmental Report for Decommissioning*, to the extent possible and the November 8, 2002 Hydrogeological Assessment Report [Reference 6-9]. The input parameter values and their source justification are provided in Appendices 6-B and 6-C.

6.7.1.4 Big Rock Point Site-Specific Calculated Physical Parameters

a. Derivation of Site-Specific Distribution Coefficients (K_ds)

Big Rock Point chose to use one of the methods discussed in Appendix O to Draft NUREG-1757, Volume 2, *Consolidated NMSS Decommissioning Guidance - Characterization, Survey, and Determination of Radiological Criteria*, for the development of input K_d values for soil [Reference 6-20]. According to the answer to Question 6, K_d values for input into site-specific dose modeling codes may be determined by using sensitivity analyses, which include an appropriate range of K_d values, to identify the importance of the K_d to the dose assessment and how the change in K_d impacts the dose (i.e., how dose changes as K_d increases or decreases).

Using the results of the sensitivity analysis, a conservative K_d value is chosen, depending on how it affects the dose (e.g., if higher K_d values result in the larger dose, an input K_d value should be selected from the upper quartile of the distribution, or if lower K_d values result in the larger dose, an input K_d value should be selected from the lower quartile of the distribution).

K_d values were derived for both the site Southern and Northern Industrial Areas using the parameters contained in Appendices 6-B and 6-C.

Following preparation of the dose models, a probabilistic RESRAD calculation was performed for each of the radionuclides individually and for individual radionuclides in each location of the Industrial Area under evaluation [Reference 6-2]. An uncertainty analysis was performed for the radionuclide's (element's) K_d value in each of the seven RESRAD zones (contaminated zone, unsaturated zones 1-5, and the saturated zone). K_d parameter distribution values for each element were obtained from Attachment C to NUREG/CR-6697. The statistical distribution type for K_d is truncated lognormal-n. The K_d parameter distribution values for each element are provided in Table 6-3.

Table 6-3. K_d Parameter Distribution Values

Element	Log Mean (μ)	Standard Deviation (σ)	Lower Quartile	Upper Quartile
H	-2.81	0.5	0.001	0.999
Mn	5.06	2.29	0.001	0.999
Fe	5.34	2.67	0.001	0.999
Co	5.46	2.53	0.001	0.999
Sr	3.45	2.12	0.001	0.999
Cs	6.10	2.33	0.001	0.999
Eu	6.72	3.22	0.001	0.999
Gd	6.72	3.22	0.001	0.999

Following completion of each RESRAD calculation, the most conservative K_d value for each of the seven RESRAD zones was determined. In accordance with the guidance contained in Appendix O to Draft NUREG-1757, the K_d value at the upper or lower quartile of the distribution resulting in the highest derived dose is an acceptable value to input into the dose code without further justification. For those radionuclides where the K_d does not have a significant impact on the dose assessment (i.e., K_d is not a sensitive parameter), the median value within the range is an acceptable input parameter. If the absolute value of the partial ranked correlation coefficient (PRCC) was greater than 0.25, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value of the peak all-pathway dose was negative, the K_d to dose correlation is negative and the K_d value at the 25% quartile was selected. If the PRCC value was positive, the K_d to dose correlation is positive and the K_d value at the 75% quartile was selected. If the absolute value of the PRCC was equal to or less than 0.25, the dose correlation is not sensitive and the K_d value at the 50% quartile was selected. Partial ranked correlation coefficient was chosen because NUREG/CR-6692 recommends that it be used when nonlinear relationships, widely disparate scales or long tails are present in the inputs and outputs [Reference 6-25].

The results of the K_d derivation are summarized in Table 6-4 for the Southern Industrial Area and in Table 6-5 for the Northern Industrial Area.

Table 6-4. Southern Industrial Area K_d Results (cm^3/g) and PRCC* Value ( )

Element	Contaminated Zone	Unsaturated Zone 1	Unsaturated Zone 2	Unsaturated Zone 3	Unsaturated Zone 4	Unsaturated Zone 5	Saturated Zone
H	0.043(-0.47)	0.060(0.04)	0.043(-0.35)	0.060(-0.01)	0.043(-0.46)	0.043(-0.31)	0.060(-0.08)
Mn	733(1.00)	156(-0.05)	155(0.02)	157(-0.05)	155(0.02)	155(-0.05)	165(-0.03)
Fe	1251(1.00)	206(-0.02)	204(-0.09)	207(-0.02)	215(-0.04)	204(0.05)	205(0.01)
Co	1284(1.00)	232(-0.09)	231(-0.04)	234(0.02)	231(0.03)	230(-0.11)	232(0.05)
Sr	131(0.98)	31.2(-0.03)	31.0(-0.06)	31.4(-0.07)	31.0(-0.03)	31.0(-0.08)	31.1(-0.02)
Cs	2130(0.99)	441(-0.04)	438(-0.05)	444(-0.05)	438(0.02)	438(0.06)	440(0.03)
Eu	7194(1.00)	817(0.03)	809(-0.06)	824(-0.03)	809(0.03)	808(-0.02)	814(-0.02)
Gd	7194(0.98)	817(0.00)	809(-0.02)	824(0.00)	809(-0.02)	808(0.02)	814(-0.06)

*PRCC = Partial ranked correlation coefficient for peak all-pathways dose

Table 6-5. Northern Industrial Area K_d Results (cm^3/g) PRCC* Value ( )

Element	Contaminated Zone	Unsaturated Zone 1	Unsaturated Zone 2	Unsaturated Zone 3	Unsaturated Zone 4	Unsaturated Zone 5	Saturated Zone
H	0.043(-0.57)	0.060(-0.05)	0.060(-0.07)	0.060(-0.04)	0.060(-0.07)	0.043(-0.35)	0.060(0.04)
Mn	733(1.00)	156(-0.05)	155(0.02)	157(-0.05)	155(0.02)	155(-0.05)	156(-0.03)
Fe	1251(1.00)	206(-0.02)	204(-0.09)	207(-0.02)	204(-0.04)	204(0.05)	205(0.01)
Co	1284(1.00)	232(-0.09)	231(-0.04)	234(0.02)	231(0.03)	230(-0.11)	232(0.05)
Sr	131(0.95)	31.2(-0.14)	31.0(-0.05)	31.4(-0.12)	31.0(-0.03)	31.0(-0.14)	31.1(0.00)
Cs	2130(0.99)	441(-0.04)	438(-0.05)	444(-0.05)	438(0.02)	438(0.06)	440(0.03)
Eu	7194(1.00)	817(0.03)	809(-0.06)	824(-0.03)	809(0.03)	808(-0.02)	814(-0.02)
Gd	7194(0.98)	817(0.00)	809(-0.02)	824(0.00)	809(-0.02)	808(0.02)	814(-0.06)

*PRCC = Partial ranked correlation coefficient for peak all-pathways dose

The derived K_d values are nearly identical for both the Southern and Northern Industrial Areas. Furthermore, the calculated dose with an input concentration of 1 pCi/g for each of the nine radionuclides in the Southern and Northern Industrial Area models with the calculated K_d values yields an identical calculated dose of 19.8 mrem/year at the initial time. This occurs because we have shown that the dominant exposure pathways are, with the exception of tritium, water independent for the six radionuclides identified to be present in BRP soils. Water independent parameters (contamination depth, contaminated zone soil constants, root depths and other parameters which affect dose from the water independent pathways) are the same for the both the Northern and Southern Industrial Areas.

b. Derivation of Site-Specific Physical Parameters

Whenever possible, site-specific physical parameter values were determined by direct measurement. If a physical parameter value could not be determined by direct measurement, a site-specific value was derived by a probabilistic sensitivity analysis using RESRAD v6.21 [Reference 6-3]. The process followed for this probabilistic sensitivity analysis is outlined in Figure 6-2.

The mathematical model used for the parameter selection process was the RESRAD model for the Northern Industrial Area. This model was selected because there are no known significant areas of surface soil, subsurface soil or groundwater contamination within the Southern Industrial Area. Also, the Northern Industrial Area RESRAD model is considered to be the most conservative because the bedrock aquifer is closer to the surface in the Northern Industrial Area than it is in the Southern Industrial Area and the silty clay layer directly above the bedrock aquifer in the Southern Industrial Area does not exist in the Northern Industrial Area. As noted in Section 6.7.2.4.a, no difference is found for doses calculated in the Northern or Southern Industrial Areas for the radionuclides found in soils at BRP. This is because doses from these radionuclides are driven by water independent pathways. Some doses from discounted radionuclides, calculated at laboratory analytical minimum detectable concentration levels (see LTP Section 6.7.2), are more dependent upon water pathways. Thus, the Northern Industrial Area parameters are more conservative for these radionuclides.

The upper two groundwater zones are not considered to be viable sources of potable water due to low yield and distance from the surface, i.e., State of Michigan statutes prohibit drilling wells this shallow. Also, it is not considered likely that a resident would decide to drop a well into the sand lens to get some water for irrigation in violation of State of Michigan statutes because, due to the shoreline location, it would be easier to run a hose to Lake Michigan to obtain irrigation water.

As shown schematically in Figure 6-2, the parameter selection process starts with the selection of the RESRAD parameter to be evaluated. For purposes of BRP dose modeling, the RESRAD pathways for radon, meat consumption and milk consumption are suppressed and their associated parameters are not used. The selected parameter must then be classified as behavioral, metabolic or physical. Some parameters may belong to more than one of these types.

If the parameters were classified as behavioral or metabolic, the default values used in RESRAD v6.21 were used for performing sensitivity analyses. These parameter values are provided in Appendix 6-D. If the parameters were classified as physical, then they were reviewed to determine if measured, site-specific values for the parameters were available. Measured, site-specific values for physical parameters are also provided in Appendix 6-D.

If measured, site-specific values for physical parameters were not available; the parameters were then ranked by priority as 1, 2, or 3 where 1 represents high priority, 2 represents medium priority and 3 represents low priority. This ranking was the second step in the procedure used by Argonne National Laboratory to develop the probabilistic RESRAD code. The parameter ranking has been documented in Attachment B to NUREG/CR-6697. The assigned priority ranking for each physical parameter for which measured, site-specific values are not available is provided in Appendix 6-D.

If the physical parameters were ranked as priority 3, the default values used in RESRAD v6.21 were used for performing sensitivity analyses. These parameter values are also provided in Appendix 6-D. Three exceptions to this assignment exist. These are for the parameters of humidity in air, indoor time fraction and inhalation rate for which statistical parameter distributions were developed.

Argonne National Laboratory developed statistical parameter distributions for the physical parameters ranked as priority 1 or 2. These parameter distributions have been documented in Attachment C to NUREG/CR-6697. The parameter distributions for BRP priority 1 and 2 physical parameters are listed in Appendix 6-E.

Once the parameter values listed in Appendix 6-D and the statistical parameter distributions listed in Appendix 6-E were loaded into RESRAD v6.21, the code was run in the probabilistic mode. The absolute value of the calculated PRCC of the peak all-pathways dose was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. PRCC was chosen because Reference 6-25 recommends it be used when nonlinear relationships, widely disparate scales or long tails are present in the inputs and outputs. The calculated PRCC values are listed in Appendix 6-E. If the absolute value of the PRCC was greater than 0.25, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.25, then the parameter was classified as non-sensitive.

Finally, values for use in dose modeling for the physical parameters with statistical distributions were selected based on sensitivity of the calculated PRCC.

If the absolute value of the PRCC was greater than 0.25, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value of the peak all-pathways dose was negative, the K_d parameter to dose correlation is negative and the K_d parameter value at the 25% quartile was selected. If the PRCC value was positive, the K_d parameter to dose correlation is positive and the K_d value at the 75% quartile was selected.

If the absolute value of the PRCC was equal to or less than 0.25, then the parameter value at the 50% quartile was selected.

The appropriate behavioral, metabolic, physical and probabilistically determined parameter values listed in Appendices 6-D and 6-E were compiled and listed in Appendix 6-F. The RESRAD v6.21 parameters listed in Appendix 6-F are appropriate for dose modeling of surface soil, subsurface soil and groundwater zone contamination at BRP.

6.7.2 Potential Dose From Discounted Radionuclides

As discussed in Section 6.4, a site-specific suite of 24 radionuclides was identified as potentially present in BRP soils. However, only six of these radionuclides were observed in soils during site characterization. Therefore, only these six radionuclides plus the three isotopes of europium were considered in dose modeling. Because of this, an evaluation was performed to ensure the potential dose contribution from these 15 discounted radionuclides does not exceed ten percent of the 10 CFR 20.1402 TEDE dose limit of 25 mrem/year for unrestricted license termination [Reference 6-6]. The radionuclides of concern include the following 15 radionuclides listed in Table 6-6.

Table 6-6. Discounted Radionuclides

C-14	Tc-99	Pu-238	Am-241
Ni-59	Ag-110m	Pu-239	Cm-243
Ni-63	I-129	Pu-240	Cm-244
Zn-65	Cs-134	Pu-241	

6.7.2.1 Derivation of Site-Specific Parameters for Discounted Radionuclides

The RESRAD dose model was prepared in the same manner as that used by EA-BRP-SC-0202 to derive K_d values, using the parameters contained in Appendix 6-F. A probabilistic RESRAD v6.21 calculation was performed for each of the 15 discounted radionuclides. An uncertainty analysis was performed for the radionuclide (element) and progeny K_d value in each of the seven RESRAD dose model zones (contaminated zone, unsaturated zones 1 - 5, and the saturated zone). K_d parameter distribution values for each element were obtained from Attachment C to NUREG/CR-6697. The statistical distribution type for K_d is truncated lognormal-n. The K_d parameter distribution values for each element are provided in Appendix 6-G.

Following completion of each RESRAD calculation, the appropriate K_d value for each of the seven RESRAD zones was determined. The K_d value at the upper, middle or lower quartile of the distribution was selected as the appropriate K_d value. If the absolute value of the PRCC was greater than 0.25, then the parameter value at either the 75% quartile or the 25% quartile was selected based on TEDE correlation with the parameter. If the PRCC value of the peak all-pathway dose was negative, the K_d value at the 25% quartile was selected. If the PRCC value was positive, the K_d value at the 75% quartile was selected. If the absolute value of the PRCC was equal to or less than 0.25 the dose correlation is not sensitive and the K_d value at the 50% quartile was selected.

The results of the K_d derivation from the RESRAD v6.21 calculations are summarized in Appendix 6-G. K_d value results were rounded to three significant figures.

