NOV 15 1973

Docket No. 50-263

Northern States Power Company
ATTN: Mr. L. O. Mayer, Director of
Nuclear Support Services
414 Nicollet Mall
Minneapolis, Minnesota 55401

Change No. 12
License No. DPR-22

Gentlemen:

Your letters dated October 26 and October 31, 1973, proposed changes to the Technical Specifications of Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant that would correct errors and inadequacies, correct discrepancies in Change No. 2 to the Technical Specifications dated January 14, 1972, and incorporate changes into the Technical Specifications which are necessary to permit operation of the Off-Gas Holdup System.

During our review, we informed your staff that certain modifications to the proposed changes were necessary to meet Regulatory requirements. These modifications have been made.

We have reviewed your request and have determined that the proposed changes do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. A copy of our related Safety Evaluation is enclosed.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to Provisional Operating License No. DPR-22 are hereby changed by deleting Change No. 2 dated January 14, 1972, and by replacing the existing pages 7, 10, 15, 30, 48, 49, 51, 62, 68, 70, 92, 93, 134, 150, 164, 168-170, 173, 173A, 177-179, 216, and 216A with the revised pages enclosed herewith.

Further discussions will be held between our staffs regarding additional operating procedures and technical specifications for operation of your radioactive waste treatment system for both liquid and off-gas releases

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to meet as low as practical considerations prior to issuance of a full-term license. The enclosed changes provide the technical specifications necessary for initial operation of the augmented off-gas system.

Sincerely,

Original Signed by
D. J. Skovholt

Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing

Enclosures:
1. Safety Evaluation
2. Revised pages: 7, 10, 15, 30, 48, 48A, 49, 51, 62, 68, 70, 92, 93, 134, 150, 164, 168, 169, 170, 170A, 170B, 173, 173A, 177, 177A, 178, 178A, 179, 179A, 179B, 216, and 216A

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NAME ▶	FDAnderson:sjh	RMDiggs	DLZiemann	DJSkovholt		
DATE ▶	11/13/73	11/13/73	11/15/73	11/15/73		

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

CHANGE NO. 12 TO TECHNICAL SPECIFICATIONS

By letters dated October 26 and October 31, 1973, Northern States Power Company (NSP) proposed changes to the Technical Specifications of Provisional Operating License No. DPR-22 for the Monticello Nuclear Generating Plant that would correct errors and inadequacies, correct discrepancies in Change No. 2 to the Technical Specifications dated January 14, 1972, and incorporate revisions to the Technical Specifications necessary for the operation of the augmented off-gas system. We have reviewed the proposed changes and the reasons for the changes submitted by NSP. Our evaluation and discussion of these changes is presented in the order of the Technical Specifications pages to be changed.

We compared the proposed change to the fuel cladding integrity safety limit (Specification 2.1.B) of 5 percent design core flow with other BWR plants of similar design, such as Vermont Yankee. We concluded that the proposed change is acceptable and is consistent with the requirements of other BWR plants.

We reviewed the proposed addition to Table 3.1.1 of a Limiting Trip Setting on the APRM downscale trip. The proposed setting should be "greater than or equal to 3/125 of full scale" rather than "less than or equal to 3/125 of full scale" as proposed. With this modification, the trip setting is consistent with the function of the APRM and is an acceptable trip setting limit. The NSP staff has been consulted and has agreed with this necessary modification.

The changes to be incorporated into the Technical Specifications by Change No. 2 dated January 14, 1972, have been compared with the proposed changes incorporated herein. We have concluded that Change No. 2, which was approved to be effective following installation of the augmented off-gas system, should be deleted in its entirety. The changes proposed by the October 31, 1973 submittal and modified by us should be incorporated into the Technical Specifications to replace those changes approved for incorporation by Change No. 2. These changes will be consistent with current Regulatory requirements for the augmented off-gas system.

We have reviewed the proposed changes to Specifications 3.2.D, 3.2.E.2, and Table 4.2.1, including Bases, regarding the radiation monitoring requirements and set points for the steam jet air ejector monitors and the reactor building vent monitors. We have modified the limit set point (Specification 3.2.D.1) for the air ejector monitors to reflect the AEC staff's changes to Specification 3.8.A.1 on gaseous effluent limits. The proposed lowering of the trip set point for the ventilation plenum is consistent with the gaseous effluent limit for the reactor building vent releases as determined by the AEC staff to meet 10 CFR Part 20 limits. The proposed change to Table 4.2.1 clarifies the calibration method to be used. The Bases have been modified to reflect our changes and justify the limits set by the proposed changes.

Tables 3.2.1 and 3.2.5 regarding HPCI High Steam Flow trip setting and time delay setting and deviations to the APRM and RBM trip settings have been revised to clarify the operation of the existing trip system.

We have reviewed the proposed changes to Figures 3.4.1 and 3.4.2 regarding liquid poison volume concentration requirements. We have concluded that the proposed changes are consistent with current Regulatory requirements and identify the chemical form of the liquid poison.

The changes to Specifications 4.6.D and 4.7.C.1.a-c with Bases which delete reporting requirements that have been fulfilled and add reporting requirements on system surveillance are appropriate and necessary to meet Regulatory requirements. Other minor changes to these technical specifications are necessary for clarification.

We have reviewed the proposed changes to Specifications 3.8 and 4.8 regarding radioactive effluents, except for the liquid effluents. Although we agreed with the intent of the proposed changes, i.e., to reflect as low as practical limits for the airborne effluents with the augmented off-gas system operating, we made major modifications to the technical specifications to include current Regulatory

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requirements on release rates, current dose analysis methods, AEC staff meteorological models, and current surveillance requirements. The Bases for the technical specifications relative to these modifications have been revised completely by the AEC staff. NSP representatives have agreed with these modifications.

The proposed change to Specification 6.7.A.2.i is consistent with current Regulatory requirements for reporting of occupational personnel radiation exposure and, therefore, is acceptable.

On the basis of our evaluation, we have concluded that the proposed changes, as modified, do not present significant hazards considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The Technical Specifications should be changed as proposed by NSP and modified by the AEC staff.

/s/

Fredric D. Anderson
Operating Reactors Branch #2
Directorate of Licensing

/s/

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Directorate of Licensing

Date: NOV 15 1973

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2.0 SAFETY LIMITS	LIMITING SAFETY SYSTEM SETTINGS
	$S = \frac{486,000\ P}{X}$ Where: P = percent of rated power X = peak heat flux - ($BTU/HR/FT^2$) shall be used. 2. IRM--Flux Scram setting shall be ≤ 15% of rated neutron flux.
B. When the reactor pressure is less than 600 psig or core flow is less than 5% of design, the reactor thermal power transferred to the coolant shall not exceed 300 MW.	B. APRM Rod Block - The APRM rod block setting shall be as shown in Figure 2.3.1 unless the combination of power and peak heat flux is above the curve in Figure 2.3.2. When the combination of power and peak flux is above the curve in Figure 2.3.2, a rod block trip setting (RB) as given by: $RB = \frac{437,400P}{X}$ where: P = percent of rated power X = peak heat flux ($BTU/HR/FT^2$) shall be used.
C. 1. The neutron flux shall not exceed the scram setting established in Specification 2.3.A for longer than 0.95 seconds as indicated by the process computer.	C. Reactor Low Water Level Scram setting shall be ≥ 10'6" above the top of the active fuel.

