ATTACHMENT 1

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station, Unit No. 1

Design Review of Plant Shielding
of Spaces for Post-Accident Operation

QUAD-1-80-040

DESIGN REVIEW OF PLANT SHIELDING OF SPACES FOR POST-ACCIDENT OPERATION
NUREG-0578 SECTION 2.1.6.b

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1.0 INTRODUCTION

1.1 NUREG-0578

The Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission established the TMI-2 Lessons Learned Task Force shortly after the TMI-2 accident in the spring of 1979. The purpose of the Task Force is to identify and evaluate those safety concerns raised by the TMI-2 accident that require generic licensing actions (beyond those already specified in I.E. Bulletins and Commission Orders). The Task Force is charged to identify, analyze, and recommend changes to both specific licensing requirements and the general licensing process for nuclear power plants based on the lessons learned.

In July 1979, the Task Force issued its first report, NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations". This report contains a set of short-term recommendations which are to be implemented in two stages over the 18 month period following publication. The recommendations are to be implemented for operating plants, plants under construction, and plants with pending construction permit applications. There are 23 specific recommendations in 12 areas where implementation is judged to provide substantial additional protection for the public health and safety.

This study by Nuclear Services Corporation specifically addressed implementation of Section 2.1.6b of NUREG-0578, "Design Review of Plant Shielding of Spaces for Post-Accident Operations" for Beaver Valley Unit 1. This particular section is concerned with post-accident control of radiation in systems outside containment. At TMI-2, systems outside the containment building contained radioactive material to the extent that personnel access was impaired due to high radiation levels. Several deficiencies were noted. For example, the licensee had little knowledge of the plant's operational leakage pathway characteristics, expected post-accident radiation levels, and area occupancy requirements.

Shielding provision for personnel access were inadequate. Section 2.1.6b recommends: (1) performing a shielding design review of systems processing primary coolant outside containment, (2) identifying any areas or equipment that are vital for post-accident occupancy or operations, and (3) taking steps to assure that access and performance will not be unduly impaired due to radiation from these systems.

2.0 SUMMARY

Dosage and occupancy requirements of the systems which have post-accident radionuclide flow paths from inside containment to outside containment are summarized in Table 2-1.

2.1 Chemical and Volume Control System

Use of the Chemical and Volume Control System (CHS) must be avoided when the reactor coolant is highly radioactive due to an accident. A pathway for the radioactive reactor coolant is through the CHS letdown line. Use of the normal letdown line will most likely result in highly radioactive liquid and gas flow in the CHS, boron recovery system, and gaseous waste system.

Therefore, an alternate letdown flow path through the existing excess letdown heat exchanger, under containment isolation, was investigated. In this scheme, the containment isolation is assumed so that the radioactive gases and liquid are contained in the containment. The charging pumps are assumed to take suction from the refueling water storage tank. The normal letdown line in CHS, however, is closed to prevent radioactive reactor coolant from flowing out of the containment into the CHS and other systems. It is further assumed that two reactor coolant pumps are running. Under these conditions, the excess letdown flow line through the excess letdown heat exchanger may be effectively used to maintain the water levels in the pressurizer and the reactor pressure vessel. The letdown flow is discharged to the containment sump through a relief valve in the bottom of the primary drain transfer tank (DG-TK-1). To ensure the letdown flow into the containment sump, it is recommended that the presently existing lock-shut manual valve in the excess letdown bypass line to the containment sump be replaced with a remotely operated valve.

2.2 Containment Isolation

Containment Isolation Train A (CIA) signal will automatically close the letdown orifice isolation valves and stop the letdown flow from the reactor coolant system. The CIA signal is initiated by a containment high pressure signal or a safety injection. This project responds to TMI Lessons Learned,

Section 2.1.6b, additional shielding required for the increased radiation levels. The response to this section does not address high containment pressure of CIA actuation in a high radiation situation without a pipe break. In the future, Duquesne may have to respond to a high radiation CIA actuation in response to radiation in the primary coolant loop.

2.3 Safety Injection System

During the safety injection system recirculation phase, the contact dose rate on the floor above the pipe tunnel in the safeguards and containment contigous areas, when the low head safety injection pump discharge lines are recirculating the radioactive containment sump water, is estimated to be 5 R/hr one hour after the accident. Similarly, the contact dose rate on top of the pipe trenches in the Auxiliary Building at Elevation 722'-6", where the charging pump lines are flowing the containment sump water in the recirculation phase, is estimated to be 22 R/hr at one hour after the accident. Since these estimates are contact dose rates, the actual total body exposure rate to personnel will be much less, and the personnel should be able to pass by the areas, as required, under administrative control.

There is an area where recirculation lines are exposed in the west-wall pathway and over the pipe trench No. 1 near the charging pump cubicles at elevation 722'-6" in the Auxiliary Building. A very high radiation background will exist in this area during the recirculation phase and personnel should be restricted from this area.

Since the boron injection tank (BIT) in the Auxiliary Building at Elevation 722'-6" has 900-gallon capacity, the radiation level outside the BIT cubicle remains high with the radioactive containment sump water flowing through the BIT during the safety injection system recirculation phase. The calculated contact dose rate at the outer surface of the 2-ft concrete shield wall is 810 R/hr at one hour after the accident. Although occupancy is not required in the vicinity of the BIT under accident conditions, strict administrative control is necessary to ensure personnel are restricted from the BIT area.

Shortly after switch over to recirculation phase, the operator must operate the HHSI/charging pump suction and discharge valves manually from the south wall of the pump cubicles to set up redundant and independent flow paths from containment sump to cold legs. Although the operator will be exposed to radiation from the pipe trench No. 2 and Boron Injection Tank, the operator will receive dosage which is appreciably less than the allowable limit of 5 Rem whole body. The following change in operating procedure, however, is recommended to reduce radiation exposure:

- o Switch from injecting through the Boron Injection Tank line to the HHSI discharge through cold legs prior to start of recirculation phase, or
- o Delay switching from injection through the Boron Injection Tank line to the HHSI discharge through cold legs during recirculation phase to allow sufficient time for radioactive decay in the Boron Injection Tank and recirculation lines.

The recombiner inlet manual valves and cross-connect instrument air valve (1IA-90) to the containment instrument air are located in the pipe penetration area of safeguards where the LHSI lines with highly radioactive containment sump water are routed. It is recommended that these valves be made remotely operable.

2.4 Containment Depressurization

Recirculation spray pumps of the Containment Depressurization System, which take suction from the containment sump, are used to maintain the containment at subatmospheric pressure following an accident. Two of the four recirculation spray pumps are located outside the containment. The outside recirculation spray pumps, components, and piping are in enclosed shielded areas which are located in the west safeguards area. There is no need for personnel access to the pump cubicle area.

2.5 Containment Vacuum and Leakage Monitoring System

The Containment Pressure Sensing Subsystem of the Containment Vacuum and Leakage Monitoring System will contain radioactive gases after an accident. Four 3/8" pipes are led from inside containment to the pressure sensing instruments which are enclosed by concrete shield walls outside the containment. Based on the estimated radiation levels and the occupancy requirements, it is concluded that personnel exposures can be maintained within the allowable limits without any shielding modifications.

2.6 Supplementary Leak Collection and Release System

The Supplementary Leak Collection and Release System ensures that radioactive leakage from the reactor containment is collected and filtered for iodine removal prior to discharge to the atmosphere at an elevated release point. The calculated radiation levels after an accident indicate that passage near the vent ducts and the shielded filter banks is permissible for personnel as the need arises. It is expected that post-accident occupancy requirements in the vicinity of system ducting and the filter bank will be minimal. Therefore, it is concluded that additional shielding for the system is not necessary. Personnel exposures can be controlled by implementing personnel access restrictions.

2.7 Post-DBA Hydrogen System

Two redundant hydrogen recombiner systems are installed in the safeguards area to maintain hydrogen content in the containment below the lower flammability limit in the event of LOCA. There is a biological shield wall between the recombiners and the control panels. Calculations indicate that an operator standing on grating in the control panel room will receive most of the radiation dosage from two low head safety injection lines which are routed approximately 14 feet below the 2-ft concrete floor which is adjacent to the grating. No appreciable shielding protection is afforded to the operator by the 2-ft concrete floor. A radiation level of 3000 R/hr at the operator's location one hour after

the accident has been determined. At one hour after the accident, the operator will have to start the H_2 analyzer from the control panel. The total amount of occupancy time required for this action should not be more than 30 minutes. At approximately 24 hours after the accident, the recombiner is started from the control panel. Approximately 30 minutes are required for startup. Based on the dose rates and personnel occupancy requirements at this location, personnel exposures far in excess of permissible limits would result. Therefore, the control panels must be relocated to a concrete floor area or the control panel room must be shielded from the LHSI lines. One additional problem area exists that requires resolution. Manual valves 101, 102. 103, and 104 on the recombiner inlet line have to be opened prior to start of the hydrogen analyzer and recombiner. These valves are located in the pipe penetration area of the safeguards area where the LHSI lines containing highly radioactive containment sump water are routed. Based on the high radiation level predicted for the LHSI lines and requirement for operation of these valves, it is expected that personnel overexposure would occur with the existing arrangement. Therefore, it is recommended that the manual valves be either changed to remotely operated valves or made to open from the floor above with shielding and using reach rods.

2.8 Post-Accident Sampling System

It was determined that the existing reactor building sampling system was not designed for the post-accident sampling. The containment isolation valves (both inside as well as outside containment) of all sampling lines are operated from one solenoid valve and a limit switch inboard (SOV-SS-000DA), and one solenoid and limit switch outboard (SOV-SS-000B). The existing sampling panel was not intended for use with source term radiation levels. Therefore, it is recommended that a post-accident shielded sampling system be built for sampling both liquid hot leg and containment air samples.

TABLE 2-1 SUMMARY OF DOSAGE AND OCCUPANCY REQUIREMENTS

SYSTEM	AREA	EQUIPMENT	SHIELDING	RADIATION LEVEL VS. TIME (HRS)* ZERO	ONE	10	24	REQUIREMENTS	RESOLUTION
CHS With Use of Letdown Line	Floor Above Penetration Room	Charging and Letdown Lines	2' Thick Concrete Floor	32 R/Hr		5.2 R/Hr	2.4 R/Hr	Unlikely	Administrative Control
	Auxiliary Bldg.	Pipe Chase (Letdown or Charging Line)	2' Thick Concrete Shield	22 R/Hr		3.6 R/Hr	1.7 R/Hr	Possibly	Administrative Control
	Auxiliary Bldg.	Outside Volume Control Tank Enclosure	3.5' Thick Concrete Wall	41 R/Hr		3.4 R/Hr	1.2 R/Hr	Possibly	Administrative Control
	Auxiliary Bldg.	Outside Charging Pump Cubicle	2' Thick Concrete Floor and Wall	3.9 R/Hr		0.7 R/Hr	0.3 R/Hr	Possibly	Administrative Control
Safety Injection System Recirculation Phase	West Safeguards	LHSI Pump Cubicle - Directly Above	2' Thick Floor		2.4 R/Hr	0.6 R/Hr	0.3 R/Hr	Unlikely	Administrative Control
	Safeguards and Cont. Contiguous Areas	Pathway Above Pipe Tunnel - LHSI Pump Lines	2' Thick Floor		4.6 R/Hr	1.0 R/Hr	0.4 R/Hr	Unlikely	Administrative Control
	Auxiliary Bldg.	Pathway Above Pipe Vault Charging Lines	2' Thick Floor		21.8 R/Hr	4.8 R/Hr	2.4 R/Hr	Possibly	Administrative Control
	Auxiliary Bldg.	Charging Pump Cubicle - Directly Above	2' Thick Floor		410 mR/Hr	96 mR/Hr	47 mR/Hr	Unlikely	Administrative Control
	Auxiliary Bldg.	Charging Pump Cubicle - Standing Aside	2' Side Wall		410 mR/Hr	96 mR/Hr	47 mR/Hr	Possibly	Administrative Control
	Auxiliary Bldg.	Outside Boron Injection Tank	2' Thick Wall		810 R/Hr	180 R/Hr	87 R/Hr	Unlikely	Administrative Control
Containment Depressurization Outside Recirc. Spray Pump System	West Safeguards	Outside Recirc. Spray Pump Cubicle - Directly Above	2' Thick Floor		2.4 R/Hr	0.6 R/Hr	0.3 R/Hr	Unlikely	Administrative Control
	West Safeguards	Recirc. Spray & LHSI Pump Lines - Pipe Tunnel	2' Thick Floor		4.8 R/Hr	1.2 R/Hr	0.6 R/Hr	Unlikely	Administrative Control

* Hours elapsed following time zero of the accident or event. Radiation level is in contact with shielding.

TABLE 2-1 SUMMARY OF DOSAGE AND OCCUPANCY REQUIREMENTS (Continued)

SYSTEM	AREA	EQUIPMENT	SHIELDING	RADIATION LEVEL VS. TIME (HRS)* ZERO	ONE	10	24	REQUIREMENTS	RESOLUTION
Containment Vacuum and Leakage Monitoring System, Containment Press. Sensing Subsystem	Contiguous Area to Containment	3/8" Pipe, Valves and Instrument (4 sets)	2' Thick Wall		6.8 mR/Hr	1.3 mR/Hr	0.5 mR/Hr	Infrequent	Administrative Control
Supplementary Leak Collection and Release System	Safeguards and Containment Contiguous Areas and Auxiliary Bldg.	Main Duct at 10' Distance	None	200 mR/Hr				Infrequent	Administrative Control
		Main Duct at 20' Distance	None	74 mR/Hr				Infrequent	Administrative Control
	Auxiliary Bldg.	Outside Filter Bank	2' Thick Wall	290** mR/Hr				Infrequent	Administrative Control
Recombiner System	Hydrogen Recombiner Control Panel	LHSI Line Below Floor and Control Panel Room on Grating	Concrete Shield Between Control Panel and Recombiner		3,000 R/Hr	1,400 R/Hr	1,000 R/Hr	One hr. Max. residency per person	Relocate control panels to concrete floor area or shield control panel room from LHSI lines
Post-Accident Sampling System (Proposed)	Auxiliary Bldg.			(Radiation Level will be Low)				Infrequent	Administrative Control

* Hours elapsed following time zero of the accident or event. Radiation level is in contact with shielding.
** Assume all iodines filtered for 30 minutes after time zero.

3.0 BASIS OF EVALUATION

3.1 NUREG-0578

NUREG-0578, Section 2.1.6b is addressed in this study and described in the Introduction Section 1.0. In analysis of the original source terms and sources for radioactive material transport outside containment, two general accident conditions are important in assessing the effects of system shielding.

In the first case, no breach in the Reactor Coolant System integrity is assumed and that water being handled originates in the reactor coolant system. The source terms used for this analysis are listed in Table 3-1 under Liquid Systems. Because of the assumption of no breach of the reactor coolant line, the fission gases are assumed to remain in the water where the dilution volume is only the contents of the reactor coolant system itself.

In the second case, it is assumed that the systems are handling water which originates in the containment sump. Because the equilibrium value of gas removal from the water has not been reached at the time immediately following the accident, a maximum source term value was assumed where all the noble gases and halogens remain the the water phase. The dilution volume is assumed to be all water in the refueling water storage tank, accumulators (3), chemical addition tank, and boron injection tank, or 65,500 ft^3 including the reactor coolant system.

The containment atmosphere was assumed to follow the guidelines as outlined in Table 3-1 Containment Air. Although an intact coolant gaseous source term inside containment would be minor when compared to a LOCA condition, the maximum source term was used for all situations (both LOCA and no breach of Reactor Coolant Systems conditions).

3.2 Source Terms

The radiation source term used for the Beaver Valley Unit 1 post-LOCA radionuclide distribution and shielding study was based on the guidelines

given in TID-14844 and TMI-2 Lessons Learned Short Term Recommendations, Section 2.1.6b as outlined in Table 3-1.

The digital computer code designated as "Origen" was used to obtain the initial radioactivity released and subsequent containment/reactor water radiation level calculations. Origen calculates detailed isotopic compositions for a range of reactor conditions including fuel irradiation, neutron activation, and radioactive decay. The particular parameters used in Origen for the Beaver Valley Unit 1 study were acquired from BVPS and are outlined in Table 3-2.

The final computer output is summarized in two printouts, one for the reactor coolant and one for containment atmosphere. Each printout is broken up into 12 groups of different mean energies in photons/second and MeV/watt-sec units. The printouts are reproduced in Table 3-3 for Liquid and Table 3-4 for Containment Air.

The liquid source terms from Table 3-3 were used as follows:

- For "intact" coolant lines (no pipe break but high radioactivity), the radioactivity was assumed to be uniformly distributed in the 9400 ft^3 reactor coolant volume. For example, dose rate calculations for the normal letdown flow path (Section 4.1.1) were based on this radioactive intact primary coolant flowing in the Chemical and Volume Control System.

- For a LOCA condition, it was assumed that all of the stored water in the refueling water storage tank, accumulators (3), chemical addition tank, and boron injection tank was injected into either the Reactor Coolant System or containment spray and ultimately appeared in the reactor containment sump. The total volume of that water is approximately 65,500 ft^3, including the reactor coolant. For the Safety Injection System and Containment Depressurization System which recirculate the reactor containment sump water, the radioactivity was assumed to be distributed in the 65,500 ft^3 of water.

- The gaseous source term from Table 3-4 was assumed to be uniformly distributed in the 1.86×10^6 ft^3 containment free volume. The resulting concentration of radioactivity was used in the dose rate calculations of the Post-DBA Hydrogen System and containment pressure sensing subsystem of the Containment Vacuum and Leakage Monitoring System.

3.3 CIA AND CIB Actuation

The Containment Isolation phase A (CIA) signal isolates all non-essential process lines on receipt of a signal from the actuated Safety Injection System (SIS). The Containment Isolation phase B (CIB) signal isolates remaining process lines (which do not include safety injection lines) on receipt of two out of four high-high containment pressure signals. Each line penetrating the containment has redundant isolation valves. Tables 3-5.1 through 3-5.8 give information on isolation valves installed outside containment. Tables 3-6.1 through 3-6.8 give similar information on isolation valves installed inside containment.

SIS and CIA are activated by any of the following.

1. Low pressurizer pressure in coincidence with low pressurizer water level.

2. High containment pressure.

3. Pressure in one steam line lower than the pressure in both of the other lines by a predetermined and set amount.

4. High steam flow in any of the three steam lines with low steam line pressure or low-low average temperature in the Reactor Coolant System.

5. Manual action.

CIB is actuated by high-high containment pressure.

3.4 Radionuclide Flow Paths

Section 2.1.6b of NUREG-0578 is concerned with post-accident control of radiation emanating from systems outside containment. Table 3-7 summarizes the systems that were considered for this study and identifies those systems as to which could transport radioactivity to areas outside the containment.

3.5 Vital Areas

Areas which must be accessible for post-accident operations (vital areas) are discussed below and are taken into consideration in system evaluations presented in Section 4.0. Accessibility to these areas is included in the discussion below.

3.5.1 Control Room

The main control room is located in the service building between the reactor and the turbine portions of the stations. The control room is in such an area as to allow easy access in an accident condition and remain relatively unaffected by the radiation levels. The area and design are shown in Figure 3-1.

3.5.2 Shutdown Panel

A shutdown panel is located two floors below the main control room. This panel provides the capability for hot shutdown if the control room is uninhabitable. Events making the control room uninhabitable will not render the shutdown panel area uninhabitable. The area is easily accessible and unaffected by accident conditions.

3.5.3 Emergency Power Supplies

These supplies are located below the main control room and are an easy access area relatively unaffected by an accident condition.

3.5.4 Instrument Areas/Local Operating Panels

There are no instruments or local operating panels in areas of high radiation deemed necessary for usage in an accident condition.

3.5.5 Radiochemistry Laboratory

The hot laboratory is surrounded by the change area, and a clean shop. These are areas of low contamination in an accident condition. Although the laboratory would be accessible for chemical analyses, the background radiation may be too high to use the existing counting equipment for accident monitoring. This subject needs to be addressed further in response to NUREG-0578, Section 2.1.8.

3.5.6 Post-Accident Sampling Station

The post-accident sampling station will be located on plan elevation 735'-6" on the back of the existing sample room wall as drawn in Figure 3-2.

3.5.7 Recombiner Room

The Recombiners and hydrogen analyzers are vital components for post-accident operations. The entrance to the recombiner panel, under accident conditions, is as drawn in Figure 3-3. The control panel room is on grating and the operator will be exposed to high radiation from the low head safety injection (LHSI) lines in the recirculation mode (maximum 3000 R/hr at one hour after accident). The LHSI lines are routed below the control panel room floor. Recommendation for relocating the control panel or shielding from the LHSI lines is provided in Section 4.6. To turn the manual valves (number 101, 102, 103, and 104), the operator must go to the 722' level. The predicted radiation level at this floor will be approximately 3000 R/hr at one hour after the accident with the system in the recirculation mode. This radiation level is too high to permit personnel access to the area. Recommendation for remote operation of these valves is provided in Section 4.6.

3.6 Occupancy and NRC Radiation Exposure Criteria

Required occupancy of vital areas (specified above in Section 3.5) for post-accident conditions must be given considerable attention in emergency

planning. These occupancy requirements must be carefully evaluated in concert with existing shielding design in vital areas and radioactive flow paths to ensure that plant personnel exposures are maintained within the limits specified by the NRC.

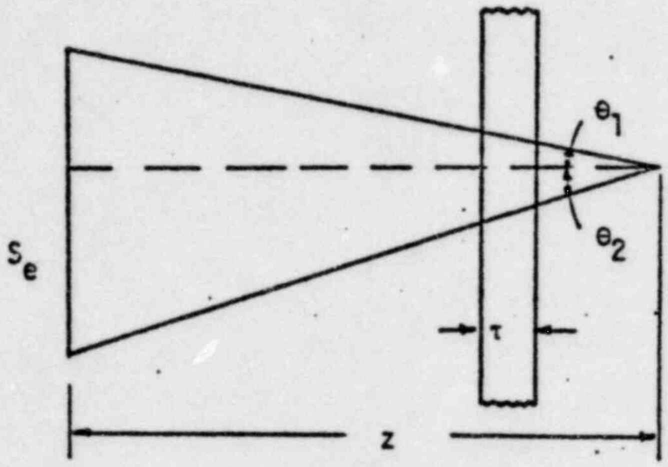
Based on the NRC definition of requirements for post-accident conditions specified in NUREG-0578, the basic dose rate (in vital areas) should be such that the guidelines of General Design Criteria 19 (GDC 19) of Appendix A of 10 CFR 50 should not be exceeded (during the course of the accident). GDC 19 limits the dose to an operator to 5 Rem whole body or its equivalent to any part of the body. These exposure limitations are factored into system evaluations in Section 4.0 of this study. Areas which require higher periods of occupancy obviously require lower dose rates than areas where lesser occupancy periods are required.

Most of the post-accident dose rates presented in Section 4.0 of this study are near contact with existing shielding. Therefore, the total body dose rate for an individual passing near (or occupying the area) will be appreciably less than the dose rate in contact with the shield. This factor is taken into consideration in Section 4.0 in assessing the adequacy of existing shielding.

