Met-Ed / GPU

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Writer's Direct Dial Number

November 13, 1979
GQL 1416

TMI Support
Attn: Richard Vollmer
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
License No. DPR-73
Docket No. 50-320
Reactor Containment Building Atmosphere Cleanup

The ultimate safe condition for the reactor and reactor containment building requires decontamination and removal of the reactor fuel. To accomplish this it is necessary that the existing reactor building atmosphere, containing Kr-85, be removed. Over the past few months we have studied the various alternatives for accomplishing removal of the Kr-85, including a comprehensive safety and environmental assessment. The results of these studies are contained in the attached report.

Of the four options examined; charcoal adsorption and storage, gas compression and storage, cryogenic processing and storage, and atmospheric purge, we strongly recommend that atmospheric purge be the means for accomplishing the disposal of Kr-85. Our studies show that the purge operation, using controlled venting through the plant exhaust stack and meteorological feedback, can be done with no significant hazard or radiation exposure either to the general population or the site. The purge can meet all technical specifications and NRC radiological criteria. A significant advantage to the purge operation is that it can be accomplished in a relatively short time compared to the two to three years required for alternatives and this short time scale, in itself, is a significant safety advantage. The time to implement alternatives to purge are such that we cannot guarantee full containment integrity and would, in fact, expect general population doses to exceed those minimum levels resulting from purge.

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While details of the proposed purging operation are contained in the report, the salient features include:

1. The controlled purge of the approximately 44,000 curies of Kr-85 is accomplished from an elevated stack with significant dilution before reaching the site boundary. In addition, purge will be permitted only under conditions of favorable meteorology. Comprehensive evaluations indicate that the maximum off-site dose resulting from total release will be less than 5 mrem. Environmental monitoring will be employed to detect the off-site ground level presence of any Kr-85 above background.

2. Controlled purging does not require storage of Kr-85 for prolonged periods of time. It accordingly is a permanent solution and eliminates all risks arising from accidents with the three alternative methods.

3. Purging requires only slight modifications to existing equipment and, hence, is an operationally desirable and safe approach.

We are cognizant of the concern on the part of some members of the surrounding communities about the venting of the Kr-85. We are convinced, however, that this is the most prudent and safest approach, with negligible radiological impact of handling the containment Kr-85. The Company will do whatever it can to provide sufficient information to the public to assure them they will be aware of the timing of releases and the results of the monitoring of both on-site and off-site radiation levels.

We will be technically ready to proceed with containment purging in approximately one month. We are requesting your approval to proceed with purging, subject to verification by NRC personnel on site of equipment and procedures, and are ready to meet with you to review the attachment or any other questions which you might have.

Very truly yours,

R. C. Arnold
Senior Vice President
Metropolitan Edison

RCA:LWH:tas

Attachment

THREE MILE ISLAND UNIT 2

REACTOR BUILDING PURGE PROGRAM

SAFETY ANALYSIS AND ENVIRONMENTAL REPORT

NOVEMBER 12, 1979

THREE MILE ISLAND UNIT 2
REACTOR BUILDING PURGE PROGRAM
SAFETY ANALYSIS AND ENVIRONMENTAL REPORT

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1.0 INTRODUCTION

1.1 Need For Reactor Building Atmosphere Cleanup

The unknown Three Mile Island Unit 2 core configuration poses a small but incalculable risk. Although much analysis has been completed that tends to bound the limits of uncertainty with regard to the core configuration, this uncertainty can best be dealt with by timely entry into the reactor building and ultimate removal of the nuclear fuel from the reactor pressure vessel.

In order to allow entry into the reactor building without significantly complicating the entry program and restricting the effectiveness of operations toward ultimate fuel removal, the reactor building atmosphere must first be cleansed of radioactive materials. Leaving the airborne materials in the atmosphere while other steps toward fuel removal proceed represents substantial risk of ultimate uncontrolled release of these materials to the environment and unacceptable increase in operations personnel exposure.

Airborne radioactivity within the reactor building has reduced considerably since the accident due to decay of the short-lived radioactive fission products such as Xenon and Iodine. The principal remaining airborne fission product is Krypton-85 which has a 10.76 year half-life. Due to this long half-life, additional delays in cleaning up the reactor building atmosphere will not materially reduce its radioactive concentration.

Several alternatives have been studied for removing the approximately 45,000 Curies of Kr-85 estimated to exist in the building. There alternates include charcoal adsorption and long term storage, gas compression and long term storage, cryogenic processing and long term storage, and atmospheric dispersion of Krypton-85 by controlled purging of the reactor building atmosphere. The optimum choice from an environmental impact standpoint when potential accidents are considered is atmospheric dispersion through controlled purging of the reactor building atmosphere.

1.2 Purpose of this Report

The purpose of this report is to document that atmospheric dispersion through controlled purging of the reactor building atmosphere can be accomplished within all applicable safety limits and radiation protection standards and that purge represents the optimum solution for reactor building atmosphere cleanup, considering the health and safety of the population around the Three Mile Island Unit 2 plant. This report presents a description of the proposed program for controlled purging, the safety analysis for this purge program, the environmental impact of this proposed purge program, and the results of studies of the less desirable alternatives for reactor building atmosphere clean-up.

1.3 Organization of Report

This report is organized in the format of a combined safety analysis and environmental report. Following this introduction, the assessment of reactor building airborne activity is presented and then the analysis of the purge program. The purge program analysis includes a summary of the purge method, the design basis for the hydrogen control system and its modifications, the design evaluation, operating description, and accident analysis for the system.

The Safety Limits for radioactive gaseous releases are discussed including some perspective on radiation exposure, and limiting conditions for the purge program.

The effects of purge operation on the environment and on operational exposure are presented, including the environmental effects of a postulated purge accident. Finally, the results of the studies on the alternatives to controlled purge are presented including the environmental effects of each alternate.

1.4 Conclusion

The studies concerning disposal of the Krypton-85 from the containment vessel result in the following conclusions:

1. There are only four potentially feasible methods for disposing of the Krypton: purging to the atmosphere, charcoal adsorption and storage, storage as a compressed gas, and cryogenic separation and storage. Of these four methods, three, charcoal adsorption, gas storage and cryogenic separation require a long schedule to implement, are of high complexity, but theoretically can provide a zero or near zero offsite dose. The purge method can be implemented very quickly, is simple, but does yield a small finite offsite dose to the general population.

2. The examination of radiation doses to the general population in the event of accidents, for each alternative, shows just the reverse of the normal dose comparison, i.e., purge has an extremely small general population accident dose, whereas the other three have relatively large general population accident doses. Of the alternatives studied, only purge to the atmosphere provides a permanent solution to the Krypton-85 problem. The other three options require treatment and storage in systems which have the potential for accidental release of Krypton-85 during processing and especially during the long storage time required.

3. The long schedule required for the storage options is considered a significant safety disadvantage. There is no assurance that containment integrity can be maintained for the 2-3 years necessary to implement storage. As shown in Section 8.6.1, if the reactor building air cooling capability is lost, the reactor building pressure could rise to 1 to 2 psig. The uncontrolled leakage of Krypton 85, if the equivalent of a 1/2 inch diameter hole is present in the

containment boundary, could result in an off-site beta dose in the range of 15 to 80 mrem in a single day or 60 to 270 mrem if the leakage occurred over a 30 day period.

4. Krypton-85 disposal is an essential prerequisite to performing disposal work within the containment leading toward cleanup of the containment structure. The delay in initiating such cleanup, which would be required by the storage options, can, in itself, be a significant safety hazard and cause large increases in radiation dose to the work force. This increased dose would arise because of additional complexity in decontamination, but at this time cannot be quantified.

5. Purge of the Krypton-85 to the atmosphere can be performed under well-controlled conditions, and such purging can meet all technical specifications and Regulatory guidance. The estimated dose to the general population, as well as dose to the onsite staff, is extremely low or insignificant.

6. Table 1-1 summarizes the radiation effects of each of the alternatives for reactor building atmosphere clean-up. The expected dose/exposure shown in this table for each alternate uses expected meteorology based on historical data. The system upset dose/exposure analysis uses conservative, 5% probable extreme meteorology as specified in Regulatory Guides for accident analysis. The coincidence of the postulated accident conditions and the extreme meteorology are highly unlikely as stated in Section 7.1.

It is recommended that cleanup of the containment atmosphere proceed through purging as the safest, and most effective permanent solution to the Krypton-85 problem.

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TABLE 1-1

Dose/Exposure Comparison For
Reactor Building Atmosphere Clean-Up Alternatives

	EXPECTED DOSE/EXPOSURE				SYSTEM UPSET/ACCIDENT DOSE/EXPOSURE			
	OFF-SITE SKIN DOSE (MREM)*		OFF-SITE WHOLE BODY DOSE*	ON-SITE WHOLE BODY DOSE	OFF-SITE SKIN DOSE (MREM)**		OFF-SITE WHOLE BODY**	OFF-SITE WHOLE BODY DOSE***
	BETA	GAMMA	PERSON-REM	PERSON-REM	BETA	GAMMA	DOSE (MREM)	PERSON-REM
PURGE	5	<0.1	0.75	<1	61	0.9	0.73	0.07
CHARCOAL	0	0	0	23.	104	1.4	1.24	0.13
GAS COMPRESSION	0	0	0	58.	1730	24	20.7	2.1
CRYOGENIC	<1	<0.1	0	570.	4090	56	49.0	5.2

* CALCULATED IN ACCORDANCE WITH RG 1.109 USING THI-2 SITE HISTORICAL METEOROLOGICAL DATA

** CALCULATED IN ACCORDANCE WITH RG 1.24 and RG 1.145 ACCIDENT X/Q VALUES FROM HISTORICAL MET. DATA

*** CALCULATED IN ACCORDANCE WITH RG 1.109 and 5% PROBABLE POPULATION DOSE GIVEN THAT THE ACCIDENT OCCURS

2.0 REACTOR BUILDING AIRBORNE ACTIVITY

Three types of samples are being collected periodically from the reactor building atmosphere to determine the nature of airborne contaminants present. The samples are for noble gas, particulate, and radio-iodine activity.

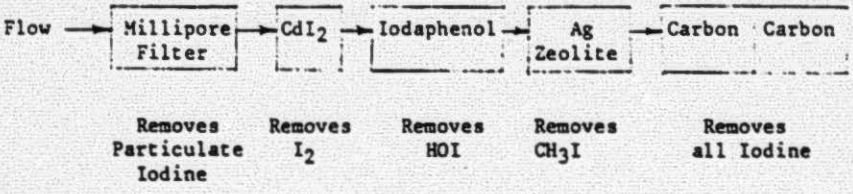
2.1 Method

In order to determine the activity content of the atmosphere in the reactor building, air samples utilizing the installed HPR 227 sample pumps and cabinet have been taken. The sample procedure allows use of a normal sample path or an alternate sample path which connects penetration R-562B to the suction of the HPR 227 sample pump via valves AH-V147 and AH-V148. Another sample path will be available in November through penetration R-626 (at elevation 358'), when the inner flange of that penetration is cut to allow camera and radiation monitor placement inside the reactor building.

To determine noble gas activity, a reactor building air sample is collected in a 6 cc glass bulb and analyzed by gamma spectroscopy. Isotopic identification is made on the basis of the discrete energy levels at which gamma rays are absorbed in a GeLi detector. The spectrum containing the various gamma peaks is then screened and compared against a library of known peaks vs. isotope to make final identification. The intensity of each peak at its discrete energy level is a function of the concentration of the respective radio nuclide. The process is commonly referred to as a "gamma scan."

To determine the particulate activity concentration in the atmosphere, a sample of the reactor building air is pumped through a 100 millipore filter. Particulate activity is removed from the air by the filter and the filter is then analyzed using gamma spectroscopy as described above.

To determine the concentrations of the different types of iodine in the atmosphere, a sample of the reactor building air is pumped through a series of filters as shown below. Separation of the different forms of iodine is accomplished based on the relative affinity of each iodine species for a specific filter media. Each filter is then analyzed using gamma spectroscopy as described above.



2.2 Source Term Evaluation

The sampling and analysis techniques described in Section 2.1 provide for the determination of noble gas activity, particulate activity and iodine in the reactor building atmosphere. The results of these samples to date are included in Table 2-1. It should be noted that the sample results, especially for the samples taken prior to June 21, vary widely. Whereas the earlier samples were drawn under less controlled conditions, the current procedure requires extensive documentation to ensure accurate sample times are used and proper volumes are drawn. In addition, retained with each sample result is the documented condition of the sample lineup and reactor building ventilation system while the sample was drawn.

From Table 2-1, it can be seen that the dominant isotope inside the reactor building at this time is Kr-85 at $\sim 0.78 \mu Ci/ml$. Particulate levels, primarily Cs-137, are on the order of $1 \times 10^{-6} \mu Ci/ml$. The radio iodine levels inside containment are rapidly dropping due to decay. Latest results indicate Iodine to be below minimum detectable activity (MDA) levels of $10^{-9} \mu Ci/ml$.

In order to determine a best estimate of the airborne radioactivity inventory in the reactor building, the results of all gas samples were reviewed and correlated. The results of this review are given in Table 2-2.

When these activity levels for Kr-85, Cs-137, and Iodine 131 are evaluated against technical specification limits for allowable instantaneous and quarterly average allowable gaseous effluent release rates, the Kr-85 concentration is shown to control allowable release rates.

2.3 Reactor Building Source Term Results

Results of reactor building air samples taken to date are shown in Table 2-1. The best estimate of reactor building airborne radioisotope activity projected to Nov. 1, 1979 is shown in Table 2-2. This estimate includes consideration of all samples taken to date and projects reduced concentrations to Nov. 1, 1979 of the shorter lived radio-isotopes.

2.4 Reactor Building Source Term Conclusions

The airborne activity sampling process is sufficiently defined and recent results are sufficiently consistent that the results given in Table 2-2 represent accurate values for airborne activity levels of the isotopes of concern namely noble gas Kr-85, particulate Cs-137 and Iodine-131. The results are sufficiently valid to serve as a basis for evaluating alternative reactor building atmosphere clean-up options.

Also, from these activity levels, it is clear that efforts should be taken to clean-up the reactor building atmosphere in order to

reduce the total exposure during manned entry of the containment resulting from the Kr-85 present.

Finally, from the low levels of particulate (Cs-137) and Iodine, it is concluded that reactor building recirculation using the reactor building purification filtration system is not necessary to achieve further reduction in Iodine and particulate levels. Air discharged from the reactor building will be filtered through the in-line particulate and charcoal filters of the hydrogen control system during purge.

TABLE 2-1

Gas/Atmosphere Sampling* of TMI 2 Using HDR 222

Sample Type	Gas	Gas	Gas	Gas	Gas	Gas	Gas	Gas	Gas	Part.	Gas	Part.	Spectra	Spectra	Part.	Gas	Gas	Gas	Gas	Part.	Gas	Part.	Same Sample*** Gas	Gas	Gas
Sample No.							7058	7056	11552	11556	12216	12290	12213	11355	14329	14371	14379	15160	18618	19658	19709	19693	20161	20164	20698
Ref.	1	1	1	1	1	1	1	1	5	5	5	5	2	2	1	1	1	1							
Time (days)	t=3	t=6	t=27	t=44	t=47	t=59	t=63	t=64	t=85			t=90		t=103		t=117		t=127							
Date	3/31	4/3	4/4	4/11	4/14	4/27	4/30	5/1	6/21	6/21	6/21	6/26	6/26	7/9	7/9	7/23	7/23	8/2	9/8	9/8	9/18		9/26	9/26	9/26
Nuclide	$T_{1/2}$**																								
Co-58	71.3									(5.5-6	(8.6-12		<9.2-4												
Co-60	5.26 yrs									(8.6 6	(1.7-11		<7.9-4			6.0 10									
Zr-95	10.73 yrs			6.3-1		6.6-1	5.5-1	9.6-2	9.8-1	7.9-1		1.0**						9.9 1	1.0-1		9.6 1		1.0 1	1.1-1	8.6-1
Zr-95m	11.98			6.9-2	9.7-3		2.2-1	5.3-2	3.2-1	8.5-2		8.6 2						9.5 1							
Zr-97m	2.19	1.6+1	5.9+0			3.1-1																			
Zr-133	5.29	6.8+2	1.7+1	9.9+0	5.5-3	8.1-1	5.5-1	5.6-2	6.8-1	2.6 2		1.5 2													
Zr-135	0.382		1.6-1																						
I-131	8.04	6.6-2	1.1-1	2.8-3	3.3-5	1.3-3	1.6-3	1.7-3	1.0-3	1.2 6	1.9 9	1.2-6	1.1-3	6.1-5	1.6-5										
Co-134	2.07 yrs									(8.1 6	2.0 9		1.0 5			9.6 10					1.6 7		6.9 11		
Co-135	3 x 10^5 yrs																								
Co-136	13.00						2.6-5																		
Co-137	30.13 yrs			6.0-2		1.6-5	6.8-5			(1.1-5	9.5 9		1.6-2			1.0 10	2.0 5	7.1-5	6.5-5		1.5 6	1.5-5	1.0 10		1.2-5
Ba-137m	2.55 min.												(1.5-1												
Ba-140	12.79			3.1-2									5.0 1												
La-140	1.68																								

* Species

I$_2$													6.0 6	1.6 6											
HOI													3.5 6	7.1 6											
ORGANIC													5.7-5	1.7-5											

COMPOSITION

N_2								86.9%								78.1%	79.1%	85.3%		80.9%				86.5%
O_2								10.5%								21.1%	20.9%	14.1%		18.3%				12.8%
H_2								0.6%								0.8%	0.0%	0.6%		0.9%				0.7%

References:

1. From J. A. Daniel to G. Bond, 5 June 1979, "Containment Air Samples," TMI Waste Management Group.
2. From P. C. Veilleux to Clay Montgomery, 11 July 1979, "Reactor Containment Spectco Samples."
3. Sample Gamma Analysis Summary Sheet.
4. Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fusion Products by A. Tobias, June 1978, Research Dept. Berkley Nuclear Laboratories.
5. Table of Isotopes, 6th Ed. by C. M. Lederer, J. M. Hollander & I. Perlman, John Wiley & Sons, Inc., 1967.

* Units are/Ml/cc for all concentrations.
** Given in days unless otherwise noted. Ref. 4 & 5.
*** Same sample but different analysis of sample.

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TABLE 2-2

Best Estimate of RB Airborne Activity
November 1, 1979

Nuclide	Concentration (μCi/cc)	Total Inventory (Ci)
Kr-85	0.78	$4.4 \times 10^{+4}$
Xe-131 m	$<2 \times 10^{-5}$	<1.14
Xe-133	$<1 \times 10^{-5}$	$<5.7 \times 10^{-1}$
I-131	$<1 \times 10^{-9}$	$<5.7 \times 10^{-5}$
Cs-134	$<1 \times 10^{-5}$	$<5.7 \times 10^{-1}$
Cs-135	$<1 \times 10^{-5}$	$<5.7 \times 10^{-1}$
Cs-136	$<1 \times 10^{-5}$	$<5.7 \times 10^{-1}$
Cs-137	$<1 \times 10^{-5}$	$<5.7 \times 10^{-1}$

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3.0 PURGE PROGRAM ANALYSIS

3.1 Purge Program Summary

The reactor building purge program will purge the reactor building atmosphere using the hydrogen control subsystem of the reactor building ventilation and purge system. The purged atmosphere containing radioactive gases will be released from the plant vent stack (160 ft. above grade level) at times when wind and other meteorological conditions are most favorable for atmospheric dispersion.

The hydrogen control subsystem is designed for use as a back-up for the hydrogen recombiner. The reactor building atmosphere is drawn through a filter train by the hydrogen control exhaust fan before being discharged to the plant vent stack. (A modification is being made to reroute flow from the inlet of the supplementary vent filters to the plant vent stack to obtain an elevated release). The filter train consists of a prefilter, HEPA filter, an activated carbon filter and another HEPA filter. The purge flow rate is controlled by a throttle valve, AH-V36. (Valve AH-V36 is being modified to provide fine motion flow control). The air may be discharged at a rate of up to 1000 cfm. (A modification is being made to increase the capacity from 150 cfm to 1000 cfm.) The flow rate, temperature and radiation level are monitored during discharge.

The replacement air to the reactor building will be supplied through valves AH-V7 and AH-V3B such that the RB atmosphere remains at slightly negative or atmopheric pressure throughout the purge process. Figure 3-1 gives a schematic of the purge system pumps, pipes and valve configuration.

The maximum purge discharge flow rate will be determined using the Technical Specification instantaneous release rate for gross gaseous activity given by paragraph 4.2 and the latest assessment of Kr-85 level from the revised RB air sampling program. This release rate is expected to be in the range of 50 to 100 cfm initially. As the Kr-85 content drops within the reactor building, due to purging, the maximum allowable purge rate will increase until the 1000 cfm limit of system capability is reached. The actual purge rate during any time interval will be based on actual wind conditions such that for unfavorable meteorology, the maximum allowable release rate will be reduced to minimize dose accumulation at off-site locations. At the start of each purge period, wind data will be recorded and predicted incremental dose at the boundary will be calculated and compared against an administrative limit of 0.1 mrem/hour beta skin dose. If the dose rate is calculated to exceed this limit, the release rate will be reduced to stay within the limit. If the allowable release rate drops below 20 cfm, no release will be allowed during the period until meteorological conditions improve.

Accumulated off-site doses will be calculated throughout the purge process using actual meteorological and release data to assure that 10CFR50 Appendix I limits are not violated and that projections of future purging will not cause the limit to be exceeded.

Plant vent stack monitoring will provide continuous feedback and a complete record of actual stack releases to track total curie activity released and to compare actual releases with expected releases based on reactor building activity.

In addition, reactor building activity will continue to be monitored to determine the Kr-85 activity for establishing future purge release rates in conformance with Technical Specification limits and to confirm that MPC limits or lower have been met at the end of the purge operation.

3.2 System Design Basis

The purge program uses the installed hydrogen control (atmospheric purge) system with modifications to increase the fan capacity to 1000 cfm, to provide variable purge flow control with interlocks for rapid isolation on equipment failure or high radioactive levels at the fan discharge.

The atmospheric purge operation was originally designed to keep the reactor building hydrogen concentration from reaching the lower flammability limit following a LOCA if the hydrogen recombiner system is not available. The atmospheric purge system is designed to 30 psig and 150° F and seismic Class I conditions. The system meets the requirements of ANSI B31.0 Code for Power Piping, Class 2, and ASME Boiler and Pressure Vessel Code, Section III, Class B.

3.3 System Design

In order to efficiently purge the reactor building of radioactive gases, the flow capacity of this system has been increased to 1000 cfm to match the filter train capacity. Variable system flow capability is added to control the atmospheric purge rate as a function of activity in the reactor building and meteorological conditions. Since the airborne activity within the reactor building will exponentially decrease with purging, it is desirable to increase the maximum flow rate to 1000 cfm during the later stages of purge operation.

The reactor building atmosphere is drawn through a filter train by the hydrogen control exhaust fan before being discharged to the Station vent. The filter train consists of a prefilter, HEPA filter, an activated carbon filter and another HEPA filter. In the original design the hydrogen purge flow rate was controlled by throttle valve AH-V25. The valve must be partially open for fan operation. The fan discharge valve, AH-V36 opens with fan start. For operation with the increased fan capacity, purge flow rate will be controlled by remote control of valve AH-V36 in place of AH-V25 for better control of flow rate over the full range of flow capacity. The air may be discharged at a rate of up to 1000 cfm. Replacement air is supplied through AH-V7. The flow rate, temperature, and radiation

level will be monitored during discharge. The atmospheric purge system is shown diagrammatically in Figure 3-2.

3.3.1 Component Description

Design performance and equipment data are provided in Table 3-1.

Reactor Building Hydrogen Control Exhaust Unit

The reactor building hydrogen control exhaust unit is located in the auxiliary building at an elevation of 328'. The unit is comprised of a bank of filters housed in a steel cabinet and an exhaust fan connected to the housing.

