UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CONSUMERS POWER COMPANY

DOCKET NO. 50-155

BIG ROCK POINT PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. DPR-6

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Consumers Power Company (the licensee) dated January 7, 1985 as revised March 14, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 2.C.(2) of Facility Operating License No. DPR-6 are hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 77, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 26, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. DPR-6

DOCKET NO. 50-155

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

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iv	--
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2-1	2-1
-	2-1a
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1.2 DEFINITIONS

Various provisions of these Technical Specifications set forth limitations and restrictions which depend upon modes of operation. The following modes of operations (1.2.1 through 1.2.6) are defined to clarify the intent of such provisions, and are not the same as, nor should they be confused with, the positions of the mode selector switch described in Section 6.1.3.

1.2.1 Power Operation - is any operation other than shutdown or cold shutdown with the reactor vessel closure bolted in place.

1.2.2 Core Alteration - is any completed planned sequence of movements or core components resulting in either a net change in the configuration of the reactor core or a net gain in core reactivity.

1.2.3 Refueling Operation - is any operation with any of the reactor vessel closures open during which a core alteration, or other operation which might increase core reactivity, is in progress.

1.2.4 Major Refueling - is any refueling operation with the head off during which four or more fuel bundles are added, exchanged or repositioned in the reactor core.

1.2.5 Shutdown - is any reactor condition meeting the following requirements:

(a) All or all but one of the control rods are fully inserted in the reactor core; and

(b) Primary system coolant water temperature is less than 212°F.

1.2.6 Cold Shutdown - is a reactor condition involving no fuel in the core, or a reactor condition meeting with the following requirements.

(a) All of the control rods are fully inserted in the core and withdrawal prevented by means of the keylock selector switch, the key to which is in the possession of the Shift Supervisor; and

(b) The reactor coolant system is at atmospheric pressure.

(c) (DELETED)

1.2.7 DOSE EQUIVALENT I-131 - Is that concentration of I-131 microcuries per milliliter, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

1.2.8 Intentionally left blank

1.2.9 CHANNEL CALIBRATION - adjustments, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The channel calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the Channel Functional Test. The channel calibration may be performed by any series of sequential, overlapping, or total channel steps such that the entire channel is calibrated.

1.2.10 CHANNEL CHECK - the qualitative assessment of channel behavior during operation by observation. The determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

1.2.11 CHANNEL FUNCTIONAL TEST

a. Analog Channels - the injection of a signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

b. Bistable Channels - the injection of a signal into the sensor to verify operability including alarm and/or trip functions.

1.2.12 SOURCE CHECK - the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.2.13 MEMBERS OF THE PUBLIC - includes all persons who are not occupationally associated with the plant.

