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May 16, 1983

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Limerick Generating Station, Units 1 and 2
Request for Information from the Radiological Assessment Branch

References: (1) A. Schwencher to E. G. Bauer, Jr. letter dated March 11, 1983.

(2) Conference call between the Radiological Assessment Branch and Philadelphia Electric Company on April 28, 1983.

File: Gov't 1-1 (NRC)

Dear Mr. Schwencer:

The attached documents are draft question response changes and draft text changes to the FSAR resulting from the discussions with Mr. Mike La Mastra, Radiological Assessment Branch reviewer, at the referenced conference call.

The changes to items 3 and 9 will be formally incorporated into the FSAR revision scheduled for May, 1983. The balance of the changes will be formally incorporated into the FSAR revision scheduled for June, 1983.

Sincerely,

B001

EJBradley

E. J. Bradley

LN/cw/G-1
cc: See Attached Service List

cc: Judge Lawrence Brenner (w/o enclosure)
Judge Richard F. Cole (w/o enclosure)
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2. High-radiation alarm light (amber)
3. Downscale alarm light (white)
4. Alarm reset (push button)
5. Meter zero adjust (on the amplifier)
6. Alarm level adjust
7. Trip check pushbutton
8. Power supply switch and "power-on" light (clear lens)
9. Indicators to show power supply voltages
10. Annunciator outputs
11. Recorder outputs

e. The radiation monitors are calibrated at regular time intervals in accordance with station procedures. Calibration methods are covered in detail in the equipment procedures manual.

f. The following annunciators are located in the control room to alert the operator:

1. Reactor enclosure area, high-radiation (Units 1 and 2)
2. Refueling floor area, high-radiation (Units 1 and 2)
3. Turbine enclosure area, high-radiation (Units 1 and 2)
4. Turbine enclosure common area, high-radiation
5. Radwaste enclosure area, high-radiation (common)
6. Refueling hoistway common area, high-radiation
7. Hot maintenance shop, high-radiation (common)
8. Unitized area, low-radiation (trouble)
9. Common area, low-radiation (trouble)

12.3.4.1.3 Local Area Monitors

In addition to the area radiation monitors described above, a total of four local area monitors is provided, located on each of the two refueling bridges and the two turbine enclosure crane cabs. The essential differences between these monitors and those area monitors described above are as follows:

a. No outputs to the control room are provided.
b. Alarms are local only.
c. No recorders are provided.
d. Local power (battery) packs are provided in the event of external power cutoff.

Portable alarming rate meter will be used in the drywell to warn plant personnel if temporary shielding is removed during fuel transfer operations. The meter will have local and remote alarms to ensure adequate personnel warning. Administrative controls will be used to prevent access to the upper drywell and unnecessary high exposures to personnel. Also, accessible portions of the spent fuel transfer canal will be clearly marked with signs stating that potentially lethal radiation fields are possible during fuel transfer.

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ITEM # 3 DSER FROM

THE RAB

INTRODUCTION & GENERAL DESCRIPTION

CHAPTER 1

FIGURES

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Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/ Particulate)
Outside	Percent (100/50/1) in reactor coolant system	Percent (10/10/0) in reactor coolant system
Inside	Larger of (100/50/1) in containment or (100/50/1) in reactor coolant system	(10/10/0) In reactor coolant system

Response

~~A radiation and shielding design review of vital areas and equipment will be completed in order to determine what corrective action is required, if any, to provide for adequate access to this equipment and to provide protection for this equipment during accident conditions.~~

~~The shielding design review will include the following areas:~~

~~a. Main Control Room~~

~~b. Technical Support Center~~

~~c. Post-Accident Sampling Area~~

~~d. Post-Accident Sample Analysis Area~~

~~e. Secondary Containment~~

~~f. Security Center~~

~~g. Radwaste Control Area~~

~~h. Remote Shutdown Panel Area~~

~~The radioactive source terms used will be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, 1.7, and SRP 15.6.5. Existing shielding for the Main Control Room is~~

Replace with insert (attached)

Replace with insert (attached)

~~described in Sections 6.4.2 and 12.3.2. Environmental Qualification of Class 1E equipment and components is described in Section 3.11.~~

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• II.B.3 POSTACCIDENT SAMPLING

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," or 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactor" release of fission products. If the review indicates that personnel could not promptly and safely obtain samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases which indicate cladding failure and isotopes which indicate fuel melting. The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

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A. Introduction

The design review of plant RADIATION AND shielding ~~of spaces~~ ~~for post accident operation~~ was performed ~~is described below.~~ as required by ~~in accordance with~~ NUREG 0737, Item II.B.2 and is described below. The purpose of the review was to identify potential problem areas and equipment which may require the development of special post-accident procedures, installation of additional permanent or temporary shielding, relocation of components or piping, or requalification of components.

¶ Areas and equipment which are vital for post-accident occupancy or operation were evaluated to determine if access and performance of required operator activities or equipment functions might be unduly impaired due to the presence of the postulated radiation source in the selected systems. Systems required or postulated to process highly radioactive fluids or gases outside the containment during post-accident conditions were selected for evaluation. Radiation levels in adjacent plant areas due to contained sources in piping and equipment of these systems were estimated. Airborne sources caused by leakage from the PRIMARY containment and systems containing post-accident sources were also included in the evaluation AS ~~to the extent~~ described ~~below~~ in Section E.

¶ The identification of vital areas and a summary of the methodology used to determine radiation doses for these areas and to equipment are presented below. Insert B (next page) The identification of essential equipment and the results of the review of equipment for the postulated

radiation sources ARE ~~will be~~ provided in the separate Environmental Qualification Report. The results of the shielding design review for vital areas, ~~and a description of corrective actions taken~~ are provided below in Section H.

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Insert B for page 1

Doses to personnel in vital areas and access paths are listed in Tables 1.13-1 and 1.13-2, respectively. Doses to equipment, used for qualification purposes, are ~~listed~~ PROVIDED IN THE ~~in Table~~ ~~2.11-3~~. EQUIPMENT QUALIFICATION REPORT

B. Vital Area Identification

Areas which may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident are designated as vital areas. A review of the Limerick Generating Station was made which determined that the following areas should be designated vital areas.

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Continuous Occupancy

1) Main Control Room

2) Technical Support Center

3) Operations Support Center

4) Security Center

Infrequent Occupancy

1) Counting Room

2) Radiochemistry laboratory

3) (Post accident Sampling Station

4) ~~Post-LOCA radiation monitoring room~~ North stack instrument room

5) HVAC panels at el. 304'

6) Radwaste control room

7) Diesel-generator area

Potential vital areas that are not listed above were excluded for the following reasons. The post-LOCA hydrogen control (recombiner) system, and containment isolation valves ~~resets, and emergency power supply (diesel generators) can all be~~ are all automatic or remotely controlled by the operator in the main control room and require no local access. There is no manual ECCS alignment area at Limerick. Instrument panels

and motor control centers ~~MCCs~~ are not included because the control and alignment of essential systems ~~equipment~~ is accomplished from ~~are established in~~ the main control room and ~~do not~~ require no ~~any~~ local action.

Three access paths to vital areas that pass through high radiation areas were also identified and included in this review.

C. Selection of Systems ~~for~~ Radiation and Shielding Review

A review ~~of the Limerick Generating Station~~ was made to determine which systems could be required to operate and/or are expected to contain highly radioactive materials following a postulated accident where substantial core damage has occurred. The results of this review are presented below.

1. Core Spray, High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), ~~and RHR~~ Residual Heat Removal (RHR), and Safeguard Piping Fill Systems

   The Core Spray, RHR, HPCI (water side) ~~and~~ RCIC (water side), and Safeguard Piping Fill, systems would contain suppression pool water being injected to the reactor coolant system. Although the HPCI and RCIC systems could also carry condensate, suppression pool water was ~~is~~ assumed for this review for conservatism. The steam sides of the HPCI and RCIC systems would operate on reactor steam.

2. RHR System (Shutdown Cooling Mode)

   The RHR system recirculates reactor water when it operates in the shutdown cooling mode. Before operation in this mode can be initiated the reactor must be depressurized to less than 75 psig. This depressurization is expected to remove substantially all of the noble gases released into the reactor water. Following a postulated serious accident the HPCI, RCIC, RHR (LPCI mode), and Core Spray Systems would inject water into the reactor coolant system. This water from the condensate tank and/or the suppression pool would dilute the reactor water prior to the initiation of shutdown cooling with the RHR system. This ~~shielding~~ review ~~will~~ assumed that there are no noble gases in the reactor water in the RHR system for the shutdown cooling mode and that the reactor water is diluted by the suppression pool water volume. ~~However, since the exact amount of dilution of the reactor water is difficult to determine, no dilution in addition to the reactor coolant volume~~ was ~~is assumed.~~

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3. Control Rod Drive (CRD) System

The operation of the CRD system was reviewed to determine if the scram discharge headers will contain highly radioactive water following a postulated accident. It ~~has been~~ was determined that they will not. Prior to a scram the CRD housings contain condensate water delivered by the CRD pumps. When a scram occurs some of this condensate water from the CRD is discharged to the scram discharge header. After the scram, some condensate and reactor water flows to the scram discharge header until it is completely filled. This takes a matter of seconds. Since the vents and drains in the scram discharge header are isolated by the scram, all discharge flow then stops. Since it is not reasonable to assume that significant core damage occurs in the first few seconds following a scram, the scram discharge header will contain only a mixture of condensate and pre-accident reactor water following this postulated accident.