The filter bank (Table 3-1) consists of the following filters listed as they occur in the flow path:

a. Pre-Filter AH-F-36
b. HEPA Filter AH-F-33
c. Activated Carbon Filter AH-F-34
d. HEPA Filter AH-F-35

Access doors are located on top of the housing for easy maintenance. There is a differential pressure switch connected across the filter bank which will initiate an alarm on high differential pressure. Each filter is provided with a differential pressure indicator.

Reactor Building Hydrogen Control Pre-Filter AH-F-36

The pre-filter is a replaceable bag filter designed for rough particle removal. It has a local differential pressure indicator.

Reactor Building Hydrogen Purge Absolute (HEPA) Filters AH-F-33 and AH-F-35

The HEPA filters (Table 3-1) are constructed of a dry fibrous high interception, sub-micron glass fiber which has an efficiency of 99.97% for particles larger than .3 microns. The filters conform to ORNL-NSIC-65. The filters are mounted in a steel frame and have alumimum separators. Each HEPA filter is fitted with a local differential pressure indicator.

Reactor Building Hydrogen Control Activated Carbon Filter AH-F-34

The activated carbon filters are designed to trap and remove gaseous contaminants (iodine) from the airstream.

The carbon filters (Table 3-1) are of activated charcoal impregnated type, and are of water repellant and fire resistant construction. The adsorbent material (MSA 85851) is housed in a stainless steel flat bed type frame. The filters are tested in accordance with ORNL-NSIC-65.

Each carbon filter is fitted with a differential pressure indicator. A sprinkler system is built-in for each carbon filter bank for fire protection. Means for detecting radiation levels and leaks are provided through a flanged rubber sock-port opening at the upstream and downstream face of each filter bank for insertion of radiation monitor probes.

Reactor Building Hydrogen Control Exhaust Fan, AH-E-34

The Reactor Building Hydrogen Control Exhaust Fan will be replaced by a fan manufactured by Buffalo Forge Company, Model No. 4RE, centrifugal type, fabricated housing, direct driven, 1000 cfm capacity at 48 inches of water static pressure at 3550 rpm. This fan is located on the 328' level of the auxiliary building and driven by a Westinghouse, explosion proof induction motor with air cooled bearings rated at 15 horsepower at 3550 rpm.

If the Reactor Building Hydrogen Control Exhaust Fan ON/OFF switch on Panel No. 25 is in the ON position, the motor can be powered from 2-11EB. If the switch is in OFF the motor can be powered from 2-21EB.

There are two red lights to indicate which of the two sources are lined up to power the fan motor and its associated valves (AH-V-25, 36, and 52). Panel No. 25 has two PULL-TO-LOCK-STOP-NORMAL-START switches for each of the two power supplies. Additionally the motor has a local START/STOP pushbutton. Motor run indication is available on Panel 25 and locally. The fan will stop with a Fire Protection System signal or when its supply valve AH-V25 is fully closed. Fan start will automatically open its discharge valve.

Reactor Building Pressure Sensing Line Penetration Isolation Valves AH-V5 and AH-V6

A solenoid operated 1" stainless steel valve with a design pressure of 100 psig and a design temperature of 300°F is provided on both sides of reactor building penetration R-562 in the pressure sensing line. AH-V5 and V6 are located on the 305' Level of the reactor and auxiliary building. These valves close with an ES signal. Control is provided locally on Panel 25. Indication is available locally and on Panels 13, 15 and 25.

Reactor Building Pressurization Valve AH-V7

An air cylinder operated, 10" carbon steel butterfly valve with an ANSI Rating of 100 psig and a design temperature of 300°F is provided in a branch connection off the reactor building purge exhaust line between reactor building penetration R-552 and the outer isolation valve AH-V4B, on the 328' Level of the auxiliary building. The valve is in full compliance with the "Draft ASME Code for Pumps and Valves for Nuclear Power", Section B, Nuclear Class II Valves. The valve fails closed with a loss of instrument air. The valve is normally locked closed with its outlet flow path blanked. It is locally controlled. The valve is in the air compressor discharge path during containment leak rate testing.

Reactor Building Hydrogen Control Valve AH-V25

A motor operated 6", carbon steel, butterfly valve with ANSI Rating of 150 psig and a design, temperature of 150°F is provided in the hydrogen control line upstream of the hydrogen control exhaust fan. The valve and fan receive their power from the same sources. The source is determined by an ON/OFF switch on Panel No. 25. The valve must be partially open for the fan to start. The valve is positioned locally and has local indication.

Reactor Building Hydrogen Control Discharge Valve AH-V36

A diaphragm operated, 6" carbon steel butterfly valve with an ANSI rating of 150 psig and a design temperature of 150°F is provided in the hydrogen control discharge line. The normally shut vent isolation valve will open when the hydrogen control exhaust fan is started. The valve fails closed with a loss of instrument air. AH-V36 is on the 328' Level of the auxiliary building.

Reactor Building Hydrogen Control Isolation Valve AH-V52

An air cylinder operated 10", carbon steel valve with ANSI rating of 100 psig and a design temperature of 300°F is provided in the hydrogen control line upstream of the hydrogen control valve, AH-V25. The valve is in full compliance with the "Draft ASME Code for Pumps and Valves for Nuclear Power." Section B, Nuclear Class II Valves. This containment isolation valve is padlocked shut and is only opened for hydrogen exhaust fan operation. The power source is similar to that described for AH-V25. The valve fails closed with loss of instrument air. AH-V52 is on the 328' Level of the auxiliary building.

3.3.2 Instrument, Controls, Alarms and Protective Devices

All controls, indicators and annunciators described are located in the Control Room on Panel 25 unless stated otherwise. All remotely controlled RB penetration isolation valves have position indicating lights on Containment Isolation Panel 15 in the Control Room. All ES operated valves have indicating lights located on Engineered Safety Features Panel 13 in the Control Room. All instrumentation, controls, annunciators and computer inputs are included in Tables 11 and 12. Reactor building air pressure indication is provided as part of the Building Spray (BS) System.

As discussed previously the power supply for the hydrogen control exhaust fan, AH-E-34, and its associated throttle valve, AH-V25, is determined by an ON/OFF switch on Panel 25. The throttle valve must be partially open (20%) for fan start. With fan start, the discharge damper opens. The flow path is instrumented with filter differential pressure alarms, radiation, flow and temperature recorders.

3.3.3 Monitoring

The reactor building radiation monitoring system will be used to obtain a sample of the building atmosphere for analysis of its isotopic composition. This system takes samples from two points in the reactor building, which are located approximately 10' 10" east and west of the north-south centerline of the reactor building dome. The samples are transmitted through two lines running from the dome down and outside to the reactor building air sample gaseous monitor schematically shown as monitor HP-R-227, "Radiation Detection and Sampling" on Figure 3-2. The sampling lines are Seismic I. Redundant inlet and discharge lines are provided for the system to prevent a single active failure of any valve from impairing the function of the monitoring system.

In the nuclear sampling laboratory, the sample will be analyzed with a gas chromatograph to determine its hydrogen content. A gamma spectrum analyzer will be used to determine the isotopic composition of the sample.

During atmospheric purging, the purge exhaust flow is continuously monitored and recorded on Panel 25, so that the exact flow to the environment is known. To replace the atmosphere exhausted from the building, a 10 in. pressurization valve (AH-V7), located outside the R.B. is provided to admit a controlled amount of outside air to the building.

3.4 Design Evaluation

Given in Section 5 are evaluations of allowable flow rates as a function of time to stay within Tech. Spec. limits, and the expected off-site exposure as a result of full reactor building purging. Also included in Sections 6 and 7 is an assessment of filter dose buildup and effect of accidental releases of radioactive gases during maximum flow. The exposure analysis shows conformance to limits in each case.

3.5 Tests and Inspections

The atmospheric purge components are not continuously operated and therefore are accessible for out of service inspection. The performance of the system components can be verified while the system is in operation. Pressure, temperature and flow instrumentation are provided as shown in Figure 3-2 to confirm performance of the system and its components. Radiation monitoring instrumentation is also provided in the system to check radioactive levels of the exhuast air. In addition, means have been provided for pre-purge DOP and freon leak tests of the filters.

The steel pipe duct work system was subjected to leak tests during manufacture, erection and after assembly in the field. Filters and filter housings were subjected to manufacturers performance and production tests as well as DOP and Freon 11 tests.

The charcoal filter will be subjected to a Freon 11 leakage test at 1000 cfm, the maximum flow expected in the system.

The HEPA filters will be subjected to an Efficiency-Penetration Test (DOP). The filter will be tested according to MIL-STD-282, May 28, 1956. Penetration will not exceed 0.1 percent of 0.3 micron diameter homogeneous particles of dioctyl-phthalate (DOP).

3.6 Materials

The ductwork is primarily mild carbon steel and has a 6 mil coat of Phenoline 368. The material for other components was given in Section 3.3.

3.7 Operation of Hydrogen Control Purge System

Controls for this system are located on HVAC Panel 25. To start this system it is necessary first to open reactor building isolation valve AH-V3A (closed on ES signal) and AH-V52 (padlocked closed). A Blind flange on valve AH-V7 must be removed prior to this operation. Open throttle valve AH-V25 to about 309° prior to starting the Hydrogen Control fan, AH-E-34. Upon starting the fan the discharge valve, AH-V36, will open. Throttle AH-V36 as desired. When the reactor building pressure is slightly below atmospheric, open AH-V7 (normally locked closed) and then open AH-V3B, to replenish the exhausted air.

The reactor building atmosphere is exhausted through isolation valve AH-V3A, a 10" branch line containing valve AH-V52, a 6" line containing throttle valve AH-V25, pre-filter, two absolute filters and an activated carbon filter. The fan then discharges through valve AH-V36 to the Station Vent.

The system is shutdown by stopping AH-E-34 and closing AH-V25, AH-V52 and AH-V3A and AH-V36.

3.8 Accident Analysis

The worst case limiting accident is inadvertent and undetected initiation of the hydrogen control purge system at full capacity with ground level release for 30 minutes during worst case meteorological conditions. The off-site dose due to this accident is analyzed in Section 7 of this report. The results show maximum off-site exposure to an individual of 61 mrem beta skin dose or 0.73 mrem whole body dose, considerably less than 10CFR100 limits of 25000 mrem whole body dose.

3.9 Failure Modes and Consequences

All failures and their consequences are evaluated with respect to increasing the probability of producing an uncontrolled radioactive release or a release at a faster rate than allowed by procedure.

3.9.1 Loss of Instrument Air

Loss of instrument air will affect only the air operated valves and dampers in the system. All air operated valves in this system fail shut upon loss of air pressure which stops all flow in the system and thereby prevents any release of radioactivity.

3.9.2 Loss of Power

Fan AH-E-34

Loss of power to this fan will reduce flow rate through the system causing a reduction in the release rate.

Valves operated by air will fail shut on loss of power to the solenoid operated air control valves. Motor operated valve AH-V-25 will fail as is on loss of power. Neither of these conditions will cause any increase in the release rate.

Instrumentation

Flow indication will be lost on loss of power, however, this indication does not control the process so no increase in release rate can occur. If an increase in flow rate occurs due to some other cause, HPR 229 will alarm as a backup indication. System operating procedures will require the operator to stop purging upon loss of flow indication.

Filter unit differential pressure alarm capability will be lost on loss of power. Flow indication will provide a backup indication of filter blockage. This alarm does not cause any automatic action. No increase in release rate will occur from loss of this instrument.

Reactor Building High Pressure or Loss of Power to AH-PS-5058 will cause AH-V-3A,B to shut thereby stopping flow in the system.

Radiation Monitor HPR 229 is being modified to cause fan shutdown upon loss of power to HPR 229.

Valve AH-V7 is also interlocked so that when the fan stops, this valve will close, isolating this potential path out of the reactor building.

3.9.3 System Leakage

In order to ensure radiation will not be released from building ducts during operation, a leak test of ducting downstream of the containment isolation valves and the filter housing will be conducted prior to system operation at 18 inches of water positive pressure in accordance with ANSI N510, Section 6.3, 6.4, or 6.5 and shall indicate maximum leakage less than 6 cfm/1000 ft^3 of system volume before acceptance.

3.9.4 Fires

The charcoal filter AH-F-34 is protected by a fire detector and an automatic deluge system which also secures fan AH-E-34. The filter housing drain is piped to floor drains which flow to a collection facility.

3.9.5 Duct Failure

The steel pipe ducting is designed for 2 psig positive pressure. The filter unit housing is designed for 11 inches H_2O negative pressure. The system is protected from high pressure transients by a .5 psig containment pressure interlock from pressure switch AH-PS-5058. Maximum internal duct pressure is limited to 1.5 psig, since the maximum differential pressure will be caused by a .5 psig reactor building pressure coupled with a minimum external pressure of -1.0 psig caused by an external atmospheric disturbance. This extreme pressure condition is within the design ratings of the duct and filter unit and will not cause a duct failure which would result in an uncontrolled radioactive release.

3.9.6 Operator Errors

Misoperation of the valves in the system could possibly increase the rate of radioactive gas release above the maximum allowable rate. However, HPR 229 would alarm and automatically stop flow in the system if the allowable release rate were exceeded.

3.10 System Modification For Purge

In order to use the hydrogen control system to purge the reactor building safely at rates up to 1000 cfm, the following modifications will be made:

1. Replace the existing fan, AH-E-34, with a fan capable of at least 1000 cfm flow rate.

2. Add manual jog control to valve AH-V36.

3. Interlock AH-V7 to close on loss of power to the fan.

4. Provide interlock to trip the fan on high activity as measured on HPR-229, or on failure of/loss of power to HPR-229.

5. Interlock AH-V3A&B to shut on high reactor building pressure.

6. Uncap the stack.

7. Provide gamma monitor probe in the hydrogen control filter housing to monitor filter activity buildup.

8. Increase the measurement range of HPR-229.

Table 3-1

Hydrogen Purge to Atmosphere
Design Performance and Equipment Data

a. Hydrogen Control Exhaust Fan

Quantity	1
Type	Centrifugal Exhauster with Direct Drive
Flow, cfm	0 to 1000
Static Pressure, in W.G.	48 neg at 3550 rpm
Fan (Motor) Speed, rpm	3550
Fan Motor Voltage/No. of Phases/Hz	460/3/60
Motor H.P.	15

b. Hydrogen Purge Air Exhaust Filter Train

One hydrogen purge air exhaust filter train in cabinet/housing containing the following filters listed in sequence with respect to air flow:

(1) Prefilter

Quantity	1
Type	Disposable bag filter
Clean Pressure drop, in W.G.	0.8
Max. Capacity, cfm	1000
Face Velocity through Filter, fpm (max.)	500
Size of Filter, inches	24x24x36
Seismic Classification	I

(2) Absolute Filter (HEPA)

Quantity	1
Clean Pressure drop, in W.G.	1.0
Max. Capacity, cfm	1000

Table 3-1 (Con't)

	Size of Filter, inches	24x24x11-1/2
	Seismic Classification	I
(3)	Carbon Filter	
	Quantity	3 per Bank
	Type	Flat-bed radio Iodine Absorption activated carbon
	Max. Capacity, cfm	1000
	Flow through cell, cfm	333
	Clean Pressure drop, in W.G.	1.0
	Size of Filters, inches	24x40x7-3/4
(4)	2nd Absolute Filter (HEPA)	
	Quantity	1
	Clean Pressure drop, in W.G.	1.0
	Capacity, cfm	1000
	Size of Filters, inches	24x24x11-1/2
	Seismic Classification	I

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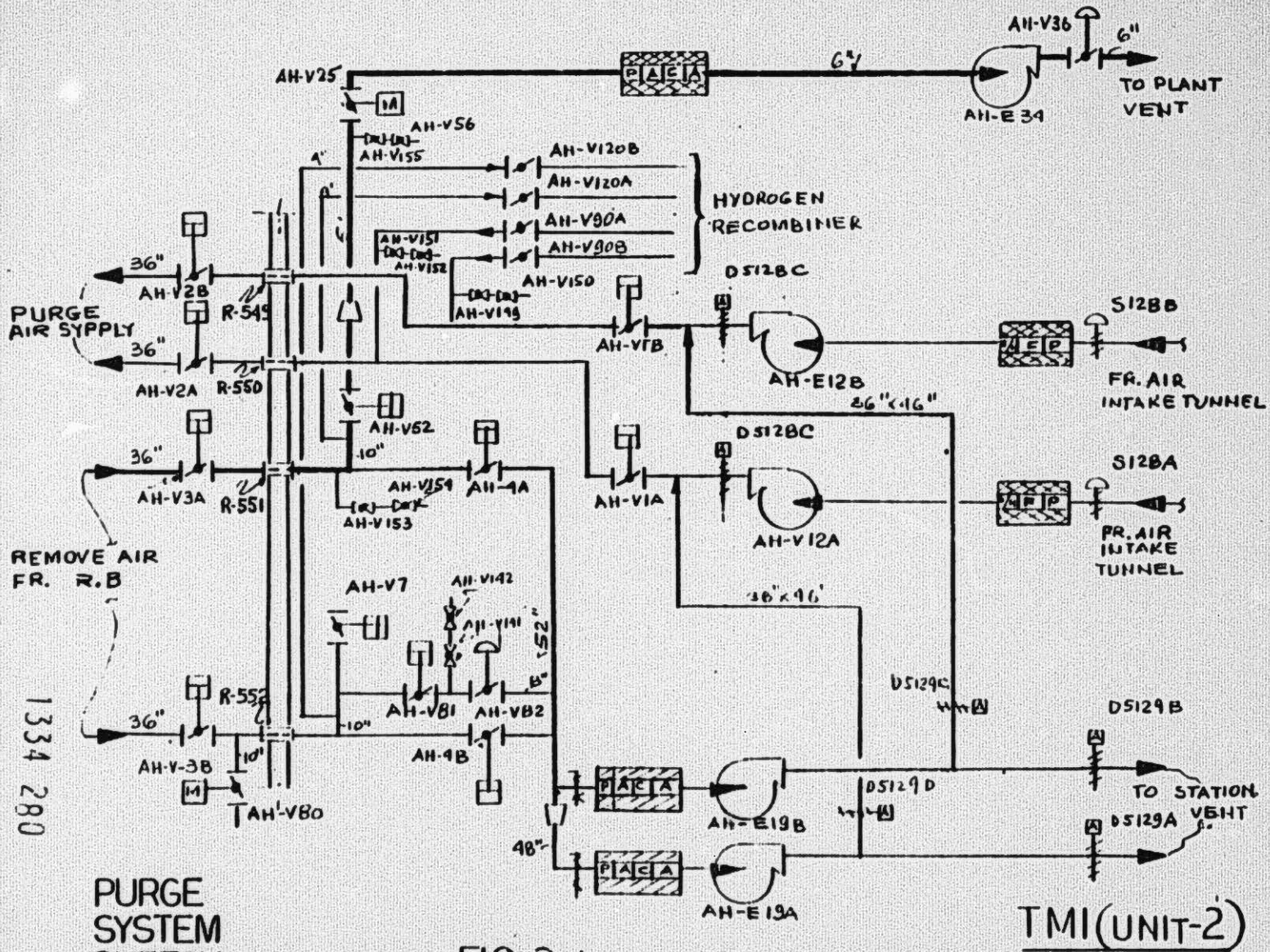
AH-V25
1A
AH-V56
AH-V155
AH-V120B
AH-V120A
AH-V90A
AH-V90B
HYDROGEN RECOMBINER
AH-V151
AH-V152
AH-V150
AH-V149
AH-V36
6"
TO PLANT VENT
AH-E34
PLACA
6"
PURGE AIR SYPPLY
36"
AH-V2B
R-549
36"
AH-V2A
R-550
AH-V1B
D512BC
AH-E12B
S12BB
FR. AIR INTAKE TUNNEL
36"x16"
AH-V52
10"
36"
AH-V3A
R-551
AH-V154
AH-V153
AH-4A
AH-V1A
D512BC
AH-V12A
S12BA
FR. AIR INTAKE TUNNEL
REMOVE AIR FR. R.B
AH-V7
AH-V142
AH-V141
36"x96"
D5129C
D5129B
R-552
36"
AH-V-3B
10"
AH-V80
AH-V81
AH-V82
AH-4B
10"
8"
152"
PLACA
AH-E19B
D5129D
TO STATION VENT
D5129A
48"
PLACA
AH-E19A
TMI (UNIT-2)
PURGE SYSTEM SKETCH
FIG. 3-1
1334 280

FIGURE 3-2

4.0 SAFETY LIMITS FOR RADIOACTIVE GASEOUS RELEASES

4.1 Radiation Exposure in Perspective

4.1.1 Sources of Radiation in the Environment

The average natural radiation exposure to persons living in the United States is estimated to be about 125 mrem per year. The source of this exposure is from cosmic rays and from naturally occurring radioactive elements in the earth. Radiation is received directly from many minerals containing uranium and thorium isotopes in the ground or in the construction materials in homes. The most significant radioisotope in food is potassium. An additional small amount of exposure is received through radioactive gases in the air.

It is estimated that an additional exposure of 100 mrem per year may be received on the average from other than natural sources such as medical X-rays, luminous dials on watches, bomb detonations in the atmosphere, and television.

4.1.2 The Person Rem Concept

It is appropriate to compare new exposures to population groups relative to their exposure to natural background. One measure of the extent of population exposure is to add all the radiation exposures received by each individual in a population group. This resulting quantity is referred to as person-rem. The annual background population exposure within a 50 mile radius of the site is computed to be about 275,000 person-rem (125 mrem times 2.2 million people). By comparison, the person-rem for reactor building purging as discussed in this report is about 100,000 times less than the natural background amount.

It should be noted that the whole body gamma doses listed for each event are most comparable to the background dose. The external body beta dose affects only the external parts of the body (skin or retina of the eyes) which are less sensitive to radiation than the whole body.

4.1.3 Effects of Radiation Exposure

For many years standards committees have spent considerable effort to determine the effect of radiation on man. As a result, a set of guidelines have been developed to define maximum levels of radiation exposure which are acceptable for any individual to receive every year. These recommendations are embodied in the government regulation entitled 10CFR20 which limits whole body exposure to less than 500 mrem per year.

Comparison of the site boundary doses from the events associated with reactor building purge considered in this report indicates that individual exposures are well below both the natural background level of 125 mrem/year and the 10CFR20 limit of 500 mrem/yr.

4.2 Environmental Technical Specifications

Section 2.1.2 of the Environmental Technical Specifications contains specifications for the release of gaseous effluents. These are:

2.1.2.a. The instantaneous release rate of gross gaseous activity except for halogens and particulates with half lives longer than eight days shall not exceed:

$$\frac{Qi}{(MPC)_i} \leq 1.5 \times 10^5 \ \frac{m^3}{sec}$$

where Qi is the release rate in Ci/sec for isotope i, and MPC_i (Ci/m^3) is the maximum permissible concentration of isotope i as defined in Appendix B, Table II, column 1, 10CFR20.

2.1.2.b. The instantaneous release of I-131 and particulates with half-lives greater than eight days, released to the environs as part of airborne effluents, shall not exceed 0.3 μCi/sec.

2.1.2.c. The release rate of gross gaseous activity shall not exceed:

$$\frac{Qi}{(MPC)_i} \leq 2.4 \times 10^4 \ \frac{m^3}{sec}$$

when averaged over any calendar quarter.

2.1.2.d. The release rate of I-131 and particulates with half-lives greater than eight days, shall not exceed 0.024 μCi/sec., when averaged over any calendar quarter.

The specifications above can be used to establish limiting reactor building release rates for instantaneous and quarterly average releases when the Kr-85 (paragraphs a. and c.) and the I-131 (paragraphs b. and d.) are known. Present estimates of airborne activity for Kr-85 of 0.78 μCi/ml and $<1. \times 10^{-9}$ μCi/ml for I-131 make the paragraphs a. and c. most limiting based on Kr-85 activity.

