Detroit
Edison
2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8000

September 18, 1981

EF2 - 54,825

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. L. L. Kintner
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington DC 20555

Reference: Enrico Fermi Atomic Power Plant-Unit 2

Subject: Equipment Environmental Qualification - Radiation Profiles.

Dear Mr. Kintner:

Per your request, the following documents are enclosed:

1. General Electric letter TDEC-4034 dated 8/25/81 and document number 22A3019 Rev.1; Radiation entries in FSAR table 3.11.5 for normal plant operation.

2. Sargent & Lundy letters SLM(NI)-223 dated 8/12/81 and SLM(NI)-207 dated 7/8/81 and attachment, Equipment Qualification Dose Analysis during Post-LOCA accident.

The above information was requested during a telephone conversation on September 15, 1981 between the NRC (Frank Akstulewicz) and Detroit Edison reqresentatives (Dick Beaudry, Lou Bregni and Len Fron).

Should you require any further information, please let me know.

Very truly yours,

W. F. Colbert

W. F. Colbert
Technical Director
Enrico Fermi Unit 2 Project

WFC/QHD/mb

B001
s
1/1

GENERAL ELECTRIC

NUCLEAR POWER

SYSTEMS DIVISION

GENERAL ELECTRIC COMPANY, 175 CURTNER AVE., SAN JOSE, CALIFORNIA 95125
MC 391, (408) 925-2588

August 25, 1981
IDEC-4034

Mr. W. F. Colbert, Project Engineer
Enrico Fermi 2 Project
The Detroit Edison Company
Documentation Control - Room 361
2000 Second Avenue
Detroit, MI 48226

Attention: Mr. L. Sherman

Gentlemen:

SUBJECT: FERMI 2 RADIATION ENVIRONMENTAL CONDITIONS

The radiation environmental condition entries in FSAR Table 3.11-5 were extracted from GE Specification 22A3019, BWR Equipment Environmental Requirements. The values in this specification were derived by a combination of analysis and extrapolation of actual operating plant measurements.

Very truly yours,

for G. Howard
C. M. Johnson
Project Manager
Enrico Fermi 2 Project

CMJ:sem/2I

cc: T. J. Evans, GE Southfield
F. Gregor, Edison
S. K..emer, Edison
R. Pratt, GE Site
T. Mintun, GE Site

GENERAL ELECTRIC

NUCLEAR ENERGY DIVISION

S-3

Document No. 22A3019 Rev. 1

General Electric Class

TRANSMITTAL

PROJECT(S) ENRICO FERMI 2 & 3

TITLE OF DOCUMENT BWR EQUIPMENT ENVIRONMENTAL REQUIREMENTS

TYPE OF DOCUMENT:
[] PURCHASE SPECIFICATION
[X] SYSTEM DESIGN SPECIFICATION
[] INSTALLATION SPECIFICATION
[] ____

REPLACES DOCUMENT NO. ____

PIPING OR COOLING SYSTEM INVOLVED ____

RESPONSIBLE ENGINEER VM DOCHEZ ISSUED BY EA HARTMAN DATE JAN 31 197?

REFERENCES

MASTER PARTS LIST (MPL) NOS. A61-4270

SPECIFICATIONS ____

DRAWINGS ____

OTHER ____

REVISION RECORD

REVISED PER (ECA, ECN, ETC.) NE 35496

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GENERAL ELECTRIC
NUCLEAR ENERGY DIVISION
ATOMIC POWER EQUIPMENT DEPARTMENT
San Jose, California

DOCUMENT NO. 22A3019 REV. 1

APPLICATION

MPL A61-4270

[X] SPECIFICATION [ ] DRAWING

TYPE DESIGN

DOCUMENT TITLE BWR EQUIPMENT ENVIRONMENTAL REQUIREMENTS

REVISIONS			
1	Per ECN NE35496. Sheets 5,6,7,8,9, 10,11,12,14. Revisions were identified with a spade. (♠) VMDochez 1-24-73 [illegible] 1-30-73 JAN 31 1973		

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REVISION STATUS SHEET

NED 805 (3-69)

1. SCOPE

1.1. This document specifies indoor environmental data to be used for design of equipment supplied by the Atomic Power Equipment Department (APED).

1.2. Seismic requirements, and vibration levels are not included in the scope of this document.

2. APPLICABLE DOCUMENTS, CODES, AND STANDARDS

2.1. General Electric Company Doc s

2.1.1. This specification, or applicable portions, thereof, represent the controlling environmental data for use in design of the specific equipment supplied by APED. If there is a conflict with respect to environmental data between this specification and other design documents, the requirements of this specification shall govern.

2.2. Codes and Standards

2.2.1. The following documents are to be used in conjunction with this specification to the extent specified herein.

a. Atomic Energy Commission (AEC) - Criterion 1 of the AEC General Design Criteria, 10CFR50, Appendix A.

3. DESCRIPTION

3.1. Incorporation of appropriate environmental design data is necessary to ensure proper functional performance of the system or equipment during all design modes of operation.

4. REQUIREMENTS

4.1. Normal Conditions

4.1.1. Normal conditions are defined as those conditions existing during routine plant operations. Environmental requirements stated i Tables as Normal or Operating are those which shall be used for design, and represen ormal, maximum and minimum expected conditions that may exist during routine plant eration.

4.1.2. Tables summarize the enviromental conditions which shall be used for component or system design within the plant locations stated. All components shall be designed to operate under the normal conditions.

4.2. Abnormal Conditions

4.2.1. Abnormal enviromental conditions are defined as those which deviate from the conditions described in Paragraph 4.1, preceding. The most significant abnormal condition is the environment during and following postulated design basis accidents. Other ambient conditions, including small continuous steam leaks which generate high temperature and/or high humidity, and test or operator-controlled conditions, shall also be considered.

4.2.2. Essential components and safety systems shall be designed to operate or be in a fail-safe condition, as given in Section A, II-III of the following tables. Essential components are those which are essential to the prevention of accidents which could affect the public health and safety or mitigate their consequences, according to the definition and interpretation of Criterion 1 of the AEC General Design Criteria, 10CFR50, Appendix A.

4.3. Tables

4.3.1. The following tables are divided into Sections A and B. Section A defines the pressure, temperature, and humidity environmental conditions. Section B defines the radiation environmental conditions.

4.3.1.1. Section A is further subdivided into three parts: Section A-I includes all equipment operating under normal conditons. Section A-II and A-III delineate the abnormal conditions for essential components. The tables in Section A-II and A-III for essential components represent an envelope of abnormal conditions in which the systems or components are required to be functional or in a fail-safe condition, as noted. The specified envelope is not based on one specific design basis accident, but on all postulated accidents relevant to this envelope.

4.4. Drywell Zones

4.4.1. A diagram follows (Figure 1) showing typical drywell zone locations within primary containment.

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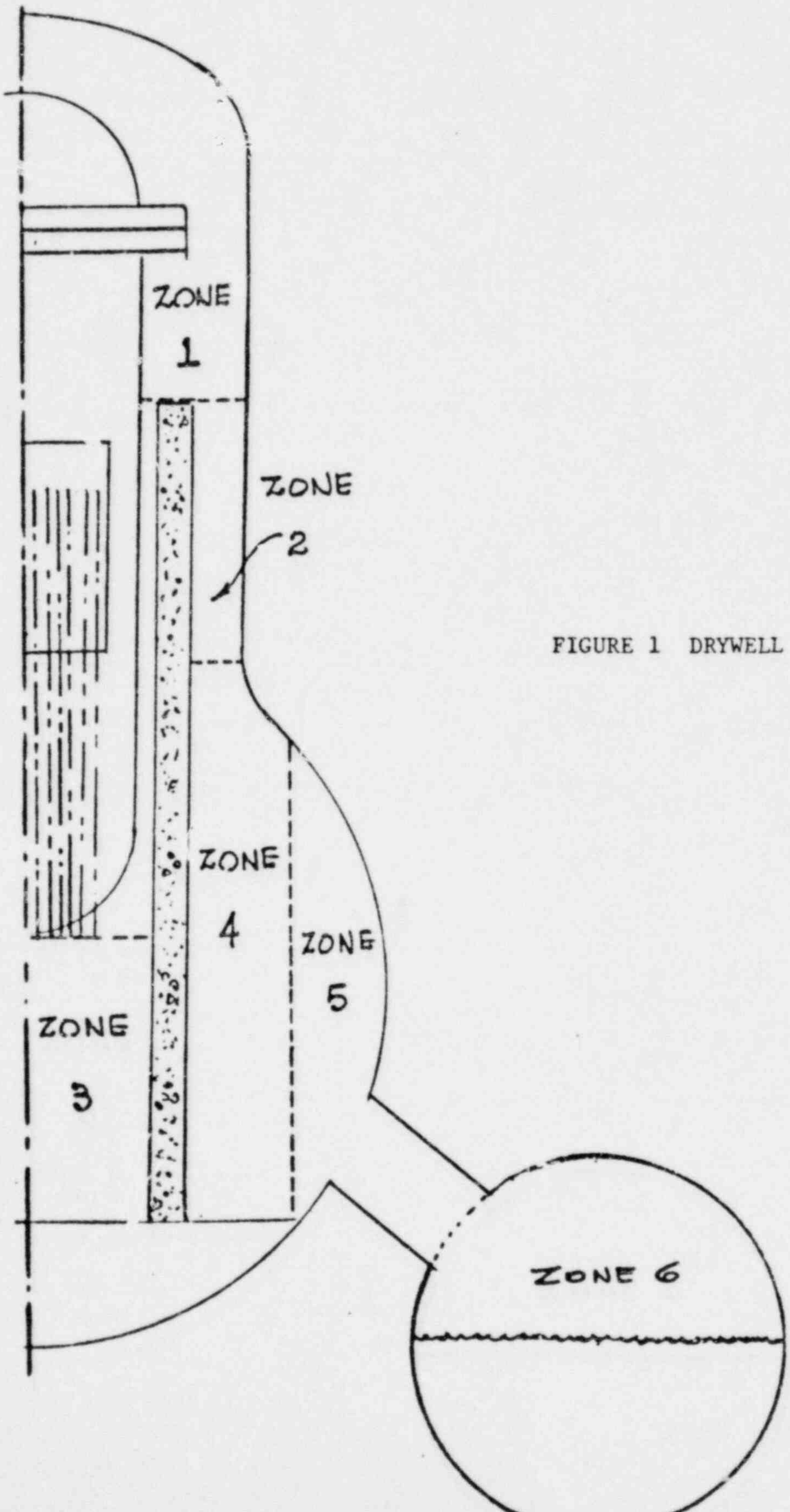

