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NORTHERN STATES POWER COMPANY

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RECEIVED JUN 16 1975 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

June 13, 1975

Mr. A. Giambusso, Director
Division of Reactor Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263 LICENSE NO. DPR-22

Draft Technical Specification
for Radioactive Effluents

A letter dated August 5, 1974 from AEC/DL requested that we submit an updated draft of proposed environmental Technical Specifications for the Monticello facility, including a Section 2.4 regarding limitation of radioactive discharges from the plant. Our letter dated August 16, 1974, transmitting the proposed environmental Technical Specifications, did not include a Section 2.4 on radioactive effluents because regulatory guidance on the format and details of this section to meet "proposed Appendix I" was still under development by the AEC. There have been ongoing discussions between the NRC and NSP since that time to provide clarification of a number of items in the proposed Section 2.4 Technical Specifications and also to assure consistency between the generic guidance and the specific Monticello systems and equipment.

In connection with the ongoing Monticello public hearing, the NRC Regulatory Staff furnished information to the ASLB which included a draft Environmental Technical Specification containing a Section 2.4-Radioactive Effluents. The Regulatory Staff also expressed their intent to issue interim effluent specifications, based upon this draft, and which would be in effect until a specification based upon the "newly adopted Appendix I" could be finalized. NSP representatives subsequently met with members of the Regulatory Staff to obtain further clarification of a number of points in the NRC draft as they relate to the specific Monticello design and operation. NSP later committed to furnish, by June 15, 1975, a revised draft of Section 2.4 to reflect those clarifications and to incorporate certain needed refinements. Attached are 40 copies of a draft Section 2.4 which was based upon the "proposed Appendix I" criteria as suggested by the Regulatory Staff.

If these Section 2.4 Effluent Technical Specifications are issued on an interim basis, presumably, as a separate Technical Specification Appendix to the Operating License, it will be necessary to delete the current Appendix A Technical Specification Sections 3.8/4.8 covering radioactive effluents, with the exception of Section 3.8.F and 4.8.F covering the environmental monitoring program which should be retained in the Appendix A Section for the time being.

DOCKETED USNRC JUN 16 1975 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

6471

We believe that it is appropriate that any interim radioactive Effluent Technical Specification become effective no sooner than 30 days following resumption of commercial power operation after completion of the Fall, 1975 Monticello refueling outage, since modification and improvements to the augmented off-gas system are scheduled to be made during the fall refueling outage. The letter issuing interim Technical Specifications should specifically reference such an implementation schedule.

Very truly yours,

L. O. Mayer

L. O. Mayer, PE
Manager, Nuclear Support Services

LOM/ECW/deb

cc: J. G. Keppler
G. Charnoff
MPCA
Attn: E. A. Pryzina

2.4 LIMITING CONDITIONS FOR OPERATION

Radioactive Effluents

Objective: To define the limits and conditions for the controlled release of radioactive materials in liquid and gaseous effluents to the environs to ensure that these releases are as low as practicable. These releases should not result in radiation exposures in unrestricted areas greater than a few percent of natural background exposures. The concentrations of radioactive materials in effluents shall be within the limits specified in 10 CFR Part 20.

To ensure that the releases of radioactive material are as low as practicable, the following design objectives apply until Technical Specifications are issued in accordance with the recently adopted Appendix I to 10 CFR Part 50:

For liquid wastes:

a. The annual dose above background to the total body or any organ of an individual should not exceed 5 mrem in an unrestricted area.

b. The annual total quantity of radioactive materials in liquid waste, excluding tritium and dissolved gases, discharged should not exceed 5 Ci.

For gaseous wastes:

c. The annual total quantity of noble gases above background discharged from the site should result in an air dose due to gamma radiation of less than 10 mrad, and an air dose due to beta radiation of less than 20 mrad, at any location near ground level which could be occupied by individuals at or beyond the boundary of the site.

d. The annual total quantity of all radioiodines and radioactive material in particulate forms with half-lives greater than eight days above background, should not result in an annual dose to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem.

e. The annual total quantity of iodine-131 discharged should not exceed 1 Ci.

