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EVALUATION OF THE ATOMICS INTERNATIONAL NUCLEAR DEVELOPMENT FIELD LABORATORY AS A LOCATION FOR REACTOR FACILITIES

AEC Research and Development Report

ATOMICS INTERNATIONAL
A DIVISION OF NORTH AMERICAN AVIATION, INC.

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EVALUATION OF THE ATOMICS INTERNATIONAL NUCLEAR DEVELOPMENT FIELD LABORATORY AS A LOCATION FOR REACTOR FACILITIES

General Editor
R. L. ASHLEY

Principal Contributors
R. L. ASHLEY
G. D. CARPENTER
E. R. LONG
R. S. LUBOMIRSKI
A. R. PICCOT
D. R. SHOEMAKER

ATOMICS INTERNATIONAL

A DIVISION OF NORTH AMERICAN AVIATION, INC.
P.O. BOX 309 CANOGA PARK, CALIFORNIA

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ABSTRACT

The Atomics International Nuclear Development Field Laboratory, located at Santa Susana, California, is evaluated with respect to its adequacy as a location for nuclear reactors. In order to adequately describe this site, much of the already existing site descriptive information is collected and additional data are provided, especially with regard to the meteorological and hydrological aspects.

The adequacy of the meteorological data which exist for the site is examined. It is concluded that, although meteorology does play a part in site development and facility hazards evaluations, the short distances between facilities and to the site boundary relegate diffusion and particularly the knowledge of diffusion parameters to a relatively unimportant consideration. As a result, it is not considered necessary to institute a meteorological measurements program.

The overall adequacy of the site is also examined. However, since criteria to judge the adequacy of a multiple (and independent) reactor facility site were not available, particularly considering on-site effects, a set of criteria which can be used in such an evaluation are developed. Considerations in this regard included: the establishment of limitations as to the initial dose as well as the dose due to residual effects, particularly ground contamination from fallout; the layout of facilities on the site, insofar as preventing interaction between facilities; and the location of adequate emergency assembly areas to be utilized in the event that evacuation is required.

In addition, a brief description of each reactor facility on the site is presented, together with information on the program presently being conducted and the future plans for use of the facility. The maximum credible accidents in each reactor facility are summarized and the radiological consequences thereof are re-evaluated using a consistent analytical model in order to place the results on a common basis.

By comparing the dose criteria which are established with the results obtained from the re-evaluation of the radiological aspects of the maximum credible accidents, including the interaction effects between facilities, it is concluded that the site is perfectly adequate to contain those reactors already in existence and those presently under construction.

Utilization of an alternate approach to develop reactor site criteria which, in effect, establishes design criteria for any future reactor facility to be located on the site, considering only the meteorological and environs population distribution aspects of the site (knowledge of the reactor design not being required), will provide assurance that the adequacy of the site will not be compromised by the installation of additional reactor facilities.

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UCLEAR DEVELOP MENT
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Looking Northeast

I. INTRODUCTION

Atomics International was requested by the Atomic Energy Commission to prepare a report presenting those aspects of its Santa Susana Nuclear Field Development Laboratory which are required to evaluate the overall suitability of the site for existing facilities and for future expansion. In addition, Atomics International was asked to provide information concerning the appropriateness of the meteorological data used in the hazards evaluations performed for the reactor facilities presently located on the site. This report provides this information.

In order to evaluate the adequacy of the site, it was necessary to establish a set of criteria by which the site could be judged. In developing these criteria, it was realized that the evaluation of a site containing multiple reactor facilities, and numerous other facilities as well, involves considerations in addition to those normally taken as potentially affecting the health and safety of the general public. An investigation of these additional considerations was undertaken and the necessary criteria developed.

One aspect of these criteria is the stipulation of limits for the prompt and delayed doses received in facilities adjacent to and/or the environs of the building in which a reactor suffers its maximum credible accident. In order to determine the radiological consequences that would result from the maximum credible accidents postulated to occur in the various reactors on site, the applicable hazards reports were reviewed. As could be expected, it was found that, subsequent to the preparation of the first hazards reports, both the meteorological data and the analytical methods used in determining the radiation exposures have changed to some extent, the change being evolutionary in nature and arising out of new developments and improvements in knowledge of these areas. As a result, it was necessary, for the purpose of this report, to re-evaluate the radiological consequences of the maximum credible accidents specified in the hazards reports. This was done primarily so that the results could be placed on a common basis, especially with regard to the microscopic data used in the calculations, the extent of fission product release from the reactor cores, and the methods by which the doses were evaluated. The results of this re-evaluation are included herein.

These results are then compared with the criteria which have been developed, and the overall adequacy of the site is established insofar as that aspect of the criteria is concerned. The other considerations relative to site adequacy, as established in this report, are also compared and appropriate conclusions drawn.

II. SITE INFORMATION

A. DESCRIPTION OF THE SITE

1. Location

The North American Aviation Field Laboratory is located in the southeastern portion of Ventura County, adjacent to the Los Angeles County line. The site is located about 29 miles from downtown Los Angeles. (See Figures II-1 and II-2.) Its distance from, and directional relationship to various surrounding populated communities is:

Canoga Park - 6 miles east-southeast

Chatsworth - 6 miles east-northeast

Woodland Hills - 8 miles southeast

Calabasas - 7 miles south

Thousand Oaks - 8.5 miles southwest

Simi - 5 miles northwest

Santa Susana - 3 miles north

Susana Knolles - 3 miles northeast

The site lies entirely within a pocket formed by the higher surrounding Simi Hills, thus affording relatively complete isolation of the Field Laboratory from direct sight of the various communities listed above. Its higher elevation, ranging from 800 to 1000 feet above the populated valley floors, serves further to enhance its isolation characteristics.

The main access road to the site originates in the San Fernando Valley near the communities of Chatsworth and Canoga Park. This road has two lanes and is paved throughout its length. Although grades are relatively steep in places, the road is designed to permit negotiation by all conceivable trucks and loads. This road has served as the access road for all construction activities since the inception of the Field Laboratory in 1948. An auxiliary paved road, adequate for passenger vehicles, leads from the site to the Simi Valley.

2. NAA Property

The North American Aviation site property includes both the Rocketdyne Propulsion Field Laboratory and the Atomics International Nuclear Development

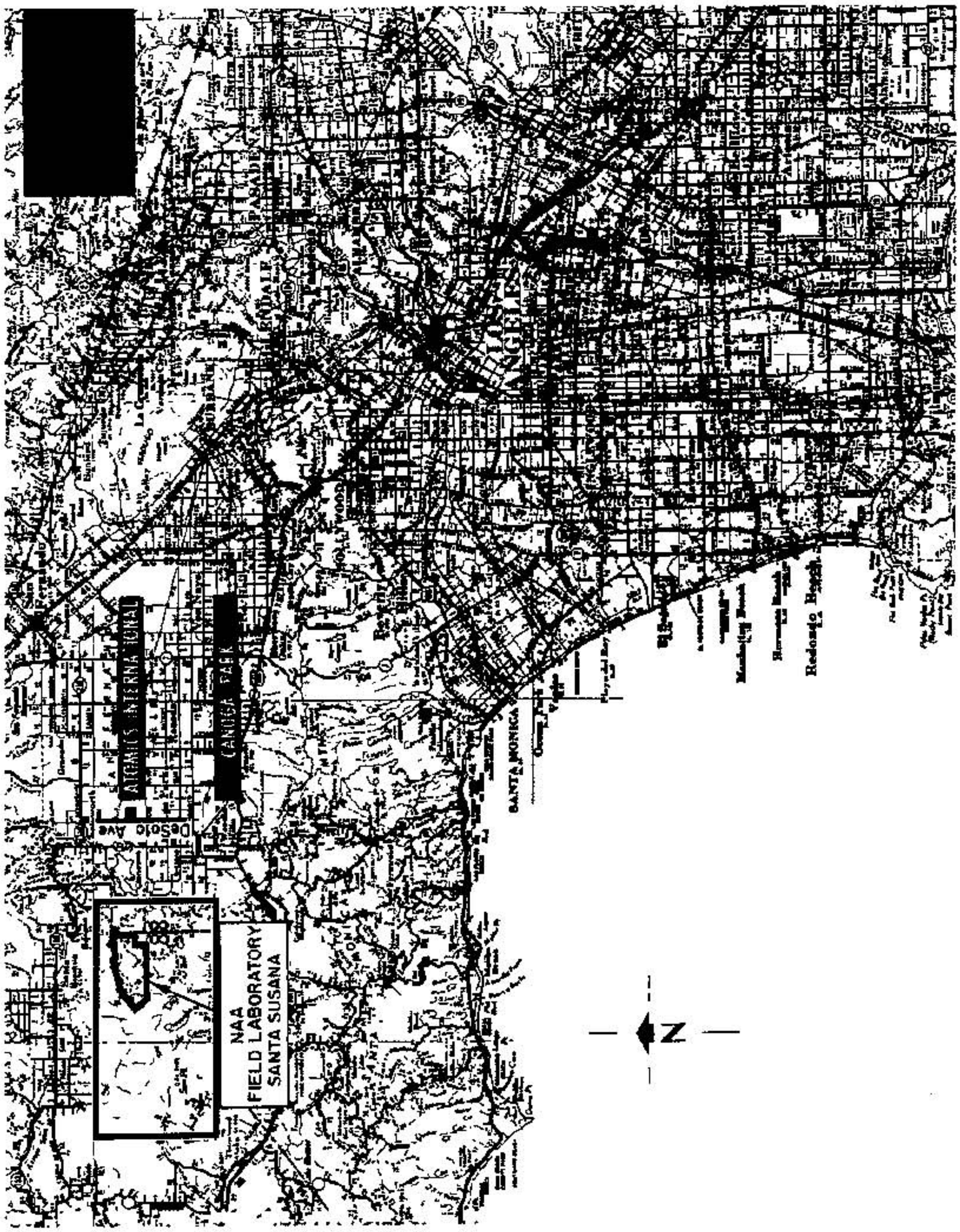

Figure II-1. Map of the General Los Angeles Area

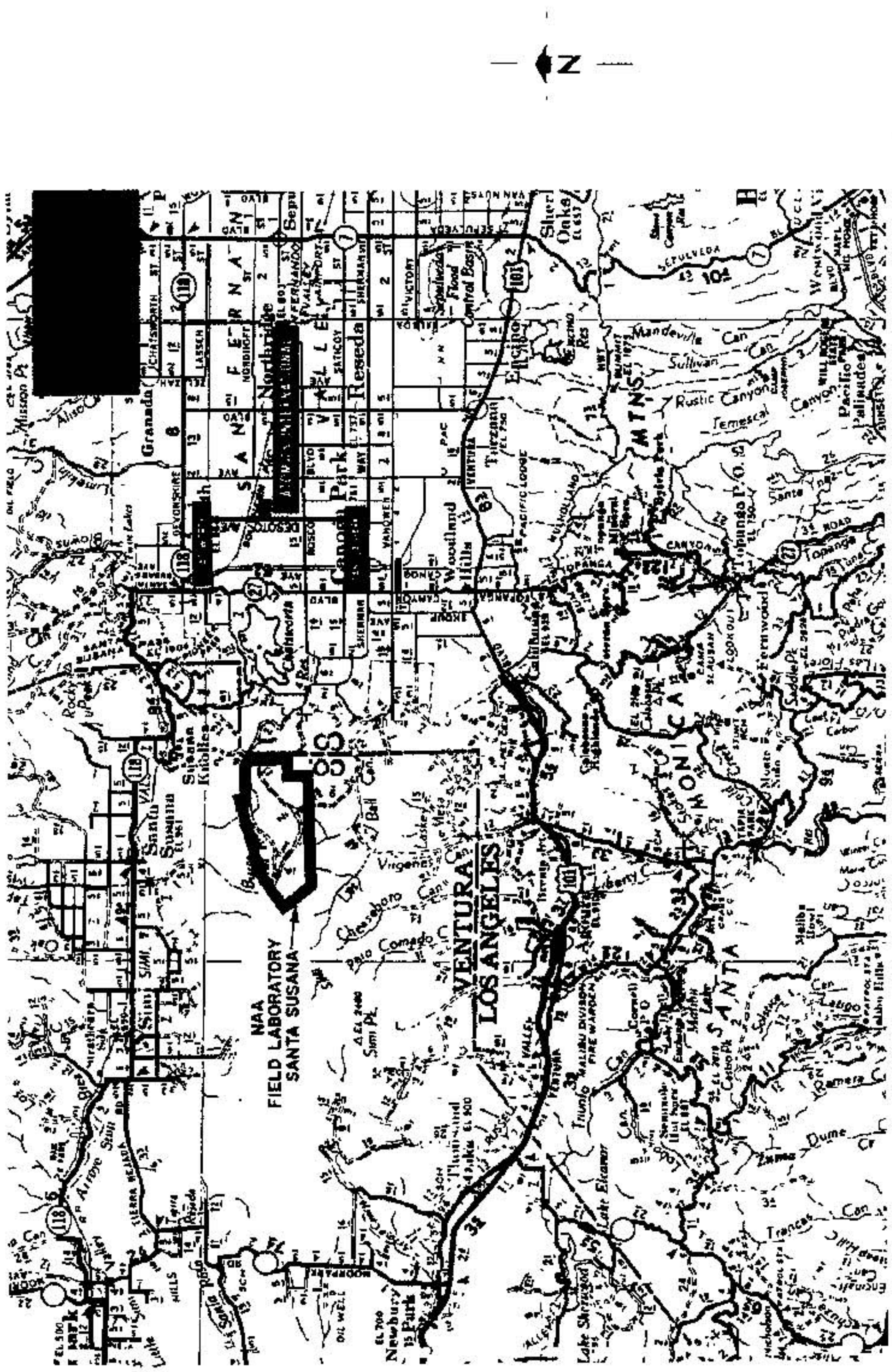

Figure II-2. Map of Site Environs

Field Laboratory. The total site encompasses approximately 1900 acres, of which about 290 acres are under the administrative control of Atomics International. Of this 290 acres, about 60 are presently under option by the Atomic Energy Commission.

The access roads enter the site at the extreme easterly edge of the Rocketdyne area. (See Figure II-22.) The primary internal road proceeds from the entrace security post in a westerly direction through the length of the Rocketdyne area to the Atomics International security station, Building 623. (See Figure II-3.)

Immediately adjacent to and east of the Atomics International entrance is the Rocketdyne Area II Administrative Support area. Fire control and emergency services are located here to serve the Rocketdyne Area II and, if necessary, to support Atomics International's emergency forces.

The Rocketdyne Propulsion Field Laboratory is divided into two major test areas. Each contains engineering, administration, and supporting activities for various test stand complexes associated with them. These test stands are isolated, either singly or in small groups, from other activities by intervening terrain and distance. This isolation is increased in relation to Atomics International's Nuclear Development Field Laboratory.

3. Nuclear Development Field Laboratory Buildings and Layout

Figure II-3 illustrates the layout of the Nuclear Development Field Laboratory. The following is a brief description of some of the significant buildings on the AI site; for the most part, this description does not include the reactor buildings, descriptions of which are included elsewhere in this report.

Building 040, the Atomics International Control Center, contains the Nuclear Development Field Laboratory headquarters for plant protection, fire department, health and safety, and emergency medical treatment. This building is equipped with the master control panel of the criticality, radiation, and fire alarm systems which permits automatic detection of these and other hazards at the Control Center. Other detection alarms which register in the Control Center include intrusion, smoke, and certain water levels important to maintaining safe operation of the facility.

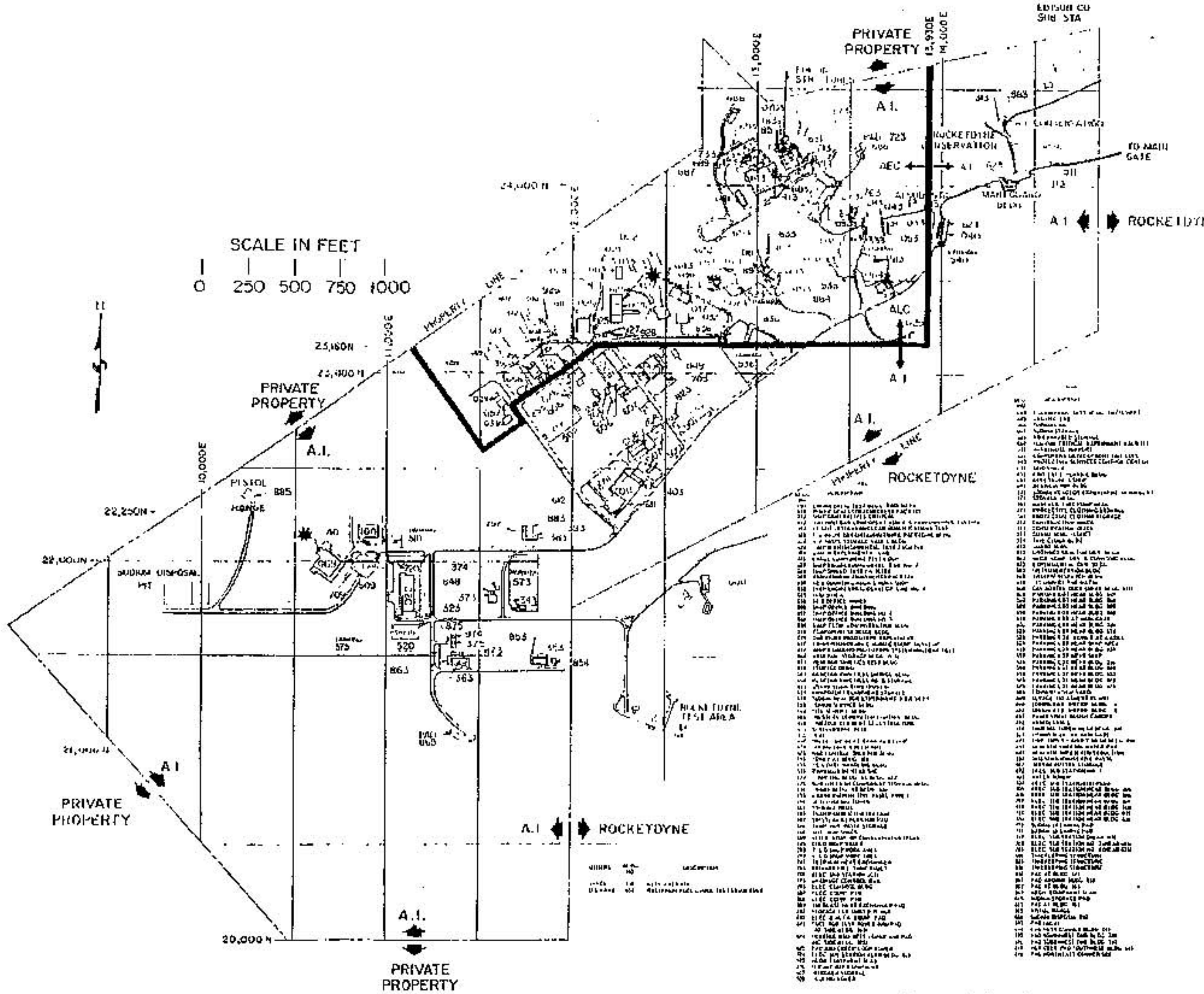

Figure II-3. Layout of Atomics International Nuclear Development Field Laboratory

Building 064, Source and Special Nuclear Materials Storage, contains a concrete shielded vault for safekeeping of most of the nuclear materials not in use at the site. This building is surrounded by an exclusion fence to prevent entry by unauthorized persons.

Building 029, directly south of Building 064, is the Radiation Measurements Facility, used for calibrating instrumentation. Rock barriers surrounding the building act as natural shielding from stray background radiation which might originate from other operating buildings in the vicinity.

The SNAP complex (comprising about 17 acres) is due west of Building 064. Buildings 036 and 037 are offices providing space for engineering support of the eastern portion of the SNAP area. Other SNAP non-nuclear buildings comprising this eastern portion are 032 and 027, where mechanical, thermal, and vibration tests are performed on reactor components; and 025, an experimental laboratory which contains various mock-ups preceding in-cell work.

The radioactive waste storage facility is located north and west of the area described above, in Buildings 021 and 022. These buildings contain packaging and decontamination areas and radioactive waste storage vaults.

Buildings 005 and 006, to the south and east of the SNAP area, are the Organics and Sodium Laboratories, respectively. These buildings are company-owned-and-operated, as distinguished from the SNAP buildings, which are Government funded.

West of Building 006 are various test loops, fuel and control rod test buildings, and a shell-side heat transfer test structure. These are in general support of the Sodium Graphite Reactor program.

The non-nuclear buildings in the western portion of the SNAP complex are 013, 057, 038, and 039. Building 013 is a components assembly and performance testing area for the various assembled parts of the reactor systems. Building 057 is used for the development of ground handling equipment for operations at the launch site. Building 038 is an office building for engineering support of the westerly portion of the SNAP complex, and Building 039 houses the AEC technical staff administrating the buildings presently under or pending construction.

Building 011, southeast of the SNAP area, contains the Atomics International general warehouse, shop, transportation, and administrative support for the Field Laboratory.

Continuing westerly along the main road are Buildings 020, 009, and 100. Buildings 009 and 100 contain critical assemblies. Building 020 is known as the Component Development Hot Cell. In it are located four large (juxtaposed) hot cells equipped with viewing windows and remote manipulators operated from a common gallery. In addition, the building contains a well-equipped machine shop for handling radioactive materials.

East and southeast of Building 020 are various small development buildings, test loops, and fuel and control rod test structures. The personnel population density in this area is extremely sparse compared to that for the other portions of the site.

Due east of this group is the Atomics International Sewage Disposal Plant, Building 600, which is located on Rocketdyne property.

The daytime population of the major buildings on the AI portion of the site is shown in Table II-1. At present there are about 600 personnel on the site during normal daytime shift hours.

4. Site Topography

The North American Aviation Field Laboratory site is situated in rugged terrain typical of that usually found in mountain areas of relatively recent geologic age. The entire terrain is generously sprinkled with exposed outcroppings rising above the more level patches on which most of the facilities are located. The Rocketdyne test stands occupy some of the higher and more remote outcrops which exist on the site.

The overall site may be described as an irregular plateau with eroded gullies at the perimeters. Elevations of the general site vary from 1800 to 2100 feet above sea level, with extremes of 1650 feet and 2250 feet. Figure II-4 indicates the rugged nature of the terrain, as the 25-foot contour intervals are very close to each other.

The Atomics International site, as distinguished from the Rocketdyne area, is located on the relatively level terrain known as Burro Flats. The surface terrain is partly mantle consisting of sand and clay soil, with a light

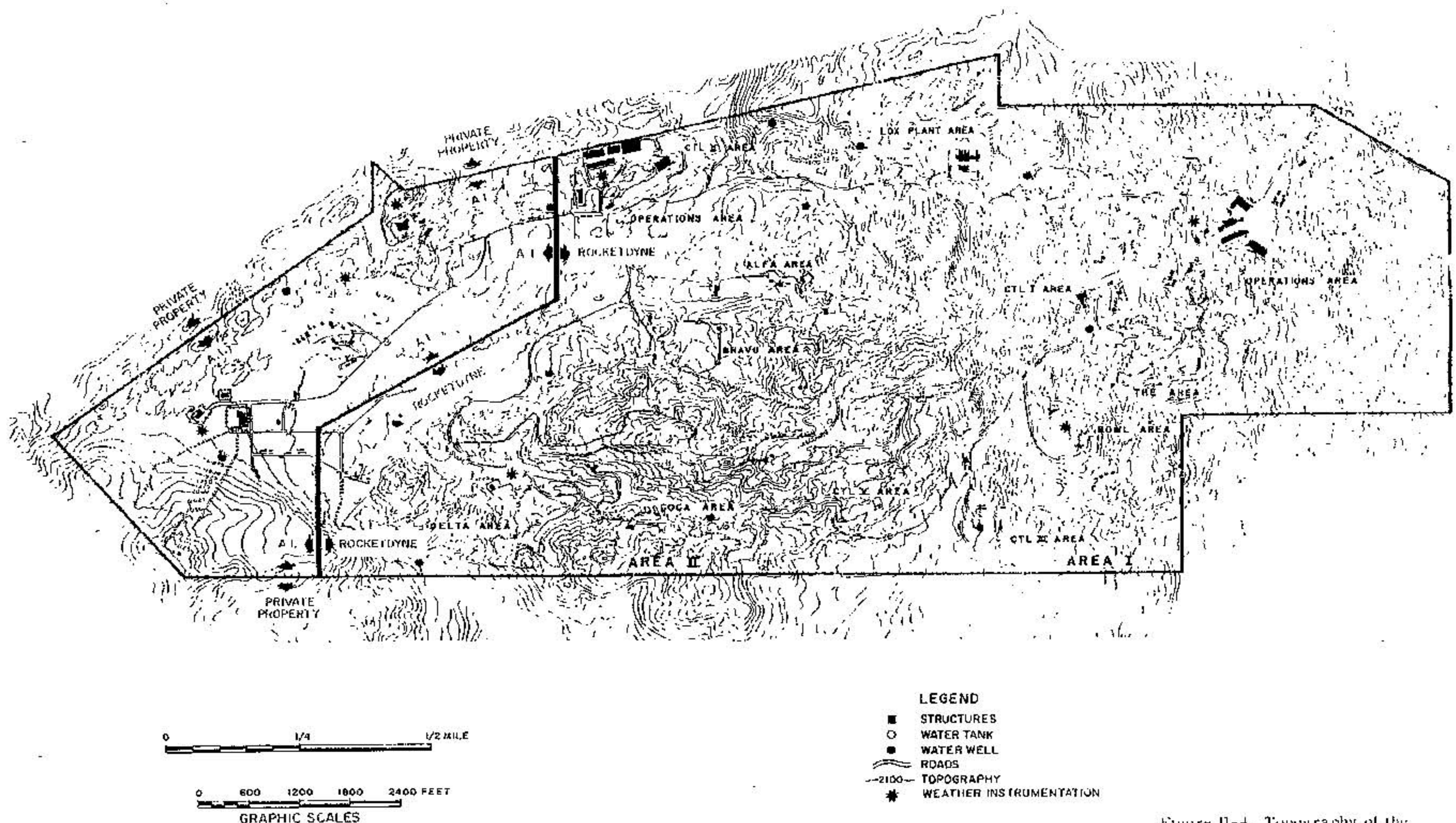

Figure II-4. Topography of the North American Aviation, Inc., Field Test Laboratory

TABLE II-1

NORMAL DAYTIME POPULATION OF BUILDINGS AT THE AI NUCLEAR FIELD DEVELOPMENT LABORATORY

Building Number	Building Name	Population
003	Engineering Test Bldg.	20
005	Organic Laboratory	32
006	Sodium Laboratory	30
009	SGR-OMR Critical Experiments Laboratory	15
010	SNAP-8 Experimental Reactor	5
011	Warehouse and offices	38
012*	SNAP Generalized Critical Bldg.	6
013	Non-Nuclear Component Assembly and Performance Test Bldg.	6
019*	Flight Systems Nuclear Test Bldg.	20
020	Components Development Hot Cell	26
021	R/A Waste Decontamination and Office Bldg.	6
022	R/A Waste Storage Vault	6
024	SNAP Environmental Test Facility	10
025	SNAP Experimental Laboratory	15
026	Large Component Test Loop	7
027	Non-Nuclear Vibration and Shock Test Bldg.	12
028	Shield Test Experiment Facility	5
030	AE-6 Counting Room and Workshop	3
032	Thermal and Vacuum Environmental Test Bldg.	12
035	AE-6 Office Annex	15
036	SNAP Office Bldg.	45
037	SNAP Office Bldg. No. 2	44
038	SNAP Office Bldg. No. 3	40
039	AEC Office Bldg.	5
040	Protective Services Control Bldg.	15
041	SRE Component Storage Bldg.	2
049	OMR Prototype Experiment	3
056*	Flight System Test Bldg.	12(est.)
057	Launch Handling and Mobile Equipment Development Bldg.	7
059*	SNAP-8 Ground Prototype System Test Bldg. (S8DS)	12
073	Reactor Kinetics Test Bldg.	6
083	Reactor Kinetics Control Bldg.	2
093	AE-6 Reactor Building	12
100	Epithermal Critical Experiment Laboratory	15
143	Sodium Reactor Experiment	30
153	Sodium Service Bldg.	2
163	Site Service Bldg.	15
343	Guard Building	3
353	Organic Reactor Development Bldg.	10
356	HNPF Fuel Handling Test Facility	3
357	HNPF Sodium Pump Loop	3
363	Mechanical Component Development Bldg.	10
373	Experimental Development Bldg.	16
375	OMR Control Shelter Bldg.	5
383	Instrumentation Bldg.	18
453	AE-6 Fuel Handling Bldg.	2
623	Guard Bldg.	1

*Presently under construction or proposed as of May 15, 1962.

covering of xerophytic mesquite and grasses. The unused ravine areas have some stands of scrub live oaks.

B. POPULATION DISTRIBUTION AND PROJECTION

1. Distribution

The greatest population density in areas contiguous to the Field Laboratory occurs in the West San Fernando Valley, particularly in the communities of Canoga Park, Chatsworth, and Woodland Hills. (See Figure II-5 and II-6.) None of these particular communities is closer than 6 miles from the test site. Other communities of much lower population density, but also contiguous to the site, are Simi and Santa Susana, located in the Simi Valley. The nearest of these is the town of Santa Susana, located about 3 miles from the Atomics International portion of the site.

All communities and scattered residential areas in the vicinity are separated from the Field Laboratory by an expanse of hills and rugged terrain, as well as by distance.

2. Population Growth and Trends*

Accessible areas in the San Fernando Valley have marked potential growth capabilities. Contributing to the growth trends are the present industrialization of the west valley, causing a buildup of labor pools and the overflow of population from more densely settled areas.

Ventura County's potentially habitable regions in the general vicinity of the Field Laboratory, including the Simi Valley, are also indicated as rapidly growing areas. Although lagging behind the San Fernando Valley growth trend, the demographic forecast is for saturation within 50 years.

The projected ultimate populations for the various communities in close proximity to the Field Laboratory are as follows:

a) San Fernando Valley (total): This entire area covers approximately 130,000 acres. Assuming a 65% residential use and an ultimate population density of 22 persons per acre, the estimated ultimate population is 1,859,000 persons.

*All information concerning population has been derived from figures released by city and county planning commissions.

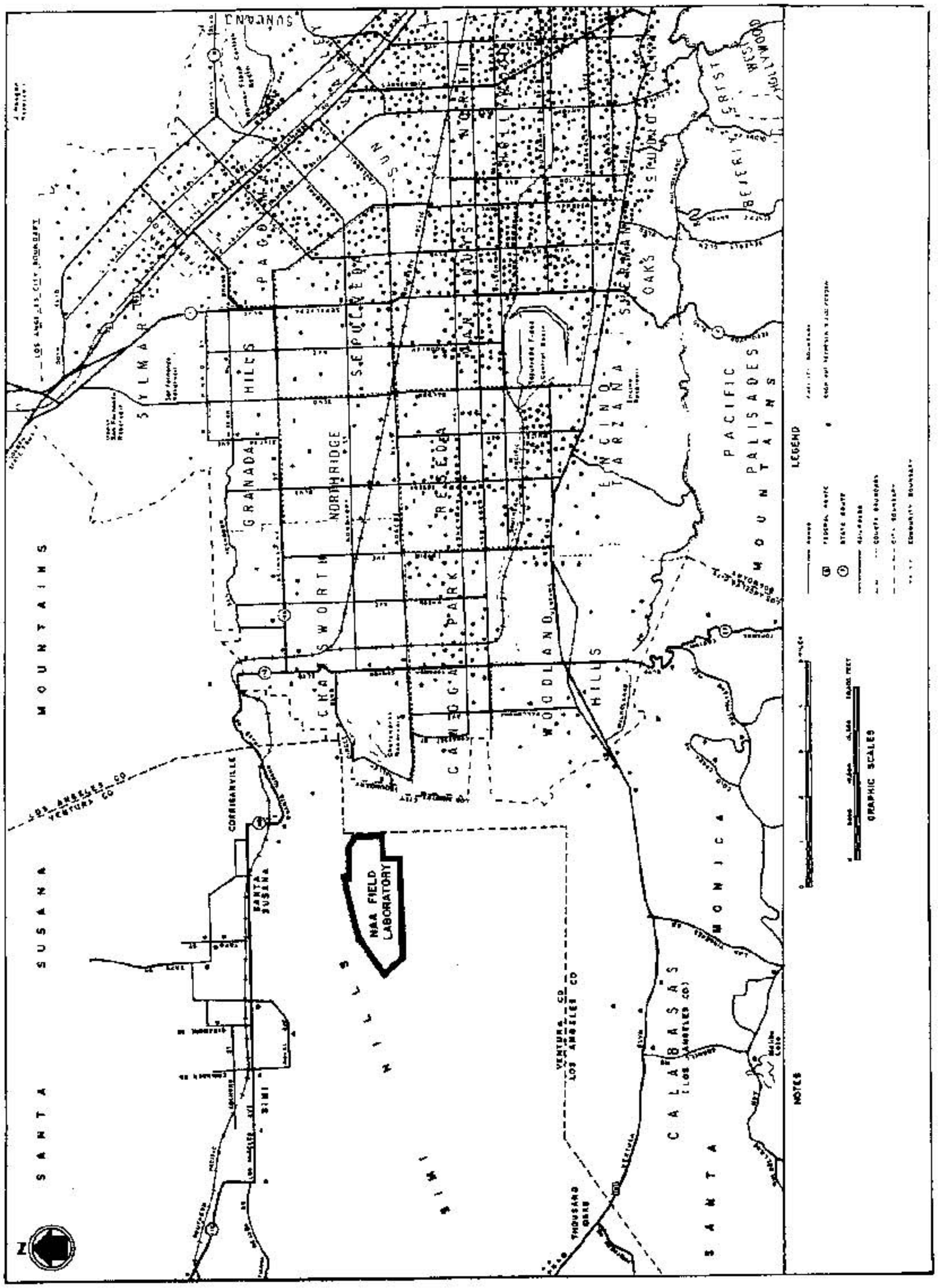

Figure II-5. Population Density for Site Environs - 1950

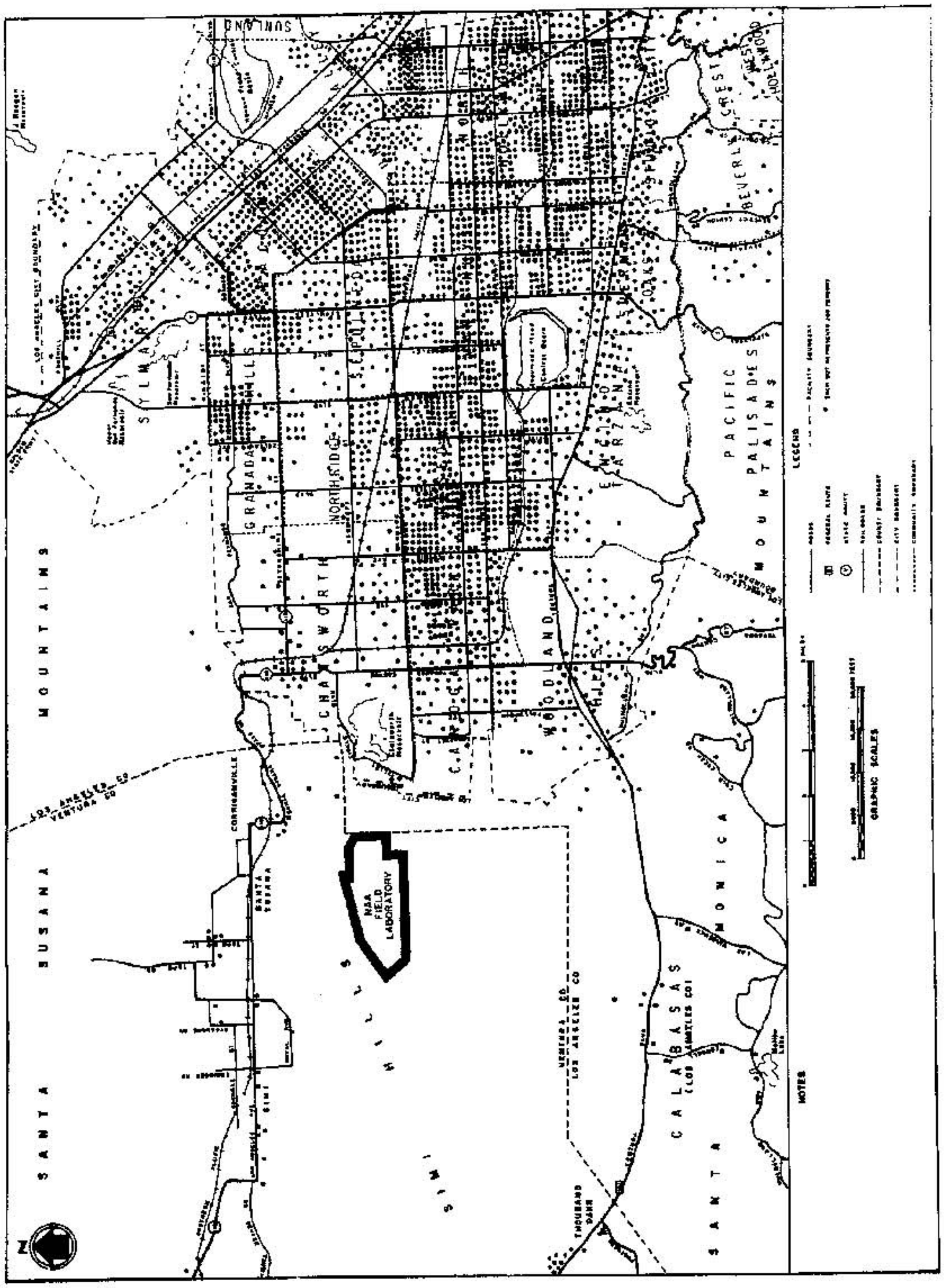

Figure II-6. Population Density for Site Environs - 1957

1) Canoga Park: The approximate area is 7,500 acres. Assuming high residential use of 80% and a density of 22 persons per acre, the ultimate population is estimated at 132,000.

2) Woodland Hills: The approximate area is 6,600 acres, with only about 70% suitable for residential occupancy because of the character of the terrain. Larger, more expensive homesites reduce the ultimate population density to 17 persons per acre, resulting in an estimated ultimate population of 78,540.

3) Chatsworth District: The district covers approximately 9,600 acreas with a relatively large percentage of unsuitable terrain and some acreage reserved for industrial development. The assumed residential development and density are 65% and 20 persons per acre, resulting in an estimated ultimate population of 124,800.

b) Calabasas Area: This area is located west of the community of Woodland Hills and southwest of the Field Laboratory site. The area comprises approximately 62,800 acreas with some rugged terrain among the gently rolling hills. Assuming about 30% suitable for residential occupancy, and typically suburban living conditions with a density of 10 persons per acre, the estimated ultimate population is 188,400 persons.

c) Simi Area: This area includes the entire Simi Valley as well as contiguous canyon areas, which, to some degree, are also suitable for habitation. The estimated population by the year 2000 is 70,000. This assumes a quasi-suburban development maintained to that year.

Population growth for the areas mentioned above are shown in Figures II-5, II-6, and II-7.

3. Population Encroachment

As population saturation is approached in various areas, it is anticipated that some remote, less desirable regions may be opened to development. However, the limits of encroachment, as predicted for the years 1970 and 1980, are still separated from the Field Laboratory by the rugged terrain that surrounds it.

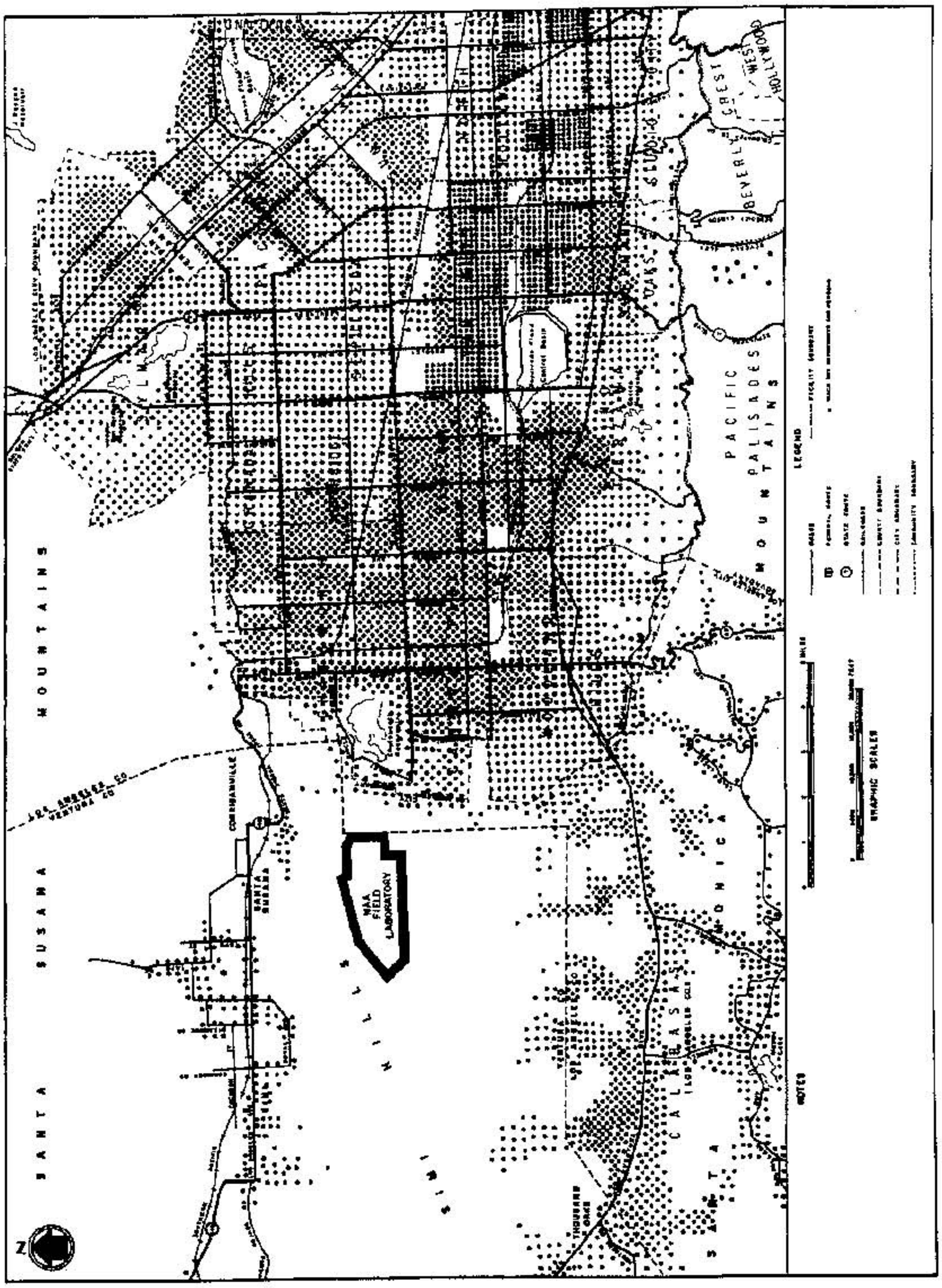

Figure II-7. Projected Population Density for Site Environs - 1980

Figure II-8 outlines the existing and predicted encroachment limits for the areas contiguous to the Field Laboratory.

C. METEOROLOGY AND CLIMATOLOGY

1. General Area Climatology

The Los Angeles basin in a semi-arid region, controlled for the most part by the semi-permanent Pacific high-pressure cell. The seasonal changes in the position of this cell influence to a great extent the weather conditions over the area. The Pacific high-pressure cell extends from Hawaii to the Southern California coast, with the associated subsidence inversion tilting downward from Hawaii to the California coastline. During the summer season, the high is displaced to the north, resulting in mostly clear skies with little precipitation. In the winter, the Pacific high moves south far enough to allow some of the Pacific lows and their associated fronts to move through the area, producing light to moderate precipitation. Due to off-shore pressure gradients and frontal passages, the winds are predominantly from a northerly or northwesterly direction.

a. Location of Weather Stations

The U.S. Weather Bureau takes complete surface observations at three locations in the Los Angeles area: The City Office in downtown Los Angeles, 29 miles southeast of the Atomics International Nuclear Development Field Laboratory; the Los Angeles International Airport, 28 miles southeast of the site; and the Lockheed Air Terminal in Burbank, 22 miles east of the site. The Burbank station is the nearest to the site but has an elevation of 700 feet, which is about 1100 feet below that of the site.

As part of the Rocketdyne Division's program, a network of three stations, each providing continuous recordings of wind speed and direction, has been established at the North American Aviation Field Laboratory. The location of these stations is shown on Figure II-4. In addition to the Rocketdyne instruments, there are a number of similar instruments located at various of the Atomics International facilities (Buildings 009, 024, and 143). (See also Figure II-4.)

b. Site Temperatures and Precipitation

The North American Aviation Field Laboratory is located in the Simi Hills, a low, narrow range separating the Simi and San Fernando Valleys.