3.7 Shielding Calculation Description

This program (SHD) is designed to calculate dosage from a line source. For a line source with shielding and assumed buildup factor of the form

$$B(\mu\tau) = A\,e^{-\alpha\mu\tau} + (1-A)\,e^{-\beta\mu\tau} \quad \text{(Ref. 1)} \qquad (3\text{-}1)$$



The flux at point P is given by

$$\phi = \frac{Se}{4\pi z} \left\{ A \left[F(\theta_1, (1 + \alpha)\,\mu\tau + F(\theta_2, (1 + \alpha)\,\mu\tau) \right] \right. \qquad (3\text{-}2)$$

$$\left. + (1 \cdot A \left[F(\theta_1, (1 + \beta)\,\mu\tau) + F(\theta_2, (1 + \beta)\,\mu\tau \right] \right\}$$

where

$$F(\theta, X) = \int_0^{\theta} e^{-X\sec\theta}\, d\theta \qquad (3\text{-}3)$$

μ = linear attenuation coefficient of shielding material

Values of μ were obtained from Table 3.8. Values o^ A, α, and β were obtained from Table 9.1.12-117 in Reference 1. Subroutine INTERP then performed linear interpolation to calculate these values at the specific energy desired.

Flux was calculated from equation 3-2 for each energy group. The flux, photons/cm^2 -sec was then converted to equivalent dose rate in mR/hr. Dose conversion factor, (MeV/cm^2 -sec)/(mR/hr), was calculated with the following assumptions:

1. DCF = 500 for photon energy between 0.3 and 1 MeV.

2. DCF is a linear function between 1 and 8 MeV when plotted on log-log paper.
   (At 1 MeV, DCF = 500; at 8 MeV, DCF = 1000)

Therefore, for photon energy E_n between 1 and 8 MeV

$$\log (\mathrm{DCF}) = \log (500) + \log (E_n) \left[\frac{\log (1000) - \log (500)}{\log (8) - \log (1)} \right] \quad (3\text{-}4)$$

or letting

$$X = \log (500) + \log (E_n) \left[\frac{\log (1000) - \log (500)}{\log (8)} \right] \quad (3\text{-}5)$$

then

$$\mathrm{DCF} = 10^X \quad (3\text{-}6)$$

Then dose rate from group n is

$$D_n = \phi_n \times E_n/\mathrm{DCF}_n \quad (3\text{-}7)$$

where
D_n = Dose rate from group n, mR/hr
ϕ_n = Flux of group n, photons/cm^2 -sec.
E_n = Photon energy of group n, MeV/photon
DCF_n = Dose conversion factor for group n (Ref. 2) (MeV/cm^2 -sec)/(mR/hr)

The total dose rate is then the sum of dose rates from all n groups.

$$\text{Total Dose Rate} = \sum_{n} D_n \tag{3-8}$$

The dose rate for each energy group was first calculated and then summed for a total dose for all 12 energy groups.

Program SHD was thoroughly checked against hand calculations. Six test runs were made on Program SHD, and the results agreed well with values given in Figure 12, Reference 3.

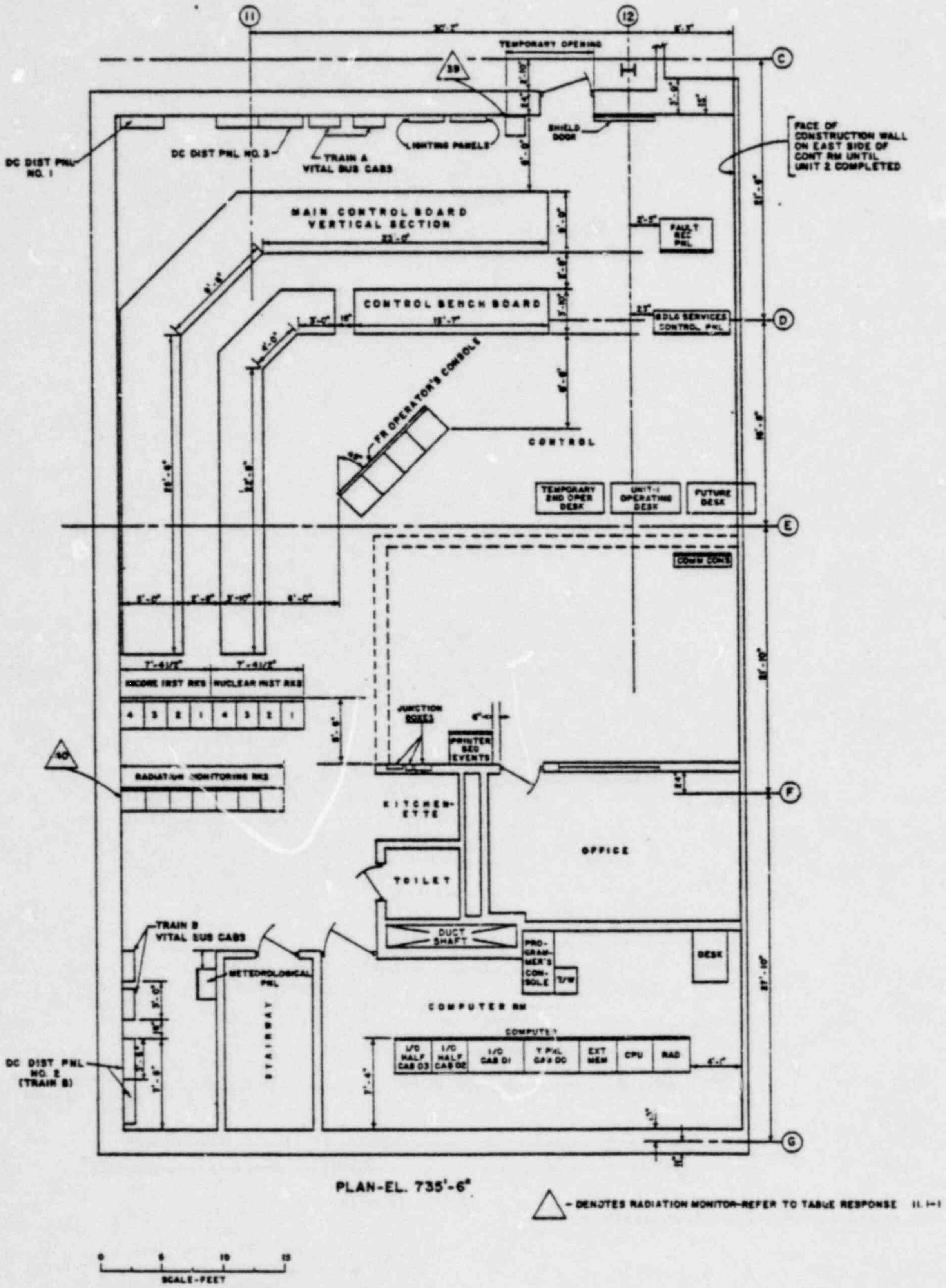
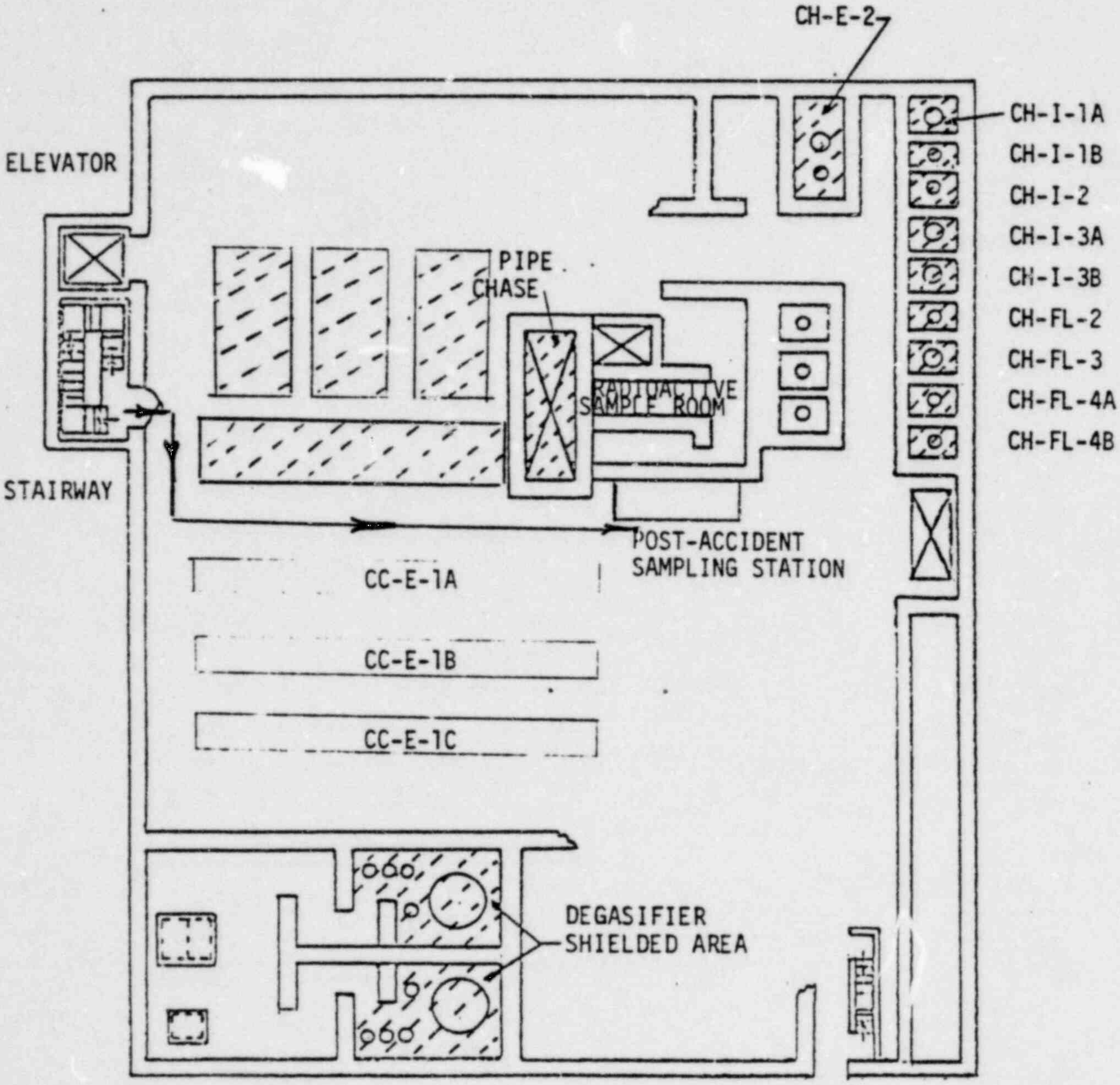


FIGURE 3-1 CONTROL ROOM



PLAN EL. 735'-6" IN AUXILIARY BUILDING

NOTE: Shaded areas with dash lines indicate potentially radioactive areas.

FIGURE 3-2 POST-ACCIDENT SAMPLING STATION

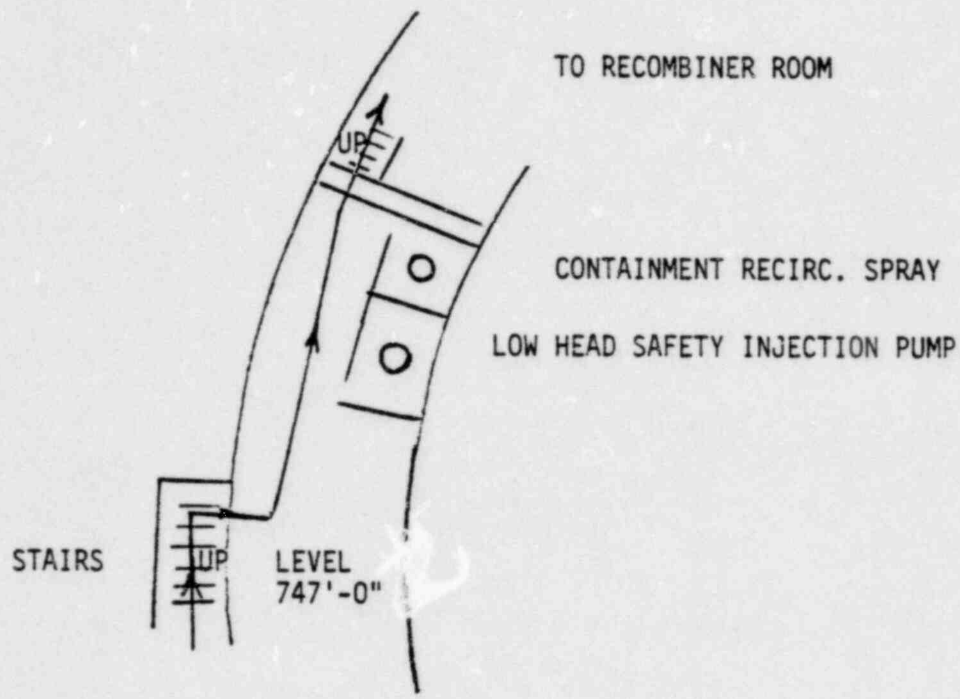


FIGURE 3-3 PATHWAY TO RECOMBINER ROOM IN ACCIDENT CONDITION

TABLE 3-1

RADIATION SOURCE TERM UNDER POST-LOCA CONDITIONS

CONTAINMENT AIR

Noble gases	100% of core inventory
Halogens	25% of core inventory

LIQUID SYSTEMS

Noble gases	100% of core inventory
Halogens	50% of core inventory
Others	1% of core inventory

TABLE 3-2

REACTOR PARAMETERS

Power	2766MW	
Burnup	1797888. MWD	
Time	650 days	
Flux	4.80×10^{13}	N/Cm^2 - Sec

Decay Periods: initial, 1 hour, 2 hours, 5 hours, 10 hours, 24 hours, 120 hours, and 720 hours

TABLE 3-3 LIQUID SOURCE

********** GAMMA SOURCE **********

PRINCIPAL PHOTON SOURCES IN GROUP 1, MEV/WATT-SEC
MEAN ENERGY = .300MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	4.22E+08	4.45E+08	4.64E+08	4.97E+08	5.05E+08	4.25E+08	2.16E+08	1.73E+07
TOTAL (PHOTONS/SEC)	3.89E+18	4.11E+18	4.28E+18	4.59E+18	4.66E+18	3.92E+18	1.99E+18	1.59E+17

PRINCIPAL PHOTON SOURCES IN GROUP 2, MEV/WATT-SEC
MEAN ENERGY = .630MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	4.07E+09	3.52E+09	2.87E+09	1.98E+09	1.71E+09	1.40E+09	5.01E+08	2.59E+07
TOTAL (PHOTONS/SEC)	1.79E+19	1.54E+19	1.26E+19	8.70E+18	7.53E+18	6.13E+18	?.20E+18	1.14E+17

PRINCIPAL PHOTON SOURCES IN GROUP 3, MEV/WATT-SEC
MEAN ENERGY = 1.100MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	1.77E+09	1.56E+09	1.34E+09	9.40E+08	6.42E+08	3.16E+08	5.56E+07	4.61E+05
TOTAL (PHOTONS/SEC)	4.45E+18	3.92E+18	3.37E+18	2.36E+18	1.61E+18	7.95E+17	2.40E+17	1.16E+15

PRINCIPAL PHOTON SOURCES IN GROUP 4, MEV/WATT-SEC
MEAN ENERGY = 1.550MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	8.11E+08	6.88E+08	5.58E+08	3.64E+08	2.68E+08	1.73E+08	7.52E+07	6.57E+06
TOTAL (PHOTONS/SEC)	1.45E+18	1.23E+18	9.96E+17	6.50E+17	4.79E+17	3.08E+17	1.34E+17	1.17E+16

PRINCIPAL PHOTON SOURCES IN GROUP 5, MEV/WATT-SEC
MEAN ENERGY = 1.590MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	4.87E+09	4.10E+08	3.48E+08	2.21E+08	1.22E+08	4.73E+07	1.38E+07	2.71E+05
TOTAL (PHOTONS/SEC)	6.77E+17	5.70E+17	4.83E+17	3.08E+17	1.69E+17	6.58E+16	1.92E+16	3.77E+14

PRINCIPAL PHOTON SOURCES IN GROUP 6, MEV/WATT-SEC
MEAN ENERGY = 2.380MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	6.07E+08	4.70E+08	3.63E+08	1.71E+08	5.03E+07	2.93E+06	1.20E+06	3.15E+05
TOTAL (PHOTONS/SEC)	7.05E+17	5.46E+17	4.22E+17	1.99E+17	5.85E+16	3.40E+15	1.39E+15	3.66E+14

PRINCIPAL PHOTON SOURCES IN GROUP 7, MEV/WATT-SEC
MEAN ENERGY = 2.750MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	4.24E+08	2.48E+08	1.44E+08	2.79E+07	1.84E+06	2.10E+03	5.96E-08	0.
TOTAL (PHOTONS/SEC)	4.27E+17	2.50E+17	1.45E+17	2.81E+16	1.85E+15	2.11E+12	6.00E+01	0.

PRINCIPAL PHOTON SOURCES IN GROUP 8, MEV/WATT-SEC
MEAN ENERGY = 3.250MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	4.60E+07	7.30E+06	2.23E+06	4.60E+05	4.80E+04	8.56E+01	1.22E-17	0.
TOTAL (PHOTONS/SEC)	3.92E+16	6.21E+15	1.90E+15	3.92E+14	4.08E+13	7.29E+10	1.04E-08	0.

PRINCIPAL PHOTON SOURCES IN GROUP 9, MEV/WATT-SEC
12 MEAN ENERGY = 3.700MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	3.27E+07	9.70E+06	2.61E+06	5.16E+04	7.46E+01	8.34E-07	0.	0.
TOTAL (PHOTONS/SEC)	2.44E+16	7.25E+15	1.95E+15	3.85E+13	5.57E+10	6.23E+02	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP 10, MEV/WATT-SEC
12 MEAN ENERGY = 4.220MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	1.15E+07	8.36E+00	4.94E-06	1.02E-24	0.	0.	0.	0.
TOTAL (PHOTONS/SEC)	7.51E+15	5.48E+09	3.24E+03	6.71E-16	0.	0.	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP 11, MEV/WATT-SEC
MEAN ENERGY = 4.700MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	9.17E+01	4.74E-01	2.45E-03	3.40E-10	1.26E-21	0.	0.	0.
TOTAL (PHOTONS/SEC)	5.40E+10	2.79E+08	1.44E+06	2.00E-01	7.41E-13	0.	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP 12, MEV/WATT-SEC
12 MEAN ENERGY = 5.250MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL (MEV/WATT-SEC)	2.01E+06	1.67E+00	8.67E-07	1.80E-25	0.	0.	0.	0.
TOTAL (PHOTONS/SEC)	1.06E+15	7.72E+08	4.57E+02	9.46E-17	0.	0.	0.	0.

TABLE 3-4 GAS SOURCE

********** GAMMA SOURCE **********

PRINCIPAL PHOTON SOURCES IN GROUP 1, MEV/WATT-SEC
MEAN ENERGY = .330MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	3.21E+09	3.52E+09	3.70E+09	[illegible]	[illegible]	[illegible]	[illegible]	1.03E+0[illegible]

PRINCIPAL PHOTON SOURCES IN GROUP 2, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	1.30E+09	[illegible]	8.15E+08	[illegible]	2.30E+08	1.09E+05

PRINCIPAL PHOTON SOURCES IN GROUP 3, MEV/WATT-SEC
MEAN ENERGY = 1.100MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	[illegible]	[illegible]	3.16E+08	[illegible]	[illegible]	2.28E+05

PRINCIPAL PHOTON SOURCES IN GROUP 4, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	[illegible]	[illegible]	1.18E+08	[illegible]	[illegible]	[illegible]

PRINCIPAL PHOTON SOURCES IN GROUP 5, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	6.49E+06	[illegible]

PRINCIPAL PHOTON SOURCES IN GROUP 6, MEV/WATT-SEC
MEAN ENERGY = 2.300MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.

PRINCIPAL PHOTON SOURCES IN GROUP 7, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	8.30E+02	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP 8, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	0.	0.	0.	0.	0.	0.	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP 9, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP 10, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

TABLE 3-4 GAS SOURCE (Continued)

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	0.	0.	0.	0.	0.	0.	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP [illegible], MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	0.	0.	0.	0.	0.	0.	0.	0.

PRINCIPAL PHOTON SOURCES IN GROUP [illegible]2, MEV/WATT-SEC
MEAN ENERGY = [illegible]MEV

	INITIAL	1. H	2. H	5. H	10. H	24. H	120. H	720. H
TOTAL	0.	0.	0.	0.	0.	0.	0.	0.

TABLE 3-5.1

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
1-D	CCR to RHS H/X 1A & RHS Pump 1A Seal Cooler	Manual	1CCR-247	Shut	Open	AS-IS	None	_____
2-D	CCR From RHS H/X 1B & RHS Pump 1B Seal Cooler	Manual	1CCR-252	Shut	Open	AS-IS	None	_____
3-	Spare							
4-D	CCR From RHS H/X 1A & RHS Pump 1A Seal Cooler	Manual	1CCR-251	Shut	Open	AS-IS	None	_____
5-D	CCR to RHS H/X 1B & RHS Pump 1B Seal Cooler	Manual	1CCR-248	Shut	Open	AS-IS	None	_____
6-B	Spare							
7-A	High Head S.I. To Hot Legs	Rem-Man	MOV-1SI-869A	Shut	Shut	AS-IS	None	_____
8-C	CCR From RCP B & C Thermal Barriers	Auto-Trip	TV-1CC-107D2	Open	Open	Shut	S-CIB	_____
9-B	CCR From Shroud Coolers	Auto-Trip	TV-1CC-111D2	Open	Open	Shut	S-CIB	_____
10-B	Spare							
11-B	Air Recirc. Cooling Water Out	Auto-Trip Auto-Trip	TV-1CC-110F2 TV-1CC-110F1	Open Shut	Open Shut	Shut Shut	S-CIB S-CIB	_____ _____
12-A	Spare							
13-D	Spare							
14-D	Air Recirc Cooling Water IN	Auto-Trip	TV-1CC-110E2	Open	Open	Shut	S-CIB	_____
15-A	Coolant System Charging	Auto-Trip	MOV-1CH-289	Open	Shut	AS-IS	S-SIS	
16-B	CCR to Shroud Coolers	Auto-Trip	TV-1CC-111A1	Open	Shut	Shut	S-CIB	_____
17-A	CCR to RCP 1B	Auto-Trip	TV-1CC-103B	Open	Open	Shut	S-CIB	_____
18-A	CCR to RCP 1C	Auto-Trip	TV-1CC-103C	Open	Open	Shut	S-CIB	_____
19-A	RCP's Seal Water Return	Auto-Trip	MOV-1CH-381	Open	Open/ Shut	AS-IS	S-CIB	_____
20-C	S.I. Accum. Makeup	Manual	1SI-41	Shut	Shut	AS-IS	None	_____

TABLE 3-5.2

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
21-B	Spare							
22-B	Spare							
23-B	Spare							
24-SgD	RHS to RWST	Manual	1RH-15	Shut	Open/ Shut	AS-IS	None	
25-B	CCR From RCP 1B & 1C Motors	Auto-Trip	TV-1CC-105D2	Open	Open	Shut	S-CIB	
26-C	CCR From RCP 1A Thermal Barriers	Auto-Trip	TV-1CC-107E2	Open	Open/ Shut	Shut	S-CIB	
27-C	CCR From RCP 1A Motor	Auto-Trip	TV-1CC-105E2	Open	Open	Shut	S-CIB	
28*-A	RCS Letdown	Auto-Trip	TV-1CH-204	Open/ Shut	Shut	Shut	S-CIA	
29-A	Pri. Drains Trans. Pump No. 1 Discharge	Auto-Trip	TV-1DG-108B	Open/ Shut	Shut	Shut	S-CIA	
30-B	Spare							
31-D	Spare							
32-C	Spare							
33-C	High Head S.I. to Hot Legs	Rem-Man	MOV-1SI-869B	Shut	Shut	AS-IS	None	
34-A	Spare							
35-A	Seal Inj. Water RCP 1A	Rem-Man	MOV-1CH-308A	Open	Open	AS-IS	S-CIB	
36-A	Seal Inj. Water RCP 1B	Rem-Man	MOV-1CH-308B	Open	Open	AS-IS	S-CIB	
37-A	Seal Inj. Water RCP 1C	Rem-Man	MOV-1CH-308C	Open	Open	AS-IS	S-CIB	
38-A	Cnmt Sump Pump Discharge	Auto-Trip	TV-1DA-100B	Open	Shut	Shut	-CIA	
39*-C	Stm Gen 1A Blowdown	Auto-Trip	TV-1BD-100A	Open	Shut	Shut	S-CIA	
40*-A	Stm Gen 1B Blowdown	Auto-Trip	TV-1BD-100B	Open	Shut	Shut	S-CIA	
41*-B	Stm Gen 1C Blowdown	Auto-Trip	TV-1BD-100C	Open	Shut	Shut	S-CIA	
42-C	Compressed Air to Fuel Handling Equipment	Manual	1SA-14	Shut	Open	AS-IS	None	
43-B	Air Activity Monitor - Out	Auto-Trip	TV-1CV-102-1	Open	Shut	Shut	S-CIA	