EA-BRP-SC-0203 derived appropriate physical parameter values for use in BRP dose modeling using probabilistic sensitivity analyses. These physical parameters included those for which site-specific measurements were not obtained; however, radionuclide specific physical parameters were derived only for radionuclides that have been observed in BRP soils (and the three isotopes of europium). Therefore, in order to calculate potential dose from the discounted radionuclides, it was necessary to derive radionuclide specific physical parameters using the same methodology as that used to perform EA-BRP-SC-0203. This parameter selection process is outlined in Figure 6-2. For purposes of BRP dose modeling, the RESRAD pathways for radon, meat consumption and milk consumption are suppressed and their associated parameters are not used. The parameter classification and priority ranking from EA-BRP-SC-0203 were retained. The assigned parameter classification and priority ranking are listed in Appendix 6-G. The parameter distribution types from Attachment C to NUREG/CR-6697 are also listed in Appendix 6-G. Once the parameter values listed in Appendix 6-F, the K_d values derived above and the statistical parameter distributions referred to in Appendix 6-G were loaded into RESRAD v6.21, the code was run in the probabilistic mode. The absolute value of the calculated PRCC was then used to classify the parameters with statistical distributions as sensitive or non-sensitive. If the absolute value of the PRCC was greater than 0.25, then the parameter was classified as sensitive. If the absolute value of the PRCC was equal to or less than 0.25, then the parameter was classified as non-sensitive. The parameter value was selected on the basis of TEDE correlation as described above for K_d value selection. The selected parameter values are listed in Appendix 6-G.

6.7.2.2 Calculation of Potential Discarded Dose from Discounted Radionuclides

The potential discounted dose was calculated based on the median minimum detectable activity (MDA) for each radionuclide from the laboratory analyses performed to analyze for each radionuclide in the entire suite of 24 radionuclides. The median MDA for each radionuclide and the associated analytical uncertainty is provided in Table 6-7 below.

Table 6-7. MDA Values for Calculation of Potential Discounted Dose

Radionuclide	MDA (pCi/g)	Radionuclide	MDA (pCi/g)
C-14	0.470 ± 0.821	Pu-238	0.0450 ± 2.09
Ni-59	2.35 ± 2.70	Pu-239	0.040 ± 0.114
Ni-63	0.400 ± 1.02	Pu-240	0.040 ± 0.114
Zn-65	0.0687 ± 0.077	Pu-241	7.25 ± 1.62
Tc-99	0.310 ± 1.09	Am-241	0.0665 ± 0.078
Ag-110m	0.0300 ± 0.016	Cm-243	0.0601 ± 0.131
I-129	0.260 ± 0.037	Cm-244	0.0601 ± 0.131
Cs-134	0.0222 ± 0.015		

The dose model used for this calculation included the parameter values provided in Appendix 6-G. Soil contamination has only been observed to occur in small areas of no more than 150 square meters (1600 square feet). Also, no contaminated soil has been observed above the characterization DCGL values at depths greater than 30 centimeters (12 inches). Therefore, the dose calculation was performed with a contamination zone thickness of 0.3 meters (12 inches) and a contamination zone area of 150 square meters (1600 square feet). The calculation was performed using RESRAD v6.21 in the deterministic mode. The RESRAD Summary Report is provided as Appendix 6-H.

The calculated peak dose from all radionuclides discounted from the site-specific suite of 24 potential radionuclides is 2.23 mrem/year and occurs 84 years after deposition of the contamination. This peak dose occurs from I-129 and 98.2 percent of the calculated dose is due to the drinking water pathway. The peak dose from I-129 occurs over a very short timeframe at which no other radionuclide calculated peak dose occurs.

6.8 DERIVED CONCENTRATION GUIDELINE LEVELS (DCGLs)

6.8.1 Screening DCGLs

The historical site assessment and site characterization have shown that the majority of the site does not contain radioactivity distinguishable from background levels in surface soils, subsurface soils or in subsurface water. This includes survey areas that are classified as Non-Impacted or Impacted Class 2 or 3. Only approximately six acres of the site are classified as Impacted Class 1 and are known or highly suspected to contain localized contamination areas.

The Impacted Class 2 or 3 areas of the site consist of a wide variety of soil types. Both surface and subsurface waters on the site flow northward into Lake Michigan. No Non-Impacted or Impacted Class 2 or 3 areas exist between the Impacted Class 1 areas with known contamination and Lake Michigan.

For the reasons presented above, screening DCGLs published by the NRC may be used to demonstrate compliance with the site release criteria presented in Section 6.2 for Impacted Class 2 or 3 areas [Reference 6-17]. Areas of the site classified as non-impacted, by definition, do not require demonstration of compliance with the site release criteria. These screening DCGL values are presented in Table 6-8 below.

Table 6-8. Screening DCGL Values Applicable to Non-Impacted or Impacted Class 2 or Class 3 Areas

Radionuclide	DCGL (pCi/g)	Radionuclide	DCGL (pCi/g)
H-3	110	Cs-134	5.7
C-14	12	Cs-137	11
Mn-54	15	Eu-152	8.7
Fe-55	10000	Eu-154	8.0
Ni-59	5500	Eu-155	280*
Co-60	3.8	Pu-238	2.5
Ni-63	2100	Pu-239	2.3
Zn-65	11*	Pu-240	2.3*
Sr-90	1.7	Pu-241	7.2
Tc-99	19	Am-241	2.1
Ag-110m	4.9*	Cm-243	3.2
I-129	0.50	Cm-244	4.2*

*These values were obtained from NUREG/CR-5512, Vol. 3, Table 6.91 for P_{crit} = 0.1.

6.8.2 Site-Specific Industrial Area DCGLs

6.8.2.1 Dose Limit Reduction Due to Tritium in Groundwater Aquifers

The hydrogeological assessment confirmed that groundwater occurs in three distinct zones beneath the Industrial Area of the BRP site. These zones include the base of the near surface native sand and gravel layer, the intermediate depth sand layer, and the uppermost fractured portion of the underlying bedrock. In the Southern Industrial Area these groundwater zones appear to be separated by clay-rich aquitard layers. These clay-rich layers thin out and become non-contiguous in the Northern Industrial Area. Tritium (but no other radionuclides) has been identified in each of these groundwater zones in the Northern Industrial Area. The maximum tritium concentrations were 957 pCi/l in the upper groundwater zone, 32,000 pCi/l in the intermediate groundwater zone under the containment sphere and 1560 pCi/l in the bedrock aquifer as measured by the groundwater surveys conducted in 2002.

The near-surface location of the two upper groundwater-bearing zones prevents their use as sources of drinking water. The limited thickness and pumping capacity of these same two units would also prevent their use as sources of non-potable water for irrigation or other purposes. The fractured bedrock zone is considered to be the main potable water aquifer that could be utilized in any future development at the site, as well as a potential source of non-potable water. For these reasons, the upper two groundwater-bearing zones are modeled for the RESRAD calculations as contaminated zones bearing tritium contamination. Although there are also much deeper aquifers in the area, use of the most shallow fractured bedrock aquifer (45-foot depth) is most conservative for use in this evaluation.

The tritium concentrations provided above are the concentrations of tritium contained in the groundwater itself. Therefore, for use in RESRAD, these concentrations must be converted into concentrations of tritium in soil. Following the guidance of "Tritium and Carbon-14 Pathway Models" which appears as Appendix L to ANL/EAD-4, User's Manual for RESRAD Version 6, the concentration of tritium in soil water is expressed as [Reference 6-27]:

$$W_{H-3}^{(cz)} = \frac{\rho_b^{(cz)} \times S_{H-3}}{p_t^{(cz)} \times R_s^{(cz)}}$$

where:

$W_{H\text{-}3}^{(cz)}$ = concentration of H-3 in soil water in the contaminated zone (pCi/cm^3),
$\rho_b^{(cz)}$ = bulk density of the contaminated zone,
$S_{H\text{-}3}$ = concentration of H-3 in contaminated soil pCi/g,
$p_t^{(cz)}$ = total soil porosity, and
$R_s^{(cz)}$ = saturation ratio.

Solving the above equation for the concentration of tritium in the contaminated soil gives:

$$S_{H-3} = \frac{W_{H-3}^{(cz)} \times p_t^{(cz)} \times R_s^{(cz)}}{\rho_b^{(cz)}}$$

Based on the groundwater tritium concentrations and the parameters listed in Appendix 6-F, the parameters required to calculate the concentration of tritium in the contaminated soil of the upper and intermediate groundwater-bearing zones are provided in Table 6-9 below.

Table 6-9. Parameters Required to Calculate the Concentration of Tritium in Contaminated Soil

Parameter	Groundwater Zone	
	Upper	Intermediate
$W_{H\text{-}3}^{(cz)}$	0.957 pCi/cm^3	32.0 pCi/cm^3
$p_t^{(cz)}$	0.274	0.317
$R_s^{(cz)}$	1*	1*
$\rho_b^{(cz)}$	1.95 g/cm^3	1.83 g/cm^3

*Assuming 100% saturation

Based on these parameters, the contaminated soil tritium concentrations resulting from solving the above equation are 0.134 pCi/g in the upper groundwater zone and 5.54 pCi/g in the intermediate groundwater zone.

RESRAD v6.21 was then used with the parameters listed in Appendix 6-F and the above soil tritium concentrations to calculate the peak dose resulting from the contaminated groundwater zones [Reference 6-5]. For the upper groundwater zone, the 1.37 meter thick soil layer above the upper groundwater zone was modeled as a cover layer with the physical properties of the unsaturated zone 1 listed in Appendix 6-F. This mathematical model resulted in a deterministic mode calculated peak dose of 0.0143 mrem/year that occurs at 2.70 years.

Three distinct zones reside above the intermediate groundwater zone. These zones were also modeled as a cover above the contaminated zone with physical properties based on Appendix 6-F. The densities of the zones were weight averaged based on zone thickness to derive an average value for the cover. This mathematical model resulted in a deterministic mode calculated peak dose of 0.421 mrem/year that occurs at 1.81 years.

Calculation of peak dose from tritium contamination of the bedrock aquifer is not as straightforward as the calculation for the upper two groundwater zones because RESRAD v6.21 does not have the ability to model contamination in the saturated zone. However, RESRAD v6.21 will calculate radionuclide concentrations in the saturated zone and use these concentrations to calculate dose from the water dependent pathways. Because of this feature, RESRAD v6.21 was used in the deterministic mode with the parameters listed in Appendix 6-F as the site-specific dose model. An arbitrary tritium concentration of 100 pCi/g was used for the contaminated zone as a source for tritium to enter the water dependent pathways. This calculation showed that, at the time of peak dose (3.05 years), essentially all of the calculated dose resulted from water dependent pathways resulting in 89.9 percent from the water pathway and 10.1 percent from the plant pathway through irrigation.

Since the bedrock aquifer is a potential source of potable water, the dose was calculated by comparison with the EPA's 40 CFR 141.16 drinking water standard. This standard is for a peak concentration of 20,000 pCi/l, which will ensure that the annual dose equivalent to the total body or to any organ will not exceed four mrem/year. Thus the dose from groundwater may be calculated as follows:

$$Dose_{DW} = C_{DW} \times \frac{4\ mrem/y}{20{,}000\ pCi/l}$$

Solving the above equation for the tritium concentration of 1560 pCi/l in the bedrock aquifer yields a calculated annual dose from drinking water of 0.312 mrem/year. Addition of the dose from the plant pathway through irrigation (0.0046 mrem/year) results in a total dose from tritium in this aquifer of 0.317 mrem/year. However, a more conservative approach is to divide 0.312 mrem/year by the RESRAD calculated factor of 89.9 percent of total dose. This yields a total dose of 0.347 mrem/year from all water dependent pathways in the bedrock aquifer.

Assuming that the calculated peak dose from each groundwater zone occurs at the same time, the calculated peak dose from each groundwater zone may be conservatively summed to yield a total dose contribution of 0.782 mrem/year from tritium contamination in each groundwater zone.

6.8.2.2 Derivation of Site-Specific DCGLs

Six radionuclides have been observed in site soils or groundwater. These include H-3, Mn-54, Co-60, Cs-137 and the HTD radionuclides Fe-55 and Sr-90. Therefore, it is appropriate to derive surface soil DCGL values for these radionuclides and demonstrate it is conservative to apply these DCGL values to subsurface soils down to depths of 10.7 meters. In addition to deriving DCGL values for the observed radionuclides, DCGL values were also derived for the three isotopes of europium, Eu-152, Eu-154, and Eu-155 that could contaminate site soils during demolition of neutron activated concrete.

To derive surface soil DCGL values, RESRAD v6.21 was used in the deterministic mode with the parameters listed in Appendix 6-F to calculate DCGL values for the above six radionuclides. A contaminated zone thickness of 1.5 meters was used for the DCGL derivation. This thickness represents the maximum depth of soil contamination observed to date. DCGL values were also derived for contaminated zone thicknesses of 0.15 and 1.0 meters. Comparison of these values with the values obtained for the 1.5 meter thickness indicates the use of a 1.5 meter thickness provides the most conservative (lowest) DCGL values. The level of conservatism is highlighted by the fact soil characterization has not detected concentrations above site-specific DCGLs at soil depths greater than 0.3 meters.

To compensate for tritium subsurface water contamination in the three groundwater zones, the calculated dose of 0.782 mrem/year from tritium contamination was subtracted from the 10 CFR 20.1402 limit of 25 mrem/year for license termination yielding a DCGL limit of 24.218 mrem/year. The RESRAD Summary Report for the 1.5-meter contaminated zone thickness calculation is included as Appendix 6-I.

The above calculation resulted in the surface soil DCGL values listed in Table 6-10 (calculated results were conservatively rounded down to three significant figures):

Table 6-10. Site-Specific Surface Soil DCGL Values

Radionuclide	DCGL Value (pCi/g)	Radionuclide	DCGL Value (pCi/g)
H-3	327	Cs-137	13.2
Mn-54	13.7	Eu-152	7.36
Fe-55	3.58E+05	Eu-154	6.79
Co-60	3.21	Eu-155	287
Sr-90	2.48		

6.8.2.3 Subsurface Soil Contamination

The site characterization process identified several discrete locations of small area and volume pockets of subsurface soil contamination. For purposes of this LTP, subsurface soil contamination is defined as any soil contamination found at depths of greater than 0.15 meters (six inches). The site characterization process did not identify subsurface soil contamination at depths greater than 1.5 meters (five feet); however, a potential exists for subsurface soil contamination down to a level of approximately 10.7 meters (35 feet) at the base of the containment sphere structure. It should be noted corings through the concrete in the basement of the containment sphere structure did not identify any contaminates in soil under the containment sphere structure other than tritium in the soil water.