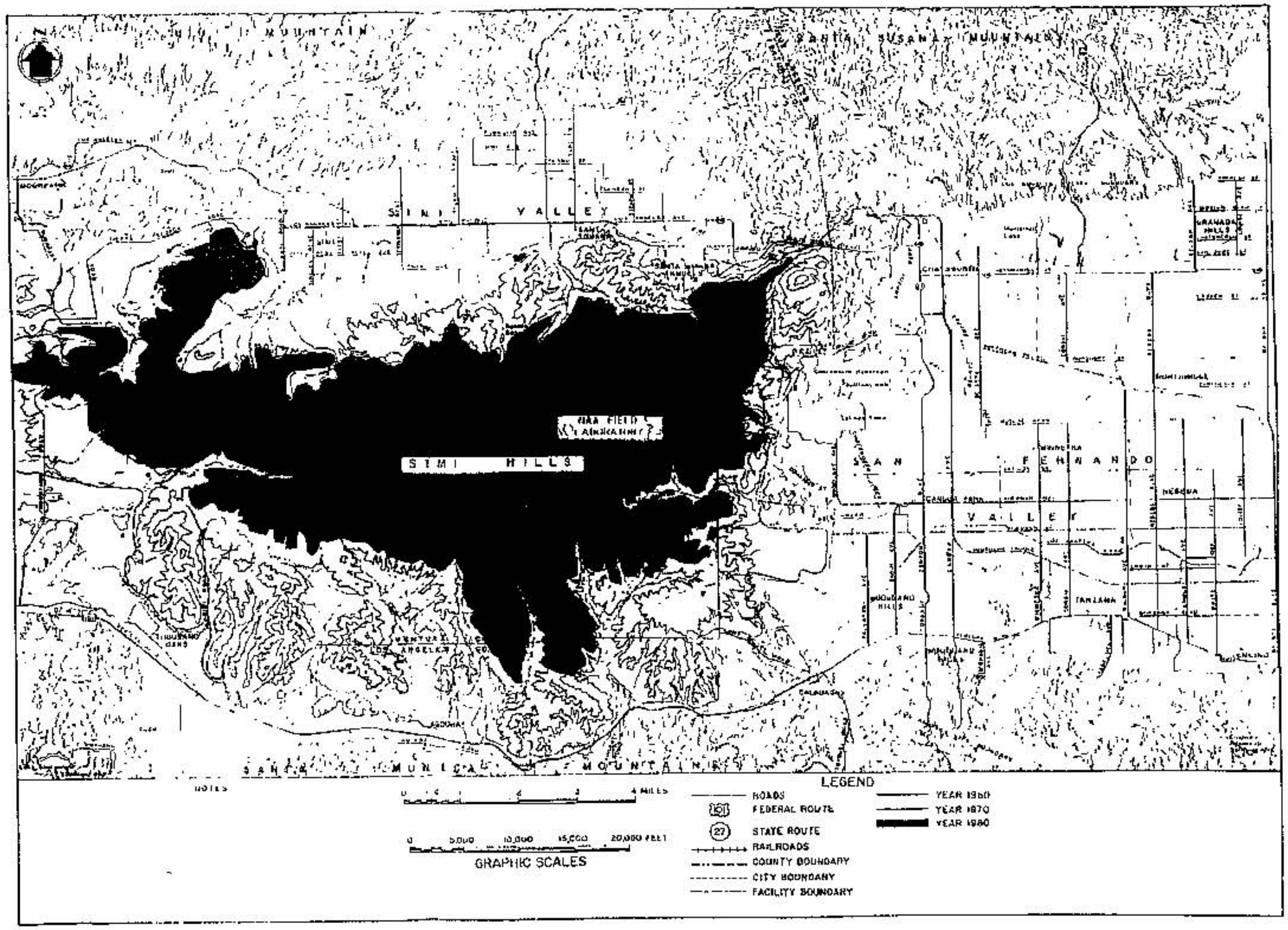

Figure II-8. Population Encroachment Limits

The low altitude and ocean influence make for a relatively mild climate throughout the year. In addition, the elevation of the site, averaging about 1800 feet M. S. L. and about 1000 feet above the surrounding valleys, provides a further moderating influence on the temperature regime. In general, minimum temperatures are higher in winter and maximum temperatures lower in summer than those at Burbank.

For approximately 30 days of the year, temperatures at the site are above 90°F, with an annual maximum of 102°F, while about 3 days each year show temperatures below 32°F, with an annual minimum of about 29°F. In a 50 year period, it is anticipated that the maximum might rise to 109°F and the minimum drop to 18°F.

The precipitation is extremely variable, with many years having less than half the normal amount. Similarly, most years have no snowfall at all, the occasional year with a moderate fall accounting for the measurable average. The statistics for average and heaviest precipitation are given in Table II-2.

TABLE II-2

MEANS AND EXTREMES IN PRECIPITATION FOR PERIOD OF RECORD

	Mean (in.)	Heaviest (50-year probable) (in.)
Annual	16.0	40
Summer half yr (May-Oct)	1.0	7
Winter half yr (Nov-Apr)	15.0	38
Rainiest month (Feb)	3.5	15
Rainiest 24 hr in 50 years	-	8.5
Annual snow	1.0	12
Maximum 24-hr snow	-	10
Number of days with 0.01 in. or more	40	-

2. Site Meteorology

Observations at and in the vicinity of the site, combined with related data from other areas, indicate average surface wind conditions as shown in Table II-3.

TABLE II-3

SURFACE WIND CONDITIONS AT THE SITE

	Summer	Winter
Prevailing afternoon direction	WNW	NW
Prevailing early morining direction	ESE	ESE
Average daytime speed	8 mph	6 mph
Average nighttime speed	3 mph	3 mph
Maximum wind (1-minute average)	30 mph at least once per yr. 55 mph once in 50 yr.	

During the early morning, the surface wind passes over Burro Flats into the Simi Valley. In the afternoon, the wind reverses and is directed generally toward the San Fernando Valley.

Upper wind flow (above 4000 feet) plays little or no role in the transport of any effluent from the site. This is due to the fact that, although some incidents might be accompanied by thermal effects, none can be postulated which would provide enough heat to cause cloud rises more than 1500 feet above ground. Information on the upper wind pattern, therefore, is presented only to provide a more complete climatological picture of the area. These upper winds are controlled almost entirely by the large- or intermediate-scale pressure systems and do not vary much over a few miles of distance. On this basis, the upper winds above Burbank may be presumed to be quite similar to the upper winds above the site. Typical upper wind data from the Burbank station are shown graphically in Figure II-9. The estimated upper winds above the site are given in Table II-4.

Information obtained from the Rocketdyne station located in the Area II Administrative Support Area (just north of the Area II Operations Area - See Figure II-4) was used to prepare the data shown in Figure II-10, which represents

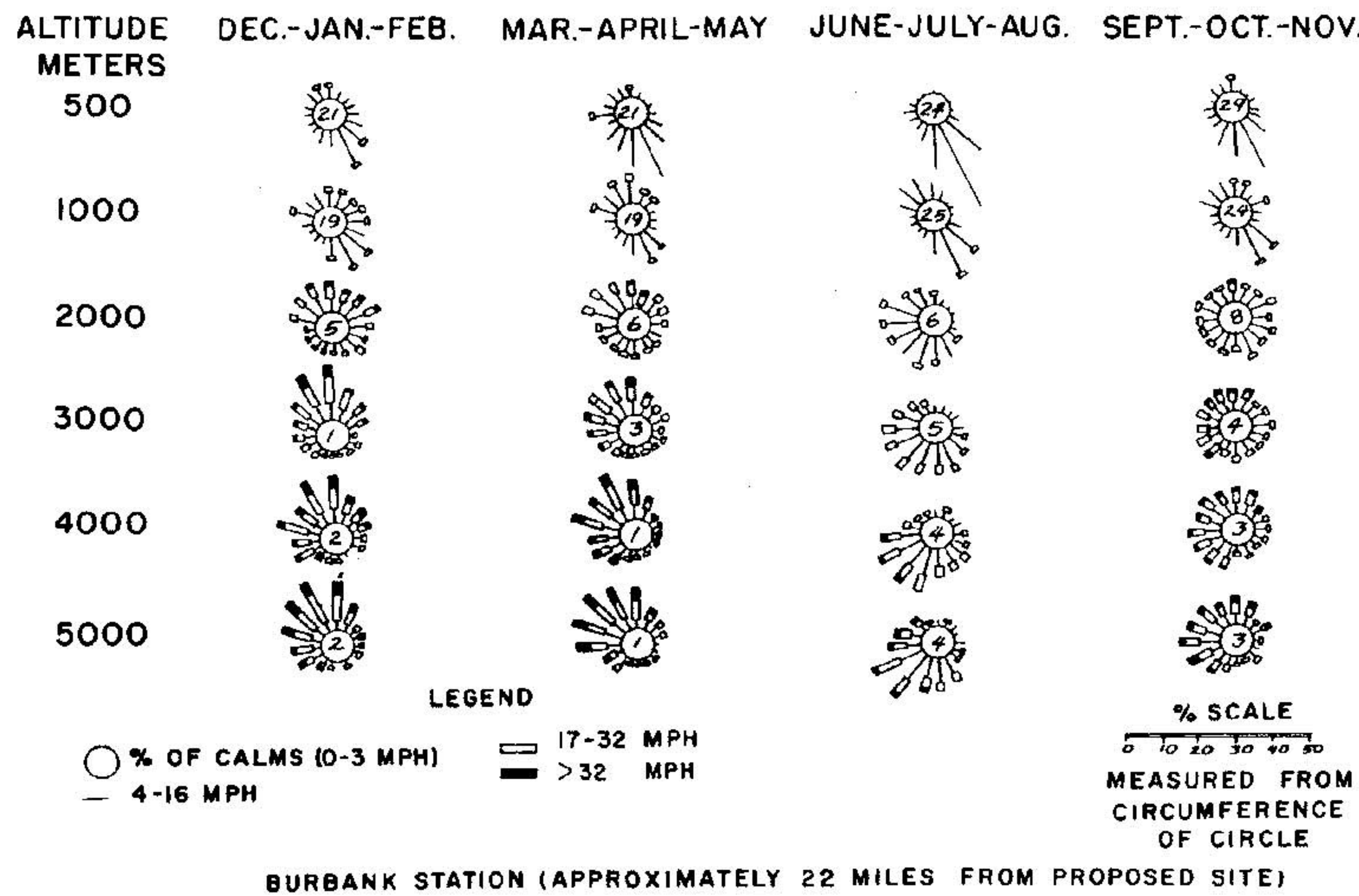

Figure II-9. Typical Upper Wind Data (Burbank Station)

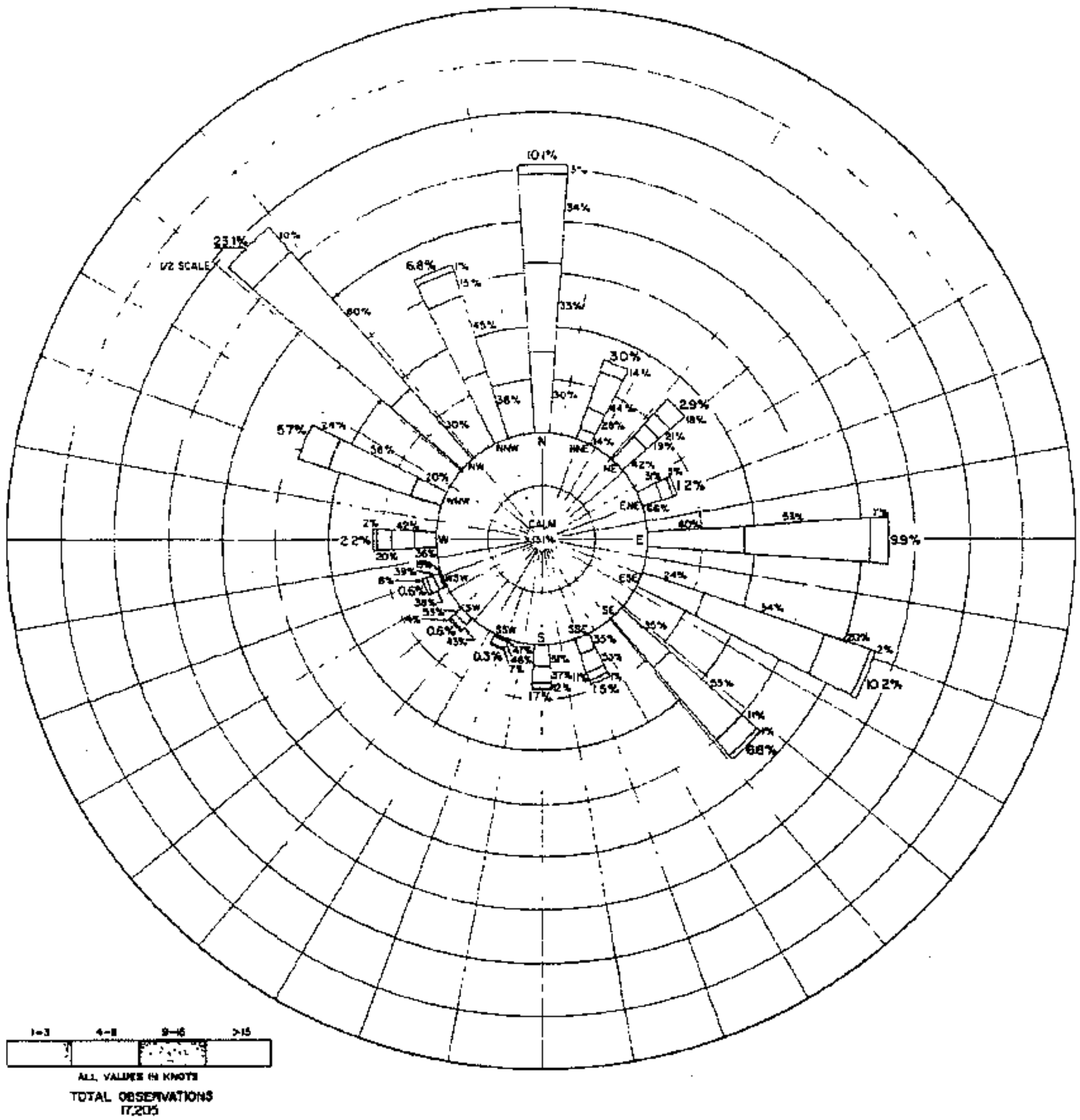

Figure II-10. Annual Surface Wind Rose (1960-1961), Rocketdyne Field Test Laboratory

TABLE II-4

UPPER WIND CONDITIONS AT THE SITE

	Summer			Winter		
Elevation (ft)	3250	9750	16,500	3250	9750	16,500
Prevailing direction	SSE	SW	SW	N	NW	NNW
Average speed in prevailing direction (mph)	5	12	15	5	15	20

the annual surface wind rose for the area. In addition, the Rocketdyne data have also been reduced to provide information as to seasonal variations, the results of which are shown for the summer and winter seasons in Figures II-11 and II-12. These wind rose data depict in greater detail the wind patterns which were summarized in Table II-3.

In addition, a study was recently performed to determine how well the wind data from Atomics International wind instruments would correlate with that obtained at Rocketdyne. Although a detailed and lengthy statistical analysis was not performed, it was evident that very good correlation existed.* This leads to the conclusion that the local AI site wind patterns are not substantially different from those indicated by the Rocketdyne instrumentation and that, therefore, the surface wind roses shown in Figure II-10 through II-12 are valid for the AI site. From a theoretical meteorological standpoint, this correlation must exist, since the flow patterns across the Simi Hills are governed by temperatures and pressure factors of a scale considerably larger than that dictated by local site topography.

Although a subsidence inversion is present in the surrounding valleys almost every day during the summer months, and frequently in other months, the elevation of the site is several hundred feet greater than the height of the average inversion base that exists over the populated areas in the valleys below.

*The original comparison of data indicated good correlation between all data except that obtained from the SRE. Subsequent investigation showed that the SRE instrument was improperly connected. When the data were corrected to account for the fact that the transmitter was out of phase with the recorder, good agreement was obtained with the data from the other instruments.

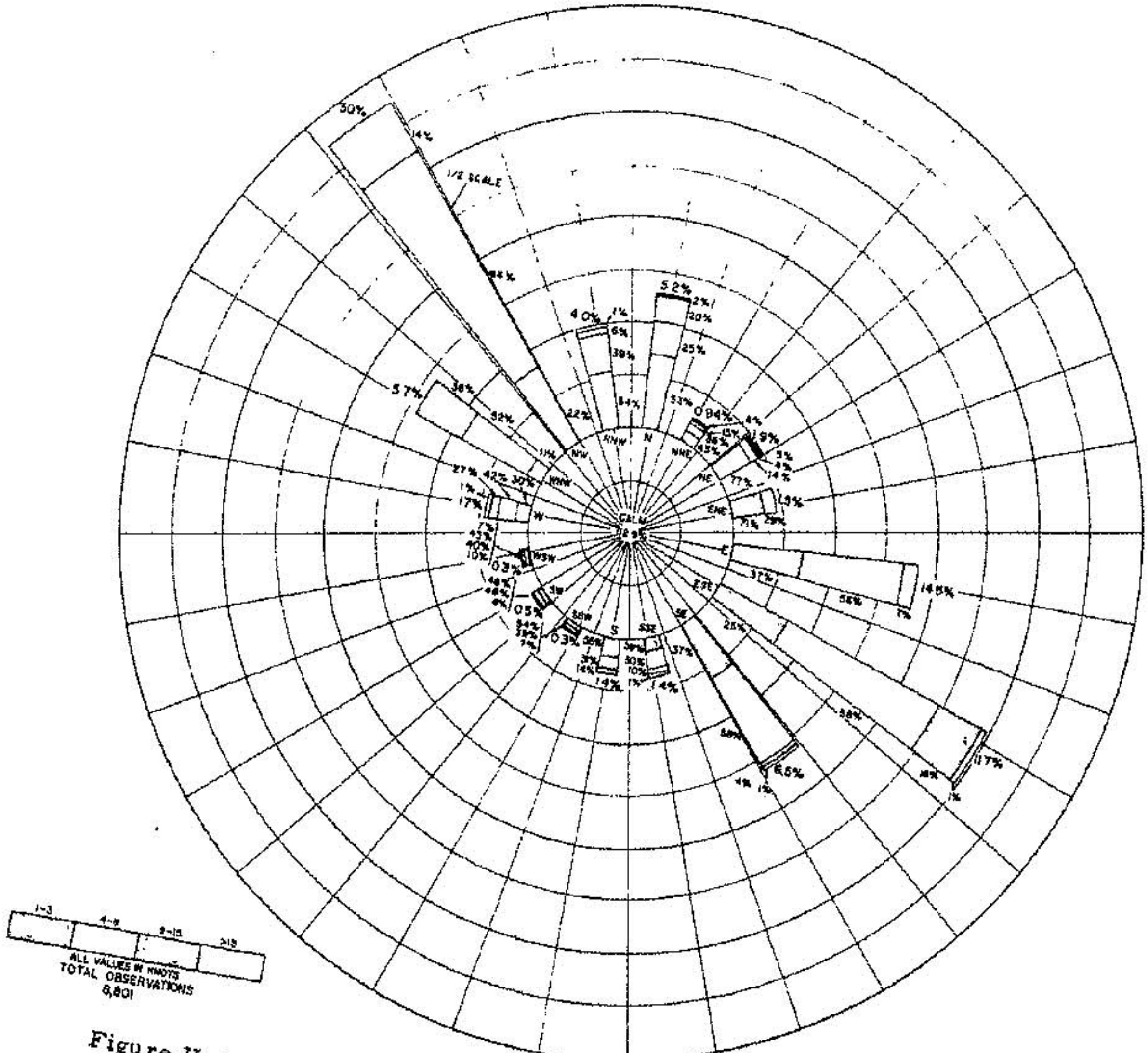

Figure II-11. Seasonal Surface Wind Rose (Summer 1960-1961), Rocketdyne Field Test Laboratory

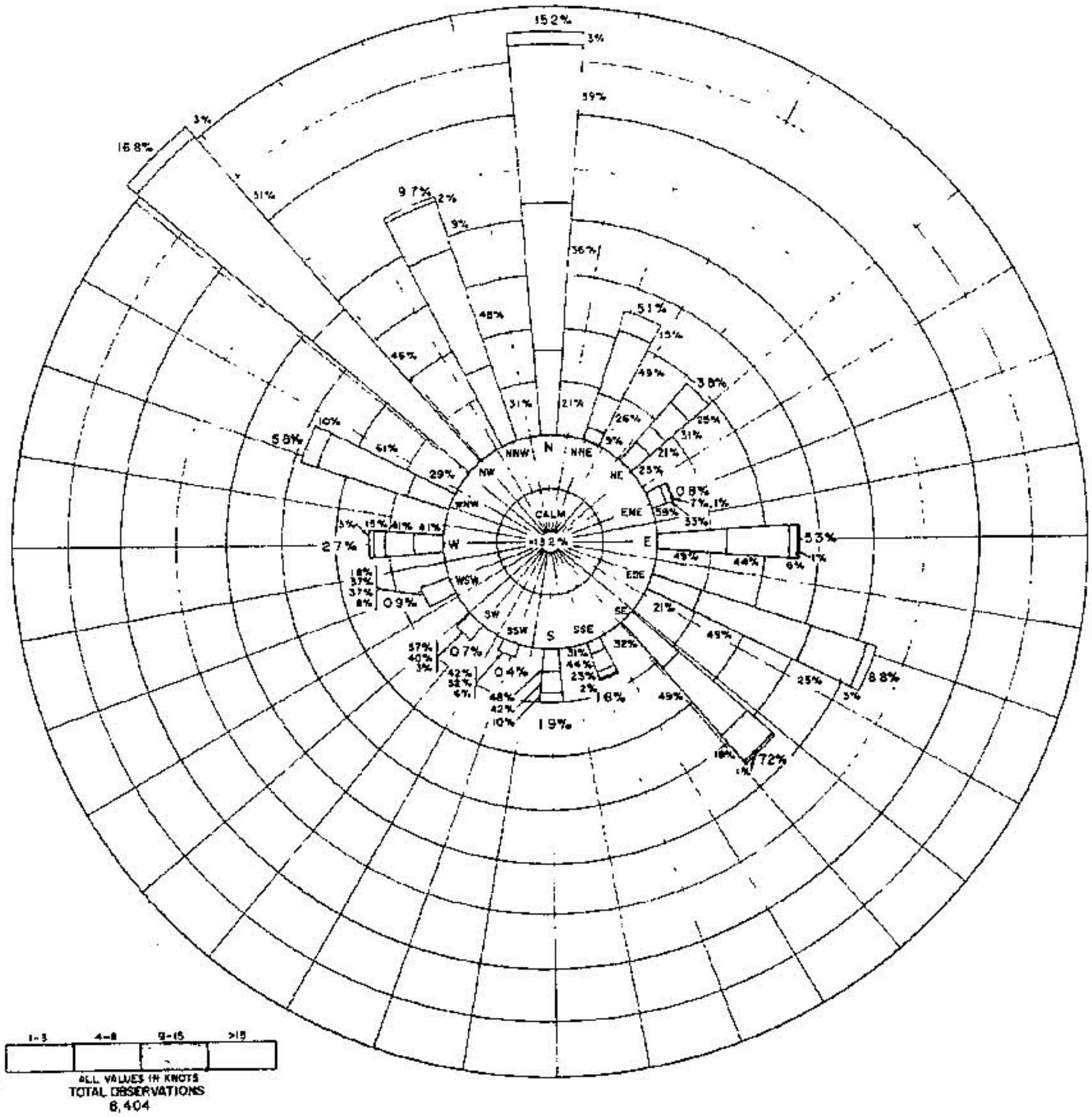

Figure II-12. Seasonal Surface Wind Rose (Winter 1960-1961), Rocketdyne Field Test Laboratory

This places the site usually within or above the inversion layer. Ground inversions are produced by radiation cooling during clear nights in fall, winter, and spring and are stronger in the wind-sheltered valleys than on the relatively exposed hills. Although radiation inversions generally are associated with a tendency for the cooled air to drain downward from the slopes of hills, downward motion from this site is negligible. Under calm, clear, nighttime conditions, the air at the site is always potentially warmer than the air at the valley floor; i.e., compressional heating upon descent would not allow much penetration of the cool, low-lying air layer. During the day, the diurnal heating tends to establish a stirred layer with near adiabatic lapse rate immediately above the hills.

The site experiences clear days and nights throughout most of the summer. In other seasons, particularly spring, when the incidence of inversions is high, stratus is present over the site. Spring and fall are periods of greatest frequency of fog, resulting in stratus clouds at the elevation of the site. During winter, higher clouds associated with large-scale weather activity are frequent. Table II-5 presents an estimate of the average number of days per month which are clear, partly cloudy, or cloudy during the daytime at the site.

TABLE II-5

AVERAGE OVERCAST CONDITIONS AT THE SITE

Month	Number of Days per Month		
	Clear	Partly Cloudy	Cloudy
January	15	8	8
March	14	9	8
May	13	11	7
July	25	5	1
September	22	7	1
November	19	8	3

3. Factors Governing Effluent Cloud Paths

Any effluent released into the atmosphere at the site will follow a course dictated by the wind flow over the site. Idealized flow over an obstruction such as the Simi Hills Range, considering only fluid dynamics, is shown in Figure II-13. This indicates that the downward penetration of a cloud into the valley is dependent upon the streamline which the cloud is forced to follow (i.e., the height above the site at which vertical equilibrium is established). From this consideration, it appears that a hot cloud, which achieves a rise due to thermal buoyancy, presents a less serious problem than a cold cloud. However, flow over the Simi Hills is not governed strictly by laminar fluid dynamics. Local meteorological conditions, for the most part, tend to inhibit the downward component of the wind toward the valley floors and consequently reduce the number of potentially unfavorable situations. This alternation of the idealized flow path is accomplished in a number of ways.

a. Wind Direction and Turbulence

The northwest and west-northwest winds at Burro Flats are primarily a flow from the Simi Valley extending into the western San Fernando Valley, caused both by the initial momentum of the valley wind and by the pressure gradients set up by more rapid heating of the San Fernando Valley than the Simi Valley. This flow has depth, and its movement across the Simi Hills can be

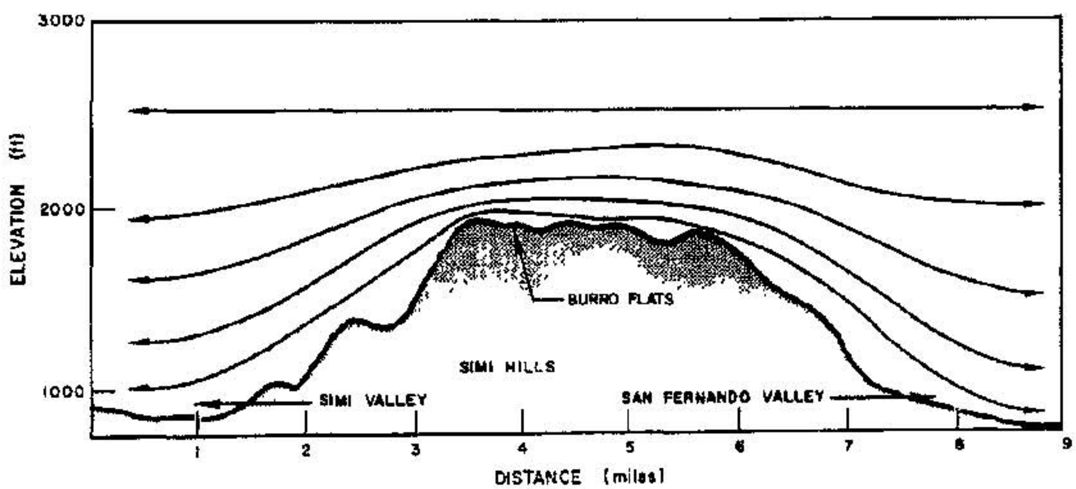

Figure II-13. Idealized Air Flow Over the Simi Hills

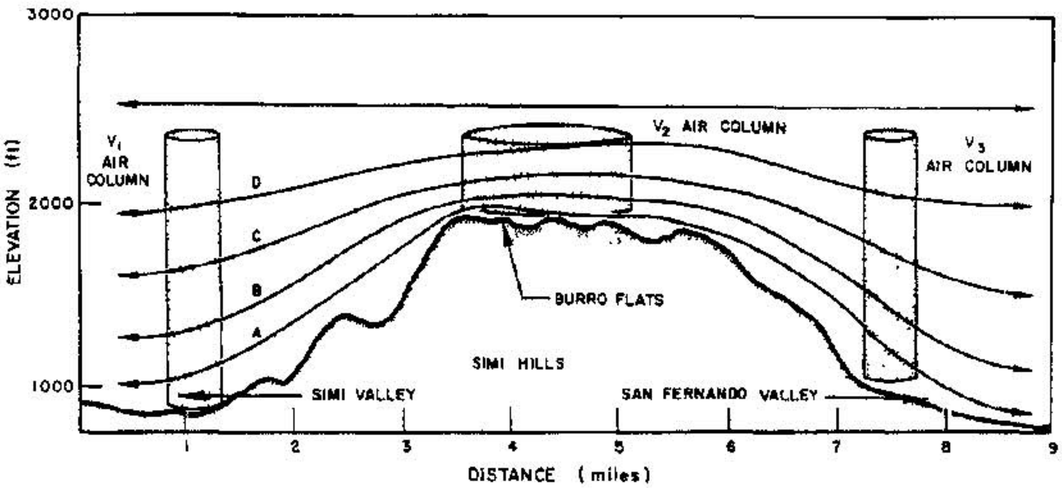

Figure II-14. Representative Air Flow Across Simi Hills

depicted as a column of air traversing an obstruction (Figure II-14). This column can be thought of as being made up of many air particles, each of which follows a curved path (in the vertical plane) dictated by the streamline on which it is located. In the zone of packed streamlines over the site, the turbulence in the flow serves as a mechanism for relatively free interchange of particles from one streamline to another. This results in a mean cloud trajectory which is higher than one would derive from purely laminar flow, with the height of the trajectory being directly related to the degree of turbulence.

It can be assumed, also, that for at least a portion of the period of the day during which northwest or west-northwest winds exist, the flow is essentially horizontal; i.e., there are few or no vertical components. This assumption is based on the fact that, initially, the northwest wind is strictly an up-valley (Simi) wind, which, in order to continue to flow, must be moving horizontally. If there were any significant vertical component to the flow, adiabatic cooling would take place. This cooling would effectively remove the mechanism (warm, buoyant air in the valley being replaced by cooler air from the plain) which fosters the air movement.

b. Elevation

The location of Burro Flats, approximately 1000 feet above the adjacent valley floors, provides an environment which, most of the time, naturally protects the San Fernando Valley population from site-generated pollution. As shown in Figure II-15, situations (a), (c), and (d) all provide a measure of protection, either by restricting diffusion due to stability over the valley, as in (a) and (c), or by achieving a high cloud trajectory as well as good diffusion by instability over Burro Flats, as in (d). However, the flow toward the Simi Valley normally does not occur under such favorable conditions. Actually, the east-southeast winds, primarily an early morning phenomenon in spring and summer, are weaker than the west-northwest winds and are associated sometimes with stable conditions over the site and somewhat unstable lapse rates over the valley. A situation commonly associated with flow toward the Simi Valley would than be represented by (b) in Figure II-15 (which would require the direction arrow to point west-northwest).

It is important to realize that east-southeast flow also is often associated with stable conditions both over the site and over the Simi Valley (as shown in Figure II-15 (c)). The conclusion can be drawn, therefore, that, although the Simi Valley is not as well protected meteorologically as the San Fernando Valley, it enjoys considerable protection due to either stable lapse rates or winds blowing in another direction.

c. Location

The flow over Burro Flats is so predominantly east to southeast, or northwest to north, that locations other than the eastern Simi and western San Fernando Valleys have little opportunity to become involved with pollution from the site. During the winter, there is a higher incidence of north to northeast winds which would carry the effluent toward the Agoura-Thousand Oaks area. However, the resulting hazard to these areas is negligible because of:

1) Distance (6 to 8 miles from Burro Flats).

2) Flow direction, which is parallel to the Simi Hills spine, fostering good diffusion and little downward component.

3) Morning conditions, which would closely resemble the situation in (a) on Figure II-15 (with direction arrow pointing southwest).

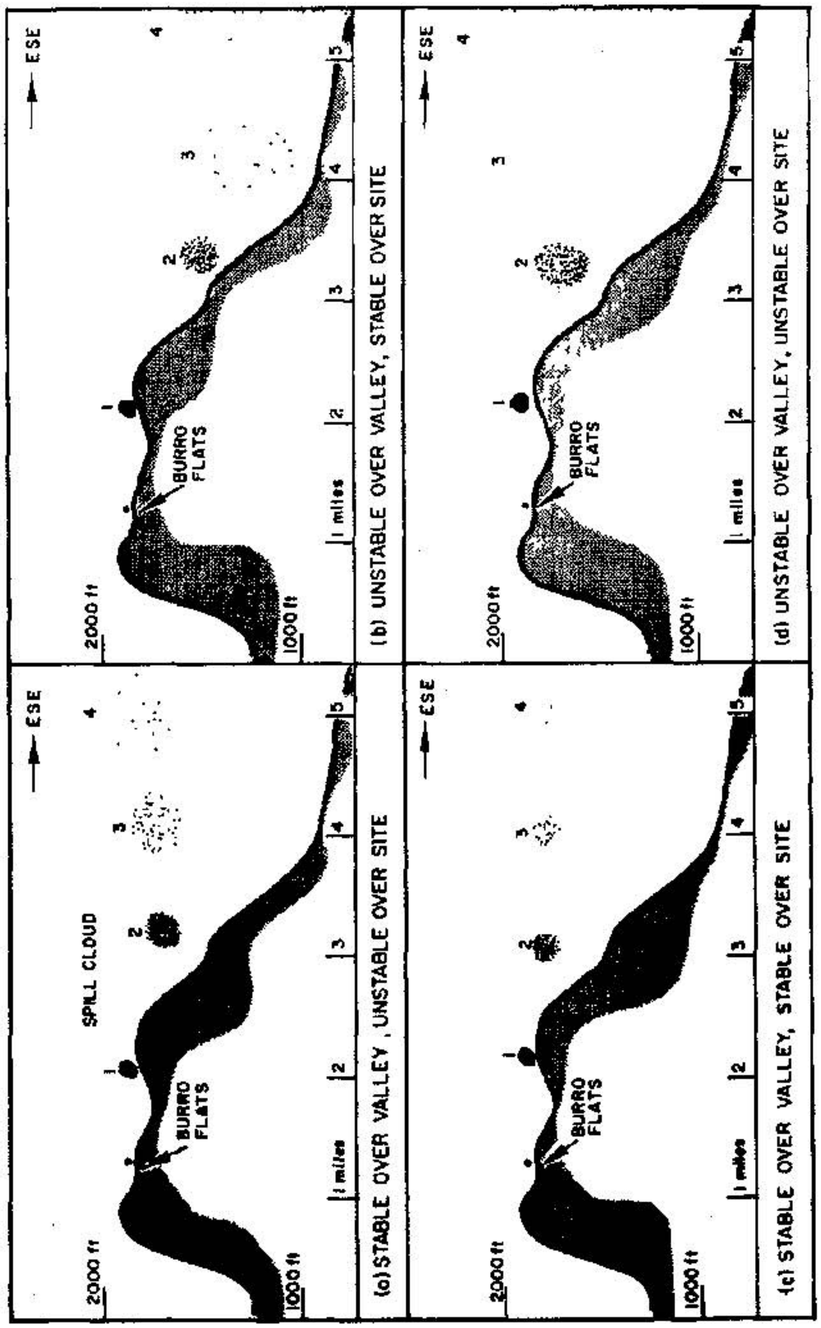

Figure II-15. Dispersion of a Puff Type Release from the Site

d. Release Time

A complete analysis of the consequences of a radioactive discharge must include the possibility of a release over an extended period of time. The effluent cloud than would take on the appearance of an elongated plume (Figure II-16), the size, shape, and path of which result from the same variations in atmospheric stability as depicted in Figure II-15.

As in the case of the instantaneous or short-duration release, the worst case is the one with stable air over Burro Flats and unstable air over the valleys, as represented by (b) in Figure II-15. This condition exists less than 5% of the normal operating time. The resulting pollution at any particular location is determined by the following:

1) The rate of release which governs the concentrations along the plume axis.
2) Residence time at a particular location, a function of the duration of the release.
3) Diffusion in the direction of wind flow (negligible in an elongated plume).

4. General Conclusions

The meteorological factors associated with the site indicate that favorable conditions exist for the proper off-site diffusion of material resulting from potential radioactivity releases. In addition, the site provides one rather unique feature which results in an additional advantage.

Specifically, because of the very irregular terrain surrounding the site, there is a large buffer zone which exists between the site and the surrounding populated communities. The width of this buffer zone is about 3 miles. As a result, the hazard at the outside boundary of this buffer zone, due to a radioactive release from the site, would be orders of magnitude less than that at the site boundary. This zone of essentially zero population density provides this site with a means of almost completely decoupling the hazard to the general public from site releases of radioactivity. Meteorological conditions which would decrease the advantage thus gained do not exist except under very rare conditions. Hence, it can be concluded that the geographic location of the site with respect to surrounding populations coupled with the area meteorology provide adequate means for protecting the general public.

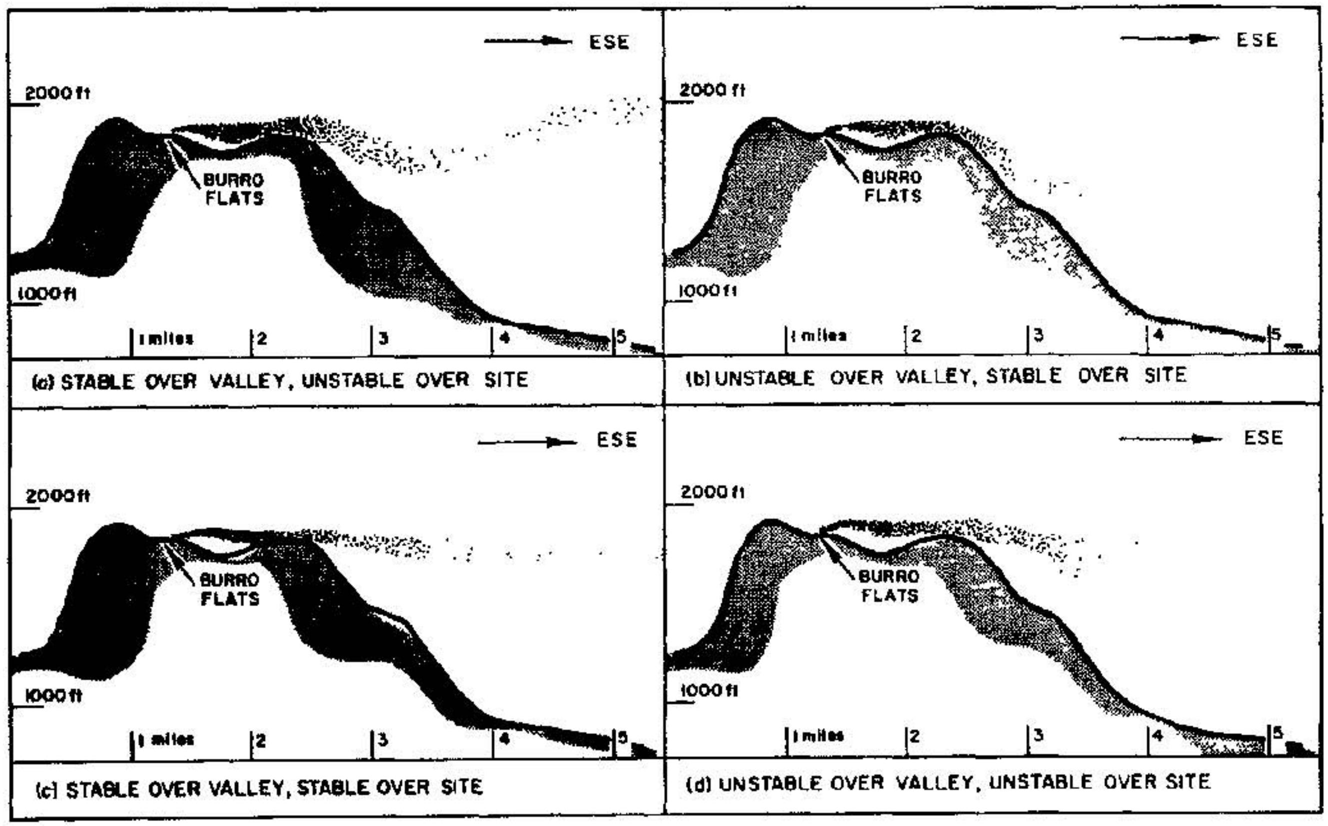

Figure II-16. Dispersion of a Continuous Release from the Site

D. GEOLOGY

1. General

Nearly the entire North American Aviation Field Laboratory property is underlain by the Chico Formation, which is Upper Cretaceous in age. This formation consists predominantly of buff- to brown-colored, massive-bedded, coarse to slightly pebbly sandstone, with occasional beds of fine sandstone and thin beds of grayish micaceous shale. The sandstone and shale are tightly cemented. The formation is several thousand feet thick.

The southwest corner of the property within the Atomics International boundary is underlain by the Santa Susana-Martinez formation of marine sandstone, conglomerate, and shale. This formation is locally water-bearing.

The strike of the beds across the property ranges from N60 E to N85 E, with all dips to the north varying from 20 to 30°.

The surface strata is 20 to 30 feet thick with numerous fractures generally parallel to the direction of dip. The fractures are most abundant in areas cut by a large number of faults. Probable depth of fracturing is 5000 to 6000 feet. The results of the severe fracturing are made evident by the many large monoliths separated from the original strata. Weathering and decomposition of surface shales and sandstone is also evident.

Burro Flats is composed of alluvial deposits of recent geologic origin. The surface deposit may vary from approximately 10 to 30 feet. The general area geology is shown in Figure II-17.

2. Faults

An extensive minor fault skirts the south boundary of the property. However, no recent activity of this fault is evident. The local faulting is considerably in evidence, crisscrossing the Chico sandstones. The fault traces exhibit either a rhombic or a triangular, or, in some instances, a wedgelike pattern, apparently due to the relief of stresses which accumulated over the entire area of Cretaceous outcrops during regional warping. The rhombic fault pattern is most common. These patterns are bounded by sets of faults extending roughly from east to west and from northeast to southwest. These are the main faults in the area.

Since the fault pattern indicates that some diagonal movement may have taken place, there is probably a slight strike-slip component involved as well

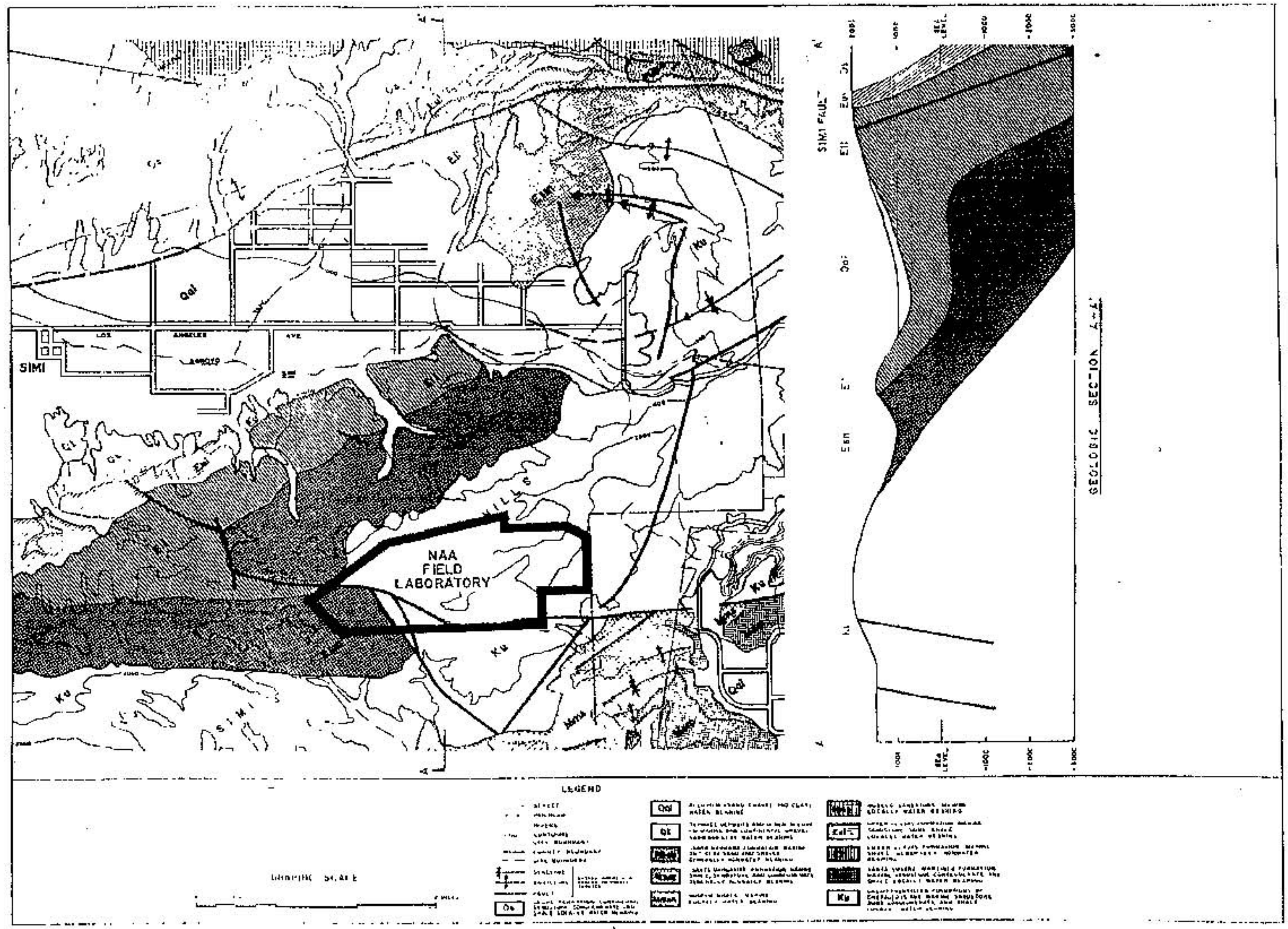

Figure II-17. Geology of the Field Test Laboratory and Environs as a vertical component. Most faults within the area of sandstone outcrops appear to have moved distances measurable in tens, rather than in hundreds of feet. The faulting relieved local tensional stresses during the regional diastrophism, and probably all faults are normal rather than reverse.

