Enclosure F
71 Pages

Consumers Energy
Big Rock Point

RADIOACTIVE EFFLUENT RELEASE REPORT

BIG ROCK POINT VOLUME 25
PART A – ODCM
PART B - PCP

Revision 29

BIG ROCK POINT NUCLEAR POWER PLANT
PROCEDURE APPROVAL AND AUTHORIZATION

Procedure No. VOLUME 25 Rev No. 29

Procedure Title: OFF-SITE DOSE CALCULATION MANUAL AND RELATED DOCUMENTS

A. OFF-SITE DOSE CALCULATION MANUAL

B. PROCESS CONTROL PROGRAM

CURRENT REVISION STATUS

Author TAGoble Date 02/09/06 Quality Review Form No. 022-06

APPLICABILITY ISSUE HISTORY

Revision No. N/A Date N/A Quality Review Form No. N/A

Approved for use

Procedure Sponsor/Designate [signature] Date 02/14/06

Authorized Period of Use N/A through N/A

BEFORE USING THIS PROCEDURE FOR WORK ACTIVITIES, VERIFY WITH THE RESPECTIVE PROCEDURE CONTROLLING DEPARTMENT THERE ARE NO OUTSTANDING TEMPORARY CHANGES

When applicable:

PROCEDURE IMPLEMENTATION HISTORY

Reviewed for System or Component Operability

Performed by ______ Title

Completed/Reviewed by ______ Title

Date ______ Time ______ Date ______ Time ______

Method of Verification

- ☐ Functional Test
- ☐ Physical Inspection
- ☐ Administrative Review

AMMS WORK ORDER NO. (if applicable) ______

OFF-SITE DOSE CALCULATION MANUAL
APPROVAL CERTIFICATION

The signature below certifies approval for revision to the Off-site Dose Calculation Manual pursuant to Defueled Technical Specifications, Section 6.6.2.4.1, Changes to the ODCM.

2/14/06

Site General Manager   Date

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ODCM SECTION I

PROCEDURAL AND SURVEILLANCE REQUIREMENTS

(Relocated Technical Specifications)

1.1.4 Surveillance Requirements

Equipment utilized for particulate air sampling shall be calibrated, source checked, maintained and operated in accordance with Radiological Protection procedures applicable to worker protection.

1.1.5 Basis

Plant systems have been dismantled and fuel is present only in sealed dry storage containers at a location separate from the dismantlement area. Equipment, concrete and other structural materials having significant levels of activation or contamination have been shipped off as radiological waste. Radioactive materials that remain are predominantly of concentrations less than detectable at the lower limits of detection established for environmental analyses of solids (sediments) in ODCM, Section I, Table 1-3.

Worker protection monitoring is performed for all atmospheres with potential to reach 0.3 DAC over a 40-hour work week, and any short-term concentration above 1.0 DAC. At maximum (Cs-137*), DAC-to-EC ratio is (6E-8)/(2E-10)=300 at the work location. One DAC equals only 0.282 EC at off-site locations, due to atmospheric dilution of 9.4E-4 times local on-site concentration at the site boundary (EA-BRP-RAE-0501).

Suspension of particulate radioactivity from surfaces has been calculated to provide higher potential concentrations than releases from bulk materials during decontamination and dismantlement activities (EA-BRP-RAE-0301). Maximum off-site dose with surface radioactivity at a level equal to ten times the Regulatory Guide 1.86 levels for beta-gamma radioactivity is 0.16 mrem/year, assuming 2000 hours per year of release.

Removal of plant-installed monitoring equipment has been performed in accordance with ODCM, Section III, Part 4.2.2. Since that time, conservative use of airborne samples from work areas as estimates of airborne release concentrations has shown no 31-day dose greater than 0.01% of the annual guideline (highest month was December 2004 at 0.0064% of the annual guideline value).

*Co-60 ratio is 200 or 350 (mean = 275), depending on chemical form.

1.2 GASEOUS EFFLUENTS DOSE RATE

1.2.1 Applicability

This section remains applicable only for tritium and radionuclides in particulate form. Surveillance requirements will cease to be applicable when analyses show no further potential for releases above the action levels of Requirement 1.2.3.

1.2.2 Requirement

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary (see Figure 2-1) shall be limited to the following:

a. For noble gases: NA

b. For tritium and for all radionuclides in particulate form with half-lives greater than eight days: Less than or equal to 1500 mrem/year to any organ.

1.2.3 Action

With the dose rate averaged over a period of one hour exceeding the above limits (>35 DAC in the on-site work area); upon discovery, promptly restore the release rate to within the above limit(s).

1.2.4 Surveillance Requirements

a. For noble gases: NA.

b. The dose rate due to tritium and all radionuclides in particulate form with half lives greater than eight days in gaseous effluents shall be determined to be within the limits of Requirement 1.2.2 above by utilizing the methodology and parameters in Section II of this ODCM.

1.2.5 Basis

This specification is provided to allow the Licensee operational flexibility in meeting the limits of 10CFR50, Appendix I. The instantaneous dose rates of this specification are higher than the implied dose rates of 10CFR20, Appendix B, Table 2, Column 1, Effluent Concentrations for Members of the Public. However, the ALARA philosophy for atmospheric effluents ensures doses to the public at and beyond the site boundary are minimized and less than the annual doses of 10CFR50, Appendix I.

The required detection capabilities for radioactive materials in gaseous effluent samples are tabulated in terms of the lower limits of detection (LLDs). See Notation c of Table 1-3.

The relationship between DAC and the 10 EC of Requirement 1.1.2 is provided in Basis 1.1.5: one DAC at the work location equals 0.282 EC at the site boundary. Therefore, the hourly maximum of 10 EC at the site boundary equals (10 EC)/(0.282 EC/DAC)=35.4 DAC at the work location.

1.3 AIRBORNE RADIOACTIVITY DOSE

1.3.1 Applicability

This section remains applicable only for doses due to tritium and radionuclides in particulate form. Surveillance requirements will cease to be applicable when analyses show no further potential for releases above the action levels of Requirements 1.3.3.b.

1.3.2 Requirements

a. The air dose due to noble gases released in gaseous effluents is not applicable.

b. The dose to a member of the public from tritium and all radionuclides in particulates form with half lives greater than eight days in gaseous effluents released to areas at and beyond the site boundary (see Figure 2.1) shall be limited to the following:

   1) During any calendar quarter: Less than or equal to 7.5 mrem to any organ, and

   2) During any calendar year: Less than or equal to 15 mrem to any organ.

1.3.3 Actions

a. With the calculated dose from the release of radioactive material exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. With projected dose in the next 31 days greater than two percent of the annual limits defined by Requirement 1.3.2.b above, institute controls to reduce release below two percent of the annual limit per month.

1.3.4 Surveillance Requirement

a. Cumulative and projected dose contributions for the current calendar quarter and current calendar year for radioactive materials shall be determined at least once per 31 days in accordance with the methodology and parameters in this ODCM.

b. Projected dose for the next 31 days shall be calculated at least once per 31 days.

1.3.5 Basis

This specification is provided to implement the guides set forth in Sections II.B and II.C of Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in airborne effluents to unrestricted areas will be kept "as low as is reasonably achievable."

The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated.

The dose calculation methodology and parameters established in Section II, Parts 1.2 and 1.3 of this ODCM are consistent with the methodology provided in Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I, Revision 1, October 1977 and Regulatory Guide 1.111, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Revision 1, July 1977.

Section II, Part 1.2.2.b, provides methodology for evaluation of releases from unconfined areas. This methodology may be used for releases from contaminated structures in the process of dismantlement, in the absence of, or in lieu of, airborne analyses in the work area itself.

2.0 LIQUID EFFLUENTS

2.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

2.1.1 Applicability

Installed liquid effluent monitoring equipment is no longer required. Effluents purposefully discharged to the lake form surface and groundwater that have been detained for sedimentation will undergo precautionary sampling and analysis for tritium and radionuclides other than dissolved or entrained noble gases.

2.1.2 Requirement

Pending determination that such precaution is not necessary, any systems containing water detained for sedimentation or other purposes shall be sampled prior to initiation of a season of release and a minimum of monthly throughout the season thereafter, from grab samples or effluent composites.

2.1.3 Action

a. With a detained water sample indicating radioactivity concentration greater than twice the lower limit of detection for environmental water samples as specified in ODCM, Section I, Table 1-1, prohibit further release of water until calculations confirm that the EC fraction is less than the requirement of ODCM, Section I, Part 2.2.2, AND

b. With a sample indicating detectable radioactivity concentration greater than twice the lower limit of detection for environmental water samples as specified in ODCM, Section I, Table 1-1, prohibit release until calculations confirm that monthly release will not exceed the requirement of ODCM, Section I, Part 2.3.2.

2.1.4 Surveillance Requirements

Analytical accuracy for liquid samples is documented in routine calibration and quality control checks sufficient to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10CFR50.

2.1.5 Basis

Sampling of detained water is provided to indicate and quantify presence of radioactive material prior to and during releases of clarified liquids such that controls may be applied, as applicable, to limit such releases.

With the exception of tritium, previous experience has shown that site drainage water has contained radioactive materials only in association with suspended soils. Water is detained to allow sedimentation. This sedimentation serves to clarify the water for release and to minimize the potential for effluent to contain detectable radioactivity. Sampling is precautionary.

2.2 LIQUID EFFLUENTS CONCENTRATION

2.2.1 Applicability

Effluents purposefully discharged to the lake from surface and groundwater that have been detained for sedimentation will undergo precautionary sampling and analysis for tritium and radionuclides other than dissolved or entrained noble gases.

2.2.2 Requirement

The concentration of radioactive material released in liquid effluents from the site to UNRESTRICTED AREAS (identified as the site boundary, see Figure 2-1) shall be limited to ten times the concentration values specified in Appendix B, Table 2, Column 2, to 10CFR20 for radionuclides other than dissolved or entrained noble gases.

2.2.3 Action

With liquid effluent concentrations in areas at or beyond the site boundary (see Figure 2-1) exceeding limits, upon discovery, promptly restore the concentration to within the above limits.

2.2.4 Surveillance Requirements

a. A representative grab sample of water to be released shall be analyzed prior to initiating a season of release.

b. Composite or grab samples of potentially radioactive effluents shall be sampled and analyzed at least monthly during release periods for radionuclides other than dissolved and entrained noble gases.

c. The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in this ODCM, to assure that the concentrations at the point of release are maintained within the limits of Requirement 2.2.2 above.

2.2.5 Basis

This requirement ensures that the concentration of radioactive materials released in effluents to unrestricted areas will be less than ten times the concentration levels specified in 10CFR20, Appendix B, Table 2, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the Section II.A design objectives of Appendix I, 10CFR50, to a member of the public, and (2) the limits of 10CFR20.1301(e) to the population. Iodines, noble gases and short-lived particulates ($t_{1/2}$ less than approximately five years) are no longer observed at Big Rock Point (BRP) due to the time period elapsed since permanent shut down.

See Notation c to Table 1-3 for further information on the LLDs referenced in this section.

2.3 LIQUID EFFLUENT DOSE

2.3.1 Applicability

All installed liquid radioactive waste treatment systems have been removed in the plant dismantlement process, after being shown not necessary for maintenance of public doses below two percent of Requirements 2.3.2.a and 2.3.2.b, below. However, effluents purposefully discharged to the lake from surface and groundwater sources will undergo precautionary sampling and analysis for tritium and radionuclides other than dissolved or entrained noble gases.