Because subsurface soil contamination has been found only in discrete pockets of small area and volume, it is not appropriate to assume subsurface soil contamination exists over the entire area of assumed surface soil contamination. Therefore, it is not appropriate to apply the same dose limit correction process for subsurface soil that was applied for subsurface water contamination in the three groundwater zones. Instead, it has been demonstrated it is conservative to apply surface soil DCGL values to subsurface soils down to depths of 10.7 meters (35 feet).

The surface soil DCGL values were used as input soil concentration values for RESRAD v6.21 peak total dose calculations for each radionuclide. Calculations were performed for the contaminated zone located at five different depths. These depths included the surface 15 cm (six inches) to represent the definition of surface soil contamination, 0.15 meter (six inches) below the surface to represent shallow subsurface contamination, 1.5 meters (five feet) below the surface, 2.74 meters (nine feet) below the surface and 10.7 meters (35 feet) below the surface. Each calculation was based on a 15 cm (six inches) thick contaminated zone. The 1.5 meters (five feet) below the surface location represented the lowest depth at which soil contamination was observed during the site characterization process. The 2.74 meters (nine feet) below the surface location represented the base of the upper two soil classifications in the site mathematical dose model. The 10.7 meters (35 feet) below the surface location represented the base of the containment sphere structure.

In each calculation, the integrity of the site mathematical dose model was maintained. Only the thickness and number of the unsaturated zones were varied to account for the depth of the contaminated zone. Physical parameters of the contaminated zone were varied to match those of the appropriate unsaturated zone in the site mathematical dose model and the cover density used was a weighted average based on zone thickness to derive an average density for the cover.

These calculations demonstrated the peak total dose for tritium is directly proportional to depth from the surface and peak total dose for the remaining radionuclides is inversely proportional to depth from the surface. This means that application of the surface soil DCGL value for tritium to subsurface soil tritium contamination is not conservative; however, the annual dose limit of 24.218 mrem is never exceeded by applying the surface soil DCGL value to subsurface soil contamination down to depths of 10.7 meters (35 feet).

The above evaluation was performed assuming the entire site mathematical dose model contaminated zone area of 8,094 square meters (two acres) and a contaminated zone thickness of 0.15 meters (six inches). Since subsurface soil contamination has only been observed to occur in discrete pockets, the calculations were repeated assuming a contaminated zone area of 100 square meters (1,080 square feet) and a subsurface contaminated zone thickness of 1.5 meters (five feet) to further validate the above observations. With the exception of the contaminated zone area, the integrity of the site mathematical dose model was again maintained.

Based on the results of these calculations provided in Table 6-11, the application of surface soil DCGLs to discrete pockets of subsurface contamination down to 10.7 meters (35 feet) below the surface is acceptable and conservative.

Table 6-11. 100 Square Meter Contaminated Zone – 1.5 Meters Thick

Radionuclide	Individual Radionuclide Dose Contribution at Time of Radionuclide Peak Total Dose with Contamination at DCGL Concentrations (mrem/y)				
	Surface*	0.15 m Subsurface	1.5 m Subsurface	2.7 m Subsurface	10.7 m Subsurface
H-3	0.32	1.75	2.26	2.24	2.42
Mn-54	18.1	2.35	0.00	0.00	0.00
Fe-55	0.47	1.96	0.00	0.00	0.00
Co-60	17.5	3.05	0.00	0.00	0.00
Sr-90	0.32	2.12	0.00	0.00	0.00
Cs-137	16.9	2.02	0.00	0.00	0.00
Eu-152	18.6	2.62	0.00	0.00	0.00
Eu-154	18.3	2.70	0.00	0.00	0.00
Eu-155	20.8	0.15	0.00	0.00	0.00

*The calculation for surface soil contamination was performed with a contamination zone thickness of 0.15 meters.

The above evaluations demonstrate it is conservative to apply surface soil DCGL values to subsurface contamination. With this application, if subsurface soil contamination is excavated in the future and spread on the surface, surface soil DCGL values will not be exceeded.

6.9 SENSITIVITY ANALYSIS OF THE BIG ROCK POINT DOSE MODEL

An evaluation was performed to determine sensitivity of the parameters used in the BRP RESRAD v6.21 dose model and to rank sensitive parameters in the order of their sensitivity to the calculated peak mean dose [Reference 6-7].

6.9.1 Sensitivity Analysis with Uncorrelated Parameters

The site-specific RESRAD v6.21 dose model used for the sensitivity analysis was based on the parameters listed in Appendix 6-F. The dose model was loaded with the statistical parameter distributions provided in Appendix 6-J and run in the probabilistic mode. The uncertainty analysis input settings for this calculation were:

- Latin Hypercube sampling
- Random seed – 1000
- Number of observations – 300
- Number of repetitions – 1
- Grouping of observations – correlated or uncorrelated

Calculations were performed for the entire site-specific suite of nine radionuclides (including the three isotopes of europium) with an input concentration of 1 pCi/g for each radionuclide and individually for each of the six radionuclides that have been observed in site soils. Results of these calculations are presented in Appendix 6-K (the ten most sensitive parameters for the suite of radionuclides and the five most sensitive parameters for individual radionuclides).

6.9.2 Sensitivity Analysis with Correlated Parameters

A few input parameters are clearly related, such as effective porosity and total porosity. NUREG/CR-6697, Attachment C identified potential correlations among RESRAD parameters assigned statistical distributions. These correlations for which RESRAD v6.21 allows correlation are provided in Table 6-12 below.

Table 6-12. Potential Parameter Correlations

Parameter	Correlated With	Positive/Negative Correlation
Distribution coefficients	Soil/plant transfer factors	Negative
Effective porosity	Total porosity	Strong positive
Erosion rate	Wind speed	Positive
Erosion rate	Runoff coefficient	Positive
Runoff coefficient	Erosion rate	Positive
Soil density	Total porosity	Negative
Soil/plant transfer factors	Distribution coefficients	Negative
Total porosity	Soil density	Negative
Total porosity	Effective porosity	Strong positive
Wind speed	Erosion rate	Positive

The calculations with uncorrelated parameters were repeated with the addition of the potential parameter correlations contained in Table 6-12 as additional parameter inputs for the uncertainty analysis. In the correlated/uncorrelated grouping, the RESRAD v6.21 user must specify the degree of correlation between each correlated parameter by inputting the correlation coefficient between the ranks of the parameters. If the correlation identified in Table 6-12 was indicated to be positive, a rank correlation coefficient of 0.5 was used. If the correlation is indicated to be negative, a rank correlation coefficient of -0.5 was used. If the correlation is indicated to be strong positive, a rank correlation coefficient of 0.9 was used.

Results of these calculations are presented in Appendix 6-K (the ten most sensitive parameters for the suite of radionuclides and the five most sensitive parameters for individual radionuclides).

6.10 DERIVATION OF AREA FACTORS

NUREG-1727, Appendix C, Section 4.3.3.5 states that the extent of residual radioactivity can be taken into account when modifying the default (baseline) scenarios. The baseline scenarios assume large areas of homogeneous surface contamination. If the area of residual radioactivity is smaller than the baseline, the licensee may propose modifying the exposure pathways to account for the effect on the critical group's activities. Two methods can be followed: (1) the licensee can reduce the calculated dose by the fraction of the default area or modify usage parameters accordingly; or (2) modify the exposure scenario and pathways to account for the size of the residual radioactivity. When the extent of residual radioactivity becomes smaller, some of the activities are no longer viable as reasonable assumptions for exposure.

The meat and milk pathways have been suppressed in the BRP site-specific dose model because it is highly unlikely that a farm of any size would ever be established on the shoreline of Lake Michigan at the site location. Therefore, area factors have been determined based on variation in the size of the residual radioactivity.

Within RESRAD v6.21, the internal area factor[2] (FA3) used to calculate dose from the plant ingestion pathway is automatically calculated to adjust for size of the contaminated zone when the contaminated fraction is set to the default entry of -1 in accordance with the equation below:

$$
\begin{aligned}
FA_3 &= A/2{,}000 \text{ when } 0 \leq A \leq 1{,}000\ m^2 \\
&= 0.5 \text{ when } A > 1{,}000\ m^2
\end{aligned}
$$

An area factor for use in elevated measurement comparison during final status surveys is defined by the equation:

$$Area\ Factor = \frac{DCGL_{EMC}}{DCGL_W}$$

where: $DCGL_W$ = Baseline DCGL value and
$DCGL_{EMC}$ = Elevated measurement comparison DCGL value

[2] Not to be confused with the area factors calculated for final status surveys.

NUREG-1505 provides the methodology for calculating area factors in Chapter 8. Chapter 8 states that the area factors should be calculated using dose pathway models and assumptions that are consistent with those used to calculate the $DCGL_W$. Area factors are computed by taking the ratio of the dose per unit concentration calculated by RESRAD for the baseline area to that calculated for various smaller areas.

RESRAD v6.21 was used in the deterministic mode to calculate dose per unit concentration values using the site-specific input parameters contained in Appendix 6-F. A contaminated zone thickness of 1.5 meters was used for the calculations to match the contaminated zone thickness used to derive DCGL values. The contaminated zone area was varied from the baseline area of 8,094 m^2 down to 1 m^2. Each time a new contaminated zone area was used, a corresponding value for length of contaminated zone parallel to the aquifer flow was calculated.

The dose per unit concentration value is termed the dose to source ratio (DSR) by RESRAD (units of mrem/year per pCi/g). The DSRs calculated by RESRAD were used to calculate area factors in accordance with the following equation:

$$AF_i = \frac{DSR_{Baseline}}{DSR_i}$$

where:

Af_i = Area Factor at EMC area i
$DSR_{Baseline}$ = *DSR at the baseline area of 8,094 m^2*
DSR_i = *DSR for EMC area i*

The results of the area factor calculations (rounded to three significant figures) are provided as Appendix 6-L. Area factors for the gamma emitting radionuclides are shown as a graphical plot in Figure 6-2 and the area factors for HTD radionuclides are shown as a graphical plot in Figure 6-3.

