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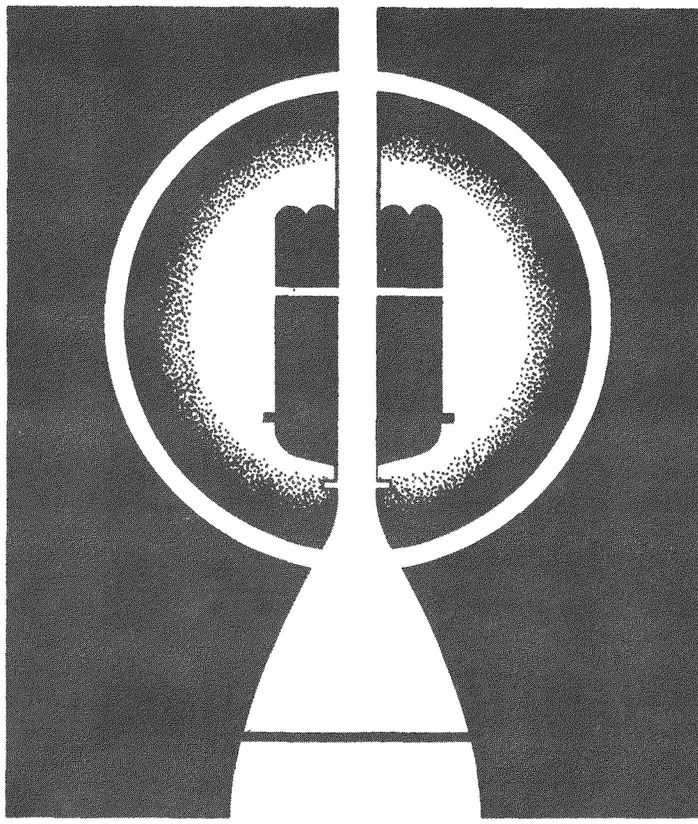

SAFETY CONSIDERATIONS RELATIVE
TO CONSTRUCTION OF E/STS 2-3

NOVEMBER 1967

AEROJET
GENERAL TIRE
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SAFETY CONSIDERATIONS RELATIVE TO CONSTRUCTION OF E/STS 2-3

NOVEMBER 1967

PREPARED BY
AEROJET-GENERAL CORPORATION
AND
WESTINGHOUSE ELECTRIC CORPORATION

CLASSIFICATION CATEGORY
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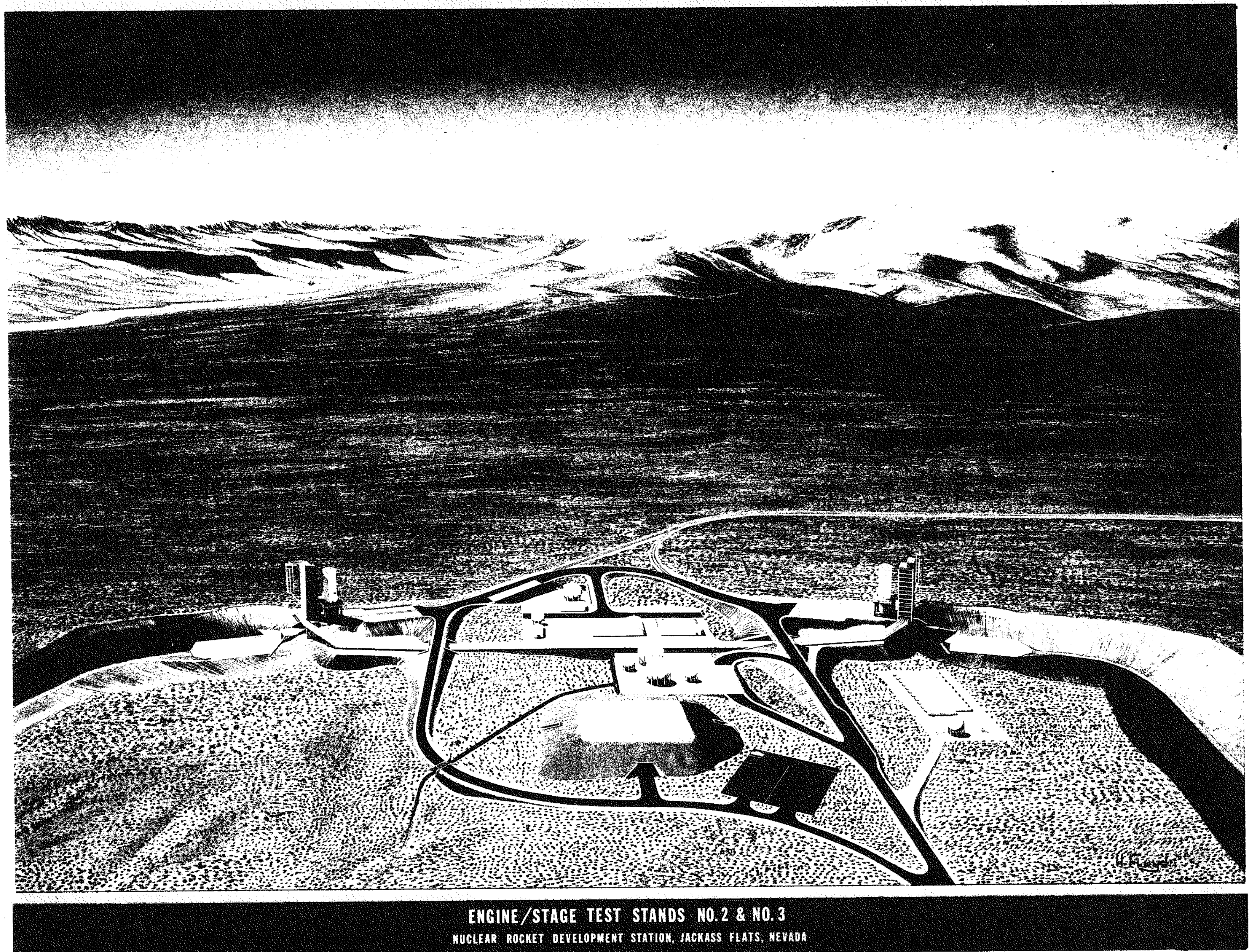

Artists Concept of E/STS 2-3

1. INTRODUCTION AND SUMMARY

The AEC-NASA Space Nuclear Propulsion Office (SNPO) is preparing plans for a nuclear-rocket propulsion-module test complex at the Nuclear Rocket Development Station (NRDS), Jackass Flats, Nevada. The test complex is designated Engine/Stage Test Stand 2 and 3 (E/STS 2-3). The purpose of the test complex is to provide the capability for testing large NERVA engine systems (nominally rated at 250,000 lb thrust) and other nuclear-propulsion flight systems. A general description of the facility is provided as Appendix B.

At a July meeting of the Government E/STS 2-3 Safety Committee, Aerojet and Westinghouse agreed to prepare a report which, based on the facility design status as of 1 August 1967, would summarize briefly the major safety considerations that relate to the initiation of E/STS 2-3 construction. The present report is the result of a combined AGC/WANL effort to satisfy this commitment. As such, it treats primarily with those safety aspects which are peculiar to the construction and utilization of E/STS 2-3 at its planned location on NRDS.

Section 2 of this report discusses the results of off-site radiological hazards studies and presents the significant conclusions therefrom relative to the protection of off-site people, the use of an effluent scrubber, and the effects of meteorological conditions during testing. Basically, the studies show that radiation exposures to off-site population groups located near the closest site boundary exceed established dose criteria both during normal test operations and under accident conditions. The use of an effluent scrubber to decontaminate the effluent gases prior to release to the atmosphere offers a solution for the normal test case (with no meteorological constraints imposed on test operations) if decontamination efficiencies of the order of 90 percent are achieved. However, for the postulated accident situation, an effluent scrubber would not assure decontamination of the effluent. Analyses indicate that test periods need to be selected with respect to meteorological conditions, such as wind direction and atmospheric stability, in order to assure that following an accident the effluent is adequately diluted before reaching uncontrolled population groups. Furthermore, monitoring, control and possible confiscation of milk out to distances of the order of 500 miles for the 500 Mw engine and 800 to 1000 miles for the 10,000 Mw engine may be required under accident conditions.

Section 3 considers the suitability of the planned location for E/STS 2-3 relative to other principal NRDS facilities. Analyses indicate that on-site personnel, if located in unprotected areas downwind of E/STS 2-3 during normal test operations or following an accident, are subject to radiation exposures in excess of established dose criteria. Similar to the case of the off-site population groups, an effluent scrubber may reduce downwind exposures to acceptable levels under normal test conditions; however, since effluent-scrubbing capability cannot be assured in the event of an accident, personnel exposed to the effluent cloud downwind from an accident could be subjected to excessive radiation exposures, whether or not a scrubber is provided. Consequently, it appears that on-site personnel must either be located in suitably protected areas if downwind of E/STS 2-3 during test operations, or evacuated from downwind areas. While evacuation of personnel from NRDS facilities may be necessary if E/STS 2-3 testing is conducted under adverse meteorological conditions, consideration of the results of the effluent studies in conjunction with the general meteorological information available for NRDS does not indicate that, in this respect, any other equally accessible location on NRDS would offer definite advantages over the presently planned location.

Section 4 of this study deals with the spacing of major facility components of the E/STS 2-3 complex. It treats with the spacing relationships of the control center, test stands and cryogenic storage areas as they are individually affected by nuclear and thermal radiation, fire, explosion and flooding. Facility safeguards for protection from fire and thermal effects are provided for by adequate spacing used in conjunction with deluge water and cooling; while distance, fragment shields and structural design features provide protection from explosion, overpressures and fragmentation. In these respects, the intra-spacing of facility components appears satisfactory provided that operational safeguards are devised and enforced and facility safeguards are operational when required. During testing, personnel are protected against radiological hazards by keeping them underground in the control center. Consideration of emergency escape in the event evacuation of the control center is necessary following a major on-stand accident during testing indicates that two alternate escape exits, each approximately 1500 ft from the active test stand, would provide for safe evacuation of control-center personnel if the accident did not occur during adverse meteorological conditions. Our review to date does not support a finding that below-grade entrances are adequately protected against flooding and therefore, it remains to be established that below-grade entrances to occupied facilities are adequately protected against flooding in the event of major ruptures of fluid piping or storage tanks.

Section 5, concerning principal E/STS 2-3 components and systems, is a design review of the control center, test stands and several of the major facility systems. Kaiser Engineers has utilized their original concepts to prepare definitive drawings and "Bid Packages" which basically describe the design of a facility. Our review of these designs to date indicates that they are basically sound from the safety viewpoint and that there are no problems which would materially affect initiation of construction of the facility as proposed. However, a number of significant problems, noted in Section 5 and its supporting references, remain to be resolved. Among these are: effective isolation of control center ventilating system sections to preclude spread of toxic gases between fire zones; control center second-floor emergency exit direct to outside area; integrity of the engine test compartment of the test stand; and the nuclear exhaust system design. A continuing review and evaluation from the safety standpoint is necessary as the design and construction proceed to assure that a facility conducive to safe and satisfactory operation is achieved.

2. OFF-SITE RADIOLOGICAL HAZARDS

2.1 INTRODUCTION

During the normal course of operations, a NERVA engine releases fission products to the hydrogen exhaust which is subsequently discharged to the atmosphere. These fission products will subject an individual downwind of the test facility to ionizing radiation. Operations at NRDS are regulated by the Space Nuclear Propulsion Office (SNPO) and the Nevada Operations Office, AEC. Established AEC standards and tentative SNPO standards to be used for the protection of both radiation workers and the general public from undue amounts of radiation are summarized in Table 2.1.

A reactor development and testing program has been carried on for a number of years at NRDS without subjecting the off-site population to radiation doses in excess of the AEC recommendations of Table 2.1. The United States Public Health Service (USPHS) conducts a program of radiological monitoring and environmental sampling in the off-site area surrounding NTS to document these releases for the AEC. As a part of this effort, they maintain dose-rate recorders, film badge stations, and air samplers at off-site locations. They also monitor the milk iodine concentrations in down-wind areas. Table 2.2 records the maximum off-site exposures as reported by the USPHS for recent reactor tests. These doses are not necessarily on the cloud centerline. A representative power level for the majority of the reactors listed on this table is about 1000 MW.

2.2 NORMAL OPERATION

In contrast to prior typical reactor test sequences of 15 minutes at 1000 MW, the NERVA program encompasses test operations of 30 minutes at 5000 MW and 45 minutes at 10,000 MW. These projected test operations result in fission product inventories and releases an order of magnitude higher than

TABLE 2.1

STANDARDS FOR RADIATION PROTECTION

	Criteria	Type of Exposure	Dose: Normal	Dose: Accident
Controlled Areas (On-Site)				
	AEC[1]	Whole body	3 rem/quarter	3 rem/quarter
		Thyroid	10 rem/quarter	10 rem/quarter
	SNPO[2]	Whole body	3.3 rem/test	12 rem/accident
		Thyroid	10 rem/test	36 rem/accident
Uncontrolled Areas* (Off-Site)				
	AEC[1]	Whole body	0.17 rem/year	0.17 rem/year
		Thyroid	0.5 rem/year	0.5 rem/year
	SNPO[2]	Whole body	0.17 rem/year	3.3 rem/accident
		Thyroid	0.5 rem/year	10 rem/accident

(1) AEC Manual, Appendix 0524 - "Standards for Radiation Protection"

(2) Letter, M. Klein to J. Jewett, "Radiation Dose Guides for Reactor/Engine Test Approvals", 26 July 1967 (Tentative)

*Based on average exposure to population sample where exposures cannot be measured and evaluated on an individual basis.

TABLE 2.2

MAXIMUM OFF-SITE DOSES FROM ROVER TESTING AS REPORTED BY USPHS

Test	Date	Time at Full Power (min)	Cloud Direction	External Gamma Dose: Distance (miles)	External Gamma Dose: Dose Rate (mr/hr)	External Gamma Dose: Dose (mr)	Thyroid Inhalation: Distance (miles)	Thyroid Inhalation: Dose (mr)	Milk Dose: Distance (miles)	Milk Dose: Milk Conc. (pc/l)	Child's* Thyroid Dose (mr)
KIWI-B4D EP-IV	5/13/64	1	N	75	0.43				93	140	17
KIWI-B4E EP-V	8/28/64	8	NE	55	0.18				20	20	2.5
EP-VI	9/10/64	2	NE						75	40	5
NRX-A2 EP-IV	9/24/64	10	SW	13	0.43						
KIWI TNT	1/12/65		SW	14	70.0	5.7	14	3.3			
NRX-A3 EP-IV	4/23/65	3-1/2	SE	45	0.025	0.01	45	0.27			
EP-V	5/20/65	13	NE	60	0.06		60	0.2	75	90	11
PHOEBUS-1A	6/25/66	10-1/2	N	4 65	30.0 0.065				125	180	21
NRX/EST EP-IV	3/16/66	15	NE	6.5	15.0	3	6.5	18			
EP-IVA	3/25/66	13-1/2	W	19		22	100	3.6	100	140	20**
NRX-A5 EP-III	6/8/66	16	SW	Not Detectable			15	<1	15	50	6
EP-IV	6/23/66	14-1/2	NE	62		1	62	5	175		40

*Assuming continuous consumption of milk at a rate of 1 liter/day.
**About 90 percent of the cow's food intake was from stored feed and 10 percent from grazing. Extrapolating the measured 140 pc/l to 100 percent grazing gives a dose of 200 mr.

those previously seen. (See Appendices E and F). These releases from normal operation under somewhat ideal lapse diffusion conditions result in estimated doses at the nearest site boundaries to the south (~12 miles) and west (~6 miles) that are higher than the AEC recommendations for individuals in uncontrolled areas. The estimated doses at the nearest off-site populated area, Lathrop Wells (~14 miles south), are also higher than the recommendations. Even considering the diffused activity doses alone with no contribution from a corrosion component, the estimated doses at Lathrop Wells are higher than the recommendation for the general population sample in an uncontrolled area. Including the corrosion component, the doses meet the uncontrolled population sample standards at 19 miles for the cloud, 43 for the child's iodine inhalation, and 130 for the milk ingestion dose.

2.3 ACCIDENTS

The estimated whole body cloud gamma dose from a loss of coolant at the conclusion of a 30-minute 500-MW run and under ideal lapse conditions would exceed the AEC recommendations for population groups in an uncontrolled area at distances less than about 29 miles. The estimated inhalation thyroid dose to a child would exceed the thyroid standards for receptors within 140 miles. The thyroid milk dose exceeds the standards for distances of up to 400 miles.

The estimated whole body gamma dose would be acceptable under the SNPO accident criteria for general population exposures for distances greater than 11 miles. The inhalation thyroid dose to a child is acceptable at a distance of 33 miles, while the milk thyroid is acceptable beyond 78 miles.

In view of the control possible over milk usage in the event of a major accident, the limiting dose under either set of criteria can be considered to be the thyroid inhalation dose to a child.

2.4 DOSE SUMMARY

The above results are given in table form in Tables 2.3 and 2.4 for the AEC criteria and Table 2.5 for the SNPO criteria of Table 2.1.

2.5 SOURCES OF ERROR

The dose estimates have three sources of error, the source term, the atmospheric diffusion model, and the dose model. Comparisons of pre-run source term estimates with post-run estimates based on radiochemical analyses of aircraft filters and fuel samples have indicated differences of up to a factor of 3. All of this difference cannot be attached to the pre-run FIPDIF estimate since the radiochemical procedures also involve uncertainties. In the majority of cases the FIPDIF estimate is higher but this cannot be considered a maxim since the post-run ratio FIPDIF estimate for the NRX/EST EP-IV-A test diffusion release was lower than that determined post-run by about this factor of 3. The corrosion estimates are based on measured releases from NRX reactors.

LASL[1] has compared the ground level air concentrations as calculated by Sutton's equation with the measurements made during reactor tests. The experimental data is widely scattered, and no direct comparison with calculated values is possible except to say that the measured values are usually lower. LASL has drawn a best fit through the experimental data and statistically calculated a set of curves for the whole body gamma dose from the cloud. Comparison of these curves with the doses predicted using Sutton's model indicate that over the range of distances in question, 10 to 100 miles, the Sutton prediction will be higher than the measured value 90% of the time at 10 miles and something greater than 99% of the time at 100 miles. But the obverse of the coin is also true. Out of ten reactor tests it is probable that the Sutton prediction at 10 miles will be lower than the measured value at least once.

(1) E/STS 2-3 Safety Committee Meeting at Oakland, Calif., 10 August 1967

TABLE 2.3

OFF-SITE DOSE SUMMARY* BASED ON AEC RADIATION PROTECTION STANDARDS
(5000 MW - 30 MIN)

	NORMAL				ACCIDENT	
Type of Exposure**	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable (Miles)	Scrubber Efficiency Required for Acceptable Dose at Distances Beyond: 12 mi	40 mi	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable (Miles)
Whole Body	0.17 rem/yr				0.17 rem/yr	
Cloud		19	72%	0%		29
Fallout - 1 hr		15	32%	0%		24
Fallout - 100 days		36	88%	0%		75
Thyroid (Child)	0.5 rem/yr				0.5 rem/yr	
Inhalation		43	89%	17%		140
Milk		130	98%	86%		420

*Lapse Conditions. Diffusion plus corrosion.
**Based on average exposure to suitable population sample.

TABLE 2.4

OFF-SITE DOSE SUMMARY* BASED ON AEC RADIATION PROTECTION STANDARDS
(10,000 MW - 45 MIN)

	NORMAL				ACCIDENT	
			Scrubber Efficiency Required for Acceptable Dose at Distances Beyond:			
Type of Exposure**	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable (Miles)	12 mi	40 mi	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable (Miles)
Whole Body	0.17 rem/yr				0.17 rem/yr	
Cloud		29	92%	0%		43
Fallout - hr		24	80%	0%		37
Fallout - 100 days		69	97%	67%		130
Thyroid (Child)	0.5 rem/yr				0.5 rem/yr	
Inhalation		78	97%	75%		240
Milk		250	99.3%	96%		800

*Lapse conditions. Diffusion plus corrosion.
**Based on average exposure to suitable population sample.

TABLE 2.5

OFF-SITE DOSE SUMMARY* BASED ON TENTATIVE SNPO RADIATION PROTECTION STANDARDS
(5000 MW - 30 MIN)

	NORMAL				ACCIDENT	
			Scrubber Efficiency Required for Acceptable Dose at Distances Beyond:			
Type of Exposure**	SNPO Radiation Dose Guides	Distance Beyond Which Dose is Acceptable (Miles)	12 mi	40 mi	SNPO Radiation Dose Guides	Distance Beyond Which Dose is Acceptable (Miles)
Whole Body	0.17 rem/yr				3.3 rem/test	
Cloud		19	72%	0%		11
Fallout - hr		15	32%	0%		8
Fallout - 100 days		36	88%	0%		18
Thyroid	0.5 rem/yr				10.0 rem/test	
Inhalation - Child		43	89%	17%		33
Milk - Child		130	98%	86%		78

*Lapse conditions. Diffusion plus corrosion.
**Based on average exposure to suitable population sample.

For comparison, the 50% probability (equal chance that the measured dose will be higher or lower) at 10 miles is a factor of 20 less than the Sutton prediction. It would thus seem that the Sutton model is conservative, but not unreasonably conservative in that there is a probability of occasionally underpredicting the measured dose at the site boundary when considering the number of reactor and engine tests planned for the NERVA program.

The dose from a defined cloud is straightforward and is considered fairly accurate. The iodine inhalation dose model is that recommended by the International Commission for Radiological Protection. The milk dose model is considered the most uncertain. The ratio of the air concentration to milk concentration used is that recommended by the USPHS.

It is concluded that the present dose prediction techniques are reasonable and not unduly conservative. In fact, there is a possibility that the estimated doses could be exceeded.

2.6 PARTICULATE HAZARD

Although the medical significance of the particulate fallout is unclear, the magnitude of the fallout concentration and the potential dose rate decrease rapidly with distance. This is due to the diminution of particulate size with distance and the greater decay time afforded by the transport time to these distances. Until the medical significance of these beta doses is determined it would seem advisable to avoid contaminating nearby off-site locations such as Lathrop Wells.

2.7 TEST PLANNING IMPLICATIONS

Based on the material contained in Table 2.3 it is concluded that operating with the wind blowing over Lathrop Wells is ruled out. Lathrop Wells is the only significant population concentration in an uncontrolled area so affected by normal operational gamma activity release. The most desirable direction for the discharge of radioactivity is northerly. Off-site doses in this direction are acceptable under all government standards, with the exception of the child's milk dose. Under the assumptions considered, the child drinking 1 liter/day from a cow fed completely on fresh pasture, the dose is marginal. Milk control for distances out to about 130 miles from the facility may be necessary.

The addition of a scrubber to the facility which removed 90% of the fission products from the effluent would result in releases comparable to those experienced in the NRX program. The NRX reactors have been tested with no apparent off-site radiological complications. In addition, there would be no potential particulate hazard. Such a system may allow testing under weather conditions no more stringent than those presently applied to NRX tests.

One other aspect of the problem, though, may still necessitate weather controls. A scrubber may not be effective in the event of a loss of coolant or excursion type accident. If a loss of coolant were caused by failure of the Ground Test Module it is possible that the Engine Test Compartment (ETC) would be breached allowing the fission products to escape directly to the atmosphere. Such an accident is most hazardous during and shortly after a high power test run when the fission product inventory is significant. If such an event occurred after a 30 minute 5000 MW test with the wind blowing over Lathrop Wells, the whole body cloud dose would be about 2 rem and the thyroid inhalation dose about 20 rem. In the sector from north-west to due east the boundary of the government controlled area is about 60 miles. At this distance the estimated doses from the above accident are on the order of 0.02 rem and 1 rem, respectively.

2.8 CONCLUSIONS

If there is no scrubber the analyses indicate a need for weather controls such that the effluent cloud will be carried to the north or north-east over government controlled area. Lapse conditions are also highly desirable. Milk monitoring off-site would be required for normal operation and diversion of milk to allow iodine decay is likely to be necessary sometime during the course of the NERVA test program. Under favorable weather conditions accident doses may be tolerable off-site, but control of milk may be required for a considerable distance downwind from the test facility. If there is a scrubber the analyses indicate that under lapse conditions the off-site doses for normal operation are acceptable and diversion of milk is not likely to be required regardless of wind direction. However, weather controls are still needed to prevent high doses at close off-site locations, e.g. Lathrop Wells, in the event of an accident. Similar to the situation without a scrubber, in the event of a major accident control over milk may be required for a considerable distance downwind from the test facility.

3. LOCATION OF E/STS 2-3 ON NRDS

This chapter summarizes the possible effect that testing reactors in E/STS 2-3 will have on other on-site facilities. Two effects are considered, radiation from cloud during passage and facility contamination from ground deposition. The above effects are dependent on the distances separating the facility in question from E/STS 2-3. Table 3.1 contains these distances. Table 3.2 contains the personnel access requirements defined by SNPO in the E/STS 2-3 Criteria Document.

If operations are conducted with the effluent carried over a site facility, the doses from normal operation under lapse conditions would be unacceptable on a quarterly basis for a single test under the AEC criteria and under the SNPO criteria which defines the AEC quarterly standards as the standards applicable to a single test. In addition, the doses from a loss of coolant are unacceptable, and the minimum cloud transport time (~5 minutes to ETS-1 at a wind velocity of 20 miles/hour) would require a very prompt evacuation response time. The alternative, that of evacuating the downwind area before a test would still lead to particulate contamination and, for a period of several days following the test, the dose rates from deposited gaseous fission products may be higher than those allowed under the uncontrolled access criteria. In the event of a loss of coolant under these weather conditions the dose rate of ETS-1 from ground deposited gaseous fission products would exceed the controlled access criteria for a period of about 100 hours following the test.

All of the above conclusions are drawn based upon doses and dose rates under ideal lapse conditions. The doses during non-ideal conditions are higher. Tables 3.3-3.5 indicate the receptor distances from the test stand at which the whole body and the adult thyroid inhalation dose would meet the AEC and tentative SNPO criteria (Table 2.1) under these lapse conditions.

TABLE 3.1

DISTANCES FROM E/STS 2-3 TO OTHER NRDS FACILITIES

Facility	Distance (feet)
ETS-1	9,000
Burial Ground	15,000
E-MAD	15,000
Test Cell "C"	18,500
Control Point	23,000
Test Cell "A"	25,000
A & E Building	25,000
R-MAD	32,000

TABLE 3.2

PERSONNEL ACCESS REQUIREMENTS

	Dose Rate
Unlimited Access	<2-1/2 mR/hour
Normal Controlled Access	<100 mR/hour
Maximum Planned Exposure	<3R/hour
Restricted Access (no planned exposure)	>3R/hour

TABLE 3.3

ON-SITE DOSE SUMMARY BASED ON AEC RADIATION PROTECTION STANDARDS
(5000 Mw - 30 min)

	Normal		Accident	
Type of Exposure	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable* (Miles)	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable* (Miles)
Whole Body	3.0 rem/qtr	6	3.0 rem/qtr	12
Adult Thyroid	10.0 rem/qtr	0.3	10.0 rem/qtr	22

*On-site distances: ETS-1 (2 mi), E-MAD (3 mi), TCC (3-1/2 mi), CP (4-1/2 mi), A&E (5 mi).

TABLE 3.4

ON-SITE DOSE SUMMARY BASED ON AEC RADIATION PROTECTION STANDARDS
(10,000 Mw - 45 min)

	Normal		Accident	
Type of Exposure	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable* (Miles)	Radiation Protection Standards, AECM 0524	Distance Beyond Which Dose is Acceptable* (Miles)
Whole Body	3.0 rem/qtr	10	3.0 rem/qtr	16
Adult Thyroid	10.0 rem/qtr	9	10.0 rem/qtr	37

*On-site distances: ETS-1 (2 mi), E-MAD (3 mi), TCC (3-1/2 mi), CP (4-1/2 mi), A&E (5 mi).

TABLE 3.5

ON-SITE DOSE SUMMARY BASED ON TENTATIVE SNPO RADIATION PROTECTION STANDARDS
(5000 Mw - 30 min)

Type of Exposure	Normal: SNPO Radiation Dose Guides (Per Test)	Normal: Distance Beyond Which Dose is Acceptable* (Miles)	Accident: SNPO Radiation Dose Guides (Per Test)	Accident: Distance Beyond Which Dose is Acceptable* (Miles)
Whole Body	3.3 rem	5	12 rem	7
Adult Thyroid	10 rem	0.3	36 rem	11

*On-site distances: ETS-1 (2 mi), E-MAD (3 mi), TCC (3-1/2 mi), CP (4-1/2 mi), A&E (5 mi).

It appears that testing at E/STS 2-3 cannot be carried out without either weather controls or absolute personnel control. The analyses indicate that the weather control must be such that the effluent will not be carried over an inhabited facility. This is true even for operations under ideal diffusion conditions since the consequences from a major accident may be unacceptable. The other choice, absolute personnel control, would require the evacuation of all people in the downwind area. Before they would be able to return to these areas decontamination with its attendant expenses, inconveniences, and delays would likely be necessary. This is true even for normal operation under ideal conditions. The decontamination required would be greater if test operations were performed under less than ideal conditions.

The results presented in this chapter bear the same qualifications with regard to the source term, atmospheric diffusion, and dose models as the results in Chapter 2, and the comments of Section 2.5 specifically apply.

4. INTRA-FACILITY LOCATION AND SPACING

The geographical placement of the major components of the E/STS 2-3 facility are of paramount importance in the overall safety evaluation. First order importance is placed on the protection of people during normal operations and in the event of postulated accident situations. While second order importance is placed on facility safety, adequate attention is given to major hazards and protection of facilities from them. There is an interrelationship between facility and personnel safety and proper thought has been devoted to continuity of operation as well as safety of personnel.

There appear to be five types of hazards which require consideration in the development of finite spacing of facility elements and the layout of the various facility components. These are: nuclear radiation, plume thermal radiation, fire, explosion and flooding. The facility components which need consideration with regard to these hazards are the control center, the test stands, and the cryogenic storage areas. The placement and spacing of these key facility components with regard to the major hazards noted are discussed in this section.

4.1 NUCLEAR RADIATION

During engine test periods, E/STS 2-3 personnel are protected against excessive radiation exposure by confinement to the control center which is located approximately 700 ft from the test stands and which is shielded to reduce the internal radiation intensity to less than 2-1/2 mr/hr during a 10,000 Mw test. The protection afforded personnel by the control center during engine testing is further discussed in Section 5.1.

During non-test periods, personnel are protected by controlling and limiting their access to high-radiation areas within the facility. The basic safety principles for personnel control that are being accommodated by the E/STS 2-3 design are identified in Appendix G.

Reference (1) defines the environmental radiation intensities expected at E/STS 2-3 during test and post-test periods due to direct radiation from a 5000 Mw NERVA engine operated for 30 minutes. These data are summarized in Appendix C. Personnel access requirements to various parts of the E/STS 2-3 are identified on page VI-B-9 of the Facility Design Criteria, Reference (2). Reference (3) establishes the minimum intra-facility separation distances based on these access requirements, on the results of the environmental radiation analyses reported in Reference (1), and on the use of a facility shield with a thickness equivalent to 6 ft of water. These minimum distances are reflected by the E/STS 2-3 site layout, Figure B-1 of Appendix B.

The separation distances established do not take into account radiation levels which can result from dispersion of radioactive material, either during normal test operations or under accident conditions. However, since the dispersal of radioactivity over the facility during normal test operations can, to some extent, be controlled by incorporation of an effluent scrubber and by restricting high-power tests to periods of favorable meteorological conditions, the only intra-facility separation distances which should be affected by dispersal of radioactive material are the emergency-tunnel exit locations. With the exception of these locations, the nuclear-based separation distances established by Reference (3) and reflected by the site layout, Figure B-1 of Appendix B, appear to be adequate from the safety standpoint, Reference (4).

Though maximum precautions are being taken in the design of the control center (see Section 5.1) to make it unnecessary to have to evacuate this location under any foreseeable accident conditions, emergency escape routes away from the facility are being provided. Emergency exit locations recommended below are based on environmental dose rates resulting from fallout contamination at the facility following a loss-of-coolant accident and on direct radiation from an unshielded engine following a full-power full-duration test. On the basis of those two accident considerations, Appendix F recommends providing two alternate emergency escape routes from the control center with exits to the above-ground radiation environment at locations approximately 1500 ft from the active test stand and orientated such that the angle formed by the exit locations and the active stand is approximately 70°. One of the recommended escape routes is through a shielded tunnel which leads to a point due south of the control center and exits at a point approximately 1500 ft from both stands. The second escape route utilizes the access tunnels to the test stands in conjunction with short extension tunnels from the stands to shielded exit points. An enclosed area for evacuation vehicles is to be provided at each emergency exit location. The area will be sized for sufficient vehicles to transport all personnel in the control center at the time of a test.

With incorporation of the above emergency escape provisions into the E/STS 2-3 design, Appendix F shows that personnel can safely evacuate the facility under favorable meteorological conditions almost immediately following an accident. For example, assuming personnel egress from the facility 0.1 hr after a maximum accident (either loss of coolant or loss of facility shield at the conclusion of a 5000-Mw 30-minute test) and assuming personnel are exposed for a time period equivalent to 15 minutes to the radiation levels which exist at the emergency exit locations, the total exposure received by personnel during escape would be 12 rem. This maximum exposure level presupposes the wind to be blowing in a favorable direction with respect to at least one of the two exit locations and it also assumes that the escape route taken is the one which has the lowest radiation level at the point of exit. It is also shown in Appendix F that, even if favorable wind conditions exist at the time of a loss-of-coolant accident, wind shifts following the accident can produce unacceptable inhalation and whole body dose rates from the effluent cloud at the exit locations for periods up to several hours after the accident. Consequently, safe escape from the facility immediately following an accident cannot be absolutely assured under all conditions (i.e., with unfavorable meteorological conditions at the time of the accident or with a wind shift following the accident resulting in variable winds alternately directed over both emergency exit locations). However, since the control center is being designed so that evacuation is not necessitated under foreseeable accident conditions, the escape provisions discussed above are considered adequate for the protection of E/STS 2-3 personnel provided testing is not conducted under adverse meteorological conditions.

4.2 PLUME THERMAL RADIATION

The thermal effects of the rocket exhaust on the facility have been partially analyzed at this time with consideration given to both analytical methods and test scale models. The analyses were conducted utilizing results from both 1/8 and 1/4 scale model tests as reported in "Comparison of Plume Thermal Effects for ETS-1," Reference (5). A sophisticated digital computer program named GASRAD has been developed by Kaiser Engineers to predict the thermal radiation from the plume. The theoretical results from this program agree quite closely with the 1/4 scale test results obtained by AGC. A summary of Kaiser Engineers' method of analysis is outlined in the Kaiser Engineers Budgetary Study, Reference (6), and their comments on AGC recommendations are outlined in Reference (7), which deals with ETS-1 exhaust plume thermal parameters.

The GASRAD predictions are based on surface emissivity and flame temperatures of the burning jet, assuming stoichiometric mixtures of hydrogen and air with adiabatic burning of the steady-state plume. The plume, incidentally, is estimated at 480 ft in length for the 4:1 ratio wet duct with the 5000 Mw system and the incident heat flux to various portions of the test stand have been predicted based on the full power run.

The incident heat flux, as would be expected, is quite high at certain selected target locations on the stand and the predicted temperature rise severe enough to require shielding or water protection. It should be noted, however, that radiant heat from the plume is not a governing factor insofar as distances to other facilities are concerned. A hydrogen explosion or fire on the stand, for instance, is more severe and hence is a governing factor. The exhaust plume produces a localized problem only.

Certain meaningful tests are planned during the activation phase of ETS-1 and Reference (7) outlines the test instrumentation deemed necessary. It is anticipated that data from these tests will be available for application to the problem of determining the necessary level of protection for safeguarding the E/STS 2-3 facility.

Several factors of the design weigh heavily in favor of satisfying the requirement to protect against thermal radiation during full power operation. First the wet diffuser concept results in significantly lower plume temperatures (on the order of 2800°F less) and the wet plume is expected to be deflected very little by an on-stand 35 mph wind (design criteria). Empirical equations were developed to determine the degree the jet would be distorted from its normal axis due to high winds blowing toward the stand. The calculations assumed 45° inclination of the wet conical plume and found that with this configuration the effect of the wind could be disregarded. This is no doubt true because of the mass effect of the injected water.

During pulse cooldown modes and low power runs, however, the flow rates are significantly less than those of full power runs. During these operations there may be entirely different criteria to analyze when considering the significance of a 35 mph wind blowing the plume toward the stand. Therefore, heat flux for this set of conditions must also be investigated to determine the protection requirements for the worst case, i.e., full power run or low flow with a 35 mph wind blowing the plume back onto the stand.

Deluge protection needs have been estimated at 1 gpm/ft^2 of the wetted surface to guard against the radiant heat, and "shadow shields" have also been discussed in the criteria outlined in the Kaiser Engineers' Budgetary Study, Reference (6). As pointed out in the above paragraphs and Reference (8), a discussion of "Test Stand Deluge Water Systems," some modification of the deluge flow rates may be in order after a complete review of this problem and the information from ETS-1 activation tests. When this has been accomplished, the desired degree of safeguarding can be provided.

4.3 FIRE

A considerable amount of orderly and progressive thought has been devoted to proper spacing and protection from fire. The two major problems considered were those of a massive fire at the main LH_2 storage area and a fire on the test stand. Of the two, it is readily apparent that a fire at the bulk storage area is the more severe from the standpoint of shear size alone. The quantity of LH_2 held in storage at the single main dewar is 1.3 million gallons as contrasted with 500,000 gallons in the module tank (on the test stand).

Fires at these two source points were evaluated in Sections 4 and 5 of Reference (9), Study Area F, and Section 3 of Reference (10), which is a preliminary report of the preceding reference, to determine the danger to personnel and the threat to other elements of the facility. In these reports it is postulated that the maximum credible accident on the test stand is the rapid fracture of the module tank due to overpressurization. It was assumed that the contents of the tank (500,000 gal) would be released and would ignite with combustion of the discharged LH_2 occurring in 5 seconds.

The fire created by this accident and the radiated heat at the adjoining test stand is not a governing factor in the determination of the separation distance between stands. The same is true in regard to the separation distance between the stand and LH_2 dewar, as, in this case, the thermal radiated energy from a fire at the stand is less than that of a fire at the dewar. Fragment dispersal or shrapnel dispersion as a consequence of the module tank rupture is a real problem, however, and in the final analysis becomes the major criteria for stand separation. The factors involved are discussed in more detail in Section 4.4 of this report. Section 4.4, Explosion, discusses the phenomena in some detail and outlines the reasons for it being the major factor in governing spacing requirements from one stand to another.

The maximum credible fire accident postulated is that of shearing a 20-in. pipe flange at the base of the LH_2 dewar and above the shutoff valve. This accident is assumed to occur while the dewar is pressurized to 75 psi just prior to an LH_2 transfer operation. The assumption is naturally for the worst set of accident conditions. Hydrogen is initially released at an assumed rate of 2500 lb/sec. With this set of criteria, and the many other assumptions of Area Study F, Reference (9), Section 4, a fire of awesome magnitude is produced and its thermal radiation is of sufficient intensity to become a governing factor for the required separation distance from the test stand to the main LH_2 storage dewar. Based on the K.E. calculations, a 380-ft separation is necessary for protection of the module, but several other factors also play an important role in the protection scheme. They are: time, insulation of the module, and deluge protection at the module. In order to understand the part that each plays, they are discussed below in some detail.

First, the module is the major concern at the test stand since it is most sensitive to heat. The insulation covering is assumed to be fabric with an inherent ability to provide 20 sec of protection against the thermal radiation from a fire at the dewar. Unless deluge spray is then applied to cool the test article, it is assumed that thermal damage will start occurring. The deluge necessary for this protection is estimated at 0.5 gpm/ft^2 and it is assumed that it will be operated within 10 sec after initiation of the fire. The spray is assumed to be completely vaporized, thus achieving its maximum heat absorption capability.

Several assumptions in the analysis may be either too conservative or overly optimistic. It is too early in design to be certain, but they need further evaluation at a later date when some of the points are firmed up. It is assumed, for instance, that the deluge system operates at the module within 10 sec after a fire is initiated at the LH_2 storage dewar and that the insulation (assumed to be a fabric) has the capability of withstanding the heat radiated from this fire for a period of 20 seconds.

It is true that the deluge could be programmed to operate in the estimated 10-sec period, but it is also true that it would take special equipment to do this. It would require extra fast detection systems and transmission circuits and finally a special activation system which would probably require prepriming of the water up to the nozzles. It might require explosive operated systems similar to those used in the chemical and solid rocket manufacturing industries to achieve the desired result. In light of this, the 10-sec operating time might be overly optimistic unless special equipment is furnished.

It would appear that another insulating medium with a higher degree of resistance might be selected for the module protection. If so, the time factor might not be so critical. At any rate, it is an area of pessimism which could be changed and seems well worth investigation.

Complete vaporization of the deluge water and conversion to steam may be too idealistic and further investigation may show that it is necessary to provide more than 0.5 gpm/ft^2 to achieve adequate protection.

The separation distance from test stand to LH_2 dewar is approximately 500 ft. The additional 100-plus ft over the calculated required separation distance of 380 ft would no doubt provide greater attenuation of the radiant heat and thus afford an additional safety factor.

In summing up the various factors previously discussed there are a calculated 2:1 factor of safety for time in actuating deluge protection, a 1.3:1 factor of safety with regard to separation distance as compared to the stated requirement, and a 1.1 factor of safety with regard to deluge water cooling and protection. The module insulation limitations are unknown at this time, thus, no safety factor can be assigned.

From the previous discussion it should be apparent that further evaluation and definition of limiting factors is necessary. Adequate facility protection can probably be provided as postulated but it might also be borderline. The questions raised should be fully investigated to reduce the uncertainties and provide a more reliable determination of the level of protection provided.

Safety of personnel from fire is considered in Reference (10), Section 3, in which calculations and extrapolations from existing documents are used to determine safe spacing distances. DOD Instruction 4145.21 was considered too conservative when dealing with the allowable burn criteria of 2 cal/cm^2 and the recommendations in Bureau of Mines Report of Investigation 5707 offered little additional help. This investigation was based on 90 liter spills and provided a formula for prediction of the resulting fire ball size after ignition. Extrapolation from this formula results, for the E/STS 2-3 case, in flame sizes greater than the "safe distances" required by DOD.

This inconsistency was believed by Kaiser Engineers to warrant further investigation and resulted in the simplified calculations as shown in Appendix D of Reference (10), the preliminary report of Study Area F. The approach taken was found to agree quite closely with work performed by Arthur D. Little, Reference (11), whose investigations studied spills of 5000 gal of LH_2 as contrasted with the 90 liter spills of the above-noted Bureau of Mines study. As a result of this work, a 390-ft separation distance from the main LH_2 storage is considered adequate if personnel move behind protection in a 10-sec period. In this time period it is predicted that second-degree burns could be incurred over an appreciable area of the body. The calculations also indicate that personnel at the test stand would not experience first-degree burns if the main LH_2 tank were 530 ft from the test stand.

The plot plans indicate approximately 500 ft from the test stand to the LH_2 dewar and approximately 350 ft from the dewar to the entrance to the control center. These protective distances, plus stringent operational controls, should offer sufficient protection to personnel to preclude substantial injury.

4.4 EXPLOSION

The separation distance between test stands and the site of the steam generator unit are both considered on the basis of possible explosions. The distances from the hydrogen dewar to other elements of the facility were also reviewed. Based on the information outlined in the A. D. Little study, Reference (11), a hydrogen vapor phase explosion at the dewar was considered improbable and therefore ruled out. As determined in the referenced studies by A. D. Little, a very strong initiating source (such as a detonating cap) is required to produce detonation of the H_2-air mixture and it was deemed improbable that such an energy source would be available in the event of a major spill at the dewar. A major fire is considered possible, however, and its effect on site layout has been pointed out in the preceding section.

The steam generators are located about midway between the two test stands with this single installation serving both locations. The generators will be fueled with a mixture of alcohol and liquid oxygen. A study of this hazard is presented in Section 4 of Study Area F, Reference (9), where the maximum potential accident is estimated to be equivalent to a 100-lb TNT confined explosion. The overpressure generated is not a determining factor for spacing, but a considerable amount of shrapnel could be generated. Fragment shields are to be constructed around this facility in order to prevent dispersal of the shrapnel and thus allow the "close in" location. Safe spacing because of overpressure is not a governing factor since it is automatically provided as a result of the greater distances required for other hazards.

Each test stand in the Study Area F is assumed to have a module provided with a 500,000 lb LH_2 storage tank. The tank is "flight-weight" and has a safety factor of only approximately 1.6:1. An overpressurization of this tank is calculated to result in an energy release of about 500 lb of TNT. This is based on the information contained in Reference (12), an Air Force report which considers data from fragmentation by overpressurization of a similar type module tank in a Saturn IV test. The Saturn IV failure was equated to about 1000 lb of TNT, or twice that calculated for the present E/STS 2-3 module tank. The fragment dispersal as a function of distance obtained for the Saturn IV test was assumed to apply directly to the E/STS 2-3 module tank, thus providing a 2:1 safety factor.

Failure of the tank would result automatically in either an explosion or fire. The former case equates to a 500-lb TNT explosion resulting in overpressures at the adjoining stand of about 0.5 psi. Test stand design is based on accommodating this loading.

Fragmentation from this postulated accident becomes the governing factor in spacing requirements. At the specified 1150 ft to an adjoining stand, it is calculated that a 1% probability exists for fragments impacting on the stand. This seems an acceptable risk when viewed from the fact that the criteria for failure or malfunction of other critical components permits a 2% probability of occurrence.

Complete protection from fragments is assumed at the steam generator location by virtue of a fragment shield installed at that location. The shield as designed would both confine fragments, in the event of internal explosion, and also protect it from fragments coming from the stand.

Design features of the LH_2 dewar must incorporate protective features that will prevent appreciable damage by fragments from a module tank failure. The dewar must also have the inherent capability to nullify damage by overpressure in the event of an explosion of the module tank or the steam generator. With regard to thermal radiation from a module tank failure, a five-second burning time has been assumed and the integrated thermal dose is calculated to show a 15 to 20:1 safety factor.

The control center is a reinforced concrete buried structure and by its construction features and underground location is protected from overpressures, fire or thermal damage from accidents at the test stand, LH_2 dewar, and steam generator. Further, the basis for structure design, Reference (13), calls for blast-resistant exterior doors.

The main electrical substation is located such that it meets the same minimum distances as the previously-mentioned facilities. These minimum distances are: at least 1150 ft from a test stand for protection from fragments, and at least 380 ft from the LH_2 dewar for protection from thermal radiation. If deemed necessary, at a later date, fragment shields could be erected and deluge protection provided for added protection.

4.5 FLOODING

Kaiser Engineers has not considered the problem of flooding in any of their special hazard studies. Actually, it is not really a factor in spacing of key elements of the facility as one would view the other factors considered in this section of our study. It does merit review, however, when it is noted that the facility water supply will be a gravity flow system. Test requirements call for some 180,000 gpm which, translated in terms of storage, requires in excess of 9,000,000 gal held for cooling, deluge and domestic use.

The proposed development scheme calls for location of the main storage tank approximately 1-1/2 miles north of the main facility and at sufficient elevation to obtain a 100 psi operating head. The various studies conducted to determine flow requirements, with various configurations of duct design, are outlined in Section 15 of Reference (6), Kaiser Engineers Budgetary Study. Pertinent drawings are Water Supply Flow Diagram, 6552-8-C2; Yard Piping General Arrangement, C4-1500 Rev. R-A; and Water Lateral Details, C4-1532 Rev. R-A.

A review of these drawings and the Central Complex Plot Plan, C1-1101, Rev. R-0, presents a fairly accurate picture of the natural and artificial grade contours as they relate to the supply tank and main lateral lines. Industrial water is routed to each of the stands through a 72-in. diameter supply line located on the north side of the facility complex and well removed from the operating areas. A 45-in. diameter spur, however, proceeds south to the cryogenic storage area to serve deluge needs at this location. A separately supplied 10-in. looped fire main supplies domestic and fire hydrant requirements throughout the complex.

It appears that rupture of any of these lines would not cause any catastrophic condition at the site as the grade lines appear to be such as to adequately convey away the bulk of any released water. Also, control valves installed at key points in these lines can be used for shutdown of an affected section. Final judgement, however, is reserved until such time that a study can be completed which deals with the possible rupture of a line with the water entering an underground structure. There should be positive assurance that the entrances to the control center and test stands are adequately protected against such an occurrence.

5. PRINCIPAL E/STS 2-3 COMPONENTS AND SYSTEMS

5.1 CONTROL CENTER

The design of the control center must of necessity receive careful attention and thorough pre-planning. During the conduct of a test or after an emergency has occurred, it will be a safe refuge from fire, radiation or explosion. To meet these requirements it must offer shelter in the event of any major accident during testing and provide protection to its occupants for some 24 hours after the occurrence.

To meet the objective stated above, numerous special features must be designed into the structure. These include radiation shielding; protection against overpressure as a result of an exterior explosion; strict limitations on the use of flammable materials of construction; protection against earthquake or weapons test damage; protection against flooding; isolation of hazardous areas and equipment; protection from fire and segregation of fire areas; ventilation systems that provide for normal requirements as well as emergency operation (preventing the distribution of radiation contaminated air or smoke and other toxic gases in event of a fire); emergency escape exits so that evacuation of the control center can be achieved under the conditions of any accident situation; and sufficient air and health and sanitation facilities to sustain the occupants safely for a period up to 24 hours with the control center building sealed.

Those features just enumerated are amplified in some detail in various reference documents. The most pertinent of these is "Safety Requirements for Protection of Personnel at the E/STS 2-3 Control Point Building During Test Operations." This document was prepared by the Safety Staff of the NRO Division of Aerojet-General Corporation and is Appendix G to this study. A second reference is "Agreements Reached at Meeting Held in Oakland on 29 March 1967 to Establish Safety Criteria for the Design of the

E/SFS 2 C Control Point Building," dated 30 March 1967, Reference (14). The major points of these documents are summarized here to illustrate the broad base criteria prepared to highlight the problem areas.

The following is extracted from Appendix G.

"Shielding against the nuclear radiation generated during power operation of the engine must be an inherent part of the building design. The basic requirement is that the earth shield reduce the radiation level such that the dose rate received by personnel inside the structure during normal operations is within the limits established for unrestricted unlimited occupation, which is normally established as 2.5 mr/hr."

"The operation of the engine on the test stand includes the use of cryogenic fluids and pressurized gases which are a source of possible explosive mixtures and fires. The control center must therefore be designed to protect the operating personnel from accidents arising from these hazards. The basic requirements include the following:

1. Fire-resistant materials of construction will be utilized insofar as possible.

2 Exits will be provided in sufficient numbers to assure personnel cannot be trapped in any area of the building. Exit doors must be provided with jam-proof hardware and be of blast-proof construction if warranted by location near an explosive source. The exits must be designed to maintain the waterproof integrity of the building to prevent water leakage and the possible flooding of all exits must be prevented.

3. The structure design must also consider ground shocks due to earthquakes and weapons tests, the primary requirement being that personnel cannot be trapped due to collapse of the main building structure or any exits therefrom."

"To prevent the personnel located in the control center from being exposed to radiation doses from radioactive effluent in the atmosphere, the following additional requirement must be met by the ventilation design: provide a means for sealing off the ventilation air intake supply duct against the atmosphere.

The close proximity of the control point to the test stands may prevent egress through the normal control point exits due to possible personnel exposure to fire, explosion or nuclear radiation hazards resulting from accident conditions. Therefore, an emergency escape route is required which protects personnel from such hazards during evacuation from the facility. A tunnel leading from the underground control point building could provide the necessary protection from the potential accidents cited."

The following items from Reference (14) set forth key safety criteria for design of the control point building as agreed to at a meeting of NERVA Safety personnel on 29 March 1967:

"1. The number of personnel located at the control point during engine test periods should be held to a minimum consistent with the essential needs of the run-day operations.

2. It is recommended that, during engine test periods and at all other times when a hazardous radiation environment can develop above ground, the control point building be sealed. Thus, the building and all access routes to it should be designed such that they can be sealed from the external environment. In addition,

the building should be provided with a self-contained air supply to support test operations personnel for periods up to 24 hours.

3. Though the air-intakes to the ventilation system will be designed for shutdown during engine test operations, they should be provided with absolute filters, air monitors, remote alarms, and automatic shutdown devices to preclude the intake of airborne particulate and combustible gases during periods in which they are operating.

4. Maximum precautions should be taken to make it unnecessary to have to evacuate the CP under foreseeable accident conditions. For example, the CP should be designed to minimize the possibilities of fire. In addition, the building and access tunnels should be divided into zones with air supplies that can be isolated from the remainder of the facility ventilation system. Automatic fire sensor and alarm devices should be provided at appropriate locations.