2.3.2 Requirement

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to areas at and beyond the site boundary (see Figure 2-1) shall be limited:

a. During any calendar quarter: to less than or equal to 1.5 mrem to the total body and to less than or equal to five mrem to any organ, and

b. During any calendar year: to less than or equal to three mrem to the total body and to less than or equal to 10 mrem to any organ.

2.3.3 Actions

a. With the calculated dose from the release of radioactive material exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. With projected dose in the next 31 days greater than two percent of the annual limit of Requirement 2.3.2.b, take action to ensure further releases do not exceed the two percent level.

2.3.4 Surveillance Requirements

a. Cumulative and projected dose contributions for the current calendar quarter and current calendar year for radioactive materials shall be determined in accordance with the methodology and parameters in this ODCM at least once per 31 days.

b. Projected dose for the next 31 days shall be calculated prior to each batch release, or at least once every 31 days if continuous release is in progress.

2.3.5 Basis

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10CFR Part 50. The Requirement 2.3.2 for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to unrestricted areas will be kept "as low as is reasonably achievable." Also, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40CFR141.

The dose calculation methodology and parameters in this ODCM implement the requirements in Section III.A of Appendix I that conformance be shown by calculational procedures based on models and data, such that the actual exposure of the public through appropriate pathways is unlikely to be substantially underestimated.

The methods specified in the ODCM for calculating the doses due to the actual and estimated release rates of radioactive materials in liquid effluents are consistent with the methodology of Regulatory Guide 1.109, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I, Revision 1, October, 1977 and Regulatory Guide 1.113, Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I, April 1977.

The requirement for action when effluents are projected to exceed two percent of the annual Appendix I guidelines in any 31 days dose due to effluents will be minimized.

3.0 TOTAL FUEL CYCLE DOSE

3.1 APPLICABILITY

Requirement and actions set forth below are applicable when liquid effluent dose exceeds twice the limits of Requirement 1.3.2.a, 1.3.2.b.1), 1.3.2.b.2), 2.3.2.a, or 2.3.2.b of this section.

3.2 REQUIREMENT

The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

3.3 ACTION

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Requirements 1.3.2.a, 1.3.2.b.1), 1.3.2.b.2), 2.3.2.a, or 2.3.2.b of this section, calculations should be made including direct radiation contributions from fuel pool systems and from outside storage tanks to determine whether the limits of Requirement 3.2 above have been exceeded. If such is the case:

Prepare and submit to the Commission within 30 days, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits.

This Special Report, as defined in 10CFR20.2203(a)(4), shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations.

3.4 SURVEILLANCE REQUIREMENTS

a. Cumulative dose contributions from liquid and gaseous effluents shall be determined to comply with Requirement 3.2 above at a minimum of once per 31 days, and in accordance with the methodology and parameters described in Section II of this ODCM.

b. Cumulative dose contributions from direct radiation from spent fuel and from radwaste storage shall be determined in accordance with the methodology and parameters in ODCM, Section II, Part 3.2, Assumptions, and 3.3, Dose Calculation.

3.5 BASIS

This specification is provided to meet the dose limitations of 40CFR190 that have been incorporated into 10CFR20 by 46FR18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

The Special Report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40CFR190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of five miles must be considered.

If the dose to any member of the public is estimated to exceed the requirements of 40CFR190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40CFR190 have not already been corrected), in accordance with the provisions of 40CFR190.11 and 10CFR20.2203(a)(4) is considered to be a timely request and fulfills the requirements of 40CFR190 until NRC staff action is completed. The requested variance relates only to the limits of 40CFR190 and shall not apply in any way to the other requirements for dose limitation of 10CFR20, as addressed in Specifications.

An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

4.0 RADIOLOGICAL ENVIRONMENTAL MONITORING

4.1 APPLICABILITY

The requirements and actions for environmental monitoring are applicable until the Final Status Survey (FSS) has been completed in accordance with the License Termination Plan (LTP). However, if any groundwater monitoring well exceeds, or is projected to exceed, 20,000 pCi/l at the time the FSS report is submitted, monitoring of groundwater will continue as specified (see Table 1-1).

4.2 REQUIREMENT

The radiological environmental monitoring program shall be conducted as specified in Table 1-1.

4.3 ACTION

a. With the radiological environmental monitoring program not being conducted as specified in Table 1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Section III of this ODCM, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents. When more than one of the radionuclides in Table 1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Reporting Level (1)}} + \frac{\text{Concentration (2)}}{\text{Reporting Level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Requirements 1.3.2 or 2.3.2 of ODCM, Section I. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the Condition shall be reported and described in the Annual Radiological Environmental Operating Report.

4.4 SURVEILLANCE REQUIREMENTS

a. The radiological environmental monitoring samples shall be collected and analyzed pursuant to the requirements of Table 1-1 and the detection capabilities required by Table 1-3.

b. A final land use census was undertaken in the last year of plant operation. No further census data is required.

c. Analytical Accuracy

Records of instrument calibrations and quality control data for the instrumentation used for environmental analyses shall be maintained. Interlaboratory comparison participation shall be required for off-site environmental sample analyses. Environmental TLD readouts will be performed by an NVLAP-accredited facility.

4.5 BASIS

a. Monitoring Program

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive resulting from post-operational conditions. This monitoring program implements Section IV.B.2 of Appendix I to 10CFR50 by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of low effluent levels due to the defueled conditions and the modeling of the environmental exposure pathways.

Upon termination of the current radiological environmental monitoring program, this program may be used to confirm that the Independent Spent Fuel Storage Installation (ISFSI) is meeting the requirements of 10CFR72.104. Specifically, calculated annual doses from plant-generated radioactive effluents and direct radiation, including ISFSI operations, cannot exceed 25 mrem to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem to any real individual beyond the controlled area.

b. Basis, Radiological Monitoring

The iodine and noble gas source terms for the defueled facility are negligible, and particulate release has been reduced more than a thousandfold, compared to power operations. Consequently, off-site sampling of the food chain, and off-site air particulate and iodine sampling for radionuclides normally linked to atmospheric releases have been deleted from the ODCM. Liquid releases during initial stages of decommissioning, although much smaller than during power operation, will continue to be monitored for food chain uptake at the most sensitive indicator location: the lake environment in the area that clarified ground and surface water effluents reach the lake.

Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. Program changes may be initiated based on operational experience. The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 1-3 are considered optimum for routine environmental measurements in industrial laboratories.

c. Land Use Census

A land use census from the final year of plant power operation is provided in ODCM, Section II, Table 2-4, for informational purposes. Use of an assumed garden and milk animal (critical receptors) at the site boundary in the downwind sector of highest D/Q, per ODCM, Section II, Table 2-4, is conservative for ground level release, with respect to any actual garden and milk animal locations.

d. Analytical Accuracy

Participation in an approved Interlaboratory Comparison Program is required for off-site environmental monitoring sample analyses. Analytical accuracy of radioactive material in effluent and other on-site samples is documented in routine calibration and quality control checks in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10CFR50. ThermoLuminescent Dosimetry (TLD) readouts are performed by a NVLAP Accredited laboratory in order to assure analytical reliability and accuracy.

TABLE 1-1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and frequency of Analysis
1. Direct Radiation[b]	21 monitoring stations either with two or more TLDs or one instrument for measuring and recording dose rate continuously, placed as follows[d]: a) Miscellaneous site locations (4) b) A ring of stations (6) at or near the Site boundary c) Balance of stations (3) placed to serve as control stations d) Outside perimeter of ISFSI (4) e) ISFSI protected area fence line (4)	Quarterly	Gamma dose quarterly
2. Waterborne			
a. Lake	1 sample near site	Semiannual (grab)	Tritium and gamma isotopic[d]
b. Well (drinking) and groundwater monitoring wells	1 sample from Site well, if in use, and 1 sample from minimum of 6 monitor wells	Semiannual (grab) Semiannual (grab)	Tritium and gamma isotopic semiannually
3. Biota			
a. Marine	1 fish or invertebrate sample where clarified, detained water enters lake	Semiannual (grab) Apr-Nov	Gamma Isotopic Semiannually
4. Lake Sediment			
a. Shoreline	1 sample where clarified, detained water enters lake	Semiannual (grab) Apr-Nov	Gamma Isotopic Semiannually
b. Shoreline	1 sample each side of 4.a (above), within ~½ mile	Semiannual (grab) Apr-Nov	Gamma Isotopic Semiannually

TABLE 1-1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Notations

a. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to the Reporting Requirements of ODCM, Section III. Alternative media and locations may be chosen for any particular pathway if designated locations or media are not available, and appropriate substitutions are made within 30 days in the radiological environmental monitoring program.

b. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. The background dosimetry requirement also may be met through use of dosimeters shared with another facility, or from data provided by another entity, such as the State of Michigan, as appropriate for this site.

c. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to licensed materials in the effluents from the facility.

d. For the purposes of this table, a TLD is considered to be one phosphor; two or more phosphors or phosphor readout zones in a packet are considered as two or more dosimeters.

TABLE 1-2
REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Level

Analysis	Water (pCi/ℓ)	Fish (pCi/kg)
H-3	20,000*	-
Mn-54**	1,000	30,000
Fe-59**	400	10,000
Co-58**	1,000	30,000
Co-60	300	10,000
Zn-65**	300	20,000
Zr-Nb-95**	400	-
I-131**	2	-
Cs-134	30	1,000
Cs-137	50	2,000
Ba-La-140**	200	-

* For drinking water samples from site domestic water well (20,000 pCi/l is the 40CFR141 value).

** Short half-lives of these radionuclides may preclude presence due to plant effluents, but these nuclides may be observed from weapons testing.

TABLE 1-3
DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS [a]
LOWER LIMIT OF DETECTION (LLD) [b, c]

Analysis	Water (pCi/-1)	Fish (pCi/kg, wet)	Sediment (pCi/kg, dry)
H-3	2,000	-	-
Mn-54*	15	130	-
Fe-59*	30	260	-
Co-58, 60	15	130	-
Zn-65*	30	260	-
Zr-Nb-95*	15	-	-
I-131*	1[d]	-	-
Cs-134	15	130	150
Cs-137	18	150	180
Ba-La-140	15	-	-

* Short half-lives of these radionuclides may preclude presence due to plant effluents, but these nuclides may be observed from weapons testing.

Notations

a. This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to the Reporting Requirements of ODCM, Section III.

b. Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

c. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability, with only five percent probability of falsely concluding that a blank observation represents a "real" signal.**

**NUREG-0473, Draft Radiological Technical Specifications for BWRs, November 1978.

TABLE 1-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS [a]
LOWER LIMIT OF DETECTION (LLD) [b, c]

Notations

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\, s_b}{E \times V \times 2.22 \times Y \times e^{-\lambda \Delta t}}$$

Where:

LLD	is the predetermined lower limit of detection as defined above, as pico-curies per unit mass or volume.
s_b	is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,
E	is the counting efficiency, as counts per disintegration,
V	is the sample size in units of mass or volume,
2.22	is the number of disintegrations per minute per pico-curie,
Y	is the fractional radiochemical yield, when applicable,
λ	is the radioactive decay constant for the particular radionuclide, and
Δt	for environmental samples is the elapsed time between sample collection or end of the sample collection period, and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculation.

d. LLD for drinking water samples.

