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VPNPD-96-099

10 CFR 50.4
10 CFR 50.90

November 20, 1996

Document Control Desk
US NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555-0001

Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT TO TECHNICAL SPECIFICATIONS
CHANGE REQUESTS 188 AND 189
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

This letter provides additional information in support of Technical Specifications Change Requests (TSCRs) 188 and 189. TSCRs 188 and 189 were submitted in letters dated June 4, 1996. Supplements to TSCRs have been submitted in letters dated August 5, 1996; September 26, 1996; October 21, 1996; and November 13, 1996. These requests propose amendments to the Point Beach Technical Specifications that were identified by analyses performed in support of Unit 2 operations following replacement of steam generators this fall.

We are providing additional information regarding the radiological consequences for the Steam Generator Tube Rupture, Rupture of a Steam Pipe, Locked Rotor, and Rod Ejection accidents as attachments to this letter. This information is in response to your request for additional information dated November 13, 1996.

Additionally, in support of the review of our proposed changes to Technical Specifications setting limits, we are hereby confirming that the instrument setpoint methodology being used for PBNP is contained in Procedure DG-I01, "Instrument Setpoint Methodology." The methodology contained in this procedure is based on guidance contained ISA S67.04 and ISA RP67.04 Part II, both approved in September 1994.

We have determined that the additional information does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of any effluent release or result in any significant increase in individual or cumulative occupational exposure. Therefore, we conclude that the proposed amendments meet the requirements of 10 CFR 51.22(c)(9) and that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared. The original "No Significant Hazards" determinations for operation under the proposed Technical Specifications remain applicable.

If you require additional information, please contact us.

Sincerely,

G M Krieser
for

A001 1/1

Bob Link
Vice President
Nuclear Power

9611270171 961120
PDR ADOCK 05000266
P PDR

Subscribed and sworn before me on this 20th day of November 1996.

Jacqueline Gorecki
Notary Public, State of Wisconsin

My commission expires 10/26/2000

cc: NRC Resident Inspector
NRC Regional Administrator
PSCW

270051

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATIONS CHANGE REQUESTS 188 AND 189

1. **Describe the modes of operation of the control room heating, ventilation, and air conditioning (HVAC) system, including the following: control room habitable volume in cubic feet, range of assumed flow rates (recirculated flow, fresh air inlet flow, filtered air flow) used for dose calculations, assumptions of inleakage and outleakage, pressurization assumed in inches of water gauge and basis for assumption, descriptions of the filters used including assumed removal capabilities. Describe any additional protective measures included in calculation of control room dose.**

Response:

Modes of operation of the control room HVAC system

Mode 1 is the normal control room HVAC operational mode. In this mode, 5% of the air flow is outside air and 95% is recirculated air. (Reference 1,2)

Mode 2 is initiated by a containment isolation signal or manually from the control room. In this mode, 100% of the air is recirculated. (Reference 1)

Mode 3 can only be initiated manually by operator action. This mode is similar to Mode 2 since 100% of the air is recirculated. However, in this mode, 25% of the recirculated air is filtered. (Reference 1,2)

Mode 4 is initiated by a control room radioactivity signal or manually from the control room. Mode 4 operation is 25% filtered outside air and 75% recirculated air and is considered a "positive pressure" mode for inleakage considerations. (Reference 1,2,9)

Control Room Habitable Volume and Air Flow Characteristics
(References 2-5)

	Mode 1	Mode 2	Mode 3	Mode 4
Control Room Volume	65243 ft^3	65243 ft^3	65243 ft^3	65243 ft^3
Total Flow Rate	19800 cfm	19800 cfm	19800 cfm	19800 cfm
Recirculated Flow	18800 cfm	19800 cfm	19800 cfm	14850 cfm
Filtered Recirculation	0	0	4950 cfm	0
Unfiltered Recirculation	18800 cfm	19800 cfm	14850 cfm	14850 cfm
Fresh Air Inlet Flow	1000 cfm	0	0	4950 cfm
Filtered Inlet Flow	0	0	0	4950 cfm
Unfiltered Inlet Flow	1000 cfm	0	0	0
Unfiltered Inleakage	65.2 cfm	65.2 cfm	65.2 cfm	10 cfm

Pressurization and basis for assumption

In mode 4, the control room envelope is pressurized to a pressure greater than 0.125 inches water gauge. In accordance the PBNP Technical Specification Section 15.4.11, monthly testing of the HVAC system is performed to ensure proper operation of all system operational modes and positive pressurization of the control room. (Reference 8)

In 1994, the computer room was added to the control room envelope. The position of the dampers associated with the computer room are checked monthly. With these two dampers opened, the pressure differential between the computer room and the turbine hall exceeds 1/8-inch w.g. by a considerable amount. (Reference 2, 6)

Filter Description and Removal Capabilities

The filter fans for the control/computer room system drive air through a filter train consisting of roughing, HEPA, and charcoal filters. The filter fans discharge the air to the inlet of the normal supply system filter. The iodine removal capabilities assumed for the dose calculations are shown in the following table.

Filter Bank Iodine Removal Efficiencies

	Filter Removal Efficiency
Elemental	90%
Methyl	90%
Particulate	99%

Additional protective measures

Potassium iodide (KI) is stored in the control room and procedure EPIP 5.2 directs its use. For the accidents addressed in this submittal, the rod ejection and the locked rotor assume the use of KI. An iodine protection factor of 10 is used. (Reference 7, 11)

Question 1 References

1. DBD-31 "Control Room HVAC and Habitability," Revision 0, §1 "Functional Description."
2. DBD-31, §3 "System Bases."
3. DBD-31, §4 "Component Bases."
4. NUREG-0800, §6.4 "Control Room Habitability System."
5. Tenera letters from B. Beasley to C. Castell "Calculation of the Effects of a Loss of Ventilation in the Control Room at the Point Beach Nuclear Plant," 8/30/91, and "Calculation of the Effects of a Loss of Ventilation in the Computer Room at the Point Beach Nuclear Plant," 7/10/91.

6. CR-HVAC and Habitability Validation Report, December 27, 1994.
7. Point Beach Nuclear Plant - Control Room Habitability Report, Stone & Webster Engineering Corporation, November 30, 1982.
8. Technical Specification Test TS-9 "Control Room Heating and Ventilation System Monthly Checks," Revision 18, June 13, 1994.
9. Radiation Monitoring System Alarm Setpoint and Response Book, RE-101 "Control Room Area Monitor," and RE-235 "Control Room Noble Gas Monitor."
10. DBD-31, §4.11 "Filters."
11. EPIP 5.2 "Radioiodine Blocking and Thyroid Dose Accounting."