5. Though maximum precautions should be taken to make it unnecessary to evacuate the CP during test periods, emergency escape provisions must be provided. The following should be considered:

a. A "hot" engine (30 days after a 9×10^6 Mw-sec test) is installed at the alternate test position.

b. At least two emergency escape routes away from the facility are available. The routes should be in opposite or nearly opposite directions. (Use of the two stand access tunnels can be considered for one of these routes.)

c. An enclosed area for evacuation vehicles should be provided at the exit of the primary emergency escape route. The area should be sized for sufficient vehicles to transport all personnel in the CP at the time of a test.

d. Point of exit for the primary emergency escape route cannot be established until effluent conditions resulting from accidents are better defined.

6. An escape route should be provided from the second floor of the CP for use during non-run periods. This route should bypass the first floor area.

7. A hazards control room equipped with instrumentation for remote readout of the environmental status of the facility and with sufficient space for safety and health physics personnel necessary for re-entry and personnel evacuation operations should be provided. The room should have adequate storage space for safety equipment.

8. Personnel decontamination facilities should be provided at the control point to prevent radioactive material from being carried into the CP area."

Since these criteria were prepared, several design reviews have been made and recommendations outlined to achieve a level of protection consistent with the above noted criteria. The design to date does not reflect incorporation of these comments and recommendations and falls short of the goal. AGC memoranda, References (15) and (16), outline those recommendations which should receive attention. The more important of these are listed briefly to illustrate the situation: segregation of ventilation and air conditioning system to avoid circulation of contaminates, smoke or toxic gases; maximum elimination of combustible materials of construction; the positive segregation of fire areas; and positive protection against the possibility of flooding.

From the foregoing illustrations, it should be clear that there are some important considerations which must still be given to the design of the control center. Corrections to the current control-center design and specification are in order to achieve appropriate protection of personnel, but none of these is deemed so difficult as to be critical. In addition, there is the problem of the emergency escape tunnels, which is treated in Section 4.1 of this report.

5.2 SAFETY SUBSYSTEMS OF THE INSTRUMENTATION AND CONTROL SYSTEM

The I&C system is a sophisticated facility control, monitoring and data acquisition system. The safety subsystems comprise a rather small but important part of the whole system and are used for both quantitative measurement and personnel warning. Those subsystems involved are: gamma radiation monitoring, criticality monitoring, atmospheric radioactivity monitoring, oxygen monitoring, combustible gas detection, fire detection, and meteorological systems. In addition, an area surveillance and warning system is provided to advise and direct personnel in event of an emergency of any kind.

The Budgetary Study, Reference (6), presented an outline of the requirements of the various systems, along with working descriptions of some of the instrumentation available to perform the intended function. Because it was stated in such general terms, it did not provide a reasonable basis for the evaluation of Kaiser Engineers' Performance Specifications of the various systems. Accordingly, SNPO-C directed that NRO Safety prepare the minimum acceptable criteria which would describe the functional intention of the systems, relate any pertinent standards and, in general, call on experience gained in activation of ETS-1. This document was prepared and is called "Basic Functional Requirements of E/STS 2-3 Safety Oriented I&C Facility Support Systems." It is attached to this study as Appendix H.

A brief review or synopsis of this criteria is in order to aid in the understanding of the requirements established for a design basis. The heart of the systems is a safety console located in the control center. This console will provide an operating control and monitoring capability for all of the safety subsystems mentioned above. This will be accomplished by circuit and sensor readouts, recordings, visual and audible alarms, failure indications, manual activation switches and alarm acknowledge and reset switches. Flashing alarm lights will indicate (on a graphic display panel) the location of an alarming sensor or activated emergency switch. Panel meters will also be provided to indicate the level of activity at the various monitoring locations of the detection systems.

Pertinent features of the safety subsystems are reliable power supply systems, electrical supervision of circuitry, use of tested and approved components, reliance on nationally accepted standards for circuit design, system malfunction alarms and alarm condition annunciation with tie-in to the NRDS fire station and the area surveillance and warning system. The latter is an adjunct to the various detection systems in that it alerts personnel, on a facility-wide basis, of hazardous conditions associated with the operation and guides them to a safe location. A klaxon sound is used for a criticality event and a siren for any other condition requiring evacuation.

A condensed summary of the individual systems is as follows:

Criticality Monitoring System

A neutron detector system with a two-out-of three detector logic is used for determination of accidental criticality. The system sensitivity is such that an alarm will be transmitted when the level of radiation from an excursion is 300 rem/hr at a distance of one foot from the source.

Gamma Monitoring System

Sensors for monitoring gamma radiation are necessary to determine radiation levels after an engine test in order to pre-plan re-entry for operational personnel. They are also located at exit points from the control center to determine activity levels in the event that emergency evacuation is necessary. These two requirements dictate that sensors be located at important operational areas and at tunnel and emergency escape route exits.

Atmospheric Radioactivity Monitoring System

Detectors for this system are required to monitor the control center incoming air to determine whether airborne activity is being introduced into the system. The units are required to monitor for radioactive particulate and gaseous beta/gamma activity.

Combustible Gas Systems

Hydrogen calibrated detectors must be installed in the duct vault, engine compartment and other enclosed areas where hydrogen could pocket and form an explosive mixture with air. They also aid in determining purge conditions.

Oxygen Detection Systems

Determining the completeness of purge is the prime requirement for these systems. They must monitor such areas as the engine shield, flare stacks and duct system to ensure that air is removed before hydrogen is introduced to these areas.

Fire Detection Systems

Fire or heat detection systems are necessary to monitor for a fire condition in all of the important areas throughout the facility. The control center and test stands are examples of areas where an early indication of fire is necessary so that fire fighting operations can be started without delay. Many types of sensors are available for use depending on the application.

The above is a brief summary of the criteria submitted to the A&E by SNPO for use in the second design preparation. Review of the final "package" of performance specifications has just recently been completed and indicates acceptance and rework of the design to incorporate the philosophy and parameters outlined in the criteria just discussed. These specifications are identified as DTL 0120, E/STS 2-3 Procurement Package No. 10 (I&C), dated 2 October 1967. Comments from review of this package by AGC-NRO Safety are given in Reference (17) and are more directed to refinement and cost savings as opposed to criteria differences. In general, we feel that the specification package is acceptable and will provide a basically sound system for safety monitoring.

5.3 TEST STANDS

Each of the two test stands is designed for a vertical-mounted, downward-firing, 5000 Mw engine system developing 260,000 lb thrust. Modifications at a later date will allow the firing (from the same position) of a 10,000 Mw system, which develops some 500,000 lb thrust.

The successful accomplishment of this firing depends on many factors including numerous safety parameters. A review of the Budgetary Study, Reference (6), shows the significant areas necessary for consideration to be: nuclear shielding to limit neutron activation of test equipment and facility and to provide protection to operations personnel during post-test periods; thermal radiation shielding to prevent damage to the module, controls systems, test stand, umbilical arms, etc., from the exhaust plume during engine testing; providing protection of the stand against fire and possible explosion of hydrogen on the stand, hydrogen fire at the main storage dewar, and explosion and resulting fragmentation from the adjoining stand; protection against seismic disturbances caused by natural or weapons test earth movements; and the dynamic loading of the wind.

Reference (13), which is the "Basis for Structural Design," indicates the basic approach and safety factors inherent in the design of the structure. Appropriate codes and standards, such as the Uniform Building Code, American Concrete Association, etc., have been listed to ensure standard accepted practice is followed. Those factors of most significance from the safety standpoint have been selected from this reference and in brief are:

Seismic Load - The load will be calculated on the basis that the site is located in a Zone III Seismic Intensity Area. This requires a structure capable of resisting a 0.1g ground acceleration for dynamic conditions. Both natural and nuclear weapons test phenomena will be countered by use of this design factor.

Wind Load - Basic wind load will be predicated on velocities of 80 mph at 30 ft aboveground and gusts of 125 mph. Appropriate increases in loading are indicated in this reference for higher elevations on the stand.

Credible Accident Loading - For design purposes, a 500-lb TNT equivalent blast at the adjacent stand is assumed, as described in Section 4 of this study. An explosion of this magnitude results in an overpressure of 0.3 psi at the planned stand separation distance of 1150 ft. This overpressure will be incorporated in the loading parameters.

Thermal Load - Incomplete information exists at this time in regard to thermal heating effects of the plume. Section 4 of this study outlines the approach to the problem and studies yet to be completed. It is assumed that the GASRAD Computer System and thermal studies will suffice to define the magnitude of thermal heating and the heat load to be guarded against. Thermal radiation shielding, water deluge and appropriately-sized cross-sections of exposed construction members are proposed for use and protection. It appears that an adequate design relative to protection against the thermal heating effects of the plume can be accomplished when the entire problem is clearly defined.

Nuclear facility radiation protection is provided by the shielding arrangement shown in Sheet No. 15 (Dwg. No. 6552-4 N-10 Rev. 0) of Reference (6), the Budgetary Study. The major sections or assemblies are the engine or facility shield, the intermediate shield, the diffuser inlet shield door and the lower shield. Together they provide radiation protection of the environment, the module, test stand facility and personnel. Preliminary studies, Reference (18), of shielding aspects of the test stands indicate that residual radiation levels, as a consequence of neutron activation of test-stand structural materials, are low enough to permit effective utilization of the test stand. However, this aspect of the design needs to be monitored during ensuing design to assure that selected structural materials do not alter this finding.

The facility shield is a water-filled shield which surrounds the engine during firing. It is floated into place in a moat and then locked into its raised position to prevent inadvertent lowering. The shield or tank is constructed of aluminum and filled with a 1% borate and water solution to provide the desired shielding. In the raised position it mates with the module support structure and the lower shield surrounding the diffuser throat to provide an environmental enclosure around the engine for altitude simulation.

Radiation shielding of the module is accomplished by use of an intermediate shield raised from below pad level to its operating position above the engine. The shield is constructed of aluminum filled with steel balls (for gamma attenuation) and provided with a circulating borate/water solution for neutron attenuation. This dual purpose shield also helps to provide protection from radiation from the diffuser section during shutdown conditions when the shield is retracted to its closed position below the pad. Protection for the open area (over the diffuser) is provided, however, by a "split-halves," horizontally-sliding door which is constructed of structural steel with an appropriately thick lead covering. During a test this shield door is withdrawn to a position outside the vertical facility shield and therefore is not subjected to significant neutron activation. Because of this, steel construction with lead overlay is appropriate and the door can then structurally support tracks for the EIV used for engine installation and removal.

The lower shield, like the intermediate shield, is constructed of aluminum and provided with a circulating borated water solution for neutron attenuation. This shield, in essence, is merely a "filler" in the area between the facility shield and the area of the diffuser supports and prevents activation and excessive radiation heating of the diffuser supports and the lower seals of the facility shield during a power run.

With respect to the shield design, integrity of the ETC is probably the greatest concern. This involves questions of seals, overpressurization, structural integrity, and cooling. Also of concern is ability to remove the shield following a test. As the design progresses, thorough review with regard to these aspects will be necessary.

5.4 NUCLEAR EXHAUST SYSTEM

The proposed nuclear exhaust system is described in some detail in Reference (6), the Budgetary Study. It can be identified as the 4:1 wet elbow concept in that document, which outlines three tentative systems. It is further described in Reference (19), which is the Duct Functional Specification prepared by the Aerojet-General Corporation.

Model studies by AGC have borne out the feasibility of this system and the specification outlines the requirements and parameters for the required system performance. A description of the system to accomplish the stated purpose is as follows:

The NERVA test stand and exhaust system are designed for a vertically oriented engine which fires downward into a duct system. The hot hydrogen gas is then turned 90° and directed away from the stand where it can be safely burned as it leaves the duct. The exhaust system also provides for altitude simulation by providing a low back pressure (2.3 psia max) on the engine during all phases of operation.

Design is based on an engine exhaust temperature of 5000°R with the nozzle exhaust stream directed downward into a vertically-mounted, water-cooled diffuser which acts as an altitude pump. The diffuser is in reality a second-throat supersonic diffuser and water is injected into the subsonic gas stream at the exit plane of the diffuser to cool the exhaust from 5000°R to 1000°R.

The expanded and cooled exhaust stream then is turned 90° (to a horizontal direction) by a wet elbow in the duct and directed through an ejector portion of the plenum before exhausting to atmosphere.

The ejector portion of the nuclear exhaust system is that portion where steam is injected to aid in pumping down the engine for altitude simulation during startup and shutdown of the engine. Steam injection also is used to preclude the formation of any hydrogen-air mixture in the duct during low flow conditions of hydrogen in the system. These low flow conditions normally occur during startup, shutdown, pulse cooling and perhaps during the mapping phase of reactor operation. The twofold purpose of the steam ejector can be readily seen and its importance as a safety check valve cannot be over-emphasized. (The operation of the steam generator system is described in the next section.) Water from the diffuser and the wet elbow portion of the plenum is drained off through a barometric drain which directs the water to a facility barometric well.

The Duct Functional Specification, Reference (19), provides the following design requirements which are of major significance from a safety standpoint: a duct system that is maintenance free and of such structural integrity as to be able to withstand one hundred 30-minute test firings; any combination of hydrogen and air in the exhaust system shall be prohibited at all times; no leakage of GH_2 from the NES; altitude simulation such that the engine test compartment will never be pressurized over ambient in event of any malfunction; safety features required to ensure reliability of the diffuser and elbow cooling water supplies to prevent duct system burnout, a diffuser section capable of withstanding the nuclear heating effects; and finally, materials of construction and fabrication techniques able to withstand the heat and cyclic effects of the engine test.

In addition to meeting the functional specifications set forth above, it is naturally necessary to assure that reactor fuel element particles cannot collect underwater in the duct in sufficient quantity to form a critical mass. It is also necessary to prevent these particles from collecting and forming a critical mass in any downstream portion of the drainage system.

The engineering design has not progressed to a point where finite answers to these problems are available, nor have failure mode analyses been performed on critical items. At this time, it appears the proposed design and approach is feasible but there are many unsolved problems both known and unknown. It is hoped that ETS-1 tests will do much to shed light on some of these areas so that the information gained can be incorporated in the downstream design effort.

5.5 STEAM GENERATOR SYSTEM

The motive driving force of the ejector (discussed in Section 5.4) is steam supplied by the steam generator system. The reliability is of paramount importance since the ejector is operated not only to pump down the engine and the test compartment, but also to prevent the influx of air into the duct. During any low hydrogen flow through the engine, it is required to operate the steam system and thereby prevent the formation of an explosive atmosphere in the duct.

Peak steam requirements are estimated to occur during engine startup and shutdown when the hydrogen flow rate is insufficient to aspirate the system. It is estimated that an engine chamber pressure of approximately 600 psi is required for self pumping. At this time, the steam flow rate can be reduced and maintained at a safety level in the event of an abort.

Steam flow required for engine startup is estimated at 4400 lb/sec with a temperature of 350°F and a molecular weight of 18. By contrast, the safety steam flow rate is only 1420 lb/sec and is that quantity required to prevent entrance of air into the duct with a sudden 35 mph gust directed against the duct exit. These conditions and the operating format just described are discussed in the previously noted Duct Functional Specification, Reference (19), which outlines the NES steam requirements and also describes the engine run power profiles along with the required steam flow rates.

The Kaiser Engineers' Budgetary Study (Reference (6), Sections 8 and 15) optimized a steam plant fueled by alcohol and utilizing liquid oxygen as the oxidizer. In that study, transfer pressures were estimated at 650 psig because of the boiler chamber pressures and it was further recommended that a pressurized transfer system for both fuel and oxidizer be used because of the reliability of operation.

Area Study F, Reference (9), indicates the steam plant will be located near the pipe runs and midway between the two test stands. Alcohol and liquid oxygen would be appropriately separated at DOD required distances and a fragment shield installed around the generators to provide shrapnel protection as described in Section 4.4 of this report.

The design of the steam system has not been undertaken at this time. From a safety standpoint, the key criteria that must be satisfied is that the reliability of the system must be such as to preclude hydrogen-air mixtures in the duct. When a design is selected, stringent safeguards must be incorporated into the system to guarantee safety and reliability of operation. The controllers, fuel and oxidizer transfer systems, steam supply lines, water supply and other facets of the system need to be critically examined to ensure reliability of operation.

5.6 CRYOGENIC FLUIDS AND HIGH PRESSURE GASES

The materials included in this category include liquid hydrogen, nitrogen and oxygen and the gas phase of hydrogen, nitrogen and helium. They are utilized as a propellant, for inerting and purging, as valve and control actuating medium, as oxidizers and for cooldown.

Hazards associated with the use of these fluids include: fire, explosion, asphyxiating atmospheres, oxygen enrichment, high pressure piping systems, disposal of flammable gases, and the storage and handling of high pressure gases and cryogenic fluids. In addition to personnel exposure to the above noted hazards, there is the problem of contact with cryogenic fluids or their containment systems at sub-zero temperatures.

A general approach and the design guidelines are outlined in Section 10 of the Budgetary Study, Reference (6). Further definition is derived from Area Study "F," Reference (9), and E/STS 2-3 Cryogenic and High

Pressure Piping and Miscellaneous Equipment (Bid Package #5 of 25 July 1967 Specification Section) which is Reference (20) to this study. In essence, these first two documents describe the operational aspects of the systems, while the latter is an equipment and materials specification. A general review of the system based on these references follows.

5.6.1 Liquid Hydrogen

Bulk storage of liquid hydrogen is maintained in a 1.3 million-gallon dewar, midway between the test stands and some 300-400 ft north of the control center. Two additional dewars of 28,000 and 101,000 gal capacity are also provided for operational requirements.

Standardized design is proposed for construction of the 80-ft diameter sphere (71-ft inside diameter) with a vacuum, perlite-insulated tank. The inner tank is proposed as stainless steel while the outer vessel is constructed of carbon steel as consistent with current fabrication techniques.

The dewar is protected by a system of relief valves and burst diaphragms in parallel. They are set at 110% and 120% of working presure, respectively. A double pair is provided with one installed for idle conditions (up to 5 psi tank pressure) and the other designed for run conditions where pressures of the order to 60-120 psi are expected. The vacuum insulated space is also protected by the same redundant scheme of pressure relief valve and burst disc to afford protection in the event of leakage from the storage sphere into the vacuum jacket.

Liquid hydrogen transfer from the dewar is accomplished by vaporizing the required quantity of the liquid in hot-water heat-exchange units and using this gas for pressurization. The pressurized fluid is then piped through super-insulated, vacuum-jacketed transfer piping to the module and other areas as required. Separate fill and withdrawal lines are provided at the dewars and they are provided with burst relief for the inner line as described above.

A complete description of the storage and transfer systems is provided in Sections 10 and 19 of Reference (6), the Budgetary Study. Our analysis of the design indicates general safety acceptance except for the lack of pressure relief for the vacuum insulated portion of the transfer pipe, the proposed location of burn stacks, proof testing of assembled systems and the treatment of a maximum credible accident at the dewar. For a discussion of this latter subject, refer to Reference (21) which deals with the problem of a massive spill at the main storage dewar. The other recommendations are outlined in Reference (20) which is a review of the piping systems.

The cryogenic and high pressure gas system requirements are such that a valve failure-mode analysis is necessary to determine that engine shutdown and cooldown can be satisfactorily performed in the event of a system malfunction. To accomplish this, it is recommended that such an analysis be made by the vendor-designer and included as a formal part of the detailed design of the cryogenic and high pressure gas systems.

Dewars and pipelines are provided with a high-level and a low-level venting system. The low-level system is defined as 0.5 lb/sec or less and is vented directly to the atmosphere. The high-level vent system is comprised of three separate flare stacks, one near each test stand and one located near the cryogenic storage area. Emergency venting, flash-off, chilldown and other high-level flows are directed to one of these burn-off points for safe disposal. Check valves are intended near the termination of the vent lines to prevent diffusion of air into the vent system. Redundant flares, ignition systems and monitoring devices will also be necessary to insure safe operation.

5.6.2 Gaseous Hydrogen

Bulk storage in high pressure tanks at 6000 psi is used for dewar pressurization, emergency cooldown and other miscellaneous uses at the module. Two tanks of 2352 and 435 ft^3 capacity (STP), complete with relief valves and rupture discs, perform the intended functions. Gas is conveyed through carbon steel pipelines to the use point. Reduced pressures of 4200 and 500 psi are used for distribution after regulation.

Venting for the GH_2 pipeline distribution system is consistent with the high-level disposal system described under Liquid Hydrogen. Storage tanks are merely vented to the atmosphere and with idle vents terminating at a safe elevation, safe or auto-ignition being incidental.

5.6.3 Liquid Nitrogen

Three dewars are provided for the storage of liquid nitrogen and have 165,000, 146,000 and 12,000 gallon capacities. They operate at 90 psi, 365 psi and 655 psi, respectively, and are provided with standard relief systems except that those transfer lines used for emergency cooldown are not provided with burst discs. These lines, instead, are provided with double parallel safety relief valves set at 120% and 140% of the operating pressures. Dewar pressurization is provided by a similar system to that utilized for hydrogen-liquid flow which is a system of hot water vaporizers designed to convert the liquid to gas.

5.6.4 Gaseous Nitrogen

Storage of GN_2 is confined to two tanks of 3030 ft^3 and 93 ft^3 (STP) which operate at pressures of 6000 psi. Distribution is at 130 or 2500 psi after regulation, depending on the intended use. The high pressure storage tanks and their fill and discharge lines are protected by pressure relief valves to prevent overpressurizing the system components.

5.6.5 Gaseous Helium

Several engineering considerations are currently underway in regard to the sizing and use of a helium system. At this time, the only comments appropriate are that the design of the system should be consistent with engineering design and practice for the utilization and distribution of high pressure inert gases.

5.6.6 Liquid Oxygen

The liquid oxygen system has not been completely defined as yet. Section 10 of Reference (6), Budgetary Study, indicates that a dewar of about 155,000 gal capacity is required for steam generator requirements. Area Study F, Reference (9), indicates the requirement for a transfer pressure of 650 psi, due to the operating chamber pressure of the generators. Pressurization is achieved through the use of high pressure nitrogen from the 6000-psi storage bank. The criteria for the cryogenic system call for a design that is identical to that of the nitrogen system. It is noted, however, that special care must be used in the installation of the vent system to avoid the discharge of O_2 in an area where this enrichment could pose a problem. Other safety and design features of insulation, line expansion, pipeline support, vents between any pair of valves, etc., must be comparable to those for liquid hydrogen or nitrogen systems.

5.7 ELECTRICAL POWER SYSTEM

Power requirements for the E/STS 2-3 operation are supplied through a 69/4.16KV Substation connected to the NRDS 69KV overhead distribution system. Bid Package #13 Electrical GFE (DTL 0110), Reference (22) to this study, describes the system. A review of this document indicates that the main substation is rated at 3750KVA, 60 cycle and is sized for a 50% increase in capacity as are other components of the distribution system.

Secondary power distribution at 4.16 KV is controlled through metal clad switch gear in the control center. Distribution transformers in the load centers are rated at 480/277V and are liquid-filled indoor type of Δ to Y configuration with the secondary neutral connected to ground. The I&C

Power Supply is of similar type but rated at 120/208V and located in the control center. It is paralleled with a stored-energy motor flywheel combination to provide 30 sec of rated output in event of major power failure.

As a consequence of prior reviews[23], a major change was effected in the latter part of September 1967 due to I&C and safety requirements for critical power. The redesign provides for an emergency diesel driven generator to be synchronized and paralleled with the main substation and provided with automatic tie-breakers to the secondary of the substation. In event of major power failure on the NRDS line or the main substation, this unit would be able to amply carry the load. It is rated at 12,000 KVA, 69/4.16 KV, 60∿, 3∅ and is physically located near the main substation. In addition, a so-called "super critical" power supply is provided in the I&C power system and is comprised of a floating battery of capacity to adequately meet power requirements for an emergency shutdown. With these major changes and refinements incorporated in the design, the emergency shutdown capability of the electrical power system appears adequate.

Except as noted at the end of this paragraph, the proposed system appears satisfactory. The design requires conformance to latest codes and standards, such as the National Electric Code (NEC), A.S.A., etc. In general, distribution wiring is in conduit where exposed and in duct where underground. Conduit is properly connected to the grounding system and all equipment is grounded through a separate ground wire carried along with the feeders. Grounding is accomplished through an extensive ground grid system with driven ground rods connected to the system to provide a maximum resistance to earth of 5 ohms. The provisions for lighting and illumination levels appear adequate. Electrical equipment for hazardous locations appears satisfactory also, with either classified or pressure purged equipment located in areas where explosive atmospheres could exist. With respect to the proposed installation of the liquid-filled transformer in the control center, it is recommended that it be enclosed in a NEC-defined vault or changed to a dry type in order to preclude generation and distribution of toxic fumes in the event of internal fault.

LIST OF REFERENCES

1. "Interim Nuclear Analysis in Support of E/STS 2-3 Site Layout Activities", RN-TM-0415, December, 1966

2. "Criteria for Preliminary Design of Engine/Stage Test Stand 2-3", prepared by SNPO, dated 20 January 1967

3. "Nuclear and Safety Analysis Interim Report", prepared by K.E., 31 January 1967

*4. Memo 7030:M1112, J. B. Philipson to A. Schaff, Jr., dated 4 August 1967, Subject: E/STS 2-3 Nuclear Based Intra-Facility Separation Distances

5. RN-S-1068, "Comparison of Combustion Plume Thermal Effects for ETS-1"

6. Kaiser Engineers Budgetary Study, May 1966

7. SNPO Memo B. J. Billings to Distribution, dated 11 April 1967, Subject: K. E. Comments on AGC Recommendations for ETS-1 Instrumentation to Determine Duct Exhaust Plume Thermal Parameters

*8. Memo 7030:M1063, J. B. Philipson to A. Schaff, Jr., dated 23 June 1967; Subject: Test Stand Deluge Water System

9. Kaiser Engineers Study Area F, dated 30 September 1966

10. Kaiser Engineers Study Area F, Preliminary Report, dated 3 August 1966

11. Interim Report on an Investigation of Hazards Associated with Liquid Hydrogen Storage and Uses, A. D. Little, Inc., Contract AF18(600)-1687 061092, 15 January 1959

12. Air Force Report URS 652-6, 31 August 1965

13. K. E. Interoffice Memo dated 19 May 1967 entitled: E/STS 2-3, Basis for Structural Design, Revision #1

14. Agreements Reached at Meeting Held in Oakland on 29 March 1967 to Establish Safety Criteria for the Design of the E/STS 2-3 Control Point Building

*15. Memo 7030:M1024, J. B. Philipson to A. Schaff, Jr., dated 15 May 1967, Subject: E/STS 2-3 Control Center Design Criteria

*16. Memo 7030:M1099, J. B. Philipson to A. Schaff, Jr., dated 24 July 1967, Subject: Control Center General Arrangements and Control Tunnel Sections and Details

LIST OF REFERENCES (cont.)

*17. Memo 7030:M1228, J. B. Philipson to A. Schaff, Jr., dated 16 Oct. 1967, Subject: Review of I & C Systems, DTL #0120 E/STS 2-3 Procurement Package No. 10

18. AGC Report RN-S-0390, "Interim Nuclear Analysis in Support of E/STS 2-3 Test Stand Preliminary Design" dated April, 1967

19. E/STS 2-3 Nuclear Exhaust System (NES) Duct Functional Specification, Technical Section, Final Issue, dated 15 August 1967

20. E/STS 2-3 Cryogenic and High Pressure Piping and Miscellaneous Equipment (Bid Package #5 of 25 July 1967 Specification Section)

*21. Memo 7030:M1105, J. B. Philipson to A. Schaff, Jr., dated 27 July 1967, Subject: LH_2 Retention Pond (E/STS 2-3 DTL 0078)

22. E/STS 2-3 DTL #0010, Bid Package 13, Electrical GFE, 15 September 1967

*23 Memo 7030:M1154, J. B. Philipson to H. Schaff, Jr., dated 1 September 1967; Subject: Safety Requirement for Minimum Emergency Power at E/STS 2-3

* NOTE: Those references marked * are contained in Appendix I to this study.

APPENDIX A

SITE DESCRIPTION

A.1 POPULATION AND LAND USE DATA

The NERVA test facility is located on the Nuclear Rocket Development Station (NRDS) in the Jackass Flats region of the Nevada Test Site (NTS) about 80 miles northwest of Las Vegas, Nevada. With the exception of the metropolitan Las Vegas area, this portion of the country is very sparsely inhabited and sustains little in the way of industry or agriculture. NTS is in Nye county, which has a population of ~4400. The populations of nearby communities are given in Table A.1. The rural population in the immediate vicinity has been added to the population of the named communities[1] Figure A.1 shows the general layout of the area and the relative distances between NRDS and these communities. Figure A.2 is a pictogram representing the human and bovine population for various sectors around NRDS. Several observations may be made:

a. The major population center is metropolitan Las Vegas, which has a population of about 250,000 and is located 70-80 miles southeast of NRDS.

b. The nearest population center is Lathrop Wells, which is about 14 miles from the test facility and has a population of about 400.

c. The immediate area to the east and north of NRDS is a part of NTS, with access and land usage controlled by the Atomic Energy Commission. Large areas to the east and north of NTS are part of the Las Vegas Bombing and Gunnery Range, a restricted area under the jurisdiction of Nellis AFB. There are no military quarters on this Range, nor do military personnel remain for extended periods of time.

d. The population within a thirty-mile radius is about 6,900, within 60 miles 10,700, and within 90 miles 261,000.

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(1) LASL H-8TCC-1, "Population and Dairy Cow Data Associated with Test Cell "C", December 1966.

TABLE A.1

POPULATION DATA

(1966 Estimate)

NEVADA

Clark County	
Glendale - Moapa	233
Moapa	73
Las Vegas (metro)	250,000
Esmeralada County	
Goldfield	154
Lincoln County	
Alamo	348
Caliente	858
Hiko	74
Panaca	580
Pioche	1,477
Nye County	
Beatty	1,535
Indian Springs	2,904
Lathrop Wells	437
Mercury	3,000
Pahrump	332
Tonopah	3,003

CALIFORNIA

Death Valley Junction	35
Lone Pine	1,400
Shoshone	140
Tecopa	200

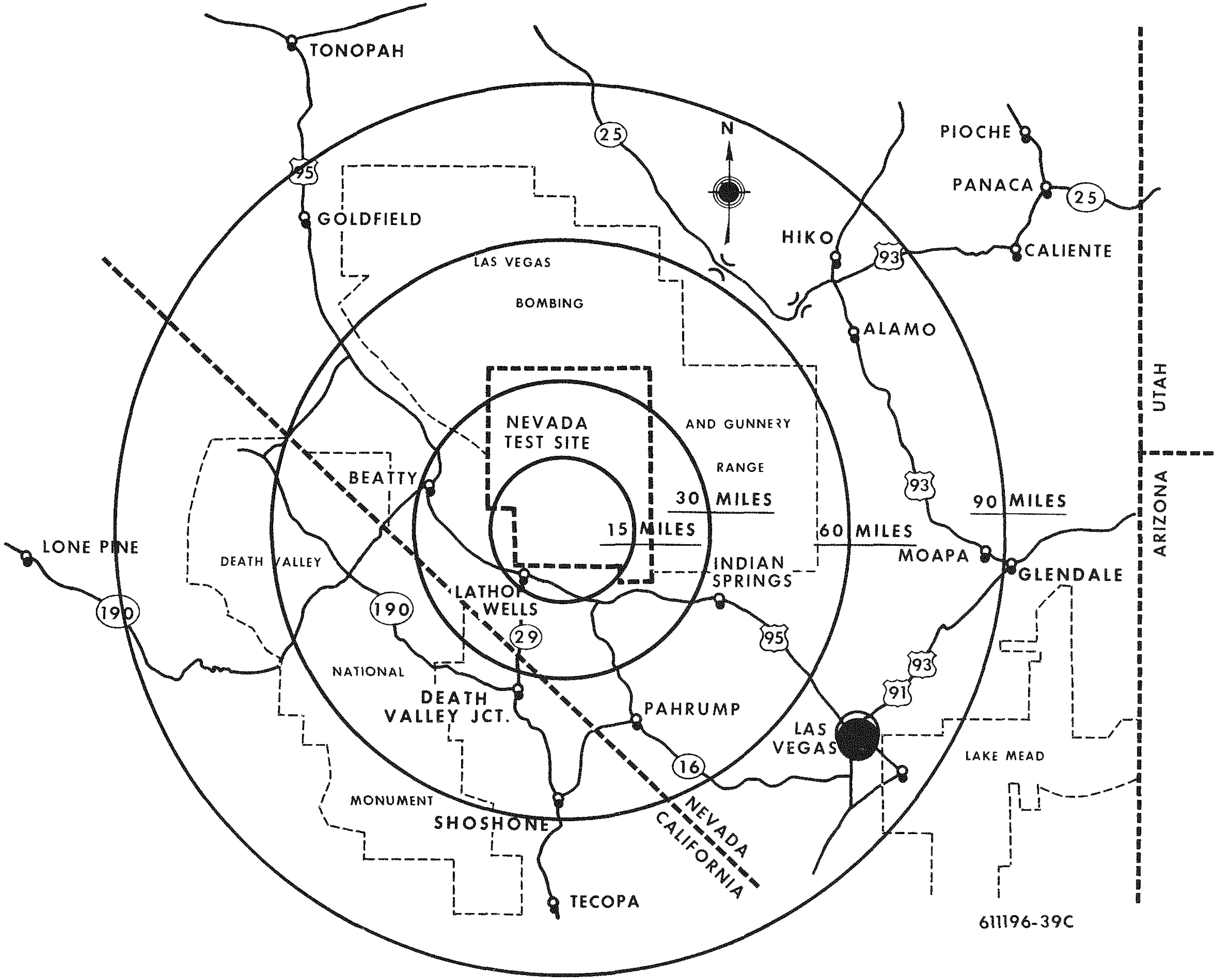

Figure A.1 Map of NTS and Vicinity

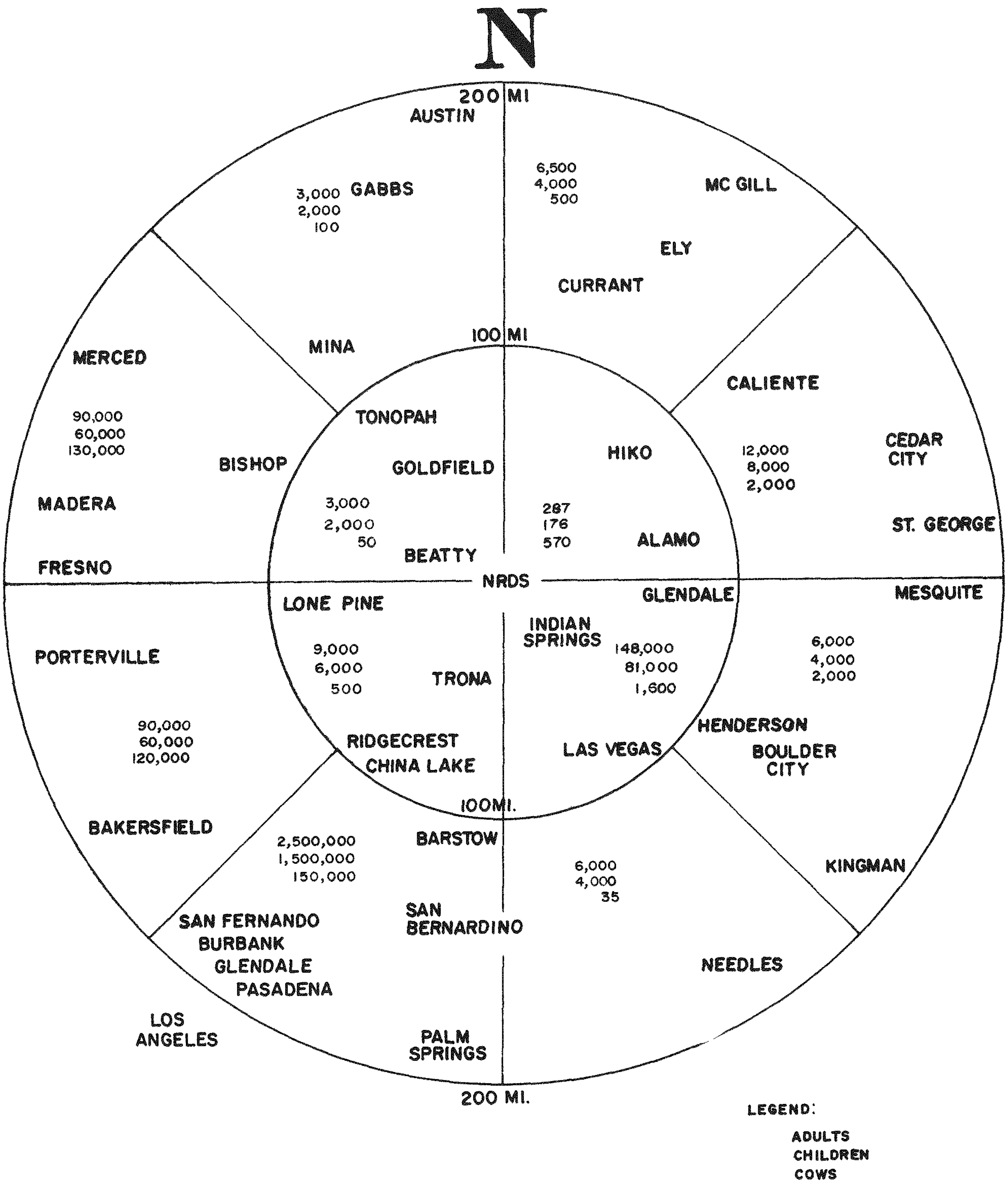

Figure A.2 Estimated Human and Dairy Cow Population Surrounding the NRDS

The southern Nevada area does contain numerous ranches and mines with a few large recreational areas including Charleston Peak (50 miles southeast of NRDS providing winter sports, summer camping, and picnic facilities) and Death Valley (45 miles southwest providing picnic facilities and winter camping). The general characteristics of vegetation and dairying practices of this area are shown in Figure A.3.

A.2 METEOROLOGY

The meteorological information presented in this section was obtained from Towers 4JA and 4, located about 5 miles to the southeast from the E/STS 2-3 site. The surface data may differ from E/STS 2-3 conditions to some extent, but the data for altitudes above 5000 feet MSL (site ~3800 feet) should be representative of conditions at E/STS 2-3 and applicable to off-site dose predictions.

A.2.1 Temperature

Surface temperature data are given in Table A.2[1]. The large mean daily range is typical of gently sloping terrain in mountainous desert areas. At comparable elevations where there is a tendency toward air stagnation the minimum temperatures tend to be considerably colder and the mean daily range may be on the order of 10°F greater. The daily minimum temperature usually occurs near sunrise with the maximum temperature occurring at about mid-afternoon.

The change of temperature with height determines the stability of the atmosphere; that is, the ability of atmosphere to suppress upward movement of an effluent. Neutral stability is represented by a temperature decrease at the rate of 5.4°F per thousand feet (dry adiabatic lapse rate). If the rate of decrease is greater than this the atmosphere is unstable and upward movement is unhampered. Instability is usually

(1) "Up-dated Weather Section, NRX-SAR", memo from H. G. Booth (ESSA, ARFRO, NRDS Branch) to B. Haertjens (LASL-J-17) dated January 31, 1967.

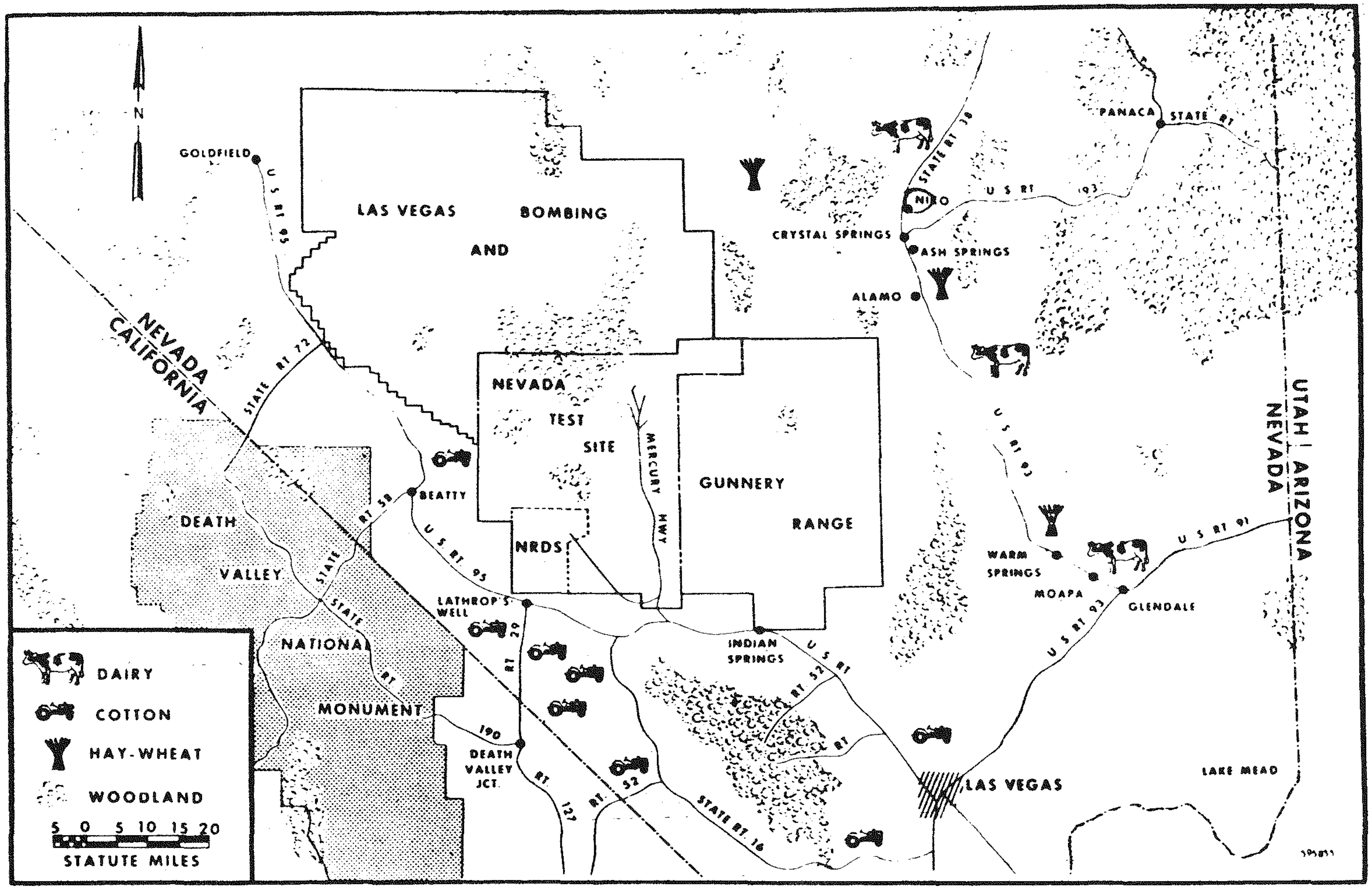

Figure A.3 Land Usage Map

TABLE A.2

MEANS AND EXTREMES OF TEMPERATURE (DEGREE F)

	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Annual
Highest	72	76	80	90	97	110	110	106	100	94	83	75	110
Mean Daily	54	58	62	72	79	90	97	94	88	77	63	58	74
Mean	43	47	50	59	66	76	82	80	74	64	51	47	62
Mean Daily Minimum	32	35	38	46	52	61	68	67	60	52	39	35	49
Lowest	7	14	22	27	32	42	54	50	45	33	22	18	7
Mean Daily Range	22	23	24	26	27	29	29	27	28	25	24	23	25

TABLE A.3

MEAN MIXING DEPTH IN FEET ABOVE GROUND

Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
1900	4900	5100	7100	9000	11500	11600	11100	7600	3600	2400	1900

confined to a shallow layer near the ground when the sun is shining brightly. Above this shallow unstable layer nearly neutral conditions exist to a height which reaches a maximum depth at a time coincident with the maximum surface temperature. The mean maximum mixing depth is given in Table A.3[(1)] as the depth of the mixed layer in feet. This mixing depth with essentially neutral lapse conditions represents the height above the ground to which contaminated air will eventually rise without appreciable retardation.

Above the mixed layer the temperature generally decreases at a rate less than the dry adiabatic. The presence of such stable layers exerts a pronounced influence on the mixing depth and thus the height to which an effluent will rise. The data presented in Table A.4[(1)] provide the distribution with height of the base of significant stable layers. A significant stable layer is defined as one which strongly inhibits vertical motion and corresponds to a lapse rate of less than 2.3°F per thousand feet. The analysis was based on observations at 0400 PST and excluded surface based stable layers.

An inversion is a more intense stable layer in which the temperature increases with height. A surface based inversion is a common occurrence at night due to radiational cooling and reaches a maximum intensity just before sunrise. The data presented in Table A.5[(1)] provide the frequency of occurrence of low level inversion at two times of day and the percent of hours with low level inversion by season.

A.2.2 Wind

A generalized analysis of the winds in the vertical direction is presented in Figure A.4[(1)]. The isolines are drawn for values of constancy and provide a relative measure of the variability of the wind on a

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(1) "Up-dated Weather Section, NRX-SAR", memo from H. G. Booth (ESSA, ARFRO, NRDS Branch) to B. Haertjens (LASL-J-17) dated January 31, 1967.

TABLE A.4

PERCENTAGE PROBABILITY OF THE BASE OF A STABLE LAYER
(EXCLUDING SURFACE BASED STABLE LAYERS) BEING BELOW SPECIFIED HEIGHTS
IN FEET ABOVE MEAN SEA LEVEL

Height (MSL)	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
18,000	55	46	44	41	32	27	16	21	43	39	67	56
16,000	53	44	43	39	28	25	13	18	38	37	65	55
14,000	49	41	39	37	23	21	10	14	31	33	60	53
12,000	43	36	33	33	18	16	7	10	23	28	53	49
10,000	37	27	25	26	13	11	4	6	14	20	42	42
9,000	32	22	20	22	11	9	3	5	11	16	37	38
8,000	26	16	15	18	9	7	2	4	9	12	30	32
7,000	19	10	10	13	6	5	2	3	6	8	23	24
6,000	12	6	6	9	4	4	1	2	4	5	15	15
5,000	6	3	3	5	2	3	1	1	2	3	8	8

TABLE A.5

FREQUENCY OF OCCURRENCE AND DURATION OF LOW LEVEL INVERSIONS BY SEASONS

Season	% of Days At 0400 PST	% of Days At 1600 PST	% of Hours	Mean Duration
Winter	92	2	54	13.0
Spring	86	0	39	9.4
Summer	89	1	37	8.9
Fall	90	0	50	12.0

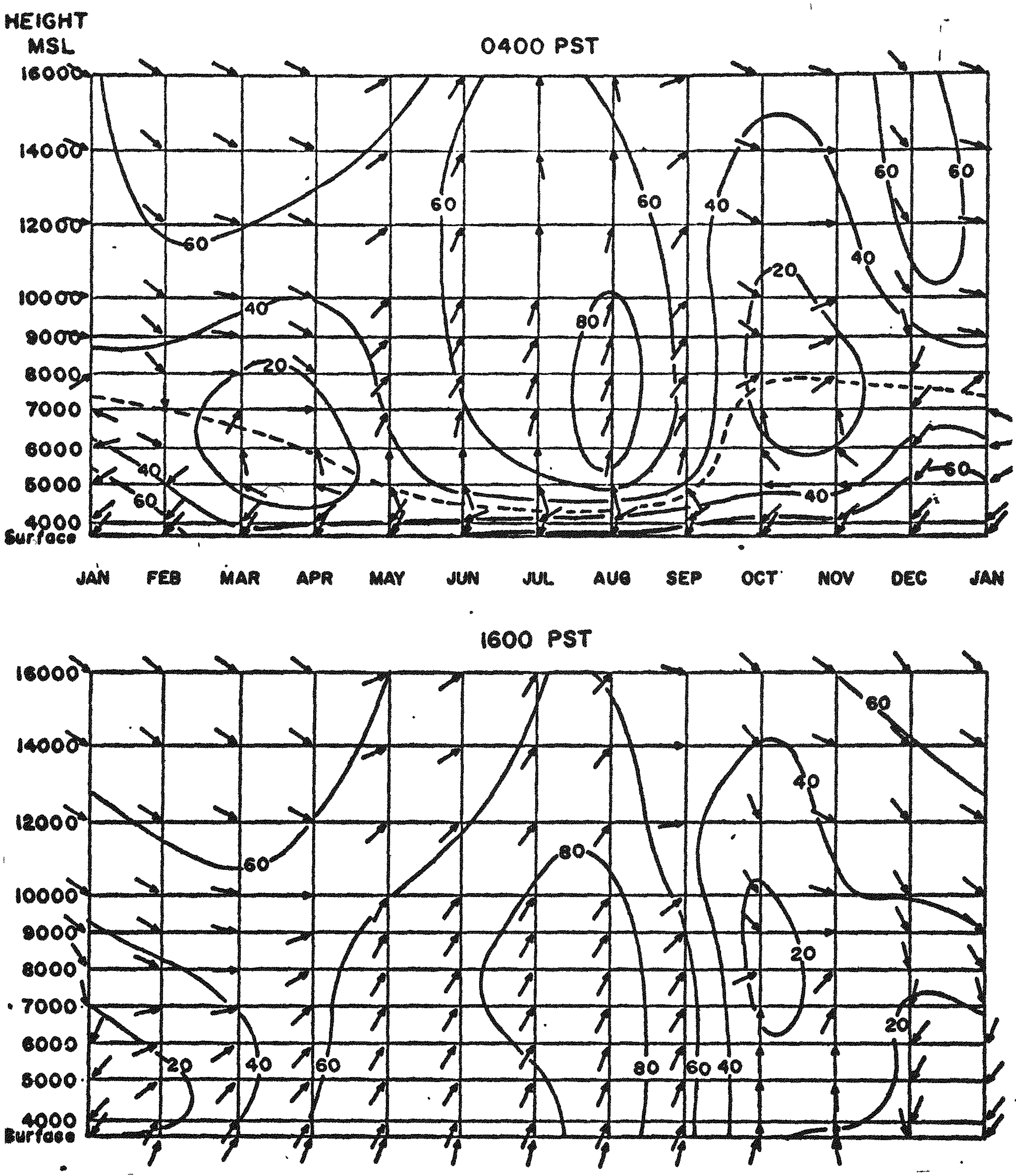

Figure A.4 Relative Variability of the Wind and Vector Mean Wind Direction as a Function of Height and Time of Day scale from zero to 100. Constancy is the percentage ratio of the magnitude of the vector mean wind to the scalar mean wind speed. The scalar mean speed is independent of the wind direction and will therefore always be equal to or greater than the magnitude of the vector mean wind. If the wind at a specified time of day for a given month always blew from the same direction the vector mean and scalar mean would be equal and the constancy would be 100. This condition is most nearly achieved during the afternoon hours in summer as indicated by constancy values above 80 in Figure A.4. With high values of constancy the vector mean wind coincides very nearly with the most frequent wind direction; that is, it approximates the middle of the distribution of wind directions.

When the wind direction is highly variable at a specified altitude the magnitude of the vector mean wind is much smaller than the scalar mean speed and low values of constancy ensue. The dashed line on the chart for the hour 0400 PST is drawn through the region of greatest wind variability (low constancy). From May through September the line provides a good approximation to the mean depth of the surface inversion; however, this analogy breaks down for the months October through April due to a generally high degree of variability up to about 10,000 feet MSL. The surface inversion is destroyed during the day as seen by the alignment of the low level vector mean winds with those at higher levels in the analysis for 1600 PST. Figure A.5 presents an analysis of frequency of wind direction versus altitude for the various seasons. With the exception of winter, there is a quite pronounced peak in the 160-240 degree sector.