The criteria used in plotting the fault lines include:

a) Linear topographic troughs or rifts.

b) Discontinuity of strata along strike.

c) Changes in strike of adjacent strata.

d) Readjusted stream and valley patterns.

3. Foundations

Because of the variable nature of the alluvial mantle and actual substrata bedding, most buildings which impose high foundation loads or require stringent stability have foundation designs based on subsurface investigations.

E. SEISMOLOGY

1. Earthquakes - General

A variety of causes have been suggested for earthquakes. The general consensus on California quakes is in favor of movement of the earth's outer crust along fault zones due to relieving of strains. It is common knowledge that California, which is part of the Pacific seismic belt, accounts for most of the seismicity of the United States. However, this does not preclude the existence of major earthquake centers in other regions of the country, as has been evidenced by the locations of several high intensity quakes which have been experienced in such widely separated areas as New England, the South Atlantic states, the Montana-Idaho region, and the Western plains.

The California region contains a number of active faults. In addition, several formerly inactive faults are now known to be active, based on recent seismic activity. An example of the latter is the White Wolf Fault in Kern County. Although its extent had been plotted many years ago, it was considered devoid of movement until the quake of July 21, 1952. It was this relatively inactive fault which triggered the damaging Tehachapi and Bakersfield quakes of the same year.

Faults in the Los Angeles vicinity (see Figure II-18) which can be classified as active (although some presently inactive faults are possibly potentially more active) are as follows:

> San Andreas - This is the dominating fault in California. It is a right lateral fault extending several hundred miles into the Pacific Ocean northwesterly of the California coast. Many of California's recent quakes have been attributed to movement along this fault. The Field Laboratory site is approximately 40 miles southwest of the San Andreas fault zone.
>
> Santa Ynez - This fault traverses east-west, approximately 35 miles north of the site. Little historical activity has been associated with this fault.
>
> San Gabriel - This fault is about 30 miles north of the site. Earthquake activity associated with it has been clearly defined. A history of recorded quakes fails to attribute disturbances of major intensity to movements initiated along this structure.
>
> Inglewood - Activity of this fault was probably responsible for the Long Beach earthquake of 1933. Considerable activity, in addition to the destructive Long Beach earthquake, has been experienced within the past few years.

A history of earthquake damage throughout the world has been maintained for many years, and attempts have been made to correlate the damage reports with known earthquake phenomena. Seismologists in the California region are in general agreement that the geologic composition of any area in question is more of a factor in sustaining damage than is the proximity of the active fault, except within the immediate confines of the fault itself.

The damage and intensity scaling system now in general use is the Modified Mercalli (M. M.) scale which is summarized below. Only grades VI - IX are included, since microregionalization of the Los Angeles area and, in fact, of the entire California seismic belt, considers VI as the absolute minimum of potential effect. The M. M. scale has been further modified by C. F. Richter, and reworded here for purposes of summarization.

> VI. Felt by all, both indoors and outside. General excitement. Trees shaken slightly. Liquid set in strong motion. Small bells ring. Damage slight in poorly constructed buildings. Fall of plaster in small amounts. Some cracked plaster. Broken dishes and glassware; also some windows. Some overturned or moved furniture.

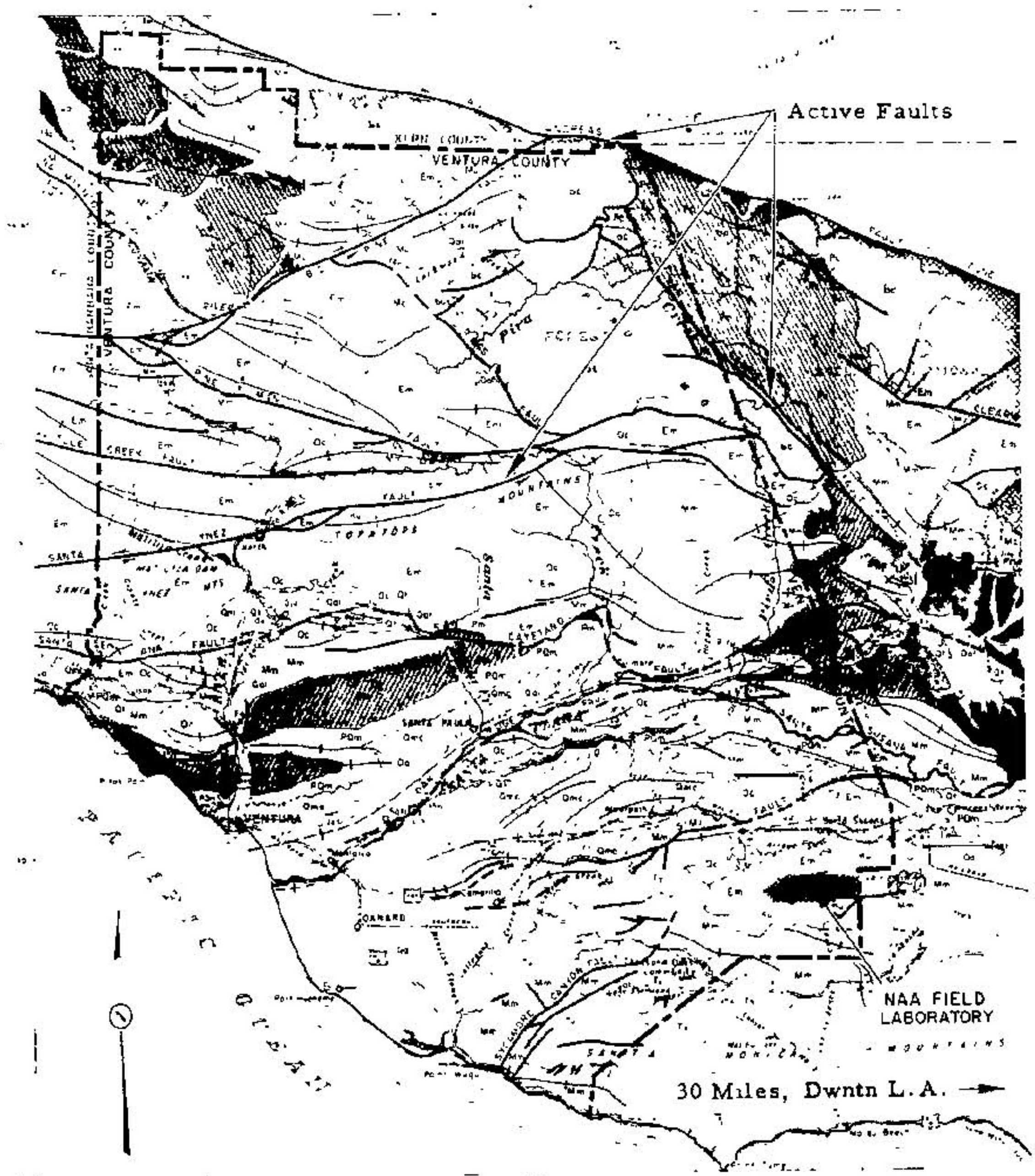

Figure II-18. Earthquake Faults in the Los Angeles Vicinity

VII. Frightened all. Also noticed by persons in motor cars. Trees and bushes shaken moderately to strongly. Waves on ponds and lakes. Water turbid. Incaving to small extent of sand or gravel tanks. Damage negligible to well-designed and constructed buildings, slight in well-built ordinary buildings, and considerable in poorly designed and constructed buildings, adobe houses, old walls, etc.

VIII. General alarm approaching panic. Trees shaken strongly; some branches broken off. Changes of flow in springs and wells. Damage slight in structures designed especially to withstand earthquakes. Considerable damage in ordinary substantial buildings - partial collapse, racked frame in wooden houses, some wall panels thrown out. Chimneys and columns twisted, sometimes thrown down.

IX. Panic general. Ground cracks conspicuously. Damage considerable to masonry structures built especially to resist earthquake forces. Serious damage to reservoirs; undergound pipes sometimes broken.

Regionalization of the Los Angeles basin consists chiefly in translating geology into seismic intensity as follows:

M.M. SCALE

IX. - Quaternary alluvium and sand dunes

VIII. - Quaternary terraces

VII. - Tertiary

VI. - Mesozoic sediments and igneous rock.

The Quaternary system is most recent. It is characterized by various forms of deposits such as that found in alluvial plains. In general, the mantle is unconsolidated and offers little structural resistance to earthquake motion. Terraces offer slightly more resistance since greater consolidation is evident.

The Tertiary system is also marked by deposits, though differing from the Quaternary in the coastal regions where the deposition is mainly marine. Cementation occurs in most of these formations offering some structural resistance.

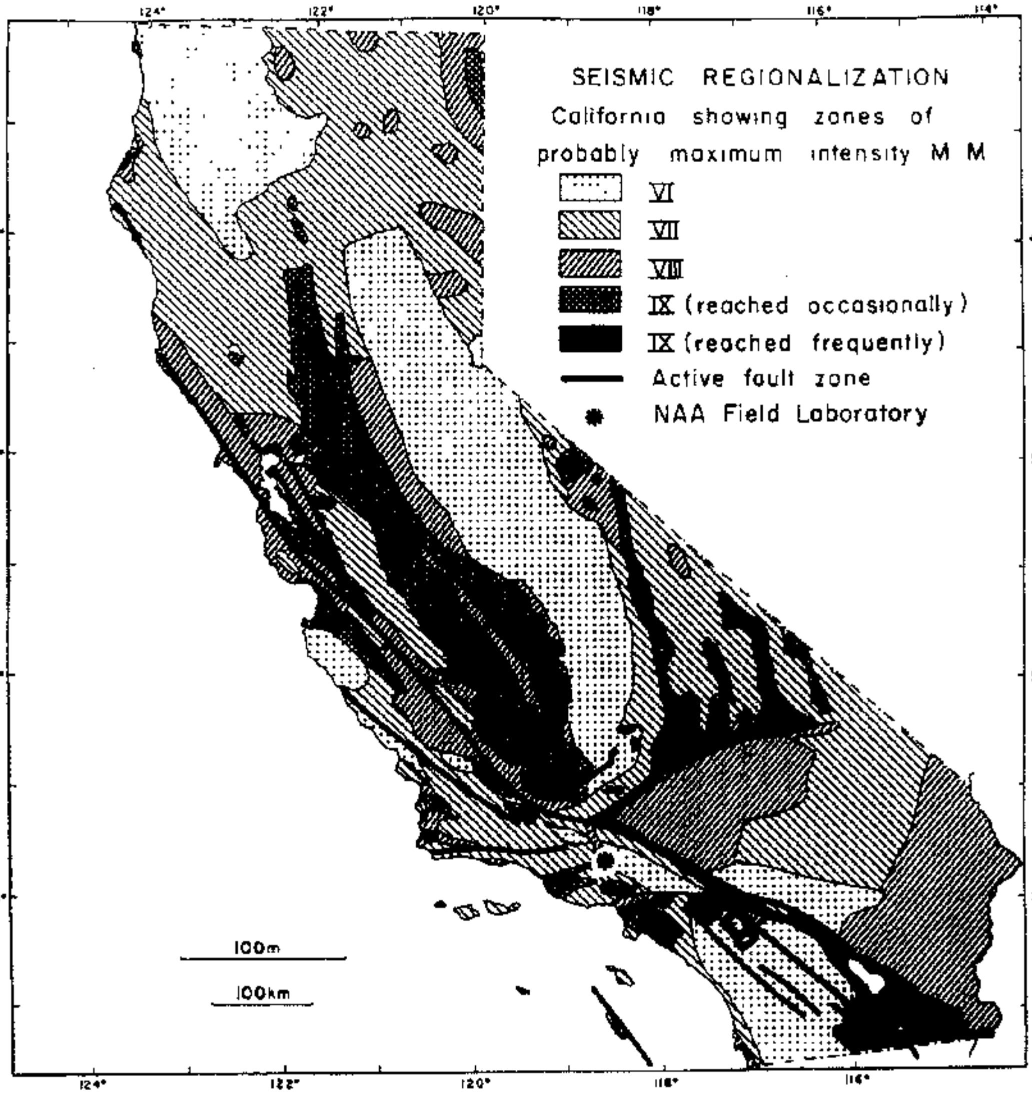

Figure II-19. Seismic Regionalization Map of California

The Mesozoic sediments include the Upper Cretaceous formations found at the site.

2. Microregionalization

Figure II-19 shows a seismic regionalization map of the State of California.* Figure II-20 shows seismic microregionalization for the Los Angeles Basin, also adopted from C. F. Richter. Although details do not include outlying areas, such as that occupied by the site, regionalization charts of California invariably place these mountainous areas in categories VI or VII. Exceptions occur when very close proximity to an active fault zone dominates the seismicity. In this regard, the site's relationship to active faults is considered remote enough for the lower classifications.

Since further refinement of microregionalization does not exist for the areas within the North American Aviation Field Laboratory boundaries, as it does not exist for most areas of the world, it is not unreasonable to assign a modified Mercalli intensity of VI to the outcrop areas and VII to the thin alluvial covering of Burro Flats.

3. Design Requirements

All buildings constructed on the Santa Susana site conform to the provisions of the Uniform Building Code (UBC). This Code stipulates that the area is in Zone 3, the area of major damage. It is obvious, from the zone map of the United States included in the UBC, that microregionalization is not considered in the arbitrary selection of zoning. Therefore, it is reasonable to assume that assignment of the highest numerical values for earthquake design in regions where actual damage is expected to be light (i.e., regions corresponding to an M.M. scale of VI or VII), such as the Field Laboratory site area, should result in conservative designs.

F. SURFACE HYDROLOGY

1. General

Surface water at the site is a result of rainfall and industrial waste. Average annual rainfall is about 17 inches, with approximately 70% occurring

*C.F. Richter, "Seismic Regionalization," Bulletion of the Seismological Society of American, 49, No. 2, p. 123-162 (April 1959).

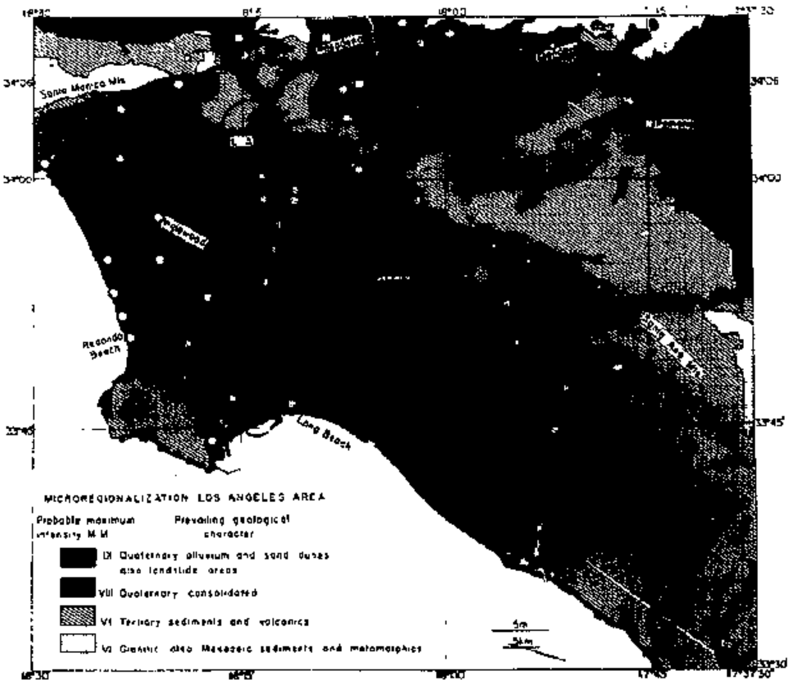

Figure II-20. Seismic Microregionalization Map
of the Los Angeles Basin and Vicinity

during the winter months of December through March. The major source of industrial waste at the Field Laboratory is rocket engine exhaust coolant, the quantity varying with the number and length of tests.

2. Reclamation

Networks of reclamation ponds (Figure II-21) are located in Rocketdyne's Areas I and II. These, in turn, overflow into water storage areas. Reclaim water lines and pumps in each area make up coolant water at the various test stands.

3. Rainfall Runoff

Calculations of the amount of rainfall runoff, considering the worst storm anticipated in a 50 year period, were prepared for the Rocketdyne test areas. Based on these calculations and observations at the site, the reclaim and storage ponds are adequate to hold most off-site drainage. That drainage which overflows the various retention points is directed off the site through two channels which eventually meet in Bell Canyon Creek. (See Figure II-22.) The creek is a natural drainage course which flows southeasterly into the Los Angeles River channel at

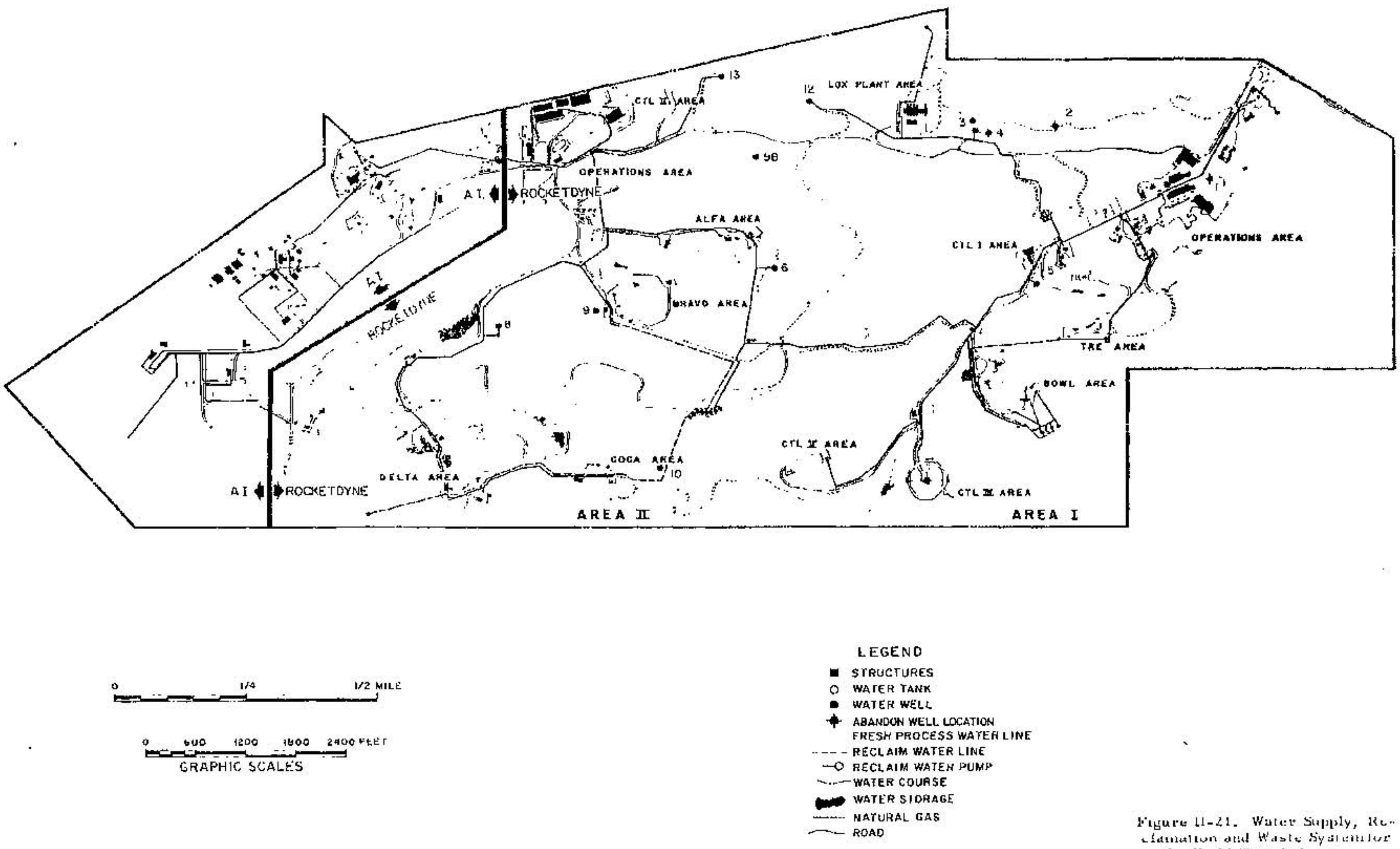

Figure II-21. Water Supply, Reclamation and Waste System for the Field Test Laboratory

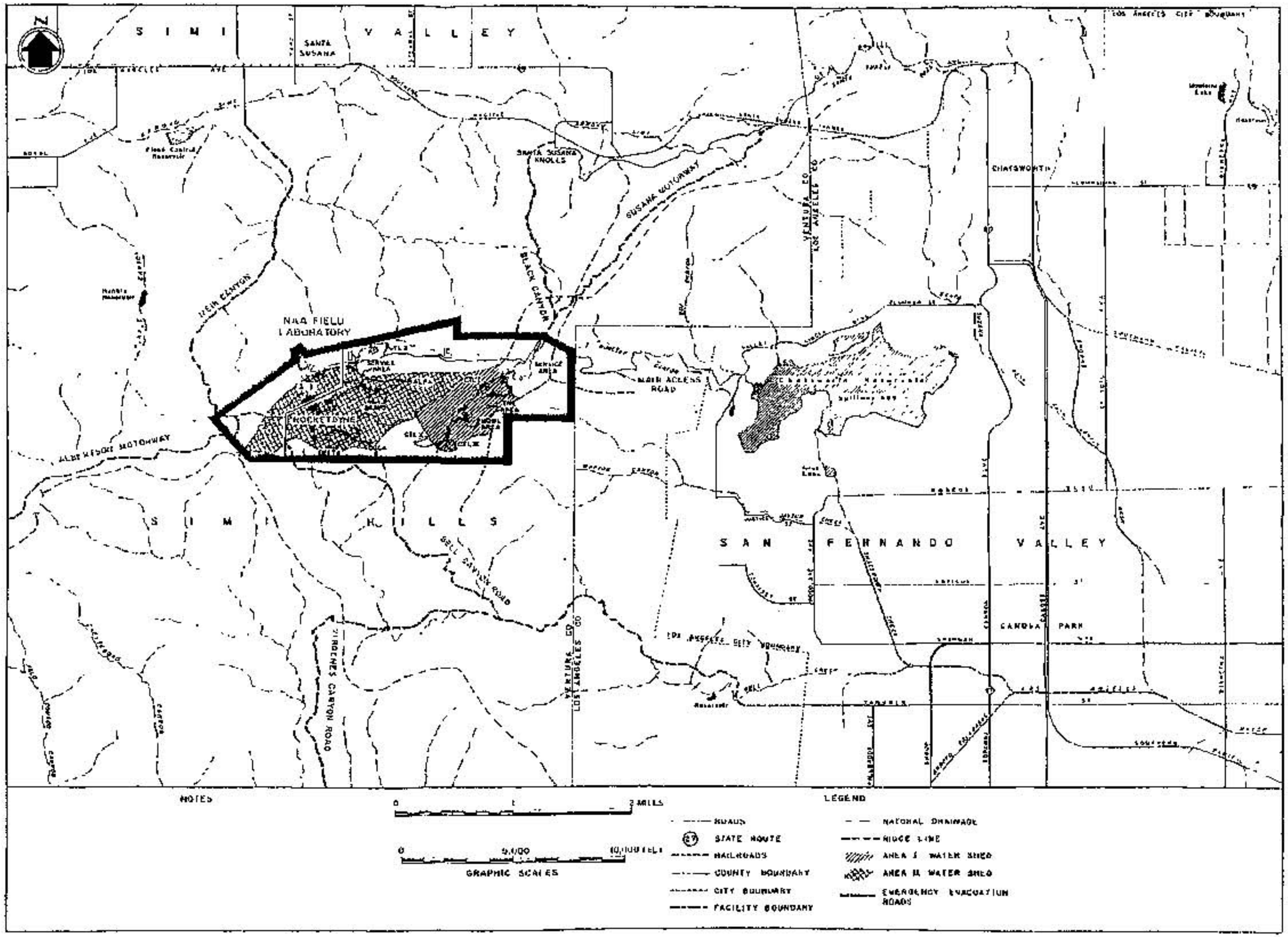

Figure II-22. Regional Surface Water Map a point west of the community of Canoga Park. The Los Angeles River drains through the Sepulveda Flood Control Basin, where it may be held up before continuing to its termination.

4. Atomics International Site Drainage

The natural drainage in the area occupied by Atomics International is southerly toward the Area II storage basins (see Figure II-21), except at the north property line, where drainage is northwesterly toward Simi Valley. It is here, along the north boundary, that the majority of Atomics International's facilities are located. In order to prevent run-off and accidental spillage from following the natural drainage pattern to Simi Valley and its various populated communities, retention basins have been constructed at the two off-site channel heads. Each is equipped with pumps of sufficient size to prevent overflow. The pumps and connected pipes direct flow from the retention basins to the Burro Flats area, where drainage courses empty into the Rocketdyne Area II storage ponds.

5. Monitoring

Area I and Area II storage pond waters and site well waters are continuously monitored for a large number of possible impurities, including radioactivity. Samples are thoroughly tested and reports submitted each month. The radioactivity content reported each month has always been at the background level.

G. GROUND WATER

1. Hydrologic Characteristics

The Chico formation underlying most of the site is tightly cemented throughout its entire thickness. Total porosity is probably somewhat less than 1%. The ground water in this formation may be attributed to the following:

a) Fault planes where movement has caused fracturing of the sandstone.

b) Joints and fractures not directly associated with faults but related to the overall faulting in the area.

c) Bedding planes where there is a change in lithology of the formation.

d) Limited permeable zones in the sediments where original cementation of the grains has not been entirely effective.

The Chico formation, as a whole, is a very poor aquifer or water-producing formation. Most of the formation shows secondary cementation which has decreased the original porosity to a very low capacity. The majority of the porosity in which water occurs in the formation is very closely associated with the fault planes, fractures, and joints throughout the entire thickness of sediments.

Field examination and photogeologic studies disclose numerous evidences of fault planes crisscrossing the Chico Sandstones. There is also some evidence that the east-west faults slightly antedate those extending in a northeast-southwest direction, since the latter appear, in most cases, to offset the former. It is probable that the east-west faults have associated with them the greatest fracturing system, and it is along these that the better producing wells have been developed. These are wells 5, 6, 12, and 13, shown on Figure II-21.

2. Structural Geology

The geologic structure which holds the water at 700 to 800 feet elevation above the surrounding valley floors is unique and quite unusual. The Cretaceous massive sandstone is bounded on the northwest, west, and partially on the south by Eocene shales. A large fault trending east-west apparently forms a seal toward the southern direction. On the east escarpment of the Santa Susana Mountains, the northwesterly dip of the formations with interbedded thin shale members apparently forms a barrier on each stratum to retain the water within the Chico sandstone. The barrier on the north is not definitely known, but it is most certainly there. Evidence of the existence of this northern barrier stems mainly from the fact that a well drilled outside the North American Aviation property, approximately one-half mile south of the Simi Valley, has a static water level of 865 feet above sea level, while wells on the property have static water levels from 1430 to 1630 feet above sea level.

3. Hydrology

The production capacity of each well is controlled by the number and association of the joints and fractures in the formation generally associated with the faults. There is apparently no known method for determining the fracture system capacity and its extent beneath the property, except by interpretation of empirical data from the history of operations. A representative historical compilation of well production on the site begins with well 3. This well, activated in October of 1948, was the first actually placed in production. It supplied the entire water demand for the North American Aviation Field Laboratory until May, 1955, when well 5 was activated. Considering that well 3 was only 210 feet deep, its production has been remarkable. It produced an accumulated total of 24.8 million gallons until it was de-activated in 1955. Well 11 has produced the least amount of water, with a total of 4.2 million gallons recorded between August 1956 and June 1958. Well 12 has produced the most water, with a total of 140 million gallons in the relatively short period of 22 months. The poorest wells, 7, 8, and 11, are all located in the Burro Flats region, an area of thin alluvial mantles overlying the Cretaceous formation.

It is interesting to note that three wells, 3, 4, and 4A, are all located within 200 feet of each other and have penetrated practically the same beds down to their equivalent depths. Well 3, at a depth of only 210 feet, was by far the best well and yet was only two-fifths the depth of well 4A. This fact points to the influence of fractures in controlling the capacity of the underground water.

The tilting of beds is such that well 5, at a previous depth of 700 feet, had penetrated practically the same beds as well 6, at a depth of 1440 feet.

Wells 9B and 12 penetrate nearly the same beds down to a similar depth and are only a horizontal distance of 1000 feet apart, along the strike of the formation. The productive index of these two wells are very much apart, with 12 the best well on the North American Aviation property, producing 100 gallons per minute per foot of drawdown, whereas 9B produces 30 gallons per minute with a total drawdown of 300 feet.

The fact that wells penetrating the same beds can have such a difference in productive capacity is evidence that the fracturing and jointing definitely controls the capacity of the well yield and the amount and extent of underground water in the area.

4. Inventory of Existing Wells

Current well production, as governed by pumping capacity, is about 1360 gallons per minute. This amount supplies both Rocketdyne, the majority user, and Atomics International. The contributing wells are 5, 6, 9A, 12, and 13. Their vital statistics are listed in Table II-6.

TABLE II-6

DATA ON CURRENTLY PRODUCING WELLS AT THE SITE

Well No.	Depth (ft from surface)	Yields (gpm)	Elevation Above Sea Level (ft)	Water Level (ft below surface)
5	2304	350	1832	560
6	1440	230	1938	649
9A	582	150	1648	270
12	700	400	1708	460
13	930	230	1665	373

Wells 5 and 9A were recently deepened to expose new water sources. Both well 5 and well 12 are being pumped considerably below their available capacities. Their full potentials, with reasonable drawdown, have not been explored.

5. Future Outlook

The North American Aviation ground water supply is derived, literally, from mining operations. Rainwater replenishment is insignificant because of the general impermeability of the structures and the lack of large surface water collection areas. However, the Cretaceous formation, with its numerous fractures and joints, extends to an estimated depth of 5,000 to 6,000 feet below the surface. Considering the relatively shallow penetration of existing wells, it is postulated that a great deal of trapped water remains within the entire formation. This can be exploited as required. In addition to this permanent supply, studies are in progress to develop an alternate supply from one of two public agencies. This, it is considered, will provide assurance that an adequate water supply will always be available.

H. UTILITIES

All utilities necessary to the operation of a Field Laboratory are available at the site. The water supply is developed under the supervision of Rocketdyne personnel and transmitted to Atomics International through two major pipelines from producing wells. These lines terminate at two 50,000 gallon storage tanks located on promontories overlooking the Burro Flats area. Other utilities are directly under the jurisdiction of Atomics International.

1. Water Supply

The domestic and cooling water supply is obtained from Rocketdyne, except for on-site potable water. Drinking water is supplied by an outside vendor in bottles delivered regularly to the site. However, a potable supply switch-over to the main well supply is contemplated, subject to economic justification. This water is presently used by Rocketdyne as a potable supply.

Recent improvements in the supply lines include several additional loops, thereby guaranteeing continuous flow and reduced line stagnation.

2. Sewage System

All sewage originating at the Atomics International site is terminated, in its raw state, at a central sewage treatment plant. Capacity of the plant at this time is 25,000 gallons per day. This is considerably in excess of the average amount of sewage delivered to the system. Expansion of the Field Laboratory beyond the capacity of the present system can be handled with the installation of one or two aeration tanks. Provision has already been made for these installations, should the need arise.

3. Power

Power is supplied to the North American Aviation Field Laboratory via 66-kv high lines of the Southern California Edison Company, which enter an Edison substation located on the property. This substation, in addition to stepping down power for the site, supplies several surrounding communities. Two lower substations located within the Atomics International site boundary are fed by 16-kv lines emanating from the main substation. These, in turn, provide 4160-volt power to various area substations throughout the site.

Complete future development of the site will necessitate the addition of new circuits with little or no increase in the local capacity.

4. Telephones

All buildings requiring service are linked to a central board located on the Atomics International site. Plans are presently in effect to change from manual to automatic switchboards and to increase capacity for future use.

5. Natural Gas

The recent installation of a natural gas supply line by the Southern Counti·s Gas Company has provided service to the entire Field Laboratory. The gas enters the property at 200 psi and proceeds to a metering station. All feeder lines from this station are at 30 psi, including the main feeder to the Rocketdyne facility. Feeders proceed to all major buildings and installations requiring gas service.

III. EVALUATION OF THE SITE AS A WHOLE AS A LOCATION FOR REACTORS

A. ESTABLISHING REACTOR SITE CRITERIA

The approach to reactor site evaluation set forth in Title 10, Part 100 of the Code of Federal Regulations (10 CFR 100) requires that the consequences of the maximum credible accident associated with some specific reactor type be related to the different population zones established in the regulation for the proposed site. If the dose criteria established in 10 CFR 100 for the various zones are met, the site can be considered suitable for the proposed reactor, at least insofar as those aspects noted are concerned. However, as an alternate approach one can just as well establish design criteria for any reactor to be situated on a particular site by evaluating the suitability of that site for releasing radioactive material, independent of the reactor type. Using this latter approach, which would actually be more in line with the formulation of site criteria, an additional set of specifications can be developed for the reactor design. These specifications would relate solely to the amount of radioactivity which any reactor, if located on the evaluated site, could release to the site environs, and would be a prime function of the facility leak rate and release height. This approach will be used here to establish both the maximum steady-state power level and maximum integrated reactor transient which can be permitted on the Santa Susana site, both as a function of the facility leak rate and release height.

B. USE OF SITE CHARACTERISTICS TO DETERMINE PERMISSIBLE RADIOACTIVITY RELEASE

In establishing the extent of radioactivity release which can be permitted from the site, knowledge of the reactor characteristics usually considered necessary to such studies will not be required. Rather, only the characteristics of the site itself need be considered. In this way the evaluation can truly be of the site and not of the reactor to be placed thereon. Chief among those site and environs characteristics which must be known are the topography, meteorology, geology and seismology, hydrology, general accessibility, population distribution, and use characteristics. Consideration is always given to all of these factors in establishing the suitability of a site for locating a reactor facility. However, for the purpose of this section, we shall restrict ourselves to the consideration of only the meteorology and population distribution aspects, since these are the only important aspects of airborne releases. In the study to be performed here, the population distribution aspects generally will be treated in the same manner as in 10 CFR 100, wherein the acceptable dose criteria for different control and population zones are established and the zones are defined. In calculating the dispersion of the released radioactive material, sufficiently pessimistic meteorological conditions appropriate to the site will be used.

It should be pointed out that the results obtained from this study will be applicable only to the extent that the assumptions which were required in the analytical model are consistent with actual cases. However, by computing the maximum permissible radioactivity releases which can be permitted from the site, this study will provide information concerning the maximum steady-state reactor power level and the maximum size of reactor transient, both as a function of facility leak rate and release height, which can be tolerated on the site. Other sections in this report will deal mainly with on-site effects, and only indirectly will they provide any quantitative insight into the "load that the site can bear."

C. REACTOR POWER LEVEL AND EXCURSION LIMITATIONS

1. Assumptions, Limitations, and Calculational Techniques

In order to provide a concise treatment of this problem, it was necessary to establish certain limitations and make several simplifying assumptions. These limitations and assumptions will, as mentioned, qualify to some extent the utility of the results. However, the main objective will be achieved: namely, the results will provide the data necessary to establish design criteria for reactor facilities to be located on site, considering the implicit limitations which the site will impose on the steady-state reactor power level and reactor transient. These assumptions, because of their importance, are tabulated below, together with some justifying remarks.

a) The maximum permissible dose at the site boundary for an incident is taken to be 25 rem whole-body-equivalent. This dose is evaluated considering the contributions from the integrated inhalation dose in 90 days resultant from a 2-hour exposure to the cloud, the external whole body dose received from a 2-hour exposure to the cloud, and the dose received due to fallout. In the case of the latter contribution, the exposure consists of the fallout dose occurring during the first 2 hours of the accident plus the exposure received in the 90-day period starting 24 hours after the accident.

b) The whole-body-equivalent dose is obtained by adding to the whole body exposure received from the cloud and fallout, 25/300 of the TID received by the thyroid due to iodine inhalation. This approach to a determination of the total dose received was utilized due to its convenience in simplifying presentation of the results; in addition, it should yield somewhat conservative results.

c) The maximum permissible dose at the nearest community is also taken to be 25 rem whole-body-equivalent. The nearest community is taken as being located 3 miles from the site. The dose is evaluated considering contributions from the integrated inhalation dose in 90 days resultant from continuous exposure to the radioactivity release for the entire duration of its passage, the external whole body exposure received from the cloud during the same period of time, and the fallout dose received during the release period and the succeeding 90 days. In this "nearest-community" case, however, the duration of radioactivity release resultant from the accidents was restricted to 24 hours.

d) The radioactivity released is assumed to consist only of noble gases and halogens. Noble gas-to-halogen release fraction ratios of 2:1 and 4:1 are considered. These release ratios are consistent with the models utilized later in this report for evaluating the radioactivity release from the specific accidents considered at the individual reactor facilities, and cover the range of variation indicated.

e) Fission product inventories from reactors operating at significant power levels were based on one year of continuous operation. The activity produced in a reactor transient is obtained using the method described in Appendix A.

f) In all calculations, it was assumed that the radioactivity was released to the environs after holdup in only one compartment, the leak rate of which is the prime variable in the analysis. In addition, this leak rate is assumed to remain constant throughout the accident. (The results of this study will not generally be applicable to cases where there is double containment, e.g., a pressure vessel and then (followed by) a low-leak-rate building, or where the leak rate from the building varies significantly during the accident due to the pressure increase in the contaminated air volume. However, application of the results to such cases would certainly be conservative in that the doses resultant from an accident would then be less than calculated.)

g) Distances of 100 and 200 meters between the release point and the site boundary were chosen, since these represent a reasonable range of the actual distances. (When the maximum downwind ground concentration occurs at distances greater than 100 or 200 meters, this distance, d_{max}, is used in order to obtain the maximum dose.)

h) Release heights of 0, 15, 30, and 45 meters were used, since these represent the range of heights which might normally be encountered.

The calculational techniques used in the evaluation of the doses received are shown in Appendix A, with one exception: the on-site whole body dose from ground deposition has been computed as shown in Appendix B, wherein the ground concentration is taken as the product of TID and V_g. The dose conversion factors which are used are consistent with those in Appendix A.

2. Summary of the Results

The results are presented in Figures III-1 through III-10, wherein the product of either the reactor steady-state thermal power (P_t) or integrated reactor transient (E), the fractional release of noble gases from the core due to the maximum credible accident (f), and $[(d)/(d + x_o)]^{2-n}$, the virtual point source correction factor, is plotted as a function of the fractional volume of contaminated air released per day from the facility (F). The curve parameter is the ratio of the noble gas-to-halogen release fraction, ratios of 2:1 and 4:1 being included. Figures III-1 and III-2 present the variation of reactor thermal power level results for the case of the site boundary being 100 and 200 meters from the release point and considering release heights of 0 and 15 meters, respectively. Figures III-3 and III-4 present similar results, but for the case where d_{max} is used for the distance between the release point and receptor for release heights of 30 and 45 meters, respectively. Similar data are presented for the variation of integrated reactor transient with leak rate; Figures III-6, III-7, III-8, and III-9 are for release heights of 0, 15, 30, and 45 meters, respectively, with the downwind distances for each release height being the same as for the power level curves. Figures III-5 and III-10 present the variation of power level and reactor transient with leak rate when the dose is evaluated at the nearest community, with a 200-foot release height assumed in both cases. (See Section VI for further considerations relating to the choice of this release height.)

Comparison of the date from the curves indicates that, for the dose criteria which have been utilized, the maximum permissible steady-state power level or integrated transient which can be accommodated on the site (for a given facility leak rate) is restricted more by the proximity of the site boundary than by the nearest community. This is an important factor, especially with regard to considerations of the adequacy of the existing meteorological data for the site, and is discussed elsewhere in the report. The curves do show, however, that if the dose criteria at the site boundary can be met by the facility design, the populated areas surrounding the site will be amply protected by virtue of the distance between them and the site.

3. Discussion of the Use of the Results

Use of the data is rather straightforward. For example, if it is proposed to locate a reactor with a steady-state power level of 0.4 Mw on the site, and studies indicate that the maximum credible accident releases 100% of the noble gas fission product inventory, then, in order to locate the facility 200 meters from the site boundary, a ground release (Figure III-1) would require a contaminated air volume change rate which is significantly less than 0.01 per day for either the 2:1 or 4:1 noble gas-to-halogen release fraction ratio. A leak rate of this magnitude would probably require a special facility containment structure. However, if it were known that only some fraction of the noble gas inventory would be released, then f, the noble gas release fraction, could be reduced and the permissible leak rate would be increased. As an example, if only 2% of the noble gas inventory were released, then a ground release would require a facility leak rate of 0.0085 (extrapolated) or 0.014 per day for the 2:1 or 4:1 noble gas-to-halogen release fraction ratios, respectively.

However, in the event that the entire core were to melt, a reasonable alternate (from an economic standpoint) to constructing a special containment structure would be to provide a stack in order to permit utilization of a higher leak rate. Referring to Figure III-2, it is seen that a stack height of 15 meters would require a contaminated air volume change rate of 0.055 or 0.11 per day for the 2:1 or 4:1 noble gas-to-halogen release fraction ratios, respectively. If these leak rates are still too restrictive, then, referring to Figure III-3, a 30-meter stack height would require leak rates of 0.21 or 0.49 per day for the respective 2:1 or 4:1 ratios. Increasing the stack height to 45 meters would still further relax the requirements; from Figure III-4, leak rates of 0.54 or 1.6 per day would be required.

It should be pointed out that all of these leak rates assume the factor $[(d)/(d + x_0)]^{2-n}$ to be unity. If a virtual point source correction were appropriate, the permissible leak rates could be increased. In order to demonstrate this point, one can assume the above reactor siting conditions and a virtual point source distance of 500 meters. The correction factor then becomes 0.150, 0.117, 0.283, and 0.400 for the ground release and for the 15, 30, and 45 meter stack heights, respectively. The permissible leak rates would then correspond to ordinate values equal to the product of the power and these correction factors, f being unity in this case. For the example being considered, i.e., with a complete noble gas release, the leak rate values are found to be less than 0.01, 0.6 and 2.0, 1.1 and 13, and 3.1 and ∞ for the 2:1 and 4:1 ratios and release heights of 0, 15, 30, and 45 meters, respectively. The reactor transient curves can be used in the same manner.

Of much greater significance, however, is the fact that the comparison of Figure III-5 with Figures III-1 through III-4 and Figure III-10 with Figures III-6 through III-9 indicates that the remoteness of the nearest community is such that, even for stack heights greater than 45 meters, the controlling location is the site boundary. For example, even for the 45 meter stack height the maximum reactor power level permitted by the location of the nearest community would be at least 130 times that dictated by the proximity of the site boundary. For the permissible reactor transient curves, this factor is reduced to about 1.4 for the low leak rates examined, but it increases to about a factor of 55 at the higher leak rates.