2.4.1 Specifications for Liquid Waste Effluents

a. The concentration of radioactive materials released in liquid waste effluents shall not exceed the value specified in 10 CFR Part 20, Appendix B, Table II, Column 2, and Notes thereto, in the condenser cooling water discharge canal.

b. The cumulative release of radioactive materials in liquid waste effluent, excluding tritium and dissolved gases, shall not exceed 10 Ci in any calendar quarter.

c. The cumulative release of radioactive materials in liquid waste effluents, excluding tritium and dissolved gases, shall not exceed 20 Ci in any 12 consecutive months.

d. The equipment installed in the liquid radioactive waste system shall be maintained and shall be operated to process radioactive liquid wastes prior to their discharge when the projected cumulative release could exceed 1.25 Ci/calendar quarter, excluding tritium and dissolved gases.

e. The maximum radioactivity to be contained in any liquid radwaste tank that can be discharged directly to the environs shall not exceed 10 Ci, excluding tritium and dissolved gases.

f. If the cumulative release of radioactive materials in liquid effluents, excluding tritium and dissolved gases, exceeds 2.5 Ci/calendar quarter, the licensee shall make an investigation to identify the causes for such releases, define and initiate a program of action to reduce such releases to the design objective levels listed in Section 2.4, and report these actions to the NRC within 30 days from the end of the quarter during which the release occurred.

g. An unplanned or uncontrolled offsite release of radioactive materials in liquid effluents in excess of 0.5 curies requires notification. This notification to the NRC shall be made within 30 days.

2.4.2 Specifications for Liquid Waste Sampling and Monitoring

a. Plant records shall be maintained of the radioactive concentration and volume before dilution of liquid waste intended for discharge and the average dilution flow and length of time over which each discharge occurred. Summaries of the quantities of releases shall be included in the Semi-Annual Radioactive Effluent Report. Estimates of the sampling and analytical errors associated with each reported value shall be included.

b. Prior to release of each batch of liquid waste, a sample shall be taken from that batch and analyzed for the concentration of each significant gamma energy peak in accordance with Table 2.4-1 to demonstrate compliance with Specification 2.4.1 using the flow rate into which the waste is discharged during the period of discharge.

c. Sampling and analysis of liquid radioactive waste shall be performed in accordance with Table 2.4-1. Prior to taking samples from a monitoring tank, at least two tank volumes shall be recirculated.

d. The radioactivity in liquid wastes shall be continuously monitored and recorded during release. Whenever these monitors are inoperable for a period not to exceed 72 hours, two independent samples of each tank to be discharged shall be analyzed and two plant personnel shall independently check valving prior to the discharge. If these monitors are inoperable for a period exceeding 72 hours, no release from a liquid waste tank shall be made and any release in progress shall be terminated.

e. The flow rate of liquid radioactive waste shall be continuously measured and recorded during release.

f. The continuous monitors listed in Table 2.4.3 shall be calibrated at least quarterly by means of a solid radioactive source which has been calibrated to a National Bureau of Standards source. Each monitor shall also have a functional test monthly and an instrument check prior to making a release.

Bases: These Specifications are applicable until Specifications prepared in accordance with Appendix I to 10 CFR Part 50 are issued by the Commission. In some cases these Specifications are more restrictive than required by Appendix I. In the event that plant availability is adversely affected by limits which are more restrictive than those calculated in accordance with Appendix I, the licensee may apply to the Commission for appropriate Technical Specification changes on a case by case basis.

Specification 2.4.1.a requires the licensee to limit the concentration of radioactive materials in liquid waste effluents released from the site to levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas. This specification provides assurance that no member of the general public will be exposed to liquid containing radioactive materials in excess of limits considered permissible under the Commission's Regulations.

Specifications 2.4.1.b and 2.4.1.c establish the upper limits for the release of radioactive materials in liquid effluents. The intent of these Specifications is to permit the licensee the flexibility of operation to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the levels normally achievable when the plant and the liquid waste treatment systems are functioning as designed. Releases of up to these levels will result in concentrations of radioactive material in liquid waste effluents at small percentages of the limits specified in 10 CFR Part 20.

Specification 2.4.1.d requires that the licensee maintain and operate the equipment installed in the liquid waste systems to reduce the release of radioactive materials in liquid effluents to as low as practicable consistent with the requirements of 10 CFR Part 50.36a. Normal use and maintenance of installed equipment in the liquid waste system provides reasonable assurance that the quantity released will not exceed the design objective. In order to keep releases of radioactive materials as low as practicable, the specification requires operation of equipment whenever it appears that the projected cumulative discharge rate will exceed one-fourth of the design objective annual quantity during any calendar quarter.

Specification 2.4.1.e restricts the amount of radioactive material that could be inadvertently released to the environment to an amount that will not exceed the Technical Specification limit.

In addition to limiting conditions for operation listed under Specifications 2.4.1.b and 2.4.1.c, the reporting requirements of Specification 2.4.1.f delineate that the licensee shall identify the cause whenever the cumulative release of radioactive materials in liquid waste effluents exceeds one-half the design objective annual quantity during any calendar quarter and describe the proposed program of action to reduce such releases to design objective levels on a timely basis. This report must be filed within 30 days following the calendar quarter in which the release occurred.

