I. CHANGES (Continued)

M. Change the second sentence of Section 5.1 Site to read "Figure 5-1 shows the plan of site."

N. Add Figure 5-1 "Palisades Nuclear Plant Site Boundary" at the end of Section 5.

O. Change Section 6.5.2.8.1 e to read:

"e. The results of actions taken to correct deficiencies identified by the audit program specified in Specifications 6.5.2.8.2 and 6.5.2.8.3 at least once every six months."

P. Add Section 6.5.2.8.3 (pages attached).

Q. Delete paragraph 4 Solid Wastes, and paragraph 5 Radiological Impact on Man of existing Section 6.9.3.1.A.

R. Add new paragraphs 4 Solid Wastes and 5 Radiological Impact on Man, to Section 6.9.3.1.A (pages attached).

S. Add new paragraph 6 PCP and ODCM changes to Section 6.9.3.1.A (pages attached)

T. Delete existing Section 6.9.3.1.B "Radiological Environmental Monitoring Program".

U. Add new Section 6.9.3.1.B entitled "Annual Radiological Environmental Operating Report" (pages attached).

V. Replace existing Table 6.9-1 with revised Table 6.9-1 (attached).

W. Add new Section 6.9.3.3 d (pages attached).

X. Add new Section 6.18 "Offsite Dose Calculation Manual ODCM" (pages attached).

Y. Add new Section 6.19 "Process Control Program PCP" (pages attached).

Z. Add new Section 6.20 "Major Modifications to Radioactive Liquid Gaseous and Solid Waste Treatment Systems" (pages attached).

AA. Delete Special Technical Specifications Pursuant to Agreement in its entirety.

BB. Remove and replace Table of Contents in its entirety.

II. DISCUSSION

The above proposed Technical Specifications changes are requested in order to satisfy NUREG-0472 and staff guidance in the development of Radiological Effluent Technical Specifications.

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II. DISCUSSION (continued)

A. These definitions are being added in response to NUREG-0472 guidance in the development of RETS.

B. These sections are being added in response to NUREG-0472 guidance and supercede our August 21, 1980 proposed Sections 3.0.1, 3.0.2, 3.0.3 and 3.0.4.

C. This page is being added for administrative purposes.

D. This page is being changed and renumbered for administrative purposes.

E. and F. These pages are being renumbered for administrative purposes.

G. and H. Information contained in existing Section 3.9 has been expanded, revised and reformatted in accordance with NUREG-0472 guidance and is now located in Section 3.24 "Radiological Effluent Releases" of this proposal.

I. All process monitor discussion has been relocated to proposed Tables 4.24-1 "Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements" and 4.24-2 "Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements".

J. and K. Information contained in existing Section 4.11 "Environmental Monitoring Program" has been reformatted and revised in accordance with NUREG-0472 and staff guidance and is now located in new proposed Section 4.11 "Radiological Environmental Monitoring".

L. This section is being added in response to NUREG-0472 guidance in the development of RETS. The Section is numbered in keeping with the Standard Technical Specifications methodology whereby surveillance requirements are the same subsection number as the LCO. (e.g. LCO 3.24.3 has as its surveillance requirement Section 4.24.3.) Blank Sections 4.20, 4.21, 4.22 and 4.23 are created (and left intentionally blank) for administrative continuity.

M. and N. A new figure was created for this submittal that clearly shows the site boundary.

O. and P. Section 6.5.2.8.3 was added and referenced in 6.5.2.8.1.e to delineate NSB responsibilities for ODCM, PCP and radiological environmental monitoring program.

Q., R., S., T. and U.
Changes being made in response to NUREG-0472 guidance.

V. Example Table 6.9-1 "Environmental Radiological Monitoring Program Summary" has been clarified.

W. A new section concerning the submittal of Special Reports referenced in these proposed specifications has been created.

X., Y. and Z.
These Sections were added in response to NUREG-0472 and staff guidance.

AA. These Sections are being deleted for the reasons specified in the cover letter.

Analysis of No Significant Hazards Consideration

The changes proposed herein constitute additional limitations, restrictions and controls not presently included in the Technical Specifications.

The operation of the facility in accordance with the above change will not: 1) involve an increase to the probability or consequences of an accident previously evaluated; 2) create the possibility a new or different kind of accident from those previously evaluated; and 3) involve a significant reduction in the margin of safety.

III. CONCLUSION

The Palisades Plant Review Committee in Special Meeting Number 84-048 on July 16, 1984 has reviewed this Technical Specification Change Request and has determined that this change does not involve an unreviewed safety question and therefore involves no significant hazards consideration. This change has also been reviewed under the cognizance of the Nuclear Safety Board. A copy of this Technical Specification Change Request has been sent to the State of Michigan official designated to receive such Amendments to the Operating License.

CONSUMERS POWER COMPANY

By

J W Reynolds
Executive Vice President,
Energy Supply

Sworn and subscribed to before me this 31st day of July 1984.

Sherry L Durfey, Notary Public
Jackson County, Michigan

My commission expires November 5, 1986.

SHERRY LYNN DURFEY
Notary Public, Jackson County, Mich.
My Commission Expires Nov. 5, 1986

TS0784-0011-NL02

Consumers Power Company

Palisades Plant - Docket 50-255

TECHNICAL SPECIFICATION CHANGE REQUEST

PROPOSED PAGES

72 Pages

TS0784-0032-NL02

PALISADES PLANT TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS - APPENDIX A

PROPOSED

PALISADES PLANT TECHNICAL SPECIFICATIONS
TABLE OF CONTENTS - APPENDIX A

PROPOSED

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PROPOSED

1.4 MISCELLANEOUS DEFINITIONS (Contd)

GASEOUS RADWASTE TREATMENT SYSTEM

A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system off gases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of gaseous and liquid effluent monitoring alarm/trip set points.

PROCESS CONTROL PROGRAM (PCP)

The PROCESS CONTROL PROGRAM shall contain the current formula, sampling, analyses, tests and determinations to be made to ensure that the processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Part 20, 10 CFR Part 71 and Federal and State regulations and other requirements governing the disposal of the radioactive waste.

SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned nor otherwise controlled by the licensee.

UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or, any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional and/or recreational purposes.

VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Feature (ESF) atmospheric clean-up systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

LIMITING CONDITIONS FOR OPERATION

3.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

3.0.1 Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the plant conditions or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

3.0.2 Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the Action requirements is not required.

3.0.3 When a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of cirsumstances in excess of those addressed in the specification, within one hour action shall be initiated to place the unit in a condition in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,

2. At least HOT SHUTDOWN within the following 6 hours, and

3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

3.0.4 Entry into a plant condition or other specified condition shall not be made unless the conditions for the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to plant conditions as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

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TO CONTAIN 8/21/80 SECTION 3.0.5 (PROPOSED)

3.1 PRIMARY COOLANT SYSTEM

Applicability

Applies to the operable status of the primary coolant system.

Objective

To specify certain conditions of the primary coolant system which must be met to assure safe reactor operation.

Specifications

3.1.1 Operable Components

a. At least one primary coolant pump or one shutdown cooling pump shall be in operation whenever a change is being made in the boron concentration of the primary coolant.

b. Four primary coolant pumps shall be in operation whenever the reactor is operated continually above 5% of rated power (exception to this specification is permitted as described in Table 2.3.1, Item 1).

c. The minimum flow for various power levels shall be as shown in Table 2.3.1.

   The measured four primary coolant pumps operating reactor vessel flow (as determined by reactor coolant pump differential pressures and pump performance curves) shall be 126.9×10^6 lb/h or greater, when corrected to 532°F.

   In the event the measured flow is less than that required above, the limits specified on Figure 2-3 shall be reduced by 1°F in inlet temperature for each 1% of reactor flow deficiency.

   Continuous operation at power shall be limited to four-pump operation. Following loss of a pump, thermal power shall be reduced as specified in Table 2.3.1 and appropriate corrective action implemented. With one or more pumps out of service, return the pumps to service (return to four-pump operation) or be in hot standby (or below) within 24 hours. Start-up (above hot standby) with less than four pumps is not permitted.

d. Both steam generators shall be capable of performing their heat transfer function whenever the average temperature of the primary coolant is above 325°F.

e. Maximum primary system pressure differentials shall not exceed the following:

   (1) Maximum steam generator operating transient differential of 1530 psi.

3.1.1 Operable Components (Continued)

(2) Hydrostatic tests shall be conducted in accordance with applicable paragraphs of Section XI ASME Boiler & Pressure Vessel Code (1974). Such tests shall be conducted with sufficient pressure on the secondary side of the steam generators to restrict primary to secondary pressure differential to a maximum of 1380 psi. Maximum hydrostatic test pressure shall not exceed 1.1 Po plus 50 psi where Po is nominal operating pressure.

(3) Primary side leak tests shall be conducted at normal operating pressure. The temperature shall be consistent with applicable fracture toughness criteria for ferritic materials and shall be selected such that the differential pressure across the steam generator tubes is not greater than 1380 psi.

(4) Maximum secondary hydrostatic test pressure shall not exceed 1250 psia. A minimum temperature of 100°F is required. Only ten cycles are permitted.

(5) Maximum secondary leak test pressure shall not exceed 1000 psia. A minimum temperature of 100°F is required.

(6) In performing the tests identified in 3.1.1.e(4) and 3.1.1.e(5), above, the secondary pressure shall not exceed the primary pressure by more than 350 psi.

f. Nominal primary system operation pressure shall not exceed 2100 psia.

g. The reactor inlet temperature (indicated) shall not exceed the value given by the following equation at steady state 100% power operation:

$$T_{inlet} \leq 538.0 + 0.03938\ (P-2060) + 0.00004843\ (P-2060)^2 + 1.0342\ (W-120.2)$$

Where: T_{inlet} = reactor inlet temperature in F°
P = nominal operating pressure in psia
W = total recirculating mass flow in 10^6 lb/h corrected to the operating temperature conditions.

Note: This equation is shown in Figure 3-0 for a variety of mass flow rates.

h. A reactor coolant pump shall not be started with one or more of the PCS cold leg temperatures $\leq$ 250°F unless 1) the pressurizer water volume is less than 700 cubic feet or 2) the secondary water temperature of each steam generator is less than 70°F above each of the PCS cold leg temperatures.

3.1 PRIMARY COOLANT SYSTEM (Cont'd)

3.1.1 Operable Components (Cont'd)

i. The PCS shall not be heated or maintained above 325°F unless a minimum of 375 kW of pressurizer heater capacity is available from both buses 1D and 1E. Should heater capacity from either bus 1D or 1E fall below 375 kW, either restore the inoperable heaters to provide at least 375 kW of heater capacity from both buses 1D and 1E within 72 hours or be in hot shutdown within the next 12 hours.

3.9 EFFLUENT RELEASE

DELETED IN ITS ENTIRETY

Information now contained in Section 3.24
"Radiological Effluent Releases"

12 Pages
3-50
3-50a
3-51
3-51a
3-52
3-52a
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3-53a
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3-57

3.24 RADIOLOGICAL EFFLUENT RELEASES

3.24.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.24.1.1 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.24-1 shall be OPERABLE with their alarm/trip set points set to ensure that the limits of Specification 3.24.3 are not exceeded. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY: At all times.

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the above specification, without delay suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable, or change the set point so it is acceptably conservative.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.24-1. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.1.1

TABLE 3.24-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line (RIA 1049)	(1)	28
b. Steam Generator Blowdown Effluent Line (RIA 0707)	(1)	29
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Service Water System Effluent Line (RIA 0833)	(1)	30
b. Turbine Building (Floor Drains) Sumps Effluent Line (RIA 5211)	(1)	30
3. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line (FIC 1051 or 1050)	(1)	31
4. CONTINUOUS COMPOSITE SAMPLERS		
a. Turbine Building Sumps Effluent Line	(1)	30

TABLE 3.24-1 (Contd)

TABLE NOTATION

ACTION 28 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.24.3.1, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving:

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcuries/ml at least once per 12 hours.

ACTION 30 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a lower limit of detection of at least 10^{-7} microcurie/ml.

ACTION 31 - With the number of channels OPERABLE less than required by the Minimum channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves or tank levels may be used to estimate flow.

3.24.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.24.2.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.24.2 shall be OPERABLE with their alarm/trip set points set to ensure that the limits of Specification 3.24.5.1 are not exceeded. The alarm/trip set points of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY: As shown in Table 3.24-2.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip set point less conservative than required by the above Specification, without delay, suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable or change the set point so it is acceptably conservative.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.24-2. Exert best efforts to return the instruments to OPERABLE status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.2.1

TABLE 3.24-2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor (RIA 1113) Providing Alarm and Automatic Termination of Release	(1)	At All Times	35
b. Effluent System Flow Rate Measuring Device (FI 1121)	(1)	At All Times	36
2. CONDENSER EVACUATION SYSTEM (RIA 0631)			
a. Noble Gas Activity Monitor	(1)	Above 210°F	37
3. STACK GAS EFFLUENT SYSTEM			
a. Noble Gas Activity Monitor (RIA 2326 or RIA 2318)	(1)	At All Times	37
b. Iodine/Particulate/Sampler/Monitor (RIA 2325)	(1)	At All Times	37
c. Sampler Flow Rate Monitor	(1)	At All Times	36
d. Hi Range Noble Gas (RIA 2327)	(1)	Above 210°F	38
4. STEAM GENERATOR BLOWDOWN VENT SYSTEM			
a. Noble Gas Activity Monitor (RIA 2320)	(1)	Above 210°F	37
5. MAIN STEAM SAFETY AND DUMP VALVE DISCHARGE LINE			
a. Gross Gamma Activity Monitor (RIA 2323 and 2324)	1 per Main Steam Line	Above 325°F	38
6. ENGINEERED SAFEGUARDS ROOM VENT SYSTEM			
a. Noble Gas Activity Monitor (RIA 1810 and 1811)	1 per room	Above 210°F	38

TABLE 3.24-2 (Cont.)

TABLE NOTATION - ACTION STATEMENTS

ACTION 35 - With the number of channels OPERABLE less than required by Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 24 hours for continuous releases or four hours for batch releases.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 38 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements, initiate the preplanned alternate method of monitoring the appropriate parameter(s), within 72 hours, and:

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

2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.3.3.d within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status

3.24.3 LIQUID EFFLUENTS CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.24.3.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (see Figure 5-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, without delay, restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.3.1.a

Refer to 4.24.3.1.b

3.24.4 LIQUID EFFLUENT DOSE

LIMITING CONDITION FOR OPERATION

3.24.4.1 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each reactor unit to UNRESTRICTED AREAS (see Figure 5-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report per Specification 6.9.3.3.d that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. This Special Report shall also include the results of radiological analyses of the drinking water source.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.4.1

3.24.5 GASEOUS EFFLUENTS DOSE

LIMITING CONDITION FOR OPERATION

3.24.5.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (see Figure 5-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

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

APPLICABILITY: At all times.

ACTION:

a. With the dose rate(s) averaged over a period of one hour exceeding the above limits, without delay, restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

Refer to 4.24.5.1.a

Refer to 4.24.5.1.b

LIMITING CONDITION FOR OPERATION

3.24.5.2 The air dose due to noble gases released in gaseous effluents to areas at and beyond the SITE BOUNDARY (see Figure 5-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report per Specification 6.9.3.3.d that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.5.2

LIMITING CONDITION FOR OPERATION

3.24.5.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine 133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents released, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5-1) shall be limited to the following:

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

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days a Special Report per Specification 6.9.3.3.d that identifies the cause(s) for exceeding the limit and define(s) the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.5.3

3.24.6 GASEOUS WASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.24.6.1 THE WASTE GAS DECAY TANK SYSTEM shall be used to reduce radioactive gaseous effluents by holding gaseous waste collected by the system for a minimum of 15 days up to 60 days.

APPLICABILITY: When gaseous waste exceeds a Xe-133 concentration of 1E-05 μCi/cc.

ACTION:

a. If a waste gas decay tank is required to be released with less than 60 days holdup time, the system waste gas tank contents shall be evaluated and the waste gas decay tank with the lowest Xe-133 concentration shall be released.

b. Gaseous waste may be discharged directly from the waste gas surge tank through a high-efficiency filter or from a waste gas decay tank with less than 15 days of holdup directly to the stack for a period not to exceed 7 days if the holdup system equipment is not available and the release rates meet Specification 3.24.5.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Not applicable.

3.24.7 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.24.7.1 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

a. With the provisions of the PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.7.1

3.24.8 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specifications 3.24.4.1.a, 3.24.4.1.b, 3.24.5.2.a, 3.24.5.2.b, 3.24.5.3.a or 3.24.5.3.b, calculations should be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.24.8.1 have been exceeded. If such is the case prepare and submit to the Commission within 30 days a Special Report per Specification 6.9.3.3.d that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request and a variance is granted until staff action on the request is complete.

b. The provisions of Specification 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

Refer to 4.24.8.1.a

Refer to 4.24.8.1.b

BASES 3.24 RADIOLOGICAL EFFLUENT RELEASES

3.24.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip set points for these instruments shall be calculated and adjusted in accordance with the methodology and parameters in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3.24.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

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

The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3.24.3 LIQUID EFFLUENTS CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its Maximum Permissable Concentration (MPC) in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

This specification applies to the release of liquid effluents from all reactors at the site.

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

3.24.4 LIQUID EFFLUENT DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." Also, for fresh-water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR Part 141. The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to man from Routine Releases of Reactor Effluents for the Purpose of Evaluating compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3.24.5 GASEOUS EFFLUENTS DOSE

3.24.5.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/yr to the total body or to less than or equal to 3000 mrems/yr to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/yr.

This specification applies to the release of gaseous effluents from all reactors at the site.

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

3.24.5.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

3.24.5.3 DOSE - IODINE-131, IODINE-133, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating

Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3 24.6 GASEOUS RADWASTE TREATMENT SYSTEM

The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable" by meeting the design objectives given in Section II.D of Appendix I to 10 CFR 50.

It is expected that releases of radioactive materials in effluents shall be kept at small fractions of the limits specified in 20.106 of 10 CFR 20. At the same time the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small fractions, but still within the limits specified in Specification 3.24.5.

3.24.7 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited t , waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents and mixing and curing times.

3.24.8 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

TABLE 4.1.3

Minimum Frequencies for Checks, Calibrations and Testing of Miscellaneous Instrumentation and Controls

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Start-Up Range Neutron Monitors	a. Check	S	a. Comparison of both channel count rate indications when in service.
	b. Test	P	b. Internal test signals.
2. Primary Rod Position Indication System	a. Check	S	a. Comparison of output data with secondary RPIS
	b. Check	M	b. Check of power dependent insertion limits monitoring system.
	c. Calibrate (1)	R	c. Physically measured rod drive position used to verify system accuracy. Check rod position interlocks.
3. Secondary Rod Position Indication System	a. Check	S	a. Comparison of output data with primary RPIS.
	b. Check	M	b. Same as 2(b) above.
	c. Calibrate (1)	R	c. Same as 2(c) above, including out-of-sequence alarm function.
4. Area Monitors Note: Process Monitor Surveillance Requirements are located in Tables 4.24-1 and 4.24-2	a. Check	D	a. Normal readings observed and internal test signals used to verify instrument operation.
	b. Calibrate	R	b. Exposure to known external radiation source.
	c. Test	M	c. Detector exposed to remote operated radiation check source.
5. Emergency Plan Radiation Instruments	a. Calibrate	A	a. Exposure to known radiation source.
	b. Test	M	b. Battery check.
6. Environmental Monitors	a. Check	M	a. Operational check.
	b. Calibrate	A	b. Verify airflow indicator.
7. Pressurizer Level Instruments	a. Check	S	a. Comparison of six independent level readings.
	b. Calibrate	R	b. Known differential pressure applied to sensor.
	c. Test	M	c. Signal to meter relay adjusted with test device.

4.11 RADIOLOGICAL ENVIRONMENTAL MONITORING

4.11.1 The radiological environmental monitoring program shall be conducted as specified in Table 4.11-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 4.11-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.3.1B a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 4.11-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents. When more than one of the radionuclides in Table 4.11-2 are detected in the sampling medium, this report shall be submitted if:

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

When radionuclides other than those in Table 4.11-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.24.4.1, 3.24.5.2 and 3.24.5.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 4.11-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Annual Radiological Environmental Report.

4.11.2 The radiological environmental monitoring samples shall be collected pursuant to Table 4.11-1 and shall be analyzed pursuant to the requirements of Table 4.11-1 and the detection capabilities required by Table 4.11-3.

4.11.3 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 9 overland meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 5) m^2 (500 ft^2) producing broad leaf vegetation.

4.11.4 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.3.1B and shall be included in a revision of the ODCM for use in the following calendar year.

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

4.11.6 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.3.1.B.

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*Broad leaf vegetation sampling of at least three different kinds of vegetation may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 4.11-1.4c shall be followed, including analysis of control samples.

TABLE 4.11-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION[b]	21 routine monitoring stations either with two or more dosimeters or with one instrument for m· isuring and recording dose rate continuously, placed as follows: An inner ring of stations, one in each overland meteorological sector (9) in the general area of the SITE BOUNDARY. An outer ring of stations, one in each overland meteorological sector (9) within the 12 km range from the site. The balance of the stations (3) to be placed to serve as control stations.	Quarterly.	Gamma dose quarterly.

TABLE 4.11-1 (Contd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
2. AIRBORNE			
Radioiodine and Particulates	Samples from 5 locations: 3 samples from within 6 km of the SITE BOUNDARY in different sectors (2.4 km-SSW, 5.6 km - ESE and 1.6 km - N). 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q (Covert - 5.6 km - SE). 1 sample from a control location in the least prevalent wind direction[c] (Grand Rapids 89 km - NNE).	Continuous sampler operation with sample collection weekly or more frequently if required by dust loading.	Radioiodine Cannister: I-131 analysis weekly for each filter change. Particulate Samples: Gross beta radioactivity analysis following filter change.[d] Gamma isotopic analysis[e] if gross beta > 1.0 pCi/m^3.

TABLE 4.11-1 (Contd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Lake (Surface)	Plant lake water inlet.	Composite sample over 1-month period.[f]	Gross beta (> 10 pCi/ℓ requires gamma) and tritium monthly.
b. Well (Drinking)	Samples from plant, state park and Covert Township park wells.	Monthly - grab sample.	Gross beta (> 10 pCi/ℓ requires gamma) and tritium monthly.
c. Lake (Drinking)	1 sample of South Haven drinking water supply.	Composite sample over 1-month period.[f]	Gross beta (> 10 pCi/ℓ requires gamma) and tritium monthly.
d. Sediment From Shoreline	1 sample from between north boundary and Van Buren State Park beach.	Semiannually.	Gamma isotopic analysis[e] semiannually.

TABLE 4.11-1 (Contd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	Samples from milking animals in 3 locations between 5-13 km distance. 1 sample from milking animals at a control location, 15-30 km distance.	Monthly.	Gamma isotopic[e] and I-131 analysis monthly.
b. Fish and Invertebrates	Sample 2 species of commercially and/or recreationally important species in vicinity of plant discharge area. 1 sample of same species in areas not influenced by plant discharge.	Sample in season or semiannually if they are not seasonal.	Gamma isotopic analysis[e] on edible portions.

TABLE 4.11-1 (Contd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[a]	Sampling and Collection Frequency	Type and Frequency of Analysis
c. Food Products	1 sample each of two principal fruit crops (blueberries and apples).	At time of harvest.[g]	Gamma isotopic analyses[e] on edible portion.
	Samples of 3 different kinds of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q if milk sampling is not performed. (SE or SSE sectors near site)	Monthly when available.	Gamma isotopic[e] and I-131 analysis.
	1 sample of each of the similar broad leaf vegetation grown 15-30 km distance in the least prevalent wind direction if milk sampling is not performed. (SSW or S sectors)	Monthly when available.	Gamma isotopic[e] and I-131 analysis.

TABLE 4.11-1 (Contd)

TABLE NOTATION

[a]Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, every effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.3.1B. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program.

[b]One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors or phosphor readout zones in a packet are considered as two or more dosimeters.

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

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

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

TABLE 4.11-1 (Contd)

TABLE NOTATION

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

[g]If harvest occurs more than once a year, sampling shall be performed during each discrete harvest.

TABLE 4.11-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, Wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, Wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR Part 141 value. If no drinking water pathway exists, a value of 30,000 pCi/ℓ may be used.

TABLE 4.11-3

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

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

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, Wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, Wet)	Sediment (pCi/kg, Dry)
Gross Beta	4	0.01				
H-3	2,000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58, 60	15		130			
Zn-65	30		260			
Zr-Nb-95	15					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	80	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15			15		

*If no drinking water pathway exists, a value of 3,000 pCi/ℓ may be used.

TABLE 4.11-3 (Contd)

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.3.1B.

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

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \text{Exp}\ (-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

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

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting.

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

TABLE 4.11-3 (Contd)

TABLE NOTATION

It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.3.1B.

[d]LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

4.11 BASES-RADIOLOGICAL ENVIRONMENTAL MONITORING

4.11.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the Radiological Assessment Branch Technical Position on Environmental Monitoring. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.11-3 are considered optimum for routine environmental measurements in industrial laboratories.

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

4.11.3 LAND USE CENSUS

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

4.11.5 INTERLABORATORY COMPARISON PROGRAM

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

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4.24 RADIOLOGICAL EFFLUENT RELEASES

4.24.1 RADIOLOGICAL LIQUID EFFLUENT MONITORING INSTRUMENTATION

4.24.1.1 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.24-1.

4.24.2 RADIOLOGICAL GASEOUS EFFLUENT MONITORING INSTRUMENTATION

4.24.2.1 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 4.24-2.

4.24.3 LIQUID EFFLUENT CONCENTRATION

4.24.3.1.a Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.24-3.

4.24.3.1.b The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.24.3.1.

4.24.4 LIQUID EFFLUENT DOSE

4.24.4.1 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

4.24.5 GASEOUS EFFLUENT DOSE

4.24.5.1.a The dose rate due to noble gases in gaseous effluents shall be determined to be within the limits of 3.24.5.1.a in accordance with the methodology and parameters in the ODCM.

4.24.5.1.b The dose rate due to iodine-131, iodine-133, tritium and all radionuclides in particulate form with half lives greater than 8 days in gaseous effluents shall be determined to be within the limits of 3.24.5.1.b in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.24-5.

4.24.5.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

4.24.5.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium and radionuclides in particulate form with half lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

4.24.6 GASEOUS WASTE SYSTEM - NONE

4.24.7 SOLID RADIOACTIVE WASTE

4.24.7.1 The PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (eg, filter sludges, spent resins, evaporator bottoms and boric acid solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.19, to assure SOLIDIFICATION of subsequent batches of waste.

4.24.8 TOTAL DOSE

4.24.8.1.a Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.24.4.1, 4.24.5.2 and 4.24.5.3 and in accordance with the methodology and parameters in the ODCM.

4.24.8.1.b Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This requirement is applicable only under conditions set forth in Specification 3.24.8.1.a.

TABLE 4.24-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line (RIA 1049)	P	P	R(3)	Q(1)(2)
b. Steam Generator Blowdown Effluent Line (RIA 0707)	D	M	R(3)	Q(1)(2)
2. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE				
a. Service Water System Effluent Line (RIA 0833)	D	M	R(3)	Q(2)
b. Turbine Building (Floor Drains) Sumps Effluent Line (RIA 5211)	D	M	R(3)	Q(2)
3. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line (FIC 1051 or 1050)	D(4)	NA	R	Q
4. TURBINE SUMP EFFLUENT COMPOSITER	D(4)	NA	NA	NA

TABLE 4.24-2

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor-Providing Alarm and Automatic Termination of Release	D(4)	P	R(3)	Q(1)(2)	*
b. WGDT Effluent Flow Rate Device	D(4)	NA	R	NA	*
2. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	Above 210°F
3. STACK GAS EFFLUENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Particulate Sampler/Monitor	W	M**	R(3)**	NA	*
c. Sampler Flow Rate Monitor	D	NA	R	NA	*
d. Hi Range Noble Gas	D	M	R(3)	Q(2)	Above 210°F
4. STEAM GENERATOR BLOWDOWN VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	Above 210°F

*At all times other than when the line is valved out and locked.

**Sampler not applicable

TABLE 4.24-2(Cont.)

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. MAIN STEAM SAFETY AND DUMP VALVE DISCHARGE LINE					
a. Gross Gamma Activity Monitor	D	M	R(3)	Q(2)	Above 325°F
6. ENGINEERED SAFEGUARDS ROOM VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(1)(2)	Above 210°F

TABLE NOTATION

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

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that cotrol room alarm annunciation occurs if either of the following conditions exists:

a. Instrument indicates measured levels above the alarm set point.
b. Circuit failure.

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

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

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

TABLE FREQUENCY NOTATION

D	At least once per 24 hours	Q	At least once per 92 days
M	At least once per 31 days	R	At least once per 18 months
P	Prior to radioactive batch release	W	At least once per week

TABLE 4.24-3

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Batch Waste Release Tanks[b]	P Each Batch	P Each Batch	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[d]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[d]	Sr-89, Sr-90	5×10^{-8}
B. Continuous Releases[e]	Continuous[f]	W Composite[f]	Principal Gamma Emitters[c]	5×10^{-7}
			I-131	1×10^{-6}
Turbine Sump	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
Steam Generator Blowdown	Continuous[f]	M Composite[f]	H-3	1×10^{-5}
Service Water			Gross Alpha	1×10^{-7}
	Continuous[f]	Q Composite[f]	Sr-89, Sr-90	5×10^{-8}

FREQUENCY NOTATION

P	Prior to batch release
M	Calendar month
Q	Calendar quarter
W	Calendar week

TABLE 4.24-3 (Contd)

TABLE NOTATION

[a]The LLD is defined, in Table 4.11-3, note C.

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

[c]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99,* Cs-134, Cs-137, Ce-141 and Ce-144.* This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.3.1.a.

*(LLD - 5E-06 because of low gamma yields.)

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

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

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

TABLE 4.24-5

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[b]	1×10^{-4}
B. Containment PURGE	P Each PURGE Grab Sample	P Each PURGE	Principal Gamma Emitters[b]	1×10^{-4}
C. Stack Gas Effluent	Continuous[c]	W[d,e] Charcoal Sample	I-131, I-133	1×10^{-12}
	Continuous[c]	W[d,e] Particulate Sample	Principal Gamma Emitters[b] (I-131, Others)	1×10^{-11}
	Continuous[c]	Q Composite Particulate Sample	Sr-89, Sr-90 and Gross Alpha	1×10^{-11}
	Continuous[c]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1E-06

TABLE 4.24-5 (Contd)

TABLE NOTATION

[a]The LLD is defined, in Table 4.11-3, note C.

[b]The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135 and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99*, Cs-134, Cs-137, Ce-141 and Ce-144* for particulate emissions. This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.3.1.a.

*(LLD 1E-10 because of low gamma yields)

[c]The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.24.5.1, 3.25.5.2 and 3.24.5.3.

[d]Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing or after removal from sampler.

[e]With channels operable on iodine monitor RIA 2325 less than required per Specification 3.24.2, sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, start-up or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

5.0 DESIGN FEATURES

5.1 SITE

The Palisades reactor shall be located on 487 acres owned by Consumers Power Company on the eastern shore of Lake Michigan approximately four and one-half miles south of the southern city limits of South Haven, Michigan. Figure 5-1 shows the plan of the site. The minimum distance to the boundary of the exclusion area as defined in 10 CFR 100.3 shall be 677 meters.