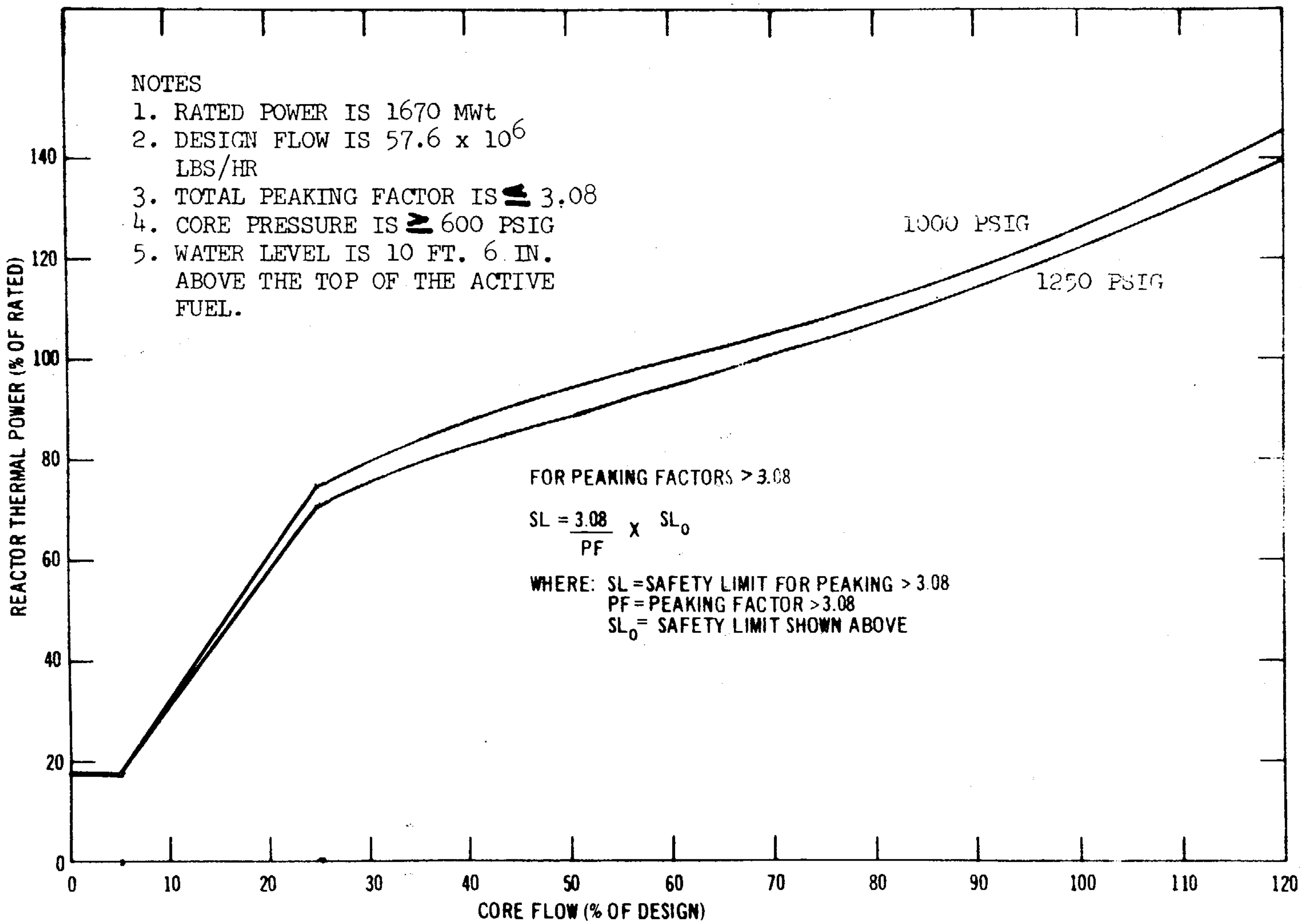

FIGURE 2.1.1 FUEL CLADDING INTEGRITY SAFETY LIMIT

Bases Continued:

The feedwater temperature assumed was the maximum design temperature output of the feedwater heaters at the given pressures and flows, which is 376°F for rated thermal power. For any lower feedwater temperature, sub-cooling is increased and the curves are conservative.

The water level assumed in the calculation of the safety limit was that level corresponding to the bottom of the steam separator skirt (7" on the level instrument is equivalent to 10'6" above the top of the active fuel at rated power). As long as the water level is above this point, the safety limit curves are applicable; i.e., the amount of steam carry under would not be increased, and, therefore, the core inlet enthalpy and sub-cooling would not be influenced.

The values of the parameters involved in Figure 2.1.1 can be determined from information available in the control room. Reactor pressure and flow are recorded and the Average Power Range Monitor (APRM) in-core nuclear instrumentation is calibrated to read in terms of percent power.

The range in pressure and flow used for Specification 2.1.A was 600 psig to 1250 psig and 5% to 100% flow respectively. Specification 2.1.B requires a restriction on power level when operating below 600 psig or 5% flow. In general, Specification 2.1.B will only be applicable during startup or shutdown of the plant. A review of all the applicable low pressure and low flow data (2, 3) has shown the lowest data point for transition boiling to have a heat flux of 144,000 BTU/HR/Ft^2. To assure applicability to Monticello fuel geometry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., critical heat flux was assumed to occur for these conditions at 72,000 BTU/HR/Ft^2. Assuming a peaking factor of 3.08, this is equivalent to a core average power of approximately 300 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions, there is increased margin.

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(2) E. Janssen - "Multirod Burnout at Low Pressure" - ASME Paper 2-HT-26, August 1962.
(3) K. M. Becker - "Burnout Conditions for Flow of Boiling Water in Vertical Rod Clusters" - AE-74 (Stockholm, Sweden), May, 1962.

TABLE 3.1.1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT REQUIREMENTS

Trip Function	Limiting Trip Settings	Modes in which function must be Operable or Operating**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Condition
		Refuel(3)	Startup	Run			
1. Mode Switch in Shutdown		x	x	x	1	1	A
2. Manual Scram		x	x	x	1	1	A
3. Neutron Flux IRM (See Note 2) a. High-High b. Inoperative	≤120/125 of full scale	x	x	x(c)	4	3	A
4. Flow Referenced Neutron Flux APRM (See Note 5) a. High-High b. Inoperative c. Downscale	See Specifications 2.3A.1 ≥3/125 of full scale			x	3	2	A or B
5. High Reactor Pressure	≤1075 psig	x	x(f)	x(f)	2	2	A
6. High Drywell Pressure	≤2 psig	x(4)	x(e,f)	x(e,f)	2	2	A
7. Reactor Low Water Level	≥7 in.(6)	x	x(f)	x(f)	2	2	A
8. Scram Discharge Volume High Level	≤32 gal.(8)	x(a)	x(f)	x(f)	2	2	A
9. Turbine Condenser Low Vacuum	≥23 in. Hg	x(b)	x(b,f)	x(f)	2	2	A or C

3.0 LIMITING CONDITIONS FOR OPERATION

4.0 SURVEILLANCE REQUIREMENTS

B. Emergency Core Cooling Subsystems Actuation

When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling sybsystems are given in Table 3.2.2.