4. Reactor Water Cleanup (RWCU) System

For a major accident with resulting core damage, the RWCU system would be isolated and would contain no highly radioactive materials beyond the second isolation valve. On a BWR this system is not needed for reactor coolant system venting. It would not be practical to use it for accident recovery after a major accident. It was therefore assumed that this system would not operate with highly contaminated reactor water.

5. Gaseous Radwaste System

For a major accident with resulting core damage, it would not be practical to use the gaseous radwaste system for accident recovery. Noble gas isotopes with long lives would cause excessive offsite doses if the gaseous radwaste system ~~were~~ was used after a design basis accident.

It was therefore assumed that this system would not operate.

6. Post Accident Sampling Lines

Sampling lines used after an accident

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would contain primary containment gas, secondary containment gas, reactor coolant (pressurized or depressurized) or suppression pool water, depending on the sampling line take-off location.

7. ~~2.~~ Containment Atmospheric Control System

The recombiner system and associated H_2-O_2 analyzer lines ~~would recirculate primary~~ containment gas after a serious accident in order to keep hydrogen and oxygen concentrations at acceptable levels.

8. ~~2.~~ Standby Gas Treatment System (SGTS) and Reactor Enclosure ~~SGT and RERS~~ Recirculation System (RERS)

The SGTS and RERS ~~Standby Gas Treatment System and Reactor Enclosure Recirculation System~~ would collect airborne activity in the secondary containment following a serious accident. Radioactivity would be collected on the filters and charcoal beds in these systems.

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9. Containment

The free volume of the primary containment is assumed to initially contain large amounts of post-accident activity. These sources, as well as those assumed for the suppression pool, are described below. Shine through the drywell and wetwell walls would cause a negligible increase to the secondary containment airborne and piping doses, and therefore was not included in this ~~radiation and shielding~~ review.

D. Source Release Fractions

The following release fractions were used as a basis for determining the concentrations for the RADIATION AND shielding review:

Source A: Containment atmosphere: 100% noble gases, 25% halogens

~~Source B: Depressurized Reactor Liquid: 50% Halogens, 1% solids~~

Source ~~C~~ B: Suppression pool liquid: 50% halogens, 1% solids

Source ~~D~~ C: Reactor ~~S~~team: 100% noble gases, 25% halogens

These release fractions were applied to the total curies available for the particular chemical species (i.e., noble gas, halogen, or solid) for an equilibrium fission product inventory for a light water reactor core.

Source D: Pressurized reactor coolant: 100% noble gases, 50% halogens, 1% solids

E. Source Term Models

The assumptions used for release fractions for the RADIATION AND shielding design review are outlined above.

These release fractions are, however, only the first step in modeling the source terms for the activity concentrations in the sytems under review. The decay time and dilution volume also affect the rationale for the selection of values for these parameters.

1. Decay Time

For conservatism, no decay time credit was taken for the radioactive decay that might occur before fission products would be transported to the various systems.

2. Dilution Volume

The volume used for dilution is important, affecting the calculations of dose rate in a linear fashion. The following dilution volumes were used with the release fractions and decay times listed above to arrive at the final source terms for the shielding reviews.

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Source A: Drywell and suppression pool free air volumes.

~~Source B: Reactor coolant system normal liquid volume (based on reactor coolant density at the operating temperature and pressure).~~

Source ~~C~~ B: The volume of the reactor coolant system (based on reactor coolant density at the operating temperature and pressure) plus the suppression pool water volume.

Source ~~D~~ C: The total reactor system steam volume.

Source D: The volume of the reactor coolant system.

3. ~~Sources Used in Piping and Equipment for Each System Under Review~~ Contained Sources and Drywell and Secondary Containment Airborne Sources

In defining the contained sources, Accident operating modes were assumed for each system. In defining the limits of the connected piping subject to contamination listed below, normally shut valves were assumed to remain shut.

- Core spray system - Source ~~C~~ B

- High pressure coolant injection system

  Liquid - Source ~~C~~ B
  Steam - Source ~~D~~ C (with credit for steam specific activity reduction due to turbine operation).

- Reactor core isolation cooling system

  Liquid - Source ~~C~~ B
  Steam - Source ~~D~~ C (with credit for steam specific activity reduction due to turbine operation).

- Residual heat removal system - Source B (all modes)

  ~~Shutdown Cooling Mode - Source B~~

  ~~All other modes - Source C~~

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- Post Accident Sampling Lines
  - Gas sample lines – Source A
  - Liquid sample lines – Source D

- Containment Atmospheric Control (Recombiner) System – Source A (Drywell free volume only)

- Drywell – Source A

- Standby Gas Treatment System, Reactor Enclosure Recirculation System, and Secondary Containment Activity

The following major assumptions were used to calculate the secondary containment airborne radiation doses and the radiation doses for the SGTS filters and the RERS filters.

a. 100% of the noble gases and 25% of the halogens are available for leakage into the secondary containment.

b. The primary to secondary containment leak rate is 0.5% per day.

c. Airborne activity in the secondary containment is confined to spaces below the refueling floor.

d. The RERS flow rate is two secondary containment air changes per hour.

e. The SGTS flow rate is one half secondary containment air change per day.

f. The RERS charcoal filter is 95% efficient with respect to halogens.

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g. The SGTS charcoal filter is 100% efficient with respect to halogens for filter loading. ~~For airborne dose calculations, it is assumed to be 99% efficient in accordance with the minimum technical specification efficiencies.~~

h. The activity inventory in the core was based on 1000 days burnup and daughter product formation was not considered. These assumptions, which have off-setting effects, were necessitated by limitations in the computer code used to treat the transport of activity from primary to secondary containment.

i. The capacities of the RERS and SGTS filters are sufficient to sustain cleanup for the duration of the accident.

4. Airborne Sources for Vital Areas

All the vital areas and access paths A, B, and C are located in the turbine enclosure, radwaste enclosure, control structure, administration building, and technical support center. The transport

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pathway of the airborne sources in these areas consists of leakage from the primary containment to the reactor enclosure, and discharge to the environment ~~through~~ VIA the reactor enclosure recirculation system (RERS) and the standby gas treatment system (SGTS). The airborne activity DISCHARGED then re-enters the buildings through the ventilation intake systems after dilution within the building wake cavity.

¶ The assumptions used in calculating the released airborne activity are the same as those listed in Section 3 above for the secondary containment except that the technical specification minimum filter efficiencies are used to maximize the activity released (i.e. RERS is 95%, and SGTS is 99%, efficient). Also, an assumed 5 gpm systems leakage (at suppression pool activity concentration) to the secondary containment is included in the analysis.

¶ The X/Q's of the airborne transport pathway to the main control room, technical support center, HVAC panels, and north stack

(9c)

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instrument room are given in FSAR section ~~for them.~~ 15.10, along with their calculational basis. ~~and~~ These X/Q values are also listed in the table below as the X/Q's for the north side of the reactor enclosure. The X/Q's used in evaluating the total integrated doses at other locations within the building wake cavity are given below, as the west end X/Q's. These values differ from the Section 15.10 values due to the differing building cross sectional areas.

	X/Q (s/m^3)	
Period (hours)	West end of reactor enclosure	North side of reactor enclosure
0 - 8	4.77×10^{-4}	3.46×10^{-4}
8 - 24	2.81×10^{-4}	2.04×10^{-4}
24 - 96	1.79×10^{-4}	1.30×10^{-4}
96 - 720	7.87×10^{-5}	5.71×10^{-5}

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F. Radiation Dose Calculation

1. Primary and Secondary Containment Doses

The sources described in Section E were used to estimate doses from the systems ~~selected to be~~ included in the RADIATION AND shielding design review. No vital areas are located in the primary or secondary containments, ~~and~~ THUS the doses described in this section were used for equipment qualification PURPOSES only.

For compartments inside the secondary containment, the post-LOCA gamma radiation levels were conservatively determined by adding the maximum piping contact dose in that compartment to the secondary containment gamma cloud dose. The total integrated gamma dose was then determined by adding the post-LOCA integrated dose to the 40 year normal operating integrated dose. The equipment quali-fication levels for all safety-related electrical components were compared to the applicable calcu-lated dose. For those components initially listed as inadequately qualified, more detailed calculations were performed, taking into account dose/distance relationships in order to determine more realistic doses. A set of dose/distance curves for each system was developed as part of this effort.

post-LOCA

9 Both gamma and beta doses were calculated for the primary containment. The doses were calculated by assuming that 100% of the core noble gas inventory, 50% of the core halogen inventory, and 1% of the core solid fission product inventory are released. These source terms are consistent with those specified in NUREG-0588 and NUREG-0737.

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The primary containment airborne dose calculations assumed that 50% of the 50% (i.e. 25%) halogen release from the core plates out instantaneously, as assumed implicitly in Regulatory Guide 1.3, Rev. 2. The airborne doses were calculated assuming source terms diluted by the primary containment (drywell and wetwell) free volume. These assumptions are consistent with those specified in NUREG-0737.

The beta doses and dose rates were calculated assuming an infinite cloud geometry.