5.2 CONTAINMENT DESIGN FEATURES

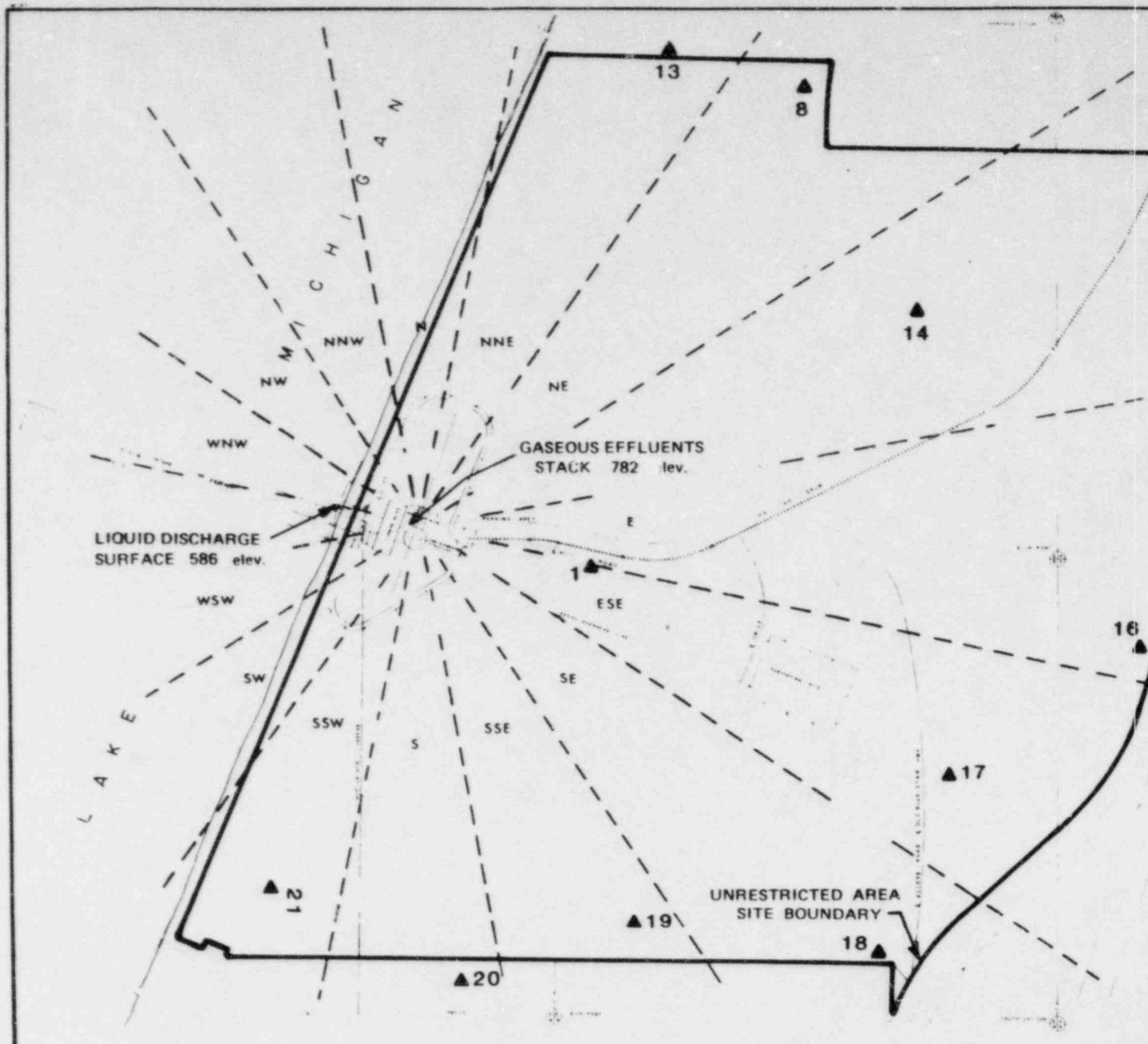
5.2.1 Containment Structure

a. The containment structure completely encloses the primary coolant system to minimize release of radioactive material to the environment should a failure of the primary coolant system occur. The prestressed, post-tensioned concrete structure provides adequate biological shielding for both normal operation and accident situations and is designed for low leakage at a design pressure of 55 psig and 283°F.

   The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a design basis loss-of-coolant accident. In this event, the total energy contained in the water of the primary coolant system is assumed to be released into the containment through a double-ended break of the largest primary coolant pipe coincident with a loss of normal and standby electrical power. Subsequent pressure behavior is determined by the engineered safety features and the combined influence of energy sources and heat sinks.

b. The external design pressure of the containment shell is 3 psig. This value is approximately 0.5 psig greater than the maximum external pressure that could be developed if the containment were sealed during a period of low barometric pressure and high temperature and, subsequently, the containment atmosphere were cooled with a concurrent major rise in barometric pressure. Vacuum breakers are therefore not provided.

c. The containment is designed as a seismic Class I structure.



SITE ENVIRONMENT
TLD STATIONS

PALISADES NUCLEAR PLANT
SITE BOUNDARY
FIGURE 5-1

QUORUM

6.5.2.7 A quorum of NSB shall consist of the Chairman and four (4) members. (The Vice Chairman may be a voting member when not acting in the capacity of Chairman.) No more than a minority of the quorum shall have line responsibility for operation of the facility. It is the responsibility of the Chairman to ensure that the quorum convened for a meeting contains appropriately qualified members or has at its disposal consultants sufficient to carry out the review functions required by the meeting agenda.

6.5.2.8 RESPONSIBILITIES

REVIEW

6.5.2.8.1 NSB shall be responsible for the review of:

a. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

b. All reportable events and other violations (of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements or of internal procedures or instructions) having nuclear safety significance.

c. Issues of safety significance identified by the Plant General Manager, the NSB Chairman, Executive Engineer - NAPO or the PRC.

d. Proposed changes in the operating license or Appendix "A" Technical Specifications.

e. The results of actions taken to correct deficiencies identified by the audit program specified in Specifications 6.5.2.8.2 and 6.5.2.8.3 at least once every six months.

f. Safety evaluations for changes to procedures, equipment, or systems and tests or experiments completed under the provisions of 10 CFR 50.59, to verify that such actions did not constitute an unreviewed safety question.

g. Maintain cognizance of PRC activities through NAPO attendance at scheduled PRC meetings or through review of PRC meeting minutes.

AUDITS

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

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

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

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

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

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

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

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

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

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

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

6.5.2.8.3 Audits of Nuclear Operations Department activities shall be performed under the cognizance of the NSB. These audits shall encompass.

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

b. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

c. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

Audit reports encompassed by Specification 6.5.2.8.3 above shall be forwarded to the NSB Vice Chairman and Secretary, and Management poistions responsible for the areas audited within thirty (30) days after completion of the audit.

AUTHORITY

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

d. Alpha Radioactivity

Total curies of gross alpha-emiting material determined to be released in liquid effluents.

e. Volumes

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

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

4. Solid Wastes

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Container burial volume,

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

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

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

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

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

5. Radiological Impact on Man

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

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

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

c. Organ doses (greater than 1 milliRem) to individuals in unrestricted areas from radioactive iodine and radioactive material in particulate form from the major pathways of exposure.

d. Total body doses (greater than 1 manRem) to the population and average doses (greater than 1 milliRem) to individuals in the population from receiving water-related pathways to a distance of 50 miles from the site.

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

6. PCP and ODCM Changes

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 4.11.4.

6.9.3.1.B <u>ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT</u>

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

The annual radiological environmental operating reports shall include summaries, interpretation and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate) and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use census required by Specification 4.11.3.

6.9.3.1.B (Continued)

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Table 4.11-4 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program including sampling methods for each sample type, a map of all sampling locations keyed to a table giving distances and directions from one reactor and the results of land use censuses required by the Specification 4.11.3, and results of the interlaboratory comparison program required by Specification 4.11.5.

DELETED IN ITS ENTIRETY

6.9.3.2 Nonroutine Reports

A report shall be submitted in the event that (a) the radiological monitoring programs are not substantially conducted as described in Section 4.11; or (b) an unusual or important event occurs from plant operation that causes a significant environmental impact or affects a potential environmental impact. Reports shall be submitted within 30 days.

TABLE 6.9-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility ______________________ Docket No ___________

Location of Facility ______________________ Reporting Period ___________
(County, State)

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range[b]	Location with Highest Annual Mean: Name Distance and Direction	Location with Highest Annual Mean: Mean (f)[b] Range[b]	Control Locations Mean (f)[b] Range[b]	Number of REPORTABLE OCCURRENCES
Air Particulates (pCi/m^3)	Gross β 416	0.003	0.08 (200/312) (0.05-2.0)	Middletown 5 miles 340°	0.10 (5/52) (0.08-2.0)	0.08 (8/104) (0.05-1.40)	1
	γ-Spec 32						
	^{137}Cs	0.003	0.05 (4/24) (0.03-0.13)	Smithville 2.5 miles 160°	0.08 (2/4) (0.03-0.13)	<LLD	4
	^{140}Ba	0.003	0.03 (2/24) (0.01-0.08)	Podunk 4.0 miles 270°	0.05 (2/4) (0.01-0.08)	0.02 (1/8)	1
	^{89}Sr 40	0.002	<LLD	-	-	<LLD	0
	^{90}Sr 40	0.0003	<LLD	-	-	<LLD	0

TABLE 6.9-1 (Continued)

Medium of Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range[b]	Location with Highest Annual Mean Name Distance and Direction	Location with Highest Annual Mean Mean (f)[b] Range[b]	Control Locations Mean (f)[b] Range[b]	Number of REPORTABLE OCCURRENCES
Fish pCi/kg (dry weight)	γ-Spec 8						
	^{137}Cs	80	<LLD	-	<LLD	90 (1/4)	0
	^{134}Cs	80	<LLD	-	<LLD	<LLD	0
	^{60}Co	80	120 (3/4) (90-200)	River Mile 35 Podunk River	See Column 4	<LLD	0

[a]Nominal Lower Limit of Detection (LLD) as defined in table notation a of Table 6.6-3 of Specification 6.6.
[b]Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses. (f)
[d]Note: The example data are provided for illustrative purposes only.

6.9.3.3 Special Report

a. Special reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable referenced specification:

Area	Specification Reference	
Prestressing, Anchorage, Liner and Penetration Tests	4.5.4 4.5.5	90 Days After Completion of the Test*
Primary System Surveillance Evaluation and Review	4.3	Five Years

* A test is considered to be complete after all associated mechanical, chemical, etc., tests have been completed.

b. Bimonthly status reports on the program to improve the reliability of the paths to prevent post-LOCA boron precipitation shall be submitted to the Division of Operating Reactors until completed.

c. Deleted.

6.9.3.3.d

Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, US Nuclear Regulatory Commission, Washington, DC 20555 within the time period specified for each report.

6.10 RECORD RETENTION

(Records not previously required to be retained shall be retained as required below commencing with the effective date of Technical Specification Change No 20. A system for efficient record retrieval shall be in effect not later than June 1976.)

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. Reportable Occurrences.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

SPECIAL TECHNICAL SPECIFICATIONS
PURSUANT TO AGREEMENT

DELETED IN ITS ENTIRETY

6.18 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.18.1 The ODCM shall be approved by the Commission prior to implementation.

6.18.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the changes(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered together with appropriate analyses or evaluations justifying the change(s);

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

   c. Documentation of the fact that the change has been reviewed and found acceptable as defined by Specification 6.18.3.

6.18.3 Licensee initiated changes to and implementing procedures for the ODCM shall become effective upon review and approval by the responsible Nuclear Operations Department per CPC 2A (Quality Assurance Program).

6.19 PROCESS CONTROL PROGRAM (PCP)

6.19.1 The PCP shall be approved by the Commission prior to implementation.

6.19.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the changes(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

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

   c. Documentation of the fact that the change has been reviewed and found acceptable as defined by Specification 6.19.3.

6.19.3 Licensee initiated changes to and implementing procedures for the PCP shall become effective upon review and approval by the responsible Nuclear Operations Department per CPC 2A (Quality Assurance Program).

6.20 MAJOR MODIFICATIONS TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS

6.20.1 Licensee initiated major modifications to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in a special report pursuant to Specification 6.9.3.3d within six months of the time the safety evaluation was reviewed by the (PRC). The discussion of each modification shall contain:

   a. A summary of the evaluation that led to the determination that the modification could be made in accordance with 10 CFR Part 50.59,

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

   c. Documentation of the fact that the modification was reviewed and found acceptable by the (PRC).

2. Shall become effective upon review and acceptance by the Plant General Manager.

PROPOSED

PALISADES NUCLEAR POWER PLANT

OFFSITE DOSE CALCULATION MANUAL

July 31, 1984

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I. GASEOUS EFFLUENTS

A. ALARM/TRIP SETPOINT METHOD

Specification 3.24.5.1 requires that MPC is not exceeded when averaged over a period not to exceed 1 hour. Based on the definition of MPC, the dose rate in unrestricted areas due to gaseous effluents from the site shall be limited at all times to the following values:

1. 500 mrem/y to the total body and 3,000 mrem/y to the skin from noble gases.
2. 1,500 mrem/y to any organ from radioiodines and particulates, due to inhalation.

Specification 3.24.2.1 requires gaseous effluent monitors to have alarm/trip setpoints to ensure that offsite concentrations, when averaged over 1 hour, will not be greater than MPC. This section of the ODCM describes the methodology that will be used to determine these setpoints.

The methodology for determining alarm/trip setpoints is divided into two major parts. The first consists of calculating an allowable concentration for the nuclide mixture to be released. The second consists of determining monitor response to this mixture in order to establish the physical settings on the monitors.

1.0 Allowable Concentration

The total MPC-fraction (R_k) for each release point will be calculated by the relationship defined by Note 1 of Appendix B, 10 CFR 20:

$$R_{(k)} = (\frac{X}{Q})\ (F)\ \frac{Ci}{MPC_k} = \leq 1.0 \qquad (1.1)$$

Ci = Concentration, at ambient temperature and pressure of nuclide i (μCi/cc)

MPC_i = The MPC of nuclide i from 10 CFR 20, Appendix B

$R_{(k)}$ = The total MPC-fraction for release point k

X/Q = Most conservative sector site boundary dispersion (4.8E-05 sec/m^3)

F = Release flow rate (83,000 cfm = 39.2 m^3/sec) for stack monitor considerations; variable for other monitors.

NOTE: If a batch release is made while a continuous release or another batch release is in progress, the sum of all values of R_k must be less than 1.0.

2.0 Monitor Response

Normal radioactivity releases consist mainly of well-decayed fission gasses. Therefore, monitor response calibrations are performed to fission gas typical of normal releases (mainly Xe-133). Response of monitors used to define fission product release rates under accident conditions may vary from that of Xe-133, however. Monitor response for the two categories of monitor is determined as follows:

1) Normal Releases (aged fission gasses)

Total gas concentration (μCi/cc) at the monitor is calculated. The calibration curve or constant for cpm/(μCi/cc) is applied to determine cpm expected. The setting for monitor alarms is established at some factor (b) greater than 1 but less than $1/R_k$ (Equation 1.1) times the allowed concentration(c):

$$s = b \times c \tag{2.1}$$

2) Accident Releases

Monitors are preset to alarm at or before precalculated offsite dose rates would be achieved under hypothetical accident conditions. These setpoints are established in accordance with Emergency Plan requirements for defining Emergency Action Levels and associated actions. Emergency Implementing Procedures contain monitor-specific curves or calibration constants for conversion between cpm and μCi/cc (or R/hr and μCi/cc), depending on monitor type, for fission product mixtures as a function of mixture decay time.

When these monitors are utilized for other than accident conditions, either an appropriately decayed "accident" conversion curve may be used, or a decayed fission gas calibration factor may be applied. In these cases, setpoints are established as in 1) above.

Setpoints of accident monitors (if set to monitor normal releases) are reset to the accident alarm settings at the end of normal release. Setpoints of other release monitors are maintained at the level used at the latest release (well below the level which would allow MPC to be exceeded at the site boundary), or are reset to approximately three times background in order to detect leakage or inadvertent releases of low level gases.

B. DOSE RATE CALCULATION

1.0 The first step involves calculating a dose rate based on the design objective source term mix used in Appendix I licensing calculations. Historical meteorological data used in licensing are used in this calculation. Doses are determined for (1) noble gases and (2) iodines and particulates. Dose rates as defined in this section are in terms of 10 CFR 50 Appendix I limits of mrem per quarter and millirem per year. All dose pathways of major importance in the Palisades environs are considered.

1.1 Equations and assumptions for calculating doses from noble gases are as follows:

1.1.1 Assumptions

1. Doses to be calculated are the maximum offsite point in air, total body and skin.
2. Exposure pathway is submersion within a cloud of noble gases.
3. Noble gas radionuclide mix is based on the historically observed source term given in Table 1.1, plus additional nuclides.
4. Basic radionuclide data are given in Table 1.2.
5. All releases are treated as ground-level.
6. Meteorological data expressed as joint-frequency distribution of wind speed, wind direction, and atmospheric stability for the period resulting in χ/Q's and D/Q's shown in Table 1.3.
7. Raw meteorological data consist of wind speed and direction measurements at 10m and temperature measurements at 10m and 60m.
8. Dose is to be evaluated at the offsite exposure points where maximum concentrations are expected to exist (overland sector site boundaries), and nearest residents.
9. Potential maximum population (resident) exposure points are identified in Table 1.4.

10. A semi-infinite cloud model is used.
11. For person exposures, credit is taken for shielding by residence (factor of 0.7).
12. Radioactive decay is considered for the plume.
13. Building wake effects on effluent dispersion are considered.
14. A sector-average dispersion equation is used.
15. The wind speed classes that are used are as follows:

Wind Speed Class Number	Range (m/s)	Midpoint (m/s)
1	0.0-0.4	0.2
2	0.4-1.5	0.95
3	1.5-3.0	2.25
4	3.0-5.0	4.0
5	5.0-7.5	6.25
6	7.5-10.0	8.75
7	>10.0	--

16. The stability classes that will be used are the standard A through G classifications. The stability classes 1-7 will correspond to A=1, B=2, . . ., G=7.
17. Terrain effects are not considered.

1.1.2 Equations

To calculate the dose for any one of the exposure points, the following equations are used.

For determining the air concentration of any radionuclide:

$$X_i = \sum_{j=1}^{9} \sum_{K=1}^{7} \left(\frac{2}{\pi}\right)^{1/2} \frac{f_{jk}\, Q_i\, p}{\Sigma zk\, u_j\, (2\pi x/n)} \exp\left(-\lambda_i \frac{x}{u_j}\right) \qquad (1.1)$$

where:

x_i = Air concentration of radionuclide i, μCi/m³.

f_{jk} = Joint relative frequency of occurrence of winds in wind speed class j, stability class k, blowing toward this exposure point, expressed as a fraction.

Q_i = Average release rate of radionuclide i, μCi/s.

p = Fraction of radionuclide remaining in plume.

Σ_{zk} = Vertical dispersion coefficient for stability class k (m).

u_j = Midpoint value of wind speed class interval j, m/s.

x = Downwind distance, m.

n = Number of sectors, 16.

λ_i = Radioactive decay coefficient of radionuclide i, s^{-1}.

$2\pi x/n$ = Sector width at point of interest, m.

For determining the total body dose rate:

$$D_{TB} = \sum_i X_i \, DFB_i \tag{1.2}$$

where:

D_{TB} = Total body dose rate, mrem/y.

X_i = Air concentration of radionuclide i, μCi/m³.

DFB_i = Total body dose factor due to gamma radiation, mrem/y per μCi/m³ (Table 1.5).

For determining the skin dose rate:

$$D_s = \sum_i X_i \, (DFS_i + 1.11 \, DFY_i) \tag{1.3}$$

where:

D_s = Skin dose rate, mrem/y.

X_i = Air concentration of radionuclide i, $\mu Ci/m^3$.

DFS_i = Skin dose factor due to beta radiation, mrem/y per $\mu Ci/m^3$ (Table 1.5).

1.11 = The average ratio of tissue to air energy absorption coefficients, mrem/mrad.

DFY_i = Gamma-to-air dose factor for radionuclide i, mrad/y per $\mu Ci/m^3$ (Table 1.5).

For determining dose rate to a point in air:

$$D_a = 3.17E\text{-}02 \sum_i X_i \,(DFY_i \text{ or } DFB_i) \qquad (1.4)$$

where:

D_a = Air dose rate mrem/yr

DFB - Air dose factor for beta radiation (Table 1.5)

3.17E-02 = Conversion from release/yr (Ci) to $\mu Ci/m^3$ divided by seconds/yr.

1.2 Equations and assumptions for calculating doses from radioiodines and particulates are as follows:

1.2.1 <u>Assumptions</u>

1. Dose is to be calculated for the critical organ, thyroid, and the critical age groups, infant (milk) and child (green, leafy vegetables).
2. Exposure pathways from iodines and particulates are milk ingestion, ground contamination, green leafy vegetables from home gardens, and inhalation.
3. The radioiodine and particulate mix is based on the historically observed source term given in Table 1.1.

4. Basic radionuclide data are given in Table 1.2.
5. All releases are treated as ground-level.
6. Mean annual average X/Q's for the period January 1, 1978 - December 31, 1982 are given in Table 1.3.
7. Raw meteorological data for ground-level releases consist of wind speed and direction measurements at 10m and temperature measurements at 10m and 60m.
8. Dose is to be evaluated at the potential offsite exposure points where maximum doses to man are expected to exist.
9. Real cow, goat and garden locations are considered.
10. Potential maximum exposure points (Table 1.4) considered are the nearest cow, goat and home garden locations in each sector.
11. Terrain effects and open terrain recirculation factors are not considered.
12. Building wake effects on effluent dispersion are considered.
13. Plume depletion and radioactive decay are considered for air-concentration calculations.
14. Radioactive decay is considered for ground-concentration calculations.
15. Deposition is calculated based on the curves given in Figure 1.1.
16. Milk cows and goats obtain 100% of their food from pasture grass May through October of each year.
17. Credit is taken for shielding by residence (factor of 0.7).

1.2.2 Equations

To calculate the dose for any one of the potential maximum-exposure points, the following equations are used.

1.2.2.1 Inhalation

Equation for calculating air concentration, X, is the same as in the Noble Gas Section 1.1.2 (Equation 1.1).

For determining the organ dose rate:

$$D_I = 1x10^6 \sum_i X_i \ DFI_i \ \ BR \qquad (1.4)$$

where:

D_I = Organ dose rate due to inhalation, mrem/y.

X_i = Air concentration of radionuclide i, μCi/m³.

DFI_i = Inhalation dose factor, mrem/pCi (Table 1.7).

BR = Breathing rate 1400 m³/y, infant; 3700 m³y, child; or 8000 m³/y adult.

1×10^6 = pCi/μCi conversion factor.

1.2.2.2 Ground Contamination

For determining the ground concentration of any nuclide:

$$G_i = 3.15x10^7 \sum_{k=1}^{7} \frac{f_k \ Q_i \ DR}{} \qquad [1\text{-}\exp -(\lambda_i \ t_b)] \qquad (1.5)$$

where:

G_i = Ground concentration of radionuclide i, μCi/m³.

k = Stability class.

f_k = Joint relative frequency of occurrence of winds in stability class k blowing toward this exposure point, expressed as a fraction.

Q_i = Average release rate of radionuclide i, μCi/s.

DR = Relative deposition rate, m^{-1} (Figure 1.2 for DR/2πX).

x = Downwind distance, m.

n = Number of sectors, 16.

$2\pi x/n$ = Sector width at point of interest, m.

λ_i = Radioactive decay coefficient of radionuclide i, y^{-1}.

t_b = Time for buildup of radionuclides on the ground, 35 y.

$3.15x10^7$ = s/y conversion factor

For determining the total body or organ dose rate from ground contamination:

$$D_G = (8,760)(1x10^6) \sum_i G_i \, DFG_i \quad (1.6)$$

where:

D_G = Dose rate due to ground contamination, mrem/y.

G_i = Ground concentration of radionuclide i, µCi/m².

DFG_i = Dose factor for standing on contaminated ground, mrem/h per pCi/m² (Table 1.8).

8,760 = Occupation time, h/y.

$1x10^6$ = pCi/µCi conversion factor.

1.2.2.3 Milk Ingestion

For determining the concentration of any nuclide (except C-14 and H-3) in and on vegetation:

$$CV_i = 3,600 \sum_{k=1}^{7} \frac{f_k \, Q_i \, DR}{(2\pi x/n)} \left(\frac{r[1-\exp(-\lambda_{Ei} t_e)]}{Y_v \lambda_{Ei}} + \frac{B_{iv} [1-\exp(-\lambda_i t_b)]}{P \lambda_i} \right) \quad (1.7)$$

where:

CV_i = Concentration of radionuclide i in and on vegetation, µCi/kg.

k = Stability class.

f_k = Frequency of this stability class and wind direction combination, expressed as a fraction.

Q_i = Average release rate of radionuclide i, µCi/s.

DR = Relative deposition rate, m^{-2} (Figure 1.1).

x = Downwind distance, m.

n = Number of sectors, 16.

$2\pi x/n$ = Sector width at point of interest, m.

r = Fraction of deposited activity retained on vegetation (1.0 for iodines, 0.2 for particulates).

λ_{Ei} = Effective removal rate constant, $\lambda_{Ei} = \lambda_i + \lambda_w$, where λ_i is the radioactive decay coefficient, h^{-1}, and λ_w is a measure of physical loss by weathering (λ_w = .0021 h^{-1}).

t_e = Period over which deposition occurs, 720 h.

Y_v = Agricultural yield, 0.7 kg/m^2.

B_{iv} = Transfer factor from soil to vegetation of radionuclide i (Table 1.9).

λ_i = Radioactive decay coefficient of radionuclide i, h^{-1}.

t_b = Time for buildup of radionuclides on the ground, 3.07×10^5 h (35y).

P = Effective surface density of soil, 240 kg/m^2.

3,600 = s/h conversion factor.

For determining the concentration of C-14 in vegetation:

$$CV_{14} = 1\times10^3 \; X_{14} \; (0.11/0.16) \qquad (1.8)$$

where:

CV_{14} = Concentration of C-14 in vegetation, μCi/kg.

X_{14} = Air concentration of C-14, μCi/m^3.

0.11 = Fraction of total plant mass that is natural carbon.

0.16 = Concentration of natural carbon in the atmosphere, g/m^3.

1×10^3 = g/kg conversion factor.

For determining the concentration of H-3 in vegetation:

$$CV_T = 1\times10^3 \; X_T \; (0.75)(0.5/H) \qquad (1.9)$$

where:

CV_T = Concentration of H-3 in vegetation, µCi/kg.

X_T = Air concentration of H-3, Ci/m³.

0.75 = Fraction of total plant mass that is water.

0.5 = Ratio of tritium concentration in plant water to tritium concentration in atmospheric water.

H = Absolute humidity of the atmosphere, g/m³.

$1x10^3$ = g/kg conversion factor.

For determining the concentration of any nuclide in cow's or goat's milk:

$$CM_i = CV_i\ FM_i\ Q_f \exp(-\lambda_i\ t_f) \tag{1.10}$$

where:

CM_i = Concentration of radionuclide i (including C-14 and H-3) in milk, µCi/l.

CV_i = Concentration of radionuclide i in and on vegetation, µCi/kg.

FM_i = Transfer factor from feed to milk for radionuclide i, d/l (Table 1.6).

Q_f = Amount of feed consumed by the milk animal per day, kg/d.

λ_i = Radioactive decay coefficient of radionuclide i, d^{-1}.

t_f = Transport time of activity from feed to milk to receptor, 2 days.

For determining the organ dose rate from ingestion of green leafy vegetables and milk:

$$D = 1x10^6 \sum_i CM_i\ DF_i\ UM \tag{1.11}$$

where:

D = Organ dose rate due to ingestion, mrem/y.

CM_i = Concentration of radionuclide i in vegetables or milk, μCi/Kg (or liters).

DFG_i = Ingestion dose factor, mrem/pCi (Table 1.7).

UM = Ingestion rate for milk, 330 l/y; for vegetables 26 Kg/yr (child), no ingestion by infant.

$1x10^6$ = pCi/μCi conversion factor.

1.2.2.4 Organ Dose Rates

For determining the total thyroid dose rate from iodines and particulates:

$$D = D_I + D_G + D_M + D_V \tag{1.12}$$

where:

D = Total organ dose rate, mrem/y.

D_I = Dose rate due to inhalation, mrem/y.

D_G = Dose rate due to ground contamination, mrem/y.

D_M = Dose rate due to milk ingestion, mrem/y.

D_V = Dose rate due to vegetable ingestion, mrem/y.

1.2.3 The maximum organ dose rate, maximum total body dose rate, and maximum skin dose rate calculated in the previous section (Sec 1.2.2) are used to calculate design basis quantities as described in Section 1.3.

1.3 Design Basis Quantities

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

a. 15 millirem to any organ of an individual from iodines and particulates.

b. 15 millirem to skin of an individual from noble gas.

c. 5 millirem to the total body of an individual from noble gas.

Design basis quantity (Ci) is the smallest value for each nuclide, calculated by dividing the dose limits (a through e, above) by the appropriate dose calculated in step 1; the result then is multiplied by the amount of radionuclide (Ci) used to conservatively estimate the doses of Section D, as listed in Table 1.1 (or assumed a hypothetical 1 Ci/year for nuclides not actually present):

$$DBQ = \frac{D_{AI}}{D_c} (C_c) \tag{1.13}$$

where:

D_{AI} = Appendix I dose limit (mrem or mrad).

D_c = Calculated dose from step 1 (mrem or mrad).

C_c = Quantity of nuclide resulting in dose D_c (Ci).

DBQ = Design Basis Quantity (Ci).

The limiting values for Design Basis Quantities for radionuclides released to the atmosphere are given in Table 1.9.

The inverse of the ratio C_c/D_c in the above equation (i.e., D_c/C_c) is a useful value, since it represents the most limiting dose per unit quantity of each nuclide released. Use of the D_c/C_c ratio in quarterly evaluation of offsite dose is discussed in section D. Values of D_c/C_c are given in Table 1.9.

1.4 Land Use Census and DBQ Changes

Specifications 4.11.3 and 4.11.4 describe the requirements for an annual land use census and revision of the ODCM for use in the following calendar year. Areas of the ODCM which will be reviewed, and changed if appropriate, are Table 1.4 (Land Use Census data by Sector), and Table 1.9 (Gaseous Design Basis Objective Annual Quantities). Changes will be effective on January 1 of the year following the year of the survey

C. DESIGN OBJECTIVE QUANTITY (DBQ) LIMITS ON BATCH AND CONTINUOUS RELEASES

1.0 Batch Releases

Prior to each batch release (waste decay tank release or containment purge), the quantity of each nuclide identified is summed with the quantity of that nuclide released since the first of the current calendar year. The cumulative total for each nuclide then is divided by the design objective quantity for each nuclide (from Table 1.9), and the resultant fractions are summed in order to assure that the sum fraction of all nuclides does not exceed 1.0:

$$\frac{Ai}{(DBQ)_i} < 1.0 \qquad (1.14)$$

The amount in any calendar quarter should not exceed 0.5. This is checked by subtracting the value obtained at the end of the previous quarter from the value obtained from the cumulative total to date, including the batch to be released.

2.0 Continuous Releases

Low level continuous releases from the vent gas collection header and other low level sources are totaled on a weekly basis and summed with any batch releases for the week in order to establish the cumulative DBQ fraction from batch plus continuous releases for the year to date. Calculations are performed in the same manner as for batch releases described in C.1.0.

3.0 Exceeding DBQ Limits

As discussed under B.1.3, the DBQ is a very conservative estimate of activity which could give doses at Appendix I limits. Because different organs are summed together and doses to different people are summed, the DBQ typically overestimates dose by about a factor of five. Thus, if calculations of DBQ fraction exceed 1.0 for year-to-date or 0.5 for the quarter, technical specifications probably still would not be exceeded. However, further discretionary releases should be deferred until an accurate assessment of dose is made by use of GASPAR computer code or by analysis of appropriate release data via the segmented gaussian dose model used in emergency planning (inhalation dose, total body external dose, and boundary dose in air). See also Section D.1.2.