2. Define each postulated source term resulting from a control rod ejection accident, steam generator tube rupture accident (SGTR), loss of reactor coolant flow accident (locked rotor), and main steam line break (MSLB). Define the receptor points used for the control room, exclusion area boundary (EAB), and low population zone (LPZ). Provide the following information for each source/receptor pair (plan view and cross-sectional figures may be helpful in describing the source, building configuration, and intake relationships).

Response:

Steam Generator Tube Rupture

For the SGTR, the complete severance of a single steam generator tube is assumed to occur. Due to the pressure differential between the primary and secondary systems, radioactive reactor coolant is discharged into the secondary system. A portion of this radioactivity is released to the outside atmosphere through either the main condenser, the atmospheric dump valves, or the safety relief valves. In addition, some of the iodine activity that is contained in the secondary coolant is released to the atmosphere as a result of steaming in the steam generators following the accident.

Input Parameters and Assumptions:

The analysis of the radiological consequences of the SGTR uses the analytical methods and assumptions outlined in the Standard Review Plan (SRP 15.6.3). One hundred and two percent of the uprated power level of 1650 MWt is used in the analysis. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the tube rupture, raising the primary system iodine concentration to 60 μCi/g of dose equivalent I-131. For the accident initiated iodine spike, the reactor trip associated with the tube rupture increases the iodine release rate from the fuel to a rate that is 500 times greater than the release rate corresponding to the maximum equilibrium primary system Technical Specification concentration of 1.0 μCi/g of dose equivalent I-131.

The noble gas activity concentration in the primary system at the time the accident occurs is based on a fuel defect level of one percent. The iodine activity concentration in the

secondary coolant of the steam generator at the time of the accident is assumed to be equivalent to the Technical Specification limit of 1.2 μCi/g of dose equivalent I-131.

The amount of primary to secondary steam generator tube leakage in the intact steam generator is assumed to be equal to the Technical Specification limit for a single steam generator of 0.35 gpm.

No credit for iodine removal is taken for steam released to the condenser prior to the reactor trip and the concurrent loss of offsite power.

An iodine partition factor of 0.01 is used in the steam generators. Partition factor = (curies of iodine per gram of steam)/(curies of iodine per gram of water).

All noble gas activity carried over to the secondary system through the steam generator tube leak is assumed to be immediately released to the outside atmosphere.

Flow through the ruptured steam generator tube is assumed to be terminated due to operator action at thirty minutes following accident initiation.

Eight hours after the tube rupture the residual heat removal system is assumed to be placed into service and there are no further steam releases to the atmosphere from the secondary system.

The total core noble gas and halogens core inventory in the reactor core are given in Table 1. The thyroid dose conversion factors, breathing rates, atmospheric dispersion factors are given in Table 2. The control room atmospheric dispersion factors are not modified for occupancy factor considerations.

TABLE 1
TOTAL CORE NOBLE GAS AND HALOGEN FUEL INVENTORIES AT SHUTDOWN

Noble Gases		Halogens	
Nuclide	**Activity(curies)**	**Nuclide**	**Activity(curies)**
Kr-85	5.4E+05	I-131	4.4E+07
Kr-85m	1.2E+07	I-132	6.3E+07
Kr-87	2.3E+07	I-133	9.0E+07
Kr-88	3.2E+07	I-134	9.9E+07
Xe-131m	4.7E+05	I-135	8.4E+07
Xe-133	8.9E+07		
Xe-133m	2.8E+06		
Xe-135	2.3E+07		
Xe-135m	1.7E+07		
Xe-138	7.5E+07		

TABLE 2

DOSE CONVERSION FACTORS (DCFs)	
Nuclide	**Thyroid DCF (rem/curie inhaled)**
I-131	1.07E+06
I-132	6.29E+03
I-133	1.81E+05
I-134	1.07E+03
I-135	3.14E+04

Time Period (hours)	**Breathing Rate (m^3/s)**
0 to 8	3.47E-04
8 to 24	1.75E-04
24 to 720	2.32E-04

Atmospheric Dispersion Factors (s/m^3)		
Site Boundary		
0 to 2 hours	5.0E-04	
Low Population Zone		
0 to 8 hours	3.0E-05	
8 to 24 hours	1.6E-05	
24 to 96 hours	4.2E-06	
96 to 720 hours	8.6E-07	
Control Room Rod Eject and MSLB release point is Unit 2 containment. SGTR and Locked Rotor release point is Unit 2 SG Safety Valves		
Time Period	Rod Eject & MSLB	SGTR & Locked Rotor
0 to 8 hours	2.1E-03	1.9E-3
8 to 24 hours	1.3E-03	1.3E-3
24 to 96 hours	8.3E-04	7.6E-4
96 to 720 hours	3.3E-04	2.9E-4

Steamline Break

The complete severance of a main steamline outside containment is assumed to occur. The affected steam generator rapidly depressurizes and releases radioiodines initially contained in the secondary coolant directly to the outside atmosphere. Radioiodines initially contained in the primary coolant that is transferred to the secondary system through steam generator tube leaks are also directly released to the outside atmosphere. A portion of the iodine activity initially contained in the intact steam generator and the noble gas activity brought to the secondary system through the steam generator tube leaks is released to the atmosphere either through the atmospheric dump valves or the safety relief valves.

Input Parameters and Assumptions:

The analysis of the radiological consequences of the steam line break uses the analytical methods and assumptions outlined in the Standard Review Plan (SRP 15.1.5). One hundred and two percent of the uprated power level of 1650 MWt is used in the analysis. (Note that an uprated power level was assumed because this analysis was originally being performed to support possible future uprating of the Point Beach reactors, not for the steam generator replacement). For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the accident raising the primary system iodine concentration to 60 μCi/g of dose equivalent I-131. For the accident initiated iodine spike, the reactor trip associated with the accident increases the iodine release rate from the fuel to a rate that is 500 times greater than the release rate corresponding to the maximum equilibrium primary system Technical Specification concentration of 1.0 μCi/g of dose equivalent I-131. The duration of the accident initiated spike is 1.6 hours.

The noble gas activity concentration in the primary system at the time the accident occurs is based on a fuel defect level of one percent. The iodine activity concentration in the secondary coolant of the steam generator at the time of the accident is assumed to be equivalent to the Technical Specification limit of 1.2 μCi/g of dose equivalent of I-131.