FIGURE 1 DRYWELL ZONES

Section A-I PRESSURE, TEMPERATURE, RELATIVE HUMIDITY ENVIRONMENTAL CONDITIONS

	Normal Conditions - Plant Operating		
Area	Pressure as Noted	Temperature °F	Relative Humidity %
I. Primary Containment (Not otherwise noted) (1)	(-)0.5 to 2.0 psig	135° Average -- Minimum 150° Maximum	40-55% Normal 90% Maximum -- Minimum
Vicinity Recirculation Pump Motors - Zone 4	Same as above	128° Average -- Minimum 135° Maximum	Same as above
Area Beneath RPV - Zone 3	Same as above	135° Average 100° Minimum (5) 165° Maximum (4)	Same as above
II. Reactor Building (Not otherwise noted)	Range from (-)0.10" to (-) 1.0" Water gage, static pressure	70° Normal 104° Maximum 40° Minimum	40% Normal 90% Maximum 20% Minimum
Reactor Building Standby Liquid Control Area	Same as above	104° Maximum 70° Minimum	Same as above
HPCI, RCIC Equipment Area	Same as above	70° Normal 104° Maximum (6) 60° Minimum	Same as above (6)
Core Spray and RHR Equipment Area	Same as above	70° Normal 104° Maximum (6) 40° Minimum	Same as above (6)

(Notes are given at end of table)

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Area	Normal Conditions - Plant Operating		
	Pressure as Noted	Temperature °F	Relative Humidity %
II. Reactor Building (con.) Steam Tunnel	Range from (-)0.10" to (-)1.0" Water gage, static pressure	70° Normal 130° Maximum 40° Minimum	40-50% Normal 90-98% Maximum 20% Minimum
III. Turbine Building (3)	Range 0.0" to (-) 0.25" water gage static pressure	70° Normal Winter 120° Maximum (Non-Elec) 40° Minimum 90° Normal (Summer) 104° Maximum (Electrical)	40% Normal 90% Maximum 20% Minimum
IV. Radwaste Building (3)	Range 0.0" to (-) 0.25" water gage static pressure	70° Normal 104° Maximum 40° Minimum	40% Normal 90% Maximum 20% Minimum
Radwaste Building Equipment Cells	0.0" to (-) 0.5" water gage static pressure	70° Normal 120° Maximum 40° Minimum	40% Normal 90% Maximum 20% Minimum
V. Control Room	Range 0.10" to 1.0" water gage static pressure	60°-90° Normal 120° Maximum 40° Minimum	40-50% Normal 60% Maximum (7) 10% Minimum

Notes:

(1) Primary containment atmosphere during normal operation may be inerted with 96 percent nitrogen, 4 percent oxygen.

(2) Whenever the residual heat removal and core spray motor and the emergency core cooling system are running, during test periods area space coolers may be required to maintain the ambient temperature listed.

(3) Components located in turbine building or radwaste building required to operate under abnormal conditions, if any, should be designed for equivalent conditions as shown for reactor building.

(4) During loss of offsite power, and other emergencies, except during Design Basis Accident temperature of and area underneath the reactor pressure vessel will be maintained at 165°F or lower for up to 30 minutes.

(5) The same minimum temperature (100°F) shall apply inside base of the shield wall. Air velocity over vessel insulation and exposed vessel parts shall be approximately 6 ft/sec.

(6) The maximum temperature and humidity will occur simultaneously in these spaces less than 1 percent of the time.

(7) During HVAC equipment failure conditions Relative Humidity may approach 90° percent for 100 hours, but 120°F would not occur simultaneously.

ESSENTIAL EQUIPMENT

Section A-II

PRESSURE TEMPERATURE, RELATIVE HUMIDITY, ENVIRONMENTAL CONDITIONS

Inside Primary Containment - Abnormal Conditions

Components must be operable under the following conditions

Condition	Component						
1	Core spray injection check valve LPCI-RHR injection check valve Reactor shutdown cooling suction Valve including operator and cable Relief valve including operator and cable Vessel level indicator Structural components (e.g. loop restraints, vessel skirt, etc.)	Temperature Pressure (1) Rel. humidity Duration (3)	340°F -2 to 56psig(2) 100 % 45 sec (3)	340°F -2 to 35psig 100 % 3 hours (3)	320°F -2 to 35psig 100 % 6 hours	250°F) to 25psig 100 % 1 day (3)	200°F 0 to 20 psig 100 % 100 days
2	Feedwater Check Valve HPCI steam line isolation valve including operator and cable RCIC steam line isolation valve including operator and cable Reactor Water cleanup suction valve including operator and cable Reactor water sample line valve including operator and cable Lines 2 inches and smaller (Isolation) Valves, Operators, Cabling) Cables to intermediate range monitors and process radiation monitor Reactor vessel head spray isolation valve including operator and cable	Temperature Pressure Rel. humidity Duration	340°F -2 to 56psig 100 % 45 sec	340°F -2 to 35psig 100 % 3 hours	320°F -2 to 35psig 100 % 6 hours		
3	Main steam isolation valve including operator and cable Main steam drain isolation valve including operator and cable Standby liquid control injection check valve	Temperature Pressure Rel. humidity Duration	340°F -2 to 56psig 100 % 45 sec	340°F -2 to 35psig 100 % 1 hour			
4	Recirculation valves (main valves, bypass valves, equalizer valve) including operators and cables	Temperature Pressure Rel. humidity Duration	310°F -2 to 56psig 100 % 45 sec	285°F -2 to 35psig 100 % 30 min			

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ESSENTIAL EQUIPMENT

Section A-II (cont.) PRESSURE TEMPERATURE, RELATIVE HUMIDITY, ENVIRONMENTAL CONDITIONS

♠ Inside Primary Containment - Abnormal Conditions

Condition	Component	Valves not required to be operable but must not fail open under the following conditions				
5	Feedwater check valve HPCI steam line isolation valve including operator and cable RCIC steam line isolation valve including operator and cable Recirculation valves (main valves bypass valves, equalizer valve) including operators and cables Reactor vessel head spray isolation Valve including operator and cable Reactor water cleanup suction valve including operator and cable Reactor water sample line valve including operator and cable Lines 2 inches and smaller (isolation valves, operators, cabling)	Temperature Pressure Rel. humidity Duration	250°F 0 to 25psig 100 % 1 day	200°F 0 to 20 psig 100 % 100 days		
6	Main steam isolation valve including operator and cable Main steam drain isolation valve including operator and cable Standby liquid control injection check valve	Temperature Pressure Rel. humidity Duration	340°F -2 to 35psig 100 % 3 hours	320°F -2 to 35psig 100 % 6 hours	250°F 0 to 25psig 100 % 1 day	200°F 0 to 20psig 100 % 100 days

Notes: (1) The equipment inside the primary containment will be subjected to 62 psig and 135°F for a maximum of 3 days during periodic leak testing.

(2) 56 psig is 90 percent of maximum containment internal pressure of 62 psig, as allowed by ASME Boiler and Pressure Vessel Code, Nuclear Vessels, Section III, Article 13, Paragraph N-1312, Sub-Paragraph (2)

♠ (3) Durations shown are termination times measured from the initiation of the postulated accident, i.e. Condition 1, the 3 hour duration, is the period from 45 seconds through 3 hours, the 1 day duration is the period from 6 hours through 1 day (24 hours).

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT
DESIGN SPECIFICATION

SPEC NO 22A3019 REV. NO. 1
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ISSUED DEC 16 1970

LEGEND SECTION A - II

This legend is a compilation of basic abnormal environmental pressures and temperatures together with the time durations expected. The full spectrum of simultaneous environmental possibilities is not presented in a series of curves, but rather as a description of the boundaries within which designated equipment must operate at discrete times during the cycles/modes of the reactor's operation.

1. Temperatures:

340°F Upper bound on maximum superheat temperature for a steam leak. This maximum can occur only when the reactor is at a pressure of 400 to 500 psi and a containment pressure of 50 psia. For higher or lower reactor pressures or lower containment pressure the temperature is less.

320°F Maximum superheat temperature during shutdown cooling line flush after reactor has been depressurized to 150 psia.

310°F Upper bound on saturation temperature at containment design pressure. This temperature applies only to the recirculation valves which must be functional only in the event of a recirculation line break. In the event of a steam leak that causes high superheat temperatures closure of the recirc valves is not required to flood the core.

285°F Saturation temperature at 35 psig (plus 4°F margin). This temperature applies only to the recirculation valves which must be functional only in the event of a recirculation line break.

250°F This represents the maximum long term temperature in the containment during the first day following a DBA.

200°F This represents the extended long term temperature in the containment following a postulated DBA.

2. Pressures:

-2 psig Negative design pressure of the primary containment

56 psig Positive design pressure of this primary containment.

35 psig The containment pressure corresponding to all the non-condensibles initially in the drywell being transferred to the wetwell.

25 psig Upper bound on long term pressure response up to one day following a postulated DBA.

20 psig Upper bound on extended long term pressure at one day and longer following a postulated DBA.

ISSUED:

3. Durations:

45 seconds	Conservative time duration to cover peak containment pressure.
30 minutes	In the event of a recirculation line break the recirc valves must be operable to insure core flooding. This represents a conservative duration.
1 hour	Applies to valves that isolate automatically on low RPV pressure or high drywell pressure. This time represents a conservative duration during which the valves must be operable.
3 hours	a) Conservative duration of time to depressurize the RPV, at a rate not exceeding 100°F/Hr, down to 150 psia.
	b) Conservative duration of time to flush shutdown cooling lines and depressurize reactor below 50 psia.

Section A-III ESSENTIAL EQUIPMENT

Outside Primary Containment - Abnormal Conditions

Condition	Component	Components must be operable under the following conditions	
♠ 1	HPCI system isolation valves including operator and cable HPCI pump, turbine, control, instrumentation and electrical equipment (5) RCIC system isolation valves, including operator and cable RCIC pump turbine, controls, instrumentation electrical equipment (5)	Temperature (4) Pressure Rel. Humidity Duration	148°F (1) 7" w.g. (1) 100% (1) 1 hour
♠ 2	Main steam isolation valves	Temperature (4) Pressure Rel. Humidity Duration	148°F (1) 7" w.g. (1) 100% (1) 1 hour
♠ 3	Feedwater isolation valves, including operator and cable Reactor water cleanup isolation valves, including operator and cable	Temperature (4) Pressure Rel. Humidity Duration	148°F (1) 7" w.g. (1) 100% (1) 1 hour
♠ 4	RHR system isolation valves, including operators and cable RHR pumps, heat exchanger, controls, instrumentation and electrical equipment (5) Core spray system isolation valves, including operator and cable Core spray pumps, controls, instrumentation and electrical equipment (5)	Temperature (4) Pressure Rel. Humidity Duration	148°F for 6 months (3) 7" w.g. for 1 hour zero inches w.g. for 6 months 100% R.H. for 1 hour 90% R.H. for 6 months
		Valves not required to be operable but must not fail open under the following conditions	
♠ 5	Reactor water cleanup isolation valves, including operator and cable	Temperature (4) Pressure Rel. humidity Duration	148°F for 1 hour (1) 7" w.g. for 1 hour (1) 100% R.H. for 1 hour (1)
♠ 6	HPCI system isolation valves, including operator and cable RCIC system isolation valves, including operator and cable Main steam isolation valves in steam tunnel, including operators.	Temperature (4) Pressure Rel. Humidity Duration	148°F for 1 hour (1) 7" w.g. for 1 hour (1) 100% for 1 hour (1)

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Section A-III (continued)

**

Notes

(1) 148°F, 100 percent R.H., and 7 inches static pressure may occur concurrently for the 1 hour as given, but R.H. and static pressure will decay after this period.