For components inside primary containment, the total integrated gamma doses were calculated by adding the post-LOCA primary containment gamma cloud dose to the 40 year normal operating dose. Dose distance relationships were not used to reduce post-LOCA doses inside primary containment.

2. Doses in Other Plant Areas Containing Safety-Related Equipment.

Doses for specified areas outside the secondary containment were also calculated, as described below, for equipment qualification purposes. For the SGTS equipment compartment, the post-LOCA gamma dose is the contact dose of the SGTS filters. For the remaining areas, the

post-LOCA gamma doses were determined by adding the cloud, filter and piping shine doses from adjacent compartments and the control structure cloud dose, as applicable. These post-LOCA doses were added to the normal operating integrated dose to determine the total integrated gamma doses. Beta doses outside secondary containment would be ~~so small that they can be considered~~ negligible for equipment qualification purposes, and therefore they were not calculated.

3. Vital Area and Access Path Doses

Calculations were performed to determine airborne and shine doses from the sources described in Section E to the vital areas and access paths. These doses were used to determine personnel exposures, using occupancy factors as described in Section G. For radiation doses due to direct shine, credit was taken for attenuation through the walls and ~~through the air~~ due to distance. Airborne doses include both gamma and beta contributions.

G. Personnel Exposure Limits

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The general basis for personnel radiation exposure guidelines was 10CFR50, Appendix A, GDC 19. The following additional radiation limit guidelines were used to evaluate occupancy and accessibility of plant vital areas and access paths.

Radiation Exposure Guidelines

Occupancy	Dose Objective
Continuous	$\leq$5 Rem for duration
Infrequent	$\leq$5 Rem for all activities
Accessway	$\leq$10 Rem/hr

These dose objectives are for personnel access only.

The dose RATE received by personnel in vital areas of continuous occupancy should be <15 mrem/h (average over 30 days). The doses ~~rates~~ for these areas WERE ~~is~~ determined using the control room occupancy factors contained in SRP 6.4, as discussed in NUREG-0737, i.e., 1.0 for 0-1 day; 0.6 for 1-4 days; and 0.4 for over 4 days.

The dose received by personnel in an infrequent occupancy of vital areas and access paths is determined by taking into account the frequency and duration of the activities anticipated for that area, and is consistent with GDC 19 limits. Average area dose rates are used to determine personnel exposure, although local hot spots may exist.

H. Results of Dose Calculations

1. Environmental Qualification of Equipment

Normal operating radiation doses and conservative post-LOCA gamma and beta doses are provided

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THE EQUIPMENT QUALIFICATION REPORT

in ~~FSAR Table 3.11-5~~

for all areas containing safety related equipment. Equipment was reviewed against these doses or, if necessary, against reduced doses calculated as described in Section F.1. The results of the review of equipment ARE ~~will be~~ provided in the Environmental Qualification Report.

2. Personnel Access

The reactor enclosure will be inaccessible after a design basis LOCA. However, this does not present a problem since it does not contain any vital areas requiring operator access.

Doses to the vital areas DESCRIBED in Section B from the sources described in Section E are provided in Table 1.13-1. Doses rates for vital area access paths which pass through high radiation areas are provided in Table 1.13-2. The airborne and

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direct shine doses are added together to show the total dose to personnel in the vital areas.

4 Peak shine dose rates are provided in Tables 1.13-1 ~~and 1.13-2~~ along with the integrated doses. The dose rate at any given time for a vital area can be estimated by multiplying the peak dose rate by an appropriate factor which can be determined by using the curves in Figures 1.13-2 and 1.13-3.

G.H Potential problem areas that were identified in the design review of plant shielding are listed in Table 1.13-2, along with the ~~corrective actions that have been taken or~~ special post-accident procedures that will be followed for these areas.

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I. Conclusion

Areas and equipment vital for post-accident occupancy or operation ~~have~~ were ~~been~~ identified, and post-accident doses ~~in doses~~ were calculated in accordance with the requirements of NUREG-0737. Potential problem areas ~~were that require~~ were identified, and solutions ~~developed~~ were determined for these problem areas. ~~were found.~~ No additional shielding is required as a result of this study.

LGS FSAR

TABLE 1.13-1

VITAL AREA RADIATION DOSES

VITAL AREAS[1]	PIPING AND CLOUD SHINE ~~DOSES~~: Major Source	PIPING AND CLOUD SHINE: Peak Dose Rate (Rem/hr)	PIPING AND CLOUD SHINE: Total Dose (Rem)	AIRBORNE WHOLE BODY DOSE (Rem)[2]	SUM OF SHINE PLUS WHOLE BODY DOSES (Rem)	DOSE OBJECTIVE (Rem)	AIRBORNE ORGAN DOSES (Rem)	DOSE OBJECTIVES (Rem)
Continuous Occupancy								
Main Control Room (SRP 6.4 occupancy)	ECCS/RHR pipe	1.1	4.2	2.3 E-1	4.4	≤ 5.0	Thyroid: 3.3 E-3 Skin β: 5.0	≤ 30 ≤ 30
Technical Support Center (SRP 6.4 occupancy)	RHR pipe/reactor enclosure cloud	2.7 E-3	1.5 E-2	2.3	2.3	≤ 5.0	Thyroid: 5.4 E-3 Skin β: 4.9	≤ 30 ≤ 30
Operations Support Center (SRP 6.4 occupancy)	(later)	(later)	(later)	(later)	(later)	≤ 5.0	Thyroid: (later) Skin β: (later)	≤ 30 ≤ 30
Security Center (SRP 6.4 occupancy)	RHR pipe/reactor enclosure cloud	1.7 E-1	7.3 E-1	(later)	(later)	≤ 5.0	Thyroid: (later) Skin β: (later)	≤ 30 ≤ 30
Infrequent Occupancy								
Counting Room (SRP 6.4 occupancy)	ECCS/RHR pipe	5.5 E-4	1.3 E-3	1.1 E-1	1.1 E-1	≤ 5.0	Thyroid: 3.7 E-1 Skin β: 8.3	≤ 30 ≤ 30
Radiochemistry Laboratory (SRP 6.4 occupancy)	ECCS/RHR pipe	9.3 E-3	3.0 E-2	1.1 E-1	1.4 E-1	≤ 5.0	Thyroid: 3.7 E-1 Skin β: 8.3	≤ 30 ≤ 30
Post Accident Sampling Station (31 min. occupancy at time t = 1 hour post-LOCA)	Recombiner/ ECCS pipe	6.6 E-1[3]	3.4 E-1[3]	6.2 E-3	3.5 E-1[3]	≤ 5.0	Thyroid: 4.6 E-3 Skin β: 7.8 E-2	≤ 30 ≤ 30
North Stack Instrument Room (1 hour occupancy)	North stack	4.0 E-2	4.0 E-2	1.6 E-2	5.6 E-2	≤ 5.0	Thyroid: 8.6 E-3 Skin β: 2.0 E-1	≤ 30 ≤ 30
HVAC Panels (1 hour occupancy)	ECCS/RHR pipe	3.3 E-1	3.3 E-1	1.6 E-2	3.5 E-1	≤ 5.0	Thyroid: 8.6 E-3 Skin β: 2.0 E-1	≤ 30 ≤ 30
Radwaste Control Room (SRP 6.4 Occupancy)	ECCS/RHR pipe	2.1 E-2	6.7 E-2	2.9 E-1	3.6 E-1	≤ 5.0	Thyroid: 3.7 E-1 Skin β: 8.3	≤ 30 ≤ 30
Diesel Generator Area (later)	(later)	(later)	(later)	(later)	(later)	≤ 5.0	Thyroid: (later) Skin β: (later)	≤ 30 ≤ 30

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TABLE 1.13-1, cont.

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(1) Occupancy factors used to calculate doses are listed in parentheses for each vital area. SRP 6.4 occupancy factors are for 30 days. For vital areas with one hour occupancy, the doses reflect one hour occupancy at the maximum post-accident dose rate, unless otherwise specified.

(2) Airborne whole body doses are specific for the listed vital areas.

(3) Doses do not include shine from sample source. See Table 11.5-6.