It is important that the factors which constitute the ordinate of the graphs are clearly recognized insofar as their overall influence on the results is concerned. The ordinate of Figures III-1 through III-4 and III-6 through III-9 is the product of reactor steady-state power level or integrated reactor transient, the fraction of the core noble gas inventory released directly to the containment or confinement structure, and the virtual point source correction factor, $[(d)/(d + x_o)]^{2-n}$. Therefore, the permissible reactor power level or transient obtained from the figures can be varied (with a constant leak rate) by a factor equal to the inverse of the product, $f[(d)/(d + x_o)]^{2-n}$, if either (or both) of these factors has a value of less than unity. (Obviously, it would also be possible to effect some trade off between power level or transient and the leak rate.) The virtual point source correction has been included in the ordinate, rather than in the calculations, due to the difficulty of assigning an average value for this correction. Since for any proposed reactor, some idea of the physical dimensions of the facility will be available, arriving at an estimate for x_o should not constitute any problem, and hence it permits more accurate evaluation of the permissible steady-state power level, integrated transient, and the leak rate. The ordinate of Figures III-5 and III-10 is the product of the reactor steady-state power level or integrated transient and the fraction of the core noble gas inventory released, only, since the effect of the virtual point source correction becomes insignificant at this downwind distance (3 miles). Therefore, in the nearest-community curves (i.e., Figure III-5 or III-10), the value for reactor power level or transient for a constant leak rate could be varied by the factor 1/f.

A second consideration which does not appear in the results but has a marked influence upon the utilization of the curves is the fact that all the results are based on a single-compartment model. If a two-compartment model is considered, the permissible reactor steady-state power level or transient will increase markedly over that indicated by the single compartment model used in the preparation of the figures. The factor of increase is substantial; e.g., for a reactor building with no stack, located 200 meters from the site boundary, and a facility leak rate of 1.0 per day (i.e., 100% per day) which would release all its noble gas fission products in a 4:1 ratio with the halogens in the event of its maximum credible accident, the maximum permissible power level would be 0.17 kw, if the building provided the only containment. On the other hand, if the same reactor were provided with secondary containment which had a leak rate of 0.1 per day, calculations indicate that the reactor steady-state power level could be increased to 2.1 kw, an increase by a factor of about 12, without increasing the dose at the site boundary.

As an example of how well the curves may be applied to facilities located at the site, one may consider the AE-6 reactor discussed in Section VI. From Figure III-1, the power level of a reactor located 200 meters from the site boundary which releases at ground level all the noble gases in a 4:1 ratio with the halogens with little or no facility containment is 58 watts. The virtual point source correction for the AE-6 is $[(200)/(200 + 1905)]^{1.5}$, or 0.0293. Correcting the value from the figure by the inverse of this factor gives a maximum permissible power level of 1.98 kw in order to produce no more than 25 rem whole-body-equivalent at 200 meters downwind. The analysis of the AE-6 maximum credible accident (see Section VI) indicates that the power level could be increased by a factor of about 10 (considering the whole-body-equivalent dose), or to about 1.2 kw without producing more than 25 rem at the site boundary, approximately 230 meters downwind. It should be pointed out that the AE-6 analysis indicated that the major fraction of the dose was received from the fission product inventory resultant from the extended period of 120 watt operation, with only minor contributions coming from the short period of 2 kw operation and the transient. Therefore, the agreement between the curve values and an actual case is quite good.

In summary, the results provide a quantitative means for assessing the magnitude of the maximum permissible reactor steady-state power level or integrated transient which the site can tolerate, and hence they provide a useful design criterion for reactors to be located on the site. Due to the limitations placed on the analytical model, the results may not have application for every specific reactor proposed for the site, but they will certainly be valid for the majority of such facilities. It should also be noted that this analysis does not provide any consideration of on-site radiation exposures. This aspect of the problem is treated in the next section; but it can be stated at this point that, if the on-site exposure criteria were the same as those for the site boundary, some lowering of the maximum permissible steady-state reactor power level and transients would result, depending on the distance from the nearest facility to the receptor. However, as long as the reactor is not located closer than 100 meters to the nearest facility, the curves would still be entirely applicable.

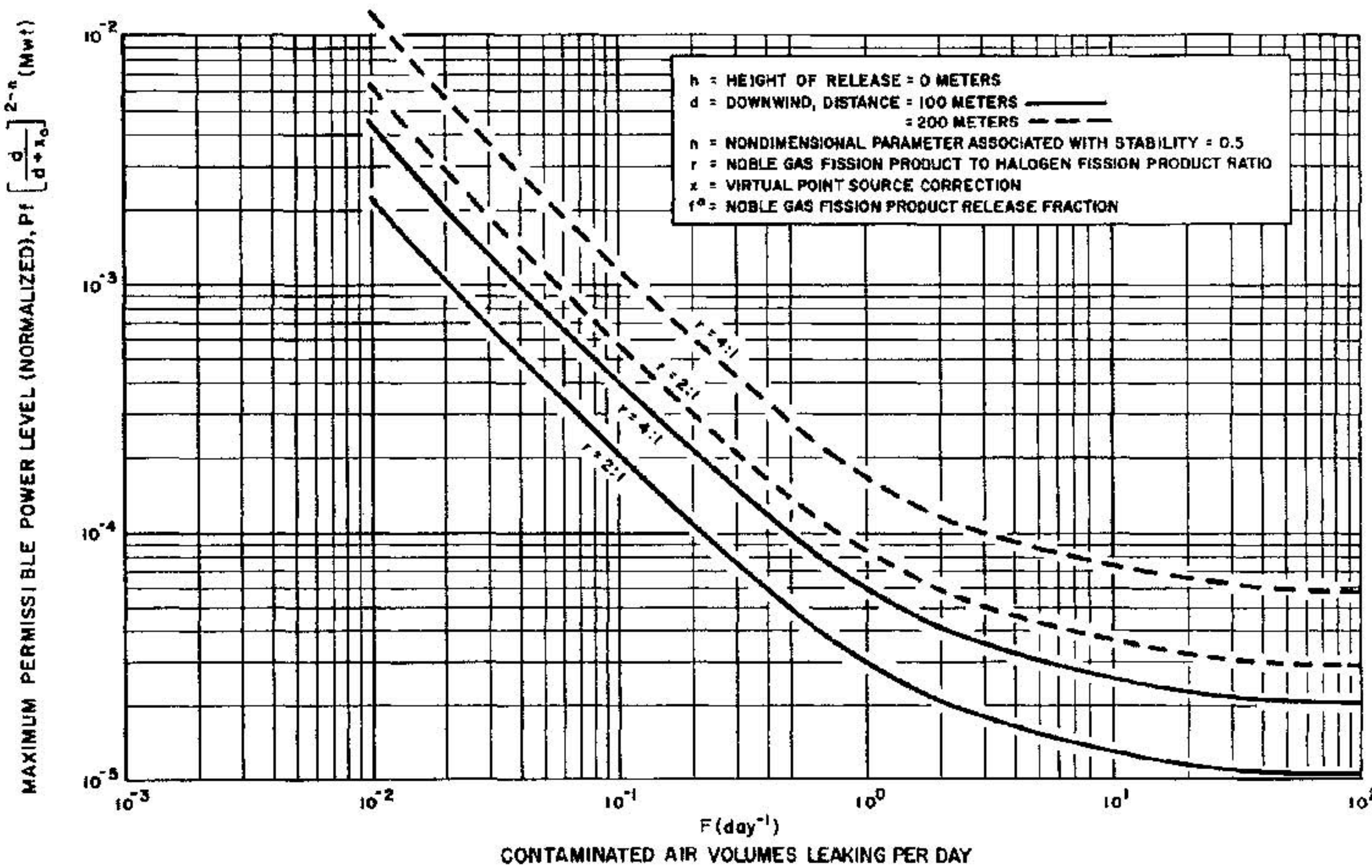

Figure III-1. Maximum Permissible Steady-State Power Level as a Function of Leakage Rate for Ground Release (Based on Dose Requirements for Site Boundary)

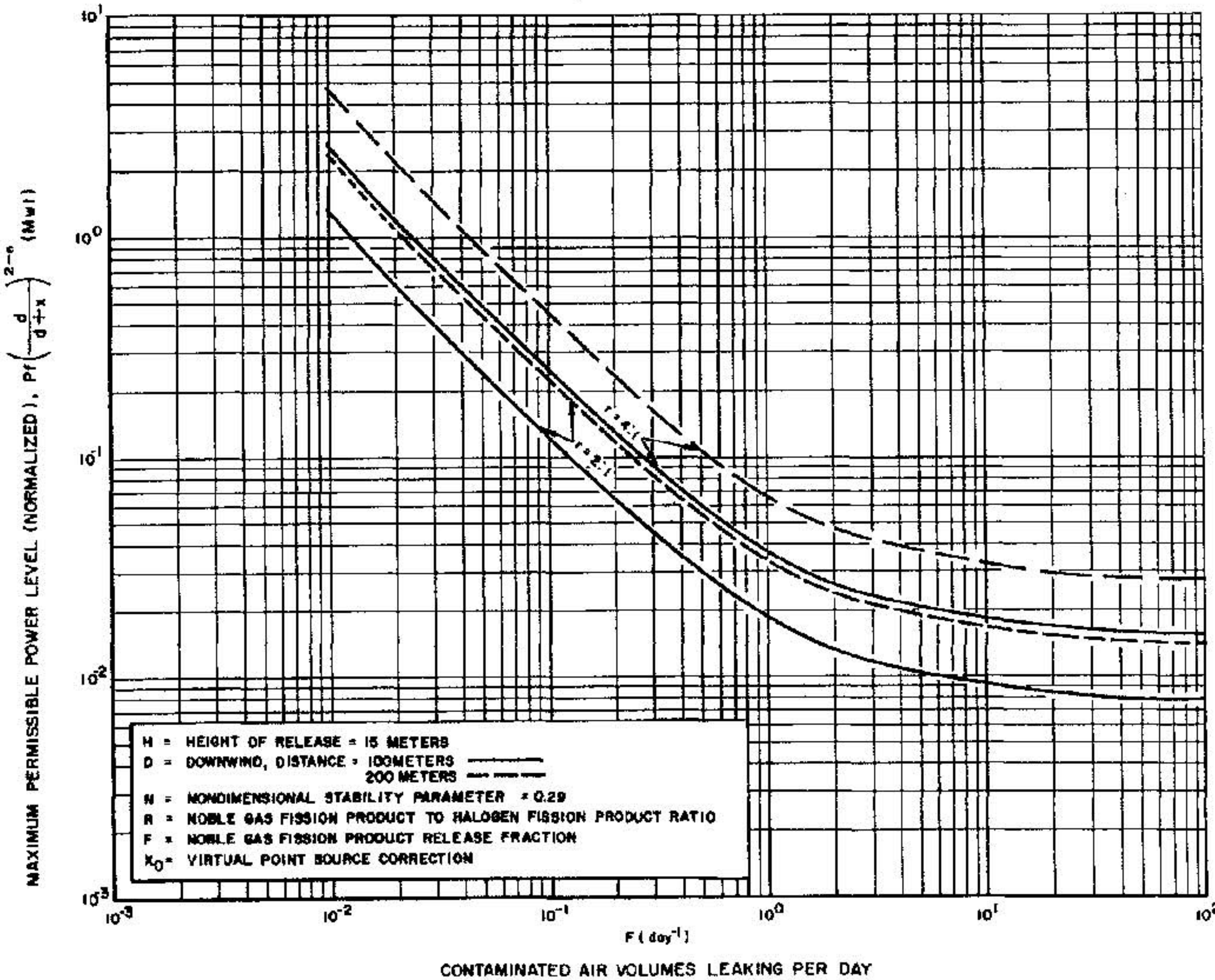

Figure III-2. Maximum Permissible Steady-State Power Level as a Function of Leakage Rate for 15 Meter Stack Release (Based on Dose Requirements for Site Boundary)

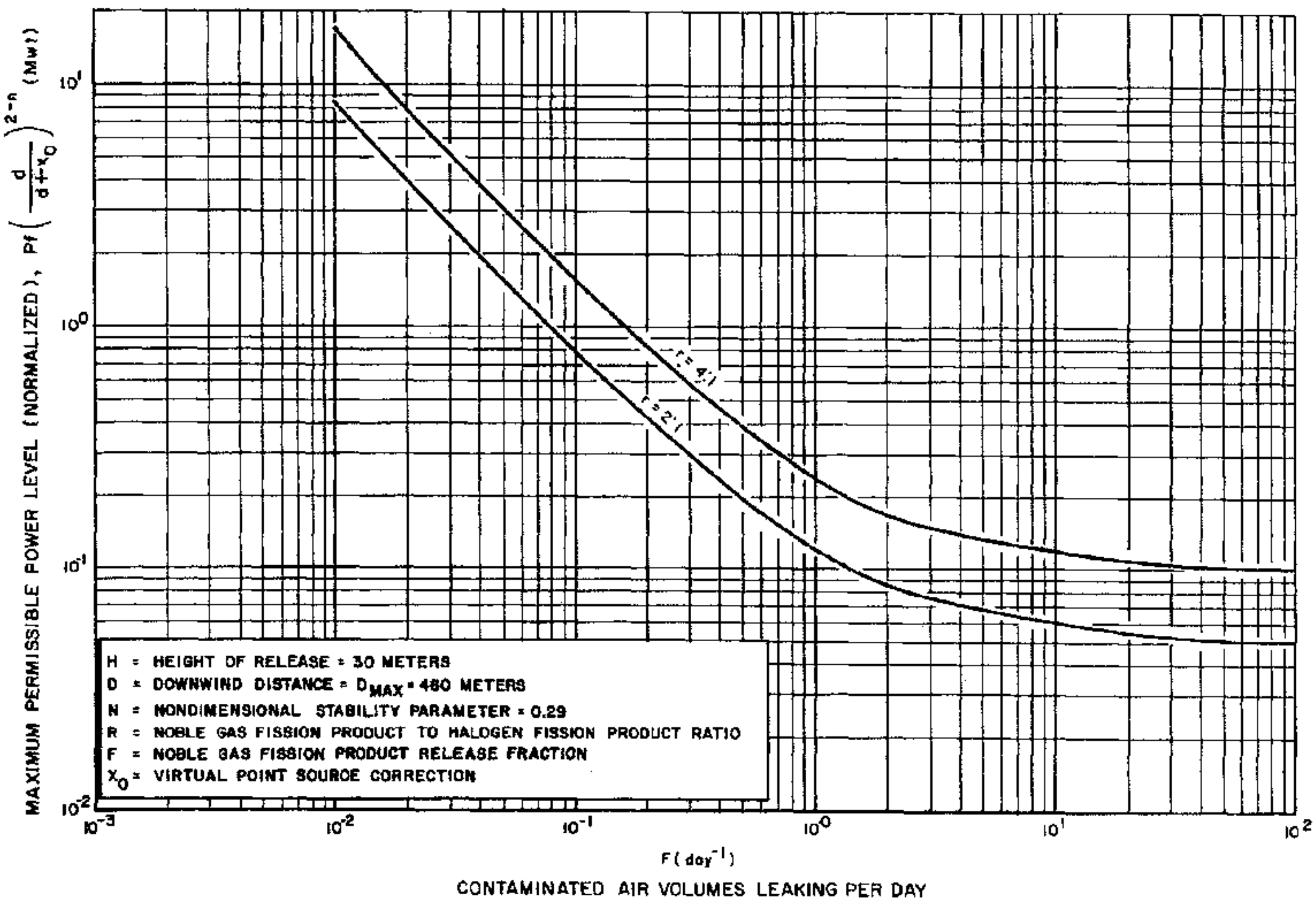

Figure III-3. Maximum Permissible Steady-State Power Level as a Function of Leakage Rate for 30 Meter Stack Release (Based on Dose Requirements for Site Boundary)

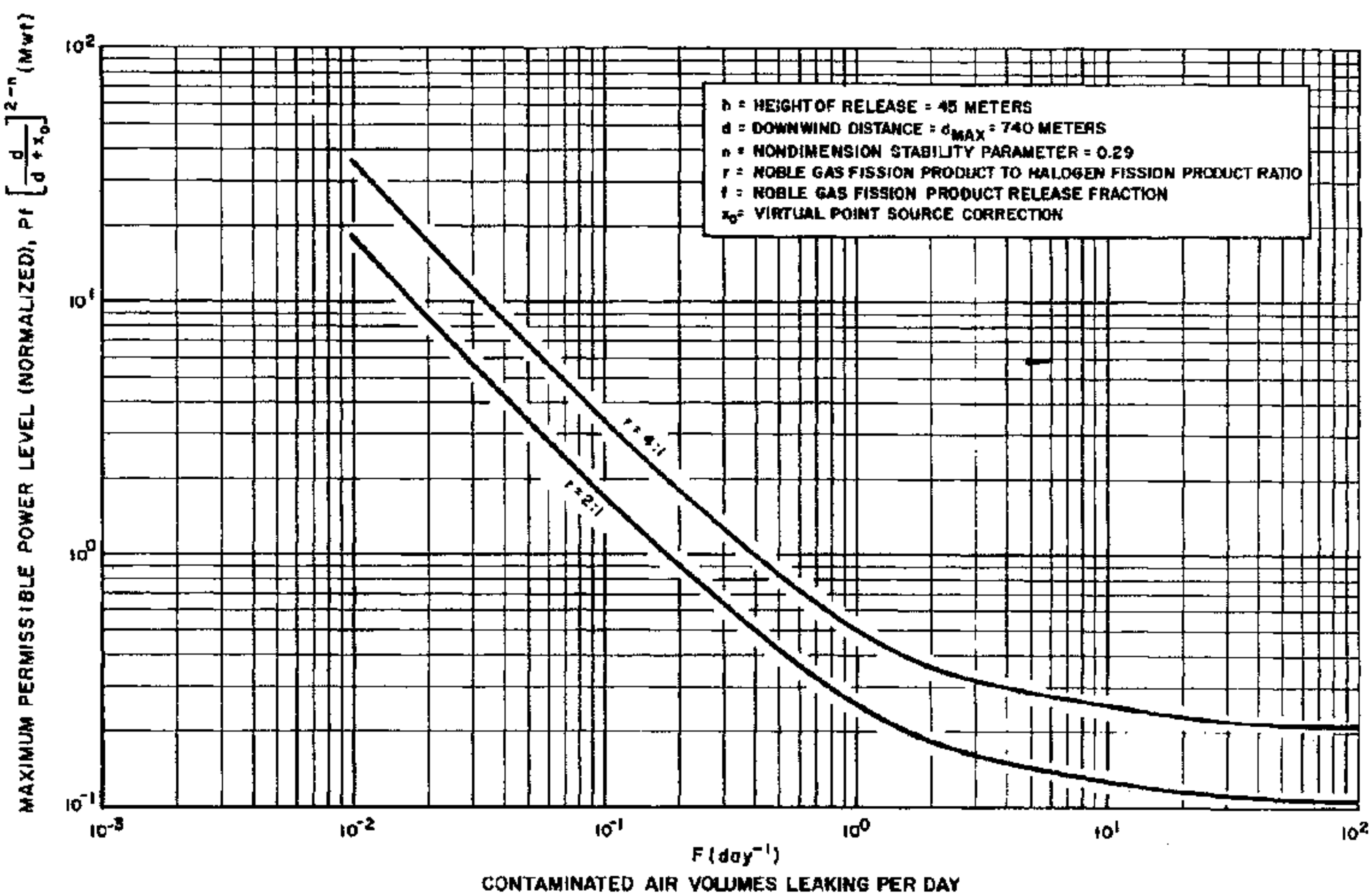

Figure III-4. Maximum Permissible Steady-State Power Level as a Function of Leakage Rate for 45 Meter Stack Release (Based on Dose Requirements for Site Boundary)

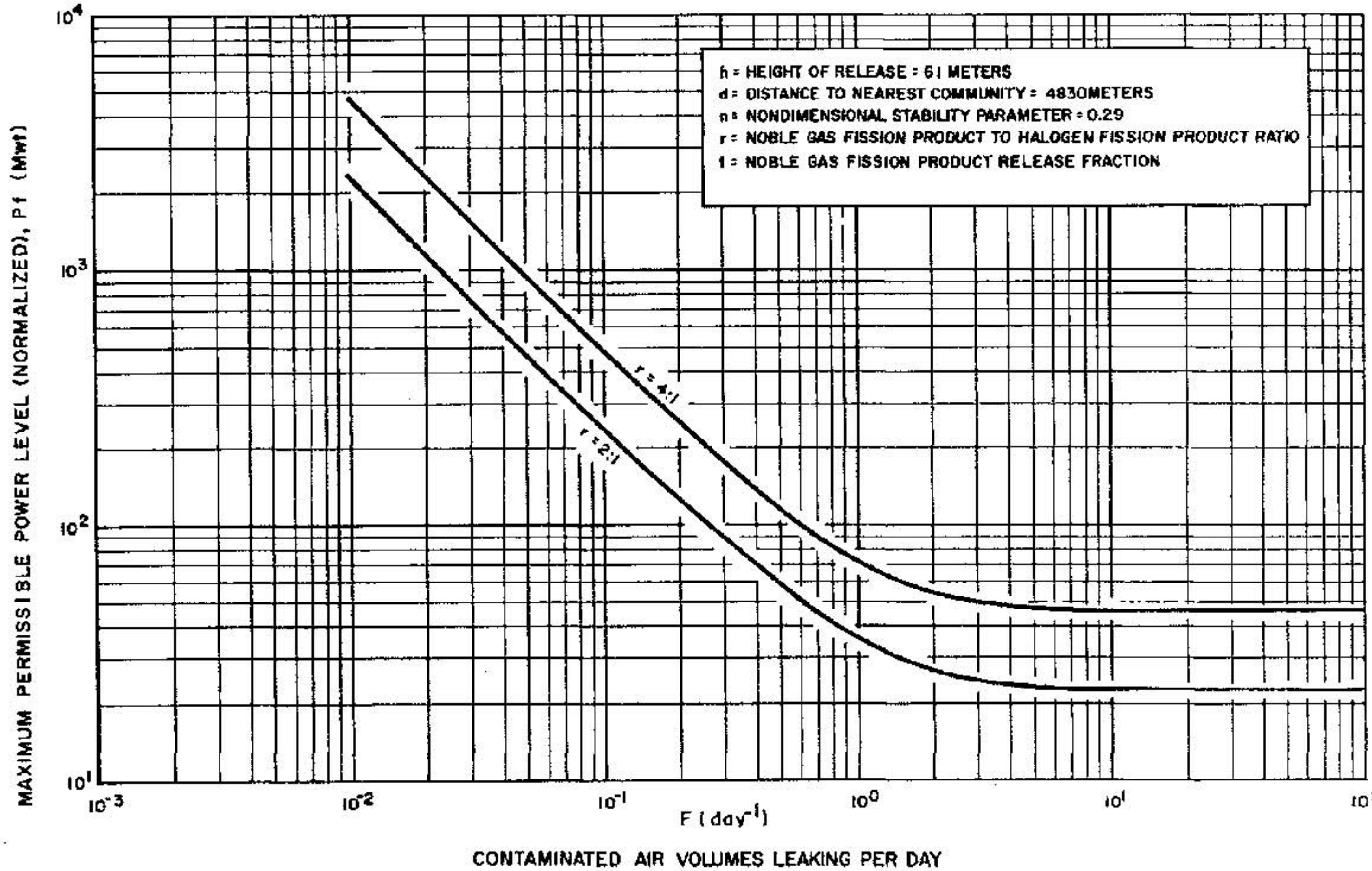

Figure III-5. Maximum Permissible Steady-State Power Level as a Function of Leakage Rate for 66 Meter Stack Release (Based on Dose Requirements for Nearest Community)

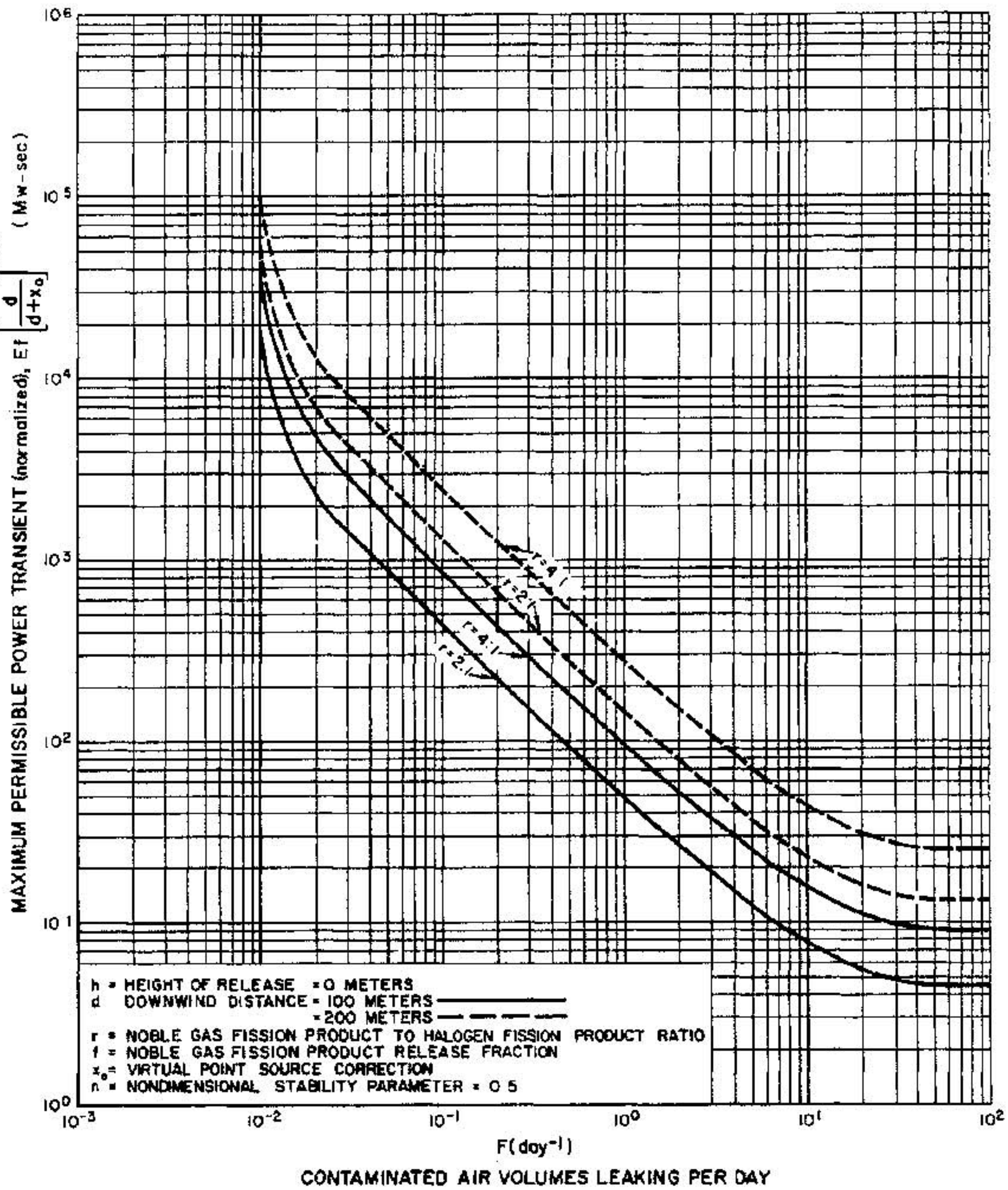

Figure III-6. Maximum Permissible Total Integrated Transient as a Function of Leakage Rate for Ground Release (Based on Dose Requirements for Site Boundary)

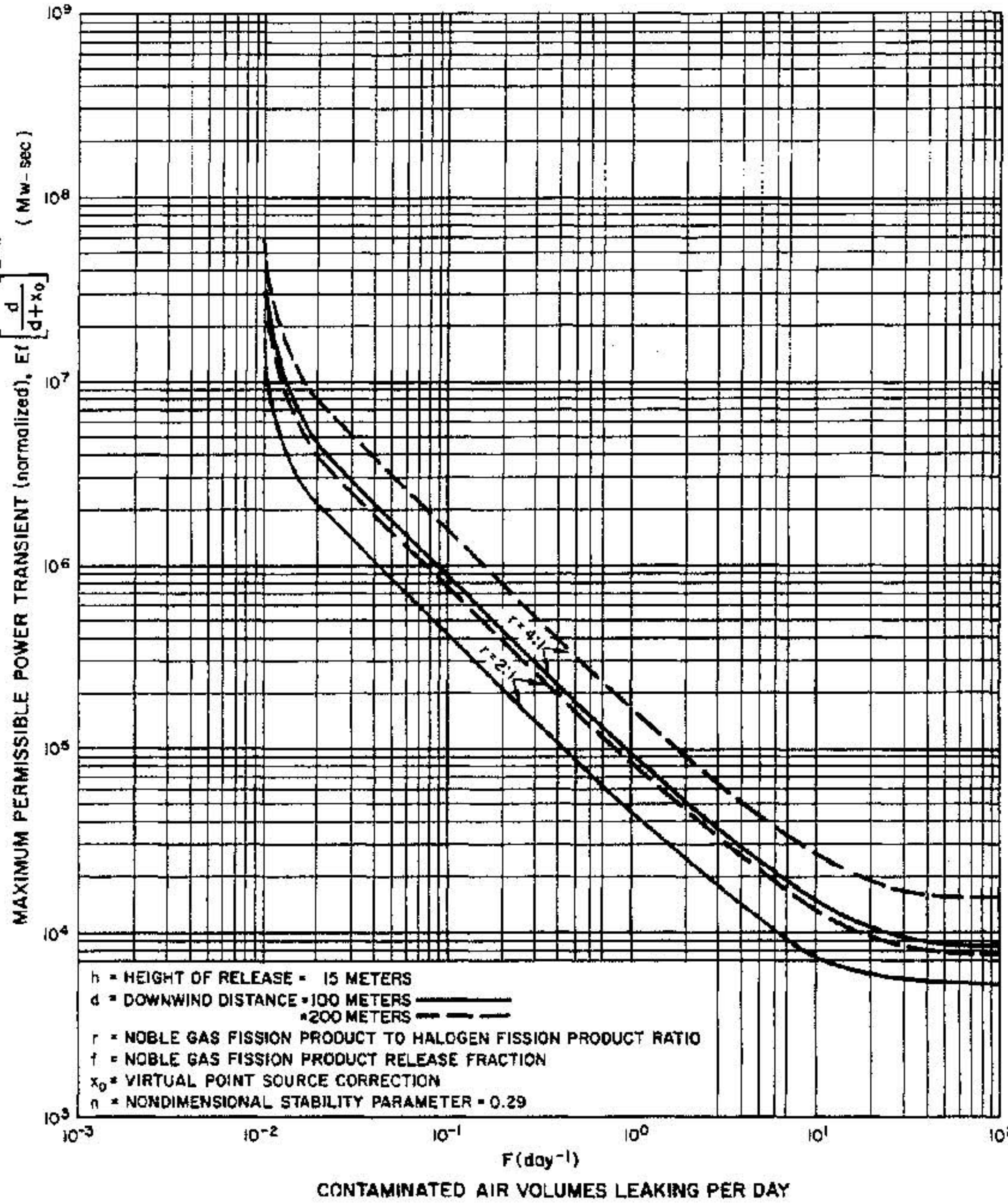

Figure III-7. Maximum Permissible Total Integrated Transient as a Function of Leakage Rate for 15 Meter Stack Release (Based on Dose Requirements for Site Boundary)

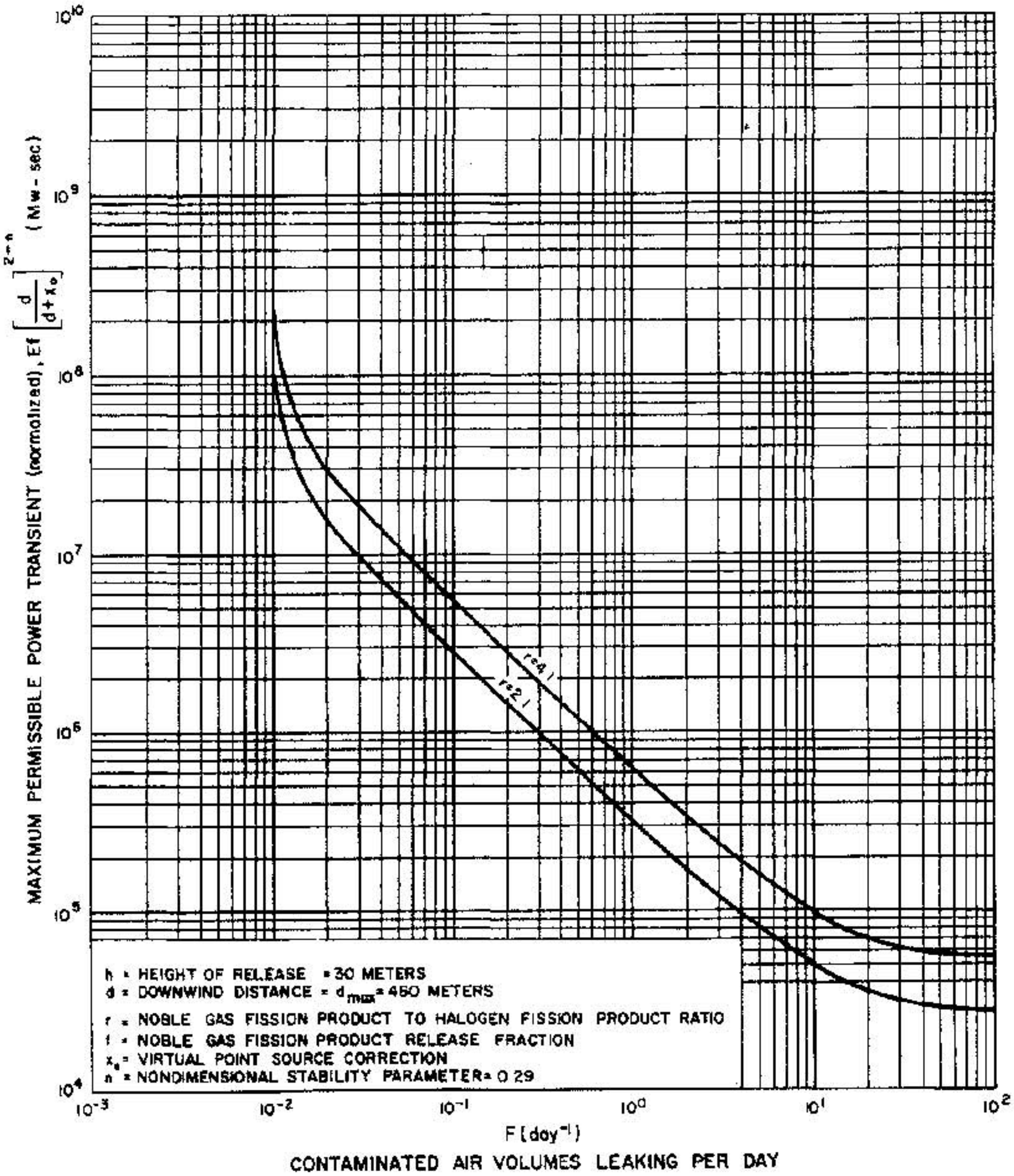

Figure III-8. Maximum Permissible Total Integrated Transient as a Function of Leakage Rate For 15 Meter Stack Release (Based on Dose Requirements for Site Boundary)

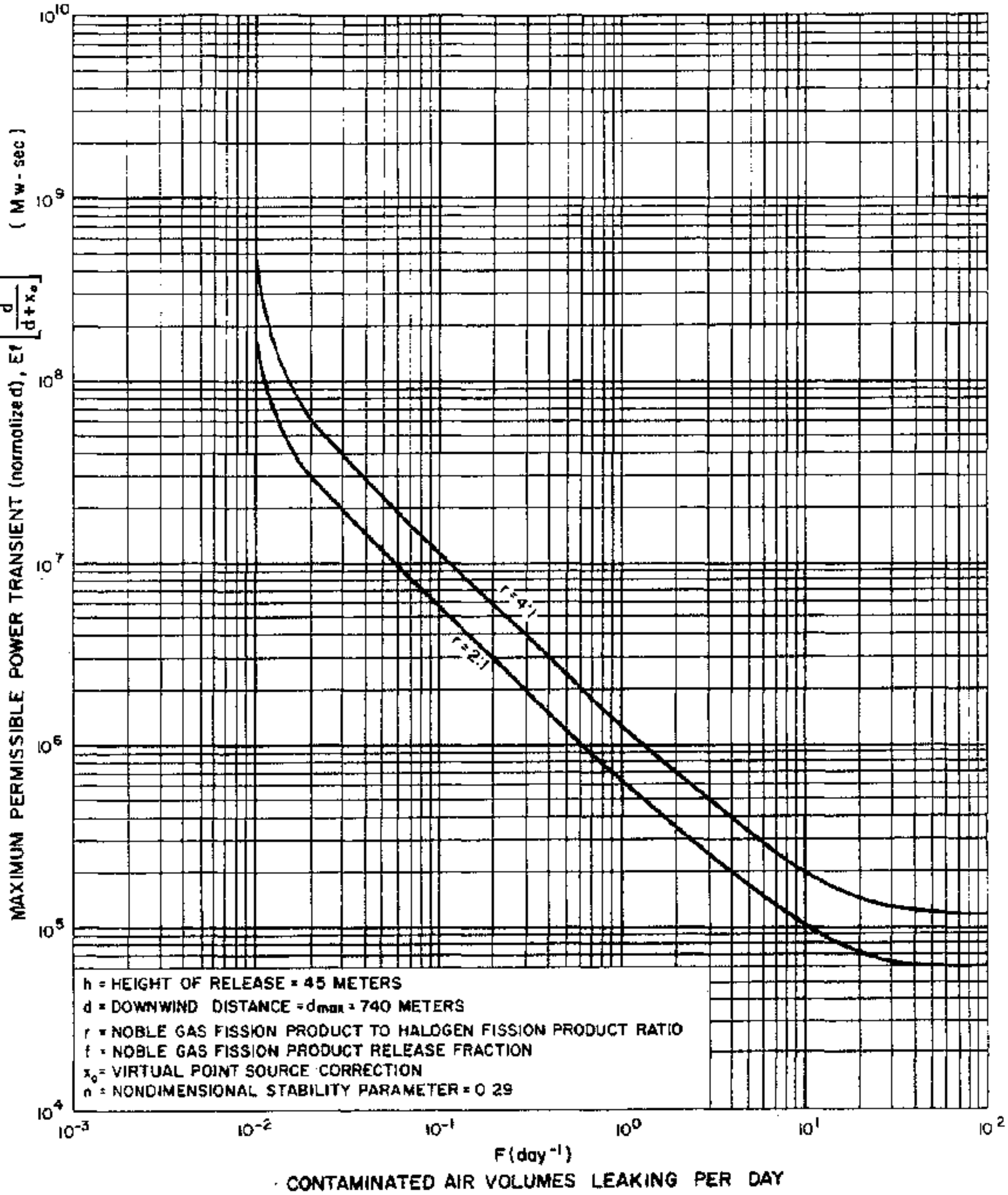

Figure III-9. Maximum Permissible Total Integrated Transient as a Function of Leakage Rate for 45 Meter Stack Release (Based on Dose Requirements for Site Boundary)

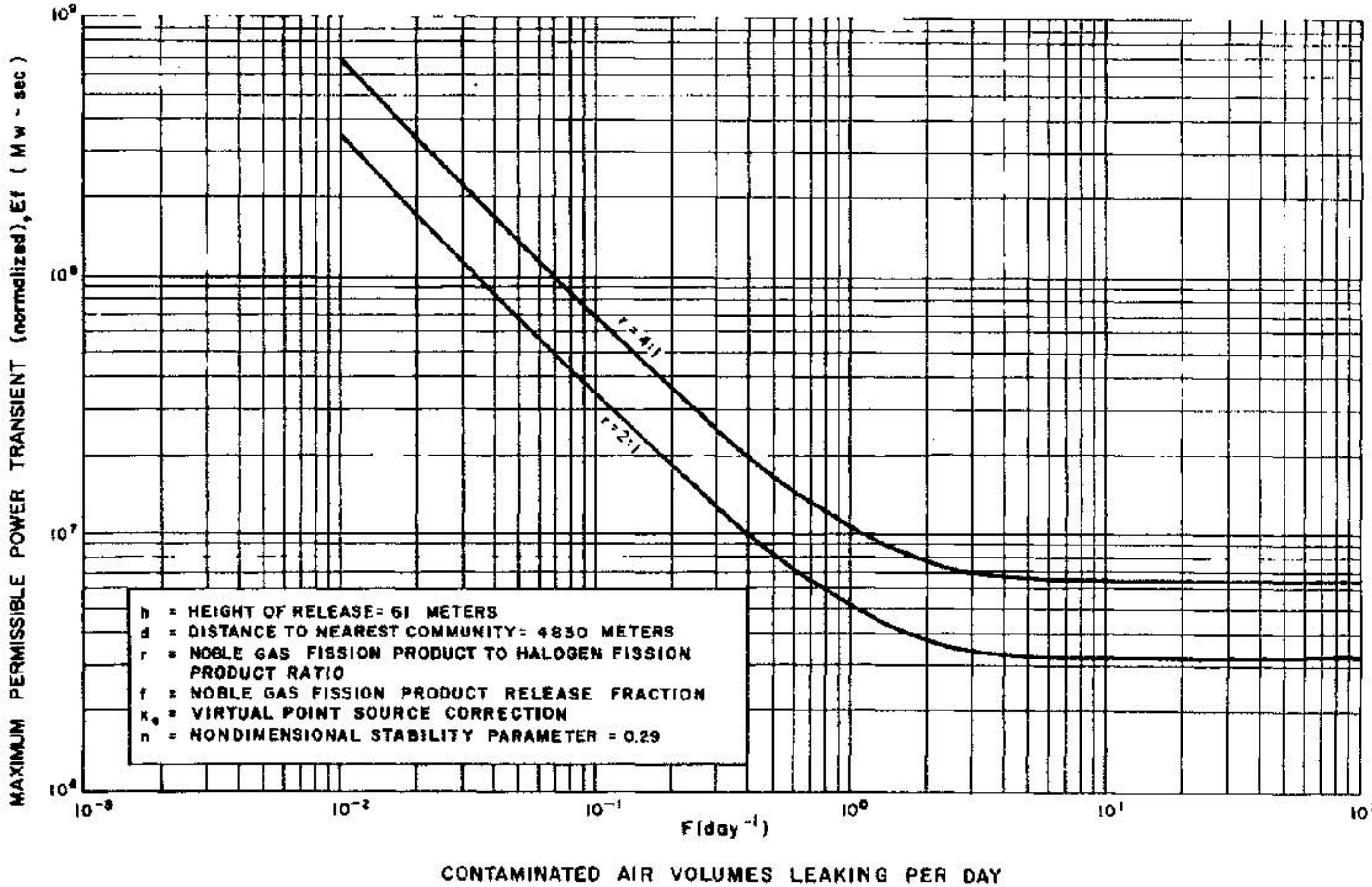

Figure III-10. Maximum Permissible Total Integrated Transient as a Function of Leakage Rate for 66 Meter Stack Release (Based on Dose Requirements for Nearest Community)

IV. DEVELOPMENT OF INTERACTION CRITERIA FOR MULTIPLE REACTOR FACILITIES ON THE SITE

A. GENERAL CONSIDERATIONS

The development of appropriate criteria upon which to judge the adequacy or suitability of a site when it contains multiple reactor facilities is a subject which heretofore has not been treated in the general literature. Much has been done, however, in the way of developing such criteria for a site which is to house either a single reactor or a series of interconnected reactors. Efforts along this latter regard have culminated in the now-revised Reactor Site Criteria (10 CFR Part 100).* The criteria set forth therein are also applicable to evaluation of the case where the multiple reactors are independent; Part 100 essentially requires that the criteria be fulfilled for the single reactor whose accident would require the largest distances to the different population and control zones. However, Part 100 is only concerned with the problem as it affects the general public, and, therefore, does not consider the problem of on-site effects.

In establishing criteria for evaluating a site on which are located multiple independent reactor facilities, it is apparent that criteria in addition to those set forth in the regulation would have to be met. First, it must be clearly recognized that the main factor to be considered is that of the interaction between a reactor facility which suffers its maximum credible accident and other facilities or operations in close proximity thereto. Here it is important to emphasize that, due to the independent and diversified nature of the programs being carried out at the site, it would be desirable to continue as many as possible of those other activities which were not involved in the accident. Obviously then, it would also be desirable that a minimum of physical damage (e.g., contamination) result to facilities and equipment being used on these other programs. Interaction, then, is brought about mainly by two causes: the dose which would be received by personnel on-site immediately following the accident, and that received in the days following the accident due to residual effects, particularly ground contamination. It is these two sources of radiation which then must be covered by the criteria to be established. However, to some extent at least, any consideration of such criteria must be related to the amount of delay which could be tolerated in a given program, since this delay period may indeed vary for the different activities performed on the site. In the case of the dose from fallout, it becomes

*As posted April 5, 1962

evident, then, that there is a direct relationship between the amount of program interruption which can be tolerated and the acceptable contamination level. Further consideration of this point will be provided in the material below.