Specification 2.4.1.g provides for reporting spillage or release events which, while below the limits of 10 CFR Part 20, could result in releases higher than the design objectives.

The sampling and monitoring requirements given under Specification 2.4.2 provide assurance that radioactive materials in liquid wastes are properly controlled and monitored in conformance with the requirements of Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50 and permit the licensee and the Commission to evaluate the plant's performance relative to radioactive liquid wastes released to the environment. Reports of the quantities of radioactive materials released in liquid waste effluents are furnished to the Commission semi-annually. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The points of release to the environment to be monitored in Section 2.4.2 include all the monitored release points listed in Table 2.4-3.

2.4.3 Specifications for Gaseous Waste Effluents

The terms used in these Specifications are as follows:

subscripts v, refers to vent releases

s, refers to stack releases

i, refers to individual noble gas nuclide

(Refer to Table 2.4-5 for the noble gas nuclides considered)

Q_T = the total noble gas release rate (Ci/sec)

= $\sum_i Q_i$ sum of the individual noble gas radionuclides determined to be present by isotopic analysis

$\overline{K}$ = the average total body dose factor due to gamma emission (rem/yr per Ci/sec)

$\overline{L}$ = the average skin dose factor due to beta emissions (rem/yr per Ci/sec)

$\overline{M}$ = the average air dose factor due to beta emissions (rad/yr per Ci/sec)

$\overline{N}$ = the average air dose factor due to gamma emissions (rad/yr per Ci/sec)

The values of $\overline{K}$, $\overline{L}$, $\overline{M}$, and $\overline{N}$ for the vent releases are determined from the isotopic analysis performed at the discharge of the steam jet air ejectors as delineated in Specification 2.4.4.c. The values of $\overline{K}$, $\overline{L}$, $\overline{M}$, and $\overline{N}$ for the stack are determined from the isotopic analysis performed at a point prior to dilution and discharge as delineated in Specification 2.4.4.c. New values should be determined each time isotopic analysis is required as follows:

$$\bar{K} = (1/Q_T) \sum_i Q_i K_i$$

$$\bar{L} = (1/Q_T) \sum_i Q_i L_i$$

$$\bar{M} = (1/Q_T) \sum_i Q_i M_i$$

$$\bar{N} = (1/Q_T) \sum_i Q_i N_i$$

where the values of K_i, L_i, M_i and N_i are provided in Table 2.4-5, and are site dependent gamma and beta dose factors.

Q = the measured release rate (Ci/sec) of the radioiodines and radioactive materials in particulate forms with half-lives greater than eight days.

a. (1) The release rate limit of noble gases from the site shall be such that

$$2.0\ (Q_{Tv}\bar{K}_v + Q_{Ts}\bar{K}_s) \leq 1$$

and

$$0.33 \left[Q_{Tv}(\bar{L}_v + 1.1\bar{N}_v) + Q_{Ts}(\bar{L}_s + 1.1\bar{N}_s)\right] \leq 1$$

(2) The release rate limit of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days, released to the environs as part of the gaseous wastes from the site shall be such that

$$3.7 \times 10^5\ Q_v + 2.5 \times 10^4\ Q_s \leq 1$$

b. Should any of the conditions of 2.4.3.b (1), (2), (3), (4), (5), or (6) be exceeded, the licensee shall take appropriate corrective action to bring the releases within these limits.

(1) The average release rate of noble gases from the site during any calendar quarter shall be such that

$$13\,(Q_{Tv}\bar{N}_v + Q_{Ts}\bar{N}_s) \leq 1$$

and

$$6.3(Q_{Tv}\bar{M}_v + Q_{Ts}\bar{M}_s) \leq 1$$

(2) The average release rate of noble gases from the site during any 12 consecutive months shall be

$$25(Q_{Tv}\bar{N}_v + Q_{Ts}\bar{N}_s) \leq 1$$

and

$$13(Q_{Tv}\bar{M}_v + Q_{Ts}\bar{M}_s) \leq 1$$

(3) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter shall be such that

$$13\,(3.7 \times 10^5 Q_v + 2.5 \times 10^4 Q_s) \leq 1$$

(4) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any period of 12 consecutive months shall be such that

$$25\,(3.7 \times 10^5 Q_v + 2.5 \times 10^4 Q_s) \leq 1$$

(5) The amount of iodine-131 released during any calendar quarter shall not exceed 2 Ci.