LGS FSAR

TABLE 1.13-2

RADIATION DOSES FOR VITAL AREA ACCESS PATHS

ACCESS PATH[2]	PIPING AND CLOUD SHINE		PEAK AIR-BORNE WHOLE BODY DOSE RATE (Rem/hr)	DOSE OBJECTIVE (Rem/hr)	PEAK AIR-BORNE ORGAN DOSE RATES (Rem/hr)
	Major Source	Peak Dose Rate (Rem/hr)			
A	ECCS pipe/ H_2 recombiner	2.1[1]	2.2 E-2	≤ 10	Thyroid: 1.2E-2 Skin β: 2.8 E-1
B	ECCS pipe/ H_2 recombiner	2.3E+1[1]	2.2 E-2	≤ 10	Thyroid: 1.2E-2 Skin β: 2.8E-1
C	ECCS pipe	2.1	2.2 E-2	≤ 10	Thyroid: 1.2E-2 Skin β: 2.8E-1

(1) The H_2 recombiner starts operating at time t=36 hours. This creates the following supplemental dose rates:

For Path A at t=36 hr, 2.5 E-1 R/hr

For Path B at t=36 hr, 1.3 R/hr

(2) Paths are shown in Figure 1.13-1

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TABLE 1.13

Solutions to Potential Vital Area Access Problems ~~and [illegible] Solutions~~

Potential Problem Area(1)	~~[illegible]~~ Solution	Remarks
~~Main Control Room~~	~~Add shielding between the control room and the HPCI/RHR pipe.~~	~~Shielding for one or two pipes should [illegible] the control room dose a suffi- cient margin below 5 [illegible]~~ Rem.
~~HVAC Panels at El. 304'~~	~~Restrict access to this area. Personnel should be allowed in this area for short durations only.~~	~~This restriction should not cause a problem for operators. Access to this area is required only when a trouble alarm sounds in the control room notifying of a problem in the HVAC isolation system. Access to the panel is required to determine the specific problem in the HVAC isolation system, which could be accomplished in a short period of time.~~
Access Path ~~[illegible]~~ B	An alternate path should be used until the dose rate falls below 10 Rem ~~rads~~/hr in the vicinity of access path ~~[illegible]~~ B.	Using Figure ~~[illegible]~~ 1.13-2, it can be seen ~~shows~~ that the dose rate for access path ~~[illegible]~~ B will fall below 10 Rem ~~rads~~/hr about 2 hours after a DBA. Access path ~~[illegible]~~ B would be used for access to the radwaste control room, radiochemical laboratory, or counting area from the control room. If access is required before the dose rates for access path B has fallen to acceptable levels, the ~~An alternate route would be for the~~ operators can exit the turbine enclosure, travel around the radwaste enclosure, and use access path C to enter the radwaste control room. ~~If PECo deter- mines that this alternate route is too inconvenient for operators, then addi- tional shielding could be provided bet- ween the HPCI/RHR pipes and access paths A and B.~~

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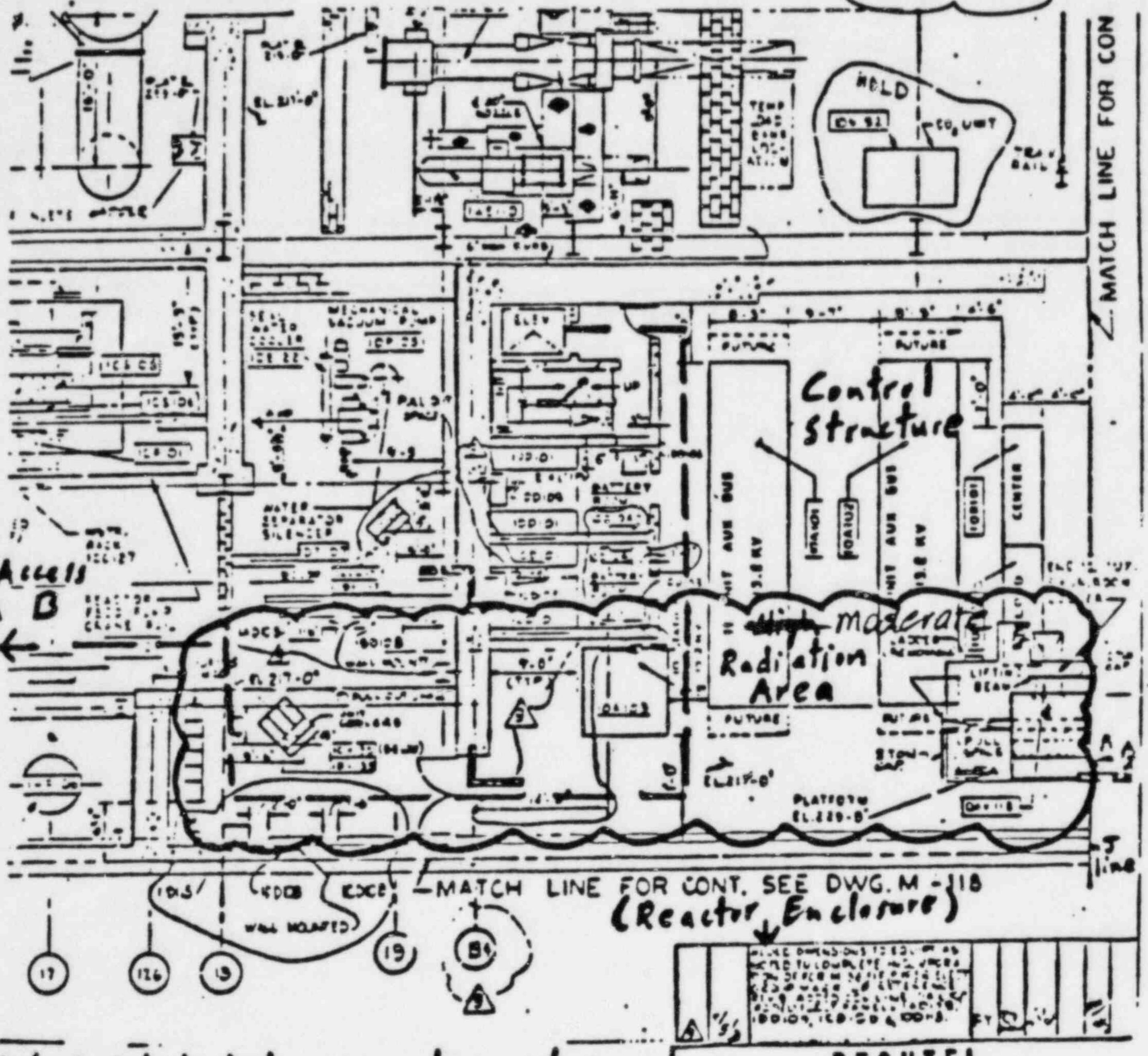
~~D-213/9~~ ~~TABLE 5~~

(1) ~~Main control room is shown in Figures 12.23 and 12.24. The HVAC panels are shown in Figures 12.25 and 12.26. The~~ Access Path B is shown in Figure 1.13-1.

DRAFT

Access Path A

Note: use latest revision of M-111



Note: Substantial area dose rates are defined as follow:
High dose rate ≥ 10 R/Hr
Moderate dose rate ≥ 1 R/Hr <10 R/Hr

BECHTEL
SAN FRANCISCO
LIMERICK GENERATING STATION UNITS 1 & 2
EQUIPMENT LOCATION
TURBINE ENCLOSURE UNIT No1
PLAN AT EL. 217'-0"
8031 | M-111 | 9

FIGURE 1.13-1 Sheet 1 of 3

Access Path B

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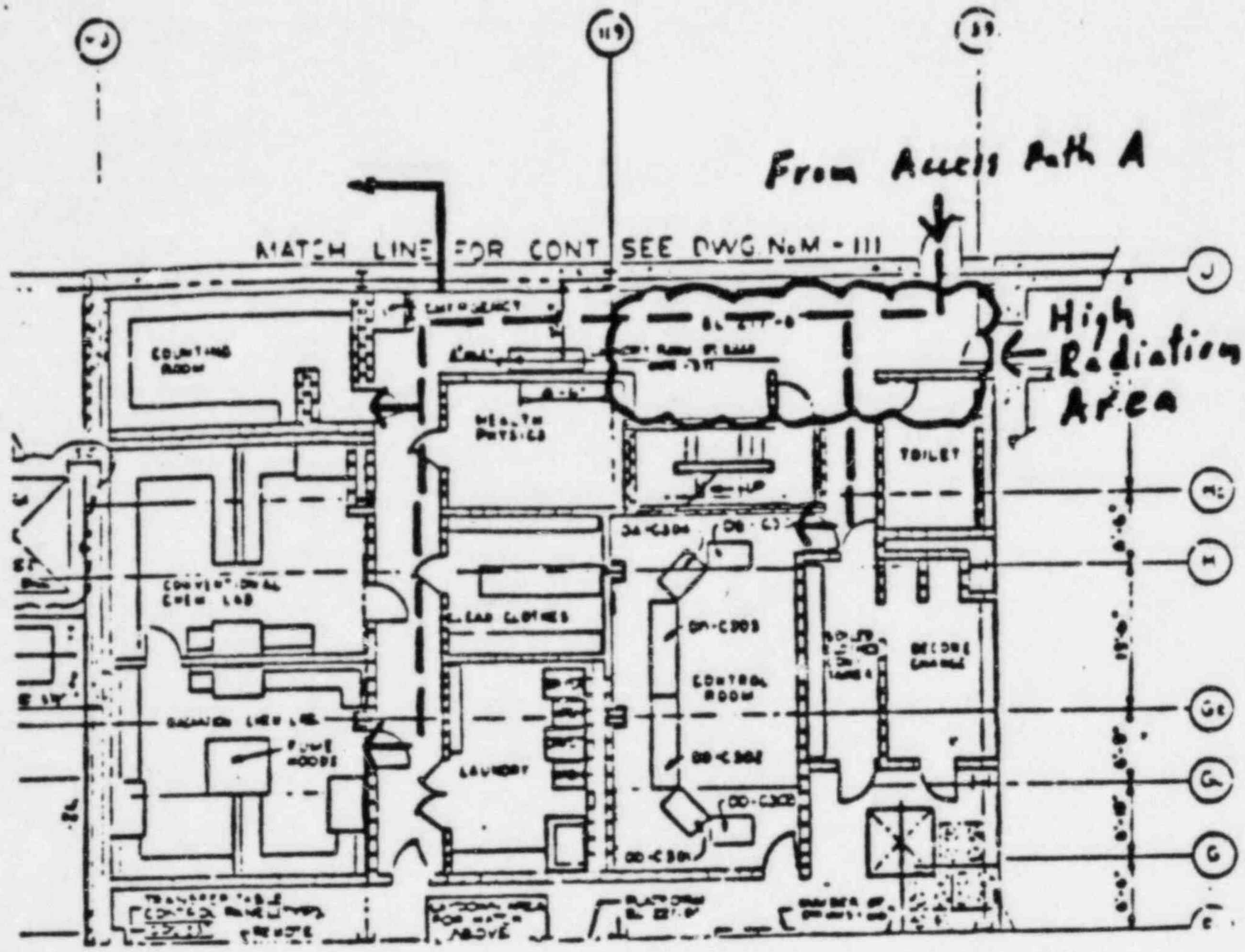