There are two other requirements which, although not so amenable to calculation, appear to result in factors which also should be considered in the evaluation of a multiple reactor facility site. The first is concerned with the general arrangement of facilities on the site. In this regard, it is felt that, in cases where wind directions are predominant, other facilities should be located across from these directions. The second requirement arises from the fact that, with a large site which contains numerous facilities, there must of necessity be numerous local emergency assembly (evacuation) areas at which personnel from specific facilities assemble in case of an emergency requiring local facility evacuation. As in the case with location of facilities, it is felt that these emergency assembly areas should generally be located so that evacuation need not occur downwind of the most probable path(s) to be taken by the radioactive effluent. This should not, however, be construed to mean that a site not having these attributes is unacceptable. Other means could be developed to overcome these deficiencies. This will also be discussed below.

B. CRITERIA FOR EXPOSURE DURING NORMAL OPERATION

The exposure limits for personnel on site due to normal operation of facilities releasing radioactive material are adequately contained in presently existing regulations. Normal practice is to treat these limits as being applicable for a single facility.

With a multiple facility site, one might consider some reduction in these values as being appropriate. Such a reduction seemingly would appear appropriate in instances where simultaneous releases occur and poor conditions for atmospheric diffusion exist. In this regard, it can first be stated that the poor conditions necessary to restrict atmospheric dispersion of radioactive effluents occur rarely on the site. Secondly, radioactive releases from the reactor facilities on the site (about half of which are critical facilities) are, for the most part, non-routine with relatively long intervals between such instances. In addition, all reactor facilities which are potentially capable of routinely releasing fission products are provided with high efficiency filters and/or hold-up (decay) tanks. Where hold-up tanks are provided, releases of radioactivity are controlled by the Health and Safety Unit at the site. Having this control they also schedule these releases so that several facilities do not release simultaneously. For other facilities, the system design requirements are such that routine radioactive releases would be small.

In all cases, the magnitude of the radioactivity concentration at the stack exit is such that little credit is taken for atmospheric dilution. In most cases, the concentration is less than MPC. Even considering the incidence of the worst meteorological conditions, averaging over the 1-year period, as set forth in the applicable regulations, will result in compliance therewith since the period of the release is finite.

Therefore, it can be said that, since the AI facilities are designed to adhere to already-established criteria for radioactive effluents and radiation exposure, with effluents being controlled to minimize simultaneous releases, personnel safety is assured during periods of normal operation. Hence, there need be no concern with regard to lowering the already-established permissible release concentrations.

C. EXPOSURE LIMITS FOR EMERGENCY SITUATIONS

1. General

As discussed earlier, there are two sources of radiation exposure which should be considered. The first of these is that received during the first few hours after the accident, during which time local or site evacuation may be involved. The second is that which would be received following resumption of routine operations in those facilities surrounding the building in which the accident occurred. This latter exposure would result primarily from the radioactive materials deposited on the ground during passage of the cloud. (Delayed exposure from inhalation of the cloud during its passage is considered, in this case, as an immediate exposure, even though it is the TID which is evaluated.) In any event, it would appear that criteria should be established to govern both the early radiation exposure and that received following resumption of normal activities.

2. Early Exposure Criteria

In establishing the criteria for exposure as a result of an accident, the total dose values decided upon were 25 rem whole body and 300 rem thyroid from iodine exposure. The 25 rem value is that of the NCRP (see NBS Handbook 69*) and is the once-in-a-lifetime accidental dose which can be received by radiation workers without affecting their radiation exposure status for normal operations. The 300 rem thyroid exposure is similarly utilized. In both cases it is recognized that these values were intended as reference values and could be used in the evaluation of the degree of personal risk involved only when the probability of occurrence is exceedingly low and the number of people involved is similarly small.

There are several relevant considerations worth dwelling on at this point. First, admittedly, criteria could be developed which would require lower total exposures than those indicated above. The argument for this stems, for the most part, from the fact that the site contains numerous facilities, each operating independently of the other, whose maximum credible accidents are also independent. In this regard it can first be stated that the probability of occurrence of the accidents which have been considered is extremely small. The probability of more than one such accident occurring simultaneously is obviously much smaller. In addition, the different reactor facilities are located such that two such simultaneous accidents will not, in general, expose the same area to the maximum doses calculated. It should also be noted that the doses resultant from these accidents do not always result in exposures which are close to the total dose values specified above unless the receptor is located on or relatively close to the cloud centerline. Since in the case of on-site exposure, there will be little chance for much lateral cloud diffusion, the cloud width will be relatively small, and, as a result, if the receptor were off the centerline, there would be some reduction in the dose received.

There is still another consideration relative to this point; namely, some element of risk is always involved and, even though a person may receive an accidental 25 rem whole body exposure, he is still permitted by currently accepted practice to return to his routine duties — where he still can potentially receive another such accidental exposure. Therefore, there appears to be an applicable precedent already established which would not require reduction of the

*"Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," Handbook 69, June, 1959. U. S. Department of Commerce, National Bureau of Standards.

total dose values chosen because of the presence of multiple potential sources of radioactivity.

A second consideration also arises out of the multiplicity of reactor facilities on the site: namely, what happens following such an accident where certain personnel on site have already received their once-in-a-lifetime emergency dose? Here again, it can be stated that the probability of two accidents occurring during the work span of an individual, where each incident is of such magnitude as to result in a once-in-a-lifetime permissible exposure, is certainly far more remote than the occurrence of one. Nevertheless, facing this situation, it can be seen that a second such exposure would certainly require reconsideration of the person's overall radiation status insofar as the 5(N-18) cumulative permissible exposure is concerned. This in itself can raise questions, e.g., how to consider the relationship between a stock-puled backlog of a given number of rem and a short-term exposure of the same magnitude. In any event, one might contrive that some time off would be appropriate. This might conceivably affect the overall program schedules of those facilities whose personnel were so affected. However, much of Atomics International's efforts are carried on at its headquarters location in Canoga Park, which is sufficiently remote from the Nuclear Development Field Laboratory that personnel located therein would receive a negligible dose from the accident. As a result, there could conceivably be some reassignment of personnel to keep critical programs going in the remote event that such action would be required, thus minimizing the effect on program schedules.

It must be recognized, however, that the overall seriousness of such an occurrence would not be too great since, first, the exposures would not all be large fractions of the total dose values specified. Secondly, the general wind conditions at the site are such that the prevalent winds are directed across the narrow dimension of the site, this distance being of the order of 1500 to 2000 feet, whereas the length of the site is in excess of 1 mile. This latter fact would tend to reduce the number of facilities involved, thus reducing the number of people involved. From these considerations, then, it is felt that no reduction of the 25 rem whole body or 300 rem thyroid dose is required for the Atomics International Nuclear Development Field Laboratory.

The last point remaining is a specification of the time span over which the observer can be permitted to receive the total doses specified. Here it was decided to utilize, without change, the time criteria specified in 10 CFR Part 100: namely, that the exposure criteria would be satisfied if the observer were to remain stationary for the first two hours following the accident Therefore, the criteria used in this evaluation will require that the maximum credible accident in one facility not cause a dose in the first two hours at either the closest adjacent facility or the point of maximum downwind ground concentration (in the event of a stack release) in excess of 25 rem whole body or 300 rem to the thyroid. In evaluation of the thyroid dose, the total integrated dose is to be evaluated assuming a 2-hour inhalation period.

3. Contamination Criteria

As mentioned earlier, the development of acceptable contamination criteria must be related to the extent of delay which can be tolerated by the specific programs effected. In addition, it is evident that the longer the acceptable program delay before resumption of routine operations, the more time would be available for area and facility decontamination. This being the case, it is felt that a single criterion would be inappropriate. All that it appears possible to do, then, is to establish upper limits for the worst case but, in any event, to treat each accident on an individual basis insofar as the contamination problem is concerned.

In establishing the maximum acceptable upper limit, it was assumed that in the quarter following the accident, the maximum permissible dose should be no greater than 3 rem. Considering that the average permissible exposure from normal operations cannot exceed 1.3 rem in the quarter (i.e., at 100 mrem/wk), this leaves 1.7 rem. Therefore, exposure to a dose of 1.7 rem in the quarter following the accident will be taken to constitute the maximum acceptable interaction criteria for the dose from residual ground contamination. Since, in any major accident of the magnitude being considered, evacuation will take place not only from the facility in which the accident occurred, but from surrounding facilities as well, it will be assumed that re-entry into the latter, in order to resume normal operations, will not occur for at least 24 hours after the accident. Hence, the quarter will be considered to start at that time, i.e., 24 hours after the accident. If the 1.7 rem value is exceeded in this period, interaction will be said to have occurred and an unacceptable situation considered to exist.

The actual extent of permissible ground contamination will, of course, depend on (1) the type of material involved, and (2) the extent of delay which can be permitted before it would be required to resume normal operations. Since the absolute magnitude will depend on the type of material dispersed on the ground, e.g., gross fission products, only halogens, or sodium, and since the amount of delay permissible will vary with the different programs, it is apparent that the dose criteria cannot be related to a specific contamination level. Therefore, since such a criteria is only appropriate on a case-by-case basis, specification of contamination limits will be handled on an individual basis in each accident study performed.

4. Other Considerations

In addition to the criteria set forth above, there are two additional factors which should be considered in either laying out a site which is to contain multiple reactor facilities or expanding such a site. First, it would be desirable, in the locating reactor facilities, that the prevalent wind conditions at the site be taken into account. For sites where predominant wind directions prevail, the reactor facilities should be located so that the number of other facilities located down-wind in the prevalent directions is minimized. To some extent, at least, there can be some trade-off of downwind facility density with distance, i.e., the further from the reactor, the greater the permissible number of downwind facilities. However, in any event, the dose criteria established in the earlier portions of this section must be met so that with the occurrence of a maximum credible accident at the reactor facility, there would be no undesirable radiation exposures at the facilities in question.

A second factor which should be considered is the location of emergency assembly areas to which people are to evacuate in the event that facility evacuation is required. Here it is felt that, because of the expanse of a multiple reactor facility site, it would be appropriate that an emergency assembly area be established for each sector of the site. In this manner, all of those facilities in that sector could evacuate to a common assembly area. The location should be relatively convenient to the facilities whose personnel would evacuate thereto in order not to require traveling too great a distance in an emergency. The distance is certainly arbitrary, but should at least be great enough that there would be a negligible exposure from any fission products remaining in the reactor facility.

In locating these emergency assembly areas, it would certainly be desirable, as with the location of other facilities, to consider the prevalent wind directions on the site and, hence, locate these areas where they would not be downwind, with the prevalent wind directions. It has been suggested by some that one possibility which would eliminate evacuation to a downwind location would be the selection of a number of emergency assembly areas for each sector of the site. In this way, evacuation to an upwind assembly area could be effected by simply noting the wind direction at the time evacuation is ordered and specifying to the personnel involved the assembly area to be used. The main objection to this scheme arises from the fact that, although multiple assembly areas could solve the problem at hand, their presence would probably create confusion in the minds of the evacuees. This confusion would arise since the evacuees would be required to know not only the location of each assembly area but also each evacuation route as well. In addition, this scheme would require the evacuee to hear not only that an evacuation is required but, in addition, which assembly area is to be used. Lastly, and what is perhaps most important, it was felt that evacuation to the assembly area should be instinctive, with as little thought as possible required; the use of multiple assembly areas would reduce significantly the degree of instinct involved. Together with the potential confusion which could exist, it was felt that the single emergency assembly area concept was superior.

One approach to minimizing exposure, once personnel have assembled at an evacuation area, would be to locate the area so that rapid evacuation to another assembly area would be feasible. Therefore, as an example, if the original assembly area were a parking lot used by the personnel in the facilities concerned, they could use their cars and quickly proceed to a second area upon receiving instructions from proper authorities. Such a procedure would certainly minimize the amount of additional radiation exposure to evacuating personnel and would also be consistent with the maintainance of a minimum of confusion during such occurrences.

It is well to indicate that compliance with these two requirements becomes difficult when several prevalent wind directions exist. For the case of a relatively small site or a site with a high facility density (all types), one cannot generally conform to both of the requirements, mainly due to the limited amount of available area. This points out that, especially with small sites and/or high facility density sites, careful attention should be given to the layout of facilities so that both the degree of interaction between facilities and the amount of unneccessary radiation exposure received by personnel are minimized.

V. PHYSICAL DESCRIPTIONS OF REACTOR FACILITIES ON SITE

A. INTRODUCTION

To provide the reader with some insight into the design, and to facilitate an understanding of the manner in which radioactivity is released in maximum credible accidents of the reactor facilities on the site, this section will provide a brief description of each building containing a reactor to be evaluated in this study. The information presented will place major emphasis on those aspects of the design more closely associated with safety, e.g., location and environment, ventilation, pressure containment capability, shielding, etc. For more detail, the reader is referred to the appropriate hazards reports which have been published and are referenced.

Information is also provided, in this section, on the current experimental program at each facility, and, where they are known with some degree of certainty, the future plans for use of the facility. The following list will aid the reader in locating specific descriptive material.

B. PHYSICAL DESCRIPTIONS OF FACILITIES

1. SNAP Critical Facility, Building 373

a. Location

Figure III-3 shows the location of the SNAP Critical Facility, Building 373, within the Atomics International Nuclear Development Field Laboratory (AI-NDFL) and its relation to other installations in the local area. The facility is located to the south of the SNAP complex, near the boundary separating the AI and Rocketdyne areas. The minimum distance from the facility to the AI exclusion area boundary is 420 feet (boundary separating AI from the Rocketdyne area to the east). The minimum distance to public land is approximately 1350 feet to the northwest (Simi Valley direction). The nearest occupied building is the Mechanical Component Development Building (Bldg 363) which is approximately 350 feet southwest of the facility (some open pad areas and test towers are located at distances closer to the facility, the nearest being the Control Rod Test Tower and pad (874) which is approximately 250 feet from the facility).

The facility floor areas are at an elevation of approximately 1810 feet above sea level, and the test cell exhaust stack is 1840 feet above sea level.

b. Description of Building and Equipment*

A major portion of the building was originally built for the testing and handling of highly explosive solid rocket fuels. Figures V-1 and V-2 illustrate the design of the facility as subsequently modified for reactor tests at low power. The building is constructed on a concrete foundation and floor slab and covers 2500 ft^2 of floor area (63 feet long and a maximum width of 45 feet). The building walls are of concrete and cemesto construction. The flat roofs have an eave height of 10 feet (except for one room which has an eave height of 14 feet).

The test areas are separated from the administrative and general operating areas by a concrete shield wall. The test area building is approximately 62 feet long by 14 feet wide and contains a 14 by 11-foot fuel storage room, a 14 by 12-foot reactor test cell, and three adjacent cells (14 by 12 foot rooms) which may be used for miscellaneous non-nuclear tests. The cells were

* O. D. Seawell, ed., "Special Purpose Power Plant Critical Facility Summary Hazards Report," NAA-SR-Memo-1946, May 15, 1957 (classified). See also Addendum 1, December 1, 1958; Addendum 2, March 11, 1959; Addendum 3, January 15, 1960; Addendum 4, July 31, 1961.

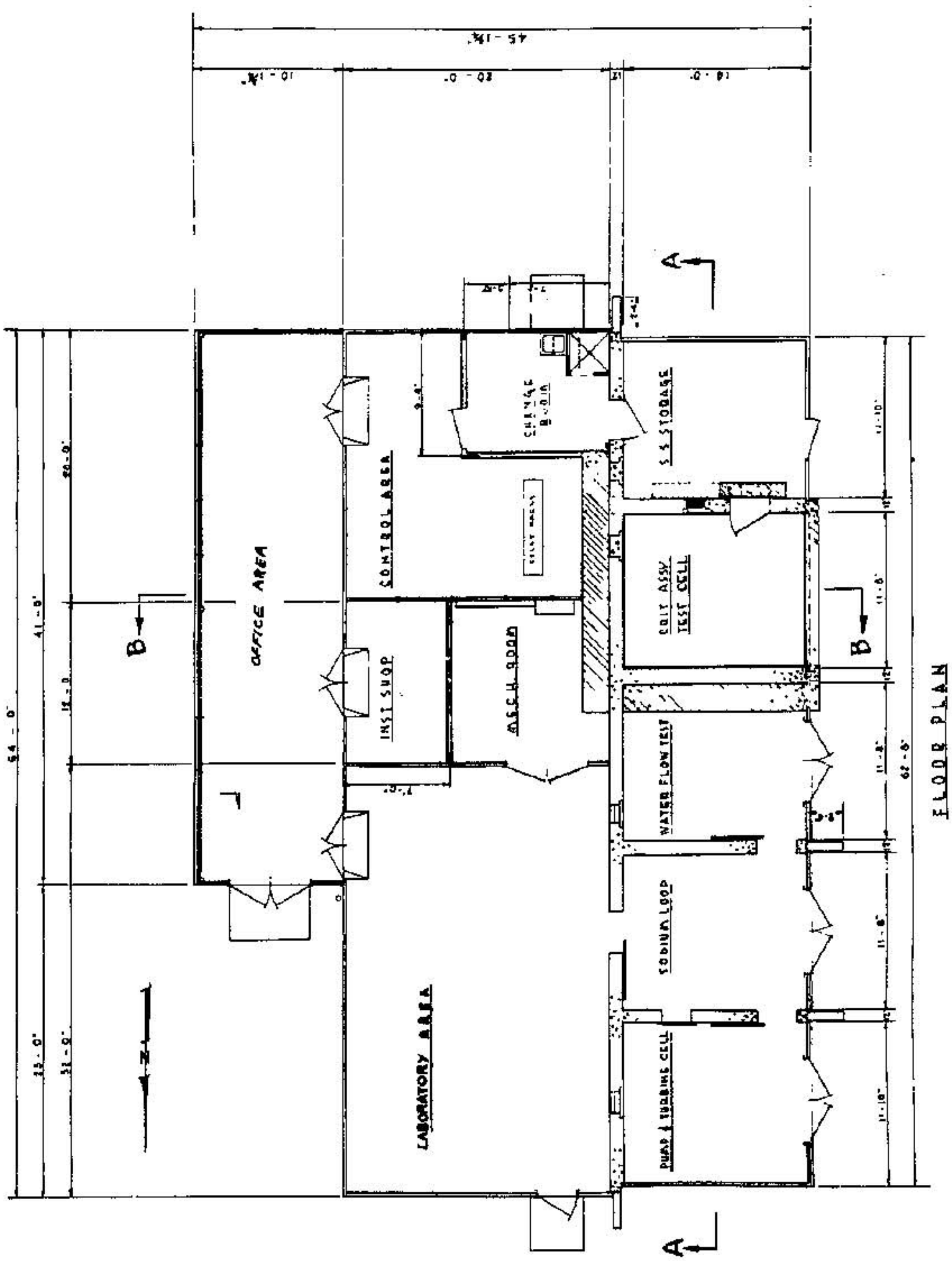

Figure V-1. Plan View of SNAP Critical Facility

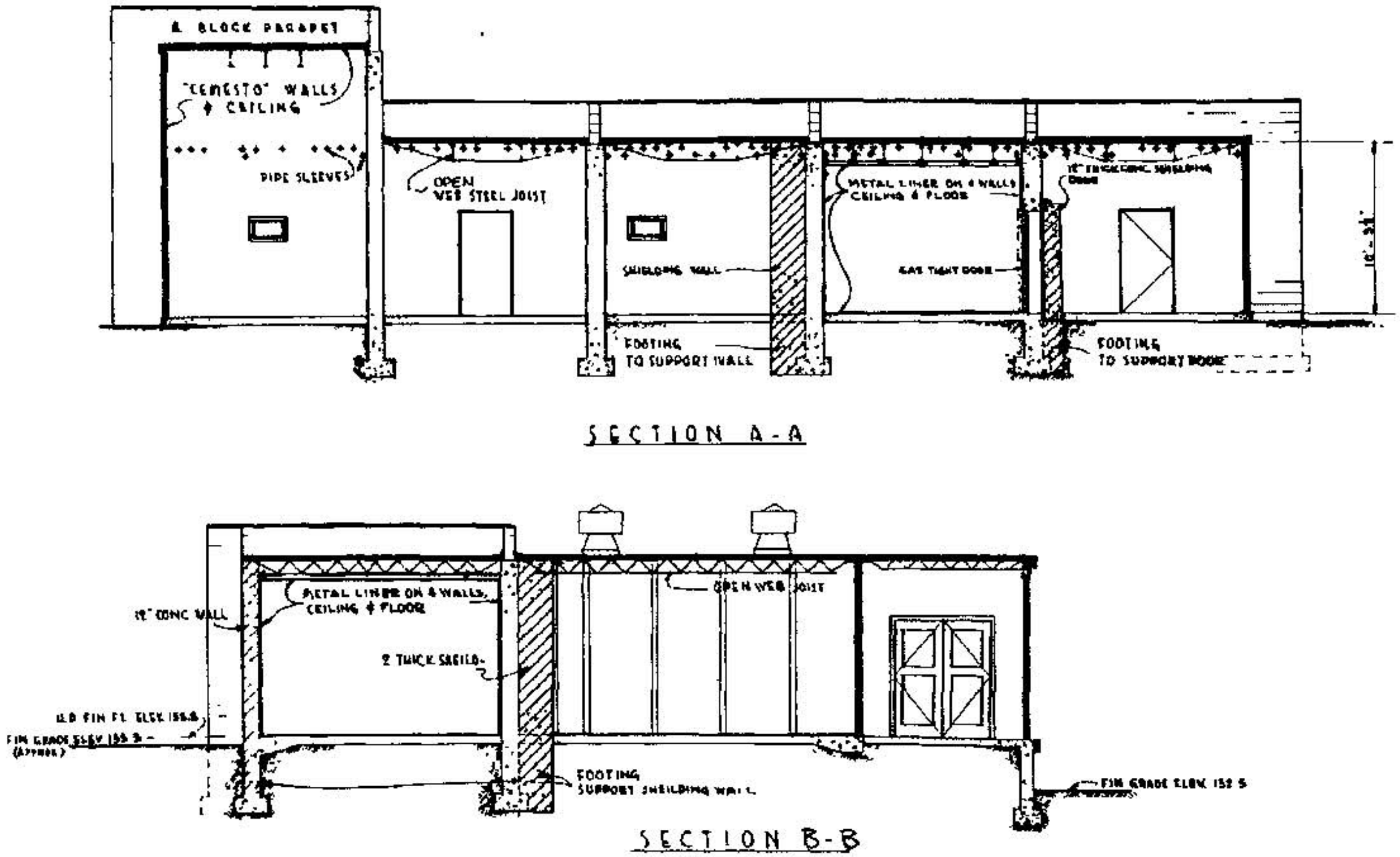

Figure V-2. Sectional View of SNAP Critical Facility originally designed as explosion blowout rooms and were provided with 1-foot thick concrete inside walls. An additional 2 feet of magnetite concrete has been added to the reactor test cell wall adjacent to the administrative area and 2 feet of ordinary concrete on the wall separating the reactor test cell and adjacent test cell. The facility roofs consist of metal joists and cemesto panels with built-up roofing material (no shielding).

The reactor test cell is provided with a gas-tight door and a welded steel lining on all four walls, the ceiling and floor. The cell liner is designed to withstand a pressure differential of 0.5 inches of Hg (0.25 psig). A leakage rate of 0.19 cfm at 0.15 inches of H_2O was determined during static leak rate tests.

The administrative and general operating area, located on the opposite side of the shield wall from the test area, consists of a 9 by 11-foot change room leading to the fuel storage room and main test cell, a 20 by 20-foot L-shaped control room (300 ft^2), a 20 by 32-foot laboratory area, a 7 by 12-foot instrument shop, a 12 by 11-foot equipment room, and a 10 by 41-foot office area. A door interlock system prevents access to the change room, fuel storage room, or reactor test cell when critical experiments are in progress. The entire facility site is surrounded by a 7-foot chain-link security fence, and an inner fence is also provided to prevent use of the areas west and south of the main test cell, where walls may not provide adequate shielding in the event of an excursion.

All sections of the building, except the reactor test cell, are heated and ventilated by conventional methods, with air discharging to the atmosphere through roof fans or gravity ventilators. The main-test-cell-ventilation system employs a combination air-filtration and air-conditioning system to enable recirculation of the cell atmosphere or discharge of the atmosphere to a stack. During normal operation, and with the cell sealed, a fan recirculates the cell atmosphere through a bank of pre- and "absolute" filters and a cooler (or heater) at a rate sufficient to turn over one room volume every 3 minutes and maintain a constant temperature of 75°F. To maintain a pressure differential of -0.25 inches of H_2O in the cell, a small stream of the cell atmosphere is drawn through a pressure-regulated exhaust valve to an exhaust blower and a 30-foot stack (roof mounted stack with exit 18 feet above roof). A dilution intake line on the blower maintains a flow rate of approximately 1000 cfm of air at the stack exit. When the reactor test cell door is opened, the ventilation system is converted automatically to a once-through system. Approximately 500 cfm of air are drawn through the cell and filters and then mixed with 500 cfm of dilution air prior to discharge to the stack (recirculation fan off).

c. Description of Experimental Program

Several critical experiments and tests on critical assemblies have been completed in the facility. A brief description of the tests and critical assemblies are listed below:

1) SCA-1. Initiated October 1957. This assembly consisted of a pseudo sphere of zirconium hydride-enriched uranium dioxide blocks. Basic reactor parameters of the SNAP 2 reactor concept were determined.

2) S2ER Critical Experiment. Initiated June 1959. The S2ER components were assembled and preliminary tests conducted at zero power.

3) SCA-2. Initiated about September 1960. A clean, cylindrical geometry core was studied, using the core and reflector components from the conduction-cooled SNAP 10 reactor.

4) SCA-3. Initiated October 1961. An assembly of about a 1 ft^3 core volume was built to study the characteristics of the SNAP 4 reactor. Plate-type fuel elements and a water coolant were used.

The experiments proposed for this facility are as follows:

1) SCA-4C. This assembly will use prototype SNAP 2/10A core and reflector components, as a final design check on the flight reactor system. The test is tentatively scheduled to begin in mid-1962.

2) S8ER Critical Experiment. The S8ER components will be used to determine the nuclear parameters of the SNAP 8 reactor design. Final adjustments will be made on the reflector shims prior to

installation of the reactor in the power test facility (Building 010). The test is tentatively scheduled to begin in mid-1962

3) SCA-4A. This assembly will provide a very flexible reflector assembly to permit optimum design of the reflector controls for the SNAP 2/10A reactor. This test is tentatively scheduled to begin late in 1962.

d. Future Plans for Use of Facility

At the conclusion of the above experiments, it is proposed that Building 373 be deactivated and that the critical experiment program be continued in Building 012.

2. SNAP Generalized Critical Facility, Building 012

a. Location

Figure III-3 shows the location of the SNAP Generalized Critical Facility, Building 012, within the AI-NDFL and its relation to other installations in the local area. The facility is situated near the northwest boundary of the AI-NDFL within the area known as the SNAP complex. The installation is located on AEC optioned land. The minimum distance from the facility to the AI-NDFL boundary (AI exclusion area boundary) is approximately 250 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1350 feet to the southeast. The nearest occupied structure is the SNAP Non-Nuclear Component Assembly and Performance Test Facility (Building 013) which is approximately 80 feet west of the reactor.

The facility floor areas are at an elevation of 1814 feet above sea level, and the stack exit is approximately 1854 feet above sea level.

b. Description of Building and Equipment*

Figure V-3 shows an isometric view of the facility, which is scheduled for completion in mid-1962. The building consists of two, major, above-grade structures connected by an enclosed passageway. To the north of the passageway is a concrete structure approximately 46 feet long by 28 feet wide with a flat deck roof 14 feet high at the eave. The structure is built on reinforced concrete footings and floor slabs and contains a 20 by 20-foot by 10-foot-high shielded, critical facility cell and a 26 by 18-foot room for the assembly of experimental equipment and the storage of fuel. Fuel is to be stored in cadmium-plated tubes imbedded in a 20-inch-thick, 1% borated, concrete wall. The structure to the south of the passageway is approximately 42 feet long by 28 feet wide.

The building is a prefabricated, rigid, steel-framed structure with corrugated metal sidings and a gable roof approximately 10 feet high at the eave and 14 feet high at the ridge. The building is located on continuous wall footings under exterior walls with spread footings and piers under all columns and a concrete floor slab. The building contains an 11 by 11 foot change room, a 15 by 24 foot instrument and experiment room, and two 9 by 10 foot offices. The

*A. W. Thiele, ed., "SNAP Critical Facility (Building 012) Summary Hazards Report," NAA-SR-MEMO-7205, April 6, 1962

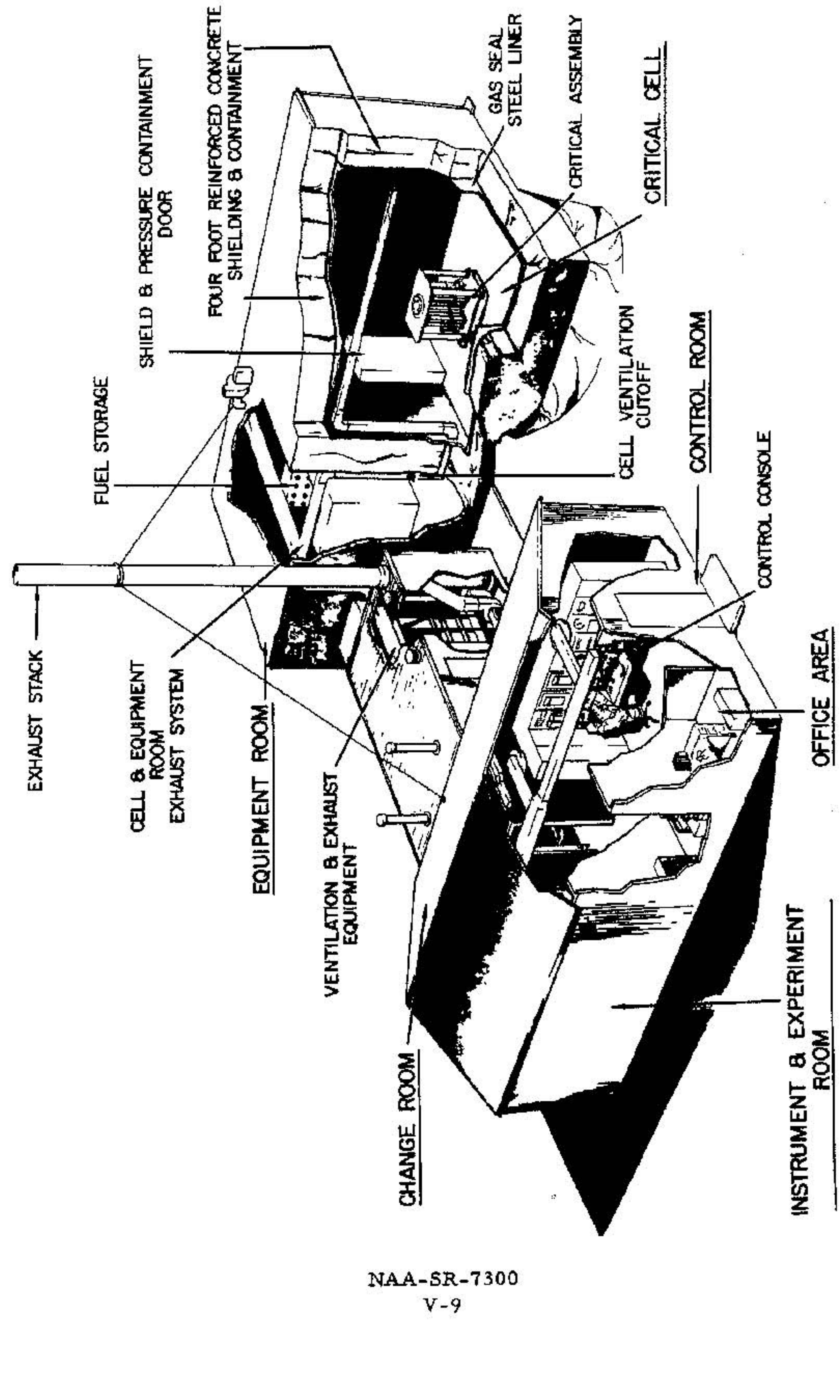

Figure V-3. Isometric View of SNAP Generalized Critical Facility passageway separating the two buildings is approximately 34 feet long and 6 feet wide. An 11 by 9 foot fan and filter room and a 15 by 9 foot electrical equipment room are located to the south side of the passageway.

The critical facility cell is a shielded room with 4 foot thick reinforced concrete walls and roof lined with a 1/4-inch steel plate. The room is sealed during operation and is designed to withstand a short-term overpressure of 100 psi without excessive deformation. A gas-tight, steel, shield door provides both the required shielding and pressure containment. The door and all penetrations are sealed to ensure that leakage from the cell will be less than 1% of the cell volume in 24 hours at a 3-inch H_2O differential. The door and all cell penetrations terminate in the adjacent equipment and fuel storage room, which is maintained at a negative pressure with respect to the cell and is exhausted through "absolute" filters and the facility stack. This arrangement provides secondary containment and control over activity leakage from the cell to the environment.

The control room and support areas are provided with conventional heating and cooling equipment for instrumentation and personnel requirements. The fuel storage and equipment room, and the critical facility cell are provided with exhaust systems, to maintain negative pressures in these areas under normal operating conditions. When the critical facility cell is in operation, the exhaust duct is valved shut. When access to the cell is desired, the cell can be vented to the stack. The entire facility is surrounded by a security fence to provide an exclusion area and to control access to the facility.

The critical assembly requirements used in the experimental program will be of varied design, in accordance with the particular experiments being conducted. In general, requirements for handling a small core volume, between 1/3 and 1 ft^3, and means for reflector control will be provided. The control room is instrumented for the operation of critical assemblics with 5 channels of nuclear safety instrumentation, instruments for the critical assembly experiments, and other supporting equipment such as radiation monitors, closed circuit television, reactor control circuits, and temperature instruments.

c. Description of Experimental Program

The program at the facility will consist of studies of nuclear parameters of reactors under development in the SNAP program. The critical loadings, reflector worths, control characteristics, temperature effects, and similar reactor parameters will be studied. The reactors will be operated at relatively low power levels to provide these data. The facility is scheduled for beneficial occupancy in mid-1962. The presently scheduled program includes studies of the SNAP 2/10A reactor, the SNAP 4 reactor, and advanced metal hydride and UC reactors.

d. Future Plans for Use of Facility

The facility will continue to be used for critical assembly studies of reactors under development on the SNAP program. The exact nature of these reactors cannot be specified further at this time.

An addition has been proposed to Building 012. This addition will provide a second critical facility cell and equipment room. These will be of a construction identical to that of the existing facility. Control room, office space, and a mock-up area will be added to provide support for the second cell. The experimental program will be similar in nature to that in the original cell. It is anticipated that this facility would be required by 1963. Since the addition will be physically attached to the existing facility, it will bear essentially the same relationship to the surrounding environment.

3. SNAP 8 Experimental Reactor (S8ER), Building 010

a. Location

Figure III-3 shows the location of the S8ER test installation (Building 010) within the AI-NDFL and its relation to other installations in the local area. The facility is situated near the northwest boundary of the AI-NDFL within the area known as the SNAP complex. The installation is located on AEC optioned land. The minimum distance from the facility to the AI-NDFL area boundary (AI exclusion area boundary) is approximately 300 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1350 feet to the southwest. The nearest occupied structure is the SNAP Generalized Critical Facility (Building 012), which is approximately 60 feet west of the reactor.

The reactor room and control room floor elevations are approximately 1812 feet above sea level, and the stack exit is 1862 feet above sea level.

b. Description of Building and Equipment*

Figure V-4 shows an isometric view of the facility as improved and modified to its present status to accommodate a SNAP 8 reactor scheduled for testing at power levels up to 600 kwt. This building was originally designed to accommodate a 50-kwt SNAP 2 experimental reactor, for power demonstration and endurance tests. These tests were completed in 1960. The building consists of a single, rigid, steel-framed superstructure with corrugated metal siding and blanket insulation. The building is 60 feet long by 24 feet wide with a 17-foot eave height. The superstructure contains a control room, mezzanine office space, and change room in the north end of the facility and a reactor room in the remaining portion (south 34 feet). The building foundation is of spread footing and grade beam construction, with a 6-inch mesh, reinforced, floor slab. The floor slab is tied into the column footings, to develop the rigid frame horizontal reactions. Some secondary coolant system equipment, the necessary power service, and a 7-1/2 ton bridge crane are located above the reactor room floor level. The 50-foot building ventilation stack, secondary NaK surge tank and air blast heat exchanger, and the equipment associated with shield-cooling systems and helium- and nitrogen-supply systems are located on concrete slab areas

*A. R. Piccot, ed. "SNAP 8 Experimental Reactor (S8ER) Final Safeguards Summary Report," NAA-SR-6958, February 28, 1962 (classified) and Addendum I, by V. Rooney, ed., April 15, 1962

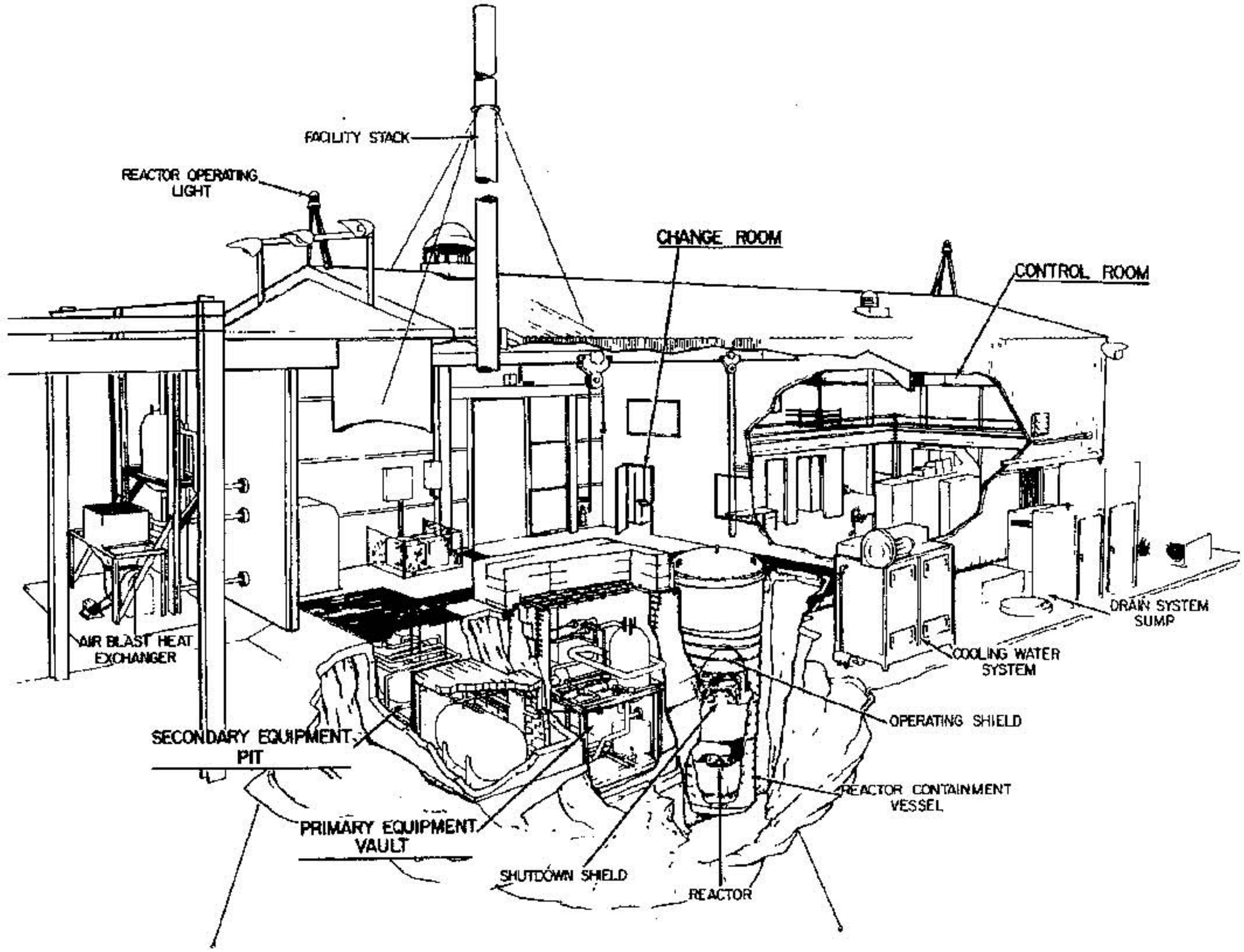

Figure V-4. Isometric View of SNAP 8 Experimental Reactor exterior to the building. The reactor room is vented directly to the stack (no filters) at a nominal rate of 5000 cfm. The reactor and containment vessel, primary NaK coolant system vault, and secondary NaK system equipment pit are located below the reactor room floor level.

The entire reactor and the control systems are contained in a shielded carbon-steel pressure vessel (containment vessel). The vessel is embedded in concrete below the reactor room floor level, with the removable vessel head (dome) protruding above the floor level. Internal shielding is provided below the vessel dome to attenuate radiation escaping from the contained reactor systems. The containment vessel is sealed, and helium at a slightly positive pressure is maintained in the vessel during reactor operation. The primary system coolant lines penetrate the containment vessel through welded seal caps and are then routed to a shielded, steel-lined vault, where the heat is transferred from the primary NaK coolant loop to a secondary, nonradioactive NaK coolant loop. A rupture diaphragm is provided on the reactor inlet coolant line in the containment vessel to prevent failure of the piping system outside of the containment vessel, in the event of system overpressure. In this manner, system overpressures will result in rupture disc relief in the containment vessel and, hence, possible pipe rupture in the primary system vault, due to overpressure, will be prevented. The containment system is designed to withstand 100-psig operation and ensure a leak rate of less than 1% per day at a 68 psig overpressure. A buried, gas hold-up tank is provided which can be used to reduce the overpressure and leak rate following potential accidents.

The primary system vault is a 9-1/2 by 8 by 11-foot deep room with the ceiling of the room at the same level as the reactor room floor. Shielding with an effective thickness of 3 feet of high density concrete (6200 lbs/yd^3) is provided for personnel protection. The vault is designed for a nominal internal pressure of 3/4 psig and will be filled with nitrogen maintained at a positive pressure of 3 inches of water during normal operation to avoid possible NaK and/or hydrogen-oxygen reactions.

The SNAP 8 core is a compact, hydrogen-moderated, beryllium-reflected assembly containing highly enriched uranium. The core contains 211 individual fuel rods, consisting of homogeneously combined zirconium, uranium, and hydrogen to form a hydrided Zr-U alloy. Each fuel rod is provided with a ceramic barrier to prevent loss of the moderator as a result of hydrogen diffusion. The core is contained in a stainless-steel core vessel approximately 9-1/2 inches in diameter by 21 inches long. Heat is removed by eutectic NaK 78 coolant which enters a plenum at the bottom of the core at 1100°F and exits from a plenum at the top of the core at 1300°F. The primary coolant inlet and outlet lines are 2-inch diameter, stainless-steel pipe. The core vessel is surrounded by 3 inches of beryllium (nominal thickness) which forms the external radial reflector. Reactor control is accomplished by rotation of portions of this reflector (control drums) toward or away from the core.

c. Description of Experimental Program

The SNAP 8 program is being developed jointly by the Atomic Energy Commission (AEC) and the National Aeronautics and Space Administration (NASA). The AEC is responsible for the reactor development, nuclear safety, and nuclear systems testing, and NASA is responsible for the power conversion system development and the overall systems engineering. The S8ER test program is the first in a series of reactor tests to be performed prior to flight testing the SNAP 8 system in 1965. The S8ER test is a test of the core only, without power-conversion-system equipment. The objectives of the test are to (1) determine the operating characteristics of the reactor; and (2) demonstrate reactor performance at 300 kwt, with a reactor coolant outlet temperature of 1300°F. Subsequent tests will be conducted at 600 kwt leading to the development of a 60-kwe system.

This phase of the reactor test program is being performed to verify the SNAP 8 reference design and provide an experimental basis for any design improvement that may be indicated. Operation of the facility is scheduled to begin in mid-1962. Prior to startup of the S8ER in Building 010, critical experiment tests will be performed on the core in the SNAP Critical Facility, Building 373. The critical test will provide data for checking calculational techniques, desired physics measurements, and the calibrations necessary prior to installation in the power test facility. The S8ER tests should be completed in 1963.

d. Future Plans for Use of Facility

The use which will be made of the facility following completion of the S8ER tests has not been determined.

4. SNAP 8 Ground Prototype Test Facility (GPTF), Building 059

a. Location

Figure III-3 shows the location of the GPTF (Building 059) within the AI-NDFL and its relation to other installations in the local area. The facility is situated on AEC-optioned land near the northwest boundary of the AI-NDFL within the area known as the SNAP complex. The minimum distance from the facility to the AI-NDFL area boundary is approximately 250 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1400 feet to the southeast. The nearest occupied structure is the Flight Systems Nuclear Test Facility (Building 019) which is approximately 60 feet northeast of the reactor.