C. Control Rod Block Actuation

The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3.

D. Air Ejector Off-Gas System

1. Except as specified in 3.2.D.2 and 3.2.D.3, both steam jet air ejector off-gas radiation monitors shall be operable during reactor power operation. The trip settings for the air ejector monitors, except as specified in 3.2.D.4, shall be set to close within 30 minutes the recombiner train inlet valve(s) at a level not to exceed the equivalent of the limits stated in Specification 3.8.A.1 for the off-gas stack after a decay time of 30 minutes.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. From and after the date that one of the two steam jet air ejector off-gas radiation monitors is made or found to be inoperable, continued reactor power operation is permissible provided the inoperable radiation monitor instrument channel is tripped. 3. Upon loss of both steam jet air ejector off-gas radiation monitors, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours. 4. If operation is necessary with the Off-Gas Holdup System recombiners bypassed, the steam jet air ejector radiation monitors shall be set to close the off-gas isolation valve instead of the recombiner inlet valves with a delay time not to exceed 15 minutes.	

Table 3.2.1 - Continued

Function	Trip Settings	Total No. of Instrument Channels Per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (1,2)	Required Conditions
b. High Drywell Pressure (5)	≤ 2 psig	2	2	D
3. Reactor Cleanup System (Group 3)				
a. Low Reactor Water Level	≥10'6" above the top of the active fuel	2	2	E
4. HPCI Steam Lines				
a. HPCI High Steam Flow	≤150,000 lb/hr with ≤60 second time delay	2(4)	2	F
b. HPCI High Steam Flow	≤300,000 lb/hr	2(4)	2	F
c. HPCI Steam Line Area High Temp.	≤200°F	16(4)	16	F
5. RCIC Steam Lines				
a. RCIC High Steam Flow	≤45,000 lb/hr	2(4)	2	G
b. RCIC Steam Line Area High Temp.	≤200°F	16(4)	16	G

Table 4.2.1 - Continued

Minimum Test and Calibration Frequency For Core Cooling, Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Note 1	Once/3 months	None
4. Steam Line High Radiation	Once/week (5)	Note 6	Once/shift
HPCI ISOLATION			
1. Steam Line High Flow	Note 1	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once/3 months	None
RCIC ISOLATION			
1. Steam Line High Flow	Note 1	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once 3/months	None
REACTOR BUILDING VENTILATION			
1. Radiation Monitors (Plenum)	Note 1	Once/3 months	Once/shift
2. Radiation Monitors (Refueling Floor)	Note 1	Once/3 months	(4)
OFF-GAS ISOLATION			
1. Radiation Monitors (Air Ejectors)	Notes (1,5)	Note 6	Once/shift

NOTES:

(1) Initially once per month until exposure hours (M as defined on Figures 4.1.1) is 2.0×10^5, thereafter according to Figure 4.1.1, with an interval not greater than three months.

Bases Continued:

3.2 For effective emergency core cooling for the small pipe break the HPCI or Automatic Pressure Relief system must function since for these breaks, reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria is met. Reference Section 6.2.4 and 6.2.6 FSAR. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high trip point or one has an upscale trip and the other a downscale trip or two downscale. There is a 30-minute delay before recombiner train inlet valve closure when the recombiners are in use and a 15-minute delay before off-gas isolation valve closure when the recombiners are bypassed in which the reactor operator may take corrective action. Both instruments are required for trip. The trip settings of the instruments are set so that the maximum stack release rate limit allowed by Specification 3.8.A.1 is not exceeded.

Four radiation monitors are provided which initiate isolation of the reactor building and operation of the standby gas treatment system. The monitors are located in the reactor building ventilation plenum and on the refueling floor. Any one upscale trip will cause the desired action. Trip settings of 3 mR/hr for the monitors in the ventilation duct are based upon initiating normal ventilation isolation and Standby Gas Treatment System operation so as not to exceed the maximum release rate limit allowed by Specification 3.8.A.1 for the reactor building vent. Trip settings of 100 mR/hr for the monitors on the refueling floor are based upon initiating normal ventilation isolation and standby gas treatment system operation so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

Although the operator will set the set points within the trip settings specified in Tables 3.2.1, 3.2.2, 3.2.3, and 3.2.4, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the following amounts.

Table 3.2.5 - Continued
Trip Function and Deviations

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	Low-Low Reactor Water Level	-3 Inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale	-2/125 of Scale
	IRM Upscale	+2/125 of Scale
	APRM Downscale	-2/125 of Scale
	APRM Upscale	See Basis 2.3 - Page 24
	RBM Downscale	-2/125 of Scale
	RBM Upscale	Same as APRM Upscale