--- See Note on sheet 1 of 3

Note: ~~The layout of this area will be revised per PLB-11443, dated 9/26/80. However, the layout revision will not affect the designation of the high radiation area shown on this figure.~~

BECHTEL SAN FRANCISCO		
LIMERICK GENERATING STATION UNITS 1 & 2 PHILADELPHIA ELECTRIC COMPANY		
EQUIPMENT LOCATION RADWASTE ENCLOSURE PLAN AT EL 217-0"		
JOB NO.	DRAWING NO.	REV.
8031	M-142	8

FIGURE 1.13-1 Sheet 2 of 3

Access Path C

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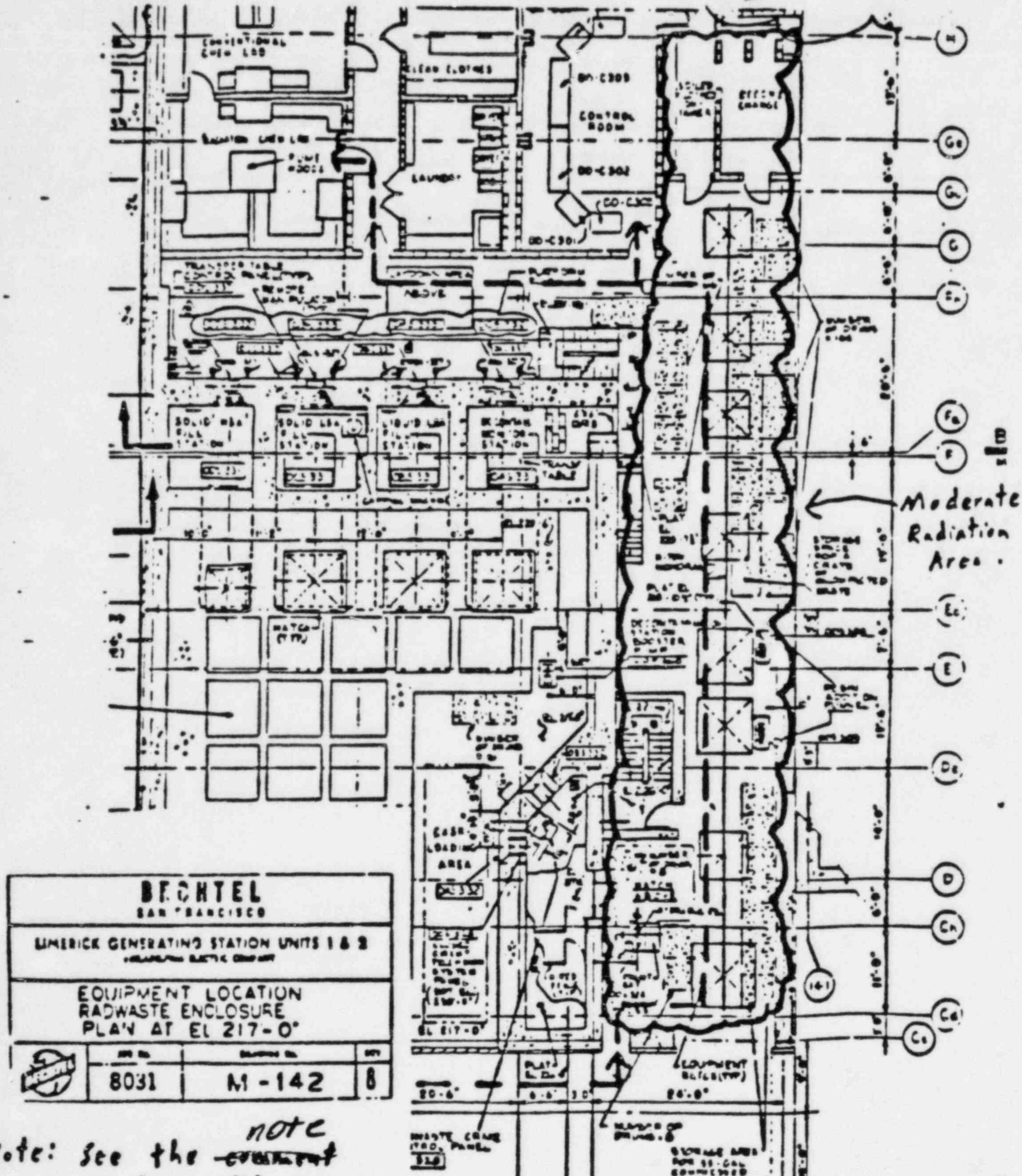

Note: See the ~~comment~~ note on ~~figure (2).~~ Sheet 1 of 3.

FIGURE 1.13-1

Sheet 3 of 3

Dose Rate Reduction Fac[illegible]rs for ECCS/RHR Piping
(0 to 24 HRS)

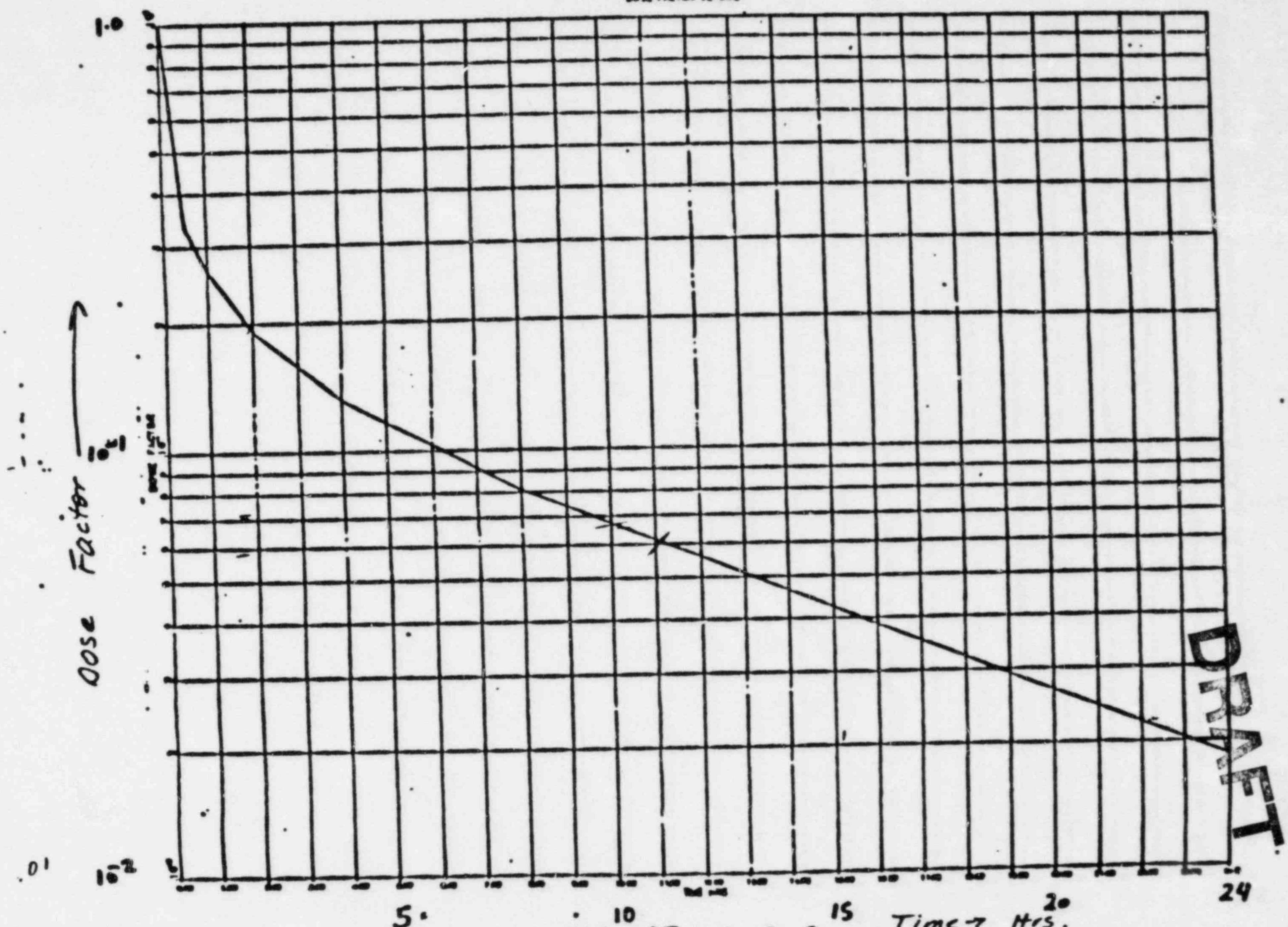

FIGURE 1.13-2

EK-179

Dose Rate Reduction Factors for ECCS/RHR Piping
(24 to 720 Hrs)