(6) The amount of iodine-131 released during any period of 12 consecutive months shall not exceed 4 Ci.

c. Should any of the conditions of 2.4.3.c(1), (2), or (3) below exist, the licensee shall make an investigation to identify the causes of the release rates, define and initiate a program of action to reduce the release rates to design objective levels listed in Section 2.4 and report these actions to the NRC within 30 days from the end of the quarter during which the releases occurred.

(1) If the average release rate of noble gases from the site during any calendar quarter is such that

$$50\,(Q_{Tv}\overline{N}_v + Q_{Ts}\overline{N}_s) > 1$$

or

$$25\,(Q_{Tv}\overline{M}_v + Q_{Ts}\overline{M}_s) > 1$$

(2) If the average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter is such that

$$50\,(3.7\times10^{5}Q_v + 2.5\times10^{4}Q_s) > 1$$

(3) If the amount of iodine-131 released during any calendar quarter is greater than 0.5 Ci.

d. Whenever gaseous wastes are being released from the offgas treatment system, at least one Main Stack monitor shall be operable and set to alarm and initiate automatic termination of offgas discharge prior to exceeding the limits of Specification 2.4.3.a above. The capability of each automatic isolation valve shall be demonstrated quarterly. If no Main Stack monitor is operating, releases from the offgas system shall be terminated within 10 hours.

e. During power operation the Reactor Building Ventilation System monitoring system shall be operable and set to alarm and initiate automatic termination of Reactor Building ventilation air discharge prior to exceeding the limits of Specification 2.4.3.a above. The capability of automatic isolation of the ventilation system shall be demonstrated quarterly. If the Reactor Building Ventilation System monitoring system is not operating, releases from the Reactor Building Ventilation System shall be terminated within 10 hours.

f. If the gross radioactivity release rate of noble gases at the steam jet air ejector monitors exceeds, for a period greater than 48 hours, the equivalent of 260,000 uCi/sec following a 30-minute decay, notify the NRC within thirty days.

g. The drywell shall be purged through the standby gas treatment system.

h. Except as specified in Specification 2.4.3.i below, at least one hydrogen monitor downstream of each operating recombiner shall be operable during power operation.

i. If the above specified downstream hydrogen monitors are not operable, offgas flow to the compressed storage subsystem shall be terminated.

j. The maximum gross radioactivity contained in one gas decay tank after 12 hours holdup that can be discharged directly to the environs shall be less than 22,000 curies of Xe-133 dose equivalent.

k. The mechanical condenser vacuum pump shall be capable of being isolated and secured on a signal of high radioactivity whenever the main steam line isolation valves are open or it shall be isolated.

l. At least once during each operating cycle automatic isolation of the mechanical condenser vacuum pump shall be verified.

m. An unplanned or uncontrolled offsite release of radioactive materials in gaseous effluents in excess of 5 curies of noble gas or 0.02 curie of radioiodine in gaseous form requires notification within 30 days.

2.4.4 Specifications for Gaseous Waste Sampling and Monitoring

a. Plant records shall be maintained of the sampling and analyses results. Summaries of the quantities of releases shall be included in the Effluent and Waste Disposal Semiannual Report. Estimates of the sampling and analytical error associated with each reported value should be included.

b. Whenever a Turbine Building roof exhauster is running, a continuous radioactive particulate and radioiodine analyzer shall be in operation on the turbine floor. The concentration of radiodine and particulate matter measured by this analyzer shall be used in conjunction with the design flow rate of the roof exhausters in determining turbine building release rates. These release rates shall be included in the "Q_V" terms in Specifications 2.4.3a through 2.4.3.c.

c. An isotopic analysis shall be made of a representative sample of gaseous activity at the discharge of the steam jet air ejectors and at a point prior to dilution and discharge.

    (1) at least monthly, and

    (2) following each refueling outage, and

    (3) if the gaseous waste monitors indicate an increase of greater than 50% in the steady state fission gas release after factoring out increases due to power changes.

d. The continuous monitors listed in Table 2-4-4 shall be calibrated at least quarterly by means of a known solid radioactive source which has been calibrated to a National Bureau of Standards source. Each monitor shall have a functional test at least monthly and an instrument check at least daily.

e. Sampling and analysis of radioactive material in gaseous waste, including particulate forms and radioiodines shall be performed in accordance with Table 2.4-2.

f. The hydrogen monitors shall be functionally tested monthly and calibrated quarterly with an appropriate gas mixture source. Each monitor shall have a sensor check at least daily.

g. Condenser air inleakage shall be evaluated weekly and used in conjunction with steam jet air ejector offgas isotopic analyses and Figure 2.4-1 to to determine that the limit of Specification 2.4.3.j is not exceeded.