A.3 CONCLUSIONS

It is obvious from Figures A.1 and A.2 that the most desirable direction for the engine effluent to be carried is toward the north. The government controlled land extends the furthest in this direction and the

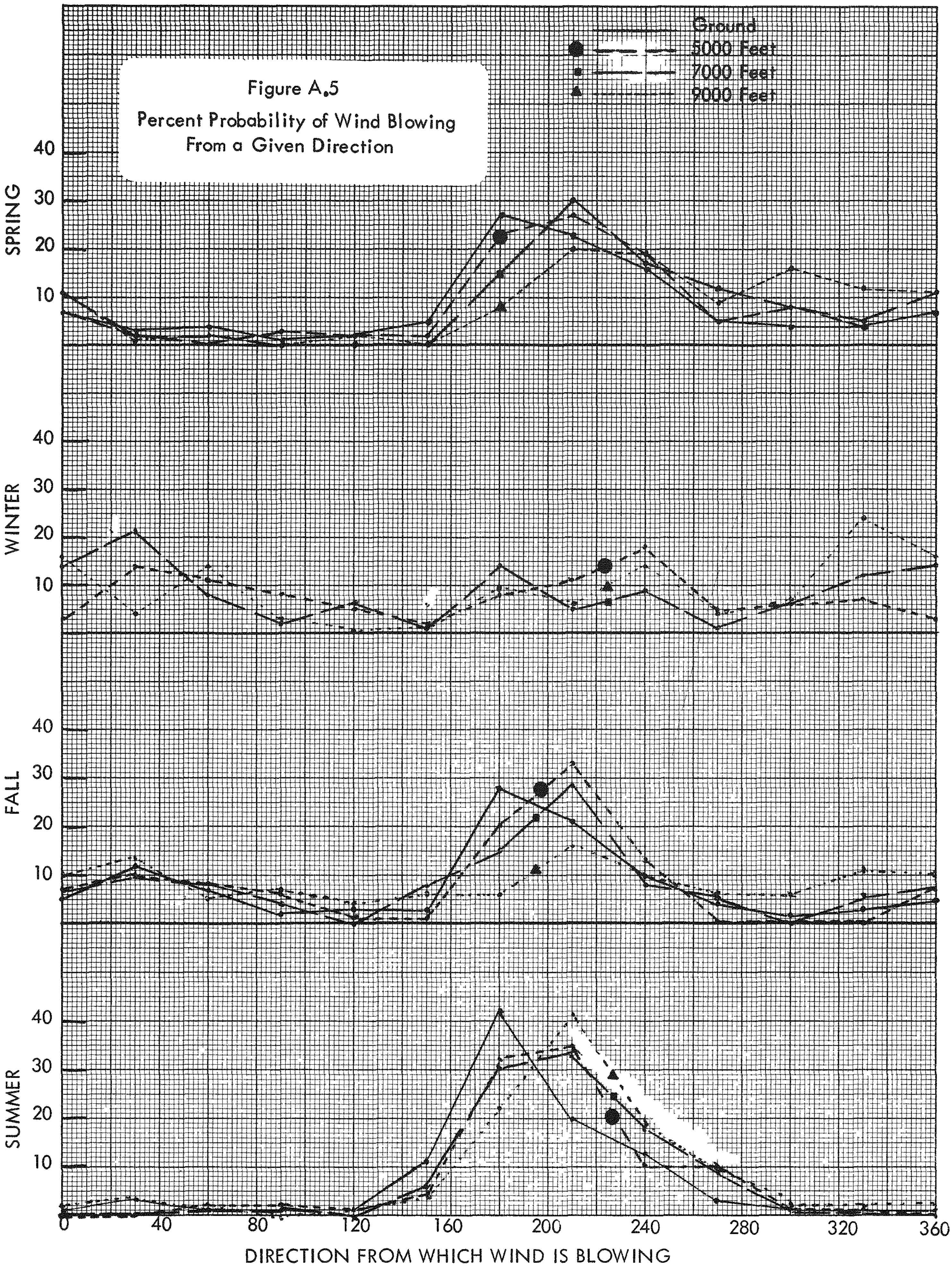
Figure A.5
Percent Probability of Wind Blowing From a Given Direction
Ground
5000 Feet
7000 Feet
9000 Feet
PERCENT PROBABILITY
SPRING
WINTER
FALL
SUMMER
40
30
20
10
0
40
80
120
160
200
240
280
320
360
DIRECTION FROM WHICH WIND IS BLOWING

off-site population density is low. As shown by Tables A.3, A.4 and A.5, the most optimum time for test operations from a standpoint of maximum mixing layer depth and minimum probability of an inversion condition is May through August. Figure A.5 indicates that for the spring, summer and fall seasons there is a high probability that the effluent will be carried in the optimum northerly direction.

APPENDIX B

E/STS 2-3 FACILITY DESCRIPTION

B.1 GENERAL DESCRIPTION

The E/STS 2-3 complex will provide ground testing capability for NERVA and larger engines, completely assembled with the associated modular propellant tank (MPT) assembly. The original design of the complex will accommodate testing of a 5000 MW engine (250,000 lbs. thrust) but provisions have been made for future modifications to permit testing of a 10,000 MW engine (500,000 lbs. thrust) at each test position.

The E/STS 2-3 complex will be located at NRDS approximately 10,000 feet west of ETS-1. Figure B-1 (Dwg. C1-1101) indicates the relative location of principal components. Location of the complex and distances between the components of the complex are dictated by access criteria and the KE study report A-2 titled "Nuclear & Thermal Radiation Analyses" (SNPN-20).

Utilization will be made of existing site facilities as follows:

a. Principal road access by extending road "H" and constructing a new road "L".

b. Connection to site domestic water supply loop at the closest feasible point on the site loop system.

c. Constructing a new power substation and feeder takeoff at the closest feasible point on the site loop system.

d. Extending the railroad system from the closest existing point (at ETS-1) to the new site to utilize the E-MAD facilities, radioactive storage yards, etc.

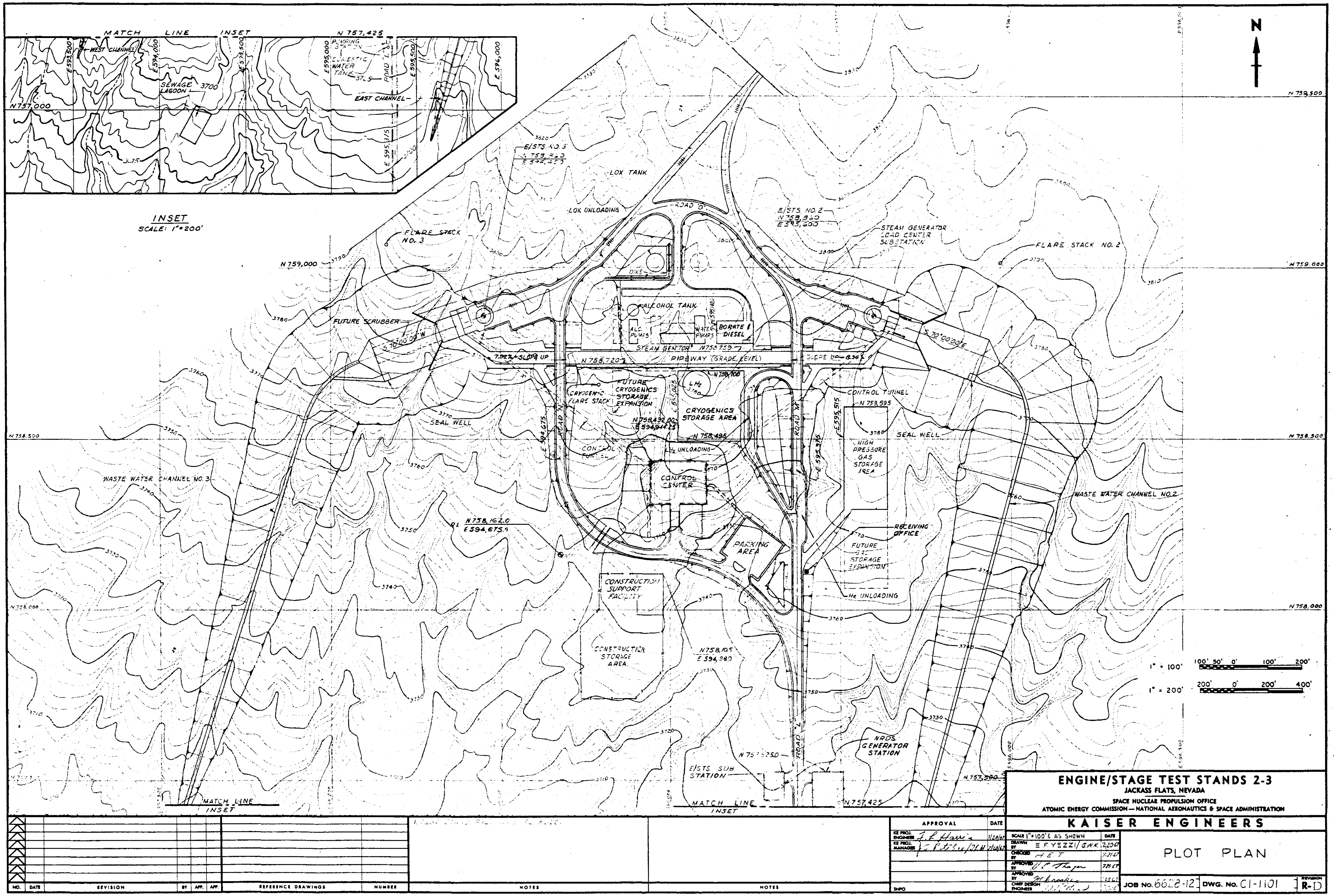

Figure B.1 - E/STS 2-3 Plot Plan (Dwg. C1-1101)

The following reference documents contain backup information and details in support of the engineering design documents, construction drawings and specifications:

a. Budgetary Study - Kaiser Engineers.

b. Revised Criteria for Design - SNPN-20, Jan. 1967.

c. Study Task Report A-2 (SNPN-20, Architect - Engineer and Related Services in Connection with Engine/Stage Test Stands 2-3).

d. Design Bases, Prepared by KE.

e. NERVA Module Program Specs (SNPO).

f. Minutes of SNPO-KE Meetings (Task Group).*

g. File of Design Information Memoranda (Task Group).*

B.2 TEST STANDS

Each test stand (2 and 3) is capable of testing a completely assembled Nuclear Ground Test Module (NGTM) at full rated power for a duration of 30 minutes (45 minutes for the future 10,000 MW engine), and in an environment simulating as nearly as feasible the actual flight environment. The overall dimensions of the stands are shown on Figure B-2, Dwg. C3-3125 attached. The two test stands are essentially identical.

B.2.1 Module Support Structure

The module support structure consists of a water-cooled aluminum ring girder rigidly connected to four fabricated steel support columns which are bolted to the concrete substructure. Future reductions in engine length from that specified in the criteria will be accommodated by shortening the columns and thereby lowering the elevation of the ring girder which supports the module from which the engine is supported. Thus, the engine exit plane elevation remains essentially constant with varying engine lengths.

*These documents are in available files of the E/STS 2-3 task group and will not be generally distributed.

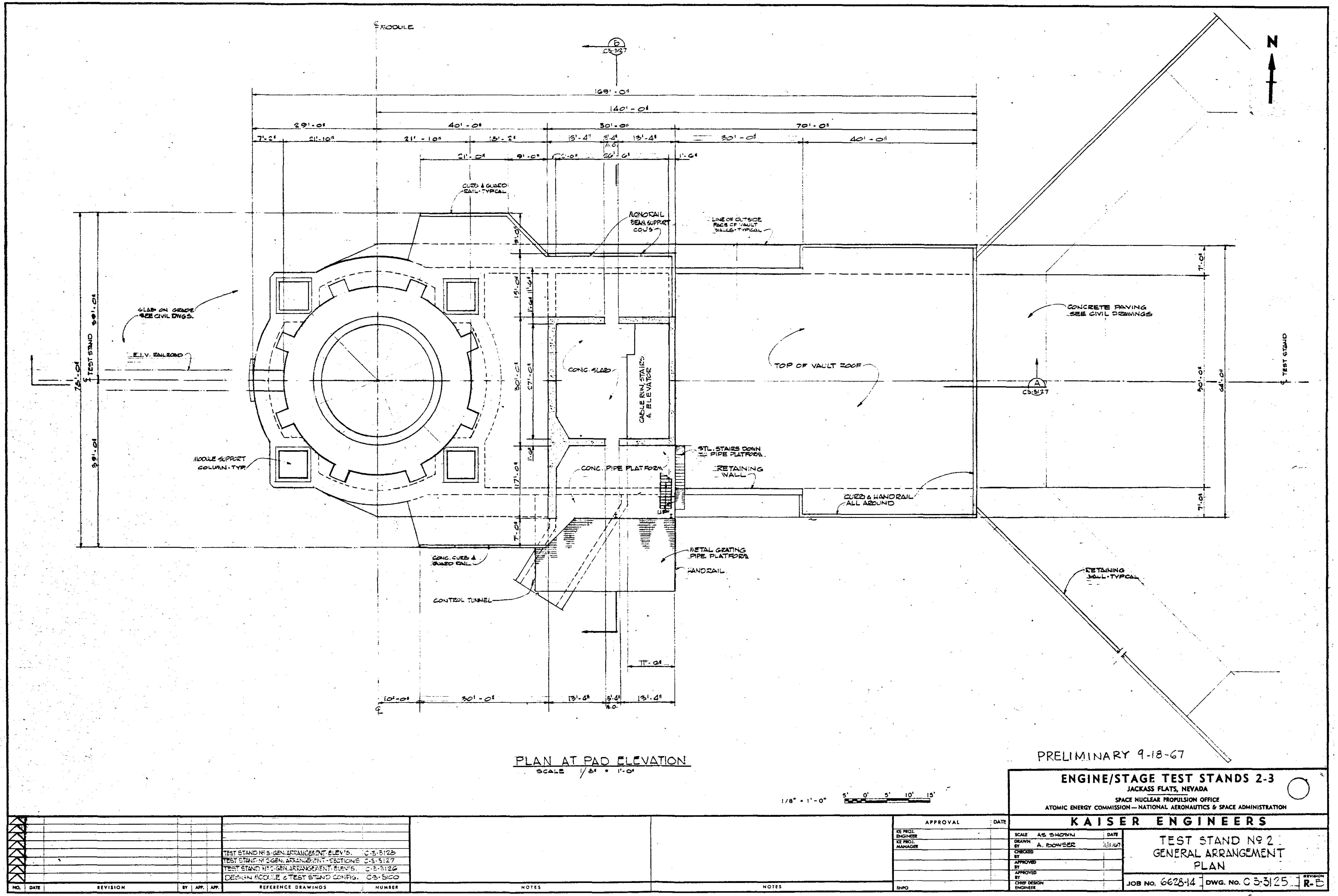

Figure B.2 - Test Stand Plan (Dwg. C3-3125)

B.2.2 Facility Shields

The facility shields, together with the module tank bottom and the water-cooled support ring, form a cylindrical enclosure which contains the NERVA engine. The enclosure is composed of three component shields; side shield, bottom shield, and intermediate shield. The functions of the enclosure are: (a) to provide a safe oxygen-free operating environment for the engine during test; (b) to permit reduction in ambient atmospheric pressure around the engine in order to partially simulate space environmental conditions; (c) to simulate nuclear radiation environment existing during a flight or mission; (d) to reduce deleterious effects of nuclear radiation on facility components located outside the engine compartment; and (e) to allow greater personnel access to facility areas following the engine shutdown.

All facility shields are fabricated from aluminum and filled with borated water. The top shield is a segmented annular tank mounted on the support structure ring girder and outside the module tank supporting skirt. It is not a part of the altitude environment chamber and serves only to attenuate radiation that penetrates the module support skirt. The cross section of the top shield is presently established as 11 ft 2 in. high by 5 ft 6 in. wide.

The side shield is an annular cylinder approximately 30 ft 6 in. inside diameter, 43 ft 2 in. outside diameter and 38 ft 9 in. high. After installation of the NERVA test engine, the shield is floated vertically upward from its storage vault beneath the test pad and sealed against the bottom of the support structure. Mechanical locks hold it in the raised position and seals between the shield wall and the cylindrical vault wall are provided. The shield is filled with borated water after being securely locked in position.

The bottom shield is permanently mounted to the structure and is an annular water-cooled cylinder approximately 14 ft 3 in. inside diameter, 30 ft 0 in. outside diameter and 5 ft 0 in. thick. Its top elevation is set so that it supports the intermediate shield in the stored position and permits a grating work platform to be placed on top of the intermediate shield and flush with the concrete test stand pad.

The intermediate shield, stored on the lower shield when not in use, is raised remotely by cables to the same elevation as the upper surface of the radiation shield located within the NERVA engine pressure vessel. If a mating engine intermediate shield is found to be necessary, the facility intermediate shield will structurally support and mate with it. The purpose of this shielding arrangement is to attenuate the direct and scatter radiation from the reactor and thus provide better simulation of flight environment radiation on the module tank bottom. After being raised to position, the intermediate shield is locked in place and connected to a supply of circulating borated water.

B.2.3 Substructure

The substructure forms the foundation for the module support structure, the service tower, and test stand access areas. It is reinforced concrete with necessary embedments to accommodate utility services, mechanical devices, and service access passages for not only the 5000 MW engine but also the 10,000 MW or growth engine.

B.2.4 Duct Vault

The duct vault is an adjunct to the test stand substructure and is sized to house either the 5000 MW nuclear exhaust duct or an exhaust duct for the growth engine. It is provided with a sealed closure door which will permit inerting of the vault and also allow for water discharge to the exterior drainage ditch through an atmospheric trap. Rails imbedded in the vault floor will support the NES duct transport and installation carriage both for the initial installation and when later remote removal of the duct becomes necessary.

V

B.2.5 Drainage Ditch

Each test stand is provided with a drainage ditch which has three primary functions: (a) to direct the hot hydrogen effluent horizontally away from the test stand and thus minimize the nuclear and thermal protection required for the test facility; (b) to conduct excess cooling water away from the facility without the use of pumps; and (c) to facilitate access to the NES and associated equipment by heavy transport equipment. It should be noted that the overall height of the facility, from the top of the service tower to the invert elevation of the vault (250 ft) has been minimized and that the test stand pad elevation and the resulting ditch invert elevation were established after an economic study that reflected the minimum estimated construction cost.

B.2.6 Service Towers

Each test stand is provided with a service tower whose gross dimensions are as shown in Figure B-3 (Dwg. C3-2103). The tower serves three distinct functions: (a) support for the service or umbilical arms that provide all fluid requirements and instrumentation and control connections to the NGTM; (b) shield the NGTM from the thermal and nuclear radiation from the NES exhaust plume; and (c) provide housing for equipment that must be located in a shielded area in close proximity to the NGTM. It is designed to be increased 46 ft in height to serve the enlarged module which contemplates the 10,000 MW nuclear engine.

Each service tower is equipped with an elevator, internal and external stairs, work platforms, a change room with monitoring facilities, ventilating system, lighting, etc. for the safety of personnel.

The substructure, in addition to supporting the service tower superstructure provides shielded housing for instrumentation and control equipment, heating, ventilating and air conditioning equipment, power

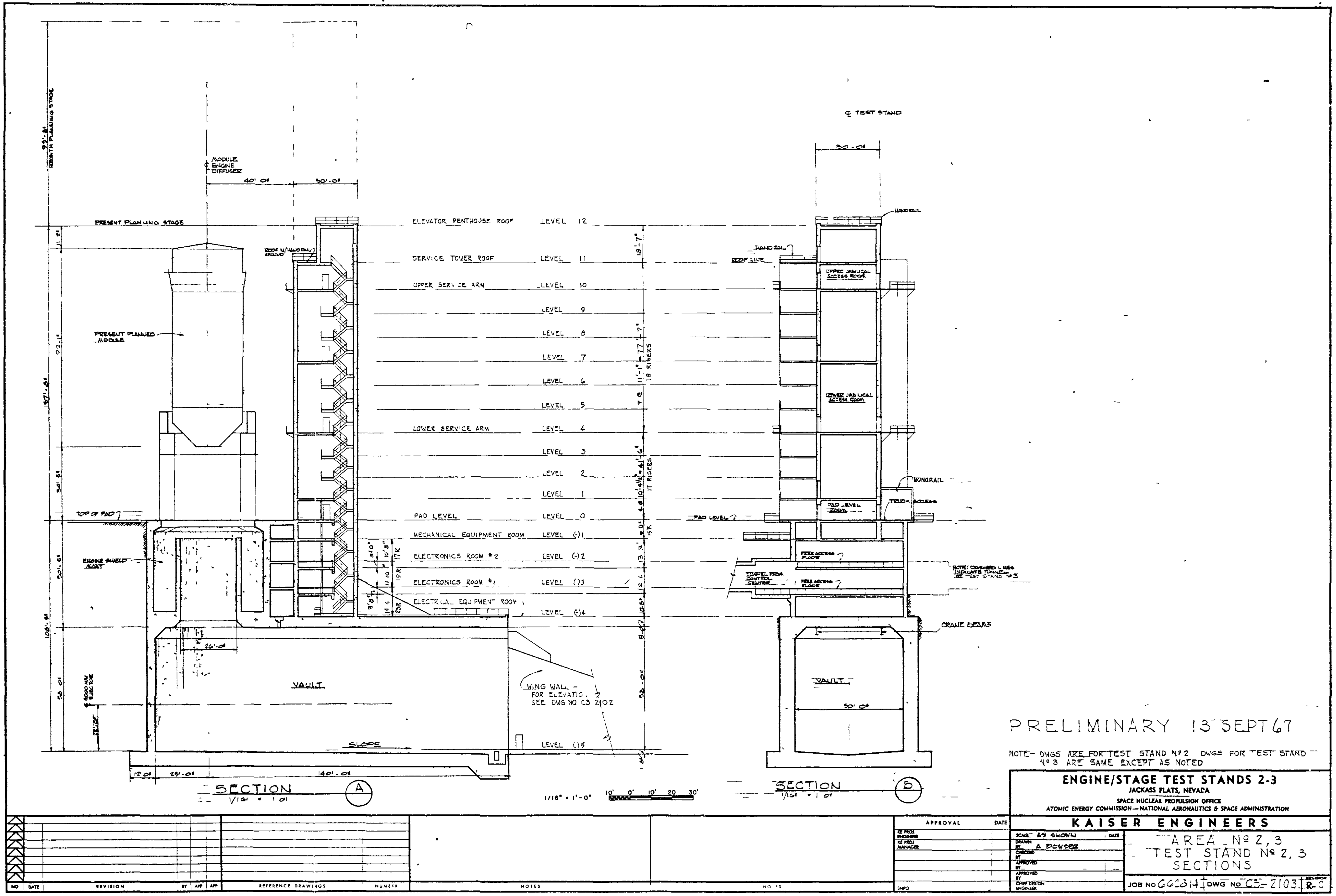

Figure B.3 - Test Stand Sections (Dwg. C3-2103) distribution panels and, through connection to the service tunnel, access to the control center.

Industrial water is used for deluge and fire protection for each service tower and umbilical arm. Provisions are made for future connections to provide deluge or fire protection water within each service tower. Domestic water is provided for personnel use only in the change room, toilets, emergency showers, etc.

Electric power is distributed as required to the interruptible-uninterruptible-, and critical-power buses within the service tower.

A high pressure CO_2 fire extinguishing system is provided in each tower substructure to protect the electronic and electrical equipment not accessible during a test run. Automatic discharge is preceded by adequate personnel warning and system capacity will provide for two separate inertions of the protected area.

Hydraulic power for operation of the intermediate shield hoist, hydraulic valve actuators and other mechanical devices is supplied from pumps located in the mechanical equipment room of the service tower.

B.2.7 Control Tunnels

Control tunnels connecting each service tower substructure with the control center serve three specific functions: (a) provide shielded, protected routes for all instrumentation, control and power systems to each test stand; (b) provide clean, dry tempered ventilation for each service tower from mechanical equipment located in a remote nonhazardous area; and (c) provide personnel access between the control center and the service tower at all times other than during test operations. The concrete tunnels are 10 ft high by 10 ft wide inside dimensions and are constructed with tee slots and supports cast in place to support cable racks, piping, etc.

B.3 CONTROL CENTER

The control center is a two-story underground structure, shielded for safe personnel occupancy under all credible testing and post-testing hazards. Figures B-4 and B-5 (Dwgs. C5-2120 and C5-2121) show floor plans, dimensions and space allocation. Personnel in the building during test operations will be limited to those whose presence is essential. During test operation, the building occupancy is expected to be limited to 50 persons.

The heating, ventilating and air conditioning for the control center is designed providing zoned conditioned air flow for electronic, electrical and mechanical equipment cooling as well as for personnel comfort. Filtered and tempered outside air is introduced to maintain a positive pressure in the building with respect to the ambient atmosphere and to supply air via the control tunnels to each service tower from which it is discharged to atmosphere. During non-test periods, in case airborne radioactive particulate matter or gaseous hydrogen is detected in the air intake, dampers close immediately to prevent the hazardous atmosphere from entering the control center. During test periods, the air intake will be closed and the control center sealed.

Safety for personnel during test operations and after a maximum credible accident (MCA) is provided as follows. Shielding over the control center roof (equivalent of four feet of concrete) satisfies the unlimited access requirement established by the facility design criteria. Evacuation after a MCA can be accomplished by busses (parked in a shielded shelter) when rad safe conditions prevail as determined by health physics personnel. During confinement after a MCA all personnel required to remain at the control center are supplied with at least a 24-hour supply of uncontaminated breathing air, stored for that purpose in tanks within the control center structure. Sealed doors in the control tunnels prevent infiltration of air through the tunnels and service towers during emergencies.

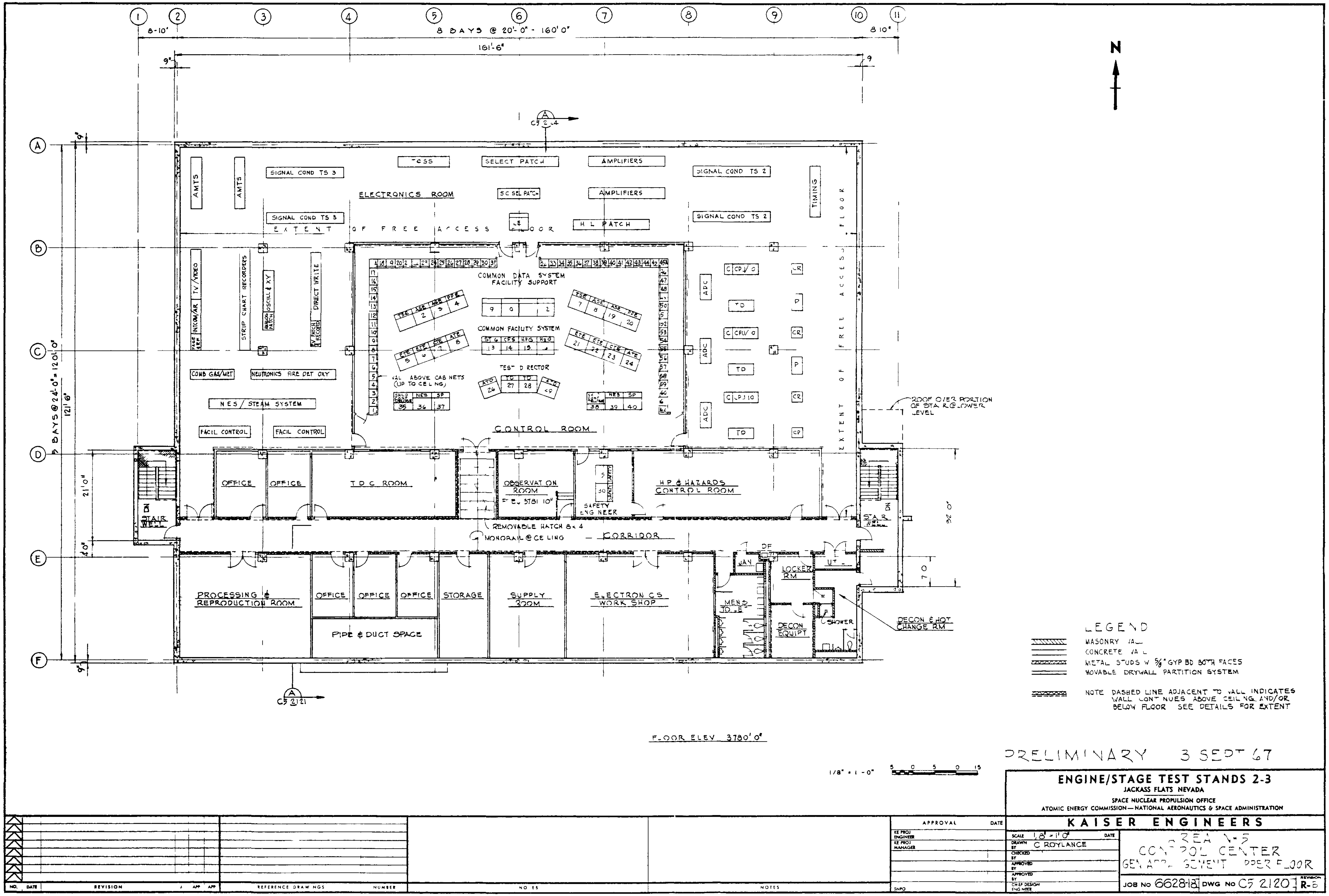

Figure B.4 - Control Center Upper Floor Plan (Dwg. C5-2120)

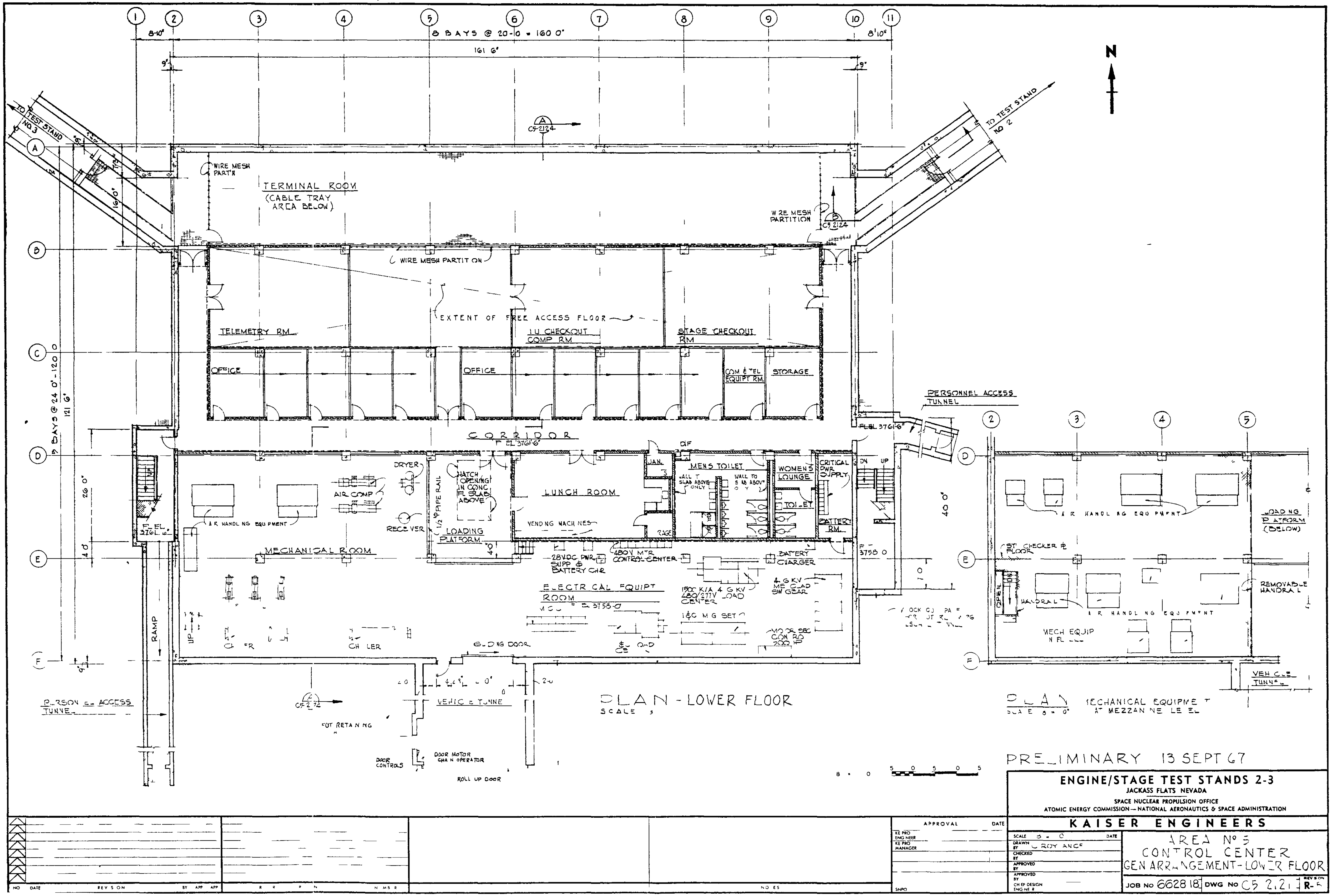

Figure B.5 - Control Center Lower Floor Plan (Dwg. C5-2121)

B.4 NUCLEAR EXHAUST SYSTEM

The nuclear exhaust system (NES) which creates altitude environment for the test article and conveys the hot hydrogen from the test stand structure, is comprised of the following three principal components: (a) exhaust duct, (b) steam generator and deluge, (c) waste water disposal.

The exhaust duct is a water tube diffuser ejector, driven by the NERVA exhaust nozzle exit gas and horizontal plenum assembly. It conducts the hydrogen from the NERVA engine vertically downward through a water injection station to a horizontal cylindrical plenum equipped with a deflector plate which turns the hydrogen stream 90° to the horizontal direction. The diffuser ejector is supported on the horizontal plenum and connected to the bottom shield by a pressure tight bellows. The total load of the assembly, including the plenum and deflector plate, is carried on support trunnions built into the concrete vault. The duct configuration is shown in Figure B-6 (Dwg. C3-4500). Industrial water required for cooling is 100,000 gpm at 100 psig at pad level for a total of 4,950,000 gallons for a 30-minute full power run, including start-up and cooldown.

The steam generator and delivery system supplies steam to power the secondary or steam ejectors that are mounted horizontally on the end of the cylindrical plenum of the duct. These ejectors provide low back pressure for the NERVA engine during startup and shutdown and maintain safe atmosphere within the duct in case of engine malfunction. The ejector steam is supplied from generators (located on grade approximately midway between the two stands). The generators supply 3700 pps of steam at 113 psia for a total time of 75 minutes. The run profile requires different flow rates for standby, safety and operating conditions and extends for a total time of 75 minutes. Total steam required, because of reduced flow rate during full power operation of the NERVA engine, is 4,350,000 pounds.

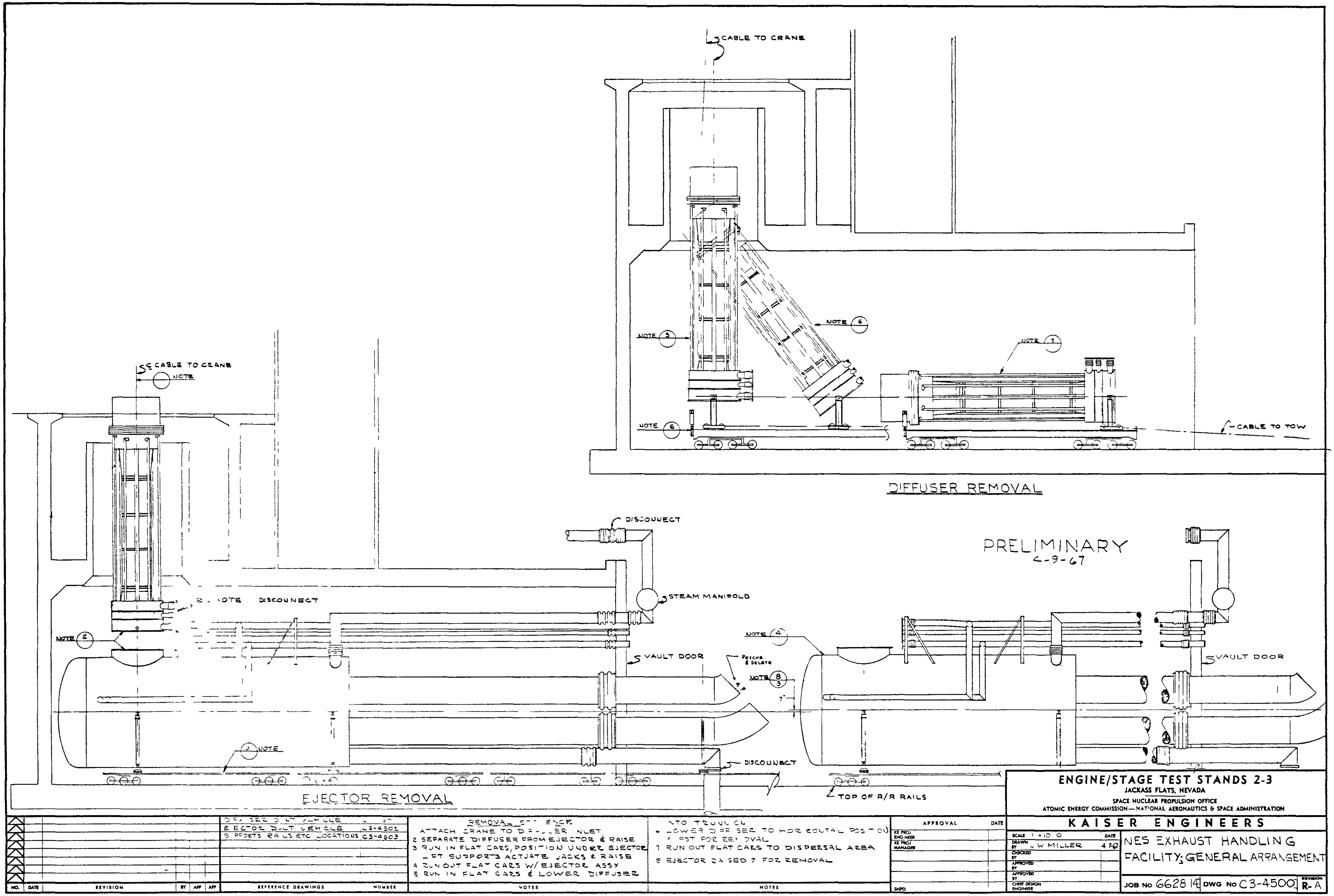

Figure B.6 - NES Configuration (Dwg. C3-4500)

B.5 FLUID SYSTEMS

Fluid systems associated with the E/STS 2-3 test complex are provided from centralized supply and storage areas and are transported through suitable controlled piping systems to the several use points either at the test stand interfaces or specific site locations.

B.5.1 Liquid Hydrogen

Liquid hydrogen is stored in the cryogenics area and is transported through vacuum jacketed lines to either of the test stands or to the vaporizers. Data pertaining to LH_2 storage and use are as follows:

Designation Storage Vessel	Gallons Capacity	PSIG Pressure	30 Min. 5000 MW	Primary Function
V-LH-1	1,362,000	90	1,225,000	NGTM Supply
V-LH-2	102,000	180	92,000	LH_2 press & purge
V-LH-3	28,000	360	25,000	Engine cooldown
V-LH-4	5,000	1,500	5,000*	Engine cooldown
V-LH-5	5,000	1,500	5,000*	Engine cooldown
Total	1,492,000 gal.		1,343,000 gal.	

*Transferred from V-LH-2 immediately prior to test run.

A tabulation of the liquid hydrogen use requirements is contained in "Ground Test Module Fluid Interface Requirements" published as an enclosure with MSFC memorandum of April 6, 1967 (R-P&VE-XN-67-37).

LH_2 instrumentation and controls will be provided under the several bid packages for storage dewars, piping, and I&C systems. The TCSS will incorporate LH_2 I&C functions that are directly related to dynamic test variables, utilizing sensors, amplifiers, etc. furnished by others.

LH_2, stored in vessel V-LH-2, is converted to gaseous hydrogen and used to purge and pressurize the NGTM. Capacity for generation of GH_2 is 16 pps at 195 psia. In addition, air heated converters along with compressors are utilized to pressurize the storage dewars and to recharge gaseous hydrogen storage vessels.

B.5.2 Gaseous Hydrogen

Gaseous hydrogen from the high pressure gas storage vessels is used to pressurize LH_2 storage vessels and to cool the NERVA engine during part of the post-operative cooldown sequence. It is also used to pressurize the on-stand LH_2 cooldown storage dewars and for all NGTM functions controlled by the pneumatic complexes.

Hydrogen storage vessels are provided as follows:

Designated Storage Vessel	Total Water Vol. - Cu. Ft.	Pressure Range PSIA	Available Storage Cap. - Pounds	Function
V-GH-1	450	6000/4200	175	Pneumatic Control Complex
V-GH-3	1400	6000/3000	1,050	Engine Cooldown & Vessel Pressurization
V-GH-2	2400	6000/500	3,531	LH_2 Vessel Pressurization & Transfer
			4,756	

B.5.3 Liquid Nitrogen

Liquid nitrogen is utilized for reactor cooldown purposes, as a source to replenish the stored gaseous nitrogen and as a pressurizing fluid for the transfer of liquid nitrogen.

Liquid nitrogen storage is provided as follows:

Vessel No.	Capacity Gallons	Pressure PSIG	Use Per 5000 MW Run - Gallons	Primary Function
V-LN-1	146,000	355	132,000	Engine cooldown
V-LN-2	11,000	655	10,000	Engine cooldown
V-LN-3	164,000	90	148,000	Nitrogen storage
Totals	321,000 gal.		290,000 gal.	

Liquid nitrogen is transported to use point interfaces as required through insulated piping.

LN_2 instrumentation and controls are provided under the several bid packages for storage dewars, piping and equipment and the facility I&C systems. The TCSS incorporates LN_2 I&C functions utilizing sensors, amplifiers, etc. furnished by others.

Liquid nitrogen vaporizers are provided to supply gaseous nitrogen as follows:

Exchgr No.	Flow Rate	Pressure	Function
H-LN-1	20 pps	380 psia	Purge requirement
H-LN-2	9 pps	105 psia	Purge requirement
R-LN-1	75,000 scfh	6000 psi	Storage recharging (Compressed from vaporizer)

B.5.4 Gaseous Nitrogen

Gaseous nitrogen is used throughout the facility for inerting, LN_2 dewar pressurization, power operated devices where compressed air would not be safe to use, line drying, seal inflating, and buffer gas.

Gaseous nitrogen is supplied to each test stand for (a) purging and inerting the engine environmental chamber and the duct vault; (b) providing power for lock actuating motors etc. associated with remotely operated mechanical devices; (c) inflating and buffering seals; and (d) operating pneumatic valves. Nitrogen supply to each test stand is 1.5 pps at 2500 psi.

Storage is provided as follows:

Vessel No.	Total Water Vol. Cu. Ft.	Pressure PSIG	Available Gas Pounds	Flow Rate Lb/Sec.	Function
V-GN-2	93	6000/2500	1,100	15.3	Valve actuator
V-GN-1	3050	6000/1200	53,000	1.8	Transfer LN_2

Nitrogen is transported to the facility use point interfaces through uninsulated piping systems.

B.5.5 Helium

Helium is used to actuate certain critical flow valves, to purge certain liquid hydrogen system components and for engine emergency cooldown. Helium compressors are provided at the unloading station for truck unloading and for recharging the storage vessels. The helium storage vessels are rated as follows:

Vessel Number	Water Vol. Cu. Ft.	Pressure PSIG	Available Gas-lbs.	Flow Rate Lbs/Sec.	Function
V-GHe-3	700	6000/4400	543	2.8	Pneumatic complex
V-GHe-2	10,000	6000/2300	20,000	165	Emergency cooling
V-GHe-1	3,000	6000/400	10,000	0.4	Purge LH_2

Helium is transported through piping systems to facility use point interfaces from the storage vessels located in the high pressure gas storage area.

Instrumentation and control for the helium systems is provided by the several component suppliers and the facility I&C systems. The TCSS design integrates all helium controls and instrumentation to meet the requirements of the operating contractor.

B.5.6 Water

Domestic water is supplied through a 10-in. diameter line from the site water supply through a booster pump to the industrial water treatment station, to a storage reservoir and to personnel use points throughout the complex. Estimated personnel demand is 12,000 gallons per day. Available supply is 144,000 gallons per day. Domestic water is supplied to the site fire hydrants. Domestic water is not provided to the test stands for other than incidental personnel use such as drinking fountains, emergency showers, etc.

Industrial water is used for steam generator process water, nuclear exhaust system cooling and water injection, test stand deluge and fire protection, borated water mixing station, and all other uses where filtered, softened, and pH controlled water is required.

Location of use	Gallons per test	Max. Flow rate g.p.m.	Static Pressure elev. 3820
Test stand	8,700,000	150,000	100 p.s.i.g. (flowing)
Steam generator	1,900,000	T.B.D.	
Borated water	900,000	0 during test	
Cryo. storage	300,000	17,800	
Totals	11,800,000	167,800* g.p.m.	

*Storage capacity = 11,370,000 gallons at 137 psig minimum head at elevation 3820

Discharge line size = 72 in.

Industrial water for the test stand is used as follows for each full power, full duration test run:

Fog at vault exit	160,000 gal.
Diffuser - ejector cooling	4,950,000 gal.
Test stand deluge	800,000 gal.
Shield lift	740,000 gal.
Diffuser (cooldown cycle)	1,730,000 gal.
Total	8,700,000 gal.

Pressure at pad level is 137 psia static or 100 psia at maximum flow rate of 125,000 gpm through the 72-in. diameter storage-tank discharge line. Make-up is supplied at 400 gpm through a 10-inch line for 22 days at 24 hr/day.

Borated water is used in all facility shield components and also in the engine intermediate shield, if required. It is stored in a 550,000 gallon tank and pumped into the shields prior to test. Circulation of borated water during test for the bottom and intermediate shields is provided by borated water transfer pumps, with the hot borated water being collected in a 400,000 gallon dump tank.

Disposal facilities for water contaminated with radioactive waste will be provided at such time as the requirements for such disposal are firmly established. Separate disposal systems are contemplated for duct effluent water and stand deluge water.

Waste industrial water is discharged through the vault ditch to natural drainage channels. Excess water removed from the horizontal plenum of the NES may contain radioactive wastes and is removed through a barometric well and buried discharge line to a waste water retention/disposal system. Uncontaminated waste cooling water and duct coolant, along with other sources, is discharged into the vault and, through a sealing weir, allowed to run in the ditch to natural drainage channels. Maximum flow rates and quantities anticipated are:

Contaminated water 40,000 gpm	(?)	gallons
Uncontaminated water 60,000 gpm	(?)	gallons

B.6 ELECTRICAL POWER AND DISTRIBUTION

Electric power at 4160 volts is supplied from two separate synchronized sources, one being the NRDS 69 KV loop line transformed to 4160 volts at the E/STS 2-3 complex substation and the other being the NRDS generating station whose primary output voltage is 4160. Power is supplied to the main distribution load center at the control center through feeders in an underground duct bank.

a. Interruptible power is supplied from the 69 KV commerical power source for general complex use. Installed capacity is 3750 KVA.

b. Uninterruptible power available for I&C systems is 250 KW and is provided by a stored energy system operating from the 4160 volt main or standby power supply to the E/STS 2-3 complex.

c. Critical power (or standby) is provided by diesel generators at the NRDS generator station and connected by separate feeder to the critical power bus at the control center Connected load on the critical power bus is 650 KVH.

d. Super critical power is provided as D.C. battery power for NERVA engine controls and other devices essential to safe operation and shutdown of the test article and the facility. Required amperage at 28 volts has not yet been specified.

Diesel generators (peaking plant equipment made available by the Nevada Power Co.) are located near the E/STS 2-3 power feeder connection to the NRDS looped power system. During test periods these generators are operated and phased in with commercial power to provide back-up for the "interruptible" power supply. A separate feeder to the control center, and separate switch-gear and distribution panels provide critical power to the secondary distribution bus at the E/STS 2-3 complex. This feeder has a capacity of 3750 KVA.

Electric power is furnished to the control center via underground ducts from the facility substation. Secondary transformers, distribution panels for the control center and other facility components are located in the control center mechanical and electrical room. Power rating of the substation is:

	Voltage	Total KVA
Interruptible power	480,208/120	3,750
Uninterruptible power (I&C)	208/120	300
Critical power (standby source)	4160	3,300
Supercritical power (D.C.)	28	TBD

Power rating of the transformers may be uprated by later addition of cooling fans. Duct bank to the control center is provided with 50% spare capacity.

Electric power is supplied to each test stand from the unit substation located at the control center via cables through the control tunnels to distribution panels in each service tower substructure. Power is distributed within the test stand as required at 480,208Y/120 volts and is classed as interruptible power with total demand rating of 2800 KVA. Uninterruptible 60 cycle power, from an electrically separate source, feeder, and distribution panel (KVA to be determined), is also available at each test stand. In addition, provisions are made for super-critical D.C. power as necessary for NERVA engine control during scram, shutdown and cooldown. Voltage and capacity are not yet determined.

B.7 INSTRUMENTATION AND CONTROL

Instrumentation and control systems include the following subsystems:

a. Instrumentation subsystem containing provisions for measuring 1500 analog signals (pressures, temperatures, etc.) and 400 events (binary or on/off signals). Also included is recording capability for 395 analog signals and 70 event signals. Recording is done on 196 magnetic tape channels, 72 optical oscillograph channels, 96 direct writing oscillograph records and 24 strip chart recorders.

b. Facility control subsystem consists of control consoles with indicating meters, controls, distribution wiring, transmission cabling, etc. All facility process systems, mechanical devices, and GSE which require remote monitoring and/or operation from the control center are part of the facility control subsystem instrumentation and controls. This subsystem, integrated with TCSS, comprises the integrated test control system.

c. Facility support subsystem includes nuclear instrumentation, fire detection, combustible gas detection, oxygen detection, area and surveillance warning systems, closed circuit TV systems, meteorological monitoring, and communications subsystems.

Facility control and instrumentation systems are provided to the test stands and include approximately 400 channels for facility operation, approximately 540 channels for engine operation and control and approximately 565 channels for stage operation and control, all in accordance with the design criteria requirements.

B.8 SUPPORT EQUIPMENT

In addition to transporting the NERVA engine and mating it with the MPT, the engine installation vehicle (EIV) will be modified to install and remove the duct cover shield. It is also to provide manipulator capability for installation and removal of bottom shield cover grating and the EIV track extension over the retracted side shield.

MPT handling and installation is by ground support equipment furnished by the Government. The MPT will be brought to the test stand in a horizontal position on rubber tired transport, erected and placed on the module support ring by means of a mobile crane working simultaneously with the transport carrier. The handling equipment will be used at either test stand and at the stage assembly area, to be delineated later.

Provisions are made for remote removal of the NES by providing railroad rails embedded in the duct vault floor. Special transport cars are moved into the vault by a prime mover (shielded as necessary) and positioned under the duct jacking points. Hydraulic jacks mounted on the cars lift the duct so that the trunnions are clear. The trunnions and the duct, complete with attached piping, all completely supported by the cars, are withdrawn horizontally through the vault exit. The vertical diffuser is disconnected from the test stand substructure and the duct horizontal plenum and then raised vertically to permit duct removal. The diffuser is then lowered through the vault and removed in the horizontal attitude by shielded equipment (Figure B-6).

In addition to engine and stage handling and the NES installation devices, facility operation will utilize other portable equipment, furnished by the Government, such as stage access scaffolds, duct cover shield and handling fixture, EIV mounted manipulators, etc.

B.9 FACILITY SUPPORT BUILDING

The facility support building houses offices, personnel service facilities, storage and supply rooms, etc. closely associated with the construction and activation and operation of the E/STS 2-3 complex. The building is designed to accommodate 115 male and 15 female occupants. A single story prefabricated rigid frame structure of 16,500 square feet, it is located as close as practible to the control center and provided with a 25-car parking lot. It is unoccupied during a nuclear test event.

APPENDIX C

DIRECT RADIATION FROM NERVA ENGINE

This appendix describes the environmental radiation intensities expected at E/STS 2-3 during test and post-test periods due to direct radiation from the NERVA engine. The data presented are based upon the results of the nuclear analyses reported in Reference (1) for a 5000 Mw NERVA configuration operated for 30 minutes. For test periods, environmental dose rates are given for the case with the facility shield in its normal test position. For post-test periods, environmental dose rates are given for both the normal shielded case and for the unshielded accident case.

The facility model used for the radiation analyses is as shown in Figure C.1. The engine model used is based on the 5000 Mw reactor configuration described in Reference (2) with a minimum reflector thickness and no flight-type shield above the reactor. Direct gamma radiation levels were calculated with the QAD-P5 point-kernel integration code, Reference (3). Gamma-ray scattering in air and in the hydrogen contained in the ground-test-module tank were calculated with the aid of the GGG computer program, Reference (4).

The gamma dose rate as a function of distance from the engine during engine operation at 5000 Mw is shown in Figure C.2. For this case, it is assumed that there is no LH_2 in the tank and that the engine is shielded by 6 feet of water as shown in Figure C.1. (Reference (1) indicates that the environmental neutron dose rate is negligible compared to the environmental gamma dose rate, and that the effect of LH_2 in the tank on the environmental gamma dose rate is relatively insignificant.)

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(1) H. O. Whittum, et al, "Interim Nuclear Analyses in Support of E/STS 2-3 Site Layout Activities," RN-TM-0415, December 1966.
(2) "NERVA I Reactor Conceptual Design Report," WANL-TME-1315, October 1965 (CRD).
(3) R. Malenfōnt and G. Graves, "LASL Computer Program: QAD-P5," personal communication.
(4) R. Malenfunt and G. Graves, "LASL Computer Program: GGG," personal communication.