Specification 2.1.2.c above assumes an average X/Q value of 4.2×10^{-5} sec/m^3. Historical meteorological data for the TMI site with elevated releases give an average X/Q value of 1.8×10^{-6} sec/m^3 for a margin factor of 23 built into the quarterly release rate limit. Since the limit is applied over any quarter, the limit for the quarter will not be any greater than one-fourth of the annual dose objective limit. Therefore a factor of roughly one hundred is achieved below the 10CFR20 objective of 500 mrem off-site whole body dose when the quarterly average technical specification limits are imposed for radioactive gaseous effluent release from the plant vent stack.

In other words, while the 10CFR20 limit (to be discussed below) states as its objective a limit of 500 mrem annual whole body dose, the application in the Technical Specification achieves a factor of 100 reduction to 5 mrem annual whole body dose limit when using the quarterly average release rate limits.

4.3 10CFR20 - Standards for Protection Against Radiation

Article 20.105, permissible levels of radiation in unrestricted areas, paragraph (a) states: "The commission will approve the proposed limits if the applicant demonstrates that the proposed limits are not likely to cause any individual to receive a dose to the whole body in any period of one calendar year in excess of 0.5 rem."

Article 20.106, Radioactivity in effluents to unrestricted areas, paragraph (d) states: "For the purposes of this section the concentration limits in Appendix B Table II of this part shall apply at the boundary of the restricted area. The concentration of radioactive material discharged through a stack, pipe or similar conduit may be determined with respect to the pipe where the material leaves the conduit. If the conduit discharges within the restricted area, the concentration at the boundary may be determined by applying appropriate factors for dilution, dispersion or decay between the point of discharge and the boundary."

Table II, column 1 of Appendix B gives the following limits for the pertinent isotopes:

Kr-85	3×10^{-7} μCi/ml
Xe-131m	4×10^{-7} μCi/ml
Xe-133	3×10^{-7} μCi/ml

These isotopic limits are used with the limiting X/Q values in the Technical Specifications given above to determine allowable gaseous releases at the point of effluent release from the plant for instantaneous and quarterly average limits. As pointed out earlier, use of these limits with the Technical Specification X/Q values to determine release rates will yield substantially lower actual values of the isotopic air concentrations in the unrestricted area because the expected X/Q values are at least an order of magnitude below the limiting X/Q values. For this reason, the expected whole body dose will be substantially below the 0.5 rem yearly limit imposed by paragraph 20.105 (a).

4.4 10CFR50 Appendix I - Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion - ALARA.

Paragraph B.1 states: "The calculated annual total quantity of all radioactive material above background to be released from each light-water-cooled nuclear power reactor to the atmosphere will not result in an estimated annual air dose from gaseous effluents at any location near ground level which could be occupied by individuals

in unrestricted areas in excess of 10 millirads for gamma radiation or 20 millirads for beta radiation."

Paragraph B.2 states: "Not withstanding the guidance of paragraph B.1:

(a) The commission may specify, as guidance on design objectives, a lower quantity of radioactive material above background to be released to the atmosphere if it appears that the use of the design objective in paragraph B.1 is likely to result in an estimated annual external dose from gaseous effluents to any individual in an unrestricted area in excess of 5 millirems to the total body; and

(b) Design objectives based upon a higher quantity of radioactive material above background to be released to the atmosphere than the quantity specified in paragraph B.1 will be deemed to meet the requirements for keeping levels of radioactive material in gaseous effluents as low as is reasonably achievable if the applicant provides reasonable assurance that the proposed higher quantity will not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin."

As stated above in discussing 10CFR20 limits, the margin provided in bounding X/Q values in the Technical Specification to limit effluent release rates will bring the 0.5 rem total body dose limit stated in 10CFR20 down to the 10CFR50 Appendix I level when actual expected site meteorology is applied in calculating actual population doses.

Nevertheless, the actual accumulated doses during purge operation shall be calculated to demonstrate conformances to the 10CFR50 Appendix I limits. Off-site dose calculations for the planned purge scenario using typical meteorological data are presented in Section 5.

4.5 Limiting Conditions for Operation of Purge Program

In determining acceptability of the reactor building purge program, based on the above requirements, the limiting conditions are as follows:

1. Release rate for Kr-85, Cs-137, and I-131 are determined from Technical Specification paragraphs 2.1.2.a. and 2.1.2.b. to establish the most limiting release rate. For the current status of reactor building isotopic content, the Kr-85 content is most limiting.

2. For the total quantity of Kr-85 to be released, the quarterly average release rate limit stated in paragraph 2.1.2.c. can be verified not to be exceeded with the Kr-85 concentration below 0.988 μCi/ml.

3. Based on actual meteorology of the site at the time of release, administratively limit release rate to minimize actual dose to the unrestricted site boundary area so that total beta and gamma dose do not exceed limits stated in 10CFR50 Appendix I. The limiting value is 15 millirems skin dose due to beta activity in the Kr-85 release.

4. Continuous monitoring of meteorology will be provided to administratively control release as a function of actual characteristics and to calculate the dose build-up within allowable Appendix I limits.

5.0 ENVIRONMENTAL EFFECTS OF PURGE OPERATION

Starting with the best estimates of reactor building airborne isotopic activity, the Technical Specifications are used to establish limiting purge release rates from the plant vent stack. Using typical historical meteorology, the site boundary beta skin dose and gamma dose are calculated using the NRC prescribed methods defined in Regulatory Guide 1.109 "Calculation of Annual Doses to Man From Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR Part 50 Appendix I" and Regulatory Guide 1.111 "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors."

Since at the time of purge operation, the I-131 has decayed to inconsequential levels, the Kr-85 contribution to beta skin dose is the most limiting radioactive source. The results of this analysis yield a peak beta skin dose of 5.0 mrems when purging is limited during unfavorable meteorological conditions. This is compared to the 10CFR50 Appendix I limit of 15 mrems. For the case in which the only limit on purge operation is the instantaneous Technical Specification limit due to Kr-85 content in the released gas, and no limit is placed on purging during unfavorable conditions, the peak skin dose is 10.0 mrems. In both cases, the peak gamma dose is less than 0.10 mrems, compared to the 10CFR50 Appendix I limit of 10 mrem.

5.1 Purge Compliance with Technical Specification

5.1.1 Method

The computer code TIDRLS and manual calculations are used to ensure that the Technical Specifications for instantaneous and quarterly average release of iodine and gaseous activity are not exceeded. The appropriate Technical Specifications are:

a. $\frac{Qi}{MPC_i} \leq 1.5 \times 10^5$ m^3/sec for instantaneous release of gross gaseous activity (applies for Kr-85)

b. Instantaneous release rate of I-131 and particulates (applies for Cs-137) with half-life greater than eight days must be ≤ 0.3 μCi/sec

c. $\frac{Qi}{MPC_i} \leq 2.4 \times 10^4$ m^3/sec for average over a quarter of gross gaseous activity (applies for Kr-85)

d. Average release over a quarter of I-131 and particulates (applies for Cs-137) with half-life greater than eight days must be ≤ 0.024 μCi/sec

Where Q_i is the release rate in Ci/sec for isotope i and MPC_i (Ci/m^3) is the maximum permissible concentration of isotope i as defined in Appendix B, Table II, Column 1, 10CFR20.

Hand calculations are used to determine an initial purge rate that would not exceed the instantaneous Technical Specifications discussed above. For gases (Kr-85, Xe-133, and Xe-131m), the allowable purge rate is determined as follows:

$$\frac{Qi}{MPC_i} = 1.5 \times 10^5 \ m^3/sec$$

$$Qi = (MPC_i)\ (1.5 \times 10^5 \ m^3/sec)$$

Then, purge rate , m^3/sec = Q/concentration, (where concentration is in Ci/m^3).

For I-131, and Cs-137 the allowable purge rate is determined as follows:

$$\text{Purge rate} = \frac{0.3 \ \mu Ci/sec \times 10^{-6}}{(\text{concentration})\ (1-\text{charcoal efficiency})}, \ m^3/sec$$

(where concentration is in Ci/m^3 as above).

After determining an initial purge rate, the decrease in containment activity inventory resulting from the purge was calculated. No credit is taken for decrease in inventory due to decay. When containment activity decreases sufficiently to allow approximately a doubling of the flow rate, a new purge rate is established. The same process is then repeated until a maximum purge rate of 1000 cfm was obtained. The decrease in activity in the RB is calculated as follows:

$$T = -\frac{M}{m} \ln \frac{C_o}{C_i}$$

where

- T = time of purge
- m = purge rate in cfm
- M = volume of containment in ft^3
- C_i = concentration at start of purge
- C_o = concentration at end of purge

Once the purge scenario is determined, it was input into the TIDRLS computer code for a detailed calculation of instantaneous and integrated releases. The integrated purge rate can be used to determine an average purge rate. Using average purge rate, the quarterly average release rate can be calculated as follows:

$$\text{Quarterly release rate} = \frac{(\text{Average release in } \frac{Ci}{sec})\ (\text{purge time in seconds})}{(\text{MPC})\ (\text{Number of seconds in quarter})}$$

5.1.2 Evaluation with Technical Specifications

Initial Purge Rate

Using the techniques described in Section 5.1.1, the initial purge rate is calculated that can be accomplished without exceeding the Technical Specification limits.

For Kr-85, the allowable instantaneous purge rate is:

Purge Rate = Q/concentration

$$Q = MPC\ (Kr\text{-}85) \times 1.5 \times 10^5\ m^3/Sec$$

From 10CFR20 Appendix B, Table II column 1,

$$MPC\ (Kr\text{-}85) = 3 \times 10^{-7}\ Ci/m^3$$

Therefore,

$$\text{Purge Rate} = \frac{3 \times 10^{-7}\ (Ci/m^3) \times 1.5 \times 10^5\ (m^3/Sec) \times 60\ (Sec/min)}{0.78\ (\mu Ci/ml) \times 10^{-6}\ (Ci/\mu Ci) \times 2.832 \times 10^4\ (ml/ft^3)}$$

$$= 122\ CFM$$

The maximum allowable initial purge rate for release of Kr-85 is 122 CFM, based on an initial Kr-85 content in the reactor building atmosphere of 0.78 μCi/ml.

For Cs-137, the allowable instantaneous purge rate is:

$$\text{Purge Rate} = \frac{0.3\ (\mu Ci/Sec) \times 60\ (Sec/Min)}{1 \times 10^{-5}\ (\mu Ci/ml) \times 2.832 \times 10^4\ (ml/ft^3)\ (1\text{-}0.9)}$$

$$= 636\ CFM$$

The maximum allowable initial purge rate for release of Cs-137 is 636 CFM, based on an initial Cs-137 particulate content in the reactor building atmosphere of 1×10^{-5} μCi/ml and a particulate filter efficiency of 90%.

For I-131, the maximum allowable purge rate is determined the same way as for Cs-137 except that charcoal efficiency replaces particulate filter efficiency. This evaluation gives a maximum allowable initial purge rate for I-131 of 636×10^4 CFM, based on an initial I-131 concentration of 1×10^{-9} μCi/ml and a charcoal efficiency of 90%.

From the above calculation, it can be seen that the limiting isotope for Tech. Spec. release limits is Kr-85.

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Purge Duration

As the concentration of radioisotopes within the reactor building is reduced while purging, the allowable purge rate can be increased up to the maximum purge flow capacity of 1000 CFM. A typical scenario for purging might consist of stepwise purge levels of 100 CFM, 200 CFM, 500 CFM, 1000 CFM. The limiting concentrations for Kr-85 to stay within instantaneous Technical Specification limits are given in the Table 5.1.2-1.

The time to reach the limiting concentrations for each purge level can be calculated as described in the previous section assuming step wise increases in purge rate each time the Kr-85 content drops below the maximum concentration for the next increased purge rate level. The final time period at 1000 CFM purge rate is the time required to reduce the Kr-85 content from 0.095 μCi/ml to the 1×10^{-5} μCi/ml MPC limit for restricted access to the reactor building per 10CFR20 Appendix B Table 1, Column 1. The time period at each purge rate and the initial and final Kr-85 levels are given in Table 5.1.2-2.

The above analysis shows that the RB can be purged in about 31 days without exceeding the instantaneous release Technical Specifications. The releases must also be compared to the quarterly average Technical Specifications.

The quarterly average Technical Specification for I-131 and Cs-137 can be easily met using the above scenario. The quarterly average Technical Specification for gaseous activity can be met, so long as the initial gaseous concentration in the RB (Kr-85) is less than 0.988 μCi/ml. This can be shown as follows:

$Q/MPC \leq 2.4 \times 10^{4}$ m^3/sec (from Tech Specs)

Where MPC = 3.0×10^{-7} μCi/ml = 3.0×10^{-7} Ci/m^3 for Kr-85, the limiting isotope

Max allowable Q = $(2.4 \times 10^{4}$ m^3/sec$)(3.0 \times 10^{-7}$ Ci/m^3
= 7.2×10^{-3} Ci/sec

$$\text{But, max allowable concentration (C in } \mu\text{Ci/ml)} = \frac{(Q)\ (\text{Number of seconds in quarter})}{\text{RB volume in ml}}$$

$$\text{Max allowable C} = \frac{(7.2 \times 10^{-3}\ \text{Ci/sec})(90\ \text{days} \times \frac{24\ \text{hr}}{\text{day}} \times \frac{60\ \text{min}}{\text{hr}} \times \frac{60\ \text{sec}}{\text{min}})}{(2.0 \times 10^{6}\ \text{ft}^3)(2.832 \times 10^{4}\ \text{ml/ft}^3)}$$

$$C = 0.988\ \frac{\mu\text{Ci}}{\text{ml}}$$

Maximum allowable concentration to be released within quarterly allowable Technical Specification limits can also be determined for Cs-137 and I-131.

For Cs-137, using

Avg. Release Rate $\leq$ 0.024 μCi/Sec,

Max. allowable concentration released =

$$\frac{0.024\ \mu\text{Ci/Sec} \times \text{Total Seconds in Quarter}}{\text{RB Volume in ml}}$$

Because a large fraction of particulate Cs-137 will be captured by the HEPA filters in the hydrogen control system, credit can be taken for filtered removal of Cs-137 prior to release. The HEPA filter at worst performance can be expected to be at least 90% efficient. Therefore

$$\text{Max. allowable concentration} = \frac{0.024\ \mu\text{Ci/Sec} \times \text{Total Seconds in Quarter}}{\text{RB Volume in ml} \times (1-\text{Filter efficiency})}$$

$$= \frac{0.024\ (\mu\text{Ci/Sec}) \times 90 \times 24 \times 60 \times 60\ (\text{Sec./Quarter})}{2.0 \times 10^6\ (\text{ft}^3) \times 2.832 \times 10^4\ (\text{ml/ft}^3) \times (1-.9)}$$

$$= 3.3 \times 10^{-5}\ \mu\text{Ci/ml}$$

Therefore the maximum allowable Cs-137 concentration in the reactor building atmosphere for full release within one quarter within current quarterly average Technical Specification limits is 3.3×10^{-5} μCi/ml for a worst case filter efficiency of 90% Cs-137 removal.

The maximum allowable I-131 concentration for full release within quarterly Technical Specification limits is determined in the same manner as for Cs-137 except that the HEPA filter efficiency is replaced by charcoal efficiency for Iodine removal. Using a 90% charcoal efficiency, the maximum allowable I-131 concentration within the reactor building atmosphere for full release within one quarter conforming to quarterly average Technical Specification limits is 3.3×10^{-5} μCi/ml.

Table 5.1.2-1

Maximum Concentration of Kr-85 vs. Purge Rate

Purge Rate (CFM)	Kr-85 Conc. (μCi/ml)
100.	<0.95
200.	0.48
500.	0.19
1000.	0.095

Table 5.1.2-2

Time to Reach Each Limiting Concentration

Purge Rate (CFM)	Initial Kr-85 Content (μCi/ml)	Final Kr-85 Content (μCi/ml)	Purge Duration (days)
100	<0.95	0.48	9.4
200	0.48	0.19	6.5
500	0.19	0.095	1.9
1000	0.095	1×10^{-5}	12.7
		Total Duration	30.5 days

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5.1.3 Results of Purge Release Within Technical Specification Limits

Table 5.1.3-1 gives the summary results for maximum allowable instantaneous purge rate within Technical Specification limits for the three radio-isotopes most limiting on purge rate.

Using the Kr-85 concentration to establish purge flows as a function of time, a full purge of the reactor building airborne radio isotope concentrations within MPC limits for restricted area access can be achieved in 31 days following the schedule shown in Table 5.1.2-2.

The quarterly average release rate Technical Specification limits can be maintained so long as the concentrations of radioisotopes within the reactor building atmosphere are less than the levels given in Table 5.1.3-2.

The expected quarterly average release based on current Kr-85 levels will be 79% of the allowable Tech. Spec. limit for full purge of the reactor building atmosphere.

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Table 5.1.3-1

Isotope	Estimated Concentration (μCi/ml)	Limiting Purge Rate (CFM)
Kr-85	0.78	122
Cs-137	$<1 \times 10^{-5}$	636
I-131	$<1 \times 10^{-9}$	636×10^{4}

Table 5.1.3-2

Isotope	Quarterly Avg. Tech. Spec. Limit	Limiting Purge Concentration (μCi/ml)
Kr-85	$\leq 7.2 \times 10^{+3}$ Ci/sec	0.988
Cs-137*	≤ 0.024 μCi/Sec	3.3×10^{-5}
I-131*	≤ 0.024 μCi/Sec	3.3×10^{-5}

*Assumes filter efficiencies of 90% for Cs-137 and I-131 removal.

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5.1.4 Conclusions for Purge Release Within Tech. Spec. Limits

1. Full purge of the reactor building atmosphere below instantaneous Technical Specification radioisotope release rate limits can be accomplished in 31 days with quarterly average radioisotope release rates below 79% of the Technical Specification limits.

2. Kr-85 is the limiting isotope for determining purge rate. So long as the initial Kr-85 content is below 0.988μCi/ml, the quarterly average limits can be met.

3. The purge rates can be allowed to vary from 100 to 1000 CFM with 9.4 days at 100 CFM, 6.5 days at 200 CFM, 1.9 day at 500 CFM and 12.7 days at the maximum purge rate of 1000 CFM.

5.2 Off-Site Dose Determination for Purge

5.2.1 Method

The PURTST and XDCALC computer programs were developed for use in a parametric study to assess the effects on offsite dose of different purge start times and various purge procedures. These programs incorporate the environmental dose calculation routines utilized in the dose assessment effort following the accident. The program computes not only beta, gamma, and thyroid doses but also checks for compliance with Technical Specification limits and MPC offsite limits each hour during the purge. Both elevated and ground level release points are evaluated. Two basic types of simulations were made. One type (XDCALC) uses a "predetermined" purge rate and the second (PURTST) uses a "variable" purge rate depending on meteorological conditions each hour.

The basic objective of the study was to make a series of computer simulations to determine integrated doses offsite while achieving the goal of reducing noble gas concentrations to below MPC in the reactor building during a purge time of one to two months, starting in October/November, 1979. Results are used to plan and evaluate the purge procedures to be utilized.

5.2.1.1 Meteorological Data

Historical data from the onsite meteorological tower taken in 1976, 1977 and 1978 were used for the same months of the year to simulate conditions in 1979. Hourly values of measured wind speed, wind direction and vertical temperature difference were available on computer files for access by the program. Specifications for instruments used to collect these data are given in Table 5.2-1. The meteorological program is designed and operated in accordance with NRC Regulatory Guide 1.23.

Vertical temperature difference was used to determine atmospheric stability (Pasquill-Gifford category) in accordance with procedures

also outlined in NRC Regulatory Guide 1.23. Wind speed measured at 100 ft. on the meteorological tower is adjusted to the 33 ft. or 160 ft level to be representative of the assumed effective heights used during the purge dose studies. The equation used for this adjustment is as follows:

$$\bar{U}_{RH} = \bar{U}_{100} \left(\frac{h_{RH}}{h_{100}} \right)^{n}$$

where h_{RH} (ft) and $\bar{U}_{RH}$ (mph) are the height and wind speed at the release height respectively, and n is a function of atmospheric stability as follows:

Pasquill-Gifford Category	Value of n
A, B, C	0.25
D	0.33
E, F, G	0.50

5.2.1.2 Initial Radioactivity in Reactor Building

Simulations for variable purge rates requires knowledge of only the initial reactor building inventories since the PURTST program computes the remaining inventories after each purge hour. These initial inventories are based on the RB air samples taken on June 26, 1979. For start dates beyond July 1, the initial inventories were reduced according to the half lives of the individual isotopes. The following table summarizes the assumed starting inventories in uCi/cc.

	Purge Start Date*		
Isotope	July 1	August 1	September 1
Kr-85	1.0	1.0	1.0
Xe-131M	0.085	0.012	0.0021
Xe-133	0.015	0.00017	negligible
I-131	0.00012	0.00002	negligible

*Note that the Kr-85 level is conservative relative to the best estimate value of 0.78 quoted in Section 2.3.

Airborne iodine levels are low in the reactor building and are expected to remain low during the purge. In addition, the purge effluent will flow through charcoal absorbers prior to discharge. If most of the iodine is in an organic form, and more penetrates the charcoal absorbers compared with the elemental iodine, it would not be of particular importance in the environmental analysis because it is not taken up in the cow-milk pathway. Therefore, the controlling isotopes are the noble gases, (primarily Kr-85), and thyroid doses due to iodine releases are not calculated.

5.2.1.3 Plant Characteristics

Effluent from the reactor building purge will be directed into the plenum of the plant vent and out the plant vent stack. If the supplementary filter system plant release point were to be used (since this system is horizontal) no plume rise due to momentum of the existing gas is possible. Effluent will always enter the turbulent wakes surrounding plant buildings and undergo initial dilution in this region. After leaving the influence of these buildings, concentrations in the plume at ground level will always decrease with distance. Plant parameters pertinent to the diffusion model are given in Table 5.2-2.

Several runs were made to determine the beneficial effects of plume rise when the plant vent stack is used. For these runs, vent parameters are necessary and are given in Table 5.2-2. For elevated plumes, terrain must be subtracted, therefore, the assumed terrain at various distances downwind in each of the 16 direction sectors is provided in Table 5.2-3.

5.2.1.4 Standards

10CFR50, Appendix I -- provides guidelines for offsite dose objectives for routine plant operation due to noble gases as follows:

Noble Gas air gamma dose	10mrad*
Noble Gas air beta dose	20mrad*
Noble Gas whole body dose	5mrem**
Noble Gas skin dose	15mrem**

*applicable at site boundary
**applicable to real person

5.2.1.5 Atmospheric Dispersion Model

For long-term ground level releases in the building wake, the sector average version of the Gaussian dispersion model is used.

$$X/Q_{gnd} = \frac{2.03}{\bar{u} \times \Sigma_z} \quad \text{where,}$$

$$\Sigma_z = \sqrt{\sigma_z^2 + \frac{cH^2}{\pi}} < \sqrt{3}\,\sigma_z$$

X/Q = concentration at ground level (Ci/m^3) ÷ release rate Q (Ci/sec)

$\bar{u}$ = wind speed at 33 ft. level (m/sec)
x = distance from plant (m)
σ_z = Pasquill-Gifford dispersion coefficient (m^3)
c = wake coefficient (=0.5)
H = building height (m)

For the vent stack release runs, plume rise above the building wake boundary under certain light wind conditions was accounted for. The model referred to as "mixed mode" from Regulatory Guide 1.111 was used for these cases. The elevated release model is defined as follows:

$$X/Q_{ele} = \frac{2.03}{\bar{u} \sigma_z} \exp\left(\frac{-(H_S + \Delta h)^2}{2 \sigma_z^2}\right)$$

where symbols are as before, and

H_S = stack exit height above local terrain (m)
Δh = plume rise due to momentum jet (m) (determined using Briggs' model)

$$X/Q_{\text{mixed mode}} = (1 - E_t)\ X/Q_{ele} + E_t\ X/Q_{gnd}$$

where E_t is determined as follows

$$\text{if } \frac{W_o}{\bar{u}} \geq 5.0,\ E_t = 0$$

$$\text{if } \frac{W_o}{\bar{u}} < 1.0,\ E_t = 1.0$$

and

$$E_t = 2.58 - 1.58\ (W_o/\bar{u}) \text{ for } 1 \leq W_o/\bar{u} \leq 1.5$$

$$E_t = 0.3 - 0.06\ (W_o/\bar{u}) \text{ for } 1.5 \leq W_o/\bar{u} \leq 5.0$$

W_o = Vertical exit velocity of plume

5.2.1.6 Dose Calculation Models

In this analysis both whole body and skin dose analyses were made. The skin dose (D_{skin}) results from both beta and gamma radiation. Since beta particles are stopped by only a few centimeters of air, one must be submerged in the plume to receive a dose. Therefore, ground level concentrations of each isotope based on X/Q must be used. Gamma dose (D_{gamma}) can be received as a result of shine from plumes aloft as well as from submersion in the plume. A finite plume dose model is used to estimate gamma dose as described later.