DRAFT

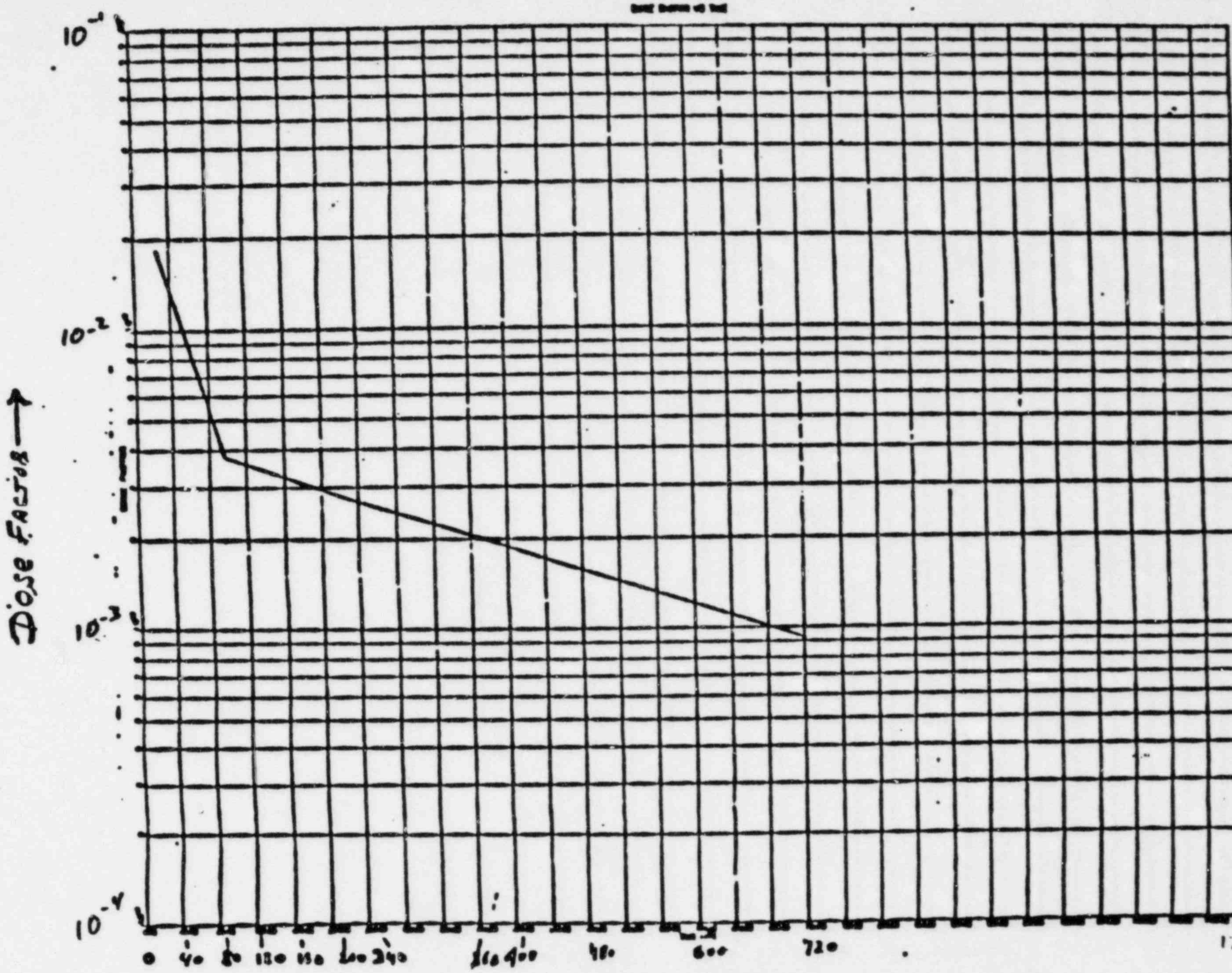

FIGURE 1.13-3

DRAFT RESPONSE TO

ITEM #4 DSER FROM

THE RAB

gases could escape from the plant structures and be drawn into the supply air intake.

There are four independent monitors, separated in accordance with IEEE 279-1971, that monitor air inside the control room intake duct. These inline monitors respond to the gross radioactivity in the vicinity of the detectors. Each monitor provides three alarm conditions: low, high, and high-high. The low and high alarms trip the control room annunciator. The high-high alarm trips the control room fresh air isolation valves and starts the control room emergency fresh air supply, which provides for the filtration of the incoming air through HEPA/charcoal filters. The trip of either monitor A or B shuts off the control room fresh air supply, and the trip of either monitors A and C, or B and D, starts the control room emergency fresh air supply. (See Section 6.4 for a more detailed discussion of control room isolation on detection of high radiation.)

11.5.2.1.5 Control Room Emergency Fresh Air Radiation Monitors

This monitoring system is actuated by the tripping of the control room fresh air supply isolation valves. Particulate and iodine radioactive isotopes are removed by the HEPA and charcoal filters. Radioactive noble gas concentration is measured and continuously recorded. These inline monitors detect gross radiation only. Two monitors, separated in accordance with IEEE 279-1971, monitor sample air from the control room emergency fresh air duct.

Each monitor provides three alarm conditions: low, high, and high-high. These alarms trip annunciators in the control room.

All requirements of this system are identical with those of the control room air supply radiation monitors except that no provisions for valve closure are made.

11.5.2.1.6 Primary Containment Post-LOCA Radiation Monitor

This monitoring system is comprised of four ion chamber sensors for the primary containment in the event of a loss-of-coolant accident (LOCA). After such a postulated accident the monitoring system measures the gross radioactivity present in the containment atmosphere. This information is transmitted to control room personnel to provide them with a basis for making safety-related decisions. ~~No trip capability or upscale annunciator trips are provided~~. A downscale annunciator is provided to indicate instrument malfunction. ~~The four sensors are separated in accordance with IEEE 279 requirements.~~

INSERT (A) ¶

(replace this sentance with the following:) The primary containment post-LOCA radiation monitoring system provides a trip signal to the containment sump pumps on an upscale alarm indication.

INSERT (A) PAGE 11.5-9 FS-229 SHT 1 OF 2

THE SENSORS ARE LOCATED IN SEPARATE AREAS OF CONTAINMENT TO PROVIDE INDEPENDENT MEASUREMENTS AND "VIEW" LARGE FRACTIONS OF THE CONTAINMENT VOLUME.

INSERT → (See Figure 12.3-16 for locations)

CONSIDERATION TO ACCESSIBILITY FOR MAINTENANCE AND CALIBRATION WAS GIVEN IN THE SELECTION OF THE SENSOR LOCATIONS.

THE SENSORS ARE LOCATED IN RELATIVELY OPEN AREAS TO PREVENT SHIELDING THAT COULD IMPAIR THEIR DETECTION FUNCTION.

THE MONITORING SYSTEM PROVIDES ENERGY RESPONSE FROM 60 keV TO 3 MeV, WITH UNIFORM RESPONSE WITHIN ± 20% FROM 80 keV TO 3 MeV.

ON SITE CALIBRATION OF THE MONITORS WILL BE PERFORMED WITH A CALIBRATED 100 mCi GAMMA SOURCE THAT WILL PROVIDE AN EFFECTIVE DOSE RATE OF 10 R/HR. A BUILT-IN CURRENT SOURCE

DRAFT

(CONTINUED)

INSERT (A) PAGE 11.5-9 FS-229 SH 2

IS PROVIDED IN THE MONITORS TO ALLOW CALIBRATION CHECKS THROUGH ELECTRONIC SIGNAL SUBSTITUTION FOR THE DECADES ABOVE 10 R/HR.