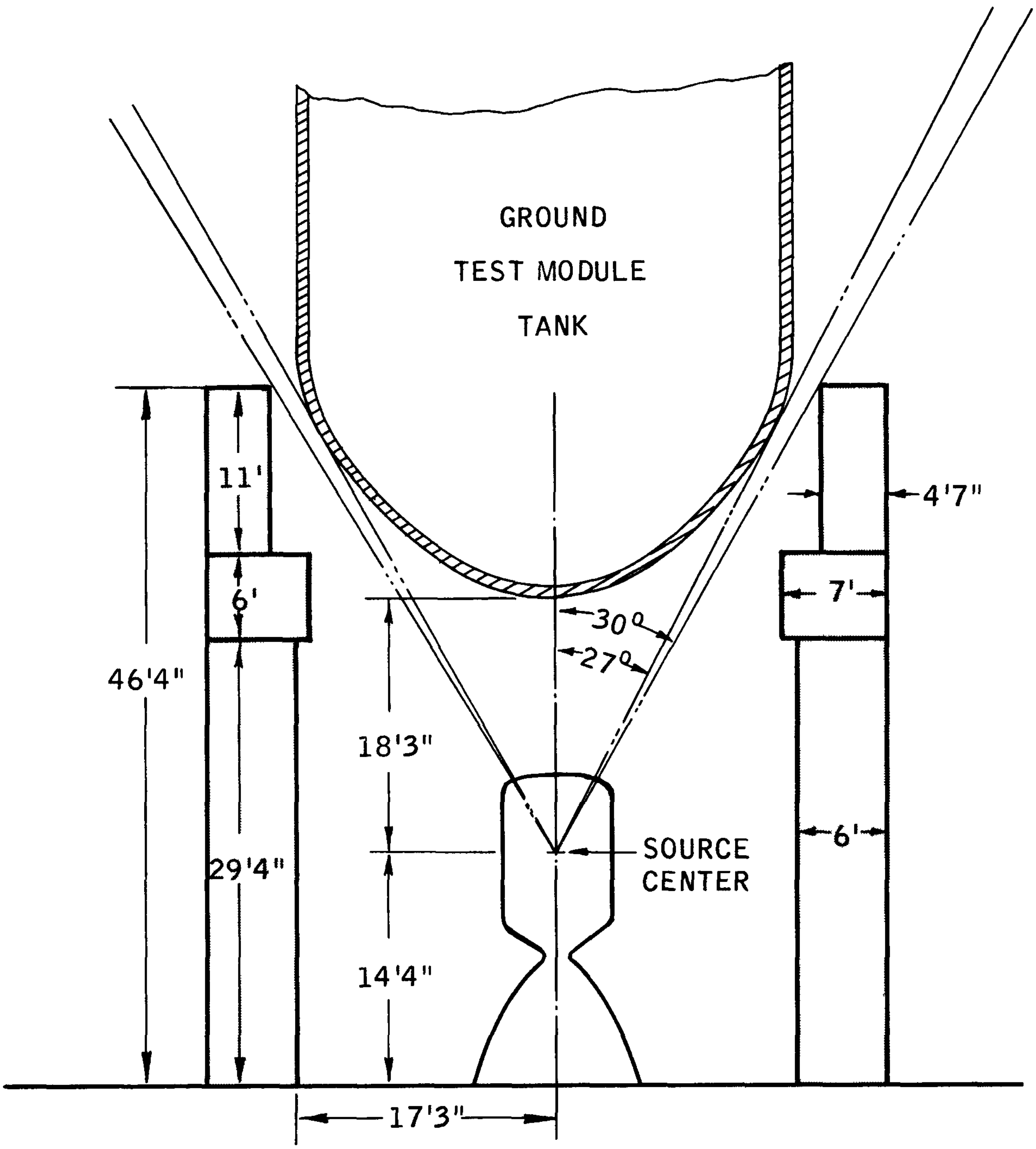

Figure C.1 - Facility Model for Radiation Analysis

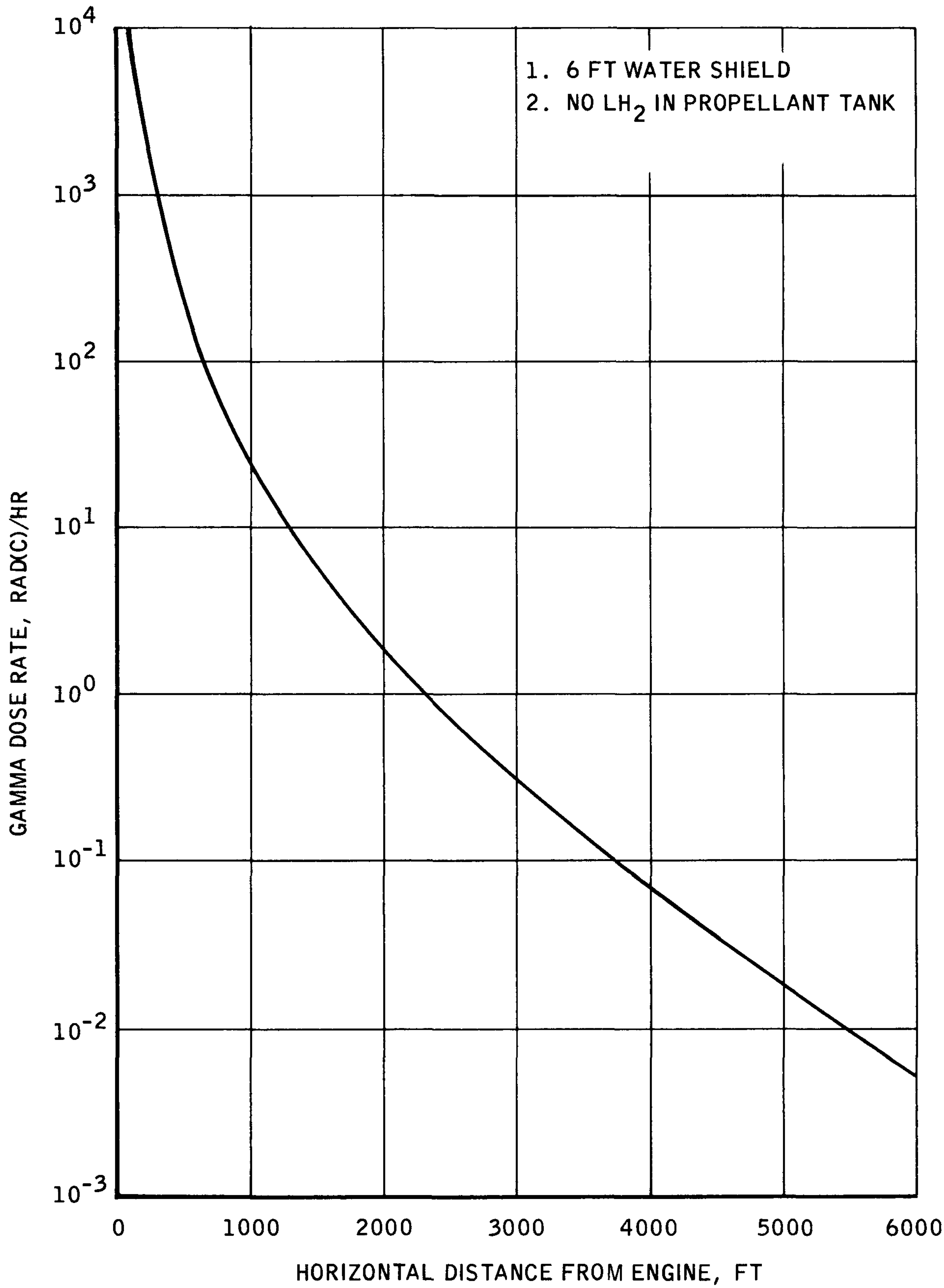

Figure C.2 - Dose Rate vs Distance from Engine Operating at 5000 Mw

The environmental dose rates as a function of distance from the engine 1 hr and 24 hr following shutdown from 30 minutes of operation at 5000 Mw are shown in Figures C.3 and C.4. Figure C.3 represents the normal post-test conditions with the facility shield in place. Figure C.4 gives the environmental dose rates under the accident condition where shield water is lost. For both of these cases, it is also assumed that no LH_2 is in the tank.

Isodose maps of the E/STS 2-3 area are presented in Figures C.5 to C.7. In each figure, the isodose contours are shown at radial increments of 200 ft from the two test positions. The figures give isodose maps for: full power engine operation, one hour after shutdown with the facility shield in place, and one hour after shutdown with no facility shield.

Finally, an important consideration in evaluating facility accessibility is the variation of fission-product activity with time after engine shutdown. This variation, as defined by Reference (5) between 0.5 hr and 100 hr after shutdown, is shown in Figure C.8. These data have been extrapolated to 0.1 hr after shutdown, and normalized to unity at one hour decay to facilitate their use in extrapolating the radiation data presented previously to various shutdown times. It should be noted, however, that the data in Figure C.8 do not account for activation sources nor do they incorporate the effects of subcritical fission multiplication due to photoneutrons. This latter effect could be significant for the 1-hr decay case and may be expected to dominate at very short times after shutdown (i.e., at times less than several minutes).

(5) J. F. Perkins and R. W. King, "Energy Release from the Decay of Fission Products," Nuclear Science and Engineering, 3, p. 726-746, 1958.

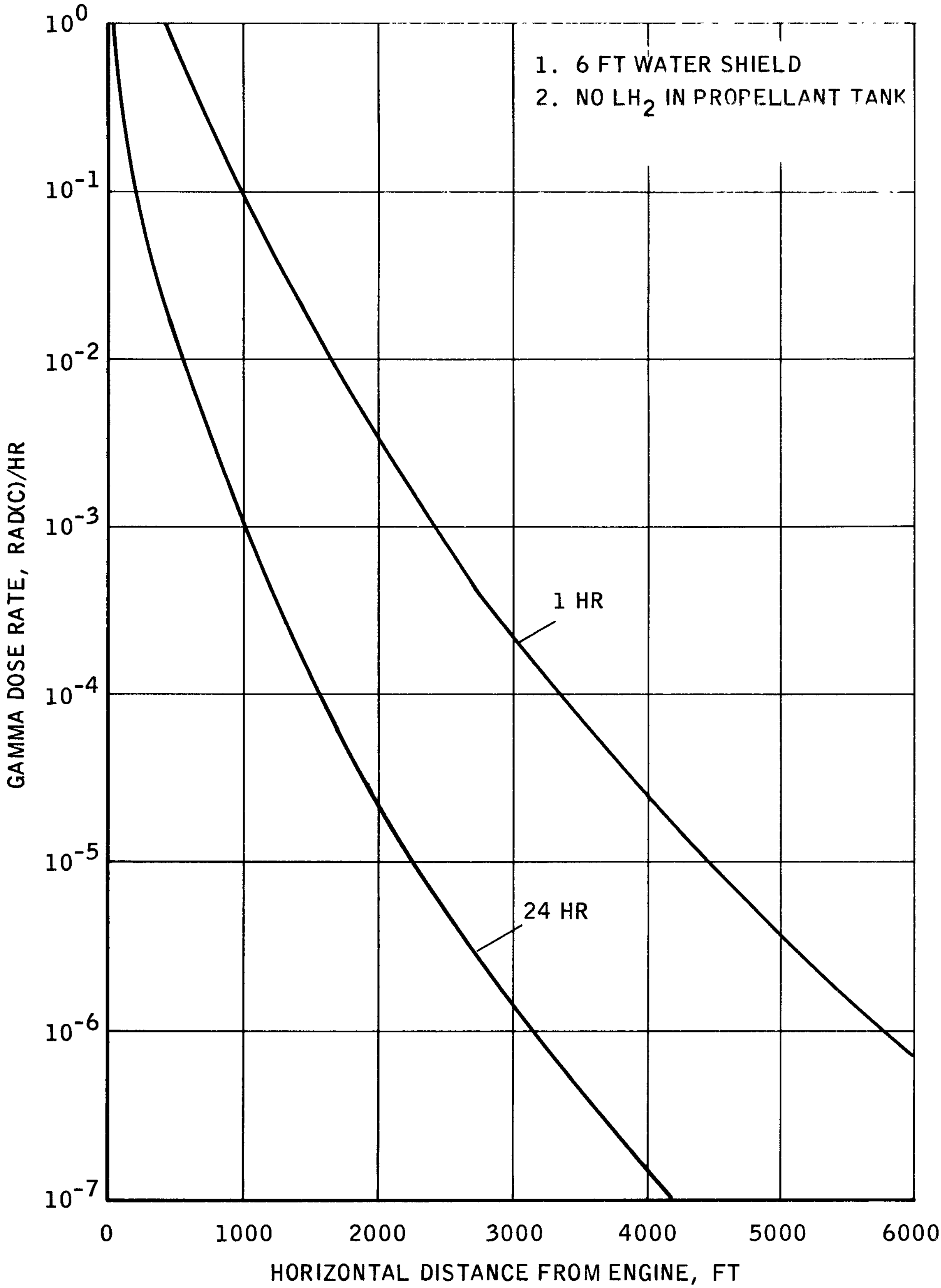

Figure C.3 - Shielded Dose Rate vs Distance 1 and 24 hr After 30 min at 5000 Mw

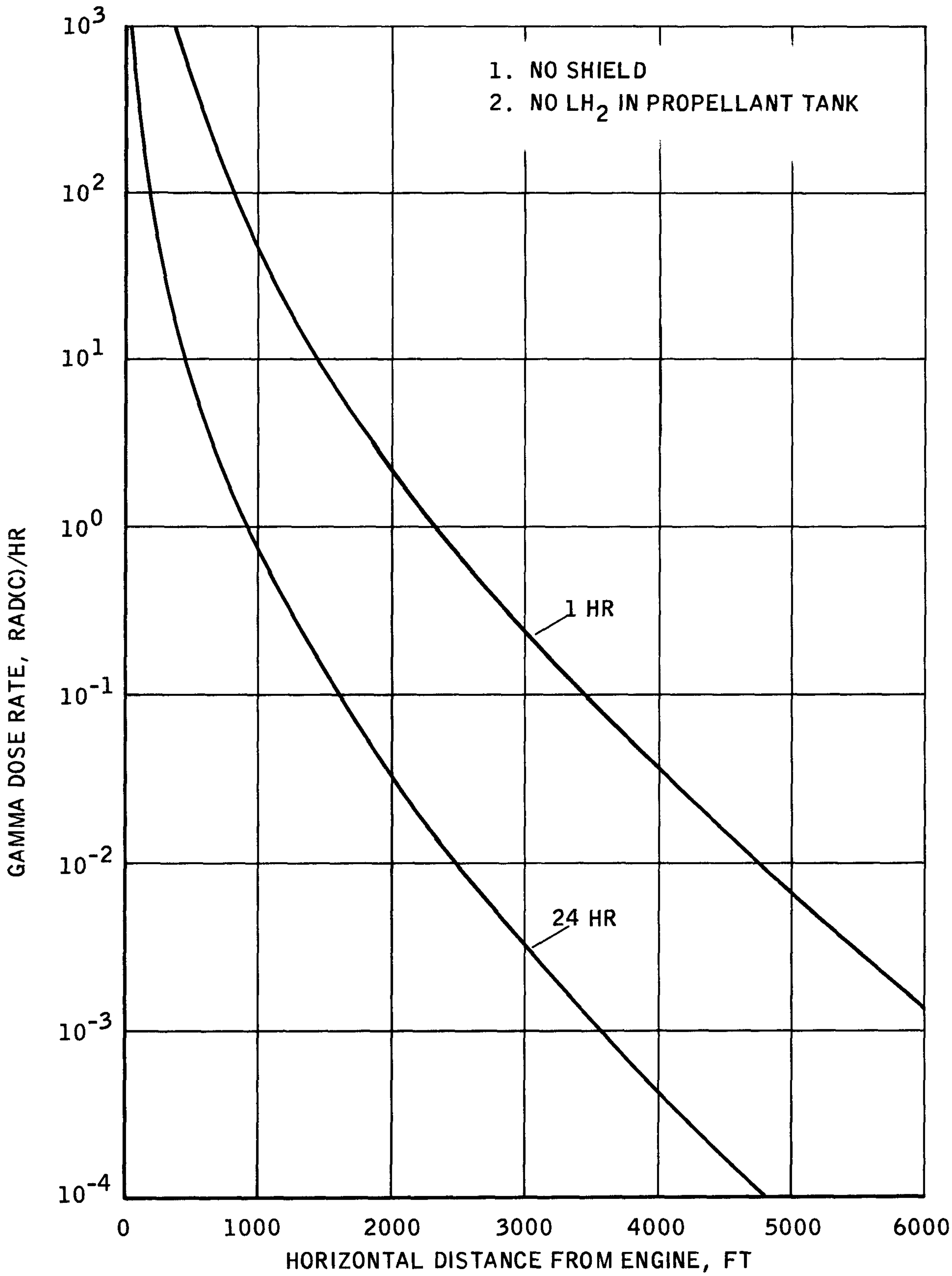

Figure C.4 - Unshielded Dose Rate vs Distance 1 and 24 hr After 30 min at 5000 Mw

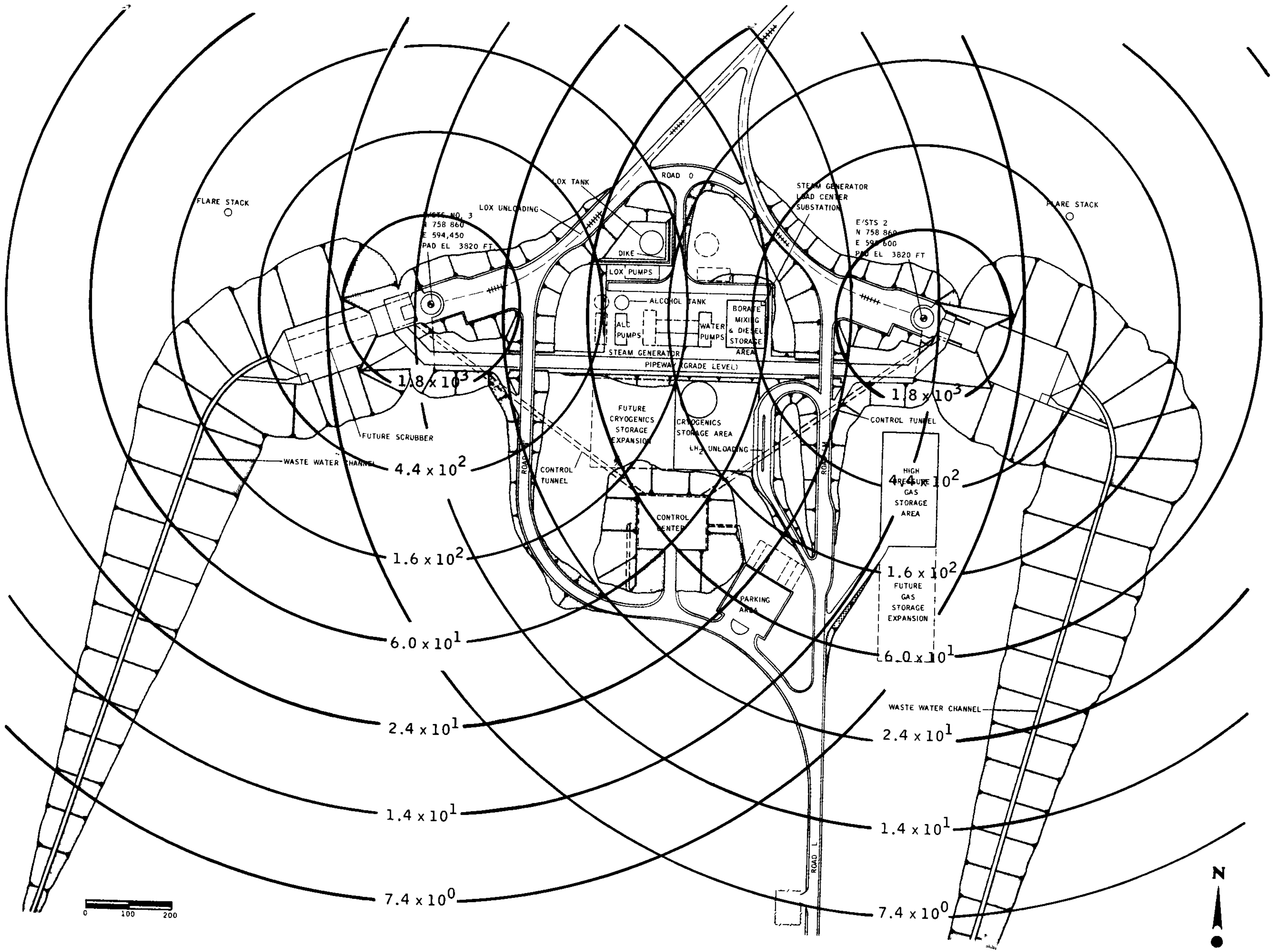

Figure C.5 - E/STS 2-3 Isodose Map During NE Operation

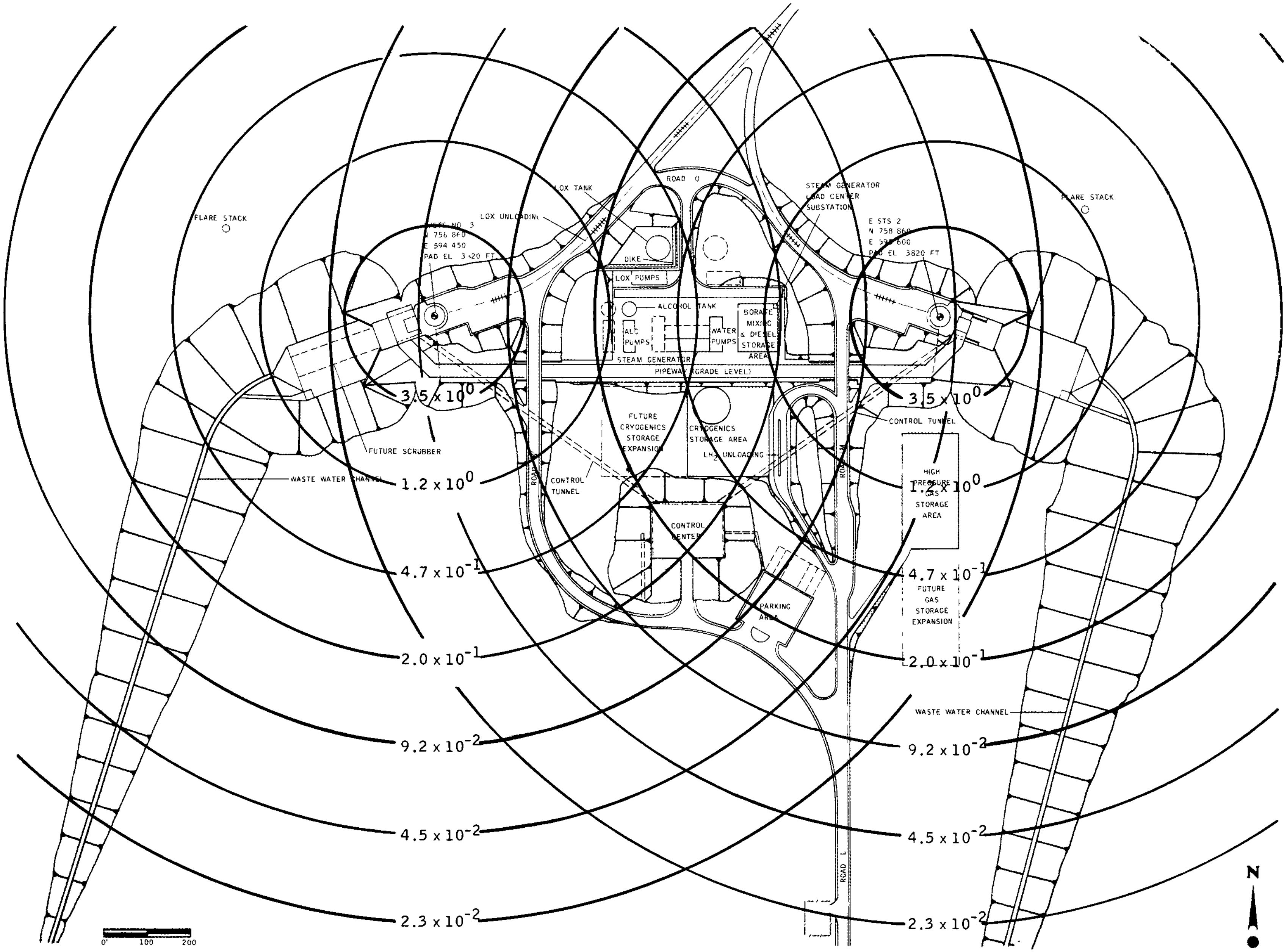

Figure C.6 - E/STS 2-3 Isodose Map 1 hr After NE Shutdown - Facility Shield in Place

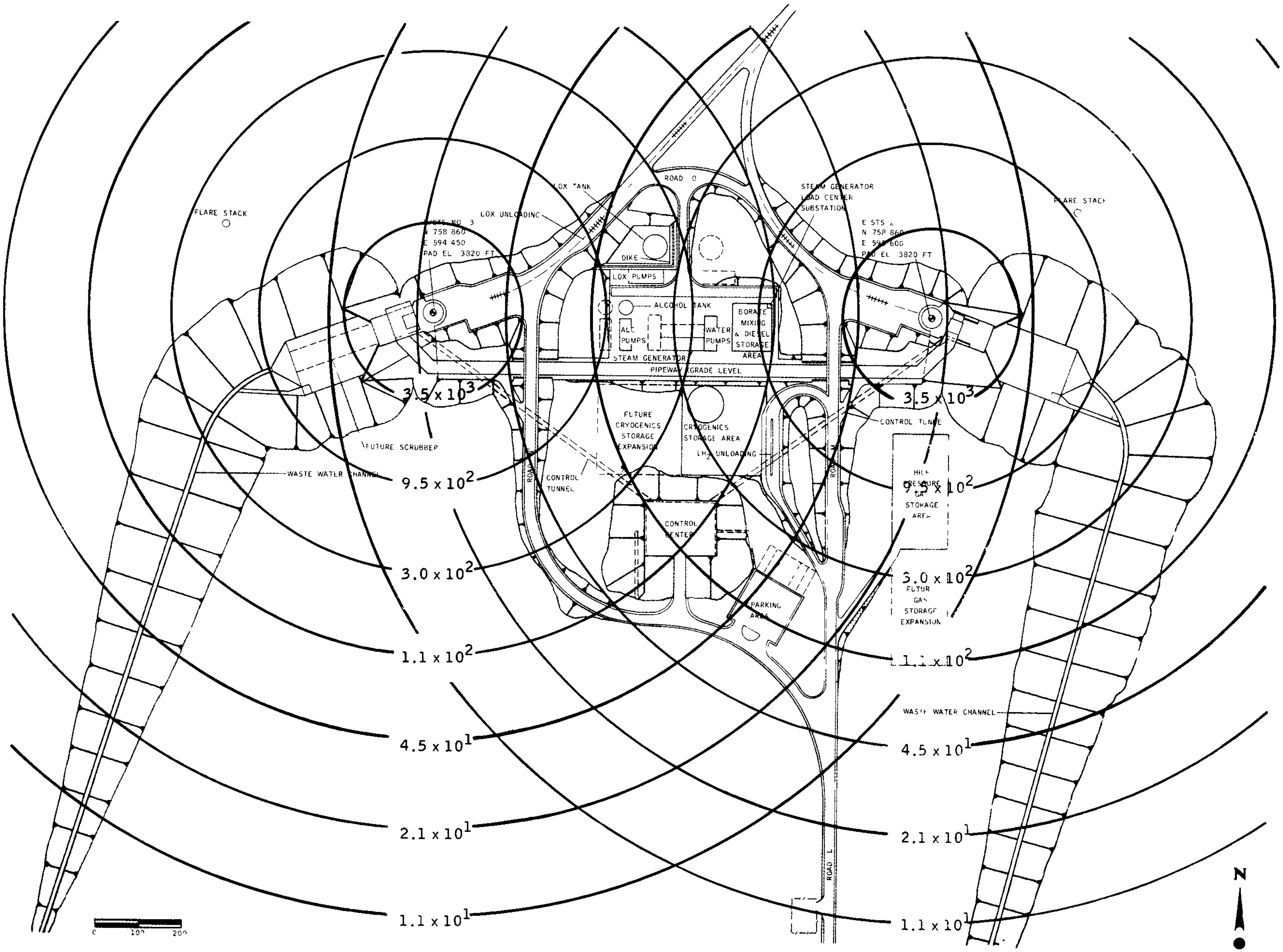

Figure C.7 - E/STS 2-3 Isodose Map 1 hr After NE Shutdown - No Facility Shield

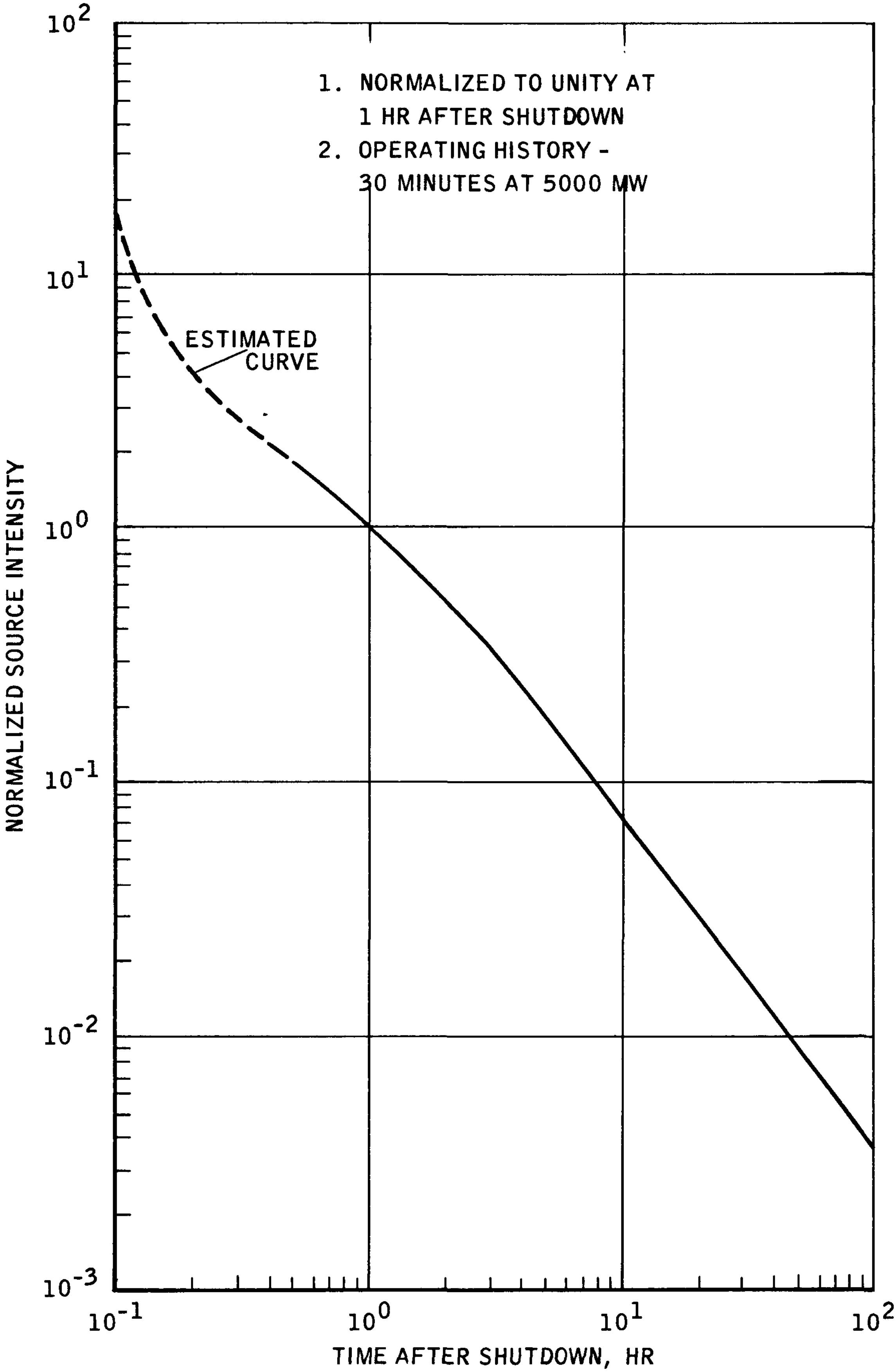

Figure C.8 - Normalized Fission Product Decay Curve

APPENDIX D

POTENTIAL NUCLEAR ACCIDENTS

D.1 CRITICALITY ACCIDENTS

D.1.1 Control Drum Runouts

The NR control drum drive scheme is considerably different than the NRX design. A "ganged" drive mechanism which rotates all control drums in unison has been selected. It consists of thin titanium straps rigidly fastened to titanium pulleys which transmit the torque between drums. The pulleys drive the drums through a decoupler spring. The system is designed such that if one drum sticks at any position (i.e., the torque required to move it is greater than that exerted by the spring), it will decouple from the drive mechanism and allow the other drums to move freely. If at any time this stuck drum is released, the decoupler spring is designed in such a manner that the drum will move to the bank position, whichever direction that may be. The number of actuators required to drive this system is tentatively set at 1 for every 4 drums. The actuators will be sized in such a way that the normally operating actuators will be able to override an actuator that may fail to supply torque, jam in position, or deliver full torque in the outward direction due to a feedback potentiometer failure[1]. The XE actuator is presently the prime candidate. This system eliminates the possibility of a single drum runout and greatly simplifies the overall control circuitry which should improve system reliability.

Bank control drum runouts from source power levels of 1 milliwatt and 1 watt were analyzed for drum velocities of 10 to 1000 degrees/second, (the maximum velocity of the XE actuator) with no scram. The principal investigative tool was the RTS code[2] which solves the one-group, space independent, reactor kinetics equation. The NR-1 nuclear parameters, as presently estimated, used in this analysis are in Table D.1.

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(1) WANL-TME-1485, "NR-1 Reactor Mechanical Design Report," September 1966 (CRD).
(2) G. R. Keepin and C. W. Cox, "Nuclear Science and Engineering," 8, 670 (1960).

TABLE D.1

ESTIMATED NR-1 NUCLEAR PARAMETERS

Drum Span	9$
Shutdown Margin	-7$
Average Neutron Lifetime	30μ seconds
Beta Effective	0.0071
Average Temperature Coefficient	-0.073¢/°R
Initial Reactor Temperature	Ambient

The available excess reactivity of two dollars is considerably less than that available in the NRX reactor due to the amount of hydrogen contained in the reactor at operating conditions.

Figures D.1 and D.2 contain the results of these analyses. The excursion energy represents the energy generated during the power spike. In terms of average core temperatures, an axial-support-damaging 2000°R corresponds to about 3×10^9 watt-seconds. At this temperature there is negligible fission product release. If the excursion were allowed to run its course, it would result in greater than 10^{10} watt-seconds of energy with an average core temperature in excess of 4500°R and concomitant high release fractions for fission products.

A single drum runout from source power, if it could occur, would be capable of introducing only about 38 cents reactivity. In view of the seven dollars shutdown this would be insufficient to cause criticality.

In practice, safety circuitry such as period scrams and fixed and floating power scrams are designed to terminate a control drum runout before reactor damage occurs.

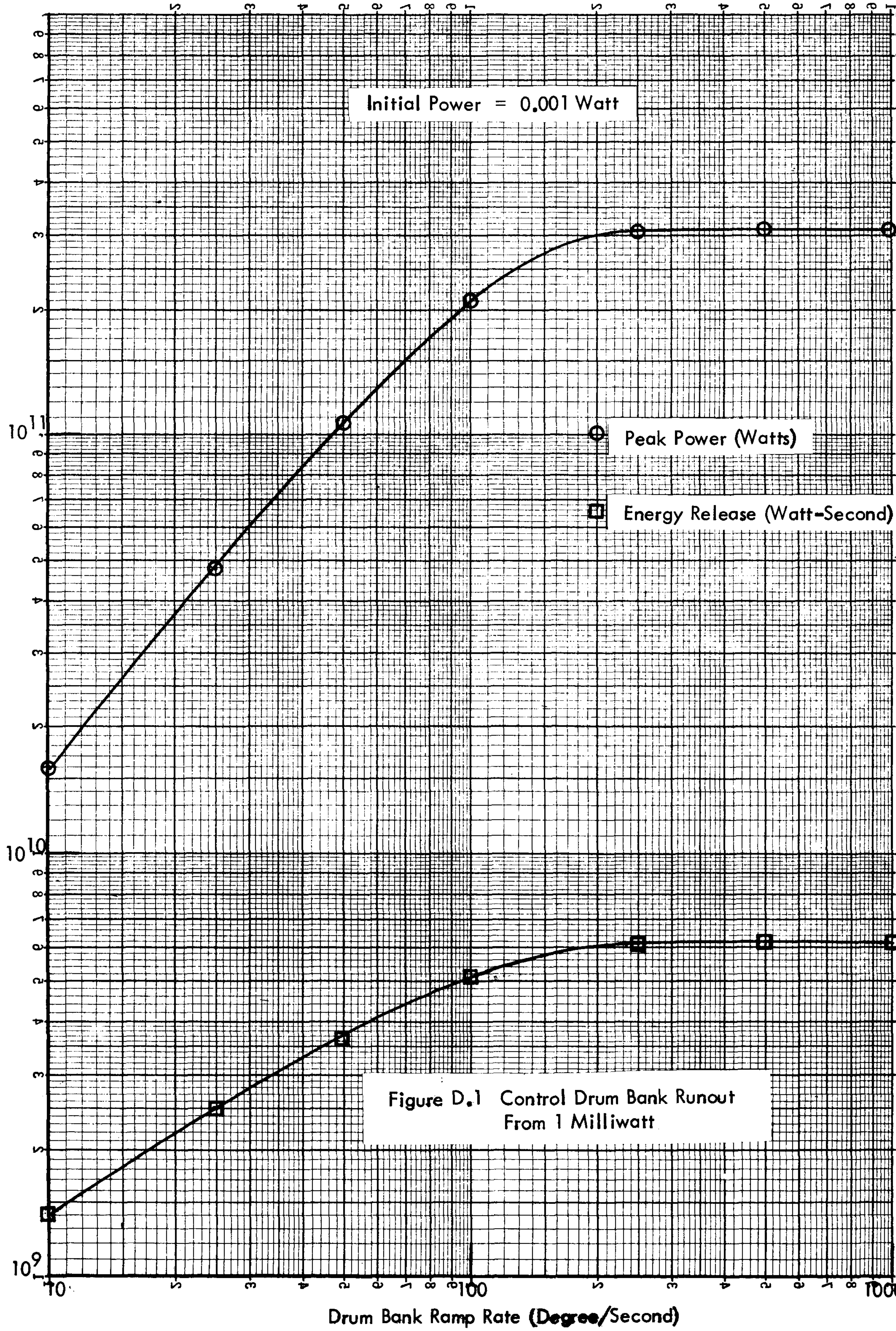

Figure D.1 Control Drum Bank Runout From 1 Milliwatt

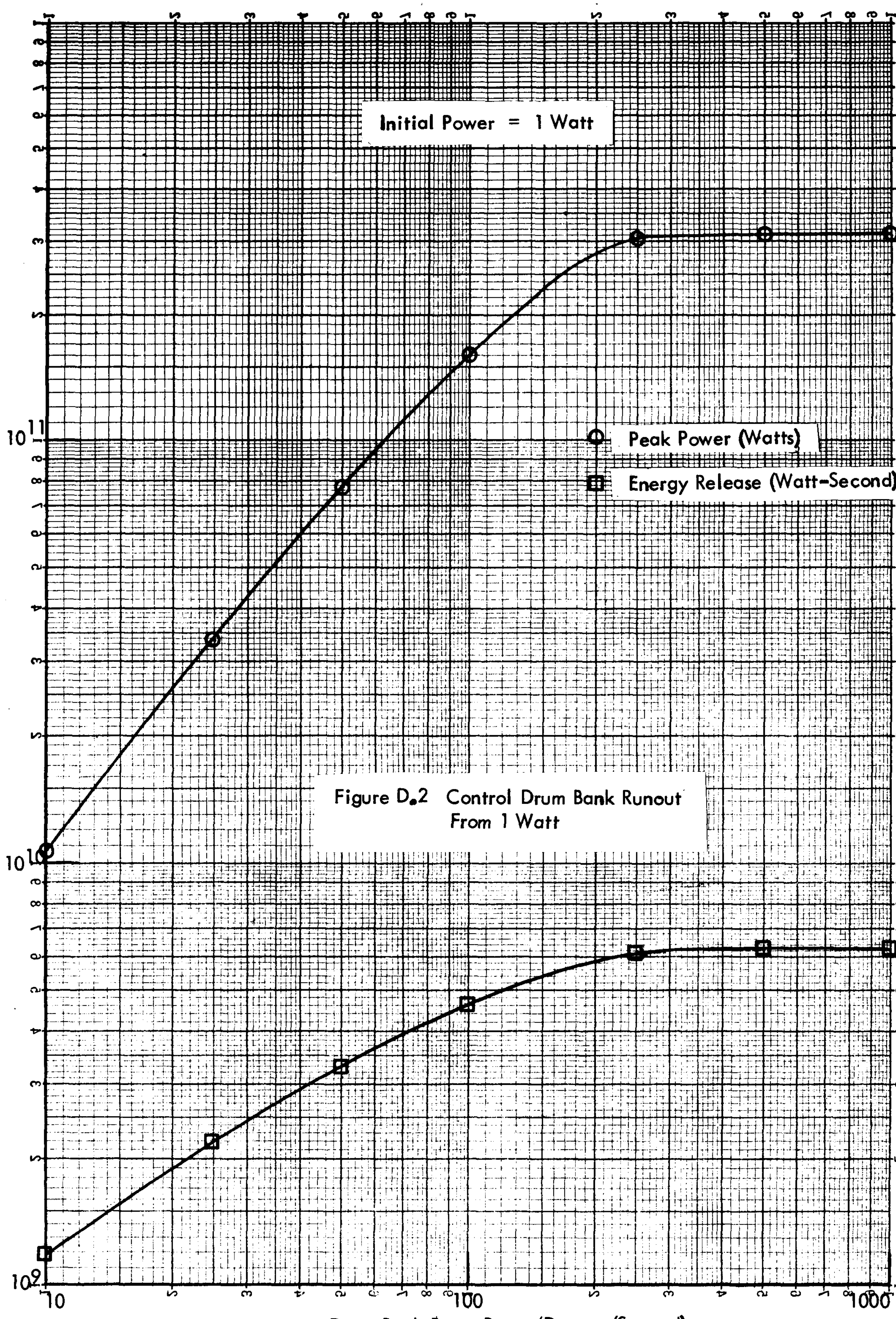

Figure D.2 Control Drum Bank Runout From 1 Watt

D.1.2 Fuel in Wet Duct

The fuel loadings for the NR-1 reactor have not been finalized as yet, and it is highly likely that the fuel loadings will change during the course of the NR and NE programs. A 500 mg/cc uranium loading has therefore been chosen as a representative loading to indicate some of the potential nuclear interactions which may occur if fuel material would be present in the exhaust duct. Analyses performed on this loading for the NRX program[(1)] indicate that the minimum number of full length intact fuel elements needed to form a critical lattice is about 30. Approximately 200 would be required if they were in a close-packed array (i.e. all sides touching other elements). A smaller number of elements yet, ~14, would be required in the optimum configuration utilizing quarter length segments. The diameter of this critical system is about 11 inches. Minimum critical dimensions of homogeneous uranium, graphite, water systems have been determined for a 500 mg/cc element[(2)]. This corresponds to keeping the carbon/uranium ratio constant and changing the amount of water. Critical values for such homogeneous systems are less than the comparable ones for heterogeneous systems and represent the minimum achievable critical masses and dimensions. These results appear in Table D.2.

Fuel elements which lost their axial support and were ejected from the reactor may or may not be intact upon reaching the bottom of the duct, especially after striking the gas deflection plate. It is not unreasonable to expect that there may be more than enough pieces, chunks, dust, and elements to form a critical mass either in the exhaust duct or the drain line. An upper estimate may be made of the energy release of an accidental criticality in the

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(1) WANL-TME-760, "Criticality of NRX-A Fuel Lattices", April 1964 (CRD).
(2) LAMS-2955, "Critical Dimensions of Uranium (93.5) - Graphite-Water Spheres, Cylinders, and Slabs", October 1963.

TABLE D.2

CRITICAL PARAMETERS FOR REFLECTED U-C-H_2O SYSTEMS

Geometrical Shape	Volume Fraction Water	Critical Radius Thickness (cm)	Critical Volume (liters)	Critical Mass (Kg - U_{235})
Sphere	0.0	35.7	19.10	89.3
Sphere	0.8	14.4	12.4	1.2
Sphere	0.92	18.1	24.6	0.9
Infinite Cylinder	0.0	24.8	-----	----
Infinite Cylinder	0.8	9.5	-----	----
Infinite Cylinder	0.92	12.4	-----	----
Infinite Slabs	0.0	24.1	-----	----
Infinite Slabs	0.8	8.3	-----	----
Infinite Slabs	0.92	12.2	-----	----

duct. The 10,000 Mw-Second release of KIWI-TNT was sufficient to disassemble the reactor. The maximum from an unrestrained assembly of the same size must be less since a critical configuration will be maintained for a shorter period of time. The overpressures from such an excursion will be less than those produced by KIWI-TNT since it will take less internal pressure to disassemble the critical configuration. In addition, the presence of a major portion of the core in the duct would mean that 4500°R hydrogen at rated flow would not be present in the diffuser section of the duct and that all of the water spray injected to cool this effluent would not be vaporized. There would thus be a water flow in the duct which could be of the order of thousands of gallons per minute, tending to wash away any obstruction in its path. Criticality could occur in drains, etc., where fuel material could be trapped. Table D.2 has been presented to enable the duct designer to prevent criticality in the drains, etc., by control of critical geometry.

D.2 LOSS OF COOLANT ACCIDENTS

The principal analytical tool for a loss of coolant analysis is the POST-OP code developed by WANL for flight safety studies[(1)]. This code has built in the NR-1 reactor geometry and includes heat flow by conduction and radiation between components, neutron heating of reactor components, and fission product decay energy deposited in the fuel material. For a loss of coolant at power the major source of heat loss is graphite sublimation. Work is progressing on a model which includes the transport graphite vapor pressure in the coolant channels and its effect on sublimation. At present vacuum sublimation data are being used. Since a retardation of sublimation leads to higher fuel temperatures, the results presented here may not be conservative but indicate the time scale of damage to the reactor due to loss of coolant. The graphite vapor pressure model may be of significance for the case of loss of coolant during the cooldown phase, but comparison of the preliminary model with the vacuum sublimation data for the case of loss of coolant after 30 minutes at full power, described below, indicates negligible differences.

(1) WANL-TME-1503, "Post-Operational Heating Analysis of NR-1", October 1966 (CRD).

Analysis of a loss of coolant 30 minutes into a full power (5000 Mw) run indicates that sublimation of the fueled graphite begins and the axial support system in the center of the core melts within a matter of seconds. The control drum poison plates begin to melt shortly thereafter. Approximately 1/8 of the plate (end away from the nozzle) never melts. The data indicates that the nozzle would melt through between 1000 and 2000 seconds after loss of coolant, with the first part to fail being the portion nearest the core. The springs in the lateral support system fail in the same time scale as the nozzle. By this time 10-20% of the core has sublimed. All of these results are based on the reactor geometry remaining intact.

It would thus seem that some portion of the core would end up in the duct. If the lateral support system does not supply sufficient bundling pressure, elements would slide through the nozzle into the duct when the axial support system fails. If the lateral support system initially maintains the core in position, the lateral support and the nozzle eventually fail.

Cooldown analysis has not been completed as yet, but comparison with the NRX reactor indicates that cooling will be required for ~4 days.

APPENDIX E

SOURCE TERMS

E.1 DIFFUSION SOURCE TERM

The estimates for the release of fission products by diffusion through the fuel bead coating and the graphite element matrix are based on the releases observed from irradiated fuel samples heated under controlled conditions in the laboratory. This raw diffusion data is fit to the theoretical Arrhenius expression for diffusion,

$$D = D_o \exp(-E/R \times 1/T),$$

where T is the absolute temperature and D_o and E/R are adjusted to the data. The release of each fission product is then calculated by the computer code, FIPDIF[1], which considers the radioactive buildup and decay of each isotope and accounts for the loss by diffusion by assuming that the diffusion of each isotope is equal to the product of its concentration, N, and the empirical diffusion constant, D, described above ($\frac{DN}{dt} = - DN$).

These diffusion constants have been determined for both uncoated fuel samples and for samples completely coated with NbC. Neither case exactly describes the fuel in the reactor, since for the majority of the elements the outer surface is uncoated but all coolant channels are coated. Therefore two estimates have been made, one with the coated constants and one with the uncoated. The most probable value and one which best agrees with the measurement of effluent cloud activity from previous tests has been determined by taking the product of the ratio of the outer (uncoated) surface area/total surface area and the uncoated prediction plus the product of the coolant channel (coated) surface area/total surface area and the coated prediction.

The code FIPDIF does not calculate the decay of the isotopes after they are released. The Source Term Program (STP)[2] computes the activity of each isotope as a function of decay time and also sums the decay energy for the gamma energy grouping used in the cloud gamma dose model.

(1) WANL-TME-958, "Interim Report on Fission Product Diffusion Code (FIPDIF)", September 1964.

(2) WANL-TME-796, "WANL Source Term Program Status Report", Volume II of II, May 1964.

E.2 CORROSION SOURCE TERM

Fission product release can occur through corrosion and erosion of the graphite fuel matrix resulting in the release of uranium and associated fission products. The largest reactivity loss experienced to date in the NRX program occurred in the NRX/EST reactor and amounted to about \$3.00[3]. Approximately 5% of the fission product inventory was lost by corrosion[4]. Present considerations indicate that the maximum reactivity loss which will be allowed in the NR program is about \$3.00. The maximum allowable weight loss per element is also comparable to the NRX/EST weight loss. Five percent is therefore considered a reasonable limit on the amount of fission products that can be lost by corrosion.

An unknown fraction of the corroded fuel material is released in particulate form. Particles found on the desert following reactor runs have exhibited fission product spectra ranging from that equivalent to normal U-235 fissioning to ones markedly deficient in the more volatile elements, such as silver and the iodine precursors. For this reason the corrosion fission products are considered to be in gaseous form for the purposes of estimating downwind doses from the effluent cloud and have been added to the diffusion source terms.

E.3 ACCIDENT SOURCE TERM

The accidental release of fission products is dependent on the type of accident and the time during the test series at which the accident occurs. An upper limit on the release fraction due to an excursion is indicated from KIWI-TNT data for which 100% of the iodine and 2/3 of the non diffusers were released. Laboratory data suggests that $\sim$2/3 of the gross gamma activity is released when irradiated samples are heated to a temperature of 5100°R (typical of maximum core temperatures predicted by the NOFLOW code for loss of coolant at power) followed by gradual cooling. A FIPDIF estimate on the loss of fission products by diffusion due to a loss of coolant immediately upon completion of a

(3) WANL-TNR-216, "NRX/EST Reactor Test Analysis Report", December 1966.
(4) WANL-TME-1476, "Effluent Studies of NRX/EST", July 1966.

30 minute run at 5000 Mw (based on the temperatures calculated by POST-OP[1]) is 100% of the iodine chains and 50-60% of the gross gamma inventory. Figure E.1 indicates the predicted behavior of this release with time after loss of coolant. It is not known as yet whether on excursion equivalent to the KIWI-TNT is possible in the NR system. Until then, an accident source term of 50% of the gross gamma inventory and 100% of the iodines is considered reasonable.

E.4 SOURCE TERM

A source term is presented in Table E.1 for two separate tests, 30 minutes at 5000 Mw and 45 minutes at 10,000 Mw. The same temperature distribution was used for the 10,000 Mw reactor as was used for the 5000 Mw model. The source term is defined as that percentage of the generated inventory that is released. The diffusion release is based on the ratio FIPDIF model described in Section E.1. A uniform 5% corrosion release has been added to the diffusion release as indicated by Section E.2. Finally, the accident source term of Section E.3 is included.

(1) WANL-TME-1503, "Post-Operational Heating Analysis of NR-1", October 1966.