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

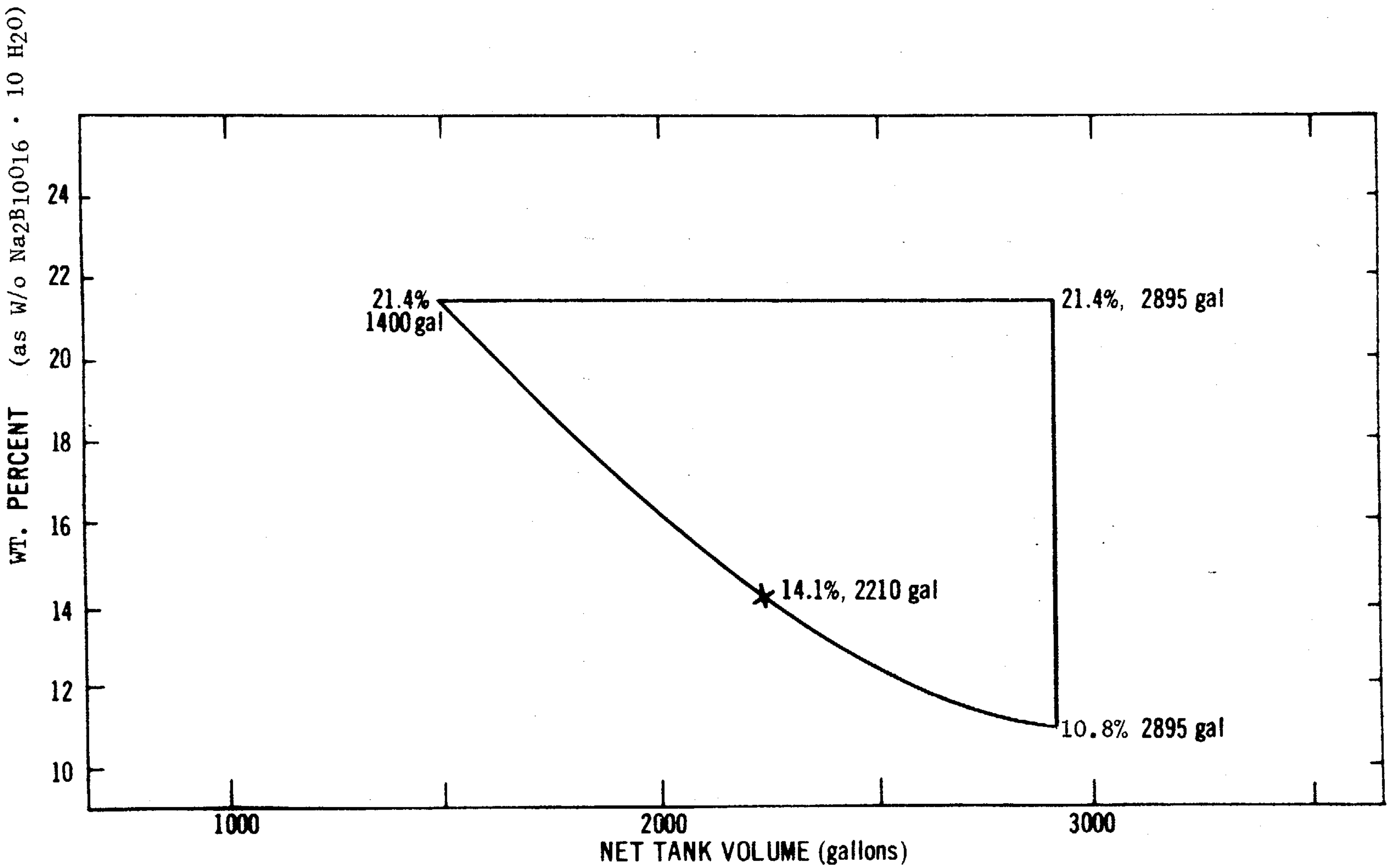

FIGURE 3.4.1. Sodium Pentaborate Solution Volume — Concentration Requirements

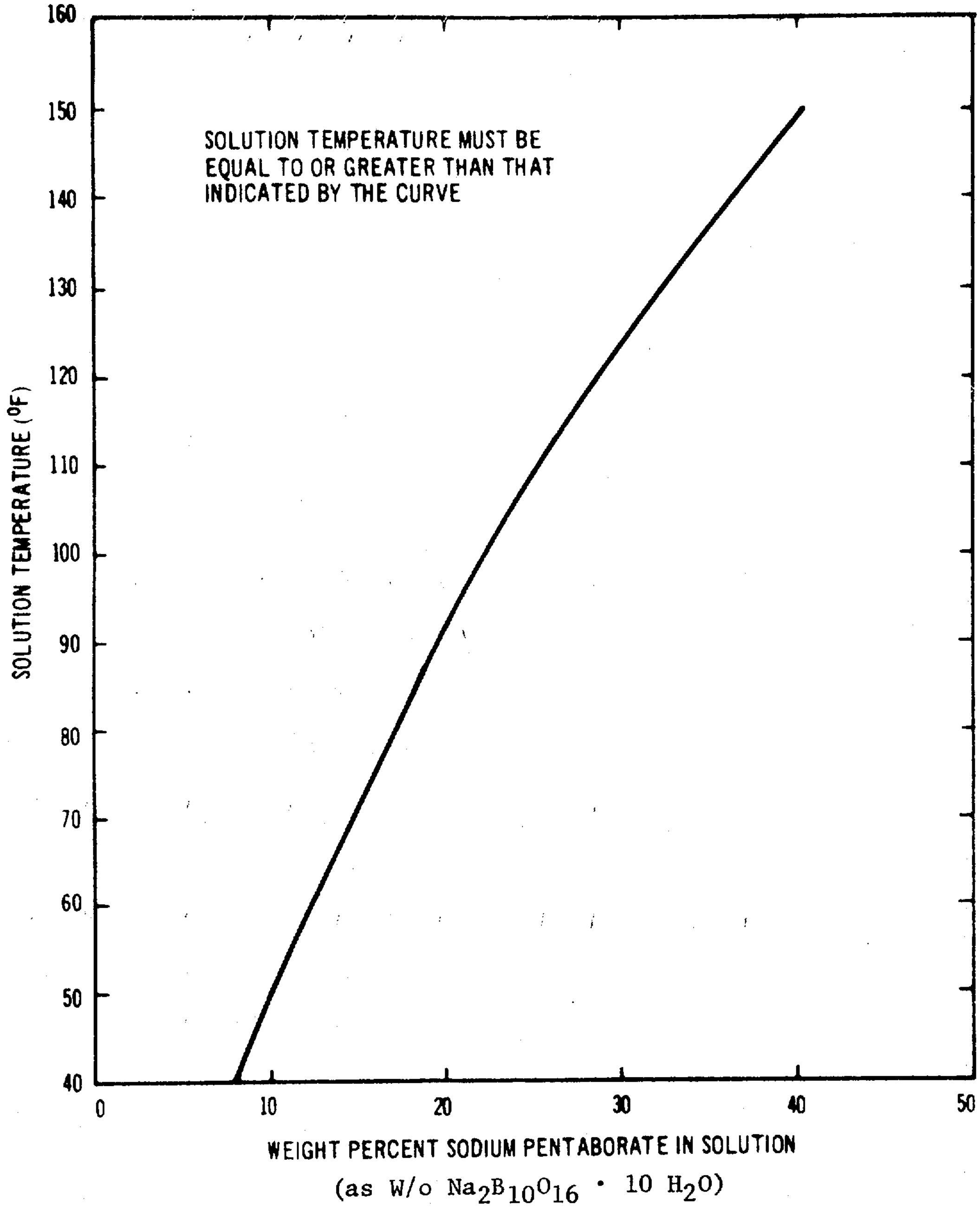

FIGURE 3.42 SODIUM PENTABORATE SOLUTION TEMPERATURE REQUIREMENTS

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such leakage was coming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

The capacity of the drywell sump pumps is 100 gpm and the capacity of the drywell equipment drain tank pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

An annual report will be prepared and submitted to the AEC summarizing the primary coolant to drywell leakage measurements. Other techniques for detecting leaks and the applicability of these techniques to the Monticello Plant will be the subject of continued study.

E. Safety and Relief Valves

Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. A tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as $\pm 1\%$ of the set pressure. An analysis has been performed which shows that with all safety valves set 1% higher than the set pressure, the reactor coolant pressure safety limit of 1375 psig is not exceeded. Safety/relief valves are used to minimize activation of the safety valves. The operator will set the pressure settings at or below the settings listed. However, the actual set points can vary as listed in the basis of Specification 2.4.