(2) Motors rated for continuous operation in an ambient temperature of 104F will operate in a higher ambient temperature with decreased life expectancy. Space cooling may be required to limit the ambient to an acceptable level.

(3) Temperature based on RHR equipment operating.

(4) Temperatures given do <u>not</u> take into account any temperature rise caused by direct steam impingement.

♠ (5) Equipment unable to withstand 100% R.H. and elevated temperatures (Motor Control Centers and Electrical Switchgear) for safety systems shall be located outside of secondary containment, or in separate rooms ventilated independently from the remainder of the reactor building.

RADIATION ENVIRONMENTAL CONDITIONS

Section B.

I. Inside Primary Containment

Equipment or Area	Radiation Type	1) Operating Dose Rate Plant Oper	System Oper	Design Basis Accident Type	Dose Rate	2) Integrated Dose Normal	Accident
Drywell, No Vessel Shield	Gamma Neutron	6.5×10^4 6.3×10^7		* LOCA	1.3×10^6	2.3×10^{10} 7.9×10^{16}	2.6×10^7
With Vessel Shield Zone 1 Above Core	Gamma Neutron	25.0 5×10^4		* LOCA	1.3×10^6	8.8×10^6 6.3×10^{13}	2.6×10^7
Zone 2 Core Region	Gamma Neutron	50.0 1.4×10^5		* LOCA	1.3×10^6	1.8×10^7 1.8×10^{14}	2.6×10^7
Zone 3 Under Vessel	Gamma Neutron	7.2 <1		* LOCA	1.3×10^6	2.5×10^6 $<1.3\times10^9$	2.6×10^7
Zone 4 Near Recirc.	Gamma Neutron	25.0 2×10^3		* LOCA	1.3×10^6	8.8×10^6 2.5×10^{12}	2.6×10^7
Zone 5 >15 ft. Recirc	Gamma Neutron	4.0 2×10^3		* LOCA	1.3×10^6	1.4×10^6 2.5×10^{12}	2.6×10^7
Zone 6 Torus	Gamma Neutron	0.1 2×10^2		* LOCA	1.3×10^6	3.5×10^4	2.6×10^7

1) Gamma Dose Rate — Rads (Carbon)/hour
   Neutron flux — Neutrons/cm^2-sec
2) Gamma Dose — Rads (Carbon)
   Neutron fluence — Neutrons/cm^2 (NVT)
   Normal Conditions — Integrated over 40 years
   Accident Conditions — Integrated over 6 months

* LOCA - Loss of Coolant Accident

- 100 percent load factor at rated power
- LOCA Analysis was based on the assumption that 100 percent of the noble gases, 50 percent of time halogens, and 1 percent of the solid fission products were released from the core

RADIATION ENVIRONMENTAL CONDITIONS

Section B. (Con.)

II. Inside Secondary Containment

Equipment or Area	Radiation Type	1) Operating Dose Rate Plant Oper	System Oper	Design Basis Accident Type	Dose Rate	2) Integrated Dose Normal	Accident
General Floor Area	Gamma	0.001		*LOCA	6.5×10^2	3.5×10^2	1.7×10^5
HPCI and RCIC Area	Gamma	0.015	0.200	*LOCA	1.6×10^2	5.3×10^3	4.5×10^4
RHR and HPCS Area	Gamma	0.015	0.030	*LOCA	1.6×10^2	5.3×10^3	4.5×10^4
24" Pipe Containing Torus Water (Typical Pipe)	Gamma	0.0		*LOCA	1.4×10^4	0.0	7.9×10^5
Cleanup Systems a) Heat Exchanger	Gamma	15.0		*LOCA	6.5×10^2	8.76×10^6	1.7×10^5
b) Pump Room	Gamma	>0.05		*LOCA	6.5×10^2	1.8×10^4	1.7×10^5
c) Filters & Tanks	Gamma	10.0		*LOCA	6.5×10^2	3.6×10^6	1.7×10^5
Steam Tunnel	Gamma	5		*LOCA	1.6×10^2	1.8×10^6	4.5×10^4
				Rod Drop	2.5×10^2		$>2.5 \times 10^2$
Standby Gas Treatment System	Gamma	0.001		*LOCA	5.7×10^5		3.8×10^4

1) Gamma Dose Rate — Rads (Carbon)/hour
Neutron flux — Neutrons/cm^2-sec
2) Gamma Dose — Rads (Carbon)
Neutron fluence — Neutrons/cm^2 (NVT)
Normal Conditions — Integrated over 40 years
Accident Conditions — Integrated over 6 months

- 100 percent load factor at rated power.

* LOCA Analysis was based on the assumption that 100 percent of the noble gases, 50 percent of time halogens, and 1 percent of the solid fission products were released from the core.

RADIATION ENVIRONMENTAL CONDITIONS

Section B. (Con.) III. Turbine Building

Equipment or Area	Radiation Type	1) Operating Dose Rate		Design Basis Accident		2) Integrated Dose	
		Plant Oper	System Oper	Type	Dose Rate	Normal	Accident
General Areas Protected by Shields	Gamma	0.001				4×10^3	
Operating Floor General	Gamma	0.005 - 0.020				77.0×10^4	
Contact HPT	Gamma	0.5				1.8×10^5	
Contact LPT	Gamma	0.1				3.5×10^4	
Equipment Bay (Heaters, condensers, etc.)	Gamma	0.05 - 5.0				1.8×10^6	
Steam Jet Air Ejector	Gamma	15 R/hr				5.3×10^6	
Condensate Treatment	Gamma	10 R/hr				3.5×10^6	

1) Gamma Dose Rate — Rads (Carbon)/hour
Neutron flux — Neutrons/cm^2-sec
2) Gamma Dose — Rads (Carbon)
Neutron fluence — Neutrons/cm^2 (NVT)
Normal Conditions — Integrated over 40 years — 100 percent load factor at rated power
Accident Conditions — Integrated over 6 months — LOCA Analysis was based on the assumption that 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solid fission products were released from the core.

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DESIGN SPECIFICATION

GENERAL ELECTRIC
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RADIATION ENVIRONMENTAL CONDITIONS

Section B. (Con.)

IV. Rad-Waste Building

Equipment or Area	Radiation Type	1) Operating Dose Rate Plant Oper	System Oper	Accident Dose Type	Dose Rate	2) Integrated Dose Normal	Accident
Control Room	Gamma		0.001			3.5×10^2	
Valve & Pump Rooms	Gamma		0.020			7.0×10^3	
Storage Tanks (Unprocessed)	Gamma		20.0			7.0×10^6	
Centrifuge	Gamma		100.0			1×10^7	

V. REACTOR CONTROL ROOM

Equipment or Area	Radiation Type	1) Operating Dose Rate Plant Oper	System Oper	Accident Dose Type	Dose Rate	2) Integrated Dose Normal	Accident
Control Room	Gamma	0.0005				1.75×10^2	3×10^0

1) Gamma Dose Rate — Rads (Carbon)/hour
Neutron flux — Neutrons/cm^2-sec
2) Gamma Dose — Rads (Carbon)
Neutron fluence — Neutrons/cm^2 (NVT)
Normal Conditions — Integrated over 40 years — 100 percent load factor at rated power
Accident Conditions — Integrated over 6 months — LOCA Analysis was based on the assumption that 100 percent of the noble gases, to percent of the halogens, and 1 percent of the solid fission products were released from the core.

Fron

SARGENT & LUNDY
ENGINEERS
55 EAST MONROE STREET
CHICAGO, ILLINOIS 60603
TELEPHONE 312-269-2000

L. C. FRON

AUG 24 1981

SLM(NI)-223
August 12, 1981
Project No. 6139-30

The Detroit Edison Company
Enrico Fermi Atomic Power Plant - Unit 2

Post-Accident Radiation Doses to Equipment;
Phase I Update

The Detroit Edison Company
Enrico Fermi - Unit 2 Project
Document Control Office - 110 S.B.
2000 Second Avenue
Detroit, Michigan 48226

Attention: Mr. R. J. Beaudry
Room No. 318 ECT

References: 1. John S. Brtis to R.J. Beaudry, "Post-Accident Radiation Doses to Equipment, Phase I Results Update", Letter No. SLM(NI)-193, Dated June 10, 1981.

2. John S. Brtis to R.J. Beaudry, "Post-Accident Radiation Doses to Equipment: Phase II Results", Letter No. SLM(NI)-207, Dated July 8, 1981.

Attached for your use is an update of the table which was transmitted in reference 1. As we have discussed, a change in our method of modeling the doses due to pipes carrying post-LOCA radioactivity resulted in our reporting slightly higher doses in the final Phase II results (reference 2) then were reported in the Phase I results (reference 1). (Specifically: the Phase II evaluation reflects the increase of dose along the 45° bisect of a 90° bend in the line over that which would be calculated for a totally straight line. The effect of taking this into account is an increase of approximately 50%.)

SARGENT & LUNDY
ENGINEERS
CHICAGO

Please place the attached table in your files and mark the earlier version as obsolete to avoid any confusion over this adjustment.

If there are any questions, please let me know.