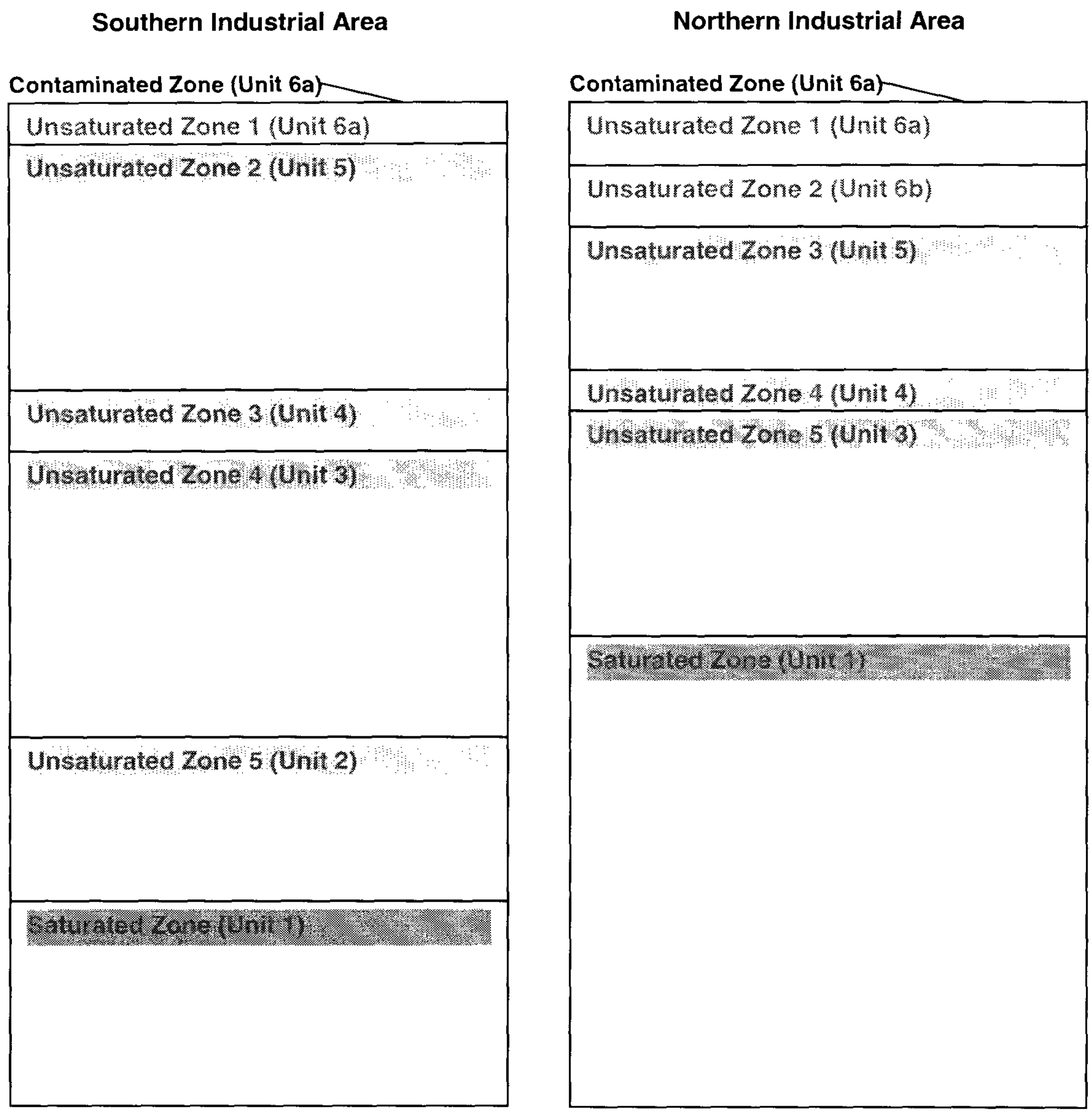

Figure 6-1. RESRAD Mathematical Dose Models

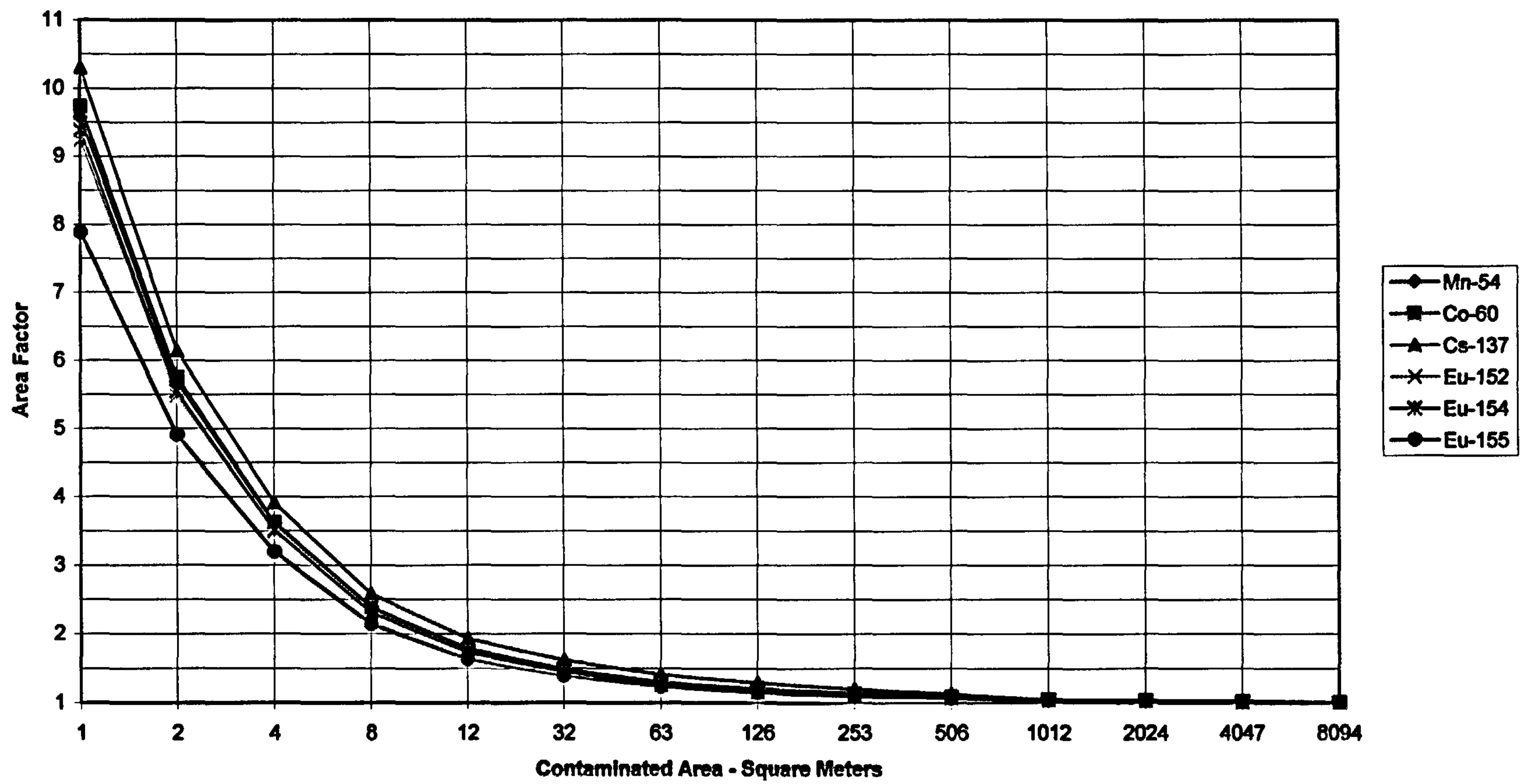

Figure 6-2. Area Factors for Gamma Emitting Radionuclides

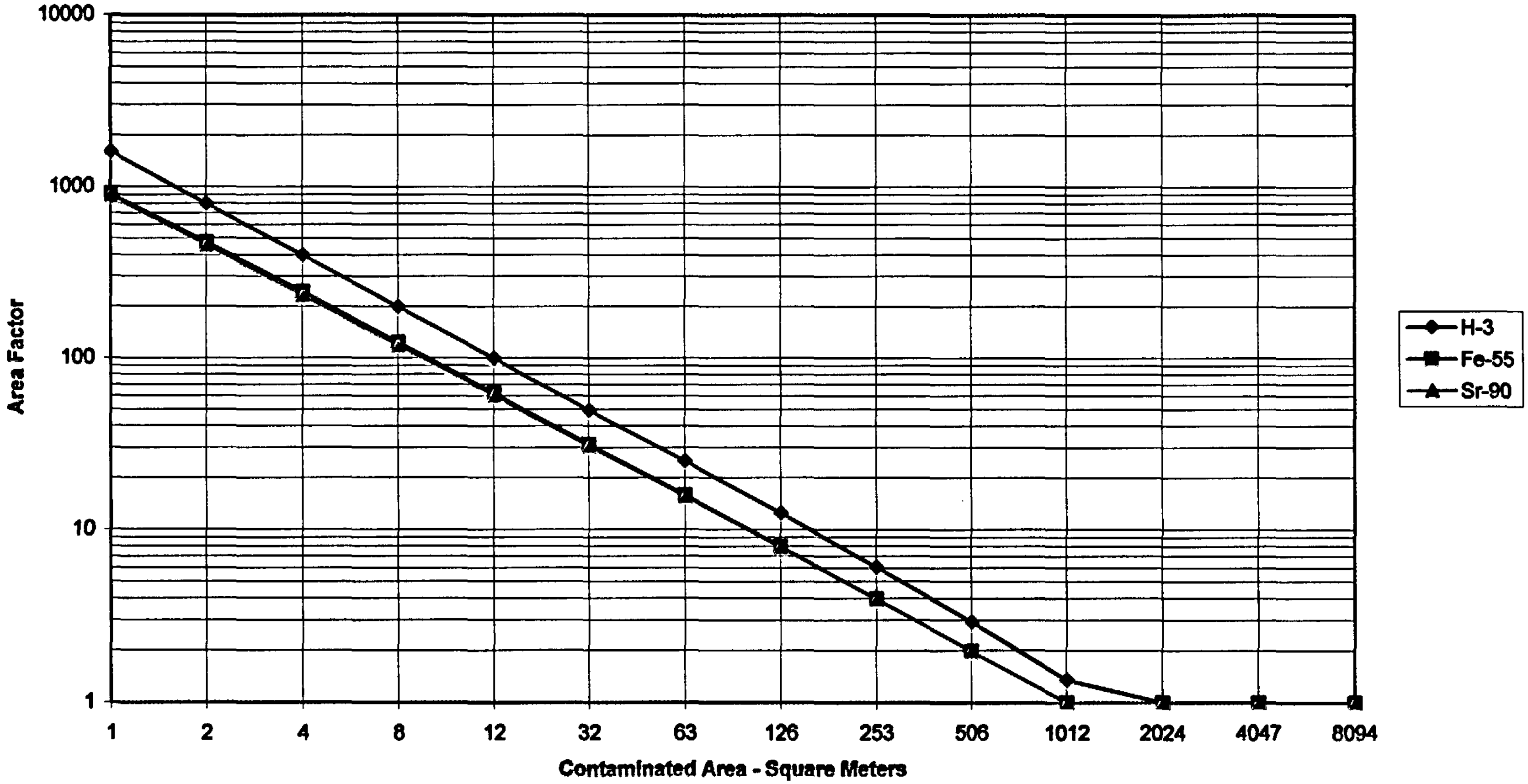

Figure 6-3. Area Factors for Hard-to-Detect Radionuclides

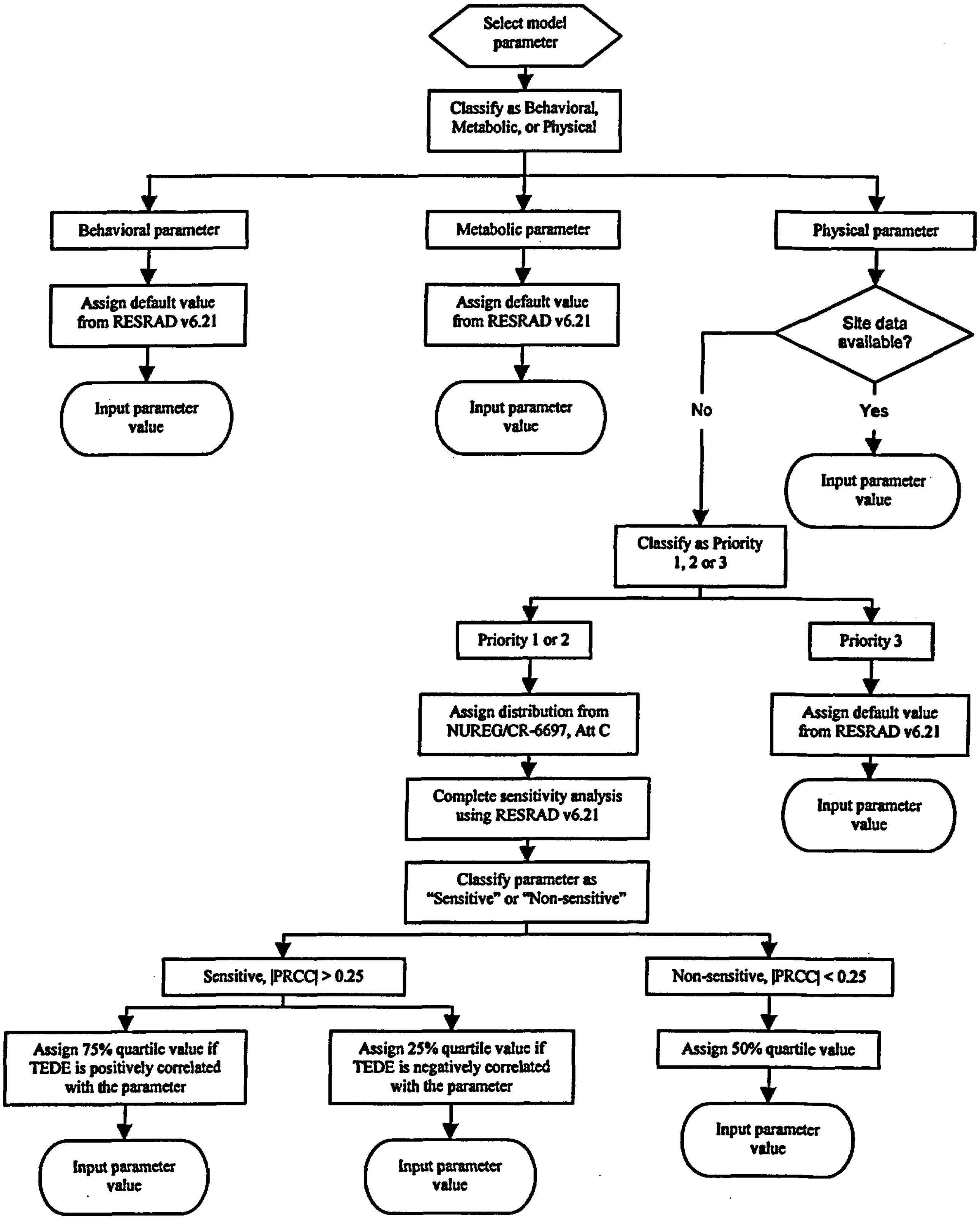

Figure 6-4. Parameter Selection Process

6.11 REFERENCES

6-1 Big Rock Point Engineering Analysis EA-BRP-LEB-02-01, Identification of the Suite of Radionuclides for Consideration During Site Characterization

6-2 Big Rock Point Engineering Analysis EA-BRP-SC-02-02, Derivation of K_d Values for use at Big Rock Point

6-3 Big Rock Point Engineering Analysis EA-BRP-SC-02-03, RESRAD Parameter Selection for use in Big Rock Point Dose Modeling

6-4 Big Rock Point Engineering Analysis EA-BRP-SC-02-04, Radionuclides Present in Onsite Soil and Water

6-5 Big Rock Point Engineering Analysis EA-BRP-SC-02-05, Effect of Subsurface Soil and Water Contamination and the Development of Surface Soil DCGLs

6-6 Big Rock Point Engineering Analysis EA-BRP-SC-03-01, Evaluation of Potential Dose from Discounted Radionuclides

6-7 Big Rock Point Engineering Analysis EA-BRP-SC-03-02, Sensitivity Analysis of the Big Rock Point RESRAD v6.21 Dose Model Parameters

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6-24 U.S. Nuclear Regulatory Commission NUREG/CR-5512, Volume 3, Residual Radioactive Contamination from Decommissioning Parameter Analysis, October 1999

6-25 U.S. Nuclear Regulatory Commission NUREG/CR-6692, Probabilistic Modules for the RESRAD and RESRAD-BUILD Computer Codes, November 2000

6-26 U.S. Nuclear Regulatory Commission NUREG/CR-6697, Development of Probabilistic RESRAD 6.0 and RESRAD-BUILD 3.0 Computer Codes, November 2002

6-27 Yu, C., et al., ANL/EAD-4, User's Manual for RESRAD Version 6, July 2001

Parameter	Classification	Description
Contamination		
Thickness of contaminated zone	Physical	The distance (m) between the uppermost and lowermost soil samples that has radionuclide concentration clearly above background.
Area of contaminated zone	Physical	Area (m^2) that contains location of soil exceeding background radionuclide concentrations.
Shape of the contaminated zone	Physical	The code has the capability to handle any shape of contaminated zone. If the shape factor flag has been set, the 12 annular area fields comprising shape factor data are input. The shape factor data are calculated by RESRAD by drawing 2 to 12 concentric circles emanating from the receptor location inside (or possibly outside) the contaminated area. The outermost circle circumscribes the entire contaminated zone. For each annular ring, the outer radius and fraction of the ring within the contaminated zone should be entered. For simple shapes (square, rectangle, triangle, doughnut), two circles are sufficient. For complicated shapes, all 12 concentric circles can be used.
Initial concentration of principal radionuclides in soil	Physical	Average radionuclide concentration (pCi/g) in the contaminated zone. The contaminated zone is treated as a uniformly contaminated volume with the same radionuclide concentration.
Initial concentration of radionuclides present in groundwater	Physical	The measured groundwater concentration (pCi/l) of principal radionuclides. The groundwater concentration should be measured at the same time the soil concentrations are measured. Groundwater concentration can be input only if time since placement of material is > 0.

Parameter	Classification	Description
Leach rate	Physical	The fraction of the available radionuclide leached from the contaminated zone per unit of time (1/yr). Radionuclide leach rates should be used if known. The nonzero leach rate is used to calculate the leaching of the radionuclide from the contaminated zone and to derive the water/soil concentration ratio and distribution coefficient. If the derived distribution coefficient is greater than zero, the input distribution coefficient would be replaced by these values.
Solubility limit	Physical	The solubility limit (mol/L) provides an additional option for calculating the distribution coefficient in the code. The solubility limit serves as an upper bound on the amount of a radionuclide released from the soil particles.
Time since placement of material	Physical	The duration (yr) between the placement of radioactive material on-site and the performance of a radiological survey. A non-zero value for this parameter is necessary to activate the groundwater concentration input box. The non-zero value of this parameter is used along with groundwater concentration to calculate the water/soil concentration ratio and effective distribution coefficient.
Times for calculation	Physical	The times in years following the radiological survey for which tabular values for single-radionuclide soil guidelines and mixture sums will be obtained. The code calculates dose at time zero and up to nine user specified times.
Contaminated zone density	Physical	Bulk density of the contaminated zone (g/cm^3).
Contaminated zone distribution coefficient	Physical	Site-specific values should be used everywhere for each radionuclide. Default values are provided by the code for most radionuclides. However, these values should be used with care because distribution coefficients can vary over many orders of magnitude.
Use plant/soil ratio	Not Applicable	The code allows distribution coefficients to be calculated from the plant root uptake factors (plant/soil concentration ratio). This option becomes active if the plant/soil ratio box is checked.