The reactor room and control room floors are at an elevation of 1815 feet above sea level, and the stack exit is approximately 1900 feet above sea level.

b. Description of Building and Equipment*

The GPTF is designed to accommodate the ground testing of the 600-kwt SNAP 8 reactor systems, including their complete electrical generating equipment. Figure V-5 shows an isometric view of the facility (as presented in the preliminary hazards summary). The facility is scheduled for completion in 1962. The superstructure consists of a main high-bay building, with attached low-bay structures on opposite sides for the housing of general support and operating functions. The buildings are rigid, steel-framed structures with insulated metal roof-decking and siding.

Foundations consist of continuous wall footing under bearing walls, with spread footing and piers designed to minimize differential settlement under all columns. The high bay structure is 62 feet long by 32 feet wide, with a 32-foot eave height. The area is serviced by a 10-ton bridge crane. The high bay area provides unobstructed access to the test vault located below the reinforced concrete floor. Access to the test vault (which contains the reactor and test systems) is through hatches and ports with removable sealed plugs or through a stairwell at the southeast end of the high bay which leads to a sealed door at the vault floor level.

*A. R. Piccot, ed., "The SNAP 8 Development System (S8DS) Test Facility Preliminary Safeguards Study," NAA-SR-6181, September 1, 1961 (classified)

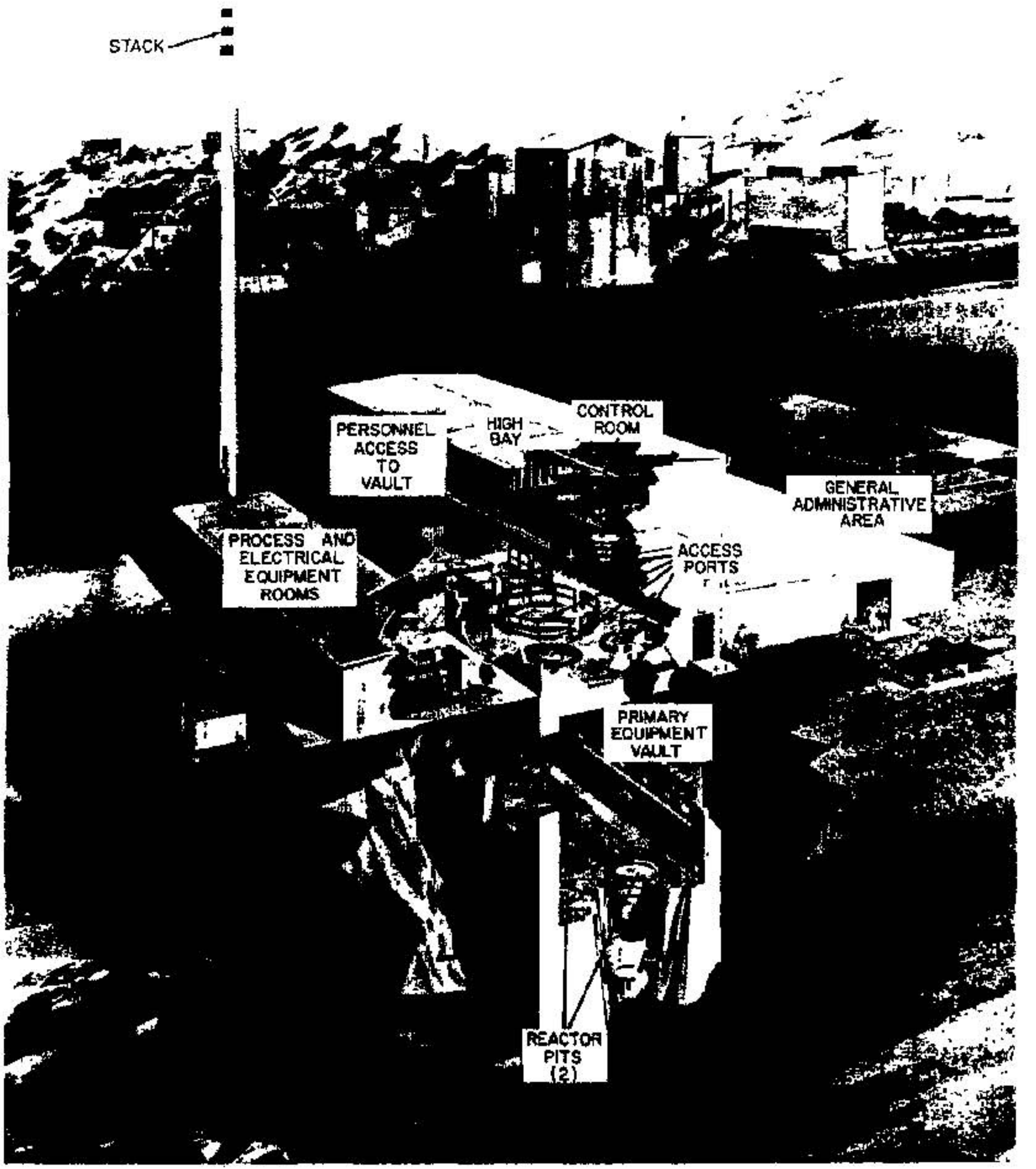

Figure V-5. Isometric View of SNAP 8 Ground Prototype Test Facility

The low-bay structure on the south side of the high-bay area contains general facility support and operating areas. The main structure is 62 feet long by 42 feet wide, with a 10-foot eave height. The area contains a 41 by 20-foot control room separated from the high bay area by a common concrete wall with windows which permit observation of activities in the high bay. A toilet area, locker room, and change room occupy approximately 400 ft^2 of floor space. The administrative area contains three offices and an 18.5 by 16-foot area for data reduction and instrument repair. A 20 by 10-foot facility equipment room is provided at the southeast end of the structure, for heating and air conditioning equipment and a water heater.

The low bay structure on the north side of the high bay area contains a process equipment room and an electrical equipment room. The structure is 62 feet long by 31 feet wide, with a 12-foot eave height. The process equipment room occupies a 30 by 40-foot area in the east end, and the electrical equipment room occupies a 30 by 21-foot area in the west end. The process equipment room is serviced by a 1/4-ton monorail hoist and contains service equipment required for control of the vault environment, portions of which may become contaminated (e.g., compressors, vacuum pump, filters, etc.).

The test vault is a shielded, steel-lined vault located below the high bay floor. The vault has a ceiling-to-floor depth of 32 feet and is approximately 40 feet long and 30 feet wide. Biological shielding is provided by 4 feet of ordinary concrete located above the ceiling. The vault is used for installation and operation of the basic components of the test systems. The contained portions include all components of the primary NaK loop, the boiler and rotating machinery package, the prototype radiator, and the reactor. The reactor will be installed in one of the two shielded reactor pits located in the west end of the vault. Removable plugs consisting of 4 feet of ordinary concrete are provided over the pits to reduce radiation levels in the vault. Each pit may be evacuated during reactor operation to further control environmental conditions surrounding the reactor and control system. A nitrogen atmosphere at 1/2 inches of H_2O negative pressure will be maintained in the test vault to prevent NaK and/or oxygen-hydrogen reactions. The nitrogen atmosphere will be cooled by a cooling coil and fans recessed in the vault wall. The vault is equipped with TV cameras and contains remote handling facilities for servicing the equipment in the vault.

The vault is designed to withstand a 4-psi overpressure and to provide a leak tightness which is capable of preventing loss of more than 1% of the vault volume in 24 hours under a differential pressure of 3 inches of H_2O. Following an accident, the leakage rate and overpressure will be reduced by compressors, which will subsequently keep the vault at a slight negative pressure and exhaust the gases to a buried hold-up tank (150-day capacity). All atmospheres released from potentially contaminated areas or equipment in the facility (high bay, vaults, hold-up tanks, etc.) are routed through pre- and "absolute" filters prior to release to the stack (nominal flow of 12,000 cfm during normal operation).

Some of the facility and process service system components are located on concrete pads outside of the building. An 85-foot facility stack is located just outside the northeast corner of the process equipment room. The secondary heat-transfer system airblast heat exchanger and radiator coolers are located on concrete pads in the same area. A kerosene heat exchanger used in conjunction with the vault nitrogen atmosphere cooling system is located on a concrete pad just north of the electrical equipment room. An area just north of the facility contains the electrical substation, liquid nitrogen storage, the buried contaminated gas hold-up tanks, and a cooling tower.

The first test system will be a complete, nuclear-powered, electrical generating system intended to produce 30 kwe (net) continuously for 10,000 hours of operation in space. The reactor will be a compact zirconium hydride, 10 wt % U (fully enriched) reactor controlled by beryllium reflector elements. (The reactor core is similar to the S8ER reactor described earlier in Section 3 above.) The reactor will be cooled with eutectic NaK 78, which enters the core at 1100°F and exits the reactor at 1300°F. The NaK then passes through a counter-current mercury boiler and back to the reactor inlet. The boiler produces superheated mercury vapor at 1200°F and 275 psia which then passes through a turbine (which drives the alternator), exhausting at 20 psia and 706°F. The turbine exhausts to a radiator-condenser which returns 560°F subcooled mercury to the boiler. The power conversion system is being developed for 30 kwe. The reactor is designed to produce 300 kwt to meet the heat source requirements for 30-kwe operation.

c. Description of Experimental Program

The SNAP 8 program is being developed jointly by the AEC and the NASA. The AEC is responsible for the reactor development, and NASA is responsible for the power conversion-system development and the overall system engineering. The S8GPT program is the second in a series of ground tests which will be performed prior to flight testing the SNAP 8 system in 1965. (The first test was described in Section 3 above.) The S8GPT will be used to test a breadboard configuration of the SNAP 8 system, consisting of the reactor, power conversion system, and the prototype radiator. The test program is tentatively segregated into four phases:

1) Power Conversion System (PCS) Shakedown Tests: In the first phase, one 30-kwe PCS, with prototype radiator and accessories will be installed in the test vault, and shakedown tests will be performed with a 375-kw electric heater as the power source.

2) Reactor Shakedown Tests: Following completion of phase 1, the SNAP 8 reactor will be placed in one of the two reactor pits, and connected to a nuclear system shakedown test loop. The primary NaK loop will remove the heat from the reactor. Heat will be transferred to a secondary NaK loop, which will be used in lieu of the PCS. Ultimate heat rejection will be by means of an airblast heat exchanger located outside of the building. The reactor shakedown tests will include checkouts ranging from zero to full-power operation (300 kwt).

3) Ninety-day Endurance Tests: Upon completion of the reactor shakedown tests, the reactor will be disconnected from the shakedown test loop and connected to the pretested PCS for the 30-kwe ground prototype system tests, which will include a 90-day endurance test.

4) Ten-thousand-hour Endurance Test: Following the tests with the PCS, a 10,000-hour endurance run is planned.

The objectives of the program are:

1) To provide early compatibility data for the nuclear and power conversion systems

2) To demonstrate startup and control methods

3) To demonstrate a 90-day endurance capability prior to the first

feasibility flight in 1965

4) To demonstrate a 10,000-hour operational capability.

The program is scheduled for completion in 1966.

d. Future Plans for Use of Facility

After completion of the 30-kwe ground prototype system tests, a series of tests similar to those in Section 3 above may be conducted on a 60-kwe system. The facility is designed to accommodate such a test.

The use which will be made of the facility following completion of S8GPTF tests has not been determined.

*

5. SNAP 8 Flight System Test Facility (FPTF), Building 056

a. Location

Figure III-3 shows the location of the FPTF (Building 056) within the AI-NDFL and its relation to other installations in the local area. The facility is situated on AEC optioned land near the northwest boundary of the AI-NDFL within an area known as the SNAP complex. The minimum distance from the facility to the AI-NDFL area boundary is approximately 325 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1250 feet to the southeast. The nearest occupied structure is the SNAP 8 Ground Prototype Test Facility (Building 059) which is approximately 200 feet northeast of the reactor.

The reactor room and control room floor are at an elevation of approximately 1814 feet above sea level, and the stack exit is approximately 1914 feet above sea level.

b. Description of Building and Equipment*

The FPTF is being designed to accommodate the ground testing of a complete 60-kwe SNAP 8 reactor system with its electrical generating system, in as near a flight configuration as possible. Figure V-6 shows an artist's conception of the facility. The facility is scheduled for completion in 1964. The superstructure consists of a main high-bay building with attached low-bay structures on opposite sides for housing general support and operating functions. The buildings are rigid, steel-framed structures with insulated, galvanized-metal sidings and poured-in-place gypsum roofs. Foundations consist of continuous wall footings under bearing walls, with spread footings and piers designed to minimize differential settlement.

The high bay structure is approximately 62 feet long by 52 feet wide, with a 40-foot eave height. The area is serviced by a 30-ton bridge crane. The area provides unobstructed access to a shielded transfer lock located in the high bay and to the shielded test vault located below the reinforced concrete floor. Access to the transfer lock is through a hatch and stepped plug in the roof, or through a massive, rolling, plug-type door in the south side of the structure.

*R. S. Lubomirski, ed., "SNAP 8 Flight System (S8FS) Test Facility Safeguards Report," NAA-SR-MEMO-7359, to be published as classified document.

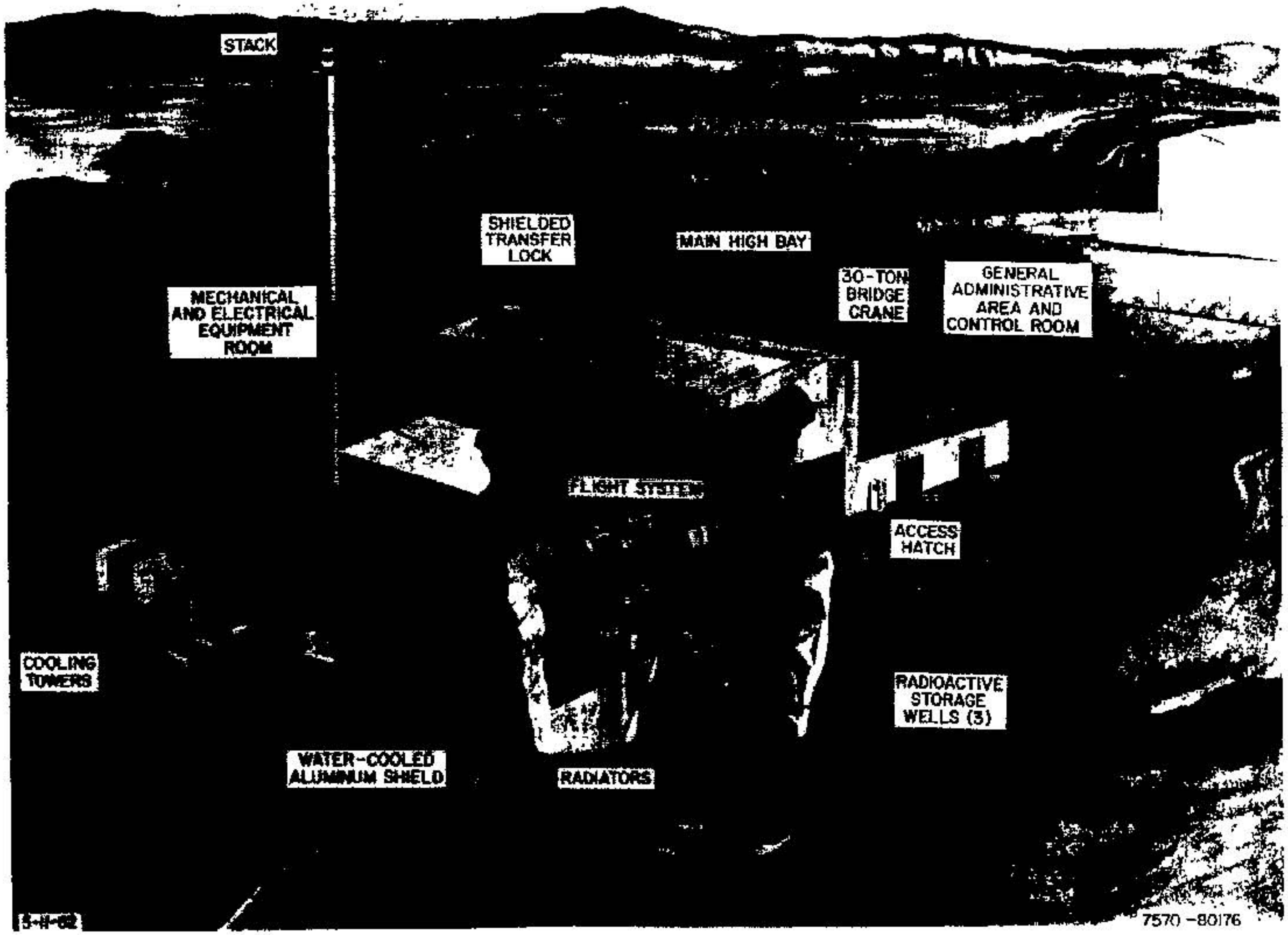

Figure V-6. Artist's Conception of SNAP 8 Flight System Test Facility

Equipment access to the test vault is through hatches, with removable sealed stepped plugs in the floor of the high bay. Personnel access to the vault for startup operations is possible via a stairwell which leads from an area near the hot change room to a door at the vault floor level. This door is sealed just prior to and during all reactor operations.

The low-bay structure on the east side of the high-bay area contains general administrative and operating areas. The main structure is approximately 62 feet long by 50 feet wide, with a 10-foot eave height. The area contains a 40 by 30-foot control room separated from the high bay area by a concrete block wall and windows which permit observation of activities in the high bay. A toilet area, locker room, and change room occupy approximately 600 ft^2 of floor space. A 14 by 24-foot office area and an 11 by 14-foot area for data storage and instrument repair are provided.

The low-bay structure to the west side of the high-bay area contains a mechanical equipment room and an electrical equipment room. The building is 62 feet long by 24 feet wide, with a 15-foot eave height. The mechanical equipment room occupies a 40 by 23-foot area in the north end, and the electrical equipment room occupies a 20 by 23-foot area in the south end. The mechanical equipment room is serviced by a 2-ton monorail crane and contains service equipment required for control of the environmental conditions in the transfer lock, vault, and reactor pit. Some local shielding is included, and ample space is provided for additional shielding on equipment which may become contaminated.

The transfer lock is a steel-lined shielded cell located in the northwest corner of the high bay. The inside dimensions are 12 feet by 28 feet by 18 feet high. The shield walls contain 4 feet of ordinary concrete. The cell will be used for transferring equipment in and out of the vault, for examining and packaging disassembled components (including an entire power conversion system), removing radioactive materials by means of shipping casks (10-ton cask provided), and remotely performing maintenance on vault equipment. The cell is equipped with a shield-viewing window and TV camera, a 5-ton bridge crane, manipulators for performing hot-cell-type work from the high bay area, etc. Access from the cell to the vault is by means of a horizontal rolling shield in the floor of the cell, which opens a hatch in the roof of the vault. The cell is designed to ensure a leak-tightness which is capable of preventing loss of more than 1% of the cell volume in 24 hours under a differential pressure of 3 inches of H_2O.

The test vault is essentially a shielded vacuum chamber with a water-cooled aluminum liner. The vault has a ceiling-to-floor depth of 32 feet and a 70-foot inside diameter (liner to liner). The ceiling will have a 9-1/2 foot concrete shield. The vault is used for installation and operation of the basic components of the test system. The contained portions include all components of the primary NaK loop, the mercury boiler and rotating machinery package, the prototype radiator, and the reactor. The vault is equipped with a 3-ton polar crane and remotely operated manipulator, polar track TV systems, roof-mounted periscope, etc., as necessary for operation and service of the equipment in the vault. The experiments will be carried out in vacuum or in partial vacuum, employing a nitrogen atmosphere. The test vault is designed for both a positive pressure of 4 psig and vacuum conditions of 0.01 atmosphere. At 4 psig overpressure, the vault is designed to leak no more than 1% of the volume per day. The design of the facility provides for a buried radioactive gas hold-up system capable of holding for several months, if necessary, the fission products released in a major accident.

Some of the facility and process service systems are located on concrete pads outside of the building. A 100-foot facility stack is located just outside the northwest corner of the mechanical equipment room along with the filters and exhaust fans. Cooling water towers, air compressors, nitrogen storage, a substation, and other miscellaneous equipment are located on concrete pads outside of the building. Buried liquid waste and gas hold-up decay tanks are located just northwest of the building.

All atmospheres released from potentially contaminated areas or equipment (high bay, transfer lock, test vault, hold-up tanks, etc.) are routed through pre- and "absolute" filters prior to release to the stack (nominal flow rate of 10,000 cfm during normal operation).

The test system will be a complete, nuclear-powered, electrical, generating system intended to produce 30 kwe (net) continuously for 10,000 hours (for use aboard a space vehicle). The reactor will be a compact, zirconium hydride 10 wt % U (fully enriched) reactor controlled by beryllium reflector elements. The system will be a flight configuration of the test system described earlier in Section 4 above (S8GPTF). The reactor will be cooled with eutectic NaK 78, which enters the core at 1100°F and exits the reactor at 1300°F. The NaK then passes through a counter-current mercury boiler and returns to the reactor inlet. The boiler produces superheated mercury vapor at 1200°F and 275 psia, which passes through a turbine (driving the alternator) exhausting at 20 psia and 706°F. The turbine exhausts to a radiator-condensor which returns 560°F subcooled mercury to the boiler. The power conversion system is being developed for 30 kwe. The reactor is designed to produce 300 kwt of thermal power to meet the heat source requirements for 30-kwe operation.

c. Description of Experimental Program

The SNAP 8 program is being developed jointly by the AEC and NASA. The AEC is responsible for the reactor development, and NASA is responsible for the power conversion system development and the overall system engineering. The S8FPTF test is the third in a series of reactor tests which will be performed prior to flight testing the SNAP 8 system in 1965 (the first and second tests were described in Sections 3 and 4 above). The FPTF will be used to ground test the prototype SNAP 8 electrical generating system in a flight configuration. The objective of the test program is to test the power plant under simulated (as near as practical) flight conditions. Tests will be conducted to prove the starting capability, stability of control of the PCS, and nuclear system thermal and environmental capability, performance, and endurance. These tests are scheduled for completion in 1964. The test program also calls for a 90-day and a 10,000-hour endurance test, to be completed by 1966.

d. Future Plans for Use of Facility

Current NASA requirements call for tests to be conducted using the 30-kwe system. Advanced planning indicates that a 60-kwe system may also be developed, and, therefore, the facility is designed to accommodate the 60-kwe system. (This system will contain two 30-kwe conversion systems and a single nuclear system.)

The use which will be made of the facility following completion of the above tests has not been determined.

6. SNAP Environmental Test Facility (SETF), Building 024

a. Location

Figure III-3 shows the location of the SNAP Environmental Test Facility (Building 024) within the AI-NDFL and its relation to other installations in the local area. The facility is situated in the eastern end of the SNAP complex on AEC optioned land. The minimum distance from the facility to the AI-NDFL boundary (AI exclusion area boundary) is approximately 500 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1250 feet to the southeast. The nearest occupied structure is the SNAP 2 Experimental Lab (Building 025), which is approximately 130 feet west of the reactor.

The control room and high bay floor (i.e., top of vault complex) are at an elevation of 1826 feet above sea level, and the stack exist is approximately 1911 feet above sea level.

b. Description of Facility and Equipment*

The installation has been designed to enable simultaneous safe operation of two SNAP reactor systems, each operating at 150 kwt. Figure V-7 illustrates the design of the facility. The facility consists of (1) a main high-bay superstructure containing a subgrade reactor vault complex, with an open pit access area adjacent to the vault, and (2) two prefabricated steel low-bay buildings attached to opposite sides of the superstructure to provide administrative and operational support areas.

The main superstructure is a rigid steel-framed building, with insulated metal siding and roofing and continuous concrete footings. The building is 60 feet long and 55 feet wide with a 30-foot eave height. This building houses the shielded vault complex containing the two reactor test cells and a transfer lock, the vault access area adjacent to the vault complex, and the high-bay area, which provides unobstructed access to the vault complex and access area. The reactor vaults have inside dimensions of 15 by 15 by 19 feet high, and are internally connected to the centrally located transfer lock (15 by 26 by 19 feet high) by means of a maze leading from each cell. A minimum of 8 feet of ordinary concrete is provided between the vaults and accessible areas. Three, removable,

*G. H. Anno, ed., "SNAP 2 Environmental Test Facility Hazards Report," NAA-SR-3513, May 1, 1959 (classified).

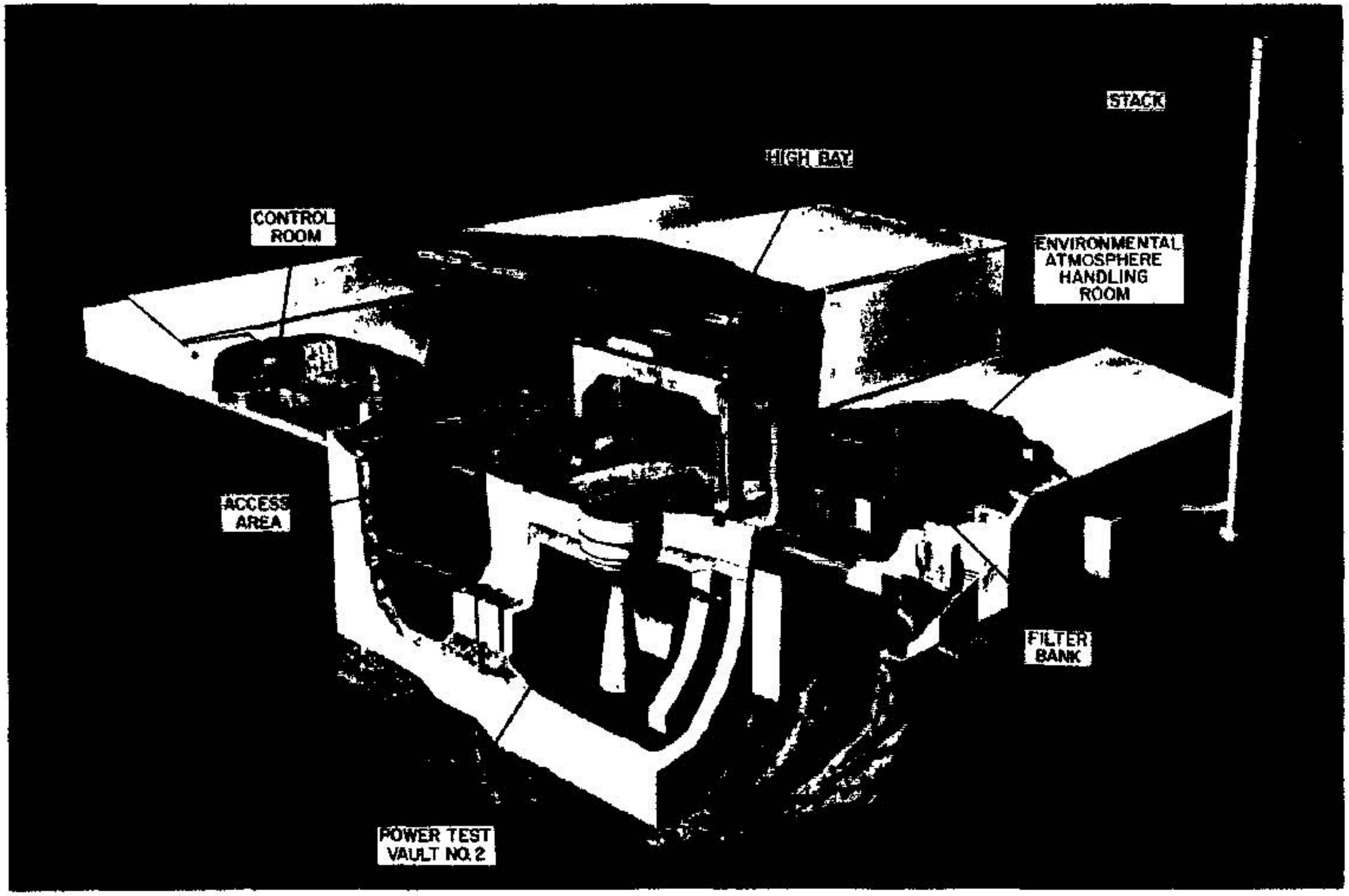

Figure V-7. Design of SNAP Environmental Test Facility stepped, shield plugs in the top of the vault complex provide access to the two reactor vaults and transfer lock. The area above the vault complex is serviced by a 20-ton gantry crane. The vault access area is a 56 by 20 by 27-foot-deep pit just north of the vault complex, which provides direct access to the operating face of each vault and the transfer lock as well as internal access through sealed doors and plugs. A lead glass window provides for observation of activities in the transfer lock. The access pit and high bay are services by a 5-ton bridge crane.

The low-bay structure on the north side of the high-bay area contains the general administrative and operating areas. This structure is approximately 64 feet long by 60 feet wide and consists of two standard prefabricated metal buildings attached to each other and the north wall of the high bay superstructure. The buildings are constructed on continuous concrete footings and a floor slab and have gable roofs with an eave height of 10 feet and ridge heights of 16 feet. One portion of the building (32 by 60 feet) is called a test mock-up area and is used as a "cold" experimental and shop area and an office area. The remaining half of the building (section adjacent to high bay) contains a 32 by 28 foot control room, a locker room and change room, and a "cold" mechanical equipment room. A window in the south wall of the control room permits observation of the high bay and vault access areas. The instrumentation necessary for safe operation of each reactor test cell occupies separate and isolated portions of the control room. Personnel leaving the high bay, access area, or vault complex are routed through the change room to enable contamination control and personnel monitoring.

The low-bay structure on the south side of the high bay area contains service equipment for the facility and vault complex. The structure is approximately 60 feet long by 34 feet wide and is a steel and concrete structure with a gable roof 13 feet high at the eave and 19 feet high at the ridge. The east portion of the building contains a 45 by 34-foot filter and hot gas compressor room. The north, west, and south walls of this room are of concrete construction, to provide radiation shielding. This room is serviced by a 4-ton bridge crane. The west portion of the building contains a 34 by 15-foot mechanical and electrical equipment room.

All interior surfaces of the vault complex are lined with 3/16-inch aluminum plate. The liners and seals on vault openings are designed and constructed to withstand a pressure differential of 4.0 psig and to provide a leak tightness of better than 1% of the contained volume in 24 hours under a differential pressure of 3 inches of H_2O. During operation a nitrogen atmosphere at a slightly negative pressure is maintained in the test vault to prevent NaK- and/or hydrogen-oxygen reactions. The nitrogen atmosphere is cooled by freon cooling coils and a nitrogen recirculation system located in the vault complex. Following an accident, any resulting overpressure or leakage to the environment will be reduced by compressors, which will pump the vault gases to buried hold-up tanks designed to contain the radioactive materials. The access area and high bay area are maintained at a slightly negative pressure with respect to ambient and other occupied areas and are exhausted through pre- and "absolute" filters to an 85-foot stack.

c. Description of Experimental Program

Power Test Vault No. 1 presently houses a SNAP-2 reactor (S2DR) which operates at a nominal power of 65 thermal kilowatts.* This program involves a test of the core without power-conversion-system equipment. The vault contains a nitrogen atmosphere which is controlled to an oxygen concentration of less than 3% by a nitrogen feed and a small exhaust stream which is routed through pre- and "absolute" filters prior to release to the stack. This phase of the test program is scheduled for completion in 1962, after which the system will be removed.

Power Test Vault No. 2 will be used to provide shielding and containment for the SNAP-10A Flight System, which operates at a nominal power of 30.5 kwt. This test (S10A-FS-1) is scheduled for initial operation in late 1962. This reactor system is to be operated in a vacuum chamber located within the test vault. Although the vault provides adequate containment, the vacuum chamber is considered the primary source of containment for the system. The environmental atmosphere in vault No. 2 can be controlled for oxygen content and pressure in the same manner as Vault No. 1. This reactor system test is scheduled for completion and removal some time in 1963.

The installation of the SNAP 10A Flight System No. 1 (S10A-FS-1) test in Vault No. 2 requires modifications to the SETF. These modifications,

*H. N. Rosenberg, ed., "Summary Safeguards Report for SNAP 2 Development System (S-2-DS)," NAA-SR-5483, November 23, 1960 (classified) which are presently under way, are described below:

1) The construction of an additional control room adjacent (immediately west) to the existing control room. This modification is scheduled for completion in mid-1962.

2) The construction of a wall and ceiling to divide a section of the lower access area, forming a room opposite Vault No. 2. This room utilizes the existing structure for three walls and the floor, and existing steel sections form a mezzanine level. This room will be used to contain vacuum pumps and other heavy equipment associated with the S10A-FS-1 experimental program.

d. Future Plans for Use of Facility

The test program for the SETF cells calls for the installation and nuclear test of the SNAP 2 flight system. This system will operate at the same thermal power level as the S2DR presently installed and operating in the SETF.

7. SNAP Flight Systems Nuclear Qualifications Test Facility, Building 019

a. Location

Figure III-3 shows the location of the Flight Systems Nuclear Qualifications Test Facility, more commonly designated as the Acceptance Test Facility (ATF), Building 019, within the AI-NDFL, and its relation to other installations in the local area. The facility is situated near the northwest boundary of the AI-NDFL, within the SNAP complex. The facility is located on AEC optioned land. The minimum distance from the facility to the AI-NDFL boundary is approximately 250 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1325 feet in a southeasterly direction. The facility is located between the SNAP 8 Ground Prototype Test Facility (Building 059) and the Non-Nuclear Component Assembly and Performance Testing Building (Building 013), which are approximately 60 feet to the southwest and northeast of the reactor, respectively.

The finished floor elevation in the facility is 1814 feet above sea level. There is no stack associated with this facility.

b. Description of Facility and Equipment*

The ATF is being constructed to accommodate acceptance testing of final SNAP flight systems just prior to shipment of the units to launch sites. The facility is scheduled for completion by mid-1962. Figure V-8 illustrates the facility design. The building consists of a main reactor high-bay superstructure, with an attached low-bay structure on one side for housing of general support and operating functions. The structures are rigid steel-framed buildings with insulated metal sidings and roofs. The overall building dimensions are approximately 83 feet long by 77 feet wide, with an additional 23 by 10-foot concrete block fuel storage room attached to the southeast corner of the building. All foundations consist of continuous wall footings under bearing walls with spread footings and piers designed to minimize differential settlement under all columns. Concrete floors are provided for all areas.

The high bay is approximately 80 feet long by 45 feet wide and is provided with a composition roof over rigid insulation and steel decking. The roof is 34 feet high at the eave. The area is serviced by a 10-ton bridge crane. The

*Compact Systems Division "Preliminary Safeguards Report — Acceptance Test Building (019)," NAA-SR-6733, November 15, 1961

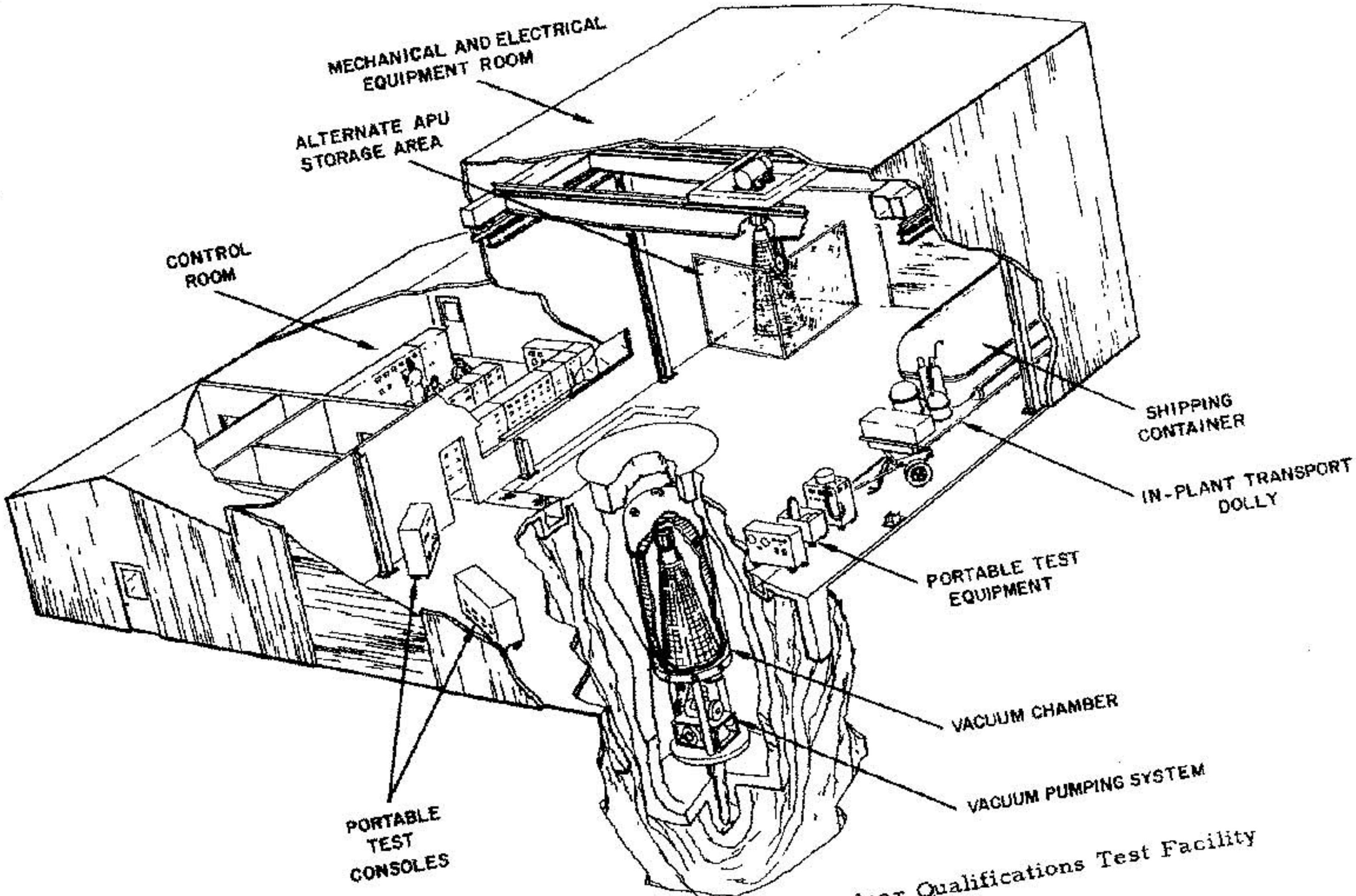

Figure V-8. Design of the Flight Systems Nuclear Qualifications Test Facility high-bay structure houses the space and equipment required for assembly, testing, and adjustment of APU power system units and provides unobstructed access to the reactor test vault located below the floor level. Access to the test vault is through a hatch equipped with removable, stepped, concrete shielding plugs in the high-bay floor (ceiling of the vault).

The low bay structure is approximately 83 feet long by 29 feet wide, with an 11-foot eave height and 16-foot ridge height. The area contains a 31 by 29-foot control room, a 20 by 28-foot mechanical and electrical equipment room, a 12 by 13-foot instrument repair room, a 12 by 17-foot toilet area and change room, and a 30 by 12-foot office area.

The test vault is a 12 by 12 by 39-foot deep concrete pit located below the high bay floor and opposite the control room (window in control room wall). Removable concrete plugs totaling 4 feet in thickness provide shielding directly over the vault. A 15-ton capacity hydraulically operated elevator is provided in the vault for handling the test system. During normal operation, the test system will be contained in a 9-foot diameter by 17-1/2 foot-high vacuum chamber (stainless-steel vessel) in the test vault. The system is designed to operate at 10^{-3} mm of Hg. The vessel wall is cooled to an average wall temperature of 100°F and will conform to the ASME Boiler and Pressure Vessel Code, Section XII for unfired pressure vessels. Vacuum pump discharge is through an "absolute" filter to a roof vent. Two quick-closing valves will isolate the vessel in the event of an incident. The vessel is designed to contain the maximum pressure generated by a 50-Mw-sec excursion in a SNAP 2 or 10A reactor system. The vacuum chamber is designed and tested to withstand both normal operations and the consequences of an excursion with no leakage. Fission product activity, therefore, will be all times be either contained or released only through the vent. The two quick closing valves would close in less than 1 sec.

Since reactors will be tested only at zero power level, there will be no significant fission product generation during normal operations. No stack is provided for the facility. The high-bay area is a controlled atmosphere incorporating both temperature and humidity control.

c. <u>Description of Experimental Program</u>

The presently planned program for the facility will include the testing of the SNAP 10A Flight Systems 2, 3, and 4 (S10A-FS-2, 3, and 4) and a SNAP 2 flight system prior to shipment to Vandenburg Air Force Base (VAFB) or another site. In addition to non-nuclear atmospheric tests, the systems will be tested in the containment vessel under controlled environmental conditions The major acceptance tests will include (1) a physical inspection and electrical and instrument check of the APU without fuel or NaK coolant, (2) a functional demonstration of correct operation and startup sequence, at ambient temperature, of the APU mechanical systems and safety and control instrumentation, (3) a dry critical loading test and measurements with the APU in the vacuum chamber, (4) installation of the core cap and welding of pump connections and lines, (5) NaK loading, (6) wet critical and nuclear acceptance tests, (7) thermal acceptance tests at temperature, (8) shield mating tests, and (9) packaging of APU, reflectors, and shields for shipment to VAFB.

d. Future Plans for Use of Facility

Present plans include acceptance testing of SNAP 2 flight systems as described above. However, to accommodate these systems, additional instrumentation will be installed to facilitate their testing. Because the present test chamber is not large enough to accommodate a SNAP 8 flight system, only fuel loading and nuclear acceptance of the SNAP 8 system are planned. Other plans for use of the facility are not presently known.

8. Shield Test Experiment Facility, Building 028

a. Location

Figure III-3 shows the location of the Shield Test Facility (Building 028) within the AI-NDFL and its relation to other installations in the local area. The facility is situated near the northwest boundary of the AI-NDFL, within the SNAP complex. The installation is located on AEC optioned land. The minimum distance from the facility to the AI-NDFL boundary (AI exclusion area boundary) is approximately 250 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1450 feet to the southeast. The nearest occupied structure is the SNAP Environmental Test Facility (Building 024), which is approximately 150 feet southeast of the reactor.

The reactor room and control room floor elevations are 1816 feet above sea level, and the stack exit is approximately 1846 feet above sea level.

b. Description of Building and Equipment*

The Shield Test Facility is designed to provide the experimental capability for exposing materials to well-defined radiation sources and thus enable determination of their shielding effectiveness for compact power system applications. Figure V-9 illustrates the facility design. The facility consists of two basic areas; (1) a Butler type building which houses the reactor installation and contains a reactor room, control room, laboratory, change room, and office area and (2) an attached, below grade, shield test room adjacent to the reactor for conducting the actual shield tests and measurements.