Yours very truly,

J. S. BRTIS

John S. Brtis
Supervisor
Shielding & Radiological
Safety Section

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NSLD File: 4C-17-A1

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SIX MONTH POST-LOCA EQUIPMENT RADIATION DOSES

	Area	NUTECH Designation	Elevation	Columns	Rows	Calculated Dose (RAD)[c]
1.	SGTS Cubicles	3.1.1	677'-6"	F to G	12 to 17	1.6×10^8
2.	HVAC Area	3.1.2	677'-6"	F to H	9 to 12	5×10^3
3.	HVAC Area	3.1.3	677'-6"	G to H	12 to 17	1.5×10^2
4.	Refueling Floor	3.2	684'-6"	A to F	9 to 17	3.7×10^5
5.	HVAC Area	3.3	659'-6"	F to H	9 to 13	3.1×10^4
6.	Switch Gear Room	3.4	643'-6"	F to H	9 to 11	3.1×10^4
7.	Rx. Bldg. 3rd Flr.	3.5	643'-6"	A to F	9 to 11	2.4×10^6 [a]
8.	Switch Gear Room	3.6	613'-6"	F to H	9 to 11	3.1×10^4
9.	Rx. Bldg. 1st Flr.	3.7	583'-6"	C to G	9 to 11	5.4×10^6 [b]
.	Rx. Bldg. 1st Flr.	3.8	583'-6"	C to G	13 to 17	5.4×10^6 [b]
.	West Corner Rms.	3.9.1	562'-0"	-	-	5.4×10^6 [b]
.	East Corner Rms.	3.9.2	562'-0"	-	-	5.4×10^6 [b]
.	CRD Pump Room	3.10	562'0"	G to H	9 to 11	3.1×10^6 [b]
.	Compressor Room	3.11	562'-0"	G to H	9 to 17	3.1×10^4
.	East Corner Room	3.12.1	540'-0"	-	-	5.4×10^6 [b]
.	West Corner Room	3.12.2	540'-0"	-	-	5.4×10^6 [b]
.	HPCI Room	3.13	540'-0"	G to H	9 to 11	5.4×10^6 [b]
.	Rx. Bldg. 2nd Flr.	4.1	613'-6"	A to C	9 to 17	5.4×10^6 [b]

SIX MONTH POST-LOCA EQUIPMENT RADIATION DOSES (Cont'd)

	Area	NUTECH Designation	Elevation	Columns	Rows	Calculated[c] Dose (RAD)
19.	Rx. Bldg. 2nd Flr.	4.2	613'-6"	C to F	9 to 12	5.4×10^6 b
20.	Rx. Bldg. 4th Flr.	4.3.1	659'-6"	A to B	11 to 13	3.7×10^5 b
21.	Rx. Bldg. 4th Flr.	4.3.2	659'-6"	E to F	9 to 10	3.7×10^5 b
22.	Drywell	None	-	-	-	γ: 1.49×10^8
						$\gamma+\beta$: 1.89×10^9
23.	Torus	None	-	-	-	γ: 4.81×10^7
						$\gamma+\beta$: 5.91×10^8
24.	Undesignated Rx. Bldg. Areas	None	All	A to G	9 to 17	5.4×10^6 a,b
25.	Steam Tunnel	None	-	F to G	11 to 13	5.4×10^6 a,b

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a. The calculated dose to the internals of the hydrogen recombiners and other equipment carrying drywell atmosphere is 9.4×10^8. The recommended dose is 1×10^9.

b. The calculated dose to the internals of primary water carrying equipment is 1.6×10^7 rad. The recommended test dose is 2×10^7 rad.

c. The maximum dose calculated in this area.

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AUG 24 1981

SLM(NI)207
July 8, 1981
Project No. 6139-30

The Detroit Edison Company
Enrico Fermi Atomic Power Plant - Unit 2

Post-Accident Radiation Doses to Equipment:
Phase II Results

The Detroit Edison Company
Enrico Fermi - Unit 2 Project
Document Control Office - 110 S.B.
2000 Second Avenue
Detroit, Michigan 48226

Attention: Mr. R. J. Beaudry
Room No. 318 ECT

Reference: John S. Brtis to R. J. Beaudry, "Post-Accident Radiation Doses to Equipment: Phase II Results," Letter Number SLM(NI)-177, dated May 28, 1981.

Dear Mr. Beaudry:

The post-accident radiation zone maps and the associated report (reference) have been revised to incorporate your comments and changes found to be necessary during our review and approval process. A copy of each is attached for your use. The revised zone maps can be identified by the "6-81" designation in their lower right hand corner. The earlier sets should no longer be used.

COPY

If there are any questions, please let me know.

Yours very truly,

JOHN S. BRTIS

John S. Brtis
Supervisor
Shielding & Radiological
Safety Section

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AUG 24 1981

EQUIPMENT QUALIFICATION RADIATION DOSE ANALYSIS FOR THE ENRICO FERMI UNIT 2 STATION

1.0 Introduction

The purpose of this report is to outline the sources and assumptions used in the NUREG-0588 post accident equipment qualification dose calculation, for the Enrico Fermi - Unit II station. Integrated beta and gamma doses are calculated for equipment exposed to airborne and aqueous mixtures of released fission products, in accordance with the guidance of NUREG-0588.

2.0 Shutdown Core Fission Product Inventories

The core fission product inventories are calculated using the RIBD (Radio Isotope Buildup and Decay) subroutine of the RUNT-II computer program.[1] A tabulation of the input parameters and assumptions are given in the attached table, "Parameters and Assumptions for NUREG-0588 Analyses." These parameters were the same as those used by General Electric to generate the shutdown activities supplied in Reference 2. The only difference is that 3430 MWth is used for this study instead of the 1 MWth used by GE.

3.0 Airborne Source Dispersion

A schematic of the release model used to calculate the activity of airborne noble gas and halogen radionuclides in various parts of the Reactor Building are shown in Figure 1. Note that initial release assumptions for a BWR are not explicitly given in NUREG-0588, so reasonable extensions of the PWR model must be made. For this study it is assumed that 100% of the core noble gas inventory is initially airborne in the combined drywell and torus free volumes. In addition 25% of the core halogen inventory is assumed to be airborne in the drywell free volume and, 25% of the core halogen inventory is assumed to be plated-out on drywell surfaces at t=0. The 25% plate-out assumption, is conservative, though somewhat arbitrary since a realistic halogen deposition model for the drywell of a BWR has not been formulated.

The drywell and torus are assumed to leak at a constant rate of 0.5 volume percent per day throughout the course of the accident. Main Steam Isolation Value (MSIV) leakage does not contribute to the secondary containment source because of the pressurized leakage control system.

When determining airborne concentrations in the reactor building outside the primary containment, it is assumed that radionuclides released from the primary containment are completely mixed in the reactor building free volume before being exhausted through the Standby Gas Treatment System (SGTS). Conversely, when calculating SGTS filter inventories and exhaust plume concentrations, it is assumed that there is no mixing in the reactor building free volume and that the sources go directly to the SGTS.

Airborne radionuclide source activities are calculated using the Baffle portion of the RUNT-II[1] program. In Baffle the core inventories calculated by RIBD are distributed throughout a multicompartment model of the source regions described above. Time dependent buildup and decay of radionuclides are taken into account in Baffle.

4.0 Liquid Sources

Baffle is also used to model the time dependent activity of halogens and fission solids mixed in the primary coolant and suppression pool water. A schematic of the release model used is shown in Figure 2. As mentioned, above, explicit release assumptions for a BWR are not provided in NUREG-0588 and thus a reasonable adaptation of the PWR methodology must be made. For the PWR case, the action of spray removal systems produces a net liquid source term composed of almost 50% of the core halogens and 1% of the core fission solids.

Since credit for spray removal is not allowed on BWR's, no comparable mechanism exists for transporting airborne halogens into the suppression pool water.

However, for this analysis the conservative assumptions of Regulatory Guide 1.7[3] (i.e., 50% of the core halogens and 1% of core fission solids mixed in the suppression pool water) are used.

The RUNT-II computer code has a special option that allows noble gas daughter products evolved from halogen parents to be removed from the source region. This option is used in the analysis of liquid source terms.

5.0 Integrated Doses

Integrated doses are also calculated using RUNT-II[1]. A source geometry model is selected from those available in the ISOSHLD[4] portion of the program and dose rates are determined at times at which source activities were calculated in Baffle. Total integrated doses are then obtained by a simple trapezoidal integration of the dose rates.

Descriptions of the modeling used in several of the integrated dose determinations are given below.

5.1 Drywell Centerline Doses from Airborne Nuclides

Integrated gamma doses in the drywell from airborne noble gas and halogen radionuclides are conservatively calculated by treating the spherical portion of the drywell bulb as a right circular cylinder with an equivalent interior volume. No credit is taken for shielding afforded by the sacrificial shield or from piping and equipment contained in the drywell. Note that gamma shine from halogens plated-out on the interior surfaces of the drywell also contributes to the centerline dose. The calculation of the plate-out shine dose is discussed in the following section.

Integrated beta doses in the drywell are calculated using a semi-infinite cloud beta dose model with the nuclide concentration in the cloud determined by mixing the assumed noble gas release in the combined drywell and wetwell free volumes and the assumed halogen release in the drywell free volume only. NUREG-0588 indicates that an infinite cloud beta dose model should be used to determine beta doses at the drywell centerline. However, it also states that half of the infinite cloud dose is appropriate for cables in cable trays to account for beta self-shielding. Since even relatively small thicknesses of dense materials provide complete self-shielding against beta particles, the semi-infinite cloud model is used.

In addition to the semi-infinite cloud beta surface doses, beta depth doses are also calculated for a 10 mil thickness of 2 gm/cm^3 material. NUREG-0588 allows credit for beta attenuation in 10 mils of 2 gm/cm^3 material when calculating exposures to surfaces in contact with the cloud in the drywell.

5.2 Drywell Centerline Gamma Dose from Plated-Out Halogens

As mentioned in the previous section, halogens plated-out on the interior surfaces of the drywell also contribute to the drywell centerline dose. For this calculation it is assumed that 25% of the core halogen inventory is instantaneously plated-out in the drywell at time zero. This source is assumed to be uniformly distributed over the steel and concrete surfaces within the drywell. The total plate-out area is given in the EF-2 FSAR in Table 6.2-8.

Because the exact orientation and position of all the plate-out surfaces in the drywell are unknown, it will be assumed that the shine dose at the center of the drywell can be approximated by the calculated dose from a spherical shell source

whose radius is equal to the interior radius of the drywell bulb. Again, shielding afforded by structural materials and piping inside the drywell is ignored.

In ISOSHLD the dose from the spherical shell source geometry is determined by using an equivalent point source located at the sphere's radial distance from the dose point.

5.3 Drywell Wall Surface Doses from Plated-Out Halogens and Airborne Sources

In addition to contributing to the drywell centerline dose, the halogens plated-out in the drywell also contribute to surface exposure doses. For this study the surface dose contribution from plated-out halogens are conservatively calculated by using an infinite plane source model with the source strength per unit area determined by spreading 25% of the core halogen inventory unformly over the total drywell plate-out area.