It should be noted that Palisades Plant to date (based on review of semiannual effluent data) has never exceeded the annual or quarterly DBQ fraction, despite its conservatism. Thus, it is not expected that an alternate to the DBQ method will be required unless the plant is in a significantly off-normal condition.

4.0 Releasing Radionuclides Not Listed in Table 1.9

Table 1.9 contains all nuclides identified to date as routine constituents of gaseous releases at Palisades Plant, plus those common to PWRs in general, even if not previously detected at Palisades. From time to time, however, other nuclides may be detected.

If the unlisted nuclide constitutes less than 10% of the MPC-fraction for the release, and all unlisted nuclides total less than 25% of the MPC-fraction, the nuclide may be considered not present.

If the unlisted nuclide constitutes greater than 10% of the MPC-fraction, or all unlisted nuclides together constitute greater than 25%, then each nuclide should be assigned a DBQ equal to the most conservative value listed for the physical form of the nuclide involved (noble gas, halogen or particulate).

Should a nuclide not listed in Table 1.9 begin to appear in significant quantities on a routine basis, revision to this ODCM should be made in order to include a design basis quantity specific to that nuclide.

D. OPTIONAL QUARTERLY DOSE CALCULATIONS

1.0 Methodology for Optional Quarterly Dose Calculations

This option may be used in place of, or in addition to, the design basis quantity (DBQ) fraction calculation described by Equation 1.4. This optional conservative calculation relates the DBQ fraction to the doses from which it was originally derived. Use of this method may assist in identification of the critical dose pathway or characteristics of the assumed critical individual (infant, child, adult), since Table 1.9 indicates these parameters.

1.1 Simplified Conservative Approach

This method utilizes a limiting dose concept such that the limiting dose for each nuclide is summed with the limiting dose for each other nuclide, regardless if such sum is physically possible. It also assumes critical pathways, such as milk and vegetables, are in effect even in winter when the pathway is absent.

As such, the method is highly conservative and significantly overestimates dose. If limits appear to be exceeded by this method, Section D.1.2 (a concise method, but requiring computer support) will be utilized.

1.1.1 Assumptions

1. All assumptions of Section 1.1 are utilized.
2. Limiting doses for each gaseous nuclide are summed, regardless of limiting decay mode (gamma or beta).
3. Limiting doses for each particulate and iodine nuclide are summed, regardless of dose point location, exposure pathway or organ affected.
4. Doses are summed for detected nuclides such that all nuclides which contribute greater than 10% individually or 25% in aggregate, to the MPC of released radioactivity, are included in the dose calculation.

1.1.2 Equations

For determining gaseous effluent dose:

$$D_G = \sum_{o}^{i} A_{iG}(D_c/C_c)_{iG} < 5 \text{ millirad/quarter, } 10 \text{ mrad/yr}$$

where:

D_G = Dose from gaseous effluents (mrad).

A_{iG} = Quantity of gaseous nuclide i released (Ci).

$(D_c/C_c)_{iG}$ = Dose per Ci factor for gaseous nuclide i (mrad/Ci).

The limit for this mixture is conservatively taken as that for gamma exposure (5 mrem/quarter, 10 mrem/year) although as indicated in Table 1.9, a majority of the gaseous effluents are beta-limiting and on an individual basis have the higher limit of 10 millirem/quarter and 20 millirem/year.

For determining iodine and particulate dose to organs:

$$D_{PI} = \sum_{0}^{i} A_{PIi}(D_c/C_c)_{PIi} \quad < 7.5 \text{ mrem/q, } 15 \text{ mrem/y}$$

where:

D_{PI} = Dose from particulates and iodines (mrem).

A_{PIi} = Quantity of particulate or iodine nuclide i released (Ci).

$(D_c/C_c)_{PIi}$ = Dose per Ci factor for particulate or iodine nuclide i (mrad/Ci).

1.2 Realistic Calculations

This methodology is to be used if the highly conservative calculations described in C.1.1, C.1.2 or D.1.0 yield values that appear to exceed applicable limits.

Doses for released particulates, iodines and noble gases will be determined by use of the NRC GASPAR computer code. The computer run will utilize the annual average joint frequency meteorological data based on not less than 3 years of meteorological measurement, and will reflect demographic and land use information from the land use survey generated in the most recent prior year. Where appropriate, seasonal adjustments will be applied to obtain realistic dose estimates since both recreational and agricultural activities can vary greatly in relation to season of the year.

An alternative to GASPAR for offsite dose calculation is the use of the Palisades Segmented Gaussian Plume Emergency offsite dose calculation program. This dose model allows evaluation of dose under the actual meteorological conditions present at the time of release. It is anticipated that the system may be used if major short-term releases such as containment purges are to be made under conditions which depart significantly from mean annual conditions.

E. GASEOUS RADWASTE TREATMENT SYSTEM OPERATION

The gaseous radwaste treatment system (GRTS) described below shall be maintained and operated to keep releases ALARA.

1. System Description

A flow diagram for the GRTS is given in Figure 1-1. The system consists of three waste-gas compressor packages, six gas decay tanks, and the associated piping, valves, and instrumentation. Gaseous wastes are received from the following: degassing of the reactor coolant and purging of the volume control tank prior to a cold shutdown, displacing of cover gases caused by liquid accumulation in the tanks connected to the vent header, and boron recycle process operation.

Design of the system precludes hydrogen explosion by means of ignition source elimination (diaphragm valves, low flow diaphragm compressors and system electrical grounding), and minimization of leakage outside the system. Explosive mixtures of hydrogen and oxygen have been demonstrated compatible with the system by operational experience over the past 13 years.

2. Determination of Satisfactory Operation

Design basis quantity fraction will be calculated for batch and continuous releases as described in Section C. These calculations will be used to ensure that the GRTS is operating as designed. Because the plant was designed to collect and hold for decay a vast majority of the high level gasses generated within the primary system, and because the 13-year operating history (to date of writing the initial ODCM) of the plant has demonstrated the system's consistent performance well below Appendix I limits, no additional operability requirements are specified.

F. RELEASE RATE FOR OFFSITE MPC (500 mRem/yr)

10 CFR 20.106 requires radioactive effluent releases to unrestricted areas be in concentrations less than the limits specified in Appendix B, Table II when averaged over a period not to exceed one year. (Note: There are no unrestricted areas anywhere within the site boundary as defined by Figure 2-1.) Concentrations at this level if present for one year will result in a dose of 500 mrem due to external exposure or inhalation depending on the nuclide(s) released. 10 CFR 50.36a requires that the release of radioactive materials be kept as low as reasonably achievable. However, the section further states that the licensee is permitted the flexibility of operation, to assure a dependable source of power, to release quantities of material higher than a small percentage of 10 CFR 20 limits but not exceeding those limits under unusual operating conditions. Appendix I to 10 CFR 50 provides the numerical guidelines on limiting conditions for operations to meet the as low as reasonably achievable requirement.

The GASPAR code has been run to determine the dose due to external radiation and inhalation. The source term used is listed in Table 1.1. The meteorology data is given in Table 1.3. Dose using annual average meteorology, to the most limiting organ of the person assumed to be residing at the site boundary with highest X/Q, is 2.15E-02 mrem (for one year). The release rate which would result in a dose rate equivalent to 500 mrem/year (using the more conservative total body limit) is the Curies/year given in Table 1.1 multiplied by 500/2.15E-02 or 1.11 Ci/sec.

G. PARTICULATE AND IODINE SAMPLING

Particulate and iodine samples are obtained from the continuous sample stream pulled from the plant stack. Samples typically are obtained to represent an integrated release from a gas batch (waste gas decay tank or containment purge, for example), or a series of samples are obtained to follow the course of a release. In any event, sample intervals are weekly, at minimum.

Because HEPA filters are present between most source inputs to the stack and the sample point, releases of particulates normally are significantly less than pre-release calculations indicate. This provides for conservatism in establishing setpoints and in estimation of pre-release design basis quantity fraction. However, for the sake of maintaining accurate release totals, monitor results (for gasses) and sample results (for particulates and iodines) are utilized rather than the pre-release estimates, for cumulative records.

Gamma analytical results for particulate and halogen filters are combined for determination of total activity of particulates and halogens released. Beta and alpha counting also is performed on the particulate filters. Beta yields of the gamma isotopes detected on particulate filters are applied to determine "identified" beta, and the "identified" count rate is subtracted from the observed count rate to give "unidentified" beta. The "unidentified" beta is assumed to be Sr-90 until results on actual Sr-90 (chemically separated from a quarterly composite of filters) are obtained. Similarly, alpha activity not identified as natural radium or thorium or their daughters is assumed as Pu-239 until results of detailed analyses are obtained from quarterly composites.

H. NOBLE GAS SAMPLING

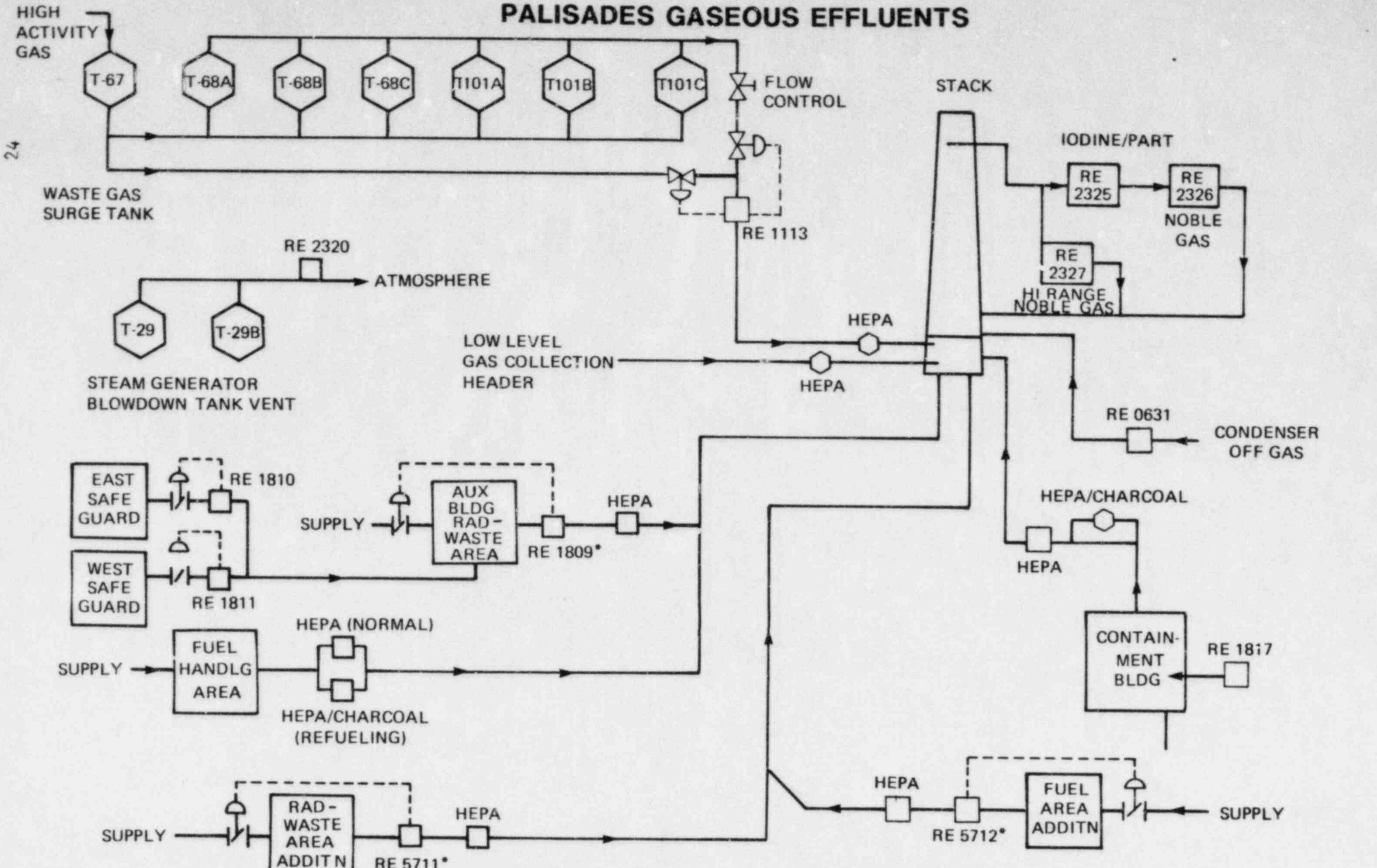
Noble gasses will be sampled from Waste Gas Decay Tanks prior to release and the Containment prior to purging. Analysis of these samples will be used for accountability of noble gasses. Off gas will be sampled at least weekly and used to calculate monthly noble gas releases. Nonroutine releases will be quantified from the stack noble gas monitor (RE 2326) which has a LLD of 1E-06 μCi/cc (if RE1815 is used because RE 2326 is out of service, the LLD will be 5E-05 μCi/cc).

I. TRITIUM SAMPLING

Tritium has a low dose consequence to the public because of low production rates. The major contributors to tritium effluents are evaporation from the fuel pool and reactor cavity (when flooded). Because of the low dose impact, gaseous tritium sampling will not be required. Tritium effluents will be estimated using conservative evaporation rate calculations from the fuel pool and reactor cavity.

FIGURE 1-1

PALISADES GASEOUS EFFLUENTS



*RE 1809, 5711 AND 5712 TRIP SUPPLY AND ONE OF TWO EXHAUSTERS. FLOW IS NOT TERMINATED

PALISADES EFFLUENT FLOW PATHS GASEOUS

TABLE 1.1

PALISADES GASEOUS AND LIQUID SOURCE TERMS, CURIES/YEAR (1)

Nuclide	Gaseous (2)	Liquid (2)
H-3	5.5	159
Kr-85	4.1	NA
Kr-85m	0.12	NA
Kr-87	8.4E-02	NA
Kr-88	2.1E-01	NA
Ar-41	3.1E-02	NA
Xe-131m	2.2	NA
Xe-133	1493	NA
Xe-133m	0.43	NA
Xe-135	1.11	NA
Xe-135m	0.3	NA
I-131	0.025	3.21E-03
I-132	2.91E-03	NA
I-133	6.5E-03	4.7E-05
I-134	4.8E-04	NA
I-135	1.84E-02	NA
Na-24	1.5E-06	NA
Cr-51	2.5E-04	3.9E-03
Mn-54	4.1E-04	7.8E-03
Co-57	2.1E-06	3.2E-05
Co-58	8.6E-04	2.9E-02
Fe-59	6.6E-06	4.1E-04
Co-60	1.1E-03	1.24E-02
Se-75	3.7E-06	NA
Nb-95	2.4E-05	4.53E-04
Zr-95	4.7E-06	1.79E-04
Mo-99	1.5E-07	NA
Ru-103	.3E-07	.1E-05
Sb-127	NA	3.5E-05
Cs-134	4.5E-05	0.7
Cs-136	NA	1.8E-06
Cs-137	2.6E-04	1.36E-02
Ba-140	2.8E-07	NA
La-140	7.5E-07	1.1E-04
Unidentified beta	3.9E-04	3.3E-03

(1) Data derived from taking the effluents released during July-December 1978 through January-June 1982 and dividing by 4.

(2) Nuclide values listed as NA have not been observed at detectable levels in these waste streams.

TABLE 1.2

BASIC RADIONUCLIDE DATA

	NUCLIDE	HALF-LIFE (days)	LAMBDA (1/s)	BETA (MEV/DIS)	GAMMA (MEV/DIS)
1	Tritium	4.49E 03	1.79E-09	5.68E-03	0.0
2	C-14	2.09E 06	3.84E-12	5.17E-02	0.0
3	N-13	6.94E-03	1.16E-03	4.91E-01	1.02E 00
4	O-19	3.36E-04	2.39E-02	1.02E 00	1.05E 00
5	F-18	7.62E-02	1.05E-04	2.41E-01	9.88E-01
6	NA-24	6.33E-01	1.27E-05	5.55E-01	4.12E 00
7	P-32	1.43E 01	5.61E-07	6.95E-01	0.0
8	AB-41	7.63E-02	1.05E-04	3.63E-01	1.28E 00
9	CR-51	2.78E 01	2.89E-07	3.75E-03	3.28E-02
10	MN-54	3.03E 02	2.65E-08	4.17E-03	8.36E-01
11	MN-56	1.07E-01	7.50E-05	7.93E-01	1.76E 00
12	FE-59	4.50E 01	1.78E-07	1.18E-01	1.19E 00
13	CO-58	7.13E 01	1.12E-07	2.05E-01	9.78E-01
14	CO-60	1.92E 03	4.18E-09	9.68E-02	2.50E 00
15	ZN-69m	5.75E-01	1.39E-05	0.0	4.15E 00
16	ZN-69	3.96E-02	2.03E-04	3.19E-01	0.0
17	BR-84	2.21E-02	3.63E-04	1.28E 00	1.68E 00
18	BR-85	2.08E-03	3.86E-03	1.04E 00	8.40E-01
19	KR-85m	1.83E-01	4.38E-05	2.53E-01	1.59E-01
20	KR-85	3.93E 03	2.04E-09	2.51E-01	2.21E-03
21	KR-87	5.28E-02	1.52E-04	1.32E 00	7.93E-01
22	KR-88	1.17E-01	6.86E-05	3.75E-01	1.96E 00
23	KR-89	2.21E-03	3.63E-03	1.23E 00	2.08E 00
24	RB-88	1.24E-02	6.47E-04	2.06E 00	6.86E-01
25	RB-89	1.07E-02	7.50E-04	0.00	2.40E 00
26	SR-89	5.20E 01	1.54E-07	5.73E-01	1.36E-04
27	SR-90	1.03E 04	7.79E-10	1.96E-01	0.0
28	SR-91	4.03E-01	1.99E-05	6.50E-01	6.95E-01
29	SR-92	1.13E-01	7.10E-05	1.95E-01	1.34E 00
30	SR-93	5.56E-03	1.44E-03	1.61E 00	6.28E-01
31	Y-90	2.67E 00	3.00E-06	9.36E-01	0.0
32	Y-91m	3.47E-02	2.31E-04	0.0	5.56E-01
33	Y-91	5.88E 01	1.36E-07	6.06E-01	3.61E-03
34	Y-92	1.47E-01	5.46E-05	1.44E 00	2.50E-01
35	Y-93	4.29E-01	1.87E-05	1.17E 00	8.94E-02
36	ZB-95	6.50E 01	1.23E-07	1.20E-01	7.35E-01
37	NB-95m	3.75E 00	2.14E-06	2.85E-01	5.87E-02
38	NB-95	3.50E 01	2.29E-07	4.50E-02	7.64E-01
39	MO-99	2.79E 00	2.87E-06	3.96E-01	1.62E-01
40	TC-99m	2.50E-01	3.21E-05	4.85E-03	1.43E-01
41	TC-99	7.74E 07	1.04E-13	8.38E-02	0.0
42	TC-104	1.25E-02	6.42E-04	0.0	0.0

TABLE 1.2 (CON'T)

BASIC RADIONUCLIDE DATA

	NUCLIDE	HALF-LIFE (days)	LAMBDA (1/s)	BETA (MEV/DIS)	GAMMA (MEV/DIS)
43	RU-106	3.67E 02	2.19E-08	1.01E-02	0.0
44	TE-132	3.24E 00	2.48E-06	1.00E-01	2.05E-01
45	I-129	6.21E 09	1.29E-15	4.02E-02	3.77E-03
46	I-131	8.05E 00	9.96E-07	1.94E-01	3.81E-01
47	MI-131	8.05E 00	9.96E-07	1.94E-01	3.81E-01
48	I-132	9.58E-02	8.37E-05	5.14E-01	2.33E 00
49	MI-132	9.58E-02	8.37E-05	5.14E-01	2.33E 00
50	I-133	8.75E-01	9.17E-06	4.08E-01	6.10E-01
51	MI-133	8.75E-01	9.17E-06	4.08E-01	6.10E-01
52	I-134	3.61E-02	2.22E-04	6.10E-01	2.59E 00
53	MI-134	3.61E-02	2.22E-04	6.10E-01	2.59E 00
54	I-135	2.79E-01	2.87E-05	3.68E-01	1.58E 00
55	MI-135	2.79E-01	2.87E-05	3.68E-01	1.58E 00
56	XE-131m	1.18E 01	6.80E-07	1.43E-01	2.01E-02
57	XE-133m	2.26E 00	3.55E-06	1.90E-01	4.16E-02
58	XE-133	5.27E 00	1.52E-06	1.35E-01	4.54E-02
59	XE-135m	1.08E-02	7.43E-04	9.50E-02	4.32E-01
60	XE-135	3.83E-01	2.09E-05	3.17E-01	2.47E-01
61	XE-137	2.71E-03	2.96E-03	1.64E 00	1.94E-01
62	XE-138	1.18E-02	6.80E-04	6.06E-01	1.18E 00
63	CS-134	7.48E 02	1.07E-08	1.57E-01	1.04E 00
64	CS-135	1.10E 09	7.29E-15	5.74E-02	0.0
65	CS-136	1.30E 01	6.17E-07	1.01E-01	2.20E 00
66	CS-137	1.10E 04	7.29E-10	2.52E-01	5.97E-01
67	CS-138	2.24E-02	3.58E-04	1.23E 00	2.30E 00
68	BA-139	5.76E-02	1.39E-04	6.54E-01	5.50E-02
69	BA-140	1.28E 01	6.27E-07	3.15E-01	1.95E-01
70	LA-140	1.68E 00	4.77E-06	5.40E-01	2.31E 00
71	CE-144	2.84E 02	2.82E-08	9.13E-02	3.24E-09
72	PB-143	1.36E 01	5.90E-07	3.14E-01	0.0
73	PR-144	1.20E-02	6.68E-09	1.21E 00	2.18E 00

Table 1.3

GROUND LEVEL RELEASE - TOP OF CONTAINMENT BUILDING
NO DECAY, UNDEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	3.009E-06	1.003E-06	5.514E-07	3.633E-07	2.013E-07	1.327E-07	9.619E-08	7.397E-08	5.927E-08	4.895E-08	4.136E-08
SSW	2.457E-06	8.083E-07	4.376E-07	2.890E-07	1.610E-07	1.068E-07	7.774E-08	6.002E-08	4.826E-08	3.996E-08	3.385E-08
SW	2.942E-06	9.490E-07	5.050E-07	3.325E-07	1.853E-07	1.236E-07	9.050E-08	7.019E-08	5.665E-08	4.707E-08	3.999E-08
WSW	2.830E-06	9.005E-07	4.735E-07	3.109E-07	1.731E-07	1.161E-07	8.548E-08	6.658E-08	5.393E-08	4.495E-08	3.830E-08
W	3.706E-06	1.175E-06	6.161E-07	4.037E-07	2.245E-07	1.508E-07	1.111E-07	8.663E-08	7.024E-08	5.859E-08	4.995E-08
WNW	4.623E-06	1.463E-06	7.674E-07	5.018E-07	2.784E-07	1.870E-07	1.378E-07	1.075E-07	8.717E-08	7.273E-08	6.202E-08
NW	7.102E-06	2.229E-06	1.165E-06	7.615E-07	4.230E-07	2.847E-07	2.102E-07	1.642E-07	1.333E-07	1.114E-07	9.505E-08
NNW	9.645E-06	2.971E-06	1.536E-06	1.001E-06	5.560E-07	3.765E-07	2.795E-07	2.193E-07	1.787E-07	1.497E-07	1.282E-07
N	8.473E-06	2.627E-06	1.374E-06	8.976E-07	4.989E-07	3.365E-07	2.490E-07	1.948E-07	1.584E-07	1.325E-07	1.132E-07
NNE	4.402E-06	1.386E-06	7.335E-07	4.808E-07	2.675E-07	1.796E-07	1.323E-07	1.032E-07	8.362E-08	6.975E-08	5.946E-08
NE	4.261E-06	1.398E-06	7.556E-07	4.958E-07	2.740E-07	1.806E-07	1.309E-07	1.007E-07	8.072E-08	6.669E-08	5.637E-08
ENE	3.472E-06	1.139E-06	6.120E-07	4.001E-07	2.203E-07	1.446E-07	1.045E-07	8.014E-08	6.409E-08	5.285E-08	4.460E-08
E	3.590E-06	1.200E-06	6.486E-07	4.228E-07	2.316E-07	1.515E-07	1.092E-07	8.356E-08	6.671E-08	5.491E-08	4.627E-08
ESE	3.375E-06	1.130E-06	6.075E-07	3.948E-07	2.155E-07	1.408E-07	1.013E-07	7.752E-08	6.185E-08	5.088E-08	4.286E-08
SE	4.250E-06	1.423E-06	7.669E-07	4.976E-07	2.710E-07	1.766E-07	1.268E-07	9.683E-08	7.713E-08	6.338E-08	5.332E-08
SSE	4.306E-06	1.433E-06	7.784E-07	5.078E-07	2.782E-07	1.820E-07	1.311E-07	1.004E-07	8.013E-08	6.596E-08	5.558E-08

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES

BEARING	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	3.559E-08	2.003E-08	1.336E-08	7.589E-09	5.099E-09	3.752E-09	2.924E-09	2.369E-09	1.976E-09	1.685E-09	1.461E-09
SSW	2.919E-08	1.656E-08	1.111E-08	6.364E-09	4.300E-09	3.178E-09	2.485E-09	2.020E-09	1.690E-09	1.444E-09	1.255E-09
SW	3.458E-08	1.983E-08	1.341E-08	7.765E-09	5.289E-09	3.934E-09	3.092E-09	2.525E-09	2.120E-09	1.817E-09	1.584E-09
WSW	3.320E-08	1.921E-08	1.308E-08	7.640E-09	5.236E-09	3.914E-09	3.089E-09	2.531E-09	2.131E-09	1.831E-09	1.600E-09
W	4.333E-08	2.514E-08	1.714E-08	1.004E-08	6.894E-09	5.160E-09	4.077E-09	3.343E-09	2.817E-09	2.423E-09	2.119E-09
WNW	5.381E-08	3.125E-08	2.133E-08	1.251E-08	8.596E-09	6.439E-09	5.091E-09	4.177E-09	3.521E-09	3.030E-09	2.650E-09
NW	8.254E-08	4.810E-08	3.289E-08	1.935E-08	1.333E-08	9.997E-09	7.914E-09	6.500E-09	5.485E-09	4.724E-09	4.134E-09
NNW	1.115E-07	6.560E-08	4.515E-08	2.679E-08	1.856E-08	1.399E-08	1.111E-08	9.153E-09	7.743E-09	6.683E-09	5.861E-09
N	9.837E-08	5.753E-08	3.944E-08	2.328E-08	1.607E-08	1.208E-08	9.574E-09	7.873E-09	6.649E-09	5.731E-09	5.020E-09
NNE	5.157E-08	2.993E-08	2.041E-08	1.195E-08	8.206E-09	6.141E-09	4.852E-09	3.979E-09	3.352E-09	2.883E-09	2.520E-09
NE	4.854E-08	2.740E-08	1.833E-08	1.046E-08	7.056E-09	5.211E-09	4.072E-09	3.309E-09	2.766E-09	2.363E-09	2.053E-09
ENE	3.835E-08	2.155E-08	1.437E-08	8.171E-09	5.504E-09	4.060E-09	3.170E-09	2.574E-09	2.150E-09	1.836E-09	1.595E-09
E	3.973E-08	2.220E-08	1.474E-08	8.330E-09	5.585E-09	4.104E-09	3.195E-09	2.588E-09	2.158E-09	1.839E-09	1.595E-09
ESE	3.679E-08	2.055E-08	1.364E-08	7.704E-09	5.165E-09	3.796E-09	2.955E-09	2.394E-09	1.996E-09	1.701E-09	1.426E-09
SE	4.573E-08	2.544E-08	1.684E-08	9.481E-09	6.347E-09	4.659E-09	3.624E-09	2.933E-09	2.444E-09	2.082E-09	1.805E-09
SSE	4.773E-08	2.668E-08	1.773E-08	1.002E-08	6.722E-09	4.941E-09	3.848E-09	3.117E-09	2.599E-09	2.215E-09	1.921E-09

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT

SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	5.681E-07	2.068E-07	9.705E-08	5.954E-08	4.147E-08	2.052E-08	7.765E-09	3.780E-09	2.378E-09	1.688E-
SSW	4.539E-07	1.653E-07	7.839E-08	4.846E-08	3.394E-08	1.695E-08	6.502E-09	3.200E-09	2.027E-09	1.447E-
SW	5.270E-07	1.906E-07	9.121E-08	5.687E-08	4.009E-08	2.026E-08	7.919E-09	3.959E-09	2.533E-09	1.821E-
WSW	4.961E-07	1.784E-07	8.609E-08	5.412E-08	3.838E-08	1.959E-08	7.780E-09	3.936E-09	2.538E-09	1.834E-
W	6.459E-07	2.315E-07	1.119E-07	7.048E-08	5.006E-08	2.563E-08	1.022E-08	5.189E-09	3.352E-09	2.427E-
WNW	8.038E-07	2.874E-07	1.388E-07	8.747E-08	6.215E-08	3.185E-08	1.273E-08	6.475E-09	4.188E-09	3.035E-
NW	1.222E-06	4.368E-07	2.117E-07	1.338E-07	9.525E-08	4.899E-08	1.968E-08	1.005E-08	6.517E-09	4.731E-
NNW	1.617E-06	5.751E-07	2.813E-07	1.793E-07	1.284E-07	6.672E-08	2.721E-08	1.406E-08	9.175E-09	6.693E-
N	1.441E-06	5.153E-07	2.507E-07	1.589E-07	1.134E-07	5.857E-08	2.367E-08	1.214E-08	7.892E-09	5.740E-
NNE	7.663E-07	2.758E-07	1.333E-07	8.392E-08	5.959E-08	3.051E-08	1.217E-08	6.176E-09	3.989E-09	2.888E-
NE	7.828E-07	2.818E-07	1.321E-07	8.108E-08	5.652E-08	2.807E-08	1.069E-08	5.247E-09	3.320E-09	2.368E-
ENE	6.349E-07	2.266E-07	1.054E-07	6.439E-08	4.473E-08	2.209E-08	8.364E-09	4.089E-09	2.583E-09	1.840E-
E	6.709E-07	2.385E-07	1.102E-07	6.703E-08	4.641E-08	2.278E-08	8.535E-09	4.136E-09	2.598E-09	1.843E-
ESE	6.290E-07	2.221E-07	1.023E-07	6.215E-08	4.299E-08	2.109E-08	7.895E-09	3.825E-09	2.403E-09	1.705E-
SE	7.929E-07	2.794E-07	1.281E-07	7.752E-08	5.349E-08	2.613E-08	9.724E-09	4.695E-09	2.944E-09	2.087E-
SSE	8.036E-07	2.865E-07	1.324E-07	8.051E-08	5.575E-08	2.738E-08	1.027E-08	4.978E-09	3.128E-09	2.220E-

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	58.10
DIAMETER	(METERS)	0.0
EXIT VELOCITY	(METERS)	0.0

REP. WIND HEIGHT	(METERS)	10.0
BUILDING HEIGHT	(METERS)	58.1
BLDG.MIN.CRS.SEC AREA	(SQ.METERS)	2000.0
HEAT EMISSION RATE	(CAL/SEC)	0.0

AT THE RELEASE HEIGHT:

VENT RELEASE MODE	WIND SPEED (METERS/SEC)
ELEVATED	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0
GROUND LEVEL	ABOVE 0.0

AT THE MEASURED WIND HEIGHT (10.0 METERS):