Beta dose to the skin is computed using the following relationship from NRC Regulatory Guide 1.109 Equation (11):

$$D_{skin} = D_{gamma} + 0.114 \Sigma_i\ Q_i\ (X/Q)\ F_i t$$

where

Q_i = release rate (Ci/sec) of isotope i
X/Q = ground level dispersion factor (sec/m^3)
F_i = dose factor Kr-85 = 1.34×10^{-3}, Xe-131m = 4.76×10^{-4} Xe-133 = 3.06×10^{-4} (units of $mrem\text{-}m^3/pCi\text{-}yr$)
t = time (hour)
0.114 = constant for units correction

Gamma dose (D_{gamma}) is computed using the finite plume model detailed as Equation (6) of Regulatory Guide 1.109. The integral "I" is solved using the method of Hamawi in accordance with the regulatory guide and is not repeated here. For Kr-85, the only significant airborne isotope, the beta dose is about 100 times higher than the gamma dose for an individual in the plume.

Population doses are computed using the same dispersion and dose routines out to 50 miles. At each population segment shown in Figures 5.2-1 and 5.2-2, the dose is multiplied by the number of people and summed to determine person-rems.

5.2.1.7 Computer Programs

The dose models were incorporated in a routine called PURTST which is used to assess the effect of varying a series of input parameters. The general program flow diagram is shown in Figure 5.2-3. The routine starts by reading the meteorological parameters for the hour. Then, using the reactor building isotopic content for the hour, doses and concentrations at the site boundary are calculated for an assumed arbitrary flow rate of 100 CFM. Based on limits established at the beginning of the run for beta and gamma dose as well as MPC for each isotope, an allowable flow rate is computed. Tech Spec limits are also checked and if the release rate would exceed limits the flow is further reduced. Flexibility is provided to bypass checks of certain limits as part of the parameter study. For example, if Tech Spec limits are to be neglected, the Tech Spec limits are set to high values so they will not produce limiting flow rates. Similar provisions are made for other limiting conditions such as dose limit.

This calculation is repeated every hour until the total period of record specified has been processed. At the end of each hour, the amount of each isotope released (based on the limiting parameter) is subtracted from the amount in the reactor building at the beginning of the hour. This provides the isotope concentration for establishing the release rate during the next hour.

After each hour, summary tables of beta and gamma dose are incremented along with the total volume purged. Thus, at the end of the specified purge time the resulting doses are available along with total effluent release. No limits have been placed on doses within a direction sector to stop purge while winds are in the loaded sector.

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Several runs were made using the XDCALC program without regard to feedback from meteorological conditions. The XDCALC computer program has been used extensively in evaluating doses resulting from the accident. The XDCALC program uses the dispersion and dose models described above and a predetermined release source term that can be specified each hour. Doses are computed at several distances near the site and integrated each hour in the appropriate direction sector.

5.2.2 Evaluation of Off-Site Doses Due to Purge

General

An evaluation has been made to assess the off-site man-rem, beta-gamma integrated dose and instantaneous dose rate to the whole body, skin and thyroid as a result of various purge scenarios. The results vary as a function of release rate and meteorology. Alteration of the existing purge path to release via the station vent stack also affects the net environmental impact associated with RB purge.

A series of runs were made using the PURTST and XDCALC programs and meteorological data for July, August, October, and November from 1976, 1977 and 1978. The results of these analyses are included in Table 5.2-5. Sensitivity studies were completed for varying purge rates, release locations and meteorological conditions.

Cases Considered

A summary of the cases evaluated is given in Table 5.2-4.

The first case listed, Case No. 1, determines the expected boundary doses when a constant activity release rate is used to purge all activity over a one month interval (Kr-85 release rate is 2.15E4 μCi/sec for one month). Estimated doses are calculated using typical July meteorology as taken from July 1976. The release point is taken to be the roof top supplementary filter vent.

The second case listed, Case No. 2, uses the same meteorology and release location as Case No. 1 but the release rate is taken from the step wise purge rate scenario calculated in Section 5.1. Variations to Case No. 2 are treated in: Case No. 11, for vent stack elevated releases; Case No. 15, for Aug. 1977 reference meteorology; and Case No. 17, for August 1978 meteorology.

The third case listed, Case No. 8, uses historical meteorology conditions to limit release rates to hourly limits for beta, gamma, and MPC off-site limits. This case uses July 1976 meteorological data and the ground level equivalent roof vent filter release point. Variations to Case No. 8 are treated in: Case No. 12, for vent stack elevated release; Case No. 13 for August 1976 meteorological data; Case No. 14, for August 1977 meteorological data; and Case No. 16 for August 1978 meteorological data.

Man-rem exposure to the surrounding population out to a radius of 50 miles from the site boundary was calculated for Case No. 1. Because the resulting man-rem was representative and quite small, this calculation was not repeated for the other cases.

For Case Nos. 2, 11, 15, and 17 the step wise purge scenario assumes 4 days at 50 CFM, 9.5 days at 100 CFM, 6.5 days at 200 CFM, 2 days at 500 CFM, 14.5 days at 1000 CFM, for a total purge duration of 36.5 days to reach MPC levels in the reactor building. Although 50 CFM was selected for the initial purge rate, since the I-131 has decayed substantially, 100 CFM can be used for the initial purge rate to stay within Technical Specification limits as shown in Section 5.1.

For Case Nos. 8, 12, 13, 14, and 16 the release is made in a series of steps in which purge rate each hour is varied in accordance with X/Q weather conditions to meet the following objectives:

- Instantaneous Dose Rate (β) < 0.3 mr/hr (at site boundary)*
- Instantaneous Dose Rate (γ) < 0.1 mr/hr (at site boundary)
- Instantaneous activity release < $1.5 \times 10^5 \frac{m3}{sec}$
- Peak hourly isotopic concentration at site boundary < 10 MPC
- Maximum purge rate = 1000 cfm
- Minimum purge rate = 20 cfm

A number of iterations have been necessary to determine the appropriate selection of these limits to achieve full purge of the reactor building.

The purge routine sets specific dose rate limits and uses real time meteorological input to meet these objectives by varying purge rate. Use of this scenario increases the length of purge, as compared to the technical specification release, but reduces total beta and gamma dose. It can be seen that the purge can be completed in 33-49 days using the dose objective release, depending on the release point and meteorology used.

To confirm the effect of meteorology for months when the purge is most likely to be carried out, Cases 18 through 20 and Cases 21 through 23 and 26 were run with October and November meteorological data respectively. Cases 18 and 23 use the predetermined purge rate schedule as defined in Section 5.1, Cases 19 through 22 use meteorological feedback data to control purge rate within specified limits on hourly gamma and beta dose rates, MPC levels, purge flow capacity and Tech. Spec. limits, Case 26 uses only the Tech. Spec. limit on isotopic release rate for Kr-85 of 1.5×10^5 x MPC for determining purge rate each hour. This is a slightly different scenario than the step wise purge increase rate case as in Case No. 2 since in Case 26, the purge rate will change slightly each hour as the KR-85 content remaining in the reactor building is reduced.

*For vent stack elevated releases, the beta dose rate limit is lowered to 0.1 mr/hr.

The total integrated whole body dose in person-rem was calculated for Case Nos. 1 and 21. Because the levels were less than 1 person-rem it was not considered necessary to complete this calculation for each of the other cases.

5.2.3 Dose Effect Results

Table 5.2-5 compares the actual calculated dose for each release scenario discussed in the previous section. From this table it can be seen that the total time to purge the reactor building to within 10CFR20, Appendix B, Table I limits for maximum permissible concentration is in the range of 31 to 49 days depending on the purge constraints imposed. In all cases, the 10CFR20 limit of 500 mrem annual whole body dose is easily met, and the meteorology selected has only a slight effect on total dose when purging is controlled by meteorological feedback. Tabulated in Table 5.2-5 for each case are the average effective purge rates, the time to reach MPC, peak skin doses at the site boundary due to beta and gamma radiation, the location of the peak doses, and the number of times specified limiting conditions are reached for the controlled purge cases.

A review of the data presented in Table 5.2-5 indicates that a release made at a rate consistent with the current TMI-2 Technical Specifications will result in a small integrated dose to a person residing at the nearest residence (700 meters from the vent stack). The total whole body man-rem expenditure considering populations out to a radius of 50 miles from the site is less than 1 man-rem. By elevating the release point, an immediate reduction in the beta dose at the nearest residence can be achieved, however, the gamma dose is only slightly reduced. The beta dose reduction is primarily a result of increasing the height of the theoretical plume centerline containing the beta emitting Krypton above the nearest residence. The mean free path of the gamma's however is much greater than that for beta's and gamma radiation dose is caused by a shine-mechanism which is relatively insensitive to the variation in distance between the source and the nearest residence.

From Cases 1, 2, 10, 15, and 17, the skin dose due to beta radiation for ground level release varies between 24 to 50 mrem with no meteorological feedback to control purge rate. By introducing controlled purging using meteorological feedback to limit purging to specified hourly dose limits, the beta skin dose varies between 4.8 and 18 mrem for ground level releases. For elevated releases at the plant vent stack, the beta skin dose varies between 3.5 to 10.0 mrem with no meteorological feedback to control purge rate. With controlled purging (using dose limits and meteorological feedback) and with elevated releases at the plant vent stack, the beta skin dose is in the 2.9 to 5.6 mrem range depending on the actual meteorology.

Table 5.2-6 gives a summary of the range of various purge scenarios calculated.

In all cases, the skin dose due to gamma radiation is below 0.35 mrem at the site boundary and the whole body integrated dose throughout a 50 mile radius from the plant is approximately 1 person-rem for the worst case scenario.

The results for elevated releases are all within the 10CFR50 Appendix I guidelines of 15 mrem annual skin dose for "as low as reasonably achievable." In particular, using elevated releases with controlled purging, skin dose can be held to within one third of the 10CFR50 Appendix I guideline limits.

Figures 5.2-4, 5, and 6 present results of purge characteristics as a function of time for several of the purge scenarios analyzed. The first figure (Fig. 5.2-4) compares the beta skin dose build-up at the limiting site boundary location as a function of time during purge for several cases. In all cases, most of the dose is generated during the early portions of the purge program when the reactor building contains the highest Kr-85 activity. After about 20 days, the purge is proceeding at maximum flow rate of 1000 CFM and the incremental dose is small from that point forward. Comparing Cases 19 and 20, the effect of tightening limits on the hourly dose objective for beta skin dose can be seen. Although reducing the hourly beta limit from 0.1 mrem/hr to 0.05 mrem/hr lowers the initial dose accumulation, the total dose is not significantly affected due to the fact that the initial savings in dose is compensated by increased incremental dose later in the purge cycle and an extension of the time at which the 1000 CFM flow rate limit is achieved.

Figure 5.2-5 shows the additional characteristic features of the Case No. 21 purge as a function of time. This case is most typical of the recommended purge program for November purging, using meteorological feedback to control purge at specified hourly dose limits with an elevated vent stack release. The figure shows purge flow rate, integrated purged volume, remaining reactor building curie inventory, and accumulated beta skin dose. Again, this figure shows that the majority of the reactor building activity has been released by 20 days and that the remaining purge is completed at maximum flow rate to reach MPC levels.

For comparison, Figure 5.2-6 shows the case of November meteorology, elevated vent stack releases but with purge rate controlled only by instantaneous Tech. Spec. release rate limits for Kr-85. In this case the accumulated skin dose is twice as large as the hourly dose rate limited purge case, the maximum purge rate is reached much sooner and the reactor building Kr-85 curie activity drops more rapidly.

The total integrated whole body dose in person-rem was calculated for Case Nos. 1 and 21. Because the levels were less than 1 person-rem it was not considered necessary to complete this calculation for each of the other cases.

5.2.4 Dose Effect Conclusions

Table 5.2-5 compares the actual calculated dose for each release scenario discussed in the previous section. Table 5.2.6 gives a comparison of the range of calculated doses for all evaluated meteorologies for use in evaluating alternative purge strategies. From these results the following conclusions are made:

1. All cases evaluated are substantially below the 10CFR20 objective of 500 mrem/yr whole body dose. Although the whole body dose is not shown, it can be taken as approximately 1.2% of beta skin dose giving results below 0.5 mrem whole body dose for all cases evaluated.

2. All cases evaluated with elevated vent stack release give skin dose below the 15 mrem/yr 10CFR50 Appendix I guideline for "As Low As Reasonably Achievable." Results give beta skin dose in the range of 2.9 to 7.5 mrem.

3. Controlled purging with meteorological feedback reduces site boundary doses by 20 to 50% for elevated release. Results for controlled purge at elevated release give beta skin dose in the range of 2.9 to 5.6 mrem. The worst case dose is approximately 1/3 of 10CFR50 Appendix I guideline.

4. The total integrated person-rem whole body dose is less than 1 person-rem to the population within 50 miles of the reactor. Therefore, in performing cost-benefit studies called for in 10CFR50 Appendix I paragraph II.D, the maximum benefit for other alternatives to controlled purging of the reactor building cannot be greater than 1 person-rem even if these alternatives achieve zero release.

5. The total time to purge the reactor building to maximum permissible concentration levels for Kr-85 is in the range of 30 to 36 days for elevated vent stack release.

6. Purge of the reactor building should be done through the station vent stack, using the meteorological feedback system described, in order to maintain off-site doses as low as reasonably achievable.

TABLE 5.2-1

TMI Site Weather Instruments

Approximate Height Above Tower Base (ft)	Sensed Parameters	Recorded Parameters	Recorders** Analog	Recorders** Digital	Instrument Description
100A	Wind speed & direction	Wind speed & direction	Esterline Angus Model L11125 servo type two channel strip chart recorder for each level. Accuracy 0.1% of full scale. Eleven inch chart width.	Varian-V71 mini-computer with automatic transmittal via phone line to disc storage.	Wind speed is Teledyne Model 50.1. Anemometers are three cup with a threshold of 0.75 mph and a distance constant of 5 ft. Accuracy is 0.15 mph or 1%, whichever is greater. Direction is measured using Teledyne 50.2. Quick-2 vanes are used with threshold of 0.93 mph, a distance constant of 3.7 ft. and damping ratio of 0.4 at initial attack angle of 10°. Direction accuracy is ±2°.
100B	Wind speed & direction	Wind speed & direction			
100	Vertical & horizontal angle	Vertical & horizontal angle	Same as above	Same as above	R.M. Young-Bivane Model 17002C. Threshold 0.4 mph, delay distance for 50% recovery is 3.2 ft. damping ratio is 0.53.
150A	Temperature	T 150-33A ft	Westronics Model W11D2. Twelve channel Selectronic potentiometric dot printing recorder. Accuracy better than 0.3% of full scale.	Same as above	Dual Rosemont platinum 4 wire RTDs Model 104 MP (calibration traceable to NBS) with 414 L linear triple bridges. Accuracy (RSS for system) is 0.1°F.
150B	Temperature	T 150-33B ft			
33A	Temperature-reference for comparison with other level.	Ambient temperature 33 ft			
33B	Temperature-reference for comparison with other level.	Ambient temperature 33 ft			
Ground level	Rainfall	Rainfall		Same as above	Belfort Model 5915 weighing rain gauge. Accuracy 1/2% of full scale.
150A	Dew point temperature	Dew point temperature	Same as above	Same as above	EG&G Model 110 SM-thermoelectric dew point system. Accuracy ±.5F.
150B	Dew point temperature	Dew point temperature			
150A	Ambient temperature	Ambient temperature			
150B	Ambient temperature	Ambient temperature			
		$T_{Amb\ 340} - T_{DP\ 340}$			
		$T_{Amb\ 33} - T_{DP\ 33}$			

** All parameters are continuously recorded.

TABLE 5.2-2

Input Data For Dispersion Modelling

METEOROLOGICAL DATA

Parameter	Characteristics
Wind speed	Measured at 100 ft, adjusted to 33 ft level for ground releases and adjusted exponentially to 160 ft for elevated releases
Wind direction	Measured at 100 ft
Stability	Based on Δt 150-33 ft and PG (A-G) dispersion categories in accordance with Regulatory Guide 1.23
σ_z (dispersion coefficient)	Based on PG curves, limited to 1000m
Site Specific Data	
Terrain height	See Table 5.2-3
Population distribution	See Figures 5.2-1 and 5.2-2
Plant Specific Data	
Vent Height	160 ft
Vent exit diameter	3.0m
Vent exit velocity	9.1 m/sec
Building Height for computation of Σ_z, the effective vertical dispersion coefficient	170 ft

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[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0	0	0
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0	0	0
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0	0	0
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0	0	0	0
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	55	65	65	65	63	50
[illegible]	[illegible]	[illegible]	[illegible]	65	[illegible]	65	65	[illegible]	65	75	90	85	65	95	65	50
[illegible]	75	90	90	65	60	90	65	60	65	75	130	125	65	162	65	50
[illegible]	75	190	130	80	105	110	65	65	65	130	165	150	168	203	65	70
[illegible]	80	200	155	80	110	110	65	70	130	152	305	150	310	310	75	80
[illegible]	100	200	200	100	130	120	75	305	170	152	310	310	350	350	100	100

*Terrain height for distances beyond 20,000 was assumed to be unchanged from the last value since higher elevations would result in little change in dispersion at greater distances.

TABLE 5.2-4

Assumptions for Purge Dose Calculations

Run No.	Purge Scenario*	Release Point***	Meteorological Data Applied**	Program Used
1	Constant Ci/sec over 1 month to release all containment activity.	GND	July '76	XDCALC
2	Tech. Spec. Limit Purge Scenario at 100, 200, 500, 1000 CFM per Section 6.1.	GND	July '76	XDCALC
8	Release rate limit to hold hourly β, γ, and MPC levels based on meteorology (- limit = 0.3 mrem/hr).	GND	July '76	PURTST
10	Hourly release rate limit at 1.5E5 instantaneous Tech. Spec. M^3/sec. limit.	GND	July '76	PURTST
11	Same as No. 2	VENT	July '76	XDCALC
12	Same as No. 8 (β limit = 0.1)	VENT	July '76	PURTST
13	Same as No. 8 (β limit = 0.3)	GND	Aug. '76	PURTST
14	Same as No. 8 (β limit = 0.3)	GND	Aug. '77	PURTST
15	Same as No. 2	GND	Aug. '77	XDCALC
16	Same as No. 8 (β limit = 0.3)	GND	Aug. '78	PURTST
17	Same as No. 2.	GND	Aug. '78	XDCALC
18	Same as No. 2	VENT	Oct. '78	XDCALC
19	Same as No. 8 (β limit = 0.1).	VENT	Oct. '78	PURTST
20	Same as No. 8 (β limit = 05, 50 CFM lower flow)	VENT	Oct. '78	PURTST
21	Same as No. 8 (β limit = 0.1)	VENT	Nov. '78	PURTST
22	Same as No. 8 (β limit = 0.3)	GND	Nov. '78	PURTST
23	Same as No. 2	VENT	Nov. '78	XDCALC
26	Hourly release rate limit at 1.5E5 instantaneous Tech. Spec. M^3/sec. limit.	VENT	Nov. '78	PURTST

* All cases except No. 1 limit purge rate to 1000 CFM maximum.
** As monthly meteorological data changes from July to August, September or later, short lived isotope initial inventory is reduced based on decay half-life.
***GND = Ground level equivalent roof filter release point, VENT = Vent stack elevated release point.

TABLE 5.2-5

Results of Purge Dose Calculations

Run No.	Avg Purge Rate (CFM)	Time MPC is reached	Peak β Skin Dose (mrem)	Pop-(1) ulated Location (m, dir)	Peak γ Dose (mrem)	Pop-(1) ulated Location (m, dir)	Whole Body Population Dose (person-rem)	Number of Times Condition Limited Flow					
								Gamma	Beta	MPC Limit	Tech Spec	Max Flow	Min Flow
1	-	30d	50.0	700,E	0.35	700,E	1.02	-	-	-	-	-	-
2	-	36d	43.0	700,E	0.35	700,E	-	-	-	-	-	-	-
8	489	49d	13.0	700,E	0.12	700,E	-	0	392	0	211	746	84
10	643	31d	35.0	700,E	0.35	700,E	-	0	0	0	343	393	0
11	-	36d	6.4	1500,SE	0.10	700,E	-	-	-	-	-	-	-
12	534	33d	5.6	1500,SE	0.10	800,ESE	-	0	50	0	311	1072	0
13	491	41d	4.8	700,E	0.02	700,E	-	0	343	0	183	858	57
14	500	42d	12.5	700,E	0.08	700,E	-	0	362	0	188	648	18
15	-	36d	25.0	700,E	0.11	700,E	-	-	-	-	-	-	-
16	500	46d	10.0	700,E	0.05	700,E	-	0	394	0	162	528	12
17	-	36d	24.0	700,E	0.10	700,E	-	-	-	-	-	-	-
18	-	36	3.5		-		-	-	-	-	-	-	-
19	-	34	2.9		0.04		-	0	183	0	210	406	2
20	-	56	2.9		0.03		-	-	-	-	-	-	-
21	-	36	5.0		0.06		0.75	0	165	0	210	411	71
22	-	48	18.0		0.09		-	-	-	-	-	-	-
23	-	36	7.5		-		-	-	-	-	-	-	-
26	-	32	10.0		-		-	0	0	0	316	416	0

(1) Location of highest dose to resident (meters, direction).

TABLE 5.2-6

Comparison of Dose for Purge Scenarios

Case Nos	Purge Scenario	Peak Boundary Skin Dose (mrem) BETA	GAMMA	Whole Body Pop. Dose (person-rem)
12,19,21	Met. Feedback, Hourly Limit Beta = 0.1, Variable Meteorology, Vent Stack Release	2.9-5.6	0.04-0.10	0.75
11,18,23	Tech. Spec. Purge Schedule at 100, 200, 500, 1000 CFM; Variable Meteorology, Vent Stack Release	3.5-7.5	0.1	
8,13,14 16,22	Met. Feedback, Hourly Limit Beta = 0.3, Variable Meteorology, Ground Level Release	4.8-18.0	0.02-0.12	
2,15,17	Tech. Spec. Purge Schedule at 100, 200, 500, 1000 CFM; Variable Meteorology, Ground Level Release	24.0-43.0	0.10-0.35	1.0

LIMITS

10CFR20 Objective	500 mrem/yr. whole body*
10CFR50 APP. I Guideline FOR ALARA	5 mrem/yr. whole body* 15 mrem/yr. skin dose (BETA & GAMMA)

* Note that whole body is approximately 1.2% of beta skin dose for Kr-85 therefore limit will not be approached so long as skin dose is met.