In ISOSHLD both beta and gamma depth doses are calculated at points 10, 30, and 50 mils below the surface of a $2gm/cm^3$ material. Since the beta and gamma point kernel dose equations for an infinite plane source are singular for zero depth, it is not possible to directly determine surface doses using ISOSHLD. Surface doses are extrapolated from depth dose results.

The contribution to the gamma dose from airborne noble gases and halogens is assumed to be half of the drywell centerline value, while the semi-infinite cloud beta dose is assumed to apply at both locations.

5.4 Torus Immersion Doses from Airborne Noble Gases

Immersion doses from noble gas radionuclides airborne in the torus air space are calculated by modeling half of the free volume in the torus as a circular cylinder. The length and radius of the cylinder are determined by the amount of the airborne source actually seen by a dose point located anywhere on the torus centerline. No credit is taken for shielding afforded by piping or other equipment contained in the torus air space. As noted in the source term development discussion, the initial release of 100% of the core noble gas inventory is assumed to be uniformly mixed in the combined drywell and torus free volumes. None of the postulated halogen release to the drywell is assumed to enter the torus air.

5.5 Immersion Doses from Halogens and Fission Solids Released into the Suppression Pool

Immersion doses from halogens and fission solids released into the suppression pool are calculated using the same cylinder model as was used in determining the air immersion doses in the torus. The only differences are that water is the attenuation medium instead of air and that source dilution factors are based on liquid rather than air volumes. The initial release of 50% of the core halogen inventory and 1% of the core fission solid inventory is assumed to be mixed in the combined primary coolant and suppression pool liquid volumes. Note that beta doses are semi-infinite cloud values.

5.6 Refueling Floor Centerline Immersion Doses

Immersion doses at the refuleing floor centerline from airborne noble gas and halogen radionuclides that have leaked from the drywell into the reactor building are estimated by modeling the volume above the refueling floor as an air filled cylinder with a height equal to the distance from the operating floor to the roof of the reactor building. Leakage from the drywell is assumed to be completely mixed in the entire volume exhausted by the Standby Gas Treatment System (SGTS). No credit is taken for shielding afforded by equipment resident in the refueling volume.

5.7 Shine Through the Drywell Wall into the Reactor Building

Gamma shine through the drywell wall into the reactor building from airborne noble gases and halogens is estimated by modeling the drywell bulb as a spherical source region surrounded by a concrete shell. The sphere radius is taken to be the interior radius of the drywell bulb, 33', and the shell thickness is taken to be the minimum drywell wall thickness, 6'. No credit is taken for source attenuation due to structures within the drywell. Doses are calculated at a point 1' outside the concrete shield wall.

5.8 Dose from Halogens and Particulates Trapped on the SGTS Filter

To calculate the gamma shine dose from halogens and particulates trapped on the charcoal filter in the Standby Gas Treatment System (SGTS), it is assumed that all radionuclides captured in the SGTS are deposited uniformly throughout the charcoal mass. In ISOSHLD the charcoal is modeled as a rectangular prism filled with a matrix of carbon and source nuclides.

No credit is taken for shielding afforded by casing materials surrounding the filter region. It is also conservatively assumed that nuclides leaking from the drywell are immediately exhausted through the SGTS without mixing in the reactor building free volume.

5.9 Shine from SGTS Filter Through 5' Thick Shield Wall

The shine dose through the concrete wall that surrounds the SGTS filter trains is calculated using the same model used to determine the dose from unshielded SGTS filters. The only differences are that a 5' thick ordinary concrete wall is placed seven feet from the center of the filter and that dose points are located outside the wall.

5.10 Doses from Unshielded Pipes Containing Halogens and Fission Solids

Gamma shine doses from unshielded pipes containing halogen and fission solid radionuclides that have been mixed with the primary coolant and suppression pool waters are calculated using a simple, water filled cylinder geometry. Pipe wall thicknesses are based on Schedule 40 pipe specifications, while the pipe length is arbitrarily taken to be 15'. Nominal pipe diameters ranging from 4" to 24" are examined with dose points positioned from 1 inch to 50' away from the pipe.

5.11 Shine from 16" Pipe Through 1.5' Thick Wall

Gamma shine doses from a 16" Schedule 40 pipe that is shielded by a 1.5' thick ordinary concrete wall are calculated using the same ISOSHLD models used for the unshielded pipe cases described previously. The only differences are that a 1.5' thick shield is placed 3.5' from the center of the pipe and that dose points are placed outside the shield wall.

5.12 Plume Immersion Dose from SGTS Releases

To calculate an immersion dose from the radionuclides released from the SGTS vent, the source nuclides are assumed to be uniformly distributed in a large, finite cylindrical volume adjacent to the wall of the reactor building. The volumetric source strength is determined by the SGTS release rate and a simple χ/Q based on building wake mixing. The radius of the cylinder is related to the reactor building cross section area, while its length is taken to be six times the distance

from the reactor building wall to the dose point, i.e., 6x50'=300'. No credit is taken for mixing in the reactor building prior to release through the SGTS.

5.13 Shine Doses from Refuleing Floor Volume to Exterior Points

If complete mixing in the reactor building free volume is assumed, the airborne noble gases and halogens contained in the volume above the refueling floor will contribute to the exposure doses received by equipment located outside the reactor building. To account for this source, the volume above the refueling floor is modeled as a rectangular prism. An exterior dose point is conservatively assumed to be located on the same elevation as the center of the refueling floor volume. No credit is taken for shielding afforded by the wall of the reactor building above the refueling floor elevation. Dose points are positioned at 10 to 500 feet from the wall of the reactor building to assess the impact of this source at various distances from the building.

5.14 Doses to the Internal Components of the Hydrogen Recombiner

The internal components of the recombiner are exposed to the undiluted drywell airborne source of noble gas and halogen nuclides. To estimate the dose to these components, the free volume within the recombiner heater box is conservatively modeled as an equivalent cylindrical source. This geometry ignores the actual piping layout in the heater box and neglects the dose contribution from the air filled piping outside the heater box. The radionuclide source concentration in the cylinder is assured to be the same as the airborne concentration in the drywell.

5.15 Shine Doses from the Hydrogen Recombiners

To estimate the dose at points away from the hydrogen recombiners from airborne noble gases and halogens passing through the device, the free volume in the recombiner heater box is modeled as a sphere source surrounded by an iron shell. The shell accounts for the metal casing and the pipe walls that define the actual air flow path through the recombiner. The contribution to the shine dose from piping located outside the heater box is accounted for by scaling the source strength such that the total activity in the sphere is equal to the total activity in all the source volume associated with the recombiner piping. Integrated doses are calculated at points 15" to 50' away from the sphere center.

5.16 External Shine Dose to the Control Room Ventilation Equipment Area

The external shine dose to the control room ventilation equipment area comes from airborne noble gas and halogen radionuclides in the reactor building. Shine from nuclides in the reactor building below the refueling floor level must pass through the 1'8" thick reactor building wall before entering the equipment area, while nuclides above the refueling floor are only attenuated by the 4" thick roof slab over the equipment area. For the purposes of this calculation the shine through the reactor building wall is ignored. The shine through the roof is modeled as a slab abutted to the reactor building wall above the refueling floor level with dose points located from 3' to 50' beyond the shield wall.

5.17 Dose at a Point Outside the Torus Liner from Airborne and Liquid Sources in the Torus

Exposure doses to points outside the torus liner come from airborne noble gases in the torus and from halogens and fission solids mixed in the torus water. Since the gaseous and liquid phases in the torus occupy approximately equal volumes, doses outside the torus are calculated by treating each phase separately, while conservatively taking the net dose to be the sum of the contributions from each phase. The dose from the airborne nuclides is determined by modeling half the torus air volume as a long, rectangular prism source. An infinite slab model is used for the liquid dose calculation because of the short mean free path for gammas in water. In both cases a thin iron shield is used to account for attenuation afforded by the torus liner. The dose point is one inch outside the torus liner.

5.18 Doses from 4" Diameter Hydrogen Recombiner Pipe Containing Drywell Air

The gamma shine dose from an unshielded 4" diameter pipe containing drywell air is calculated using a simple, air filled cylinder geometry. The pipe wall thickness is based on schedule 40 pipe specifications, while the length is arbitrarily taken to be 15'. Dose points are positioned from 1 inch to 50 feet away from the pipe.

6.0 Radiation Zone Maps

It should be pointed out that the enclosed zone maps are based on a conservative interpretation of NUREG-0588 for the Enrico Fermi Unit - 2 nuclear power plant. Doses to the equipment, integrated over a six month period postaccident, were calculated using the models given in section 5 of this report. These doses were then used to create a set of radiation zone maps for the integrated dose to equipment located in the Reactor Building and specified areas of the Auxiliary Building.

6.1 Airborne Sources in Reactor Building

In determining the zones, consideration was first given to the airborne noble, halogen and particulate sources. The dose due to beta particles was calculated using an infinite cloud formulation which gave a value of 3.1×10^5 rad everywhere in the Reactor Building HVAC Boundary. The dose due to gamma rays was based on a finite cloud immersion dose model for the refueling floor. Doses to smaller rooms were scaled as the volume to the 1/3 power based on an unattenuated spherical source model. Values for various volumes are given in Table 1.

6.2 Shine Doses from Refueling Floor Airborne Sources

The Shine Dose to the Auxiliary and Reactor Buildings were based upon the model given in section 5.16. The 4 inches of concrete shielding would keep these doses to a value of less than 10^4 rad which would render them insignificant.

6.3 Doses from Unshielded Pipes Containing Radioactive Liquids

Doses from pipe sources were based on models described in section 5.10. Values obtained from these models were used to generate a set of "Dose vs. Distance" curves for varying diameter pipes. These curves are shown in Figure 3. Doses for pipes of diameters not modeled were obtained from calculations based upon the next larger pipe. Zones were defined by obtaining the highest zone that could be realistically shown (usually only to within 5 feet of the source) on the plot.

6.4 Unshielded Pipes Containing Radioactive Airborne Sources

These doses were plotted in the same way as radioactive liquid sources except a different graph was used to find the Doses. This graph is shown in Figure 4.

6.5 Shine Doses from the Hydrogen Recombiners

The Doses from the Hydrogen Recombiner use the model described in section 5.15. These doses were plotted in the same way as the Radioactive Liquid sources except that the Doses were read from a diff   nt graph. This graph is enclosed as Figure 5.

6.6 Sh ne from SGTS Filters Sources

Doses from the SGTS filter bank were based on the model described in section 5.8. These doses were plotted in the same manner as those for Radioactive Liquids and the graph is enclosed as Figure 6.