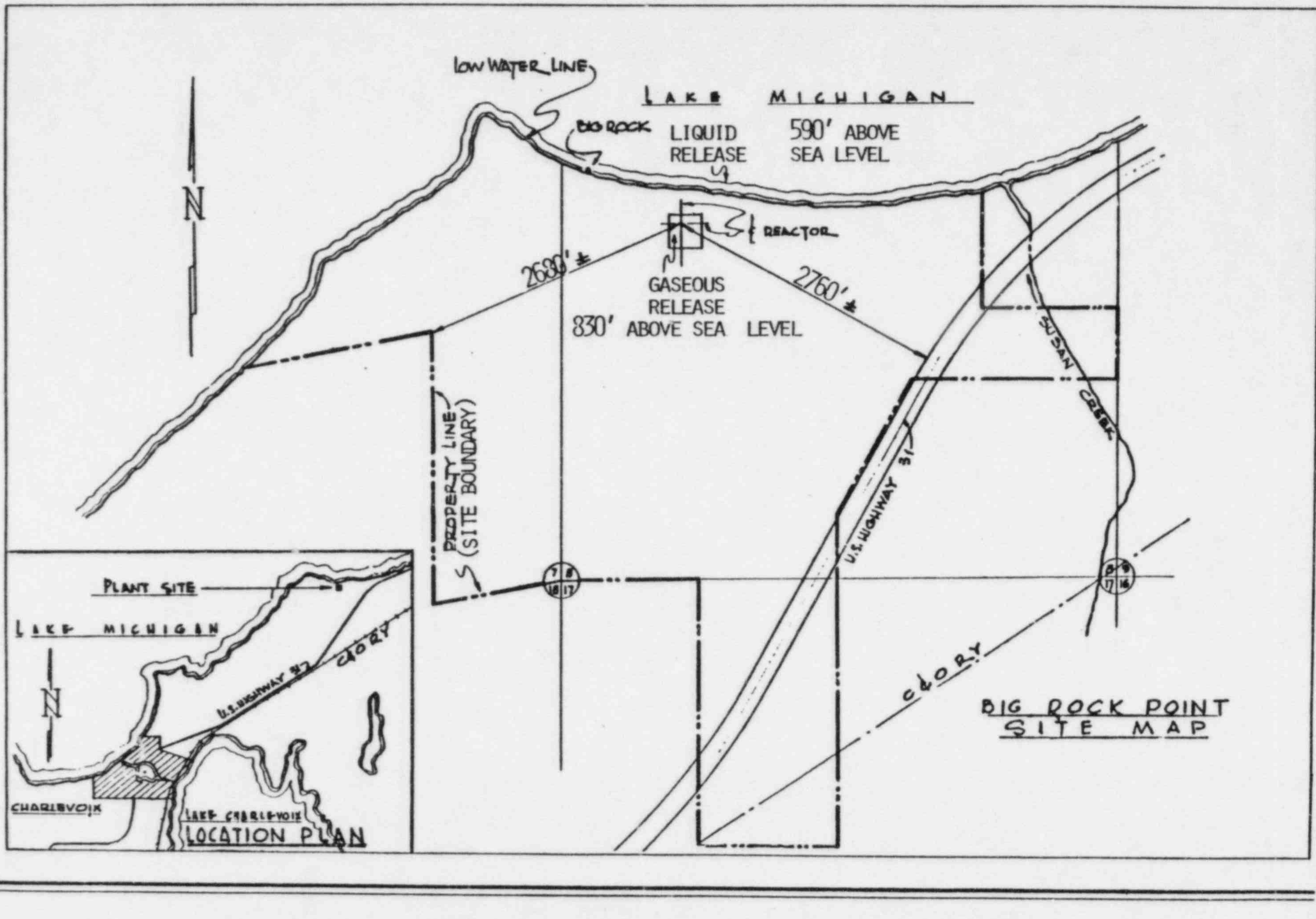
2.0 SITE

2.1 LOCATION - shall be in Charlevoix County, Michigan, shall be about 4 miles northeast of Charlevoix, Michigan, and shall be about 11 miles west of Petoskey, Michigan.

2.2 BOUNDARIES - shall surround about 600 acres, and the nearest landside property line shall be about 2680 feet, and the nearest shoreline property line shall be about 200 feet from the plant's containment sphere. (Reference Figure 2.1)

2.3 PRINCIPAL ACTIVITIES - shall be those associated with the operation of the Big Rock Point Nuclear Plant and may include other activities directly connected with the generation, transmission and distribution of electric energy.

FIGURE 2.1



LOW WATER LINE
LAKE MICHIGAN
BIG ROCK
LIQUID RELEASE
590' ABOVE SEA LEVEL
REACTOR
2680' ±
2760' ±
GASEOUS RELEASE
830' ABOVE SEA LEVEL
PROPERTY LINE (SITE BOUNDARY)
SUSAN CREEK
U.S. HIGHWAY 31
CLORY
7 8 18 17
8 9 17 16
N
BIG ROCK POINT SITE MAP
PLANT SITE
LAKE MICHIGAN
U.S. HIGHWAY 31
CLORY
CHARLEVOIX
LAKE CHARLEVOIX
LOCATION PLAN

6.3.2 Refueling Operation Controls

Interlocks shall be provided to prevent all motion with any of the refueling cranes (namely, jib cranes, transfer cask winch) which are positioned over the reactor vessel whenever any control rod is not fully inserted in the core and the mode selector switch is in the "refuel" position.

6.3.3 Operating Requirements

(a) All reactor refueling safety system sensors and trip devices shall be functionally tested at each major refueling shutdown and shall be maintained in the specified condition during all refueling operations.

(b) The refueling operation controls including position interlocks shall be functionally tested at each major refueling shutdown.

6.4 PLANT MONITORING SYSTEMS

The plant monitoring systems include the process radiation monitoring systems, the area monitoring system, and the reactor water level monitors in the Reactor Depressurizing System.

6.4.1 Process Radiation Monitoring Systems

(a) Air Ejector Off-Gas Monitoring System

(see Section 13.1)

(b) Stack-Gas Monitoring System

(see Section 13.1)

(c) Emergency Condenser Vent Monitor

The emergency condenser vent shall be monitored to detect a significant release of radioactive material. Monitoring shall be supplied by two independent gamma sensitive instrumentation channels employing scintillation crystal sensing devices. These channels shall have a range of 0.1 to 100 mr/hr and shall be provided with an alarm which shall annunciate in the control room to inform the operator of a release of radioactive material.

One of the emergency condenser vent monitors shall be in service at all times during power operation. The monitors shall be set to alarm at approximately 10 mr above the maximum expected background during operation of the emergency condenser. The calibration shall be checked at least monthly.

(d) Process Liquid Monitor System

(see Section 13.1)

6.4.1 (Contd)

(e) <u>In-Plant Radio-Iodine Measurements Under Accident Conditions</u>

Procedures for determining airborne radio-iodine concentrations in occupied areas shall be implemented and technicians shall be trained on an annual basis. Maintenance of the sampling equipment shall occur at least semi-annually and maintenance of the analytical equipment shall occur at least monthly.