5.0 TEMPORARY LIQUID STORAGE IN OUTSIDE TANKS

5.1 REQUIREMENT

The concentration of radioactive material contained in any unprotected outside tank* used for temporary liquid radwaste storage shall be limited such that the mixture of radionuclides do not exceed 8000 times the Effluent Concentration (EC) as listed in 10CFR20, Appendix B, Table 2, Column 2.

$$\frac{C_a}{EC_a} + \frac{C_b}{EC_b} + \ldots + \frac{C_i}{EC_i} = < 8000$$

5.2 ACTION

With the quantity of radioactive material in any of the tanks exceeding the above concentration, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank activity to within the limit, and describe the events leading to this condition in the next Radiological Effluent Release Report.

5.3 SURVEILLANCE REQUIREMENT

The concentration of radioactive material contained in each tank shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per seven days when radioactive materials are being added to the tank.

* An outdoor unprotected liquid storage tank is any vessel which does not have dikes or walls capable of holding the vessel contents, or does not have overflows or drains which route leaks, overflows or other losses back to a containment system. A vessel or container that is intended for use as a containment system or packaging is not considered an outdoor unprotected storage tank.

5.4 BASIS

This requirement will provide reasonable assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations, averaged over not more than one year, would be less than 10 EC per 10CFR20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the Unrestricted Area (the dilution between BRP and the Charlevoix drinking water supply has been established as 800). Such an uncontrolled release would not exceed the 10CFR20 annual dose limits for the public or 10CFR50, Appendix I guidelines due to the short interval over which that any such release would occur.

6.0 DEFINITIONS AND SURVEILLANCE REQUIREMENT TIME INTERVALS

6.1 APPLICABILITY

Requirements do not apply to equipment removed from service for dismantlement.

6.2 REQUIREMENT

Unless otherwise specified, each surveillance requirement shall be performed with time extensions not to exceed 25%.

6.3 ACTION

A surveillance not performed within the defined interval (including the allowable extension) shall constitute noncompliance with the operability requirements.

6.4 SURVEILLANCE REQUIREMENT

NA

6.5 BASIS

Terms and Surveillance interval extensions are as defined in Defueled Technical Specifications.

ODCM SECTION II

METHODOLOGIES FOR REQUIREMENTS IMPLEMENTATION

1.0 GASEOUS EFFLUENTS

1.1 CALCULATION OF ALLOWABLE CONCENTRATION AND DOSE

This section of the ODCM describes appropriate methods for determination of allowable concentrations and dose from gaseous effluents.

1.1.1 Allowable Concentration

The total EC-fraction (R) for the release point will be calculated by the relationship defined by Note 1 of Appendix B, 10CFR20.

$$R = \left(\frac{X}{Q}\right)(F) \qquad \sum_i \frac{A_i}{EC_i} = \leq 10 \qquad (1.1)$$

Where:

R = The total EC-fraction for the release point.

X/Q = Most conservative sector site boundary dispersion (8.10E-08 sec/m; for ground level releases from Table 2-4).

F = Release flow rate cubic feet per minute (cfm).

A_i = The measured or calculated concentration, at ambient temperature and pressure, of nuclide i (µCi/cc) at the release point.

EC_i = The EC of nuclide i from 10CFR20, Appendix B.

1.1.2 Monitor Response

Due to lack of installed instrumentation, this section has been deleted.

1.2 DOSE CALCULATION

1.2.1 Dose Pathway Analyses

Doses are calculated for radionuclides detected in accordance with the analytical requirements of ODCM, Section I. All dose pathways of major importance in the BRP environs are considered. Dose calculation is by direct use of the NRC-approved GASPAR computer program, or by use of the constants of Table 2-6. Site-specific parameters for GASPAR use in such analyses are provided in Tables 2-4 and 2-5.

1.2.2 Equations for Determining Air Concentration

a. For determining the air concentration at distance x:

$$X_i = \sum_{j=1}^{7} \sum_{k=1}^{7} \left(\frac{2}{\pi}\right)^{1/2} \frac{f_{jk}\, Q_i\, P}{\Sigma_{zk}\, u_j \left(2\pi x/n\right)} \exp\left(-\lambda_i \frac{x}{u_j}\right) \exp\left(\frac{-h^2}{2\sigma_{zk}^2}\right) \qquad (1.3)$$

Where:

X_i	=	Air concentration of radionuclide i, µCi/m;.
f_{jk}	=	Joint relative frequency of occurrence of winds in wind speed class j, stability class k, blowing toward this exposure point, expressed as a fraction.
Q_i	=	Average release rate of radionuclide i, µCi/s.
P	=	Fraction of radionuclide remaining in plume.
Σ_{zk}	=	Vertical dispersion coefficient for stability class k (m).
u_j	=	Midpoint value of wind speed class interval j, m/s.
x	=	Downwind distance, m.
n	=	Number of sectors, 16.
λ_i	=	Radioactive decay coefficient of radionuclide i, s^{-1}.
2Ξx/n	=	Sector width at point of interest, m.

h = Release height (formerly 73 meters).

σ^2_{zk} = Vertical dispersion coefficient of stability class k.

b. Special Case for Ground Level Release with Unknown Release Rate

When dismantlement of significantly contaminated structures (those with surface contamination levels greater than ten times the Table I, Acceptable Surface Contamination Levels, of Regulatory Guide 1.86) occurs in such a way that contaminated dust is released to the environment, there may be no identifiable release rate (parameter Q_i of Equation 1.2) by which calculations may be performed using the previously discussed methods. However, by use of Figure 2-2, particulate air concentration at the site boundary may be calculated from Equation 1.3:

$$X_{id2} = (X_{id1})\,(X/Q)_{d2}/(X/Q)_{d1}^{*} \tag{1.3}$$

And

$$Q_i = X_{id2}/(X/Q)_{d2} \tag{1.4}$$

Where:

X_{id2} = Site boundary air concentration ($\mu Ci/m^3$)

X_{id1} = Measured air concentration at distance d_1 from source (typically from downwind portable air sampler), in $\mu Ci/m^3$

$(X/Q)_{d2}$ = Dispersion constant at site boundary (ground level value from Table 2-4)

$(X/Q)_{d1}$ = Dispersion constant at sample point (from Figure 2-2)

Q_i = Release Rate ($\mu Ci/sec$)

* $(x/Q_{d2}/(X/Q)_{d1}$=9.4E-04 may be used as default for dilution to site boundary for samples taken within work areas.

1.2.2 Design Basis Quantities

The design basis quantity (DBQ) of a radionuclide emitted to the atmosphere is the amount of that nuclide, when released in one year, which would result in a dose not exceeding any of the following:

a. The 10CFR50 Appendix I limit of 15 millirems to any organ of an individual from iodines and particulates with half-life greater than eight days.

b. Fifteen-millirad air dose for beta radiation from noble gas (this is conservatively less than the 10CFR50 Appendix I limit of 20 millirad).

c. Five-millirad air dose for gamma radiation from noble gas (this is conservatively less than the 10CFR50 Appendix I limit of 10 millirad).

Design basis quantity values corresponding to the above annual limits are provided in Table 2-9. Design basis quantity (Ci) is the smallest value for each nuclide, calculated by dividing the dose limits (as indicated in Sections 1.2.2.a through 1.2.2.c above) by the appropriate dose calculated in Section 1.2; the result then is multiplied by the amount of radionuclide (Ci) used to conservatively estimate the doses as listed in Table 2-1 (or assumed a hypothetical 1 Ci/year for nuclides not actually present):

$$DBQ = \frac{D_{AI}}{D_C}(C_C) \qquad (1.5)$$

Where:

D_{AI} = Appendix I dose limit for organ dose (mrem) and conservative values less than the Appendix I limits as specified in b. and c. above for beta and gamma air dose (mrad).

D_C = Calculated dose from GASPAR.

C_C = Quantity of nuclide resulting in dose D_C (Ci).

DBQ = Design Basis Quantity (Ci). The limiting values for Design Basis Quantities for radionuclides released to the atmosphere are given in Table 2-6.

The inverse of the ratio C_C/D_C in the above equation (i.e., D_C/C_C) is a useful value, since it represents the most limiting dose per unit quantity of each nuclide released.

1.3 DESIGN BASIS QUANTITY (DBQ) LIMITS

1.3.1 Design Basis Quantity Fraction

Per Specification 2.3 of ODCM, Section I, the cumulative DBQ fraction for nuclides released is summed at least every 31 days (or alternatively, GASPAR may be run) to assure that the sum of the fractions of all nuclides released does not exceed 1.0 year to date or 0.5 in any calendar quarter, and forecast not to exceed 0.02 over the next 31 days. Forecasts should be based on historical performance, consideration of upcoming scheduled tasks and operational parameters, or a combination of factors. The equation for forecasting the cumulative dose by means of DBQs is provided below.

$$\sum_i \frac{A_i}{(DBQ)_i} < 1.0\ \text{Annual}, \quad < 0.5\ \text{Quarter}, \quad < 0.02\ \text{Next 31Day} \qquad (1.6)$$

Where:

A_i = Activity of nuclide i

DBQ_i = Design Basis Quantity of nuclide i

1.3.2 Exceeding DBQ Limits

The DBQ is a very conservative estimate of activity which could give doses at 10CFR50, Appendix I limits. Because different organs are summed together and doses to different individuals are summed, the DBQ typically overestimates dose by about a factor of five. Thus, if calculations of the DBQ fraction exceed 1.0 for year to date, 0.5 for the quarter, or 0.02 forecast for the following 31 days, an accurate assessment of dose should be made by use of the NRC GASPAR computer code (either by running the code or by normalizing a prior run's dose output on a nuclide by nuclide basis). If GASPAR is not utilized to determine accurate doses, Actions of Specifications 2.3 or 3.3, ODCM, Section I, shall be performed based on the conservative DBQ fraction results of Equation 1.6.

1.3.3 Releasing Radionuclides not Listed in Table 2-6

Table 2-6 contains all nuclides identified to date as routine constituents of gaseous releases at BRP Plant, plus those common to boiling water reactors in general, even if not previously detected at BRP. From time to time, however, other nuclides may be detected.

If the unlisted nuclide constitutes less than ten percent of the EC-fraction for the release, and all unlisted nuclides total less than 25% of the EC-fraction, the nuclide may be considered not present. If the unlisted nuclide constitutes greater than ten percent of the EC-fraction, or all unlisted nuclides together constitute greater than 25%, then each nuclide should be assigned a DBQ equal to the most conservative value listed for the physical form of the nuclide involved (noble gas, halogen or particulate).

Should a nuclide not listed in Table 2-6 begin to appear in significant quantities (in concentrations greater than ten percent of its EC) on a routine basis, revision to this ODCM should be made in order to include a design basis quantity specific to that nuclide.

1.4 GASEOUS RADWASTE TREATMENT SYSTEM OPERATION

1.4.1 System Description

The gaseous radwaste system has been removed during the dismantlement process.

1.5 OFF-SITE RELEASE RATE

10CFR50.36a requires that the release of radioactive materials be kept as low as reasonably achievable. Appendix I to 10CFR50 provides the numerical guidelines on limiting conditions for operation to meet the as low as reasonably achievable requirement.