The amount of primary to secondary steam generator tube leakage in each steam generator is assumed to be equal to the Technical Specification limit for a single steam generator of 0.35 gpm.

The steam generator connected to the severed steam pipe is assumed to boil dry within the initial two hours following the break. The entire liquid inventory of this steam generator is assumed to steam off and all of the iodine initially in this steam generator is released to the atmosphere. Additionally, iodine carried over to the secondary side of this steam generator by steam generator tube leaks is assumed to be released directly to the atmosphere with no credit taken for iodine retention in the steam generator.

No credit for iodine removal is taken for steam released to the condenser prior to the reactor trip and the concurrent loss of offsite power.

An iodine partition factor of 0.01 is used in the intact steam generators. Partition factor = (curies of iodine per gram of steam)/(curies of iodine per gram of water).

All noble gas activity carried over to the secondary system through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

Eight hours after the accident the residual heat removal system is assumed to be placed into service and there are no further steam releases to the atmosphere from the secondary system.

Locked Rotor

An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. Fuel clad damage is assumed to occur as a result of the reduced flow. Fission products are assumed to leak from the primary system to the secondary system through steam generator tube leaks due to a pressure differential between the primary and secondary systems. A fraction of this activity is released to the outside atmosphere through either the atmospheric dump valves or the safety relief valves. Some of the iodine activity contained in the secondary coolant prior to the accident is released to the outside atmosphere as a result of steaming of the steam generators following the accident.

Input Parameters and Assumptions:

The analysis of the radiological consequences of the locked rotor uses the analytical methods and assumptions outlined in the Standard Review Plan (SRP 15.3.3 - 15.3.4). One hundred and two percent of the uprated power level of 1650 MWt is used in the analysis. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the accident raising the primary system iodine concentration to 60 μCi/g of dose equivalent I-131. Since fuel failure is assumed, it is not necessary to assume an accident initiated iodine spike.

The noble gas activity concentration in the primary system at the time the accident occurs is based on a fuel defect level of one percent. The iodine activity concentration in the secondary coolant of the steam generator at the time of the accident is assumed to be equivalent to the Technical Specification limit of 1.2 μCi/g of dose equivalent of I-131.

It is conservatively assumed that one hundred percent of the fuel rods in the core suffer sufficient damage that their gap activity is released to the primary system. Ten percent of the total core activity for both iodines and noble gases is assumed to be in the fuel-clad gap.

The amount of primary to secondary steam generator tube leakage in each steam generator is assumed to be equal to the Technical Specification limit for a single steam generator of 0.35 gpm.

No credit for iodine removal is taken for steam released to the condenser prior to the reactor trip and the concurrent loss of offsite power.

An iodine partition factor of 0.01 is used in the intact steam generators. Partition factor = (curies of iodine per gram of steam)/(curies of iodine per gram of water).

All noble gas activity carried over to the secondary system through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

Eight hours after the accident the residual heat removal system is assumed to be placed into service and there are no further steam releases to the atmosphere from the secondary system.

Control Rod Ejection

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. Fission products are assumed to leak into the containment through the failed control rod mechanism housing. Fission products are also assumed to leak from the primary system to the secondary system through steam generator tube leaks due to a pressure differential between the primary and secondary systems.

Two release paths are evaluated for the control rod ejection accident. One path is the leakage of activity from the containment building. The second path is the release of radioactivity from the secondary side of the steam generators with steam released through the safety valves.

Input Parameters and Assumptions:

The analysis of the radiological consequences of the control rod ejection uses the analytical methods and assumptions outlined in Regulatory Guide 1.77. For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the rod ejection raising the primary system iodine concentration to 60 μCi/g of dose equivalent I-131. Since fuel failure is assumed, it is not necessary to assume an accident initiated iodine spike.

It is conservatively assumed that ten percent of the fuel rods in the core suffer sufficient damage that their gap activity is released to the primary system. It is also conservatively assumed that 0.25 percent of the fuel in the core melts as a result of the accident.

One hundred percent of the noble gases and iodines in the gap of the failed fuel plus 100 percent of the noble gases and 50 percent of the iodines in the fuel melted by the accident is assumed mixed throughout the primary system instantaneously.

One hundred percent of the noble gases and iodines in the gap of the failed fuel plus 100 percent of the noble gases and 50 percent of the iodines in the fuel melted by the accident is assumed mixed throughout the containment instantaneously. Fifty percent of the iodines releases to containment are assumed to plateout instantaneously on containment surfaces.

The noble gas activity concentration in the primary system at the time the accident occurs is based on a fuel defect level of one percent. The iodine activity concentration in the secondary coolant of the steam generator at the time of the accident is assumed to be equivalent to the Technical Specification limit of 1.2 μCi/g of dose equivalent of I-131.

The amount of primary to secondary steam generator tube leakage in each steam generator is assumed to be equal to the Technical Specification limit for a single steam generator of 0.35 gpm.

No credit for iodine removal is taken for removal of iodine in the containment due to containment sprays.

An iodine partition factor of 0.01 is used in the intact steam generators. Partition factor = (curies of iodine per gram of steam)/(curies of iodine per gram of water).

All noble gas activity carried over to the secondary system through steam generator tube leakage is assumed to be immediately released to the outside atmosphere.

The containment leaks for the first twenty-four hours at its design leak rate of 0.4 percent per day. Thereafter, the containment leak rate is 0.2 percent per day.

Primary and secondary system pressures are equalized at 1500 seconds after accident initiation thus terminating primary to secondary leakage in the steam generators.

2a. Height of physical top of vent or stack or other assumed release point

For each postulated source term resulting from the control rod ejection accident, steam generator tube rupture accident (SGTR), loss of reactor coolant flow accident (locked rotor), and main steam line break (MSLB), the assumed release point is described below.