Bases: The release of radioactive materials in gaseous waste effluents to unrestricted areas shall not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as practical in accordance with the requirements of 10 CFR Part 50.36a. These specifications provide reasonable assurance that the resulting annual air dose from the site due to gamma radiation will not exceed 10 mrad, the annual air dose from the site due to beta radiation will not exceed 20 mrad from noble gases, that no individual in an unrestricted area will receive an annual dose to the total body greater than 5 mrem or an annual skin dose to the total body greater than 15 mrem from noble gases, and that the annual dose to any organ of an individual from radioiodines and radioactive material in particulate form with half-lives greater than eight days will not exceed 15 mrem from the site.

At the same time these specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided with a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. Even with this operational flexibility, the annual releases will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience taking into account a combination of system variables including defective fuel, primary system leakage, and the performance of the various waste treatment systems.

Specification 2.4.3.a(1) limits the release rate of noble gases from the site so that the corresponding annual gamma and beta dose rate above background to an individual in an unrestricted area will not exceed 500 mrem to the total body or 3000 mrem to the skin in compliance with the limits of 10 CFR Part 20.

For Specification 2.4.3.a(1), gamma and beta dose factors for the individual noble gas radionuclides have been calculated for the plant gaseous release points and are provided in Table 2.4-5. The expressions used to calculate these dose factors are based on dose models derived in Section 7 of Meteorology and Atomic Energy-1968 and model techniques provided in Draft Regulatory Guide 1.AA.

Dose calculations have been made to determine the site boundary location with the highest anticipated dose rate from noble gases using on-site meteorological data and the dose expressions provided in Draft Regulatory Guide 1.AA. The dose expression considers the release point location, building wake effects, and the physical characteristics of the radionuclides.

The offsite location with the highest anticipated annual dose from released noble gases is 700 meters in the SSE direction.

The release rate Specifications for radioiodine and radioactive material in particulate form with half-lives greater than eight days are dependent on existing radionuclide pathways to man. The pathways which were examined for these Specifications are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, and 3) deposition onto grassy areas where milch animals graze with consumption of the milk by man, which was determined to be the most limiting pathway. Methods for estimating doses to the thyroid via these pathways are described in Draft Regulatory Guide 1.AA.

The offsite location with the highest anticipated thyroid dose rate from radioiodines and radioactive material in particulate form with half-lives greater than eight days was determined using on-site meteorological data and the expressions described in Draft Regulatory Guide 1.AA.

Specification 2.4.3.a(2) limits the release rate of radioiodines and radioactive material in particulate form with half-lives greater than eight days so that the corresponding annual thyroid dose via the most restrictive pathway is less than 1500 mrem.

For radioiodines and radioactive material in particulate form with half-lives greater than eight days, the most restrictive location is a dairy farm located 3700 meters in the NNE direction (vent $X/Q = 4.3 \times 10^{-7} sec/m^3$; stack $X/Q = 2.5 \times 10^{-8}$ sec/m^3).

Specification 2.4.3.b establishes upper offsite levels for the releases of noble gases and radioiodines and radioactive material in particulate form with half-lives greater than eight days at twice the design objective annual quantity during any calendar quarter, or four times the design objective annual quantity during any period of 12 consecutive months. In addition to the limiting conditions for operation of Specifications 2.4.3.a and 2.4.3.b, the reporting requirements of 2.4.3.c provide that the cause shall be identified whenever the release of gaseous effluents exceeds one-half the design objective annual quantity during any calendar quarter and that the proposed program of action to reduce such release rates to the design objectives shall be described.

Specification 2.4.3.d and 2.4.3.e are in accordance with Design Criterion 64 of Appendix A to 10 CFR Part 50.

Specification 2.4.3.f is intended to monitor the performance of the core. An increase in the activity levels of gaseous releases may be the result of defective fuel. Since core performance is of utmost importance in the resulting doses from accidents, a report must be filed within 30 days following the specified increase in activity level at the steam jet air ejector.

Specification 2.4.3.g requires that the drywell atmosphere receive treatment for the removal of gaseous iodine and particulates during purging.

Specification 2.4.3.h requires that hydrogen concentration upstream of the compressed radioactive gaseous storage tanks shall be monitored at all times.

Specification 2.4.3.i requires offgas flow to the compressed storage tanks to be terminated in the event that the hydrogen monitors downstream of the recombiners are inoperable. This prevents the possible accumulation of an explosive mixture in a gas storage tank.