6.7 Other Sources

Doses that were shielded by thick shields (i.e., greater than 5 feet of concrete) all prove to be insignificant. These included shine from the Primary Containment, SGTS filters, the Torus, and Refueling Room shielded by Reactor Building walls and floors. The Doses from the SGTS Plume also was insignificant. The Primary Containment area, due to its large sources, was designation as Zone I. The Torus area, due to contact dose to the torus of 5.1×10^6 Rad, was designated as Zone II.

7.0 Conclusions

Postaccident doses were calculated for all equipment that could contain radioactive sources. These doses were combined to give the total integrated dose to any point in the Reactor Building and designated areas of the Auxiliary Building, and were then plotted on a set of General Arrangement Figures.

To use these drawings, one should locate the area of interest an ind the zone that you are in. If there are no highly ra active sources, (this can be determined by inspection of the "EQ" set of figures), use the highest value of that zone as the dose value.

If there are Highly Radioactive Sources in your area, you will have to read your Dose from the tables given for the Highly Radioactive Sources on each drawing. If, however, you are in contact with a radioactive source, you should use the Doses given under Notes 4, 5 and 6 of the first page of the figures. In this manner the dose to any equipment can be found.

References

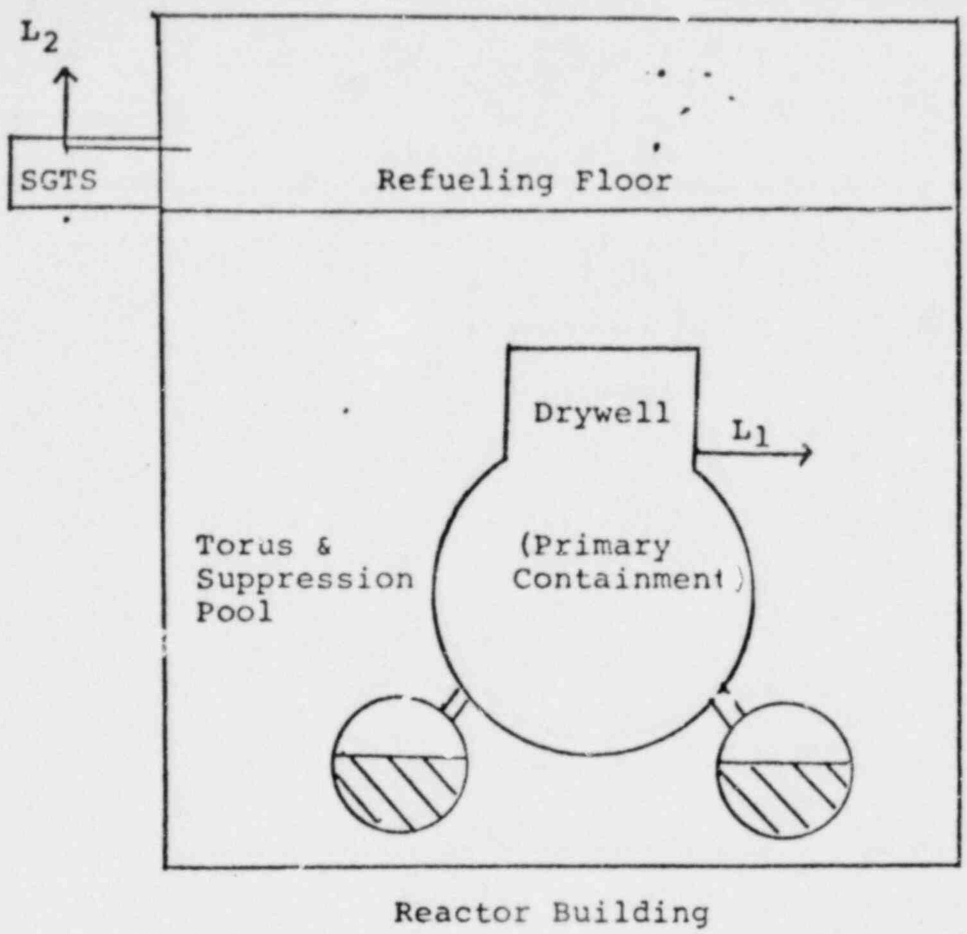
1) Pichurski, D. J. "RUNT-II A Computer Program for Determining Time Integrated Doses," S&L Program Number RAC 09.8.034-2.20, December 1979

2) General Electric (GE) Letter to BWR Owner's Group, "Radiation Source Term Information for NUREG-0578 Implementation," November 1979

3) Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978

4) Pichurski, D. J. "ISOSHLD A Computer Program for General Purpose Isotope Shielding Analysis; S&L Program Number ISO09.8.029-2.20", January 1981

5) Enrico Fermi-2 Final Safety Analysis Report (EF-2 FSAR)