Parameter	Classification	Description
Contaminated zone field capacity	Physical	Volumetric moisture content of soil at which (free) gravity drainage ceases. This is the amount of moisture that will be retained in a column of soil against the force of gravity. The field capacity is one of several hydrogeological parameters used to calculate water transport through the unsaturated part of the soil. The user can use this input to specify a minimum moisture content for each partially saturated region. It is also called specific retention, irreducible water content, or residual water content.
Contaminated zone erosion rate	Physical, Behavioral	Average volume of soil material removed from one place to another by water, wind, or moving ice per unit of ground surface area per unit time.
Contaminated zone total porosity	Physical	Ratio of the pore volume to the total volume of the contaminated zone.
Contaminated zone hydraulic conductivity	Physical	The measure of the soil's ability to transmit water (m/year) when submitted to a hydraulic gradient. The hydraulic conductivity depends on the soil grain size, the structure of the soil matrix, the type of soil fluid, and the relative amount of soil fluid (saturation) present in the soil matrix.
Contaminated zone b parameter	Physical	An empirical and dimensionless parameter that is used to evaluate the saturation ratio (or the volumetric water saturation) of the soil according to a soil characteristic function called the conductivity function.
Carbon-Model Parameters		
Thickness of evasion layer of C-14 in soil	Physical	The maximum soil thickness layer through which C-14 can escape to the air by conversion to CO_2. C-14 below this depth is assumed to be trapped.
C-14 evasion flux rate from soil	Physical	The fraction of the soil inventory of C-14 that is lost to the atmosphere per unit time.
C-12 concentration in local water	Physical	The stable carbon concentration in g/cm^3 of C-12 in water.
C-12 concentration in contaminated soil	Physical	The stable carbon concentration in g/cm^3 of C-12 in the contaminated soil
Fraction of vegetation carbon absorbed from soil	Physical	The fraction of total vegetation carbon obtained by direct root uptake from the soil.

Parameter	Classification	Description
Fraction of vegetation carbon absorbed from air	Physical	The fraction of total vegetation carbon assimilated from the atmosphere through photosynthesis.
C-12 evasion flux rate from soil	Physical	The fraction of C-12 in soil that escapes to the atmosphere per unit time.
Grain fraction in beef cattle feed	Behavioral	The fraction of grain (non-leafy) vegetation in the livestock diet (0.8). The balance is assumed to be leafy vegetation: hay or fodder.
Grain fraction in milk cow feed	Behavioral	The fraction of grain (non-leafy) vegetation in the livestock diet (0.2). The balance is assumed to be leafy vegetation: hay or fodder.
Soil		
Cover depth	Physical	Distance (m) from the ground surface to the contaminated zone.
Density of cover material	Physical	Bulk density of the cover material (g/cm^3).
Cover total porosity	Physical	The ratio of the void space volume to the total volume of the porous medium.
Cover volumetric water content	Physical	The fraction of the total volume of the porous medium that is occupied by water.
Cover radon diffusion coefficient	Physical	The effective (or interstitial) radon diffusion coefficient is the ratio of the diffusive flux density of radon activity across the pore area to the gradient of the radon activity concentration in the pore space (m^2/s). Entering –1 for any diffusion coefficient will cause the code to calculate the diffusion coefficient based on the porosity and water content of the medium.
Cover erosion rate	Physical, Behavioral	The average volume of cover material that is removed per unit of ground surface area and per unit of time. Erosion rates can be estimated by means of the universal soil loss equation.
Number of unsaturated zones	Physical	Number of unsaturated zones. An unsaturated zone is defined as a horizontal uncontaminated layer located between the contaminated zone and the aquifer. The code allows a maximum of five unsaturated zones.

Parameter	Classification	Description
Unsaturated zone thickness	Physical	The thickness of the uncontaminated unsaturated zone that lies below the bottom of the contaminated zone and above the groundwater table. The code has provisions for up to five different horizontal strata within this zone. Each stratum is characterized by six radionuclide independent parameters: thickness, density, total porosity, effective porosity, b parameter, and hydraulic conductivity.
Unsaturated zone density	Physical	Bulk density of the unsaturated zone soil (g/cm^3).
Unsaturated zone distribution coefficient	Physical	Site-specific values should be used everywhere for each radionuclide. Default values are provided by the code for most radionuclides. However, these values should be used with care because distribution coefficients can vary over many orders of magnitude.
Unsaturated zone total porosity	Physical	Ratio of the pore volume to the total volume of the unsaturated zone.
Unsaturated zone effective porosity	Physical	The effective porosity is the ratio of the pore volume where water can circulate to the total volume. It is used along with other hydrological parameters to calculate the water transport breakthrough times.
Unsaturated zone field capacity	Physical	Volumetric moisture content of soil at which (free) gravity drainage ceases. This is the amount of moisture that will be retained in a column of soil against the force of gravity. The field capacity is one of several hydrogeological parameters used to calculate water transport through the unsaturated part of the soil. The user can use this input to specify a minimum moisture content for each partially saturated region. It is also called specific retention, irreducible water content, or residual water content.
Unsaturated zone hydraulic conductivity	Physical	The measure of the soil's ability to transmit water when submitted to a hydraulic gradient. The hydraulic conductivity depends on the soil grain size, the structure of the soil matrix, the type of soil fluid, and the relative amount of soil fluid (saturation) present in the soil matrix.

Parameter	Classification	Description
Unsaturated zone soil-specific b parameter	Physical	An empirical and dimensionless parameter that is used to evaluate the saturation ratio (or the volumetric water saturation) of the soil according to a soil characteristic function called the conductivity function.
Water		
Density of saturated zone	Physical	Bulk density of the saturated zone (g/cm^3).
Saturated zone distribution coefficient	Physical	Site-specific values should be used everywhere for each radionuclide. Default values are provided by the code for most radionuclides. However, these values should be used with care because distribution coefficients can vary over many orders of magnitude.
Saturated zone total porosity	Physical	Ratio of the pore volume to the total volume of the saturated zone.
Saturated zone effective porosity	Physical	The effective porosity is the ratio of the pore volume where water can circulate to the total volume. It is used along with other hydrological parameters to calculate the water transport breakthrough times.
Saturated zone field capacity	Physical	Volumetric moisture content of soil at which (free) gravity drainage ceases. This is the amount of moisture that will be retained in a column of soil against the force of gravity. The field capacity is one of several hydrogeological parameters used to calculate water transport through the saturated part of the soil. It is also called specific retention, irreducible water content, or residual water content. (The field capacity and b parameter of the saturated zone are used only if the water table drop rate is positive.)
Saturated zone hydraulic conductivity	Physical	The measure of the soil's ability to transmit water when submitted to a hydraulic gradient. The hydraulic conductivity depends on the soil grain size, the structure of the soil matrix, the type of soil fluid, and the relative amount of soil fluid (saturation) present in the soil matrix.

Parameter	Classification	Description
Saturated zone hydraulic gradient	Physical	The change in hydraulic head per unit of distance in the groundwater flow direction. In an unconfined (water table) aquifer, the horizontal hydraulic gradient of groundwater flow is approximately the slope of the water table. In a confined aquifer, it represents the difference in potentiometric surfaces over a unit distance.
Saturated zone b parameter	Physical	An empirical and dimensionless parameter that is used to evaluate the saturation ratio (or the volumetric water saturation) of the soil according to a soil characteristic function called the conductivity function.
Length of contaminated zone parallel to the aquifer flow	Physical	The distance (m) between two parallel lines perpendicular to the direction of aquifer flow, one at the upgradient edge of the contaminated zone and the other at the downgradient edge.
Water table drop rate	Physical	The rate at which the depth of the water table is lowered. If the water table drop rate is greater than zero, the unsaturated zone thickness will be created or increased. The saturation of this newly created unsaturated zone is estimated by the hydrological parameters of the saturated zone. The code does not allow a negative water table drop rate.
Well-pump intake depth (below water table)	Physical	The screened depth of a well within the aquifer (the saturated zone).
Well pumping rate	Behavioral, Physical	The volume of water removed from the groundwater aquifer annually for all domestic purposes in m^3/y.

Parameter	Classification	Description
Model: non-dispersion or mass balance	Not Applicable	Two models are used in the code for calculating the water/soil concentration ratio for the groundwater pathway: a mass-balance (MB) model and a nondispersion (ND) model. The MB model assumes that a well is located at the center of the contaminated zone, and the ND model assumes that the well is located at the downgradient edge of the contaminated zone. In the MB model, it is assumed that all of the radionuclides released from the contaminated zone are withdrawn through the well. In the ND model, it is assumed that the saturated zone is a single homogenous stratum, and the water withdrawn introduces only a minor perturbation in the water flow.
Evapotranspiration coefficient	Physical	The ratio of the total volume of water leaving the ground via evapotranspiration to the total volume of water available within the root zone of the soil during a fixed period of time.
Humidity in air	Physical	Average absolute humidity in air. It is used in the tritium model to calculate tritium concentration in air.
Average annual wind speed	Physical	The overall average of the wind speed, measured near the ground.
Precipitation rate	Physical	Average volume of water in the form of rain, snow, hail or sleet that falls per unit area and per unit time at the site (m/yr)
Irrigation mode	Behavioral	Method of irrigation (overhead or ditch).
Irrigation rate	Behavioral	The average volume of water that is added to the soil at the site, per unit of surface area and per unit of time. Irrigation is the practice of supplying water artificially to the soil in order to permit agricultural use of the land in an arid region or to compensate for occasional droughts in semidry or semi humid regions. It is the average annual irrigation rate.
Runoff coefficient	Physical	Fraction of annual precipitation that does not infiltrate into the soil and is not transferred back to the atmosphere through evapotranspiration.
Watershed area for nearby stream or pond	Physical	The site-specific area that drains into the nearby pond.

Parameter	Classification	Description
Accuracy for water soil computation	Not Applicable	The fractional accuracy desired (convergence criterion) in the Romberg integration used to obtain water/soil concentration ratios.
Ingestion		
Fruit, vegetable, and grain consumption rate	Metabolic, Behavioral	The dietary factor for fruit, vegetable, and grain consumption by humans. The default is based on national averages.
Leafy vegetable consumption	Metabolic, Behavioral	The dietary factor for leafy vegetable consumption by humans. The default is based on national averages.
Milk consumption	Metabolic, Behavioral	The dietary factor for milk consumption by humans. The default is based on national averages.
Meat and poultry consumption	Metabolic, Behavioral	The dietary factor for meat and poultry consumption by humans. The default is based on national averages.
Fish consumption rate	Metabolic, Behavioral	The dietary factor for fish consumption by humans. The default is based on national averages.
Other seafood consumption rate	Metabolic, Behavioral	The dietary factor for other seafood consumption by humans. The default is based on national averages.
Aquatic food contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake for the fish pathway. The remaining balance is from off-site sources, which are assumed to be uncontaminated. The default value of 0.5 indicates that 50% of aquatic food is being obtained from on-site sources. Setting the value to 0 will turn off the fish pathway entirely.
Soil ingestion rate	Metabolic, Behavioral	The average annual quantity of soil ingested for the soil ingestion pathway.
Drinking water intake	Metabolic, Behavioral	The drinking water ingestion rate.
Storage time for fruits, non-leafy vegetables, and grain	Behavioral	The storage times are used to calculate radioactive ingrowth and decay adjustment factors for food and feed due to storage. The code has values for fruits, non-leafy vegetables, and grain (one category), leafy vegetables, milk, well and surface water, livestock fodder, meat, fish and crustacea; and mollusks.
Storage time for leafy vegetables	Behavioral	See above.
Storage time for milk	Behavioral	See above.
Storage time for meat	Behavioral	See above.
Storage time for fish	Behavioral	See above.

Parameter	Classification	Description
Storage time for crustacea and mollusks	Behavioral	See above.
Storage time for well water	Behavioral	See above.
Storage time for surface water	Behavioral	See above.
Storage time for livestock fodder	Behavioral	For livestock fodder the storage time is an annual average. The default value is obtained by assuming 6 months of outside gazing and 6 months of silage fodder with an average silo time of 3 months.
Drinking water contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake for the drinking water pathway. The remaining balance (if value <1) of the drinking water is from off-site sources, which are assumed to be uncontaminated. Setting the value to zero will turn off the drinking water pathway entirely.
Household water contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated household water for use in calculating radon exposure. The remaining balance (if value <1) of the household water is from off-site sources, which are assumed to be uncontaminated. The default value of 1 indicates that all household water is from an on-site source.
Livestock water contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake of livestock water for the meat and milk pathway. The remaining balance (if value <1) of the livestock water is from off-site sources, which are assumed to be uncontaminated. The default value of 1 indicates that all livestock water is from an on-site source.
Irrigation water contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake of irrigation water for the plant, meat and milk pathways. The remaining balance (if value <1) of the irrigation water is from off-site sources, which are assumed to be uncontaminated. The default value of 1 indicates that all irrigation water is from an on-site source.

Parameter	Classification	Description
Plant food contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake for the plant pathway. The appropriate values range from 0 to 1, although a negative value can be input. The remaining balance is from off-site sources, which are assumed to be uncontaminated. The default value of -1 specifies that the contaminated fraction of plant food will be calculated from the appropriate area factor in the code. Setting the value to 0 will turn off the plant pathway entirely.
Meat contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake for the meat pathway. The appropriate values range from 0 to 1, although a negative value can be input. The remaining balance is from off-site sources, which are assumed to be uncontaminated. The default value of -1 specifies that the contaminated fraction of meat will be calculated from the appropriate area factor in the code. Setting the value to 0 will turn off the meat pathway entirely.
Milk contaminated fraction	Behavioral, Physical	Allows specification of the fraction of contaminated intake for the milk pathway. The appropriate values range from 0 to 1, although a negative value can be input. The remaining balance is from off-site sources, which are assumed to be uncontaminated. The default value of -1 specifies that the contaminated fraction of milk will be calculated from the appropriate area factor in the code. Setting the value to 0 will turn off the milk pathway entirely.
Livestock fodder intake rate for meat	Metabolic	The daily intake of fodder by livestock kept for meat consumption. The code uses the area factor to calculate the contaminated intake.
Livestock fodder intake rate for milk	Metabolic	The daily intake of fodder by livestock kept for milk consumption. The code uses the area factor to calculate the contaminated intake.
Livestock water intake rate for meat	Metabolic	The daily intake of water by livestock kept for meat consumption. The code uses the area factor to calculate the contaminated intake.