Specification 2.4.3.j limits the maximum gross activity in one decay tank on the basis that accidental release of its contents to the environs by operator error after 12 hours decay should not result in exceeding the dose equivalent to the maximum quarterly release rate specified in Specification 2.4.3.c.1. Staff analysis of an elevated release under accident meteorology for a minimum release period of 8 hours indicated a release of 22,000 curies of Xe-133 or the dose equivalent would result in an air dose from the site of 20 mrad from noble gases.

Calculations have been performed to determine the relationship between steam jet air ejector offgas activity and composition and condenser air inleakage. These calculations were used to determine the curves presented in Figure 2.4-1. The results of the measurement of condenser air inleakage and the average air ejector offgas release rate are used in conjunction with the most recent offgas isotopic analysis to determine if the maximum permitted Xe-133 dose equivalent tank radioactivity contents may be exceeded. This analysis is adequate to initiate corrective action in the unlikely event that the tank radioactivity limit is being approached.

Specifications 2.4.3.K and 2.4.3.1 require that the mechanical vacuum pump be provided with automatic isolation capability to limit the release of activity from the main condenser during an accident.

Specification 2.4.3.m provides for reporting release events which, while below the limits of 10 CFR Part 20, could result in releases higher than the design objectives.

The sampling and monitoring requirements given under Specification 2.4.4 provide assurance that radioactive materials released in gaseous waste effluents are properly controlled and monitored in conformance with the requirements of Design Criteria 60 and 64. These requirements provide the data for the licensee and the Commission to evaluate the plant's performance relative to radioactive waste effluents released to the environment. Reports on the quantities of radioactive materials released in gaseous effluents are furnished to the Commission semiannually. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The points of release to the environment to be monitored in Section 2.4.4 include all the monitored release points as provided for in Table 2.4-4.

These Specifications are applicable for the interim period until the date that Specifications prepared in accordance with new Appendix I become effective. In some cases these Specifications are more restrictive than required by Appendix I. In the event that plant availability is adversely affected by limits which are more restrictive than those calculated in accordance with Appendix I, the licensee may apply to the Commission for appropriate Technical Specification changes on a case by case basis.

2.4.5 Specifications for Solid Waste Handling and Disposal

a. Measurements shall be made to determine or estimate the total curie quantity and principle radionuclide composition of all radioactive solid waste shipped offsite.

b. Summaries of radioactive solid waste shipments, volumes, principle radionuclides, and total curie quantity, shall be included in the Effluent and Waste Disposal Semiannual Report.

Bases: The requirements for solid radioactive waste handling and disposal given under Specification 2.4.5 provide assurance that solid radioactive materials stored at the plant and shipped offsite are packaged in conformance with 10 CFR Part 20, 10 CFR Part 71, and 49 CFR Parts 170-178.

TABLE 2.4-1

RADIOACTIVE LIQUID SAMPLING AND ANALYSIS

Liquid Source	Sampling Frequency	Type of Activity Analysis	Detectable Concentrations (uCi/ml)[a]
A. Monitor Tank Releases	Each Batch	Principal Gamma Emitters	5×10^{-7}[b]
	One Batch/Month	Dissolved Gases[e]	10^{-5}
	Weekly Composite[c]	Ba-La-140, I-131	10^{-6}
	Quarterly Composite[c]	Sr-89, Sr-90 H-3 Gross Alpha	5×10^{-8} 10^{-5} 10^{-7}
B. Primary Coolant	Weekly[d]	I-131, I-133	10^{-6}

[a] The detectability limits for activity analysis are based on the technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable, and when nuclides are measured below the stated limits, they should also be reported.

[b] For certain mixtures of gamma emitters, it may not be possible to measure radionuclides in concentrations near their sensitivity limits when other nuclides are present in the same in much greater concentrations. Under these circumstances, it will be more appropriate to calculate the concentrations of such radionuclides using measured ratios with those radionuclides which are routinely identified and measured.

[c] A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged.

[d] The power level and cleanup or purification flow rate at the sample time shall also be reported.

[e] For dissolved noble gases in water, assume a MPC of 4×10^{-5} uCi/ml of water.

TABLE 2.4-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS

Gaseous Source	Sampling Frequency	Type of Activity Analysis	Detectable Concentrations (uCi/ml)[a]
A. Drywell Purges	Within 24 hours of Each Purge	Particulate Gross Beta	10^{-11}
		I-131	10^{-12}
B. Main Stack and Reactor Building Vent Releases	Monthly (Gas Samples)	Principal Gamma Emitters[g]	10^{-4}[b, c]
		H-3[f]	10^{-6}
	Weekly (Charcoal Sample)	I-131	10^{-12}[d]
	Weekly Composite (Charcoal Sample)	I-133, I-135	10^{-10}
	Weekly (Particulates)	Principal Gamma Emitters (at least for Ba-La-140 and I-131)	10^{-11}[d]
	Quarterly Composite[e] (Particulates)	Gross Alpha Sr-89, Sr-90	10^{-11} 10^{-11}

[a] The above detectability limits for activity analysis are based on technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable, and when nuclides are measured below the stated limits, they should also be reported.