VENT RELEASE MODE	WIND SPEED (METERS/SEC) STABLE CONDITIONS	WIND SPEED (METERS/SEC) UNSTABLE/NEUTRAL CONDITI
ELEVATED	LESS THAN 0.0	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0	BETWEEN 0.0 AND 0.
GROUND LEVEL	ABOVE 0.0	ABOVE 0.0

Table 1.3 (Contd)

GROUND LEVEL RELEASE - TOP OF CONTAINMENT BUILDING
2.260 DAY DECAY, UNDEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) DISTANCE IN MILES

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	3.008E-06	1.002E-06	5.506E-07	3.626E-07	2.007E-07	1.322E-07	9.571E-08	7.353E-08	5.886E-08	4.856E-08	4.099E-08
SSW	2.455E-06	8.074E-07	4.369E-07	2.883E-07	1.604E-07	1.063E-07	7.731E-08	5.963E-08	4.788E-08	3.961E-08	3.352E-08
SW	2.940E-06	9.479E-07	5.042E-07	3.318E-07	1.847E-07	1.231E-07	9.001E-08	6.973E-08	5.622E-08	4.666E-08	3.960E-08
WSW	2.828E-06	8.995E-07	4.727E-07	3.101E-07	1.725E-07	1.156E-07	8.499E-08	6.612E-08	5.349E-08	4.454E-08	3.790E-08
W	3.704E-06	1.174E-06	6.151E-07	4.028E-07	2.237E-07	1.501E-07	1.105E-07	8.606E-08	6.969E-08	5.807E-08	4.946E-08
WNW	4.620E-06	1.461E-06	7.661E-07	5.007E-07	2.774E-07	1.861E-07	1.370E-07	1.068E-07	8.648E-08	7.208E-08	6.139E-08
NW	7.098E-06	2.226E-06	1.163E-06	7.598E-07	4.215E-07	2.834E-07	2.090E-07	1.631E-07	1.323E-07	1.103E-07	9.407E-08
NNW	9.640E-06	2.967E-06	1.533E-06	9.989E-07	5.541E-07	3.747E-07	2.779E-07	2.178E-07	1.773E-07	1.484E-07	1.268E-07
N	8.469E-06	2.624E-06	1.371E-06	8.957E-07	4.973E-07	3.351E-07	2.477E-07	1.936E-07	1.572E-07	1.313E-07	1.121E-07
NNE	4.400E-06	1.385E-06	7.323E-07	4.798E-07	2.667E-07	1.789E-07	1.316E-07	1.025E-07	8.302E-08	6.917E-08	5.891E-08
NE	4.259E-06	1.396E-06	7.546E-07	4.948E-07	2.732E-07	1.799E-07	1.303E-07	1.001E-07	8.017E-08	6.616E-08	5.567E-08
ENE	3.470E-06	1.137E-06	6.108E-07	3.991E-07	2.194E-07	1.438E-07	1.038E-07	7.949E-08	6.349E-08	5.228E-08	4.406E-08
E	3.588E-06	1.199E-06	6.476E-07	4.219E-07	2.308E-07	1.508E-07	1.086E-07	8.301E-08	6.620E-08	5.443E-08	4.581E-08
ESE	3.373E-06	1.129E-06	6.065E-07	3.940E-07	2.148E-07	1.402E-07	1.008E-07	7.700E-08	6.137E-08	5.043E-08	4.243E-08
SE	4.247E-06	1.421E-06	7.655E-07	4.965E-07	2.700E-07	1.757E-07	1.260E-07	9.613E-08	7.648E-08	6.277E-08	5.274E-08
SSE	4.303E-06	1.431E-06	7.772E-07	5.067E-07	2.773E-07	1.813E-07	1.304E-07	9.973E-08	7.953E-08	6.539E-08	5.504E-08

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) DISTANCE IN MILES

BEARING	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	3.524E-08	1.973E-08	1.310E-08	7.363E-09	4.897E-09	3.567E-09	2.751E-09	2.207E-09	1.822E-09	1.538E-09	1.320E-09
SSW	2.887E-08	1.629E-08	1.087E-08	6.153E-09	4.111E-09	3.004E-09	2.323E-09	1.868E-09	1.544E-09	1.305E-09	1.122E-09
SW	3.421E-08	1.951E-08	1.312E-08	7.514E-09	5.062E-09	3.724E-09	2.896E-09	2.339E-09	1.942E-09	1.647E-09	1.421E-09
WSW	3.282E-08	1.889E-08	1.278E-08	7.381E-09	5.001E-09	3.696E-09	2.884E-09	2.336E-09	1.944E-09	1.653E-09	1.428E-09
W	4.285E-08	2.473E-08	1.677E-08	9.711E-09	6.595E-09	4.882E-09	3.815E-09	3.094E-09	2.579E-09	2.194E-09	1.898E-09
WNW	5.320E-08	3.073E-08	2.085E-08	1.209E-08	8.220E-09	6.090E-09	4.763E-09	3.866E-09	3.225E-09	2.746E-09	2.376E-09
NW	8.159E-08	4.727E-08	3.214E-08	1.867E-08	1.273E-08	9.447E-09	7.397E-09	6.011E-09	5.018E-09	4.276E-09	3.703E-09
NNW	1.103E-07	6.448E-08	4.412E-08	2.579E-08	1.774E-08	1.322E-08	1.039E-08	8.473E-09	7.094E-09	6.061E-09	5.261E-09
N	9.734E-08	5.663E-08	3.862E-08	2.256E-08	1.541E-08	1.147E-08	8.998E-09	7.325E-09	6.125E-09	5.227E-09	4.532E-09
NNE	5.104E-08	2.947E-08	1.999E-08	1.159E-08	7.873E-09	5.831E-09	4.560E-09	3.701E-09	3.086E-09	2.627E-09	2.273E-09
NE	4.806E-08	2.700E-08	1.797E-08	1.015E-08	6.777E-09	4.954E-09	3.832E-09	3.082E-09	2.550E-09	2.157E-09	1.855E-09
ENE	3.783E-08	2.112E-08	1.399E-08	7.843E-09	5.214E-09	3.796E-09	2.926E-09	2.346E-09	1.936E-09	1.632E-09	1.401E-09
E	3.930E-08	2.184E-08	1.442E-08	8.058E-09	5.344E-09	3.884E-09	2.991E-09	2.396E-09	1.976E-09	1.666E-09	1.429E-09
ESE	3.638E-08	2.020E-08	1.334E-08	7.448E-09	4.938E-09	3.588E-09	2.763E-09	2.214E-09	1.826E-09	1.539E-09	1.320E-09
SE	4.518E-08	2.498E-08	1.644E-08	9.141E-09	6.046E-09	4.386E-09	3.372E-09	2.698E-09	2.223E-09	1.872E-09	1.604E-09
SSE	4.722E-08	2.626E-08	1.735E-08	9.701E-09	6.438E-09	4.682E-09	3.608E-09	2.892E-09	2.387E-09	2.014E-09	1.728E-09

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT

SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	5.673E-07	2.062E-07	9.657E-08	5.913E-08	4.110E-08	2.023E-08	7.541E-09	3.595E-09	2.216E-09	1.541E-(
SSW	4.532E-07	1.648E-07	7.797E-08	4.809E-08	3.360E-08	1.668E-08	6.293E-09	3.027E-09	1.875E-09	1.308E-(
SW	5.262E-07	1.900E-07	9.071E-08	5.644E-08	3.970E-08	1.994E-08	7.670E-09	3.750E-09	2.347E-09	1.651E-(
WSW	4.953E-07	1.778E-07	8.560E-08	5.369E-08	3.798E-08	1.927E-08	7.523E-09	3.719E-09	2.343E-09	1.656E-(
W	6.449E-07	2.308E-07	1.113E-07	6.994E-08	4.95[illegible]E-08	2.522E-08	9.894E-09	4.912E-09	3.104E-09	2.199E-(
WNW	8.025E-07	2.865E-07	1.380E-07	8.679E-08	6.15[illegible]E-08	3.133E-08	1.232E-08	6.127E-09	3.878E-09	2.751E-(
NW	1.220E-06	4.353E-07	2.105E-07	1.327E-07	9.427E-08	4.817E-08	1.903E-08	9.503E-09	6.029E-09	4.284E-(
NNW	1.614E-06	5.732E-07	2.797E-07	1.778E-07	1.271E-07	6.561E-08	2.632E-08	1.330E-08	8.496E-09	6.071E-(
N	1.438E-06	5.137E-07	2.494E-07	1.578E-07	1.123E-07	5.767E-08	2.295E-08	1.153E-08	7.346E-09	5.236E-(
NNE	7.652E-07	2.750E-07	1.326E-07	8.332E-08	5.904E-08	3.005E-08	1.180E-08	5.867E-09	3.712E-09	2.632E-(
NE	7.818E-07	2.810E-07	1.314E-07	8.053E-08	5.603E-08	2.766E-08	1.039E-08	4.991E-09	3.094E-09	2.162E-(
ENE	6.337E-07	2.258E-07	1.047E-07	6.379E-08	4.419E-08	2.166E-08	8.039E-09	3.826E-09	2.355E-09	1.636E-(
E	6.698E-07	2.377E-07	1.096E-07	6.652E-08	4.595E-08	2.242E-08	8.266E-09	3.916E-09	2.406E-09	1.670E-(
ESE	6.281E-07	2.214E-07	1.018E-07	6.167E-08	4.256E-08	2.075E-08	7.641E-09	3.618E-09	2.223E-09	1.543E-(
SE	7.916E-07	2.784E-07	1.273E-07	7.687E-08	5.291E-08	2.567E-08	9.387E-09	4.423E-09	2.710E-09	1.877E-(
SSE	8.024E-07	2.856E-07	1.317E-07	7.992E-08	5.521E-08	2.695E-08	9.950E-09	4.721E-09	2.904E-09	2.018E-(

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	58.10	REP. WIND HEIGHT	(METERS)	10.0
DIAMETER	(METERS)	0.0	BUILDING HEIGHT	(METERS)	58.1
EXIT VELOCITY	(METERS)	0.0	BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	2000.0
			HEAT EMISSION RATE	(CAL/SEC)	0.0

AT THE RELEASE HEIGHT:

VENT RELEASE MODE	WIND SPEED (METERS/SEC)
ELEVATED	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0
GROUND LEVEL	ABOVE 0.0

AT THE MEASURED WIND HEIGHT (10.0 METERS):

VENT RELEASE MODE	WIND SPEED (METERS/SEC) STABLE CONDITIONS	WIND SPEED (METERS/SEC) UNSTABLE/NEUTRAL CONDITI!
ELEVATED	LESS THAN 0.0	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0	BETWEEN 0.0 AND 0.(
GROUND LEVEL	ABOVE 0.0	ABOVE 0.0

Table 1.3 (Contd)

GROUND LEVEL RELEASE - TOP OF CONTAINMENT BUILDING
8.000 DAY DECAY, DEPLETED

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES

SECTOR	0.250	0.500	0.750	1.000	1.500	2.000	2.500	3.000	3.500	4.000	4.500
S	2.848E-06	9.154E-07	4.912E-07	3.178E-07	1.708E-07	1.098E-07	7.781E-08	5.866E-08	4.616E-08	3.749E-08	3.119E-08
SSW	2.325E-06	7.379E-07	3.898E-07	2.528E-07	1.366E-07	8.829E-08	6.288E-08	4.759E-08	3.757E-08	3.060E-08	2.552E-08
SW	2.784E-06	8.663E-07	4.498E-07	2.909E-07	1.572E-07	1.022E-07	7.320E-08	5.565E-08	4.411E-08	3.605E-08	3.015E-08
WSW	2.678E-06	8.221E-07	4.217E-07	2.719E-07	1.469E-07	9.604E-08	6.913E-08	5.278E-08	4.198E-08	3.442E-08	2.887E-08
W	3.507E-06	1.073E-06	5.488E-07	3.532E-07	1.905E-07	1.247E-07	8.986E-08	6.868E-08	5.469E-08	4.486E-08	3.766E-08
WNW	4.374E-06	1.335E-06	6.835E-07	4.390E-07	2.362E-07	1.546E-07	1.115E-07	8.521E-08	6.786E-08	5.569E-08	4.676E-08
NW	6.720E-06	2.035E-06	1.037E-06	6.662E-07	3.589E-07	2.354E-07	1.700E-07	1.302E-07	1.038E-07	8.526E-08	7.166E-08
NNW	9.127E-06	2.712E-06	1.368E-06	8.758E-07	4.718E-07	3.113E-07	2.260E-07	1.738E-07	1.391E-07	1.146E-07	9.661E-08
N	8.018E-06	2.398E-06	1.224E-06	7.852E-07	4.233E-07	2.783E-07	2.014E-07	1.545E-07	1.233E-07	1.014E-07	8.534E-08
NNE	4.165E-06	1.266E-06	6.533E-07	4.207E-07	2.270E-07	1.486E-07	1.070E-07	8.180E-08	6.512E-08	5.342E-08	4.484E-08
NE	4.032E-06	1.276E-06	6.731E-07	4.337E-07	2.325E-07	1.494E-07	1.059E-07	7.986E-08	6.287E-08	5.108E-08	4.251E-08
ENE	3.286E-06	1.039E-06	5.451E-07	3.500E-07	1.869E-07	1.196E-07	8.445E-08	6.351E-08	4.988E-08	4.044E-08	3.360E-08
E	3.397E-06	1.096E-06	5.777E-07	3.699E-07	1.965E-07	1.253E-07	8.828E-08	6.625E-08	5.194E-08	4.205E-08	3.489E-08
ESE	3.194E-06	1.031E-06	5.411E-07	3.454E-07	1.828E-07	1.164E-07	8.196E-08	6.146E-08	4.815E-08	3.896E-08	3.231E-08
SE	4.021E-06	1.299E-06	6.830E-07	4.353E-07	2.299E-07	1.460E-07	1.025E-07	7.675E-08	6.004E-08	4.852E-08	4.019E-08
SSE	4.074E-06	1.308E-06	6.934E-07	4.442E-07	2.361E-07	1.505E-07	1.061E-07	7.958E-08	6.239E-08	5.051E-08	4.191E-08

ANNUAL AVERAGE CHI/Q (SEC/METER CUBED) — DISTANCE IN MILES

BEARING	5.000	7.500	10.000	15.000	20.000	25.000	30.000	35.000	40.000	45.000	50.000
S	2.645E-08	1.406E-08	8.926E-09	4.679E-09	2.944E-09	2.047E-09	1.516E-09	1.172E-09	9.365E-10	7.665E-10	6.396E-10
SSW	2.169E-08	1.162E-08	7.420E-09	3.920E-09	2.479E-09	1.731E-09	1.286E-09	9.976E-10	7.987E-10	6.551E-10	5.476E-10
SW	2.570E-08	1.392E-08	8.955E-09	4.784E-09	3.051E-09	2.143E-09	1.601E-09	1.247E-09	1.003E-09	8.253E-10	6.921E-10
WSW	2.466E-08	1.348E-08	8.729E-09	4.705E-09	3.019E-09	2.131E-09	1.598E-09	1.249E-09	1.007E-09	8.306E-10	6.980E-10
W	3.220E-08	1.764E-08	1.144E-08	6.185E-09	3.976E-09	2.811E-09	2.110E-09	1.651E-09	1.332E-09	1.100E-09	9.253E-10
WNW	3.998E-08	2.193E-08	1.424E-08	7.702E-09	4.956E-09	3.506E-09	2.634E-09	2.062E-09	1.665E-09	1.375E-09	1.157E-09
NW	6.132E-08	3.374E-08	2.195E-08	1.191E-08	7.680E-09	5.442E-09	4.093E-09	3.208E-09	2.591E-09	2.142E-09	1.803E-09
NNW	8.288E-08	4.602E-08	3.013E-08	1.649E-08	1.070E-08	7.612E-09	5.746E-09	4.517E-09	3.658E-09	3.031E-09	2.556E-09
N	7.311E-08	4.038E-08	2.634E-08	1.435E-08	9.274E-09	6.585E-09	4.961E-09	3.894E-09	3.149E-09	2.607E-09	2.197E-09
NNE	3.833E-08	2.101E-08	1.363E-08	7.367E-09	4.737E-09	3.349E-09	2.514E-09	1.968E-09	1.588E-09	1.311E-09	1.103E-09
NE	3.608E-08	1.924E-08	1.224E-08	6.447E-09	4.074E-09	2.842E-09	2.111E-09	1.637E-09	1.311E-09	1.075E-09	8.987E-10
ENE	2.847E-08	1.511E-08	9.580E-09	5.021E-09	3.164E-09	2.202E-09	1.632E-09	1.264E-09	1.010E-09	8.274E-10	6.907E-10
E	2.952E-08	1.558E-08	9.843E-09	5.131E-09	3.221E-09	2.235E-09	1.654E-09	1.278E-09	1.020E-09	8.345E-10	6.960E-10
ESE	2.734E-08	1.442E-08	9.105E-09	4.745E-09	2.978E-09	2.067E-09	1.529E-09	1.182E-09	9.433E-10	7.718E-10	6.438E-10
SE	3.397E-08	1.784E-08	1.124E-08	5.834E-09	3.655E-09	2.533E-09	1.871E-09	1.445E-09	1.153E-09	9.422E-10	7.854E-10
SSE	3.547E-08	1.873E-08	1.184E-08	6.173E-09	3.877E-09	2.692E-09	1.992E-09	1.540E-09	1.229E-09	1.006E-09	8.391E-10

CHI/Q (SEC/METER CUBED) FOR EACH SEGMENT — SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	5.084E-07	1.764E-07	7.867E-08	4.643E-08	3.130E-08	1.453E-08	4.852E-09	2.074E-09	1.181E-09	7.699E-
SSW	4.063E-07	1.409E-07	6.354E-08	3.778E-08	2.561E-08	1.199E-08	4.057E-09	1.752E-09	1.004E-09	6.578E-
SW	4.717E-07	1.625E-07	7.392E-08	4.433E-08	3.025E-08	1.433E-08	4.940E-09	2.168E-09	1.255E-09	8.286E-
WSW	4.441E-07	1.521E-07	6.977E-08	4.219E-08	2.896E-08	1.385E-08	4.850E-09	2.154E-09	1.256E-09	8.337E-
W	5.783E-07	1.974E-07	9.068E-08	5.494E-08	3.777E-08	1.812E-08	6.372E-09	2.841E-09	1.661E-09	1.104E-
WNW	7.196E-07	2.450E-07	1.125E-07	6.818E-08	4.689E-08	2.252E-08	7.934E-09	3.544E-09	2.074E-09	1.380E-
NW	1.094E-06	3.723E-07	1.715E-07	1.043E-07	7.186E-08	3.463E-08	1.226E-08	5.499E-09	3.226E-09	2.150E-
NNW	1.448E-06	4.902E-07	2.279E-07	1.397E-07	9.686E-08	4.715E-08	1.695E-08	7.688E-09	4.541E-09	3.041E-
N	1.290E-06	4.393E-07	2.031E-07	1.239E-07	8.558E-08	4.141E-08	1.476E-08	6.652E-09	3.915E-09	2.616E-
NNE	6.860E-07	2.352E-07	1.080E-07	6.543E-08	4.497E-08	2.158E-08	7.590E-09	3.385E-09	1.979E-09	1.316E-
NE	7.007E-07	2.403E-07	1.071E-07	6.323E-08	4.267E-08	1.987E-08	6.681E-09	2.878E-09	1.648E-09	1.080E-
ENE	5.682E-07	1.932E-07	8.544E-08	5.018E-08	3.373E-08	1.562E-08	5.209E-09	2.231E-09	1.273E-09	8.309E-
E	6.005E-07	2.034E-07	8.934E-08	5.226E-08	3.502E-08	1.613E-08	5.329E-09	2.265E-09	1.287E-09	8.382E-
ESE	5.631E-07	1.895E-07	8.295E-08	4.845E-08	3.244E-08	1.493E-08	4.928E-09	2.094E-09	1.190E-09	7.752E-
SE	7.097E-07	2.383E-07	1.038E-07	6.043E-08	4.035E-08	1.849E-08	6.066E-09	2.568E-09	1.455E-09	9.464E-
SSE	7.193E-07	2.443E-07	1.073E-07	6.278E-08	4.207E-08	1.938E-08	6.411E-09	2.728E-09	1.551E-09	1.010E-

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	58.10	REP. WIND HEIGHT	(METERS)	10.0
DIAMETER	(METERS)	0.0	BUILDING HEIGHT	(METERS)	58.1
EXIT VELOCITY	(METERS)	0.0	BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	2000.0
			HEAT EMISSION RATE	(CAL/SEC)	0.0

AT THE RELEASE HEIGHT:

VENT RELEASE MODE	WIND SPEED (METERS/SEC)
ELEVATED	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0
GROUND LEVEL	ABOVE 0.0

AT THE MEASURED WIND HEIGHT (10.0 METERS):

VENT RELEASE MODE	WIND SPEED (METERS/SEC) STABLE CONDITIONS	WIND SPEED (METERS/SEC) UNSTABLE/NEUTRAL CONDITI
ELEVATED	LESS THAN 0.0	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0	BETWEEN 0.0 AND 0.
GROUND LEVEL	ABOVE 0.0	ABOVE 0.C

Table 1.3 (Contd)

GROUND LEVEL RELEASE - TOP OF CONTAINMENT BUILDING

RELATIVE DEPOSITION PER UNIT AREA (M**-2) AT FIXED POINTS BY DOWNWIND SECTORS

DIRECTION FROM SITE	0.25	0.50	0.75	1.00	1.50	2.00	2.50	3.00	3.50	4.00	4.50
S	2.696E-08	9.115E-09	4.680E-09	2.874E-09	1.433E-09	8.690E-10	5.875E-10	4.257E-10	3.237E-10	2.550E-10	2.065E-10
SSW	1.809E-08	6.119E-09	3.142E-09	1.929E-09	9.618E-10	5.833E-10	3.944E-10	2.858E-10	2.173E-10	1.712E-10	1.386E-10
SW	1.921E-08	6.496E-09	3.336E-09	2.048E-09	1.021E-09	6.193E-10	4.187E-10	3.034E-10	2.307E-10	1.818E-10	1.471E-10
WSW	1.621E-08	5.480E-09	2.814E-09	1.728E-09	8.614E-10	5.224E-10	3.532E-10	2.560E-10	1.946E-10	1.533E-10	1.241E-10
W	2.248E-08	7.601E-09	3.903E-09	2.396E-09	1.195E-09	7.246E-10	4.899E-10	3.550E-10	2.699E-10	2.127E-10	1.722E-10
WNW	3.084E-08	1.043E-08	5.355E-09	3.288E-09	1.639E-09	9.942E-10	6.722E-10	4.871E-10	3.704E-10	2.918E-10	2.362E-10
NW	4.866E-08	1.646E-08	8.449E-09	5.188E-09	2.586E-09	1.569E-09	1.061E-09	7.685E-10	5.844E-10	4.604E-10	3.727E-10
NNW	6.104E-08	2.064E-08	1.060E-08	6.507E-09	3.244E-09	1.968E-09	1.330E-09	9.640E-10	7.330E-10	5.775E-10	4.675E-10
N	5.189E-08	1.755E-08	9.009E-09	5.532E-09	2.758E-09	1.673E-09	1.131E-09	8.195E-10	6.231E-10	4.909E-10	3.974E-10
NNE	2.813E-08	9.512E-09	4.884E-09	2.999E-09	1.495E-09	9.067E-10	6.130E-10	4.442E-10	3.378E-10	2.661E-10	2.154E-10
NE	4.173E-08	1.411E-08	7.245E-09	4.449E-09	2.218E-09	1.345E-09	9.095E-10	6.591E-10	5.011E-10	3.948E-10	3.196E-10
ENE	3.500E-08	1.184E-08	6.078E-09	3.732E-09	1.861E-09	1.128E-09	7.629E-10	5.528E-10	4.204E-10	3.312E-10	2.681E-10
E	3.886E-08	1.314E-08	6.748E-09	4.143E-09	2.066E-09	1.253E-09	8.470E-10	6.138E-10	4.667E-10	3.677E-10	2.977E-10
ESE	3.857E-08	1.304E-08	6.697E-09	4.112E-09	2.050E-09	1.243E-09	8.407E-10	6.092E-10	4.632E-10	3.650E-10	2.955E-10
SE	5.156E-08	1.743E-08	8.951E-09	5.497E-09	2.740E-09	1.662E-09	1.124E-09	8.143E-10	6.191E-10	4.878E-10	3.949E-10
SSE	4.978E-08	1.683E-08	8.643E-09	5.307E-09	2.646E-09	1.605E-09	1.085E-09	7.862E-10	5.978E-10	4.710E-10	3.813E-10

DIRECTION FROM SITE	5.00	7.50	10.00	15.00	20.00	25.00	30.00	35.00	40.00	45.00	50.00
S	1.708E-10	8.370E-11	5.251E-11	2.654E-11	1.607E-11	1.077E-11	7.718E-12	5.796E-12	4.506E-12	3.600E-12	2.938E-12
SSW	1.146E-10	5.618E-11	3.525E-11	1.782E-11	1.078E-11	7.231E-12	5.181E-12	3.890E-12	[illegible].025E-12	2.416E-12	1.972E-12
SW	1.217E-10	5.965E-11	3.743E-11	1.892E-11	1.145E-11	7.677E-12	5.501E-12	4.130E-12	3.212E-12	2.565E-12	2.094E-12
WSW	1.027E-10	5.032E-11	3.157E-11	1.596E-11	9.659E-12	6.476E-12	4.640E-12	3.485E-12	2.709E-12	2.164E-12	1.766E-12
W	1.424E-10	6.979E-11	4.379E-11	2.213E-11	1.340E-11	8.982E-12	6.436E-12	4.833E-12	3.758E-12	3.002E-12	2.450E-12
WNW	1.954E-10	9.576E-11	6.009E-11	3.037E-11	1.838E-11	1.232E-11	8.831E-12	6.631E-12	5.156E-12	4.119E-12	3.362E-12
NW	3.083E-10	1.511E-10	9.480E-11	4.792E-11	2.900E-11	1.944E-11	1.393E-11	1.046E-11	8.135E-12	6.498E-12	5.304E-12
NNW	3.867E-10	1.895E-10	1.189E-10	6.010E-11	3.638E-11	2.439E-11	1.748E-11	1.312E-11	1.020E-11	8.151E-12	6.653E-12
N	3.287E-10	1.611E-10	1.011E-10	5.109E-11	3.092E-11	2.073E-11	1.486E-11	1.116E-11	8.674E-12	6.929E-12	5.655E-12
NNE	1.782E-10	8.733E-11	5.480E-11	2.770E-11	1.676E-11	1.124E-11	8.054E-12	6.048E-12	4.702E-12	3.756E-12	3.066E-12
NE	2.644E-10	1.296E-10	8.130E-11	4.109E-11	2.487E-11	1.667E-11	1.195E-11	8.972E-12	6.976E-12	5.572E-12	4.548E-12
ENE	2.218E-10	1.087E-10	6.819E-11	3.447E-11	2.086E-11	1.399E-11	1.002E-11	7.526E-12	5.852E-12	4.674E-12	3.815E-12
E	2.462E-10	1.207E-10	7.571E-11	3.827E-11	2.316E-11	1.553E-11	1.113E-11	8.356E-12	6.497E-12	5.190E-12	4.236E-12
ESE	2.444E-10	1.198E-10	7.515E-11	3.798E-11	2.299E-11	1.541E-11	1.104E-11	8.293E-12	6.448E-12	5.151E-12	4.204E-12
SE	3.266E-10	1.601E-10	1.004E-10	5.077E-11	3.073E-11	2.060E-11	1.476E-11	1.108E-11	8.619E-12	6.885E-12	5.619E-12
SSE	3.154E-10	1.546E-10	9.698E-11	4.902E-11	2.967E-11	1.989E-11	1.425E-11	1.070E-11	8.321E-12	6.647E-12	5.426E-12

RELATIVE DEPOSITION PER UNIT AREA (M**-2) BY DOWNWIND SECTORS

SEGMENT BOUNDARIES IN MILES

DIRECTION FROM SITE	.5-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
S	4.863E-09	1.502E-09	5.979E-10	3.267E-10	2.076E-10	8.919E-11	2.766E-11	1.096E-11	5.854E-12	3.623E-1
SSW	3.264E-09	1.009E-09	4.013E-10	2.193E-10	1.394E-10	5.987E-11	1.857E-11	7.358E-12	3.929E-12	2.432E-1
SW	3.466E-09	1.071E-09	4.261E-10	2.328E-10	1.480E-10	6.357E-11	1.971E-11	7.812E-12	4.172E-12	2.582E-1
WSW	2.924E-09	9.033E-10	3.594E-10	1.964E-10	1.248E-10	5.362E-11	1.663E-11	6.591E-12	3.519E-12	2.178E-1
W	4.055E-09	1.253E-09	4.985E-10	2.724E-10	1.731E-10	7.437E-11	2.306E-11	9.141E-12	4.881E-12	3.021E-1
WNW	5.564E-09	1.719E-09	6.841E-10	3.738E-10	2.376E-10	1.021E-10	3.165E-11	1.254E-11	6.698E-12	4.146E-1
NW	8.779E-09	2.712E-09	1.079E-09	5.898E-10	3.748E-10	1.610E-10	4.993E-11	1.979E-11	1.057E-11	6.541E-1
NNW	1.101E-08	3.402E-09	1.354E-09	7.398E-10	4.702E-10	2.020E-10	6.263E-11	2.482E-11	1.325E-11	8.204E-1
N	9.361E-09	2.892E-09	1.151E-09	6.289E-10	3.997E-10	1.717E-10	5.324E-11	2.110E-11	1.127E-11	6.974E-1
NNE	5.074E-09	1.568E-09	6.238E-10	3.409E-10	2.167E-10	9.307E-11	2.886E-11	1.144E-11	6.108E-12	3.781E-1
NE	7.528E-09	2.326E-09	9.255E-10	5.058E-10	3.214E-10	1.381E-10	4.282E-11	1.697E-11	9.062E-12	5.609E-1
ENE	6.315E-09	1.951E-09	7.764E-10	4.242E-10	2.696E-10	1.158E-10	3.592E-11	1.423E-11	7.602E-12	4.705E-1
E	7.011E-09	2.166E-09	8.619E-10	4.710E-10	2.994E-10	1.286E-10	3.987E-11	1.580E-11	8.440E-12	5.224E-1
ESE	6.959E-09	2.150E-09	8.555E-10	4.675E-10	2.971E-10	1.276E-10	3.958E-11	1.569E-11	8.377E-12	5.185E-1
SE	9.301E-09	2.874E-09	1.143E-09	6.248E-10	3.971E-10	1.706E-10	5.290E-11	2.097E-11	1.120E-11	6.930E-1
SSE	8.980E-09	2.774E-09	1.104E-09	6.033E-10	3.834E-10	1.647E-10	5.107E-11	2.024E-11	1.081E-11	6.691E-1

VENT AND BUILDING PARAMETERS:

RELEASE HEIGHT	(METERS)	58.10	REP. WIND HEIGHT	(METERS)	10.0
DIAMETER	(METERS)	0.0	BUILDING HEIGHT	(METERS)	58.1
EXIT VELOCITY	(METERS)	0.0	BLDG.MIN.CRS.SEC.AREA	(SQ.METERS)	2000.0
			HEAT EMISSION RATE	(CAL/SEC)	0.0

AT THE RELEASE HEIGHT:

VENT RELEASE MODE	WIND SPEED (METERS/SEC)
ELEVATED	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0
GROUND LEVEL	ABOVE 0.0

AT THE MEASURED WIND HEIGHT (10.0 METERS):

VENT RELEASE MODE	WIND SPEED (METERS/SEC) STABLE CONDITIONS	WIND SPEED (METERS/SEC) UNSTABLE/NEUTRAL CONDITI
ELEVATED	LESS THAN 0.0	LESS THAN 0.0
MIXED	BETWEEN 0.0 AND 0.0	BETWEEN 0.0 AND 0.
GROUND LEVEL	ABOVE 0.0	ABOVE 0.0

TABLE 1.4

PALISADES, 1983 SURVEY

Distance to the nearest residence, garden, beef cow, dairy cow and goat in each sector.