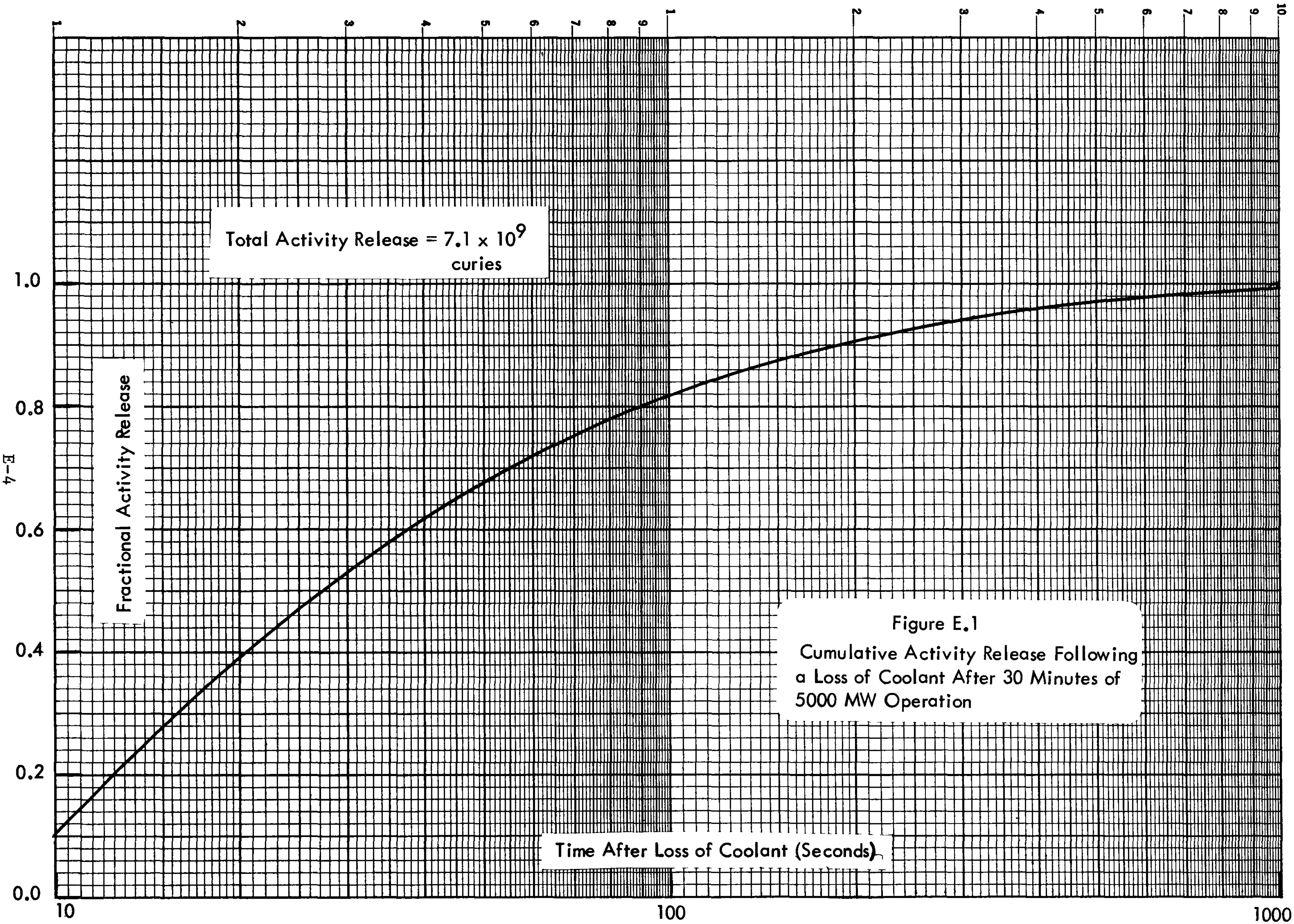


Figure E.1
Cumulative Activity Release Following a Loss of Coolant After 30 Minutes of 5000 MW Operation

TABLE E.1

SOURCE TERMS

NORMAL OPERATION*	PERCENT RELEASED 1-131	1-132	1-133	1-134	1-135	Gamma
30 minutes at 5000 Mw	11.6	9.3	8.4	8.0	8.0	9.7
Diffusion Only	6.6	4.3	3.4	3.0	3.0	4.7
45 minutes at 10,000 Mw	13.2	10.7	9.8	9.2	9.3	10.9
Diffusion Only	8.2	5.7	4.8	4.2	4.3	5.9
ACCIDENT	100	100	100	100	100	50

*Based on ratioed coated/uncoated FIPDIF model and 5% loss by corrosion.

APPENDIX F

DOSES FROM EFFLUENT

F.1 ATMOSPHERIC EFFECTS

The reliable prediction of downwind air concentrations of released contaminants is not straightforward. The two most common analytical expressions which may be related by an algebraic identity are due to Sutton and Pasquill, and result in an expression for which the cloud concentration follows a Gaussian distribution as related to the cloud centerpoint.

$$X = \frac{Q}{\pi^{3/2}\, C^3\, x^{3/2(2-n)}} \exp - \left[\frac{x^2 + y^2 + h^2}{C^2\, x^{2-n}} \right] \qquad (1)$$

where

X = air concentration (curie/m^3)

Q = curies released

C = virtual diffusion coefficient ($m^{n/2}$)

n = stability parameter (dimensionless)

u = wind velocity (m/sec)

x = distance downwind (meters)

y = distance crosswind (meters)

h = height of release (meters)

The ground deposition of activity from the cloud at any point is related to the time integrated concentration as expressed above. This integration results in

$$X_t = \frac{2\,Q}{\pi\, u\, C^2\, x^{2-n}} \exp - \left[\frac{y^2 + h^2}{C^2\, x^{2-n}} \right] \qquad (2)$$

where

X_t = curies-sec/m^3

As the cloud rises through the atmosphere to its effective stabilization height atmospheric turbulence and wind shear cause it to act to some extent as a line source of contamination. LASL experience from monitoring the KIWI and NRX tests indicates that this effect may be approximated considering two components to the cloud, one containing 1% of the activity and released at ground level and the other containing the remainder of the activity and released at the stabilized height of the cloud. The resultant equation for the time integrated exposure then has the form

$$X = \frac{2\,Q}{\pi u\ C^2\ x^{2-n}} \exp - \left[y^2/C^2\ x^{2-n} \right] \left[0.99\ e^{-h^2/C^2\ x^{2-n}} + 0.01 \right] \qquad (3)$$

Equation (3) may readily be solved to calculate isoconcentration curves by identifying the crosswind distance, y, where the concentration is p percentage of the centerline value.

$$y_p = \left(C^2\ x^{2-n} \ln \frac{100}{p} \right)^{1/2} \qquad (4)$$

Ground deposition is proportional to the integrated concentration (Equation 3) since the product of a pseudo deposition velocity (m/sec) and the integrated concentration (curies-sec/m^3) yields curies/m^2.

The actual situation in nature is not as simple as the idealized situation described by Sutton's equation. The wind may not blow constantly from one direction at all levels in the atmosphere. Figure F.1 represents a case where the cloud for the first 16,000 feet forms a classic pattern, but is then sheared by the wind into two directions initially almost 90° apart. Figure F.2 presents a case where there was a definite overall general direction to the wind, but ranging from south-southwest at the surface to southwest and westerly at higher elevations.

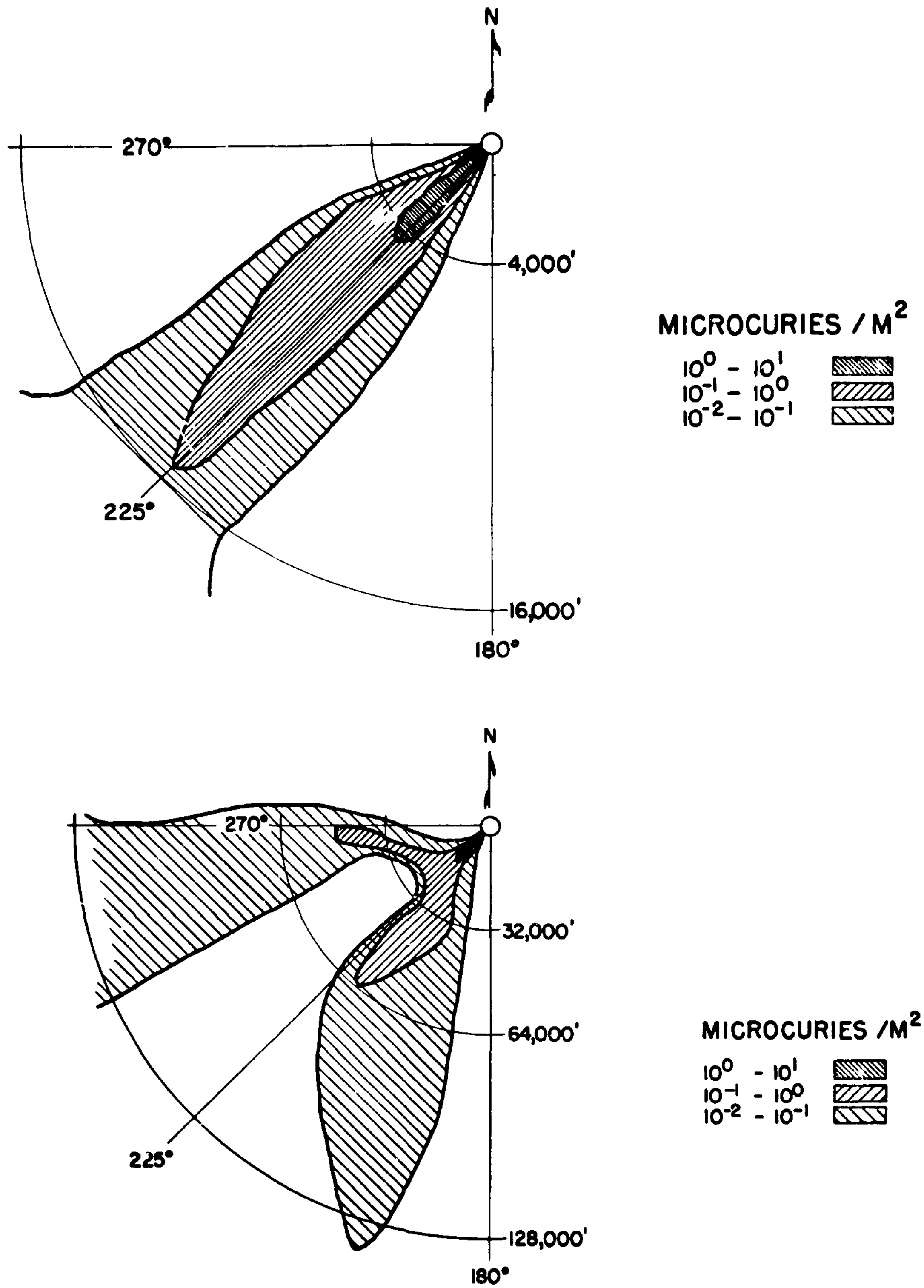


Figure F.1

NRX A-2 Activity on resin-coated trays. Beta activity corrected to estimated time of cloud passage.

(LA-3394-MS, "Radiation Measurements of the Effluent from the NRX-A2 and NRX-A3 Reactors", Figure 21)

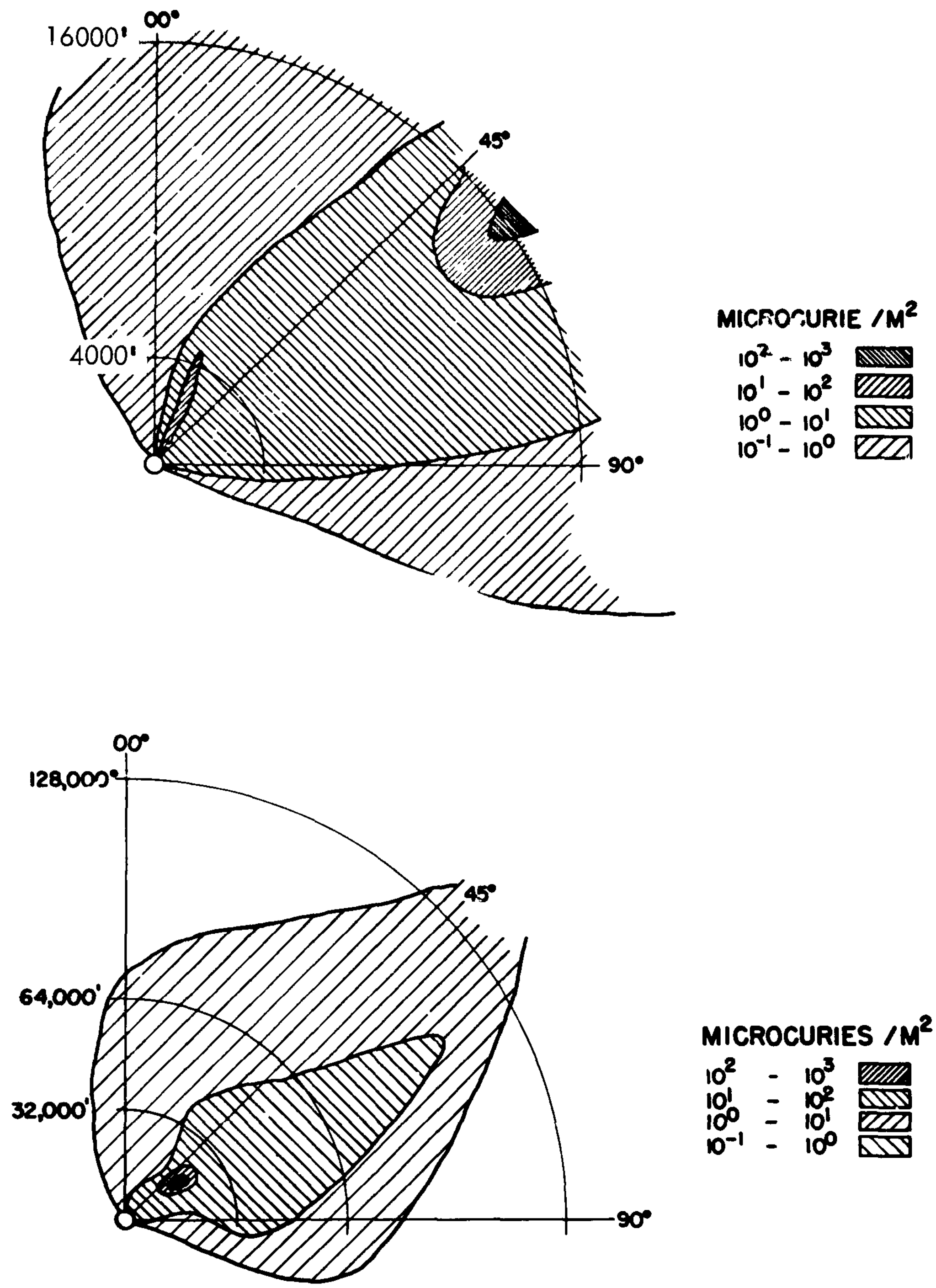


Figure F.2

NRX A-3 EP-5 Activity on resin-coated trays. Beta activity corrected to estimated time of cloud passage.

(LA-3394-MS, "Radiation Measurements of the Effluent from the NRX-A2 and NRX-A3 Reactors", Figure 29)

Figure F.3 indicates what may happen when test operations are conducted under very variable weather conditions. The resultant ground deposition pattern is not as simple as Figure F.1. In fact, the major portion of the cloud went to the northeast, not to the northwest where the highest ground concentrations occurred. The operation of reactors for longer times (30 to 45 minutes versus 15 minutes) will increase the probability that the cloud may be torn into several pieces and the pieces carried several different directions by changing winds.

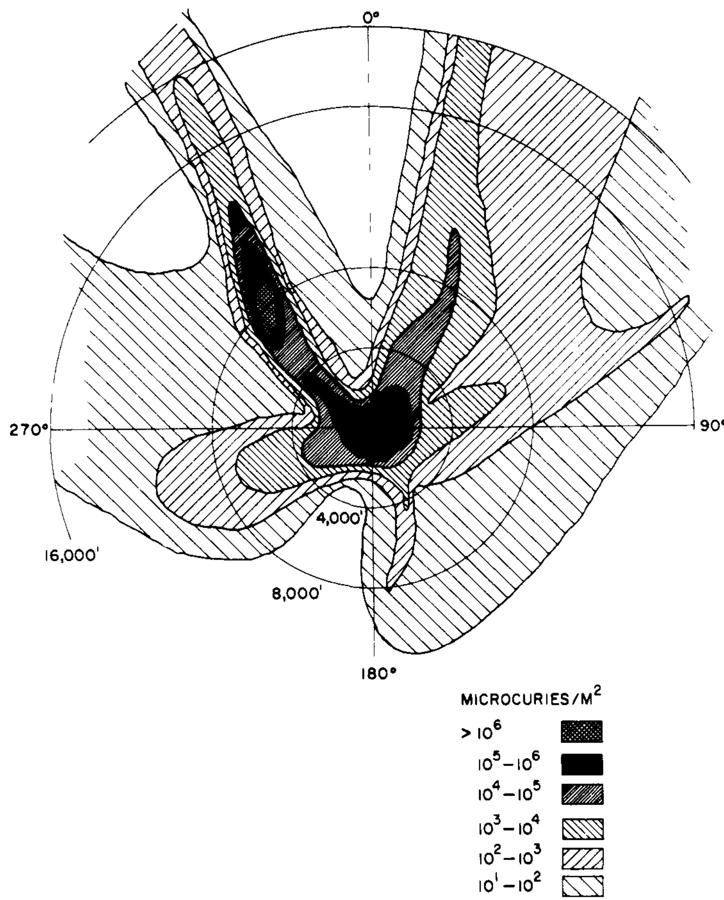
It is not to be expected, however, that either broadening the cloud path by local variability in the winds or the breaking the cloud into two or three distinct pieces will result in more activity deposited within a given radius of the test stand. Rather, the integrated concentration will be approximately the same and either the activity will be spread out over a larger area within one cloud path or will be divided among the several paths.

F.2 ESTIMATED OFF-SITE DOSES

F.2.1 Gaseous Fission Product Model

The diffusion of gaseous fission products in the atmosphere is based on Sutton's model. This model has given reasonable, usually conservative results when used to estimate the off-site doses due to ROVER testing. The whole body gamma doses from the passing cloud and ground deposition are calculated by the computer code GAMMA, which incorporates Sutton's model and includes decay of the cloud activity, depletion of the cloud by ground deposition, and a linear dose buildup factor to account for air scattering. The source term for this code is calculated by the FIPDIF and STP codes, described previously. The effect of cloud depletion and ground deposition of gaseous fission products is included by using a pseudo deposition defined as

$$V_g = \frac{\text{amount deposited/m}^2\text{ of horizontal surface/sec}}{\text{volumetric concentration above this surface}}$$



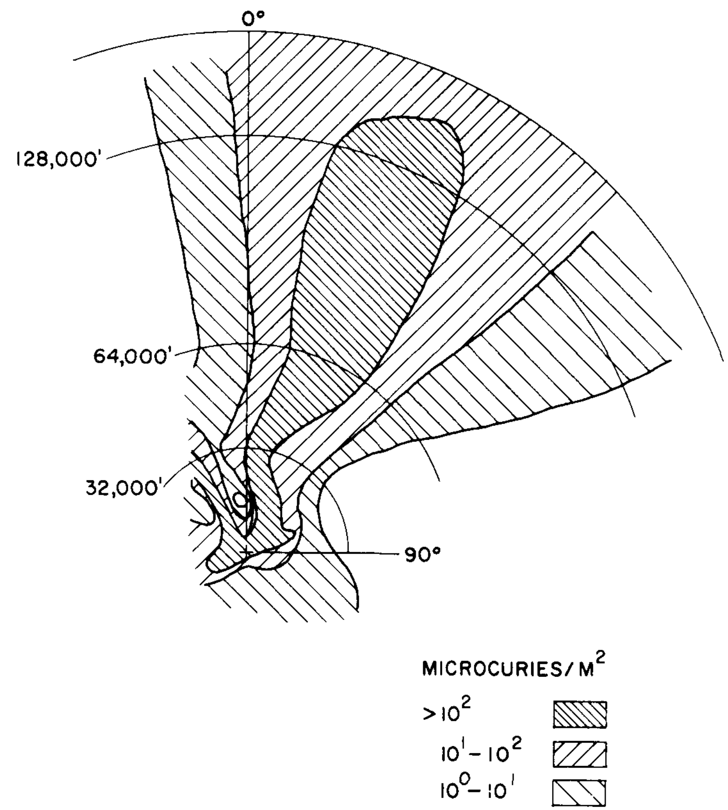


Figure F.3

Phoebus 1A 321 activity on resin-coated trays.

Beta activity corrected to estimated time of cloud passage.

(LA-3396, "Radiation Measurements of the Effluent from the PHOEBUS 1A-321 Reactor", Figure 20)

The gamma dose from ground deposition is directly proportional to the value of V_g, which for gross fission products, varies from 0.1 cm/sec to about 2.5 cm/sec in the literature[1,2]. LASL experimental measurements of deposition velocities of gaseous fission products during ROVER testing at NRDS have shown considerable variability, but a median representative value seems to be about 1 cm/sec. One cm/sec has been used in this report.

The thyroid inhalation doses are computed by the RISC program, again based on Sutton's model. The adult thyroid dose is that received by a standard man with a thyroid mass of 20 grams and a breathing rate of 20 m^3/day. The dose conversion factors for the iodine isotopes are as recommended by the Internal Committee on Radiological Protection[3]. Due to differences in breathing rate and thyroid mass, the dose to a child's thyroid is considered to be 2.5 times that to an adult. It is thus more limiting than the adult dose.

Contaminated food supplies may subject the population to radiation doses from internally absorbed radioisotopes. Of primary concern in this regard is the dose to the thyroid from 1-131 ingested with milk produced by cows feeding on contaminated forage. Due to the smaller size of the thyroid (2 grams) and higher per capita milk consumption (1 liter/day), the critical organ is considered to be a child's thyroid. The equation used to compute the dose is

DOSE (rem) = CURIES * f * TID * K

where CURIES = peak curies of 1-131 (assumed constant due to long half-life compared to transport time)

f = 131 chain release fraction

TID = integrated air concentration (by Sutton's model)

(1) NUS-122, "A Preliminary Evaluation of the Environmental Safety Aspects of Nuclear Rocket Flight Operations", January 1963.
(2) UCRL-14702, "Deposition Velocities of Aerosols and Vapors on Pasture Grass", March 1966.
(3) Health Physics, Volume 3, "Report of ICRP Committee II on Permissible Dose for Internal Radiation", June 1960.

The components of K include the deposition velocity, the conversion from iodine on the grass to iodine in the milk, the milk consumption rate, and the dose conversion factor. There is wide variation in the values of the above quantities in the literature, but fairly representative values and the ones used here are

Milk conversion factor[1]	$= \frac{4 \times 10^{-4} \text{ pc/lit}}{\text{pc/m}^3 \text{ sec}}$
Iodine effective half-life on grass	= 4.8 days
Ingestion rate	= 1 liter/day
1-131 dose conversion factor	= 1.936×10^7 rads/curie

Combining these values with the value for the peak curies of 1-131 (0.02 curies/MW-sec of operation) reduces the dose equation to

DOSE (rem) = 1000 * f * Integrated Power (MW-Sec) * TID

F.2.2 Particulate Fission Product Dose Model

At the present time the fractionation of the corrosion fission products between gaseous and particulate forms is unknown. Fission product particulate deposition was first observed following the NRX-A3 test, but a systematic effort at establishing ground concentrations and particle size distributions was not made until the Phoebus 1B test. In Table F.1 are shown the size distributions and ground concentrations as observed following the Phoebus 1B full power test. Pan American surveyed one location forty-five miles downwind from Test Cell "C" and seventeen particles were found in 12 m^2. The Public Health Service found one particle at 82 miles. It exhibited a dose rate of 20 mr/hr combined β-γ at contact at collection time (~3 days post-test). Of the 86 particles found by Pan Am at distances greater than 12 miles which were measured for size, 52% were less than 40μ in size, 34% in the 50-90μ range, and 13% > 90μ. All particles found on the 45 mile arc (furthest arc surveyed) were less than 40μ in size.

(1) Recommended by the United States Public Health Service.

TABLE F.1

PARTICULATE SIZE DISTRIBUTION AND GROUND CONCENTRATION

(Phoebus 1B Data)(1)

Distance	Most Likely Size Range (μ)
45 miles	10-40
25	10-40
18	10-40
12	50-90
6	>90
3	>90
8,000 feet	>90

Distance	Highest Ground Concentration Particles/10 m^2
45 miles	14
25	14
23	12
15	12
5	33
2.5	24
16,000 feet	44
8,000	58
4,000	104
2,000	372

(1) Compiled from material presented at the Ten-Day Critique Meeting for Phoebus 1B, held in Las Vegas, Nevada, on March 10, 1967.

A model is not available to predict the downwind size distributions and ground concentrations of particulate from an engine test conducted at E/STS 2-3. The cooling of the effluent by the water spray injection in the duct and the inclination of the ejector exhaust from the vertical will undoubtedly influence the ground pattern, preventing a direct extrapolation from the Phoebus 1B data. All of the differences in the test system should tend to reduce the velocity of the effluent and thus decrease the height to which the particles rise, decreasing the fall time and the downwind impact position.

Particulates in the size range observed following previous reactor tests do not contain enough activity to pose a gamma radiation hazard. They may, however, produce quite high localized beta doses. Studies by the U.S. Naval Radiological Defense Laboratory indicate that the beta skin point dose rate from a 75 micron fuel bead fragment one hour after release from the reactor (after operation at 5000 MW for 30 minutes) could be of the order of several million rads/hour to the active skin layer (100μ below the surface). The dose from other sized particles depends roughly on the particle diameter to the 2.5 power, and the dose rate varies approximately inversely with time. For example, the skin beta point dose rate at 50 miles due to a 40μ UC_2 particle from a 30 minute 5000 MW test would be on the order of 300,000 R/hour (assuming a 12 mph wind velocity). These numbers may be misleading in that the majority of particles found exhibit much less radioactivity than would be expected based on their size alone.

The medical significance of such a dose is not clear. The majority of the beta energy is absorbed within 1 centimeter of the particle, while the average dose varies markedly with the irradiated area considered in the calculation. Such particles can produce small lesions in the skin. LASL has performed a series of experiments with irradiated fuel beads on the skin of a monkey involving integrated point doses up to 52,000 rads. Seventy-two hours after placement of the beads on the skin it was not possible to distinguish definitive gross qualitative differences in the lesions (small, shallow vesicles) as a function of exposure dose for those doses above about 25,000 rads. Ninety days past exposure no lesions were visible or palpable at any irradiated site. LASL has concluded that all lesions produced by this series of experiments were of an inconsequential medical nature[1]. One possible long term effect of skin irradiation is tumor formation. An equation [number of tumors/particle = 2.3×10^{-7} (point dose in rads)] can be obtained from the information presented in Figure VII-2 of USNRDL-TR-1010[2]. This equation would indicate that the interaction of any one particle with any given individual would result in a low probability of tumor formation. When many particles and many individuals are involved, however, the probability of some tumor formation increases proportionately. The information in the USNRDL report is based on experiments involving irradiation of relatively large areas of mouse skin. Based on analysis of medical information obtained over a period of time on the Rongelap islanders accidentally exposed to fallout from the Bikini weapons test series of 1954, it would appear that the above relationship does not apply to the dose received from weapon debris. Using the above formula one would expect to see 166 tumors in the Rongelap people within 10 years of the exposure. The October 1966 medical survey did not disclose any indication of tumors. Based on the medical history of the inhabitants of Rongelap and neighboring islands it would appear that beta burns are not a problem from an exposure to particulate matter from weapons testing at locations where the infinite whole body dose is less than 10 rad. Consideration must be made of the possible effects of particulate deposition when evaluating the effects of an engine test, particularly if the wind is blowing toward a nearboy populated area such as Lathrop Wells. If an off-site individual received such a beta dose, it would be an isolated instance. No estimate is made of the possible beta dose to such an individual, except as mentioned previously. Beta doses are not included in the estimated off-site doses of the next section.

(1) LA-3365-MS, "Some Biological Aspects of Radioactive Microspheres", August 1965.
(2) USNRDL-TR-1010, "Radiological Considerations in Nuclear Flight Safety", April 1966.

F.2.3 Estimated Off-Site Doses

The estimated off-site gamma doses are based on the models of this section and the source terms of Table E.1. Tests have in the past been conducted under lapse conditions, that condition of the atmosphere conducive to rapid dilution of the effluent. Experience has indicated that a wind speed of 12 mph is typical under such conditions. In the past the typical cloud height has been 1200-1500 meters. It is probable that the effluent from a NERVA engine will not rise to these heights due to the lower temperature of the effluent at the ejector exit. A release height of 1000 meters has therefore been used. Examination of Figure F.8 indicates that this has a minor effect on the off-site doses. The estimated gamma doses from the cloud, gaseous F.P. deposition, iodine inhalation, and milk are given in Table F.2 for the 5000 MW normal operation test, both for the combined diffusion and corrosion and for the diffusion alone. The analogous table for the 10,000 MW operation is Table F.3. The estimated accident doses from these two operations are presented in Table F.4. For comparative purposes the whole body gamma doses from the cloud appear on Figure F.4 along with the AEC recommended standards. The iodine inhalation doses appear in Figure F.5. As mentioned previously, these doses are given for lapse conditions, a release height of 1000 meters, and a wind speed of 12 miles/hour. Figures F.6 through F.11 indicate the effect of other weather conditions on the estimated centerline doses. The model parameters for these conditions are given in Table F.5. The activity release used for this series of figures is the full release of the fission products generated by 10^4 MW-Sec of operation. The number 10^4 is not itself significant, the primary intent being illustration of downwind atmospheric diffusion for various conditions. The effect of the stability of the atmosphere may be seen in Figures F.6 and F.7 while the effect of changes in the cloud release height is shown in Figures F.8 and F.9. Figures F.10 and F.11 illustrate the dependence of the dose on wind velocity. The reduction in dose for receptor positions off the cloud centerline is shown in Figures F.12 and F.13.

TABLE F.2

ESTIMATED OFF-SITE GAMMA DOSES (REM) FROM NORMAL OPERATION AT 5000 MW FOR 30 MINUTES UNDER LAPSE CONDITIONS

TYPE OF DOSE		DISTANCE (MILES) 12	20	40	100
Combined Diffusion and Corrosion					
Gamma:	Cloud	6.0×10^{-1}	1.5×10^{-1}	1.8×10^{-2}	1.1×10^{-3}
	Gaseous F.P. Deposition - 1 Hour	2.5×10^{-1}	7.2×10^{-2}	1.4×10^{-2}	7.8×10^{-4}
	Gaseous F.P. Deposition - 100 Days	1.4×10^{0}	5.7×10^{-1}	1.4×10^{-1}	2.2×10^{-2}
Thyroid:	Inhalation (child)	4.5×10^{0}	2.1×10^{0}	6.0×10^{-1}	8.5×10^{-2}
	Milk	2.3×10^{1}	1.1×10^{1}	3.5×10^{0}	7.4×10^{-1}
Diffusion Only					
Gamma:	Cloud	2.9×10^{-1}	7.6×10^{-2}	1.0×10^{-2}	6.8×10^{-4}
	Gaseous F.P. Deposition - 1 Hour	1.3×10^{-1}	3.8×10^{-2}	6.3×10^{-3}	4.6×10^{-4}
	Gaseous F.P. Deposition - 100 Days	7.8×10^{-1}	3.2×10^{-1}	8.2×10^{-2}	1.2×10^{-2}
Thyroid:	Inhalation (child)	1.9×10^{0}	9.0×10^{-1}	2.5×10^{-1}	3.7×10^{-2}
	Milk	1.3×10^{1}	6.3×10^{0}	2.0×10^{0}	4.2×10^{-1}

TABLE F.3

ESTIMATED OFF-SITE GAMMA DOSES (REM) FROM NORMAL OPERATION
AT 10,000 MW FOR 45 MINUTES UNDER LAPSE CONDITIONS

TYPE OF DOSE		DISTANCE (MILES) 12	20	40	100
Combined Diffusion and Corrosion					
Gamma:	Cloud	2.0×10^{0}	5.1×10^{-1}	6.2×10^{-2}	4.0×10^{-3}
	Gaseous F.P. Deposition - 1 Hour	8.5×10^{-1}	2.5×10^{-1}	4.9×10^{-2}	2.7×10^{-3}
	Gaseous F.P. Deposition - 100 Days	4.9×10^{0}	2.0×10^{-0}	5.1×10^{-1}	7.6×10^{-2}
Thyroid:	Inhalation (child)	1.6×10^{1}	7.5×10^{0}	2.0×10^{0}	3.0×10^{-1}
	Milk	7.8×10^{1}	3.8×10^{1}	1.2×10^{1}	2.5×10^{0}
Diffusion Only					
Gamma:	Cloud	1.1×10^{0}	2.9×10^{-1}	3.9×10^{-2}	2.7×10^{-3}
	Gaseous F.P. Deposition - 1 Hour	4.8×10^{-1}	1.5×10^{-1}	2.4×10^{-2}	1.8×10^{-3}
	Gaseous F.P. Deposition - 100 Days	3.0×10^{0}	1.2×10^{0}	3.2×10^{-1}	4.8×10^{-2}
Thyroid:	Inhalation (child)	8.0×10^{0}	3.7×10^{0}	1.0×10^{0}	1.5×10^{-1}
	Milk	4.8×10^{1}	2.3×10^{1}	7.5×10^{0}	1.6×10^{0}

TABLE F.4

ESTIMATED OFF-SITE GAMMA DOSES (REM) FROM LOSS OF COOLANT UNDER LAPSE CONDITIONS

TYPE OF DOSE		DISTANCE (MILES) 12	20	40	100
30 Minutes at 5000 MW					
Gamma:	Cloud	2.5×10^{0}	5.6×10^{-1}	6.1×10^{-2}	3.7×10^{-3}
	Gaseous F.P. Deposition - 1 Hour	1.2×10^{0}	3.0×10^{-1}	4.3×10^{-2}	3.0×10^{-3}
	Gaseous F.P. Deposition - 100 Days	7.1×10^{0}	2.5×10^{0}	6.1×10^{-1}	9.2×10^{-2}
Thyroid:	Inhalation (child)	7.5×10^{1}	3.0×10^{1}	7.0×10^{0}	1.0×10^{0}
	Milk	2.8×10^{2}	1.1×10^{2}	3.3×10^{1}	6.5×10^{0}
45 Minutes at 10,000 MW					
Gamma:	Cloud	6.7×10^{0}	1.6×10^{0}	2.0×10^{-1}	1.1×10^{-2}
	Gaseous F.P. Deposition - 1 Hour	3.3×10^{0}	8.5×10^{-1}	1.3×10^{-1}	8.8×10^{-3}
	Gaseous F.P. Deposition - 100 Days	1.1×10^{1}	7.3×10^{0}	1.8×10^{0}	2.8×10^{-1}
Thyroid:	Inhalation (child)	2.2×10^{2}	8.6×10^{1}	2.1×10^{1}	3.0×10^{0}
	Milk	8.3×10^{2}	3.4×10^{2}	9.9×10^{1}	1.9×10^{1}

TABLE F.5

SUTTON DIFFUSION MODEL PARAMETERS

Parameter	Lapse	Moderate Inversion	Inversion
Diffusion Coefficient c^2 (meters)^n	0.1*	0.04	0.01*
Stability Factor n (dimensionless)	0.23*	0.33	0.5*
Mean Wind Speed u (meters/secon)	5.36*	2.68	1.34
Effective Release Height (Normal Operation) h (meters)	1500	600	600
Effective Release Height (Accident h (meters)	0	0	0
Effective Release Height (Capped Inversion) h (meters)	----	----	150

*These parameters have been recommended by the USWB, Las Vegas.

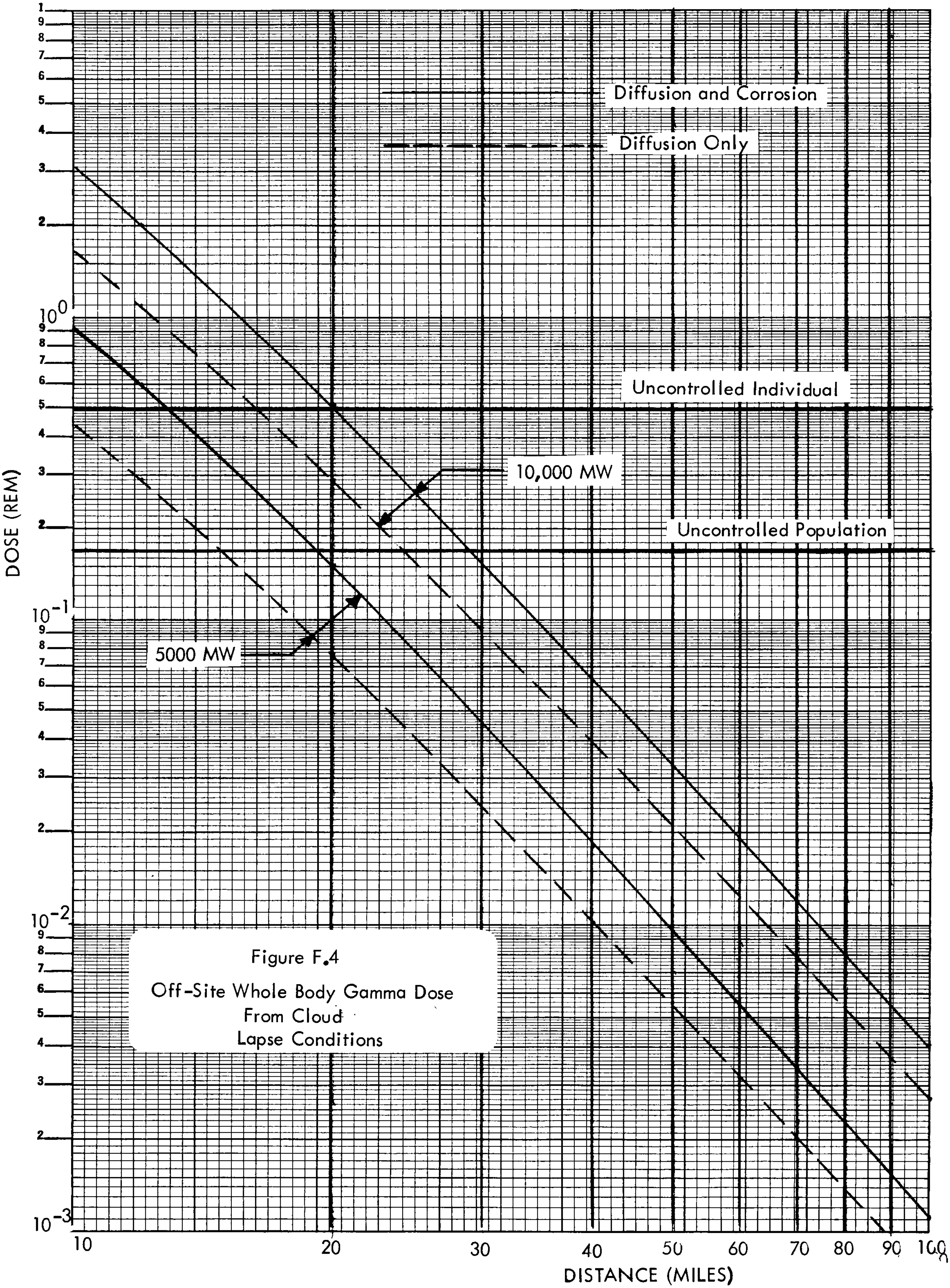


Figure F.4
Off-Site Whole Body Gamma Dose
From Cloud
Lapse Conditions

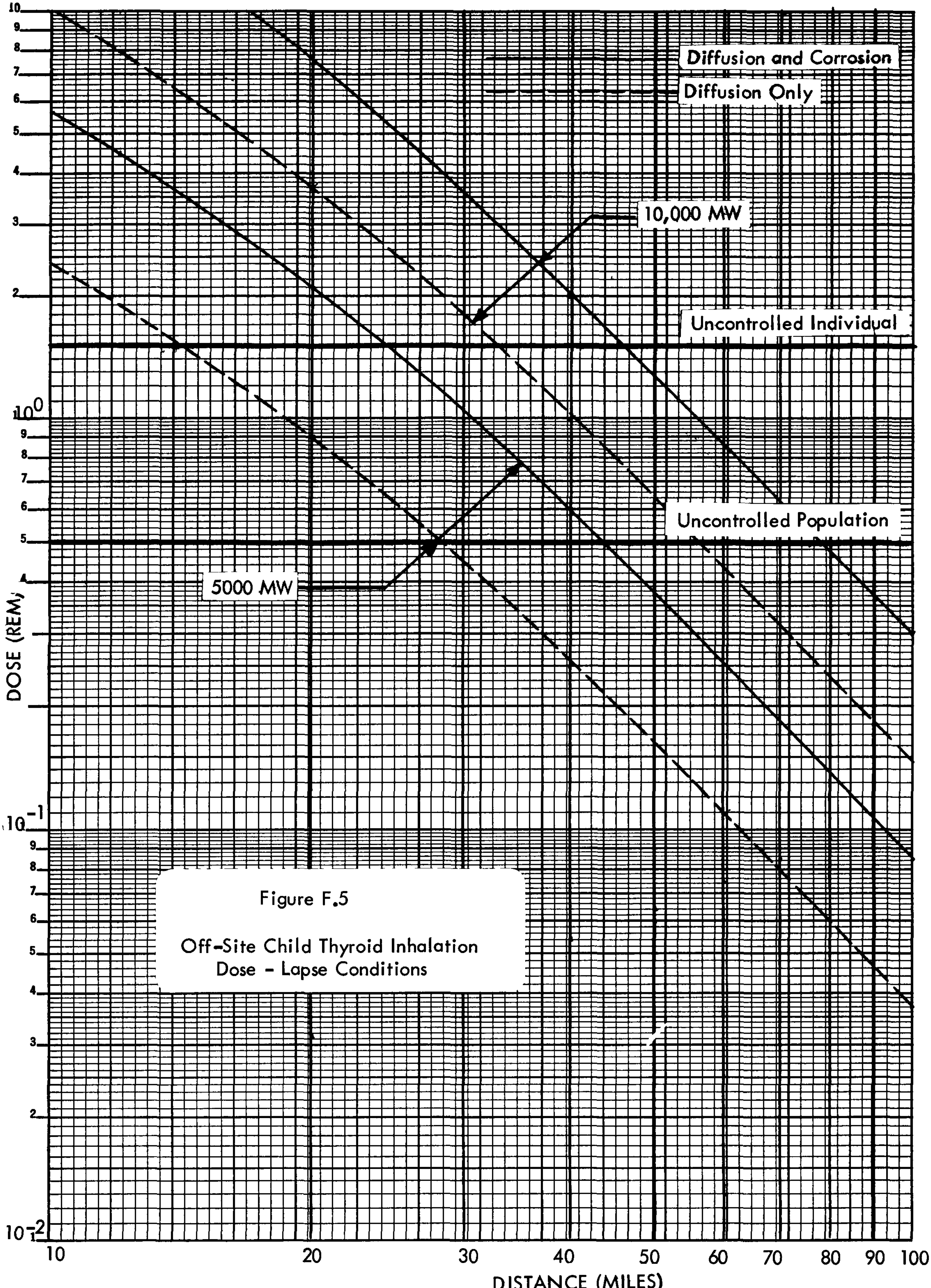


Figure F.5

Off-Site Child Thyroid Inhalation Dose - Lapse Conditions

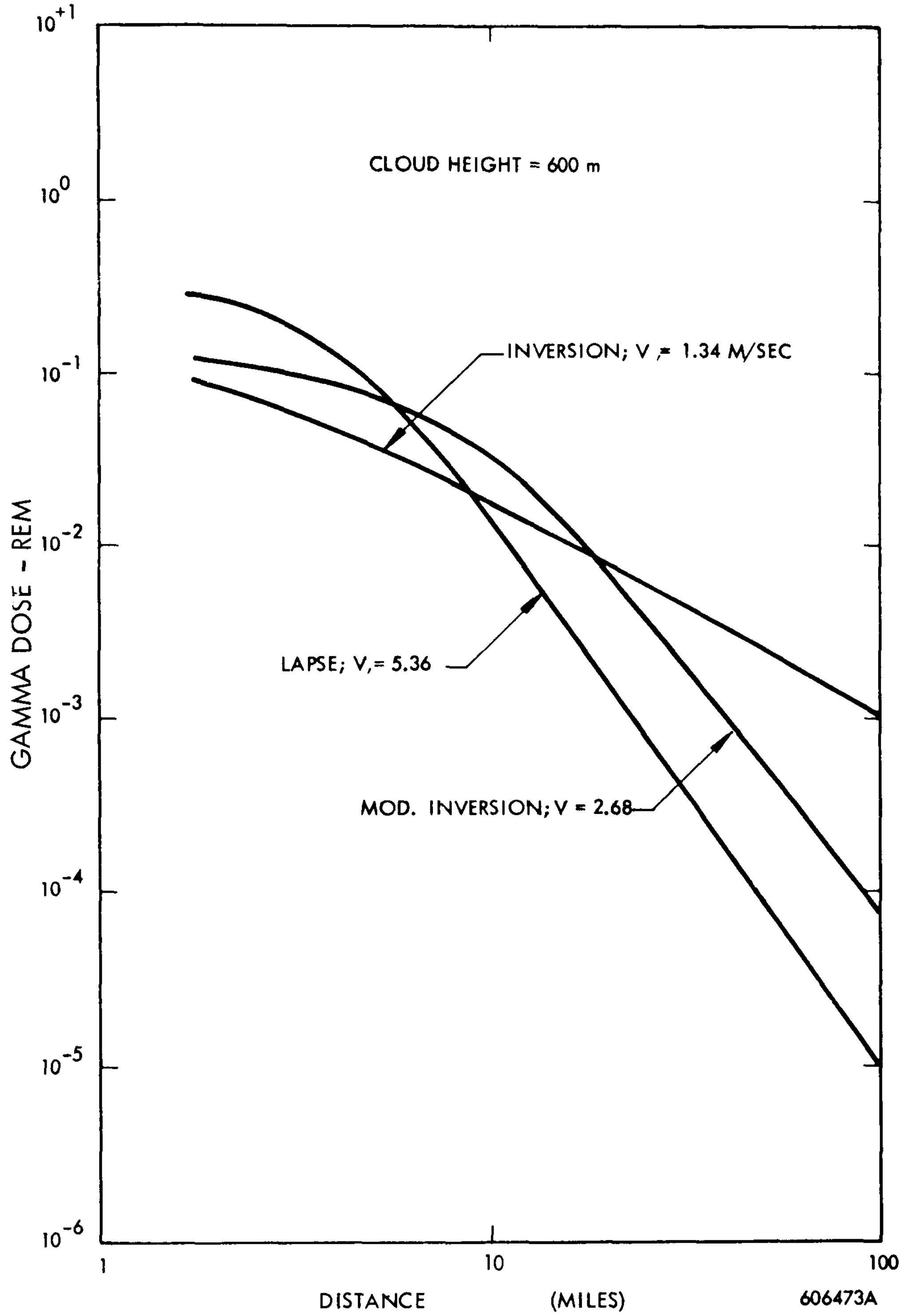


Figure F.6 Effect of Weather Conditions on the Gamma Dose from the Passing Cloud Full Release of 10^4 MW-Second Inventory

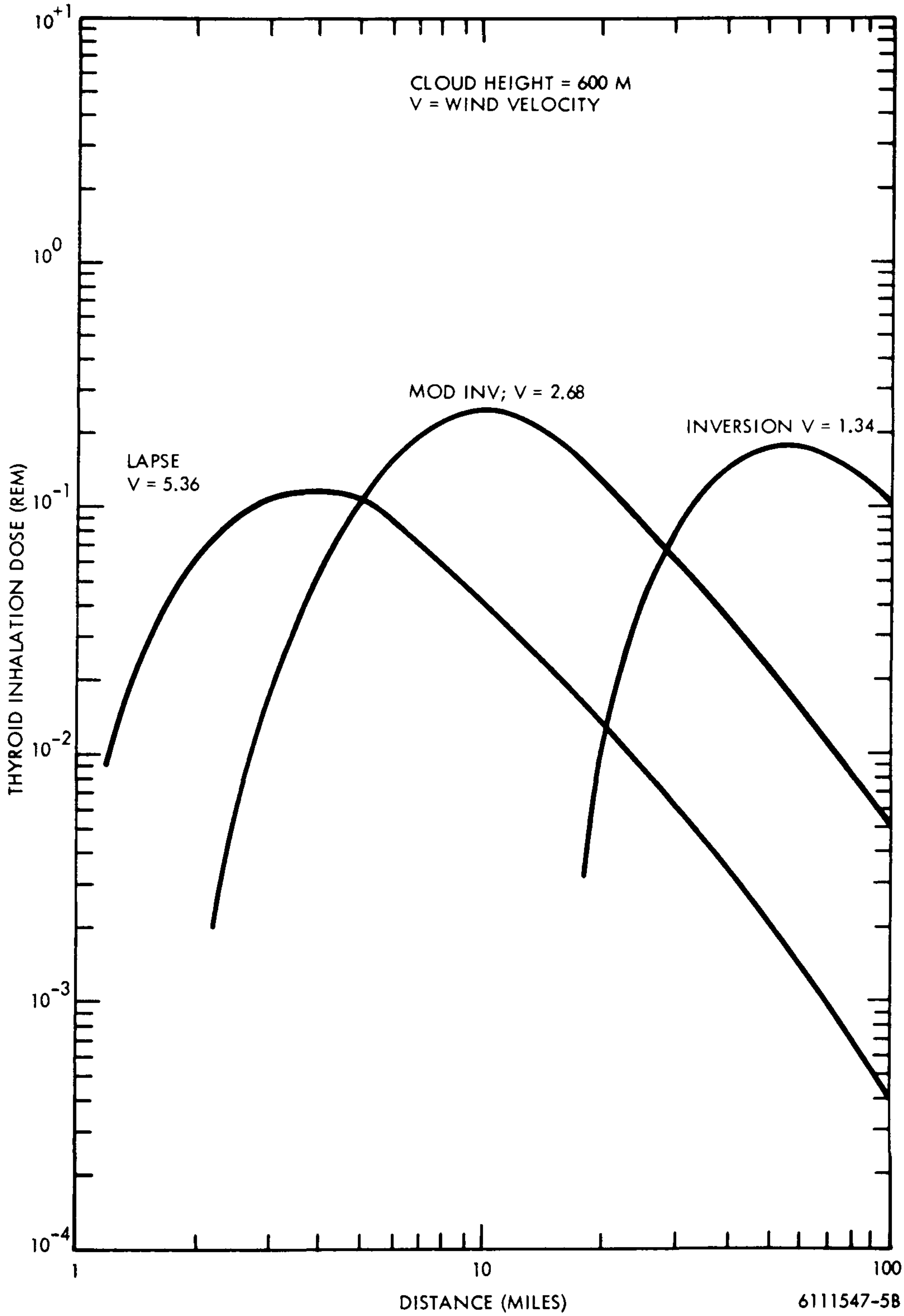


Figure F.7 Effect of Weather Conditions on Adult Thyroid Inhalation Dose - Full Release of 10^4 MW-Second Inventory

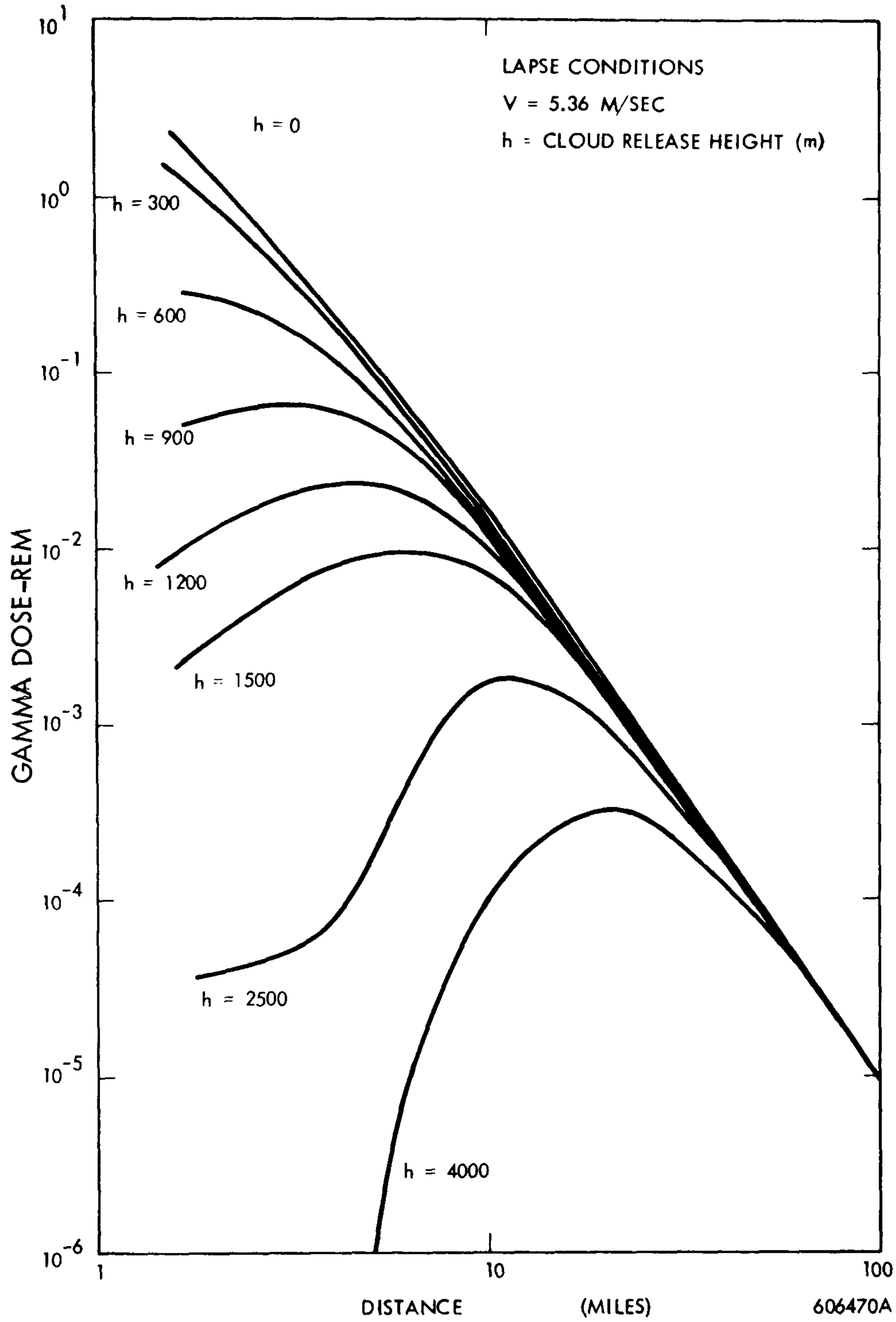


Figure F.8 Effect of Cloud Release Height on the Gamma Dose from the Passing Cloud Full Release of 10^4 MW-Second Inventory