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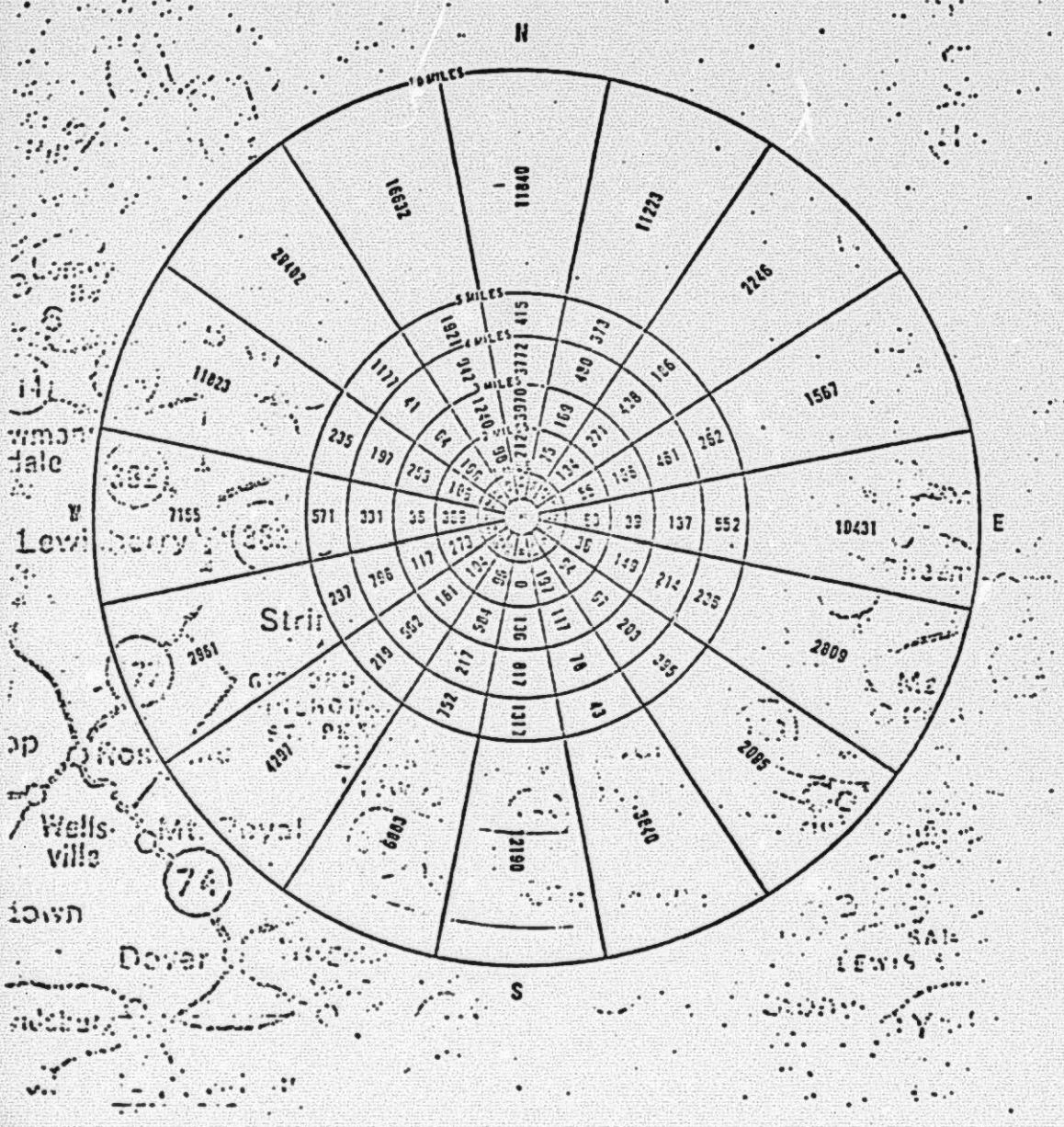
Figure 5.2-1



DISTANCE	TOTAL	CUM. TOTAL
0-1 MI.	658	
1-2 MI.	2,017	2,675
2-3 MI.	7,579	10,254
3-4 MI.	9,676	19,930
4-5 MI.	8,691	28,621
5-10 MI.	137,474	166,095

• EXCLUSION RADIUS

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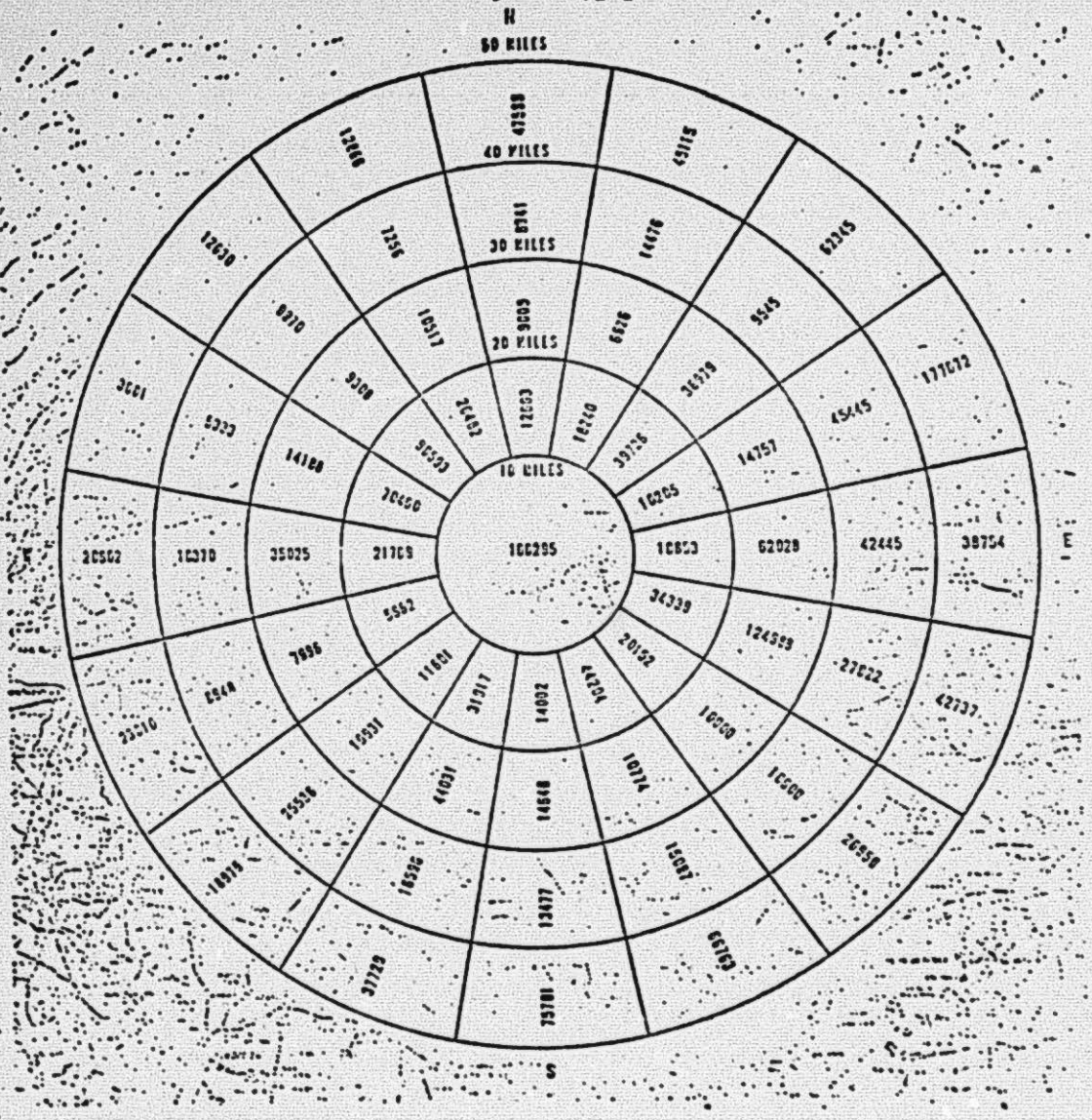
POPULATION DISTRIBUTION
0 TO 10 MILES – 1980

THREE MILE ISLAND NUCLEAR STATION UNIT

Figure 5.2-2

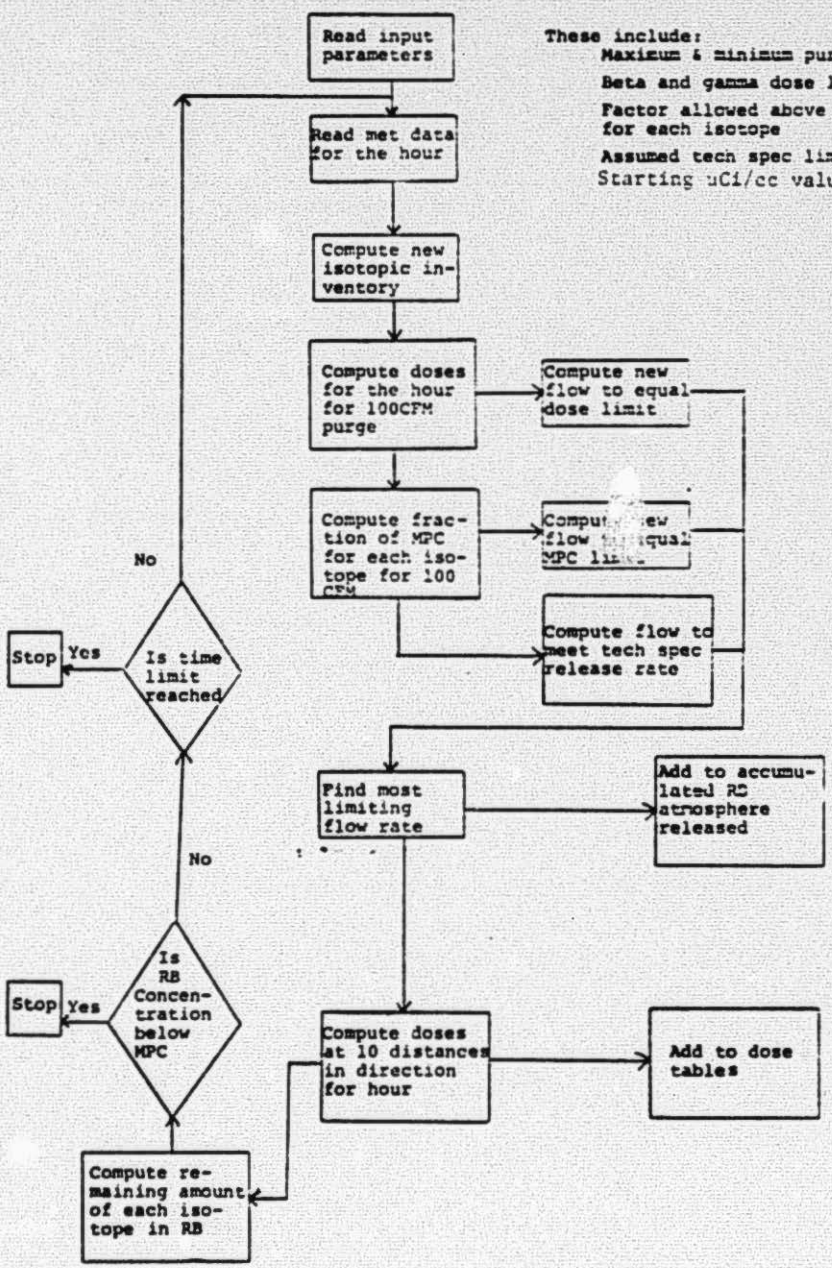


DISTANCE	TOTAL	CUM. TOTAL
0-10 MI.	166,295	
10-20 MI.	577,298	745,593
20-30 MI.	433,031	1,178,624
30-40 MI.	273,857	1,452,481
40-50 MI.	703,210	2,155,691

POPULATION DISTRIBUTION
0 TO 50 MILES – 1930
THREE MILE ISLAND NUCLEAR STATION UNIT

Figure 5.2-3

Generalized Flow Diagram for PURTST Program



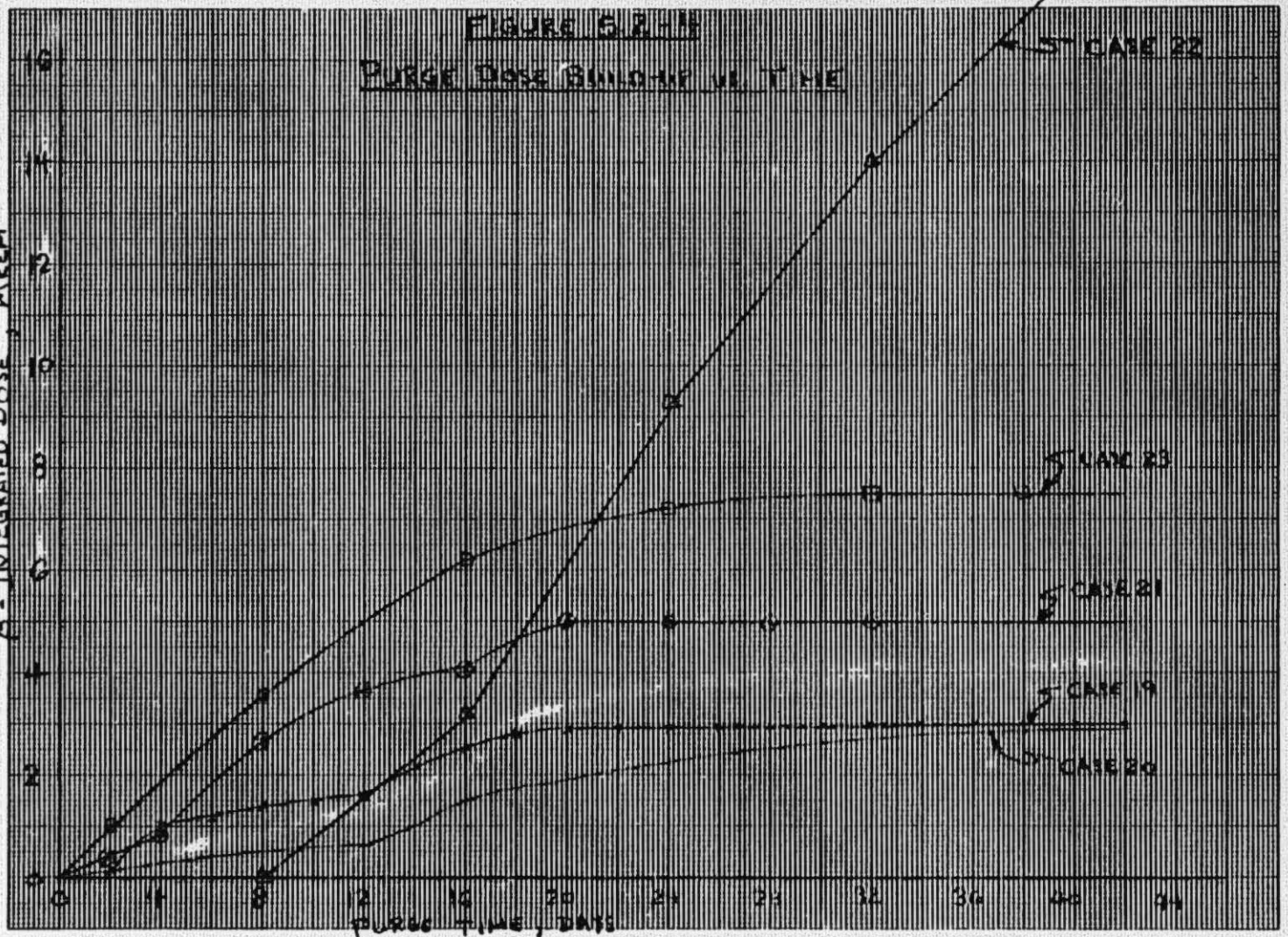


FIGURE 5.2.1-6
SELECTED CHARACTERISTICS VS PURGE TIME
RUN NO. 216
CURIES REMAINING IN REACTOR BLDG
BETA SKIN DOSE - mr/pm
TOTAL VOLUME PURGED, 10⁶ ft³
PURGE TIME, DAYS
BETA
DOSE
ACTIVITY
RELEASED

6.0 OPERATIONAL EFFECTS OF PURGE OPERATION

In order to complete the assessment of the radiological impact of purging the reactor building atmosphere using the hydrogen control system flow path, the filter dose rate has been evaluated to show acceptability for on-site exposure. The filter dose rate analysis for the purge process was completed assuming that the recirculation system was not operated.

To be conservative, airborne activity was assumed to include radioisotopes present as of July 1, 1979. At the time of actual purge, the shorter lived I-131 will be many orders of magnitude lower.

During a RB purge using the systems in the modes as described in this report, the most significant man-rem expenditure to personnel on site will be incurred during filter change out activities. The following evaluation defines the change out sequence, estimates the required man-hour expenditures and utilizing the theoretical dose rate derived from the predicted curie buildup in the filter trains from the TIDRLS program, quantifies the total man-rem expenditures.

6.1 Method

The computer code TIDRLS calculates time dependent radioactive transport into and out of a single node. The node used in this analysis was the TMI-2 containment building. It is a versatile code that may be used for ventilation studies, associating reactor primary coolant activity with an unidentified leakage rate for setting conditions of operation, release of radioactivity, dose rate to personnel in control room, filter inventories of fission products and so on.

Individual isotopes and initial concentrations are read into the code which then identifies the family to which it belongs and performs all calculations for both parent and daughter isotopes. Provisions are made to include recirculation and purge from the node. The node volume is assumed to have separate liquid and vapor regions and each isotope may be assigned a partition factor for transport between liquid and vapor regions. All gaseous daughters of isotopes in the liquid region are transported to the vapor region automatically. A gaseous daughter of a parent isotope trapped in a filter is released from the filter.

The radiological assessment calculation for filter change out used the TIDRLS code with recirculation through HEPA filters only (no iodine removal). The code was run for one hour intervals in the recirculation mode and filter fission product inventory at each time step was obtained. The individual isotopes and their initial concentrations were based on actual air sample data. A review of the fission product inventory at the time of the accident and their respective decay schemes determined that additional isotopes beyond

those identified by gamma spectroscopic analyses performed to date do not reasonably exist. However, additional isotopes were assumed to be present in the filter change out dose rate calculations to develop a worst case calculation. The inventory of any isotope assumed to be present was at its minimum detectable concentration for this worst case.

The dose rate calculation assumed the fission products trapped by the filters were evenly distributed across the face of the filter, and the filter was a disc source. The dose rate is given by the equation

$$D = Ko(E)\ E\ \frac{Sa}{2}\ [E_1\ (b) - E_1\ (b \sec \theta)]$$

where

D	= dose rate (r/hr)
Ko(E)	= dose rate per unit energy flux (r/hr per Mev/cm^2 sec)
E	= photon energy (Mev)
b	= optical length (cm)
$E_1(b)$	= exponential integral
θ	= angle between center of filter and point of measurement
S_A	= source (photon/cm^2 - sec)

The calculation of dose rate was done for each gamma present in the filter.

A filter efficiency of 90% was used in calculation of filter fission product accumulation.

6.2 Filter Dose Evaluation

6.2.1 H_2 Control Filter Change Out

During the purge operation using the Hydrogen Control System, both HEPA and charcoal filters will be used. After being bagged nearby the filter housing, the filters will be transported to the equipment hatch. One man will be required to handle the HEPA filter while two men are required to handle the charcoal filters because of their greater weight.

6.2.2 Filter Dose Rates

Using the best estimate RB airborne activity dose rates on the the H_2 control filters are calculated. The purge rate was based on the Tech. Spec. scenario given in Section 5.1.

Assuming a purge without prior recirculation, the analysis showed that dose rate on the H_2 control HEPA and iodine (charcoal) filters would be less than the design changeout point of 1 r/hr at

the end of a 35 day purge. This analysis assumed that iodine, gases and particulates in the RB would all be below MPC after this purge.

Since filter changeout can be accomplished with dose rates up to 1 r/hr and since the filter dose rates expected are well below this value, the doses received by workers performing the filter removal should be acceptable.

6.3 Results

Table 6.3-1 shows the dose rates that will occur on the hydrogen control system exhaust filter, if the reactor building is purged without prior recirculation. The maximum buildup on the HEPA filter is 340 mr/hr at the end of the 840 hour purge period. The charcoal filter will have a maximum dose rate less than 1.2 R/hr after about two weeks of operation when July 1979 I-131 concentration are assumed. The dose rate decreases from 1.2 R/hr after two weeks as a result of Iodine-131 decay. At the end of the purge the dose rate would be less than 350 mr/hr. Using the I-131 levels expected to be present in November 1979, the charcoal filter doses will be several orders of magnitude lower.

6.4 Conclusions

The dose rates calculated from best estimates of fission products in containment atmosphere during July 1979 for hydrogen HEPA and charcoal filters should not come close to the 1R/hr design basis for filter changeout.

Since the dose rate levels do not require filter changeouts during the operation of the purge system, the impact on man-rem expenditures will be minimal.

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TABLE 6.3-1

Hydrogen Purge Filter Dose Rate*

Isotope	Energy Mev	% Yield	Conversion Factor (R/HR)/Ci	Filter Inventory At 120 HRS (Ci)	Dose Rate At 120 HRS (R/HR)	Filter Inventory At 348 HRS (Ci)	Dose Rate At 348 HRS (R/HR)	Filter Inventory At 552 HRS (Ci)	Dose Rate At 552 HRS (R/HR)	Conversion Factor (R/HR)/Ci	Filter Inventory At 840 HRS (Ci)	Dose Rate At 840 HRS (R/HR)
Cs-137	0.662	85	4.539	7.94×10^{-3}	0.036	2.77×10^{-2}	0.126	3.98×10^{-2}	0.181	4.539	4.67×10^{-2}	0.212
Cs-134	0.570	23	1.085	1.82×10^{-3}	0.002	6.31×10^{-3}	0.007	9.80×10^{-3}	0.011	1.085	1.05×10^{-2}	0.011
	0.605	98	4.856	1.82×10^{-3}	0.009	6.31×10^{-3}	0.031	9.80×10^{-3}	0.048	4.856	1.05×10^{-2}	0.051
	0.796	99	6.324	1.82×10^{-3}	0.012	6.31×10^{-3}	0.040	9.80×10^{-3}	0.062	6.324	1.05×10^{-2}	0.066
Cs-136	0.818	100	6.497	4.44×10^{-6}	0	9.35×10^{-6}	0	9.30×10^{-6}	0	6.497	5.29×10^{-6}	0
	1.05	82	6.586	4.44×10^{-6}	0	9.35×10^{-6}	0	9.30×10^{-6}	0	6.586	5.29×10^{-6}	0
	1.23	20	1.862	4.44×10^{-6}	0	9.35×10^{-6}	0	9.30×10^{-6}	0	1.862	5.29×10^{-6}	0
Total HEPA					0.059		0.206		0.302			0.340
I-131	0.364	82	2.469	0.275	0.679	0.423	1.044	0.318	0.785	2.469	0.122	0.306
	0.637	7	0.362	0.257	0.100	0.423	0.153	0.318	0.115	0.362	0.122	0.044
Total Charcoal					0.779		1.197		0.900			0.350

*Based on RB concentrations in Table 2-2
Purge Rate Not Constant - Changing According to Section 5.1.

7.0 ENVIRONMENTAL EFFECTS OF PURGE ACCIDENT

The purge accident is discussed in Section 3. This section contains the analysis of the environmental effects of the postulated purge accident.

The accident analysis is performed consistent with Regulatory Guide 1.145. "Atmospheric Dispersion Models for Potential Accident Consequences Assessments at Nuclear Power Plants."

7.1 Description of Accident

The postulated purge accident that leads to worst case radiation dose to the environment is the extreme condition of uncorrected inadvertent initiation of the modified hydrogen control system at 1000 cfm for 30 minutes before any controlled purging of the reactor building has been completed. The likelihood of this accident is extremely low because of the interlocks and procedures in place that allow purge system operation only when planned.

Analysis of this accident using the conservative Regulatory Guide 1.145 meteorology makes the calculated environmental exposure even less likely.

7.1.1. Reactor Building Release During Accident

For an assumed 30 minutes purge system flow at 1000 cfm with a building concentration of 1 μCi/ml of Kr-85 prior to any controlled purge activity gives a total accident curie release of:

$$\text{Curies} = 1000 \frac{ft^3}{min} \times 30 \text{ min} \times \frac{2.83 \times 10^4 \text{ ml}}{ft^3} \times \frac{1 \mu Ci}{ml} \times \frac{1 \text{ Ci}}{10^6 \text{ Ci}}$$

$$= 850 \text{ Ci released.}$$

7.2 Accident Dispersion Model

The accident dispersion parameter X/Q has been computed for TMI Unit 2 using the methodology outlined in the NRC's Regulatory Guide 1.145 which accounts for distance to the site boundary in each of 16 direction sectors and takes into account the reduction in dose due to wind meander under low wind speed conditions. Calibrated SF_6 diffusion tests conducted at the site in 1971 (reported in Amendment 24 to the Unit #1 application) demonstrated the existence of this meander effect.