Parameter	Classification	Description
Livestock water intake rate for milk	Metabolic	The daily intake of water by livestock kept for milk consumption. The code uses the area factor to calculate the contaminated intake.
Livestock intake of soil	Metabolic	Daily intake of soil by livestock kept for meat or milk consumption.
Mass loading for foliar deposition	Physical	The average mass loading of airborne contaminated soil particles in a garden during the growing season.
Depth of soil mixing layer	Physical	Used in calculating the depth factor for dust inhalation and soil ingestion pathways and for foliar deposition for the plant, meat, and milk ingestion pathways. The depth factor is the fraction of the resuspendable soil particles at the ground surface that are contaminated. It is calculated by assuming that mixing of soil will occur in the soil mixing layer.
Depth of roots	Physical	The maximum root depth below the ground surface.
Groundwater fractional usage for drinking water	Behavioral, Physical	The four groundwater fractional usage parameters (drinking water, household water, livestock water, and irrigation water) are included primarily for all groundwater (well or spring) and surface water (pond or river) scenarios. Hence, the fractions will usually be set at 1 or 0. A value of 1 specifies 100% groundwater usage and 0 selects 100% surface water usage.
Groundwater fractional usage for household water	Behavioral, Physical	See above.
Groundwater fractional usage for livestock water	Behavioral, Physical	See above.
Groundwater fractional usage for irrigation water	Behavioral, Physical	See above.
Wet weight crop yield for non-leafy plants	Physical	The weight of the edible portion of plant food produced per unit land area for different food classes. The code has wet weight crop yield for non-leafy, leafy, and fodder. Non-leafy and leafy vegetables are for human consumption; fodder is for animal consumption.
Wet weight crop yield for leafy plants	Physical	See above.
Wet weight crop yield for fodder	Physical	See above.

Parameter	Classification	Description
Length of growing season for non-leafy vegetables	Physical	The exposure time to contamination for the plant food during the growing season. The contamination can reach the edible portion of the plant food through foliar deposition, root uptake, and water irrigation. The code has length of growing season for non-leafy vegetables, leafy vegetables, and fodder.
Length of growing season for leafy vegetables	Physical	See above.
Length of growing season for fodder	Physical	See above.
Translocation factor for non-leafy vegetables	Physical	The fraction of the contamination that is retained on the foliage that is transferred to the edible portion of the plant. The code has three food categories, non-leafy (includes non-leafy vegetables, fruit, and grain) and leafy vegetables for human and fodder for animal consumption (in all three values).
Translocation factor for leafy vegetables	Physical	See above.
Translocation factor for fodder	Physical	See above.
Weathering removal constant	Physical	The weathering process would remove contaminants from foliage of the plant food. The process is characterized by a removal constant and reduces the amount of contaminants on foliage exponentially during the exposure period.
Dry foliar interception fraction for non-leafy vegetables	Physical	The fraction of deposited radionuclides that is retained on the foliage of the plant food. Both the dry deposition (from airborne particulates) and the wet deposition processes (from irrigation) are considered. The code has wet as well as dry foliar interception fraction for non-leafy, leafy (for human consumption), and fodder (for animal consumption).
Dry foliar interception fraction for leafy vegetables	Physical	See above.
Dry foliar interception fraction for fodder	Physical	See above.
Wet foliar interception fraction for non-leafy vegetables	Physical	See above.
Wet foliar interception fraction for leafy vegetables	Physical	See above.

Parameter	Classification	Description
Wet foliar interception fraction for fodder	Physical	See above.
Slope factor - external	Metabolic	The ratio of cancer risk per year to the radionuclide concentration in the soil. The slope factors are based on the EPA methodology of calculating cancer risk.
Slope factor - inhalation	Metabolic	The ratio of cancer risk to the radionuclide activity inhaled.
Slope factor - ingestion	Metabolic	The ratio of cancer risk to the radionuclide activity ingested.
Plant transfer factor	Physical	The ratio of radionuclide concentration in edible portions of the plant at harvest time to the dry soil radionuclide concentration. It is assumed that the same root uptake transfer factors can be used for leafy and non-leafy vegetables. The code has element-specific values.
Meat transfer factor	Physical	The ratio of radionuclide concentration in beef to the daily intake of the same radionuclide in livestock feed or water (pCi/kg)/(pCi/d). The code has element-specific values for meat.
Milk transfer factor	Physical	The ratio of radionuclide concentration in milk to the daily intake of the same radionuclide in livestock feed or water (pCi/kg)/(pCi/d). The code has element-specific values for milk.
Bioaccumulation factor for fish	Physical	The ratio of radionuclide concentration in the aquatic food to the concentration of the same radionuclide in water (pCi/kg)/(pCi/L). The code has the element-specific aquatic bioaccumulation factors for fish and crustacea and mollusks.
Bioaccumulation factor for crustacea and mollusks	Physical	The ratio of radionuclide concentration in the aquatic food to the concentration of the same radionuclide in water (pCi/kg)/(pCi/L). The code has the element-specific aquatic bioaccumulation factors for fish and crustacea and mollusks.
Occupancy (Inhalation & External Parameters)		
Inhalation rate	Metabolic, Behavioral	The annual air intake in m^3/yr. The default value of 8,400 m^3/yr is recommended by the International Commission on Radiological Protection (1975).

Parameter	Classification	Description
Inhalation dose conversion factors	Metabolic	Radionuclide-specific values from FGR-11. Usually values for more than one inhalation class are listed per radionuclide. The three classes D, W, and Y correspond to retention half-times of less than 10 days, 10 to 100 days, and greater than 100 days; respectively. For some gaseous radionuclides (e.g., H-3, C-14, Ni-59, and Ni-63), inhalation classes other than D, W, and Y are also listed. The most conservative dose conversion factor is chosen as the default. The values can be changed if chemical forms are known or more appropriate data are available.
Ingestion dose conversion factors	Metabolic	Radionuclide-specific values from FGR-11. Ingestion dose conversion factors depend on the chemical form, which determines the fraction of a radionuclide entering the gastrointestinal tract that reaches body fluids. The code lists these fractions along with the dose conversion factor. The most conservative values are chosen as the default for the dose conversion factor. The values can be changed if chemical forms are known or more appropriate data are available.
Mass loading for inhalation	Physical, Behavioral	The air/soil concentration ratio or average mass loading of airborne contaminated soil particles. The code uses this parameter along with the area factor for inhalation pathway dose estimation. This average mass-loading factor includes short periods of high mass loading and sustained periods of normal activity on a typical farm.
Indoor dust filtration factor	Physical, Behavioral	Describes the effect of the building structure on the level of contaminated dust existing indoors. This is the fraction of the outdoor contaminated dust that will be available indoors.
External gamma shielding factor	Physical	Describes the effect of building structure on the level of gamma radiation existing indoors. It is the fraction of the outdoor gamma radiation that will be available indoors. The shielding factor value is used in calculating the occupancy factor.
Building foundation thickness	Physical	Average thickness of the building shell structure in the subsurface of the soil.

Parameter	Classification	Description
Building foundation density	Physical	The solid phase of mass to the total volume.
Building foundation total porosity	Physical	The ratio of the void space volume to the total volume of the porous medium.
Building foundation volumetric water content	Physical	The fraction of the total volume of the porous medium that is occupied by water.
Building foundation radon diffusion coefficient	Physical	The effective (or interstitial) radon diffusion coefficient is the ratio of the diffusive flux density of radon activity across the pore area to the gradient of the radon activity concentration in the pore space. Entering –1 for any diffusion coefficient will cause the code to calculate the diffusion coefficient based on the porosity and water content of the medium.
Contamination radon diffusion coefficient	Physical	See building foundation radon diffusion coefficient above.
Radon vertical dimension of mixing	Physical	The height into which the plume of radon is uniformly mixed in the outdoor air (above).
Building air exchange rate	Physical, Behavioral	The building exchange rate (or ventilation) is defined as the number of the total volumes of air contained in the building being exchanged with outside air per unit of time.
Building height	Physical	The average height of rooms in the building.
Building indoor area factor	Physical	The fraction of the area built on the contaminated soil. Values greater than 1 indicate contribution from adjacent walls. A default value of zero is assumed, which forces the code to calculate this time dependent area factor by assuming a floor area of 100 m^2 and walls extending into the contaminated area. This factor is time dependent because of erosion.
Foundation depth below ground surface	Physical	The vertical distance in the soil immediately from the bottom of the basement floor slab to the ground surface. A default value of –1 is used in the code; in this case the code adjusts the depths so that the foundation depth will not extend into the contaminated zone at each of the times at which dose is computed.

Parameter	Classification	Description
Radon-222 emanation coefficient	Physical	The fraction of the total radon generated by radium decay that escapes soil. (Depends on such parameters as porosity, particle size distribution, mineralogy, and moisture content.)
Radon-220 emanation coefficient	Physical	See radon-222 emanation coefficient (above).
Indoor time fraction	Behavioral	The average fraction of time during which an individual stays inside the house.
Outdoor time fraction	Behavioral	The average fraction of time during which an individual stays outdoors on the site.
Exposure duration	Behavioral	The exposure duration is the span of time, in years, during which an individual is expected to spend time on the site. This value is used in calculating lifetime cancer risk from exposure to radionuclide contamination. It is also used to calculate time-integrated dose if exposure duration is less than a year.

Parameter	Class	Units	Parameter Value	Source Justification
Contamination				
Thickness of contaminated zone	P	m	0.15	Assumed depth of contaminated surface soil.
Area of contaminated zone	P	m^2	8,094	Surface area of the Southern Industrial Area.
Shape of the contaminated zone	P	-	Circular	Default Physical value.
Initial concentration of principal radionuclides in soil	P	pCi/g	10	Assumed value used for DCGL generation.
Initial concentration of radionuclides present in groundwater	P	pCi/L	Not Used	Groundwater tritium concentration was not used as an input in the K_d determination.
Leach rate	P	1/yr	0	Default Physical value to invoke the calculation of this parameter via a first-order leaching model that uses the value of the soil/water distribution coefficient in the contaminated zone.
Solubility limit	P	mol/L	0	Default Physical value – not used by RESRAD with leach rate flag set to 0.
Time since placement of material	P	yr	0	Default Physical value assumed acceptable for purposes of DCGL generation.
Times for calculation	P	yr	1, 3, 10, 30, 100, 300, 1000	Default values applicable to the BRP site.
Contaminated zone (Unit 6a) density	P	g/cm^3	1.75	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone (Unit 6a) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value..
Use plant/soil ratio	NA	Check box	No	For purposes of DCGL calculation, the code should not be allowed to calculate the distribution coefficient from the plant root uptake factors.
Contaminated zone (Unit 6a) field capacity	P	-	0.263	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone erosion rate	P,B	m/yr	0.001	Default Physical/Behavioral value.
Contaminated zone (Unit 6a) total porosity	P	-	0.351	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone (Unit 6a) hydraulic conductivity	P	m/yr	536	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone (Unit 6a) b parameter	P	-	4.05	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Carbon-Model Parameters				
Thickness of evasion layer of C-14 in soil	P	m	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-14 evasion flux rate from soil	P	1/s	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 concentration in local water	P	g/cm^3	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 concentration in contaminated soil	P	g/g	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Fraction of vegetation carbon absorbed from soil	P	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Fraction of vegetation carbon absorbed from air	P	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 evasion flux rate from soil	P	1/s	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Grain fraction in beef cattle feed	B	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Grain fraction in milk cow feed	B	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
DCF correction factor for gaseous forms of C-14	P	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Soil				
Cover depth	P	m	0	The contamination is assumed to be on surface soil.
Density of cover material	P	g/cm^3	Not Used	A cover is not used in the site-specific dose model.
Cover total porosity	P	-	Not Used	A cover is not used in the site-specific dose model.
Cover volumetric water content	P	-	Not Used	A cover is not used in the site-specific dose model.
Cover radon diffusion coefficient	P	m^2/s	Not Used	A cover is not used in the site-specific dose model.
Cover erosion rate	P,B	m/yr	Not Used	A cover is not used in the site-specific dose model.
Number of unsaturated zones	P	-	5	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) thickness	P	m	1.22	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 2 (Unit 5) thickness	P	m	7.32	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Unsaturated Zone 3 (Unit 4) thickness	P	m	1.52	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 4 (Unit 3) thickness	P	m	9.14	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 5 (Unit 2) thickness	P	m	6.10	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) density	P	g/cm^3	1.75	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 2 (Unit 5) density	P	g/cm^3	1.92	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 3 (Unit 4) density	P	g/cm^3	1.83	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 4 (Unit 3) density	P	g/cm^3	1.85	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 5 (Unit 2) density	P	g/cm^3	2.17	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated Zone 2 (Unit 5) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated Zone 3 (Unit 4) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated Zone 4 (Unit 3) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated Zone 5 (Unit 2) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated Zone 1 (Unit 6a) total porosity	P	-	0.351	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 2 (Unit 5) total porosity	P	-	0.287	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 3 (Unit 4) total porosity	P	-	0.317	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 4 (Unit 3) total porosity	P	-	0.313	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Unsaturated Zone 5 (Unit 2) total porosity	P	-	0.194	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) effective porosity	P	-	0.088	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 2 (Unit 5) effective porosity	P	-	0.037	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 3 (Unit 4) effective porosity	P	-	0.029	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 4 (Unit 3) effective porosity	P	-	0.054	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 5 (Unit 2) effective porosity	P	-	0.033	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) field capacity	P	-	0.263	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 2 (Unit 5) field capacity	P	-	0.250	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 3 (Unit 4) field capacity	P	-	0.287	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 4 (Unit 3) field capacity	P	-	0.258	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 5 (Unit 2) field capacity	P	-	0.161	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) hydraulic conductivity	P	m/yr	536	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.
Unsaturated Zone 2 (Unit 5) hydraulic conductivity	P	m/yr	5.17	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.
Unsaturated Zone 3 (Unit 4) hydraulic conductivity	P	m/yr	1050	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.
Unsaturated Zone 4 (Unit 3) hydraulic conductivity	P	m/yr	15.6	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.
Unsaturated Zone 5 (Unit 2) hydraulic conductivity	P	m/yr	8.20E-03	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 1 (Unit 6a) soil-specific b parameter	P	-	4.05	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Unsaturated Zone 2 (Unit 5) soil-specific b parameter	P	-	4.90	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 3 (Unit 4) soil-specific b parameter	P	-	4.05	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 4 (Unit 3) soil-specific b parameter	P	-	4.90	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated Zone 5 (Unit 2) soil-specific b parameter	P	-	7.12	November 2002 Version 0.0 Hydrogeological Assessment Report.
Water				
Density of saturated zone (Unit 1)	P	g/cm^3	1.52	Environmental Report for Decommissioning (ERD) identifies the saturated zone as highly fractured Traverse limestone; Table 2.1 of the RESRAD Data Collection Handbook provides a value for sand that was used as an approximation for fractured limestone.
Saturated zone (Unit 1) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Saturated zone (Unit 1) total porosity	P	-	0.30	RESRAD Data Collection Handbook Table 3.2 value for limestone.
Saturated zone (Unit 1) effective porosity	P	-	0.14	RESRAD Data Collection Handbook Table 3.2 value for limestone.
Saturated zone (Unit 1) field capacity	P	-	0.16	In accordance with the RESRAD Data Collection Handbook, field capacity equals the total porosity minus the effective porosity.
Saturated zone (Unit 1) hydraulic conductivity	P	m/yr	1520	November 2002 Version 0.0 Hydrogeological Assessment Report.
Saturated zone (Unit 1) hydraulic gradient	P	-	0.02	Default value for generic use of RESRAD. It was not possible to measure the hydraulic gradient for the Southern Industrial Area.
Saturated zone (Unit 1) soil-specific b parameter	P	-	4.05	RESRAD Data Collection Handbook Table 13.1 value for sand as an approximation for fractured limestone.
Length of contaminated zone parallel to the aquifer flow	P	m	102	Diameter of an 8,094 m^2 contaminated zone.
Water table drop rate	P	m/yr	0.001	Default Physical value.