TABLE 3-5.3

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
44-B	Air Activity Monitor - IN	Series	TV-1CV-101A	Open	Open	Shut	S-CIA	_____
		Auto-Trips	TV-1CV-101B	Open	Open	Shut	S-CIA	_____
45-B	Pri. Grade Water to Pzr. Relief Tank	Auto-Trip	TV-1RC-519	Open	Open	Shut	S-CIA	_____
46-A	Charging Fill Header	Rem-Man	FCV-1CH-160	Shut	Shut	Shut	None	
47-B	Instrument Air	Manual	1IA-9[illegible]	Shut	Shut	AS-IS	None	_____
48-B	Primary Vent Header	Auto-Trip	TV-1D[illegible] 109A1	Open	Shut	Shut	S-CIA	_____
49-C	Nitrogen Supply to Pzr. Relief Tank	Auto-Trip	TV-1RC-101	Open	Shut	Shut	S-CIA	_____
50-C	Spare							
51*-C	Spare							
52*-C	Spare							
53-C	Nitrogen Supply to S.I. Accumulators	Auto-Trip	TV-1SI-101-1	Shut	Shut	Shut	S-CIA	_____
54-B	Spare							
55*-1-A	S.I. Acuum. Sample	Auto-Trip	TV-1SS-109A2	Open	Open	Shut	S-CIA	_____
55*-2-A	CNMT Leakage Monitoring Open Taps	Series	TV-1LM-100A1	Open	Open	Shut	S-CIA	_____
55*-3-A	Spare							
55*-4-A	Pzr. Relief Tank Gas Sample	Auto-Trip	TV-1SS-111A2	Open	Open	Shut	S-CIA	_____
56*-1-A	Pzr. Liquid Sample	Auto-Trip	TV-1SS-100A2	Open	Open	Shut	S-CIA	_____
56*-2-A	RCS Cold Leg Samples	Auto-Trip	TV-1SS-102A2	Open	Open	Shut	S-CIA	_____
56*-3-A	RCS Hot Leg Samples	Auto-Trip	TV-1SS-105A2	Open	Open	Shut	S-CIA	_____
56*-4-A	Stm Gen 1A Blowdown Sample	Auto-Trip	TV-1SS-117A	Open	Open	Shut	S-CIA	_____
57*-1A	Cnmt Leakage Monitoring Open Taps	Series	TV-1LM-100A1	Open	Open	Shut	S-CIA	_____
		Auto-Trip	TV-1LM-100A2	Open	Open	Shut	S-CIA	_____
57*-2-A	CNMT Leakage Monitoring Open Taps							

TABLE 3-5.4

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
57*-3-A	CNMT Leakage Monitoring System - Pressurized	Manual	1CV-35	Shut	Shut	AS-IS	None	______
57*-4-A		Manual	1CV-36	Shut	Shut		None	______
58-B	CCR to RCP 1A	Auto-Trip	TV-1CC-103A	Open	Open/ Shut	Shut	S-CIB	______
59-C	Spare							
60-SgD	Low Head S.I. to Hot Legs	Rem-Man	MOV-1S1-890A	Shut	Shut	AS-IS	None	______
61-SgD	Low Head S.I. to Cold Legs	Rem-Man	MOV-1S1-890C	Open	Shut	AS-IS	None	______
62-SgD	Low Head S.I. to Hot Legs	Rem-Man	MOV-1S1-890B	Shut	Shut	AS-IS	None	______
63-SgD	Quench Spray Pump Disch 360° Header	Parallel	MOV-1QS-101B	Shut	Shut	AS-IS	O-CIB	______
64-SgD	Quench Spray Pump Disch 360° Header	Parallel	MOV-1QS-101A	Shut	Shut	AS-IS	O-CIB	______
65	Fuel Transfer Tube	Manual	LATER	Shut	Open	AS-IS	None	______
66-SgD	Outside Recirc Spray Pump 2A Suct. From CNMT	Rem-Man	MOV-1RS-155A	Open	Open	AS-IS	O-CIB	______
67-SgD	Outside Recirc Spary Pump 2B Suct. From CNMT	Rem-Man	MOV-1RS-155B	Open	Open	AS-IS	O-CIB	______
68-SgD	Low Head S.I. Pump 1A Suction From CNMT Sump	Rem-Man	MOV-1S1-860A	Shut	Shut	AS-IS	None	______
69-SgD	Low Head S.I. Pump 1B Suction From CNMT Sump	Rem-Man	MOV-1S1-860B	Shut	Shut	AS-IS	None	______
70-SgD	Outside Recirc Spray Pump 2B Discharge	Auto-Trip	MOV-1RS-156B	Open	Open	AS-IS	O-CIB	______
72-SgD	Argon Supply (RP-14A)	LATER	LATER	LATER	LATER	LATER	LATER	______

TABLE 3-5.5

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
73*-SgD	Main Steam Loop 1A	Auto-Trip	TV-1MS-101A	Open	Shut	Shut	S-SLI	_____
	Main Steam Line Drain	Auto-Trip	TV-1MS-111A	Open	Open	Shut	S-SLI	_____
	Main Steam Atmos. Dump	PCV	PCV-1MS-101A	Shut	Open	Shut	None	_____
	Main Steam Safety Valves	Safety Valves	Safety Valves	Shut	Shut	Shut	None	_____
	Main Steam to Aux. Feed Pump	Rem-Man	MOV-1MS-105	Open	Shut	AS-IS	None	_____
74*-SgD	Main Steam Loop 1B	Auto-Trip	TV-1MS-101B	Open	Shut	Shut	S-SLI	_____
	Main Steam Line Drain	Auto-Trip	TV-1MS-111B	Open	Open	Shut	S-SLI	_____
	Main Steam Atmos. Dump	PCV	PCV-1MS-101B	Shut	Open	Shut	None	_____
	Main Stm Safety Valves	Safety Valves	Safety Valves	Shut	Shut	Shut	None	_____
	Main Steam to Aux. Feed Pump	Rem-Man	MOV-1MS-105	Open	Open	AS-IS	None	_____
75*-SgD	Main Steam Loop 1C	Auto-Trip	TV-1MS-101C	Open	Shut	Shut	S-SLI	_____
	Main Steam Line Drain	Auto-Trip	TV-1MS-111B	Open	Open	Shut	S-SLI	_____
	Main Steam Atmos. Dump	PCV	PCV-1MS-101B	Shut	Open	Shut	None	_____
	Main Stm Safety Valves	Safety Valves	Safety Valves	Shut	Shut	Shut	None	_____
	Main Steam to Aux. Feed Pump	Rem-Man	MOV-1MS-105	Open	Open	AS-IS	None	_____
76*-SgD	Feedwater Loop 1A	Non-Return	MOV-1FW-156A	Open	Shut	Shut	S-FWI	_____
	Aux Feedwater Loop 1A	Rem-Man	MOV-1FW-158A	Open	Open	AS-IS	None	_____
77*-SgD	Feedwater Loop 1B	Non-Return	MOV-1FW-156B	Open	Shut	Shut	S-FWI	_____
	Aux Feedwater Loop 1B	Rem-Man	MOV-1FW-158B	Open	Open	AS-IS	None	_____
78*-SgD	Feedwater Loop 1C	Non-Return	MOV-1FW-156C	Open	Shut	Shut	S-FWI	_____
	Aux Feedwater Loop 1C	Rem-Man	MOV-1FW-158C	Open	Open	AS-IS	None	_____
79-SgD	RW to 1A Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-104A	Open	Open	AS-IS	None	_____
80-SgD	RW to 1C Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-104C	Open	Open	AS-IS	None	_____

TABLE 3-5.6

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
81-SgD	RW to 1B Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-104B	Open	Open	AS-IS	None	______
82-SgD	RW to 1D Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-104D	Open	Open	AS-IS	None	______
83-SgD	RW from 1A Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-105A	Open	Open	AS-IS	None	______
84-SgD	RW from 1C Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-105C	Open	Open	AS-IS	None	______
85-SgD	RW from 1B Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-105B	Open	Open	AS-IS	None	______
86-SgD	RW from 1D Recirc. Spray Heat Exch.	Rem-Man	MOV-1RW-105D	Open	Open	AS-IS	None	______
87-SgD	Post DBA Hydrogen Control	Manual	1HY-110	Shut	Shut	AS-IS	None	______
88-SgD	Discharge to CNMT	Manual	1HY-111	Shut	Shut	AS-IS	None	______
89-SgD	Main Condenser Ejector Vent	Auto-Trip	TV-1SV-100A	Shut	Shut	Shut	S-CIB	______
90-SgD	CNMT Purge Exhaust	Auto-Trip	VS-D-5-3A	Shut	Open	AS-IS	S-RM	
91-SgD	CNMT Purge Supply	Auto-Trip	VS-D-5-5A	Shut	Open	AS-IS	S-RM	______
92-A	CNMT Vacuum Pump 1B & H_2 Recomb. Suction	Series	TV-1CV-150C	Open	Open	Shut	S-CIA	______
		Auto-Trips	TV-1CV-150D	Open	Shut	Shut	S-CIA	______
		Manual	1HY-102	Shut	Shut	AS-IS	None	______
93-B	CNMT Vacuum Pump 1A & H_2 Recomb. Suction	Series	TV-1CV-150A	Open	Open	Shut	S-CIA	______
		Auto-Trips	TV-1CV-150B	Open	Shut	Shut	S-CIA	______
		Manual	1HY-101	Shut	Shut	AS-IS	None	______
94-C	CNMT Vacuum Ejector Suction	Manual	HCV-1CV-151-1	Shut	Shut	AS-IS	None	______
95-C	Spare							
96-B	High Head S.I. to Cold Legs	Rem-Man	MOV-1SI-836	Shut	Shut	AS-IS	None	______

TABLE 3-5.7

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
97*-1-A	RHS Inlet Sample	Auto-Trip	TV-1SS-104A2	Open	Open	Shut	S-CIA	______
97*-2-A	RHS Outlet Sample	Auto-Trip	TV-1SS-103A2	Open	Open	Shut	S-CIA	______
97*-3-A	CNMT Leakage	Series	TV-1LM-100A1	Open	Open	Shut	S-CIA	______
	Monitoring Open Taps	Auto-Trips	TV-1LM-100A2	Open	Open	Shut	S-CIA	______
97*-4-A	Stm Gen 1C Blowdown Sample	Auto-Trip	TV-1SS-117C	Open	Open	Shut	S-CIA	______
98-1-C	Oxygen	Capped	Capped	N/A	N/A	N/A	N/A	
98-2-C	Argon	Capped	Capped	N/A	N/A	N/A	N/A	
98-3-C	Acetylene	Capped	Capped	N/A	N/A	N/A	N/A	
89-4-C	Spare							
99-C	Spare							
100-B	Spare							
101-B	Spare							
102-B	Spare							
103-A	Refueling Cavity Purif Outlet	Manual	1PC-37	Shut	Open	AS-IS	None	______
104-A	Refueling Cavity Purif Inlet	Manual	1PC-10	Shut	Open	AS-IS	None	______
105*-1-B	Stm Gen 1B Blowdown Sample	Auto-Trip	TV-1SS-117B	Open	Open	Shut	S-CIA	______
105*-2-B	Pzr Vapor Sample	Auto-Trip	TV-1SS-112A2	Open	Open	Shut	S-CIA	______
105*-3-B	Spare							
105*-4-B	Spare							
106-SgD	S.I. Accum. Test Line	Auto-Trip	TV-1SI-889	Shut	Shut	Shut	S-CIA	______
[illegible]-C	Spare							
[illegible]-B	Spare							
1[illegible]9-C	Spare							
110-1-C	Press Dead Weight	Series	1RC-277	Shut	Shut	AS-IS	None	______
	Calibrator PT-RC-455A	Manual	1RC-278	Shut	Shut	AS-IS	None	______
110-2-C	Spare							
110-3-C	Spare							

TABLE 3-5.8

CONTAINMENT PENETRATION CHECKLIST

(Outside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
110-4-C	Spare							
111-C	Diluted Fuel Bldg Exhaust	Manual	VS-D-4-6A	Shut	Open	AS-IS	None	______
112-C	Fuel Bldg Exhaust	Parallel	VS-D_9-1A	Shut	Shut	AS-IS	O-RM	______
		Auto-Trip	VS-D-9-1B	Shut	Shut	AS-IS	O-RM	______
113-1-A	B.I. Tank to Cold Legs	Parallel	MOV-1S1-867C	Shut	Shut	AS-IS	O-S1S	______
		Auto-Trips	MOV-1S1-867D	Shut	Shut	AS-IS	O-S1S	______
113-2-A	B.I.T. Bypass to cold Legs	Manual	1S1-91	Shut	Shut	AS-IS	None	______
113-3-A	Spare							
113-4-A	Spare							

TABLE 3-6.1

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
1-D	CCR to RHS H/X 1A & RHS Pump 1A Seal Cooler	Rem-Man	MOV-1CC-112A2	Shut	Open	AS-IS	None	______
2-D	CCR From RHS H/X 1B & RHS Pump 1B Seal Cooler	Rem-Man	MOV-1CC-112B3	Shut	Open	AS-IS	None	______
3-	Spare							
4-D	CCR From RHS H/X 1A & RHS Pump 1A Seal Cooler	Rem-Man	MOV-1CC-112A2	Shut	Open	AS-IS	None	______
5-D	CCR to RHS H/X 1B & RHS Pump 1B Seal Cooler	Rem-Man	MOV-1CC-112B2	Shut	Open	AS-IS	None	______
6-B	Spare							
7-A	High Head S.I. to Hot Legs	Check	1S1-83	Shut	Shut	AS-IS	None	
8-C	CCR From RCP B & C Thermal Barriers	Auto-Trip	TV-1CC-107D1	Open	Open/Shut	Shut	S-CIB	______
9-B	CCR From Shroud Coolers	Auto-Trip	TV-1CC-111D1	Open	Open	Shut	S-CIB	______
10-B	Spare							
11-B	Air Recirc Cooling Water Out	Auto-Trip	TV-1CC-110D	Open	Open	Shut	S-CIB	______
12-A	Spare							
13-D	Spare							
14-D	Air Recirc Cooling Water IN	Auto-Trip	TV-1CC-110E3	Open	Open	Shut	S-CIB	______
15-A	Coolant System Charging	Check	1CH-31					
16-B	CCR to Shroud Coolers	Auto-Trip	TV-1CC-111A2	Open	Shut	Shut	S-CIB	______
17-A	CCR to RCP 1B	Auto-Trip	TV-1CC-103B1	Open	Open/Shut	Shut	S-CIB	______
18-A	CCR to RCP 1C	Auto-Trip	TV-1CC-103C1	Open	Open/Shut	Shut	S-CIB	______
19-A	RCP's Seal Water Return	Auto-Trip	MOV-1CH-378	Open	Open/Shut	AS-IS	S-CIB	______

TABLE 3-6.2

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
20-C	S.I. Accum. Makeup	Check	1SI-42	Shut	Shut	AS-IS	None	
21-B	Spare							
22-B	Spare							
23-B	Spare							
24-SgD	RHS to RWST	Manual	1RH-14	Shut	Open/ Shut	AS-IS	None	____
25-B	CCR From RCP 1B & 1C Motors	Auto-Trip	TV-1CC-105D1	Open	Open/ Shut	Shut	S-CIB	____
26-C	CCR From RCP 1A Thermal Barriers	Auto-Trip	TV-1CC-107E1	Open	Open/ Shut	Shut	S-CIB	____
27-C	CCR From RCP 1A Motor	Auto-Trip	TV-1CC-105E1	Open	Open	Shut	S-CIB	____
28*-A	RCS Letdown	Auto-Trip	TV-1CH-200A	Open/ Shut	Shut	Shut	S-CIA	____
		Auto-Trip	TV-1CH-200B		Shut	Shut	S-CIA	____
		Auto-Trip	TV-1CH-200C		Shut	Shut	S-CIA	____
		Rem-Man	MOV-1CH-142	Shut	Open	AS-IS	None	____
29-A	Pri. Drains Trans Pump No. 1 Discharge	Auto-Trip	TV-1DG-108A	Open/ Shut	Shut	Shut	S-CIA	____
30-B	Spare							
31-D	Spare							
32-D	Spare							
33-C	High Head S.I. to Hot Legs	Check	1SI-84	Shut	Shut	AS-IS	None	
34-A	Spare							
35-A	Seal Inj. Water RCP 1A	Check	1CH-181	Open	Open	AS-IS	S-CIB	
36-A	Seal Inj. Water RCP 1B	Check	1CH-182	Open	Open	AS-IS	S-CIB	
37-A	Seal Inj. Water RCP 1C	Check	1CH-183	Open	Open	AS-IS	S-CIB	
38-A	Cnmt Sump Pump	Auto-Trip	TV-1DA-100A	Open/	Shut	Shut	S-CIA	____

TABLE 3-6.3

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
39*-C	Stm Gen 1A Blowdown	System	System					
40*-A	Stm Gen 1B Blowdown	System	System					
41*-B	Stm Gen 1C Blowdown	System	System					
42-C	Compressed Air to Fuel Handling Equipment	Check	1SA-15	Shut	Open	AS-IS	None	
43-B	Air Activity Monitor - Out	Auto-Trip	TV-1CV-102	Open	Shut	Shut	S-CIA	
44-B	Air Activity Monitor - IN	None	None					
45-B	Pri. Grade Water to Pzr. Relief Tank	Check	1RC-72	Open	Open	Shut	S-CIA	
46-A	Charging Fill Header	Check	1CH-170	Shut	Shut	Shut	None	
47-B	Instrument Air	Check and	1IA-91 and	Shut	Shut	AS-IS	None	
		Manual	1IA-91-1	Open	Open	AS-IS	None	
48-B	Primary Vent Header	Auto-Trip	TV-1DG-109A2	Open	Shut	Shut	S-CIA	
49-C	Nitrogen Supply to Pzr. Relief Tank	Check	1RC-68	Open	Shut	Shut	S-CIA	
50-C	Spare							
51*-C	Spare							
52*-C	Spare							
53-C	Nitrogen Supply to S.I. Accumulators	Auto-Trip	TV-1SI-101-2	Shut	Shut	Shut	S-CIA	
54-B	Spare							
55*-1-A	S.I. Accum. Sample	Auto-Trip	TV-1SS-109A1	Open	Open	Shut	S-CIA	
55*-2-A	CNMT Leakage Monitoring Open Taps	None	None					
55*-3-A	Spare							
55*-4-A	Pzr. Relief Tank Gas Sample	Auto-Trip	TV-1SS-111A1	Open	Open	Shut	S-CIA	
56*-1-A	Pzr. Liquid Sample	Auto-Trip	TV-1SS-100A1	Open	Open	Shut	S-CIA	

TABLE 3-6.4

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
56*-2-A	RCS Cold Leg Samples	Auto-Trip	TV-1SS-102A1	Open	Open	Shut	S-CIA	____
56*-3-A	RCS Hot Leg Samples	Auto-Trip	TV-1SS-105A1	Open	Open	Shut	S-CIA	____
56*-4-A	STM Gen 1A Blowdown Sample	System	System					
57*-1A	CNMT Leakage Monitoring Open Taps	None	None					
57*-2-A	CNMT Leakage Monitoring Open Taps	None	None					
57*-3-A 57*-4-A	CNMT Leakage Monitoring System - Pressurized	Auto-Trip	TV-1LM-101A TV-1LM-101B	Open	Open	Shut	S-CIA	____
58-B	CCR to RCP 1A	Auto-Trip	TV-1CC-103A1	Open	Open/ Shut	Shut	S-CIB	____
59-C	Spare							
60-SgD	Low Head S.I. to Hot Legs	Check	1S1-13	Shut	Shut	AS-IS	None	
61-SgD	Low Head S.I. to Cold Legs	Checks	1S1-10,11,12	Open	Shut	AS-IS	None	
62-SgD	Low Head S.I. to Hot Legs	Check	1S1-14	Shut	Shut	AS-IS	None	
63-SgD	Quench Spray Pump Disch 360° Header	Check	1QS-4	Shut	Shut	AS-IS	O-CIB	
64-SgD	Quench Spray Pump Disch 360° Header	Check	1QS-3	Shut	Shut	AS-IS	O-CIB	
65	Fuel Transfer Tube	Flange	Flange	Shut	Open/ Shut	AS-IS	None	
66-SgD	Outside REcirc. Spray Pump 2A Suct From CNMT	None	None					
67-SgD	Outside Recirc. Spray Pump 2B Suct from CNMT	None	None					

TABLE 3.6-5

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
68-SgD	Low Head S.I. Pump 1A Suction from CNMT Sump	None	None					
69-SgD	Low Head S.I. Pump 1B Suction from CNMT Sump	None	None					
70-SgD	Outside REcirc Spray Pump 2B Discharge	Check	1RS-101	Open	Open	AS-IS	O-CIB	
72-SgD	Argon Supply (RP-14A)	LATER	LATER	LATER	LATER	LATER	LATER	
73*-SgD	Main Steam Loop 1A	System	System					
	Main Steam Line Drain	System	System					
	Main Steam Atmos. Dump	System	System					
	Main Stm Safety Valves	System	System					
	Main Steam to Aux. Feed Pump	System	System					
74*-SgD	Main Steam Loop 1B	System	System					
	Main Steam Line Drain	System	System					
	Main Steam Atmos. Dump	System	System					
	Main Stm Safety Valves	System	System					
	Main Steam to Aux. Feed Pump	System	System					
75*-SgD	Main Steam Loop 1C	System	System					
	Main Steam Line Drain	System	System					
	Main Steam Atmos. Dump	System	System					
	Main Stm Safety Valves	System	System					
	Main Steam to Aux. Feed Pump	System	System					
76*-SgD	Feedwater Loop 1A	System	System					
	Aux Feedwater Loop 1A	System	System					
77*-SgD	Feedwater Loop 1B	System	System					
	Aux Feedwater Loop 1B	System	System					

TABLE 3-6.6

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
78*-SgD	Feedwater Loop 1C	System	System					
	Aux Feedwater Loop 1C	System	System					
79-SgD	RW to 1A Recirc Spray Heat Exch.	System	System					
80-SgD	RW to 1C Recirc Spray Heat Exch.	System	System					
81-SgD	RW to 1B Recirc Spray Heat Exch.	System	System					
82-SgD	RW to 1D Recirc Spray Heat Exch.	System	System					
83-SgD	RW from 1A Recirc Spray Heat Exch.	System	System					
84-SgD	RW from 1C Recirc Spray Heat Exch.	System	System					
85-SgD	RW from 1B Recirc Spray Heat Exch.	System	System					
86-SgD	RW from 1D Recirc. Spray Heat Exch.	System	System					
87-SgD	Post DBA Hydrogen Control	LATER	LATER					
88-SgD	Discharge to CNMT	LATER	LATER					
89-SgD	Main Condenser Ejector Vent	Check	1AS-278					
90-SgD	CNMT Purge Exhaust	Auto-Trip	VS-D-5-3B					
91-SgD	CNMT Purge Supply	Auto-Trip	VS-D-5-5B					
92-A	CNMT Vacuum Pump 1B & H_2 Recomb. Suction	None	None					
93-B	CNMT Vacuum Pump 1A & H_2 Recomb. Suction	None						