6.4.2 <u>Area Monitoring System</u>

(a) Fixed gamma monitors employing scintillation type detectors shall be installed as follows: (1) two on the refueling deck and (2) one in the control room. Each monitor shall have the following:

(i) A range consistent with expected radiation levels in the area to be monitored. (0.01 mr to 10 mr or 0.1 mr to 100 mr or 1 mr to 1,000 mr.)

(ii) An output indicated and recorded in the control room.

(iii) An adjustable high radiation alarm which shall be annunciated in the control room.

The area monitors described above shall normally be in operation; however, individual monitors may be taken out of service for maintenance and repairs. Adequate spare parts shall be on hand to allow necessary repairs to be made promptly. During monitor outages temporary monitoring shall be provided. Calibration of monitors shall be performed at least monthly. Alarm trip points shall be set at a radiation level approximately twice the normal maximum indicated radiation level, but normally not less than one decade above the lowest scale reading.

(b) The two area monitors located on the refueling deck shall provide gamma monitoring of the fuel storage areas and refueling operations. Local alarms shall be provided for these monitors, and alarm setting shall be in accordance with the provisions of 10 CFR 70. In the event that both of these monitors become inoperable during power operation or fuel handling activities, the containment ventilation isolation valves shall be closed.

However, notwithstanding the requirements of Section 70.24(a)(1), alarm settings may be raised above 20 mR/hr as long as the overall detection criterion in Section 70.24(a)(1) is satisfied and the requirements specified in paragraph 6.4.2(a) above are met.

6.4.2 (Contd)

(c) The containment atmosphere shall be monitored by two high range gamma monitors. The monitors are designed to measure gamma radiation in containment under accident conditions from 1 R/hr to 1E+06 R/hr. The monitors are located external to the containment sphere. The readouts of the monitors are located in the control room.

Both high range containment atmosphere gamma monitors shall normally be in service during power operation. If either monitor is inoperable, restore to operable status within 72 hours or, in lieu of any other report required by Specification 6.9.2 (administrative controls), prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3 (administrative controls) within the next 30 days outlining the cause of the inoperability and the plans for restoring the system to operable status. A channel check shall be performed for each monitor at least once per month, and channel calibration shall be performed at each refueling outage. The channel calibration for all ranges above 10R/hr may be performed by electronic signal substitution.

(d) Portable gamma dose-rate measuring instruments and portable neutron dose-rate measuring instruments shall be provided for establishing permissible working limits. These instruments shall be calibrated at least once every three months. Portable high range gamma measuring instruments shall be calibrated from 0-20 R/hr every three months and on all scales every six months.

6.4.3 Reactor Water Level Monitors in the Reactor Depressurization System

Four narrow range water level monitors are provided in the main control room as part of the Reactor Depressurizing System to be used for detection of adequate core cooling during accident situations.

At least two reactor water level indicators in the Reactor Depressurization System shall be operable during power operation.

7.6 (Contd)

System or Function Undergoing Test	Frequency of Routine Tests	Reference Procedure Within These Specifications
Containment sphere access airlocks leakage rate	6 months or less	
Control rod performance	At each major refueling shutdown**	Section 5.2.2
Liquid poison system component operability	Two months or less during power operation. One squib test-fired each 12 months.	Section 5.2.3
Reactor safety system scram circuits requiring plant shutdown to check	At each major refueling shutdown*	Section 6.1.5
Reactor safety system scram circuits not requiring plant shutdown to check	One month or less	Section 6.1.5
Containment sphere isolation trip circuits	At each major refueling shutdown*	Section 6.1.5
Emergency Condenser Trip Circuits	At each major refueling shutdown*	Section 6.1.5
Control rod withdrawal permissive interlocks function	At each major refueling shutdown*	Section 6.2.2
Refueling operation controls function	At each major refueling shutdown	Section 6.3.3
Calibration and functional test of air ejector off-gas and stack-gas monitors	Per Table 13-2	Section 13.2
Calibration of emergency condenser vent monitors	One month or less	Section 6.4.1
Calibration of process liquid monitors	Per Table 13-2	Section 13.2

*But no less than once a year

**But no less than once every 20 months

7.6(Contd)

System or Function Undergoing Test	Frequency of Routine Tests	Reference Procedure Within These Specifications
Calibration of area monitoring system	One month or less	Section 6.4.2
Channel comparison check of reactor level indicating instruments in the Reactor Depressurization System	One month or less	Section 6.4.3
Calibration of reactor level indicating instruments in the Reactor Depressurizing System	At each major refueling shutdown	Section 6.4.3
Steam drum safety valve position monitor check	One month or less	Section 4.1.2
Calibration of steam drum safety valve position monitors	At each major refueling shutdown	Section 4.1.2
High radiation trip closure of the containment ventilation isolation valves	At each major refueling shutdown	Section 6.4.2
Channel Check of High Range Containment Gamma Monitors	One month or less	Section 6.4.2
Channel calibration of High Range Containment Gamma Monitors	At each major refueling shutdown	Section 6.4.2

RESPONSIBILITIES (Continued)

j. Review of any accidental, unplanned or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Plant Superintendent and to the Nuclear Safety Board (NSB).