The Butler building is a rigid, steel-framed, trussless, gable-roofed structure, with insulated metal siding and roofs, continuous concrete footings, and a concrete floor slab. The structure is 40 feet long and 40 feet wide, with an eave height of 14 feet and a ridge height of 21 feet. The reactor room is 20 feet long and 17 feet wide and provides unobstructed access to the reactor, which is located below the reactor room floor level. The reactor (water cooled) and its control and safety systems are immersed in a 5-foot diameter by 20-foot-deep, water-filled aluminum tank located in the center of the room; the top of the tank is approximately 3 feet above floor level. Radiation shielding is provided

*R. L. Tomlinson, ed., "SNAP Shield Test Experiment Final Hazards Summary," NAA-SR-5896, March 17, 1961

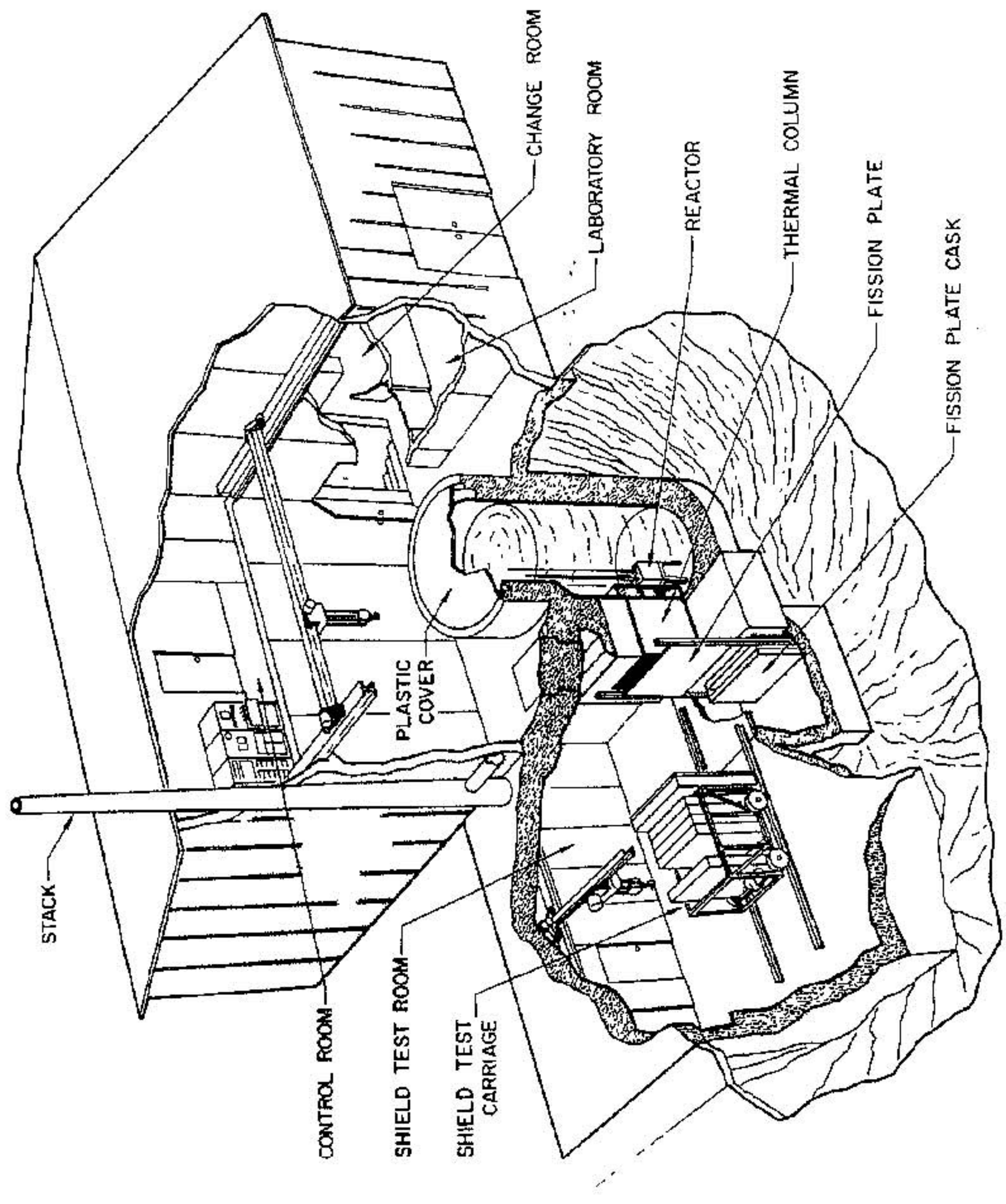

Figure V-9. Design of the Shield Test Experiment Facility by gravel, concrete, and earth surrounding the tank and a minimum of 16 feet of water over the core during reactor operation. The coolant pumps, heat exchanger, dimineralizers, and refrigerant systems are located is the reactor room. A 1-ton, hand operated, chain hoist mounted on a bridge crane provides handling services.

The 23 by 20-foot control room is adjacent to the north wall of the reactor room, which contains a window to enable observation of reactor room activities. The reactor console and instrumentation necessary for safe plant operation are located in one area of the control room, while the instrumentation necessary for actual shielding measurements occupies a separate area in the control room.

The laboratory is a 20 by 13-foot area used for fabrication and repair of detectors and contains a hood, dry box, sink, electronic test gear, and other laboratory equipment. The office area is 20 by 14 feet long and has desk space for the normal full-time staff of four experimental physicists. Access from the office area and control room to the laboratory or reactor room is through a 9 by 13-foot locker room and a 7 by 13-foot change room. A stairwell at the exit of the change room also provides controlled personnel access to the shield test room via an underground tunnel and access door at the test room floor level.

The shield test room is a shielded underground vault located on the west side of the reactor pool and tank. The vault is separated from the reactor by a concrete shield wall which contains a 4-foot-long and 5-foot-square graphite thermal column in line with the reactor. One end of the column opens into a 33-foot-long by 20-foot wide and 18-foot-high room. Within the room, a 2-foot diameter circular (fission) plate of fully enriched uranium may be remotely positioned on the centerline of the thermal column or inserted into a lead shield cask. The uranium plate converts low-energy neutrons leaving the thermal column into a fission spectrum of neutrons and gamma rays to be used in shield attenuation experiments. A remotely positioned shield test carriage is provided for mounting test specimens adjacent to the fission plate. Radiation detection equipment is mounted on the shield test carriage and connected to the control room through flexible cables. The entire operation is viewed by means of a TV system. A ramp and 19 by 9-foot exterior door provides truck access to the vault. The vault is provided with a 7-1/2-ton, pendant-controlled, travelling bridge crane for handling shielding and casks.

The reactor is a water-cooled, zirconium-hydride moderated reactor similar to TRIGA. It produces a maximum thermal power of 50 kw. Four rods control the reactor, three of which utilize B_4C as a poison, the other being a "grey" regulating rod composed of stainless steel. The control rod system is activated by electromagnetic clutches designed for fail-safe operation.

The reactor room and shield test vault are maintained at a negative pressure with respect to ambient and the other occupied areas and are exhausted through pre- and "absolute" filters to a 30-foot stack.

c. Description of Experimental Program

The reactor, fission-plate testing program has been completed. It consisted of experimentally obtaining those physics parameters necessary to prove the safety of the reactor system. The following measurements were obtained:

1) Reactor criticality
2) Reactivity worth of fuel rods and core components
3) Core and reflector void reactivity measurements
4) Isothermal temperature coefficients
5) Reactor and thermal column flux traverse
6) Reactor power calibration
7) Power coefficient measurements
8) Fission plate flux mapping
9) Fission plate power calibration.

The radiation evaluation of shield materials presently underway consists of exposing slabs of shielding materials to be used in the SNAP 2, 8, and 10A programs to the fission spectrum from the fission plate and measuring their attenuation characteristics for various energies of neutrons. The parameters to be studied on each material are the energy and angular neutron distribution as a function of shield thickness. The neutron energies to be investigated range from 0.5 to 10 Mev. A collimator will be used to obtain angular discrimination, while energy discrimination will be achieved with a fast neutron spectrometer.

d. Future Plans for Use of Facility

Initially, the shield materials to be evaluated will consist of those materials currently under consideration for the SNAP 2, 8, and 10A systems.

As the program progresses, other more exotic materials will be evaluated for higher powered systems, where biological rather than instrument-payload shielding problems predominate.

The possibility exists for reorientation of the facility program for the radiation testing of transistorized payload components, because the fission plate is capable of providing the high intensity of radiation necessary for such tests at low facility operating costs.

9. Kinetic Experiment Water Boiler Facility (KEWB) Test Facility, Buildings 073 and 083

a. Location

Figure III-3 shows the location of the KEWB reactor building (Building 073) and its associated control building (Building 083) within the AI-NDFL and its relation to other installations in the local area. The installation is located slightly northeast of the central portion of the AI-NDFL and northeast of the SNAP complex. The site is located on AEC optioned land. The minimum distance from the facility to the AI-NDFL boundary is approximately 625 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1150 feet to the southeast. The nearest occupied structures are SNAP office building No. 2 (Building 037), which is approximately 150 feet west of the reactor, and the AE-6 Reactor Building, (Building 093), which is approximately 250 feet east of the reactor. The floor area in the control building is 1860 feet above sea level and the stack exit is 1908 feet above sea level.

b. Description of Buildings and Equipment*

The KEWB installation was constructed to provide the means for experimentally determining the dynamic behavior of homogeneous, water-boiler type reactors and thereby enabling prediction of the safety aspects of this type of reactor. Figure V-10 shows a photograph of the area and facilities. The reactor is buried in the foreground behind the stack, and the control building is the smaller of the two buildings in the background (the large building (093) houses the AE-6 Reactor.) The installation consists of the reactor testing building, which houses the reactor (Building 073), and a separate control building (083) where the experiments are conducted. The reactor test building is separated from the control building by approximately 200 feet.

The reactor test building is a reinforced underground concrete structure which houses the reactor test core, control rods, gas handling and fuel storage systems, and necessary detection instrumentation. The building is 15 by 25 by 10 feet high, with outside walls and floor of reinforced concrete 8 inches thick. The roof is a 1-foot-thick reinforced concrete slab. Six feet of earth

*Reactor Kinetics Staff, "Change of KEWB Reactor Cores — Evaluation of Significance With Regard to Associated Hazards," NAA-SR-Memo 4928, Feb. 4, 1960

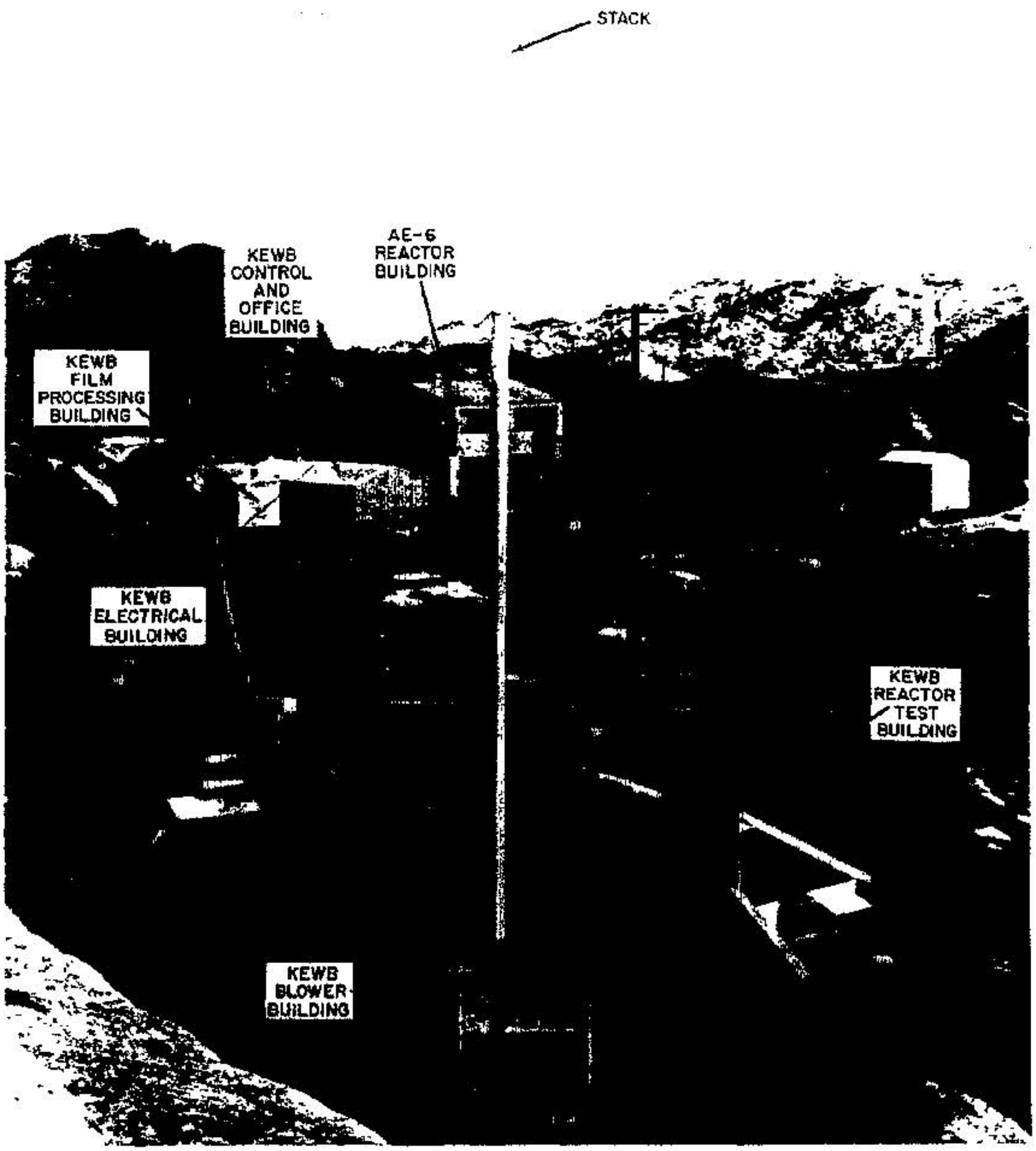

Figure V-10. Photograph of Kinetic Experiment Water Boiler Area and Facilities cover the building for shielding purposes. The interior is divided into three rooms which are separated by 2-foot-thick concrete walls. A 12 by 15-foot room contains the core-reflector-control and poison-rod systems. An adjacent 6-1/2 by 7-1/2-ft gas and fuel handling room contains the gas recombination apparatus, fuel storage tank, distillate chamber, gas ballast reservoir, and other miscellaneous hardware for adjusting operating parameters. A 6-1/2 by 7-1/2 foot valve gallery contains about 50 valves, which may be operated manually to provide for movement of fluids and gases in the system. Access to the building is through an above-grade change room and a covered ramp and stairwell which leads to the valve gallery.

The three rooms in the building communicate with each other through the ventilation system, which allows air to be drawn from the building through "absolute" filters and a 60-foot dilution stack located near the building. When ventilation is required, air is allowed to enter the building through any or all of three inlets, one being located in each of the rooms. The inlet and exhaust ducts are equipped with Keystone valves. The blower system used to ventilate the building has a capacity of 2000 cfm and is equipped with a bypass duct which permits high dilution of radioactive gases entering the stack. During normal operation, or in the event of release of activity from the reactor system, the ventilation system valves are closed to confine activity in the building. Leakage from the building has been determined experimentally to be approximately 15% per day at a constant overpressure of 1 psi.

The control building is a 12 by 20-foot, prefabricated, sheet-steel building located 200 feet from the reactor test building. A separate structure is attached to the building for office functions. All recording and control instrumentation necessary to operate the reactor in a normal fashion, plus the special instrumentation needed for initiating and recording transient operation, are located in this structure.

A small sheet-steel building, called the "electrical building," is located near the reactor test building. All preamplifiers and power supplies that need to be near the detectors in the reactor room, to reduce cable loss and noise, are located in this building. Hold-up for radioactive gases released from the core and liquid wastes is provided by buried tanks.

c. Description of Experimental Program

Initial construction of the facility was completed in 1955, and operation began in July 1956. Transient testing of the first core began in the fall of 1956 and was concluded in the summer of 1959. During this 3-year period, more than 900 transient tests were conducted under a variety of initial core conditions and for various reactivity inputs and injection rates. Following these tests, the spherical "A" core (first core) was replaced by the cylindrical "B" core. The experimental program was continued with a series of experiments utilizing the cylindrical core geometry, and a companion effort involving the direct measurement of fuel solution density as a function of time during the excursion was also conducted simultaneously.

The KEWB reactor is presently operating as a neutron burst facility and, as such, is available to all AEC and military contractors. Approximately 30% of KEWB's operating time is presently devoted to the investigation of void formation in organic reactor transients. The majority of the remaining operating time is used by outside organizations such as Space Technology Laboratory, University of California, and North American Aviation Space and Information Systems Division.

d. Future Plans for Use of Facility

Future plans call for continued operation as a burst facility, with no major modifications contemplated to either the reactor or facility.

10. AE-6 Reactor Building, Building 093

a. Location

Figure III-3 shows the location of the AE-6 Reactor, Building 093, within the AI-NDFL and its relation to other installations in the local area. The facility is located slightly east of the central portion of the AI-NDFL and east of the SNAP complex. The minimum distance from the facility to the AI-NDFL boundary (AI-exclusion area boundary) is approximately 750 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1050 feet to the southeast. The nearest occupied structure is the Kinetic Experiment Water Boiler Control Building (Building 083), which is approximately 50 feet north of the reactor

b. Description of Building and Equipment*

The AE-6 and its associated experimental facilities are specifically designed to provide a thermal neutron source for evaluating neutron behavior in subcritical exponential-type assemblies, and for irradiating foils and other materials. Figure V-11 shows the site and general building structure. The installation consists of a reactor building, housing the reactor and control room, and a separate building used for fuel handling and storage.

The reactor building contains a high-bay area and a control room. The high-bay area is a steel framed superstructure covered with sheet metal siding and roofing. The area has lateral dimensions of 30 by 40 feet, with a maximum floor-to-ceiling height of 29 feet, and is serviced by a 1/2-ton overhead crane. The floor consists of 6 inches of concrete, except under the reactor and concrete shield enclosure, where 30 inches of reinforced concrete has been provided.

The reactor is situated in the center of the high-bay area and consists of a sperhical core, a graphite reflector and thermal column, a mild steel reflector enclosure, the gas handling systems, the control and safety rod systems, the cooling systems, and the concrete shield enclosure. The reactor is a homogeneous, solution-type reactor, designed to operate at 2.5 kw of thermal power. The core and gas handling system are contained in a stainless-steel

* G. L. Blackshaw and C. H. Skeen, "Safeguards Summary for the AE-6 Reactor," NAA-SR-Memo-5304, July 7, 1960.

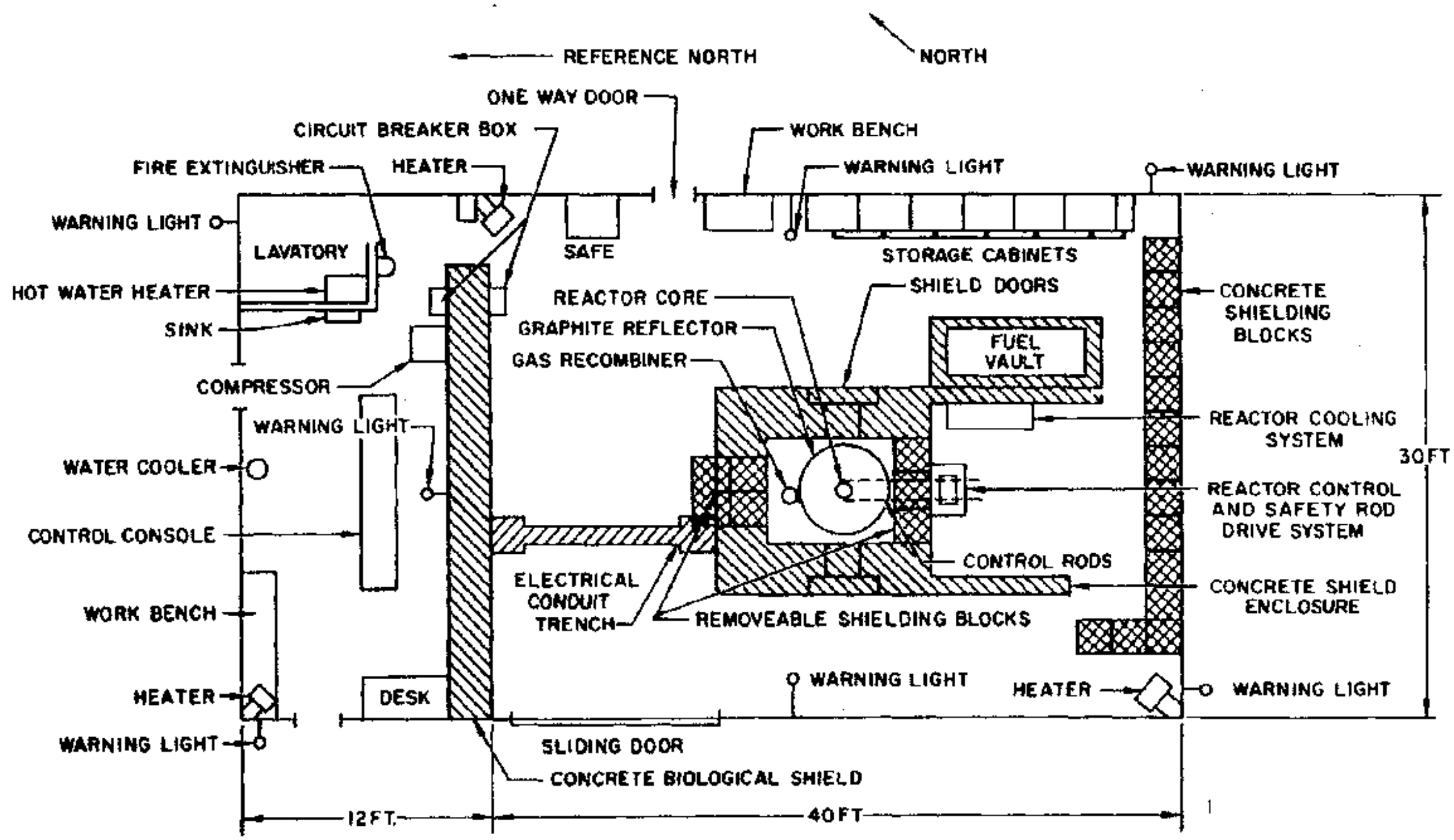

Figure V-11. Floor Plan of AE-6 Reactor Building system which provides the primary barrier against release of radioactivity. Secondary containment is provided by the mild steel graphite reflector enclosure tank and the concrete shield walls which completely surround the primary system, except on top, where the neutron beam from the thermal column enters a tank containing subcritical lattice assemblies. When this neutron beam is not used, shielding is provided. Channels are also provided in the reflector, through which the neutron flux is available in various magnitudes for experimental purposes.

A concrete fuel storage vault is provided adjacent to the critical assembly for storage of assembled fuel elements to be used in exponential experiments. Additional concrete blocks are provided in the facility and are available for stacking around the reactor or walls of the high-bay area. Ventilation is provided by opening and closing windows and roof vents. (No stack is included in the facility.)

c. Description of Experimental Program

The AE-6 facility is a modification of the old WBNS (Water Boiler Neutron Source) formerly located at the Downey, California Plant of North American Aviation. In 1956, operation began on the present site at the Nuclear Development Field Laboratory. Most of the past experimental program involves using the vertical thermal column as a driving source for subcritical lattices. Nuclear parameters have been measured for a wide variety of fuel lattices in D_2O-, organic- and graphite-moderated systems. Other experiments, such as danger coefficient measurements and a small number of service irradiations, have been carried out from time to time.

At the present time, a fission plate has been installed in the thermal column to provide a source of fission spectrum neutrons. The slowing-down distribution of these neutrons is being measured in a number of nonmultiplying systems such as water, organic, graphite, water-zirconium, water-iron, water-aluminum, etc.

d. Future Plans for Use of Facility

Future plans call for the installation of a second (in this case, horizontal) thermal column in June 1962. The actual experimental program is expected to proceed as in the past, that is, the measurement of nuclear parameters in a variety of subcritical or nonmultiplying systems.

11. Sodium Graphite Reactor Critical Facility (SGR Critical), Building 009

a. Location

Figure III-3 shows the location of the SGR Critical Facility (which is part of Building 009) within the AI-NDFL and its relation to other installations in the local area. The facility is located in the southwestern portion of the AI-NDFL. There are, at present, no other AI facilities to the west, northwest, or southwest. The minimum distance from the facility to the AI-NDFL boundary is approximately 650 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundaries are approximately 1300 feet in a southeasterly direction and 1400 feet in the southwesterly direction. The nearest occupied structure (besides the Organic Moderated Reactor Critical Facility, which is also part of Building 009) is the Epithermal Critical Experiment Laboratory, Building 100, approximately 200 feet northeast of the facility.

Grade floor elevations are approximately 1835 feet above sea level. The stack exit is approximately 1880 feet above sea level.

b. Description of Building and Equipment*

The facility is designed to accommodate a series of critical experiments which will provide an extension of knowledge of the sodium-graphite-reactor concept. The facility (see Figure V-12) consists of a high-bay building, which houses the critical assembly cell and a fuel-and-graphite-storage area, and an adjoining low-bay area which houses the control room, offices, and miscellaneous supporting activities. The high bay is a concrete structure approximately 70 feet long by 40 feet wide, with a 4-inch-thick reinforced concrete roof deck on steel framing, with an eave height of 39 feet. A concrete block penthouse, which houses the critical assembly control rod drive mechanisms, is located on the roof over the critical assembly cell. A 5-ton capacity overhead crane runs north and south over the entire high-bay area, to service both the critical cell and storage area.

* D. E. Fletchall, ed., "Sodium Graphite Reactor Critical Experiment Safeguards Summary," NAA-SR-3404, April 1959.

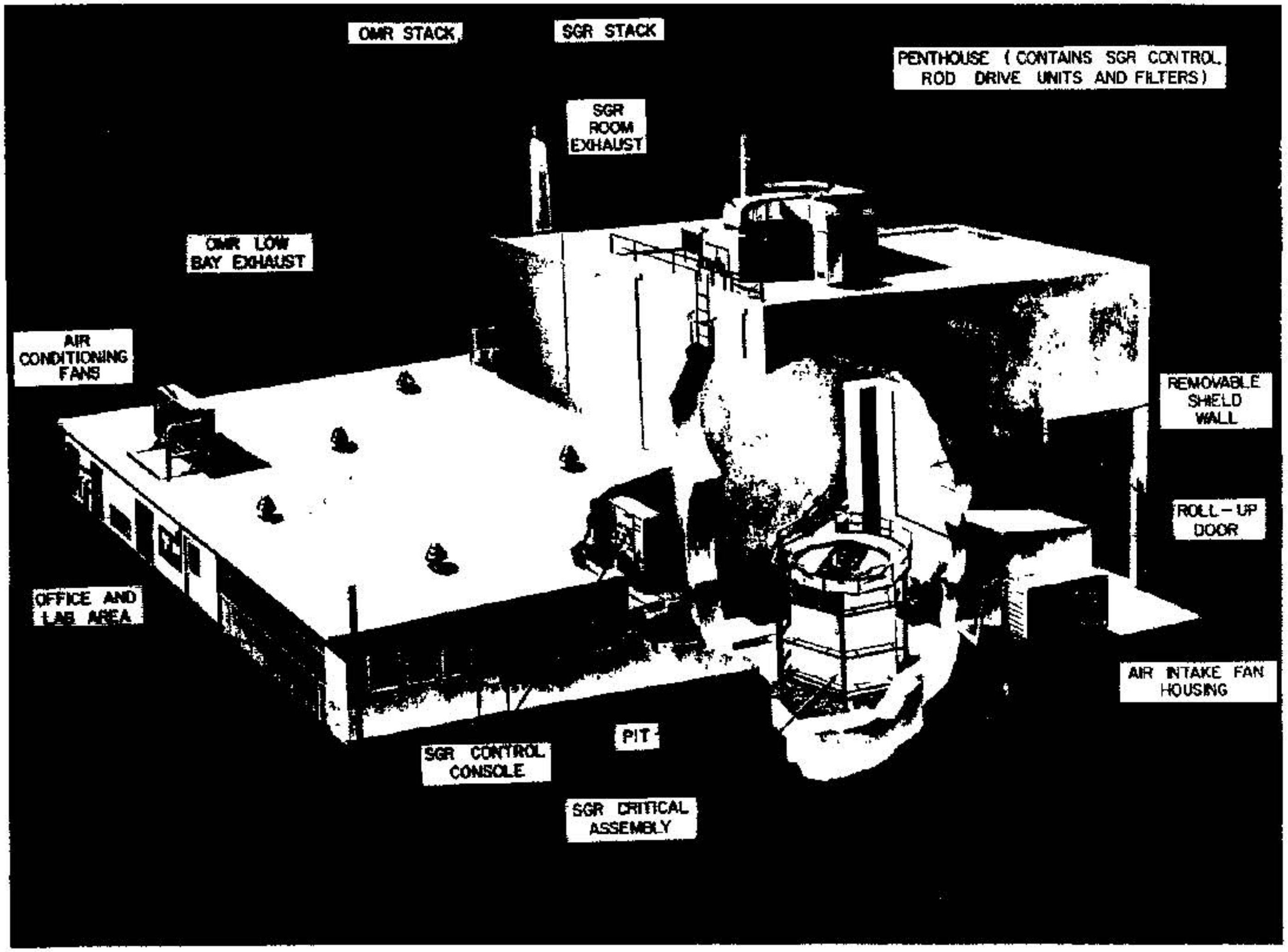

Figure V-12. Artist's Conception of SGR-OMR Critical Facility

The critical assembly cell has dimensions of 36 by 36 feet, and a floor-to-ceiling height of 46 feet, with the floor 10 feet below grade level. A 10-foot-deep hexagonal pit, 14 feet across the flats, is located in the center of the critical cell floor to provide access to the underside of the critical assembly. A manhole and passageway leads to the pit. The three sides of the critical cell away from the storage area are shielded by poured-concrete walls extending to the roof. The roof affords no shielding. A section of the 2-foot-thick wall between the critical cell and storage areas consists of removable concrete tiers to facilitate handling of critical assembly materials. A 10-foot clearance between the top of this wall and the ceiling is provided for crane travel. A service door is provided on the northwest corner for truck entry to the graphite storage area. The fuel storage area is 14 by 22 feet, and the graphite storage area is 29 by 24 feet. These areas are separated by 1-foot-thick concrete walls 18 feet high. The floors of these areas are at grade level and the ceiling height is 35 feet. The Organic Moderated Reactor Critical Facility assembly room is adjacent to the west wall of the SGR Critical Facility assembly room (common shield wall, 4 feet thick).

The low-bay building is a single-story structure attached to the south side of the SGR and OMR high-bay areas. The building is a conventional steel frame structure with insulated sheet metal siding and interior walls of metal-lath and plaster. The roof consists of gravel and tar on rigid insulation over metal decking. The entire building is 108 feet long and 70 feet wide, with the west half occupied and used by the OMR project and the east half by SGR project experiments. The portion used by the SGR operations personnel contains a change room, control room, radiochemistry laboratory, general laboratory area, offices, and miscellaneous storage and facility support areas. The change room is located at the entrance of a corridor leading to the critical assembly room and is adjacent to the radiochemistry lab. The control room is located next to the south shield wall of the critical cell. Surveillance of cell operations is provided by nuclear instrumentation and a closed-circuit television channel.

The ventilation system is designed so that air flow is away from non-radioactive areas to the critical assembly areas. The low-bay area is serviced by conventional heating and ventilating equipment which maintains a positive pressure in the area. The radiochemistry laboratory area is maintained slightly negative with respect to surrounding areas, by exhausting air through a fume hood and "absolute" filters. The SGR and OMR critical assembly areas have separate ventilation systems. These two areas and their ventilation systems are isolated and completely independent of each other. The air flow remains towards these assembly areas, even if the doors leading from the low-bay areas to such assembly areas are held open simultaneously.

Air intake to the SGR critical assembly room is filtered, to maintain the purity of the graphite, and is forced through a duct which opens near the floor of the critical cell. The area is exhausted through roof mounted, motor-driven exhausters. There is one main exhauster with an "absolute" filter unit and seven smaller exhausters without filters. The absolute" filter may be bypassed by operation of motor-driven dampers, and the small exhausters may be closed in a similar manner. During reactor shutdown periods the system is operated without exhaust filters, (all dampers open) to maintain a high air change rate. During reactor operation air is exhausted through the filter unit only. Negative pressures, with respect to ambient and surrounding occupied areas, are maintained under all conditions. The controls for the dampers and blowers are interlocked with the reactor controls, so that the system must be in proper condition prior to reactor startup.

The SGR Critical Assembly has been designed functionally to permit experiments on graphite-moderated reactors of various sizes and fuel-element configurations. The basic assembly consists of a cylindrical array of hexagonal graphite logs in which fuel and poison elements may be placed in a wide variety of lattice spacings and arrangements. The overall size of the assembly may be varied up to 14 feet in diameter and 17 feet in height.

To study temperature effects, provision is made for air-heating of the assembly.

c. Description of Experimental Program

The facility began nuclear operation in January 1960, and a variety of experiments have been performed on clean, graphite-moderated-reactor systems. Experiments have also been performed in support of the Hallam Nuclear Power Facility program. At the present time, critical experiments are underway with slightly enriched metal fuel, to study lattice parameters of SGR systems.

d. Future Plans for Use of Facility

A new UC-fueled core with canned Na coolant is planned for startup in July 1962. Control characteristics, critical mass, temperature coefficients, and other reactor parameters will be studied.

12. Organic Moderated Reactor Critical Facility (OMR Critical), Building 009

a. Location

Figure III-3 shows the location of the OMR Critical Facility (which is part of Building 009) within the AI-NDFL and its relation to other installations in the local area. The facility is located in the southwestern portion of the AI-NDFL. There are presently no other AI facilities to the west, northwest, or southwest. The minimum distance from the facility to the AI-NDFL boundary is approximately 650 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundaries are approximately 1300 feet in a southeasterly direction and 1400 feet in a southwesterly direction. The nearest occupied structure (besides the Sodium Graphite Reactor Critical Facility, which is part of Building 009) is the Epithermal Critical Experiment Laboratory, Building 100, which is approximately 200 feet northeast of the facility.

Grade floor elevations are approximately 1835 feet above sea level. The stack exit is approximately 1880 feet above sea level.

b. Description of Building and Equipment*

The facility is designed to accommodate a series of critical assembly experiments which will provide a basic body of knowledge concerning the nuclear properties of organic moderated reactor systems. The facility (see Figure V-13) consists of a high bay which houses a concrete-shielded critical assembly room, and an adjoining low-bay area which houses a control room, laboratory, offices, and miscellaneous support and utility areas. The entire facility is approximately 110 feet long by 63 feet wide.

The critical assembly room (high-bay area) is a concrete structure with shield walls extending to the roof eave (37 feet). Shield thicknesses vary from 4 feet to 1 foot, depending on the height and nature of activities in adjoining areas. External fences surrounding the facility provide control over personnel and limit the approach to hazardous areas. The critical assembly room is approximately 35 feet square and has a clear ceiling height of 33 feet. A

* G. B. Zwetzig, ed., "Organic Moderated Reactor Critical Experiment Hazards Summary," NAA-SR-3220, December 15, 1958.

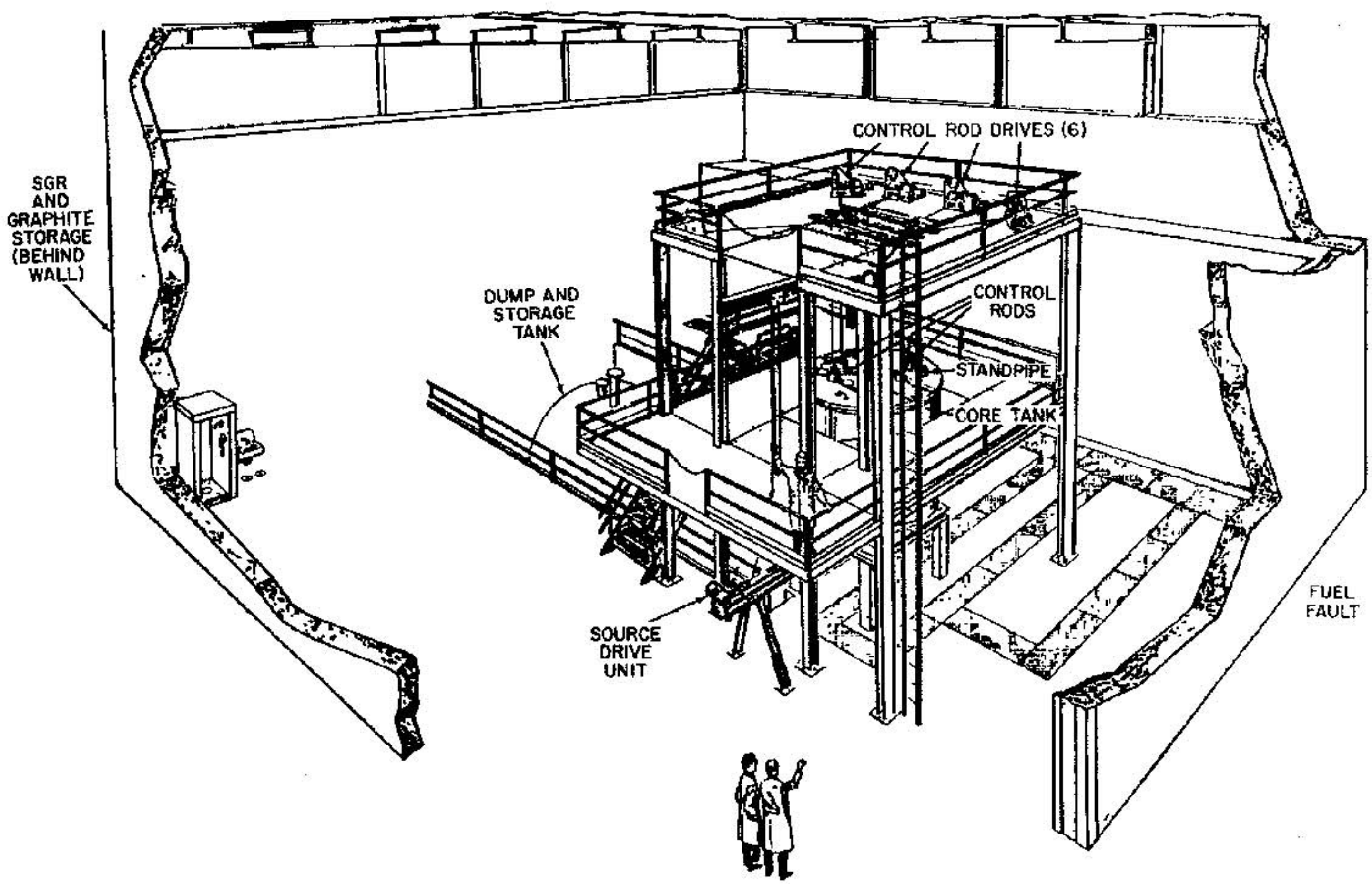

Figure V-13. Design of OMR Critical Facility

5-ton bridge crane services the area, and a service door is provided on the northeast corner for truck entry to the main high bay. The roof consists of built-up roofing over rigid insulation on metal decking (no shielding). A 19 by 12 by 10-foot-deep pit which abuts the east wall is provided in the concrete floor for the moderator drain and storage tank.

A fuel storage room is provided west of the assembly room, adjacent to the west shield wall and truck entry area. The fuel storage room faces a corridor leading from the control room in the low-bay area to the assembly room. A change room, opening on the same corridor, is provided to control traffic from the control room and low-bay area to the fuel storage room and assembly room.

The low bay is a single story structure constructed of insulated metal siding and metal lath and plaster interior. The roof is constructed of built-up roofing over insulation on metal decking (as over the high bay), except for the counting room, which has a 2-foot-thick reinforced-concrete roof and walls. In addition to the counting room, the building contains the control room, machine shop, laboratory area, mechanical equipment area, office area, lavatory area, and miscellaneous utility areas. The control room is located next to the south shield wall of the assembly room and controls access to the high-bay area.

The ventilating system for the critical assembly room is separated from that used in the remaining portion of the building. The system for the critical assembly room maintains the high bay at a negative pressure relative to surrounding areas. During reactor operation, the exhaust is discharged through a stack terminating 45 feet above grade level (10 feet above roof). This effluent is continuously monitored and is automatically diverted through pre- and "absolute" filters, if higher than permissible activity levels are detected. With the filters in use, approximately 3500 cfm of air are discharged to the stack. When the reactor is off, the high-bay area is exhausted through four roof-mounted power exhausters equipped with motorized dampers in order to provide a high air change rate when personnel are in the high bay. Interlocks require that the exhausters be turned off and the dampers closed prior to reactor startup. A conventional ventilating system maintains the other building areas at a positive pressure with respect to the assembly room.

The critical assembly consists of a core vessel, thermal shield, fuel and control elements, source shield and drive mechanism, a moderator storage and drain tank with connecting lines, and an access stand for the assembly. The core is contained in a 6 foot diameter by 8-1/2-foot-high mild-steel tank, which is supported 5 feet above the floor by a massive steel stand. The core mock-up utilizes slightly enriched fuel in a heterogeneous, organic-moderated lattice. The uranium metal fuel elements which have been used are of the flat plate, box-type, or of cylindrical or other special design. The moderator and reflector is a commercial mixture of terphenyl isomers which is maintained in a liquid state by means of electric heaters. A moderator dump system with a quick-opening valve is provided to drain the core vessel in the event of scram or shutdown. Experiments can be performed in the temperature range from 325° to 600°F, in an unpressurized system.

Boron carbide-filled shim and safety rods, activated by cables, are used for control.

c. Description of Experimental Program

A general evaluation program for OMR systems using plate and circular-type, slightly enriched, metal fuel elements has been underway in the facility since early 1959. The critical loading, control characteristics, and other reactor parameters have been studied. Void coefficient measurements are also scheduled to be performed on this system. Critical experiments in support of the Piqua Nuclear Power Facility have also been performed.

d. Future Plans for Use of Facility

The future experimental program will be in support of the Prototype Organic Power Reactor project (using UO_2 fuel). It is possible that Doppler coefficient studies will be conducted, in which case an increase in power level to 1 kw will be required.

The program may possibly be reoriented to include a test with the SNAP 10A transient test reactor. In this test, the reactor would be brought to critical and control drum calibrations performed prior to shipment to Idaho in 1963 for the transient tests. In addition, a two-region graphite critical assembly, in support of the Advanced Sodium Graphite Reactor project, may possibly be installed in the existing OMR tank.

13. Sodium Reactor Experiment (SRE), Building 143

a. Location

Figure III-3 shows the location of the SRE, Building 143, and associated experimental facilities, within the AI-NDFL and its relation to other installations in the local area. The installation is located in the eastern portion of the AI-NDFL, near the northwest boundary of the area. The site is located on AEC-optioned land. The minimum distance from the SRE building to the AI-NDFL boundary is approximately 330 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1500 feet to the southeast. The nearest occupied structure is the Engineering Test Building (Building 003), which is approximately 280 feet east of the reactor.

The reactor room floor elevations are approximately 1850 feet above sea level. The stack exit is approximately 1925 feet above sea level.

b. Description of Building and Equipment*

The SRE was designed to provide the means for exploring and improving the technology for generating nuclear power with sodium-cooled, graphite-moderated reactors. The installation presently consists of a 20-Mwt reactor equipped with a heat transfer system, service systems, a hot cell, control room and administration area, and a steam power plant. Figure V-14 illustrates the type of facilities and their design.

The reactor, its heavily shielded heat transfer systems, and the hot cell are located below grade level and are housed by a 10,000 ft^2 superstructure. The high bay superstructure is approximately 102 feet long by 48 feet wide and 52 feet high. The structure is a conventional type building of reinforced-steel, 5.5-inch-thick concrete tilt-up panels. The roof is composed of poured gypsum, insulation, tar, and gravel and is about 6 inches thick.

The reactor is located in a shielded vault below the center of the reactor room floor. The entire core is contained in a 1-1/2-inch stainless-steel vessel 19 feet deep and 11 feet in diameter, which is in turn contained in a 1/4-inch mild-steel outer tank 19 feet deep and 12-1/2 feet in diameter. The upper surface of the top shield is at the reactor room floor level. This removable

* A. I. Staff, "Hazards Summary for Thorium-Uranium Fuel in the Sodium Reactor Experiment," NAA-SR-3175 (Revised), July 1, 1959.

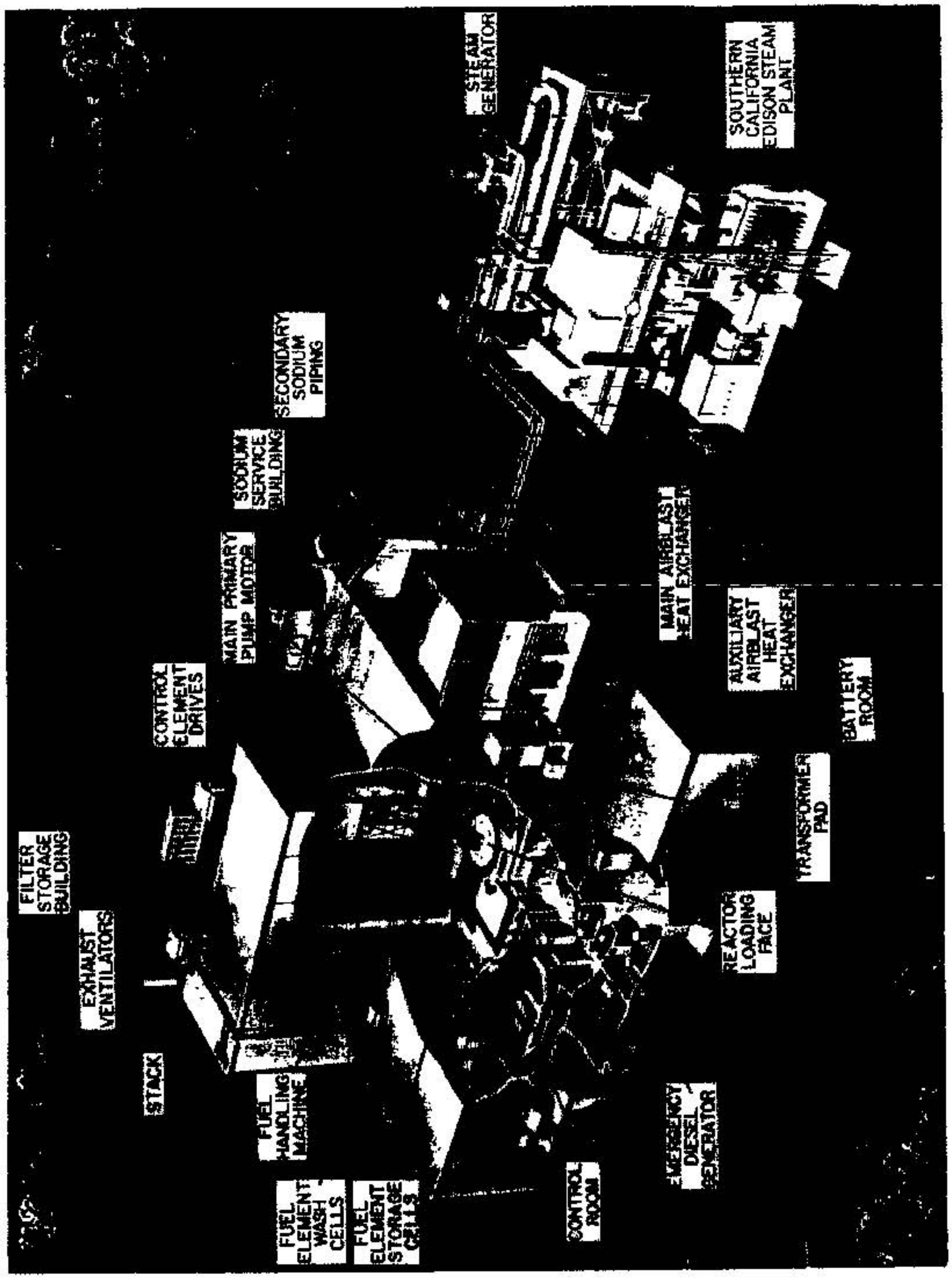

Figure V-14. Design of Sodium Reactor Experiment Facility shield is circular and rotatable and contains stepped plugs which permit access to the core vessel for handling core components and fuel.