SECTOR	RESIDENCE	GARDEN	BEEF COW	DAIRY COW	GOAT
N	>5 mi	>5 mi	>5 mi	>5 mi	>5 mi
NNE	1.0 mi	1.5 mi	>5 mi	>5 mi	>5 mi
NE	1.5 mi	1.5 mi	3.9 mi	3.0 mi	>5 mi
ENE	1.3 mi	1.6 mi	4 mi	4.0 mi	>5 mi
E	1.0 mi	2.0 mi	3.5 mi	2.5 mi	>5 mi
ESE	.9 mi	.9 mi	3.1 mi	>5 mi	3.0 mi
SE	.9 mi	1.2 mi	>5 mi	>5 mi	>5 mi
SSE	.75 mi	1.5 mi	>5 mi	>5 mi	>5 mi
S	.5 mi	1.5 mi	>5 mi	>5 mi	>5 mi
SSW	.75 mi	1.5 mi	>5 mi	>5 mi	>5 mi

TABLE 1.5

DOSE FACTORS FOR SUBMERSION IN NOBLE GASES

	DFB[1]	DFY[2]	DFS[1]	DFB[2]
Kr-85m	1.17(+3)[3]	1.21(+3)	1.46(+3)	3.86(+3)
Kr-85	1.61(+1)	1.69(+1)	1.34(+3)	3.83(+3)
Kr-87	5.92(+3)	6.05(+3)	9.73(+3)	2.01(+4)
Kr-88	1.47(+4)	1.50(+4)	2.37(+3)	5.72(+3)
Kr-89	1.66(+4)	1.59(+4)	1.01(+4)	1.88(+4)
Xe-131m	9.15(+1)	1.53(+2)	4.76(+2)	2.18(+3)
Xe-133m	2.51(+2)	3.17(+2)	9.94(+2)	2.90(+3)
Xe-133	2.94(+2)	3.46(+2)	3.06(+2)	2.06(+3)
Xe-135m	3.12(+3)	3.30(+3)	7.11(+2)	1.45(+3)
Xe-135	1.81(+3)	1.88(+3)	1.86(+3)	4.84(+3)
Xe-137	1.42(+3)	1.48(+3)	1.22(+4)	2.50(+4)
Xe-138	8.83(+3)	9.00(+3)	4.13(+3)	9.25(+3)
Ar-41	8.84(+3)	9.76(+3)	2.69(+3)	5.54(+3)

1. mrem/y per $\mu Ci/m^3$
2. mrad/y per $\mu Ci/m^3$
3. 1.17(+3) = 1.17×10^3

TABLE 1.6

STABLE ELEMENT TRANSFER DATA

ELEMENT	F_m - MILK (COW)	F_m - MILK (GOAT)	F_f - MEAT
H	1.0E-02	1.7E-01	1.2E-02
C	1.2E-02	1.0E-01	3.1E-02
Na	4.0E-02	4.0E-02	3.0E-02
P	2.5E-02	2.5E-01	4.6E-02
Cr	2.2E-03	2.2E-03	2.4E-03
Mn	2.5E-04	2.5E-04	8.0E-04
Fe	1.2E-03	1.3E-04	4.0E-02
Co	1.0E-03	1.0E-03	1.3E-02
Ni	6.7E-03	6.7E-03	5.3E-02
Cu	1.4E-02	1.3E-02	8.0E-03
Zn	3.9E-02	3.9E-02	3.0E-02
Rb	3.0E-02	3.0E-02	3.1E-02
Sr	8.0E-04	1.4E-02	6.0E-04
Y	1.0E-05	1.0E-05	4.6E-03
Zr	5.0E-06	5.0E-06	3.4E-02
Nb	2.5E-03	2.5E-03	2.8E-01
Mo	7.5E-03	7.5E-03	8.0E-03
Tc	2.5E-02	2.5E-02	4.0E-01
Ru	1.0E-06	1.0E-06	4.0E-01
Rh	1.0E-02	1.0E-02	1.5E-03
Ag	5.0E-02	5.0E-02	1.7E-02
Te	1.0E-03	1.0E-03	7.7E-02
I	6.0E-03	6.0E-02	2.9E-03
Cs	1.2E-02	3.0E-02	4.0E-03
Ba	4.0E-04	4.0E-04	3.2E-03
La	5.0E-06	5.0E-06	2.0E-04
Ce	1.0E-04	1.0E-04	1.2E-03
Pr	5.0E-06	5.0E-06	4.7E-03
Nd	5.0E-06	5.0E-06	3.3E-03
W	5.0E-04	5.0E-04	1.3E-03
Np	5.0E-06	5.0E-06	2.0E-04

TABLE 1.7

INHALATION DOSE FACTORS FOR INFANT
(MREM PER PCI INHALED)
Page 1 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	4.62E-07	4.62E-07	4.62E-07	4.62E-07	4.62E-07	4.62E-07
C 14	1.89E-05	3.79E-06	3.79E-06	3.79E-06	3.79E-06	3.79E-06	3.79E-06
NA 24	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06
P 32	1.45E-03	8.03E-05	5.53E-05	NO DATA	NO DATA	NO DATA	1.15E-05
CR 51	NO DATA	NO DATA	6.39E-08	4.11E-08	9.45E-09	9.17E-06	2.55E-07
MN 54	NO DATA	1.81E-05	3.56E-06	NO DATA	3.56E-06	7.14E-04	5.04E-06
MN 56	NO DATA	1.10E-09	1.58E-10	NO DATA	7.86E-10	8.95E-06	5.12E-05
FE 55	1.41E-05	8.39E-06	2.38E-06	NO DATA	NO DATA	6.21E-05	7.82E-07
FE 59	9.69E-06	1.68E-05	6.77E-06	NO DATA	NO DATA	7.25E-04	1.77E-05
CO 58	NO DATA	8.71E-07	1.30E-06	NO DATA	NO DATA	5.55E-04	7.95E-06
CO 60	NO DATA	5.73E-06	8.41E-06	NO DATA	NO DATA	3.22E-03	2.28E-05
NI 63	2.42E-04	1.46E-05	8.29E-06	NO DATA	NO DATA	1.49E-04	1.73E-06
NI 65	1.71E-09	2.03E-10	8.79E-11	NO DATA	NO DATA	5.80E-06	3.58E-05
CU 64	NO DATA	1.34E-09	5.53E-10	NO DATA	2.84E-09	6.64E-06	1.07E-05
ZN 65	1.38E-05	4.47E-05	2.22E-05	NO DATA	2.32E-05	4.62E-04	3.67E-05
ZN 69	3.85E-11	6.91E-11	5.13E-12	NO DATA	2.87E-11	1.05E-06	9.44E-06
BR 83	NO DATA	NO DATA	2.72E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	2.86E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	1.46E-08	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	1.36E-04	6.30E-05	NO DATA	NO DATA	NO DATA	2.17E-06
RB 88	NO DATA	3.98E-07	2.05E-07	NO DATA	NO DATA	NO DATA	2.42E-07
RB 89	NO DATA	2.29E-07	1.47E-07	NO DATA	NO DATA	NO DATA	4.87E-08
SR 89	2.84E-04	NO DATA	8.15E-06	NO DATA	NO DATA	1.45E-03	4.57E-05
SR 90	2.92E-02	NO DATA	1.85E-03	NO DATA	NO DATA	8.03E-03	9.36E-05
SR 91	6.83E-08	NO DATA	2.47E-09	NO DATA	NO DATA	3.76E-05	5.24E-05
SR 92	7.50E-09	NO DATA	2.79E-10	NO DATA	NO DATA	1.70E-05	1.00E-04
Y 90	2.35E-06	NO DATA	6.30E-08	NO DATA	NO DATA	1.92E-04	7.43E-05
Y 91m	2.91E-10	NO DATA	9.90E-12	NO DATA	NO DATA	1.99E-06	1.68E-06
Y 91	4.20E-04	NO DATA	1.12E-05	NO DATA	NO DATA	1.75E-03	5.02E-05
Y 92	1.17E-08	NO DATA	3.29E-10	NO DATA	NO DATA	1.75E-05	9.04E-05

TABLE 1.7 (CONT'D)

INHALATION DOSE FACTORS FOR INFANT
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T. BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.07E-07	NO DATA	2.91E-09	NO DATA	NO DATA	5.46E-05	1.19E-04
ZR 95	8.24E-05	1.99E-05	1.45E-05	NO DATA	2.22E-05	1.25E-03	1.55E-05
ZR 97	1.07E-07	1.83E-08	8.36E-09	NO DATA	1.85E-08	7.88E-05	1.00E-04
NB 95	1.12E-05	4.59E-06	2.70E-06	NO DATA	3.37E-06	3.42E-04	9.05E-06
MO 99	NO DATA	1.18E-07	2.31E-08	NO DATA	1.89E-07	9.63E-05	3.48E-05
TC 99m	9.98E-13	2.06E-12	2.66E-11	NO DATA	2.22E-11	5.79E-07	1.45E-06
TC101	4.65E-14	5.88E-14	5.80E-13	NO DATA	6.99E-13	4.17E-07	6.03E-07
RU103	1.44E-06	NO DATA	4.85E-07	NO DATA	3.03E-06	3.94E-04	1.15E-05
RU105	8.74E-10	NO DATA	2.93E-10	NO DATA	6.42E-10	1.12E-05	3.46E-05
RU106	6.20E-05	NO DATA	7.77E-06	NO DATA	7.61E-05	8.26E-03	1.17E-04
AG110m	7.13E-06	5.16E-06	3.57E-06	NO DATA	7.80E-06	2.62E-03	2.36E-05
TE125m	3.40E-06	1.42E-06	4.70E-07	1.16E-06	NO DATA	3.19E-04	9.22E-06
TE127m	1.19E-05	4.93E-06	1.48E-06	3.48E-06	2.68E-05	9.37E-04	1.95E-05
TE127	1.59E-09	6.81E-10	3.49E-10	1.32E-09	3.47E-09	7.39E-06	1.74E-05
TE129m	1.01E-05	4.35E-06	1.59E-06	3.91E-06	2.27E-05	1.[illegible]0E-03	4.93E-05
TE129	5.63E-11	2.48E-11	1.34E-11	4.82E-11	1.25E-10	2.14E-06	1.88E-05
TE131m	7.62E-08	3.93E-08	2.59E-08	6.38E-08	1.89E-07	1.42E-04	8.51E-05
TE131	1.24E-11	5.87E-12	3.57E-12	1.13E-11	2.85E-11	1.47E-06	5.87E-06
TE132	2.66E-07	1.69E-07	1.26E-07	1.99E-07	7.39E-07	2.43E-04	3.15E-05
I 130	4.54E-06	9.91E-06	3.98E-06	1.1[illegible]E-03	1.09E-05	NO DATA	1.42E-06
I 131	2.71E-05	3.17E-05	1.40E-05	[illegible]E-02	3.70E-05	NO DATA	7.56E-07
I 132	1.21E-06	2.53E-06	8.99E-07	1.21E-04	2.82E-06	NO DATA	1.36E-06
I 133	9.46E-06	1.37E-05	4.00E-06	2.54E-03	1.60E-05	NO DATA	1.54E-06
I 134	6.58E-07	1.34E-06	4.75E-07	3.18E-05	1.49E-06	NO DATA	9.21E-07
I 135	2.76E-06	5.43E-06	1.98E-06	4.97E-04	6.05E-06	NO DATA	1.31E-06
CS134	2.8[illegible]E-04	5.02E-04	5.32E-05	NO DATA	1.36E-04	5.69E-05	9.53E-07
CS136	3.45E-05	9.61E-05	3.78E-05	NO DATA	4.03E-05	8.40E-06	1.02E-06
CS137	3.92E-04	4.37E-04	3.25E-05	NO DATA	1.23E-04	5.09E-05	9.53E-07
CS138	3.61E-07	5.58E-07	2.84E-07	NO DATA	2.93E-07	4.67E-08	6.26E-07
BA139	1.06E-09	7.03E-13	3.07E-11	NO DATA	4.23E-13	4.25E-06	3.64E-05

TABLE 1.7 (CONT'D)

INHALATION DOSE FACTORS FOR INFANT
(MREM PER PCI INHALED)
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NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	4.00E-05	4.00E-08	2.07E-06	NO DATA	9.59E-09	1.14E-03	2.74E-05
BA141	1.12E-10	7.70E-14	3.55E-12	NO DATA	4.64E-14	2.12E-06	3.39E-06
BA142	2.84E-11	2.36E-14	1.40E-12	NO DATA	1.36E-14	1.11E-06	4.95E-07
LA140	3.61E-07	1.43E-07	3.68E-08	NO DATA	NO DATA	1.20E-04	6.06E-05
LA142	7.36E-10	2.69E-10	6.46E-11	NO DATA	NO DATA	5.87E-06	4.25E-05
CE141	1.98E-05	1.19E-05	1.42E-06	NO DATA	3.75E-06	3.69E-04	1.54E-05
CE143	2.09E-07	1.38E-07	1.58E-08	NO DATA	4.03E-08	8.30E-05	3.55E-05
CE144	2.28E-03	8.65E-04	1.26E-04	NO DATA	3.84E-04	7.03E-03	1.06E-04
PR143	1.00E-05	3.74E-06	4.99E-07	NO DATA	1.41E-06	3.09E-04	2.66E-05
PR144	3.42E-11	1.32E-11	1.72E-12	NO DATA	4.80E-12	1.15E-06	3.06E-06
ND147	5.67E-06	5.81E-06	3.57E-07	NO DATA	2.25E-06	2.30E-04	2.23E-05
W 187	9.26E-09	6.44E-09	2.23E-09	NO DATA	NO DATA	2.83E-05	2.54E-05
NP239	2.65E-07	2.37E-08	1.34E-08	NO DATA	4.73E-08	4.25E-05	1.78E-05

TABLE 1.7 (CON'T)

INHALATION DOSE FACTORS FOR CHILD
(MREM PER PCI INHALED)
Page 1 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	3.04E-07	3.04E-07	3.04E-07	3.04E-07	3.04E-07	3.04E-07
C 14	9.70E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06
NA 24	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06
P 32	7.04E-04	3.09E-05	2.67E-05	NO DATA	NO DATA	NO DATA	1.14E-05
CR 51	NO DATA	NO DATA	4.17E-08	2.31E-08	6.57E-09	4.59E-06	2.93E-07
MN 54	NO DATA	1.16E-05	2.57E-06	NO DATA	2.71E-06	4.26E-04	6.19E-06
MN 56	NO DATA	4.48E-10	8.43E-11	NO DATA	4.52E-10	3.55E-06	3.33E-05
FE 55	1.28E-05	6.80E-06	2.10E-06	NO DATA	NO DATA	3.00E-05	7.75E-07
FE 59	5.59E-06	9.04E-06	4.51E-06	NO DATA	NO DATA	3.43E-04	1.91E-05
CO 58	NO DATA	4.79E-07	8.55E-07	NO DATA	NO DATA	2.99E-04	9.29E-06
CO 60	NO DATA	3.55E-06	6.12E-06	NO DATA	NO DATA	1.91E-03	2.60E-05
NI 63	2.22E-04	1.25E-05	7.56E-06	NO DATA	NO DATA	7.43E-05	1.71E-06
NI 65	8.08E-10	7.99E-11	4.44E-11	NO DATA	NO DATA	2.21E-06	2.27E-05
CU 64	NO DATA	5.39E-10	2.90E-10	NO DATA	1.63E-09	2.59E-06	9.92E-06
ZN 65	1.15E-05	3.06E-05	1.90E-05	NO DATA	1.93E-05	2.69E-04	4.41E-06
ZN 69	1.81E-11	2.61E-11	2.41E-12	NO DATA	1.58E-11	3.84E-07	2.75E-06
BR 83	NO DATA	NO DATA	1.28E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 84	NO DATA	NO DATA	1.48E-07	NO DATA	NO DATA	NO DATA	LT E-24
BR 85	NO DATA	NO DATA	6.84E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	5.36E-05	3.09E-05	NO DATA	NO DATA	NO DATA	2.16E-06
RB 88	NO DATA	1.52E-07	9.90E-08	NO DATA	NO DATA	NO DATA	4.66E-09
RB 89	NO DATA	9.33E-08	7.85E-08	NO DATA	NO DATA	NO DATA	5.11E-10
SR 89	1.62E-04	NO DATA	4.66E-06	NO DATA	NO DATA	5.83E-04	4.52E-05
SR 90	2.73E-02	NO DATA	1.74E-03	NO DATA	NO DATA	3.99E-03	9.28E-05
SR 91	3.28E-08	NO DATA	1.24E-09	NO DATA	NO DATA	1.44E-05	4.70E-05
SR 92	3.54E-09	NO DATA	1.42E-10	NO DATA	NO DATA	6.49E-06	6.55E-05
Y 90	1.11E-06	NO DATA	2.99E-08	NO DATA	NO DATA	7.07E-05	7.24E-05
Y 91m	1.37E-10	NO DATA	4.98E-12	NO DATA	NO DATA	7.60E-07	4.64E-07
Y 91	2.47E-04	NO DATA	6.59E-06	NO DATA	NO DATA	7.10E-04	4.97E-05
Y 92	5.50E-09	NO DATA	1.57E-10	NO DATA	NO DATA	6.46E-06	6.46E-05

TABLE 1.7 (CONT'D)

INHALATION DOSE FACTORS FOR CHILD
(MREM PER PCI INHALED)

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	5.04E-08	NO DATA	1.38E-09	NO DATA	NO DATA	2.01E-05	1.05E-04
ZR 95	5.13E-05	1.13E-05	1.00E-05	NO DATA	1.61E-05	6.03E-04	1.65E-05
ZR 97	5.07E-08	7.34E-09	4.32E-09	NO DATA	1.05E-08	3.06E-05	9.49E-05
NB 95	6.35E-06	2.48E-06	1.77E-06	NO DATA	2.33E-06	1.66E-04	1.00E-05
MO 99	NO DATA	4.66E-08	1.15E-08	NO DATA	1.06E-07	3.66E-05	3.42E-05
TC 99m	4.81E-13	9.41E-13	1.56E-11	NO DATA	1.37E-11	2.57E-07	1.30E-06
TC101	2.19E-14	2.30E-14	[illegible]	NO DATA	3.92E-13	1.58E-07	4.41E-09
RU103	7.55E-07	NO DATA	2.90E-07	NO DATA	1.90E-06	1.79E-04	1.21E-05
RU105	4.13E-10	NO DATA	1.50E-10	NO DATA	3.63E-10	4.30E-06	2.69E-05
RU106	3.68E-05	NO DATA	4.57E-06	NO DATA	4.97E-05	3.87E-03	1.16E-04
AG110m	4.56E-06	3.08E-06	2.47E-06	NO DATA	5.74E-06	1.48E-03	2.71E-05
TE125m	1.82E-06	6.29E-07	2.47E-07	5.20E-07	NO DATA	1.29E-04	9.13E-06
TE127m	6.72E-06	2.31E-06	8.16E-07	1.64E-06	1.72E-05	4.00E-04	1.93E-05
TE127	7.49E-10	2.57E-10	1.65E-10	5.30E-10	1.91E-09	2.71E-06	1.52E-05
TE129m	5.19E-06	1.85E-06	8.22E-07	1.71E-06	1.36E-05	4.76E-04	4.91E-05
TE129	2.64E-11	9.45E-12	6.44E-12	1.93E-11	6.94E-11	7.93E-07	6.89E-06
TE131m	3.63E-08	1.60E-08	1.37E-08	2.64E-08	1.08E-07	5.56E-05	8.32E-05
TE131	5.87E-12	2.28E-12	1.78E-12	4.59E-12	1.59E-11	5.55E-07	3.60E-07
TE132	1.30E-07	7.36E-08	7.12E-08	8.53E-08	4.79E-07	1.02E-04	3.72E-05
I 130	2.21E-06	4.43E-06	2.28E-06	4.99E-04	6.61E-06	NO DATA	1.38E-06
I 131	1.30E-05	1.30E-05	7.37E-06	4.39E-03	2.13E-05	NO DATA	7.68E-07
I 132	5.72E-07	1.10E-06	5.07E-07	5.23E-05	1.69E-06	NO DATA	8.65E-07
I 133	4.48E-06	7.49E-06	2.08E-06	1.04E-03	9.13E-06	NO DATA	1.48E-06
I 134	3.17E-07	5.84E-07	2.69E-07	1.37E-05	8.92E-07	NO DATA	2.58E-07
I 135	1.33E-06	2.36E-06	1.12E-06	2.14E-04	3.62E-06	NO DATA	1.20E-06
CS134	1.76E-04	2.74E-04	6.07E-05	NO DATA	8.93E-05	3.27E-05	1.04E-06
CS136	1.76E-05	4.62E-05	3.14E-05	NO DATA	2.58E-05	3.93E-06	1.13E-06
CS137	2.45E-04	2.23E-04	3.47E-05	NO DATA	7.63E-05	2.81E-05	9.78E-07
CS138	1.71E-07	2.27E-07	1.50E-07	NO DATA	1.68E-07	1.84E-08	7.29E-08
BA139	4.98E-10	2.66E-13	1.45E-11	NO DATA	2.33E-13	1.56E-06	1.56E-05

TABLE 1.7 (CONT'D)

INHALATION DOSE FACTORS FOR CHILD
(MREM PER PCI INHALED)
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NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	2.00E-05	1.75E-08	1.17E-06	NO DATA	5.71E-09	4.71E-04	2.75E-05
BA141	5.29E-11	2.95E-14	1.72E-12	NO DATA	2.56E-14	7.89E-07	7.44E-08
BA142	1.35E-11	9.73E-15	7.54E-13	NO DATA	7.87E-15	4.44E-07	7.41E-10
LA140	1.74E-07	6.08E-08	2.04E-08	NO DATA	NO DATA	4.94E-05	6.10E-05
LA142	3.50E-10	1.11E-10	3.49E-11	NO DATA	NO DATA	2.35E-06	2.05E-05
CE141	1.06E-05	5.28E-06	7.83E-07	NO DATA	2.31E-06	1.47E-04	1.53E-05
CE143	9.89E-08	5.37E-08	7.77E-09	NO DATA	2.26E-08	3.12E-05	3.44E-05
CE144	1.83E-03	5.72E-04	9.77E-05	NO DATA	3.17E-04	3.23E-03	1.05E-04
PR143	4.99E-06	1.50E-06	2.47E-07	NO DATA	8.11E-07	1.17E-04	2.63E-05
PR144	1.61E-11	4.99E-12	8.10E-13	NO DATA	2.64E-12	4.23E-07	5.32E-08
ND147	2.92E-06	2.36E-06	1.84E-07	NO DATA	1.30E-06	8.87E-05	2.22E-05
W 187	4.41E-09	2.61E-09	1.17E-09	NO DATA	NO DATA	1.11E-05	2.46E-05
NP239	1.26E-07	9.04E-09	6.35E-09	NO DATA	2.63E-08	1.57E-05	1.73E-05

TABLE 1.7 (CON'T)

INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)
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NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.58E-07	1.58E-07	1.58E-07	1.58E-07	1.58E-07	1.58E-07
C 14	2.27E-06	4.26E-07	4.26E-07	4.26E-07	4.26E-07	4.26E-07	4.26E-07
NA 24	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06
P 32	1.65E-04	9.64E-06	6.26E-06	NO DATA	NO DATA	NO DATA	1.08E-05
CR 51	NO DATA	NO DATA	1.25E-08	7.44E-09	2.85E-09	1.80E-06	4.15E-07
MN 54	NO DATA	4.95E-06	7.87E-07	NO DATA	1.23E-06	1.75E-04	9.67E-06
MN 56	NO DATA	1.55E-10	2.29E-11	NO DATA	1.63E-10	1.18E-06	2.53E-06
FE 55	3.07E-06	2.12E-06	4.93E-07	NO DATA	NO DATA	9.01E-06	7.54E-07
FE 59	1.47E-06	3.47E-06	1.32E-06	NO DATA	NO DATA	1.27E-04	2.35E-05
CO 58	NO DATA	1.98E-07	2.59E-07	NO DATA	NO DATA	1.16E-04	1.33E-05
CO 60	NO DATA	1.44E-06	1.85E-06	NO DATA	NO DATA	7.46E-04	3.56E-05
NI 63	5.40E-05	3.93E-06	1.81E-06	NO DATA	NO DATA	2.23E-05	1.67E-06
NI 65	1.92E-10	2.62E-11	1.14E-11	NO DATA	NO DATA	7.00E-07	1.54E-06
CU 64	NO DATA	1.83E-10	7.69E-11	NO DATA	5.78E-10	8.48E-07	6.12E-06
ZN 65	4.05E-06	1.29E-05	5.82E-06	NO DATA	8.62E-06	1.08E-04	6.68E-06
ZN 69	4.23E-12	8.14E-12	5.65E-13	NO DATA	5.27E-12	1.15E-07	2.04E-09
BR 83	NO DATA	NO DATA	3.01E-08	NO DATA	NO DATA	NO DATA	2.90E-08
BR 84	NO DATA	NO DATA	3.91E-08	NO DATA	NO DATA	NO DATA	2.05E-13
BR 85	NO DATA	NO DATA	1.60E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	1.69E-05	7.37E-06	NO DATA	NO DATA	NO DATA	2.08E-06
RB 88	NO DATA	4.84E-08	2.41E-08	NO DATA	NO DATA	NO DATA	4.18E-19
RB 89	NO DATA	3.20E-08	2.12E-08	NO DATA	NO DATA	NO DATA	1.16E-21
SR 89	3.80E-05	NO DATA	1.09E-06	NO DATA	NO DATA	1.75E-04	4.37E-05
SR 90	1.24E-02	NO DATA	7.62E-04	NO DATA	NO DATA	1.20E-03	9.02E-05
SR 91	7.74E-09	NO DATA	3.13E-10	NO DATA	NO DATA	4.56E-06	2.39E-05
SR 92	8.43E-10	NO DATA	3.64E-11	NO DATA	NO DATA	2.06E-06	5.38E-06
Y 90	2.61E-07	NO DATA	7.01E-09	NO DATA	NO DATA	2.12E-05	6.32E-05
Y 91m	3.26E-11	NO DATA	1.27E-12	NO DATA	NO DATA	2.40E-07	1.66E-10
Y 91	5.78E-05	NO DATA	1.55E-06	NO DATA	NO DATA	2.13E-04	4.81E-05
Y 92	1.29E-09	NO DATA	3.77E-11	NO DATA	NO DATA	1.96E-06	9.19E-06

TABLE 1.7 (CONT'D)

INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)
Page 2 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	1.18E-08	NO DATA	3.26E-10	NO DATA	NO DATA	6.06E-06	5.27E-05
ZR 95	1.34E-05	4.30E-06	2.91E-06	NO DATA	6.77E-06	2.21E-04	1.88E-05
ZR 97	1.21E-08	2.45E-09	1.13E-09	NO DATA	3.71E-09	9.84E-06	6.54E-05
NB 95	1.76E-06	9.77E-07	5.26E-07	NO DATA	9.67E-07	6.31E-05	1.30E-05
MO 99	NO DATA	1.51E-08	2.87E-09	NO DATA	3.64E-08	1.14E-05	3.10E-05
TC 99m	1.29E-13	3.64E-13	4.63E-12	NO DATA	5.52E-12	9.55E-08	5.20E-07
TC101	5.22E-15	7.52E-15	7.38E-14	NO DATA	1.35E-13	4.99E-08	1.36E-21
RU103	1.91E-07	NO DATA	8.23E-08	NO DATA	7.29E-07	6.31E-05	1.38E-05
RU105	9.88E-11	NO DATA	3.89E-11	NO DATA	1.27E-10	1.37E-06	6.02E-06
RU106	8.64E-06	NO DATA	1.09E-06	NO DATA	1.67E-05	1.17E-03	1.14E-04
AG110m	1.35E-06	1.25E-06	7.43E-07	NO DATA	2.46E-06	5.79E-04	3.78E-05
TE125m	4.27E-07	1.98E-07	5.84E-08	1.31E-07	1.55E-06	3.92E-05	8.83E-06
TE127m	1.58E-06	7.21E-07	1.96E-07	4.11E-07	5.72E-06	1.20E-04	1.87E-05
TE127	1.75E-10	8.03E-11	3.87E-11	1.32E-10	6.37E-10	8.14E-07	7.17E-06
TE129m	1.22E-06	5.84E-07	1.98E-07	4.30E-07	4.57E-06	1.45E-04	4.79E-05
TE129	6.22E-12	2.99E-12	1.55E-12	4.87E-12	2.34E-11	2.42E-07	1.96E-08
TE131m	8.74E-09	5.45E-09	3.63E-09	6.88E-09	3.86E-08	1.82E-05	6.95E-05
TE131	1.39E-12	7.44E-13	4.49E-13	1.17E-12	5.46E-12	1.74E-07	2.30E-09
TE132	3.25E-08	2.69E-08	2.02E-08	2.37E-08	1.82E-07	3.60E-05	6.37E-05
I 130	5.72E-07	1.68E-06	6.60E-07	1.42E-04	2.61E-06	NO DATA	9.61E-07
I 131	3.15E-06	4.47E-06	2.56E-06	1.49E-03	7.66E-06	NO DATA	7.85E-07
I 132	1.45E-07	4.07E-07	1.45E-07	1.43E-05	6.48E-07	NO DATA	5.08E-08
I 133	1.08E-06	1.85E-06	5.65E-07	2.69E-04	3.23E-06	NO DATA	1.11E-06
I 134	8.05E-08	2.16E-07	7.69E-08	3.73E-06	3.44E-07	NO DATA	1.26E-10
I 135	3.35E-07	8.73E-07	3.21E-07	5.60E-05	1.39E-06	NO DATA	6.56E-07
CS134	4.66E-05	1.06E-04	9.10E-05	NO DATA	3.59E-05	1.22E-05	1.30E-06
CS136	4.88E-06	1.83E-05	1.38E-05	NO DATA	1.07E-05	1.50E-06	1.46E-06
CS137	5.98E-05	7.76E-05	5.35E-05	NO DATA	2.78E-05	9.40E-06	1.05E-06
CS138	4.14E-08	7.76E-08	4.05E-08	NO DATA	6.00E-08	6.07E-09	2.33E-13
BA139	1.17E-10	8.32E-14	3.42E-12	NO DATA	7.78E-14	4.70E-07	1.12E-07

TABLE 1.7 (CONT'D)

INHALATION DOSE FACTORS FOR ADULTS
(MREM PER PCI INHALED)
Page 3 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	4.88E-06	6.13E-09	3.21E-07	NO DATA	2.09E-09	1.59E-04	2.73E-05
BA141	1.25E-11	9.41E-15	4.20E-13	NO DATA	8.75E-15	2.42E-07	1.45E-17
BA142	3.29E-12	3.38E-15	2.07E-13	NO DATA	2.86E-15	1.49E-07	1.96E-26
LA140	4.30E-08	2.17E-08	5.73E-09	NO DATA	NO DATA	1.70E-05	5.73E-05
LA142	8.54E-11	3.88E-11	9.65E-12	NO DATA	NO DATA	7.91E-07	2.64E-07
CE141	2.49E-06	1.69E-06	1.91E-07	NO DATA	7.83E-07	4.52E-05	1.50E-05
CE143	2.33E-08	1.72E-08	1.91E-09	NO DATA	7.60E-09	9.97E-06	2.83E-05
CE144	4.29E-04	1.79E-04	2.30E-05	NO DATA	1.06E-04	9.72E-04	1.02E-04
PR143	1.17E-06	4.69E-07	5.80E-08	NO DATA	2.70E-07	3.51E-05	2.50E-05
PR144	3.76E-12	1.56E-12	1.91E-13	NO DATA	8.81E-13	1.27E-07	2.69E-18
ND147	6.59E-07	7.62E-07	4.56E-08	NO DATA	4.45E-07	2.76E-05	2.16E-05
W 187	1.06E-09	8.85E-10	3.10E-10	NO DATA	NO DATA	3.63E-06	1.94E-05
NP239	2.87E-08	2.82E-09	1.55E-09	NO DATA	8.75E-09	4.70E-06	1.49E-05