Parameter	Class	Units	Parameter Value	Source Justification
Well-pump intake depth (below water table)	P	m	10	Default value applicable to the BRP site.
Well pumping rate	B, P	m^3/yr	250	Default Behavioral/Physical value.
Model: non-dispersion or mass balance	NA	-	MB	The mass-balance model was chosen as the most conservative since it assumes that all of the radionuclides released from the contaminated zone are withdrawn through the well.
Evapotranspiration coefficient	P	-	0.24	Northern Michigan value taken from Figure 12.1 of the RESRAD Data Collection Handbook.
Humidity in air	P	g/m^3	8	Default Physical value.
Average annual wind speed	P	m/s	5	Environmental Report for Decommissioning.
Precipitation rate	P	m/yr	0.8	Environmental Report for Decommissioning.
Irrigation mode	B	-	Overhead	Behavioral value - ditch irrigation is not used in the Midwest.
Irrigation rate	B	m/yr	0.2	Behavioral default value.
Runoff coefficient	P	-	0.2	Default Physical value.
Watershed area for nearby stream or pond	P	m^2	2306717	ERD - the entire BRP site drains into Lake Michigan.
Accuracy for water soil computation	NA	-	0.001	Default value applicable to the BRP site.
Ingestion				
Fruit, vegetable, and grain consumption rate	M, B	kg/yr	160	Metabolic/Behavioral default value.
Leafy vegetable consumption	M, B	kg/yr	14	Metabolic/Behavioral default value.
Milk consumption	M, B	L/yr	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Meat and poultry consumption	M, B	kg/yr	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Fish consumption rate	M, B	kg/yr	5.4	Metabolic/Behavioral default value.
Other seafood consumption rate	M, B	kg/yr	0.9	Metabolic/Behavioral default value.
Aquatic food contaminated fraction	B, P	-	0.5	Default Behavioral/Physical value applicable to the BRP site.
Soil ingestion rate	M, B	g/yr	36.5	Metabolic/Behavioral default value.
Drinking water intake	M, B	L/yr	510	Metabolic/Behavioral default value.

Parameter	Class	Units	Parameter Value	Source Justification
Storage time for fruits, non-leafy vegetables, and grain	B	d	14	Behavioral default value.
Storage time for leafy vegetables	B	d	1	Behavioral default value.
Storage time for milk	B	d	1	Behavioral default value.
Storage time for meat	B	d	20	Behavioral default value.
Storage time for fish	B	d	7	Behavioral default value.
Storage time for crustacea and mollusks	B	d	7	Behavioral default value.
Storage time for well water	B	d	1	Behavioral default value.
Storage time for surface water	B	d	1	Behavioral default value.
Storage time for livestock fodder	B	d	45	Behavioral default value.
Drinking water contaminated fraction	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Household water contaminated fraction	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Livestock water contaminated fraction	B, P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Irrigation water contaminated fraction	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Plant food contaminated fraction	B, P	-	-1	Default Behavioral/Physical value applicable to the BRP site.
Meat contaminated fraction	B, P	-	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Milk contaminated fraction	B, P	-	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Livestock fodder intake rate for meat	M	kg/d	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Livestock fodder intake rate for milk	M	kg/d	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Livestock water intake rate for meat	M	L/d	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Livestock water intake rate for milk	M	L/d	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Livestock intake of soil	M	kg/d	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.

Parameter	Class	Units	Parameter Value	Source Justification
Mass loading for foliar deposition	P	g/m^3	1E-4	Default Physical value applicable to the BRP site.
Depth of soil mixing layer	P	m	0.15	Default Physical value applicable to the BRP site.
Depth of roots	P	m	0.9	Default Physical value applicable to the BRP site.
Groundwater fractional usage for drinking water	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Groundwater fractional usage for household water	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Groundwater fractional usage for livestock water	B, P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Groundwater fractional usage for irrigation water	B, P	-	1	Behavioral/Physical default value applicable to the BRP site.
Wet weight crop yield for non-leafy plants	P	kg/m^2	0.7	Default Physical value applicable to the BRP site.
Wet weight crop yield for leafy plants	P	kg/m^2	1.5	Default Physical value applicable to the BRP site.
Wet weight crop yield for fodder	P	kg/m^2	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Length of growing season for non-leafy vegetables	P	yr	0.17	Default value.
Length of growing season for leafy vegetables	P	yr	0.25	Default value.
Length of growing season for fodder	P	yr	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Translocation factor for non-leafy vegetables	P	-	0.1	Default Physical value applicable to the BRP site.
Translocation factor for leafy vegetables	P	-	1	Default Physical value applicable to the BRP site.
Translocation factor for fodder	P	-	1	Default Physical value applicable to the BRP site.
Weathering removal constant	P	1/yr	20	Default Physical value applicable to the BRP site.
Dry foliar interception fraction for non-leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Dry foliar interception fraction for leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Dry foliar interception fraction for fodder	P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.

Parameter	Class	Units	Parameter Value	Source Justification
Wet foliar interception fraction for non-leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Wet foliar interception fraction for leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Wet foliar interception fraction for fodder	P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Slope factor - external	M	(risk/yr)/(pCi/g)	Nuclide specific	Metabolic default value.
Slope factor - inhalation	M	risk/pCi	Nuclide specific	Metabolic default value.
Slope factor - ingestion	M	risk/pCi	Nuclide specific	Metabolic default value.
Plant transfer factor	P	-	Element specific	Default Physical value applicable to the BRP site.
Meat transfer factor	P	(pCi/kg)/(pCi/d)	Element specific	Default Physical value applicable to the BRP site.
Milk transfer factor	P	(pCi/L)/(pCi/d)	Element specific	Default Physical value applicable to the BRP site.
Bioaccumulation factor for fish	P	(pCi/kg)/(pCi/L)	Element specific	Default Physical value applicable to the BRP site.
Bioaccumulation factor for crustacea and mollusks	P	(pCi/kg)/(pCi/L)	Element specific	Default Physical value applicable to the BRP site.
Occupancy (Inhalation & External Parameters)				
Inhalation rate	M, B	m^3/yr	8,400	Metabolic/Behavioral default value.
Inhalation dose conversion factors	M	mrem/pCi	Nuclide Specific	Metabolic default value.
Ingestion dose conversion factors	M	mrem/pCi	Nuclide Specific	Metabolic default value.
Mass loading for inhalation	P, B	g/m^3	1E-4	Default Physical/Behavioral value applicable to the BRP site.
Indoor dust filtration factor	P, B	-	0.4	Default Physical/Behavioral value applicable to the BRP site.
External gamma shielding factor	P	-	0.7	Default Physical value applicable to the BRP site.
Building foundation thickness	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building foundation bulk density	P	g/m^3	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building foundation total porosity	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.

Parameter	Class	Units	Parameter Value	Source Justification
Building foundation volumetric water content	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building foundation radon diffusion coefficient	P	m^2/s	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Contaminated soil zone radon diffusion coefficient	P	m^2/s	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Radon vertical dimension of mixing	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building air exchange rate	P, B	1/hr	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building (room) height	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building indoor area factor	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Foundation depth below ground surface	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Radon-222 emanation coefficient	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Radon-220 emanation coefficient	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Indoor time fraction	B	-	0.5	Behavioral default value.
Outdoor time fraction	B	-	0.25	Behavioral default value.
Exposure duration	B	yr	30	Behavioral default value.

Parameter	Class	Units	Parameter Value	Source Justification
Contamination				
Thickness of contaminated zone (Unit 6a)	P	m	0.15	Assumed depth of contaminated surface soil.
Area of contaminated zone (Unit 6a)	P	m^2	8,094	Surface area of the Northern Industrial Area.
Shape of the contaminated zone (Unit 6a)	P	-	Circular	Default Physical value.
Initial concentration of principal radionuclides in soil	P	pCi/g	10	Assumed value used for DCGL generation.
Initial concentration of radionuclides present in groundwater	P	pCi/L	Not Used	Groundwater tritium concentration was not used as an input in the K_d determination.
Leach rate	P	1/yr	0	Default Physical value to invoke the calculation of this parameter via a first-order leaching model that uses the value of the soil/water distribution coefficient in the contaminated zone.
Solubility limit	P	mol/L	0	Default Physical value – not used by RESRAD with leach rate flag set to 0.
Time since placement of material	P	yr	0	Default Physical value assumed acceptable for purposes of DCGL generation.
Times for calculation	P	yr	1, 3, 10, 30, 100, 300, 1000	Default values applicable to the BRP site.
Contaminated zone (Unit 6a) density	P	g/cm^3	1.75	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone (Unit 6a) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Use plant/soil ratio	NA	Check box	No	For purposes of the sensitivity analysis, the code should not be allowed to calculate the distribution coefficient from the plant root uptake factors.
Contaminated zone (Unit 6a) field capacity	P	-	0.263	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone (Unit 6a) erosion rate	P,B	m/yr	0.001	Default Physical/Behavioral value.
Contaminated zone (Unit 6a) total porosity	P	-	0.351	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Contaminated zone (Unit 6a) hydraulic conductivity	P	m/yr	536	November 2002 Version 0.0 Hydrogeological Assessment Report.
Contaminated zone (Unit 6a) b parameter	P	-	4.05	November 2002 Version 0.0 Hydrogeological Assessment Report.
Carbon-Model Parameters				
Thickness of evasion layer of C-14 in soil	P	m	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-14 evasion flux rate from soil	P	1/s	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 concentration in local water	P	g/cm^3	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 concentration in contaminated soil	P	g/g	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Fraction of vegetation carbon absorbed from soil	P	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Fraction of vegetation carbon absorbed from air	P	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 evasion flux rate from soil	P	1/s	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Grain fraction in beef cattle feed	B	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Grain fraction in milk cow feed	B	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
DCF correction factor for gaseous forms of C-14	P	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Soil				
Cover depth	P	m	0	The contamination is assumed to be on surface soil.
Density of cover material	P	g/cm^3	Not Used	A cover is not used in the site-specific dose model.
Cover total porosity	P	-	Not Used	A cover is not used in the site-specific dose model.
Cover volumetric water content	P	-	Not Used	A cover is not used in the site-specific dose model.
Cover radon diffusion coefficient	P	m^2/s	Not Used	A cover is not used in the site-specific dose model.
Cover erosion rate	P,B	m/yr	Not Used	A cover is not used in the site-specific dose model.
Number of unsaturated zones	P	-	5	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Unsaturated zone 1 (Unit 6a) thickness	P	m	1.37	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 2 (Unit 6b) thickness	P	m	1.37	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 3 (Unit 5) thickness	P	m	4.11	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 4 (Unit 4) thickness	P	m	0.61	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 5 (Unit 3) thickness	P	m	7.32	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 1 (Unit 6a) density	P	g/cm^3	1.75	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 2 (Unit 6b) density	P	g/cm^3	1.95	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 3 (Unit 5) density	P	g/cm^3	1.92	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 4 (Unit 4) density	P	g/cm^3	1.83	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 5 (Unit 3) density	P	g/cm^3	1.85	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 1 (Unit 6a) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated zone 2 (Unit 6b) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated zone 3 (Unit 5) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated zone 4 (Unit 4) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated zone 5 (Unit 3) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Unsaturated zone 1 (Unit 6a) total porosity	P	-	0.351	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 2 (Unit 6b) total porosity	P	-	0.274	November 2002 Version 0.0 Hydrogeological Assessment Report.

Parameter	Class	Units	Parameter Value	Source Justification
Unsaturated zone 3 (Unit 5) total porosity	P	-	0.287	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 4 (Unit 4) total porosity	P	-	0.317	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 5 (Unit 3) total porosity	P	-	0.313	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 1 (Unit 6a) effective porosity	P	-	0.088	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 2 (Unit 6b) effective porosity	P	-	0.026	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 3 (Unit 5) effective porosity	P	-	0.037	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 4 (Unit 4) effective porosity	P	-	0.029	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 5 (Unit 3) effective porosity	P	-	0.054	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 1 (Unit 6a) field capacity	P	-	0.263	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 2 (Unit 6b) field capacity	P	-	0.248	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 3 (Unit 5) field capacity	P	-	0.250	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 4 (Unit 4) field capacity	P	-	0.287	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 5 (Unit 3) field capacity	P	-	0.258	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 1 (Unit 6a) hydraulic conductivity	P	m/yr	536	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.
Unsaturated zone 2 (Unit 6b) hydraulic conductivity	P	m/yr	554	November 2002 Version 0.0 Hydrogeological Assessment Report laboratory test.
Unsaturated zone 3 (Unit 5) hydraulic conductivity	P	m/yr	5.17	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.
Unsaturated zone 4 (Unit 4) hydraulic conductivity	P	m/yr	1046	November 2002 Version 0.0 Hydrogeological Assessment Report field (slug) test.