PRC review may be performed through a routing of the item subject to the requirements of Specification 6.5.1.7.

AUTHORITY

6.5.1.7 The PRC shall:

a. Recommend in writing to the Plant Superintendent approval or disapproval of items considered under Specifications 6.5.1.6.a. through d. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6.a through e. above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President - Nuclear Operations and to the Vice Chairman of NSB of any disagreement between the PRC and the Plant Superintendent; however, the Plant Superintendent shall have responsibility for the resolution of such disagreements pursuant to Specification 6.1.1 above.

The PRC Chairman may recommend to the Plant Superintendent approval of those items identified in Specifications 6.5.1.6 a. through d. above based on a routing review provided the following conditions are met: (1) at least five PRC members, including the Chairman and no more than 2 alternates, shall review the item, concur with determination as to whether or not the item constitutes an unreviewed safety question, and provide written comments on the item; (2) all comments shall be resolved to the satisfaction of the reviewers providing the comments; and (3) if the PRC Chairman determines that the comments are significant, the item (including comments and resolutions) shall be recirculated to all reviewers for additional comments.

The item shall be reviewed at a PRC meeting in the event that: (1) comments are not resolved; or (2) the Plant Superintendent overrides the recommendations of the PRC; or (3) a proposed change to the Technical Specifications involves a safety limit, a limiting safety system setting or a limiting condition for operation; or (4) the item was a reportable event as defined in 10 CFR 50.73.

RECORDS

6.5.1.8 The PRC shall maintain written minutes of each PRC meeting and shall provide copies to the NSB.

6.5.2 NUCLEAR SAFETY BOARD (NSB)

6.5.2.1 The Nuclear Safety Board is responsible for maintaining a continuing examination of nuclear safety-related corporate and plant activities and defining opportunities for policy changes related to improved nuclear safety performance. The NSB shall operate in accordance with a written charter approved by the Vice President - Nuclear Operations, which designates the membership, authority, and rules for conducting the meetings.

FUNCTION

6.5.2.2 The NSB shall function to provide review of designated activities in the areas specified in Specification 6.5.2.3.

COMPOSITION

6.5.2.3 The NSB shall consist of members appointed by the Vice President - Nuclear Operations. NSB shall be chaired by the Executive Director of Nuclear Assurance, the Vice Chairman, or a duly appointed alternate. The Director - Nuclear Safety, shall be the Vice Chairman and Secretary.

Collectively, the personnel appointed to NSB shall be competent to conduct reviews in the following areas:

a. Nuclear Power Plant Operations

b. Nuclear Engineering

c. Chemistry and Radiochemistry

d. Metallurgy

e. Instrumentation and Control

f. Radiological Safety

g. Mechanical and Electrical Engineering

h. Quality Assurance Practices

An individual appointed to NSB may possess expertise in more than one of the above specialties. These individuals should, in general, have had professional experience in their specialty at or above the Senior Engineer level.

ALTERNATE MEMBERS

6.5.2.4 Alternate members may be appointed in writing by the Vice President - Nuclear Operations to act in place of members during any legitimate and unavoidable absences. The qualifications of alternate members shall be similar to those of members.

CONSULTANTS

6.5.2.5 Consultants shall be utilized as determined by the NSB Chairman or Vice Chairman to provide expert advice to the NSB. NSB members are not restricted as to sources of technical input and may call for separate investigation from any competent source.

AUDITS

6.5.2.8.2 Audits of operational nuclear safety-related facility activities shall be performed under the cognizance of NSB. These audits shall encompass:

a. The conformance of plant operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire facility staff at least once per 12 months.

c. The performance of activities required by the operational quality assurance program (CPC-2A QAPD) to meet the criteria of Appendix "B," 10 CFR 50, at least once per 24 months.

d. The Site Emergency Plan and implementing procedures at least once per 12 months.

e. The Site Security Plan and implementing procedures (as required by the Site Security Plan) at least once per 12 months.

f. Any other area of plant operation considered appropriate by NSB or the Vice President - Nuclear Operations.

g. The plant Fire Protection Program and implementing procedures at least once per 24 months.

h. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

i. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

Audit reports encompassed by 6.5.2.8.2 above shall be forwarded to the NSB Vice Chairman and Secretary, and Management positions responsible for the areas audited within thirty (30) days after completion of the audit.

AUTHORITY

6.5.2.9 The NSB Chairman shall report to and advise the Vice President - Nuclear Operations of significant findings associated with NSB activities and of recommendations related to improving plant nuclear safety performance.

6.9.1.9 (Contd)

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c. (administrative controls) above designed to contain radioactive material resulting from the fission process.

NOTE: Sealed sources or calibration sources are not included under this item. Leakage of valve packing or gaskets within the limits for identified leakage set forth in Technical Specifications need not be reported under this item.

6.9.2 UNIQUE REPORTING REQUIREMENTS

6.9.2.1 Annual Radiological Environmental Operating Report

Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The annual radiological environmental operating reports shall include summaries, interpretation and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate) and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use census required by Specification 13.2.3.