The required safety valve steam flow capacity is determined by analyzing the pressure rise accompanying the main steam flow stoppage resulting from a MSIV closure with the reactor at 1670 MWt. The analysis assumes no MSIV closure scram, but a reactor scram from indirect means (high flux). The relief and safety valve capacity is assumed to total 83.9% (47% relief and 36.9% safety) of the full power steam generation rate. This capacity corresponds to assuming that four safety/relief valves (47%) and four safety valves (36.9%) operated.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
C. Secondary Containment	C. Secondary Containment

C. Secondary Containment

1. Secondary containment integrity, shall be maintained during all modes of plant operation except when all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.3.A is met.

   b. The reactor water temperature is below 212° and the reactor coolant system is vented.

   c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

C. Secondary Containment

1. Secondary containment surveillance shall be performed as indicated below:

   a. Secondary containment capability to maintain at least a 1/4 inch of water vacuum under calm wind ($2 < u < 5$ mph) conditions with a filter train flow rate of $\leq$4,000 scfm, shall be demonstrated at each refueling outage prior to refueling. This surveillance testing should be reported in the semiannual operating reports.

Bases Continued:

The acceptable values for local leak rate tests have been specified in terms of standard cubic feet per hour (scf/hr) for purposes of clarity. Following is the list of equivalent values given in terms of an allowable percentage of the allowable operational leak rate (L_{to}).

17.2 scf/hr = 5% L_{to}
@ 41 psig

34.4 scf/hr = 10% L_{to}
@ 41 psig

103.2 scf/hr = 30% L_{to}
@ 41 psig

where L_{to} = .75 L_t (the maximum allowable leak rate)
and L_t = 1.2 weight percent of the contained air at the test pressure of 41 psig.

Results of loss of coolant accident analyses indicate that fission products would not be released directly to the environs because of leakage through the main line isolation valves due to holdup in the steam system complex. Although this effect shows that an adequate margin exists with regard to release of fission products, the results of leak tests on the main steam line isolation valves will be closely followed in order to determine the adequacy of these valves to perform their intended function. **A summary report of the results of main steam line isolation valve leakage tests and closure time measurements will be prepared and submitted to the AEC following completion of periodic main steam line isolation valve leakage tests.**

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

3.0 LIMITING CONDITIONS FOR OPERATION

3.8 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the gaseous and liquid radioactive effluents from the plant.

Objective:

To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that any material released is kept as low as practicable and, in any event, is within the limits of 10CFR Part 20.

Specification:

A. Airborne Effluents

A set of equations are given to express the airborne effluent limits. The symbols stand for the following:

Q1 = release rate from the off-gas stack

QRS = release rate from the reactor building vent

4.0 SURVEILLANCE REQUIREMENTS

4.8 RADIOACTIVE EFFLUENTS

Applicability:

Applies to the periodic monitoring and recording of radioactive effluents.

Objective:

To ascertain that radioactive releases are being kept as low as practicable and within allowable values.

Specification:

A. Airborne Effluents

1. The maximum release rates of gross radioactivity shall not exceed a rate Q, in curies/sec:

$$Q1\left(\frac{\overline{E}_{\gamma}}{0.18}\right) + Q_{RS}\left(\frac{\overline{E}_{\gamma}}{0.028} + \frac{\overline{E}_{\beta}}{0.019}\right) \leq 1$$

2. The release rates of gross radioactivity shall not exceed 16 percent of the limit in Specification 3.8.A.1 averaged over any calendar quarter.

3. The maximum release rate of radioiodine 131 (I-131) shall not exceed a rate Q, in microcuries/sec:

$$\frac{Q1}{25} + \frac{QRS}{0.9} \leq 1$$

4. The release rate of I-131 shall not exceed 4 percent of the limit in Specification 3.8.A.3 averaged over any calendar quarter.

5. The maximum release rates of radioactive particulates with half-lives greater than 8 days shall not exceed a rate Q, in microcuries/sec:

$$\frac{Q1}{9.5\times10^{9}\ \overline{MPC}_a} + \frac{Q_{RS}}{2\times10^{8}\ \overline{MPC}_a} \leq 1$$

where $\overline{MPC}_a$ is the composite maximum permissible concentration in air in uCi/ml determined using Appendix B, Table II, Column 1 and Notes of 10 CFR 20.

1. Radioactive gases released from the off-gas stack and reactor building vent shall be continuously monitored. Station records of off-gas stack release rates of gross gaseous radioactivity shall be maintained on an hourly basis to assure that the specified rates are not being exceeded, and to yield information concerning general integrity of the fuel cladding. Records of isotopic analysis shall be maintained. The off-gas stack and reactor building vent monitoring system shall be functionally tested monthly and calibrated quarterly with an appropriate standard radiation source. Each monitor, as described, shall have a sensor check at least daily.

2. A steam jet air ejector off-gas sample shall be taken and an isotopic analysis for at least six fission product gases; Xe-138, Xe-135, Xe-133, Kr-88, Kr-85m, Kr-87 shall be made at least weekly and following each refueling or other occurrence which could alter significantl the mixture of radionuclides.

6. The release rates of radioactive particulates with half-lives greater than 8 days shall not exceed 8 percent of the limit in Specification 3.8.A.5 averaged over any calendar quarter.

7. If the maximum release rate limits of Specifications 3.8.A.1, 3.8.A.3, or 3.8.A.5 are not met following a routine surveillance check, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

8. If the limits of Specification 3.8.A.2, 3.8.A.4, or 3.8.A.6 are exceeded, appropriate corrective action such as an orderly reduction of power shall be initiated to bring the releases within these limits.

9. If the release rates exceed four percent of the limits in Specification 3.8.A.1 averaged over any calendar quarter or two percent of the limits in Specifications 2.8.A.3 or 3.8.A.5 averaged over any calendar quarter, the following actions shall be taken:

3. Gaseous release of tritium shall be calculated on a quarterly basis from tritium concentration of the condensate. Vaporous tritium shall be calculated from a representative sample. The sum of these two values shall be reported as the total tritium release.

4. Radioiodine and radioactive particulates with half-lives greater than 8 days released from the off-gas stack and reactor building vent shall be continuously sampled. Station records of release of all radioiodine 131 and particulates with half-lives greater than 8 days shall be maintained on the basis of all stack and vent cartridges counted. The charcoal cartridges shall be counted weekly when the measured release rate of radioiodine 131 activity is less than the rate of Specification 3.8.A.4; otherwise the cartridges shall be counted daily. The particulate filters shall be counted weekly when the measured release rate of particulate radioactivity with half-lives greater than 8 days is less than the rate of Specification 3.8.A.6; otherwise the activity shall be counted daily.