TABLE 3-6.7

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
94-C	CNMT Vacuum Ejector Suction	Manual	HVC-1CV-151	Shut	Shut	AS-IS	None	
95-C	Spare							
96-B	High Head S.I. to Cold Legs	Check	1SI-95	Shut	Shut	AS-IS	None	
97*-1-A	RHS Inlet Sample	Auto-Trip	TV-1SS-104A1	Open	Open	Shut	S-CIA	
97*-2-A	RHS Outlet Sample	Auto-Trip	TV-1SS-103A1	Open	Open	Shut	S-CIA	
97*-3-A	CNMT Leakage Monitoring Open Taps	None	None					
97*-4-A	Stm Gen 1C Blowdown Sample	System						
98-1-C	Oxygen	Capped	Capped	N/A	N/A	N/A	N/A	
98-2-C	Argon	Capped	Capped	N/A	N/A	N/A	N/A	
98-3-C	Acetylene	Capped	Capped	N/A	N/A	N/A	N/A	
98-4-C	Spare							
99-C	Spare							
100-B	Spare							
101-B	Spare							
102-B	Spare							
103-A	Refuleing Cavity Purif Outlet	Manual	1PC-38	Shut	Open	AS-IS	None	
104-A	Refueling Cavity Purif Inlet	Manual	1PC-9	Shut	Open	AS-IS	None	
105*-1-B	Stm Gen 1B Blowdown Sample	System	System					
105*-2-B	Pzr Vapor Sample	Auto-trip	TV-1SS-112A1	Open	Open	Shut	S-CIA	
105*-3-B	Spare							
105*-4-B	Spare							
106-SgD	S.I. Accum. Test Line	Auto-Trip	MOV-1SI-842	Shut	Shut	AS-IS	S-CIA	
107-C	Spare							
108-B	Spare							

TABLE 3-6.8

CONTAINMENT PENETRATION CHECKLIST

(Inside Containment)

PENET. NO. AREA	SERVICE	ISOLATION VALVES PROVIDED	ISOLATION VALVE IDENTIFICATION NUMBER	ISOLATION VALVE POSITIONS				INIT.
				NORMAL	SHUT DOWN	FAIL.	ESF ACTION	
109-C	Spare							
110-1-C	Press Dead weight Calibrator PT-RC-455A	None	None					
110-2-C	Spare							
110-3-C	Spare							
110-4-C	Spare							
111-C	Diluted Fuel Bldg Exhaust	Manual	VS-D-4-6B	Shut	Open	AS-IS	None	____
112-C	Fuel Bldg Exhaust	Manual	VS-D-9-2	Shut	Shut	AS-IS	O-RM	____
113-1-A	B.I. Tank to cold legs	Check	1S1-94	Shut	Shut	AS-IS	O-SIS	
113-2-A	B.I.T Bypass to cold legs	Check	1S1-94	Shut	Shut	AS-IS	None	
113-3-A	Spare							
113-4-A	Spare							

TABLE 3-7

IDENTIFICATION OF RADIONUCLIDE FLOW PATHS

SYSTEM	RADIONUCLIDE FLOW PATH	COMMENTS
Reactor Plant Vents & Drain	None	Containment sump, primary drain transfer tank and gaseous releases are isolated by CIA
Supplementary Leak Collection and Release System	Yes	Any leak from containment into the safeguards area and areas contiguous to containment is vented through the offgas system
Boron Recovery System (BR)	None	Provided that Chemical and Volume Control System (CHS) is not used
Radiation Monitoring System (RM)	None	
Containment Structure and Air Lock	None	Personnel air lock and equipment hatch are closed and not available for personnel access
Post LOCA Sampling System	Yes	Not in existence but required by U.S. NRC as defined in NUREG-0578 for post-accident conditions
Liquid Waste System (LW)	None	Connected to vent and drain systems which have no flow path from containment
Solid Waste/Decontamination (SW)	None	Connected to CHS, BR, LW, and fuel pool
Gaseous Waste (GW)	None	Connected to BR. As long as CHS remains isolated GW should not be used.
Area Ventilation	None	Separate air supply provided in control room if accident occurs
Chemical and Volume Control (CHS)	None	Except charging pumps in SIS recirculation phase. Also pump seal injection water would be used if any RC pumps are operable.

table

TABLE 3-7 (Continued)

SYSTEM	RADIONUCLIDE FLOW PATH	COMMENTS
Reactor Coolant System (RCS)	None	Inside containment
Safety Injection System (SIS)	Yes	Recirculate containment sump water in recirculation phase
Containment Depressurization, Recirculation Spray	Yes	Recirculate containment sump water through spray (2 pumps outside containment)
Containment Vacuum Leakage Monitoring System, Containment Pressure Sensing Subsystem	Yes	Pressure taps inside containment and sample line extending outside containment
Residual Heat Removal System (RHR)	None	Inside containment
Post DBA Hydrogen	Yes	Draw containment air for hydrogen monitoring and recombination of H_2 and O_2

TABLE 3-8

TOTAL GAMMA RAY LINEAR ATTENUATION COEFFICIENTS OF CONCRETE (2.30 g/cm^3 density)

GAMMA ENERGY, MeV	μ, cm^{-1}
0.5	0.2017
1.0	0.1473
2.0	0.1034
3.0	0.0841
4.0	0.0732
5.0	0.0663
6.0	0.0615
8.0	0.0554
10.0	0.052

4.0 SYSTEM EVALUATION AND RECOMMENDATIONS

This section describes the systems that are necessary for mitigating an accident but which have the potential of carrying radioactivity outside the reactor containment. Existing shielding, resultant radiation levels, occupancy requirements, and proposed resolution of problem areas are discussed. Drawings, diagrams, and tables are included for these systems to supplement the discussions. Specific flow paths, locations, and values of radiation levels are included.

4.1 Normal and Letdown Flow-Paths Under Accident Condition

If the reactor coolant is highly radioactive due to an accident, one pathway out of the reactor containment would be through the Chemical and Volume Control System (CHS). During the TMI-2 accident, the letdown line in the CHS was reopened after isolation which resulted in flow of radionuclides into the CHS and subsequent release to systems not designed for this high radioactivity. The CHS should, therefore, be isolated from the reactor coolant system (RCS) if the reactor coolant is highly radioactive. For a minor accident, where lesser amounts of radioactivity are involved, it may be possible to use the normal letdown line in the CHS without releasing radionuclides into the CHS and other systems to the extent that a serious radiation hazard is created. This section considers the uses of the normal and alternate letdown flow paths.

4.1.1 Normal Letdown Flow Path

Use of the normal letdown line is not a safe practice, and therefore should not be used when the radiation level is high in the RCS. Radiation hazard with highly radioactive reactor coolant flowing through the normal letdown line is delineated in this section.

The charging and letdown functions of the CHS are employed to maintain a programmed water level in the reactor coolant system pressurizer during all phases of plant operation. This is achieved by means of continuous feed (charging) and bleed (letdown) process. During normal operation, the charging rate is controlled by the charging line flow control valve. The reactor coolant system pressurizer water level error signal resetting the flow controller setpoint in the proper direction satisfies operational requirements resulting from the selection of one or combinations of three flow restriction orifices in the letdown line.

The letdown coolant flow path leads to the purification system where the coolant normally flows through one of the two mixed bed demineralizers, the reactor coolant filters, and into the volume control tank (VCT). The charging pumps take suction from the VCT and return the cooled, purified reactor coolant to the RCS via the charging line where flow rate is controlled by the flow control valve.

A minimum flow for charging pump protection is provided by continuously diverting a portion of the charging pump discharge back to the VCT through the charging pump orifice and the seal water heat exchanger. Another portion of the charging flow is diverted to the reactor coolant pump seals via a seal water injection filter.

The potential radionuclide flow paths resulting from use of the CHS letdown line are shown in a simplified flow diagram, Figure 4-1. To minimize the radionuclide flow paths within the CHS, all the charging flow can be taken from the VCT by diverting all of the letdown flow into the VCT. In this operating scheme, the letdown flow is circulated back to the RCS from the charging line through the VCT. Since the main objective is to maintain a sufficient water level in the reactor vessel and not the purification of the reactor coolant water, the demineralizers and reactor coolant filter may be bypassed.

Radiation levels and occupancy requirements in the CHS are summarized in Table 4-1.

If the level in the VCT increases, the letdown flow may have to be diverted to the degasifier of the boron recovery system. If the pressure builds up in the VCT, the gaseous content of the VCT is released to the degasifier through the pressure relief line. The degasifier and associated equipment in the feed, overhead, and discharge lines (exchangers, heater, condenser, chiller, and cooler)

are in a shielded enclosure. The personnel access to the degasifier shielded enclosure should be controlled. Degassed water from the degasifier, which is relatively clean and low in radioactivity, can be either pumped back to the VCT or stored in the coolant recovery tank.

Radioactive gases from the degasifier are directed by the system pressure gradient to the gaseous charcoal delay subsystem upstream of the overhead gas compressor. The overhead compressor directs the radioactive gas stream to the gas surge tank. Most of the gas flow is reduced in pressure and returned to the VCT. A quantity of gas is periodically discharged from the surge tank to one of the three decay tanks for eventual release to the atmosphere via the process vent on top of the cooling tower.

The release of the letdown flow into the degasifier should be held to a minimum to prevent spread of the radioactive gases to the gaseous waste system. Radioactive gases are difficult to contain even in closed systems and often result in airborne contamination. Use of the normal letdown line will most likely result in radioactive liquid and gas flow in the CHS, boron recovery system, and gaseous waste system.

Figures 4-2A, 2B, and 2C indicate potentially radioactive areas in the Auxiliary Building due to radionuclide transport through the CHS.

4.1.2 Alternate Letdown Flow Path Under Containment Isolation

In this scheme, the containment is assumed to be isolated so that the radioactive gases and liquid will be contained in the reactor containment. The RCS is charged through the normal charging line; the charging pumps take suction from the refueling water storage tank. The normal letdown line, however, is closed to prevent radioactive reactor coolant from flowing out of the containment into the Chemical

and Volume Control System (CHS) and other systems. Figure 4-3 presents a simplified flow diagram of the alternate letdown flow paths for containment isolation.

The Containment Isolation Train A (CIA) signal will automatically close the letdown orifice isolation valves and stop the letdown flow from the RCS. The CIA signal is initiated by a containment high pressure signal or a safety injection signal (SIS). A containment high pressure signal simultaneously actuates safety injection. This project for Duquesne responds to TMI Lessons Learned, Section 2.1.6.b, which addresses additional shielding required for post-LOCA radiation levels. This report does not, therefore, address whether high radiation without a pipe break will result in a CIA. In the future, Duquesne may have to respond to the question of whether high primary coolant loop radiation will activate a CIA. This study assumes that a CIA signal does occur and in the event of a fuel related accident, therefore, containment isolation is actuated.

The safety injection signal is activated simultaneously with CIA. The safety injection signal will start the high head safety injection/charging pumps. The suction of the high head safety injection/charging pumps is diverted from the volume control tank to the refueling water storage tank. Isolation valves on the reactor coolant pump seal water return line are also automatically closed. Other sources of water such as emergency boration could add to the accumulated water volume.

The automatic closing of the letdown line, volume control tank discharge line, and reactor coolant pump seal water return line by SIS and CIA will isolate the CHS from the RCS. The CHS is isolated at the containment boundary except for the charging pumps and the piping in the safety injection flow path.

With the CHS isolated, overflow from the RCS resulting from the safety injection and other water input sources will be confined within the reactor containment. If there is no pipe break, the excess water in the RCS will overflow from the pressurizer into the pressurizer relief tank through the pressure control valves. The pressurizer relief tank content can be transferred to the primary drain transfer tank by manually opening the motor operated valve (MOV-RC-523) in the discharge line. As the primary drain transfer tank fills up, the relief valve will open to relieve pressure in the tank. The liquid which is forced out will drain into the reactor containment sump.

All gases and liquids from the pressurizer overflow will be confined within the reactor containment. The pressurizer relief tank vent line contains two containment isolation trip valves which will close on a CIA signal. The primary drain transfer tank vent, tied into the pressurizer relief tank vent line, is also isolated. The primary drain transfer tank discharge is isolated automatically on a CIA signal by two in-line containment isolation trip valves. Similarly, the reactor containment sump discharge is also isolated automatically on a CIA signal by two in-line isolation trip valves. Isolation of the vent and drain lines assures that the pressurizer overflow remains in the reactor containment.

Rupture discs are installed on the pressurizer relief tank to protect the tank from overpressurization. High-pressure gas from the pressurizer could rupture the discs. Installation of a relief valve should be considered to prevent rupturing the discs and thus reducing the amount of activity release to the containment atmosphere through water and steam released from ruptured discs.

The alternate route to letdown from the reactor coolant system utilizes the excess letdown heat exchanger (CH-E-4). The letdown flow can be discharged from the reactor coolant loop (or loops)

into the primary drain transfer tank. This can be achieved by opening one or more of the motor operated valves (MOV-RC-557A, 557B, and 557C) on the reactor coolant legs and lining up the excess letdown divert valve (HCV-CH-389) to the drain system (see Figure 4-3).

With the hypotheses that there is either no break or only a minor pipe break, the CIA is activated and two of the three reactor coolant pumps are running, the excess letdown flow-line may be effectively used to maintain the water levels in the pressurizer and the reactor pressure vessel. It is assumed that the charging pump (or pumps) are running, the minimum flow and test line in the pump discharge is opened, and the normal charging line through the regenerative heat exchanger (CH-E-3) to the RC-P-1B cold leg is opened. Under these conditions, letdown outflow has to balance not only with inflow through the charging line, but also with seal water leakage into the RCS from the reactor coolant pumps. Since letdown through CH-E-4 is limited by line flow resistance, the charging flow rate must be reduced with flow control valve FCV-CH-122 in the charging line to balance the letdown outflow.

To reduce the charging flow rate, the following operator actions may be necessary: (1) leave only one charging pump running, and (2) open the needle valve which is in parallel with the orifice in the minimum flow and test line in the pump discharge.

It is assumed that cooling water is available to CH-E-4. The letdown flow is discharged to the containment sump through a relief valve in the bottom of the primary drain transfer tank (DG-TK-1). Since the capacity of DG-TK-1 (700 gallons) is limited, the letdown flow depends on proper operation of the relief valve. To ensure flow of the letdown stream into the containment sump, it is recommended that the present locked-shut manual valve in the excess letdown bypass line to the containment sump be replaced with a

remotely operated valve. The remotely operated valve would be operated from the Control Room and normally key-locked.

4.2 Safety Injection System

The Safety Injection System operates in two phases; the injection phase and the recirculation phase. The injection phase involves any combination of the following operations: 1) injection of borated water into the reactor vessel from the passive accumulators, 2) initial injection of borated water into the reactor vessel from the boron injection tank by the high head safety injection (HHSI)/charging pumps taking suction from the refueling water storage tank (RWST), and 3) continued injection with the high head safety injection/charging pumps and with the low head safety injection (LHSI) pumps both drawing borated water from the refueling water storage tank.

The transfer from safety injection phase to recirculation phase will automatically take place on a low level signal from the RWST. The recirculation phase involves recirculation of coolant and injection water released to reactor containment back to the reactor vessel from the reactor containment sump, by using the LHSI pumps and HHSI/charging pumps if the reactor pressure is still high.

In the recirculation phase, if the reactor containment sump water is highly radioactive, the radioactive water is recirculated through the Safety Injection System. Figure 4-4 presents a schematic drawing which shows the flow paths of the radioactive water in the recirculation phase.

Redundant motor operated valves (MOV's) in the minimum flow and test line from the HHSI/charging pump discharge are closed on safety injection signal, so the potential radionuclide flow path through the minimum flow line into the volume control tank of the Chemical and Volume Control System is effectively cut off. Similarly, redundant MOV's in the reactor coolant pump seal water return line, which is taken from the HHSI/charging

pump discharge, are closed on containment isolation signal A (CIA). During the automatic switchover from the injection to the recirculation phase, redundant MOV's in the minimum flow line from the LHSI pump discharge are automatically closed and the HHSI/charging pump suction is realigned from the RWST to the discharge of the LHSI pumps. Therefore, the potential radionuclide flow path from the LHSI pump discharge to the RWST is blocked off. A pressure relief valve is installed in each of the LHSI pump discharge lines plus the common LHSI lines to the RCS cold leg loops. If the LHSI pump discharge line is blocked while the pump is running, the reactor containment sump water may be drained into the safeguards area sump through the pressure relief lines. LHSI pump discharge high pressure alarm may be needed to warn the operator of blocked pump discharge. However, blockage of the LHSI pump discharge line is not likely to occur, because the MOV's in the connecting lines between the LHSI pump discharge and HHSI/charging pump suction are open upon receipt of the switchover signal during the transfer of injection mode to recirculation mode. The design effectively prevents any radioactive water from leaking into other systems from the Safety Injection System.

Table 4-2 summarizes predicted post-accident radiation levels and occupancy requirements during the recirculation phase. The LHSI pumps (two pumps) are housed in separate pump cubicles in the west safeguards area. Each pump cubicle is totally enclosed by shielded concrete walls and concrete floor above the pump cubicle. The only access into the pump cubicle is by removing the shielded plug on the floor above and descending to the pump floor by use of a ladder or through the open doorway on the 747' level. Streaming should not be a problem out the open doorway; it should be administratively controlled such as by roping it off. The suction line to each LHSI pump from the reactor containment sump is routed to the pump cubicle through a dedicated penetration in the reactor containment wall. Similarly, the discharge lines (cold and hot legs) from the LHSI pumps back to the reactor vessel are routed through dedicated penetrations in the reactor containment wall. The LHSI pumps are redundant pumps, each with 100 percent capacity. There is no need for personnel

access into the shielded pump cubicles during an accident condition. Based on the indicated radiation levels and no expected personnel access requirements, additional shielding for the pump cubicles is not necessary. Potential high radiation areas during the recirculation phase is indicated in Figure 4-5.

MOV's in the LHSI pump suction lines are installed in the valve pit which is on the 687'-11" level in the west safeguards area. The valve pit is well shielded from floor level 747', where access to the valve pit is possible by use of a ladder. These MOV's can be manually opened or closed with a reach rod from the shield floor above, if necessary.

The LHSI pump discharge line to the suction of HHSI/charging pumps (recirculation lines) runs under the concrete floor which is used as personnel pathway in the safeguards area and areas contiguous to containment and runs in the shielded concrete pipe vaults in the Auxiliary Building. Radiation levels in the pathways in the safeguards area and Auxiliary Building indicate that these areas are accessible for passing through under administrative control (Figure 4-5). Exception is the area where the recirculation lines are exposed in the west-wall pathway and over the pipe trench No. 1 near the charging pump cubicles at 722'-6" elevation in the Auxiliary Building. This area will be highly radioactive during the recirculation phase and should be administratively controlled to prevent the personnel from passing by the exposed recirculation lines. Radiation levels in these areas decrease rapidly with time (Table 4-2).

Shortly after switchover to recirculation phase, the operator must operate the HHSI/charging pump suction and discharge valves manually from the south wall of the pump cubicles to set up redundant and independent flow paths from containment sump through HHSI discharge to cold legs. The operator will be exposed to radiation from the pipe trench No. 2 and also from the Boron Injection Tank which is about 30 feet away. Assuming that it takes about 5 minutes for the valve switching

operation, the operator will receive dosage which is appreciably less than the allowable limit of 5 Rem whole body. The following change in operating procedure, however, is recommended to reduce radiation exposure:

- o Switch from injection through the Boron Injection Tank line to the HHSI discharge through cold legs prior to start of the recirculation phase, or

- o Delay switching from injection through the Boron Injection Tank line to the HHSI discharge through cold legs during recirculation phase to allow sufficient time for radioactive decay in the Boron Injection Tank and recirculation lines.

There is no need for personnel to be on the floor level below the pathway where the LHSI pump discharge lines are routed in the safeguards area except in the pipe penetration area where access into this area is needed to line up the hydrogen recombiners (inlet manual valves 101 and 102 or 103 and 104) and cross-connect instrument air to the containment instrument air (Valve No. ISA-90). It must be assumed that any accident in containment will result in loss of instrument air in containment. The only way to restore instrument air into containment is by cross-connecting it from the station instrument air. The radiation level is estimated at 3,000 R/hr at one hour after accident in the pipe tunnel where the LHSI lines are routed. It is recommended that the recombiner inlet manual valves and the instrument air valve (ISA-90) be made remotely operable.

The HHSI/charging pumps (three pumps) are housed in separate pump cubicles in the Auxiliary Building at floor level 722' 6". Each cubicle is enclosed with the shielded walls and the floor above the cubicle (elevation 735'-6"). Personnel access into the pump cubicles is not needed during the recirculation phase when highly radioactive sump water may be pumped. Access to other areas of the Auxiliary Building is readily available on the floor levels 722'-6" and 735'-6" by passing around the pump cubicles. If containment isolation signal B (CIB) is activated, the HHSI/charging

pump cubicles are automatically vented to the Supplementary Leak Collection and Release System. It may be a good safety precaution to rope off the floor area directly above the pump cubicles and make certain that the shielded plugs (for personnel access to the pump cubicles) are firmly in place.

The HHSI/charging pump discharge line is routed from the Auxiliary Building through the area contiguous to containment to the reactor vessel through a dedicated containment wall penetration. In the Auxiliary Building the pump discharge line is routed through the recessed pipe tunnel at floor level 722'-6" and the shielded concrete pipe vault. In the area contiguous to containment, the pump discharge line runs in the pipe tunnel below floor level 735'-6". There is no equipment on the floor below elevation 735'-6" in the area contiguous to containment and personnel access is not needed during operation of the Safety Injection System recirculation phase.

The high head safety injection line (MOV-SI-836 to cold leg loops) is opened during the recirculation phase. The injection line through the boron injection tank (BIT), which is used during the injection phase, is left open during the recirculation phase. Since the BIT has a 900-gallon capacity, the radiation level outside of the BIT cubicle remains high with the radioactive sump water flowing through the BIT during the recirculation phase. The calculated contact dose rates at the outer surface of the 2-ft. concrete shield wall are as follows: 810 R/Hr at one hour after shutdown, 180 R/hr at 10 hours after shutdown, and 87 R/Hr at 24 hours after shutdown (Table 4-2). The BIT is located in the Auxiliary Building at elevation 722'-6". Although occupancy is not required in the vicinity of the BIT under an accident condition, strict administrative control is necessary to keep personnel away from the BIT. This will ensure that personnel exposures are maintained within the limits specified in Section 3.6.

There is a pressure relief valve on the boron injection recirculation return line. The pressure relief line is routed to the floor drain via

boron injection surge tank. However, the pressure relief valve setting is higher than the shutoff head of the HHSI/Charging pump.

Based on: (1) review of the pipe equipment layouts, (2) the flow paths of the Safety Injection System in the recirculation phase, (3) predicted radiation levels, and (4) personnel occupancy requirements, it is concluded that the existing shielding is sufficient to protect the personnel from over-radiation exposure during an accident condition with strict administrative controls. It is believed that administrative controls can be appropriately implemented to maintain personnel exposure well within limits specified in Section 3.6. To make certain that equipment and instruments in the Safety Injection System are in good operating condition at all times, the plant preventive maintenance procedures should be reviewed and revised as necessary and surveillance and other tests should be performed as required. Roping off or barricading affected areas and administratively controlling access to these areas are considered effective means for controlling personnel exposure. Consideration should be given to developing methods for draining and decontaminating any pump and the pump cubicle if maintenance becomes necessary due to pump malfunction.