TABLE 1.8

EXTERNAL DOSE FACTORS FOR STANDING ON CONTAMINATED GROUND

(mrem/hr per pci/m^2)

ELEMENT	TOTAL BODY	SKIN
H-3	0.0	0.0
C-14	0.0	0.0
Na-24	2.50E-08	2.90E-08
P-32	0.0	0.0
Cr-51	2.20E-10	2.60E-10
Mn-54	5.80E-09	6.80E-09
Mn-56	1.10E-08	1.30E-08
Fe-55	0.0	0.0
Fe-59	8.00E-09	9.40E-09
Co-58	7.00E-09	8.20E-09
Co-60	1.70E-08	2.00E-08
Ni-63	0.0	0.0
Ni-65	3.70E-09	4.30E-09
Cu-64	1.50E-09	1.70E-09
Zn-65	4.00E-09	4.60E-09
Zn-69	0.0	0.0
Br-83	6.40E-11	9.30E-11
Br-84	1.20E-08	1.40E-08
Br-85	0.0	0.0
Rb-86	6.30E-10	7.20E-10
Rb-88	3.50E-09	4.00E-09
Rb-89	1.50E-08	1.80E-08
Sr-89	5.60E-13	6.50E-13
Sr-91	7.10E-09	8.30E-09
Sr-92	9.00E-09	1.00E-08
Y-90	2.20E-12	2.60E-12
Y-91m	3.80E-09	4.40E-09
Y-91	2.40E-11	2.70E-11
Y-92	1.60E-09	1.90E-09
Y-93	5.70E-10	7.80E-10
Zr-95	5.00E-09	5.80E-09
Zr-97	5.50E-09	6.40E-09
Nb-95	5.10E-09	6.00E-09
Mo-99	1.90E-09	2.20E-09
Tc-99m	9.60E-10	1.10E-09
Tc-101	2.70E-09	3.00E-09
Ru-103	3.60E-09	4.20E-09
Ru-105	4.50E-09	5.10E-09
Ru-106	1.50E-09	1.80E-09
Ag-110m	1.80E-08	2.10E-08
Te-125m	3.50E-11	4.80E-11
Te-127m	1.10E-12	1.30E-12
Te-127	1.00E-11	1.10E-11

TABLE 1.8 (CON'T)

ELEMENT	TOTAL BODY	SKIN
Te-129m	7.70E-10	9.00E-10
Te-129	7.10E-10	8.40E-10
Te-131m	8.40E-09	9.90E-09
Te-131	2.20E-09	2.60E-06
Te-132	1.70E-09	2.00E-09
I-130	1.40E-08	1.70E-08
I-131	2.80E-09	3.40E-09
I-132	1.70E-08	2.00E-08
I-133	3.70E-09	4.50E-09
I-134	1.60E-08	1.90E-08
I-135	1.20E-08	1.40E-08
Cs-134	1.20E-08	1.40E-08
Cs-136	1.50E-08	1.70E-08
Cs-137	4.20E-09	4.90E-09
Cs-138	2.10E-08	2.40E-08
Ba-139	2.40E-09	2.70E-09
Ba-140	2.10E-09	2.40E-09
Ba-141	4.30E-09	4.90E-09
Ba-142	7.90E-09	9.00E-09
La-140	1.50E-08	1.70E-08
La-142	1.50E-08	1.80E-08
Ce-141	5.50E-10	6.20E-10
Ce-143	2.20E-09	2.50E-09
Ce-144	3.20E-10	3.70E-10
Pr-143	0.0	0.0
Pr-144	2.00E-10	2.30E-10
Nd-147	1.00E-09	1.20E-09
W-187	3.10E-09	3.60E-09
Np-239	9.50E-10	1.10E-09

TABLE 1.9

PALISADES PLANT
1984 GASEOUS DESIGN OBJECTIVE ANNUAL QUANTITIES

Nuclide	Pathway-Site - Age*	Organ	Dc/Cc Dose Factor mrem/Ci	Design Objective Annual Quantity (Ci)
H-3	V-1-C	Total Body	1.23E-04	4.07E+04
C-14	V-1-C	Bone	1.07E-01	1.40E+02
Ar-41	P-1-γ	Total Body	1.96E-04	2.55E+04
Cr-51	V-1-A	GI Tract	2.40E-03	6.25E+03
Mn-54	V-1-T	GI Tract	2.48E-01	6.05E+01
Fe-55	V-1-C	Bone	2.11E-01	7.11E+01
Mn-56	V-1-C	GI Tract	2.51E-07	5.98E+07
Co-57	V-1-T	GI Tract	8.69E-02	1.73E+02
Co-58	V-1-C	Total Body	5.16E-02	9.69E+01
Fe-59	V-1-T	GI Tract	2.46E-01	6.10E+01
Co-60	V-1-C	Total Body	3.07E-01	1.63E+01
Ni-65	V-1-C	GI Tract	1.12E-07	1.34E+08
Zn-65	V-1-C	Total Body	4.67E-01	1.07E+01
Kr-83m	P-1-β	Skin	4.75E-07	3.16E+07
Kr-85	P-1-β	Skin	4.29E-05	3.50E+05
Kr-85m	P-1-γ	Total Body	2.60E-05	1.92E+05
Kr-87	P-1-β	Skin	4.60E-04	3.26E+04
Kr-88	P-1-γ	Total Body	3.26E-04	1.53E+04
Rb-88	V-1-C	Total Body	2.25E-32	2.22E+32
Kr-89	P-1-γ	Total Body	3.68E-04	1.36E+04
Sr-89	V-1-C	Bone	9.28E+00	1.62E+00
Sr-90	V-1-C	Bone	3.82E+02	3.93E-02
Nb-95	V-1-T	GI Tract	1.10E-01	1.36E+02
Zr-95	V-1-T	GI Tract	3.25E-01	4.62E+01
Mo-99	M-4-I	Kidney	4.54E-03	3.30E+03
Tc-99m	V-1-C	GI Tract	4.98E-07	3.01E+07
Ru-103	V-1-T	GI Tract	1.41E-01	1.06E+02
Sb-124	V-1-T	GI Tract	7.91E-01	1.90E+01
Sb-125	V-1-T	GI Tract	4.46E-01	3.36E+01
Te-127	V-1-T	GI Tract	4.45E-05	3.37E+05
I-131	M-4-I	Thyroid	7.55E+00	1.99E+00
Xe-131m	P-1-β	Skin	1.89E-05	7.94E+05
I-132	V-1-C	Thyroid	3.62E-07	4.14E+07
I-133	M-4-I	Thyroid	7.00E-02	2.14E+02
Xe-133	P-1-γ	Total Body	6.52E-06	7.67E+05
Xe-133m	P-1-β	Skin	3.96E-05	3.79E+05
Cs-134	V-1-C	Liver	7.07E+00	2.12E+00
I-134	V-1-C	Thyroid	2.79E-13	5.38E+13
I-135	V-1-C	Thyroid	4.71E-04	3.18E+04
Xe-135	P-1-γ	Total Body	4.02E-05	1.24E+05
Xe-135m	P-1-γ	Total Body	6.92E-05	7.23E+04

Nuclide	Pathway-Site - Age*	Organ	Dose Factor mrem/Ci	Design Objective Annual Quantity (Ci)
Cs-136	M-3-I	Total Body	7.50E-02	6.67E+01
Cs-137	V-1-C	Bone	6.91E+00	2.17E+00
Xe-137	P-1-β	Skin	4.24E-04	3.54E+04
Xe-138	P-1-γ	Total Body	1.96E-04	2.55E+04
Ba-140	V-1-C	Bone	4.68E-02	3.21E+02
La-140	V-1-A	GI Tract	6.95E-03	2.16E+03
Ce-141	V-1-T	GI Tract	1.29E-01	1.16E+02
Ce-144	V-1-T	GI Tract	3.44E+00	4.36E+00

*Codes are as follow:

Pathways

V - Green leafy vegetable ingestion
P - Plume submersion
M - Milk ingestion

Site locations

1 - Residence with garden, 0.5 mi, South sector
3 - Milk (goat), 3.0 miles, ESE sector
4 - Milk (cow), 2.5 miles, East sector

Age Groups

A - Adult
T - Teen
C - Child
I - Infant
β - All ages, beta skin exposure
γ - All ages, gamma total body exposure

II. LIQUID EFFLUENTS

A. CONCENTRATION

1. RETS Requirement

Specification 3.24.3.1 of the Radiological Effluent Technical Specifications (RETS) requires that the concentration of radioactive material released at any time from the site to unrestricted areas shall be limited to the Maximum Permissible Concentration (MPC) specified in 10 CFR 20, Appendix B, Table II, Column 2 for nuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} μCi/ml total activity. To ensure compliance, the following approach will be used for each release.

2. Prerelease Analysis

Most tanks will be recirculated through two volume changes prior to sampling for release to the environment to ensure that a representative sample is obtained. The appropriate recirculation time for those tanks too large to provide two volume changes will be the time that the suspended particulate concentration reaches steady state. Either a one-time test, or prior sampling data, may be used to determine appropriate recirculation time.

Prior to release, a grab sample will be analyzed for each release, and the concentration of each radionuclide determined.

$$C = \sum_{i=1}^{n} C_i \qquad (2.1)$$

where:

C = Total concentration in the liquid effluent at the release point, μCi/ml.

C_i = Concentration of a single radionuclide i, μCi/ml.

3. Total MPC-Fraction

The Total MPC-Fraction (R_j) for each release point will be calculated by the relationship defined by Note 1 of Appendix B, 10 CFR 20:

$$R = \sum_i \frac{Ci}{MPC_i} \tag{2.2}$$

where:

C_i = Effluent concentration of radionuclide i, μC/ml.

MPC_i = The MPC of radionuclide i, as specified in Section 2.1.1, μCi/ml.

R = The Total MPC-Fraction for the release point.

The sum of the ratios at the discharge to the lake must be ≦1 due to the releases from any or all concurrent releases. The following relationship will assure this criterion is met:

$$f_1(R_1-1) + f_2(R_2-1) + f_3(R_3-1) \leqq F \tag{2.3}$$

where:

f_1, f_2, f_3 = The effluent flow rate (gallons/minute) for the respective releases, determined by plant personnel.

R_1, R_2, R_3, = The Total MPC-Fractions for the respective releases as determined by Equation 2.2.

F = Minimum required dilution flow rate. Normally, a conservatively high dilution flow rate is used, that is, flow rate used = $(b_i)(F)$ where b_i is a conservative factor greater than 1.0.

B. INSTRUMENT SETPOINTS

1. Setpoint Determination

The setpoint for each liquid effluent monitor will be established using plant instructions. Concentration, flow rate, dilution, principal gamma emitter, geometry and detector efficiency are combined to give an equivalent setpoint in counts per minute (cpm). The physical and technical description, location and identification number for each liquid effluent radiation detector is contained in Figure 2-2.

The respective alarm/trip setpoints at each release point will be set such that the sum of the ratios at each point, as calculated by Equation 2.2, will not be exceeded. The value of R is directly related to the total concentration calculated by Equation 2.1. An increase in the concentration would indicate an increase in the value of R. A large increase would cause the limits specified in Section 2.1.1 to be exceeded. The minimum alarm/trip setpoint value is equal to the release concentration, but for ease of operation it may be desired that the setpoint (S) be set above the effluent concentration (C) by the same factor (b) utilized in setting dilution flow. That is:

$$S = b \times C \qquad (2.4)$$

Liquid effluent flow paths and release points are indicated in Figure 2.1.

2. Post-Release Analysis

A post-release analysis will be done using actual release data to ensure that the limits specified in Section 1 were not exceeded.

A composite list of concentrations (C_i), by isotope, will be used with the actual liquid radwaste (f) and dilution (F) flow rates (or volumes) during the release. The data will be substituted into Equation 2.3 to demonstrate compliance with the limits in Section 1. This data and setpoints will be recorded in auditable records by plant personnel.

C. DOSE

1. <u>RETS Requirement</u>

Specification 3.24.4.1 the Radiological Effluent Technical Specification (RETS) requires that the quantity of radionuclides released be limited such that the dose or dose commitment to an individual from radioactive materials in liquid effluents released to unrestricted areas from each reactor (see Figure 2-2.) will not exceed:

a. During any calendar quarter, 1.5 mrem to the total body and 5 mrem to any organ, and

b. During any calendar year, 3 mrem to the total body and 10 mrem to any organ.

To ensure compliance, quantities of activity of each radionuclide released will be summed for each release and accumulated for each quarter as follows in Section 2.

2. Release Analysis

Calculations shall be performed for each batch release, and weekly for continuous releases according to the formula:

$$A_i/C_i \leq 0.5 \qquad (2.5)$$

where:

A_i = Cumulative quarterly activity of nuclide i identified in liquid release (Ci).

C_i = Design objective annual quantity of radionuclide i from Table 2.2.

Radionuclides may be omitted from the summation if they fall under the criteria of allowed omission specified by Note 5 to Appendix B, 10 CFR 20.

The design basis quantities are derived in such a conservative manner that doses may be greatly overestimated by this technique. As a consequence of this conservatism, and in light of historically consistent operations with releases well below annual design basis quantities, the Palisades Plant technical specifications do not require monthly dose projections. Instead, if at any time, calculations by Equation (2.5) results in values greater than 0.5 for a given quarter or 1.0 for year-to-date, the NRC LADTAP code will be run to ensure that Specification 3.24.4.1 has been met.

Values for the design basis quantities (Ci), and the dose per Curie $(D_c/C_c)_i$ for each nuclide i shown in Table 2.2, were calculated as follows in Sections 2.1 and 2.2

2.1 Water Ingestion

The dose to an individual from ingestion of radioactivity from any source is described by the following equation:

$$D_j = \sum_{i=1}^{i} (DCF)_{ij} \times I_i \text{ rem} \qquad (2.11)$$

where:

D_j = Dose for the j^{th} organ from radionuclides released rem.

j = The organ of interest.

$(DCF)_{ij}$ = Adult ingestion dose commitment factor for the j^{th} organ from the i^{th} radionuclide rem/μCi, see attached as Table 2.1.

I_i = Activity ingested of the i^{th} radionuclide, μCi.

I_i is described by:

$$I_i = \frac{(A_i)(V)(365)}{(1000)(d)} \ \mu\text{Ci} \qquad (2.12)$$

where:

365 = Days per year.

A_i = Annual activity released of i^{th} radionuclide μCi.

V = Average rate of water consumption (730 ml/d ICRP 23, p. 358).

d = Dilution water flow for year.

1000 = Dispersion factor from discharge to nearest drinking water supply.

The dose equation then becomes:

$$D_j = \frac{266}{d} \sum_{i=1}^{i} (DCF)_{ij} \times A_i \text{ mrem} \qquad (2.13)$$

2.2 Fish Ingestion

The dose to an individual from the consumption of fish is described by Equation 2.13. In this case the activity ingested of the i^{th} radionuclide (I_i) is described by:

$$I_i = \frac{A_i \; B_i \; F}{15d} \; \mu Ci \qquad (2.14)$$

where:

A_i = Annual released of i^{th} radionuclide, μCi.

B_i = Fish concentration factor of i^{th} radionuclide $\frac{\mu Ci/gm}{\mu Ci/ml}$, see Table 2.0.

F = Amount of fish eaten per year (21 kg).

15 = Dispersion factor from discharge to fish exposure point.

d = Dilution water flow for year.

Substitution of Equation 2.14 into Equation 2.11 gives:

$$D_j = \frac{1400}{d} \sum_{i=1}^{i} A_i \times B_i \times DCF_i \text{ mrem} \qquad (2.15)$$

3. Annual Analysis

A complete analysis utilizing the NRC computer code LADTAP with the total source release will be done annually in conjunction with the annual environmental report. This analysis will provide estimates of dose to the total body and various organs in addition to the dose limiting organs considered in the method of Section 2. The following approach is utilized in LADTAP. The dose to the j^{th} organ from m radionuclides, D_j, is described by:

$$D_j = \sum_{i=1}^{m} D_{ij} \text{ rem} \tag{2.16}$$

$$= \sum_{i=1}^{m} (DCF)_{ij} \times I_i \text{ rem} \tag{2.17}$$

where:

D_{ij} = Dose to the j^{th} organ from the i^{th} radionuclide, rem.

j = The organ of interest (bone, GI tract, thyroid, liver, kidney, lung or total body).

$(DCF)_{ij}$ = Adult ingestion dose commitment factor for the j^{th} organ from the i^{th} radionuclide, rem/μCi, see Table 2.1.

I_i = Activity ingested of the i^{th} radionuclide, μCi.

I_i for water ingestion is described by:

$$I_i = \frac{A_i \, V\tau}{\nu \, d} \ \mu\text{Ci} \tag{2.18}$$

and for fish ingestion I_i is described by:

$$I_i = \frac{A_i \, B_i \, F\tau}{\nu \, d} \ \mu\text{Ci} \tag{2.19}$$

where:

A_i = Activity released of j^{th} radionuclide during the year, μCi.

V = Average rate of water consumption (730 ml/d).

τ = Number of days during the year (365 d).

ν = Dispersion factor from point of discharge to point of exposure.

d = Dilution water volume (ml).

B_i = Fish concentration factor of the i^{th} radionuclide, $\frac{\mu Ci/gm}{\mu Ci/ml}$.

F = Amount of fish eaten per day (57.5 gm).

D. OPERABILITY OF LIQUID RADWASTE EQUIPMENT

The Palisades liquid radwaste system is designed to reduce the radioactive materials in liquid wastes prior to their discharge (by recycle or shipment for disposal) so that radioactivity in liquid effluent releases to unrestricted areas (see Figure 2-1) will not exceed Specification 3.24.4.1. Maintaining the cumulative fraction of allowable release for each batch release and weekly for continuous releases assures compliance with this requirement. In addition, 13 years of operating experience (to the date this ODCM was first adopted) has shown that design basis quantities never have been exceeded.

E. RELEASE RATE FOR OFFSITE MPC (500 mRem/yr)

10 CFR 20.106 requires radioactive effluent releases to unrestricted areas be less than the limits specified in Appendix B, Table II when averaged over a period not to exceed one year. Concentrations at this level, if ingested for one year, will result in a dose of 500 millirem to the total body or its equivalent to internal organs. In addition, 10 CFR 50.36a requires that the release of radioactive materials be kept as low as is reasonably achievable. However, the section further states that the licensee is permitted the flexibility

of operation, to assure a dependable source of power, to release quantities of material higher than a small percentage of 1 CFR 20 limits but not exceeding those limits under unusual operating conditions. Appendix I to 10 CFR 50 provides the numerical guidelines on limiting conditions for operations to meet the as low as is reasonably achievable requirement.

The LADTAP code has been run to determine the dose due to drinking water at plant discharge concentration (1,000 x nearest drinking water intake concentration). The source term used is given in Table 1.1. Dose to the most limiting organ of the person hypothetically drinking this water is (TBD) mrem. The release rate which would result in a dose rate equivalent to 500 mrem/year (using the more conservative total body limit) is the Curies/year given in Table 1.1 (TBD) times 500(TBD) or (TBD) Ci/yr = (TBD) Ci/sec.

III. URANIUM FUEL CYCLE DOSE

A. SPECIFICATION

In accordance with Specification 3.24.8.1, if either liquid or gaseous quarterly releases exceed the quantity which would cause offsite doses more than twice the limit of Specifications 3.24.4.1, 3.24.5.2 or 3.24.5.3, then the cumulative dose contributions from combined release plus direct radiation sources (from the reactor unit and radwaste storage tanks) shall be calculated. The dose is to be determined for the member of the public projected to be the most highly exposed to these combined sources.

B. ASSUMPTIONS

1. The full time resident determined to be the maximally exposed individual (excluding infant) is assumed also to be a fisherman. This individual is assumed to drink water and ingest local fish at the rates specified in Sections II C.2.1 and II C.2.2.

2. Amount of shoreline fishing (at accessible shoreline adjacent to site security fence) is conservatively assumed as 48 hours per quarter (average of approximately ½ hour per day each day of the quarter) for the second and third quarters of the year, 36 hours for the fourth quarter and 18 hours for the first quarter.

C. DOSE CALCULATION

Maximum doses to the total body and internal organs of an individual shall be determined by use of LADTAP and GASPAR computer codes, and doses to like organs and total body summed. Added to this sum will be a mean dose rate, calculated or measured for the shoreline due to plant presence during the quarter in question, times the assumed fishing time:

$$D_{40} = D_G + D_L + (R_T)(T) \qquad (2.20)$$

where:

D_{40} = 40 CFR 190 Dose (mrem)

D_G = Limiting dose to an individual from gaseous source term (mrem)

D_L = Limiting dose to an individual from liquid source term (mrem)

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

T = Assumed shoreline fishing time for the quarter in question

IV. RADWASTE SYSTEM MODIFICATIONS

1.0 Definition of Major Radwaste System Modification

1.1 Purpose

The purpose of this definition is to assure that Technical Specification 6.20 will be satisfied under clearly identifiable circumstances, and with the objective that current radwaste system capabilities are not jeopardized.

1.2 Definition

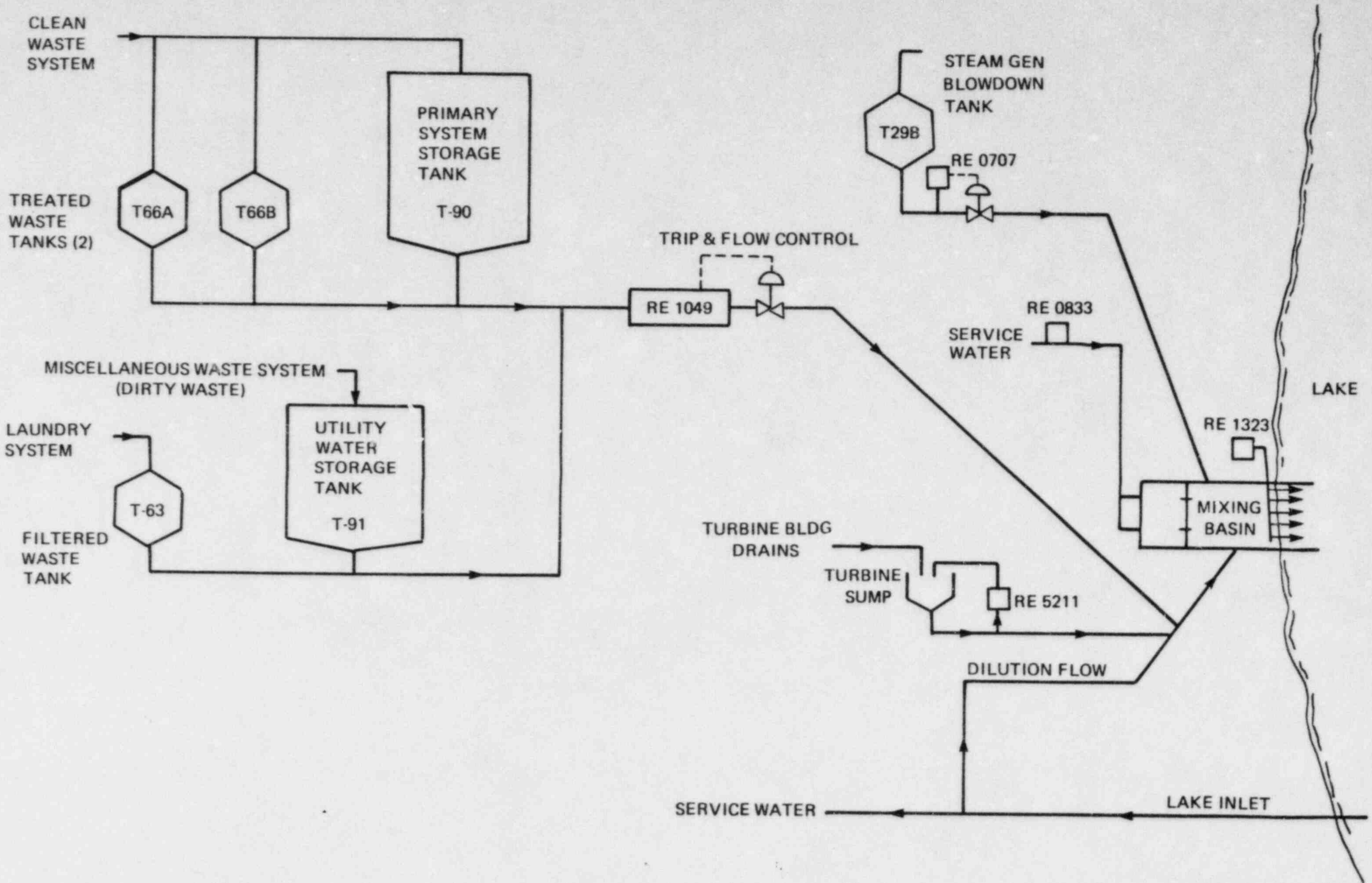
A major radwaste system modification is a modification which would remove (either by bypassing or physical removal) or replace with less efficient equipment, any components of the radwaste system:

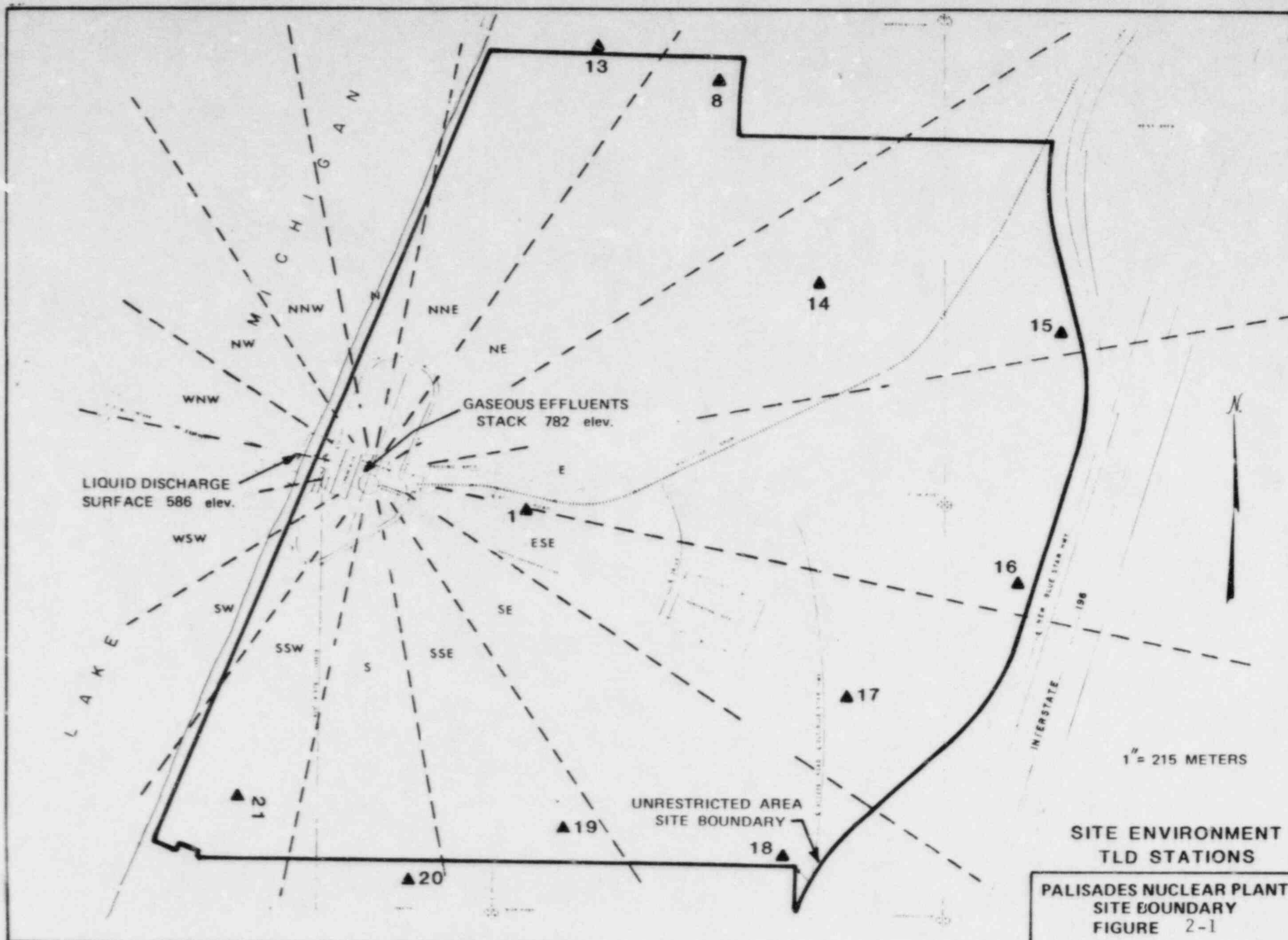
1) Letdown filters or demineralizers
2) Vacuum degassifier
3) Miscellaneous or clean waste evaporators
4) The present waste gas compressor/decay tank system
5) Fuel pool filters/demineralizers
6) Radwaste polishing demineralizers
7) Radwaste solidification system

Improvements or additions to improve efficiency will not be considered major modifications unless a complete substitution of equipment or systems is made with equipment of unrelated design. Examples would be 1) replacement of mechanical degassifier with steam; jet degassifier 2) replacement of waste gas system with cryogenic system 3) replacement of ashphalt solidification with cement system 4) change from deep bed resins to Powdex 5) etc.