[b] Analyses shall also be performed following each refueling, startup, or similar operational occurrence which could alter the mixture of radionuclides.

[c] For certain mixtures of gamma emitters, it may not be possible to measure radionuclides at levels near their sensitivity limits when other nuclides are present in the sample at much higher levels. Under these circumstances, it will be more appropriate to calculate the levels of such radionuclides using observed ratios with those radionuclides which are measurable.

TABLE 2.4-2 Notes (continued)

[d] When the average daily gross radioactivity release rate exceeds that given in 2.4.3.c(1) or where the steady-state gross radioactivity release rate increases by 50% over the previous corresponding power level steady-state release rate, the iodine and particulate collection devices for the release point whose contribution exceeds 50% of these rates shall be removed and analyzed to determine the change in iodine-131 and particulate release rate. The analyses for this release point shall be done daily following such change until it is shown that a pattern exists which can be used to predict the release rate after which it may revert to weekly sampling.

[e] To be representative of the average quantities and concentrations of radioactive materials in particulate form released in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent streams.

[f] Calculated from H-3 concentration of the condensate.

[g] Isotopic analysis performed in accordance with Specification 2.4.4.c at the discharge of the steam jet air ejectors and at a point prior to dilution and discharge of gaseous waste from the offgas system. Concentrations of gross radioactivity in the Reactor Building vent are expected to be below the minimum detectable levels with the existing analytical equipment. Therefore, isotopic analyses of samples from the vent will not normally be performed.

TABLE 2.4-3

LOCATION OF EFFLUENT MONITORS AND SAMPLERS REQUIRED BY TECHNICAL SPECIFICATIONS

Release Point	Radiation Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Measurement						High Liquid Level Alarm
					Gross Activity	I	Dissolved Gases	Alpha	H-3	Isotopic Analysis	
Floor Drain Sample Tank				X		X	X	X	X	X	X
Laundry Drain Tank				X		X	X	X	X	X	X
Liquid Radwaste Discharge Pipe	X		X		X						
Service Water Discharge Pipe	X		X		X						

TABLE 2.4-4

BOILING WATER REACTOR GASEOUS WASTE SYSTEM

LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLERS REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Radiation Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Measurement				
					Noble Gas	I	Particulate	H-3	Alpha
Condenser/Air Ejector (before gas treatment system)	X	X	X	X	X				
Offgas Treatment System Effluent				X	X				
Main Stack	X	X	X	X	X	X	X	X[b]	X
Reactor Building Ventilation System	X	X	X	X[c]	X	X	X	X[b]	X
Turbine Building Operating Floor			X			X	X		
Mechanical Vacuum Pump[a]		X							

[a] Isolation on main steam line high radiation.

[b] Calculated from H-3 concentration of the condensate.