The annual radiological environmental operating reports 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 reports shall also include the following: a summary description of the radiological environmental monitoring program including sampling methods for each sample type, a map of all sampling locations keyed to a table giving distances and directions from the reactor and the results of land use census required by the Specification 13.2.3., and results of the interlaboratory comparison program required by Specification 13.2.4.

6.9.2.2 Semiannual Radioactive Effluent Release Report

A. Radioactive Effluent Release

A report shall be submitted to NRC within 60 days after January 1 and July 1 of each year specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21, Revision 1, dated June 1974.

1. Gaseous Effluents

   a. Gross Radioactivity Releases

      (1) Total gross radioactivity (in curies), including noble and activation gases released.

      (2) Maximum gross radioactivity release rate during any one-hour period.

      (3) Total gross radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

      (4) Percent of Technical Specifications limit.

   b. Iodine Releases

      (1) Total iodine radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

      (2) Percent of Technical Specifications limits for I-131 released.

   c. Particulate Releases

      (1) Gross radioactivity (β, γ) released (in curies) excluding background radioactivity.

      (2) Gross alpha radioactivity released (in curies) excluding background radioactivity.

      (3) Total gross radioactivity (in curies) of nuclides with half-lives greater than eight days.

      (4) Percent of Technical Specifications limit for particulate radioactivity with half-lives greater than eight days.

6.9.2.2.A (Contd)

2. Liquid Effluents

   a. Gross radioactivity (β, γ) released (in curies) excluding tritium and average concentration released to the unrestricted area.

   b. Total tritium and alpha radioactivity (in curies) released and average concentration released to the unrestricted area.

   c. Total dissolved gas radioactivity (in curies) and average concentration released to the unrestricted area.

   d. Total volume (in liters) of liquid waste released.

   e Total volume (in liters) of dilution water used prior to release from the restricted area.

   f. The maximum concentration of gross radioactivity (β, γ) released to the unrestricted area (averaged over the period of release).

   g. Total radioactivity (in curies) by nuclide released, based on representative isotopic analyses performed.

   h. Percent of Technical Specifications limit and 10 CFR, Part 20 concentration limits for unrestricted areas.

3. Solid Waste

   The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite 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 (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms),

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

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

6.9.2.2.A (Contd)

4. Radiological Impact on Man

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include potential doses to individuals and populations calculated using measured effluent and averaged meteorological data in accordance with the methodologies in the Offsite Dose Calculation Manual.

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

b. The maximum offsite air doses (greater than 1 milliRad) due to beta and gamma radiation at locations near ground level from gaseous effluents.

c. Organ doses (greater than 1 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 1 manRem) to the population and average doses (greater than 1 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 1 manRem) to the population and average doses (greater than 1 milliRem) to individuals in the population from gaseous effluents to a distance of 50 miles from the site.

5. PCP and ODCM Changes

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Process Control Program (PCP) and to the Offsite Dose Calculation Manual (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 13.2.3.

6.14 PROCESS CONTROL PROGRAM (PCP)

6.14.1 Changes to the PCP shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to support the rationale for the change;

b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

c. Documentation of the fact that the change has been reviewed and approved by the responsible Nuclear Operations Department per CPC 2A (Quality Assurance Program).

6.15 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.15.1 Changes to the ODCM shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

a. Sufficiently detailed information to support the rationale for the change;

b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

c. Documentation of the fact that the change has been reviewed and approved by the responsible Nuclear Operations Department per CPC 2A (Quality Assurance Program).

13.1 RADIOLOGICAL EFFLUENT RELEASES

13.1.1 RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION OPERABILITY AND SURVEILLANCE REQUIREMENTS

13.1.1.1 The radioactive effluent monitoring instrumentation channels shown in Table 13-1 shall be Operable with their alarm/trip set points set to ensure that the limits of Specification 13.1.2.1 or 13.1.3.1 are not exceeded. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the above specification, suspend the release of radioactive effluents monitored by the affected channel or declare the channel inoperable, or change the set point so it is acceptably conservative.

b. With less than the minimum number of radioactive effluent monitoring instrumentation channels Operable, take the action shown in Table 13-1. Exert best efforts to return the instruments to Operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

13.1.1.2 Each radioactive effluent monitoring instrumentation channel shall be demonstrated Operable by performance of the Channel Check, Source Check, Channel Calibration and Channel Functional Test operations at the frequencies shown in Table 13-2.

BASES FOR 13.1.1.1 - RADIOACTIVE EFFLUENT MINOTRING INSTRUMENTATION

The radioactive effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in effluents during actual or potential releases of liquid effluents. The alarm/trip set points for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20.