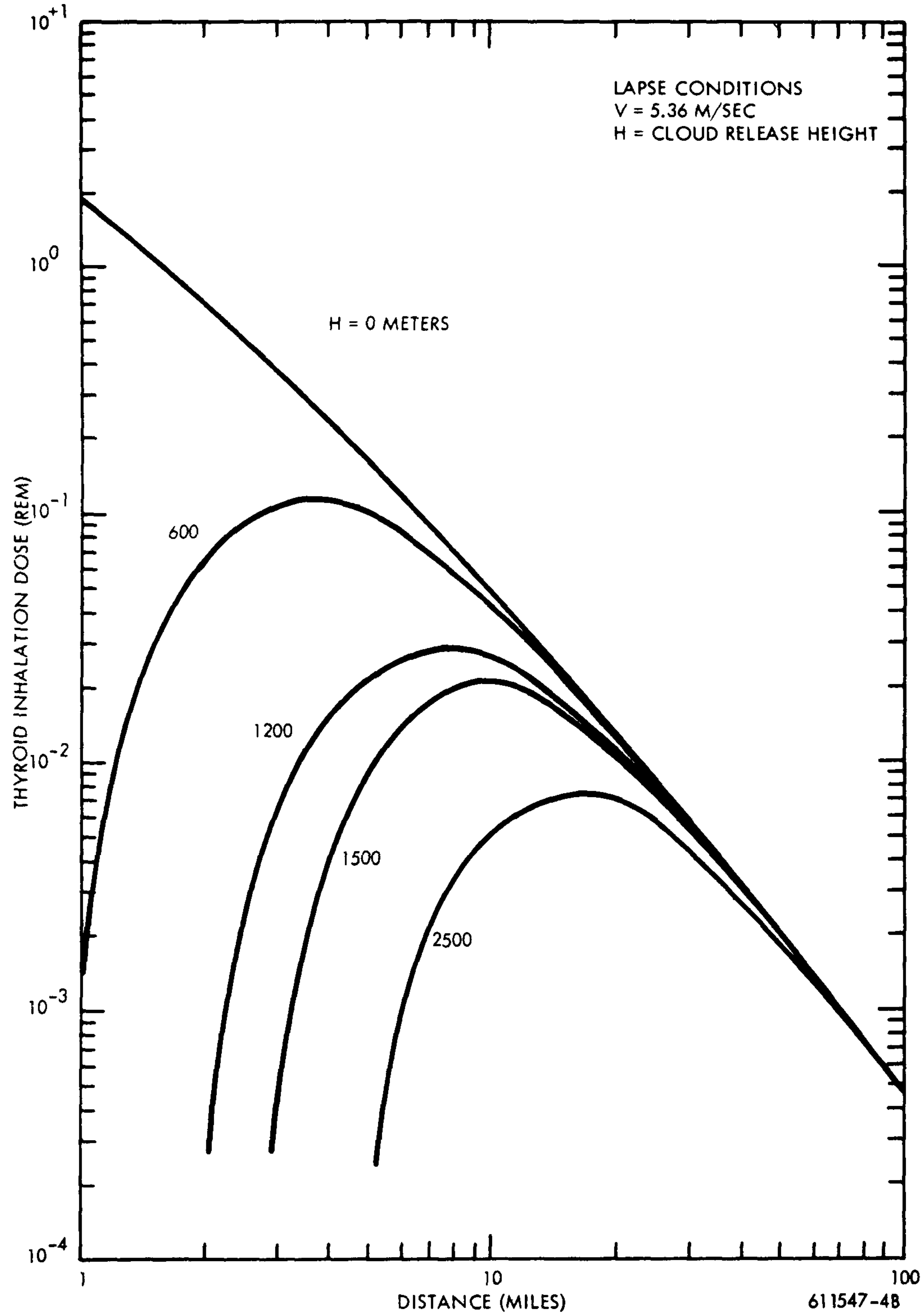


Figure F.9 Effect of Cloud Release Height on Adult Thyroid Inhalation Dose - Full Release of 10^4 MW-Second Inventory

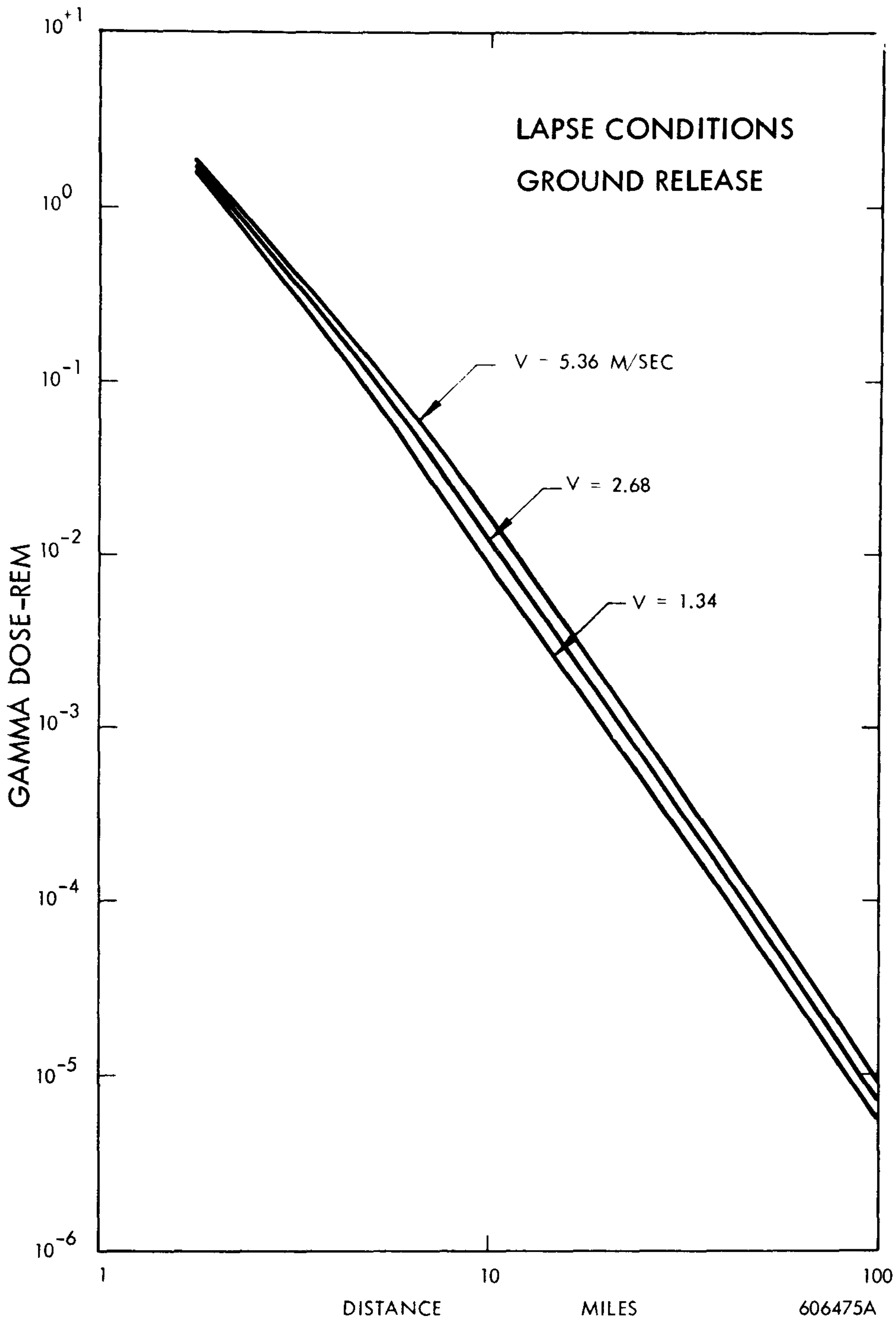


Figure F.10 Effect of Wind Velocity on the Gamma Dose from the Passing Cloud
Full Release of 10^4 MW-Second Inventory

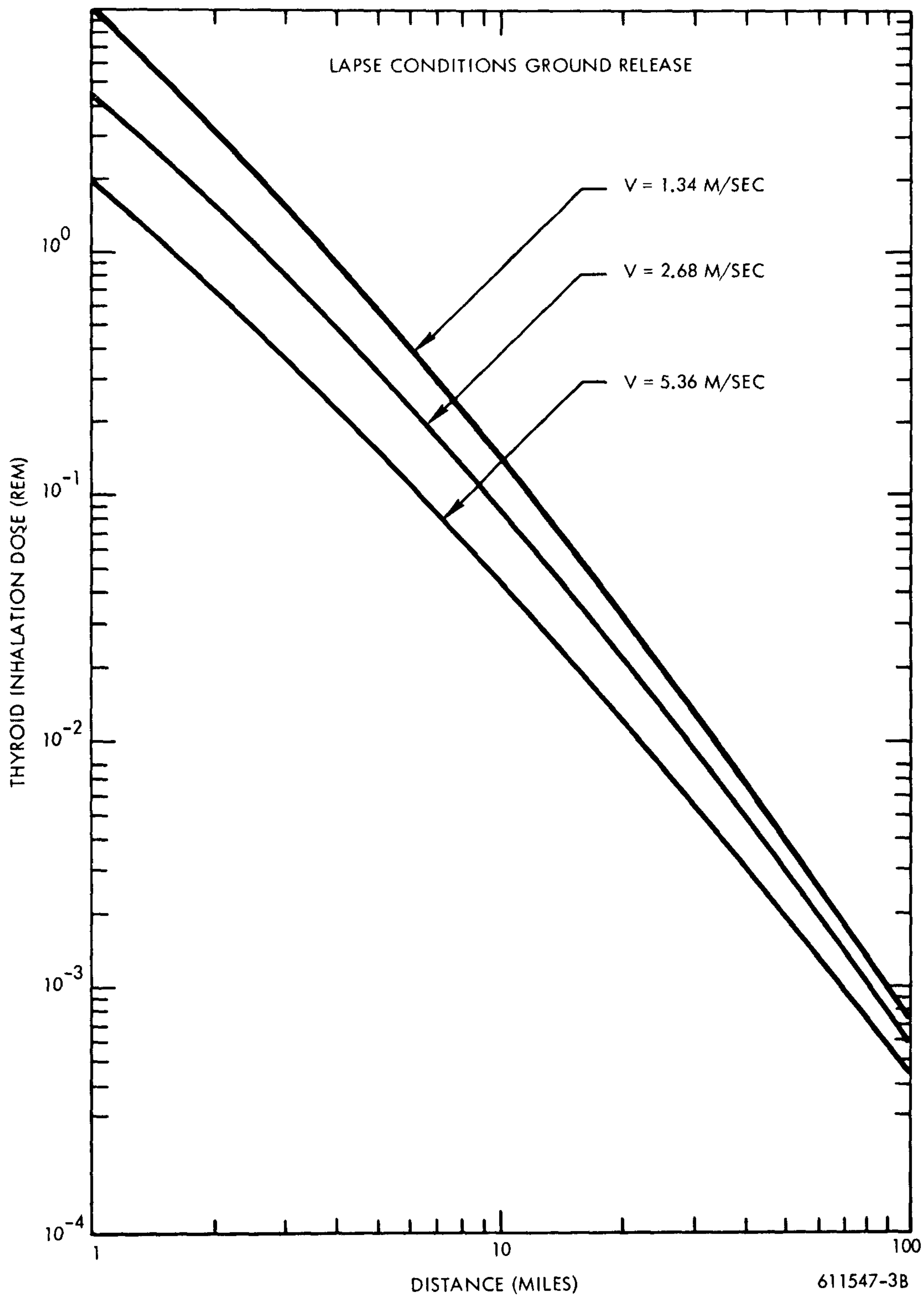


Figure F.11 Effect of Wind Velocity on Adult Thyroid Inhalation Dose - Full Release of 10^4 MW-Second Inventory

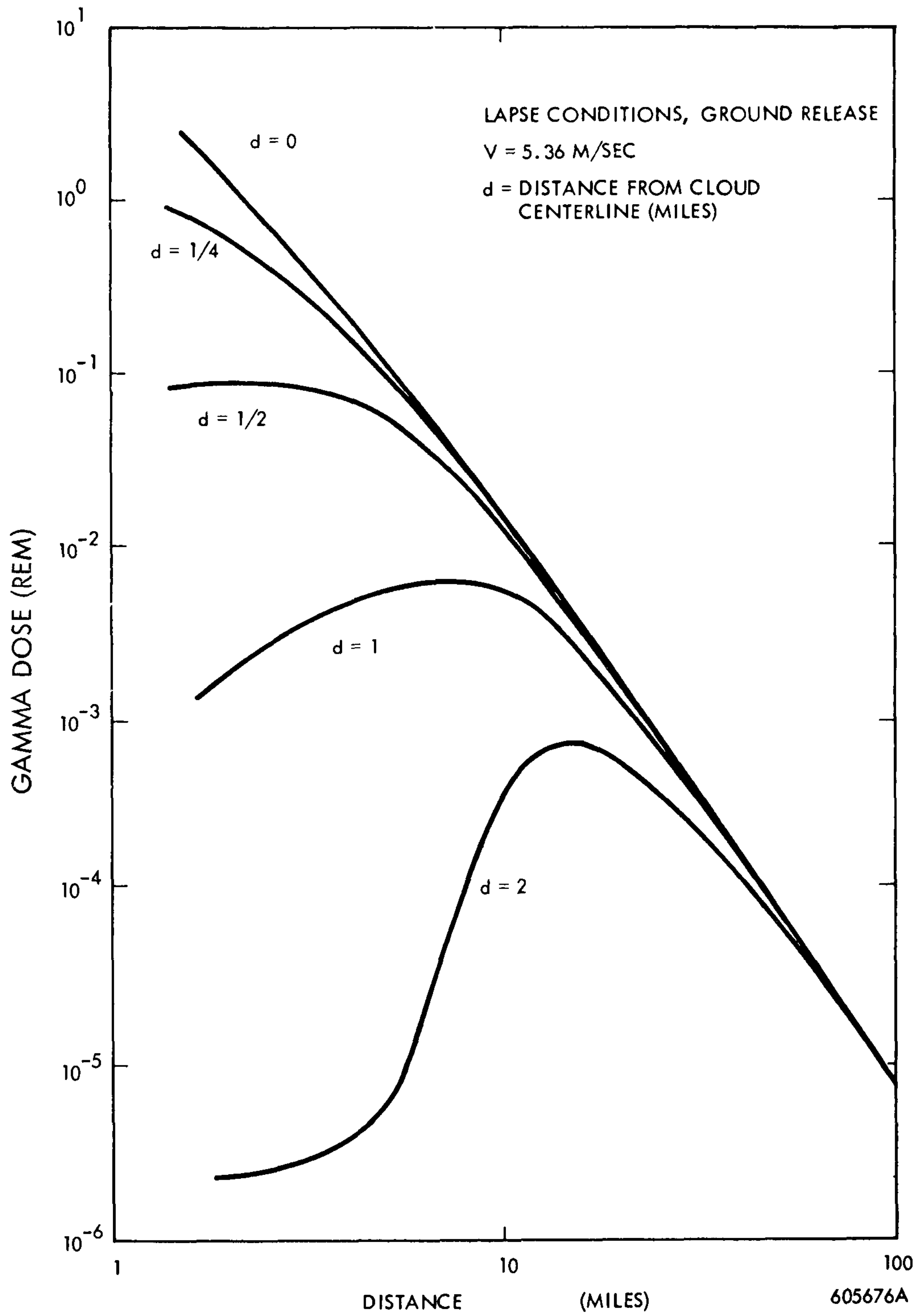


Figure F.12 Off-Centerline Gamma Dose from the Passing Cloud Ground Release of 10^4 MW-Second Inventory

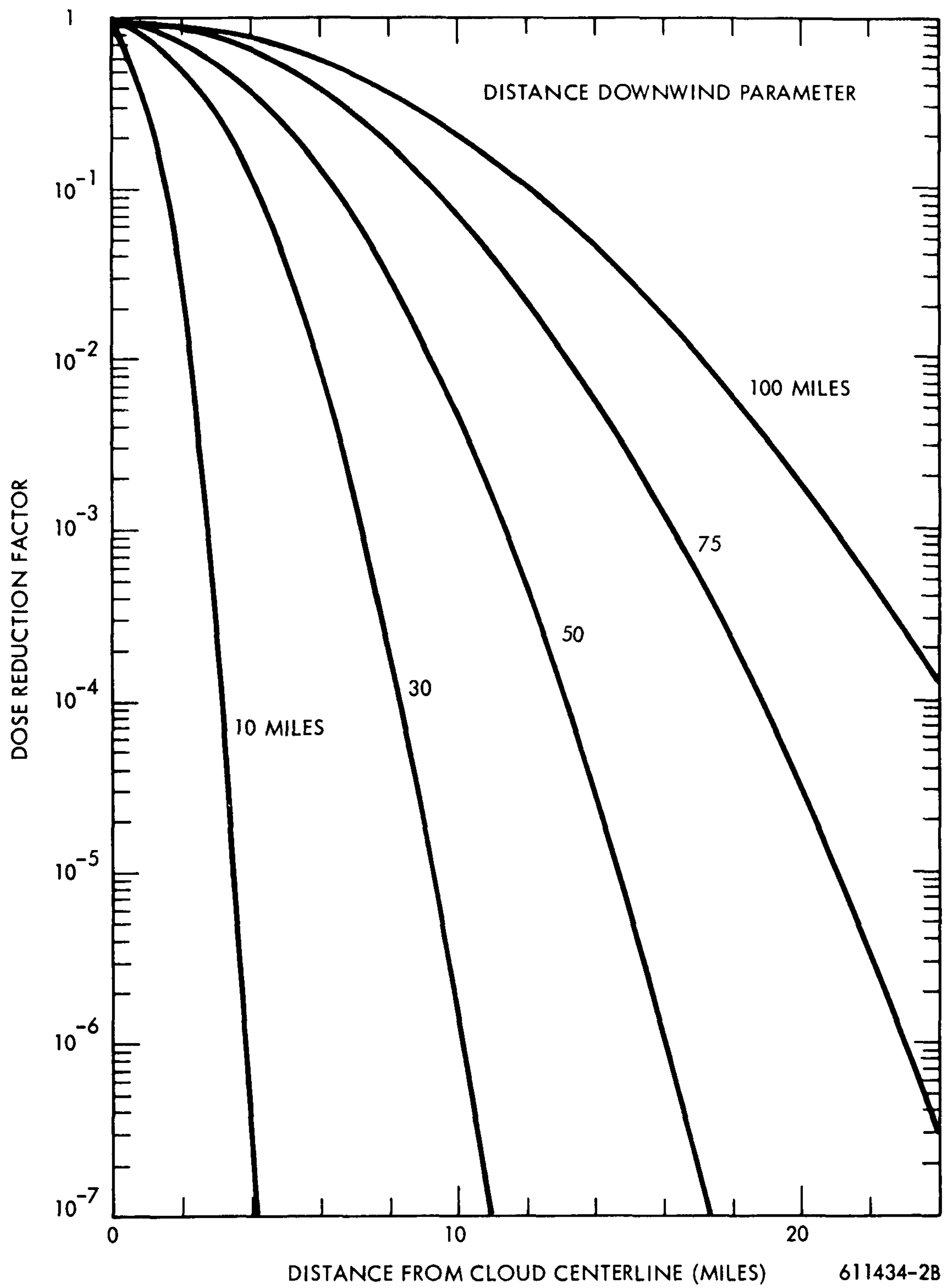


Figure F.13 Reduction in Iodine Dose versus Distance from Cloud Centerline for Lapse Conditions

Some conclusions may be drawn from these figures. Off-site doses are increased for operation during other than lapse conditions, the gamma dose from the cloud received at 100 miles under inversion conditions is two orders of magnitude higher than that received under lapse. Previous reactor tests in Nevada have been conducted under lapse conditions because of this rapid dilution of the effluent in the atmosphere. The off-site doses are affected very little by changes in the effective release height. This reduces the uncertainty in the off-site dose estimate introduced by the unknown effects of the exhaust duct on the cloud release height. The gamma dose is almost independent of the wind velocity. The cloud track at 100 miles is fairly broad, there being a negligible dose reduction for distances of less than a couple miles from the cloud centerline.

F.3 ESTIMATED ON-SITE DOSES

F.3.1 Radiation from the Effluent Cloud

The source term, atmospheric, and dose models are presented in Section F.1.1. The only difference between that description and the model used here is that 1% of the cloud is assumed released at ground level. This adjustment is based on downwind measurements following reactor tests which indicate that the close in doses are higher than would be predicted by the elevated model alone. The results are presented for lapse conditions and a wind speed of 12 miles/hour. Figure F.14 includes the whole body gamma dose from the cloud and the thyroid inhalation dose from normal operation at 5000 MW for 30 minutes. (The 10,000 MW 45 minute doses would be roughly 3 times those shown.) Also indicated on the figure are the tentative SNPO guides (per test) for radiation workers in controlled areas (Table 2.1). Again, as in Section F.2, results are given for combined corrosion and diffusion and for diffusion only. Relative changes in the dose for other weather conditions would be as shown by Figures F.6 to F.11.

The loss of coolant accident is discussed in Section E.3. Figure F.15 contains the estimated doses from this incident, along with the SNPO standards for essential radiation workers under accident conditions.

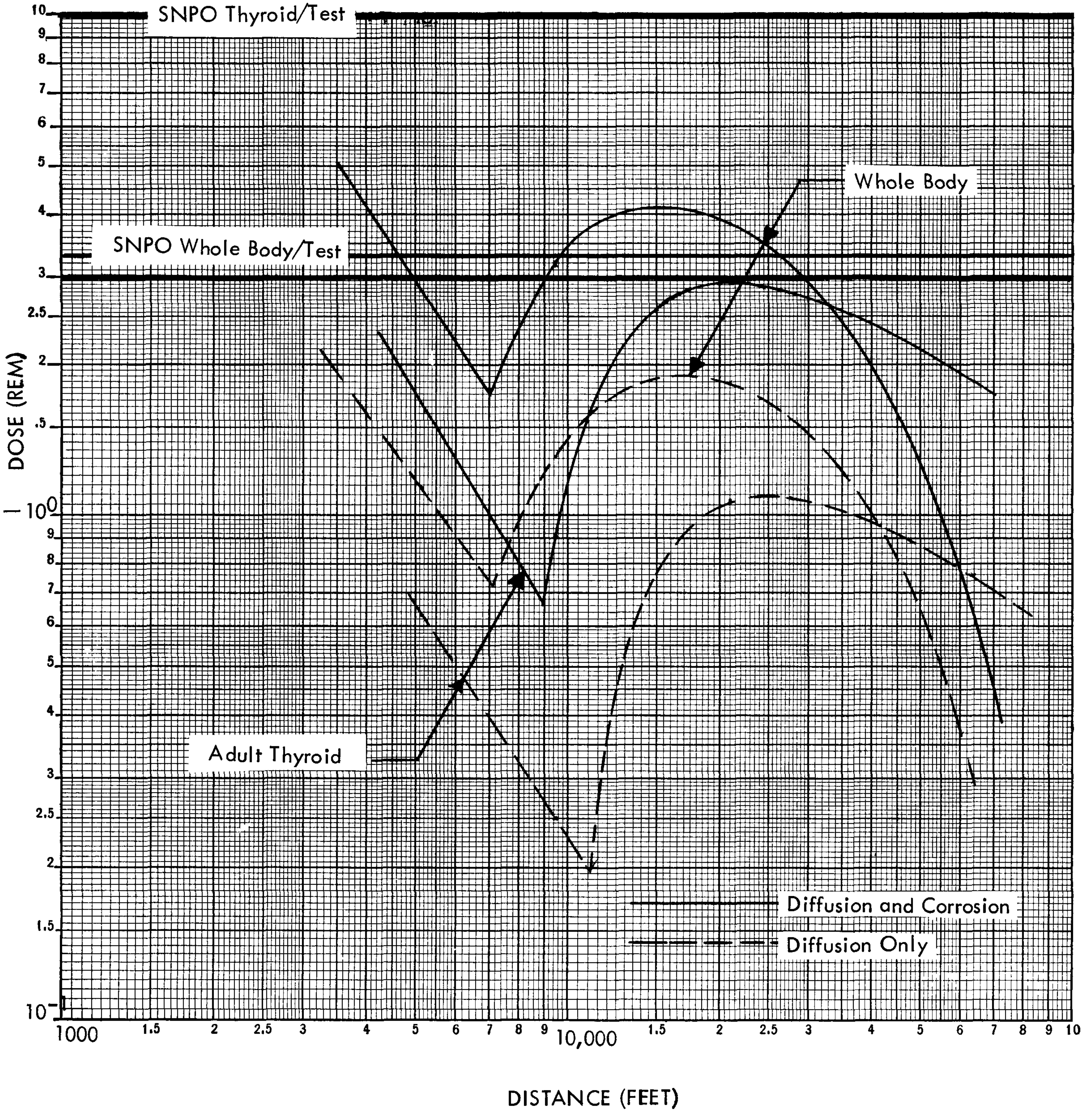
F.3.2 Radiation from Ground Deposition

F.3.2.1 Normal Operation Deposition Pattern

The normal deposition patterns are again based on the models of Section F.1.1. Results are presented for a normal operation of 30 minutes at 5000 MW and a deposition velocity of 1 cm/second under lapse conditions. Figure F.16 presents the result of deposited gaseous activity for various times post event.

Figure F.14

Dose from Normal Operation Effluent
Cloud 30 Minute at 5000 MW
Lapse Conditions



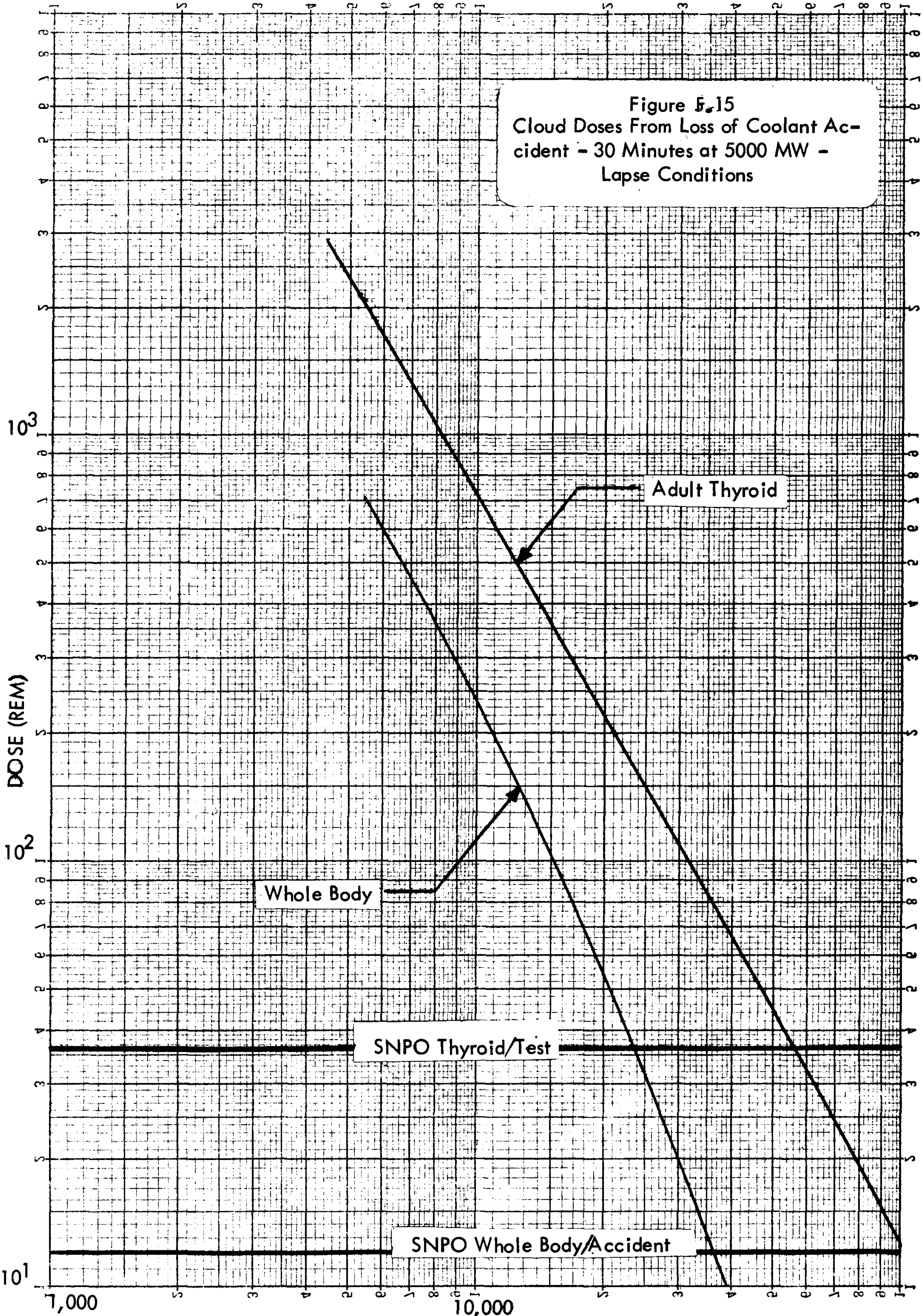


Figure F.15
Cloud Doses From Loss of Coolant Accident - 30 Minutes at 5000 MW - Lapse Conditions
Adult Thyroid
Whole Body
SNPO Thyroid/Test
SNPO Whole Body/Accident
DOSE (REM)
10^3
10^2
10^1
1,000
10,000
DISTANCE (FEET)

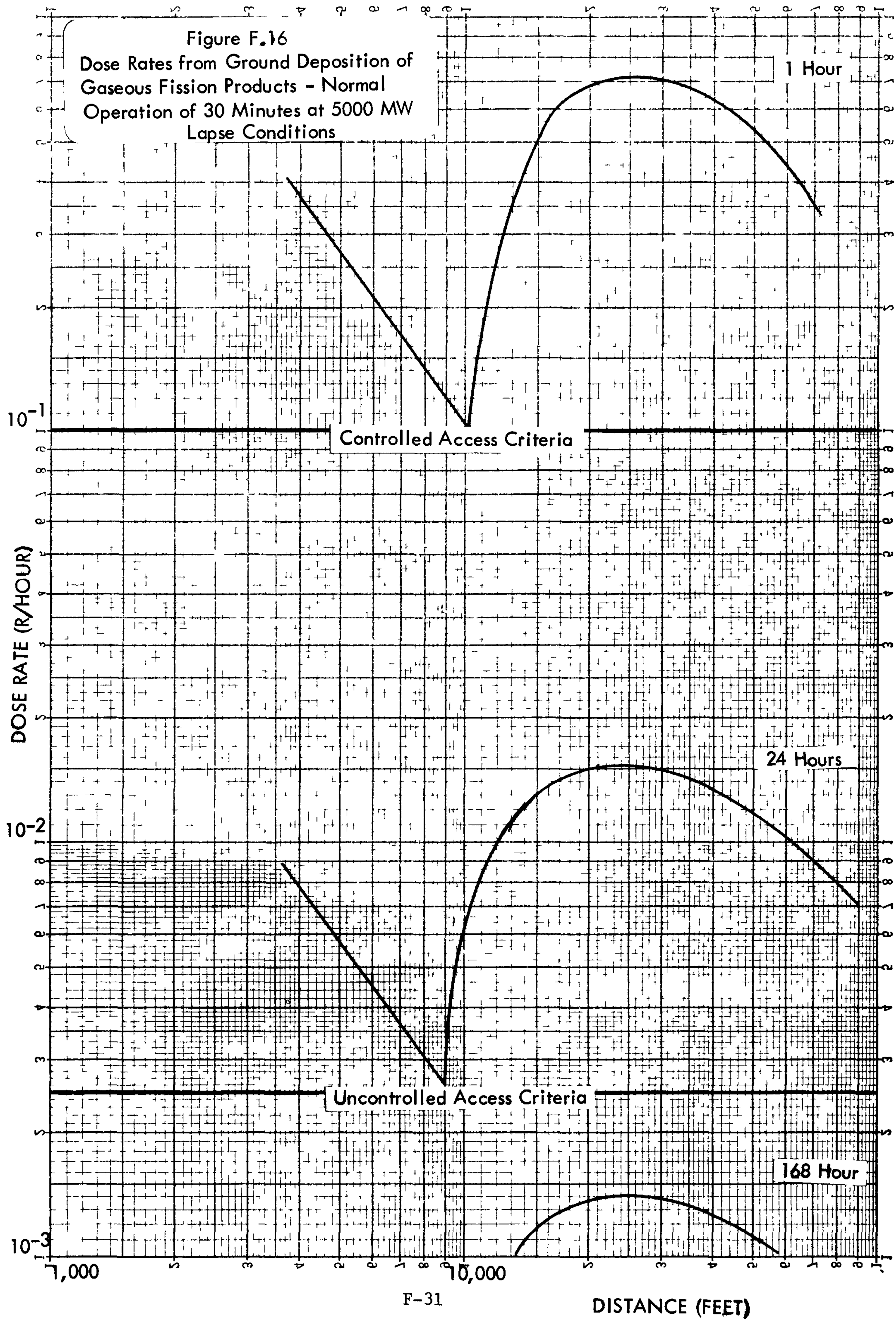


Figure F.16
Dose Rates from Ground Deposition of Gaseous Fission Products - Normal Operation of 30 Minutes at 5000 MW
Lapse Conditions
1 Hour
Controlled Access Criteria
24 Hours
Uncontrolled Access Criteria
168 Hour
DOSE RATE (R/HOUR)
10^{-1}
10^{-2}
10^{-3}
1,000
10,000
DISTANCE (FEET)

Table F.1 (Particulate Ground Concentration) indicates that, as would be expected, the on-site particulate concentration from the Phoebus 1B test was much higher than off-site. The significance of beta doses from such particulate is discussed in Section F.2.2. The on-site dose, due to the larger size particulate present and the much shorter transport (decay) time would be higher than those indicated in that section. Due to the number of people in the site facilities and the high particulate ground concentration the probability that many people would interact with many particles is quite high. All contaminated areas would have to have controlled access to prevent people from tracking radioactive material to uncontaminated areas, into cars, off-site etc.

F.3.2.2 Accident Deposition Pattern

This section looks at the deposition from the viewpoint of long term contamination. There are three accidents considered: excursion (scaled-up KIWI-TNT), loss of fuel material (scaled-up Phoebus 1A), and a loss of coolant.

Excursion

Figure F.17 presents the measured activity on resin coated trays following the KIWI-TNT event. Scaled to the inventory of a 30 minute 5000 MW[1] run, the $10^2 \mu$ curies/m^2 isoconcentration line represents ~300 R/hour at 1 hour post-event. Comparison of Figure F.17 and Table 3.1 indicates that if the wind was blowing toward the A&E Building the dose rate from fallout would be between 30 and 300 R/hour 1 hour post-event.

The highest dose rate would be at a distance corresponding to Test Cell "C", where the rate would be 5 R/hour 24 hours post-event. Considering radioactive decay only (no atmospheric weathering, fallout

(1) Based on the results in LA-3519-MS, "KIWI Transient Nuclear Test Dose Rate Survey", August 1966.

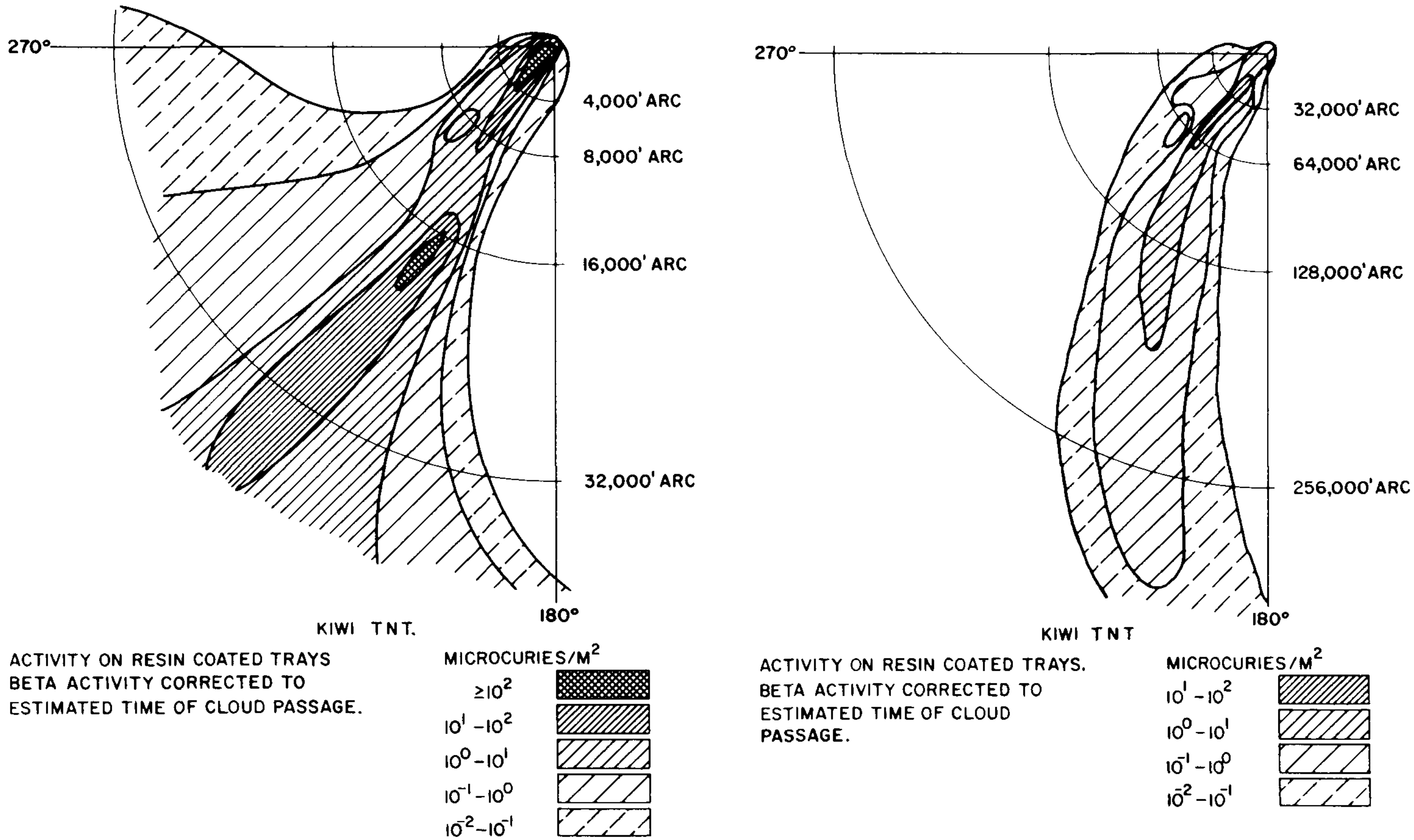


Figure F.17 KIWI-TNT Activity on Resin Coated Trays

(LA-3395-MS, "Radiation Measurements of the Effluent from the KIWI TNT Experiment", Figure 18) covered by blowing sand, etc.) it would take about 13,000 hours (540 days) for the dose rate to fall below 2-1/2 R/hour at this point. It would take 650 hours (27 days) to meet the normal controlled access criteria of 100 mR/hour.

Loss of Fuel Material

Loss of fuel material (major pieces, not the fine particulate caused by corrosion) has occurred on several prior project ROVER reactor experiments, most recently during the partial loss of coolant during Phoebus 1A.

Figure F.3 reveals the radiation pattern on resin coated trays after the event(1). Large pieces would fall close to the reactor. Such pieces may or may not be carried through the duct and into the atmosphere. The hot spot, though, on the 8000-foot arc (310°) is from airborne material. Such material would very likely be carried through the duct in the hydrogen stream. An analysis has been made of this radiation pattern scaled to the 5000 MW 30 minute operation reactor inventory(2). This appears as Figure F.18. Comparison with Table 3.1 indicates that the NRDS facility having the highest dose rate if the wind was blowing toward it is ETS-1 (9000 feet). The dose rate 24 hours post-event would be 1.5 R/hour. It would take about 225 hours (10 days) for the dose rate to meet the controlled access criteria of 100 mR/hour.

Loss of Coolant

The model used for the loss of coolant source term and down-wind dispersion model was discussed in Appendix E. The following results are based on a ground level release of material, a wind speed of 12 miles/hour, lapse conditions, a deposition velocity of 1 cm/second, and a

(1) LA-3396-MS, "Radiation Measurements on the Effluent of the Phoebus 1A-321 Reactor", June 1966.
(2) RN-TM-0415, "Interim Nuclear Analyses in Support of E/STS 2-3 Site Layout Activities", December 1966.

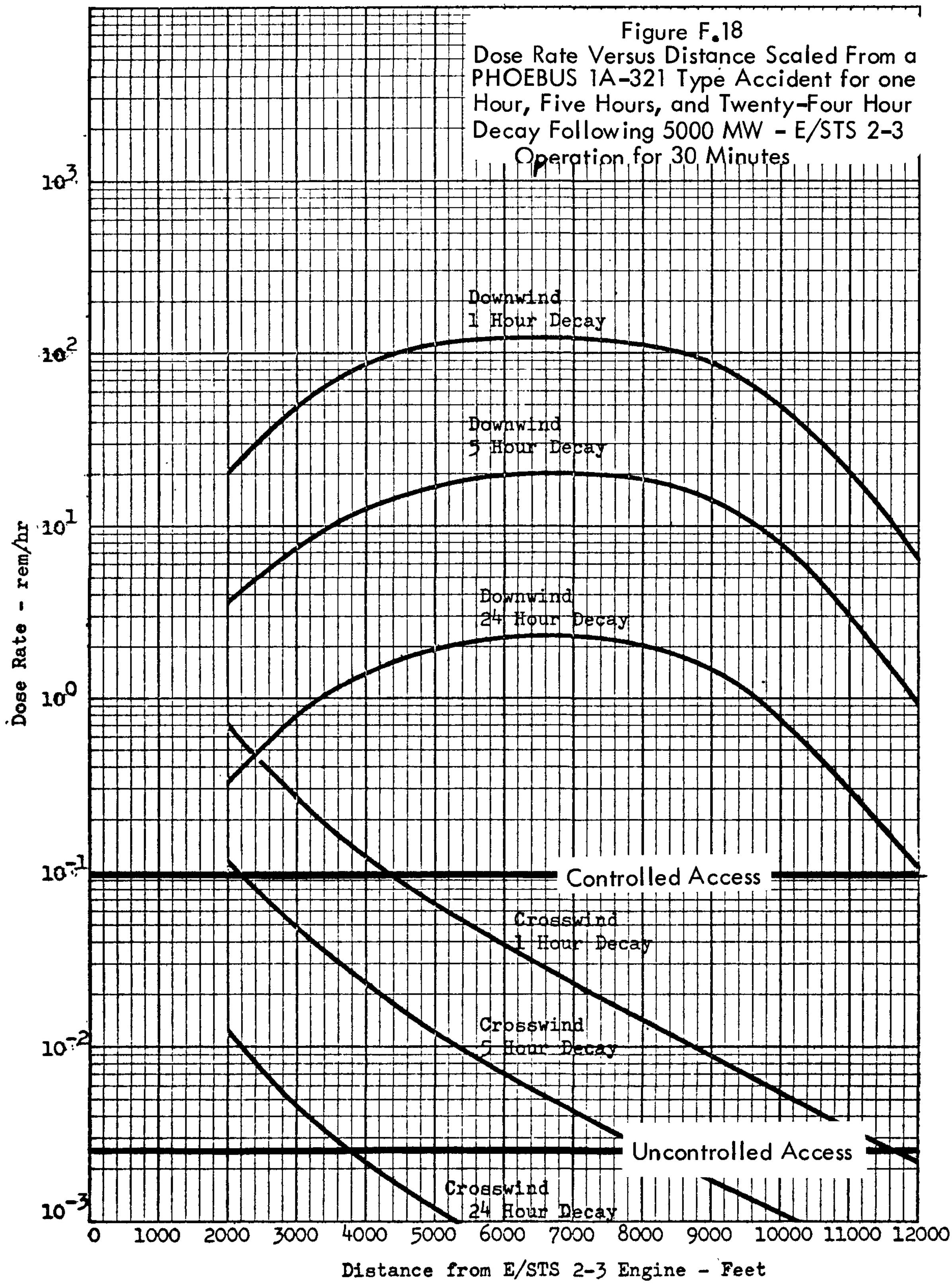


Figure F.18
Dose Rate Versus Distance Scaled From a PHOEBUS 1A-321 Type Accident for one Hour, Five Hours, and Twenty-Four Hour Decay Following 5000 MW - E/STS 2-3 Operation for 30 Minutes
Downwind
1 Hour Decay
Downwind
5 Hour Decay
Downwind
24 Hour Decay
Controlled Access
Crosswind
1 Hour Decay
Crosswind
5 Hour Decay
Uncontrolled Access
Crosswind
24 Hour Decay
10^3
10^2
10^1
10^0
10^{-1}
10^{-2}
10^{-3}
Dose Rate - rem/hr
0
1000
2000
3000
4000
5000
6000
7000
8000
9000
10000
11000
12000
Distance from E/STS 2-3 Engine - Feet

release fraction of 50% of the total inventory. As shown on Figure F.19, the highest dose rate is at ETS-1, about 25 R/hour at 1 hour post-event and 0.5 R/hour at 24 hours post-event. It would take about 100 hours for the dose rate to be less than 100 mR/hour. This dose rate is roughly inversely proportional to the wind speed and directly proportional to the deposition velocity.

F.3.2.3 Control Center

The control center is designed to provide a safe location from which to control operations at the test stand. As such it must fulfill three functions:

1. Provide adequate shielding during normal operation to maintain an interior dose rate of less than 2-1/2 mR/hour.

2. Provide adequate shielding and life support capability in the event of a major nuclear accident.

3. Provide a means of egress following a major nuclear accident if the life support capability were compromised.

Shielding to satisfy requirements 1 and 2 is provided in the form of an equivalent 4-1/2 feet of concrete placed over the control center. Life support capability is discussed in Section 5.1.

Kaiser Engineers have considered[(1)] the dose rates in the control center for normal operation of a 10,000 MW engine and concluded that 1-1/2 feet of concrete is adequate to meet the unrestricted access criteria.

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(1) Kaiser Report 67-37-R, "Nuclear Shielding, Radiation, Thermal and Safety Analyses", October 1967.

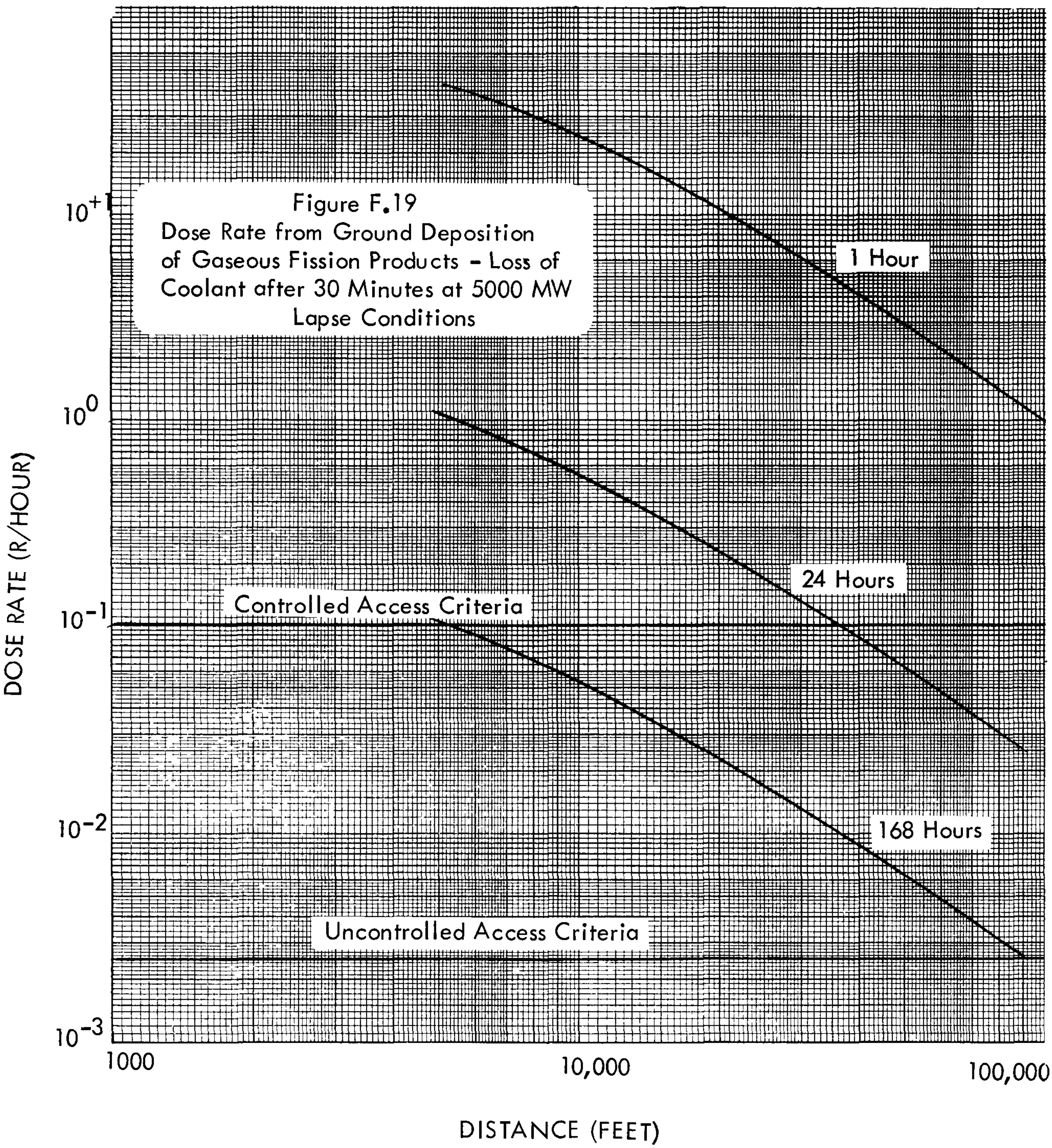
Figure F.19
Dose Rate from Ground Deposition
of Gaseous Fission Products - Loss of
Coolant after 30 Minutes at 5000 MW
Lapse Conditions
1 Hour
24 Hours
168 Hours
Controlled Access Criteria
Uncontrolled Access Criteria
DOSE RATE (R/HOUR)
10+1
10^0
10-1
10-2
10-3
1000
10,000
100,000
DISTANCE (FEET)

The adequacy of the present shield design to reduce the doses within the control center to acceptable levels in the event of an accident which resulted in surface contamination was determined by comparison with a postulated accident. This postulated accident is the inventory from a 5000-MW 30-minute run deposited uniformly and instantaneously over the control center roof shield (~20,000 feet^2). The resulting 24 hour (from time after shutdown to 24 hours plus this time) integrated dose is given in Figure F.20. The maximum integrated dose from this postulated accident is about 18 rem. Table F.6 presents the maximum estimated ground concentrations of fission products from an excursion or loss of coolant. Comparison with the postulated accident indicates that the shield will adequately protect the operating personnel in the control center in the event of a major reactor accident. A discussion of requirement 3 comprises the remainder of this section.

Definition of an escape mode is dependent upon the radiation field around the control center. Two sources for this field exist, direct radiation from an unshielded shutdown reactor and radiation from fall-out. The unshielded shutdown reactor is discussed in Appendix C. Examination of Figure C.7 indicates that the direct radiation dose from the reactor at the control center or the alternate test stand would be very high if egress was required immediately upon loss of shield and engine shutdown. Since the direct radiation field is uniform on any given arc around the reactor, the only way to achieve a lower dose rate is to provide a greater distance between the exit point and the reactor. Making two basic assumptions, that it would take 0.1 hour to get to the exit from the control center and that an individual would be exposed to the direct radiation field at this exit for a time period equivalent (dose wise) to a maximum of 15 minutes, an exit point located approximately 1500 feet from the unshielded reactor would result in an integrated dose of less than 12 rems. The 12-rem exposure is selected on the basis of the proposed SNPO accident criteria given in Table 2.1. At 1500 feet the dose rate at 0.1 hours is about 100 R/hour.