4.3 Containment Depressurization System

The Containment Depressurization System is designed to cool and depressurize the containment to subatmospheric pressure in less than 60 minutes following a design basis accident (DBA). The Recirculation Spray Subsystem of the Containment Depressurization System is capable of maintaining containment at subatmospheric pressure for several months following a DBA. There are four recirculation spray pumps which take suction from the containment sump. Two of the recirculation spray pumps are located outside the containment. If the sump water is highly radioactive, the radioactive water is recirculated through the recirculation spray pumps. The outside recirculation spray pumps, components and piping are contained within the annular safeguards structure adjacent to the west side of

containment. Potential radionuclide flow paths through the recirculation spray subsystem, using the outside recirculation spray pumps, are shown in Figure 4-6.

Table 4-3 summarizes radiation levels in the outside recirculation spray pump system. The outside recirculation spray pumps (two pumps) are housed in separate pump cubicles in the west safeguards area (see Figure 4-5). Each pump cubicle is totally enclosed by shielded concrete walls and concrete floor above the pump cubicle. Access into the pump cubicle is by removing the shielded plug on the floor above and descending to the pump floor by use of a ladder or by entering through an open doorway into the cubicle on the 747' level. Streaming should not be a problem, but the open doorway should be administratively controlled, such as by roping it off. The suction line to each outside recirculation spray pump of the containment sump is routed to the pump cubicle through a dedicated penetration in the containment wall. The outside recirculation spray pumps are redundant pumps. The pumps and valves are all remotely operable and located in the west safeguards area which is not normally occupied. There is no need for personnel access into the shielded pump cubicles during an accident condition. Additional shielding for the pump cubicles is therefore not necessary.

MOV's in the outside recirculation spray pump suction lines are installed in the valve pit which is on the 687'-11" level. The valve pit is well shielded from floor level 747' where access to the valve pit requires the use of a ladder. These MOV's can be manually opened or closed with a reach rod from the shielded concrete floor above, if necessary.

It is not anticipated that the outside recirculation spray pumps will have to function for more than a few days at most; all need for them should end within a few months.

4.4 Containment Vacuum and Leakage Monitoring System

The Containment Vacuum and Leakage Monitoring System consists of five (5) subsystems, as follows:

1. Containment Pressure Sensing Subsystem (Safety-Related)

2. Reference Volume Subsystem (Non-Safety Related)

3. Containment Atmosphere Monitoring Subsystem (Safety Related)

4. Containment Vacuum Subsystem (See Note 1, below)

5. Containment Blowdown Subsystem (Non-Safety Related)

NOTE 1: Containment Vacuum Subsystem must be actuated, per Technical Specification, to reduce the containment atmosphere pressure to a minimum of 8.9 psia prior to becoming critical and during all power reactor operation. The subsystem must maintain the containment atmosphere within the specified subatmospheric range during reactor operation or the reactor must be shut down. This system <u>should not function after either a LOCA or TMI-2 type event</u>.

The safety-related functions of the Containment Pressure Sensing Subsystem are:

a) Monitor the containment atmosphere pressure for post-LOCA or TMI-2 type event, and

b) Actuate the Containment Isolation System when a pre-set pressure is achieved inside containment.

The safety-related function of the Containment Atmosphere Monitoring subsystem is to provide periodic evaluation of the concentration of radioactive particulates and noble gases in the containment atmosphere. The existing equipment is probably inadequate for the post-event situation and isolates on CIA and should remain isolated.

The Containment Pressure Sensing Subsystem is the only subsystem of those indicated above that will contain radioactive gases after a LOCA or TMI-2 type event. Referring to Figure 4-7, the four 3/8" pipes between the inside of containment to the pressure sensing instruments outside of containment pass through containment at an azimuth of approximately 20° on the far side of the shield wall to the right of pumps CV-P-1A and CV-P-1B. The system contains flow rate restricting orifices and particulate filters inside containment, with manual valves and instruments outside containment.

The source term is the volume of containment atmosphere contained within 4-3/8" diameter pipes, each 30 feet long. This radiation source is conservatively approximated by a single 1" diameter pipe located 3'-0" above the 722'-6" floor elevation and 3' horizontally from the "Receptor" outside the 2' thick concrete shield wall. Figure 4-8 shows the potentially radioactive area of the Containment Pressure Sensing Subsystem.

Table 4-4 provides pertinent data for evaluating radiation hazards for post-accident release of gases to the Containment Vacuum and Leakage Monitoring System. Based on the radiation levels (contact with shielding) and the occupancy requirements, it is concluded that personnel exposures can be maintained within the allowable limits specified without any shielding modifications.

4.5 Supplementary Leak Collection and Release System

The function of the Supplementary Leak Collection and Release System is to ensure that radioactive leakage from the reactor containment following a Design Basis Accident (DBA), radioactive release due to a fuel handling accident, or radioactive material released in the waste gas storage area or areas contiguous to containment is collected and filtered for iodine removal prior to discharge to the atmosphere at an elevated release point.

The Supplementary Leak Collection and Release System consists of two 100 percent capacity leak collection exhaust fans. Air is exhausted from the fuel building, waste gas storage area, blowdown tank room, personnel access hatch area, east and west cable vaults, pipe tunnel, north and west safeguards, MCC-1 area and the Hydrogen Recombiner-1 room.

During normal operation, the exhaust to the fan does not go through filters. On a containment isolation phase A signal (CIA) or a high-high radiation signal from the monitor serving areas contiguous to the containment, from the fuel building monitor, the waste gas storage area monitor, or the leak collection system exhaust monitor, flow is diverted so that it first flows through one of the two parallel main filter banks before flowing to the leak collection exhaust fans.

During normal power operation, the three charging pump cubicles are exhausted by the Auxiliary Building Ventilation System A. In the event of a containment isolation phase B (CIB) signal, the isolation dampers to the Auxiliary A Exhaust System are closed and the parallel dampers which connect the charging pump cubicles to the Supplementary Leak Collection and Release System are opened. This assures that any leakage or release from the charging pump cubicles is filtered via the main filter bank prior to release at the elevated release point.

Figure 4-9 presents a simplified flow diagram of the Supplementary Leak Collection and Release system. The primary potential for radionuclide transport is through leaks through containment penetrations. Other potential sources of leakage are the external recirculating loops of the Safety Injection System and Recirculation Spray Subsystem. However, leakage from the external recirculation loops (pump seals, valves, and flanges) is negligibly small compared to the containment leak rate, unless there is failure of a pump seal or excess leakage from a valve or flange (BVPS FSAR, Tables 6.3-9 and 6.4-4). Pump seal failure and excess leakages from valves and flanges are not considered in the dose rate calculation. The containment leakage rate was assumed to be 0.10 percent by weight of the containment air per 24 hours (limit of Technical Specications, Appendix A).

Table 4-5 summarizes radiation levels from the Supplementary Leak Collection and Release System. The radiation levels from the main ducts were calculated at the distances of 10 ft and 20 ft. Dose rates are the estimated maximum values. As the reactor containment is depressurized with spray cooling, the radiation levels will decrease rapidly. The radiation level of the filter bank with two-foot thick shielded wall was based on iodine removal for 30 minutes at the assumed maximum containment leak rate. The 30 minutes duration was used with the assumption that the containment can be depressurized in approximately 30 minutes after a LOCA. Figure 4-10 illustrates radiation levels from the Supplementary Leak Collection and Release System.

The radiation levels indicate that passage near the vent ducts and around the shielded filter banks is permissible as the need arises. It is expected that post-accident occupancy requirements in the vicinity of system ducting and filter bank will be minimal. Although the exhaust fans are in an open area at elevation 768'-7" of the Auxiliary Building, the radiation level is much lower than in the main ducts upstream of the filter banks. Therefore, it is concluded that addition of shielding is not necessary. Personnel exposures can be controlled by implementing personnel access restrictions.

4.6 Hydrogen Recombiner

Two redundant Hydrogen Recombiner Systems are installed to maintain hydrogen content in the containment below the lower flammability limit in the event of a LOCA. A simplified flow diagram of the Hydrogen Recombiner System is shown in Figure 4-11. The Hydrogen Recombiner room consists of two hydrogen recombiner units, two hydrogen analyzers and one control and power panel for each of these systems. It is assumed that the hydrogen analyzer is operated within one hour after the accident and the hydrogen recombiner at approximately 24 hours after the accident. There is a biological shield wall between the recombiners and control panels. The shield wall is made of concrete with a thickness of 1.5 feet and extends 12 ft. from the containment wall. Two 6" Low Head Safety

Injection (LHSI) discharge pipes run 167" below the control panel area, but the control panel area is on grating. Operator in the control panel room will be directly exposed to the LHSI lines. The radiation dosage received by an operator working inside the control panel area is summed over contributions from the H_2 recombiner units, the LHSI pipes, and several other pipes that go in and out of the recombiner room.

The source terms for radiation and shielding calculations were conservatively obtained as follows:

1) The total amount of radiation for each of 12 mean energy groups was obtained from the Origen Code for 2766 MW and 650 days of operation, then divided by containment free volume, 1.86×10^6 ft^3, to give the concentration of radioactivity level per unit volume of the gas inside containment.

2) For each recombiner unit, effects from two different sources are combined: one source is the tank containing the reaction chamber with heating and cooling pipes around it; the other is the vessel containing the blower and motor.

3) The gas volume in the reaction chamber and pipes surrounding the chamber is estimated as follows: the reaction chamber is assumed to be a right cylinder with 12" I.D. and 4'-0" height; the pipes surrounding it are estimated to have a total length of 140'; thus the total gas volume in the tank is about 6 ft^3. This amount of radioactive gas is then assumed to be a uniformly distributed line source of 3.3' located at the edge of tank closer to the control panel.

4) The blower and motor are mounted in a sealed stainless steel vessel of 20" I.D., and 47-1/2" in length. The gas volume is conservatively estimated as 8.7 $ft.^3$. Again, this amount of gas is assumed as a line source 4' long located at the edge of the vessel closer to control panel.

5) There are many pipe lines inside the recombiner room. Eight 3/8" pipes and two 1' pipes are assumed to contain radioactive gas. (For a list of these pipes refer to Table 4-6). For conservatism, each pipe is assumed to be 30' long and instead of going in the direction perpendicular to shield wall, it is assumed all of them are parallel to the shield wall and 12' away from the operator.

Because the biological shield wall extends only 12' from the containment wall, back-scattering of photons from the side wall should be considered. However, based on the calculation, most of the dosage that an operator would receive would be from two LHSI lines; e.g., at 1 hour after the accident he would receive a maximum of 3,000 R/hr from the LHSI lines versus 25 mR/hr from the recombiner room due to operation of the hydrogen analyzers. Thus, even if the dose rate from back-scattering is on the order of 25 mR/hr, it will be negligible compared with that from the LHSI lines.

At one hour after the accident, an operator will have to start the hydrogen analyzer and check the readings on the control panel. The occupancy time required for this activity should not exceed a total of 30 minutes. At approximately 24 hours after the accident the recombiner will be started. Approximately 30 minutes is required for this startup. Table 4-7 presents the radiation level and occupancy requirement in hydrogen recombiner control area. Figure 4-12 presents the recombiner room layout and radiation level at the control panel.

The estimated dose rate is maximum 3,000 R/hr at one hour after accident in the control panel area where operator is standing on the grating and directly exposed to the LHSI lines. The operator will be exposed to a maximum dose of 1,500 Rem during the occupancy time of 30 minutes. The radiation exposure far exceeds the limit of General Design Criteria 19 (i.e., 5 Rem total body). Therefore, the control panels must be relocated to a concrete floor area or the control panel room must be shielded from the LHSI lines.

Manual valves (101 & 102 or 103 & 104) on recombiner line have to be opened first to start the hydrogen analyzer and recombiner. These valves are located in the pipe penetration area of safeguards where the LHSI lines with radioactive containment sump water are routed. The radiation level is estimated at 3,000 R/hr at one hour after accident in the pipe tunnel where the LHSI lines are routed. Therefore, the operator cannot manually open the recombiner inlet manual valves. It is recommended that the manual valves be either changed to remotely operated valves or open from the floor above with shielding and using reach rods.

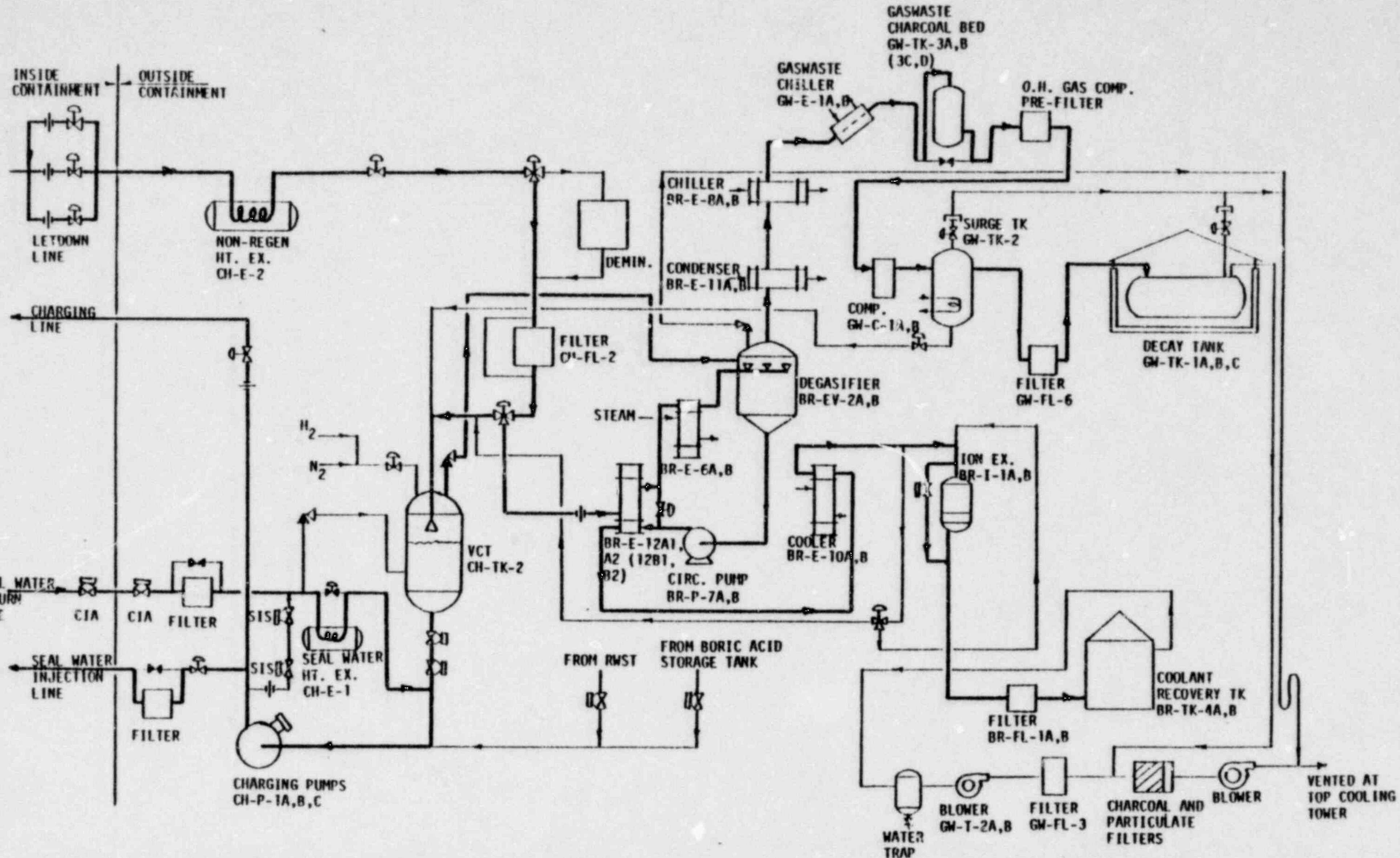
4.7 Post-Accident Sampling System

The existing reactor building sampling system draws liquid samples from the primary coolant cold leg, hot leg, pressurizer relief tank liquid and gas, residual heat removal and safety injection accumulator water as shown in Figure 4-13. Upon CIA actuation, the containment isolation valves close inside and outside containment. The inside and outside containment isolation valves are operated from one solenoid valve and a limit switch as shown on Figure 4-13. Therefore, if only one sample is required, all eight containment isolation valves must be opened. The liquid sample panel now in existence does not have facilities for handling such a highly radioactive sample.

There are no provisions for taking a sample of post-accident containment air in existence at this time. During an accident condition the line which contains a radiation monitor in the Containment Vacuum and Leakage Monitoring System is isolated, thereby leaving no method to monitor containment activity. Although the hydrogen analyzers are turned on within one hour after the accident, there is no method of extracting a sample of the containment from this system either. Table 4-8 describes basic sampling requirements as outlined in NUREG 0578.

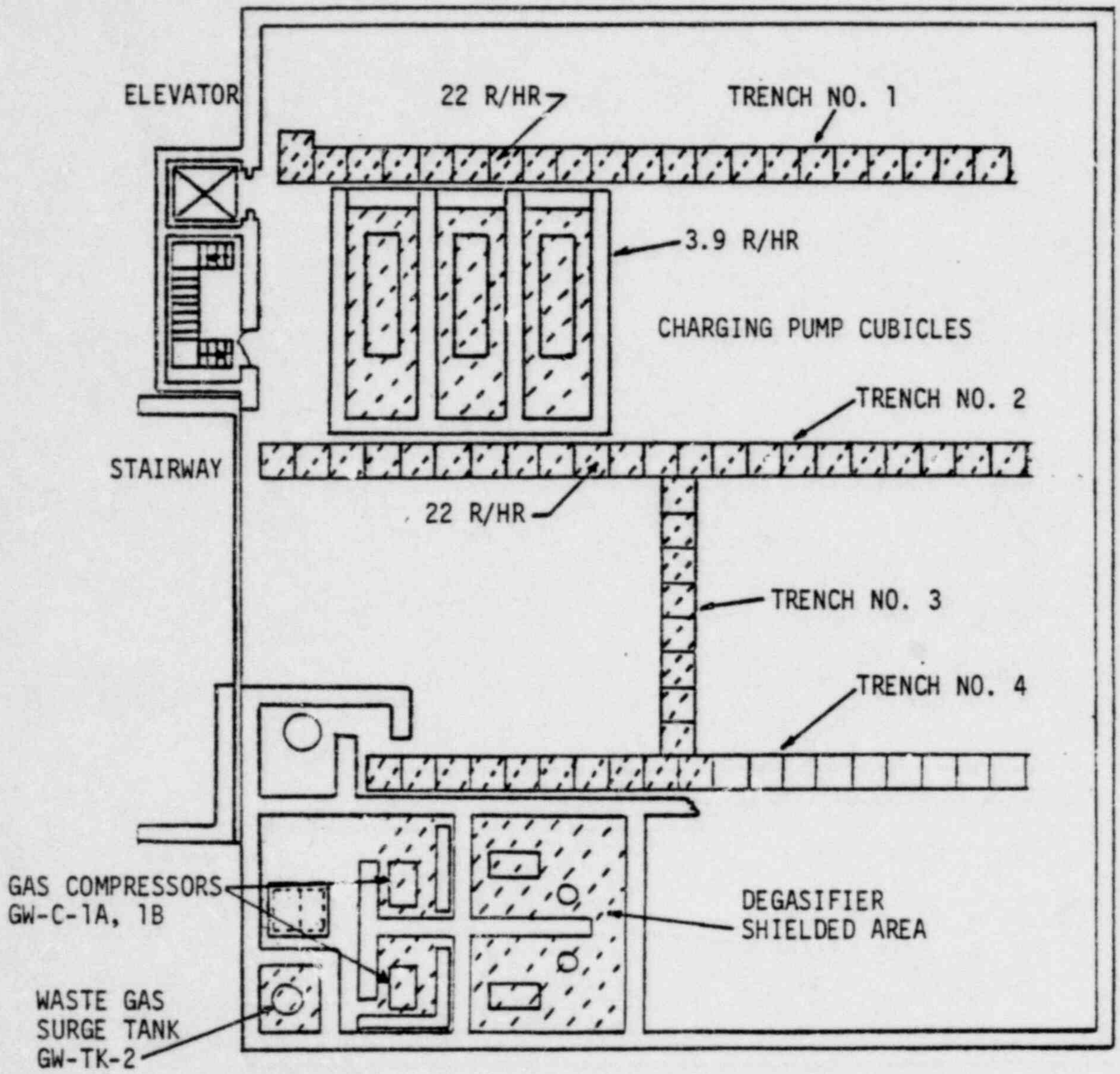
It is therefore recommended that a post-accident sampling system be built for sampling both liquid hot leg and containment air samples. Design of such a sampling system is not within the scope of this project.

However, a conceptual design is being proposed and a copy of the simplified flow diagram, plan, and evaluation views, location drawings, and arrangement drawings are included in Figures 4-14, 4-15, and 4-16.



NOTE: Potentially radioactive flow paths are indicated by heavy line. There are three knock out pots.

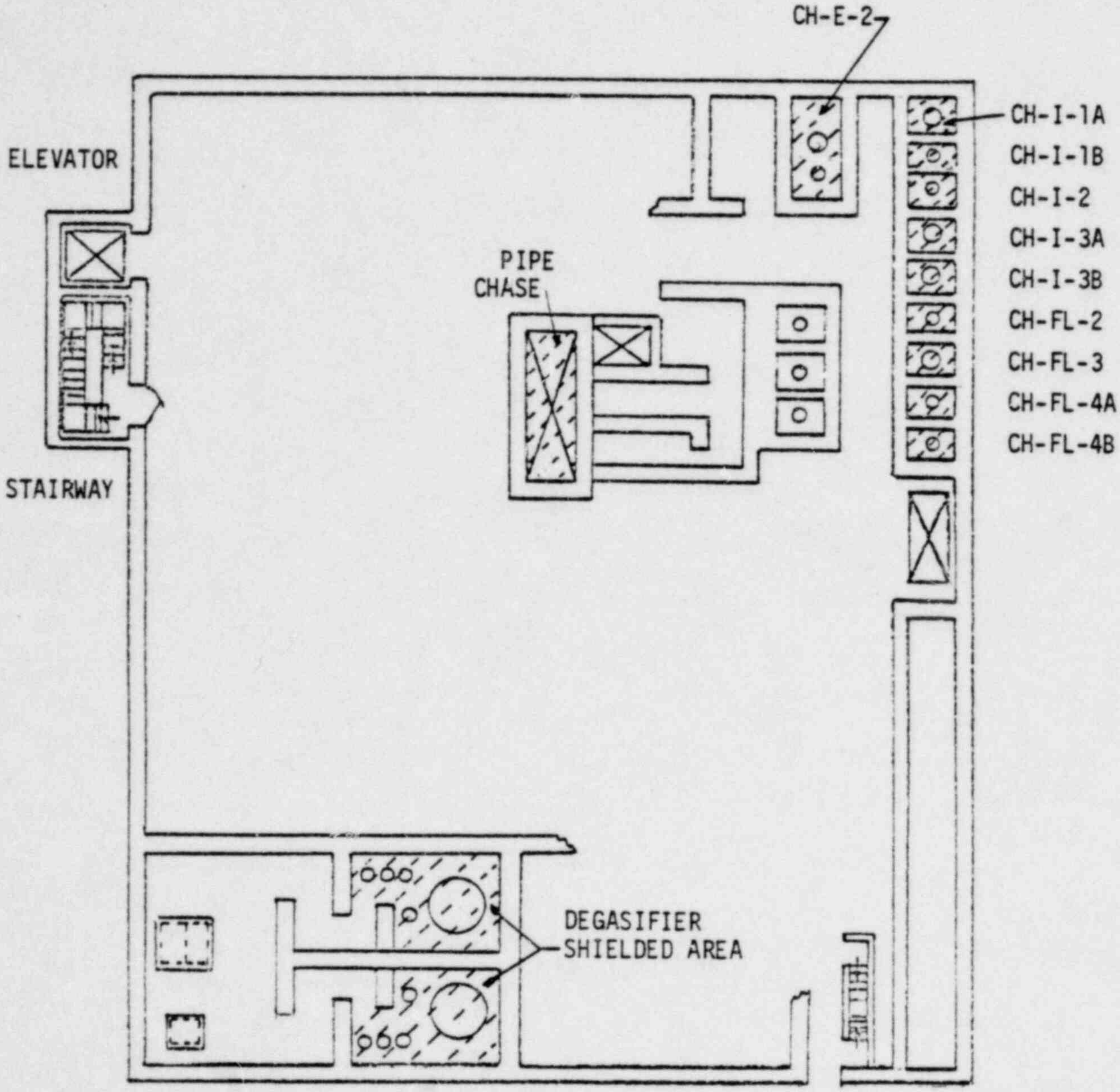
FIGURE 4-1 POTENTIAL RADIONUCLIDE FLOW PATHS WITH USE OF NORMAL CHS LETDOWN LINE



PLAN EL. 722'-6" IN AUXILIARY BUILDING

Note: (1) Shaded areas with dashed lines indicate potential radiation areas
(2) Radiation levels are estimated at the time immediately after accident, radiation readings are in contact with shielding.