Based on two years of site meteorological data, each of which had a combined recovery rate of more than 90%, the appropriate X/Q for use in short-term calculations (i.e., less than two hours) was determined to be 6.8×10^{-4} sec/m^3.

Following is a summary of methods used and input data.

Meteorological Data

A two-year period of site meteorological data was used with wind speed and direction from the 100 ft level and ΔT taken between 150 ft and 33 ft. Speed was adjusted to be representative of the 33 ft level utilizing a power law relationship with the exponent being determined as a function of stability as follows:

$$\bar{u}_{33ft} = \bar{u}_{100ft}\left(\frac{H_{33}}{H_{100}}\right)^n$$

where n = 0.25 for Pasquill Stability Classes A, B and C
0.33 for Pasquill Stability Class D
0.50 for Pasquill Stability Classes E, F and G

and H = height (m)

Data recovery (percent) by parameter was as follows:

Parameter	First Year (7/1/76-6/30/77)	Second Year (7/1/77-6/30/78)
Wind speed	96.0	93.4
Wind direction	95.0	93.0
delta-T	95.4	93.5
Combined	93.5	91.0

Diffusion Class

The Pasquill diffusion class was determined using vertical temperature difference (ΔT) and the categories given in NRC Regulatory Guide 1.23. Values of σ_y and σ_z were determined as a function of distance and stability class using the standard Pasquill-Gifford curves. The distance to the site boundary in each of the 16 direction sectors was variable with direction and was taken as the minimum distance to the site boundary in the central or either adjacent sector and are given in Table 7-1.

Values of Σ_y were computed as follows:

$$\Sigma_y = M\sigma_y \qquad (x \leq 800m)$$
$$\Sigma_y = (M-1)\sigma_{y_{800m}} + \sigma_y \qquad (x > 800m)$$

where x is distance

M = from 1.145 figure 3, f (wind speed)

Equations

Regulatory Guide 1.145 requires the use of three diffusion equations as follows:

$$(1)\quad X/Q = \frac{1}{\bar{u}\left(\pi\sigma_y\sigma_z + \frac{A}{2}\right)}$$

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(2) $X/Q = \frac{1}{\bar{u}\ (3\pi\sigma_y\sigma_z)}$

(3) $X/Q = \frac{1}{\bar{u}\ A}$

where:

X/Q = relative concentration (sec/m^3)
$\bar{u}$ = wind speed at 33 ft (m/sec)
σ_y = lateral plume speed coefficient (m)
σ_z = vertical plume spread coefficient (m)
A = smallest vertical plume cross-sectional area of containment ($\sim 2000 m^2$)

For each calculation, the following procedure is used to determine the appropriate X/Q. Determine the maximum of equation (1) and (2). Then determine the minimum of that equation and equation (3) if the wind speed is less than 6 m/sec and the stability is not unstable. This value is used for all calculations that follow.

Calculations

Values of site boundary X/Q were determined for each hour of the two year data base using the above equations. Cumulative probability distributions were then made for each direction and separately for the combined data independent of direction. An envelope was contructed around all 16 direction dependent curves and the 0.5% probable value (i.e., the value exceeded no more than 0.5% of the time) was determined to be 6.8×10^{-4} sec/m^3. A second value required by the Regulatory Guide 1.145 procedure at the 5% level on the direction independent curve was determined to be 6.3×10^{-4} sec/m^3. According to the Regulatory Guide, the first value must be used since it is higher.

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Table 7-1

Assumed Distance to Site Boundary
in Each Direction

Direction (from plant to site boundary)	Distance (m)
N	650
NNE	650
NE	630
ENE	610
E	610
ESE	610
SE	625
SSE	710
S	925
SSW	705
SW	705
WSW	705
W	1400
WNW	1400
NW	650
NNW	650

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7.3 Environmental Dose Consequences From Purge Accident

From Sections 7.1 and 7.2, the site boundary cloud concentration becomes:

$$\frac{\text{Curies}}{M^3} = \frac{850 \text{ Ci}}{30 \text{ min} \times \frac{60 \text{ sec.}}{\text{min}}} \times 6.8 \times 10^{-4} \text{ sec/m}^3$$

$$= 3.21 \times 10^{-4} \text{ Ci/m}^3$$

From Regulatory Guide 1.24, the beta and gamma air dose for Kr-85 activity becomes:

$$\text{Beta Air Dose (mrads)} = 0.23 \times 0.67 \times 3.21 \times 10^{-4} \times 1800 \text{ sec} \times 10^3 \frac{\text{mr}}{\text{r}}$$

$$= 98 \text{ mrads}$$

$$\text{Gamma Air Dose (mrads)} = 0.25 \times .0052 \times 3.21 \times 10^{-4} \times 1800 \times 10^3$$

$$= 0.75 \text{ mrads}$$

Using Regulatory Guide 1.109 Table B-1 to convert from air dose to whole body dose gives:

whole body dose due to accident = 0.73 mrem.

This whole body dose is compared to 10CFR100 limits of 25000 mrem maximum allowable total radiation whole body dose to demonstrate that the accident consequences are well within 10CFR100 accident limits.

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8.0 ALTERNATIVE TO REACTOR BUILDING PURGE PROGRAM

As discussed in Section 1, the unknown Three Mile Island Unit 2 core configuration poses a small but incalculable risk. It is possible that reactor building entry could be made and additional reactor building investigations completed (all necessary steps toward final disposition of the damaged reactor core) without cleaning-up the airborne radioactivty in the reactor building. Because of the hazards involved and added precautions required when operating in an airborne contaminated environment, several methods were examined for clean-up of the airborne activity prior to recommending the controlled purge option. This section discusses each of the evaluated alternatives.

8.1 No Atmospheric Clean-Up

It is often tempting to conclude that no further action should be taken to reduce airborne radioactivity inside the reactor building. In view of the other acceptable alternatives available, the decision to take no action is not justified. The current level of activity is 80,000 times the maximum permissible concentration for restricted access per 10CFR20 Appendix B Table I, column 1, for Krypton-85. Since the activity is well defined, it may be possible to develop adequate shielding for a reactor building entrant to complete some assessment of reactor building conditions with this high radiation environment. The risks to the entrant are quite high however and the opportunity to obtain useful information is very poor in this condition.

More importantly, as activity continues in and around the reactor building, the likelihood of unplanned accidental releases of the contained gases under conditions of undesireable meteorology remains high. Although the reactor building is presently adequately sealed, the ability to maintain this sealed system indefinitely is questionable.

8.2 Design Basis for Alternate Atmosphere Treatment and Storage Systems

The design bases for the alternate systems considered in this section are as follows:

Current Noble Gas Activity Within Containment

The noble gas activity within containment at this point in time consists entirely of the isotope Kr-85, with a half-life of 10.7 years. All other radioactive isotopes of xenon and krypton have decayed to negligible quantities.

The concentration of Kr-85 within containment has been determined, based on sampling performed during the Summer of 1979, to be less than 1 μCi/ml.

Required Concentration of Kr-85 After Cleanup

The systems should reduce the Kr-85 concentration to the maximum permissible concentration (MPC) for occupational exposure, or $1 \times 10^{-5} \mu Ci/ml$ (see 10CFR20). While some work could proceed with higher concentrations of Kr-85, a concentration of 1 MPC or less is considered essential for extensive recovery work inside of the containment building.

Containment Volume and Required Process Volume

The containment volume is two million cubic feet. At least 11.5 containment volumes must be processed using a "bleed and feed" type of operation to reduce the containment Kr-85 concentration to 1 MPC. This amounts to 23 million cubic feet of processed volume.

It should be noted that the process volume is based on the assumption of perfect mixing of clean incoming gas with the gas inside of containment. A higher process volume could be required if this assumption is not realized.

Design Bases Release of Kr-85 from the Site

The design basis for the gas compression and charcoal absorption systems is zero off-site release.

Zero release inherently cannot be achieved by the cryogenic system, which can remove a part but not all of the krypton from the process stream. The design basis off-site release for this system is 10^{-3} of the total containment inventory, or about 60 curies of Kr-85. This value is based on the removal efficiency specified in the existing system equipment specification.

Seismic Design Category

The seismic category for system components, supports, and buildings is Class 1. This category is the same as that for the TMI-2 containment building and, accordingly, is considered required for components which would contain the Kr-85 if it were transferred from containment. It should be noted that Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water Cooled Nuclear Power Plants," imposes less stringent seismic design requirements on current gaseous radioactive waste systems. This Regulatory Guide is not, however, considered appropriate for the situation at TMI-2.

Design Code

The design, fabrication, and installation of pressure boundary components is in accordance with the requirements of the ASME Code, Section III, Division 1, Class 3. Again, this code is more stringent than would be required of current gaseous radioactive waste

systems by Regulatory Guide 1.143. It is considered appropriate, however, because it is consistent with the code for the existing containment vessel.

8.3 Charcoal Adsorption and Storage System

The first alternative considered for reducing the airborne activity in the reactor building is to draw off the reactor building atmosphere into a charcoal bed storage container so that the noble gases would remain adsorbed to the charcoal. This charcoal would then remain in storage indefinitely. In order to maintain the reactor building pressure within acceptable limits, the atmosphere is continuously replenished with outside air so the airborne concentration is reduced in exponential fashion.

In order to achieve MPC levels within the reactor building the equivalent of 11.5 times the reactor building atmospheric volume must be processed.

8.3.1 System Description (See Figures 8.3-1 and 8.3-2)

Gas withdrawn from containment is passed through HEPA and charcoal filters and then gas dryers which are needed to remove essentially all moisture. The gas then passes through tanks of activated charcoal in series which absorb the Kr-85. Once "break-through" occurs, the tanks are isolated and are used for storing the Kr-85 at ambient temperature and atmospheric pressure. Charcoal loses its ability to absorb krypton when it is exposed to significant humidity, i.e., in excess of about 3 percent.

The total required charcoal weight is 34,000 tons (this represents approximately 40 percent of the total U. S. annual production). The charcoal volume is 2,000,000 cubic feet, which is equivalent to the TMI-2 containment volume.

Storage tanks, rather than piping, are used to facilitate initial loading of the charcoal. A manhole would be required at the top of each tank for loading, and a second manhole would be required at the bottom of the tank for eventual disposal of the charcoal. Each tank would be provided with isolation valves, primarily for humidity control during filling operations. The valves would also be closed once containment cleanup operations were complete.

The size of tank selected was based on fabricating the tanks in a shop and shipping them to the site. Twelve-foot diameter and 50-foot length represent about the upper limit to shop fabricated tanks.

Four hundred and fifty such tanks would be required. Each would be an atmospheric tank, designed in accordance with Section III, Subsection ND, Class 3 components. The Code requires a minimum wall thickness for such tanks of 3/16 inch. The total tank metal weight would be 6,100,000 pounds.

The building required to house the tanks (see Figure 8.3-2) would be 700 feet long, 150 feet wide, and 60 feet high. The charcoal provides significant self-shielding so that shielding is not considered necessary for the tanks.

8.3.2 Design Alternates Considered

Several alternates were considered for the charcoal adsorption system, and were rejected. In summary:

a. Operation at Lower Temperature

- Summary description of alternate -- Operate the charcoal at a lower temperature to increase its adsorption capability and, accordingly, decrease the required amount of charcoal. A number of BWRs, for example, employ systems which operate at 0°F. At this temperature, the absorption capability is about 2.5 times greater than at 70°F.

- Basis for rejecting the alternate -- The required refrigeration equipment increases system complexity. Malfunction of the equipment could cause an increase in charcoal temperature and therefore cause an uncontrolled release of Kr-85. The disadvantages are considered to outweigh the advantage of decreased charcoal volume, particularly since the required amount of charcoal with a refrigerated system would still be very large (about 15 percent of the total U. S. yearly production).

b. Regenerate the Charcoal and Store Gas Enriched in Krypton

- Summary description of alternate -- Employ two parallel trains of charcoal, each with a few days holdup time for krypton. Process with one train until "break-through" occurs, while regenerating the alternate train. Store the regeneration product gas, which is enriched in krypton, using separate storage vessels. Potential regeneration techniques which are under development include (1) cycle the bed temperature, and (2) cycle the bed pressure.

- Basis for rejecting the alternate -- Laboratory scale tests show that such a system is potentially feasible, particularly temperature cycling systems which operate at cryogenic temperatures. However, such systems have not been employed for large scale applications, so that further engineering and development would be required. Accordingly, such systems are not considered practical for near term use at TMI-2.

8.3.3 Cost and Schedule Estimate

- Cost Range (for component procurement, installation, building erection and materials, design and analysis, testing and checkout, charcoal, contingency)

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$120,000,000 to $160,000,000
(Note: More than $60,000,000 of this cost is the charcoal itself.)

- Schedule Range (for building and equipment design, procurement, erection, installation, testing)

  30 months to 40 months
  (Note: This schedule presumes charcoal would be available as required. A national commitment of U. S. production capacity would be required for this.)

8.3.4 System Evaluation

- The charcoal adsorption system achieves full treatment and storage of the reactor building atmophere with zero radioactive release assuming no equipment failures or operator errors.

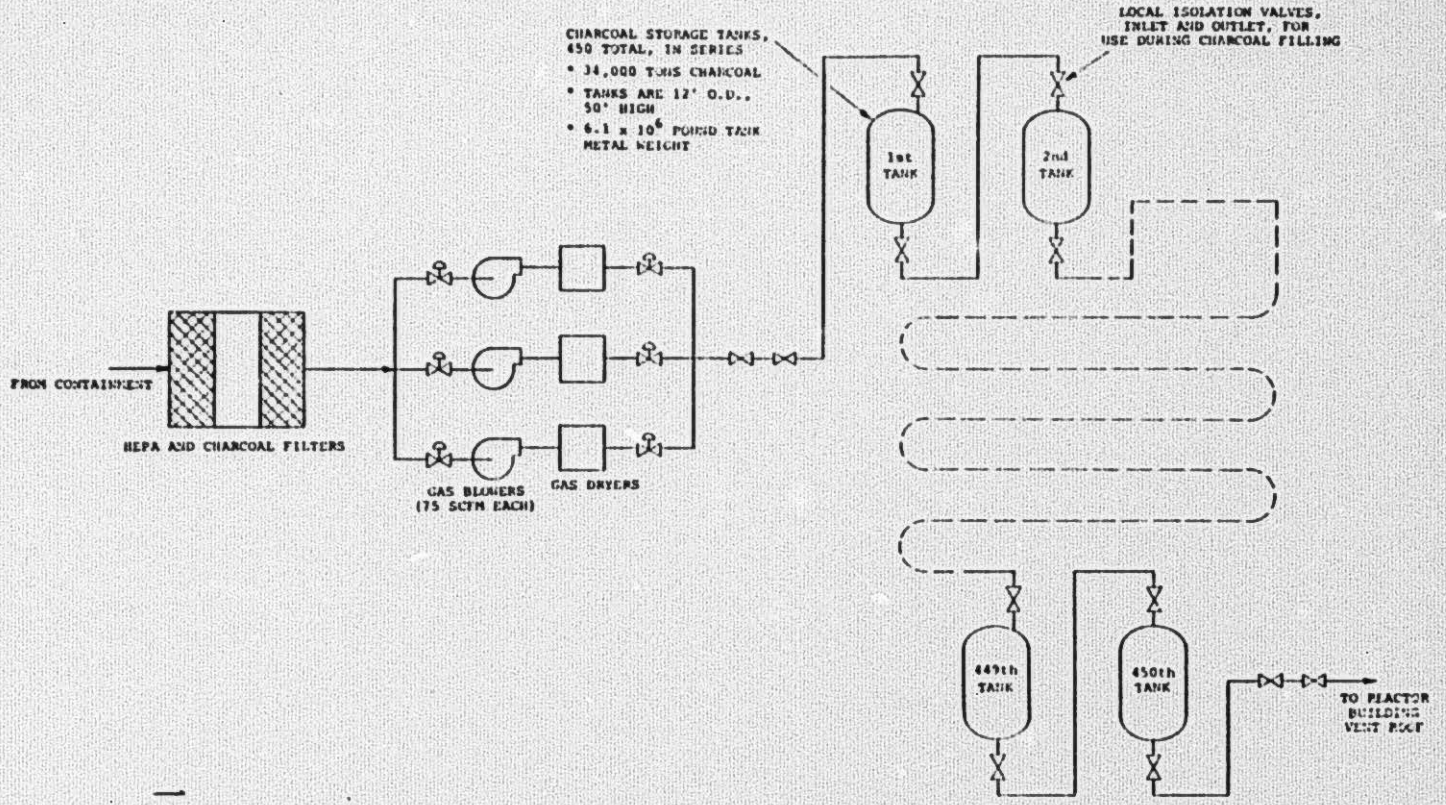
- This system is vulnerable to uncontrolled release during processing and long-term storage. In particular, charcoal loses its adsorption capability when exposed to moisture. Accordingly, gas dryer malfunction during processing, or contact of the charcoal by humid air during storage could result in inadvertent krypton release. Potential for fire also exists with charcoal, and could result in an uncontrolled release of krypton.

- Use of a charcoal system does not resolve the problem of ultimate disposal of Kr-85. Long-term storage for more than one hundred years, and off-site shipment are each considered less safe than controlled release of the Kr-85 by purging containment. Off-site shipment would be particularly impractical for this system because of the large volume of material.

- The extensive time required to build and install the charcoal adsorption system would increase the likelihood of inadvertent and uncontrolled leakage from the existing containment building, and thereby cause higher exposure to personnel. This extensive time delay to complete system installation would also delay TMI-2 cleanup operations. Finally, the cost of the charcoal adsorption system is high and no commensurate benefits are received.

8.3.5 Charcoal Adsorption System Conclusions

- In summary, when compared to controlled purging of the containment building, the alternate charcoal absorption system is considered to be less safe -- it is less reliable, and clearly has the potential for uncontrolled releases of radioactivity with higher radiation exposures.



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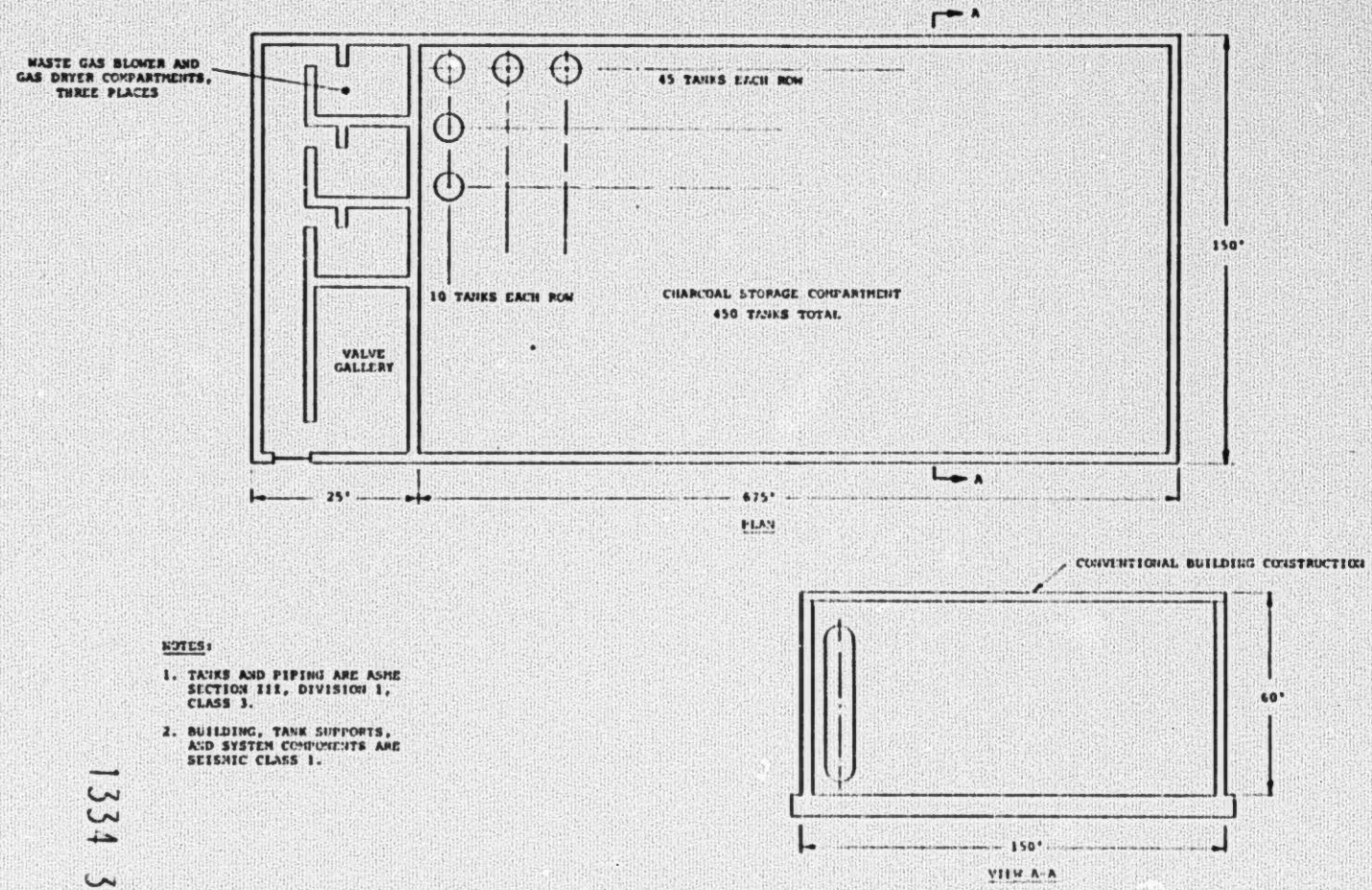
TMI-2
CONTAINMENT ATMOSPHERE CHARCOAL ADSORPTION
CONCEPTUAL SYSTEM DESIGN

FIGURE 8.3-1



NOTES:

1. TANKS AND PIPING ARE ASME SECTION III, DIVISION 1, CLASS 3.
2. BUILDING, TANK SUPPORTS, AND SYSTEM COMPONENTS ARE SEISMIC CLASS 1.

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CONCEPTUAL LAYOUT
CHARCOAL STORAGE ARRANGEMENT

FIGURE H.3-2

8.4 Gas Compression and Storage System

The second alternative considered for reducing the airborne activity in the reactor building is to draw off the reactor building atmosphere into a pressurized storage container so that this entire building atmosphere including the radioactive noble gases remains in pressurized storage indefinitely. The total volume to be stored is 23 million cubic feet.

8.4.1 System Description (See Figures 8.4-1, 8.4-2, and 8.4-3)

Gas is withdrawn from containment using three compressors with a total capacity of 225 scfm. This permits containment cleanup in 71 days if the system operates with no malfunctions and if the total process volume does not exceed 23 million cubic feet.

The gas passes first through HEPA and charcoal filters which are provided to remove any particulate radioactivity and minimize contamination of downstream components. Such filters would be required for each of the containment cleanup systems including the purge system. Accordingly, the cost and schedule associated with these filters were excluded from the evaluation.

The storage container for the compressed gas is 36-inch O.D. standard wall (0.375-inch thick) carbon steel piping.

The design pressure for this piping is about 340 psig in accordance with the ASME Code, Section III, Subsection ND. At this pressure, a total pipe volume of 1,000,000 cubic feet is required for storage of the processed gas. The total required length of 36-inch piping is 150,000 feet. The pipe weight is 21,000,000 pounds.

The piping is divided into two major sections to minimize shielding. The high activity piping section includes 20 percent of the piping and contains 90 percent of the krypton-85. Six inches of concrete shielding are required. The high activity section is subdivided into five units to (1) ensure that the highest activity piping is at the center of the building (see Figure 8.4-2) and, accordingly, is shielded by outer piping; and (2) minimize the amount of uncontrolled Kr-85 release in the event of leakage. The building which houses the high activity piping and the gas compressors is 260 feet long, 90 feet wide, and 30 feet high.