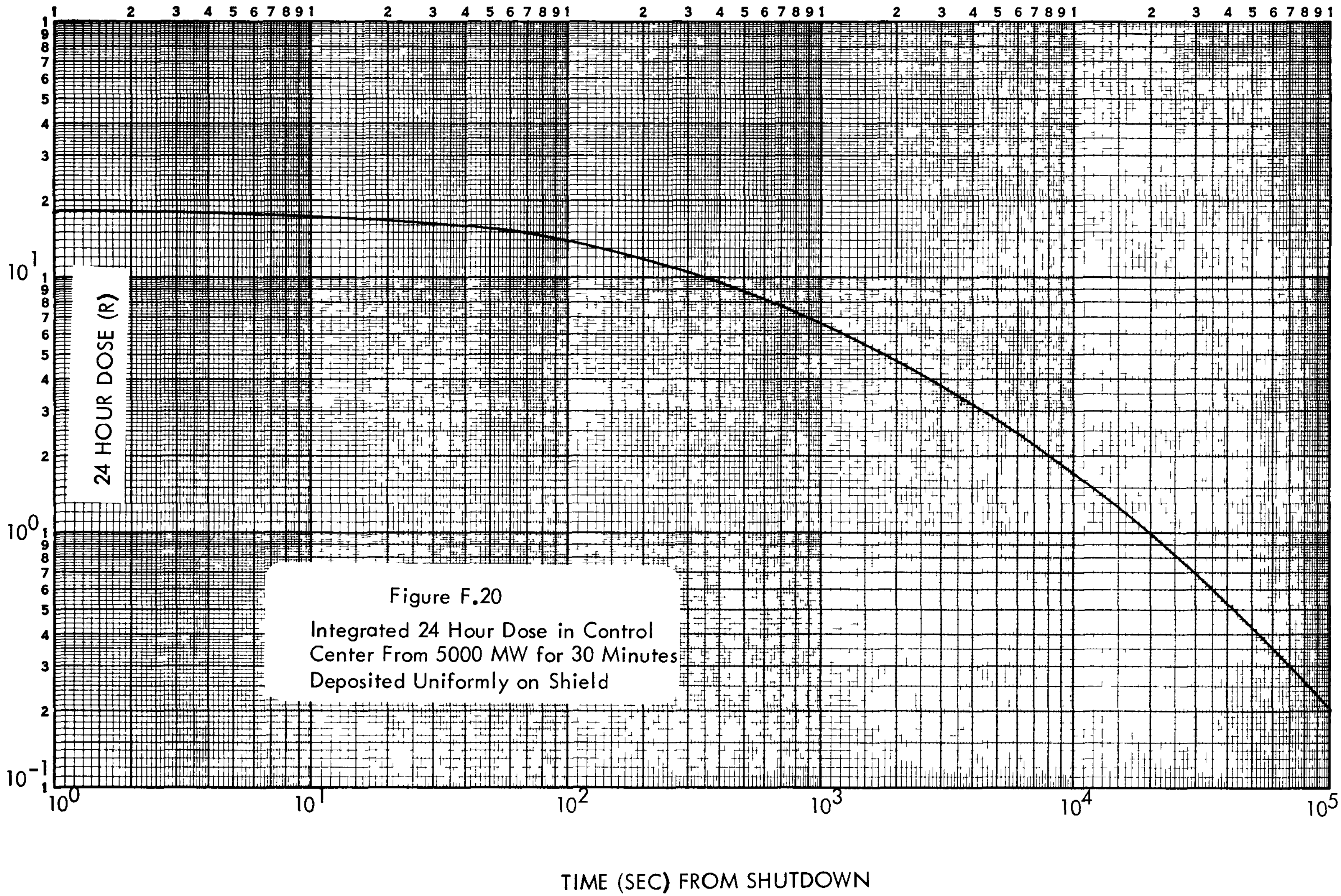

Figure F.20

Integrated 24 Hour Dose in Control Center From 5000 MW for 30 Minutes Deposited Uniformly on Shield

TABLE F.6

GROUND CONCENTRATIONS AND SURFACE DOSE RATES
24 HOURS AFTER OPERATION

	Ground Concentration MEV/m^2 sec	Surface Dose Rate R/Hour
Loss of Coolant	5.0×10^{11}	297
Excursion (Downwind)	5.8×10^{11}	350
(Crosswind)	8.3×10^{9}	5
Postulated Accident	2.8×10^{14}	1.7×10^{5}

Test stand to control center - 750 feet.

Thus the direct radiation field defines the mode of escape as a tunnel with an exit 1500 feet from the reactor. An alternate escape route, the other test stand, exits about 1200 feet from the reactor. Since due to fallout contamination it is desirable to separate the exits as much as possible, and since the location of the test stand is fixed, the orientation of the emergency tunnel should be due south. This provides two exits, about 70 degrees apart when viewed from either test stand, with the lowest direct radiation field at an exit of about 100 R/hour 0.1 hours after shutdown. Evacuation vehicles provided in shielded locations at the exit points are desirable from the standpoint of reducing the time spent in the radiation field.

Such an escape system must also provide safe egress with respect to high radiation fields due to radioactive fallout. Two types of accidents as sources of ground contamination are considered; a scaled-up KIWI-TNT excursion (although such an excursion is not considered likely in the NERVA reactor) and a loss of coolant, both for a 5000-Mw 30-min fission product inventory.

The scaled-up KIWI-TNT results appear in terms of isodose contours for 0.1, 1, and 24 hours post-event in Figures F.21 - F.23. Indicated on the figures in the least optimum configuration are the test stands, control center, and postulated escape tunnel. As can be seen the ground deposition pattern can be considered as an airborne ground release pattern superimposed on a circular pattern. This circular pattern is apparently due to ejected material too heavy to be appreciably affected by the wind and has been observed following KIWI tests which resulted in expulsion of portions of the core due to axial support failures. The airborne cloud presents a simple, well-defined pattern due to the short time span of the release and the well organized structure of the winds aloft.

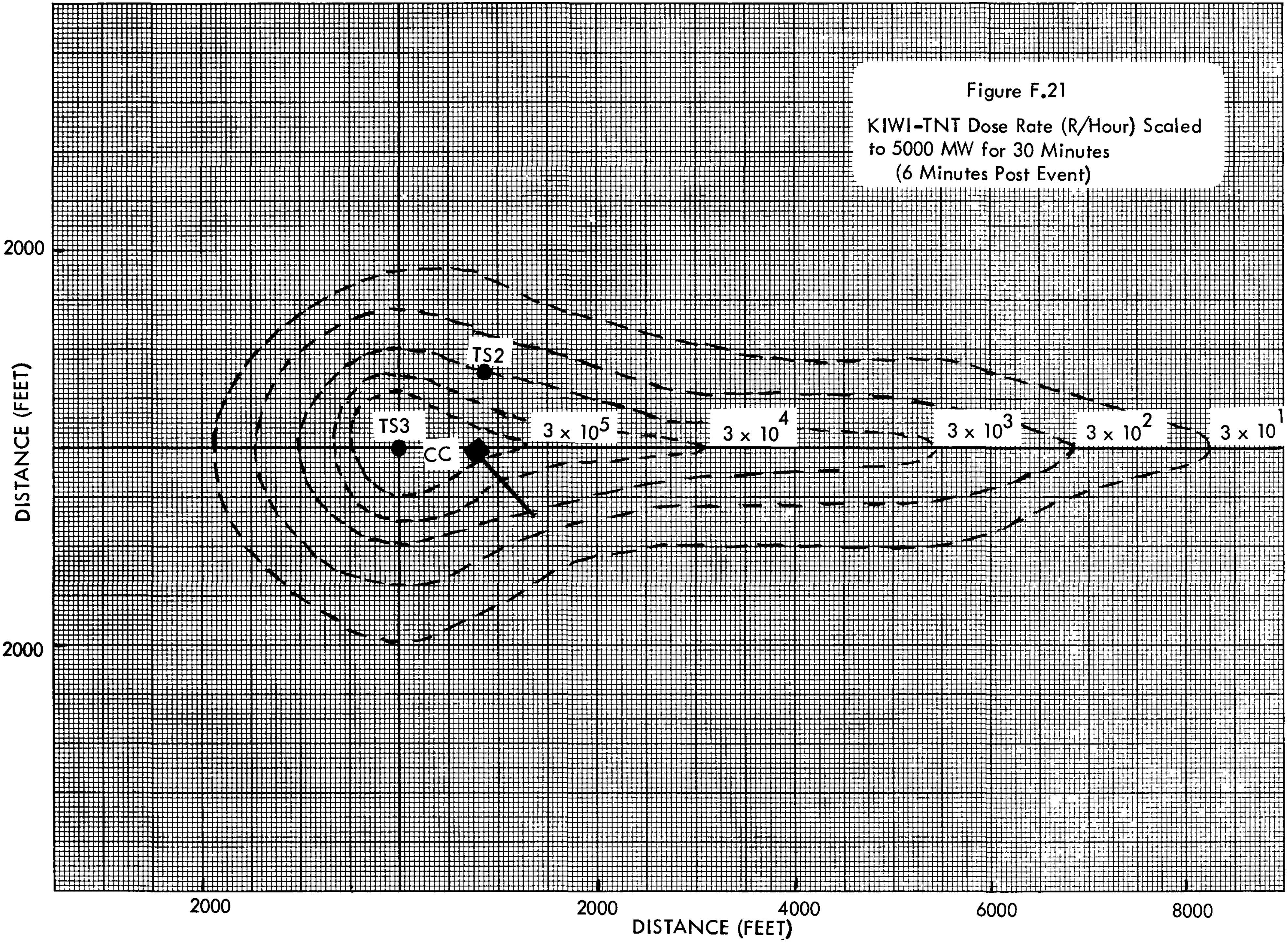

Figure F.21

KIWI-TNT Dose Rate (R/Hour) Scaled to 5000 MW for 30 Minutes (6 Minutes Post Event)

[redacted]

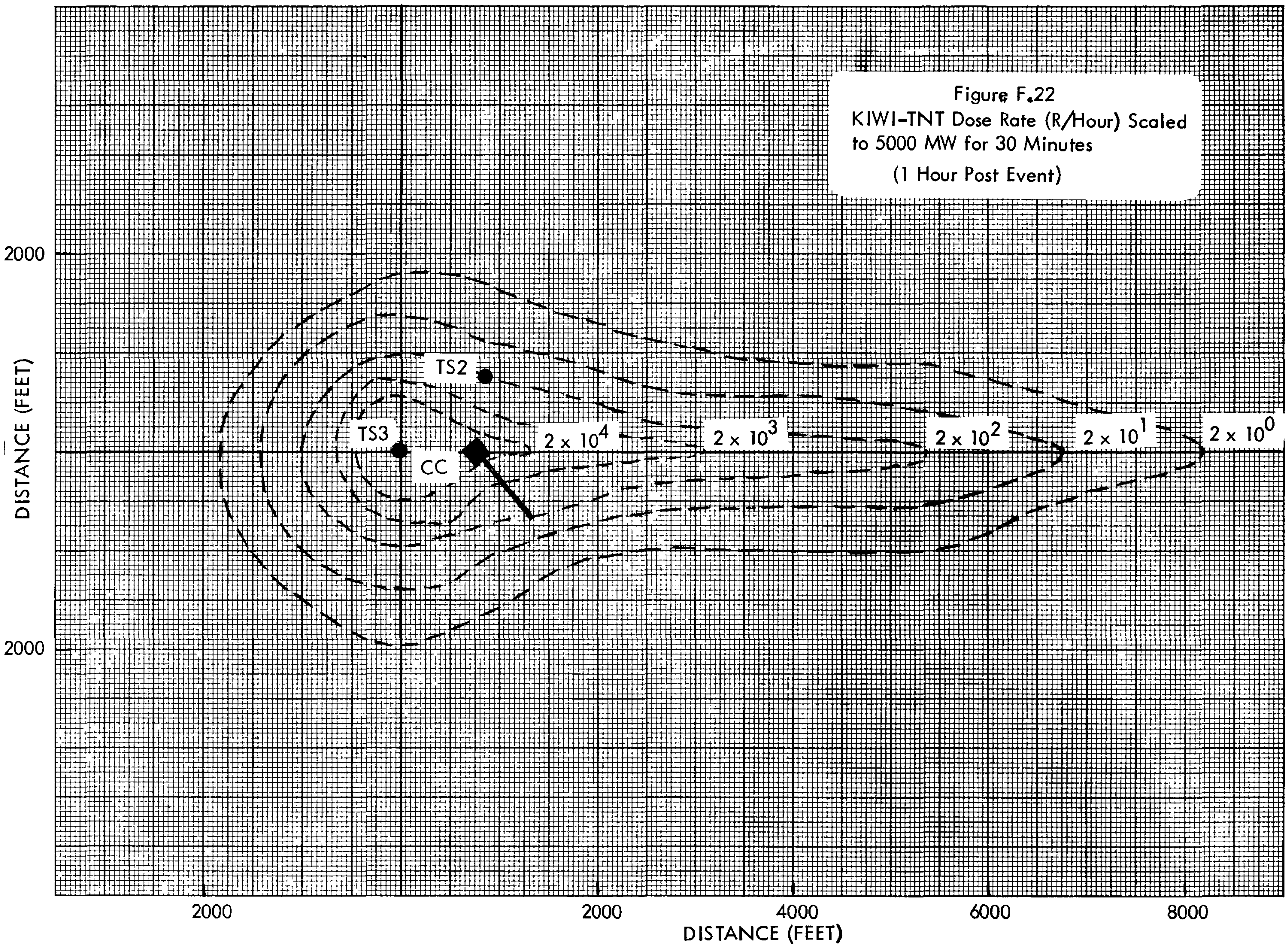

Figure F.22
KIWI-TNT Dose Rate (R/Hour) Scaled to 5000 MW for 30 Minutes
(1 Hour Post Event)

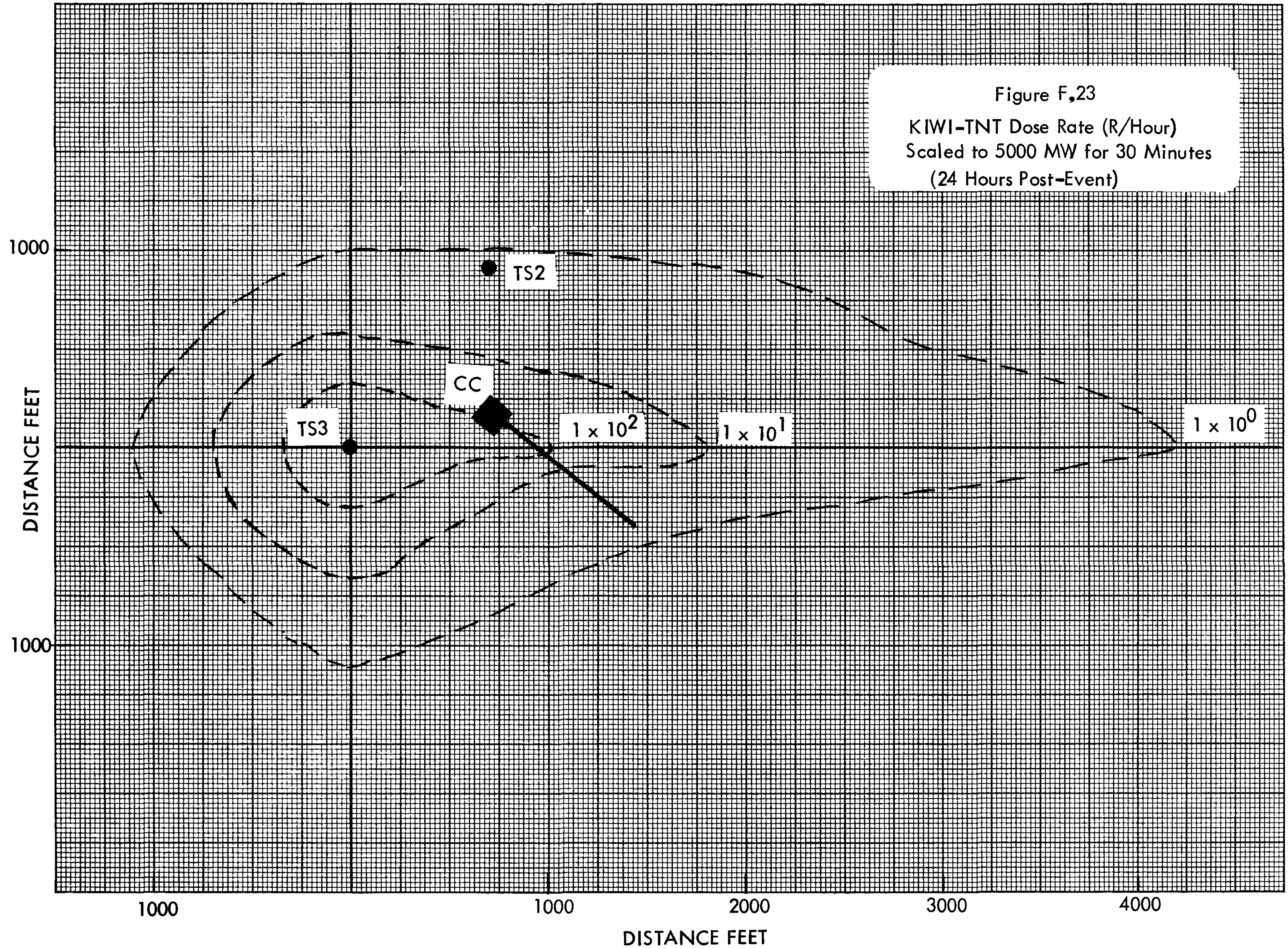

Figure F.23
KIWI-TNT Dose Rate (R/Hour)
Scaled to 5000 MW for 30 Minutes
(24 Hours Post-Event)

The dose rate at the tunnel or alternate test stand exit under this least optimum arrangement (with the wind blowing directly over the control center) would be on the order of 3000 R/hour at 0.1 hours (Figure F.21). This would be unacceptable under the 100 rem/hour or 12 rem criteria defined for the direct radiation case. The dose rate would be acceptable if the wind blew in any quadrant other than the one centered on the control center. Examination of Figure F.22 indicates that even under this least optimum arrangement egress is permissible 1 hour after the event. Figure F.23 indicates that egress directly from the control center would be permissible 24 hours post-event.

The loss of coolant source term is estimated to be 50% of the gross inventory and 100% of the iodines. The ground deposition pattern was estimated using equations 3 and 4 of Section F.1 with $h = 0$. Under the conditions of complete loss of coolant the majority of released fission products would be in gaseous form. In addition, there would be no coolant flow to carry released particulate up into the atmosphere. It is considered that a loss of coolant would occur during or upon the conclusion of a scheduled run, and therefore under favorable weather conditions.

Measured deposition velocities for gaseous fission products are in the range of about 0.1 cm/second to 2 cm/second in the literature. LASL has calculated deposition velocities of fission products released during NERVA testing from the activity on resin coated trays compared to the activity measured by air samplers at the same location. These calculated velocities vary widely, but a median value is about 1 cm/second. Since this value is consistent with those observed elsewhere, 1 cm/second is used for this model. In addition, a wind velocity of 5.36 m/second (12 miles/hour) under lapse conditions is assumed. The ground deposition is directly proportional to the deposition velocity and inversely proportional to the wind velocity.

Comparison of NRX deposition patterns with Sutton predictions indicates that the actual pattern is usually braoder, usually by a factor of two and on occasion up to a factor of four. Figure E.1 indicates that the major portion of the activity is released in ~300 seconds. Since this release time is shorter than a typical NRX run, the maximum expected spread over the Sutton prediction is considered to be a factor of two. The spreading was accomplished by doubling the angle of spread between the isodose curves and the release point and then reducing the isodose value along the spread curves by a factor of two. The results are given in Figure F.24. It should be noted that the dose rate at the emergency exit locations under adverse wind conditions is higher than would be predicted by the simple Sutton model.

The dose rate at either exit at 0.1 hour under the least optimum arrangement is about 750 R/hour (Figure F.24). Changing the wind direction by only about 20 degrees would put an exit in a field of less than 100 R/hour. The dose rate at 1 hour (~75 rem/hour) is acceptable for escape.

The above results are, however, for lapse conditions. Operations under less ideal diffusion conditions could result in higher dose rates at both exits. While Sutton-predicted isoconcentrations for three stability classifications (given in Figure F.25) indicate the deposition pattern is narrower, the actual variable wind conditions that may occur at NRDS under more stable atmospheric conditions may lead to a much broader deposition pattern.

Summarizing, egress following an excursion or a loss of coolant is permissible under the least optimum weather conditions with constant wind direction (relatively low wind variability under lapse or inversion stability classifications) 1 hour post-event. Egress would be permitted 0.1 hour post-event under favorable weather conditions, defined as the wind not blowing in the quadrant centered on the control center.

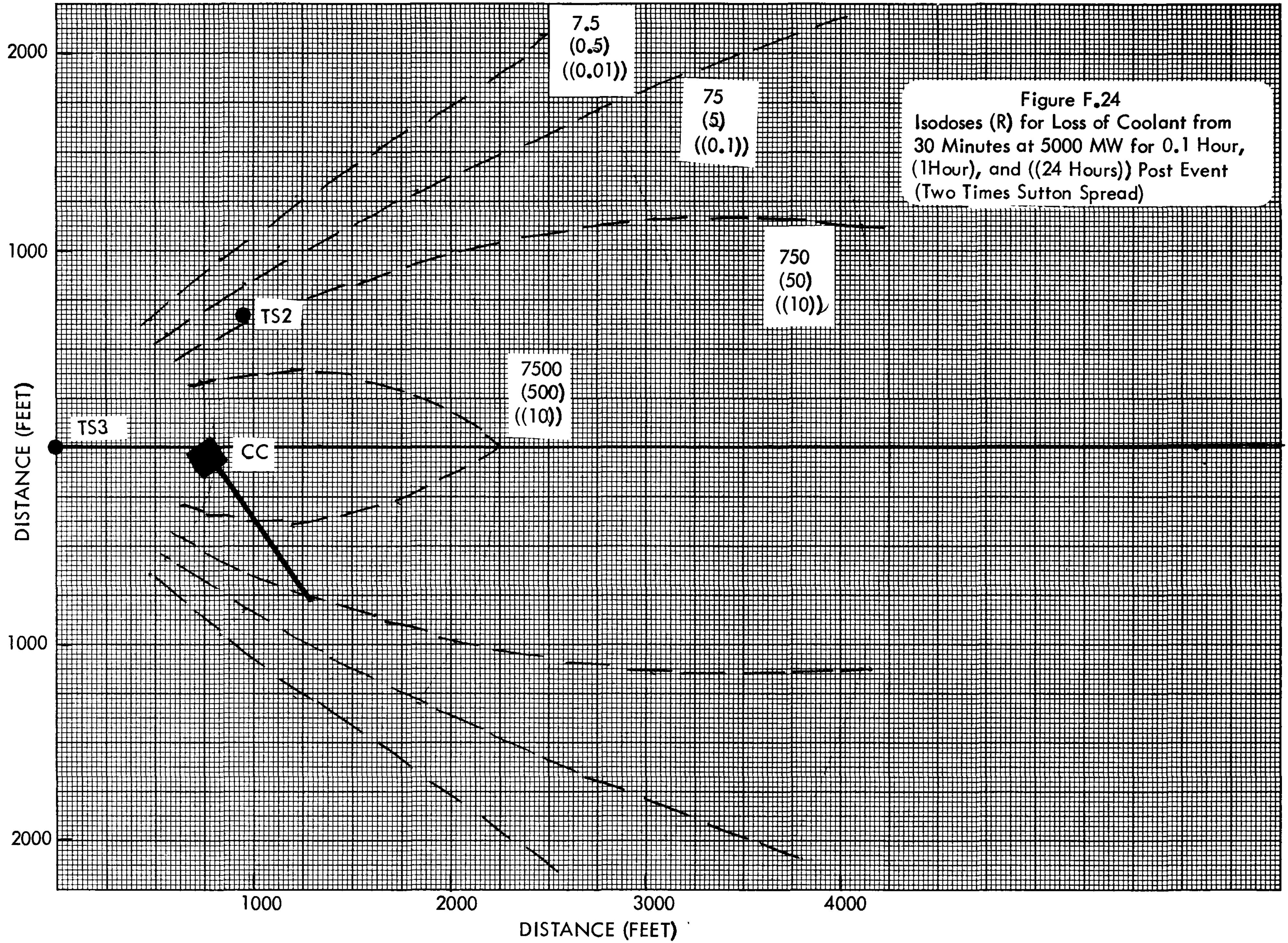

Figure F.24
Isodoses (R) for Loss of Coolant from 30 Minutes at 5000 MW for 0.1 Hour, (1Hour), and ((24 Hours)) Post Event (Two Times Sutton Spread)

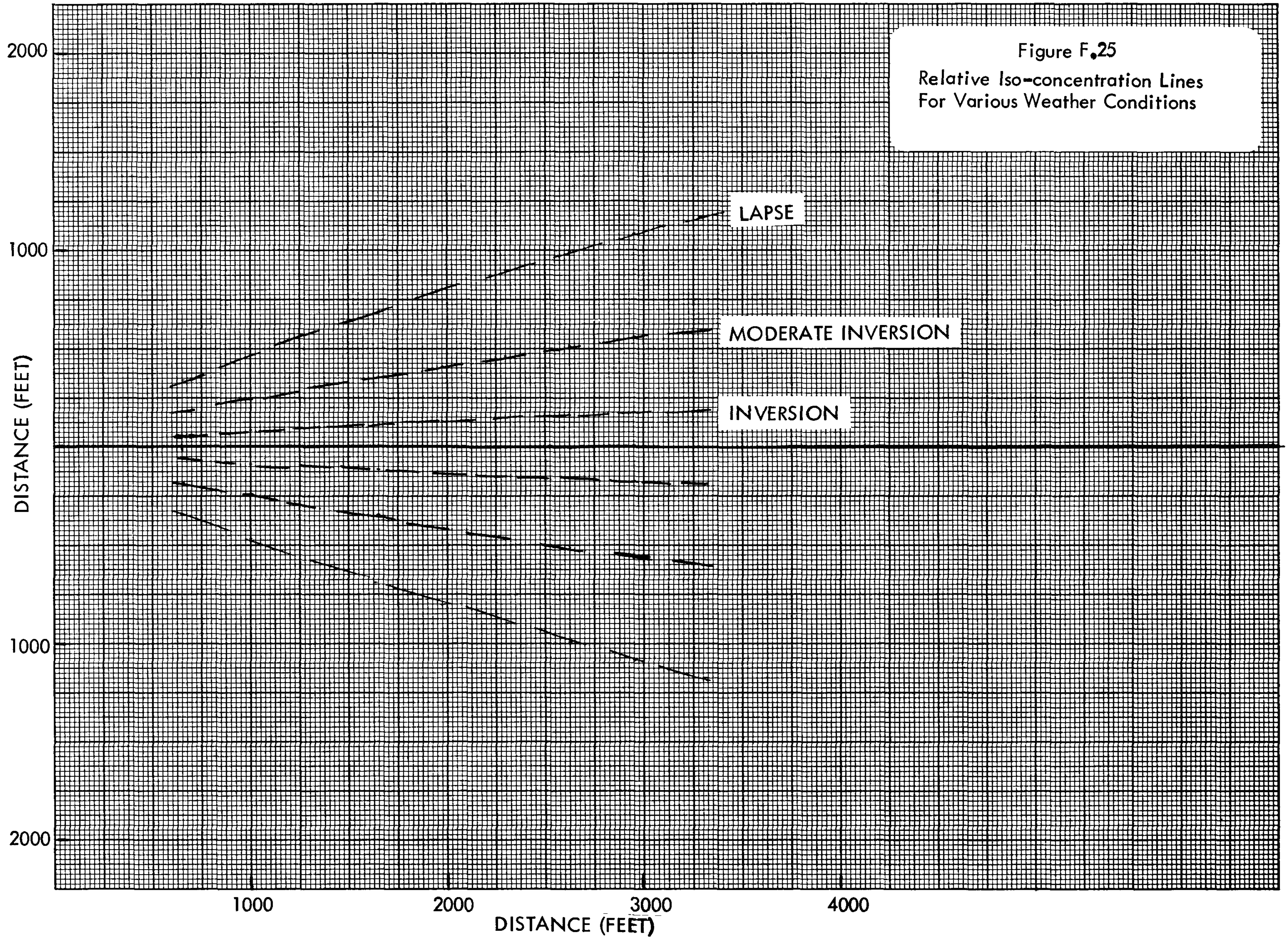

Figure F.25
Relative Iso-concentration Lines
For Various Weather Conditions

One other factor may compromise egress in the event of a reactor accident. A reactor which has suffered a loss of coolant may release fission products for some time (Figure E.1). After personnel have left the shielded areas and are proceeding to a safe location the wind may shift and carry the effluent cloud over them. Figures F.26 and F.27 illustrate this effect for 0.3 miles (~1600 feet), 1 mile, and 3 miles from the reactor. This dose rate is primarily dependent only on the fission product release rate and is roughly independent of the wind velocity. Therefore, in general, the curves shown in the figures are applicable for winds of various velocity. However, it should be noted that the cloud transport time varies inversely with wind velocity, and this will determine the time after loss of coolant at which the maximum dose occurs. Egress has been postulated to begin no sooner than 0.1 hour (360 seconds). However, the figures do indicate that under variable wind conditions facility egress immediately following an accident could result in excessive radiation exposures to personnel.

It is concluded from the results of this section that a major nuclear excursion or loss of coolant presents a potentially greater hazard than direct radiation from the reactor in the event of loss of shield. This potentially greater hazard, however, is associated with the least optimum wind directions under lapse conditions and/or a wind shift such that the effluent is carried over personnel during egress. It is also concluded that two control center exits, one at the alternate test stand and the other a tunnel extending south from the control center with an exit 1500 feet from both test stands and with roads from the exits extending away from the facility will permit egress with an integrated dose of less than 12 rem from direct radiation or ground deposition. Vehicles provided in shielded locations at the exits are required for safe movement of personnel away from the exit locations.

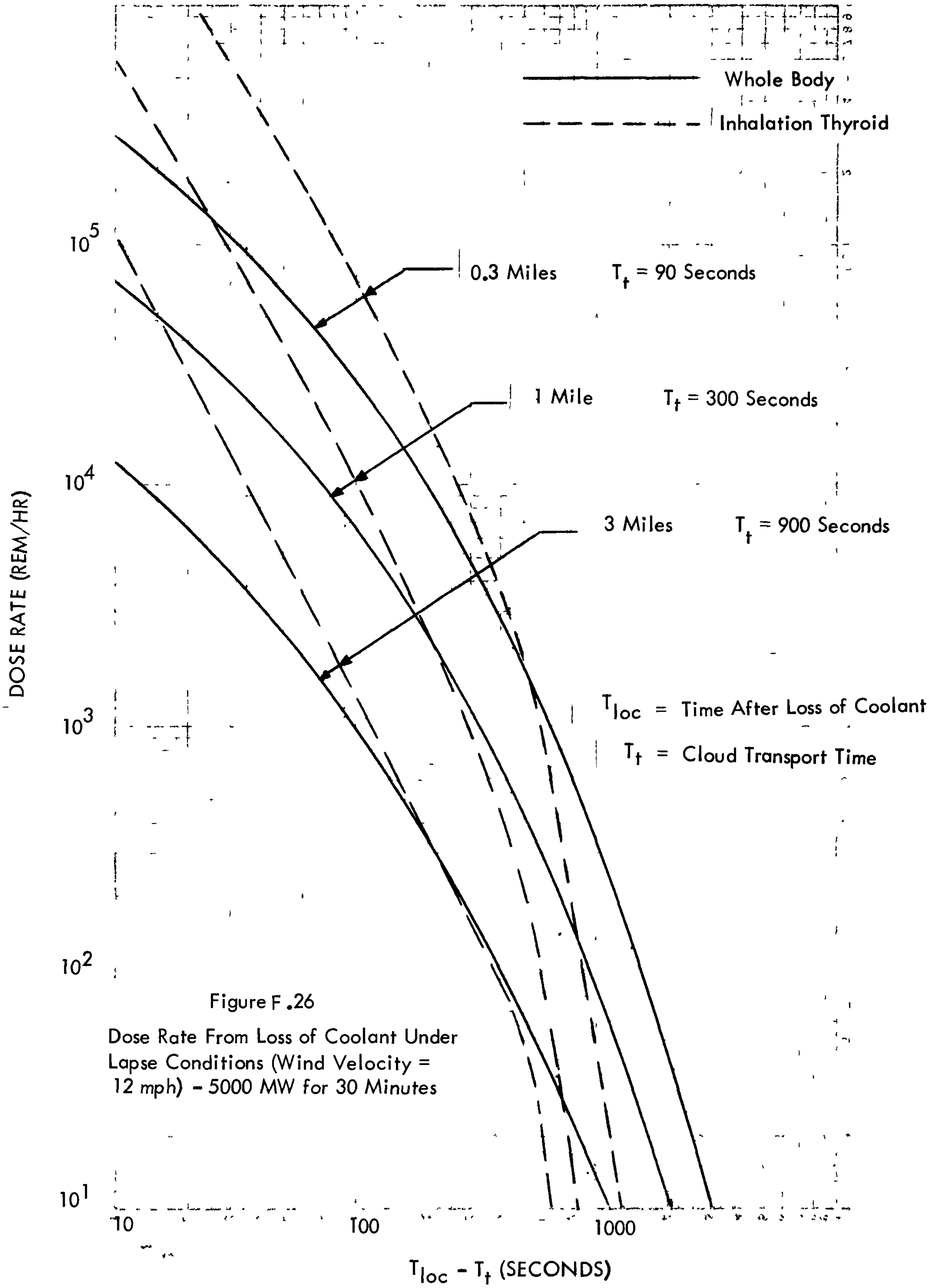

Figure F.26

Dose Rate From Loss of Coolant Under Lapse Conditions (Wind Velocity = 12 mph) - 5000 MW for 30 Minutes

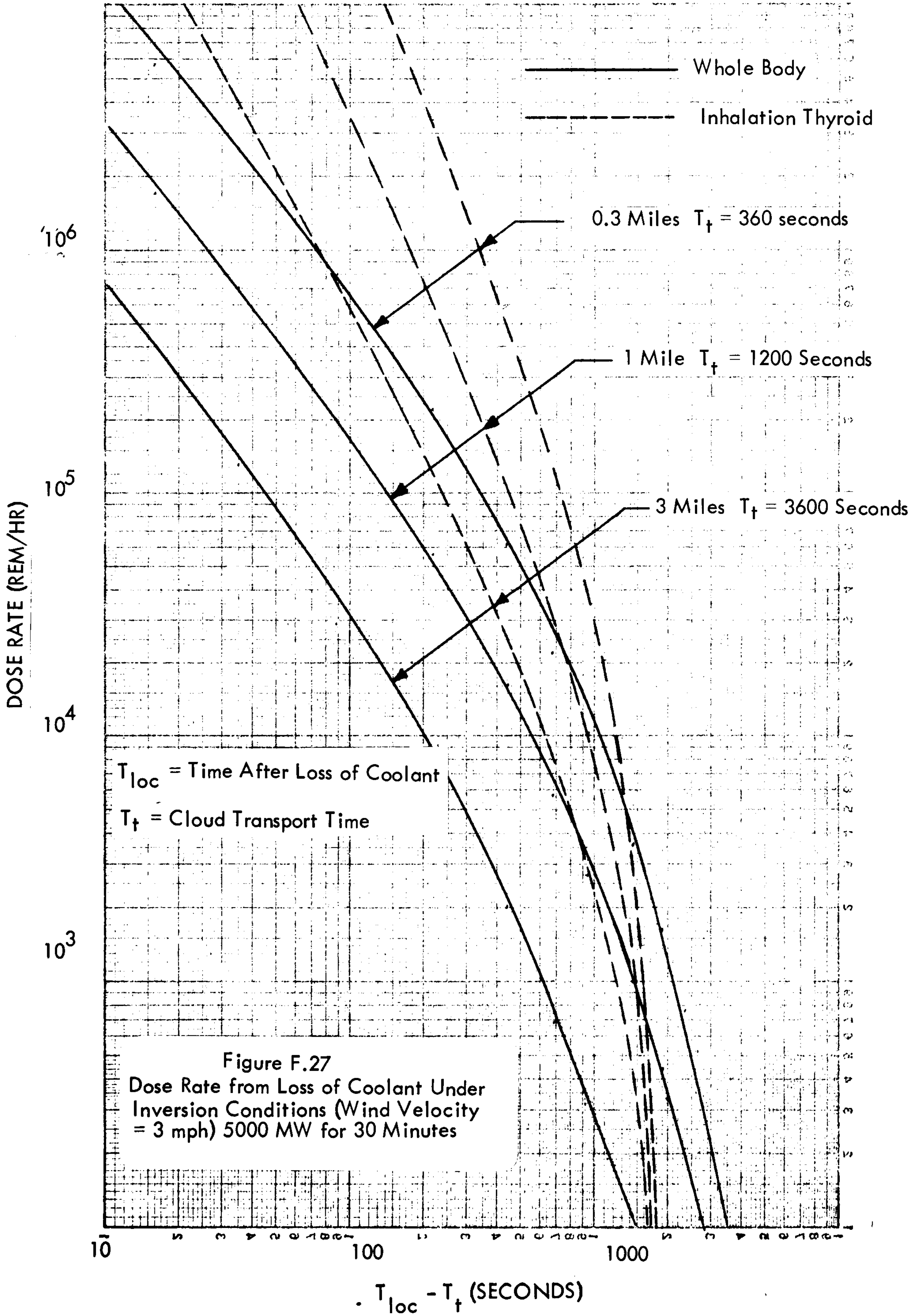

Figure F.27
Dose Rate from Loss of Coolant Under Inversion Conditions (Wind Velocity = 3 mph) 5000 MW for 30 Minutes

APPENDIX G

SAFETY REQUIREMENTS FOR PROTECTION OF PERSONNEL

AT THE E/STS 2-3 CONTROL POINT BUILDING

DURING TEST OPERATIONS*

* AGC Memo 7030:M1085, J. B. Philipson to A. Schaff, dated Nov. 3, 1966,
Subj: E/STS 2-3 Control Point Building Personnel Protection

G.1 INTRODUCTION

Engine testing at the E/STS 2-3 facility will require that a large number of personnel be assembled in a control center for varying periods of time to conduct the test operations. To achieve the test objectives in a successful manner, it is necessary that every consideration be given to providing the operating personnel with a safe working environment.

In the discussion which follows, an attempt has been made to set forth some of the minimum basic safety requirements for the protection of the personnel at the control point building during normal operations and under accident conditions. These requirements have as their only objective to assure that under any circumstances personnel cannot be trapped in the control point building without a means for survival and rescue.

It will be noted that there is no single requirement which will satisfy the basic objective but rather, the solution consists of a combination of requirements interrelated to the extent that trade-off studies must be made to determine the most feasible approach in satisfying the basic objective.

Once the basic plan of approach has been established, the resultant conceptual design must again be evaluated to determine if the basic criteria have been met and the ultimate objective achieved.

G.2 BUILDING CONSTRUCTION

G.2.1 Normal Building Considerations

As a minimum, the normal design features of an office building with respect to factors such as heating, ventilation, humidity control, and adequate lighting are essential considerations for the control center. Normal building design also includes emergency features such as a minimum number of emergency exits, emergency lighting, and fire protection equipment based on the size of the building and the type and location of equipment. These criteria are set forth in standard building and electrical codes.

G.2.2 Special Building Considerations

a. Radiation Shielding

Shielding against the nuclear radiation generated during power operation of the engine must be an inherent part of the building design. The amount of shielding required to protect personnel is a function of the source strength, and distance from the source.

Since it is desirable from an operational standpoint to locate the control center in relatively close proximity to the test stands in order to maintain short signal leads, etc., this close proximity requires special building design considerations for the protection of personnel with respect to radiation.

From an economical standpoint satisfactory shielding may be provided by an underground structure utilizing earth as a protective shield. This is the concept which has been presented in the E/STS 2-3 facility design.

The basic requirement is that the underground structure and the earth shield reduce the radiation level such that the dose rate received by personnel inside the structure during normal operations is within the limits established for unrestricted, unlimited occupation which is normally established as 2.5 mr/hr.

b. Fire and Explosion Protection

The operation of the engine on the test stand involves the use of cryogenic fluids and pressurized gases which are a source of possible explosive mixtures and fires. The control center must therefore be designed to protect the operating personnel from accidents arising from these hazards.

The basic requirements include the following:

(1) Fire resistent materials of construction will be utilized insofar as possible.

(2) Exits will be provided in sufficient numbers to assure personnel cannot be trapped in any area of the building. The location of exits should take into consideration above-ground hazards, and sources of possible obstruction. Exit doors must be provided with jam-proof hardware and be of blastproof construction if warranted by location near an explosive source. The exits must be designed to maintain the waterproof integrity of the building to prevent water leakage and the possible flooding of all exits must be prevented. The exits must be weather stripped to reduce air leakage both into and out of the control center to a minimum. Since maintaining building integrity is important during engine operations, due to the radioactive nature of the engine gaseous effluent, a system must be employed to indicate, at a central location, the opened or closed status of all exits from the control point.

(3) Fire protection equipment must be provided in sufficient quantities and located in specific locations clearly identified and easily accessible. The equipment supplied must be capable of controlling fires from normal combustible materials and those originating from electrical equipment.

c. Flood and Earthquake

Underground structures are extremely vulnerable to flooding conditions and in view of the large quantities of liquids stored and used during engine operations, it is a requirement that measures be employed to direct liquids from all sources away from the control point to prevent flooding conditions. The building design must, in addition, include an impervious membrane to prevent water seepage from natural sources and building penetrations must be adequately sealed to maintain building integrity.

The structure design must also consider ground shocks due to earthquakes and weapons tests, the primary requirement being that personnel cannot be trapped due to collapse of the main building structure or any exits therefrom.

G.2.3 Ventilation

a. Normal Operation

The requirements for the ventilation system, in addition to providing normal temperature, humidity and dust control, must include other considerations due to the close proximity of the control center to the test stands.

(1) Air Intake Supply

During normal power operation of the engine, a quantity of radioactive effluent is discharged to the atmosphere containing both radioactive gases and particulates. Experience during NRX testing has indicated that although particulate fallout tends to follow the wind direction, there is a certain amount of random distribution in the immediate vicinity of the test stand. To prevent this radioactive effluent from entering the ventilation system air intake supply duct, the following requirements must be considered in the design of the system:

(a) Locate the air intake supply duct upwind of the prevailing wind direction and sufficiently remote from the test stands to prevent radioactive effluent takeup during normal operations.

(b) Provide air filters near the air intake duct entrance which will remove the radioactive particulates from the air supply.

(c) Provide air sampling detector systems downstream of the air filters for the detection of radioactivity and combustible gases in the air supply. The air sampling detector systems will provide for readout and alarm in the central control room and must be of fail-safe design employing redundant power sources and equipment as may be required.

b. Emergency Operation

(1) Air Intake Supply

If, during normal power operation of the engine, a change in wind direction occurs, the radioactive effluent could be directed toward the control point and the ventilation system air intake supply duct. This would create a hazardous condition due to an increased possibility for radioactive effluent takeup in the air supply. The condition could be further aggravated if; (1) an accident occurred at the test stand which increased the amount of radioactive effluent, and (2) further unfavorable weather conditions prevailed at the time of the accident.

To prevent the personnel located in the control center from being exposed to radiation doses from radioactive effluent in the atmosphere, the following additional requirement must be met by the ventilation system design: provide a means for sealing off the ventilation air intake supply duct against the atmosphere. The system used must be capable of fail-safe operation from the central control room employing redundant power sources and equipment as may be required to assure operation under both normal and accident conditions.

(2) Emergency Air Supply

Under the accident conditions postulated in the foregoing discussion, it may be necessary for personnel to remain in the control point building with the air intake supply duct closed for a considerable length of time prior to evacuation. As indicated under the building construction requirements, the building integrity must be maintained by the scaling of all building penetrations and air leakage kept to a minimum by the weather stripping of all doors. Assuming these conditions have been met, the air entrapped in the control point building will furnish personnel air requirements for a period of time dependent on the number of people and volume of air entrapped. However, since it is possible that other emergency conditions might occur which could cause rapid depletion or contamination (non-radioactive) of the entrapped air supply, complete dependence cannot be placed on this source of air. To assure an adequate air supply under all conceivable accident conditions, the facility design must consider the following:

(a) A control center layout in which the area(s) least susceptible to fire or other air contaminant accidents can be isolated from the other areas of the building. This would also require that the ventilation system to the isolated area(s) could be blocked to prevent contamination from the other areas. The isolated area could then be supplied with an uncontaminated backup air supply to accommodate the control center personnel until that time when the control center can be safely evacuated. It would be necessary that personnel from all areas of the control center could reach the isolated area(s) and that the isolated area(s) provide unrestricted access to an emergency evacuation exit.

(b) A method for control of the temperature in the control point or isolated area(s) to a reasonable value during the emergency conditions.

G.2.4 Emergency Escape Routes

From the preceding discussion, it was required that the control point building be capable of accommodating personnel under accident conditions for that period of time until a safe evacuation can be accomplished. Therefore, the design of the control center and its ventilation system are interdependent on the design of the emergency escape route from the standpoint of timely evacuation.

The close proximity of the control point to the test stands may prevent egress through the normal control point exits due to possible personnel exposure to fire, explosion, or nuclear radiation hazards resulting from accident conditions. Therefore an emergency escape route is required which protects personnel from such hazards during evacuation from the facility. A tunnel leading from the underground control point building could provide the necessary protection from the potential accidents cited.

The factors to be considered in the design of an emergency exit are enumerated in the following discussion.

a. Personnel Access and Egress

It is possible that accident conditions could exist within the control point building at the time evacuation is desired which would prevent the use of a particular access to the emergency exit. Alternate entrances to the emergency tunnel from the control point building must be planned on the basis of the possible accident areas such that personnel are assured a path of egress from the building.

The tunnel itself must be located to avoid accident areas that could possibly cause the tunnel to be blocked. From this standpoint, it is necessary to consider the need for alternate exits from the tunnel and their location with respect to the test stands, cryogenic areas and high pressure gas storage areas. For example, the exit locations must take into consideration thermal radiation from fire hazards and fragmentation missiles from explosion hazards. Although accidents of this nature are of relatively short duration they are important considerations since they could occur just at the time personnel are leaving the emergency exit. In addition to the fire and explosion hazards, the tunnel and its exits must be protected from flooding by both natural and accident conditions which requires analysis of the site water storage facilities and drainage characteristics. The results of such analysis will dictate the protective measures required to assure that the emergency exits cannot be flooded.

b. Nuclear Radiation Exposure

Under accident conditions, exposure of personnel to high nuclear radiation fields is a major concern and has a direct bearing on the emergency exit locations as well as the design of the control point ventilation system.

The sources of the high radiation field against which personnel must be protected during their evacuation from the facility are: (1) direct radiation originating from an unshielded "hot" reactor on the test stand; (2) fuel particulate contamination of the ground in the vicinity of the facility, and (3) residual airborne radioactivity which may persist in the area due to adverse weather conditions.

(1) Direct Radiation

From the standpoint of direct radiation from an unshielded reactor on the test stand, the hazard to personnel is dependent on the source strength at the time of the accident, the period of decay before personnel enter the radiation field, the distance of the people from the source, and any intermediate shielding which may be interposed between the people and the radiation source. A calculation of these effects can be made which provides a measure of the dose rate versus distance from the source. Assignment of a reasonable time factor for personnel to evacuate the tunnel exit area and a reasonable radiation dose to personnel under accident conditions then provides the information necessary to locate the tunnel exit with respect to the direct radiation source.

(2) Radioactive Fallout

Radioactive particulate contamination of the ground in the vicinity of the facility is difficult to determine analytically and directly applicable experimental data are not available. However, considering that the effluent from the reactor must pass through the down-firing duct system and be ejected to the atmosphere at a low angle directed away from the test stands, it is most likely that large fuel fragments would be trapped in the duct or remain in the immediate vicinity of the duct exit. Small size particles may be dispersed randomly over a large area with some tendency to follow the predominant wind direction. Because of this fact, locating the tunnel exit upwind of the prevailing wind direction should reduce the possibility of a radiation dose to personnel from this source of radioactivity.

Under normal conditions, the amount of particulate fallout from the engine effluent may be reduced by the use of an air scrubber device on the effluent exhaust duct. However, accident conditions can be postulated leading to shield separation in which case an air scrubber at the effluent exhaust duct would have little or no effect in reducing particulate fallout or the radiation dose therefrom.

(3) Residual Airborne Radioactivity

Exposure to personnel from residual airborne radioactivity is a consideration which is predominantly dependent on the prevailing weather conditions at the time of the accident. The radioactive cloud would be dispersed along the path of the prevailing wind direction at the time of the accident and at a speed directly proportional to the wind speed. Therefore, as in the case of particulate fallout, the tunnel exit should be located away from the test stands upwind of the prevailing wind direction in order to reduce the possibility of radiation exposure to personnel from the residual airborne activity.

Since the wind direction at the time of an accident cannot be predicted, and since the predominant hazard from residual airborne radioactivity is ingestion of radioiodine, personnel being evacuated under these conditions can be protected by the use of individual self-contained air supplies (Scot-Air Pac type) available at the control point.

As indicated in the preceding discussion, the only source of nuclear radiation from accident conditions not dependent on existing weather conditions is the direct radiation source. It was further indicated that even under adverse weather conditions, nuclear radiation from particulate contamination and residual airborne activity is probably less of a hazard than from the direct radiation source. Therefore, since test operations are normally conducted during favorable weather conditions, it appears reasonable to design the emergency tunnel on the basis of radiation dose rates from the direct radiation source present under accident conditions.

In order to fix the location of the emergency tunnel exit with respect to the test stands, it is necessary to establish a permissible radiation dose to personnel during the evacuation from the facility. An integrated dose of 3 rem is recommended as the allowable evacuation dose based on the U.S. A.E.C. "Standards for Radiation Protection" which allows personnel a dose of 3 rem/quarter. This value was chosen although it could result in a quarterly over-exposure to some personnel, depending on age and previous exposure history. However, it is most probable that the 3 rem/quarter limit would permit the majority of operating personnel to continue the performance of their normal occupational duties.

The integrated dose to personnel is time dependent from two standpoints: (1) the dose rate as a function of distance from the source decreases with time after the accident, and (2) the time spend in a radiation field of given dose rate determines total dose.

The first time factor will establish the dose rate at a given distance from the source at the time personnel must be evacuated. The second time factor is dependent on the time required for personnel to leave the vicinity of the emergency exit. Assuming that the emergency exit area is provided with adequate facilities to expedite an evacuation by emergency vehicles, a reasonable maximum time for evacuation of an individual from the high radiation field is 15 minutes.

Based on the recommended allowable dose to personnel of 3 rem and an evacuation time of 15 minutes, the dose rate to which personnel can be exposed is 12 R/hr. This leaves two variables for the designer to work with; (1) the time after the accident when personnel must be evacuated, and (2) the emergency tunnel length. With the allowable dose rate fixed at 12 R/hr, the designer must determine the best method for meeting this dose rate criteria by selecting adequate values for the two variables.

For example, if the ventilation system is designed to guarantee an adequate air supply to sustain personnel in the control point building for one hour after an accident, data for the dose rate as a function of distance one hour after a 30-minute reactor run at 5000 Mw indicates that a dose rate of 12 R/hr would be present at a distance of 1400 ft from the test stand.

However, if the ventilation system was designed to provide an adequate air supply for a shorter period of time, the tunnel length would have to be increased to meet the dose rate criteria.

The designer must bear in mind, however, that the tunnel length and exit design must also satisfy the requirements previously discussed herein.

APPENDIX H

BASIC FUNCTIONAL REQUIREMENTS FOR E/STS 2-3

SAFETY ORIENTED I&C FACILITY SUPPORT SYSTEMS*

* AGC Memo 7030:M1095, J. B. Philipson to A. Schaff, dated 21 July 1967, Subj: E/STS 2-3 Instrumentation and Communication, Safety Oriented Facility Support Systems

Because of the potentially hazardous nature of E/STS 2-3 operations, special precautions must be taken to control personnel within the complex under both normal and emergency operating conditions. The purpose of this plan is to establish a basis for personnel control and to identify the requirements for a safety system that will adequately support this control.

H.1 SAFETY PRINCIPLES FOR PERSONNEL CONTROL

The basic safety principles that will be followed to control personnel are:

a. Primary control of personnel will be coordinated through a centralized facility location where information on the status of environmental conditions within the complex will be continuously available and where the capability for manual control of audible and visual signals and a means for communicating with facility personnel will be provided.

b. Personnel will be made continuously aware of access restrictions to areas during normal operations and of hazardous conditions and their locations during emergencies. A visual indication and audible alarm system designed for both automatic and manual activation will be provided for this purpose.

c. Personnel will be able to clearly distinguish an accidental criticality from all other hazards. Klaxons will be used as the audible indication for an accidental criticality condition. Sirens will be utilized for audible indication of all other hazards requiring area-wide evacuation. These alarms will be audible throughout the test complex.

d. Evacuation of personnel will always be effected by the safest direction of travel. Visual indicators that clearly show the direction of safe travel will be provided.

e. Unnecessary area-wide evacuation of all personnel to distant protected areas will be minimized. As a result, alarm logic of the audible and visual warning system will be designed for movement of personnel based on the presence of localized or widespread hazard effects. Further control will be coordinated through voice communications from the control point building.