Parameter	Class	Units	Parameter Value	Source Justification
Unsaturated zone 5 (Unit 3) hydraulic conductivity	P	m/yr	15.6	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 1 (Unit 6a) soil-specific b parameter	P	-	4.05	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 2 (Unit 6b) soil-specific b parameter	P	-	4.38	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 3 (Unit 5) soil-specific b parameter	P	-	4.90	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 4 (Unit 4) soil-specific b parameter	P	-	4.05	November 2002 Version 0.0 Hydrogeological Assessment Report.
Unsaturated zone 5 (Unit 3) soil-specific b parameter	P	-	4.90	November 2002 Version 0.0 Hydrogeological Assessment Report.
Water				
Density of saturated zone (Unit 1)	P	g/cm^3	1.52	Environmental Report for Decommissioning (ERD) identifies the saturated zone as highly fractured Traverse limestone; Table 2.1 of the RESRAD Data Collection Handbook provides a value for sand, which was used as an approximation for fractured limestone.
Saturated zone (Unit 1) distribution coefficient	P	cm^3/g	Nuclide specific	Default Physical value.
Saturated zone (Unit 1) total porosity	P	-	0.30	RESRAD Data Collection Handbook Table 3.2 value for limestone.
Saturated zone (Unit 1) effective porosity	P	-	0.14	RESRAD Data Collection Handbook Table 3.2 value for limestone.
Saturated zone (Unit 1) field capacity	P	-	0.16	In accordance with the RESRAD Data Collection Handbook, field capacity equals the total porosity minus the effective porosity.
Saturated zone (Unit 1) hydraulic conductivity	P	m/yr	1.52E+03	November 2002 Version 0.0 Hydrogeological Assessment Report.
Saturated zone (Unit 1) hydraulic gradient	P	-	0.0007	November 2002 Version 0.0 Hydrogeological Assessment Report.
Saturated zone (Unit 1) soil-specific b parameter	P	-	4.05	RESRAD Data Collection Handbook Table 13.1 value for sand as an approximation for fractured limestone.

Parameter	Class	Units	Parameter Value	Source Justification
Length of contaminated zone parallel to the aquifer flow	P	m	102	Diameter of an 8,094 m^2 circular contaminated zone.
Water table drop rate	P	m/yr	0.001	Default Physical value.
Well-pump intake depth (below water table)	P	m	10	Default value applicable to the BRP site.
Well pumping rate	B, P	m^3/yr	250	Default Behavioral/Physical value.
Model: non-dispersion or mass balance	NA	-	MB	The mass-balance model was chosen as the most conservative since it assumes that all of the radionuclides released from the contaminated zone are withdrawn through the well.
Evapotranspiration coefficient	P	-	0.24	Northern Michigan value taken from Figure 12.1 of the RESRAD Data Collection Handbook.
Humidity in air	P	g/m^3	8	Default Physical value.
Average annual wind speed	P	m/s	5	Environmental Report for Decommissioning.
Precipitation rate	P	m/yr	0.8	Environmental Report for Decommissioning.
Irrigation mode	B	-	Overhead	Behavioral value - ditch irrigation is not used in the Midwest.
Irrigation rate	B	m/yr	0.2	Behavioral default value.
Runoff coefficient	P	-	0.2	Default Physical value.
Watershed area for nearby stream or pond	P	m^2	2306717	ERD - the entire BRP site drains into Lake Michigan.
Accuracy for water soil computation	NA	-	0.001	Default value applicable to the BRP site.
Ingestion				
Fruit, vegetable, and grain consumption rate	M, B	kg/yr	160	Metabolic/Behavioral default value.
Leafy vegetable consumption	M, B	kg/yr	14	Metabolic/Behavioral default value.
Milk consumption	M, B	L/yr	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Meat and poultry consumption	M, B	kg/yr	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Fish consumption rate	M, B	kg/yr	5.4	Metabolic/Behavioral default value.
Other seafood consumption rate	M, B	kg/yr	0.9	Metabolic/Behavioral default value.
Aquatic food contaminated fraction	B, P	-	0.5	Default Behavioral/Physical value applicable to the BRP site.

Parameter	Class	Units	Parameter Value	Source Justification
Soil ingestion rate	M, B	g/yr	36.5	Metabolic/Behavioral default value.
Drinking water intake	M, B	L/yr	510	Metabolic/Behavioral default value.
Storage time for fruits, non-leafy vegetables, and grain	B	d	14	Behavioral default value.
Storage time for leafy vegetables	B	d	1	Behavioral default value.
Storage time for milk	B	d	1	Behavioral default value.
Storage time for meat	B	d	20	Behavioral default value.
Storage time for fish	B	d	7	Behavioral default value.
Storage time for crustacea and mollusks	B	d	7	Behavioral default value.
Storage time for well water	B	d	1	Behavioral default value.
Storage time for surface water	B	d	1	Behavioral default value.
Storage time for livestock fodder	B	d	45	Behavioral default value.
Drinking water contaminated fraction	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Household water contaminated fraction	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Livestock water contaminated fraction	B, P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Irrigation water contaminated fraction	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Plant food contaminated fraction	B, P	-	-1	Default Behavioral/Physical value applicable to the BRP site.
Meat contaminated fraction	B, P	-	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Milk contaminated fraction	B, P	-	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Livestock fodder intake rate for meat	M	kg/d	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Livestock fodder intake rate for milk	M	kg/d	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.
Livestock water intake rate for meat	M	L/d	Not Used	The meat ingestion pathway is suppressed in the site-specific dose model.
Livestock water intake rate for milk	M	L/d	Not Used	The milk ingestion pathway is suppressed in the site-specific dose model.

Parameter	Class	Units	Parameter Value	Source Justification
Livestock intake of soil	M	kg/d	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Mass loading for foliar deposition	P	g/m^3	1E-4	Default Physical value applicable to the BRP site.
Depth of soil mixing layer	P	m	0.15	Default Physical value applicable to the BRP site.
Depth of roots	P	m	0.9	Default Physical value applicable to the BRP site.
Groundwater fractional usage for drinking water	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Groundwater fractional usage for household water	B, P	-	1	Default Behavioral/Physical value applicable to the BRP site.
Groundwater fractional usage for livestock water	B, P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Groundwater fractional usage for irrigation water	B, P	-	1	Behavioral/Physical default value applicable to the BRP site.
Wet weight crop yield for non-leafy plants	P	kg/m^2	0.7	Default Physical value applicable to the BRP site.
Wet weight crop yield for leafy plants	P	kg/m^2	1.5	Default Physical value applicable to the BRP site.
Wet weight crop yield for fodder	P	kg/m^2	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Length of growing season for non-leafy vegetables	P	yr	0.17	Default value.
Length of growing season for leafy vegetables	P	yr	0.25	Default value.
Length of growing season for fodder	P	yr	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Translocation factor for non-leafy vegetables	P	-	0.1	Default Physical value applicable to the BRP site.
Translocation factor for leafy vegetables	P	-	1	Default Physical value applicable to the BRP site.
Translocation factor for fodder	P	-	1	Default Physical value applicable to the BRP site.
Weathering removal constant	P	1/yr	20	Default Physical value applicable to the BRP site.
Dry foliar interception fraction for non-leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Dry foliar interception fraction for leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.

Parameter	Class	Units	Parameter Value	Source Justification
Dry foliar interception fraction for fodder	P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Wet foliar interception fraction for non-leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Wet foliar interception fraction for leafy vegetables	P	-	0.25	Default Physical value applicable to the BRP site.
Wet foliar interception fraction for fodder	P	-	Not Used	The meat and milk ingestion pathways are suppressed in the site-specific dose model.
Slope factor - external	M	(risk/yr)/(pCi/g)	Nuclide specific	Metabolic default value.
Slope factor - inhalation	M	risk/pCi	Nuclide specific	Metabolic default value.
Slope factor - ingestion	M	risk/pCi	Nuclide specific	Metabolic default value.
Plant transfer factor	P	-	Element specific	Default Physical value applicable to the BRP site.
Meat transfer factor	P	(pCi/kg)/(pCi/d)	Element specific	Default Physical value applicable to the BRP site.
Milk transfer factor	P	(pCi/L)/(pCi/d)	Element specific	Default Physical value applicable to the BRP site.
Bioaccumulation factor for fish	P	(pCi/kg)/(pCi/L)	Element specific	Default Physical value applicable to the BRP site.
Bioaccumulation factor for crustacea and mollusks	P	(pCi/kg)/(pCi/L)	Element specific	Default Physical value applicable to the BRP site.
Occupancy (Inhalation & External Parameters)				
Inhalation rate	M, B	m^3/yr	8,400	Metabolic/Behavioral default value.
Inhalation dose conversion factors	M	mrem/pCi	Nuclide Specific	Metabolic default value.
Ingestion dose conversion factors	M	mrem/pCi	Nuclide Specific	Metabolic default value.
Mass loading for inhalation	P, B	g/m^3	1E-4	Default Physical/Behavioral value applicable to the BRP site.
Indoor dust filtration factor	P, B	-	0.4	Default Physical/Behavioral value applicable to the BRP site.
External gamma shielding factor	P	-	0.7	Default Physical value applicable to the BRP site.
Building foundation thickness	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building foundation bulk density	P	g/m^3	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.

Parameter	Class	Units	Parameter Value	Source Justification
Building foundation total porosity	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building foundation volumetric water content	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building foundation radon diffusion coefficient	P	m^2/s	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Contaminated soil zone radon diffusion coefficient	P	m^2/s	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Radon vertical dimension of mixing	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building air exchange rate	P, B	1/hr	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building (room) height	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Building indoor area factor	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Foundation depth below ground surface	P	m	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Radon-222 emanation coefficient	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Radon-220 emanation coefficient	P	-	Not Used	The Radon Exposure Pathway is not used because the NRC does not regulate exposure to radon.
Indoor time fraction	B	-	0.5	Behavioral default value.
Outdoor time fraction	B	-	0.25	Behavioral default value.
Exposure duration	B	yr	30	Behavioral default value.

Parameter	Class[1]	Priority[2]	Units	Value/Distribution	Basis
Contamination					
Thickness of contaminated zone	P	2	m	0.15	Assigned value
Area of contaminated zone	P	2	m^2	8,094	Assigned value
Shape of the contaminated zone	P	3	-	Circular	Default Physical value
Initial concentration of principal radionuclides in soil	P	2	pCi/g	1	Assigned value
Initial concentration of radionuclides present in groundwater	P	3	pCi/L	0	Not Used
Leach rate	P	3	1/yr	0	Default Physical value to invoke the calculation of this parameter via a first-order leaching model that uses the value of the soil/water distribution coefficient in the contaminated zone.
Solubility limit	P	3	mol/L	0	Default Physical value – not used by RESRAD with leach rate flag set to 0
Time since placement of material	P	3	yr	0	Default Physical value assumed acceptable for purposes of DCGL generation
Times for calculation	P	3	yr	1, 3, 10, 30, 100, 300, 1000	Default values applicable to the BRP site
Contaminated zone density	P	1	g/cm^3	1.75	Site-specific measured physical parameter
Contaminated zone distribution coefficient for H-3	P	1	cm^3/g	0.043	EA-BRP-SC-0202
Contaminated zone distribution coefficient for Mn	P	1	cm^3/g	733	EA-BRP-SC-0202
Contaminated zone distribution coefficient for Fe	P	1	cm^3/g	1251	EA-BRP-SC-0202
Contaminated zone distribution coefficient for Co	P	1	cm^3/g	1284	EA-BRP-SC-0202
Contaminated zone distribution coefficient for Sr	P	1	cm^3/g	131	EA-BRP-SC-0202
Contaminated zone distribution coefficient for Cs	P	1	cm^3/g	2130	EA-BRP-SC-0202

Parameter	Class[1]	Priority[2]	Units	Value/Distribution	Basis
Contaminated zone distribution coefficient for Eu	P	1	cm^3/g	7194	EA-BRP-SC-0202
Contaminated zone distribution coefficient for Gd	P	1	cm^3/g	7194	EA-BRP-SC-0202
Use plant/soil ratio	NA	3	Check box	No	For purposes of the sensitivity analysis, the code should not be allowed to calculate the distribution coefficient from the plant root uptake factors.
Contaminated zone field capacity	P	3	-	0.263	Site-specific measured physical parameter
Contaminated zone erosion rate	P,B	2	m/yr	Continuous logarithmic	NUREG/CR-6697, Attachment C
Contaminated zone total porosity	P	2	-	0.351	Site-specific measured physical parameter
Contaminated zone hydraulic conductivity	P	2	m/yr	536	Site-specific measured physical parameter
Contaminated zone b parameter	P	2	-	4.05	Site-specific measured physical parameter
Carbon-Model Parameters					
Thickness of evasion layer of C-14 in soil	P	2	m	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-14 evasion flux rate from soil	P	3	1/s	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 concentration in local water	P	3	g/cm^3	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 concentration in contaminated soil	P	3	g/g	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Fraction of vegetation carbon absorbed from soil	P	3	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.

Parameter	Class[1]	Priority[2]	Units	Value/Distribution	Basis
Fraction of vegetation carbon absorbed from air	P	3	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
C-12 evasion flux rate from soil	P	3	1/s	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Grain fraction in beef cattle feed	B	3	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Grain fraction in milk cow feed	B	3	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
DCF correction factor for gaseous forms of C-14	P	3	-	Not Used	Carbon-14 has not been identified in any contaminated surface, subsurface or aquifer samples.
Soil					
Cover depth	P	2	m	0	The contamination is assumed to be on surface soil.
Density of cover material	P	1	g/cm^3	Not Used	A cover is not used in the site-specific dose model.
Cover total porosity	P	3	-	Not Used	A cover is not used in the site-specific dose model.
Cover volumetric water content	P	3	-	Not Used	A cover is not used in the site-specific dose model.
Cover radon diffusion coefficient	P	3	m^2/s	Not Used	A cover is not used in the site-specific dose model.
Cover erosion rate	P,B	2	m/yr	Not Used	A cover is not used in the site-specific dose model.
Number of unsaturated zones	P	3	-	5	Site-specific measured physical parameter
Unsaturated zone 1 thickness	P	1	m	1.37	Site-specific measured physical parameter
Unsaturated zone 2 thickness	P	1	m	1.37	Site-specific measured physical parameter
Unsaturated zone 3 thickness	P	1	m	4.11	Site-specific measured physical parameter
Unsaturated zone 4 thickness	P	1	m	0.61	Site-specific measured physical parameter
Unsaturated zone 5 thickness	P	1	m	7.32	Site-specific measured physical parameter
Unsaturated zone 1 density	P	2	g/cm^3	1.75	Site-specific measured physical parameter