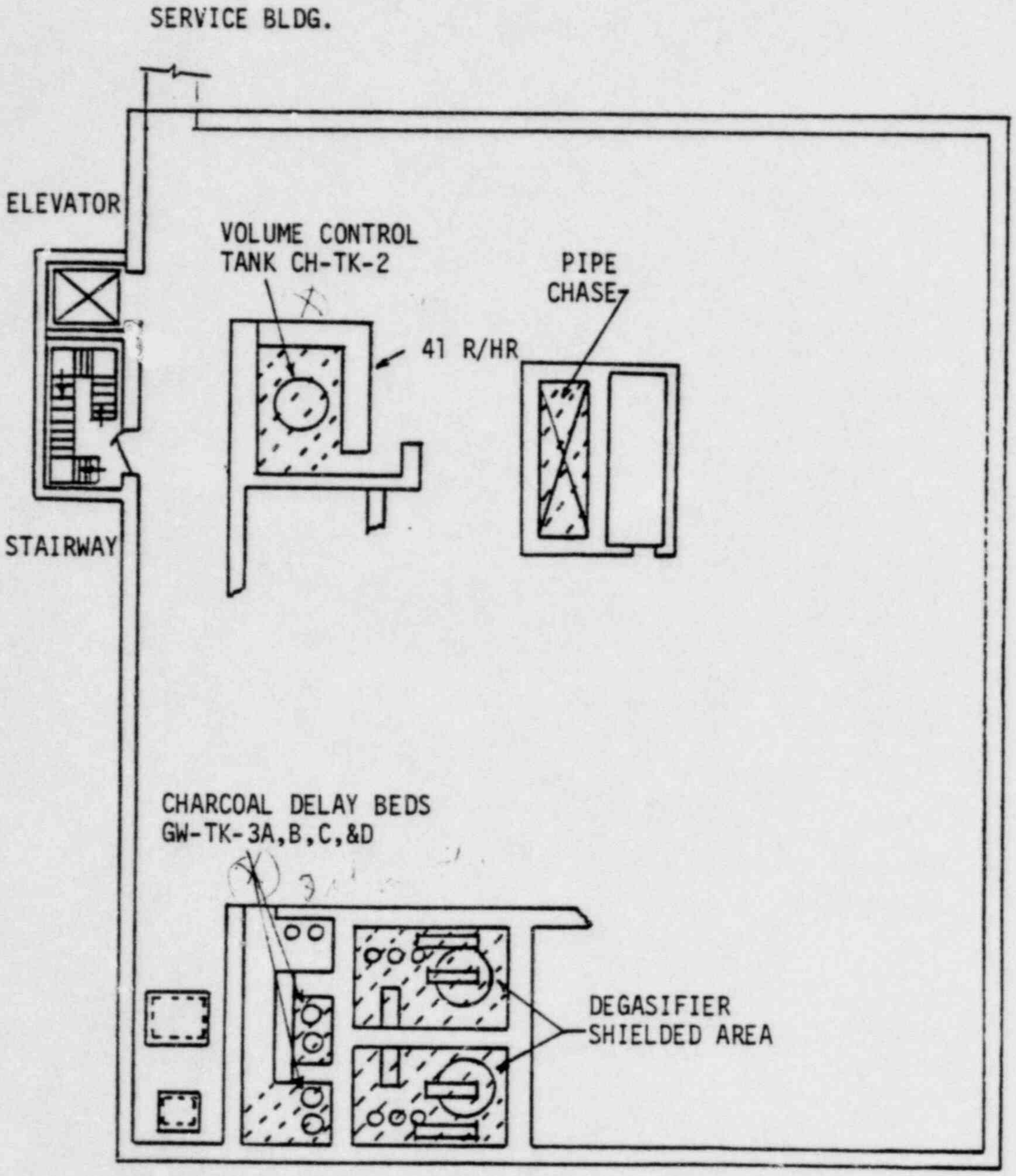
FIGURE 4-2A POTENTIAL HIGH RADIATION AREAS FROM USE OF NORMAL CHS LETDOWN LINE



PLAN EL. 735'-6" IN AUXILIARY BUILDING

NOTE: Shaded areas with dashed lines indicate potential radiation areas.

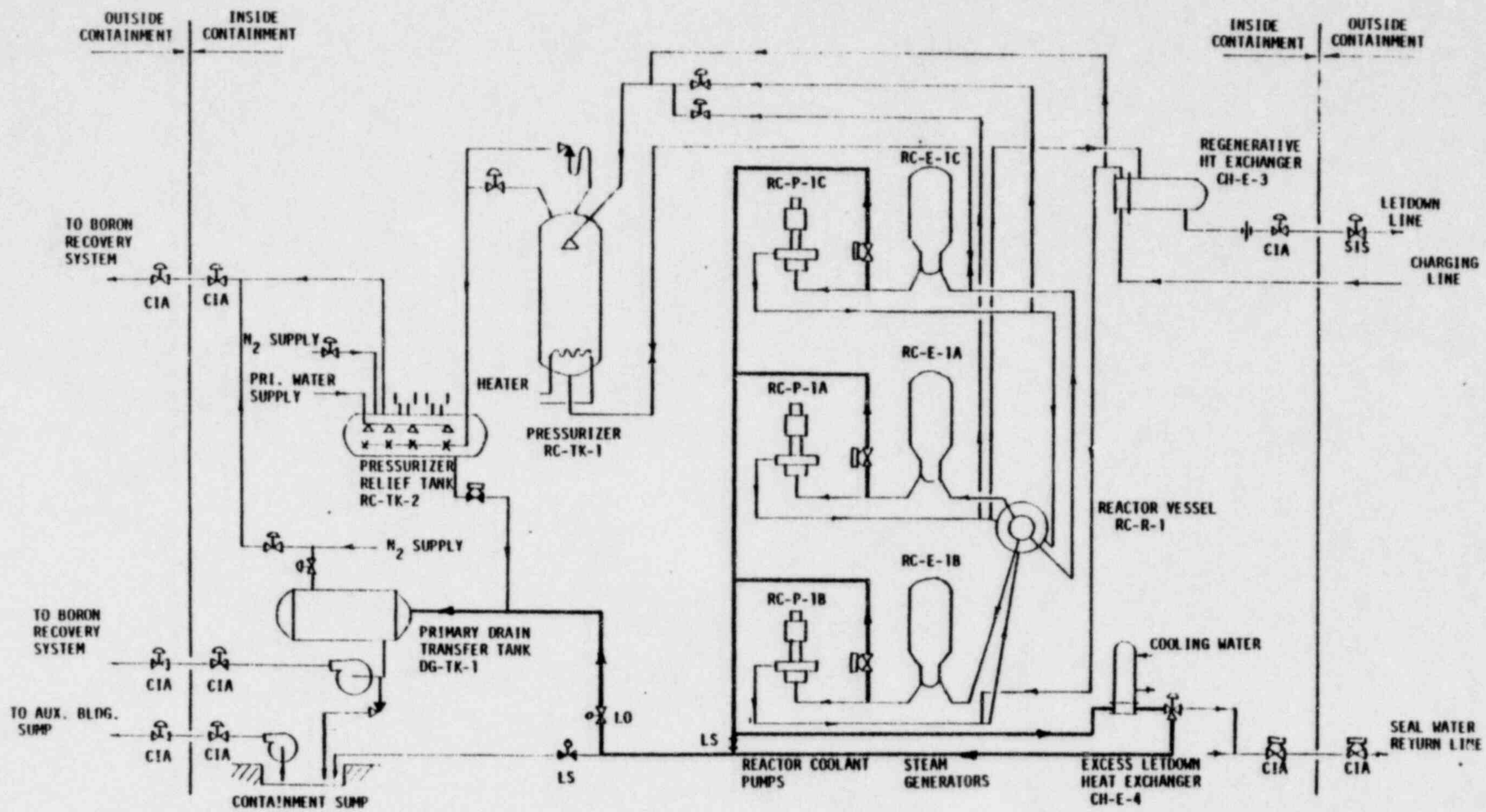
FIGURE 4-2B POTENTIAL HIGH RADIATION AREAS FROM USE OF NORMAL CHS LETDOWN LINE



PLAN EL. 752'-6" IN AUXILIARY BUILDING

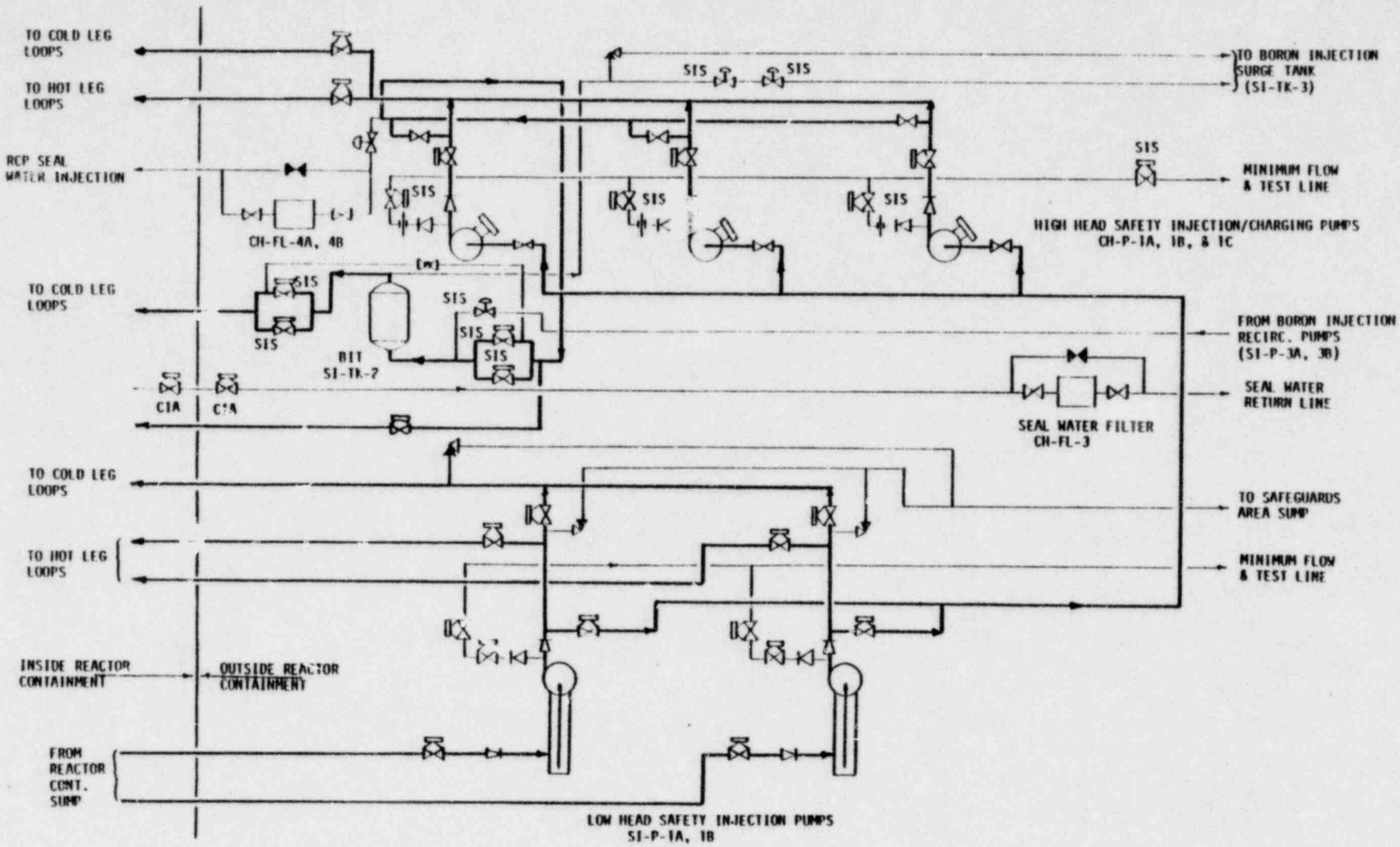
Note: Radiation level is estimated at the time immediately after accident and in contact with shielding.

FIGURE 4-2C POTENTIAL HIGH RADIATION AREAS FROM USE OF NORMAL CHS LETDOWN LINE



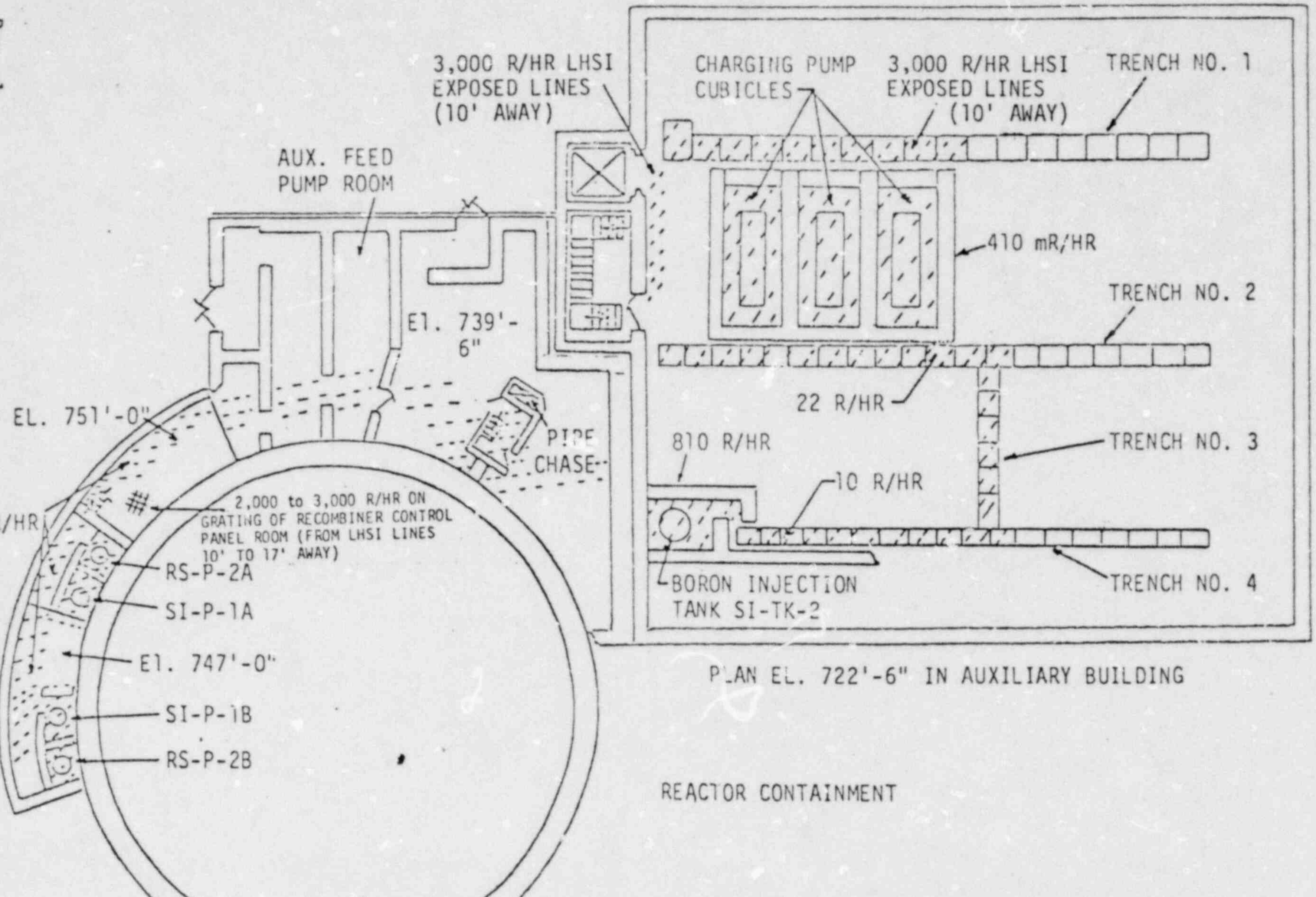
NOTE: Letdown flow path inside containment is indicated by heavy lines.

FIGURE 4-3 ALTERNATE LETDOWN FLOW PATH FOR CONTAINMENT ISOLATION



NOTE: Potentially radioactive fluid flow paths are indicated by heavy lines.

FIGURE 4-4 POTENTIAL RADIONUCLIDE FLOW PATHS IN SAFETY INJECTION SYSTEM RECIRCULATION PHASE



NOTE: (1) Shaded areas with dashed lines indicate potential radiation areas.
(2) Radiation levels are estimated at one hour after accident and in contact with shielding

FIGURE 4-5 POTENTIAL HIGH RADIATION AREAS DURING SAFETY INJECTION SYSTEM RECIRCULATION PHASE

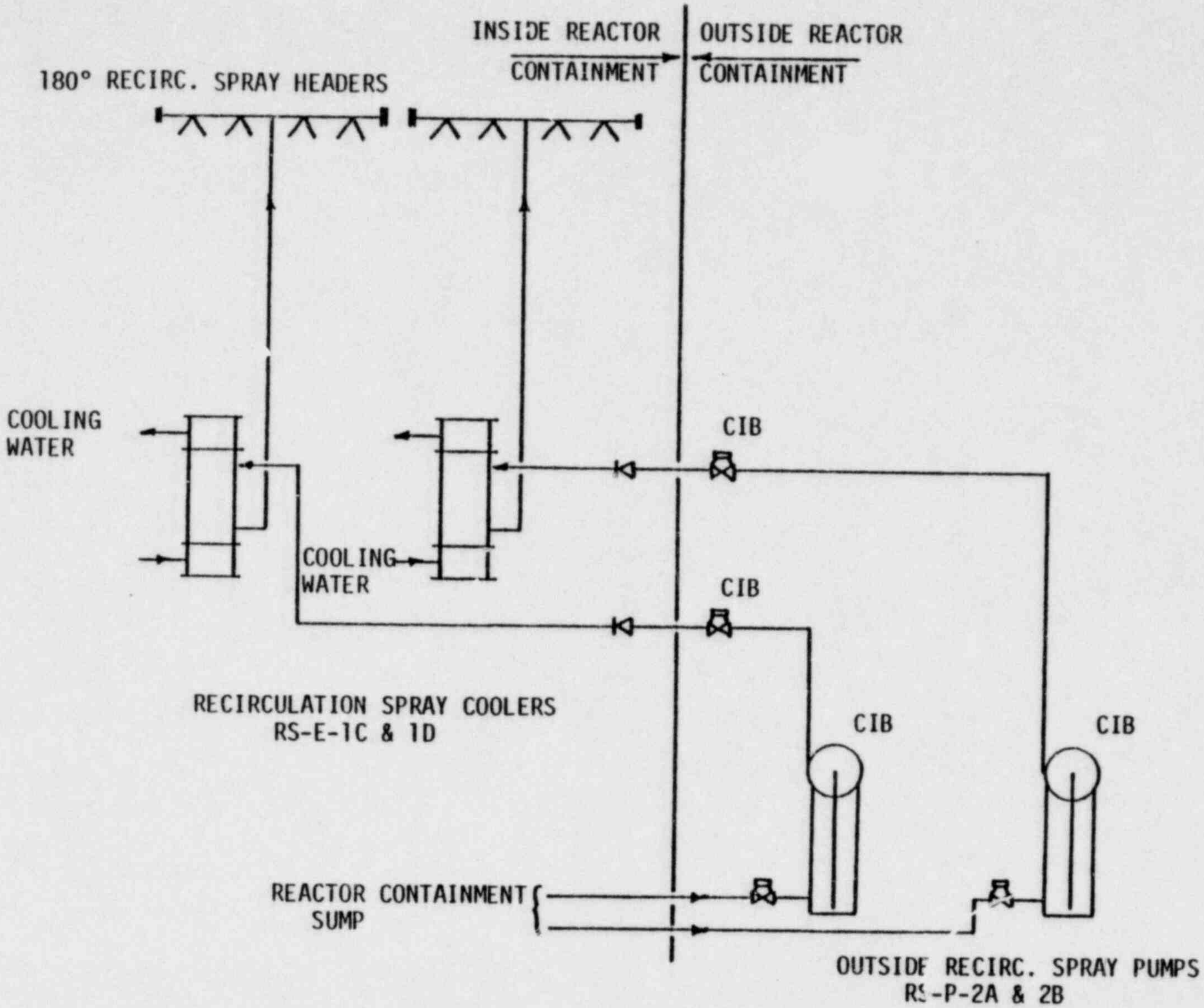


FIGURE 4-6 POTENTIAL RADIONUCLIDE FLOW PATHS THROUGH RECIRCULATION SUBSYSTEM WITH OUTSIDE RECIRCULATION SPRAY PUMPS

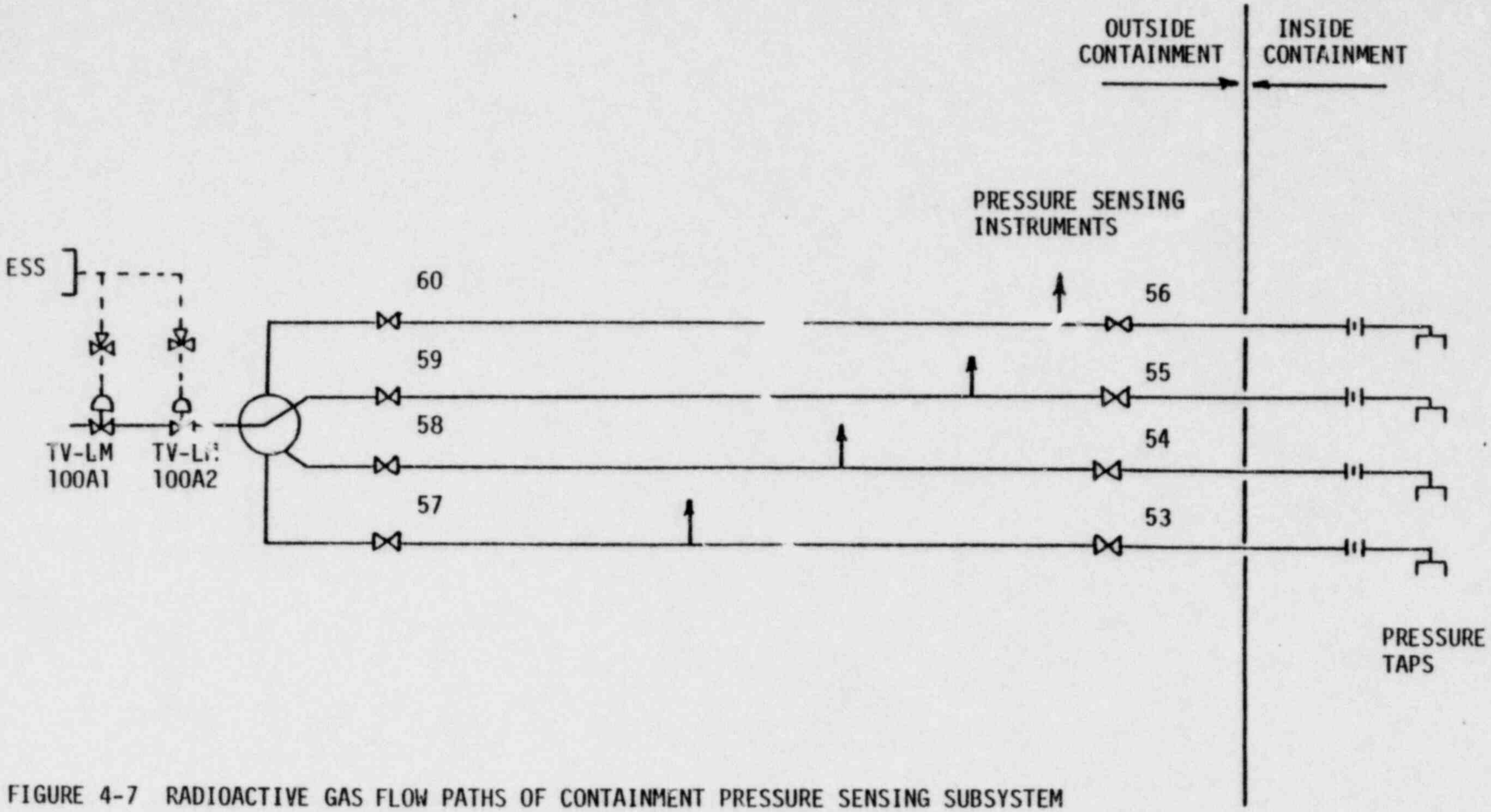


FIGURE 4-7 RADIOACTIVE GAS FLOW PATHS OF CONTAINMENT PRESSURE SENSING SUBSYSTEM

Note: (1) Shaded area with dashed lines indicates potential radiation area.
(2) Radiation level is estimated at one hour after accident and in contact with shielding.