Control Rod Ejection Accident

The release point is the containment structure at its design leak rate and the atmospheric steam dump valve or the safety relief valves. The assumed release height

is ground level. For the control room atmospheric dispersion model, the release point for the containment structure is assumed to be at an elevation of 81 feet 7.5 inches which is the mean elevation height of the containment structure. (Reference 3) The release point for the steam generator releases is the atmosphere at elevation 170 feet through vent pipes out the top of the containment facade. (Reference 1)

Steam Generator Tube Rupture Accident

The release point is the atmospheric steam dump valve or the steam generator safety relief valves. These valves release to the atmosphere at elevation 170 feet through vent pipes out the top of the facade. (Reference 1)

Loss of Rea:tor Coolant Flow Accident

The release point is the atmospheric steam dump valve or the steam generator safety relief valves. These valves release to the atmosphere at elevation 170 feet through vent pipes out the top of the facade. (Reference 1)

Main Steam Line Break

The release point for the severed steam pipe is the atmosphere at elevation 88 feet outside the containment building. The release point for the intact steam pipe is the atmospheric steam dump valve or the steam generator safety relief valves which vent to the atmosphere at elevation 170 feet through vent pipes out the top of the facade. (References 1 and 2)

Question 2a References

1. Bechtel Drawing M-44.
2. Bechtel Drawing M-11.
3. Bechtel Drawing M-9.

2b. Dimensions of buildings and other structures adjacent to or separating the release point and the intake/receptor

The dimensions of the buildings and structures adjacent to or separating the release point from the intake/receptor are shown on the attached drawing which is a modified version of reference 3. The elevation of the top of the structures are indicated on the drawing by capital H followed by the height in feet. (Example H162') Ground level near the plant is 26 feet.

The smallest plane cross-sectional area presented by the containment building is 1640 square meters. (Reference 1, p. 2.6-11)

The maximum adjacent building height either upwind or downwind from the release point is 41.3 m. (Reference 2)

Question 2b References

1. Point Beach Nuclear Plant Final Safety Analysis Report.
2. VPNPD-92-265, "Point Beach Nuclear Plant Meteorological Data," dated July 27, 1992.
3. Wisconsin Electric Drawing PBC-231 "Plot Plan."

2c. Horizontal straight line distance between the source and the receptor

The horizontal straight line distance between the source and receptor (control room intake) for the various release points are listed below.

Source	Straight Line Distance (m)	Path	Reference
U2 Containment	31.1	On the line through the center of the containment building and the receptor.	2
U1 Containment	46.0	On the line through the center of the containment building and the receptor.	1
U1 A Safeties	44.8	On the line between the midpoint of the line that the valves lie on and the receptor.	1
U1 B Safeties	54.3	On the line between the midpoint of the line that the valves lie on and the receptor.	1
U2 A Safeties	34.3	On the line between the midpoint of the line that the valves lie on and the receptor.	2
U2 B Safeties	35.5	On the line between the midpoint of the line that the valves lie on and the receptor.	2

Question 2c References

1. Bechtel Drawing M-1.
2. Bechtel Drawing M-2001.

2d. Height of the physical intake or receptor

The control room ventilation intake is located at elevation 111' 9". (Reference 1)

Question 2d Reference

1. Bechtel Drawing M-9.

2e. Direction (in degrees) from the intake to the source, as follows: assuming one is standing at the intake looking toward the source, looking toward the North is 360°, toward the East is 90°, toward the South is 180°, etc. This grid system must be congruent with the meteorological data collection grid system;

Source	Direction (degrees)	Path	Reference
U2 Containment	269°	On the line through the center of the containment building and the receptor.	2
U1 Containment	200°	On the line through the center of the containment building and the receptor.	1
U1 A Safeties	204°	On the line between the midpoint of the line that the valves lie on and the receptor.	1
U1 B Safeties	184°	On the line between the midpoint of the line that the valves lie on and the receptor.	1
U2 A Safeties	255°	On the line between the midpoint of the line that the valves lie on and the receptor.	2
U2 B Safeties	287°	On the line between the midpoint of the line that the valves lie on and the receptor.	2

Question 2e References

1. Bechtel Drawing M-1.
2. Bechtel Drawing M-2001.

2f. An hourly listing of 5 years of onsite meteorological data representative of long-term conditions at the site. Data should be provided in a form consistent with the format described in Appendix A to section 2.3.3, "Onsite Meteorological Measurements Programs," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition."

A cumulative joint frequency distribution of wind speed and direction by atmospheric stability class is provided in the table below. The direction is indicated by the sector that the wind is blowing from. Only 3 years (1991-1993) of information was readily available from the plant process computer system. The joint frequency distribution for 1991 through 1993 is shown in the following tables.

The data represents the number of hourly observations in the sector, stability class, and wind speed ranking. The following table indicates the wind speed range corresponding to each wind speed ranking in the stability class tables.

Wind Speed Ranking	Wind Speed Range
1	0.22 < wind speed <= 0.50 m/s
2	0.50 < wind speed <= 0.75 m/s
3	0.75 < wind speed <= 1.00 m/s
4	1.00 < wind speed <= 1.50 m/s
5	1.50 < wind speed <= 2.00 m/s
6	2.00 < wind speed <= 3.00 m/s
7	3.00 < wind speed <= 5.00 m/s
8	5.00 < wind speed <= 7.00 m/s
9	7.00 < wind speed <= 10.00 m/s
10	10.00 < wind speed <= 13.00 m/s
11	13.00 < wind speed <=18.00 m/s
12	18.00 m/s < wind speed

Stability Class A

Wind Direction Sectors	Wind Speed Ranking											
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
N	6	16	4	0	0	2	0	0	0	0	0	0
NNE	0	2	1	1	0	0	0	0	0	0	0	0
NE	0	0	3	3	0	0	1	0	0	0	0	0
ENE	0	0	2	0	0	0	0	0	0	0	0	0
E	0	1	0	2	0	0	1	0	0	0	0	0
ESE	1	2	2	1	0	0	1	0	0	0	0	0
SE	1	0	1	0	0	2	0	0	0	0	0	0
SSE	0	0	2	2	0	2	0	0	0	0	0	0
S	2	2	1	3	1	3	1	0	0	0	0	0
SSW	1	2	2	3	1	3	0	0	0	0	0	0
SW	2	0	3	1	0	1	1	0	0	0	0	0
WSW	0	1	5	2	4	7	0	0	0	0	0	0
W	0	3	1	0	7	11	0	0	0	0	0	0
WNW	0	0	3	2	4	7	0	0	0	0	0	0
NW	1	0	1	0	3	2	1	0	0	0	0	0
NNW	2	0	0	0	0	0	1	0	0	0	0	0