The GASPAR code has been run to determine a conservative relationship between release rate in Ci/sec and annual dose due to external radiation and inhalation. The source term used is listed in Table 2-1. Dose using annual average meteorology, to the individual with most limiting off-site dose (whole body) assumed to be residing at the residence with highest X/Q, is 0.105 mrem for one year. The release rate which would result in a dose equivalent of 500 mrem/y (using the total body limit upon which ODCM, Section I Requirement 3.1 is based) is the Curies/Year given in Table 2-1 (1.29E04) multiplied by 500/0.105 or 1.95 Ci/sec.

The above is a one-time calculation during power operation and is informational only. Methods of exposure calculations to be used for specific releases over specific periods are described in Section II, Part 1.2.

1.6 PARTICULATE SAMPLING

Particulate samples are obtained from the continuous air monitors and grab samples from another areas of potential airborne radioactivity utilized through the course of decommissioning for work area monitoring.

Gamma and beta counting is performed on particulate filters. Beta yields of the gamma isotopes detected on individual particulate filters at the time of sampling are applied to determine "identified" beta, and after suitable decay time for decay of radon and thoron daughters, the "identified" count rate is subtracted from the observed count rate to give "unidentified" beta. The "unidentified" beta is assumed to be Sr-90 until results of an optional analysis specific to Sr-90 (from a quarterly composite of filters) are obtained. Alternatively, Cs-137, if detected, may be utilized as a surrogate for Sr-90, with Sr-90 equal to 3.0% of the Cs-137 value.

1.7 NOBLE GAS MONITORING AND SAMPLING

Noble gas monitoring and sampling are no longer required, since all spent fuel is stored at the ISFSI facility in hermetically sealed containers.

1.8 TRITIUM SAMPLING

Tritium has a low dose consequence to the public because of the very small fraction of allowable quantity which is available for release. The major contributor to tritium gaseous effluents is evaporation from the barrels of groundwater that are evaporated rather than released as liquid, due to detectable gamma radioactivity levels (contamination sometimes occurs from contact with plant equipment and structural materials during dismantlement operations). Because of the low dose impact, and due to the continuous reduction in tritium source term since cessation of to reactor operation, gaseous tritium sampling will not be required. Gaseous tritium release is estimated using conservative evaporation rate calculations from the fuel pool (702 uCi/day) at a time when its water contained significant levels of tritium.

TABLE 2-1
BIG ROCK POINT GASEOUS AND LIQUID SOURCE TERMS, CURIES/YEAR [1]

Nuclide	Gaseous	Liquid	Nuclide	Gaseous	Liquid
H-3	1.21E+01	8.63E+00	Sb-124	NA	4.01E-04
N-13	1.53E+03	NA	I-131	1.94E-03	1.57E-04
Na-24	3.52E-04	1.12E-06	Xe-131m	4.38E-01	NA
Cr-51	2.82E-04	6.84E-03	I-132	8.07E-03	NA
Mn-54	5.50E-05	2.60E-02	I-133	1.99E-02	NA
Mn-56	1.70E-04	NA	Xe-133	2.01E+02	8.86E-05
Co-58	1.65E-06	6.17E-04	Xe-133m	6.00E+00	NA
Fe-59	2.81E-06	9.05E-03	Cs-134	4.04E-07	1.75E-02
Co-60	1.89E-04	4.21E-02	I-134	1.24E-02	NA
Zn-65	3.16E-05	9.01E-04	I-135	3.00E-02	NA
Br-82	8.11E-03	NA	Xe-135	1.11E+03	NA
Kr-83m	2.61E+02	NA	Xe-135m	1.15E+03	NA
Kr-85	9.55E-01	NA	Cs-136	4.74E-05	NA
Kr-85m	3.12E+02	NA	Cs-137	1.51E-04	2.04E-01
Kr-87	1.19E+03	NA	Xe-137	1.11E+03	NA
Kr-88	7.80E+02	NA	Cs-138	3.17E-01	NA
Kr-89	6.96E+02	NA	Xe-138	6.03E+03	NA
Sr-89	NA	2.27E-04	Ba-139	1.32E-03	NA
Kr-90	7.76E+02	NA	Xe-139	1.04E+03	NA
Sr-90	NA	2.22E-03	Ba-140	1.86E-03	NA
Kr-91	6.68E+00	NA	La-140	7.80E-03	5.04E-05
Sr-91	5.61E-03	NA	Xe-140	7.23E+01	NA
Sr-92	NA	1.54E-06	Hg-203	1.32E-06	NA
Nb-95	1.91E-06	NA	Np-239	1.44E-04	NA
Mo-99	3.10E-05	NA	Unidentified		
Ag110m	1.57E-05	6.88E-05	Beta	2.42E-03	6.76E-02

[1] Operating plant data derived from taking the effluents released during January-June 1980 through July-December 1983 and dividing by four. "NA" indicates that the nuclide was not detected in that effluent stream.

TABLE 2-2
BASIC RADIONUCLIDE DATA

	NUCLIDE	HALF-LIFE (days)	LAMBDA (l/s)	BETA [1] (MEV/DIS)	GAMMA [1] (MEV/DIS)
1	Tritium	4.49E 03	1.79E-09	5.68E-03	0.0
2	C-14	2.09E 06	3.84E-12	4.95E-02	0.0
3	N-13	6.94E-03	1.16E-03	4.91E-01	1.02E 00
4	O-19	3.36E-04	2.39E-02	1.02E 00	1.05E 00
5	F-18	7.62E-02	1.05E-04	2.50E-01	1.02E 00
6	Na-24	6.33E-01	1.273-05	5.55E-01	4.12E 00
7	P-32	1.43E 01	5.61E-07	6.95E-01	0.0
8	Ar-41	7.63E-02	1.05E-04	4.64E-01	1.28E 00
9	Cr-51	2.78E 01	2.89E-07	3.86E-03	3.28E-02
10	Mn-54	3.03E 02	2.65E-08	3.80E-03	8.36E-01
11	Mn-56	1.07E-01	7.50E-05	8.29E-01	1.69E 00
12	Fe-59	4.50E 01	1.78E-07	1.18E-01	1.19E 00
13	Co-58	7.13E 01	1.12E-07	3.41E-02	9.78E-01
14	Co-60	1.92E 03	4.18E-09	9.68E-02	2.50E 00
15	Zn-69m	5.75E-01	1.39E-05	2.21E-2	4.16E-01
16	Zn-69	3.96E-02	2.03E-04	3.19E-01	0.0
17	Br-84	2.21E-02	3.63E-04	1.28E 00	1.77E 00
18	Br-85	2.08E-03	3.86E-03	1.04E 00	6.60E-02
19	Kr-85m	1.83E-01	4.38E-05	2.53E-01	1.59E-01
20	Kr-85	3.93E 03	2.04E-09	2.51E-01	2.21E-03
21	Kr-87	5.28E-02	1.52E-04	1.32E 00	7.93E-01
22	Kr-88	1.17E-01	6.86E-05	3.61E-01	1.96E 00
23	Kr-89	2.21E-03	3.63E-03	1.36E 00	1.83E 00
24	Rb-88	1.24E-02	6.47E-04	2.06E 00	6.26E-01
25	Rb-89	1.07E-02	7.50E-04	1.01E 00	2.05E 00
26	Sr-89	5.20E 01	1.54E-07	5.83E-01	8.45E-05
27	Sr-90	1.03E 04	7.79E-10	1.96E-01	0.0
28	Sr-91	4.03E-01	1.99E-05	6.50E-01	6.95E-01
29	Sr-92	1.13E-01	7.10E-05	1.95E-01	1.34E 00
30	Sr-93	5.56E-03	1.44E-03	9.20E-01	2.24E 00
31	Y-90	2.67E 00	3.00E-06	9.36E-01	0.0
32	Y-91m	3.47E-02	2.31E-04	2.73E-02	5.30E-01
33	Y-91	5.88E 01	1.36E-07	6.06E-01	3.61E-03
34	Y-92	1.47E-01	5.46E-05	1.44E 00	2.50E-01
35	Y-93	4.29E-01	1.87E-05	1.17E 00	8.94E-02
36	Zr-95	6.50E 01	1.23E-07	1.16E-01	7.35E-01
37	Nb-95m	3.75E 00	2.14E-06	1.81E-01	6.06E-02
38	Nb-95	3.50E 01	2.29E-07	4.44E-02	7.64E-01

TABLE 2-2
BASIC RADIONUCLIDE DATA

	NUCLIDE	HALF-LIFE (days)	LAMBDA (l/s)	BETA [1] (MEV/DIS)	GAMMA [1] (MEV/DIS)
39	Mo-99	2.79E 00	2.87E-06	3.96E-01	1.50E-01
40	Tc-99m	2.50E-01	3.21E-05	1.56E-02	1.26E-01
41	Tc-99	7.74E 07	1.04E-13	8.46E-02	0.0
42	Tc-104	1.25E-02	6.42E-04	1.60E 00	1.95E 00
43	Ru-106	3.67E 02	2.19E-08	1.01E-02	0.0
44	Te-132	3.24E 00	2.48E-06	1.00E-01	2.33E-01
45	I-129	6.21E 09	1.29E-15	5.43E-02	2.46E-02
46	I-131	8.05E 00	9.96E-07	1.90E-01	3.81E-01
47	I-132	9.58E-02	8.37E-05	4.89E-01	2.24E 00
48	I-133	8.75E-01	9.17E-06	4.08E-01	6.02E-01
49	I-134	3.61E-02	2.22E-04	6.16E-01	2.59E 00
50	I-135	2.79E-01	2.87E-05	3.68E-01	1.55E 00
51	Xe-131m	1.18E 01	6.80E-07	1.43E-01	2.01E-02
52	Xe-133m	2.26E 00	3.55E-06	1.90E-01	4.15E-02
53	Xe-133	5.27E 00	1.52E-06	1.35E-01	4.60E-02
54	Xe-135m	1.08E-02	7.43E-04	9.58E-02	4.32E-01
55	Xe-135	3.83E-01	2.09E-05	3.17E-01	2.47E-01
56	Xe-137	2.71E-03	2.96E-03	1.77E 00	1.88E-01
57	Xe-138	9.84E-03	8.15E-04	6.65E-01	1.10E 00
58	Cs-134	7.48E 02	1.07E-08	1.63E-01	1.55E 00
59	Cs-135	1.10E 09	7.29E-15	5.63E-02	0.0
60	Cs-136	1.30E 01	6.17E-07	1.37E-01	2.15E 00
61	Cs-137	1.10E 04	7.29E-10	1.71E-01	5.97E-01
62	Cs-138	2.24E-02	3.58E-04	1.20E 00	2.30E 00
63	Ba-139	5.76E-02	1.39E-04	8.96E-01	3.53E-02
64	Ba-140	1.28E 01	6.27E-07	3.15E-01	1.71E-01
65	La-140	1.68E 00	4.77E-06	5.33E-01	2.31E 00
66	Ce-144	2.84E 02	2.82E-08	9.13E-02	1.93E-02
67	Pr-143	1.36E 01	5.90E-07	3.14E-01	0.0
68	Pr-144	1.20E-02	6.68E-04	1.21E 00	3.18E 02
69	Eu-152	4.84E 03	1.66E-09	8.22E-02	1.06E 00

[1] Average energy per disintegration values were obtained from ICRP Publication Number 38, Radionuclide Transformations: Energy and Intensity of Emissions, 1983 and NUREG/CR-1413 (ORNL/NUREG-70), A Radionuclide Decay Data Base - Index and Summary Table, D.C. Kocher, May 1980.