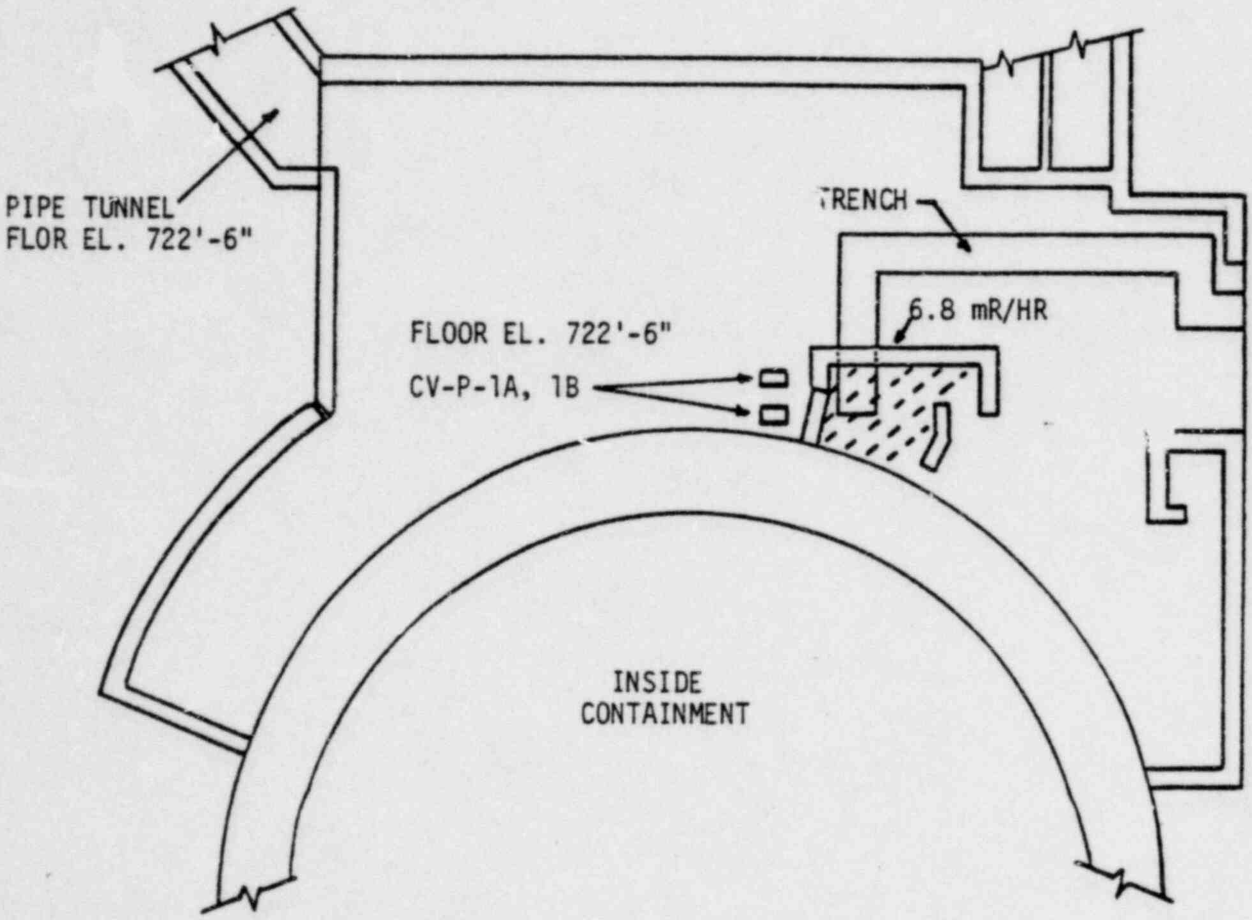
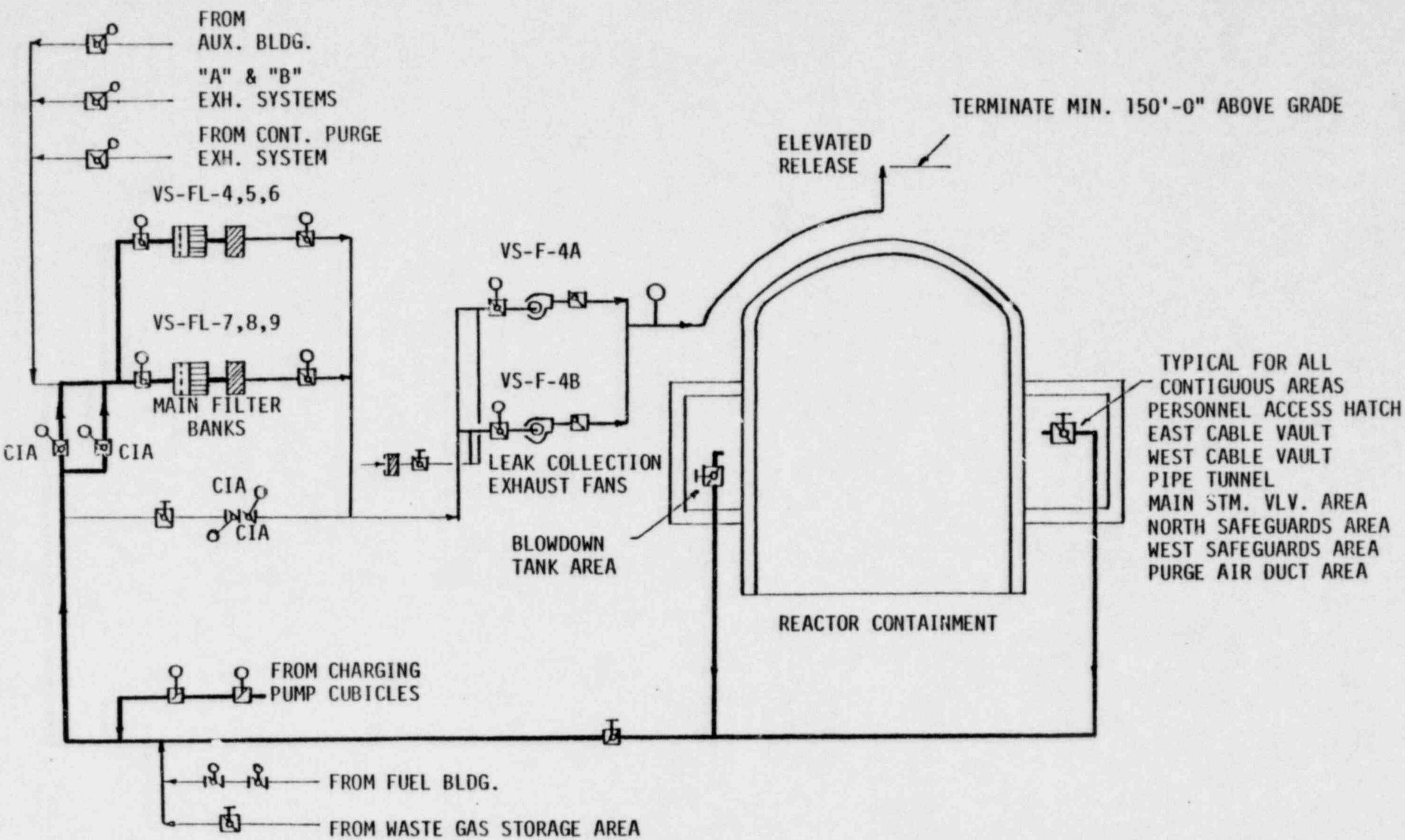


FIGURE 4-8 POTENTIAL HIGH RADIATION AREA OF CONTAINMENT PRESSURE SENSING SUBSYSTEM



NOTE: Potentially radioactive fluid flow paths are indicated by heavy lines.

FIGURE 4-9 POTENTIAL RADIONUCLIDE FLOW PATHS IN SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

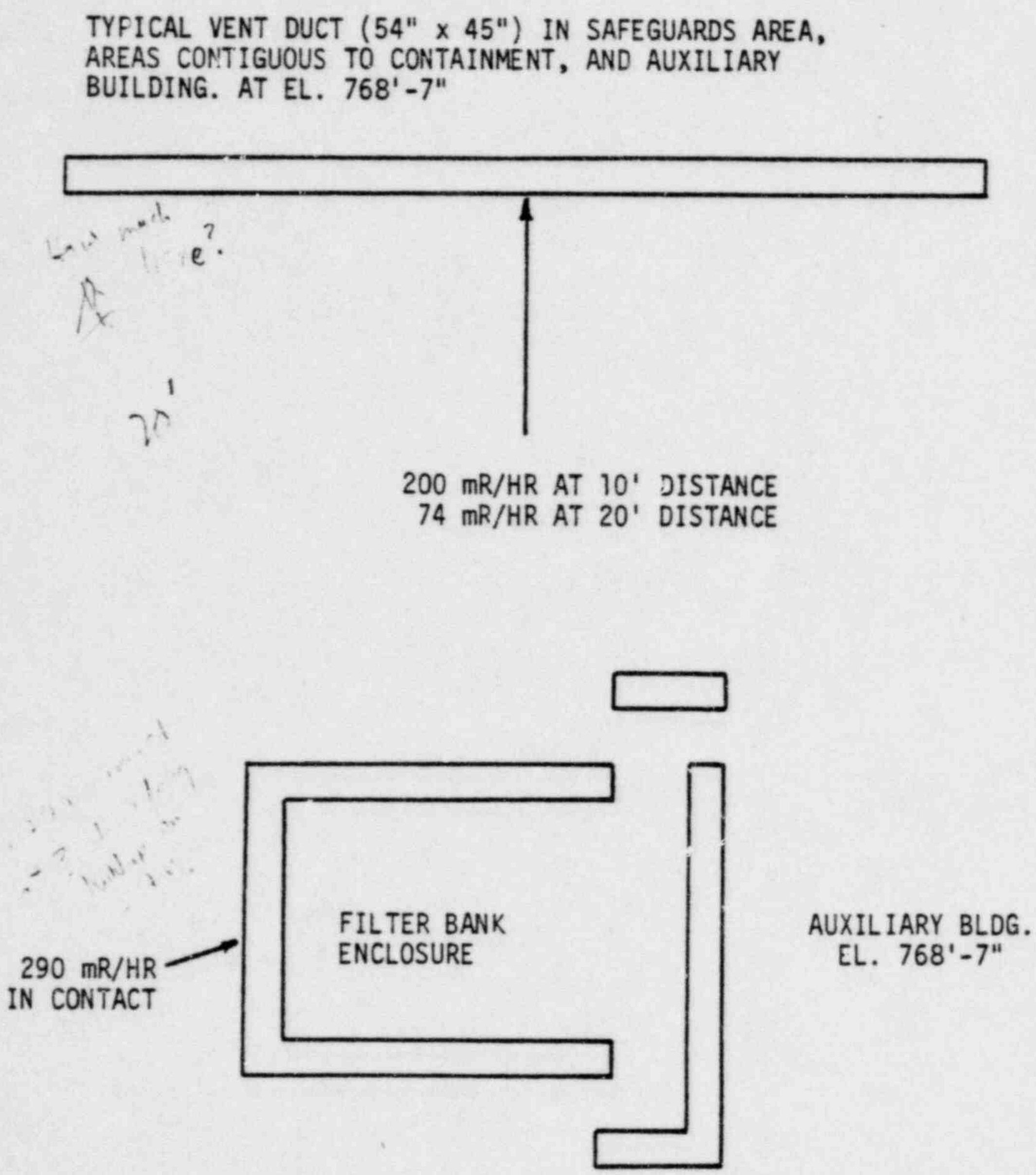


FIGURE 4-10 POTENTIAL HIGH RADIATION AREAS OF SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

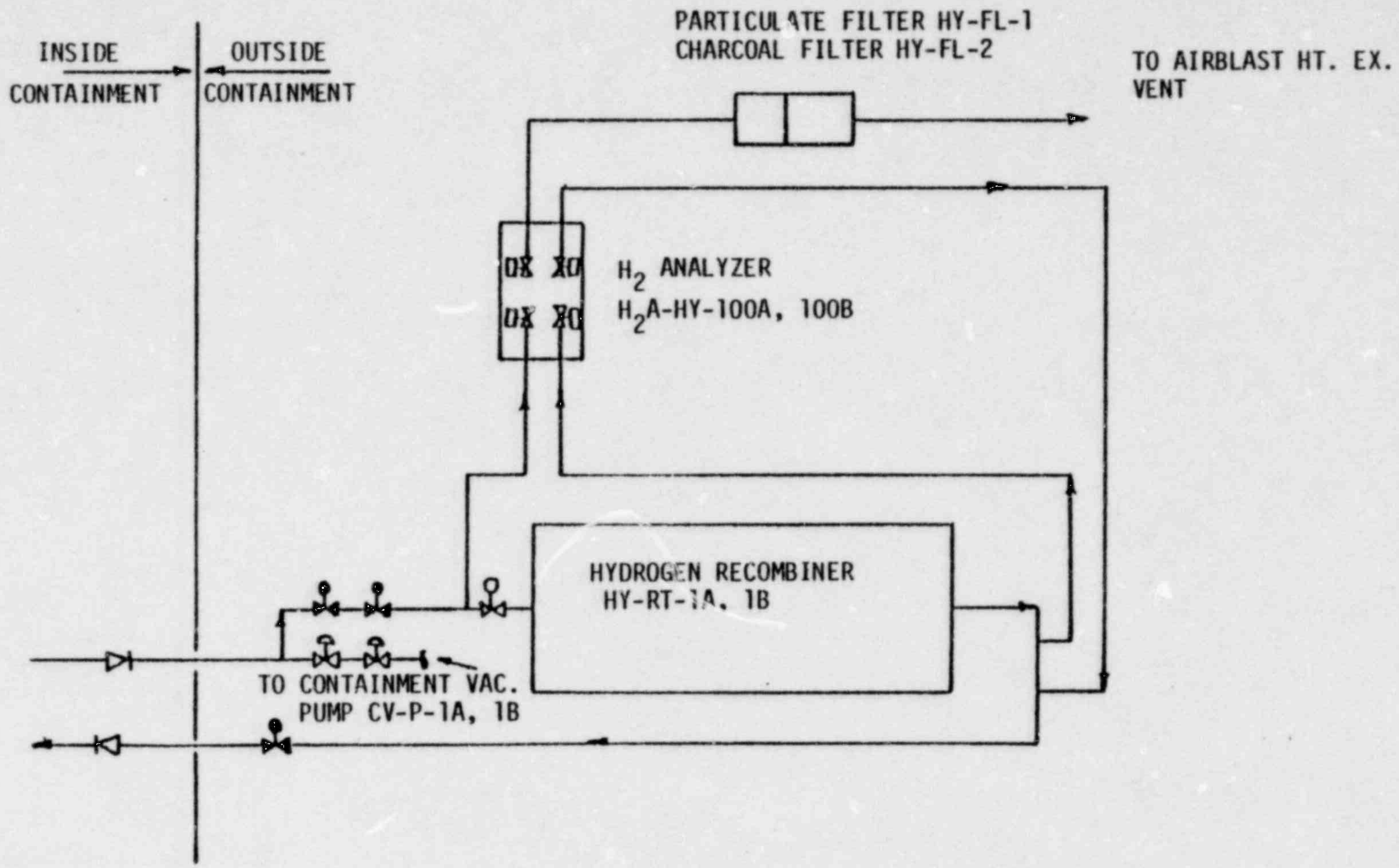
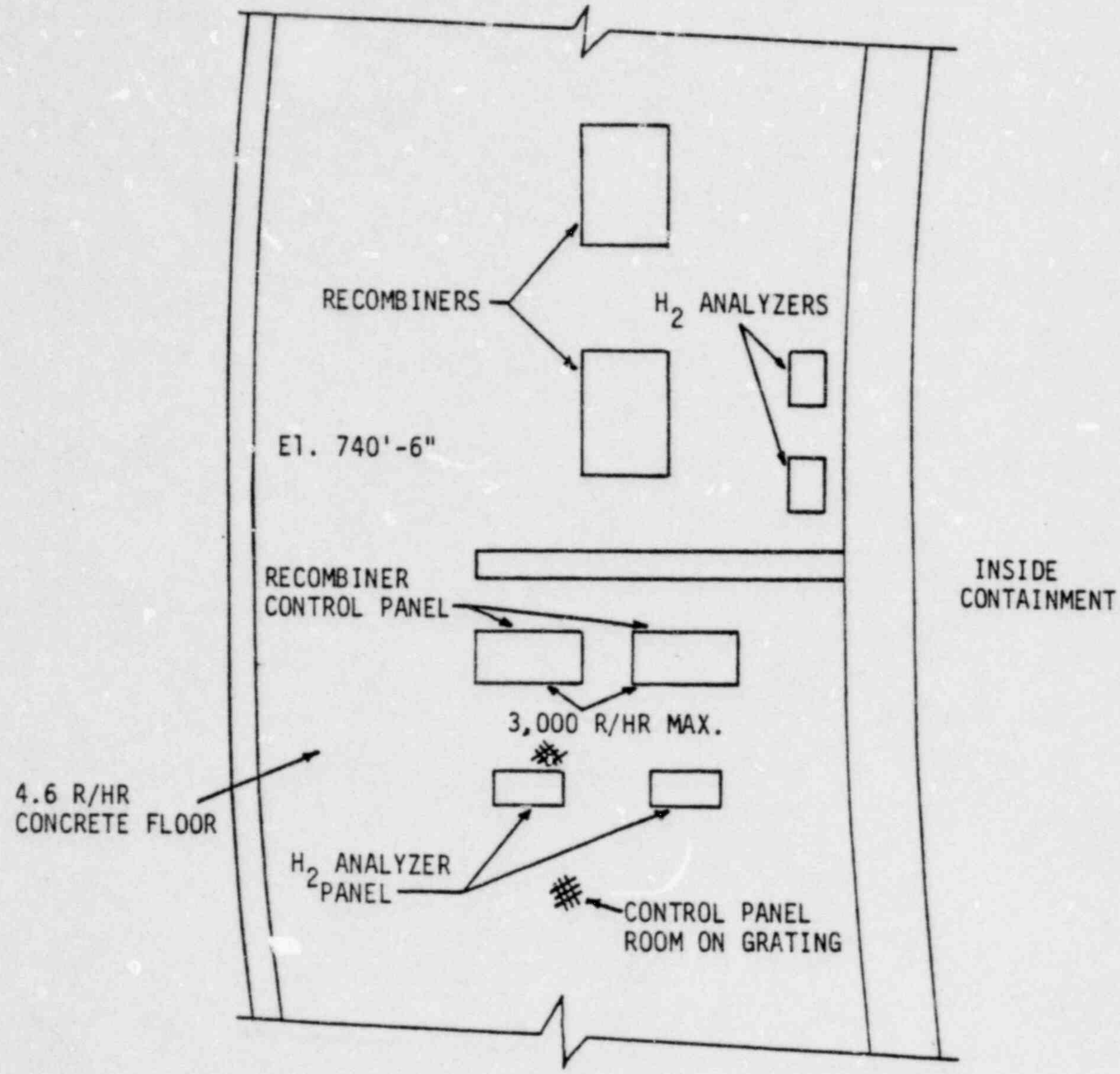


FIGURE 4-11 SIMPLIFIED FLOW DIAGRAM OF HYDROGEN RECOMBINER SYSTEM



NOTE: Radiation level shown is at control panel area and one hour after accident.