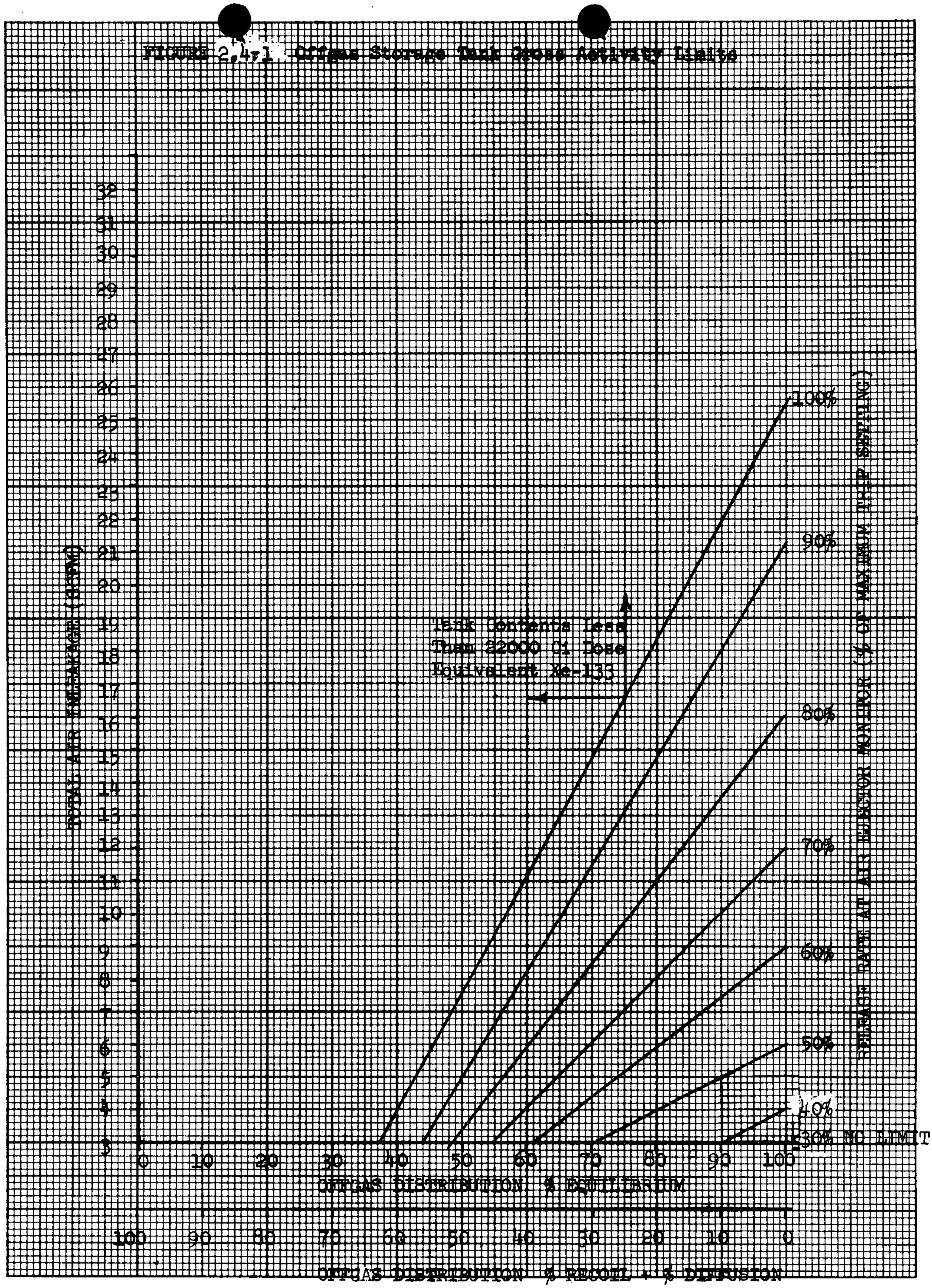
[c] For iodine and particulate only.

TABLE 2.4-5

GAMMA AND BETA DOSE FACTORS FOR

Monticello

Noble Gas Radionuclide	Doese Factors for Vent				Dose Factors For Stack			
	K_{iv} Total Body rem/yr / Ci/sec	L_{iv} Skin rem/yr / Ci/sec	M_{iv} Beta Air rad/yr / Ci/sec	N_{iv} Gamma Air rad/yr / Ci/sec	K_{is} Total Body rem/yr / Ci/sec	L_{is} Skin rem/yr / Ci/sec	M_{is} Beta Air rad/yr / Ci/sec	N_{is} Gamma Air rad/yr / Ci/sec
Kr-83m	2.0×10^{-4}	0	1.6	0.13	2.0×10^{-5}	0	0.063	5.3×10^{-3}
Kr-85m	2.0	8.0	11	2.1	0.59	0.32	0.43	0.6
Kr-85	0.023	7.4	11	0.024	8.6×10^{-3}	0.29	0.43	9.1×10^{-3}
Kr-87	6.6	54	57	6.9	2.5	2.1	2.3	2.7
Kr-88	15	13	16	16	6.3	0.52	0.64	6.6
Kr-89	9.4	56	58	9.9	2.1	2.2	2.3	2.2
Xe-131m	0.69	2.6	6.1	0.89	0.15	0.10	0.24	0.18
Xe-133m	0.54	5.5	8.1	0.75	0.12	0.22	0.33	0.14
Xe-133	0.63	1.7	5.8	0.79	0.13	0.067	0.23	0.14
Xe-135m	3.8	3.9	4.1	4.1	1.2	0.16	0.16	1.3
Xe-135	2.9	10	14	3.0	0.94	0.41	0.54	0.99
Xe-137	1.1	67	70	1.2	0.25	2.7	2.8	0.26
Xe-138	9.3	23	26	9.8	3.1	0.91	1.0	3.2



TSB FIGURE 2.4-1