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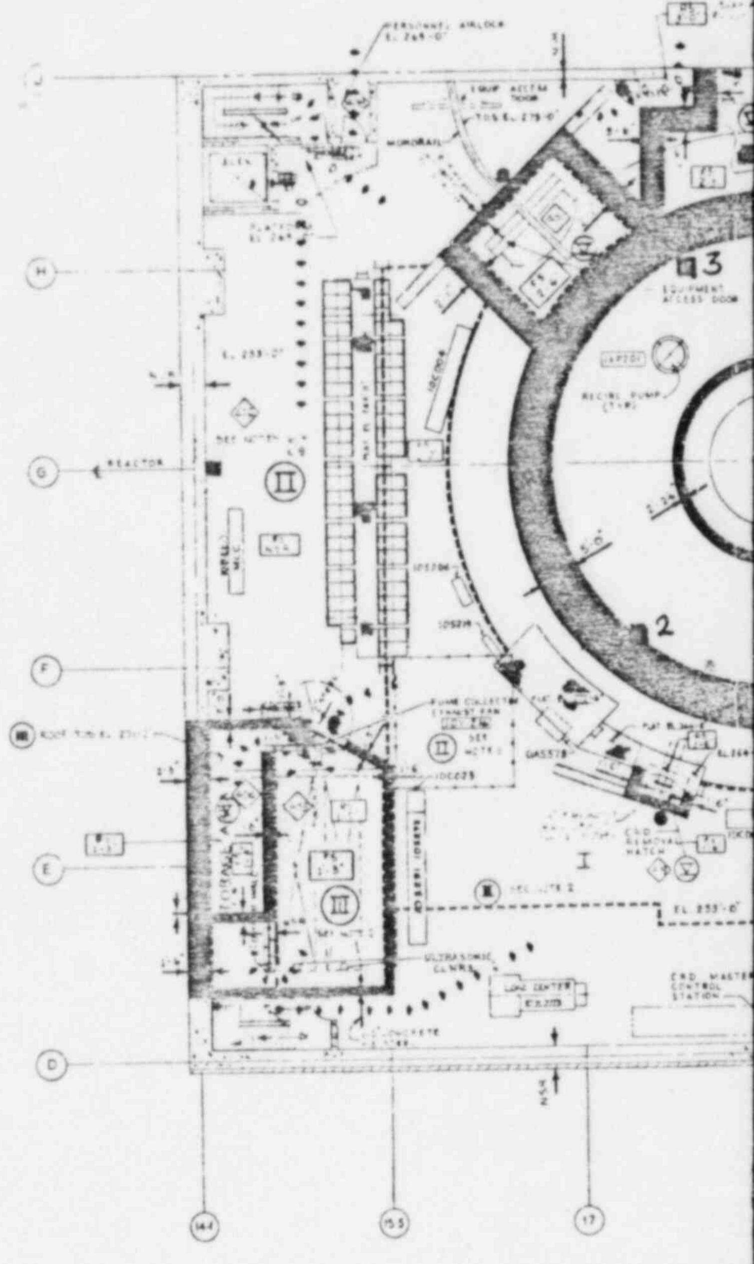
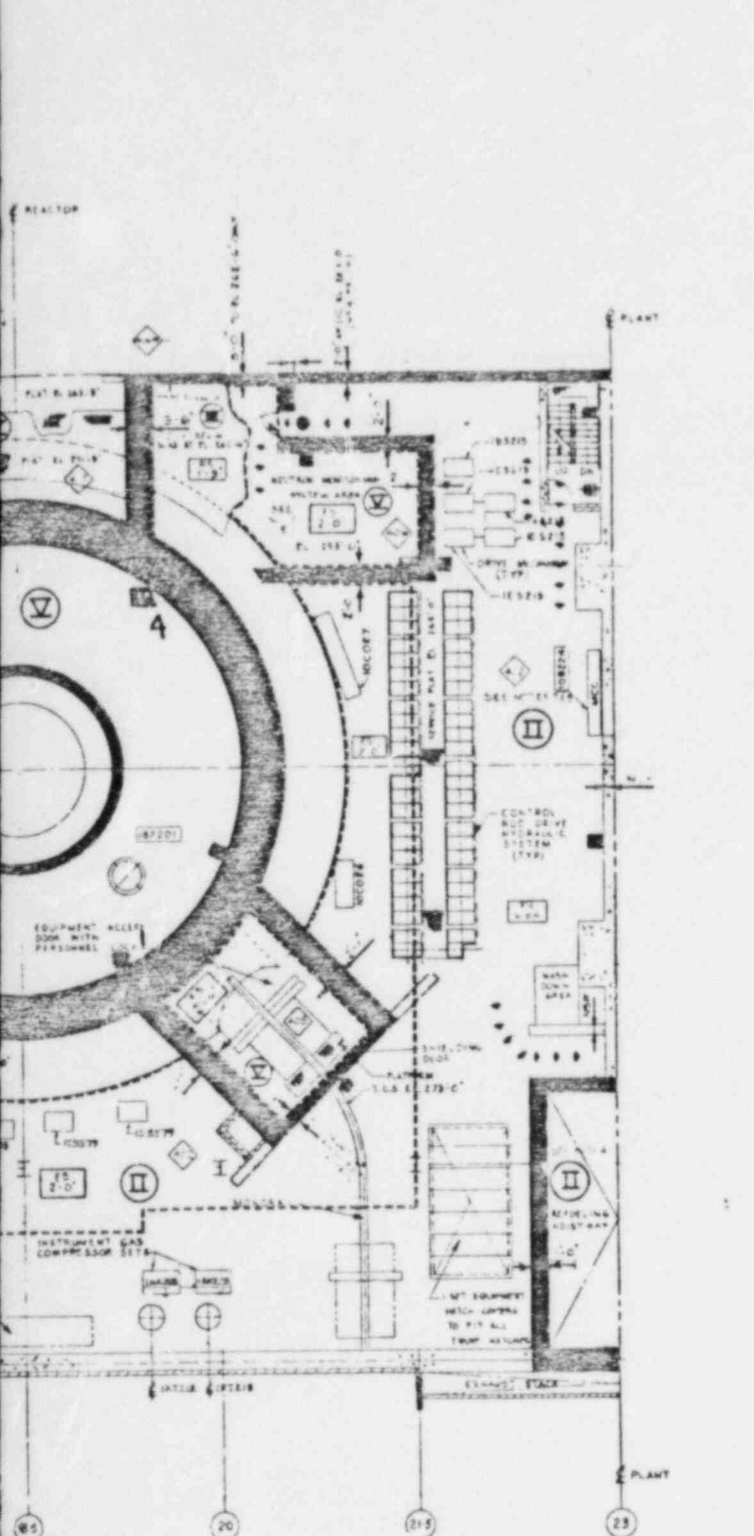
■ indicates approximate locations of high range drywell area radiation monitors.

#3 el. 258′ #1 el. 272

#4 el. 258′ #2 el. 282





LEGEND

A. PLANT RADIATION ZONES

DESIGNATION	MAX. DESIGN DOSE RATE (mR/HR)	DESCRIPTION
I	≤ 0.5	NO RADIATION SOURCES, NO RADIOLOGICAL CONTROL REQUIRED
II	≤ 2.5	LOW RADIATION SOURCES, RADIOLOGICAL CONTROL REQUIRED
III	≤ 15.0	LOW TO MODERATE RADIATION SOURCES, RADIOLOGICAL CONTROL REQUIRED
IV	< 100	MODERATE RADIATION SOURCES, RADIOLOGICAL CONTROL REQUIRED
V	≥ 100	HIGH RADIATION SOURCES, RADIOLOGICAL CONTROL REQUIRED

B. DRAWING SYMBOLS

1. PERSONNEL ALERT BARRIER
2. LOCKED PERSONNEL BARRIER
3. AREA RADIATION MONITOR
4. TYPICAL TRAFFIC PATTERN TO RADIATION AREAS
5. UPPER RADIATION ZONE NUMERAL FOR FULL POWER OPERATION
6. LOWER RADIATION ZONE NUMERAL FOR 24 HRS AFTER SHUTDOWN OF OPERATION
7. ROOM NUMBER OF RADIATION AREA
8. NSR SIGNIFIES NO SHIELDING REQUIREMENT
9. FS SIGNIFIES FLOOR SLAB
10. RS SIGNIFIES ROOF SLAB
11. EXTENT OF FLOOR SLAB SHIELDING REQUIREMENT
12. EXTENT OF WALL SLAB SHIELDING REQUIREMENT
13. PERSONNEL AND EQUIPMENT DECONTAMINATION STATIONS ARE LABELLED
14. PERSONNEL ACCESS BARRIER FOR RADIOLOGICAL CONTROL

C. SCALE

D. NOTES

9. THESE AREAS MAY BE AS HIGH AS ZONE V NEAR THE SCRAM DISCHARGE PIPING

LIMERICK GENERATING STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

SHIELDING AND RADIATION ZONING DRAWING, REACTOR ENCLOSURE PLAN AT EL 253'-0"

FIGURE 12.3-16

DRAFT RESPONSE TO
ITEM #5 DSER FROM
THE RAB

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3. High- and low-volume portable samplers capable of attaching filters and charcoal cartridges for particulate and iodine monitoring.

The ventilation system monitors are located at positions which provide representative air concentrations and a rapid indication of abnormal conditions. Those systems which require HEPA filtration have monitors upstream of the filters. Both the inline Geiger-Müller tube and beta scintillator, and offline particulate, iodine, and noble gas monitoring configurations are utilized. Readout and annunication are provided in the main control room and/or radwaste control room. Emergency dc power is provided in the event of loss of offsite power. The detectors are calibrated routinely and after any maintenance work is performed on the detector.

Continuous air monitors (CAMs) are located in freely accessible areas where airborne radioactivity is most likely to exist. These CAMs are mobile and can be moved from area to area as deemed necessary by plant conditions or maintenance operations. CAMs incorporate either fixed or movable filters for the collection of particulate activity, which is monitored directly by a detector. Readout is recorded in CPM. The filters can be removed for further analysis using counting room instrumentation. Audible and visual alarms indicate when set point levels have been exceeded. The detectors are calibrated routinely and after any maintenance work is performed on the detector.

The CAM's primary function is to indicate trends and sudden changes in airborne activity. Typical locations are solid waste handling areas, spent fuel pool areas, and the reactor operating floor and turbine building. The monitoring system is capable of detecting ten MPC-hours of particulate and iodine radioactivity from compartments which have a possibility of containing airborne radioactivity, and which normally may be occupied by personnel. A flexible hose can be attached to the monitor intake and inserted into a cavity or work area to detect the presence of localized airborne activity. Conformance to Regulatory Guide 8.2 is discussed in Section 12.5.1. The guidance of Regulatory Guide 8.25 will be followed.