TABLE 2-3

USNRC COMPUTER CODE – XOQDOQ, VERSION 2.0 RUN DATE: 940629
**** BIG ROCK POINT X0QD0Q82 *** USING 01/01/89 – 12/31/93 MET DATA ****
ELEVATED RELEASE – 240' STACK

RELEASE ID	TYPE OF LOCATION	DIRECTION	DISTANCE FROM SITE (MILES)	(METERS)	X/Q (SEC/M³) NO DECAY UNDEPLETED	X/Q (SEC/M³) 2.26 D DECAY UNDEPLETED	X/Q (SEC/M³) 8.0 D DECAY DEPLETED	D/Q (PER SQ. METER)
A	SITE BOUNDARY	E	0.57	917.	4.91E-08	4.90E-08	4.85E-08	1.25E-09
A	SITE BOUNDARY	ESE	0.52	837.	4.11E-08	4.10E-08	4.07E-08	1.16E-09
A	SITE BOUNDARY	SE	0.55	885.	3.09E-08	3.08E-08	3.05E-08	8.19E-10
A	SITE BOUNDARY	SSE	0.58	933.	2.25E-08	2.24E-08	2.22E-08	5.12E-10
A	SITE BOUNDARY	S	0.68	1094.	2.07E-08	2.06E-08	2.03E-08	3.68E-10
A	SITE BOUNDARY	SSW	0.71	1143.	1.95E-08	1.94E-08	1.91E-08	3.65E-10
A	SITE BOUNDARY	SW	0.50	805.	3.58E-09	3.57E-09	3.55E-09	1.16E-10
A	MAXIMUM CHI/Q	S	2.00	3219.	3.37E-08	3.35E-08	3.31E-08	1.34E-10
A	MAXIMUM CHI/Q	SSW	2.00	3219.	2.73E-08	2.71E-08	2.67E-08	1.31E-10
A	MAXIMUM CHI/Q	SW	2.50	4023.	1.85E-08	1.84E-08	1.82E-08	3.84E-11
A	MAXIMUM CHI/Q	WSW	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	W	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	WNW	15.00	24140.	1.12E-07	7.29E-08	8.26E-08	1.30E-10
A	MAXIMUM CHI/Q	NW	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	NNW	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	N	50.00	80467.	0.00E+00	0.00E+00	0.00E+00	0.00E+00
A	MAXIMUM CHI/Q	NNE	2.00	3219.	4.47E-08	4.46E-08	4.43E-08	1.90E-10
A	MAXIMUM CHI/Q	NE	1.50	2414.	4.63E-08	4.62E-08	4.54E-08	5.46E-10
A	MAXIMUM CHI/Q	ENE	1.50	2414.	5.32E-08	5.30E-08	5.22E-08	5.79E-10
A	MAXIMUM CHI/Q	E	1.00	1609.	5.40E-08	5.39E-08	5.30E-08	8.96E-10
A	MAXIMUM CHI/Q	ESE	0.75	1207.	4.29E-08	4.28E-08	4.21E-08	9.15E-10
A	MAXIMUM CHI/Q	SE	1.50	2414.	3.40E-08	3.38E-08	3.32E-08	3.60E-10
A	MAXIMUM CHI/Q	SSE	1.50	2414.	3.59E-08	3.58E-08	3.52E-08	2.57E-10

VENT AND BUILDING PARAMETERS:
RELEASE HEIGHT (METERS) 73.10 REP. WIND HEIGHT (METERS) 71.3

NOTE: Big Rock Point meteorological data was gathered from sensors mounted on the 73-meter stack. Sensors were mounted into the prevailing wind direction. Because of interference to the wind flow by the stack when winds were from the 71E to 159E (east, towards Lake Michigan), the meteorological data recorded in these sectors are considered invalid. For dose calculational purposes, this effectively invalidates six (6) lakeward sectors (WSW, W, WNW, NW, NNW, and N). Therefore zeros are recorded in Table 2-3 for these sectors. However, the program which calculates the annual average X/Q (Chi/Q) requires input of the full years met data. Any data recorded for these six sectors are input in the WNW sector to satisfy the program. Values of X/Q listed in the WNW sector are invalid.

TABLE 2-4
CONSERVATIVE BIG ROCK POINT GASPAR INPUT PARAMETERS

Critical Receptors

Location	Release Type	Sector	Distance (miles)	X/Q (sec/m^3)	X/Q Decay (sec/m^3)	X/Q Decay and Dep (sec/m^3)	D/Q ($1/m^2$)
Residence/	Stack	E	0.57	4.91E-08	4.90E-08	4.85E-08	1.25E-09
Garden	Ground	E	0.57	8.10E-08	8.09E-08	8.00E-08	2.06E-09
Site	Stack	E	0.57	4.91E-08	4.90E-08	4.85E-08	1.25E-09
Boundary	Ground	E	0.57	8.10E-08	8.09E-08	8.00E-08	2.06E-09
Beef Cattle	Stack	E	0.57	4.91E-08	4.90E-08	4.85E-08	1.25E-09
	Ground	E	0.57	8.10E-08	8.09E-08	8.00E-08	2.06E-09
Dairy Cow	Stack	E	0.57	4.91E-08	4.90E-08	4.85E-08	1.25E-09
	Ground	E	0.57	8.10E-08	8.09E-08	8.00E-08	2.06E-09

NOTE: Long term X/Q and D/Q values for ground level release have been calculated for this table and for Figure 2-2 by multiplying the ratio of the elevated (stack) long term to short term X/Q at 0.57 miles times the short term ground level X/Q at various distances (in Table 2-4 above only at 0.57 miles).

Final (1998) Land Use Census illustrates actual land use:

Sector	Residence	Garden	Dairy Cow	Beef Cattle	Goat
WSW	2.5 mi	>5 mi	>5 mi	>5 mi	>5 mi
SW	1.1 mi	2.7 mi	>5 mi	>5 mi	>5 mi
SSW	1.3 mi	>5 mi	>5 mi	>5 mi	>5 mi
S	1.9 mi	2.1 mi	>5 mi	>5 mi	>5 mi
SSE	1.7 mi	1.7 mi	>5 mi	1.7 mi	>5 mi
SE	1.8 mi	1.8 mi	4.5 mi	1.7 mi	>5 mi
ESE	1.5 mi	1.8 mi	>5 mi	3.2 mi	>5 mi
E	1.4 mi	2.4 mi	3.5 mi	3.2 mi	>5 mi
ENE	2.3 mi	>5 mi	>5 mi	>5 mi	>5 mi

TABLE 2-5
BIG ROCK POINT SITE-SPECIFIC PARAMETERS FOR GASPAR

891.0	=	Meters from facility to the NE corner of US 31*
0.33	=	Fraction of year leafy vegetables are grown
0.58	=	Fraction of year cows are on pasture
0.76	=	Default value fraction of crop from garden
1.0	=	Default value fraction of daily intake of cows, goats and beef cattle derived from pasture while on pasture
2,160	=	Hours hold-up for stored feed
50	=	Kg/d feed consumed by cow and beef cattle
0.67	=	Fraction of year goats are on pasture
6.0	=	Kg/d feed consumed by goat
0.58	=	Fraction of year beef cattle are on pasture
2.0	=	Days transport time from feed to milk receptor

*NOTE: Independent Spent Fuel Storage Installation is located 300 meters from site boundary, but source of gaseous effluents is plant dismantlement, not release from ISFSI. Fuel containers at ISFSI are hermetically sealed, and provide no detectable gaseous effluent.

TABLE 2-6
BIG ROCK POINT GASEOUS DESIGN OBJECTIVE ANNUAL QUANTITIES BASED ON TABLE 2-4 CRITICAL RECEPTORS FOR STACK RELEASE

Nuclide	Organ	Dose Factor mrem/Ci	Design Objective Annual Quantity (Ci)
H-3	Total Body - C	1.06E-05	4.72E+05
C-14	Bone – C	8.70E-03	1.72E+03
Cr-51	GI Tract - T	7.03E-04	2.13E+04
Mn-54	GI Tract – T	9.00E-02	1.67E+02
Fe-55	Bone – C	4.15E-02	3.61E+02
Co-58	GI Tract – T	4.25E-02	3.53E+02
Fe-59	GI Tract – T	7.37E-02	2.04E+02
Co-60	GI Tract – T	1.00E+00	1.50E+01
Zn-65	Total Body - C	2.82E-01	1.77E+01
Sr-89	Bone – C	1.42E+00	1.06E+01
Sr-90	Bone – C	5.55E_01	2.70E-01
Zr-95	GI Tract – T	8.85E-02	1.69E+02
Sb-124	GI Tract - T	1.59E-01	9.43E+01
Ar-41	Total Body	9.07E-06	5.51E+05
Kr-83m	Skin	2.20E-08	6.82E+08
Kr-85	Skin	2.21E-06	6.79E+06
Kr-85m	Total Body	1.24E-06	4.03E+06
Kr-87	Skin	2.07E-05	7.25E+05
Kr-88	Total Body	1.54E-05	3.25E+05
Kr-89	Total Body	2.27E-06	2.20E+06
Xe-131m	Skin	9.29E-07	1.16E+07
Xe-133	Total Body	3.20E-07	1.56E+07
Xe-133m	Skin	1.94E-06	7.73E+06
Xe-135	Total Body	1.95E-06	2.56E-06
Xe-135m	Total Body	2.22E-06	2.25E+06
Xe-137	Skin	3.80E-06	3.95E+06
Xe-138	Total Body	6.02E-06	8.31E+05
I-131	Thyroid - I	1.11E+01	1.35E+00
I-133	Thyroid – I	1.08E-01	1.39E+02
Cs-134	Liver – C	2.10E+00	7.14E+00
Cs-136	Total Body - I	5.04E-02	9.92E+01
Cs-137	Bone – C	2.14E+00	7.01E+00
Ba-140	Bone – C	107E-02	1.40E+03
Ce-141	GI Tract - T	2.10E-02	7.15E+02
Ce-144	GI Tract – T	5.02E-01	2.99E+01
N-13	Total Body – C	6.81E-08	7.34E+07
Na-24	Total Body - I	8.19E-04	6.11E+03
Mn-56	GI Tract – C	2.16E-04	6.94E+04
Co-57	GI Tract – T	1.47E-02	1.02E+03
Ni-63	Bone – C	2.54E+00	8.45E+00
Ni-65	GI Tract – T	1.35E-04	1.11E+05

TABLE 2-6
BIG ROCK POINT GASEOUS DESIGN OBJECTIVE ANNUAL QUANTITIES BASED ON TABLE 2-4 CRITICAL RECEPTORS FOR STACK RELEASE

CAUTION: To obtain ground-level release dose factors, multiply Table 2-6 values by 1.65.

To obtain ground-level release design objective annual quantities, multiply by 0.61.