TABLE 13-1

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE[a]	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Air Ejector Off Gas	(1)[b] Power Operation	2
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent	(1)	1
b. Circulating Water Discharge	(1)	3
3. FLOW RATE INDICATING DEVICE		
a. Liquid Radwaste Effluent Line	(1)	4
4. CANAL SAMPLE COLLECTION TANK		
a. Circulating Water Discharge	(1)	3
5. STACK GAS EFFLUENT SYSTEM		
a. Noble Gas Activity Monitor	(1)	2
b. Iodine/Particulate Sampler	(1)	5
c. Sampler Flow Rate Monitor	(1)	5
d. Hi Range Noble Gas Monitor[c]	(1)	6

[a] At all times unless otherwise noted
[b] Applicable during power operation only
[c] Hi Range Noble Gas Monitor RE-8284 will have an alarm set point in accordance with Emergency Plan Implementing Procedures

TABLE 13-1 NOTATION

ACTION 1- With the channel inoperable effluent releases may continue via this pathway provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 13.1.2.2, and

b. The release rate calculations and discharge line valving are verified by at least two technically qualified members of the Facility Staff.

ACTION 2 - With the Air Ejector Off Gas and Stack Noble Gas channels inoperable effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-4} microcuries/ml at least once per 24 hours.

ACTION 3 - With the channel inoperable, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml

ACTION 4 - With the channel inoperable, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves or tank levels may be used to estimate flow.

ACTION 5 - With the channel inoperable, effluent releases via this pathway may continue provided Reactor Coolant grab samples are taken at least once per 24 hours and these samples are analyzed for I-131 and Cs-134/137 activity within 24 hours for correlation to stack release activity.

ACTION 6 - With the channel inoperable, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

1) either restore the inoperable channel(s) to Operable status within 7 days of the event, or

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to Operable status.

TABLE 13-2

RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Air Ejector Off Gas	D	M	R(5)	Q(1)(2)
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent	D(6)	SM	R(3)	Q(2)
b. Circulating Water Discharge	D	M	R(3)	Q(2)
3. FLOW RATE INDICATING DEVICE				
a. Liquid Radwaste Effluent Line	D(4)	NA	R	NA
4. CANAL SAMPLE COLLECTION TANK	D(4)	NA	NA	NA
5. STACK GAS EFFLUENT SYSTEM				
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)
b. Iodine/Particulate Sampler	W	NA	NA	NA
c. Sampler Flow Rate Monitor	D	NA	R	NA
d. Hi Range Noble Gas	D	M	R(5)	Q(2)

TABLE 13-2
NOTATION

NOTATION	FREQUENCY
D	At least once per 24 hours
W	At least once per 7 days
SM	At least twice per 31 days
M	At least once per 31 days
Q	At least once per 92 days
R	At least once per 18 months
NA	Not applicable

(1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway occurs if instrument indicates measured levels above the alarm/trip set point.

(2) The Channel Functional Test shall also demonstrate that control room alarm annunciation occurs if the instrument indicates measured levels above the alarm set point.

(3) a. The Channel Calibration shall be performed using one or more of the reference standards traceable to the National Bureau of Standards or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range.

b. For subsequent Channel Calibration, sources that have been related to the (a) calibration may be used.

(4) Channel Check shall consist of verifying indication of flow during periods of release. Channel Check shall be made at least once per 24 hours on days on which continuous or batch releases are made.

(5) As in (3), but for ALARA purposes, ranges requiring greater than a 0.2 Ci calibration source will meet source calibration requirements on all lower ranges, and will be calibrated electronically on higher ranges.

(6) Channel Check shall be made at least once per 24 hours on days on which continuous or batch releases were made.

13.1.2. LIQUID EFFLUENTS CONCENTRATION

13.1.2.1 The concentration of radioactive material released in liquid effluents from the site to areas at or beyond the site boundary (see Figure 2.1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents from the site to areas at or beyond the site boundary (see Figure 2.1) exceeding the above limits, upon discovery promptly restore the concentration to within the above limits.

13.1.2.2 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 13-3.

13.1.2.3 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 13.1.2.1.

BASES FOR 13.1.2.1 - LIQUID EFFLUENT CONCENTRATIONS

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to unrestricted areas will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, 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, 10 CFR Part 50, to a member of the public, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its Maximum Permissible Concentration (MPC) is air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICPR) Publication 2.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L A, "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal Chem 40, 586-93 (1968), and Hartwell, J K. "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

13.1.3 GASEOUS EFFLUENTS DOSE RATE

13.1.3.1 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: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, for iodine-133, for tritium and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the dose rate averaged over a period of one hour exceeding the above limits, upon discovery promptly restore the release rate to within the above limit(s).