3.0 LIMITING CONDITIONS OF OPERATION	4.0 SURVEILLANCE REQUIREMENTS
a. Investigate to identify the causes for such release rates. b. Define and initiate a program to reduce such release rates to the as low as practical levels. c. Provide a report describing these actions within 30 days as an unusual event (See Specification 6.7.B.2). 10. At least one of the two stack monitors, including the charcoal cartridge and particulate filter, shall be operable at all times that the stack is releasing effluents to the environs. 11. If both stack monitors are made or found inoperable, the reactor shall be placed in the hot standby condition within 24 hours. 12. Except as specified in 3.8.A.13, the off-gas stack and reactor building vent monitors shall have automatic isolation set points consistent with Specification 3.8.A.1 and alarm set points consistent with Specification 3.8.A.2. 13. If operation is necessary with the Off-gas Holdup System recombiners bypassed, the off-gas stack monitors shall serve only an alarm function.	5. A determination shall be made of the total I-131 released weekly. An analysis shall be performed on a sample at least monthly for I-133 and I-135. 6. A determination shall be made of the total radioactive particulates with half-lives greater than 8 days released weekly. The particulate filters shall be removed and analyzed for gross beta particulate radioactivity with half-lives greater than 8 days. Monthly, a composite of those filters used during the month shall be prepared and analyzed for the principal gamma emitting radionuclides. 7. Analysis for Sr-89 and Sr-90 shall be made quarterly. Gross alpha radioactivity shall be determined quarterly.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Mechanical Condenser Vacuum Pump

1. 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.

2. If the limits of 3.8.B.1 are not met following a routine surveillance check, the mechanical condenser vacuum pump shall be kept in an isolated condition until repairs are made.

4.0 SURVEILLANCE REQUIREMENTS

B. Mechanical Condenser Vacuum Pump

1. At least once during each operating cycle, verify automatic isolation of the mechanical condenser vacuum pump.

3.0 LIMITING CONDITIONS FOR OPERATION

3. Two independent samples of each tank shall be taken and analyzed for gross beta-gamma activity and the valve line-up checked prior to discharge of liquid effluents.

4. If the limits of 3.8.C cannot be met, radioactive liquid effluents shall not be released.

D. Radioactive Liquid Storage

The maximum gross radioactivity in liquid storage in the Waste Sample, Floor Drain Sample, Waste Surge, and Condensate Storage Tanks shall be less than 30 curies except for tritium and dissolved noble gases. If this condition cannot be met, the liquids in these tanks shall be recycled to tanks within the radwaste facility until the condition is met.

E. Augmented Off-Gas System

1. If the hydrogen concentration in the off-gas downstream of the recombiners reaches four percent, the recombiner off-gas flow shall be stopped automatically by closing the valves upstream of the recombiners.

2. Except as specified in Specification 3.8.E.3 below, at least one hydrogen monitor upstream and one hydrogen monitor downstream of each operating recombiner shall be operable during power operation.

4.0 SURVEILLANCE REQUIREMENTS

3. The performance and results of independent samples and valve checks shall be logged.

D. Radioactive Liquid Storage

1. A sample shall be taken, analyzed, and recorded within 72 hours of each addition to a liquid waste storage tank to which Specification 3.8.D. applies.

2. If the sample analysis indicates that the total radioactivity in the liquid waste storage tanks of Specification 3.8.D exceeds 30 curies, except for tritium and dissolved noble gases, the liquids in these tanks shall be recycled to reduce the radioactivity to less than 30 curies within 24 hours of this sampling.

E. Augmented Off-Gas System

1. 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.

2. Tank radiation monitors shall be calibrated quarterly by correlation with tank sample analyses. Monitor readings shall be recorded every eight hours to determine that the limit of Specification 3.8.E.4 is not exceeded.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3. If the above specified upstream hydrogen monitors are not operable, continued operation of a recombiner is permissible if the Hydrogen Inventory Processor is set to provide a constant signal representative of the worse case hydrogen concentration. If the above specified downstream hydrogen monitors are not operable, an orderly reactor shutdown shall be initiated to transfer the Off-gas System to the recombiner bypass mode.

4. The maximum gross radioactivity contained in one gas decay tank after 12 hours hold-up that can be discharged directly to the environs shall be less than 22,000 curies of Xe-133 dose equivalent. If these conditions cannot be met, the stored radioactive gas shall be recycled within 24 hours to other gas decay tanks until the condition is met.

5. During normal plant operation, radioactive gaseous waste shall have a minimum holdup of 12 hours except for low radioactivity gaseous waste resulting from purge and fill operations associated with refueling and reactor startup. Holdup times for radioactive gaseous waste in the gas decay tanks shall be maximized consistent with plant operation.

3. If a tank radiation monitor is inoperable, a sample from the gas decay tank shall be taken, analyzed, and recorded every 24 hours. If no additions to a tank have occurred since the last sample, the tank need not be sampled until the next addition.

F. Environmental Monitoring Program

The environmental monitoring program given in Table 4.8.1 shall be conducted.

Bases:

A. Airborne Effluents

Detailed dose calculations for several locations off site have been made and are described in Appendix B of the FSAR. These calculations consider site meteorology, buoyancy characteristics, and isotopic content of the effluent. Independent dose calculations for several locations off site have been made by the AEC staff, and the most critical one was chosen to set the maximum release rate. This point is 600 meters to the south-southeast at the site boundary. The method utilized onsite meteorological data developed by the licensee and utilized diffusion assumptions appropriate to the site.

The method utilized by the staff is described in Section 7-5.2.5 of "Meteorology and Atomic Energy - 1968", equation 7.63 being used. The results of these calculations are conservative and thus chosen to be used as the basis of establishment of the limits. Based on these calculations, a continuous release rate of gross radioactivity in the amount of $0.18/\bar{E}_\gamma$ curies/sec from the off-gas stack would not result in annual whole body doses in excess of the limits specified in 10 CFR Part 20 of 500 mRem. The $\bar{E}_\gamma$ determination need consider only the average gamma energy per disintegration since the controlling whole body dose is due to the cloud passage over the receptor and not cloud submersion in which the beta dose could be additive for a skin dose.

The dose analysis performed by the AEC staff for radioactive releases from the reactor building vent included an evaluation of the beta dose as well as the gamma dose. The staff assumed that such releases would be equivalent to ground level releases which could result in a beta dose from cloud submersion. The methods utilized are the same as used for the off-gas stack releases to determine the gamma dose contribution while the beta dose contribution was determined using the method described in Section 7.4 of "Meteorology and Atomic Energy - 1968", equation 7.21 being used. Therefore, the gamma dose contribution was determined on the basis of a finite cloud passage and the beta dose contribution on the basis of a semi-infinite cloud submersion, both for a ground level release. Based on these calculations, a continuous release rate of gross radioactivity in the amount of $0.028/(\bar{E}_\gamma + 1.5\ \bar{E}_\beta)$ curie/sec from the reactor building vent will not result in offsite annual doses in excess of the limits specified in 10 CFR Part 20 of 500 mRem for the whole body or 1500 mRem for the skin in the most critical sector (south-southeast) at the site boundary (600 m).

In order to limit gross radioactivity releases in gaseous effluents to as low as practicable, quarterly average release rates have been established which would require investigative actions at 4 percent of the maximum release rate and plant actions at 16 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 8, respectively, above the as low as practical objectives of 2 percent of 10 CFR Part 20 limits.

Detailed meteorological calculations for several locations off site have been made by the AEC staff and the most critical 22.5° sector was determined to be at 600 m to the south-southeast at the site boundary. The annual average diffusion parameter value for the off-gas stack release was determined to be 1.5×10^{-7} sec/m^3 and for the reactor building vent release to be 7.2×10^{-6} sec/m^3.