Nuclide	Organ	Dose Factor mrem/Ci	Design Objective Annual Quantity (Ci)
Br-82	Total Body – I	4.73E-03	1.06E+03
Rb-88	Total Body – C	1.29E-06	3.88E+06
Sr-91	Bone – I	6.23E-02	2.41E+02
Sr-92	GI Tract – C	3.88E-04	3.87E+04
Nb-95	GI Tract – A	1.79E-01	8.38E+01
Mo-99	Kidney – I	6.81E-03	2.20E+03
Tc-99	GI Tract – A	5.47E-01	2.74E+01
Tc-99m	GI Tract - T	1.67E-05	8.98E+05
Ru-103	GI Tract – A	2.66E-01	5.64E+01
Ru-105	GI Tract – C	1.74E-04	8.62E+04
Sb-125	GI Tract – T	7.05E-02	2.13E+02
Te-127	GI Tract – T	1.31E-04	1.15E+05
I-129	Thyroid – A	7.41E+01	2.02E-01
I-132	Thyroid - C	3.09E-04	4.85E+04
I-134	Thyroid – C	7.69E-05	1.95E+05
I-135	Thyroid - C	1.40E-03	1.07E+04
La-140	GI Tract – A	2.42E-03	6.20E+03
Tc-101	GI Tract - I	8.94E06	1.68E+06
Ag-110m	GI Tract – T	9.17E-01	1.64E+01
Cs-138	Total Body - C	1.16E-02	4.31E+02
Ba-139	GI Tract – C	8.59E-05	1.75E+05
Eu-152	Skin	8.19E-01	1.83E+01
Np-239	GI Tract – A	6.27E-04	2.39E+04
Pu-238	Bone – T	3.75E+01	4.00E-01
Pu-239	Bone – T	4.34E+01	3.46E-01
Pu-241	Bone – T	9.17E-01	1.64E+01
Am-241	Bone – T	1.51E+01	9.93E-01
Cm-242	Lung - T	8.32E-01	1.80E+01
Cm-244	Bone – T	9.70E+00	1.54E+00

NOTE: Ground-level releases result in doses that are 1.65 times the doses from stack releases.

2.0 LIQUID EFFLUENTS

2.1 ALLOWABLE CONCENTRATION

2.1.1 ODCM Requirement

Requirement 2.2.2 of ODCM, Section I specifies that the concentration of radioactive material released at any time from the site to unrestricted areas shall be limited to ten times the effluent concentration specified in 10CFR20, Appendix B, Table 2, Column 2 for nuclides other than dissolved or entrained noble gases. Plant effluents contain no more dissolved or entrained noble gases. The following approach will be used.

2.1.2 Pre-release Analysis

Prior to first release for a season from water detained for sedimentation, a grab sample will be obtained and analyzed.

2.1.3 Total Release-fraction

The total release-fraction (R_j) will be calculated by the relationship defined as:

$$R_j = \sum_{i=1}^{n} \frac{C_{ij}}{10 \times EC_i} \qquad (2.1)$$

Where:

C_{ij} = Undiluted effluent concentration of radionuclide i, as determined in Section II, Part 2.1.2, µCi/ml, at release point j.

EC_i = The 10CFR20 effluent concentration limit (EC) of radionuclide i, as specified in Section II, Part 2.1.1, µCi/ml (Big Rock Point still uses MPC terminology in some Liquid Analyses Programs, although EC values are utilized).

R_j = The total release-fraction for the release point.

2.2 INSTRUMENT ALARM SET POINTS

No installed monitors are required.

2.3 LIQUID EFFLUENT DOSE CALCULATION

2.3.1 Applicability

This section provides methodology for calculating doses in compliance with Section I, Requirement 2.3.2.

2.3.2 Calculation of Design Basis Quantity Fraction

Calculations shall be performed according to the formula:

$$\sum_i \frac{A_i}{(DBQ)_i} = \text{Fraction of DBQ} \qquad (2.2)$$

Where:

A_i = Cumulative or projected 31-day, quarterly or annual activity of nuclide i identified or expected in liquid release (Ci).

$(DBQ)_i$ = Design objective annual quantity of radionuclide i from Table 2-8 (Ci).

2.3.3 Use of LADTAP

As an alternative to use of the DBQ fraction, the NRC LADTOP code may be run to confirm that ODCM, Requirement 2.3.2 of ODCM, Section I has been met.

2.3.4 Dose Calculation

Values for the design basis quantities (Ci), and the dose per curie $(D_C/C_C)_i$ for each nuclide i shown in Table 2-8, were calculated by use of LADTAP. Site-specific values for LADTAP runs are found in Tables 2-8 and 2-9.

F = Amount of fish eaten per year; adult - 21 kg, teen - 16 kg, child - 6.9 kg and infant - none.

15 = Dispersion factor from discharge to fish exposure point.

Releasing Radionuclides Not Listed in Table 2-9

Table 2-9 contains all nuclides identified to date as routine constituents of liquid releases at BRP Plant, plus those common to boiling water reactors in general, even if not previously detected at BRP. From time to time, however, other nuclides may be detected.

If the unlisted nuclide constitutes less than ten percent of the EC-fraction for the release, and all unlisted nuclides total less than 25% of the EC-fraction, the nuclide may be considered not present.

If the unlisted nuclide constitutes greater than ten percent of the EC-fraction, or all unlisted nuclides together constitute greater than 25%, then each nuclide should be assigned a DBQ equal to the most conservative value listed for the physical form of the nuclide involved.

Should a nuclide not listed in Table 2-9 begin to appear in significant quantities on a routine basis, revision to this ODCM should be made in order to include a design basis quantity specific to that nuclide.

2.3.5 Annual Analysis

A complete analysis utilizing the NRC computer code LADTAP (either by running the code, or by normalization of data to results of a previous year's run) for the total release source will be done annually in conjunction with the annual environmental report. This analysis will provide estimates of dose to the total body and various organs in addition to the dose limiting organs.

2.4 FUNCTIONALITY OF LIQUID RADWASTE EQUIPMENT

All installed liquid radwaste systems have been removed from service for dismantlement.

2.5 OFF-SITE RELEASE RATE

10CFR50.36a requires that the release of radioactive materials be kept as low as is reasonably achievable. Appendix I to 10CFR50 provides the numerical guidelines on limiting conditions for operations to meet the as low as is reasonably achievable requirement.

Table 2-9 specifies a Design Basis Annual Quantity for the limiting radionuclide currently present on-site (Cs-137) equal to 6.62E-03 Ci/y, or release rate of 18.4 uCi/day. The other two radionuclides currently detected in plant demolition debris are Co-60 and Eu-152. Both of these radionuclides have Design Basis Annual Quantities approximately order of magnitude higher Cs-137.

Annual analyses are run for the report specified in Part 1.5 of ODCM, Section III. LADTAP is used to calculate estimates of dose to the total body and limiting organs.

Radionuclides of highest dose consequence will remain predominately Cs-137 and Co-60 throughout the decommissioning interval. Iodine-131 and Cs-134, which have been important dose contributors during power operations, no longer are present in effluents in detectable concentrations due to decay.

TABLE 2-7
BIOACCUMULATION FACTORS
(μCi/g per μCi/ml)

Element	Freshwater Fish
H	9.0E-01
C	4.6E+03
Na	1.0E+02
P	1.0E+03
Cr	2.0E+02
Mn	4.0E+02
Fe	1.0E+02
Co	5.0E+01
Ni	1.0E+02
Cu	5.0E+01
Zn	2.0E+03
Br	4.2E+02
Rb	2.0E+03
Sr	3.0E+01
Y	2.5E+01
Zr	3.3E+00
Nb	3.0E+04
Mo	1.0E+01
Tc	1.5E+01
Ru	1.0E+01
Rh	1.0E+01
Te	4.0E+02
I	1.5E+01
Cs	2.0E+03
Ba	4.0E+00
La	2.5E+01
Ce	1.0E+00
Pr	2.5E+01
Nd	2.5E+01
W	1.2E+03
Np	1.0E+01

TABLE 2-8
BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

Design objective annual quantities for liquid effluents were calculated utilizing the computer code LADTAP, a program for calculating radiation exposure to man from routine releases of nuclear reactor liquid effluents (Reference NUREG/CR-1276).

Input parameters used are as follows:

Pathway	Age Group	Usage	Dilution	Process Times (Hr)
Fish	Adult	21.0 kg/yr	15.0	24.0
	Teen	16.0	15.0	24.0
	Child	6.9	15.0	24.0
	Infant	0.0	15.0	24.0
Drinking	Adult	730.0 L/yr	800.0	4.6
	Teen	510.0	800.0	4.6
	Child	510.0	800.0	4.6
	Infant	330.0	800.0	4.6
Shoreline	Adult	12.0 hr/yr	2.0	0.0
	Teen	67.0	2.0	0.0
	Child	14.0	2.0	0.0
	Infant	0.0	2.0	0.0
Swimming	Adult	12.0 hr/yr	2.0	0.0
	Teen	67.0	2.0	0.0
	Child	14.0	2.0	0.0
	Infant	0.0	2.0	0.0
Boating	Adult	100.0 hr/yr	15.0	0.0
	Teen	100.0	15.0	0.0
	Child	50.0	15.0	0.0
	Infant	0.0	15.0	0.0

The usage figures are obtained from Regulatory Guide 1.109 and are default values. Dilutions and the process time for drinking water were taken from the NUS study dated June 4, 1976. Including the process times shown in this table, total transit time for sport fish is 168 hours, commercial fish is 240 hours, and drinking water is 28.6 hours.

TABLE 2-8

BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP

The following input parameters are used when running LADTAP for BRP:

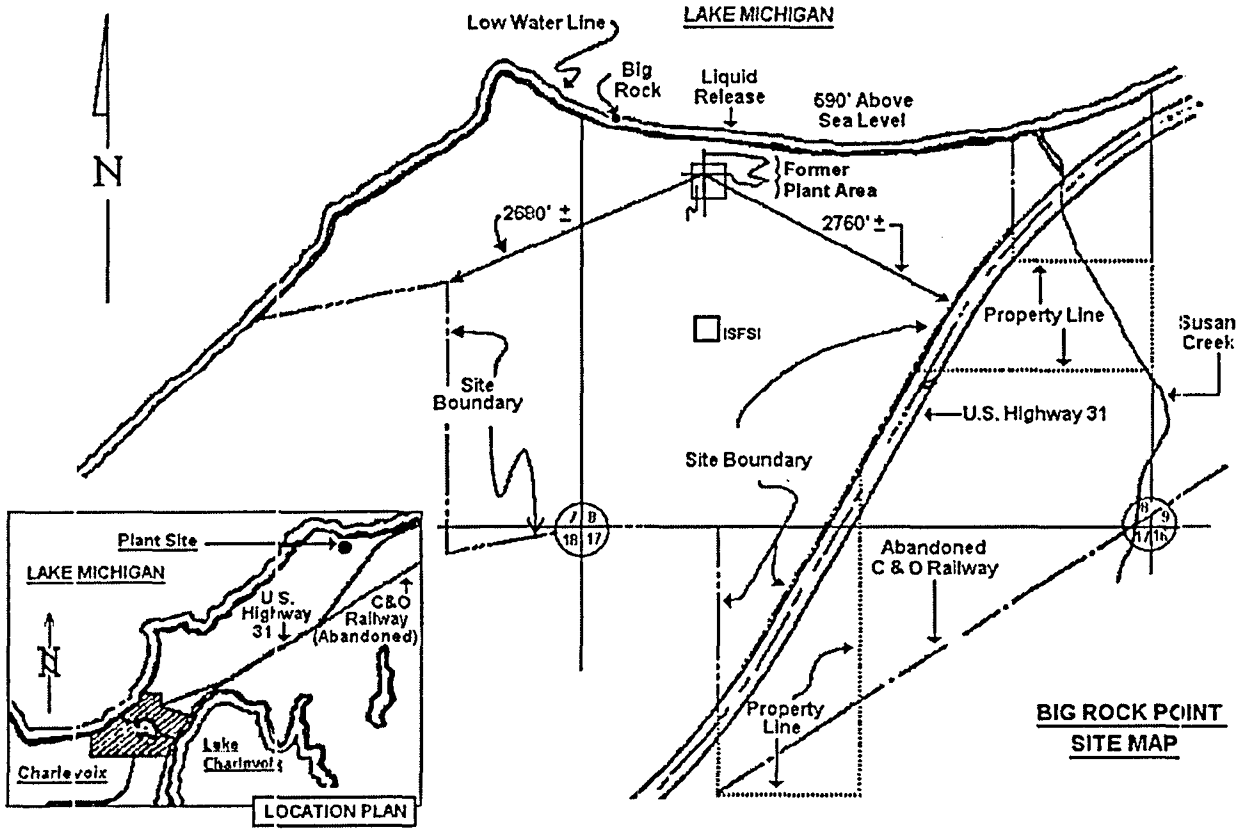
1. 50-mile population - 1.94E05
2. Shore width factor - 0.3
3. Total discharge (ft^3/sec) – Actual value
4. Transit time for all pathways – See note for Table 2-2.
5. Sport fish harvest (kg/yr) - 3.29E05
6. Commercial fish harvest (kg/yr) - 1.70E05
7. Invertebrate and algae consumption - 0
8. Drinking water population - 7.07E03
9. Shoreline population usage (man-hours) - 3.8E07
10. Swimming population usage (man-hours) - 1.2E07
11. Boating population usage (man-hours) - 3.7E07
12. Dilution factor to drinking water intake - 800