H.2 PERSONNEL CONTROL AT E/STS 2-3

H.2.1 Normal Conditions

The procedural controls that will be used for the protection of personnel at E/STS 2-3 during normal operating periods are:

H.2.1.1 Routine Maintenance Periods

During periods involving routine facility maintenance when the engine is not on the stand, establishment of specific exclusion areas will not normally be required. Therefore, personnel access to the various areas, both above and below ground, will be permissible. However, some localized controls on entry to areas in the vicinity of the test stand or major storage areas may be necessary due to residual radiation from previous testing or other hazards such as the presence of high pressures, cryogenic fluids, etc.

H.2.1.2 Countdown Operations

During countdown operations involving either the installation of the engine at the stand or checkout of the engine or facility systems, the area of personnel exclusion will depend on the hazards involved. During these periods, personnel authorized for access to E/STS 2-3 will be restricted to the immediate area of the control center unless specifically authorized access to other areas.

H.2.1.3 Engine Test Periods

Entry to the E/STS 2-3 complex and access to aboveground areas will not be permitted during engine testing. Personnel at the facility will be restricted to the control center or other approved protected locations as approved by the test director.

H.2.1.4 Post-Test Operations

Exclusion areas will remain in effect at all aboveground areas following a test run shutdown. Due to the possibility of either high gamma radiation from the engine following shutdown or fission product contamination in aboveground areas, personnel will not be allowed to exit to aboveground areas until clearance is given by the test director after environmental surveys by Rad/Safe personnel. After clearance is given, re-entry personnel who are on authorized access lists and are properly clothed and monitored, will be permitted to enter exclusion areas to perform post-test maintenance and securing of systems as required. During removal and transport of the hot engine, all personnel will be evacuated to protected areas where adequate shielding exists.

H.2.2 Emergency Conditions

Hazards associated with high-pressure gas systems, combustible gases, and cryogenics can result in both localized and widespread emergency conditions. Residual radiation levels from previous testing may also be hazardous. Hazards such as oxygen deficiency in personnel-occupied areas and combustible-gas accumulation in areas isolated from cryogenic storage vessels and connecting lines will affect only localized areas. Widespread areas will probably be affected only if fire or explosion were to occur after formation and ignition of combustible gases in close proximity to storage vessels or connecting lines.

If only a localized area is affected, personnel will be evacuated only from the immediate area. If widespread areas are or might be affected, personnel will be evacuated to designated areas. Thereafter, personnel will follow instructions given through the communications system at the control point building.

With an engine on the test stand, radiation hazards of unpredictable magnitude may exist if accidental criticality occurs. Since this condition could potentially affect an extensive area, all personnel will be immediately evacuated to designated tunnel areas, avoiding the test stand area enroute. Personnel will, thereafter, await further instructions given through the communications system at the control center. During a test run, all personnel will be located in the control center or in other protected locations. If an emergency occurs, personnel will remain in these areas pending further instructions.

H.3 SAFETY SYSTEM REQUIREMENTS

Based on the preceding safety principles and methods for personnel control, the requirements for the design and use of the safety systems at E/STS 2-3 have been established. The purpose of the safety systems will be to provide warning of impending or existing hazardous conditions at E/STS 2-3, so that appropriate corrective action may be taken to protect both personnel and facilities. The system will have seven subsystems; (1) surveillance and warning; (2) fire protection, (3) oxygen detection; (4) combustible gas detection; (5) radiation monitoring, (a) atmospheric, (b) gamma; (6) criticality monitoring; and (7) meteorological conditions. The basic requirements for these subsystems are as follows:

H.3.1 Surveillance and Warning System

A surveillance and warning system should be provided to supplement communications capability in alerting personnel of hazardous conditions associated with both normal and emergency conditions and for guiding these personnel to safe locations by the safest route. This system shall consist primarily of area and local visual and audible alarms. For purposes of this document, area refers to the general facility complex; local refers to a specific or confined location within the facility complex.

Automatic activation of the surveillance and warning system shall be accomplished by interconnection with the various sensing devices of the detector systems utilized to warn of abnormal conditions, as described in the following sections. Capability for manual activation of the system shall be provided at both the control center and aboveground areas. Specifically, the manual controls at the control center should be consolidated in a console (Safety Console) and should consist of switches for activating any area visual or audible alarm including main facility road entry or road blocks within the facility complex. These switch controls provide a visual and audible means, during normal operations, for alerting personnel of such conditions as exclusion areas during countdown operations, system checkouts, or other operations involving potentially-hazardous environments and to guide personnel to safe locations by the safest route. The controls can also be utilized as necessary to warn personnel of abnormal conditions in the event such conditions have not been detected by sensing devices of the safety systems. The aboveground manual controls shall consist of emergency switches located strategically throughout the test complex. These controls provide for activation of area visual and audible alarms by any individual observing an abnormal condition which has gone undetected by the sensing devices. Operation of a single emergency switch should activate sufficient area visual and audible alarms to assure general evacuation of aboveground personnel. Signals shall be transmitted to the NRDS Fire Station, as discussed in the following sections.

Area visual alarms shall indicate the general area affected for normal or emergency conditions and shall serve to indicate the safest direction of travel. These alarms shall be of sufficient intensity to be visible both during daylight and darkness. Local visual alarms shall be visible only in the immediate area of the hazard.

Area audible alarms shall be of sufficient intensity to be audible throughout the test complex. Klaxons shall be utilized for accidental criticality of the nuclear engine; sirens for other hazards.

Local clearly audible alarms shall be clearly discernible from area alarms and are required to be audible only in the immediate area of the hazard. Their intensity should be appreciably reduced, as compared with an area audible alarm, so as not to disrupt operations in other than the affected area. If possible, local alarms should be standardized.

Failure of safety system sensors shall be indicated by a trouble alarm at the safety console at the control center. No audible alarms shall be utilized for this purpose.

Several references have been made above and in the following sections to the safety console at the control center. This console should serve as a centralized control for the safety systems. Therefore, sufficient appurtenances should be provided at this console to afford the console operator a means of an awareness of operating status of safety systems, indications of abnormal conditions, and a means for energizing area visual and audible alarms when necessary. These appurtenances should include readouts, recordings, visual and audible alarms, system failure indications, manual activation switches, and alarm acknowledge and reset switches. A safety graphic display shall be provided near the console with flashing visual alarms showing location of an alarmed sensor or activated emergency switch. Lights shall also be provided on this display to indicate whether doors or entrances to the control center have been closed prior to testing. Early corrective action, as necessary, can thus be taken.

H.3.2 Fire Protection and Detection

Fire protection at this facility has been analyzed and determined to fall into one of two categories. The first is, in general, detecting a fire in its incipient stage and advising a responsible authority who will take some pre-planned action. The second category is that of detecting a fire and then automatically actuating a fire control system. These two types of systems must be fully automatic in operation, fail-safe and in continuous monitoring status. They must also be highly reliable and trouble-free.

H.3.2.1 Detection Systems

Automatic detection systems will be installed in all areas that contain combustible materials or electrical equipment, the malfunction of which would cause considerable loss of control or collected information. Areas such as cable termination points, cable plenums, electrical equipment rooms, control center, LH_2 storage area and unloading area, water supply pump station and locations of similar hazard or operating importance shall be provided with automatic detectors.

In general, the type of detector selected for protection at a given location shall be predicated on the type of fire anticipated and the hazard to be guarded. As an example, smoke detectors or product of combustion detectors would be used in electrical equipment areas to give early indication of an electrical fault, since a fire here would be evidenced by smoke some time prior to visible flames or the evolution of large quantities of heat. A similar process of deductive reasoning would indicate that combination fixed temperature rate-of-rise detectors would be used at the water supply pump house, since a fire here would most probably be that of hydrocarbon fuels or ordinary combustible materials. Their involvement would give rise to the rapid evolution of a great amount of heat, which is easily and simply detected by a thermostat type detector. Other types which may be considered for other locations are fixed temperature units of pneumatic systems; however, engineering consideration most govern the final selection to fulfill the particular need for the area covered.

A complete system will include sensors or detectors, power supply, wiring, alarm and trouble signs, annuciators, relays, housing cabinets, etc. All components shall be approved for this service by U. L., Inc., and the installation shall conform to the requirements of the NFPA Standard No. 72A.

The fire detection systems will be of the normally closed electrically-supervised circuit type. Failure of the current supply or a break in the circuit shall cause a distinctive trouble signal, but not a fire alarm. The trouble signal shall be visual only at the control panel in the control center. The trouble light shall remain lighted until the condition is cleared. Emergency power for the system shall be provided for at least two hours of operation in the event of primary power failure.

In addition to a local fire alarm, a signal will also be indicated at the control panel in the control center and the NRDS Fire Station. The audible alarm signal at the control center will be differentiated from all other signals; however, a common audible fire alarm signal may be used for all fire systems. The visual alarm shall indicate the location or zone of the signaling device on a graphic display panel. A fire in the cryogenic area shall automatically actuate the area alarm system in addition to the control center and fire station alarms.

Manual fire alarm boxes shall be located throughout the facility for ease in transmission of a fire alarm to the control center and the NRDS Fire Station. They should be placed at areas of principal hazard (such as cryogenic storage areas, control center concentrations of electrical equipment), and properly marked to facilitate their use for the prompt notification of fire. In addition, they should be located on regular routes of personnel travel. They may logically be a part of heat, fire or smoke detection systems and thereby utilize the components of those systems for power, wiring, zone notification, supervision and alarm transmittal. There is no requirement for coded, or non-interfering fire alarm boxes since the facility is so limited in size.

H.3.2.2 Fire Protection

Automatic damage control systems will be installed for the protection of hazardous areas and the guarding of facilities or components from the effects of a large-scale fire. The types of systems contemplated for use and their anticipated areas of coverage are outlined as follows:

a. Deluge Systems

Water spray systems in this category shall be provided in areas of high hazard, where total flooding for containment, cooling, and/or extinguishment is required. The systems must be engineered so that flow rates and densities of discharge are consistent with the hazard protected or the exposure to be guarded. Those areas included in this category include LO_2 and LH_2 unloading stations, LH_2 storage area, test stand and its appurtenances exposed to the test module, the module propellant tank and the supporting structure and similar areas of high hazard of exposure to fire. It should be noted that the deluge systems required for operational purposes are not a part of this discussion.

The design and installation of these systems will be in accordance with the requirements of the NFPA Standard No. 15, and will consist of distribution piping and spray nozzles to achieve the desired protection. Overlapping spray patterns are required on the surfaces of all protected facilities or structures to insure proper coverage. Actuation will be by local manual and automatic means as well as remotely from the control center. The local automatic actuation of the deluge shall be by the fastest and most reliable system consistent with the state-of-the-art. The manual actuation devices in the area should be easily accessible, well marked, and held to a minimum number consistent with the physical size and scope of the hazard areas. Annunciation of operation will be shown on the graphic display panel at this location and by alarm at the NRDS Fire Station.

b. CO_2 Systems

Total flooding CO_2 extinguishing systems may be valuable in certain enclosed electronic areas. These extinguishing systems must be fully automatic in operation, and should close doors, ventilating louvers and all other openings to the room in order that smothering by total flooding can be achieved. They shall be supplied from high-pressure cylinders in a so-called "two-shot" system, wherein two complete sets of cylinders are provided with one-half of them reserved for a second application or for use while the other bank is out of service. The installation and design shall comply with the requirements of NFPA Standard No. 12. If economic justification indicates a low pressure storage system to be more suitable, it will be acceptable; however, the total quantity of liquid storage must also be predicated upon two separate applications of gas for total flooding.

The system shall be automatically actuated by a smoke or product of combustion detector system and manually operated by local actuation devices as well as remotely from the control panel in the control center. Because of the hazard to personnel, when large quantities of CO_2 are discharged into a confined area (oxygen deficiency and reduced visibility) suitable safety requirements must be met. It should also be noted that panic type hardware is required on the personnel doors to facilitate egress during operation of the CO_2 systems. In addition, there must be local warning signs and audible pre-discharge warning alarms. A time delay relay must also be incorporated into the system, whereby an adjustable time interval from 0-60 seconds may be set to allow for personnel evacuation from the area prior to actual CO_2 discharge. The detection system, per se, will incorporate the safeguards and annunciation features outlined in the section on Fire Detection.

H.3.3 Oxygen Detection

An automatic oxygen detection system is required to determine the efficiency of purging operations in areas such as: the exhaust duct, engine shield area, engine compartment, etc. The requirement is one primarily of an operational safety nature and will not be used in a dependent way to control the entrance of personnel. When it is required to assign people to tasks in areas where an oxygen deficiency is known or suspected, the area will be monitored by trained personnel with portable equipment. Safeguards can then be taken locally if the area is one of low oxygen balance. In no case, however, will personnel safety be dependent upon the "readout" of a fixed detector and installed systems to determine the oxygen percentage in an enclosed area.

It is envisioned that the primary use of a fixed installed system will be that of oxygen sampling to determine the efficiency and completeness of purging in an operational sense.

H.3.4 Combustible Gas Detection

A combustible gas detection system will be required in operational areas where explosive concentrations of hydrogen could accumulate. In general, these areas would be without natural or forced ventilation and where hydrogen could pocket and form a combustible mixture with air. Areas which should be considered include the engine, stage, some portions of the test stand, duct vault, possible enclosures at the bulk storage or unloading area and at the main air inlets of the control center. The system shall consist of a series of remote analyzers located in the areas of principal hazard, with each unit monitoring a specific area of interest. The range of operation shall be from 0-100% of the LEL of hydrogen air mixtures and the units shall be capable of operation in the radiation and pressure environment expected at their installation locations. System components will be U. L., Inc. approved for this service and the remote analyzers must be suitable for operation in an explosive atmosphere without malfunction of initiation or propagation of explosion.

The explosive concentrations of the monitored areas will be displayed at the graphic display panel in the control center. Appropriate alarm set-points are necessary for the actuation of an audible alarm at the control center when concentrations exceed safe values or pre-set percentages.

H.3.5 Radiation Monitoring

H.3.5.1 Atmospheric Radioactivity Monitoring System

During normal engine testing, and the early part of post-shutdown periods or following a nuclear accident, fission products will be released to the environment. Personnel will normally be restricted to below-ground (control center) spaces during these periods. Therefore, provisions shall be made to monitor these personnel-occupied spaces to determine whether such airborne activity is introduced through air intakes or other openings thereby resulting in personnel exposure.

Monitoring units should be provided immediately downstream of the filters of the air intakes and at other locations where airborne fission products may enter and not be detected by the air intake monitoring unit. Appropriate corrective action can be taken if the presence of activity is indicated. These units should be capable of monitoring for radioactive particulate and gaseous beta/gamma activity.

To provide a warning to personnel, the monitoring units should be equipped with an alarm output. Since the control center is a confined area within the facility complex, this alarm output should be connected to local but not to area audible and visual alarms of the surveillance and warning system. (See Section H.3.1, "Surveillance and Warning," for definition of local and area alarms.) The local alarms can be incorporated in the chassis of the monitoring unit. An audible and visual alarm should also be transmitted to the Safety Console of the control center to alert the console operator. A capability should be provided at this console for acknowledgment and reset of this alarm.

In order for the console operator to observe early indications of the presence of airborne activity prior to the alarm indication, a readout meter should be provided at the console. The output of this meter should be connected to a recorder, also located at the console, to provide a continous recording of the airborne activity background.

Stable, reliable, continuous-operating type monitoring units should be utilized since extended periods of operation will probably be necessary. An emergency power supply should be provided for the units to assure continued operation during critical periods. Failure of the monitoring unit should be indicated at the control center by a visual trouble signal which is distinguishable from alarms indicating environmental hazards.

Since maximum permissible concentrations for fission products are of low magnitude (AEC Manual Chapter 0524), the detector chambers shall be shielded to reduce the contributing effect of the ambient radiation background and assure true indication of the presence of significant concentrations of airborne radioactivity. Approximate sensitivity ranges should be:

Particulate - (10^{-10} to 10^{-6} μc/cc)
Gaseous - (10^{-8} to 10^{-4} μc/cc)

H.3.5.2 Gamma Radiation Monitoring System

During engine testing, personnel are restricted to the below-ground control center which is sufficiently shielded to reduce both the gamma and neutron radiation background to a negligible level. Therefore, it is not necessary to provide radiation monitoring instrumentation at this location. Following engine testing with the engine on the test stand, a high-level gamma radiation environment exists aboveground as a result of direct and scattered gamma radiation from the engine, activated structures and equipment, and possible fission product fallout. The intensity of this radiation is appreciably reduced upon removal of the engine; however, residual levels of a hazardous magnitude may still exist. Personnel re-entry to these aboveground areas is required following engine testing. Specific re-entry locations and timing are dependent on operational requirements. Therefore, gamma monitoring units should be installed at the predetermined locations as a remote means for indication from the control center whether personnel re-entry (including Rad/Safe) is feasible. (Portable survey instruments, operated by Rad/Safe personnel, are utilized as the primary means for controlling the actual re-entry of personnel to these areas.)

In the event a major nuclear accident occurs on a test stand, it may be necessary for personnel to evacuate from the control center. Gamma detectors should therefore be provided at the exits from the control center tunnels to ascertain the radiation intensity, so that an early decision can be made to determine the best evacuation route. The detector units should not be affected by shielding afforded by the tunnel exit structure.

Based on the above-discussed functional requirements, readout meters and recorders for the gamma detectors are necessary only at the safety console at the control center. No readouts at the detector positions are required. The gamma detector outputs are not to be interconnected with any visual or audible alarms of the surveillance and warning system since the function of the detector units is to indicate magnitude of radiation intensity and not to provide audible and visual warnings to personnel of abnormal conditions.

Reliable continuous-operating type detector units capable of operation in temperatures ranging from 0° to 140°F should be utilized. The units should be equipped with internal radiation check sources, solenoid-operated from the safety console at the control center. This provides a convenient remote means for checking out detector operation and response. Failure of any individual detector shall be indicated by a visual trouble signal at the safety console which is readily distinguishable from visual alarm signals incorporated in this console.

Detector sensitivity ranges should be established based on approximate environmental gamma radiation levels predicted for both normal and accident conditions including consideration for radiation decay. These predictions are available from isodose maps for E/STS 2-3 operations. Accuracy of the detectors should be ± 15% of the actual intensity over the entire detection range.

H.3.6 Criticality Monitoring System

During certain pre-test and post-test periods with the engine at the test stand, personnel will be required to enter areas at or near the test stand. During the pre-test periods (before engine firing) the radiation contribution of the engine is negligible. During post-test periods, the contribution of the engine to the radiation environment is of a high magnitude due to decay gamma and special precautionary measures are taken to control re-entry of personnel to these areas.

During both pre-test and post-test periods, consideration must be given to the possibility of occurrence of an accidental criticality of the engine. In this case, both neutron and gamma radiations of a high magnitude would be generated. Since such an uncontrolled event can occur almost instantaneously, personnel in the immediate vicinity of the engine may not have sufficient time to evacuate to preclude a high radiation exposure. However, other personnel may be at sufficient distances away from the engine such that, if warned of this condition can evacuate to safe areas and avoid a high exposure. Therefore, a means should be provided to detect an accidental criticality and immediately alert all facility personnel so that evacuation to safe areas can be immediately initiated and appropriate action taken by control personnel.

A criticality monitoring system is the most appropriate means for detecting the accidental criticality condition. This system should be capable of detecting this occurrence with or without the engine shield tank in place around the engine and for both the pre-test and post-test periods mentioned above.

Though not a part of the criticality monitoring system, it should be noted that dosimetry measurements as a result of an accidental criticality are provided by dosimeter units installed in accord with an AEC requirement, AEC Manual Chapter 0545, Nuclear Accident Dosimetry Program.

Since the post-test period involves high ambient gamma decay radiation, the detector units should be capable of discriminating against gamma intensities of approximately 10^4 R/hr or greater. Neutron detectors should, therefore, be utilized with a minimum number of three detectors incorporating an alarm logic of two-out-of-three to minimize extraneous alarms. (This logic requires that the alarm set point of at least two of the three detectors must be exceeded for activation of a warning alarm.) To assure immediate indication of the criticality event, the detectors should have a response time of 0.1 second or less.

To establish a basis for determing detector sensitivity and alarm set points to assure indication of a criticality condition, reference is made to the criteria of Title 10, Part 70, Paragraph 70.24 of the Code of Federal Regulations. Although the requirements of 10CFR70 are not directly applicable for E/STS 2-3 operations, they do represent basic guidelines which are the product of considerable experience in the field of reactor safety. The primary requirement of this document from a system sensitivity standpoint is that the system shall energize an alarm system when the radiation level resulting from a nuclear excursion is 300 rem/hr at a distance of 1-ft from the source of the radiation. When applied to E/STS 2-3 operation, this criterion shall be interpreted as 1-ft from the reactor pressure vessel when the engine shield tank is not in place around the engine and 1-ft from the outer surface of the engine shield tank when it is in position around the engine. In selecting specific detector sensitivity, type, and placement to meet this criterion, careful consideration should be given to interposing shielding between detector and engine, and to existing shutdown (post-test) photoneutrons.

To meet the above-mentioned need for adequately warning facility personnel to an accidental criticality and guiding these personnel to safe locations, the alarm output of the detector system should be appropriately interfaced with the surveillance and warning system. Since this hazardous condition will, in all probability, affect the general facility areas at E/STS 2-3, activation of the alarm output of the detector shall initiate area visual and audible alarms including a visual alarm at the facility entrance. A signal shall also be transmitted to the NRDS Fire Station to indicate the accidental criticality. As indicated in Section H.3.1, the klaxon shall be utilized as the area audible warning signal for accidental criticality in order to assure differentiation from other hazards. Area visual alarms shall clearly indicate the affected test stand and safe direction for evacuation. Monitoring system readouts, recordings, visual and audible alarm, manual alarm activation capability, alarm acknowledge and reset capability shall be provided at a centralized control console at the control center.

To assure continued operation of the monitoring and alarm systems, emergency power supplies shall be provided to adequately supply the systems during loss of primary power. Failure of any single detector channel shall be indicated by a visual signal at the control center console with failure of any channel permitting continued operation of the remaining channels. A capability shall also be provided at the console for deactivating the detector system during engine testing to preclude actuation of the warning alarm and possible damage to the system.

H.3.7 Meteorological System

The capability for evaluation of meteorological conditions need only be simple in terms of information acquired, since detailed weather information and predictions are provided by the U. S. Weather Bureau. The meteorological system should provide the capability to assess only those local conditions which could affect the safety of operations. The required information includes wind speed and direction for selected locations within the test complex, e.g., at the control center, test stands and at the duct exits. The information shall be displayed at the control center.

APPENDIX I

REFERENCE DOCUMENTS

This appendix contains copies of the AGC memos which are referenced in the body of the report and which support the discussion and conclusion contained therein. The memos provided are:

I-1 Memo 7030:M1024, J. B. Philipson to A. Schaff, Jr., dated 15 May 1967; Subject: "E/STS 2-3 Control Center Design Criteria".

I-2 Memo 7030:M1063, J. B. Philipson to A. Schaff, Jr., dated 23 June 1967; Subject: "Test Stand Deluge Water System".

I-3 Memo 7030:M1099, J. B. Philipson to A. Schaff, Jr., dated 24 July 1967; Subject: "Control Center General Arrangements and Control Tunnel Sections and Details".

I-4 Memo 7030:M1105, J. B. Philipson to A. Schaff, Jr., dated 27 July 1967; Subject: "LH_2 Retention Pond (E/STS 2-3 DTL 0078)".

I-5 Memo 7030:M1112, J. B. Philipson to A. Schaff, Jr., dated 4 August 1967; Subject: "E/STS 2-3 Nuclear Based Intra-Facility Separation Distances".

I-6 Memo 7030:M1154, J. B. Philipson to A. Schaff, Jr., dated 1 September 1967; Subject: "Safety Requirement for Minimum Emergency Power at E/STS 2-3".

I-7 Memo 7030:M1228, J. B. Philipson to A. Schaff, Jr., dated 16 October 1967; Subject: "Review of I & C Systems, STL No. 0120 E/STS 2-3 Procurement Package No. 10".

00-007 022

INTER-OFFICE MEMO

AEROJET-GENERAL CORPORATION

TO: A. Schaff, Jr.

DATE: MAY 15 1967

7030:M1024
HRB/DSD/smh
55264/54824

FROM: J. B. Philipson

SUBJECT. E/STS 2-3 Control Center Design Criteria

DISTRIBUTION. S. S. Bacharach, C. A. DeLorenzo, D. S. Duncan, C. F. Leyse, R. V. Lichtenberger, B. Mandell, C. M. Rice, W. D. Stinnett, J. A. Vreeland, W. O. Wetmore, W. L. Winegar

References:
(a) Memo 7860:0416M, W. L. Winegar to Distribution, dated 1 May 1967; Subject: E/STS 2-3 Document Review
(b) Drawing 67-4-1, E/STS 2-3 Control Center - Sections
(c) Drawing 67-4-2, E/STS 2-3 Control Center - Bottom Floor Plan
(d) Drawing 67-4-3, E/STS 2-3 Control Center - Top Floor Plan
(e) E/STS 2-3 Control Center Design Criteria, dated 24 April 1967

In response to the Reference (a) request, NRO Safety has reviewed the Reference (b), (c) and (d) drawings, and the Reference (e) design criteria for E/STS 2-3. This review was limited to the safety criteria aspects of the control center design and, as such, did not consider either the interrelationships of other facility structures, systems or components, or the adequacy of the specific design data contained in the referenced documents. Safety evaluations of the Control Center design and its interrelationships with the facility design and operation will be performed on a continuing basis as additional information becomes available. On this basis, the following comments are transmitted:

1. No mention is made of providing separate liquid waste disposal systems for the radioactive and non-radioactive wastes. The referenced documents specify a decontamination room at the Control Center for personnel who have performed work in radioactively contaminated areas. It should be emphasized that it is standard practice in such areas to provide radioactive liquid drain systems, which are independent of the systems utilized for disposal of domestic liquids, to preclude cross-contamination. Such disposal provisions have been made for the personnel decontamination areas of the R-MAD and E-MAD facilities at NRDS. Therefore, it is recommended that these provisions be incorporated in the design criteria for E/STS 2-3.

2. The Reference (e) document indicates that the decontamination area shall be served by an independent air conditioning system with no return to an air handling unit. This criteria should be expanded to include a requirement for filtering the exhaust air from this area and for exhausting this air to the outside environment so as to preclude release of contamination to other occupied areas at the Control Center.

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3. The design criteria specifies use of a single air intake fan to supply fresh air to the Control Center and tunnel areas. It is also indicated that a positive pressure shall be maintained at all times in these areas. Thus, normal fresh air for personnel occupying these areas and maintenance of positive pressure at these same locations is dependent primarily on continuous operation of this single fan. Careful consideration should be given to determining the adequacy of a single fan for these purposes in view of the large air handling capacity required and possible malfunction of this single unit. It should be noted that two air intake fans, one of 13,000 cfm capacity and one of 13,500 cfm capacity, are provided at ETS-1 for serving personnel-occupied areas and tunnels.

4. It is indicated that controls for all remote air conditioning control instruments will be mounted on a control panel at a convenient location in the air conditioning room which is in the mechanical area, a floor level below the control room. It is noted that the fresh air system will include air monitors and an automatic shutdown capability when the monitors detect either radioactivity or combustible gases in the air stream. However, it should be emphasized that it may be highly desirable to shut down the fresh air system immediately following any accident at above-ground areas and prior to entry of contaminants to the fresh air system. Undue delay in shutdown may result with controls provided only in the air conditioning room. Therefore, consideration should be given to provision for fresh air intake controls at a strategic location in the Control Room where action can be taken to effect shutdown immediately following an accident. Similar air intake controls are provided at the LSE Console in the Control Point Building at ETS-1.

5. With regard to emergency lighting, it should be noted that lighting of the Control Room and other areas of the Control Center should be sufficient to permit continuance or safe termination, as appropriate, of a test operation in the event of failure of the main lighting power supply.

6. Reference (e) should include shielding criteria for protection of Control Center personnel against the external radiation environment under both normal and accident conditions.

7. It is not clear where the engines and associated equipment to be used for generation of electrical power will be located. It is recommended these units be located outside the Control Center to reduce the potential for fire and contamination of the Control Center air with toxic fumes.

8. The criteria should establish a requirement for preparation of an engineering design report which shows how the principal design criteria have been satisfied and gives the bases for the detailed design.

9. It is recommended that the following changes be made to Paragraph 2 on Page 2 of Reference (e):

a. Change second sentence to read: "The contractor's analysis shall consider protection of personnel against major nuclear and non-nuclear accidents at both the test stand and in the cryogenic storage area."

b. Change the third sentence to read: "Major non-nuclear accidents shall be defined by the contractor."

c. Add a sentence following (b) above as follows: "Major nuclear accidents will be defined by the Customer."

10. The surfaces of walkways used for personnel access to the decontamination room should be protected with special preparations (paint coatings, etc.) for ease of decontamination. Therefore, it is recommended that this provision be included in the Reference (e) document.

ORIGINAL SIGNED BY

J. B. Philipson
Manager, Safety Division
Nuclear Rocket Operations

INTER-OFFICE MEMO

AEROJET-GENERAL CORPORATION

TO:	A. Schaff, Jr.	DATE:	JUN 2 3 1967
FROM:	J. B. Philipson		7030:M1063 GMD/smh:54332
SUBJECT:	Test Stand Deluge Water System		

DISTRIBUTION: C. F. Leyse, P. E. Neal, G. M. Orihood, C. M. Rice, P. W. Rowe

References:
(a) Memo 7400:7547, A. Schaff, Jr. to Distribution, dated 26 May 1967; Subject: E/STS 2-3 Transmittal Letter #0064
(b) Criteria for Preliminary Design of E/STS 2-3, dated 20 January 1967
(c) Final Project Report Budgetary Study, dated May 1966

In response to the request in Reference (a), NRO Safety has reviewed the KE Drawing C3-5101, Test Stand Deluge Water System Flow Diagram, enclosed therewith.

In connection with this review, we have also reviewed the criteria outlined in References (b) and (c) and find that these do not provide an adequate base for our detailed evaluation of the flow diagram. With respect to deluge, Reference (b) only notes (on page VI-D-72) that test stand deluge is included in the fire protection subsystem. Reference (c) indicates that water flow densities of one gpm/ft^2 of exposed area will be supplied for protection of the exposed portions of the stand, shield tank, vault, etc., but does not supply any additional guidance except to state that the deluge system will be designed on the basis of practices at the NRDS and MTF facilities. We were not able to locate and review the documentation of these practices to determine their appropriateness. The following comments relative to the criteria for the deluge systems are provided for your consideration.

1. The deluge systems are installed for two separate functions, i.e., protection against structural damage by fire and protection from the radiant heat of the exhaust plume, the latter serving an operational function while the former is clearly a fire protection function. We suggest that the criteria identify these systems as "operational" or "fire protection" systems as appropriate, so that they may be better evaluated as to water demand, density of discharge, automatic and/or manual control and other pertinent features.

   For example, we would expect the purpose of the deluge systems for the Ejector Vault Exit Zone and the Service Tower Face Zone to be protection of these areas from thermal effects of the plume during a normal test. Therefore, these portions of the deluge system could be manually operated and not a part of the Fire Protection System. The water flow rates and coverage would be designed accordingly.

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Unclassified
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CLASSIFYING OFFICER DATE

By contrast, we would expect the module side of the tower to be provided with deluge as part of the Fire Protection System to prevent damage in event of a fire. This system and those protecting the shield tank, module zone, the upper and lower service arms and the tower platform zones would provide deluge protection in event of fire on the stand or at the cryogenic storage area. Their coverage and water flow rates would be designed to provide protection for the worst case situation.

2. We suggest that the basic criteria for design of the deluge systems follow the recommendations of the National Fire Protection Association. The design, thus, would conform to the requirements of the National Board of Fire Underwriters Standard No. 15, Water Spray System for Fire Protection. The design of deluge protection for our test stands at Sacramento and ETS-1 have followed their criteria and those at Sacramento have been demonstrated to be satisfactory from the performance standpoint.

3. While the density of water discharge of one gpm/ft^2 of wetted surface may be appropriate for fire protection needs, we do not know that this is appropriate for protection from the radiant heat of the hydrogen plume. The ETS-1 activation program includes tests that will provide information on protection against radiant heat from the plume. We suggest that establishment of water discharge densities for the "operational" systems await evaluation of the ETS-1 test results.

ORIGINAL SIGNED BY

C. F. Leyse

for J. B. Philipson
Manager, Safety Division
Nuclear Rocket Operations

INTER-OFFICE MEMO

AEROJET-GENERAL CORPORATION

TO: A. Schaff, Jr.

DATE: 24 July 1967
7030:M1099
GMO:jh

FROM: J. B. Philipson

SUBJECT: Control Center, General Arrangements and Control Tunnel Sections and Details

DISTRIBUTION: C. F. Leyse, P. E. Neal, G. M. Orihood, C. M. Rice, P. W. Rowe, W. D. Stinnett, W. O. Wetmore, W. L. Winegar

Reference:
(a) Memo 7400:7569, A. Schaff, Jr. to Distribution, dated 27 June 1967; Subject: E/STS 2-3 Document Transmittal Letter #0081
(b) Memo 7030:M1024, J. B. Philipson to A. Schaff, Jr., dated 15 May 1967; Subject: E/STS 2-3 Control Center Design Criteria
(c) Agreements Reached at Meeting Held in Oakland on 29 March 1967 to Establish Safety Criteria for the Design of the E/STS 2-3 Control Point Building
(d) Memo 7030:M10895 J. B. Philipson to A. Schaff, Jr., dated 3 Nov 1966; Subject: E/STS 2-3 Control Point Building Personnel Protection
(e) E/STS 2-3 Control Center Design Criteria, dated 24 April 1967

In response to the request in Reference (a), NRO Safety has reviewed the four Kaiser Engineers drawings which relate to the subject. These drawings are not too definitive; however, their review raises several questions we have discussed in previous recommendations concerning the Control Center. These recommendations, which are given in References (b), (c), and (d), should be re-examined for consideration in the pertinent portion of the design effort. In the following paragraphs we discuss several aspects of the design to insure they are given the proper consideration.

The first is isolation of the electrical room and terminal room. Both of these areas should be segregated from the remainder of the building by a one-hour fire resistive separation. They should also be provided with segregated or separated ventilation systems such that a fire in either of these areas cannot extend to other portions of the building or discharge smoke and fumes to other areas and thereby preclude emergency occupation of this building.

The second aspect is that of make-up air for supply throughout the building. It appears from the drawing of the lower floor that a single outside air supply system is contemplated. We call to your attention that the design criteria (Reference (e)) indicated that a positive pressure in the Control Room and tunnel areas shall be maintained at all times. A single fan versus a dual system should be carefully evaluated in light of this requirement.

A third aspect for consideration is emergency egrees from the Control Center. The upper floor arrangement does not make clear that the personnel access tunnels will be protected against flooding, making them unsuitable for use and jeopardizing the integrity of the building. We naturally expect that this important point will be examined and made a part of the design.

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Unclassified
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A fourth aspect we wish to discuss is that of the power transformer installed in the electrical equipment room. This contemplated installation is not in consonance with maximum safety of individuals in the Control Center and the installation should be made outside at grade or in an interior vault. This subject will receive more attention in a discussion of electrical matters, presented on Document Transmittal Letters 0084, 0085, 0086, and 0087.

In summary, we wish to emphasize that the Control Center is not only an operations building, but is also a place of refuge. Therefore, it should receive every consideration for elimination of hazards and the providing of safeguards for the personnel who will be expected to remain there during testing.

ORIGINAL SIGNED BY

C. F. Leyse

for J. B. Philipson
Manager, Safety Division
Nuclear Rocket Operations

00-007-022

INTER-OFFICE MEMO — AEROJET-GENERAL CORPORATION

TO: A. Schaff, Jr.

DATE: 27 July 1967
7030:M1105
GMO:jh:5-4332

FROM: J. B. Philipson

SUBJECT: LH_2 Retention Pond (E/STS 2-3 DTL #0078)

DISTRIBUTION: C. A. DeLorenzo, C. F. Leyse, P. E. Neal, G. M. Orihood, C. M. Rice, P. W. Rowe, W. D. Stinnett, W. O. Wetmore, W. L. Winegar

Reference:
(a) Memo 7400:7559, A. Schaff, Jr. to Distribution, dated 7 June 1967; Subject: E/STS 2-3 Document Transmittal Letter #0078
(b) Memo B. J. Billings to Distribution dated 5 June 1967; Subject: E/STS 2-3 Document Rransmittal (DTL 0078)
(c) Criteria for Preliminary Design of E/STS 2-3, dated 20 Jan 1967
(d) Final Project Report Budgetary Study, dated May 1966
(e) Supplementary Project Report, Study Area F, dated 30 Sept 1966
(f) SNPO Document Transmittal Letter SNPO 0078-R, B. J. Billings to J. S. Ritchie, Kaiser Engineers, dated 23 June 1967, re KE Dwg No. CL-SKC-5 - Proposed LH_2 Retention Pond

In response to the request in Reference (a), NRO Safety has reviewed the Kaiser Engineers proposal to retain and burn hydrogen (from a major dewar leak) in a retention pond adjacent to the LH_2 dewar in the cryogenic storage areas, as described in Reference (b) and enclosures thereto. In connection with this review, we have also reviewed the appropriate portions of the criteria and the budgetary study results in References (c), (d), and (e). As a result of this review we find the KE proposal undesirable and we concur with the SNPO reply in Reference (f). The following comments are provided for your consideration.

The accident, as postulated in Reference (e), is a shearing of the 20-inch pipe flange at the base of the main LH_2 storage dewar when the dewar is pressurized to 75 psig. KE assumes that, allowing five seconds for closing off the pressure source, the dewar contents are discharged onto the ground under a decaying pressure head. The KE Leak Flow Curve, enclosed with Reference (b), indicates an initial LH_2 flow rate of about 2500 lb/sec, decreasing to about 200 lb/sec in 7 minutes, after which gravity flow results in emptying the tank in about one hour after the break. The KE proposal involves collecting and burning the leaked LH_2 in a retention pond relatively near the LH_2 dewar. The KE documentation does not include an analysis of the consequences to the cryogenic storage area of such a conflagration; however, we would anticipate that because of the proximity of the retention pond the LH_2 dewar and associated vital equipment would be jeopardized.

In view of the above, we suggest that several alternate approaches be considered for minimizing the consequences of the postulated accident. While discussed further below, they are briefly:

1. Design the LH_2 dewar (and its associated piping) with features that facilitate termination of flow from the dewar in the event of the postulated accident.

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CLASSIFYING OFFICER DATE

2. Provide steep drain for the spilled LH_2 from the area under the dewar and collect the spilled LH_2 in a "deep well" at a more remote location than that shown for the retention pond on the KE Drawing C1-SKC-5, enclosed with Reference (b).

3. Do not intentionally ignite leaking hydrogen, but take measures to eliminate ignition sources in the spill and drain area.

With respect to Item 1 above, it appears that some steps could be taken in the design of the main LH_2 storage dewar and its associated piping to provide for termination of LH_2 leakage shortly after occurrence of a major leak such as the postulated accident. For example, the LH_2 fill-discharge line could enter the bottom of the dewar outer shell, proceed upward through the vacuum-insulated space between the inner and outer shells, enter the liquid tank through the top, and project downward into the liquid, terminating near the bottom of the liquid tank, similar to the main LH_2 dewar at ETS-1. A "siphon break' line could be incorporated between the fill-discharge line and the ullage volume. The "siphon-break" line would be open at all times except during planned fill or discharge operations, and would be opened in the event of the postulated accident during fill or discharge operations. It appears that the above and other schemes to terminate LH_2 leakage in the event of the postulated accident warrant consideration.

With respect to Item 2 above, it is noted that KE found that the amount of LH_2 in the leak did not evaporate before it reached the Road "H" culvert and therefore they proposed to retain the LH_2 in the retention pond and burn it there. If evaporation of the LH_2 does not keep pace with the leakage rate, then LH_2 accumulates so that when the leak is terminated a pool of LH_2 (and source of fire) remains. In such case, it would appear that consideration should be given to providing steep drainage away from the cryogenic storage area so as to minimize the time this area is susceptible to fire from the leaked LH_2. Further, rather than collect the LH_2 in a relatively large area retention pond in close proximity to the cryogenic storage area, it would appear that consideration should be given to collecting the LH_2 in an open "deep well" at a more remote location. If ignited, the LH_2 in the deep well should provide a fire of lesser magnitude than that of a large retention basin and thus reduce the problem of protecting the LH_2 dewar and adjacent facilities.

With respect to Item 3 above, we cannot find in the KE documentation adequate justification for deliberate ignition of the spilled LH_2. KE has not established that intentional ignition can be accomplished with due protection to personnel and major facilities from the ignition and subsequent fire, nor has KE shown its advantages over non-ignition. Until such a basis for intentional ignition is provided, we adhere to the current generally accepted philosophy that ignition sources should be removed as far as practical from potential LH_2 spill areas and collection areas in the hope that the LH_2 will evaporate harmlessly in the atmosphere.

ORIGINAL SIGNED BY
C. F. Leyse

for J. B. Philipson
Manager, Safety Division
Nuclear Rocket Operations

00-007 02?

INTER-OFFICE MEMO

AEROJET-GENERAL CORPORATION

TO: A. Schaff, Jr.

DATE: AUG 4 1967

7030:M1112
DSD/smh:54824

FROM: J. B. Philipson

SUBJECT: E/STS 2-3 Nuclear-Based Interfacility Separation Distances

DISTRIBUTION: C. A. DeLorenzo, D. S. Duncan, C. F. Leyse, P. E. Neal, G. M. Orihood, C. M. Rice, W. D. Stinnett, J. A. Vreeland, W. O. Wetmore

References:

(a) Kaiser Engineers Report for E/STS 2-3, dated 31 January 1967; Subject: Nuclear and Safety Analysis Interim Report

(b) "Criteria for Preliminary Design of Engine/Stage Test Stand 2-3," Prepared by SNPO, dated 20 January 1967

(c) AGC Report No. RN-TM-0415, dated December 1966; Subject: Interim Nuclear Analysis in Support of E/STS 2-3 Site Layout Activities

(d) AGC Letter, W. O. Wetmore to W. E. Johnson, dated 18 July 1967; Subject: E/STS 2-3 Radiological Environment Safety Report

The Reference (a) report establishes minimum interfacility separation distances based on the personnel access requirements given in Reference (b), on results of environmental radiation analyses reported in Reference (c), and on the use of a facility shield with a thickness equivalent to 6 ft of water. The separation distances established do not take into account radiation levels which can result from dispersion of radioactive material either during normal test operations or under accident conditions. However, as stated in Reference (a) and concurred with herein, the only interfacility separation distance which is expected to be affected by dispersion of radioactive material is the emergency-tunnel exit location. With the exception of this location, the nuclear-based separation distances established by Reference (a) appear to be adequate from the safety standpoint.

Westinghouse is currently analyzing the on-site effluent distribution of particulate for both normal test and accident conditions, Reference (d). Their preliminary results are expected to be available for review on 9 August.

ORIGINAL SIGNED BY

J. B. Philipson
Manager, Safety Division
Nuclear Rocket Operations

00-007-022

INTER-OFFICE MEMO AEROJET-GENERAL CORPORATION

TO: A. Schaff, Jr. DATE: SEP 1 1967

7030:M1154
DSD/smh:5482

FROM: J. B. Philipson

SUBJECT: Safety Requirement for Minimum Emergency Power at E/STS 2-3

DISTRIBUTION: V. M. H. Chang, G. D. Cowles, D. S. Duncan, R. E. Huffman, E. W. Leachty, C. F. Leyse, R. V. Lichtenberger, B. Mandell, G. M. Orihood, L. K. Porter, C. M. Rice, W. D. Stinnett, W. O. Wetmore, W. L. Winegar

Reference: (a) Memo 7860:0598, G. D. Cowles to Distribution, dated 31 August 1967; Subject: Critical Electrical Power Requirements, E/STS 2-3

As requested by Reference (a), the critical electrical power requirements for E/STS 2-3 have been reviewed by NRO Safety. The critical power load identified therein appears to be more than adequate to satisfy safety requirements if the critical power generation system is suitably isolated from the primary power system and the critical power is properly distributed to components within the facility. Specifically, the basic safety requirement for emergency power at E/STS 2-3 is:

> Redundant or backup power shall be provided in such a manner that no single failure in the electrical power system can preclude safe termination of engine test operations (shutdown plus cooldown), jeopardize or cause major damage to the facility or test article, or result in undue hazard to personnel.

This requirement necessitates the use of a separate and independent electrical power system which will be on-line and available for critical instrumentation, controls and facility components in the event of failure of the primary power system. The specific items of equipment which must be provided with backup or emergency power have not been established at the present time; however, an example of the type of equipment which requires such power is discussed in Section 4.9 of the ETS-1 Facility Safety Report, RN-S-0378.

ORIGINAL SIGNED BY

J. B. Philipson
Manager, Safety Division
Nuclear Rocket Operations

CLASSIFICATION CATEGORY
[illegible]
[illegible]
CLASSIFYING OFFICER [illegible]

CO-007-022

INTER-OFFICE MEMO

AEROJET-GENERAL CORPORATION

TO: A. Schaff, Jr.

DATE: OCT 16 1967
7030:M1228
GMO:jh

FROM: J. B. Philipson

SUBJECT: Review of I&C Systems, DTL 0120 E/STS 2-3 Procurement Package No. 10

DISTRIBUTION: C. F. Leyse, P. E. Neal, G. M. Orihood, L. K. Porter, C. M. Rice, W. D. Stinnett, W. L. Winegar, W. O. Wetmore

Reference: (a) Basic Functional Requirements for E/STS 2-3 Safety Oriented I&C Facility Support Systems, dated 21 July 1967

At your request, NRO Safety has reviewed the subject Procurement Package No. 10 on the I&C Systems using Reference (a) as a basis for review. Our comments in accord with the specification and drawing numbering system are as follows:

6.4.3 <u>Nuclear Criticality Monitoring System</u>

There is no requirement for magnetic tape recording.

6.4.3.2 <u>Proposed System Description</u>

There is no need to have the three detectors spaced equally around the periphery of the shield. They can all be located together on the side wall of the test stand where they will be out of the way and simplify the installation.

c. There is no requirement for magnetic tape recording.

h. An individual meter should be provided for each detector channel, instead of only a single meter with channel selection push buttons to cover all channels.

A key switch should be provided to cut off the high voltage to the detectors during a test.

6.4.3.3 <u>Detector Range</u>

There appears to be a typographical error. The detector range is most likely intended to be 2.5×10^{-2} to 2.5×10^{3} nv rather than 2.5×10^{2} to 2.5×10^{3}.

6.4.4 <u>Gamma Radiation Area Monitoring System</u>

There is no requirement for magnetic tape recording.

6.4.4.2. <u>Proposed System Description</u>

c. There is no requirement for magnetic tape recording.

CLASSIFICATION CATEGORY
Unclassified
C. F. Leyse 10/16/67
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e. A separate readout meter for each channel is preferred over a single meter with channel selection push buttons. However, it is required that individual channel meters be provided for those detectors that measure radiation levels at the escape exits from the control room.

A key switch should be provided to cut off the high voltage to the detectors during a test.

6.4.4.3 Detector Ranges

b. A range 1 to 10^6 mr/hr is recommended.

c. A range of 1 to 10^6 mr/hr is recommended except for the detectors at the pad, for which a range of 1 mr/hr to 10^6 r/hr is recommended.

Drawing No. B1-8408 Gamma System Block Diagram

1. Delete detectors proposed for installation in the tunnel.

2. Provide an additional detector at the vehicle loading area of the Control Center.

3. Provide an additional detector at pad level, on the module side at each test stand.

4. At each test stand provide an additional detector outside at the stairwell exit planned for use as an emergency escape exit from the tunnel.

5. Provide an additional detector at the main substation.

6. Provide two additional detectors which will be installed at the exits of the two emergency escape tunnels.

6.4.5 Atmospheric Radioactivity Monitoring System

There is no requirement for magnetic tape recording.

There should be no connection to the Area Surveillance and Warning System. Alarms will be local (in the Control Center) only.

6.4.5.2 Proposed System Description

c. There is no requirement for magnetic tape recording.

e. An individual meter should be provided for each detector channel instead of only a single meter each for particulate and gaseous activity with channel selection push buttons.

Drawing No. B1-8407 Atmospheric System Block Diagram

1. Delete connection to Area Surveillance and Warning System.

2. A "Spares" capability may be necessary for expansion of the system in the Control Building when the ventilation system is completely defined.

6.4.6 Portable and Fixed-Installation Radiation Survey Instruments

The equipment and instrumentation outlined in this section can be deleted and procured at a later date as no detail design effort is required for their installation.

6.5 Oxygen Detection System

Delete the requirement for oxygen detection with respect to "high relative concentrations (above atmospheric ambient) in the vicinity of oxygen storage vessels and transfer systems."

6.5.2 Proposed System Description

The electrolytic cell type sensor will saturate in the radiation environment and is not suitable. A paramagnetic, wind type is more suited for this application.

b. No requirement for connection to the Area Surveillance and Warning System.

c. No channel readouts are required on the Safety Engineer Console unless the channel monitors a purged area where personnel re-entry is required.

If channel readouts are provided on the Safety Engineer Console, a separate meter for each detector channel should be provided instead of a single meter with channel selection push buttons.

6.5.1 Functional Description

a. Detector or sensor elements located in purged areas requiring personnel re-entry shall be provided with local alarms to indicate a less than normal oxygen percent in the environment. Where installed, they shall be provided with an alarm cutout (at the Safety Engineer Console) to prevent continuous operation of the alarm during the purged period.

Drawing No. B1-8409 Oxygen Detection Block Diagram

1. Delete connection to Area Surveillance and Warning System.

2. Delete sensors in tunnels.

3. Delete sensors at LOX storage and transfer area.

6.6 Meteorological System

From a safety standpoint there is no need for meteorological information other than wind speed and direction.

6.7 Combustible Gas Mixture Monitoring System

Add the following sentence, "All system components will be U.L. Inc. approved for this service."

6.7.2 Proposed System Description

b. No requirement for connection to the Area Surveillance and Warning System.

c. An individual meter should be provided for each detector channel instead of the single panel meter with channel selection push buttons.

Drawing No. B1-8410 Combustible Gas Monitoring Block Diagram

1. Delete all sensors at the LH_2 dewars.

2. Delete all sensors at the LH_2 Fill Station.

3. Delete all sensors at the Gaseous Storage Area.

4. Delete all sensors in the tunnels.

5. We don't understand the philosophy of the sensors being located in the various rooms of the test stand from -4 level to the 10th level and therefore can't state if this is a satisfactory arrangement. It is our feeling that the rooms should be either properly sealed or ventilated to prevent H_2 accumulation, thereby making detection unnecessary.

6. We don't understand why six sensors are necessary for the Control Center air supply duct. One, downstream of the filters, should suffice.

7. One detector should be located on each floor of the Control Building.

6.9 Fire Detection System

6.9.1 Type II Detection System

Delete the installation on the vessels, piping and equipment in the cryogenic area, in the high pressure gas storage area, and perhaps at the piping and valves on the pipeway. More needs to be known about proposed location and physical conditions of the latter before appropriate recommendations can be made.

6.9.3 Type II Monitoring System

Delete the requirement for thermocouples at the cryogenic storage area, the high pressure gas storage area, and pipeways.

6.10 Area Surveillance and Warning System

6.10.1 System Description

Delete interconnection to Atmospheric Radioactivity Monitoring System.

17U Fire Detection System

17U-2 Applicable Publications

Add 2.3 National Fire Protection Association, Standard No. 72A, Local Protective Signaling Systems.

17U-4.1.1 General

Add: "The sensor shall be a U.L. Inc. approved component."

17U-4.8.1 General

An audible trouble alarm should not be provided.

Add a new section (4.9) to cover the emergency power system.

17U-5 Design of Type II Monitoring System

This system is not required for temperature monitoring of vessels or general service.

Drawing No. B1-8412 Fire Detection Block Diagram

1. The ionization type detectors for Test Stand Area will have to be replaced with a suitable type which will not saturate in the radiation environment.

2. It is not necessary to have two types of fire detectors in the same area (floor level 4, etc.).

3. Delete Thermocouples on E&W faces of Test Stands 3 and 2, respectively, and on outside of engine shield.

4. Delete requirement for a manual fire alarm station at each floor level in the test stands. At the principal working levels should suffice

5. Ionization type detectors are not suited for Steam Generator Area - should be thermal type.

6. Thermocouples are not required on cryogenic or high pressure gas storage vessels.

7. Delete requirement for ionization type and thermal detector type units in same room in Control Center (several locations where currently proposed).

8. Provide capability for local fire alarms to notify personnel in the area of fire origin.

Drawing No. B1-8413 Area Surveillance and Warning Block Diagram

Klaxon audible signal is reserved for criticality events only, and fire alarm signal must be tied into the audible siren signal.

ORIGINAL SIGNED BY

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