The method utilized by the staff to determine annual thyroid dose of 1500 mRem to a child for I-131 releases from the off-gas stack and the reactor building vent is given in Regulatory Guide 1.42. Based on this method, the maximum I-131 concentration in milk from an existing cow would occur in the northwest sector at 1.5 miles which has an annual average diffusion parameter value of 1.7×10^{-8} sec/m^3 for the off-gas stack and 4.8×10^{-7} sec/m^3 for the reactor building vent. Based on these calculations, a continuous release rate of I-131 from the off-gas stack of 25 uCi/sec or from the reactor building vent of 0.9 uCi/sec could result in an annual thyroid dose of 1500 mRem to a child drinking this milk.

In order to limit I-131 releases in the gaseous effluents to as low as practical, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and plant actions at 4 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 4, respectively, above the as low as practical objective of 1 percent of 10 CFR Part 20 limits.

The AEC staff performed an analysis similar to that used to determine the maximum release rate of I-131 for the radioactive particulates with half-lives greater than 8 days. A reduction factor of 700 on the $\overline{MPC_a}$ to allow for possible ecological chain effects similar to those associated with the cow-milk-child thyroid for radioiodine was used. The annual average diffusion parameters at 600 m in the south-southeast sector given previously were used for both the off-gas stack and reactor building vent releases. Based on these calculations, a continuous release rate of radioactive particulates with half-lives greater than 8 days in the amount of 9.5×10^9 $\overline{MPC_a}$ uCi/sec from the off-gas stack or 2×10^8 $\overline{MPC_a}$ uCi/sec from the reactor building vent would not result in annual organ doses in excess of the limits specified in 10 CFR Part 20.

In order to limit radioactive particulate releases in gaseous effluents to as low as practical, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and plant actions at 8 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 8, respectively, above the as low as practical objectives of 1 percent of 10 CFR Part 20 limits.

Measurements of the gross radioactivity from the off-gas stack must be continuously monitored for possible changes in the release rates from the augmented off-gas system. Additional measurements are made continuously at the steam jet air ejector to evaluate the core condition and the quantity of radioactivity being added to the augmented off-gas system. The measurements obtained by sampling and isotopic analysis define the releases to the environs. Quarterly analysis for tritium is adequate to define such releases to the environs.

The measurements and methods used for releases from the reactor building vent are similar to those described for releases from the off-gas stack. The main difference is the need to determine beta as well as gamma energies for the radioactive effluents and the possible need to use off-gas stack data to evaluate the predicted low levels of release from the reactor vent. Batch releases may be made during drywell purging or other special conditions when continuous monitoring is not available. For such conditions, sampling and analysis are required before releases are made and meteorological conditions may be used, if practical, to reduce possible environmental impact for such releases. If the samples indicate high concentrations of either radioiodines and/or radioactive particulates, the releases shall be filtered by the Standby Gas Treatment System.

The average gamma energy per disintegration used in the equation of Specification 3.8.A.1 will be based on the average composition of gases determined by the latest isotopic analyses on the releases from the gas decay tanks and off-gas stack. Considering the above, Specification 3.8.A.1 gives equations to be used in the airborne effluents from the off-gas stack and reactor building vent which will assure that offsite doses are not in excess of the limits specified in 10 CFR Part 20. The gamma energy per disintegration for those radioisotopes determined to be present from the isotopic analyses shall be as given in "Table of Isotopes", C. M. Lederer, J. M. Hollander, and I. Perlman, Sixth Edition, 1967. For Kr-89 and Xe-138, the gamma energy per disintegration shall be as given in "Energy Release from the Decay of Fission Products", Nuclear Science and Engineering: 3,726-746 (1958) until values are published in "Table of Isotopes". Using these reference gamma energies per disintegration with the composition of radiogases in the off-gas stack releases, the average gamma energy per disintegration, $\overline{E}_\gamma$, will be determined.

Isotopic analysis will be performed on samples taken from the steam jet air ejector. These samples will be used in an isotopic analysis for Xe-138, Xe-135, Xe-133, Kr-88, Kr-87, and Kr-85m, which is calculated to be approximately 90 percent of the noble gas emission. The remaining noble gases will be calculated from empirical ratios with the measured gases. Such calculations will be made for the various gases down to a release rate of 100 uCi/sec. Argon 41 will not be measured routinely since it cannot be measured in the presence of the other noble gases. Using the composition of radiogases at the steam jet air ejector, the average energy per disintegration for gamma and beta may be determined for the reactor building vent releases.

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.

Measurement of the gross radioactivity from the duct to the vent is based upon an equivalent dose rate for the release rate in curies per second. Since an isotopic analysis cannot be made routinely of the vent effluent, the assumption is made that the isotopic composition in the vent will be the same as determined at the steam jet air ejector. Therefore, the average gamma energy per disintegration, $\bar{E}_\gamma$, and the average beta energy disintegration, $\bar{E}_\beta$, to be used in the equation of Specification 3.8.A.1, will be based upon the average composition of gases from the air ejector unless the reactor building vent sample can be analyzed for its isotopic composition. The $\bar{E}_\gamma$ shall be determined as previously discussed for the off-gas stack using the same reference data. The beta energy per disintegration for those radioisotopes determined to be present from the appropriate isotopic analysis shall be given in USNRDL-TR-802, <u>II. Spectra of Individual Negatron Emitters</u> (Beta Spectra), H. Hogan, P. E. Zigman, and J. L. Mockin. The average beta energy shall be used as the beta energy per disintegration for each radioisotope evaluated. Using these reference beta energies per disintegration with the appropriate composition of radiogases in the vent, the average beta energy per disintegration, $\bar{E}_\beta$, will be determined.

The AEC staff has performed an analysis to determine the equivalent dose rate (mR/hr) to the release rate given in Specification 3.8.A.1 if a typical off-gas mixture from the air ejector with 30 minutes delay and some fuel failures is assumed. The relative $\bar{E}_\gamma$ assumed was 0.7 Mev/dis and the $\bar{E}_\beta$ was 0.3 Mev/dis. The resulting gamma dose rate for the radiation monitor equivalent to this release limit in the vent (or unit's duct) was determined to be 3.3 mR/hr. Although only $\bar{E}_\gamma$ is required for the determination of the release rate from the off-gas stack, $\bar{E}_\beta$ is required to be applied for surveillance of releases from the reactor building vent.

Determination of the $\bar{E}_\gamma$ and $\bar{E}_\beta$ values to be used in Specification 3.8.A.1 shall be performed weekly from the appropriate isotopic analysis until consistent values are obtained and quarterly thereafter unless changes are observed in either gaseous release rates of gross radioactivity or holdup time in the gas decay tanks. The quarterly determinations of $\bar{E}_\gamma$ and $\bar{E}_\beta$ should be used in the evaluation of compliance with the quarterly release limits.

The release of radioiodine from the off-gas stack and reactor building vent is monitored by the use of charcoal cartridges which integrate the releases over the sampling period of one to seven days. Frequency of removal is dependent upon the release level measured on the previously removed charcoal cartridge. The analysis performed for I-133 and I-135 indicates the contribution of these radioiodines to the possible inhalation doses.