TABLE 2-9
BIG ROCK POINT DESIGN OBJECTIVE ANNUAL QUANTITIES FOR LIQUID EFFLUENTS AS DETERMINED BY LADTAP*

Nuclide	Dose Conversion Factors (mrem/Ci)	Individual/Organ	Design Objective Annual Quantity (Curies)
H-3	5.11E-04	Adult/TB	5.87E+03
Sc-46	2.71E+00	Teen/TB	1.11E+00
Mn-54	1.83E+01	Adult/GI(LLI)	5.47E-01
Fe-55	1.20E-00	Child/Bone	8.36E00
Co-57	6.12E-01	Teen/TB	4.91E00
Co-58	1.52E+00	Teen/TB	1.97E00
Co-60	6.33E+01	Teen/TB	4.74E-02
Zn-65	4.67E+01	Child/TB	6.43E-02
Sr-89	4.21E+01	Child/Bone	2.37E-01
Sr-90	7.29E+02	Adult/Bone	1.38E-02
Zr-95	8.34E-01	Teen/TB	3.60E+00
Ag-110m	1.04E+01	Teen/TB	2.90E-01
Cd-113m	1.61E+01	Adult/GI(LLI)	6.23E-01
Sb-124	2.04E+00	Teen/TB	1.48E+00
Sb-125	6.84E+00	Teen/TB	3.39E-01
Te-127m	3.73E+01	Teen/Kidney	2.68E+01
Cs-134	7.63E+02	Adult/TB	3.93E-03
Cs-137	4.54E+02	Adult/TB	6.62E-03
Ce-144	8.91E-01	Adult/GI(LLI)	1.12E+01
Eu-152	4.33E+01	Teen/TB	6.91E-02

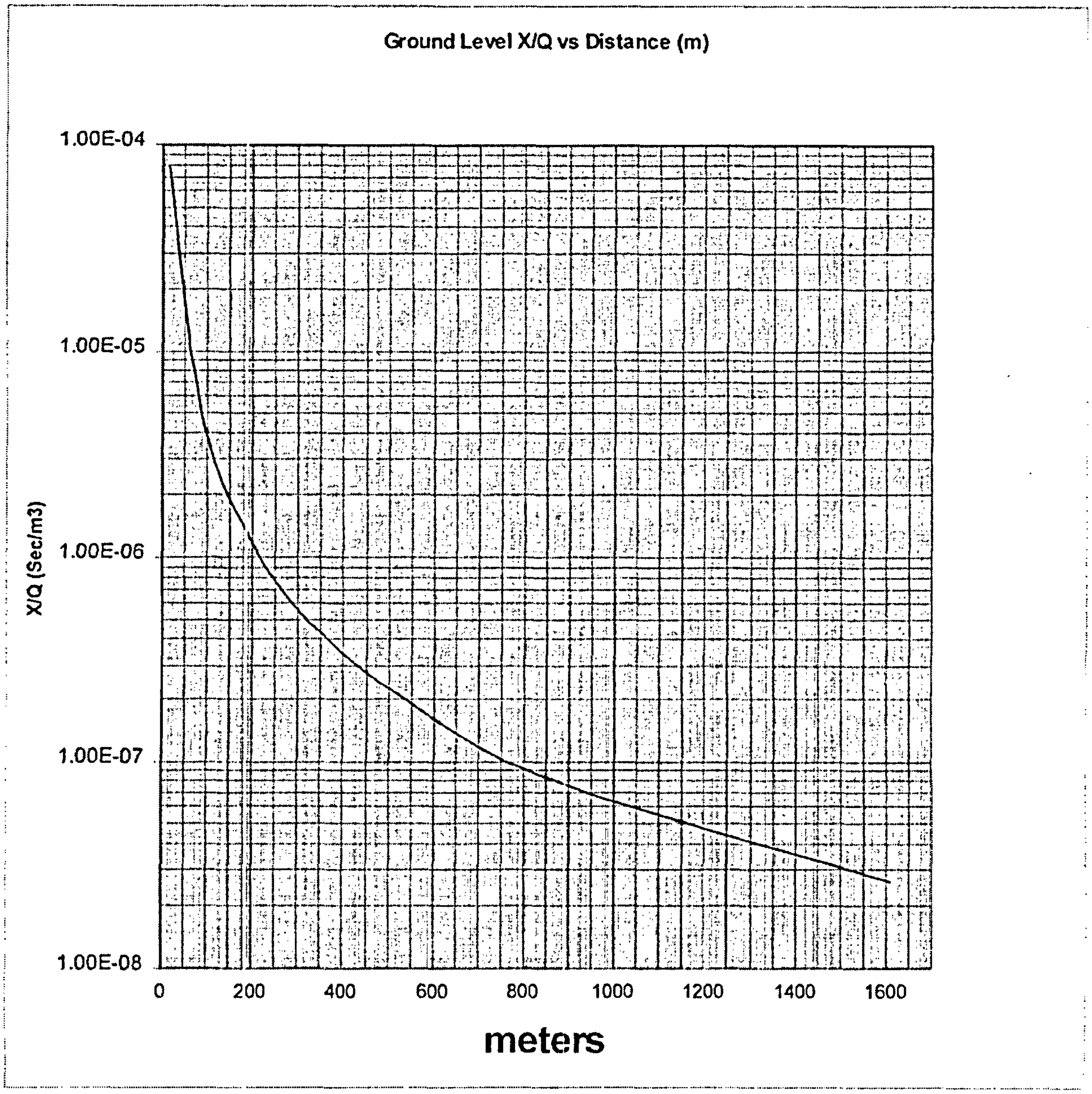
* Based on a current average discharge flow rate of 224 gpm.

FIGURE 2-1



* Gaseous release point is at ground level (590' above sea level) now that the stack has been removed for dismantlement.

FIGURE 2-2
GROUND LEVEL X/Qs FOR GROUND LEVEL RELEASE CALCULATIONS



3.0 CALCULATION OF URANIUM FUEL CYCLE DOSE

3.1 APPLICABILITY

Uranium Fuel Cycle dose is calculated when effluent doses exceed twice the levels specified in either ODCM, Section I, Requirement 1.3.2 or 3.2.2.

3.2 ASSUMPTIONS

3.2.1 The full time resident determined to be the maximally exposed individual (excluding infant) is assumed also to be a fisherman. This individual is assumed to drink water and ingest local fish at the rates indicated by the input parameters are summarized in Table 2-8.

3.2.2 Amount of shoreline fishing at accessible shoreline adjacent to the site is conservatively assumed as 48 hours per quarter (average of approximately 1/2 hour per day each day of the quarter) for the second and third quarters of the year, 36 hours for the fourth quarter and 18 hours for the first quarter.

3.2.3 The dose contribution due to uranium fuel cycle sources other than the plant (but including ISFSI) is ignored in the calculation. This is based on the lack of any operations that fall in the "cycles" definition within a five-mile radius of BRP.

3.3 DOSE CALCULATION

Maximum doses to the total body and internal organs of an individual shall be determined by use of LADTAP and GASPAR computer codes or optionally, by conservatively multiplying the mrem/Ci factors of Tables 2-6 and 2-9 by quantity released. Doses to like organs and total body shall be summed (organs are conservatively summed with total body when Tables 2-6 and 2-9 are utilized). Added to this sum will be a mean dose rate, calculated or measured for the shoreline due to the plant (ISFSI) during the quarter in question, times the assumed fishing time:

NOTE: For this calculation, the total body is conservatively assumed to be an additional organ.

$$D_{40_i} = D_G + D_L + (R_T)(T) \tag{3.1}$$

Where:

D_{40_i} = 40CFR190 dose to organ (i) (mrem).

D_G = Cumulative dose to an individual organ from gaseous releases (mrem).

D_L = Cumulative dose to an individual organ from liquid releases (mrem).

R_T = Mean dose rate (direct radiation component) calculated to be applicable to Lake Michigan shoreline adjacent to plant site (mrem/hr).

T = Assumed shoreline fishing time for the quarter in question (hours) (see 3.2.2 above).

ODCM SECTION III

REPORTING AND MAJOR MODIFICATION REQUIREMENTS

1.0 RADIOLOGICAL EFFLUENT RELEASE REPORT

The Radioactive Effluent Release Report shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and Process Control Program, and (2) in conformance with 10CFR50.36a and Section IV.B.1 of Appendix I to 10CFR50.

1.1 ESTIMATE OF UNCERTAINTY

The report shall include an estimate of the uncertainty associated with the measurement of radioactive effluents. This error term is included to provide an estimate of the uncertainty and is not to be considered the absolute error associated with the measurements or to be used in determining compliance with these requirements.

Estimates for liquid releases will be based on a statistical analysis of a series of sample results (weighed appropriately for counting statistics) taken once a year from a minimum of one typical liquid radwaste tank or other liquid effluent source. The error term for particulates released to the atmosphere will be based on the counting statistics for one air particulate sample taken during the year.

The report shall include an estimate of the lower level of detection (in uCi/ml) if the unidentified portion of the release exceeds ten percent of the total annual releases. This estimate of the lower level of detection will be made for gamma emitting isotopes listed in Appendix B of Regulatory Guide 1.21 (June 1974) and will be provided based on a typical background spectrum.

1.2 SUPPLEMENTAL INFORMATION

a. Batch Releases

Although both liquid and gaseous releases occur with variable rates and have variable durations, they shall be treated as continuous releases, rather than batches.

b. Abnormal Releases

The number of abnormal releases and number of curies of radioactive material released to the environment during abnormal releases should be reported.