FIGURE 2-1

PALISADES RADWASTE EFFLUENTS - LIQUID





PALISADES NUCLEAR PLANT
SITE BOUNDARY
FIGURE 2-1

FIGURE 2-1

TABLE 2.0

BIOACCUMULATION FACTORS

(pCi/kg per pCi/liter)

ELEMENT	FRESHWATER FISH
H	9.0E-01
C	4.6E 03
NA	1.0E 02
P	1.0E 05
CR	2.0E 02
MN	4.0E 02
FE	1.0E 02
CO	5.0E 01
NI	1.0E 02
CU	5.0E 01
ZN	2.0E 03
BR	4.2E 02
RB	2.0E 03
SR	3.0E 01
Y	2.5E 01
ZR	3.3E 00
NB	3.0E 04
MO	1.0E 01
TC	1.5E 01
RU	1.0E 01
RH	1.0E 01
TE	4.0E 02
I	1.5E 01
CS	2.0E 03
BA	4.0E 00
LA	2.5E 01
CE	1.0E 00
PR	2.5E 01
ND	2.5E 01
W	1.2E 03
NP	1.0E 01

TABLE 2.1

ADULT INGESTION DOSE FACTORS
(MREM/PCI INGESTED)
Page 1 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
H 3	NO DATA	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07
C 14	2.84E-06	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07
NA 24	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06
P 32	1.93E-04	1.20E-05	7.46E-06	NO DATA	NO DATA	NO DATA	2.17E-05
CR 51	NO DATA	NO DATA	2.66E-09	1.59E-09	5.86E-10	3.53E-09	6.69E-07
MN 54	NO DATA	4.57E-06	8.72E-07	NO DATA	1.36E-06	NO DATA	1.40E-05
MN 56	NO DATA	1.15E-07	2.04E-08	NO DATA	1.46E-07	NO DATA	3.67E-06
FE 55	2.75E-06	1.90E-06	4.43E-07	NO DATA	NO DATA	1.06E-06	1.09E-06
FE 59	4.34E-06	1.02E-05	3.91E-06	NO DATA	NO DATA	2.85E-06	3.40E-05
CO 58	NO DATA	7.45E-07	1.67E-06	NO DATA	NO DATA	NO DATA	1.51E-05
CO 60	NO DATA	2.14E-06	4.72E-06	NO DATA	NO DATA	NO DATA	4.02E-05
NI 63	1.30E-04	9.01E-06	4.36E-06	NO DATA	NO DATA	NO DATA	1.88E-06
NI 65	5.28E-07	6.86E-08	3.13E-08	NO DATA	NO DATA	NO DATA	1.74E-06
CU 64	NO DATA	8.33E-08	3.91E-08	NO DATA	2.10E-07	NO DATA	7.10E-06
ZN 65	4.84E-06	1.54E-05	6.96E-06	NO DATA	1.03E-05	NO DATA	9.70E-06
ZN 69	1.03E-08	1.97E-08	1.37E-09	NO DATA	1.28E-08	NO DATA	2.96E-09
BR 83	NO DATA	NO DATA	4.02E-08	NO DATA	NO DATA	NO DATA	5.79E-08
BR 84	NO DATA	NO DATA	5.21E-08	NO DATA	NO DATA	NO DATA	4.09E-13
BR 85	NO DATA	NO DATA	2.14E-09	NO DATA	NO DATA	NO DATA	LT E-24
RB 86	NO DATA	2.11E-05	9.83E-06	NO DATA	NO DATA	NO DATA	4.16E-06
RB 88	NO DATA	6.05E-08	3.21E-08	NO DATA	NO DATA	NO DATA	8.36E-19
RB 89	NO DATA	4.01E-08	2.82E-08	NO DATA	NO DATA	NO DATA	2.33E-21
SR 89	3.08E-04	NO DATA	8.84E-06	NO DATA	NO DATA	NO DATA	4.94E-05
SR 90	7.58E-03	NO DATA	1.86E-03	NO DATA	NO DATA	NO DATA	2.19E-04
SR 91	5.67E-06	NO DATA	2.29E-07	NO DATA	NO DATA	NO DATA	2.70E-05
SR 92	2.15E-06	NO DATA	9.30E-08	NO DATA	NO DATA	NO DATA	4.26E-05
Y 90	9.62E-09	NO DATA	2.58E-10	NO DATA	NO DATA	NO DATA	1.02E-04
Y 91m	9.09E-11	NO DATA	3.52E-12	NO DATA	NO DATA	NO DATA	2.67E-10
Y 91	1.41E-07	NO DATA	3.77E-09	NO DATA	NO DATA	NO DATA	7.76E-05
Y 92	8.45E-10	NO DATA	2.47E-11	NO DATA	NO DATA	NO DATA	1.48E-05

TABLE 2.1 (CONT'D)

ADULT INGESTION DOSE FACTORS
(MREM/PCI INGESTED)
Page 2 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
Y 93	2.68E-09	NO DATA	7.40E-11	NO DATA	NO DATA	NO DATA	8.50E-05
ZR 95	3.04E-08	9.75E-09	6.60E-09	NO DATA	1.53E-08	NO DATA	3.09E-05
ZR 97	1.68E-09	3.39E-10	1.55E-10	NO DATA	5.12E-10	NO DATA	1.05E-04
NB 95	6.22E-09	3.46E-09	1.86E-09	NO DATA	3.42E-09	NO DATA	2.10E-05
MO 99	NO DATA	4.31E-06	8.20E-07	NO DATA	9.76E-06	NO DATA	9.99E-06
TC 99m	2.47E-10	6.98E-10	8.89E-09	NO DATA	1.06E-08	3.42E-10	4.13E-07
TC101	2.54E-10	3.66E-10	3.59E-09	NO DATA	6.59E-09	1.87E-10	1.10E-21
RU103	1.85E-07	NO DATA	7.97E-08	NO DATA	7.06E-07	NO DATA	2.16E-05
RU105	1.54E-08	NO DATA	6.08E-09	NO DATA	1.99E-07	NO DATA	9.42E-06
RU106	2.75E-06	NO DATA	3.48E-07	NO DATA	5.31E-06	NO DATA	1.78E-04
AG110m	1.60E-07	1.48E-07	8.79E-08	NO DATA	2.91E-07	NO DATA	6.04E-05
TE125m	2.68E-06	9.71E-07	3.59E-07	8.06E-07	1.09E-05	NO DATA	1.07E-05
TE127m	6.77E-06	2.42E-06	8.25E-07	1.73E-06	2.75E-05	NO DATA	2.27E-05
TE127	1.10E-07	3.95E-08	2.38E-08	8.15E-08	4.48E-07	NO DATA	8.68E-06
TE129m	1.15E-05	4.29E-06	1.82E-06	3.95E-06	4.80E-05	NO DATA	5.79E-05
TE129	3.14E-08	1.18E-08	7.65E-09	2.41E-08	1.32E-07	NO DATA	2.37E-08
TE131m	1.73E-06	8.46E-07	7.05E-07	1.34E-06	8.57E-06	NO DATA	8.40E-05
TE131	1.97E-08	8.23E-09	6.22E-09	1.62E-08	8.63E-08	NO DATA	2.79E-09
TE132	2.52E-06	1.63E-06	1.53E-06	1.80E-06	1.57E-05	NO DATA	7.71E-05
I 130	7.56E-07	2.23E-06	8.80E-07	1.89E-04	3.48E-06	NO DATA	1.92E-06
I 131	4.16E-06	5.95E-06	3.41E-06	1.95E-03	1.02E-05	NO DATA	1.57E-06
I 132	2.03E-07	5.43E-07	1.90E-07	1.90E-05	8.65E-07	NO DATA	1.02E-07
I 133	1.42E-06	2.47E-06	7.53E-07	3.63E-04	4.31E-06	NO DATA	2.22E-06
I 134	1.06E-07	2.88E-07	1.03E-07	4.99E-06	4.58E-07	NO DATA	2.51E-10
I 135	4.43E-07	1.16E-06	4.28E-07	7.65E-05	1.86E-06	NO DATA	1.31E-06
CS134	6.22E-05	1.48E-04	1.21E-04	NO DATA	4.79E-05	1.59E-05	2.59E-06
CS136	6.51E-06	2.57E-05	1.85E-05	NO DATA	1.43E-05	1.96E-06	2.92E-06
CS137	7.97E-05	1.09E-04	7.14E-05	NO DATA	3.70E-05	1.23E-05	2.11E-06
CS138	5.52E-08	1.09E-07	5.40E-08	NO DATA	8.01E-08	7.91E-09	4.65E-13
BA139	9.70E-08	6.91E-11	2.84E-09	NO DATA	6.46E-11	3.92E-11	1.72E-07

TABLE 2.1 (CONT'D)

ADULT INGESTION DOSE FACTORS
(MREM/PCI INGESTED)
Page 3 of 3

NUCLIDE	BONE	LIVER	T.BODY	THYROID	KIDNEY	LUNG	GI-LLI
BA140	2.03E-05	2.55E-08	1.33E-06	NO DATA	8.67E-09	1.46E-08	4.18E-05
BA141	4.71E-08	3.56E-11	1.59E-09	NO DATA	3.31E-11	2.02E-11	2.22E-17
BA142	2.13E-08	2.19E-11	1.34E-09	NO DATA	1.85E-11	1.24E-11	3.00E-26
LA140	2.50E-09	1.26E-09	3.33E-10	NO DATA	NO DATA	NO DATA	9.25E-05
LA142	1.28E-10	5.82E-11	1.45E-11	NO DATA	NO DATA	NO DATA	4.25E-07
CE141	9.36E-09	6.33E-09	7.18E-10	NO DATA	2.94E-09	NO DATA	2.42E-05
CE143	1.65E-09	1.22E-06	1.35E-10	NO DATA	5.37E-10	NO DATA	4.56E-05
CE144	4.88E-07	2.04E-07	2.62E-08	NO DATA	1.21E-07	NO DATA	1.65E-04
PR143	9.20E-09	3.69E-09	4.56E-10	NO DATA	2.13E-09	NO DATA	4.03E-05
PR144	3.01E-11	1.25E-11	1.53E-12	NO DATA	7.05E-12	NO DATA	4.33E-18
ND147	6.29E-09	7.27E-09	4.35E-10	NO DATA	4.25E-09	NO DATA	3.49E-05
W 187	1.03E-07	8.61E-08	3.01E-08	NO DATA	NO DATA	NO DATA	2.82E-05
NP239	1.19E-09	1.17E-10	6.45E-11	NO DATA	3.65E-10	NO DATA	2.40E-05

TABLE 2.2

LIQUID EFFLUENT
DESIGN OBJECTIVE ANNUAL QUANTITIES

Nuclide	Limiting Dose	Dose Factor (mRem/Curie)	Design Objective Annual Quantity (Curies)
H-3	Total Body	1.31E-06	2.29E+06
Cr-51	GI Tract	1.49E-03	6,730
Mn-54	GI Tract	6.22E-02	161
Fe-55	GI Tract	1.21E-02	827
Fe-59	GI Tract	3.77E-02	265
Co-58	GI Tract	8.38E-03	1,190
Co-60	Total Body	4.50E-01	6.67
Rb-86	Total Body	2.28E-01	13.2
Sr-89	Bone	1.07E-02	935
Sr-90	Bone	1.64E-01	61.0
Mo-99	GI Tract	1.11E-03	9,020
Te-99m	GI Tract	6.88E-05	145,000
Te-127m	GI Tract	1.01E-01	99.2
Te-127	GI Tract	2.85E-02	259
Te-129m	GI Tract	2.57E-01	38.9
Te-131m	GI Tract	3.73E-01	26.8
Te-132	GI Tract	3.42E-01	29.2
I-130	Thyroid	5.28E-02	190
I-131	Thyroid	3.24E-01	30.9
I-132	Thyroid	1.32E-02	755
I-133	Thyroid	4.05E-02	254
I-135	Thyroid	2.83E-02	353
Cs-134	Total Body	2.83E+00	1.06
Cs-136	Total Body	3.94E-01	7.61
Cs-137	Total Body	1.67E+00	1.80
Ba-140	GI Tract	1.86E-03	5,390
La-140	GI Tract	2.57E-02	390
Np-239	GI Tract	2.66E-03	3,750

PROPOSED

Consumers Power Company
Palisades Plant - Docket 50-255

PROCESS CONTROL PROGRAM (PCP)

July 31, 1984

PALISADES PROCESS CONTROL PROGRAM (PCP)

1.0 ASPHALT VOLUME REDUCTION SYSTEM

The Palisades Plant utilizes a Waste Chem volume reduction and solidification system (VRS) to process various radioactive liquid waste streams.

The process utilizes thermal energy (heat) to evaporate water from the radioactive wastes thus reducing waste volume to anhydrous waste residue. This residue is then encapsulated in a thermoplastic matrix (asphalt).

The end product is a monolithic, freestanding solid with no free liquid. Fifty-five gallon drums are used to contain the encapsulated waste for temporary storage, transport and burial.

1.1 VARIABLES INFLUENCING SOLIDIFICATION

The purpose of this section is to identify and define those process variables which have a direct effect on the ability of the final product to form a freestanding monolith with no free liquid.

The following variables influence the properties and consistency of the final product:

a. Asphalt type

b. Waste chemical species used as feed

c. Ratio of waste-to-asphalt, and

d. Process temperature

1.2 ASPHALT TYPE

Asphalt utilized in the system shall conform to ASTM-D-312-71, Type III. This is an oxidized petroleum-based asphalt, such as Witco Chemical Company's Pioneer 221. The specifications for this asphalt are provided in Appendix A.

This grade of asphalt has a low residual volatile content and a high molecular weight. At room temperature, and at all normal ambient temperature conditions, this material is a freestanding, monolithic, solid.

Utilization of an asphalt complying with ASTM-D-312-71, Type III, is the means by which process control of this variable is achieved.

1.3 WASTE CHEMICAL SPECIES

The type and relative quantity (waste-to-asphalt ratio) of waste chemicals being incorporated into the asphalt matrix has a direct influence on the properties of the final product. Encapsulation of inorganic salts and solids typically "stiffen" and harden the end product, whereas organic liquids have the opposite tendency. When the specified ratio of waste-to-asphalt is maintained, final product properties for typical power plant wastes are independent of the waste type.

However, certain chemical specifications are required as an outer bounds to limit end product tendencies to soften at lower temperatures.

A maximum limit of 1% oil by weight will be applied to the waste feed streams. Most oils found in power plants are low viscosity fluids, which are liquid at room temperature. Based on calculations for a typical waste stream with 20% solids by weight and 1% oil by weight, Waste Chem has found the total concentration of oil in the end product would be approximately 2.5%. This would then lower the end product softening point by approximately 5°F, or approximately 2°F lower per percent of oil. This is within an acceptable range and, therefore, is the basis for the limit of 1% oil in the feed stream.

Other chemical specifications on feed streams are specified below. These are required primarily for calculating waste-to-asphalt ratio which is important to end product, and equipment protection (which will have no discernable effect on the end product).

REQUIRED ANALYSIS

Concentrates	Resin/Powdex
pH (Equip Limit)	pH (Equip Limit)
% Solids	% Slurry
Sp Gravity	
Oil %	

1.4 WASTE-TO-ASPHALT RATIO

The ratio of waste to-asphalt contained in the end product has the most significant effect on the viscosity and physical consistency of that product. Process control is achieved by placing limitations on the range of waste-to-asphalt ratios allowable for each waste type.

Waste-to-asphalt ratios (mass) and evaporative rates should not exceed the verification test values specified for the waste feeds as follows:

Feed	Ratio of Waste-to-Asphalt in the End Product
1. Boric Acid Concentrates at 120 L/hr Evaporative Rate	≤ 1.0/1.0
2. Spent Resins at 80 L/hr Evaporative Rate	≤ .67/1.0
3. Powdex at 80 L/hr Evaporative Rate	≤ .67/1.0

Should the ratio of waste-to-asphalt be increased above the range specified in the foregoing table, the end product viscosity will increase and may exhibit a grainy texture. This could lead to "pyramiding" of the product in the container, thereby decreasing the container filling efficiency. In all cases, the product will cool to form a freestanding monolith. If lower than specified waste loadings are realized, the end product properties will approach that of pure asphalt. Again, solidification is assured; however, toward this end of the spectrum, economical volume reduction may not be realized.

Maximum concentrate feed rate can be determined by the following formula:

$$\text{Conc Feed Rate} = \frac{0.528\ \text{GPM}}{(1.0 - \text{Solids Fraction})\ (\text{Sp Gravity})}$$

NOTE: 0.528 gpm = 120 L/hr evaporative rate.

The corresponding asphalt feed is calculated by:

$$\text{Asphalt Feed (GPM)} = \frac{(\text{Conc Feed Rate GPM})\ (\text{Solids Fraction})\ (\text{Sp Gravity})}{(\text{Waste-to-Asphalt Ratio})}$$

where the recommended waste-to-asphalt ratio is 1.0.

NOTES:

1. The density of Type III asphalt is 1.0 so a density correction is not needed.

2. The minimum asphalt flow is 0.065 gpm because of lubrication requirements of the twin screws.

3. If either the concentrate or asphalt flows cannot be met, the calculated flows can be ratioed to new values to maintain the 1.0/1.0 waste-to-asphalt ratio as long as the maximum concentrate flow or the minimum asphalt flows is not exceeded.

Maximum bead resin or Powdex can be determined by the following formula:

$$\text{Resin Feed} = \frac{0.35 \text{ GPM}}{(1.0 - \text{Solid Fraction})}$$

NOTES: 1. Solid fraction = slurry fraction ÷ 2. Example is a 50% slurry = 25 weight %.

2. 0.35 gpm = 80 L/hr evaporative rate.

The corresponding asphalt feed is calculated by:

$$\text{Asphalt Feed (GPM)} = \frac{(\text{Resin Feed})\ (\text{Solid Fraction})}{(\text{Waste-to-Asphalt Ratio})}$$

Where the recommended waste-to-asphalt ratio is 0.67, the notes on the preceding asphalt calculation apply.

The operator can also visually confirm that the quality of the end product is approximately being maintained. A CCTV camera "views" the discharge from the extruder-evaporator, and a TV monitor located in the Solid Radwaste Building Control Room allows the operator to observe the physical consistency of the product as it is discharged into the container. At evaporative rates higher than specified, there will be excessive steaming at discharge nozzle. At higher waste-to-asphalt loading the discharge will appear grainy and stringy.

1.5 PROCESS TEMPERATURES

A proper temperature profile along the length of the extruder-evaporator is required to provide adequate evaporative (process) capacity, and to assure that free water is not discharged from the machine.

Process temperature profiles for waste feeds should be maintained as recommended below:

Waste Type	Process Temperature (°F)					
Zones:	1	2	3	4	5/6	7
Boric Acid Concentrates LWS Concentrates Chemical/Laundry Waste	300°	280°	280°	300°	300°	*
Spent Resins/Powdex	300°	**	**	**	**	*

NOTE: No zone shall be maintained below 240°F.

Low temperature alarms are provided to alert the operator to a low temperature out-of-specification condition which could potentially lead to the discharge of free water. If the out-of-specification condition persists for two (2) minutes, the extruder-evaporator is automatically tripped to prevent free water from being discharged into the container. Free water cannot be discharged in the interim, since the residual heat of the extruder-evaporator itself is sufficient to effect evaporation.

Verification of the absence of free water and product solidification will be made on a minimum of one container from every tenth batch or run. Verification is recommended to be done on every tenth drum shipped. The required container shall be examined through a removable lid bung or equivalent means for solidification by checking penetration with a solid tool and inverted for a minimum of eight hours to check for free water. Evidence of free water other than a few drops of condensation shall be cause for rejection and evaluation system product.

2.0 DEWATERING SOLIDS IN HIGH INTEGRITY CONTAINERS (HIC)

2.1 Solids such as bead resin, filter cartridges and powdered resin (Powdex) may be dewatered and shipped in HICs per approved vendor procedures and the HIC certificate of compliance.

2.2 High integrity containers are approved by the individual burial ground agreement states as meeting 10 CFR 61 waste form stability requirements.

2.3 Free water determination shall be verified by the successful completion and documentation of the vendors approved dewatering procedure.

3.0 DELAWARE CUSTOM MATERIAL (DCM) - SILICATE CEMENT

Liquid wastes can be solidified by the DCM method. The silicate solidifies and the cement gives structural strength.

3.1 For solidification, acquire a representative sample of waste. Before following the guidelines outlined below, determine the type of waste to be solidified, example, lab waste, laundry waste, decon solutions, boric acid, oil, etc. Sample for pH, boric acid, visible organics and radioactivity.

Use analysis to determine the proper laboratory procedure to test.

All batches shall be lab tested prior to solidification in a larger container unless sample analysis (pH ± 0.2 and Boron ± 20%) matches the analysis of a waste type which has previously passed lab test criteria.

For all oil waste, do not exceed 50% by volume. Oil must be emulsified with a detergent or boric acid in some type of neutral aqueous waste or tap water.

NOTE: Oil cannot be shipped to Barnwell, South Carolina.

For spent resins, liquid absorbent, or other earthen-like material, dilute with an equal volume of concentrate or tap water to solidify.

All results must be recorded initially, at approximately 24 hours and approximately 48 hours after testing. Grade observations to evaluate sample mixes. The 48-hour test can be omitted if the 24-hour test is good.

The quantity of chemicals added to solidify radwaste shall be within 20% of the quantity as determined by the laboratory test of Step G.

3.2 SOLIDIFICATION AND FREE WATER DETERMINATION

Solidification shall be considered successful if, 48 hours after completion of Appendix A solidification, there is not standing water on the waste surface and the surface is not penetrated more than 2" with a 1" diameter rod. If deeper penetration is possible, then the drum can still be considered solid if the penetration hole remains open after the rod is withdrawn.

Silicate cement shall cure for a minimum of 28 days prior to shipment for disposal. For silicate cement drums, the following shall be done:

Each drum shall be inspected for absence of detectable freestanding liquid after curing at least 28 days. With the drum lid installed, invert each drum and allow drum to remain upside down for at least 24 hours.

After 24 hours, inspect each drum by placing upright and removing the lid. The RMC Supervisor or designate and a QC Inspector shall inspect each drum for presence of liquid. Drums which failed the 48-hour solidification evaluation should be capable of passing at this point. If no detectable freestanding liquid is present, the drum can be prepared for shipment. RMC Supervisor and QC Inspector document if no detectable freestanding liquid is present.

In the event liquid is observed, those drums with liquid shall be drained of all liquid. When no further liquid can be drained from the drum in a 24-hour period, the drum shall be core-bored or overpacked with two bags of approved absorbent and inspected by QC and RMC to verify that the drum is dry. After this verification (and documentation) the drum may be prepared for shipment.

Inspect the drum lid and gasket for defects prior to lid installation. Install lid. Use a different lid if defects are found which prevent a tight seal between drum and lid.

4.0 10 CFR 61 REQUIREMENTS

4.1 10 CFR 61 classification requirements will be met using Wastetrak computer software program using the scaling factor methodology of

AIF/NESP-027, Methodologies for Classification of Low-Level Radioactive Wastes From Nuclear Power Plants, 1983.

The scaling factors will be updated by an ongoing analysis program of actual waste streams. The program will initiate with semiannual samples of available waste streams and may be modified to longer intervals if the data base warrants. Waste streams should include, if available; bead resin, concentrates, reactor coolant, clean waste, filter crud and compacted trash.

4.2 10 CFR 61 waste form stability requirements will be met by generic testing of the asphalt/waste stream product. The generic waste streams will be boric acid, bead resin and chemical regenerative wastes.

4.3 Documentation of the waste stream analysis, waste form stability and computer software scaling factor security shall be maintained by the Radiological Services Department.

APPENDIX A

Consumers Power Company
Palisades Plant

PROCESS CONTROL PROGRAM (PCP)

ASPHALT TECHNICAL DATA SUMMARY

WITCO CHEMICAL - PIONEER 221

1. Basic Constituent

Pioneer 221 is an oxidized petroleum base asphalt. Oxidation is accomplished by air blowing at temperatures ranging from 200°C (392°F) to 300°C (572°F). Air blowing results in a product with minimum volatile content (0.1%), greater inertness and higher molecular weight.

2. Flash Point

The Flash Point of Pioneer 221 is in excess of 288°C (549°F). The Flash Point is determined by the Cleveland Open Cup (ASTM D92-71) method. It is the lowest temperature at which surface vapors will momentarily ignite when a test flame is passed over the surface.

3. Fire Point

The Fire Point of Pioneer 221 is in excess of 300°C (572°F). The Fire Point, like the Flash Point, is determined by the Cleveland Open Cup (ASTM D92-72) method. It is the lowest temperature at which the surface vapors will burn for at least 5 seconds before going out, the vapors being ignited as in the test for Flash Point.

4. Ignition Point

The Ignition Point of Pioneer 221 is approximately 400°C (752°F).

The Ignition Point is the lowest temperature at which the heat loss from the combustible mixture is exceeded by the heat produced in the chemical reaction. It is thus the lowest temperature at which combustion begins and continues in an air environment.

5. Softening Point

The Softening Point of Pioneer 221 is in the temperature range of 88-94°C (190-201°F).

The Softening Point is determined by the Ring and Ball method (ASTM D-36-70).

6. Viscosity

The Viscosity of Pioneer 221 in the temperature range from 250°F to 400°F is presented in attached graph.

The graph is based on the following data from Witco Chemical:

Saybolt Furol Viscosity

at 205°C	54 sec
at 177°C	161 sec

7. Penetration

The Penetration of Pioneer 221 by ASTM Method D-5-73 for various temperatures is given below:

25°C (77°F)	22-30 dmm
46°C (115°F)	40-60 dmm
0°C (32°F)	13-18 dmm

The abbreviation "dmm" means one-tenth of a millimeter. The number of dmm's represents needle penetration under standard conditions of loading and time for a given temperature.

8. Specific Gravity

The Specific Gravity of Pioneer 221 is approximately 1.0 gram per cc.

Specific Gravity is determined by ASTM Method D-70-72, which employs a pyenometer. A pyenometer is a container of known volume which is weighed empty and filled with sample.

9. Solubility

Pioneer 221 may be considered to be entirely waterproof and insoluble in water. Pioneer 221 is soluble in petroleum solvents such as naphtha, mineral spirits and kerosene, in addition to carbon tetrachloride, carbon disulfide and trichlorethylene.

PIONEER 221 LAMINATING & INDUSTRIAL ASPHALT
PIONEER E-7465 FOR SALT CARTON MANUFACTURERS

Pioneer 221 is an all-purpose, tough, medium softening point asphalt for use in laminating paper, foil-to-paper, as a base pigment for paints and varnishes, or in the manufacture of sealers and adhesives.

Pioneer 221 complies with federal specifications set forth by the Food & Drug Administration for use in packaging and sealing food products and will not stain, or impart an odor or taste when used properly in connection with packaging products.

PHYSICAL CHARACTERISTICS

Softening Point	190-210°F
Penetration @ 77°F	20-30 dmm
Ductility @ 77°F	2.5 cms +
Solubility CCL	99.0% +
Flash Point (C.O.C.)	550°F +
Weight Per Gallon	8.3 lbs
Use Temperature	400°F ± 25°
Viscosity @ 400°F	.94 secs
Viscosity @ 375°F	.174 secs
Viscosity @ 350°F	.360 secs

Packaging: Bulk - Tankwagon (5000 gal), tank car (10,000 gal)
Packages - 100 lb cartons

TABLE 1 - COMPARISON OF ELEMENTAL ANALYSES AND OTHER CHEMICAL PROPERTIES OF ROAD TAR, COAL-TAR PITCH AND PETROLEUM ASPHALTS

MATERIAL GRADE	ASPHALT CEMENT AC-10	ROAD TAR RT-12	ROOFING ASPHALT TYPE III	ROOFERS PITCH TYPE A
ELEMENTAL ANALYSIS, percent:				
C	85.8	92.2	86.0	92.8
H	9.7	5.2	9.9	5.1
N	0.6	1.5	0.5	1.5
O	0.5	1.0	0.7	---
S	2.8	0.6	2.9	1.53
C/H atomic ratio	0.74	1.49	0.73	1.53
MOLECULAR WEIGHT, (Numbers Avg)	1030	420	1160	497
CARBON ATOM DISTRIBUTION: (percent of total carbon)				
AROMATIC CARBON	34	80	37	79
NAPHTHENE CARBON	23	15	23	18
PARAFFIN CARBON	43	5	40	3

ATTACHMENT A

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, MI 49201 • (517) 788-0550

June 7, 1982

Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 -
BIG ROCK POINT PLANT AND
DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT

ADDITIONAL INFORMATION PERTAINING TO RADIOLOGICAL EFFLUENTS TECHNICAL SPECIFICATIONS (RETS)

Consumers Power Company agreed in a meeting with the NRC, held in Grand Rapids on January 4-7, 1982 to supply Franklin Research Center, which was contracted by the NRC to review and evaluate Consumers Power Company's RETS submittals, additional information to amplify and corroborate Consumers Power Company's previous RETS submittals. Consumers Power Company letter dated February 3, 1982 provided a portion of the requested information. It is the intent of this correspondence to furnish the NRC/FRC with additional information related to the following technical areas:

1. For Big Rock Point, provide rationale to demonstrate that 5 mR/hr corresponds to a reasonable curie limit in each tank (ie, provide basis for number of curies per 5,000-gallon tank). Please see the enclosed Appendix (1) for Consumers Power Company's response.

2. For Palisades, provide justification for omitting Fe-55 analysis. Please see the enclosed Appendix (2) for Consumers Power Company's response.

3. For Big Rock Point and Palisades, perform $100/\bar{\bar{E}}$ calculations to prove that curie limits in liquid and gas tanks are not needed. Please see the enclosed Appendix (3) for Consumers Power Company's response.

To briefly summarize the information provided, the following synopsis is given:

Appendix (1)

Exposure rates of 5 mR/hr and above correspond to reasonable curie limits for tanks considered at Big Rock Point (T15A and T15B which are waste hold tanks and T-40 which is the condensate storage tank). Exposure rates of 5 mR/hr above background for each of these tanks corresponds to a curie content which is a small fraction of the design basis quantity limits (for liquid releases). If one of these tanks ruptured with a 5 mR/hr reading and released its contents to the environment, the offsite doses would be a small fraction of 10 CFR 50, Appendix I limits. Big Rock Point liquid tanks may be controlled as described in this Appendix.

Appendix (2)

Effluent calculations must include any radionuclide which contributes greater than 10% of the MPC fraction in a release (10 CFR 20, Note 5 to Appendix B). It is shown that Fe-55 is below 0.25% of total MPC fraction and, therefore, may be excluded from effluent reporting requirements.

Appendix (3)

The last $\bar{E}$ determination at Palisades was used to calculate the gaseous radionuclide ratio present in primary coolant. The activities corresponding to these ratios were increased to the Technical Specification limit and compared to design basis quantity limits (for gaseous releases). If one gas tank contained the entire primary coolant gaseous activity based on operating at the Technical Specification limit, it would not exceed the design basis quantity curie limit. Curie limits are not needed on gas tanks at Palisades (as shown in Appendix 3) since this represents a conservative upper limit. Big Rock Point has no gas tanks.

The liquid tanks of concern at Palisades (SIRW, T-90, T-91 and T-2) are all large enough to hold the entire primary coolant volume. This volume (10,900 ft^3) was used in conjunction with the last $\bar{E}$ calculation to derive the total activity in the coolant at the maximum average concentration of 0.1 µCi/gm dose equivalent I-131. If this total activity were in one tank, it would be roughly 1.28 times the design basis quantity curie limits (for liquid releases). This amount, however, does not take credit for the demineralizers and filters that reduce the activity by a factor of at least ten, prior to entry to the tanks. It is extremely unlikely that any tank would contain more than the design basis quantity curie limit and total activity limits for these tanks should not be imposed.

The enclosed Appendix (4) provides the Design Calculation Checklist as required in Corporate Procedure HPS-09.

As agreed upon with the NRC, one additional commitment remains to be fulfilled by Consumers Power Company. A draft Offsite Dose Calculation Manual (ODCM) will be transmitted to the NRC/FRC when completed by October 1982. This ODCM will be commensurate with the Sequoyah model ODCM, except for modifications for plant-specific applications.

Please contact us if you have any questions not addressed in this correspondence.

David J VandeWalle (Signed)

David J VandeWalle
Nuclear Licensing Administrator

CC Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point
NRC Resident Inspector - Palisades

APPENDIX 1

<u>T15A and T15B, WASTE HOLD TANKS</u>

Tanks T15A and T15B are 5000 gallon waste hold tanks. If one of these tanks ruptures, the spillage would be released directly to the environment. If we assume that the tank contains the design basis quantity curie limit and calculate the exposure rate at the surface of the tank, we can limit the curie content of the tank by simply measuring the exposure rate and comparing to the exposure rate obtained at the design basis quantity limit.

Table 1 lists the radionuclides of interest, design basis quantities, and parameters needed in subsequent calculations. Also attached is a data sheet for the waste hold tank.

To determine the photon flux density at the surface of the tank (at the midpoint) the method described in Section 5.8 of "Concepts of Radiation Dosimetry" by Kenneth R. Kase and Walter R. Nelson was used for the no shield case.