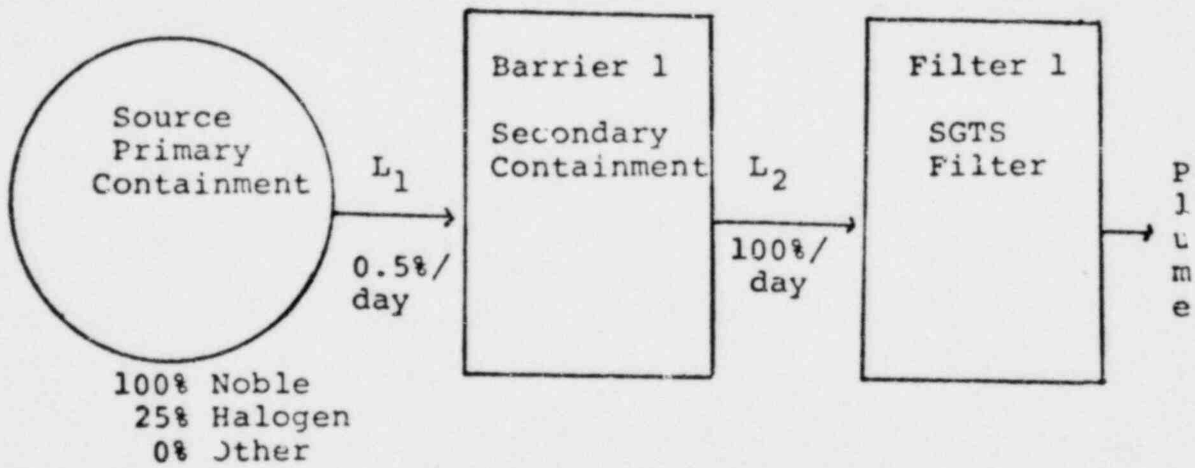


Figure 1 Airborne Source Dispersion Model

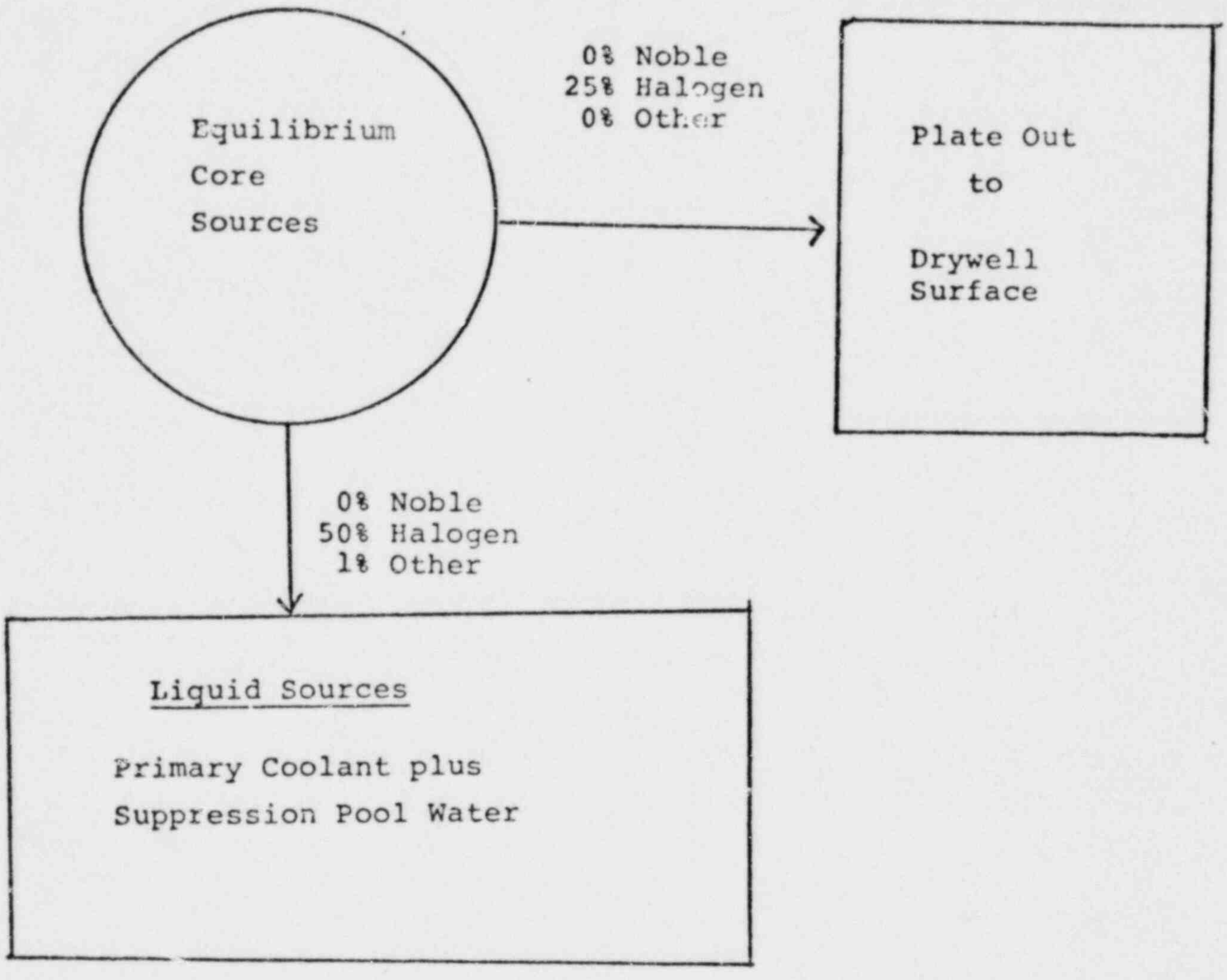


Figure 2 Plate Out and Liquid Sources

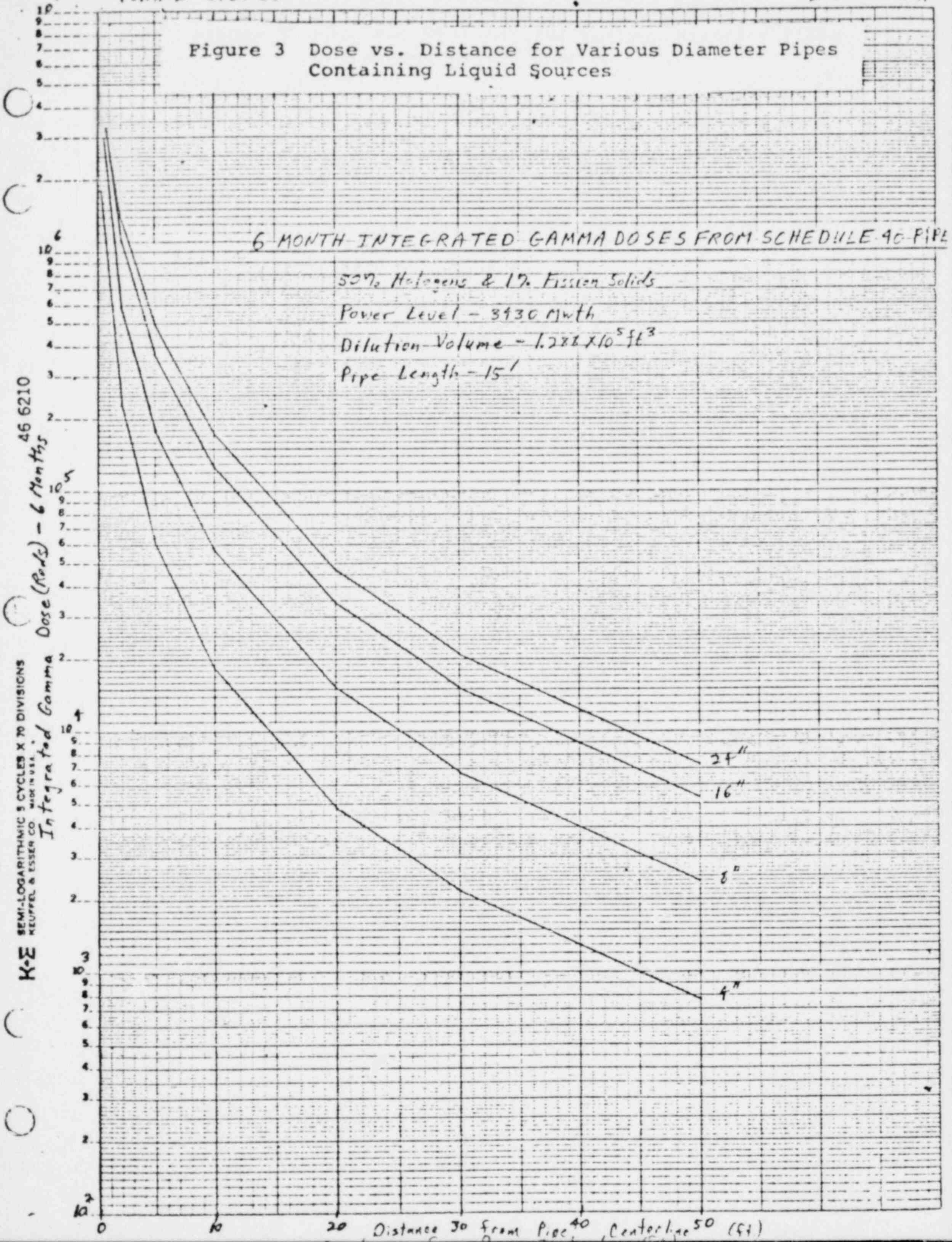


Figure 3 Dose vs. Distance for Various Diameter Pipes Containing Liquid Sources

Figure 4 Dose vs Distance for Airborne Pipe Sources

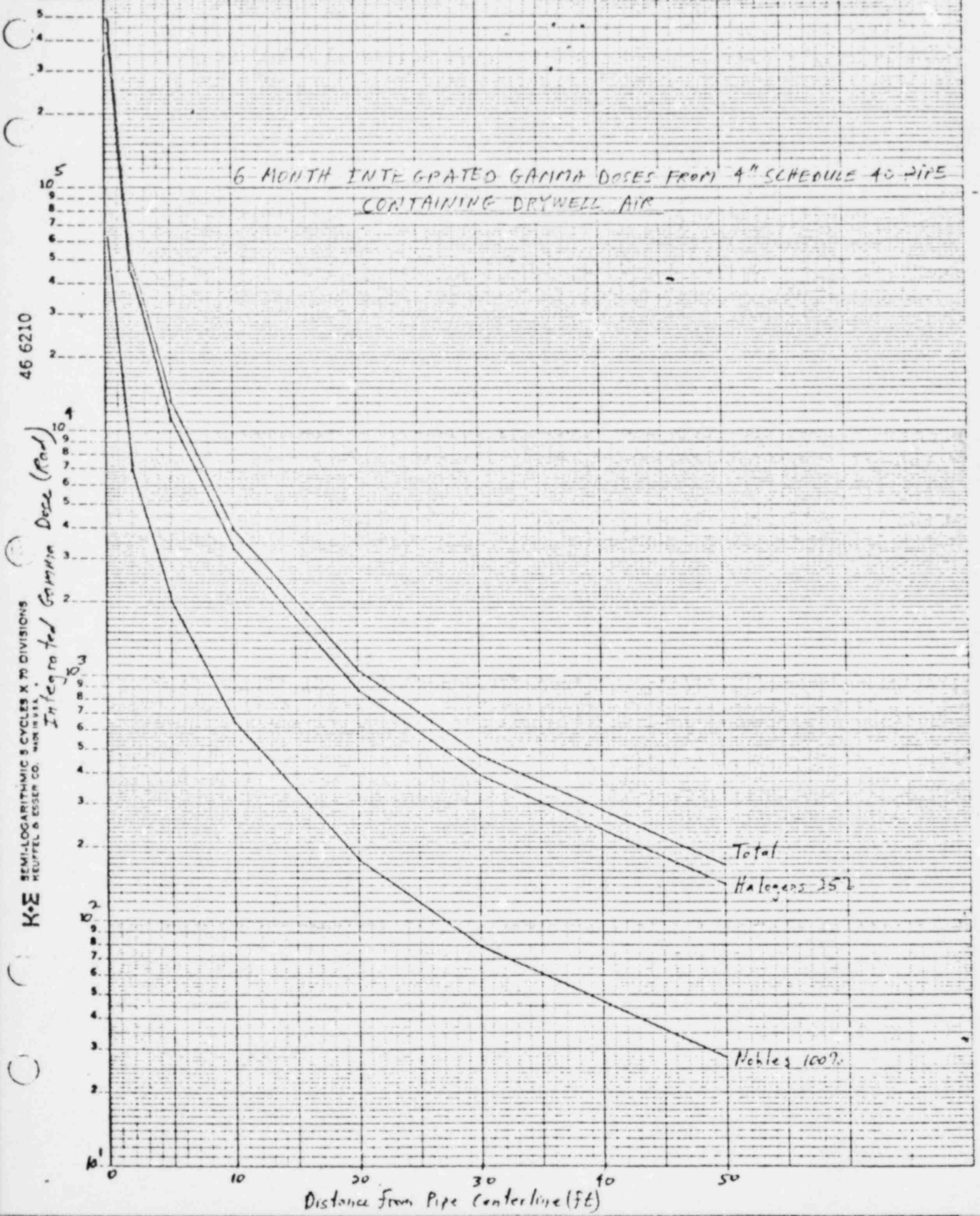


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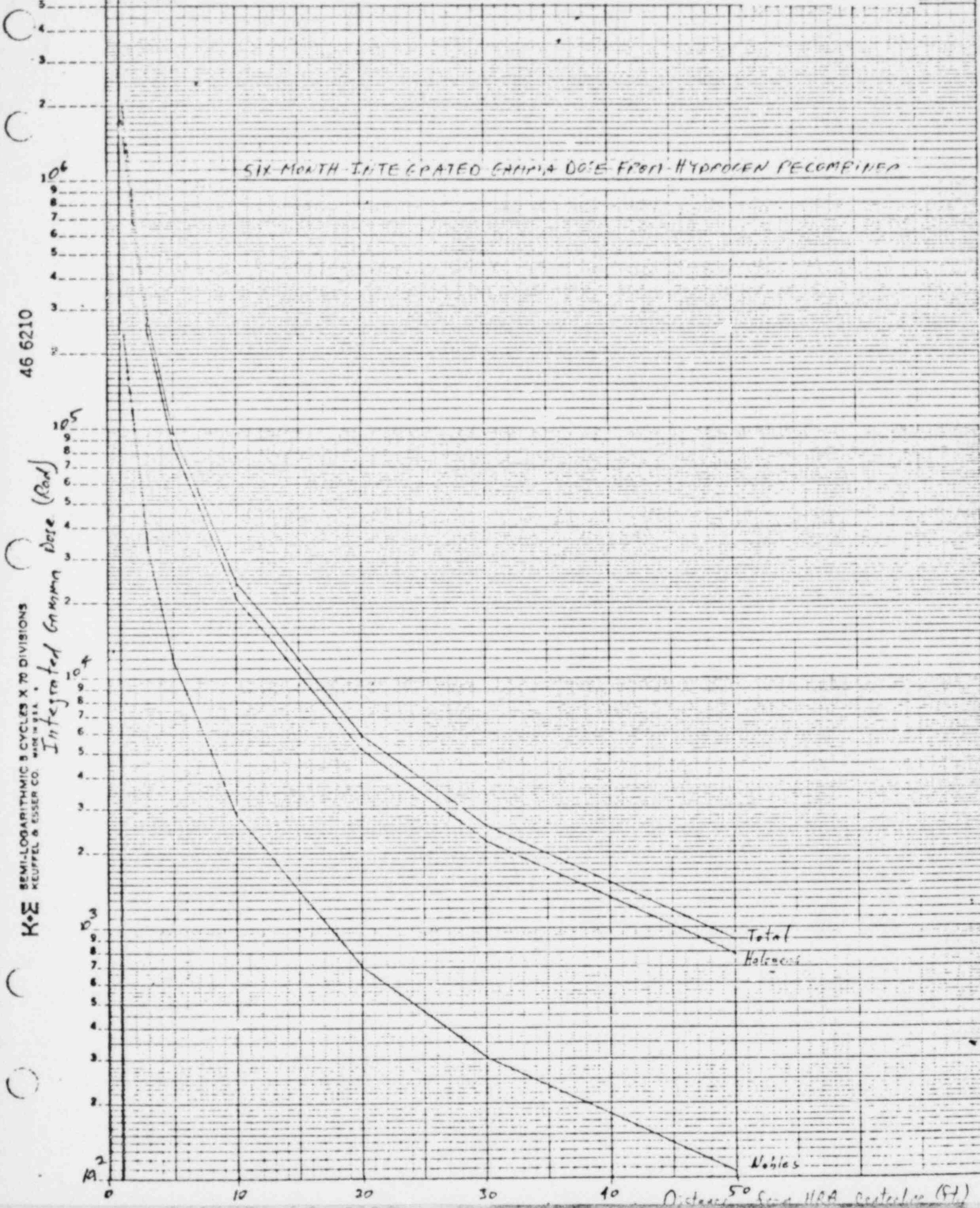
Figure 5 Dose vs Distance for Hydrogen Recombiner Source