FIGURE 4-12 RECOMBINER ROOM LAYOUT

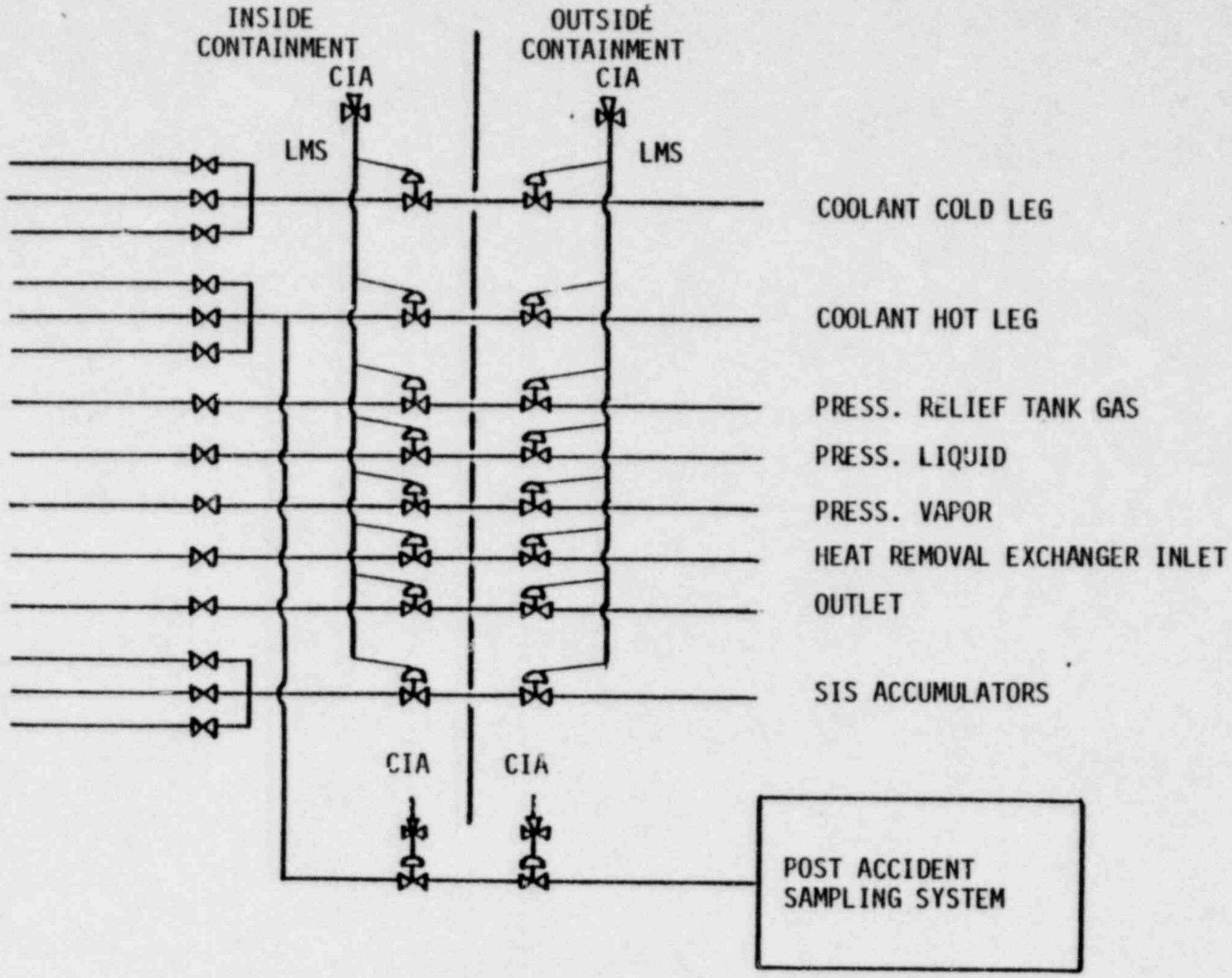
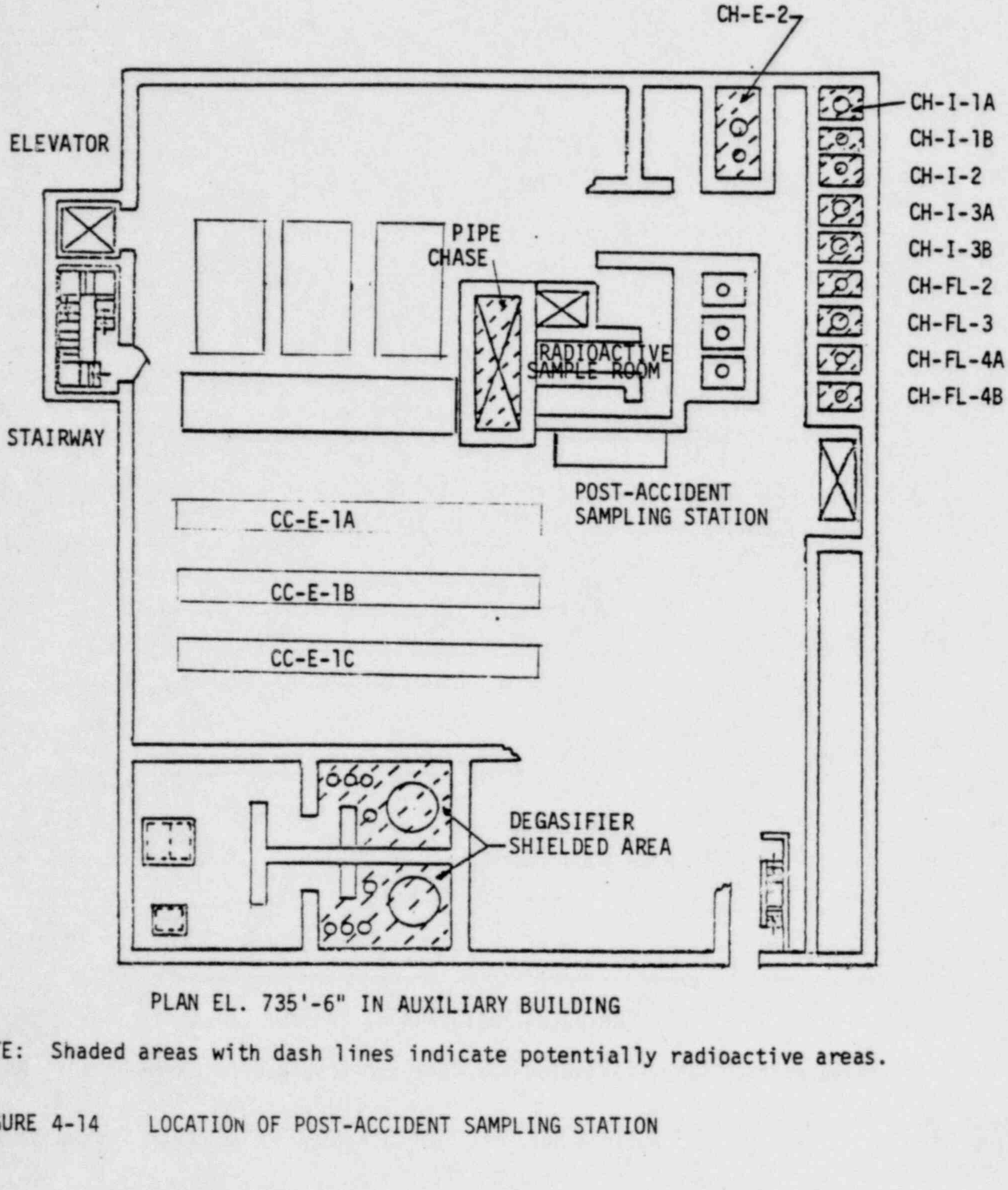


FIGURE 4-13 EXISTING REACTOR BUILDING SAMPLING SYSTEM AND PROPOSED POST-ACCIDENT SAMPLING SYSTEM



PLAN EL. 735'-6" IN AUXILIARY BUILDING

NOTE: Shaded areas with dash lines indicate potentially radioactive areas.

FIGURE 4-14 LOCATION OF POST-ACCIDENT SAMPLING STATION

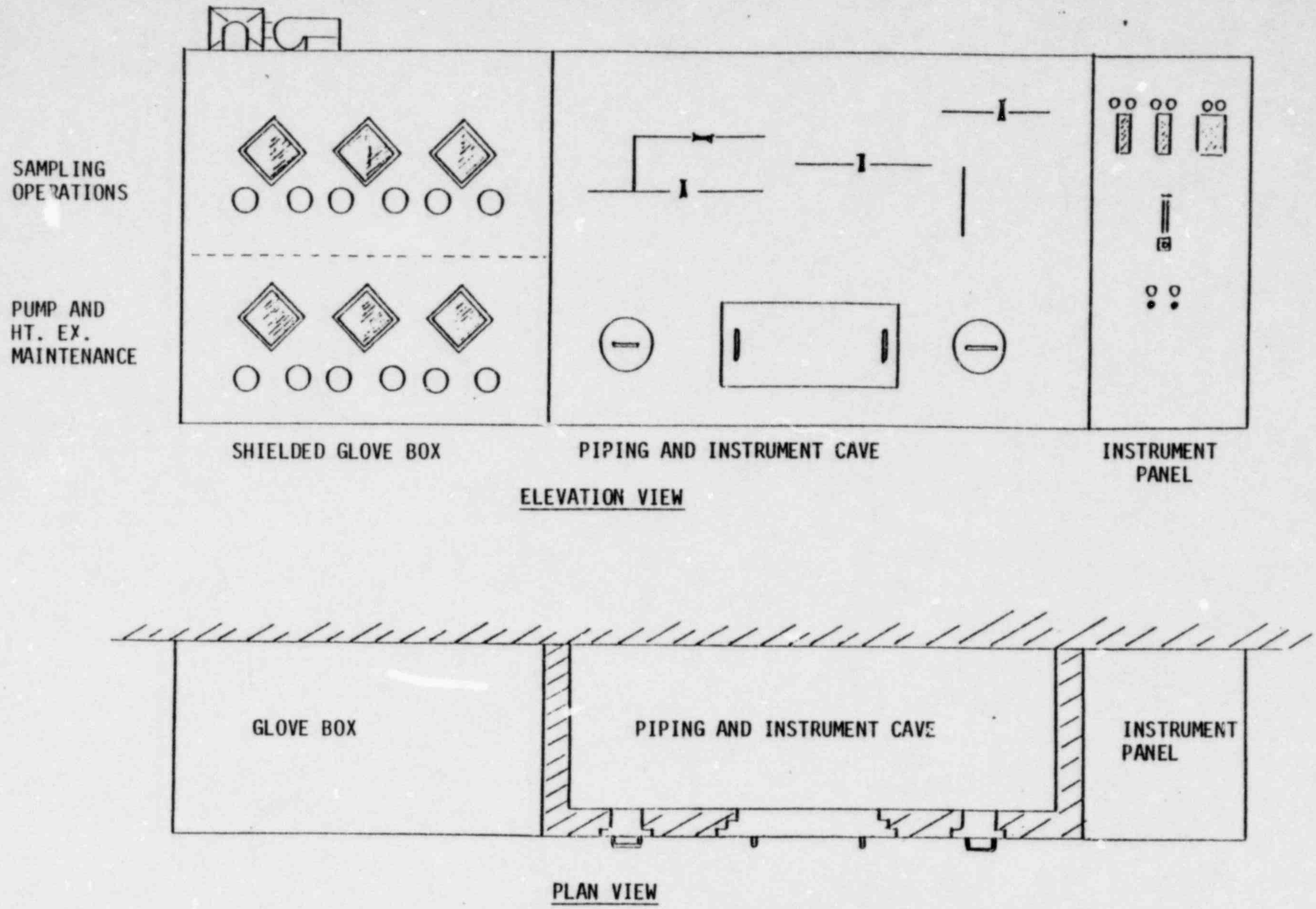
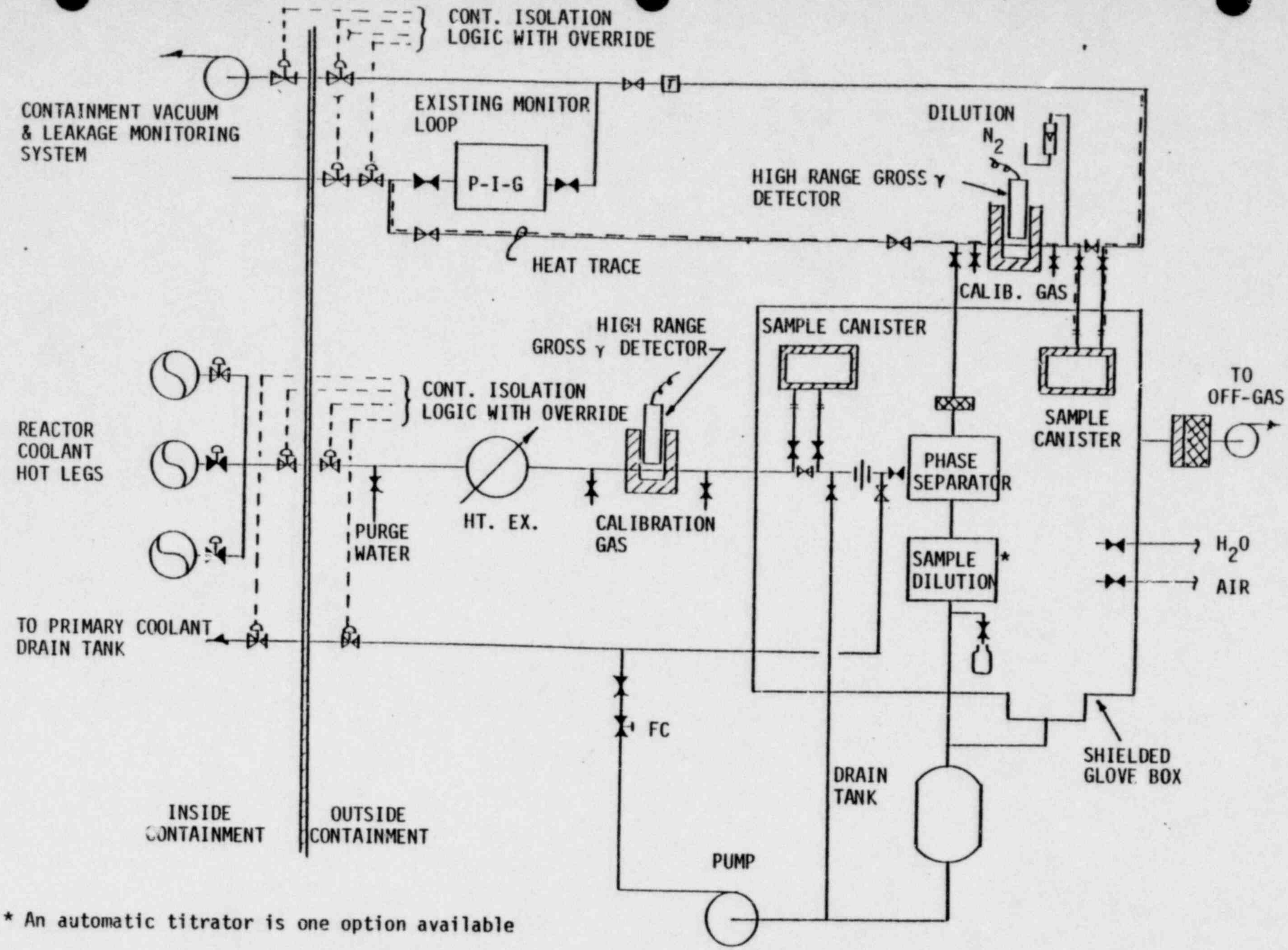


FIGURE 4-15 ARRANGEMENT DRAWING FOR POST-ACCIDENT SAMPLING SYSTEM



* An automatic titrator is one option available

FIGURE 4-16 FLOW DIAGRAM OF POST-ACCIDENT SAMPLING SYSTEM

TABLE 4-1 RADIATION LEVELS IN CHS WITH USE OF LETDOWN LINE

AREA	DESCRIPTION OF EQUIPMENT	EXISTING SHIELDING AND THICKNESS	RADIATION LEVEL VS. TIME (HRS)[1]			OCCUPANCY REQUIREMENTS	RESOLUTION
			Zero	10 hrs	24 hrs		
Floor above Penetration Room (Area Contiguous to Containment)	Charging and Letdown Lines	2 ft. thick concrete floor	32 R/Hr	5.2 R/Hr	2.4 R/Hr	Unlikely	Administrative Control
Auxiliary Bldg.	3-inch Pipe Section in Pipe Chase (Letdown or Charging Line)	2 ft. thick concrete shield	22 R/Hr	3.6 R/Hr	1.7 R/Hr	Possibly	Administrative Control
Auxiliary Bldg.	Outside of Volume Control Tank (CH-TK-2) Cubicle	3.5 ft. thick concrete shield wall	41 R/Hr	3.4 R/Hr	1.2 R/Hr	Possibly	Administrative Control
Auxiliary Bldg.	Outside of Charging Pump (CH-P-1A, 1B, & 1C) Cubicle	2 ft. thick concrete floor and wall	3.9 R/Hr	0.67 R/Hr	0.33 R/Hr	Possibly	Administrative Control

NOTE: 1. Hours elapsed following time 0 of the accident or event.
Radiation levels in contact with shielding.

TABLE 4-2 RADIATION LEVELS IN SAFETY INJECTION SYSTEM

AREA	DESCRIPTION OF EQUIPMENT	EXISTING SHIELDING AND THICKNESS	RADIATION LEVEL VS. TIME (HRS)[1]			OCCUPANCY REQUIREMENTS	RESOLUTION
			1 hr	10 hrs	24 hrs		
West Safeguards	LHSI Pump Cubicle-directly above	2 ft. thick floor	2.4 R/Hr	0.6 R/Hr	0.3 R/Hr	Unlikely	Administrative Control
	LHSI Pump Cubicle standing aside	2 ft. thick floor and side wall	7.7 mR/Hr	0.9 mR/Hr	0.3 mR/Hr	Unlikely	Administrative Control
	Recir. Spray & LHSI Pump Discharge Lines-Pipe Tunnel Floor	2 ft. thick floor	4.8 R/Hr	1.2 R/Hr	0.6 R/Hr	Unlikely	Administrative Control
Safeguards and Contiguous Areas to Containment	Pathway above pipe tunnels	2 ft. thick floor	4.6 R/Hr	1.0 R/Hr	0.4 R/Hr	Unlikely	Administrative Control
Auxiliary Bldg.	Pathway above pipe vault	2 ft. thick floor	21.8 R/Hr	4.8 R/Hr	2.4 R/Hr	Possibly	Administrative Control
Auxiliary Bldg.	Charging Pump Cubicle directly above	2 ft. thick floor	410 mR/Hr	96 mR/Hr	47 mR/Hr	Unlikely	Administrative Control
	Charging Pump Cubicle standing aside	2 ft. thick floor and side wall	1.2 mR/Hr	0.1 mR/Hr	0.04 mR/Hr	Possibly	Administrative Control
	Outside Boron Injection Tank	2 ft. thick shield wall	810 R/Hr	180 R/Hr	87 R/Hr	Unlikely	Administrative Control
	30 Feet from Boron Injection Tank at South Wall of Charging Pump Cubicle	2 ft. thick Boron Injection Tank Shield wall	15.5 R/Hr	3.7 R/Hr	1.8 R/Hr	Likely	Administrative Control

NOTE: 1. Hours elapsed following time 0 of the accident or event.
Radiation levels in contact with shielding

TABLE 4-3

RADIATION LEVELS IN OUTSIDE RECIRCULATION SPRAY PUMP SYSTEM

AREA	DESCRIPTION OF EQUIPMENT	EXISTING SHIELDING AND THICKNESS	RADIATION LEVEL VS TIME (HRS)[1] 1 hr	10 hrs	24 hrs	OCCUPANCY REQUIREMENTS	.RESOLUTION
West Safeguards	Outside Recirc Spray Pump Cubicle Directly Above	2 feet thick concrete floor	2.4 R/hr	0.6 R/hr	0.3 R/hr	Unlikely	Administrative Control
	Outside Recirc Spray Pump Cubicle Standing Aside	2 feet thick concrete floor and side wall	7.7 mR/hr	0.9 mR/hr	0.3 mR/hr	Unlikely	Administrative Control
	Recirc Spray & LHSI Pump Discharge Lines - Pipe Tunnel Floor	2 feet thick concrete floor	4.8 R/hr	1.2 R/hr	0.6 R/hr	Unlikely	Administrative Control

NOTE: 1. Hours elapsed following time 0 of the accident or event.
- Radiation levels in contact with shielding.

TABLE 4-4: SUMMARY, RADIATION LEVEL OUTSIDE CONTAINMENT FOR CONTAINMENT VACUUM AND LEAKAGE MONITORING SYSTEM,

AREA	DESCRIPTION OF EQUIPMENT	EXISTING SHIELDING AND THICKNESS	RADIATION LEVEL VS TIME (HRS)[1]			OCCUPANCY REQUIREMENTS	.RESOLUTION
			1 hr	10 hrs	24 hrs		
Containment Atmosphere Monitoring System Outside Containment	3/8" pipe, valves and instruments 4 sets 3 ft. from pipe	2 feet thick concrete wall	6.8 mR/HR	1.3 mR/HR	0.45 mR/HR	Infrequent	Administrative Control
Ref. Volume Subsystem	Not safety related						N/A
Containment Atmosphere Monitoring System	Existing System may not be adequate						Investigate range capability
Cont. Depress. subsystem	Not safety related						N/A
Cont. Blowdown subsystem	Not safety related						N/A

NOTE: [1] Hours elapsed following time 0 of the accident or event and all radiation levels in R/hr.
- Radiation levels in contact with shielding

TABLE 4-5: RADIATION LEVELS IN SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM

AREA	DESCRIPTION OF EQUIPMENT	EXISTING SHIELDING AND THICKNESS	RADIATION LEVEL VS TIME (HRS)[1]			OCCUPANCY REQUIREMENTS	RESOLUTION
			Zero				
Containment Contiguous Areas and Safe-guards area	Main Duct (at 10 ft distance)	None	200 mR/HR			Unlikely	Administrative Control
	Main Duct (at 20 ft distance)	None	74 mR/HR			Unlikely	Administrative Control
Auxiliary Bldg.	Main Duc (at 10 f distance)	None	200 mR/HR			Infrequent	Administrative Control
	Main Duct (at 20 ft distance)	None	74 mR/HR			Infrequent	Administrative Control
Auxiliary Bldg.	Filter Bank	2 ft thick concrete wall		290* mR/HR		Infrequent	Entrance of the shielded enclosure should be controlled.

NOTE: [1] Hours elapsed following time 0 of the accident or event.
* Assume all iodines filtered out for 30 minutes after time zero.

TABLE 4-6

HYDROGEN RECOMBINER PIPING

PIPE SIZE	PIPE NUMBER	APPROXIMATE LENGTH IN RECOMBINER ROOM
3/8"	HY-14-N8	284"
3/8"	HY-13-N8	368"
3/8"	HY-15-N8	368"
3/8"	HY-17-N8	311"
3/8"	HY-5-N8	181"
3/8"	HY-3-N8	202"
3/8"	HY-4-N8	181"
3.8"	HY-7-N8	74"
2"	HY-35-151-Q2	372"
2"	HY-36-151-Q2	240"

TABLE 4-7 RADIATION LEVELS IN HYDROGEN RECOMBINER CONTROL AREA

AREA	DESCRIPTION OF EQUIPMENT	EXISTING SHIELDING AND THICKNESS	RADIATION LEVEL VS TIME (HRS)[1]			OCCUPANCY REQUIREMENTS	.RESOLUTION
			1 hr	10 hrs	24 hrs		
Hydrogen Recombiner Control Panel	Control panel room on grating and directly exposed to LHSI lines	1.5 ft biological concrete shield between control panel and recombiner	3000* R/HR	1400* R/HR	1000 R/HR	1 hr. max residency per person.	Relocate control panels to concrete floor area or shield control panel room from LHSI lines

NOTE: [1] Hours elapsed following time 0 of the accident or event - Radiation levels near contact with shielding

* At this time H_2 recombiner is not operating, no contribution from recombiner units is considered.

TABLE 4-8 SAMPLING REQUIREMENTS FROM NUREG 0578

LIQUID SAMPLE

Press./Unpress. reactor coolant samples within 1 hour of incident

GAS SAMPLE

Containment atmosphere sample within 1 hour of incident

5.0 REFERENCES

5.1 "Engineering Compendium on Radiation Shielding," Volume II Shielding Material, edited by R. G. Jaeger and others, Springer-Verlag, New York, 1975.

5.2 "Nuclear Reactor Engineering," by S. Glasstone and A. Sesouske, D. Van Nostrand Company, June 1963.

5.3 "Nuclear Engineering Handbook," edited by Harold Etherington, McGraw Hill Book Company, 1958.

5.4 Beaver Valley Unit 1, Final Safety Analysis Report.

5.5 Duquesne Letter on "Plant Shielding Review and Post-Accident Sampling," dated December 19, 1979.

5.6 NUREG-0578 (TMI-2 Lessons Learned Task Force Report).

5.7 Response to NUREG-0578 Item 2.1.6.a Part II, M.O. Sanford, December 1979, METN-2005 Westinghouse Owners Group.

5.8 Operating Manual Figure Number 6-1 through 46-1.

5.9 Operating Manuals 6 through 46

5.10 10 CFR 50, Appendix A, Criterion 19

5.11 TID 14844, Calculation of Distance Factors for Power and Test Reactor Sites, March 1962.