Stability Class B

Wind Direction Sectors	Wind Speed Ranking											
	1	2	3	4	5	6	7	8	9	10	11	12
N	3	1	1	2	6	6	12	1	0	0	0	0
NNE	0	0	0	2	2	3	5	1	0	0	0	0
NE	0	0	1	1	1	1	0	0	0	0	0	0
ENE	0	0	0	6	1	3	0	0	0	0	0	0
E	0	0	0	2	2	2	0	0	0	0	0	0
ESE	0	0	0	0	3	4	2	0	0	0	0	0
SE	0	0	0	2	5	7	2	1	0	0	0	0
SSE	0	0	0	5	0	8	4	0	0	0	0	0
S	0	0	3	5	3	12	3	0	0	0	0	0
SSW	0	0	0	3	7	10	3	0	0	0	0	0
SW	0	0	0	6	8	18	6	0	0	0	0	0
WSW	0	0	0	7	6	20	8	0	0	0	0	0
W	0	0	0	7	6	28	29	0	0	0	0	0
WNW	0	0	0	4	10	31	47	0	0	0	0	0
NW	0	0	0	3	4	12	18	0	0	0	0	0
NNW	0	0	0	1	0	4	0	0	0	0	0	0

Stability Class C

Wind Direction Sectors	Wind Speed Ranking											
	1	2	3	4	5	6	7	8	9	10	11	12
N	1	0	0	6	9	26	30	3	0	0	0	0
NNE	0	0	0	4	10	16	13	1	0	0	0	0
NE	0	0	0	4	5	10	5	1	0	0	0	0
ENE	0	0	0	7	14	13	3	0	0	0	0	0
E	0	0	0	2	18	21	3	2	0	0	0	0
ESE	1	0	0	4	16	37	8	1	0	0	0	0
SE	0	0	0	7	23	30	29	2	0	0	0	0
SSE	0	0	2	11	21	34	33	1	0	0	0	0
S	0	0	1	10	24	42	23	3	0	0	0	0
SSW	0	0	0	6	14	27	25	1	0	0	0	0
SW	0	0	0	7	20	24	30	1	0	0	0	0
WSW	0	0	0	4	22	47	100	16	0	0	0	0
W	0	1	0	13	29	92	156	29	3	0	0	0
WNW	0	0	0	4	47	118	161	24	0	0	0	0
NW	0	0	0	3	22	59	78	5	1	0	0	0
NNW	0	0	0	0	6	14	19	1	2	0	0	0

Stability Class D

Wind Direction Sectors	Wind Speed Ranking											
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
N	0	2	2	20	56	125	134	53	12	3	0	0
NNE	0	0	0	21	36	49	38	19	9	0	0	0
NE	1	2	7	26	40	30	19	5	3	1	0	0
ENE	0	3	8	21	27	32	16	5	0	0	0	0
E	0	0	4	20	31	43	28	7	1	0	0	0
ESE	0	4	2	13	40	128	170	12	1	0	0	0
SE	0	1	2	25	62	138	252	41	0	0	0	0
SSE	0	4	3	12	43	117	231	117	9	0	0	0
S	0	1	3	25	100	242	357	64	4	0	0	0
SSW	0	1	1	21	36	60	100	25	2	1	0	0
SW	0	1	4	15	38	88	147	56	18	0	0	0
WSW	0	0	4	22	65	228	379	169	62	2	0	0
W	0	1	4	28	69	315	492	154	13	0	0	0
WNW	0	0	3	25	83	254	300	36	1	0	0	0
NW	0	0	7	18	51	193	482	124	16	0	0	0
NNW	0	0	0	6	16	33	89	42	12	1	0	0

Stability Class E

Wind Direction Sectors	Wind Speed Ranking											
	1	**2**	**3**	**4**	**5**	**6**	**7**	**8**	**9**	**10**	**11**	**12**
N	1	5	6	26	53	229	443	265	120	29	1	0
NNE	0	2	10	22	38	92	133	74	61	16	1	0
NE	0	6	9	27	41	81	98	39	38	9	0	0
ENE	0	5	8	36	35	104	48	47	28	9	0	0
E	1	5	3	19	22	79	114	55	9	1	0	0
ESE	1	2	7	16	12	81	247	41	16	1	0	0
SE	1	4	11	30	26	41	149	146	34	1	0	0
SSE	2	6	6	29	47	108	372	512	316	20	1	0
S	0	5	16	63	125	268	570	328	92	8	0	0
SSW	1	1	12	57	148	229	383	180	67	14	0	0
SW	0	2	7	46	118	290	444	228	125	11	0	0
WSW	0	2	3	24	68	327	470	114	47	3	0	0
W	1	0	4	30	63	149	157	1	1	0	0	0
WNW	0	5	6	22	27	32	17	1	0	0	0	0
NW	1	2	4	24	34	55	197	137	47	0	0	0
NNW	1	2	4	10	10	26	131	143	55	1	0	0

Stability Class F

Wind Direction Sectors	Wind Speed Ranking											
	1	2	3	4	5	6	7	8	9	10	11	12
N	2	3	4	17	2	42	293	358	209	73	13	0
NNE	0	1	4	9	2	15	136	126	146	80	17	0
NE	0	3	8	12	12	35	78	55	31	10	0	0
ENE	3	4	3	11	3	25	61	58	31	6	0	0
E	3	4	5	5	3	8	45	55	9	1	0	0
ESE	3	2	5	11	0	12	36	26	15	0	0	0
SE	3	1	7	10	3	2	0	0	0	0	0	0
SSE	0	0	10	27	8	10	19	10	17	1	0	0
S	1	4	10	58	35	52	143	62	23	1	0	0
SSW	0	1	7	77	47	113	92	50	23	0	0	0
SW	3	5	9	34	43	116	46	18	12	1	0	0
WSW	2	5	7	23	20	98	114	0	0	0	0	0
W	0	3	10	12	9	14	15	0	0	0	0	0
WNW	2	0	4	8	2	1	1	0	0	0	0	0
NW	1	4	4	6	2	2	1	1	0	0	0	0
NNW	0	2	4	8	1	2	7	20	4	0	0	0

Stability Class G

Wind Direction Sectors	Wind Speed Ranking											
	1	2	3	4	5	6	7	8	9	10	11	12
N	0	4	6	2	0	0	37	147	91	13	2	0
NNE	0	0	5	3	1	0	20	39	31	13	2	0
NE	3	1	1	1	0	0	14	15	4	3	0	0
ENE	0	5	6	3	0	3	14	10	0	0	0	0
E	0	0	3	5	0	2	1	2	1	3	0	0
ESE	2	0	5	2	0	1	0	1	1	0	0	0
SE	2	4	7	3	0	0	0	0	0	0	0	0
SSE	1	3	2	8	0	1	2	0	0	0	0	0
S	1	8	21	20	2	4	3	1	0	0	0	0
SSW	0	5	7	28	7	7	4	0	0	0	0	0
SW	1	3	9	7	1	6	0	0	0	0	0	0
WSW	0	6	2	3	1	3	1	0	0	0	0	0
W	1	4	2	1	5	1	0	0	0	0	0	0
WNW	3	3	4	0	0	0	0	0	0	0	0	0
NW	0	2	1	3	0	0	0	0	0	0	0	0
NNW	0	2	3	3	0	2	1	0	0	0	0	0

Question 2f References

1. NPL 93-0571, "Point Beach Nuclear Plant Meteorological Data," dated November 10, 1993.
2. VPNPD-92-265, "Point Beach Nuclear Plant Meteorological Data," dated July 27, 1992.
3. NPL 94-0230, "Point Beach Nuclear Plant Meteorological Data," dated June 13, 1994.