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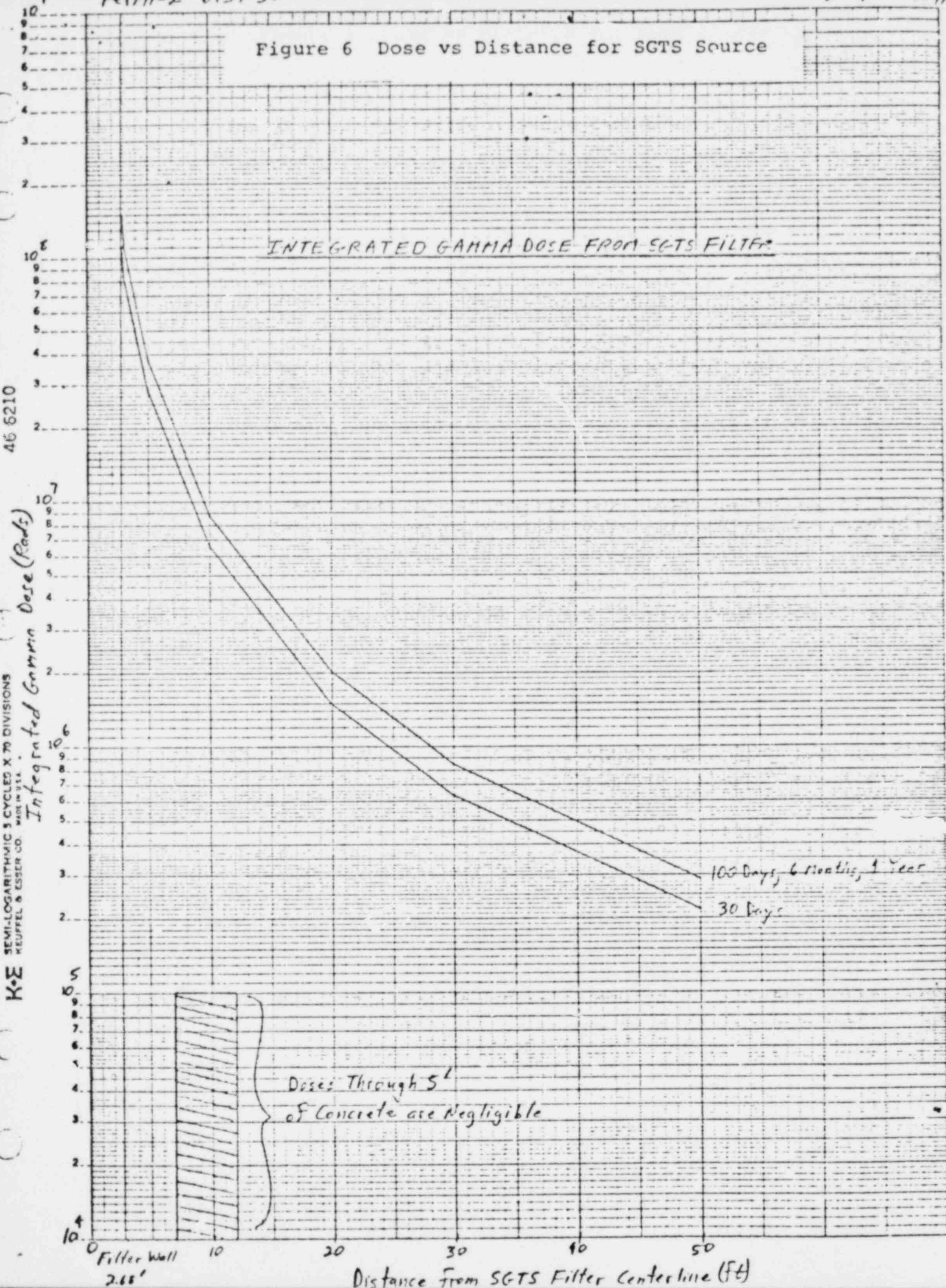

Table 1 Airborne Gamma Doses versus Room Volume

Volume (ft^3)	Gamma Dose (Rad)	% of Refueling Room Volume
1.0+06*	6.2+04	100
5.0+05	5.0+04	50
2.5+05	4.0+04	25
1.0+05	3.0+04	10
5.0+04	2.3+04	5
1.0+04	1.4+04	1
5.0+03	1.1+04	1/2

*Read 1.0×10^6

DETROIT EDISON COMPANY, FERMI-2

PROJECT NO. 6139-30

PARAMETERS AND ASSUMPTIONS FOR NUREG-0588 ANALYSES

ITEM	VALUE	REFERENCES	COMMENTS
NSSS			
Power Level	3430 MWt	FSAR p. 15B 6-37	For Source Development
Burnup	1095 days	Reference 2	Full Power Operation
Conversion Ratio	.250	Reference 2	
PU239/U235 Fissions	4×10^{-2}	Reference 2	
U-235 ABS. XSECTION	325 b	Reference 2	
Source Code	RIBD	Reference 2	
Water Density	0.74 g/cc	FSAR Fig. 4.3-1	Normal Operation
Steam Flow	1.416×10^{7} lb/hr	FSAR p. 4.4-28	Normal Operation
Core Coolant Flow	1.0×10^{8} lb/hr	FSAR p. 4.4-28	Normal Operation
Steam Pressure	1035 psia		
Core Ave. Thermal Flux	2.9×10^{13}	NEDO-20948	
SOURCE RELEASE			
Drywell			
Airborne Nobles	100%	Ref. 1	Diluted in the drywell plus wetwell free air volume
Airborne Halogens	25%	Ref. 1*	
Airborne Others	0%	Ref. 1	
Plated Out Halogens	25%	Ref. 1*	Distributed over the surface area of the Drywell

* A non-mechanistic & conservative interpretation of NUREG-0588 guidance

PARAMETERS AND ASSUMPTIONS FOR THE NUREG-0588 ANALYSIS (Cont'd)

Item	Value	Reference	Comments
Wetwell			
Airborne Nobles	100%	Ref. 1	Diluted in the drywell plus wetwell free air volume
Airborne Halogens	0%	Ref. 1	
Airborne Others	0%	Ref. 1 & 6	
Waterborne Nobles	0%	Ref. 1	Diluted in the RPV plus the suppression pool water volume for the systems drawing from the suppression pool
Waterborne Halogens	50%	Ref. 1	
Waterborne Others	1%	Ref. 1	
Airborne Source Mitigation			
Suppression Pool Air Volume	130,900 ft^3	FSAR p.6.2-4	
Drywell Free Air Volume	163,730 ft^3	FSAR Tbl. 6.2-1	
Total Primary Air Dilution Volume	294,630 ft^3	From above	
Primary Containment Leak Rate	0.5%/day	FSAR p. 15B.6-37 & Ref. 6	
Secondary Containment Free Volume	2,800,000 ft^3	FSAR p. 6.2-106	
Refueling Floor Volume	995,600 ft^3	Ref. 3	
Drywell Plate-Out Area	41,280 ft^2	FSAR Tbl. 6.2-8	
STANDBY GAS TREATMENT SYSTEM			
SGTS Flow Rate	100%/day	FSAR p. 15B.6-37	...of secondary containment volume
SGTS Filter Efficiencies	99% Iodines	FSAR p. 15B.6-37	
SGTS Filter Efficiencies	99% Particulates	FSAR p. 15B.6-37	
SGTS Charcoal Weight	3600 lb	FSAR Tbl. 6.2-11	
SGTS Charcoal Volume	140 ft^3	FSAR Tbl. 6.2-11	
SGTS Charcoal Bed Depth	6"	FSAR Tbl. 6.2-11	

PARAMETERS AND ASSUMPTIONS FOR THE NUREG-0588 ANALYSIS (Cont'd)

Item	Value	Reference	Comments
Waterborne Source Mitigation			
Reactor Coolant Liquid Volume	11,390 ft^3	FSAR Tbl. 6.2-1	
Suppression Pool Water Volume	117,450 ft^3	FSAR p. 6.2-4	
Total Liquid Source Dilution Volume	128,840 ft^3	From above	
Plume Immersion			
χ/Q	4.37×10^{-4} sec/m^3	Ref. 3	
Hydrogen Recombiner			
Free Air Volume	11,800 in^3	Ref. 3	

Systems Assumed to be Contaminated

Reactor Pressure Boundary to the Second Isolation Valves (Primary Liquid)

Drywell (Primary gases)

Wetwell (Torus Water & gases)

HPCI (Torus Water and Containment Atmosphere on Turbine Side)

CSS (Torus Water)

RHR/LPCI (Reactor Liquid)

RCIC (Torus Water and Containment Atmosphere on Turbine Side)

CRD System (Torus Water) - Scram discharge header & holdup pipe

Hydrogen Recombiner (Primary Containment Atmosphere)

Secondary Containment Atmosphere (Primary Containment Leakage & Evolution from ESF Equipment Leakage)

Secondary Containment Drains & Sumps (ESF Equipment Leakage)

SGTS (Secondary Containment Atmosphere)

Primary Sample Systems (Reactor liquid & gases)

Torus Water Management System (Torus Water)

PARAMETERS AND ASSUMPTIONS FOR THE NUREG-0588 ANALYSIS (Cont'd)

ISOLATION

All non ESF paths are assumed to be isolated with respect to the primary containment at the second isolation valve.

Other than SGTS & sampling, all paths out of the reactor building are assumed to be isolated.

Isolation valve leakage will be ignored.

RWCU is isolated post-LOCA.

References

1. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979

2. F. E. Gregor to F. Tsai, "Information Transmitted Concerning the Technical Support Center Design", Letter No. EF2-48409, dated March 12, 1980

3. Pichurski, D. J., "NUREG-0588 Post Accident Equipment Qualification Doses", EF2-TMI-EQ-04, March, 1981.

4. Notes of Meeting March 20, 1980, "Discussion of the Progress of the Post-Accident TMI Related Work Being Performed by Sargent & Lundy." (attached)

5. Enrico Fermi-2 Final Safety Analysis Report (EF-2 FSAR)

6. Notes of Meeting, December 18, 1980, "Equipment Qualification," Dated December 31, 1981. (Attached)