Insert 1

~~Radioactive airborne concentrations are determined by analysis of grab samples obtained routinely throughout the plant~~. Low- and high-volume samples with filter paper or charcoal cartridges are used. These are described in more detail in Section 12.5.3.1.3.

Insert ①

ventilation monitors and CAMS are used as trending devices and will indicate areas and times needing special samples taken.

Alarm set points are set at very low levels to ensure close respiratory controls. CAMs, however cannot account for inversion conditions or properly identify isotopic content of the air. When a setpoint is reached grab air samples are taken and analyzed in the counting lab. The MPC hours are isotopically calculated by the computer program or done manually. Appropriate actions can then be taken based on accurate data.

Potentially airborne accessible walk areas are air sampled at regular intervals. Areas not routinely sampled require an RWP for entry. Air samples are taken upon initial entry. The survey/sampling frequency is designed for that area and controlled by the RWP.

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DRAFT RESPONSES

TO ITEM #6 DSER

FROM THE RAB

SENIOR HEALTH PHYSICIST

DRAFT

NAME: Kenneth H. Taylor II

EDUCATION AND TRAINING

1974 to 1976	U. S. Navy Nuclear Propulsion Schools: Engineering Laboratory Technician (13 wks.), Reactor Plant Operator (26 wks.), Nuclear Power (24 wks.) Machinist Mate (16 wks.)
1981	Purdue University, B.S. in Environmental Health
1982	Purdue University, M.S. in Health Physics
1982	Engineer Orientation Training Program - Four week course in the fundamentals of Nuclear Power Plant Operation

WORK EXPERIENCE

6/82 to present — Senior Health Physicist
Limerick Generating Station

Assist in pre-startup development of Health Physics & Chemistry group and radiation exposure control and measurement program. Participated in Peach Bottom Atomic Power Station HP&C activities, including procedure development, plant operations, and activity evaluations.

1978 to 6/82 — Assistant Laboratory Manager/Senior Laboratory Technician - School of Civil Engineering, Purdue University

Supervised design, fabrication, operation, calibration and repair of equipment in the Hydraulics and Systems Engineering Laboratory.

Cost estimating and purchasing.

Maintained field data collection stations.

Developed and installed a computerized remote sensing data collection system.

Figure 2.1.1(a)

1976 - 1978 Safety Inspector/Engineering Laboratory Technician
USS Eisenhower, United States Navy

Responsible for safety and industrial hygiene related to reactors, propulsion plants and electrical/electronic equipment.

Developed control program for benzene products to comply with emergency standard.

DRAFT

Assisted in development and implementation of safety procedures.

Edited weekly and monthly internal safety publications.

Technical guidance to departmental safety personnel.

1974-1976 Staff Instructor/Engineering Laboratory Technician
U. S. Navy

Instructed at D1G Reactor Prototype...radiological controls, reactor plant chemistry, plant theory and operation, emergency and casualty procedures.

Performed radiation, contamination and airborne radioactivity surveys; exposure monitoring and control, personnel dosimetry, radioactive waste processing and accountability, decontamination and related functions in support of reactor refueling and plant overhaul.

Figure 2.1.1(b)

DRAFT RESPONSES

TO ITEM #7 DSER

FROM THE RAB

DRAFT

revise as indicated

TABLE 471.6-1

HEALTH PHYSICS AND CHEMISTRY INSTRUMENTATION

INSTRUMENT	SENSITIVITY/RANGE	Quantity Need Now	Quantity Need Later
Direct Current Plasma Jet ~~Flame~~ Spectrophotometer	1 ppm	1	0
Gas Chromatograph	H_2 - 30 ppm ; O_2 - 20 ppm	1	0
Ion Chromatograph	1 ppb	1	0
Photoelectric Colorimeter	1 ppm	1	0
Turbidimeter	0.1 JTU	4	0
U.V. Spectrophotometer	1 ppb	1	0
T.O.C. Analyzer	50 ppb	1	0
Germanium Counting System	10 min. count (μCi): Co^{60}, Cs^{134}, Cs^{137}, I^{131} $1x10^{-4}$ Zn^{65} Noble Gases - $1x10^{-3}$	3	1
Proportional Counter	Beta Efficiency 10-50% (E dependent)	3	1
Whole Body Counter System	<1/20 of the International Committee of Radiation Protection (ICRP) Pub. 30 Maximum Permissible Body Burden (MPBB) levels.	1	0
Radiosurveillance ~~Portable Survey~~ Instruments			
RM-14 with HP 210T	0 - 50 to 50,000 cpm	50	50
RM-16 with HP 200	100 cpm/ to 1M cpm		
RM-16 with RD 17A	0 - 0.1 mr/hr to 1R/hr		
E-520 with HP 270	0 - 0.2 mr/hr to 2000 mr/hr		
Teletector	0 - 0.1 mr/hr to 1000 R/hr	3	3
E-520 with HP 230	0 - 0.2 mr/hr to 2000 mr/hr	N	
PRM-6 with AC-3	0 - 500 cpm to 500k cpm	2	2
PRS-2P - NRD	0 - 0.2 mr/hr to 10 R/hr		
RO2A	0 - 50 mr/hr to 50 R/hr	10	10
RO7	1 mr/hr to 20,000 R/hr	2	0
RO2	0 - 5000 mr/hr	25	25

replace with Insert A

Insert A

DRAFT

Radiosurveillance Instruments	Sensitivity/Range	Quantity Needed Now	Later
RO-2	0-5000 mR/hr	25	25
RO-2A	0-50 R/hr	10	10
E-520	0.2-2000 mR/hr	11	11
E-140N	0-50,000 cpm	11	11
HP270 probe	.1 mR/hr-10 R/hr	15	15
HP210AL probe	——	15	15
HP210T probe	——	5	5
Teletector	0-1000 R/hr	3	3
EC4-X (6 cables)	0.01 mR/hr-10,000 R/h	10	10
DA1-6 probe (for EC4-X)		10	10
PRM-6 with AC-3	500-500,000 cpm	2	2
PNR 4	5-5000 mR/hr	4	0
HP280 - 3 in. sphere		2	0
R07 + BM midrange detector + BH high range detector, + BX5 - 5 ft. extender, + 2 (15 ft.) cables	0-200 Rhr; 0-20,000 R/hr	2	0
Under water probe	0-5000 R/hr 5-5000 mR/hr	2	1
RM-14 with HP210T	5-50,000 cpm	50	50
Alpha + Beta CAMS		8	5
Iodine CAMS		3	2
Lo-Vol air samplers	——	50	30
Hi-Vol air samplers	——	20	10

Insert A continued

DRAFT

[illegible] Instruments	Sensitivity/Range	Quantity Needed Now	Later
Portal monitors	approx MDA [illegible]	4	4
Self Reading Dosimeters:			
Charging Units		10	
0-200 mR	0-200 mR	3000	as
0-500 mR	0-500 mR	1000	needed
0-1500 mR	0-1500 mR	500	
Portable MCA + HP Ge detector + 24 hr. dewar		1	1
E-530N with HP220	.02-20R/hr	2	1
Shepherd 142-10 panoramic calibrator	0-1,600 mR/hr	1	0
Shepherd 89 calibrator	.01 mR/hr-1200 R/hr	1	0
Condenser R meter	——	1	--

DRAFT RESPONSE TO
ITEM #9 DSER FROM
THE RAB

Improved Inplant Iodine Instrumentation Under Accident Conditions (III.D.3.3)

Each Licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentrations in areas within the facility where plant personnel may be present during an accident.

RESPONSE

DRAFT

Sampling methods and procedures will be implemented at Limerick Generating Station which will permit the measurement of in-plant iodine concentrations during accident conditions. A description of this method is as follows:

The sampling method uses portable air samplers with a combination particulate filter and iodine sampling cartridge sampling head. The sampling heads use a glass fiber particulate filter and a CESCO style (2.25" dia. x 1.04" thickness) iodine charcoal cartridge. The cartridge normally used is the CESCO type charcoal cartridge. When long sampling times are required a larger capacity charcoal cartridge is available. During emergency conditions, with high xenon or krypton concentrations potentially present, either a silver zeolite or a silver impregnated silica-gel adsorber cannister will be employed.

Iodine activity on the sample cartridge will be determined by gamma isotopic analysis using a computer based multi-channel analyzer with high resolution intrinsic germanium detectors located in the Limerick Counting Room. The Counting Room is located in the Radwaste Enclosure at elevation 217'. An assessment of the NUREG-0737 shielding study indicates that the Counting Room dose rates and airborne radioactivity concentrations are low enough to permit sample analysis during accident conditions.

Isotopic analysis will permit iodine identification in the presence of xenon and krypton. If the analysis of iodine becomes impossible due to interference (high background) from xenon or krypton, then either silver zeolite cartridges will be used, or the charcoal cartridge will be purged with clean bottled nitrogen or breathing air to reduce the interference. If the use of silver zeolite or the silica-gel adsorber does not sufficiently reduce the xenon or krypton interference, the silver zeolite cartridges will also be purged with clean bottled nitrogen or bottled breathing air available on site.

The Health Physics technical staff will be trained in the implementation of this postaccident procedure.

GWM/dg/W/6