The release of radioactive particulates with half-lives greater than eight days from the off-gas stack and reactor building vent is monitored by the use of particulate filters which integrate the releases over the sampling period of one to seven days. All other aspects of particulate release measurements are similar to those discussed for radioiodine release measurements. The analysis performed for Sr-89 and Sr-90 and gross alpha radioactivity indicates the contribution of these radioisotopes to the gross particulate radioactivity.

B. Mechanical Vacuum Pump

The purpose of isolating the mechanical vacuum pump line is to limit release of activity from the main condenser during a control rod drop accident. During the accident, fission products would be transported from the reactor through the main steamlines to the main condenser. The fission product radioactivity would be sensed by the main steamline radioactivity monitors and initiate isolation.

C. Liquid Effluents

The radioactive liquid effluents from the Monticello plant will be controlled on a batch basis with each batch being processed by such method or methods appropriate for the quality of materials determined to be present. Those batches in which the radioactivity concentrations are sufficiently low to allow release to the discharge canal are diluted with condenser circulating water in order to achieve the allowable concentrations set forth in 10 CFR Part 20. The radioactive liquid will be sampled and analyzed for gross radioactivity prior to release to the discharge canal, thus providing a means of obtaining information on effluents to be released so that appropriate release rates will be established.

Liquid effluent release will be controlled in terms of the concentrations in the discharge canal. In the case of unidentified mixtures, such concentration limits are based on the assumption that the entire content is made up of the most restrictive isotope in accordance with 10 CFR Part 20. Such a limit assures that even if a person obtained all of his daily water intake from such a source, the resultant dose would not exceed that specified in 10 CFR Part 20. Since no such use of the discharge canal is made and considerable natural dilution occurs prior to any locations where such usage could occur, this assures that offsite doses from this source will be far less than the limits specified in 10 CFR Part 20.

If radioactive effluents are released to unrestricted areas on a radionuclide basis, the MPC shall be determined and controlled in the cooling water discharge canal in accordance with Appendix B, Table II, Column 2 of 10 CFR Part 20 and Note 1 thereto.

Detailed meteorological calculations for several locations off site have been made by the AEC staff and the most critical 22.5° sector was determined to be at 600 m to the south-southeast at the site boundary. The annual average diffusion parameter value for the off-gas stack release was determined to be 1.5×10^{-7} sec/m^3 and for the reactor building vent release to be 7.2×10^{-6} sec/m^3.

The method utilized by the staff to determine annual thyroid dose of 1500 mRem to a child for I-131 releases from the off-gas stack and the reactor building vent is given in Regulatory Guide 1.42. Based on this method, the maximum I-131 concentration in milk from an existing cow would occur in the northwest sector at 1.5 miles which has an annual average diffusion parameter value of 1.7×10^{-8} sec/m^3 for the off-gas stack and 4.8×10^{-7} sec/m^3 for the reactor building vent. Based on these calculations, a continuous release rate of I-131 from the off-gas stack of 25 uCi/sec or from the reactor building vent of 0.9 uCi/sec could result in an annual thyroid dose of 1500 mRem to a child drinking this milk.

In order to limit I-131 releases in the gaseous effluents to as low as practical, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and plant actions at 4 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 4, respectively, above the as low as practical objective of 1 percent of 10 CFR Part 20 limits.

The AEC staff performed an analysis similar to that used to determine the maximum release rate of I-131 for the radioactive particulates with half-lives greater than 8 days. A reduction factor of 700 on the MPC_a to allow for possible ecological chain effects similar to those associated with the cow-milk-child thyroid for radioiodine was used. The annual average diffusion parameters at 600 m in the south-southeast sector given previously were used for both the off-gas stack and reactor building vent releases. Based on these calculations, a continuous release rate of radioactive particulates with half-lives greater than 8 days in the amount of 9.5×10^9 $\overline{MPC_a}$ uCi/sec from the off-gas stack or 2×10^8 $\overline{MPC_a}$ uCi/sec from the reactor building vent would not result in annual organ doses in excess of the limits specified in 10 CFR Part 20.

In order to limit radioactive particulate releases in gaseous effluents to as low as practical, quarterly average release rates have been established which would require investigative actions at 2 percent of the maximum release rate and plant actions at 8 percent of the maximum release rate. These release rates are significantly below 10 CFR Part 20 limits and are factors of 2 and 8, respectively, above the as low as practical objectives of 1 percent of 10 CFR Part 20 limits.

The frequency for monitoring or sampling has been established so that if the maximum amount of gross radioactivity is exceeded, action can be taken to reduce the radioactivity to a level below the specified limit.

F. Environmental Monitoring Program

It is recognized that a precise determination of environmental dose from a certain emission from the stack is only possible by direct measurement. Such information will be provided by the environmental monitoring program conducted at and around the site. If the stack emission ever reaches a level such that it is measureable in the environment, such measurements will provide a basis for adjusting the proposed stack limit long before the effect in the environment is of any concern for permissible dose. In this regard, it is important to realize that averaging emission rate over a period of one calendar year as permitted by 10 CFR Part 20 represents a very large safety margin between conditions at any one instant (any minute, hour, or day) and the long-term dose of interest.

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive materials in environmental media as determined by an environmental monitoring program indicate the likelihood of public intakes in excess of 1% of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups, and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variation of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

i. Occupational Personnel Radiation Exposure

(1) A tabulation of the number of occupational personnel exposures for plant operations personnel (permanent and temporary) in the following exposure increments for the reporting period: less than 100 mRem, 100-250 mRem, 250-500 mRem, 500-750 mRem, 750-1000 mRem, 1-2 Rem, 2-3 Rem, 3-4 Rem, 4-5 Rem, 5-6 Rem, and greater than 6 Rem.

(2) A tabulation of the number of personnel receiving more than 500 mRem exposure in the reporting period according to duty function [e.g., routine plant surveillance and inspection (regular duty), routine plant maintenance, special plant maintenance (describe maintenance), routine fueling operation, special refueling operation (describe operation), and other job-related exposures].

(3) A tabulation annually of the number of personnel receiving more than 3 Rem and the major cause(s).

B. Non-Routine Reports

1. Abnormal Occurrence Reports

Notification shall be made within 24 hours by telephone and telegraph to the Director of the Regional Regulatory Operations Office (cc to the Director of Licensing), followed by a written report within 10 days to the Director of Licensing (cc to the Director of the Regional Regulatory Operations Office) in the event of the abnormal occurrences as defined in Section 1.0. The written report on these abnormal occurrences, and to the extent possible, the preliminary telephone and telegraph notification, shall: (a) describe, analyze and evaluate safety implications, (b) outline the measures taken to assure that the cause of the condition is determined, (c) indicate the corrective action (including any changes made to the procedures and to the quality assurance program) taken to prevent repetition of the occurrence and of similar occurrences involving similar components or systems, and (d) evaluate the safety implications of the incident in light of the cumulative experience obtained from the record of previous failures and malfunctions of similar systems and components.