2g. Describe the mode of operation of the control room HVAC for each accident for the 30-day control room operator dose calculations.

For each postulated source term resulting from the control rod ejection accident, steam generator tube rupture accident (SGTR), loss of reactor coolant flow accident (locked rotor), and main steam line break (MSLB), the mode of operation of the control room HVAC is described below.

<u>Control Rod Ejection Accident</u>

Initially the system is in Mode 1. On containment isolation, the system is automatically shifted to Mode 2 which occurs within 5 minutes. When the dose rates in the control room exceed the high radiation alarm setpoint, the system is automatically shifted to Mode 4 and remains there until the radiation release has ended. The shift to Mode 4 will occur within 30 minutes. For simplicity, the analysis assumes a shift from Mode 1 to Mode 4, on a radiation monitor automatic shift after 30 minutes, then remaining there for the rest of the event. The shift to Mode 2 after 5 minutes is not taken into account which is conservative because Mode 1 continues to have intake. Mode 2 is 100% recirculation.

<u>Steam Generator Tube Rupture Accident</u>

Initially the system is in Mode 1. On containment isolation, which occurs within 10 minutes, the system is automatically shifted to Mode 2. When the dose rates in the control room exceed one of the radiation alarm setpoints, the system is automatically shifted to Mode 4 and remains there until the radiation release has ended. The shift to Mode 4 will occur within 30 minutes. In order to bound the total dose received by the operators, the analysis is done two ways. The first includes a shift from Mode 1 to Mode 2 after 10 minutes, and remaining in Mode 2. The second includes a shift from Mode 1 to Mode 4 after 30 minutes, then remaining in Mode 4 for the rest of the event.

<u>Loss of Reactor Coolant Flow Accident (Locked Rotor)</u>

Initially the system is in Mode 1. A containment isolation signal is not actuated in this accident so Mode 2 is not used in this scenario. The dose rates in the control room

trip the control room radiation monitors within 30 minutes, switching the system to Mode 4 where it remains until the radiation release for the rest of the event.

Main Steam Line Break

Initially the system is in Mode 1. On containment isolation the system is automatically shifted to Mode 2. A containment isolation signal is received early in the accident shifting the system to Mode 2 within 5 minutes. The analysis is done two ways. The first has a shift from Mode 1 to Mode 2 after 5 minutes, and remaining in Mode 2. The second calculation has a shift from Mode 1 to Mode 4, on a control room radiation monitor automatic shift after 30 minutes, then remaining there for the rest of the event.

Question 2g References

1. Radiation Monitoring System Alarm Setpoint and Response Book, RE-101 "Control Room Area Monitor," and RE-235 "Control Room Noble Gas Monitor."
2. Technical Specification Test TS-9 "Control Room Heating and Ventilation System Monthly Checks," Revision 18, June 13, 1994.

3. What are the basis and assumptions for the X/Q's used in your past and current control room habitability assessments ?

Response:

The atmospheric dispersion factors used in the past control room habitability assessments are listed below:

Accident Period (s/m^3)			
0 to 8 Hours	**8 to 24 Hours**	**24 to 96 Hours**	**96 to 720 Hours**
2.15E-03	1.46E-03	5.16E-04	1.29E-04

A review of the documentation could only determine the horizontal distance, the wind direction factors, and the wind speed factors used in the calculation of these factors. The underlying methodology used could not be determined. The listed values do incorporate the control room occupancy factors suggested in reference 2.

The atmospheric dispersion factors used in the current control room habitability assessments are listed below. (Reference 1) The listed values do not take into account control room occupancy factors.

Accident Period (s/m^3)			
0 to 8 Hours	**8 to 24 Hours**	**24 to 96 Hours**	**96 to 720 Hours**
2.1E-03	1.3E-03	8.3E-04	3.3E-04

The five percentile X/Q used for the first time interval is calculated using Equation 7 on page 28 of reference 3 and the methodology of NUREG 1.145 (reference 4). The values used for the subsequent time intervals were calculated using the methodology of reference 2.

Question 3 References

1. Wisconsin Electric Calculation 95-0254, "Point Beach Nuclear Plant Control Room Ventilation Intake Atmospheric Dispersion Factor," 11/11/96.
2. Murphy, K.G., and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design For Meeting General Criterion 19, " U. S. Atomic Energy Commission, 13th AEC Air Cleaning Conference.
3. NUREG/CR-5055, Ramsdell, J.V., "Atmospheric Diffusion for Control Room Habitability Assessments," USNRC, May 1988.
4. U.S. Nuclear Regulatory Commission Regulatory Guide 1.145, "Atmospheric Dispersion Models For Potential Accident Consequence Assessments At Nuclear Power Plants," Revision 1, November 1982, (Reissued February 1983).

4. **What is the basis for the atmospheric dispersion estimates used in the dose assessment for the EAB, LPZ, and any other off-site locations ? For each of the 16 compass directions, what is the minimum distance from each assumed release point to the site boundary or to the shoreline for those cases in which the site boundary extends past the shoreline ?**

Response:

The atmospheric dispersion estimates used in the dose assessment for the offsite locations were calculated using the methodology of Regulatory Guide 1.145, "Atmospheric Dispersion Models For Potential Accident Consequence Assessments At Nuclear Power Plants" and are documented in reference 3 The distances used for the LPZ and the EAB distances in the 16 compass directions are shown in the table below (References 1 and 2).