A low activity pipe section contains 80 percent of the total piping and 10 percent of the krypton-85. No shielding is required for this piping. The building which houses the low activity piping is 220 feet long, 160 feet wide, and 60 feet high.

8.4.2 Design Alternates Considered

Various alternates were considered for the gas compression system, and were rejected. In summary:

a. Storage in Higher Pressure Piping

- Summary description of alternate -- Store the containment atmosphere in high pressure piping, in order to reduce the

total storage volume. For example, thick-walled (1.0 inch) 36-inch piping would permit storage at 1,070 psig, and reduce the storage volume (or total pipe length) by a factor of three.

- Basis for rejecting the alternate

  — The total pipe weight is not reduced by this alternate; the reduction in pipe length by three is balanced by the increase in wall thickness. Accordingly, pipe procurement costs would not be reduced.

  — Standard wall piping is the most readily available. Accordingly, use of thick wall piping would increase construction time.

  — The likelihood of uncontrolled leakage, e.g., through system valves, is increased at higher pressure.

b. Use of a Single Large Storage Container

- Summary description of alternate -- Employ a large container instead of piping. For example, a vessel with 2×10^6 ft^3 volume (equal to the TMI-2 containment volume) could contain the processed volume at a pressure of about 170 psig. The required wall thickness for such a vessel fabricated of carbon steel would exceed 8 inches.

- Basis for rejecting the alternate -- Such an alternate would likely be significantly more costly and take longer to construct than a system which employs standard wall piping.

c. Use of Standard High Pressure Gas Storage Bottles

- Summary description of alternate -- Employ standard 2,500 psig vessels which are used for storage and transport of commercial gas (e.g., O_2 and H_2).

- Basis for rejecting the alternate -- More than 80,000 such vessels would be required. The pipe and valve system for filling these bottles would be very complex and, accordingly, the likelihood of uncontrolled leakage would be significantly increased.

8.4.3 Cost and Schedule Estimate

- Cost Range (for component procurement, installation, building erection and materials, design and analysis, testing and checkout, contingency)

  $50,000,000 to $75,000,000

- Schedule Range (for building and equipment design, procurement, erection, installation, testing)

  25 months to 35 months

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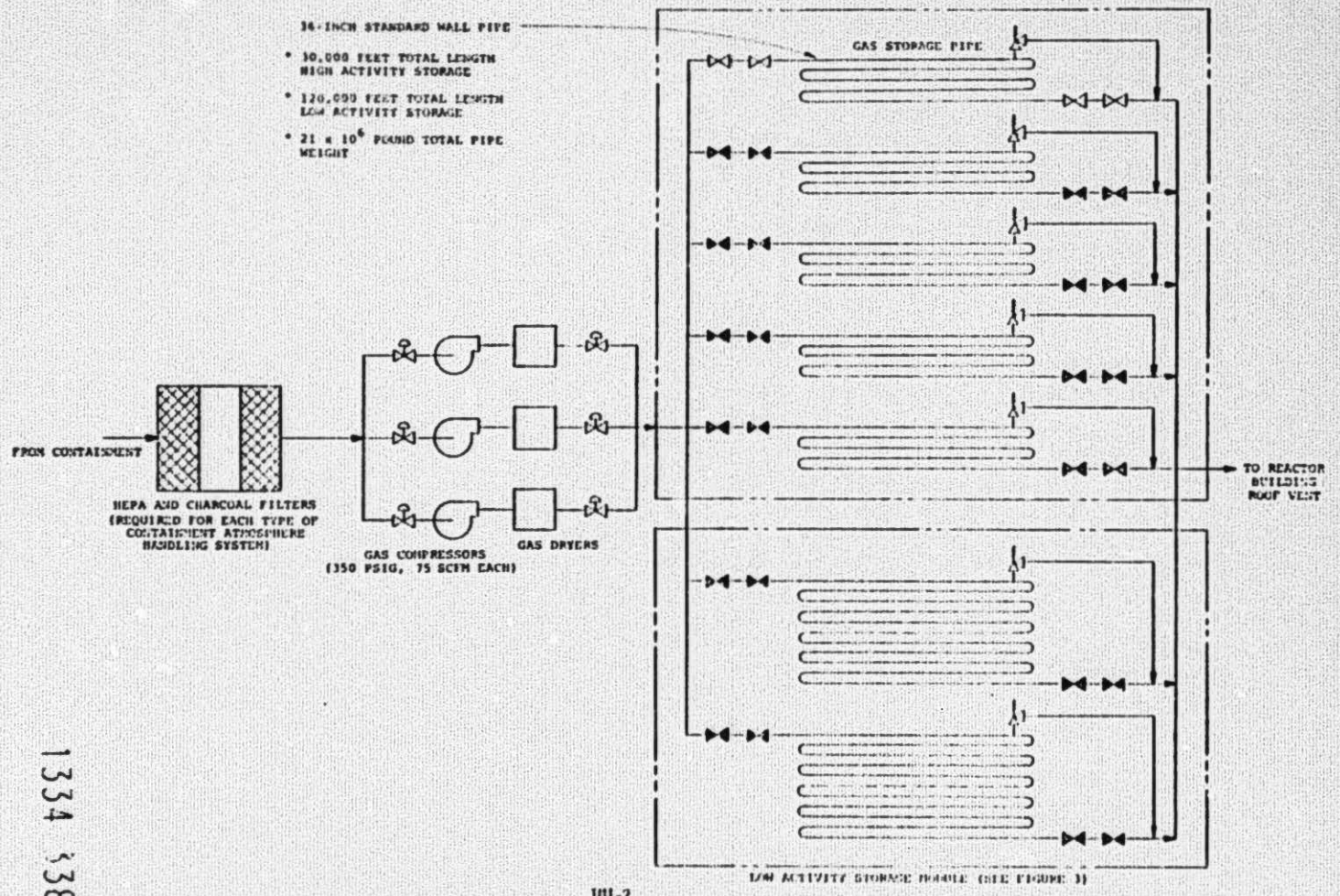

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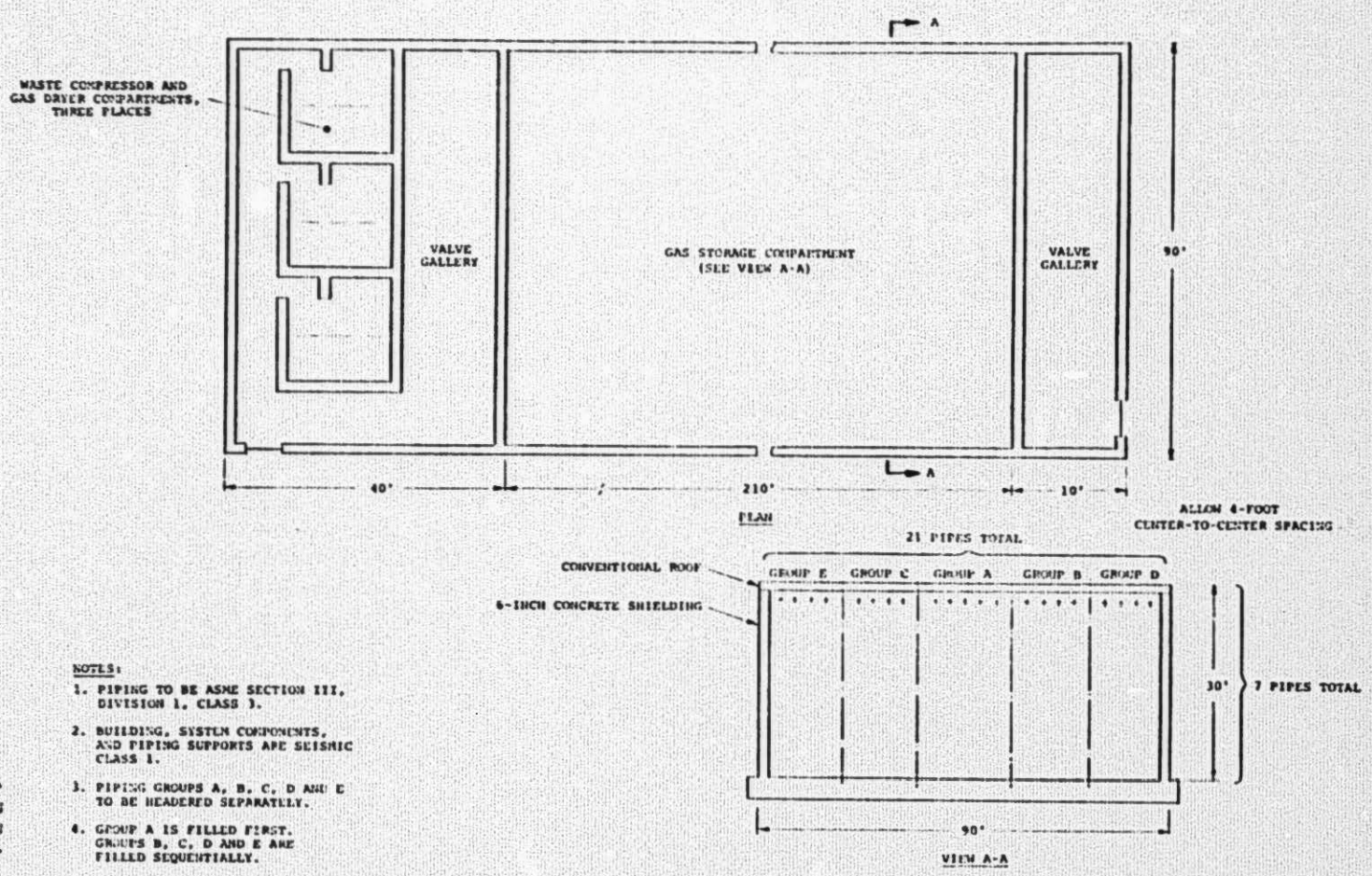
III-2.

CONTAINMENT ATMOSPHERE GAS COMPRESSION CONCEPTUAL SYSTEM DESIGN

FIGURE 8.4-1

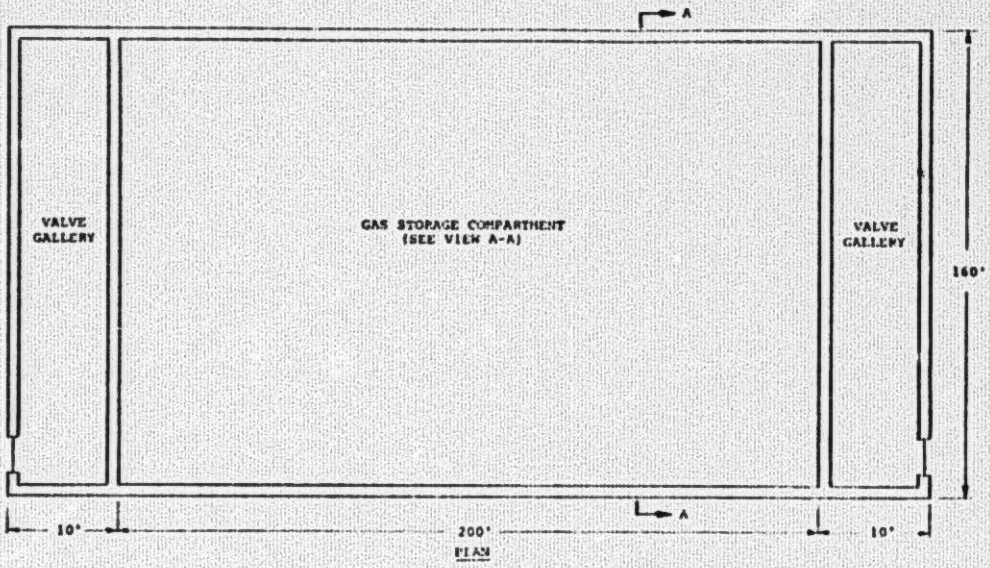


NOTES:

1. PIPING TO BE ASME SECTION III, DIVISION 1, CLASS 3.
2. BUILDING, SYSTEM COMPONENTS, AND PIPING SUPPORTS ARE SEISMIC CLASS 1.
3. PIPING GROUPS A, B, C, D AND E TO BE HEADERED SEPARATELY.
4. GROUP A IS FILLED FIRST. GROUPS B, C, D AND E ARE FILLED SEQUENTIALLY.

CONCEPTUAL LAYOUT
HIGH ACTIVITY STORAGE ARRANGEMENT

FIGURE 8.4-2



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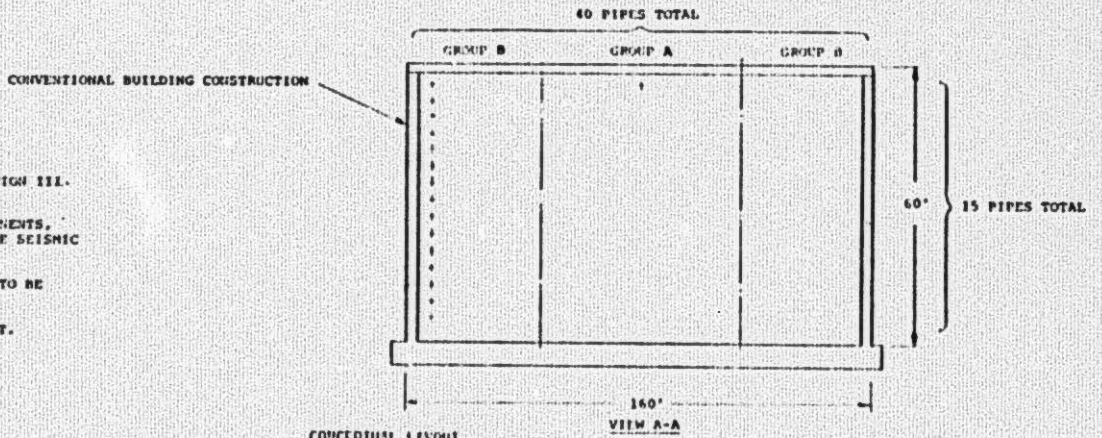

VIEW A-A

NOTES:

1. PIPING TO BE ASME SECTION III. DIVISION 1, CLASS 3.
2. BUILDING, SYSTEM COMPONENTS, AND PIPING SUPPORTS ARE SEISMIC CLASS 1.
3. PIPING GROUPS A AND B TO BE HEADERED SEPARATELY.
4. GROUP A IS FILLED FIRST.

CONCEPTUAL LAYOUT
LOW ACTIVITY STORAGE ARRANGEMENT
FIGURE 8.4-3

8.4.5 Gas Compression System Conclusions

- When compared to controlled purging of the containment building, the alternate gas compression system is considered to be less safe -- it is less reliable and clearly has the potential for uncontrolled releases of radioactivity with higher radiation exposures.

8.5 Cryogenic Processing and Storage System

The third alternative considered for reducing the airborne activity in the reactor building (primarily due to Krypton-85) is to draw off the reactor building atmosphere into a cryogenic processing system. This system would separate the noble gases from the remaining gases cryogenically and the noble gases containing Krypton would be stored indefinitely in highly concentrated form. The total volume to be processed through the system is 23 million cubic feet.

Gas removed from containment passes through a cryogenic treatment system where most krypton is removed. The purified gas is discharged from the plant. (Note: The cryogenic unit effluent gas flow rate is greater than the input flow rate, because liquid nitrogen used for cooling vaporizes in the cryogenic units. The effluent must be discharged rather than recycled to containment in order to prevent containment pressure buildup.) Liquid krypton, xenon, argon, and methane are periodically withdrawn from the system, allowed to vaporize, and are stored at ambient temperature in storage vessels.

8.5.1 System Description (See Figures 8.5-1, 8.5-2, 8.5-3, and 8.5-4)

Gas withdrawn from containment is passed through HEPA and charcoal filters and then through the various remaining components which are shown in Figures 8.5-1 and 8.5-2. It is noted that all of the components shown in Figures 8.5-1 and 8.5-2, except for the catalytic recombiners and their associated preheaters and aftercoolers, are part of an existing system at a new BWR. The system has not been placed in operation. It is being scrapped and replaced by a new conventional type of charcoal system. We understand the major reasons for this decision by the utility which currently owns the system are:

- Lifetime costs of the cryogenic system, including the cost of hydrogen and liquid nitrogen supply, maintenance, and operation, were considered to likely exceed the cost of a charcoal system.

- The cryogenic system was considered ill-suited for transient operations. For example, its ability to respond to sudden changes in input flow rate, such as could be caused by opening a vent valve to the main condenser, is questionable.

- No significant operating experience is available on a cryogenic system. It was considered likely that operation of this system would be significantly less reliable than a charcoal system because it contains many more valves, instruments, and other active components than a charcoal system.

However, this is one of the cryogenic systems which could be made available in a reasonable period of time, and was therefore chosen as a typical cryogenic system for this evaluation.

The cryogenic system consists of three separate trains. The input flow rate is 75 scfm per train. After removal of oxygen by the recombiners, the flow rate is 62 scfm. The effluent flow rate is 103 scfm, higher than the input, because some of the liquid nitrogen used for cooling the cryogenic units is vaporized in the units.

The purified gas is discharged from the site via the reactor building roof vent. The cryogenic system can remove 99.9 percent of the krypton from the input gas in accordance with the original equipment specification. The quantity of Kr-85 discharged is approximately 60 curies.

Shop tests were performed to establish the purification efficiency of the cryogenic units. These tests showed a removal efficiency by krypton greater than the value of 99.9 percent required by the equipment specification. However, the test was not performed under actual operating conditions. For example, pure nitrogen was employed for the process gas rather than a gas containing moisture, carbon dioxide and argon in addition to nitrogen. Also, the bottom of the removal column (the "reboiler section") contained a mixture of nitrogen and krypton rather than a mixture of argon, methane, and krypton. Accordingly, it is considered that testing under actual operating conditions would be required to prove out this system.

Liquid krypton and xenon are removed from the cryogenic unit when they reach a concentration of 20 percent in the bottom of the removal column. The remaining 80 percent consists of argon (76 percent) and methane (4 percent). The liquified gases are vaporized and stored at ambient temperature.

The volume of stored gas would be about 800 standard cubic feet. This estimate is based on a concentration of Kr-85 within containment of 1 μCi/ml, which for a containment volume of 2,000,000 cubic feet, amounts to about 60,000 curies of Kr-85. This is about 60 percent of the total fuel inventory of Kr-85 expected after 90 days of operation. The total inventory of noble gas expected after 90 days of operation is 256 standard cubic feet. The estimate of 800 standard cubic feet total is, accordingly, the volume corresponding to about 60 percent of the total noble gas volume in a 20 percent rich mixture of noble gas with argon and methane.

Many different components are employed for this system (see Figures 8.5-1 and 8.5-2). The function of each component is summarized as follows:

- Catalytic recombiners (Figure 8.5-1) -- Remove oxygen from the incoming air to prevent ozone buildup from irradiation of oxygen. Ozone in contact with light hydrocarbons, e.g., CH_4, can detonate. (Note: The system supplier indicated this is a concern for operating BWR applications. He would have to perform an evaluation to determine if catalytic recombiners are required for the TMI-2 application.)

- Hydrogen storage vessels (Figure 8.5-1) -- Provide hydrogen to the catalytic recombiners, 8,000,000 scf total required.

- Liquid nitrogen storage vessels (Figure 8.5-1) -- Provide liquid nitrogen for cooling the cryogenic units, 150,000 gallons total required.

- Krypton and xenon storage vessels (Figure 8.5-1) -- Store the Kr-85.

- Storage secondary container (Figure 8.5-1) -- Prevent Kr-85 release in the event of storage vessel, piping, or valve failure.

- Cryogenic unit feed compressors (Figure 8.5-1) -- Provide the required gas flow.

- Trace recombiners (Figure 8.5-2) -- Remove trace quantities of oxygen (up to 0.5 percent by volume) which may be present in the inlet gas.

- Prepurifiers (Figure 8.5-2) -- Remove water vapor and carbon dioxide from the gas stream to prevent plugging of the cryogenic columns.

- Cooldown heat exchanger (Figure 8.5-2) -- Reduce temperature of inlet gas (to -292°F) and increase temperature of outlet gas (to -40°F) and hydrogen.

- Removal column (Figure 8.5-2) -- Remove krypton from input gas (also Xe, A, and CH_4) by counter flow of liquid nitrogen (at -307°F) and the inlet gas.

- Condenser heat exchanger (Figure 8.5-2) -- Liquify gas output from the removal column.

- Phase separator (Figure 8.5-2) -- Remove excess hydrogen for recycle to the catalytic recombiners.

- Decay column (Figure 8.5-2) -- Provide three-hour decay time of the effluent gas before it is released.

- Cold box (Figure 8.5-2) -- Contain all components which operate at cryogenic temperatures to prevent uncontrolled Kr-85 release in the event of equipment malfunctions.

- Ambient heater (Figure 8.5-2) -- Heat up effluent gas to the temperature required for prepurifier bed regeneration (+330°F for H_2O and CO_2 removal).

Figure 8.5-3 shows a conceptual design of the secondary storage container for the krypton storage vessels. This is, in effect, a small size containment vessel with two-foot thick reinforced concrete walls, a stainless steel liner, typical piping penetrations with double isolation valves for the inlet and outlet headers, and a four-foot diameter equipment hatch for storage vessel installation. It is designed for an internal pressure of 20 psig to withstand the pressure resulting from failure of all the storage vessels. (This peak pressure would be 16 psig.)

Figure 8.5-4 shows the building arrangement for the system equipment. It corresponds to the arrangement of the existing system, with minor modifications to:

- Incorporate the secondary storage container,

- Provide space for catalytic recombiner equipment, and

- Provide for above grade construction rather than below grade construction which was employed at the existing facility.

Two feet of concrete shielding are required for the product storage vessels and the cold box components.

8.5.2 Design Alternates Considered

No alternate designs were considered because, as described above, the evaluation is based on an existing design which is currently available. (Note that even though this is an existing system, there is no actual operating experience with this system, or with similar equipment at any commercial light water power reactor.)

8.5.3 Cost and Schedule Estimates

- Cost Range (for component procurement, installation, building erection and materials, design and analysis, testing and checkout, utilities, contingency)

  $10,000,000 to $15,000,000
  (Note: This cost presumes that the cryogenic units will be available as surplus at a small fraction of their original cost or of the cost of new equipment.)

- Schedule Range (for building and equipment design, procurement, erection, installation, testing)

  20 months to 30 months

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8.5.4 System Evaluation

- This system is less costly and would require less time to install than the gas compression or charcoal systems. It is considered, however, to be the least safe and most unreliable of any of the alternate systems evaluated for a number of reasons.

  -- The system produces highly concentrated Kr-85. Any leakage or component failure could result in significantly greater amounts of uncontrolled radioactivity release than the other systems. Also, the radiation levels of the equipment and, accordingly, the exposure of plant personnel during maintenance and operation would be higher than for the other systems.

  -- The system is subject to plugging as a result of component malfunctions which result in inadequate moisture or carbon dioxide removal. More than 100 automatic valves are used for prepurifier regeneration cycle control, and must function correctly. Electrical power supplies to the ambient heaters (for the final prepurifier regeneration step) and cooling water supplies to upstream aftercoolers must also work. It is considered likely that upsets will occur. In the event of plugging, the system must be thawed and purged to return to operation. The likelihood of uncontrolled Kr-85 release during such an off-standard operation is considered significant.

  -- Unless further analyses by the equipment supplier prove otherwise, catalytic recombiners would be required for system operation. Recombiners have been unreliable at many BWRs. For example, less than one week of continued operation of new off-gas systems has been accomplished in the last several years as a result of difficulties with the recombiner system at several operating BWRs.

  -- The system operates with excess hydrogen to ensure all oxygen is removed. Hydrogen leakage, e.g., from the hydrogen recycle circuit valves, could result in hydrogen burning or detonation.

  -- Packed type valves, rather than diaphragm valves, are used throughout the system. The system operates at a pressure of about 85 psig. Accordingly, leakage will likely occur. An alternate would be to replace all valves, but this would require extensive refurbishment.

- This system was designed to remove greater than 99.9 percent of the noble gas activity from the input stream. Accordingly, some off-site release would occur even if this system functioned properly, i.e., this is not a zero release system.