13.1.3.2 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

13.1.3.3 The dose rate due to iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 13-3.

BASES FOR 13.1.3.1 - GASEOUS EFFLUENT DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For members of the public who may at times be within the SITE BOUNDARY, the occupancy of that member of the public will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. Examples of calculations for such members of the public, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to less

than or equal to 500 mrems/yr to the total body or to less than or equal to 3000 mrems/hr to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/yr.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L A, "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal Chem 40, 586-93 (1968), and Hartwell, J K, "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

TABLE 13-3
RADIOACTIVE WASTE SAMPLING AND ANALYSIS PROGRAM

Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Liquid Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	P Each Batch	P	Dissolved and Entrained Gases (Xe 133)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
B. Circulating Water Discharge[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	M Grab Sample	M	Dissolved and Entrained Gases (Xe 133)	1×10^{-5}
	Continuous[f]	M Composite[f]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Continuous[f]	Q Composite[f]	Sr-89, Sr-90	5×10^{-8}
C. Air Ejector Off Gas	W Power Operation Grab Sample	W	Principal Gamma Emitters[g] (Noble Gases)	1×10^{-4}
D. Stack Gas Effluent	Continuous[h]	W[i] Charcoal	I-131, I-133	1×10^{-12}
	Continuous[h]	W[i] Particulate	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[h]	Q Composite Particulate	Sr-89, Sr-90 and Gross Alpha	1×10^{-11}
	Continuous[h]	Continuously by Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-5}

TABLE 13-3 NOTATION

NOTATION	FREQUENCY
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
P	Completed prior to each release

a. 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 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot e^{-\lambda \Delta t}}$$

Where:

LLD is the predetermined lower limit of detection as defined above, as microcuries 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,

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for plant effluents is the elapsed time between the midpoint of sample collection and time of counting.

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

It should be recognized that the LLD is defined as a predetermined limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement.

b. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed to assure representative sampling.

Table 13-3 Notation (Contd)

c. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99,* Cs-134, Cs-137, Ce-141 and Ce-144.* This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.2.2 (Administrative Controls).

*(LLD = 5×10^{-6} because of low gamma yields.)

d. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; eg, from a volume of a system that has an input flow during the continuous release.

f. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144** for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.2.2 (Administrative Controls).

** (LLD = 1×10^{-10} because of low gamma yields)

h. The ratio of the sample flow to be sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 13.1.3.1, 13.1.4.2 and 13.1.4.3.

i. Samples shall be changed at least once per 7 days and analyses shall be completed within 72 hours after changing or after removal from sampler.

13.1.4 EFFLUENT DOSE

13.1.4.1 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 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

13.1.4.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the site boundary (see Figure 2.1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

13.1.4.3 The dose to a member of the public from iodine-131, iodine-133, tritium and all radionuclides in particulates form with half lives greater than 8 days in gaseous effluents released to areas at and beyond the site boundary (see Figure 2.1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ, and

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

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.

13.1.4.4 Cumulative 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 the ODCM at least once per 31 days.

BASES FOR 13.1.4.1 EFFLUENT DOSE - LIQUIDS

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition 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, for fresh-water sites with drinking water supplies that can be potentially affected by plant operations, 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 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM 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 equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents 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 10 CFR 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.

BASES FOR 13.1.4.2 - EFFLUENT DOSE - NOBLE GASES

This specification is provided to implement the requirements of Section II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The limiting Condition for Operation implements the guides set forth in Section II.B 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 gaseous 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 the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents 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

BASES FOR 13.1.4.2 (Contd)

Evaluating Compliance with 10 CFR Part 50, 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. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

BASES FOR 13.1.4.3 EFFLUENT DOSE - IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Section II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section 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 materials in gaseous effluents to unrestricted areas will be kept "as low as is reasonable achievable." The ODCM calculational methods specified in 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 ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials 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 10 CFR Part 50, Appendix 1," 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. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the site boundary. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

13.1.5 SOLID RADIOACTIVE WASTE

13.1.5.1 The solid radwaste system shall be used in accordance with a Process Control Program to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the Process Control Program not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

BASES FOR 13.1.5 - SOLID RADIOACTIVE WASTE

The process parameters included in establishing the Process Control Program may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents and mixing and curing times.

13.1.6 TOTAL DOSE

13.1.6.1 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 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 13.1.4.1.a, 13.1.4.1.b, 13.1.4.2.a, 13.1.4.2.b, 13.1.4.3.a, or 13.1.4.3.b, calculations should be made including direct radiation contributions from the reactor unit and from outside storage tanks to determine whether the above limits of Specification 13.1.6.1 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 10 CFR Part 20.405c, 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.

13.1.6.2 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specification 13.1.4.4 and in accordance with the methodology and parameters in the ODCM.

13.1.6.3 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 13.1.6.1.

BASES FOR 13.1.6.1 - TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. 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 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. 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 40 CFR Part 190 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 5 mi must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications and 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.

13.2 RADIOLOGICAL ENVIRONMENTAL MONITORING

13.2.1 The radiological environmental monitoring program shall be conducted as specified in Table 13.3-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 13.3-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.2.1 (administrative controls) 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 13.3-2 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 13.3-2 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 13.3-2 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 Specifications 13.1.2.1, 13.1.3.1 and 13.1.3.2 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.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 13.3-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Annual Radiological Environmental Report.

BASES FOR 13.2.1 RADIOLOGICAL ENVIRONMENTAL MONITORING

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of members of the public resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. 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 13.3-3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is a predetermined limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L A, "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal Chem 40," 586-93 (1968), and Hartwell, J K, "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-214 (June 1975).