Sector	EAB Distance (m)	LPZ Distance (m)
N	4000	9000
NNE	1200	9000
NE	1200	9000
ENE	1200	9000
E	1200	9000
ESE	1200	9000
SE	1200	9000
SSE	1300	9000
S	1270	9000
SSW	1290	9000
SW	1520	9000
WSW	1320	9000
W	1300	9000
WNW	1630	9000
NW	2040	9000
NNW	2010	9000

Question 4 References

1. Point Beach Nuclear Plant FSAR Chapter 14.
2. Point Beach Nuclear Plant FSAR Appendix I.
3. Wisconsin Electric calculation 96-127, "Point Beach Nuclear Plant Site Dispersion Factors For Accident Conditions," Revision 1, 11/1/96.

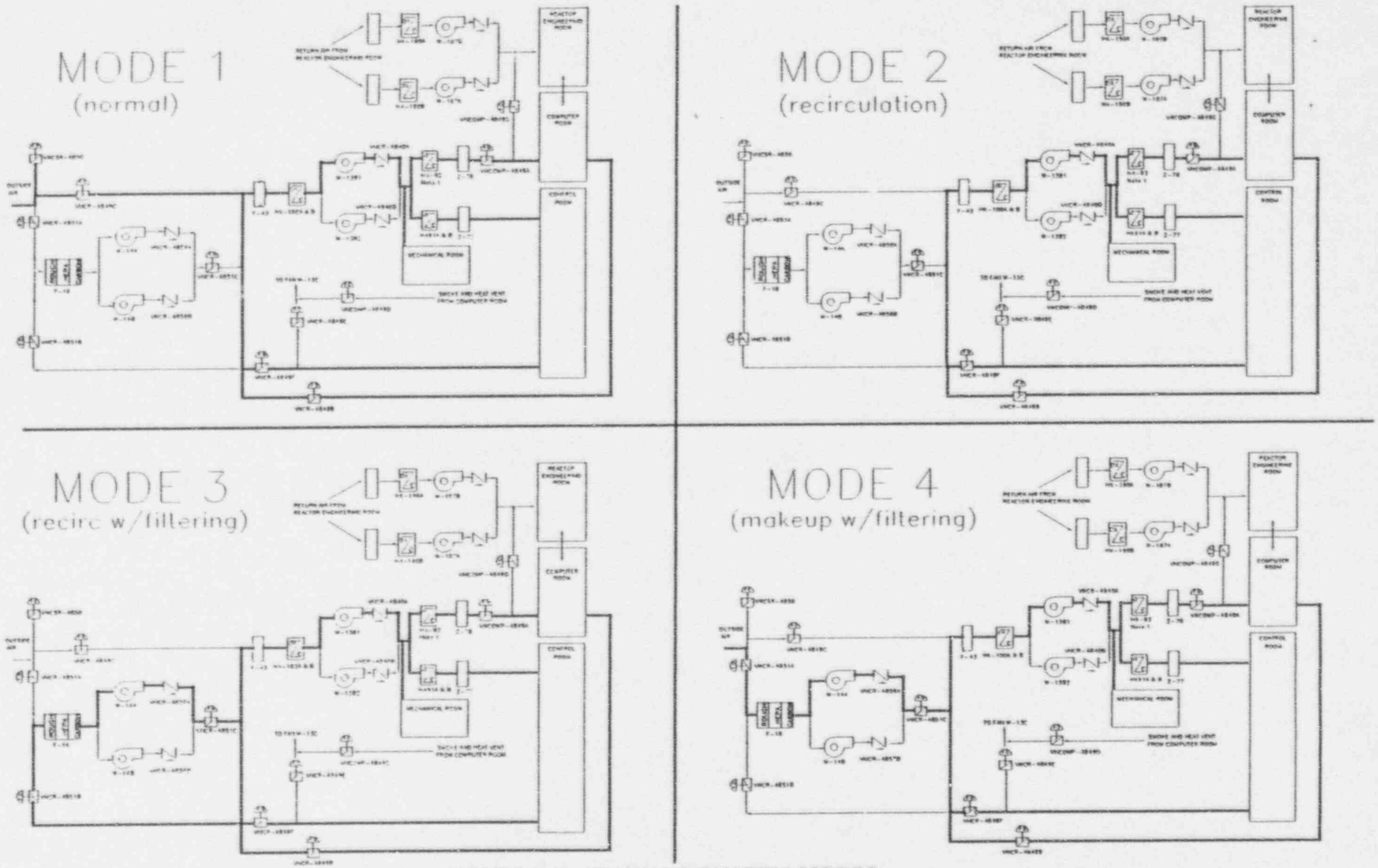