Further, the specific removal efficiency which could be achieved has not been demonstrated. A shop test was performed, but not under actual operating conditions. For example, pure nitrogen was employed for the process gas, rather than a gas containing moisture, carbon dioxide and argon in addition to nitrogen. Also, the bottom of the removal column contained a mixture of nitrogen and krypton, rather than a mixture of argon, methane, and krypton. Accordingly, it is considered that testing under actual operating conditions would be required to prove out this system before considering using it at TMI-2.

- There is no significant operating experience with a cryogenic distillation system at any operating light water reactor. Accordingly, this is not a proven technology for reactor application.

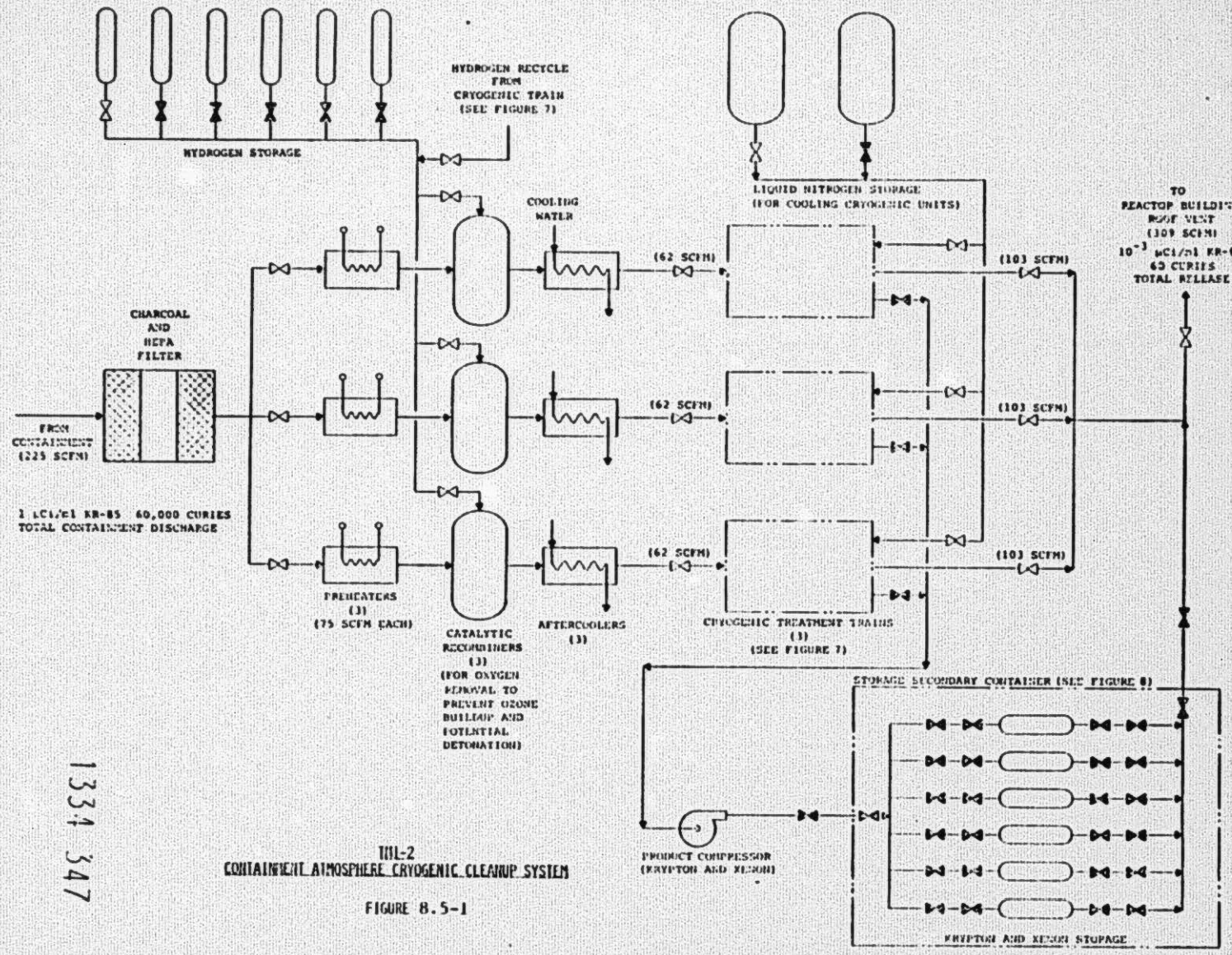
- As with the other systems, use of a cryogenic system does not resolve the problem of ultimate disposal of Kr-85. Long-term storage for over one hundred years and off-site shipment are considered particularly undesirable for this system due to the highly concentrated form of the Kr-85. Venting of the Kr-85 over a long period of time, as with the other systems, only varies the rate of personnel exposure - not the total exposure - and increases the likelihood of an uncontrolled release.

- Even though the major process components for this system are available at an existing facility, the time required to achieve system operation would not be significantly less than for the gas compression system. In particular, time is required to design and erect the building which houses the system, provide required utilities hookups, provide interconnecting piping for various system components, and to test the system.

- The extensive time required to build and install the cryogenic treatment system would increase the likelihood of inadvertent and uncontroiled leakage from the existing containment building, and thereby cause higher exposure to personnel. This extensive time delay to complete system installation would also delay TMI-2 cleanup operations. Finally, the cost of the cryogenic treatment system is high and no commensurate benefits are received.

8.5.5 Cryogenic Processing System Conclusions

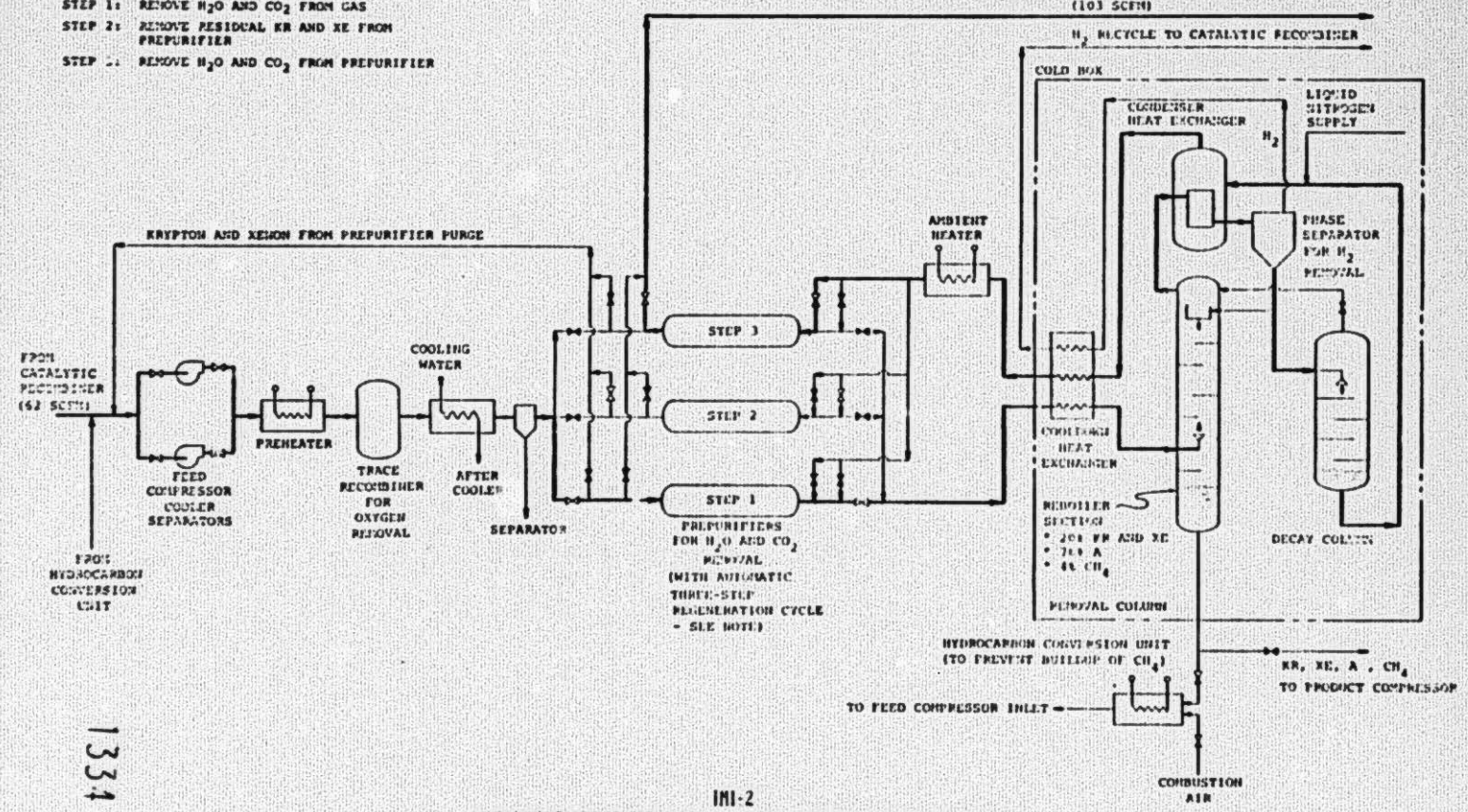
- When compared to controlled purging of the containment building, the alternate cryogenic treatment system is considered to be less safe -- it is less reliable, and clearly has the potential for uncontrolled releases of radioactivity with higher radiation exposures.



TBL-2
CONTAINMENT ATMOSPHERE CRYOGENIC CLEANUP SYSTEM

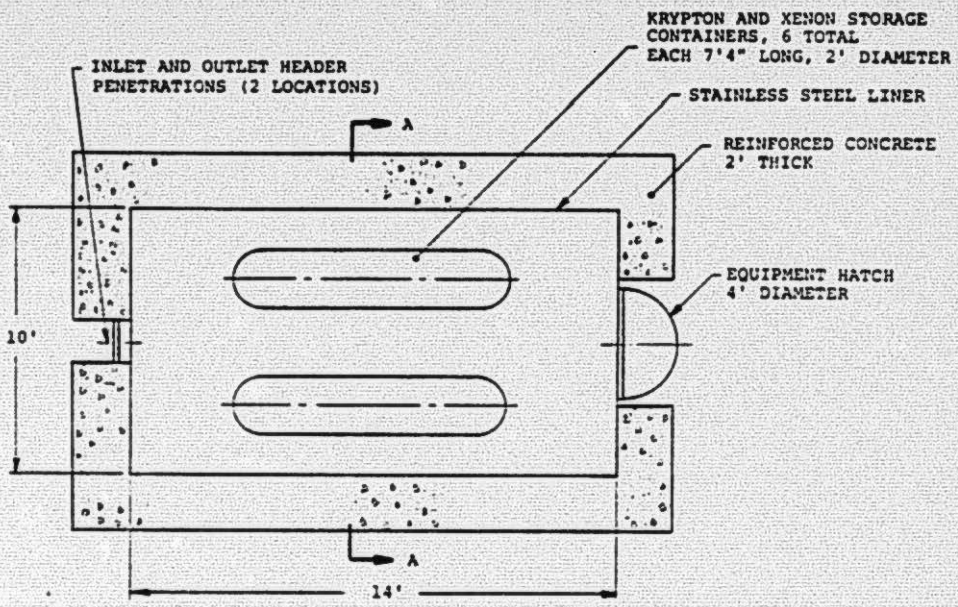
FIGURE 8.5-1

NOTE:
PREPURIFIER REGENERATION CYCLE STEPS ARE:
STEP 1: REMOVE H_2O AND CO_2 FROM GAS
STEP 2: REMOVE RESIDUAL KR AND XE FROM PREPURIFIER
STEP 3: REMOVE H_2O AND CO_2 FROM PREPURIFIER



IMI-2
CONTAINMENT ATMOSPHERE CLEANUP SYSTEM CRYOGENIC TREATMENT TRAIN
(ONE OF THREE)

FIGURE 8.5-2



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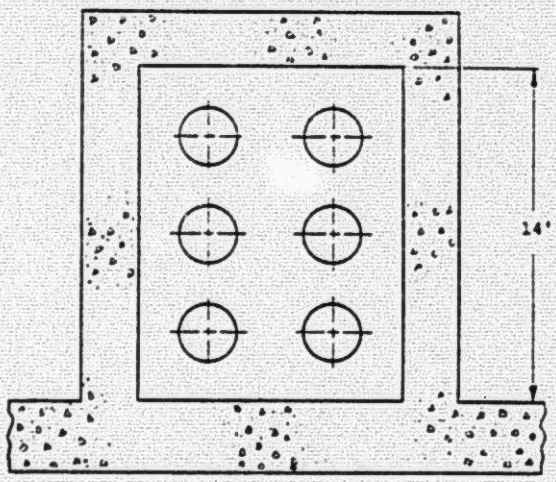

VIEW A-A

NOTES:

- SECONDARY CONTAINER TO BE SEISMIC CLASS 1.
- INTERNAL DESIGN PRESSURE = 20 PSIG

TMI-2
CRYOGENIC TREATMENT SYSTEM
KRYPTON STORAGE SECONDARY CONTAINER
CONCEPTUAL ARRANGEMENT

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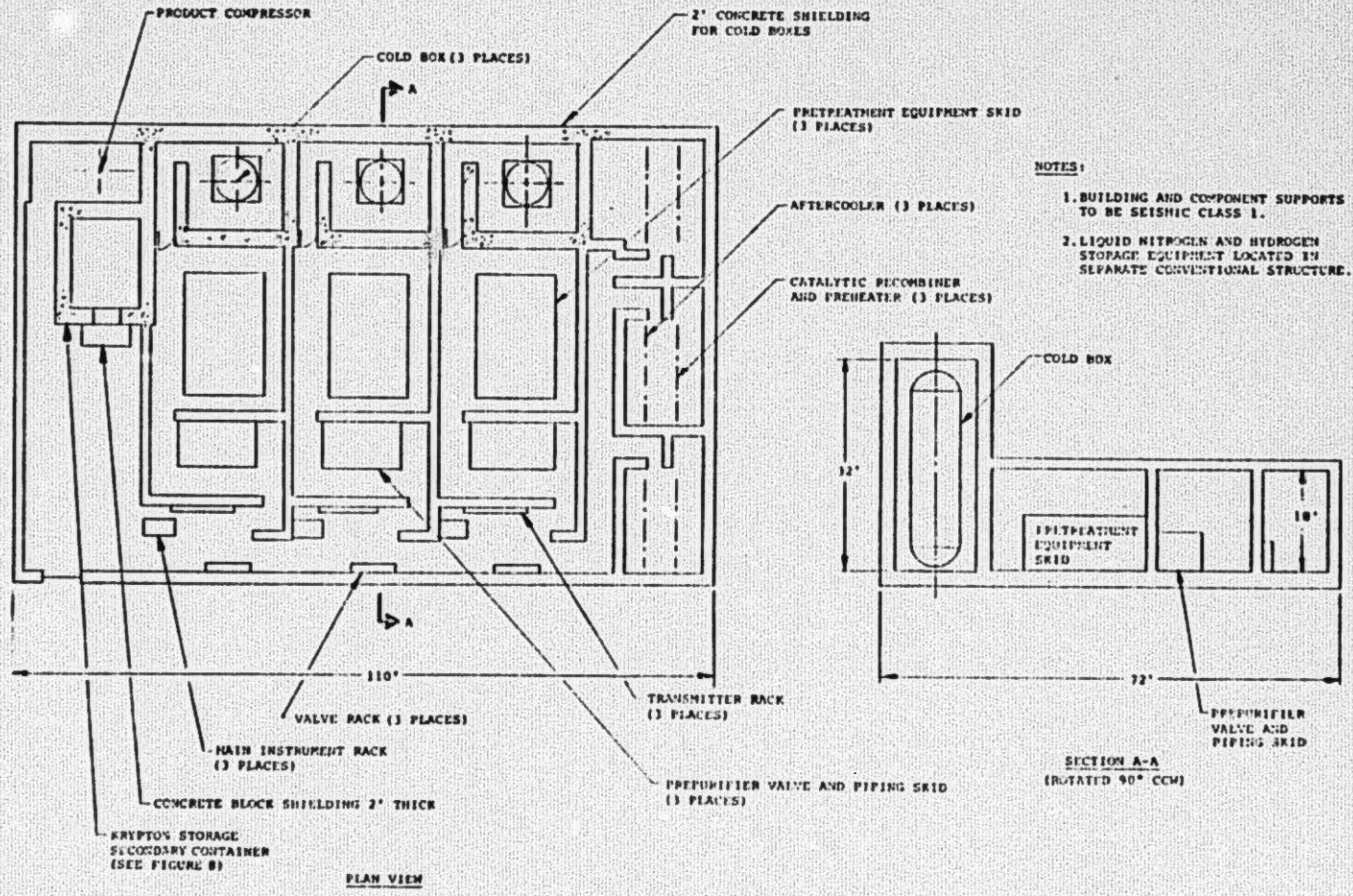
FIGURE 8.5-3



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TMI-2
CRYOGENIC TREATMENT SYSTEM
BUILDING AND EQUIPMENT LAYOUT

FIGURE 8.5-4

8.6 Environmental Effects of Alternates

The radiological impact on the environment of each alternate considered is evaluated in terms of normal operation and accident conditions. Off-site exposure is calculated for each condition. In addition, on-site dose is evaluated for each alternate considering maintenance and surveillance during processing and storage.

8.6.1 Normal Operation

Since each system is designed to collect and retain the noble gas, no substantial effect is expected due to normal operation. Because of the schedule delay associated with each, the alternatives to direct controlled reactor building purging carry with them the potential for accidental uncontrolled release of noble gases from the reactor building.

A schedule delay for each alternative has been calculated to be in the range of 2 to 4 years. Likelihood of system leakage over this period is high enough that with 10% of the noble gases leaking during worst case meteorology, the off-site dose due to any of the alternatives would be 10 times as great as the 5.0 mrem boundary beta dose calculated for the controlled purge of the entire noble gas volume.

An evaluation of possible pressure buildup in the TMI-2 containment has been made, since such a pressure buildup could be the driving force for leakage of noble gas from the containment. This evaluation is based on the following assumptions:

(a) The electrical air circulation fans within containment fail either due to failure of their motors or failures in their associated electrical circuitry. (Note: The containment temperature is currently being held at approximately 100°F with the fans running.)

(b) The steam generator still continues to remove the bulk of the primary system's decay heat and the only heat transferred from the primary system to the containment atmosphere is the heat losses through the primary system insulation.

(c) Solar heat is an additional source of heat to the containment buiding.

Using the above described assumptions, the evaluation indicates that containment pressure can rise to between 1 and 2 psig. This positive containment pressure would be the driving force for emitting the noble gases out of any defects in the containment boundary.

A study was performed to determine the impact of a 1 to 2 psig containment pressure and various size containment leaks, to establish

the potential off-site doses that could result. As background information, the present containment design is based on an allowable leakage of 0.2 percent per day under a design pressure of 60 psi. This allowable design leakage is equivalent to having an 0.13-inch diameter hole in the containment. Taking this design basis hole size, the doses resulting from the leakage through it with a containment pressure of 1 to 2 psig can be determined. Further, an assessment of the effect of increased leakage due to additional holes of 1/8-inch, 1/4-inch, and 1/2-inch diameter that could result from seal deterioration, cracks, corrosion, etc. has been made. The total off-site dose due to leakage caused by 1 to 2 psig pressure in containment is as follows:

Containment Condition	Off-Site Dose During a One-Day Period, mrem (1)		Off-Site Dose During a 30-Day Period, mrem (1)	
	D_F	D_I	D_A	D_T
Design Basis (DB) leak (equivalent to 0.13-inch hole)	1-5	0.01-0.04	4-19	0.04-0.16
DB + 1/8-inch hole	2-9	0.02-0.08	8-34	0.07-0.29
DB + 1/4-inch hole	5-23	0.04-0.19	20-84	0.17-0.71
DB + 1/2-inch hole	18-76	0.15-0.64	66-280	0.55-2.37

(1) The high dose numbers are based on using NRC meteorological parameters in Regulatory Guide 1.4, while the lower doses are based on using more realistic meteorology from the TMI-2 FSAR.

As can be seen from this table, the doses from a leaking containment, even if only a design basis leak exists, are greater than the total 5 mrem off-site dose calculated for the entire controlled purging operation of the containment.

8.6.1.1 Charcoal Adsorption and Storage System

The charcoal adsorption system is designed for full noble gas retention on charcoal beds and therefore no off-site dose is calculated, assuming no operator error or equipment failures. The on-site whole body dose due to maintenance and surveillance during processing and storage is calculated to be 23 person-rem.

8.6.1.2 Gas Compression and Storage System

The gas compression system is designed for full retention of the reactor building atmosphere and therefore no off-site dose is

calculated assuming no operator errors or equipment failure. The on-site whole body dose due to maintenance and surveillance during processing and storage is calculated to be 58 person-rem.

8.6.1.3 Cryogenic Processing and Storage System

The design basis for the cryogenic system is to achieve 99.9% noble gas removal. This results in 0.1% release of the Kr-85 to the environment. Based on the analysis of the off-site dose for controlled purging of the reactor building, assuming ground level release and average meteorological conditons the site boundary beta skin dose is estimated to be 0.05 mrem. The on-site whole body dose due to maintenance and surveillance during processing and storage is calculated to be 570 person-rem.

8.6.2 Accident Conditions

Regulatory Guide 1.24 "Assumptions Used For Evaluating the Potential Consequences of a Pressurized Water Reactor Building Gas Storage Tank Failure" specifies that release of the entire contents of a single storage tank is to be postulated to occur over a two hour period for accident analysis.

The atmospheric dispersion model used in this analysis is the same as that described in Section 7.2 in accordance with Regulatory Guide 1.145. The calculated accident X/Q value for each accident analyzed below is 6.8×10^{-4} sec/m^3.

8.6.2.1 Charcoal Adsorption and Storage System

The noble gas is stored in 450 charcoal tanks with successively decreasing noble gas activity in each tank as the withdrawn activity in the reactor building decreases due to the feed and bleed process. The highest activity tank contains 1430 curies. Using the same method as used in Section 7.3, the site boundary cloud concentration becomes:

$$\frac{\text{Curies}}{\text{M}^3} = \frac{1430}{2 \text{ hrs.} \times 60 \times 60} \times 6.8 \times 10^{-4} \text{ sec/m3}$$

$$= 1.35 \times 10^{-4} \text{ curies/m}^3$$

The resulting site boundary doses are:

Beta Air Dose = 150 mrads, Beta Skin Dose = 64 mrems
Gamma Air Dose = 1.3 mrads, Gamma Skin Dose = 1.4 mrems
Whole Body Dose = 1.2 mrems

8.6.2.2 Gas Compression and Storage System

For the gas compression and storage system, 42% of the noble gas activity is stored in the high activity storage volume. Using

the same method as used in Section 7.3, the site boundary cloud concentration becomes:

$$\frac{\text{Curies}}{M^3} = \frac{23900}{2 \times 60 \times 60} \times 6.8 \times 10^{-4} \text{ sec/m}^3$$

$$= 2.26 \times 10^{-3} \text{ Ci/m}^3$$

The resulting site boundary doses are:

Beta Air Dose = 2510 mrads, Beta skin dose = 1730 mrems
Gamma Air Dose = 21.2 mrads, Gamma skin dose = 24 mrems
Whole Body Dose = 20.7 mrems

8.6.2.3 Cryogenic Processing and Storage System

For the cryogenic processing and storage system, the entire concentrated noble gas volume is stored in a single compartment. Using the same method as used in Section 7.3, the site boundary cloud concentration becomes:

$$\frac{\text{Curies}}{M^3} = \frac{56,600}{2 \times 60 \times 60} \times 6.8 \times 10^{-4} \text{ sec/m}^3$$

$$= 5.35 \times 10^{-3} \text{ Ci/m}^3$$

The resulting site boundary doses are:

Beta Air Dose = 5930 mrads, Beta skin dose = 4090 mrems

Gamma Air Dose = 50 mrads, Gamma skin dose = 56 mrems

Whole Body Dose = 49 mrems

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