1.3 GASEOUS EFFLUENTS

a. Gases (all plant contributions are zero).

b. Iodines (all plant contributions are zero).

c. Particulates

1) Total curies of radioactive material in particulate form with half-lives greater than eight days determined to have been released.

2) Average release rate (uCi/s) of radioactive material in particulate form with half-lives greater than eight days.

3) Percent of limit for radioactive material in particulate form with half-lives greater than eight days.

4) Total curies for each of the radionuclides in particulate form determined to be released based on analyses performed.

5) Total curies of gross alpha radioactivity of plant origin determined to be released.

d. Tritium

1) Total curies of tritium estimated to be released in gaseous effluent.

2) Average release rate (uCi/s) of tritium.

3) Percentage of applicable limits for tritium.

1.4 LIQUID EFFLUENTS

a. Mixed Fission and Activation Products

1) Total curies of radioactive material determined to be released in liquid effluents (not including tritium, dissolved and/or entrained gases, and alpha-emitting material).

2) Average concentrations (uCi/ml) of mixed fission and activation products released to unrestricted areas, averaged over the quarterly periods covered by the report.

3) Percent of applicable limit of average concentrations released to unrestricted areas.

4) Quarterly sums of total curies for each of the radionuclides determined to be released in liquid effluents based on analyses performed.

b. Tritium

1) Total curies of tritium determined to be released in liquid effluents.

2) Average concentrations (uCi/ml) of tritium released to unrestricted areas, averaged over the quarterly periods covered by the report.

3) Percent of applicable limit of average concentrations released to unrestricted areas.

c. Dissolved and/or Entrained Gases (all plant contributions are zero)

d. Alpha Radioactivity

Total curies of gross alpha-emitting materials of plant origin determined to be released in liquid effluents.

e. Volumes

1) Total measured volume (liters), prior to dilution, of liquid effluent released.

2) Total determined volume, in liters, of dilution water used during the period of the report.

1.5 SOLID WASTE

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10CFR61) shipped off-site during the report period:

a. Container burial volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Source of waste and processing employed (i.e., dewatered spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (i.e., LSA, Type A, Type B, Large Quantity); and

f. Solidification agent or absorbent (i.e., cement, asphalt).

1.6 RADIOLOGICAL IMPACT ON MAN

The Radioactive Effluent Release Report shall include potential doses to individuals and populations calculated using measured effluent and averaged meteorological data in accordance with the methodologies of ODCM, Section II.

a. Total body and significant organ doses (greater than 1 millirem) to individuals in restricted areas from receiving water-related exposure pathways.

b. Maximum off-site air doses greater than one millirad due to beta and gamma radiation at locations near ground level from gaseous effluents.

c. Organ doses greater than one millirem to individuals in unrestricted areas from radioactive iodine and radioactive material in particulate form from the major pathways of exposure.

d. Total body doses greater than one manrem to the population and average doses greater than one millirem to individuals in the population from receiving water-related pathways to a distance of 50 miles from the site.

e. Total body doses greater than one manrem to the population and average doses greater than one millirem to individuals in the population from gaseous effluents to a distance of 50 miles from the site.

1.7 ODCM CHANGES

The Radiological Effluent Release Report shall include any changes made during the reporting period to the Off-site Dose Calculation Manual (ODCM), including identification of new locations for dose calculations and/or environmental sampling (ODCM, Section II, Table 2-4 and ODCM, Section I, Table 1-1, respectively).

2.0 RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

The Radiological Environmental Operating Report covering the operation of the unit during the pervious calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations and analysis of trends of results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the ODCM and Sections IV.B.2, IV.B.3 and IV.C of Appendix I to 10CFR50.

The Radiological Environmental Operating Report shall include summaries, interpretation and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with previous environmental surveillance report data of similar type and an assessment of the observed impacts of the Plant operation on the environment.

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The report also shall provide a summary description of the environmental monitoring program and a map depicting sample locations.

3.0 NON-ROUTINE REPORTS

A report shall be submitted to the NRC in the event that:

a. The Radiological Environmental Monitoring Program is not substantially conducted as described in ODCM, Section I, Part 4.0.

b. Dose to an off-site individual is estimated to have exceeded the requirements of 40CFR190, per ODCM, Section I, Part 3.0.

c. An unusual or important event occurs due to plant operations that causes a significant radiological environmental impact or affects a potential environmental impact.

d. Non-routine reports shall be submitted within 30 days of determining that an event has occurred to require such report.

4.0 MAJOR MODIFICATION OF RADIOACTIVE WASTE SYSTEMS

4.1 APPLICABILITY

All installed radioactive waste systems have been dismantled. Dismantlement was performed after determining that removal would not cause effluents to exceed Specifications 1.3 or 2.3 of ODCM, Section I, or cause effluents to exceed two percent of the annual limit on a monthly basis.

4.1.1 Major modifications of the radwaste system, as defined in Requirement 4.2.2 below, shall be reported to the NRC pursuant to 10CFR50.59. The discussion of each modification shall contain:

a. A summary of the evaluation that led to the determination that the modification could be made in accordance with 10CFR50.59.

b. A description of the equipment, components and processes involved, and the interfaces with other Plant systems.

c. Documentation of the fact that the modification was reviewed and found acceptable by the NRC.

4.1.2 Shall become effective upon review and acceptance by the Site General Manager.

4.2 DEFINITION OF MAJOR RADWASTE SYSTEM MODIFICATION OR EQUIVALENT ODCM REVISION

4.2.1 Purpose

The purpose of this definition is to assure that modifications (including removal) of radwaste systems during decommissioning or revisions to ODCM do not increase the potential for dose due to effluents to exceed the quarterly or semiannual limits of Specifications 1.3 or 2.3, ODCM, Section I or cause effluent dose to exceed two percent of the annual limit on a monthly basis.

4.2.2 Definition

A major radwaste system modification is a modification which would cause reduction in effluent treatment, monitoring or control that would result in inability to meet the quarterly or semiannual off-site dose limits of Specifications 1.3 (airborne release) or 2.3 (liquid release), of ODCM, Section I. In addition, a major modification would be determined by inability of new equipment or techniques to maintain effluents below two percent of the annual limits in a month, provided that the original equipment would have been capable of meeting that goal.

Improvements or additions to reduce radioactivity in effluents shall not be considered major modifications, nor will any dismantlement activities that do not cause effluents to exceed two percent of the Appendix I limits in any 31-day interval.

1.0 PROGRAM OVERVIEW

The Process Control Program (PCP) is intended to provide a general description of methods for controlling the processing and packaging of radioactive waste for burial. Regardless of the waste class, the resulting waste form when shipped for burial shall meet the following requirements:

1. Waste must be packaged in containers acceptable to the burial site as designated in the waste acceptance criteria.

2. The packaging or waste shall not contain any liquid except as allowed by the disposal site license and waste acceptance criteria.

3. Waste must not contain or be capable of generating, quantities of toxic gases, vapors or fumes harmful to persons transporting, handling or disposing of the waste.

4. Waste must not be pyrophoric. Pyrophoric materials in waste shall be treated, prepared and packaged to be nonflammable.

5. Waste containing hazardous, biological, pathogenic or infectious material must be treated to reduce the potential hazard from the non-radiological materials and be acceptable to the burial site as designated by the waste acceptance criteria.

For Class B and C waste, the waste form should maintain gross physical properties and identity over a 300-year period. To ensure that Class B and C wastes maintain stability, the following minimum conditions should be met:

1. The waste should be a solid form or in a container or structure that provides stability after disposal.

2. The waste or packaging shall not contain any liquid except as authorized by the disposal site license.

3. The waste or container should be resistant to degradation caused by radiation effects.

4. The waste or container should remain stable under the compressive loads inherent in the disposal environment.

5. The waste or container should remain stable if exposed to moisture or water after disposal.

6. The as generated waste should be compatible with the stabilization medium or container.

2.0 <u>WASTE STABILIZATION</u>

Radioactive waste may be inherently stable, or may be stabilized by solidification, encapsulation or use of a High Integrity Container (HIC).

2.1 Inherently stable waste is generally associated with activated or irradiated steel or concrete. The waste shall be packaged as required by the disposal site waste acceptance criteria. Additionally, contact will be made with the disposal site to ensure acceptance of the inherently stable waste classification.

2.2 If solidification or encapsulation of waste is required, the processing of the waste and tests for acceptable solidification/encapsulation shall be documented in procedures specific to the intended method.

2.3 High Integrity Containers are generally used for providing stability of waste streams including resin, filter media and non inherently stable waste, such as highly contaminated metals. General criteria for selection and use of a HIC for stability include:

1. The HIC is acceptable for use at the disposal facility.

2. The HIC is used in accordance with the applicable certificate of compliance issued for the HIC.

3. The HIC is acceptable to the operator of the disposal facility and an engineered barrier is available, if required.

<u>NOTE</u>: Due to the location and possible release path to Lake Michigan, no dewatering or liquid processing activities will be permitted in the Radwaste Storage Facility.

4. For HICs potentially containing liquids, such as resin and filter HICs, the HIC shall be dewatered and acceptable dewatering testing shall be documented in an approved procedure. After acceptable dewatering has been completed, absorbent or filler material may be added to HICs to prevent shifting of waste in the container during transport.

3.0 WASTE CLASSIFICATION

3.1 Waste sampling and classification procedures shall be sufficient to identify the actual activities in the waste form within a factor of ten. If scaling factors are used to establish, the scaling factors will be established at the following frequencies.

3.2 Scaling factors for activated components shall be decay corrected to within one year of the expected disposal date. After initial determination, no significant additional neutron irradiation will occur to increase the waste stream specific activity.

3.3 For Dry Active Waste, including demolition debris and system components, the scaling factor should be determined every two years or if a significant isotopic ratio shift is suspected to have occurred. Big Rock Point has undergone systems decontamination and all normally radioactive systems are assumed to contain the same isotopic make up.

3.4 For liquid waste system filters, the scaling factors should be determined every two years or if a significant isotopic ratio shift is suspected to occur. The major source of liquids at Big Rock Point is the Spent Fuel Pool. Sampling of the spent fuel pool filters is assumed to be representative of the filter waste streams.

3.5 For liquid waste system resins, the scaling factors should be determined for each resin shipping campaign, which is expected to occur on an infrequent basis due to the limited quantity of resin being produced.

3.6 Documentation of the waste stream analysis, waste form and scaling factor determination will be maintained by the Big Rock Point Radwaste Project Manager.

4.0 CONTRACTED VENDOR SERVICES

4.1 Vendors used for processing of waste streams shall be selected from the Consumers Energy Approved Suppliers List.

5.0 RADIOACTIVE LIQUID WASTE SYSTEM DESCRIPTION

5.1 The plant liquid radwaste system consists of installed or temporary pumps, piping or hoses, tanks, filters and demineralizers. The system is designed to process water to levels to allow discharge to the environment.

5.2 Resins and filter media from system operation may be packaged and shipped off site for processing by an approved vendor or dewatered/dried and shipped off site for disposal at an approved disposal site.

5.3 Water generated from decommissioning activities (i.e., decontamination or dust control) may be processed using evaporators. All process water evaporation will be conducted in a controlled environment equipped with filtration and monitoring in accordance with the ODCM Section 1, Subsection 1.2. Solid waste generated from evaporation will be processed in accordance with this PCP.