13.2.2 The radiological environmental monitoring samples shall be collected pursuant to Table 13.3-1 and shall be analyzed pursuant to the requirements of Table 13.3-1 and the detection capabilities required by Table 13.3-3.

13.2.3 A land use census shall be conducted and shall identify within a distance of 5 miles the location in each of the 9 overland meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation. The land use census shall also identify within a distance of 3 miles the locations in each of the overland sectors of all milk animals and all gardens of greater than 50 m^2 producing broad leaf vegetation.

*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 13.3-1.4c shall be followed, including analysis of control samples.

13.2.3 (Contd)

The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.2.1 (administrative controls) and shall be included in a revision of the ODCM for use in the following calendar year.

BASES FOR 13.2.3 - LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the radiological environmental monitoring program are made if required by results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfied the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 40 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (16 kg/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20% of the garden was used for growing broad leaf vegetation (ie, similar to lettuce and cabbage), and (2) a vegetation yield of 2 kg/m^2.

13.2.4 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.

BASES FOR 13.2.4 - INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sampling matrices are performed in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

13.2.5 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.2.1.

TABLE 13.3-1

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]	16 routine monitoring stations either with two or more thermoluminescent dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: Miscellaneous site locations (4) An inner ring of stations, (6) in the general area of the SITE BOUNDARY. An outer ring of stations, (3) in the 3- to 5-mi range from the site. The balance of the stations (3) to be placed to serve as control stations.	Quarterly.	Gamma dose quarterly.

TABLE 13.3-1 (Continued)

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
2. AIRBORNE			
Radioiodine and Particulates	Samples from 6 locations: 3 samples from within 5 mi of the SITE BOUNDARY in different sectors. 2 samples from the vicinity of communities having the highest calculated annual average ground level D/Q. (Petoskey-10.5 mi E, and Boyne City-12 mi SE) 1 sample from a control location, as for example 50 mi distant and in the least prevalent wind direction.[c] (Traverse City-50 mi SSW)	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly for each filter change. Particulate Sampler: Gross beta radioactivity analysis following[d] filter change; Gamma isotopic analysis[e] of the composite quarterly if gross beta >1.0 pCi/m³.

TABLE 13.3-1 (Continued)

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
3. WATERBORNE			
a. Lake	1 sample from Plant Lake Water Inlet (service water from intake bay).	Composite sample over 1-month period.[f]	Gross beta and tritium monthly. Gamma analysis if gross beta > 10 pCi/L.
b. Well (drinking)	1 sample from Plant well,	Monthly - grab sample.	Gross beta and tritium monthly. Gamma analysis if gross beta > 10 pCi/L.
c. Lake (drinking)	1 sample of Charlevoix Drinking Water Supply.	Daily composite sample over 1 month period.[f]	Gross beta and tritium monthly. Gamma analysis if gross beta > 10 pCi/L.
d. Sediment from Shoreline	1 sample from 1 to 3 miles east of site.	Semiannually.	Gamma isotopic analysis[e] semiannually.

TABLE 13.3-1 (Continued)

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
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations between 2-7 mi from the site.	Monthly when animals are on pasture (May through November)	Gamma isotopic[e] and I-131 analysis monthly.
	1 sample from milking animals at a control location, 10-20 mi distant.		
b. Fish and Invertebrates	Sample 2 species of commercially and/or recreationally important species in vicinity of plant discharge area.	Sample in season, or semiannually if they are not seasonal.	Gamma isotopic analysis[e] on edible portions.
	1 sample of same species in areas not influenced by plant discharge (Ludington).		

TABLE 13.3-1 (Continued)

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
c. Broad leaf vegetation			
	Samples of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available during May thru November.	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 10-20 mi distant in the least prevalent wind direction if milk sampling is not performed.	Monthly when available during May thru November.	Gamma isotopic[e] and I-131 analysis.

TABLE 13.3-1 (Continued)

TABLE NOTATION

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 Specification 6.9.2.1 (administrative controls). It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions 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. For the purposes of this table, a thermoluminescent dosimeter (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.

c. The purpose of this sample is to obtain background information. If it is not practical to establish control locations in accordance with the distance and wind direction criteria, other sites that provide valid background data may be substituted.

d. Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

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

f. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid discharged and in which the method of sampling employed results in a specimen that is representative of the liquid released (continuous composites or daily grab composites which meet this criteria are acceptable).

TABLE 13.3-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
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	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR Part 141 value.

TABLE 13.3-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b,c]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4	0.01				
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]	0.07		1	60	
Cs-134	15	0.05	130	15	80	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

TABLE NOTATION

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 Specification 6.9.2.1 (administrative controls).

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 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

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

Where:

LLD is the predetermined lower limit of detection as defined above, as picocuries 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 picocurie,

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.

TABLE 13.3-3 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as a predetermined limit representing the capability of a measurement system and not as an after the fact limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.2.1 (administrative controls).

d. LLD for drinking water samples.