The flux density (ϕ) for a right-circular cylinder source is given by the following formula:

$$\phi = \frac{S_v R_0^2 \, F(\theta, b_2)}{2(a + Z)}$$

Where: ϕ = photon flux density (photons/cm^{2}sec)

S_v = source strength (photons/cm^{3}sec)

R_0 = radius of cylinder (cm)

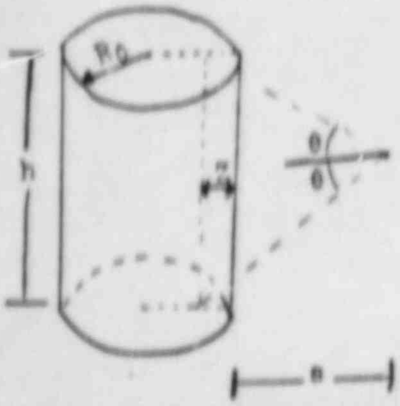
a = distance from cylinder to dose point (cm)

$F(\theta, b)$ = function to approximate the flux from the cylinder as it relates to a line source

Z = self absorption distance (cm)

b_2 = $\mu_s Z$ where μ_s is the total scatter coefficient for water (cm^{-1})

Graphically, we have:



Where: R_0 = 122 cm (4 feet)
a = 0
h = 488 cm (16 feet)
μ_s = 0.074 cm^{-1} from p. 138 of Rad. Health Handbook for 0.905 MeV gammas
b_2 = $\mu_s Z$ = 2.88 from Figures A.21 and A.22 of "Concepts of Radiation Dosimetry"

$F(\theta, b_2)$ = 4E-2 from Figure A.14 of "Concepts of Radiation Dosimetry" (using $\theta = 81^\circ$)

S_V = 6.26E + 5 photons/cm^{3}sec (from Table 1)

The photon flux density at the surface of the tank is then:

$$\phi = \frac{(6.26E + 5 \text{ photons/cm}^3\text{sec})(122\text{cm})^2(4E\text{-}2)}{2(0 + 38.9\text{cm})}$$

$$= 4.79E + 6 \text{ photons/cm}^2\text{sec}$$

Page 132 of the Rad Health Handbook gives a graph of particle fluence to give 1 R/hr. The graph lists 5.8E + 5 photons/cm^{2}sec to give 1 R/hr for 0.905 MeV photons. The exposure rate at the surface of the tank is then:

$$R/hr = \frac{4.79E + 6}{5.8E + 5} = 8.26 \text{ R/hr}$$

The only thing not considered thus far is the shielding provided by the tank wall. This is corrected by the following relationship:

$$D = D_o e^{-\mu\rho t}$$

Where: D = Exposure rate - shielded

D_o = Exposure rate - unshielded

μ = Attenuation coefficient for iron and 0.905 MeV photons = 0.0634 cm^2/g

ρ = Density of iron = 7.86 g/cm^3

t = thickness of wall = ¼" = 0.635 cm

$$D = 8.26\ e^{-(0.0634)(7.86)(0.635)} = 6.02 \text{ R/hr}$$

Therefore, if one waste hold tank contains the design basis quantity curie limit, the contact exposure rate at the tank midpoint is 6.02 R/hr. A reading of 5mR/hr would certainly correspond to a resonable curie limit in the tank. In fact, 1/100 of the limit would read 60 mR/hr.

T-40, CONDENSATE STORAGE TANK

The same methodology is used as for the waste hold tanks. Table 1 data was used with a volume of 25,000 gallons (9.45E + 7 cc) instead of 5000 gallons.

To find the photon flux density we use the following parameters:

$$
\begin{aligned}
R_0 &= 305 \text{ cm (10 feet)} \\
a &= 0 \\
h &= 366 \text{ cm (12 feet)} \\
\mu_s &= 0.074 \text{ cm}^{-1} \\
b_2 &= \mu_s Z = 3.86 \\
F(\theta, b_2) &= 1.3E - 2 \text{ (using } \theta = 74.1^\circ) \\
S_V &= 6.26E + 5 \; (5000/25{,}000) = 1.25E + 5
\end{aligned}
$$

The photon flux density at the surface of the tank is then:

$$\phi = \frac{(1.25E + 5)(305)^2(1.3E - 2)}{2(0 + 52.2)} = 1.45E + 6 \text{ photons/cm}^2\text{sec}$$

The exposure rate at the surface of the tank is:

$$R/hr = \frac{1.45E + 6}{5.8E + 5} = 2.50 \text{ R/hr}$$

The wall thickness of the tank is assumed to be one inch of aluminum. It is probably not this thick and assuming one inch is conservative. The exposure rate after passing through the wall is:

$$R/hr = 2.5\, e^{-(0.0648)(2.699)(2.54)} = 1.6 \text{ R/hr}$$

Where: 2.5 = unshielded exposure rate (R/hr)
0.0648 = attenuation coefficient (cm^2/g)
2.699 = density of aluminum (g/cm^3)
2.54 = wall thickness (cm)

A reading of 5 mR/hr at the midpoint of the condensate storage tank (6 feet above grade) corresponds to a reasonable curie limit in the tank.

TABLE 1

NUCLIDE	ENERGIES (MeV)	YIELD (%)	DESIGN BASIS QUANTITY (Ci)	PHOTONS/cc sec
Cr-51	0.320	9.0	6320.0	1.11E + 6
Mn-54	0.835	100.0	129.0	2.53E + 5
Co-58	0.511	30.0	887.0	2.27E + 6
	0.810	99.0		
	0.865	1.4		
	1.67	0.6		
Fe-59	0.143	0.8	709.0	1.44E + 6
	0.192	2.8		
	1.095	56.0		
	1.292	44.0		
Co-60	1.173	100.0	10.0	3.92E + 4
	1.332	100.0		
Zn-65	0.511	3.4	366.0	3.75E + 5
	1.115	49.0		
Sr-89	0.91	0.009	588.0	1.04E + 2
Sr-90	-	-	61.0	
Ag-110m	0.658	96.0	63.03*	3.66E + 5
	0.68	16.0		
	0.706	19.0		
	0.764	23.0		
	0.818	8.0		
	0.885	71.0		
	0.937	32.0		
	1.384	21.0		
	1.505	11.0		
Sb-124	0.603	97.0	321.2*	1.19E + 6
	0.644	7.0		
	0.72	14.0		
	0.967	2.4		
	1.048	2.4		

TABLE 1 (continued)

NUCLIDE	ENERGIES (MeV)	YIELD (%)	DESIGN BASIS QUANTITY (Ci)	PHOTONS/cc sec
Sb-124 (cont)	1.31	3.0		
	1.37	5.0		
	1.45	2.0		
	1.692	50.0		
	2.088	7.0		
I-131	0.08	2.6	23.9	4.60E + 4
	0.284	5.4		
	0.364	82.0		
	0.637	6.8		
	0.723	1.6		
Cs-134	0.57	23.0	0.829	3.67E + 3
	0.605	98.0		
	0.796	99.0		
	1.038	1.0		
	1.168	1.9		
	1.365	3.4		
Cs-137	0.662	85.0	1.39	2.31E + 3
La-140	0.329	20.0	455.0	1.67E + 6
	0.487	40.0		
	0.815	19.0		
	0.923	10.0		
	1.596	96.0		
	2.53	3.0		
		Σ Yield = 1673.7		8.77E + 6

$$\overline{E} = \frac{\Sigma(E_i)(\text{Yield})}{\Sigma\ \text{Yield}} = \frac{1514.67}{1673.7} = 0.905$$

$$\text{Limit} = \frac{8.77E+6}{14} = 6.26E+5$$

Notes

1. Nuclides were obtained from the 1981 Semiannual Effluent Reports for Big Rock Point. These were all the nuclides identified in liquid effluents.

TABLE 1 (concluded)

2. Volume of waste hold tank is 5000 gallons = 1.89E + 7 cc.

3. Energies and yields are from the Radiological Health Handbook (1970).

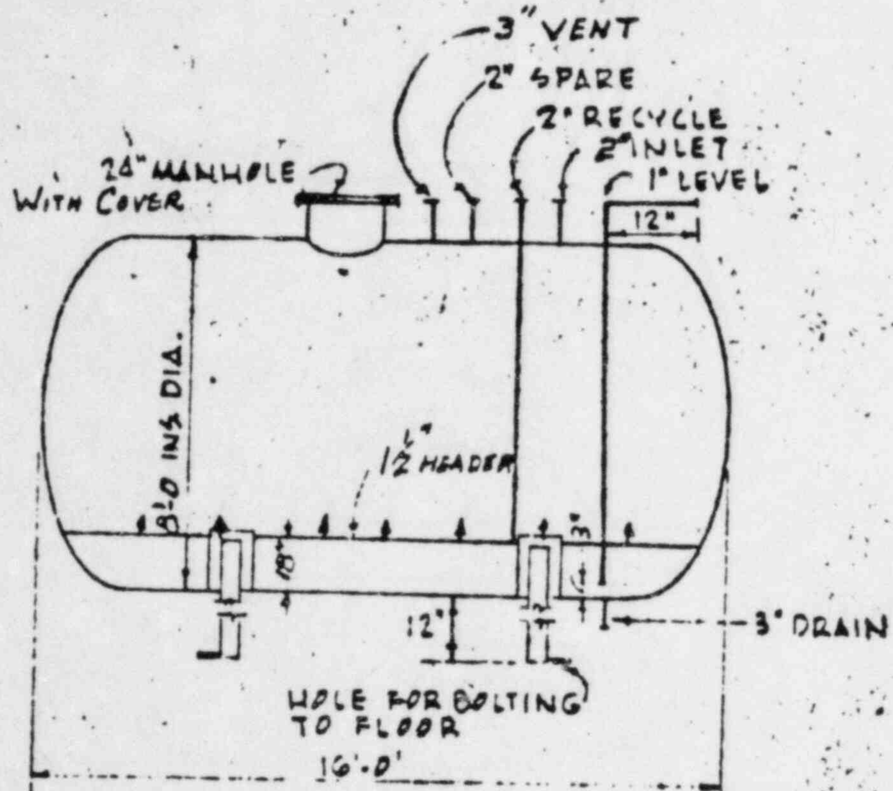
4. Design Basis Quantities are from our submittal to the NRC dated November 13, 1978. Those with an asterisk are from memo SBB 82-03.

5. $$\text{Photons/cc sec} = \frac{(\text{Design Basis Quantity}_i\text{Ci})(\Sigma\%\ \text{yield}_i)(10^6\ \mu\text{Ci/Ci})(3.7\text{E} + 4\ \text{photons/sec}\ \mu\text{Ci})}{(100)(1.89\text{E} + 7\text{cc})}$$

Where: Design Basis Quantity_i = Design Basis Quantity for Radionuclide i

Σ % Yield_i = Summation of % Photon yields for Radionuclide i

6. The design basis quantity listed for each radionuclide would give 10CFR50, Appendix I, dose limits as if only that radionuclide were present. In other words, if the sum of all 14 radionuclides were used, the total dose would be 14 times 10CFR50, Appendix I limits. Therefore, the tank would contain the design basis quantity curie limit, if one fourteenth of each radionuclide were in the tank. This is why the total photons/cc sec is divided by 14.



NOZZLE AND BOSS LOCATIONS ARE SCHEMATIC ONLY. FINAL ARRANGEMENT WILL BE MARKED ON VENDOR'S DRAWINGS.

SHELL: MTL/THICKNESS	ASTM A-283-GR-C/1/4 MIN.	OPERATING PRESS(PSIG)/TEMP(F)	ATM/150
INTERNALS (TRAYS ETC.)		DESIGN PRESS(PSIG)/TEMP(F)	ATM/200
INSUL CLIPS?		S.V. SETTING/HYDRO TEST PRESS	NONE/5 PSIG
		CORROSION ALLOWANCE	3/16" (MIN.)
HEADS (DISHED)	ASTM A-283 GR-C/1/4 MIN.	FLANGE RATING/FACING (ASA STD)	150#/F.F.
NOZZELS -		FITTINGS RATING	SCH. 80 PIPE
		JOINT EFF. SHELL	-
ALL NOZZELS SHALL BE 6" LONG WITH STANDARD 150# ASA FLAT FACED FLANGE EXCEPT LEVEL NOZZLE WHICH SHALL BE SOCKET WELD		STRESS RELIEVE?/RADIOGRAPHY	NO/NO
		WIND LOADING	NONE
		EARTHQUAKE BRACING	0.025
		STD. APPURTENANCE DWGS.	
ALL TANKS SHALL HAVE LIFTING LUGS			
SPECIFIC GRAVITY OF SOLN. 1.0		APPLICABLE CODES & SPECS.	UNDERWRITERS LAB STANDARDS FOR ABOVE GROUND STORAGE TANKS UL-142
MANUFACTURER	NILES STEEL TANK CO.		
SERIAL NO		WELDING SPECIFICATION	
WEIGHT NET/FLOODED			
(PO NO) (REQ'N NO.)	3159-M-32		
SPEC NO	3159-M-32		
COST CODE	3.045-83% + 3.044-17%		

MARK "WASTE HOLD TANKS #1 & #2"

REV	DESCRIPTION	ENG	DR	CHK	SUPV	MATL	APPROVAL	DATE
A	DWGS. NO. CHGD. ADDED DRAWINGS - CHANGED NOZZLES - ADD MANHOLE COVER		SKC					9/25/66
0								

BECHTEL CORPORATION
POWER DIVISION ENGINEERING

TANK DATA SHEET

WASTE HOLD TANKS (2)

JOB NO 3159
FORMER M-32
KEY 0740A90032
SHEET 4 OF 59
REV A

Information Copy Only

APPENDIX 2

BASIS FOR OMISSION OF Fe-55 ANALYSIS REQUIREMENTS

Iron-55 production can be shown to give rise to activity levels not exceeding a factor of 38 times that of Iron-59 in reactor vessel cc ponents or primary coolant of an operating reactor:

$$\frac{\text{Activity (Fe-55)}}{\text{Activity (Fe-59)}} = \frac{NV_o\sigma_4 N_4(1-e^{-\lambda t})}{NV_o\sigma_8 N_8(1-e^{-\lambda t})} = \frac{\sigma_5 N_4}{\sigma_9 N_8} = 38 \text{ at } t = \infty$$

Where: NV_o = thermal neutron fluence rate (n/sec)

σ_4 = thermal cross section Fe-54 (2.3 barns) to produce Fe-55

σ_8 = thermal cross section Fe-58 (1.16 barns) to produce Fe-59

N_4 = Atoms Fe-54 (5.8% total Fe)

N_8 = Atoms Fe-58 (0.3% total Fe)

$1-e^{-\lambda t}$ = equilibrium fraction (1.0 at $t = \infty$)

Iron-59 data for radioactive liquid discharges over the past 6 to 9 years from the Palisades and Big Rock Point plants have been reviewed (see Table I attached) and the percentage contribution of activity due to Iron-59 has been found to average 0.42% for Palisades and 0.62% for Big Rock Point. These percentages agree well with primary coolant analyses which indicate that Fe-59 is in the range of 0.3% to 0.9% of activity from coolant, based upon radionuclides with half-lives greater than approximately 10 days (those typically found in effluent releases). In all cases, tritium and dissolved gasses have been excluded from the calculations. Inclusion of tritium in the total activity released reduces the percentage radioactivity contribution from Fe-59 to well below 0.1% and Fe-55 to below 3.8% in all cases.

On a basis of Maximum Permissible Concentration (MPC per 10CFR20, Appendix B, Table II), Fe-55/Fe-59 (soluble) = $8 \times 10^{-4}/6 \times 10^{-5}$ = 13.3 and Fe-55/Fe-59 (insoluble) = $2 \times 10^{-3}/5 \times 10^{-5}$ = 40. Thus, Iron-55 is from 13.3 to 40 times less limiting than Iron-59 from an MPC (or dose) standpoint. Iron-59 has been observed only in insoluble form where separation by solubility has been performed (primary coolant $\bar{E}$ analyses). This indicates that the maximum quantity of Iron-55 (38 times the quantity of Iron-59) provides an MPC-fraction essentially equal to that of Iron-59 in effluent releases 38 x Fe-59/(40)(MPC Fe-59) = 0.95 x Fe-59 MPC fraction.

During the periods when Fe-59 was the highest percentage of released activity at Big Rock Point and Palisades (approximately 1% - see Table I), the fraction of total MPC accounted for by Fe-59 was 0.25%. Since we have shown above that in the worst case (infinite activation time), the MPC-fraction of Fe-55 approaches that of Fe-59, the maximum contribution of Fe-55 to the total MPC-fraction released also is 0.25%, calculated in accordance with Note 5 to Appendix B of 10CFR20. This note states: "... a radionuclide may be considered as not present in a mixture if (a) the ratio of the concentration of that radionuclide in the mixture (C_A) to the concentration limit for that radionuclide specified in Table II of Appendix B (MPC_A) does not exceed 1/10 (i.e. $C_A/MPC_A \leq 1/10$) and (b) the sum of such ratios for all radionuclides considered as not present in the mixture does not exceed 1/4 i e. ($C_A/MPC_A + C_B/MPC_B .... + \leq 1/4$)." Since it is shown that Fe-55 is less than approximately 0.25% of the total MPC-fraction of the releases, there should be no further analysis or reporting required for Fe-55.

TABLE I

C: Fe-59 AND TOTAL ACTIVITY FROM SEMIANNUAL EFFLUENT REPORTS

	PAL Fe-59	PAL Total	PAL %	BRP Fe-59	BRP Total	BRP %
1973	0.055	23.3	0.24			
	0.068	27.8	0.24			
1974	0.00059	3.08	0.02			
	0.00077	5.27	0.015			
1975	0.0027	3.28	0.082			
1976	0.0016	0.40	0.40	0.00114	0.566	0.20
	0.0018	0.43	0.42	-	0.205	-
1977	ND	0.088	-	-	0.103	-
	ND	0.0048	-	-	0.392	-
1978	ND	0.059	-			
	0.00036	0.038	0.95	0.00029	0.149	0.20
1979	0.00022	0.028	0.79	0.0050	0.573	0.87
	0.0011	0.10	1.10	-	0.33	-
1980	ND	0.0041	-	-	0.695	-
	ND	0.0046	-	0.00206	0.225	0.92
1981	ND	0.0046	-	0.00237	0.264	0.90
			0.426 ± 0.39			0.618 ± 0.38

APPENDIX 3

ACTIVITY IN PALISADES' GAS TANKS

The last Palisades $\overline{E}$ calculation (dated June 6, 1981) was used to obtain the maximum primary coolant gaseous activity at the technical specification limit of 1.0 μCi/gm of dose equivalent I-131. The iodine activities June 6, 1981 were obtained from Chemistry Form F6.2.2 (PCS Activity Analysis) for the week of June 1, 1981 to June 7, 1981. These activities were converted to dose equivalent I-131 and compared to the 1.0 μCi/gm limit. The results are provided in Table 1. Since the actual dose equivalent I-131 is 11.7 times below the technical specification limit, all the other nuclides were multiplied by this amount to get the activities at the limit (1.0 μCi/gm dose equivalent I-131 will be reached before $100/\overline{E}$).

Table 2 is a tabulation of the total gaseous activity in the primary coolant at the technical specification limit and the fraction of the design basis quantity for each radionuclide. The total fraction of the design basis quantities is from Table 2, 25.8%. Curie limits on the gas tanks should, therefore, not be imposed.

ACTIVITY IN PALISADES' LIQUID TANKS

The same type of methodology is applied to the liquid tanks as the gas tanks. During the past 10 years of operation of the Palisades facility, the dose equivalent I-131 concentration in the primary coolant has rarely been above 0.1 μCi/gm (except during occasional iodine spikes). Only during the past year have we seen activities approaching 0.1 μCi/gm.

The actual activity concentrations in the June 6, 1981 $\overline{E}$ calculation were multiplied by 1.17 to reach the maximum average dose equivalent I-131 concentration of 0.1 μCi/gm. The filtrate and crud activities were compared to the design basis quantity limits and added to give a total fraction of design basis quantity (see Table 3). If the entire primary coolant volume at the maximum average activity were contained in one tank, it would represent 128% of the design basis quantity curie limit.

It is extremely unlikely that one tank would contain this much activity, especially considering the removal of activity by filters and demineralizers prior to entry to the tank. This conservative assumption alone would account for at least a factor of 10 reduction.

In view of the above, curie limits on Palisades' liquid tanks should not be imposed.

TABLE 1

Radionuclide	Actual Conc. ($\mu Ci/ml$)[1]	X	DCF[2] ($rad/\mu Ci$)	=	DEF[3] (rad/ml)
I-131	0.0338		1.48		0.05002
I-132	0.1397		0.0535		0.00747
I-133	0.1150		0.4		0.0460
I-134	0.1965		0.025		0.004913
I-135	0.1467		0.124		0.01819

Dose equivalent $\Sigma = 0.127$ rad/ml

$$\text{Dose Equivalent I-131} = \frac{0.127}{1.48} = 0.0855\ \mu Ci/ml$$

$$\text{Factor to achieve Tech. Spec. limit} = \frac{1.0}{0.0855} = 11.7$$

Notes:

(1) From Chemistry Work Sheet F6.2.2 for June 6, 1981

(2) Dose conversion factors from TID-14844

(3) Dose equivalency factors

TABLE 2

Radionuclide	Actual Conc. ($\mu Ci/ml$)(1)	Half Life (hr)	Total Activity (Ci)(2)	DBQ (Ci)(3)	Ci/DBQ(4)
Ar-41	2.087E - 2	1.83	1.278E + 0	2,830	4.52E - 4
Kr-85	5.687E - 4	9.43E + 4	4.60E + 0(5)	19,800	2.33E - 4
Kr-85m	3.151E - 2	4.4	2.75E + 1	16,600	1.66E - 3
Kr-87	3.840E - 2	1.27	3.18E - 1	2,560	1.24E - 4
Kr-88	4.597E - 2	2.8	1.363E + 1	1,700	8.02E - 3
Kr-89	-	-	-	-	-
Xe-131m	9.014E - 4	283.2	5.058E + 0	49,700	1.02E - 4
Xe-133	3.638E - 1	126.5	2.76E + 3(5)	70,900	3.89E - 2
Xe-133m	8.729E - 3	54.2	4.327E + 1	24,800	1.74E - 3
Xe-135	1.140E - 1	9.14	2.652E + 2	12,800	2.07E - 2
Xe-135m	1.294E - 2	0.26	9.65E - 7	27,778(6)	-
Xe-137	-	-	-	-	-
Xe-138	-	-	-	-	-
I-131	1.0E + 0	192	2.957E - 1(7)	1.59	1.86E - 1
					Σ 2.58E - 1 or 25.8%

Notes:

(1) Actual concentrations are from June 6, 1981 Palisades $\overline{E}$, where the filtrate and crud were added to get the total.

(2) Total gaseous activity is calculated by the following:

$$\text{Total Activity (ci)} = (\text{Actual Conc.}, \mu Ci/ml)(10{,}900\ ft^3)(2.832E + 4\ ml/ft^3)(1.6)(11.7)(e^{-\lambda t})(10^{-6} Ci/\mu Ci)$$

Where: 10,900 ft^3 = volume of primary coolant, FSAR, Table 4.1.

1.6 = correction in total gas activity in primary coolant (see JLB 82-28, Palisades-Calculation of Steam Generator Leak Rate).

TABLE 2 (concluded)

Notes:

11.7 = increase to achieve Tech. Spec. limit concentrations (see Table 1).

$e^{-\lambda t}$ = decay factor in the Degas process. $\lambda = 0.693/T_{½}$, t = 12 hours. The degassing of the primary system can be done in as short a time as 12 hours.

(3) DBQ = Annual Design Basis Quantities.

(4) Ci/DBQ - Fraction of design basis quantity for each radionuclide.

(5) In addition to the calculation of Note (2) above, the activity in the pressurizer is also added for Kr-85 and Xe-133. Only these radionuclides were considered since their abundance and longer half-lives allows the accumulation in the pressurizer. The additional activity was calculated as follows:

$$Ci = (\text{Actual Conc.}, \mu Ci/ml)(10)(700\ ft^3)(2.832E + 4\ ml/ft^3)(11.7)(e^{-\lambda t})(10^{-6} Ci/\mu Ci)$$

Where: 10 = gaseous activity in the pressurizer is conservatively assumed to be a factor of ten higher than in the primary coolant.

700 ft^3 = gas volume of the pressurizer FSAR, Table 4.8.

(6) Taken from memo SBB 81-19.

(7) I-131 activity was calculated as follows:

$$Ci = (1/1000\ \mu Ci/ml\ \text{maximum})(10{,}900\ ft^3)(2.832E + 4\ ml/ft^3)(e^{-\lambda t})(10^{-6} Ci/\mu Ci)$$

Where: 1/1000 = the fraction of iodine that evolves from the primary coolant in the degassing process (partition coefficient).

TABLE 3

FILTRATE (DEGASSED)

Radionuclide	Actual Conc. (μCi/ml)	Total Activity (Ci)(1)	DBQ (Ci)(2)	Ci/DBQ(3)
F-18	2.909E - 2	6.45E - 1	(5)	-
Na-24	2.307E - 3	5.11E - 2	551.5(6)	9.27E - 5
Co-58	7.725E - 5	1.76E - 3	1190	1.47E - 6
Rb-88	9.543E - 2	2.11E + 0	4335.3(6)	4.87E - 4
Rb-89	4.374E - 2	9.69E - 1	(5)	-
Mo-99	6.526E - 3	1.44E - 1	9020	1.60E - 5
Tc-99m	2.968E - 3	6.58E - 2	145,000	4.53E - 7
I-131	1.0E - 1	1.89E + 0(4)	30.9	6.12E - 2
I-132	-	-	755	-
I-133	-	-	254	-
I-134	-	-	1234.6(6)	-
I-135	-	-	353	-
Cs-134	2.042E - 4	4.56E - 3	1.06	4.30E - 3
Cs-136	6.505E - 5	1.40E - 3	7.61	1.40E - 2
Cs-137	5.522E - 4	1.23E - 2	1.80	6.83E - 3
Cs-138	2.282E - 1	5.05E + 0	1298.7(6)	3.89E - 3
La-140	2.829E - 5	5.85E - 4	390	1.50E - 6
				Σ 7.82E - 2

CRUD (PARTICULATE)

Radionuclide	Actual Conc. (μCi/ml)	Total Activity (Ci)	DBQ (Ci)	Ci/DBQ
Cr-51	8.672E - 4	1.92E - 2	6730	2.85E - 6
Mn-54	2.313E - 5	5.12E - 4	161	3.18E - 6
Mn-56	2.097E - 4	4.68E - 3	1612.9(6)	2.90E - 6
Fe-59	2.640E - 5	5.85E - 4	265	2.21E - 6
Co-57	1.122E - 6	2.48E - 5	683.4(6)	3.63E - 8
Co-58	1.067E - 3	2.36E - 2	1190	1.99E - 5
Co-60	2.481E - 5	5.50E - 4	6.67	8.25E - 5

TABLE 3 (continued)

CRUD (PARTICULATE)

Radionuclide	Actual Conc. (μCi/ml)	Total Activity (Ci)[1]	DBQ (Ci)[2]	Ci/DBQ[3]
Ni-56	2.186E - 6	4.84E - 5	(5)	-
Rb-88	4.840E - 4	1.08E - 2	4335.3[6]	2.48E - 6
Zr-95	2.573E - 5	5.70E - 4	581	9.80E - 7
Zr-97	2.986E - 5	6.61E - 4	3232.8[6]	2.05E - 7
Nb-95	3.900E - 5	8.63E - 4	855	1.01E - 6
Nb-97	4.018E - 5	8.89E - 4	4316.6[6]	2.06E - 6
Mo-99	4.414E - 5	9.77E - 4	9020	1.08E - 7
Tc-99m	4.061E - 5	9.00E - 4	145,000	6.20E - 9
Tc-101	-	-	(5)	-
Ru-103	1.420E - 5	3.15E - 4	1094.9[6]	2.88E - 7
Sb-122	3.777E - 6	8.37E - 5	(5)	-
Te-132	2.861E - 5	6.33E - 4	29.2	2.16E - 5
Cs-137	2.234E - 6	4.95E - 5	1.8	2.75E - 5
Cs-138	6.713E - 4	1.49E - 2	1298.7[6]	1.14E - 5
Ba-139	4.034E - 4	8.89E - 3	64,377.7[6]	1.38E - 7
La-140	9.338E - 6	2.07E - 4	390	5.31E - 7
Ce-141	6.858E - 6	1.52E - 4	8108.1[6]	1.87E - 8
Ce-143	1.887E - 5	4.18E - 4	(5)	-
Ce-144	4.778E - 6	1.06E - 4	1923.1[6]	5.50E - 8
W-187	4.510E - 5	9.99E - 4	50.5[6]	1.98E - 5
Re-188	2.543E - 5	5.63E - 4	(5)	-
Np-239	6.311E - 5	1.40E - 3	3750	3.74E - 7
Ni-65	9.763E - 5	2.11E - 3	5154.6[6]	4.08E - 7
Sr-85	9.801E - 8	2.16E - 6	(5)	-
				Σ 2.03E - 4

Total fraction of Design Basis Quantity (Filtrate + Crud) = (7.82E - 2) + (2.03E - 4) = 7.84E - 2 for 5000 gallons

Fraction of Design Basis Quantity for entire primary coolant volume $= (7.84E - 2)\left(\frac{3.09E + 8}{1.8926E + 7}\right) = 1.28E + 0$ or 128%

TABLE 3 (concluded)

Notes:

(1) Total activity is calculated as follows:

$$Ci = (\text{Actual Conc. } \mu Ci/ml)(1.8926E + 7\ ml)(1.17)(10^{-6} Ci/\mu Ci)$$

Where: 1.8926E + 7 = ml in 5000 gallons

1.17 = increase in activities to reach the maximum average of 0.1 μCi/gm dose equivalent I-131

(2) DBQ = Design Basis Quantity

(3) Ci/DBQ = Fraction of design basis for a 5000 gallon volume. The total primary coolant volume is 3.09E + 8 ml and the total fraction of design basis quantity for the primary coolant is 1.28, as listed above.

(4) This quantity is for dose equivalent I-131 which considers the other iodine radionuclides. The maximum average of 0.1 μCi/gm is assumed.

(5) Design Basis Quantities were not found for these radionuclides, however, their contribution is expected to be insignificant.

(6) Design Basis Quantities are from memo SBB 81-15.

APPENDIX 4

DESIGN CALCULATION CHECKLIST	REF DOCUMENT	COMMENTS
The design calculation input shall include items such as:		
A. Basic functions of each structure, system and component and/or purpose of calculation referencing same.	PALISADES FSAR,	
B. Performance requirements for: 1. system, or 2. calculation	PALISADES $\bar{E}$ CALCULATION DATED 6/6/81	
C. Documents 1. codes 2. standards 3. regulatory requirements	10CFR20, App. B, TABLE II 10CFR50 App. I	ALSO, SEMIANNUAL EFFLUENT REPORTS FOR BIG ROCK POINT.
D. Design or hypothetical conditions: 1. FSAR description 2. Regulatory Guide 3. Standard Review Plan 4. other	RADIOLOGICAL HEALTH HANDBOOK, 1970	"CONCEPTS OF RADIATION DOSIMETRY" BY KASE AND NELSON, SECTION 5.8
E. Mitigating factors	IOM, SAB82-03 AND 81-19 ON DESIGN BASIS QUANTITIES	ALSO, SUBMITTAL TO NRC DATED 11/13/78 ON DESIGN BASIS QUANTITIES.
1. backup systems 2. operator action 3. procedural requirements 4. Tech Spec requirements 5. commitments	TECH SPEC LIMITS ON PRIMARY COOLANT ACTIVITY AT PALISADES	DEMINERALIZERS AND FILTERS IN THE RADWASTE SYSTEM.
F. Environmental factors 1. pressure, temp, humidity 2. corrosion 3. meteorology 4. dilution 5. concentration factors 6. duration of exposure		