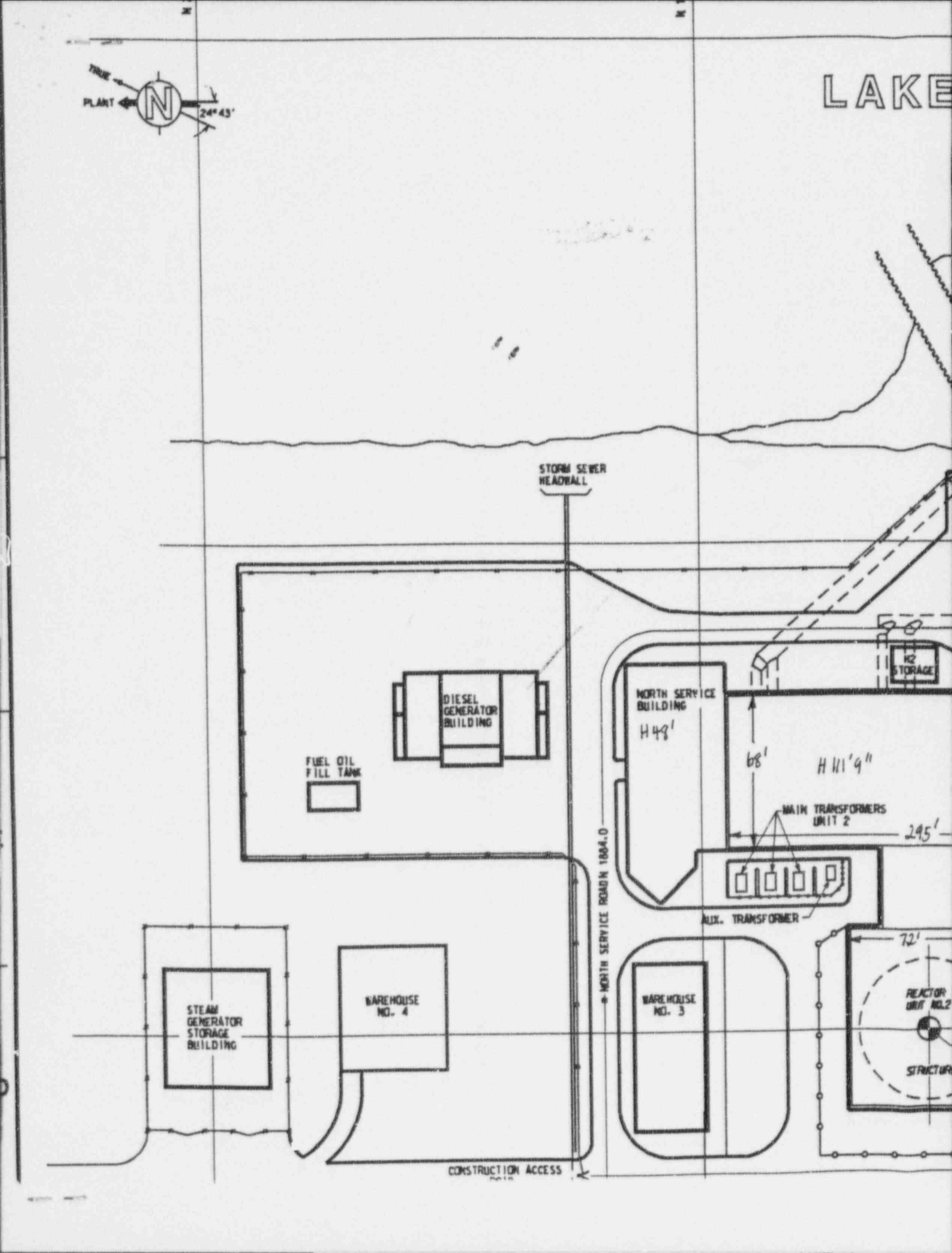
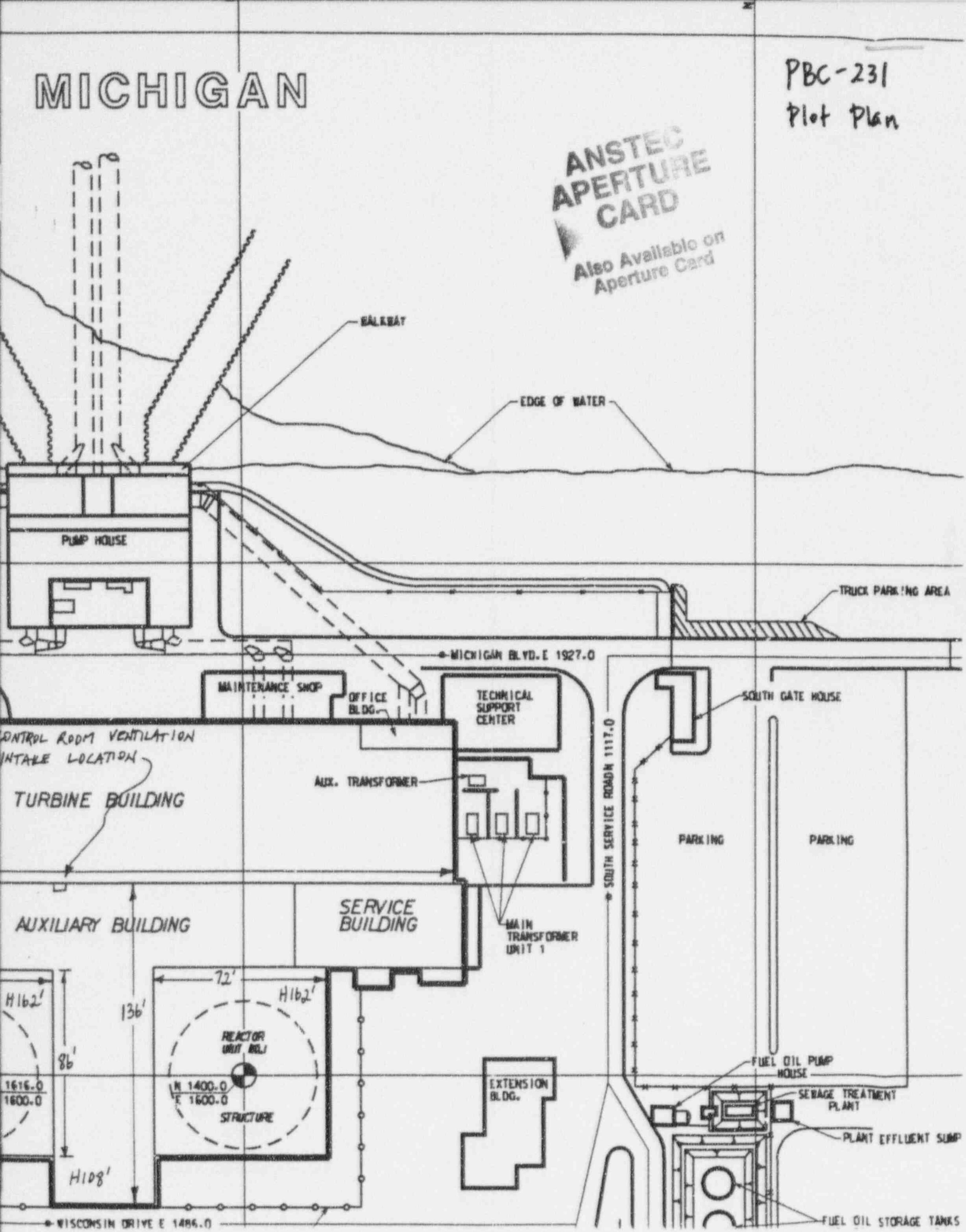


FIGURE 1-2, CR-HVAC SYSTEM MODES
(MAJOR FLOWPATHS SHOWN DARKENED)



TRUE
PLANT
N
24° 43'
LAKE
STORM SEWER
HEADWALL
DIESEL
GENERATOR
BUILDING
FUEL OIL
FILL TANK
NORTH SERVICE
BUILDING
H48'
68'
H 111'9"
H2
STORAGE
MAIN TRANSFORMERS
UNIT 2
295'
AUX. TRANSFORMER
NORTH SERVICE ROAD N 1884.0
72'
REACTOR
UNIT NO.2
STRUCTURE
STEAM
GENERATOR
STORAGE
BUILDING
WAREHOUSE
NO. 4
WAREHOUSE
NO. 3
CONSTRUCTION ACCESS

PBC-231
Plot Plan



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