The two primary loops are installed in separate nitrogen-filled concrete-walled galleries located below the reactor room floor level. Motors for the mechanical sodium pumps and for control rod drives are located above reactor room floor level for ease of maintenance. The secondary sodium coolant lines extend from the intermediate heat exchangers to a concrete slab outside the building, where the airblast heat exchangers are located. The secondary coolant can also be diverted to the Southern California Edison mercury-steam plant, which is located approximately 70 feet east of the SRE, for conversion of heat to steam and useful electric power.

A 75-ton-capacity bridge crane is designed to move within the reactor room and is capable of supporting lead-shielded casks (~50 tons) used for the removal of radioactive elements from the reactor core. At the west end of the reactor room special facilities are installed, again below floor level, for the cleaning and storage of these elements (fuel cleaning cells and storage tubes).

A shielded hot-cell complex, consisting of two cells, with an operating area in front and a personnel air lock leading to a service area in the rear, is also provided in the west end of the high bay below the floor level. The cells are equipped with three windows and six manipulators. Access holes and plugs are provided in the ceiling to receive elements from the bottom surface of the fuel handling cask, which discharges from the reactor floor level. These cells are used for inspection, disassembly, and packaging of SRE fuel elements and components, and for nondestructive testing of fuel. Personnel access to the cell complex is from a door in the high bay (or exterior door) which opens to a stairwell leading to the cell operating area. A portion of the hot cell complex extends beyond the high bay wall and is covered at grade level by a concrete deck and access plugs.

The control room and administration area is contained in a conventional steel frame building (construction similar to the high bay) which is attached to the south side of the high-bay superstructure. The building has outside dimensions of about 102 by 46 feet and contains a grade level area and a mezzanine area. The grade floor contains a 41 by 19-foot control room, with viewing windows on the reactor room side. A 45 by 24-foot electrical distribution room to the east of the control room contains electrical equipment. A 45 by 15-foot area to the west of the control room contains toilets and change room facilities. A 60 by 20-foot area in front of the control room is used as a visitor's observation gallery and contains an office area and an experimental equipment room.

The second floor is devoted almost entirely to administrative and office areas, except for a 20 by 16-foot heating and ventilating equipment room. During recent modifications, a single story concrete block structure, containing about 2500 ft^2 of floor space, was added to the south side (and to a portion of the east side) of the above building. This area contains a lobby, a health physics room, an air conditioning room, a boiler room, an electrical equipment annex, and a battery room.

In addition to the above structure, separate underground vaults and storage tanks are provided outside of the building for primary sodium storage, liquid waste hold-up, and radioactive gas hold-up and decay.

Fresh air is supplied to the high bay through cleanable filters and unit heaters and is exhausted through two roof-mounted power ventilators, which maintain the high bay at a negative pressure with respect to ambient (5 air changes per hour). Standby high-efficiency filters ("absolute") have been added on the downstream side of each exhauster, which may be cut in during an emergency, if needed.

The control room, offices, and administrative areas are maintained at a slightly positive pressure with respect to ambient. The hot cells are not designed to be airtight, and a conventional high-volume air system is used to ventilate the cells through pre- and "absolute" filters to a dilution stack. In the event of an emergency, dampers can be closed to isolate the cells. Because the cell service and operating areas are also vented to the dilution stack through pre- and "absolute" filters, any leakage from the cell is therefore filtered before being discharged to the environment. The cells are normally maintained at a negative pressure with respect to the cell operating area, and the operating area is maintained slightly negative with respect to the high bay. The dilution stack is also used in conjunction with the gas hold-up decay tanks, to provide for safe disposal of radioactive gas to the environment (controlled dilution and release).

c. Description of Experimental Program

The SRE began operations with sodium and the first core (unalloyed uranium) in April 1957. During the next two years the systems were tested and the facility was used to generate power and electricity at power levels up to 21 Mwt. Operation with reactor outlet temperatures of 1000°F and higher, and steam temperatures of up to 1000°F was accomplished during this period, demonstrating the capability of sodium-graphite-reactor systems to achieve modern steam temperatures. Following the SRE fuel element damage incident of July 1959, the uranium metal core was removed and subsequently replaced with a thorium-uranium alloy fuel loading. Power operation and system testing with this core has been in progress since September 1960. The experimental program presently includes testing of all components in the plant with particular emphasis on fuel irradiation.

d. Future Plans for Use of Facility

The emphasis on fuel irradiation will increase in the future, with the projected Power Expansion Program. In this program, it is planned to increase the reactor power level from the current 20 Mwt to 25 to 30 Mwt, with a further possible increase to as high as 40 Mwt, in the more distant future. Of particular interest is the fact that a change to uranium-carbide fuel is planned. Operation of selected fuel elements at power densities as high as 90 to 100 kw/gm and exposure to 40,000 Mwd/T at the point of maximum burnup is expected.

14. Epithermal Critical Experiments Laboratory (ECEL), Building 100

a. Location

Figure III-3 shows the location of the ECEL installation (Building 100) within the AI-NDFL and its relation to other installations in the local area. The facility is located in the southwestern portion of the AI-NDFL and to the southwest of the SNAP complex. The minimum distance from the facility to the AI-NDFL boundary is approximately 650 feet. This boundary lies in a northwesterly direction (Simi Valley direction). The nearest Rocketdyne area boundary is approximately 1150 feet to the southeast. The nearest occupied structure is the SGR-OMR Critical Experiments Facility (Building 009), which is approximately 200 feet southwest of the reactor.

The finished floor elevation in the ECEL is 1827 feet above sea level, and the stack exit is approximately 1877 feet above sea level (two stacks).

b. Description of Building and Equipment*

The ECEL is a critical facility designed to investigate the characteristics and nuclear properties of epithermal neutron energy systems utilizing thorium fuels. Figure V-15 illustrates the facility design. The facility is a steel and concrete structure 98 feet long by 72 feet wide, with a shielded critical assembly room in one section of the facility and a subassembly room, fuel fabrication area, control room, offices, laboratories, fuel storage vault, equipment room, change room, etc. in the remaining section. The facility is surrounded by a security fence, to control access to the area.

The critical assembly room is constructed of poured-in-place reinforced concrete with inside dimensions of 32 by 48 feet, and a 31-foot floor-to-ceiling height. The south wall (adjacent to the control room) is 5 feet thick to a height of 12 feet and 3 feet thick to the ceiling. The east wall adjacent to the subassembly room is of the same thickness, except that a 6 by 8-foot opening and shield door are provided on this side for personnel and lift truck access. The north and west walls are 4 feet thick to a height of 12 feet and 2 feet thick to the ceiling. An emergency exit labyrinth and a removable, stepped, shield

* D. T. Eggen, et al., "Epithermal Critical Experiments Preliminary Safeguards Report," AI-4120, August 12, 1959 and Supplement A.

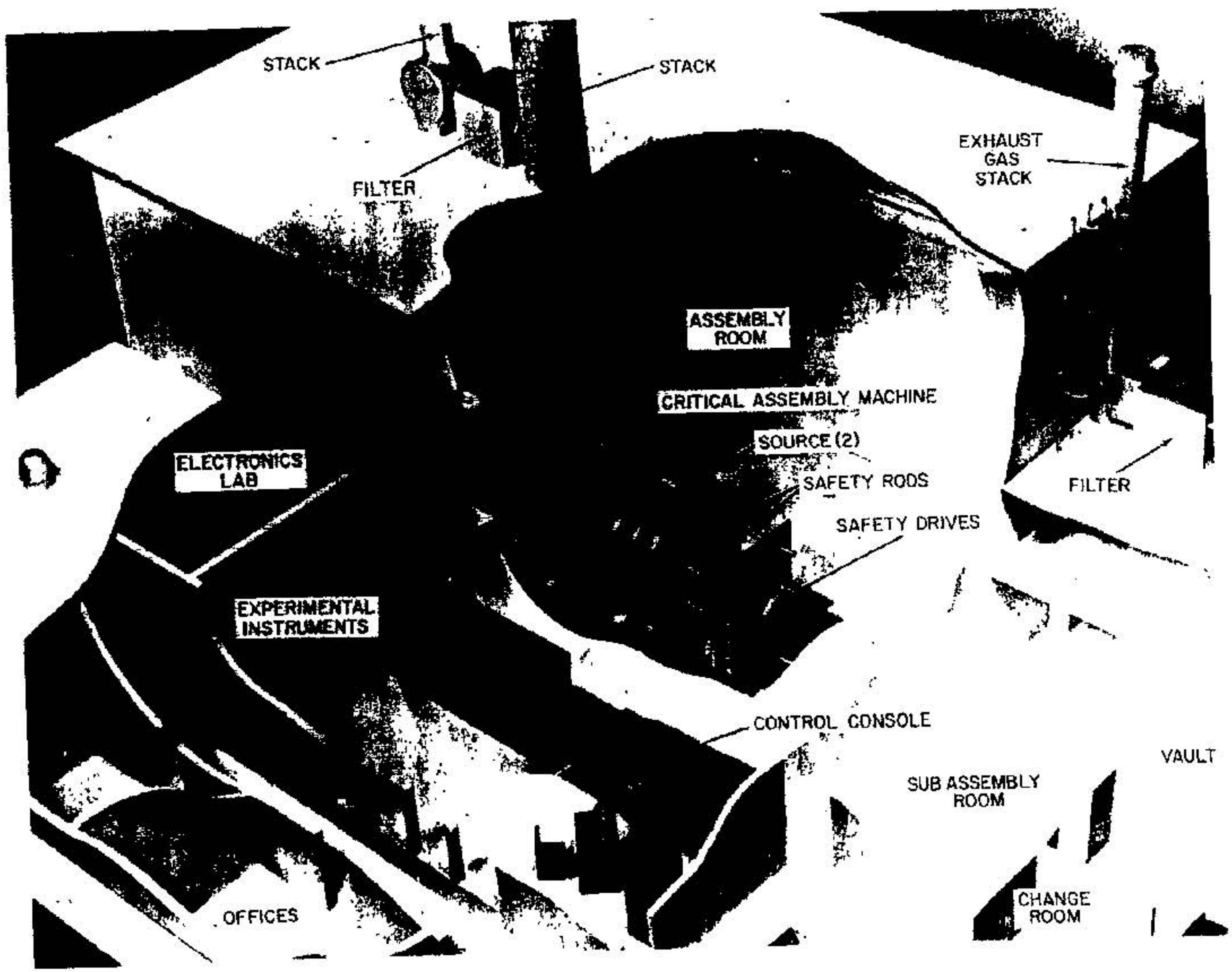

Figure V-15. Design of the Epithermal Critical Experiments Laboratory plug are provided in this wall for personnel and truck access. The roof is a self-supporting, 12-inch-thick, reinforced-concrete slab .th a standard, hot mop roof deck. The room is equipped with a 2-ton capacity bridge crane.

The remaining portion of the building is an L-shaped structure adjacent to the south and east wall of the critical assembly room. The outer south wall is 98 feet long and the outer east wall is 69 feet long. This portion of the building is constructed of insulated metal siding exterior and metal-lath and plaster interior, except for a 10 by 13 by 8-foot-high fuel storage vault, which is constructed of 1-foot concrete walls and ceiling. The roof decks are constructed of insulated metal and built-up roofing, with a 12-1/2 foot eave height. The east wing contains a subassembly room adjacent to the west wall of the critical assembly room and a receiving room, hot chemistry laboratory, and counting room on the outside west wall. The subassembly room and receiving room are separated by the fuel storage vault and a connecting passageway. The subassembly room is 36 feet long by 13 feet wide, the receiving room is 22 feet long by 16 feet wide, and the hot chemistry laboratory is 19 feet long by 16 feet wide. A 14 by 11-foot change room is provided for control of personnel traffic to the critical assembly room, subassembly room, fuel storage vault, hot chemistry laboratory, and the receiving room.

The east wing also contains a 28 by 16-foot counting room, a small material storage room, facility equipment room, health physics office, toilets, and entry vestibule, and control station.

The wing opposite the south face of the critical assembly room contains a 33 by 14-foot control room, a 22 by 14-foot instrument service and laboratory area adjacent to the critical assembly room wall, and a 56 by 12-foot office area in the remaining portion of the wing.

To contain radioactivity within the critical assembly room, the room and its openings have been designed for minimum leakage. During normal operations, the atmosphere will be maintained at a negative pressure of 0.2 inch of H_2O, by exhausting in-leakage air through the exhaust system and stack at a rate of approximately 85 to 170 cfm. The exhaust system will collect air from the vicinity of the critical assembly machine and draw it through a damper and pre- and "absolute" filters to the 50-foot stack. The room atmosphere will also be recirculated at a rate of 5,000 cfm through an air conditioner, to maintain the room at a constant temperature of 75°F and a maximum relative humidity of 25%. On loss of power or a scram signal, the exhaust dampers automatically close and seal the assembly room. Calculations indicate that the room integrity will be maintained during and following an excursion of up to 150 Mw-sec. In the event of a larger excursion, a blow-out diaphragm in the vent system would rupture, venting the excursion-produced activity up the stack.

The subassembly room, fuel storage vault, fuel fabrication area and change room are maintained at a slightly positive pressure with respect to the critical assembly room, but negative with respect to other building areas, by a separate ventilation system consisting of pre- and "absolute" filters, an exhaust blower, and a separate 50-foot stack.

The critical assembly machine is essentially a base on which two tables are mounted. One table is stationary and the other movable. One-half of the reactor is built up on each table. The two half assemblies are brought together remotely by driving the movable table toward the fixed one. The reactor is built up by loading drawers containing the reactor materials into 2-inch by 2-inch matrix tubes, which are stacked in a square array and clamped rigidly to the tables. Safety and regulating elements are core materials loaded in special drawers and moved in their respective matrix tubes by rod drives. The safety rods are withdrawn from the core under scram conditions by pneumatic pistons and springs.

c. Description of Experimental Program

The basic objectives of the critical experiments program, which started in the latter part of 1960, are to obtain experimental verification of the theoretical predictions for a full-scale nuclear electric power station and to measure experimentally the breeding ratio in such a system. Nuclear parameters are measured, using various core compositions and geometries of U^{233}, thorium, aluminum (as a substitute for sodium), graphite, and stainless-steel. A series of at least three Advanced Epithermal Thorium Reactor (AETR) type cores will be run in spherical geometry with the U^{233} test regions at the center. Also planned for investigation is a slab reactor case which will allow a study of the thorium blanket-core interface and a cylindrical core design which will be used to study engineering mockups of fuel and control elements which are long in one dimension.

d. Future Plans for Use of Facility

Measurements related to the neutron lifetime in the different assembly configurations may be made, as well as other experiments to aid in the analysis and interpretation of proposed AETR cores. N major modifications are presently planned for the facility.

VI. RE-EVALUATION OF ACCIDENT ANALYSES

A. STATEMENT OF CONSIDERATIONS

One of the prime requisites of a study such as that undertaken in this report is to ensure that, as much as possible, all evaluations are based on a consistent set of assumptions. Since the reports describing the hazards and maximum credible accidents attendant to each reactor facility at the AI-NDFL have been published over a period of years, methods and data used in evaluating the radiological consequences associated with the postulated accidents have been evolutionary in nature and thus have varied from report to report. The changes can be attributed to continuing efforts at Atomics International to improve our knowledge of the site (including the site meteorological characteristics) the techniques used in hazards assessment, and the methods of their utilization. In order to unify these data and methods, the radiation exposure aspects of each maximum credible accident have been re-analyzed, using a consistent set of assumptions and analytical techniques.

The physical nature of the maximum credible accidents and the events leading to them has been described in the appropriate hazards and safeguards reports. These basic accident concepts remain unchanged in this study and are briefly summarized here for the benefit of the reader who is unfamiliar with the original reports. Column 2 of Table VI-2 lists the reference reports. A bibliography of the hazards summary reports used in these studies is included in Appendix C.

B. ASSUMPTIONS REGARDING RADIOACTIVITY RELEASE

The general philosophy underlying the choice of assumptions upon which the re-evaluation of the radiological consequences of the maximum credible accident postulated for each reactor facility at the AI Nuclear Development Field Laboratory is based is to utilize data which provide the required degree of conservatism only in situations wherein considerable uncertainty exists. For example, although there is substantial evidence to indicate that iodine is not released in large quantities in fuel meltdowns occurring in sodium, NaK, or organic cooled reactors, generally, no credit is taken for this in the re-evaluation of iodine releases. However, it should be mentioned that Atomics International is presently engaged in an extensive experimental program to determine the extent of iodine release in such systems. When this information becomes available, it will be applied to the evaluation of the hazards of maximum credible accidents and hence will remove some of the unnecessary conservatism built into the present analysis. Another example in this area is the extent of gross fission product release postulated for the different accident conditions. In this regard, AI is presently conducting a program to evaluate the actual degree of fission product release from SNAP fuel elements. Because, however, sufficient experimental data were not available when this report was being prepared, the assumed core releases were based on criteria reflecting a conservative approach.

With regard to the choice of meteorological parameters used to evaluate the extent of diffusion involved in carrying the radioactive release to the receptor, here again, both relatively pessimistic meteorological conditions and their associated micro-parameters have been utilized. The values utilized have been discussed with competent meteorologists employed by the Rocketdyne Division of North American Aviation, and are felt to provide a sufficient degree of conservatism.

To place the results on a consistent basis, the following criteria were applied in re-evaluating the radiological consequences of the maximum credible accidents.

1. Core Inventories

In all cases, core inventories were re-evaluated to place all fission product inventories produced in the accidents on a common basis. Also, the iodine inventory was evaluated separately for each case. The models for evaluation of both the gross fission product and iodine inventories from the postulated transient and/or steady-state operation are outlined in Appendix A.

In the case of all the SNAP reactor cores, it should be pointed out that the dynamics of the transients (caused by ramp insertions) are such as to result in the reactor core operating at significant power levels for relatively long periods of time ($\sim$10 to 20 sec). However, the fission product inventories generated in these cases were calculated as for a burst, thus introducing a conservative error.

2. Core Releases

Owing to the variety of reactor cores at the AI-NDFL site, and the different core conditions achieved as a result of the postulated maximum credible accidents, several different models for core releases were necessary. They are grouped as follows:

a. SNAP 2, 10A, and 8 Critical Assemblies

The fuel rods for the SNAP critical assemblies consist of a homogeneous mixture of γ-phase hydride of Zr + 10 wt % of 93% enriched uranium alloy. The dissociation effect of zirconium hydride at elevated temperatures provides the SNAP reactors with an important inherent safety feature. The extent of dissociation is a function of temperature; at about 1530°F for SNAP 2/10A systems, and 1830°F for SNAP 8 systems, the hydrogen gas pressure increases to the point where calculations indicate that the fuel element cladding ruptures, resulting in a permanent hydrogen loss and reactor shutdown.

Very little is known regarding the extent of fission product release from fuel elements where only the cladding ruptures (i.e., no fuel melting). Because critical experiments do not result in significant fuel burnup, it is believed that the magnitude of the fission product release may very well depend on a diffusion mechanism. A useful approximation for establishing the extent of the release may be made by assuming that the diffusion of volatile fission products in the fuel matrix is inversely proportional to the square root of the molecular weight of the diffusing species. The diffusion model is limited by the number of impacts against a barrier, and is valid when the openings in the barrier are larger than the molecules of the diffusing species. Although this model is not strictly applicable to the case at hand, its application should give conservative results. Because the percentage of hydrogen released from the dissociation of the zirconium hydride matrix as a function of transient temperature can be fairly closely defined,* the fraction of fission products released simultaneously may be approximated by the inverse square root of their molecular weight. This approximation was used for establishing the extent of the fission product release from the accidents postulated to occur in the SNAP critical assemblies.

*D. L. Henry, "The Preparation and Properties of the Hydrides of Zirconium, Yttrium, Cerium, and Their Alloys," NAA-SR-3700, (Classified Report, Title Unclassified, July 1959).
C. L. Huffine, "Evaluation of Clad Hydrided Zirconium as Solid Moderator," APEX 335, November 29, 1956.

b. SNAP 2, 10A, and 8 Power Reactors

A release comparable to that from a fuel meltdown was assumed for the accidents postulated to occur in these power reactor systems. In this regard, it should be pointed out that extensive experimental work by W. V. Johnston* indicates that uranium-zirconium fuel subjected to high burnup and high temperature during the accident may release its contained fission products to an extent approaching that from molten fuel elements.

In addition, Oak Ridge experiments involving the melting of certain types of irradiated fuel elements have been reported by Couchman.† Of all the fuel elements tested, the STR fuel element most closely resembles the SNAP fuel element from the standpoint of composition, since it also contains a zirconium matrix. The release from the STR fuel element were therefore used to determine the amount of fission product activity which would escape from a SNAP power reactor core. The release consists of: 100% of the noble gases, 26% of the halogens, 10% of the cesiums, 1% of the strontiums, and 0.5% of the bariums.

c. Homogeneous Solution Cores

There is considerable experimental evidence available§ which indicates that, because of the high solubility of iodine in water, only gases will normally be released in significant quantities from a homogeneous core undergoing a transient. If, however, the transient is accompanied by a violent dispersion of the solution over a large area, then there is some uncertainty as to the extent of iodine retention in the dispersed and quickly evaporating liquid. Therefore, it was deemed necessary to assume that, under such conditions, the fission product release would be analogous to that from a molten fuel element.

d. Other Cores

The recommendations in the May 1961 version of the proposed 10 CFR Part 100 were followed for cores other than those noted above. A core release

*William V. Johnston, "The Effect of Transients and Longer Time Anneal on Irradiated Uranium-Zirconium Alloys," KAPL-1965, 1958.

†J. C. Couchman, "Graphic and Tabular Aids for Reactor Hazards Evaluation," FZM-2277, June 9, 1961.

§R. S. Hart et al., "Change of KEWB Reactor Cores - Evaluation of Significance with Regard to Associated Hazards," NAA-SR-MEMO-4928, February 4, 1960.

of 100% of the noble gases, 50% of the halogens, and 1% of the other activity was assumed for the maximum credible accidents with these other cores. It was also assumed, as in the proposed 10 CFR Part 100, that one-half of the released iodine would condense and settle out on colder surfaces within the reactor room. Recent data* indicate that this fraction is too low, i.e., that a much larger fraction of the released iodines would condense within the building and hence not be available for release to the atmosphere. If, however, the maximum credible accident also postulated oxidation of the fuel element, the iodine release was increased from 50 to 100%. Again, one-half of this amount was assumed to deposit within the confines of the building.

3. Building Release

Because facility characteristics differ considerably from each other, no general statement can be made about the mode and duration of the atmospheric release; rather, each facility was treated individually. Whenever the physical conditions of the plant permitted, following the maximum credible accident, advantage was taken of the ventilation system and air filters to reduce the release of the nonvolatile portion of the fission products. In all cases, the iodine was assumed not to deposit in the filters.

In most of the postulated accidents, since they involve transients, an instantaneous release of fission products from the core to the building was assumed. If the accident was not assumed to destroy the integrity of the building, the atmospheric release was made a function of the building ventilation or leak rates. In the case where the building roof was assumed to be removed by the force of the accident, only a short time lag of a few seconds was allowed for the formation and release of the radioactive cloud.

Assumptions made concerning facility holdup time and atmospheric release rates, and the resulting exposures, are discussed in detail in Appendix A of this report. (Appendix B discusses the SRE calculations, only.) It should be noted in the mathematical models used that the fission product holdup time in the facility markedly affects only the whole-body gamma dose calculations. The method used to determine the thyroid dose and the dose from ground deposition, based only on iodine isotopes, assumes that no significant radioactive decay takes place while the material is being transported to the receptor.

*J. B. Morris et al., "The Removal of Low Concentrations of Iodine from Air on a Plant Scale," AERE-R 3917, December 1961.

C. APPLICABILITY OF SITE METEOROLOGICAL DATA TO RADIOACTIVITY DISPERSAL CALCULATIONS

As mentioned in Section II, the site location provides certain unique features which result in a large degree of protection for the general public from radioactivity releases generated at the site. The general topography beyond the site boundary is quite irregular and is essentially uninhabited by the general public for a distance of about 3 miles, the nearest residence being located about 2 miles from the site. The estimates of expected population growth in the Valley areas do not indicate that the population density in this zone will change appreciably (see Figure II-7).

The site meteorology indicates that prevailing wind directions are toward the Simi Valley in the morning and the San Fernando Valley in the afternoon. For the most part, radioactivity concentrations in these valleys would be very low due either to a high cloud trajectory or good diffusion conditions (see Figures II-15 and II-16). With stable conditions on-site and unstable conditions over the valleys below, the large margin of safety inherent at other times is somewhat reduced. However, as was demonstrated in Section III, because of the large transport distances involved before the radioactive material arrives at the nearest boundary of a populated community, the doses received would be quite small compared to those received at the site boundary. (The maximum permissible dose at the site boundary, as used in this study, remains unchanged from that set forth in 10 CFR Part 100, namely, 25 rem whole body or 300 rem thyroid (TID) from exposure during the first 2 hours after the accident.) Obviously, the doses received in any of the populated communities, even considering longer exposure (also per 10 CFR Part 100), would still be several orders of magnitude less than at the site boundary, due to the long transport distance involved, i.e., at least 3 miles, which would provide a more than adequate safety margin. Furthermore, this would indicate that the doses at the site boundary are the controlling factor in determining the adequacy of the site insofar as the hazard to the general public is concerned. (See also Section III.)

As indicated in Section II.C, the wind data obtained from the Rocketdyne weather instruments provide a good picture of mean air flow patterns. In addition, it has also been shown that there is good agreement between measurements made at AI facilities and the Rocketdyne data, thus indicating that information obtained from one location is generally applicable to the other.

What appear to be lacking, however, are specific measurements of the diffusion and stability coefficients associated with the different meteorological conditions occurring on either the AI or Rocketdyne sites. Normally one might consider this information requisite to the preparation of an adequate hazards analysis. Here, again, when proper consideration is given to the site location and utilization, the necessity for local experimental data on which to base the choice of the proper micrometeorological parameters becomes questionable, and, in fact, appears not even to be required.

To amplify this point, emphasis must once again be placed on the fact that safety requirements dictate that doses be below already established permissible limits at the site boundary, only a few hundred feet away from most reactor locations. With such a short distance available for diffusion, it is obvious that differences in the parameters associated with the specific meteorological conditions assumed to exist at the time of the accident would have only a minor influence on the doses at the boundary. It is obvious, also, that provisions must be made to ensure that doses at the site boundary be within limits, even under poor meteorological conditions.

These rigid safety requirements demand that, under all meteorological conditions, releases be kept small. With this as the prime criterion, it follows that, if the on-site hazard is small, then site boundary doses would be below limits and, because of the long transport distances involved, doses at populated areas would be insignificant.

It would appear then, that for the Atomics International site, the role of meteorology is small. Short distances between reactor locations and to the site boundary require that releases, whether planned or unplanned, always be small, which relegates diffusion, and particularly the knowledge of diffusion parameters, to a relatively unimportant consideration. Populated areas are well protected by the limited magnitude of the releases and long cloud trajectories.

Meteorological considerations are not entirely absent, however. Wind rose data, obtained from the Rocketdyne site and found to be valid for the AI site as well, will be used for purposes of future facility planning and for the establishment of emergency on-site evacuation areas, where possible.

In conclusion, it can be stated that a substantial meteorological knowledge of the AI site exists. It shows, that within the scope of current operations, there is a high degree of safety for all populations. Unless the type of activity at the site is to change drastically, there seems to be no compelling reason to institute a more detailed meteorological program.

D. EVALUATION OF ATMOSPHERIC DISPERSION OF RELEASED RADIOACTIVITY

1. Choice of Meteorological Parameters

A detailed description of the site, the meteorological environment, and the population distribution was given in Section II of this report. Because the site is located in a mountainous area and the areas populated by the general public are located some distance removed in the valleys below, it was found that different unfavorable weather conditions would be required to produce the maximum dose at each location (i.e., on-site and at the nearest community). Therefore, a discontinuity in meteorological conditions was postulated to exist between these regions, and the dispersion of radioactivity from the release point to a receptor in each region was evaluated separately, assuming unfavorable meteorology in each case. The two regions considered will be referred to as the "on-site area" and the "off-site area."

The atmospheric dispersion of radioactivity is based on what is considered to be unfavorable meteorological conditions for the site. In addition, the micrometeorological parameters chosen for the purpose of this study are, in themselves, conservative within the specific ranges in which they apply. A summary of the parameters to be used in the calculations of the diffusion of radioactivity released from the reactor facilities on site is given in Table VI-1. Their selection was based on observations of local conditions* and the recommendations of various other sources.†

a. The On-Site Area

The on-site area is defined as the Atomics International Nuclear Development Field Laboratory. Access to this area is controlled. For further details the reader may refer to Section II.

*M. Tarpinian, "Air Flow and Diffusion Studies in the Simi Hills," NAA-SR-MEMO-848, December 14, 1953.

†H. Weiss, Meteorologist, Rocketdyne Division, North American Aviation, Inc., Personal Communication, March and April 1962. See also "Meteorology and Atomic Energy," AECU-3066, U. S. Department of Commerce Weather Bureau, July 1955.

TABLE VI-1

METEOROLOGICAL CONDITIONS AND PARAMETERS

Region	Release Type	Weather Conditions	C	C_y	C_z	n	$\bar{u}$
On-Site	Elevated	Fumigation	0.16	0.21	0.12	0.29	3
On-Site	Ground Level	Inversion	-	0.05	0.03	0.5	0.5
Off-Site	All Types	Mild Lapse	0.10	-	-	0.29	3

(1) Elevated Releases

For elevated releases, local experience* has indicated that the unfavorable condition for atmospheric dissemination of elevated releases of radioactivity is the weather type known as "fumigation." The base of the temperature inversion layer, which is present quite often on the site, generally rises to an elevation of 200 to 1000 feet above the site during the day and falls to ground level during the night. If the released radioactive cloud or plume is trapped just below the inversion base, upward diffusion would be very limited. In addition, because neutral or mild lapse conditions usually exist between the inversion base and the ground, diffusion of the cloud downward toward the ground would be fairly rapid. Therefore, to minimize the upward diffusion of the fission products, the inversion base will be assumed pessimistically to occur 5 meters above the height at which the release takes place.

In this study, elevated releases were assumed only for facilities where the stack height is at least 2-1/2 times the height of the building above ground level and then only in cases where the assumed accident would not impair the functioning of the ventilation system.

(2) Ground Release

Calculations of the atmospheric dispersion of the radioactive material from accidents in facilities without a stack, or with a stack less than 2-1/2 times the height of the building, were based on a ground level release model.

*M. Tarpinian, op cit.
H. Weiss, op cit.

This approach is conservative, at least at the closer distances, since, in many instances, some credit could have been taken for effective stack heights, owing to the effluent discharge velocity and to temperature difference with ambient air. Another conservative factor introduced in the calculations was the assumption of rather pessimistic meteorological parameters during postulated ground level releases (see Table VI-1).

b. The Off-Site Area

The off-site area was assumed to be all land lying beyond the Atomics International Nuclear Development Field Laboratory boundary. Except for the Rocketdyne area, this land is unpopulated for a distance of at least 2 miles and it is unlikely that there will be any significant population increase. The nearest community boundary starts at a distance of about 3 miles from the site, in Simi Valley, and is several hundred feet below the site elevation. Because of these conditions, it is believed that the unfavorable weather condition for off-site dissemination of radioactivity may be approximated by assuming that a mild lapse condition exists which results in an effective release height of about 200 feet relative to the valley floor. Doses at the closest boundary of the nearest community were based on this assumption. Since it is believed that the above-mentioned effective release height is independent of the type of on-site release, no credit for stack releases was taken at off-site distances.

2. Calculation of Exposures

To evaluate properly the consequences of atmospheric dissemination of radioactivity for each postulated accident, exposures and doses were computed for three general points of interest. These points are:

a) On-site, i.e., at the nearest facility (ground level release) or point of maximum concentration (elevated source release).

b) The nearest site boundary (ground level release) or point of maximum ground concentration if beyond the nearest boundary (elevated source release).

c) Off-site, i.e., at the nearest community boundary.

Some clarification of the on-site exposures is in order here. In a ground release, the radioactivity concentration downwind from the accident source may be assumed to decrease with increasing distance. It is, therefore, of interest to know if the radioactivity concentration reaching the nearest facility is low enough so as not to cause excessive interference with the facility schedule and overall project program. For an elevated source release, the downwind point of maximum ground concentration is of interest because any point between the source and point of maximum ground concentration may be assumed to be exposed to a lesser degree. Hence, if the distance to the nearest facility is less than the distance to the point of maximum ground concentration, the exposure also will be smaller. Similarly, depending on the wind direction, if the downwind site boundary is closer to the source than the distance to the point of maximum ground concentration, the point of maximum ground concentration may not necessarily occur on the site. This is reflected in the tabulation in Table VI-2.

The on-site computations include a dose to the thyroid, a whole body gamma dose from cloud immersion, a fallout dose, and a direct gamma dose from the reactor building, if applicable and significant. Calculations of the doses received from all sources mentioned, with the exception of that from fallout, were based on exposure times not exceeding 2 hours. Since the fallout dose is based solely on iodine isotopes (only insignificant amounts of strontium and other particulate fission products are released from the buildings in the postulated accidents), the calculations were based on continuous exposure throughout the life of the radioactivity. The resultant dose, however, was reduced by a factor of 5 to account for the fact that personnel on site would only be exposed a portion of the time (~40 hr/wk) and that of that time, some shielding would be effective.*

Computations for the nearest site boundary were based on the same parameters and include the thyroid dose, the whole-body gamma dose from cloud immersion, and the whole-body dose from ground deposition of iodine. The criteria for exposure times were the same as for the on-site calculations, with the only exception being that the dose from ground contamination was not reduced by the factor of 5, since occupancy at the site boundary could conceivably be uninterrupted.

*The SRE calculations handled this somewhat differently because of the short half-life of the released activity (Na^{24}). For more detail, see Appendix B.

The off-site computations were based on a receptor distance of 3 miles, which is the distance from the site to the closest boundary of the nearest community. Generally, the calculations included the thyroid dose, the whole body gamma dose from the cloud, and the dose from ground deposition. In calculating all off-site doses, exposure times were taken to correspond to the duration of the accident. The dose from fallout was evaluated assuming continuous exposure. In some cases, however, when doses at the nearest site boundary were already relatively small, the off-site doses were simply noted as being negligible.

The calculated total integrated doses are quite conservative, especially the iodine doses from transients (due to the use of cumulative iodine fission yields). The thyroid dose was calculated assuming no inventory depletion from the cloud by fallout, and the gamma cloud dose was based, in most cases, on an infinite hemispherical cloud volume, which is an over-estimate, especially at distances close to the facility. Also, the dose from ground deposition of iodine was calculated on the assumption that the receptor will remain exposed continuously through the whole radioactive life of the iodine isotopes. Although, in the case of on-site exposure to ground contamination, a factor of 5 reduction was utilized, the resultant doses are still felt to be rather conservative.

Appendix A lists all equations used in these computations and the assumptions on which they are based, with the exception of some of the SRE calculations which are included in Appendix B.

E. ACCIDENT ANALYSIS FOR EACH REACTOR FACILITY

The following list is included to aid the reader in locating specific analytical material.

1. SNAP Critical Facility (Building 373) Containing SCA-4A or 4C Critical Assemblies

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The maximum credible accident for the case when the facility contains the SCA-4A or 4C critical assemblies postulates that, during the performance of remote operations in the sealed vault and contrary to the administrative operational procedures, both passive control drums of the reactor are positioned and locked in such a configuration that criticality can be achieved on the safety element alone. This is assumed to happen with the reactor in its maximum shim configuration. Both active control drums are assumed to be full out. The first reflector control drum is assumed to be driven in, leaving the reactor \$3.00 subcritical. Then the second reflector control drum is assumed to be driven in, the reactor becomes critical, and reactivity continues to be inserted as a ramp function. As the reflector approaches its full-in position, the insertion rate increases from an average \$0.05/sec to a design maximum of \$0.08/sec. In the remotely credible event of a simultaneous failure of all the scram mechanisms and failure of the operator to take corrective action, the ramp insertion will continue unchecked. About 16 seconds after the initiation of the transient, the fuel temperature will reach 1530°F and the hydrogen pressure from dissociation of the zirconium hydride fuel matrix will rupture the cladding. A peak temperature of 2100°F will be reached at 20 seconds and the reactor will be permanently shut down at 26 seconds, having liberated an estimated 35% of the total hydrogen content of the core. The transient described above was calculated to correspond to a \$3.70 step reactivity insertion, which would result in a total energy release of 50 Mw-sec.

(2) Core Fission Product Inventories

The 50 Mw-sec transient will result in the production of about 0.9×10^8 curies of gamma activity 1 second after reactor shutdown. Since the

*R. L. Brehm, ed., "Summary Hazards Report and Operations Manual for SNAP Critical Assemblies 4A and 4C (SCA-4A-4C)," NAA-SR-MEMO-7011, January 12, 1962.

hazard from the fission product activity accumulated during the periods of operation at low power levels was considered insignificant, its contribution was neglected.

(3) Mode and Degree of Fission Product Release from Core

The quantity of fission products released from the core was based on the following criteria:

a) Only volatile fission products, which are assumed to constitute 20% of the total activity, are released.

b) The diffusion coefficient of gases in the zirconium matrix is inversely proportional to the square root of the molecular weight of the diffusing species.

Under these conditions and the already established hydrogen release of 35%, it was estimated that 0.6% of the total fission product inventory, or 5.4×10^5 curies, would be released to the cell atmosphere.

(4) Release from Building

It was assumed that the cell overpressure created by the core hydrogen release and/or by possible hydrogen combustion ruptures the roof of the cell, and that the cell atmosphere would be released to the outside in the form of a puff. The initial cloud volume was assumed to be equivalent to the cell volume, i.e., 1540 ft^3. It was also assumed that the cloud would diffuse at ground level. Assuming further that all these events occur instantaneously, the activity released from the building would be 5.4×10^5 curies.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is the OMR Control Shelter, Building 375, located about 350 feet to the southwest. The calculated doses at that location are 127 rem to the thyroid, 10.5 rem whole body gamma from the cloud, and 0.73 rem from ground deposition.

(2) Doses at the Nearest Site Boundary

The nearest site boundary is located about 420 feet to the east. This boundary is the line dividing the AI and Rocketdyne sites. The calculated doses at the 420-foot distance are 100 rem to the thyroid, 7.1 rem whole body gamma from the cloud, and 0.57 rem from ground deposition. Note that reduction by a factor of 5 of the dose from ground deposition is assumed to apply at the Rocketdyne site as well as for the on-site areas.

The nearest boundary to private land is located about 1350 feet to the northwest; the dose at that point from the cloud and from inhalation would then be significantly less than at the Rocketdyne boundary; however, the dose from ground deposition would be somewhat greater due to the requirement for omitting the factor of 5 reduction for occupancy.

(3) Doses at the Nearest Community

Doses at the nearest community were calculated to be 3.8×10^{-2} rem to the thyroid, 4.3×10^{-5} rem whole body gamma from immersion in the radioactive cloud, and 2.9×10^{-4} rem from ground deposition.

2. SNAP Critical Facility (Building 373) Containing the SNAP 8 Critical Assembly

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The maximum credible accident for the case when the facility contains the SNAP 8 critical assembly is assumed to be an accidental reactivity insertion that is associated with the continuous inward rotation of one reflector control drum. The maximum design rate of reactivity insertion by this method is 2¢/sec. Under normal operating conditions, this rate will be mechanically limited to a total of 28¢ excess reactivity. However, in the extremely unlikely event of a procedural error during approach to criticality, a total insertion of $1.00 could result. If, simultaneously with such an error, both the period and power scram chennals failed to respond, ultimate reactor shutdown would occur through loss of hydrogen subsequent to bursting of the fuel cladding. Cladding rupture would occur at about 1830°F and a hydrogen loss of 2 to 4%, occurring within 2 seconds after bursting, would be sufficient to permanently shut down the reactor. Under the postulated conditions, one-half of the fuel elements in the core were calculated to have reached the critical temperature for cladding rupture, and the total hydrogen loss from the core for the accident was computed to be 13.3%. The associated energy release was calculated to have an upper limit of 60 Mw-sec.

(2) Core Fission Product Inventories

The fission product gamma activity, corresponding to an energy release of 60 Mw-sec, was calculated to be 1.1×10^8 curies 1 second after reactor shutdown. Since the hazard from the fission product activity accumulated from the periods of operation at low power levels was considered insignificant in comparison, its contribution was neglected.

(3) Mode and Degree of Fission Product Release from Core

The released fission products are assumed to be comprised of volatiles only, these volatiles constituting 20% of the total core activity. The

*A. W. Thiele, ed., "SNAP 8 Critical Experiment Summary Hazards Report and Operations Manual," NAA-SR-MEMO-7029, February 1, 1962.

released core fission product fraction was based on the assumption, discussed earlier in this section, that the diffusion coefficient of gases in the zirconium hydride matrix is inversely proportional to the square root of the molecular weight of the diffusing species. On this basis it was calculated that 1.2% of the core volatile activity will be released. This corresponds to the release of 2.6×10^5 curies.

(4) Release from Building

It was assumed that the released hydrogen would ignite and that the resultant overpressure would remove the building roof, instantaneously releasing to the atmosphere a radioactive cloud in the form of a puff. (The ignition of hydrogen did not induce burning of the fuel elements.) The initial radioactive cloud volume was assumed to be equivalent to the cell volume, i.e., 1540 cubic feet. Diffusion of the released activity was assumed to occur as from a ground level release.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is the OMR Control Shelter, Building 375, located about 350 feet to the southwest. The calculated doses at that location are 61 rem to the thyroid, 5.2 rem whole body gamma from the cloud, and 0.35 rem from ground deposition.

(2) Doses at the Nearest Site Boundary

The nearest boundary is located about 420 feet to the west. This boundary is the dividing line between the AI and Rocketdyne sites. The calculated doses at the 420-foot distance are 48 rem to the thyroid, 3.5 rem whole body gamma from the cloud, and 0.27 rem from ground deposition, Note that reduction by a factor of 5 of the dose from ground deposition is assumed to apply at the Rocketdyne site, as well as for the on-site area.

The nearest boundary to private land is located about 1350 feet to the northwest; the dose at that point from the cloud and from inhalation would then be significantly less than at the Rocketdyne boundary. The dose from ground deposition, however, would be somewhat greater owing to the requirement for omitting the factor of 5 reduction for occupancy.

(3) Doses at the Nearest Community

Doses at the nearest community were calculated to be 1.8×10^{-2} rem to the thyroid, 2.1×10^{-5} rem whole body gamma from cloud immersion, and 1.4×10^{-4} rem from ground deposition.

3. SNAP Generalized Critical Facility (Building 012)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

Because a number of different critical assemblies will be placed in this facility from time to time, it will not be possible to define a maximum credible accident in the usually accepted sense. Rather, an upper bound on the magnitude of acceptable accident will be established by virtue of the facility design criteria.

The design pressure for the facility critical cell is 110 psi. A pressure of this magnitude could be generated by a nuclear excursion followed by burning of reactor fuel material. Such a hypothetical accident would be achieved with a core, containing 100 kg of uranium carbide fuel, undergoing a 600-Mw-sec transient, following which the fuel burns as a result of the temperatures achieved in the power transient. A similar pressure would be created by the burning of all the hydrogen which would be released from 100 kg of zirconium hydride-uranium fuel in the event of a 600 Mw-sec transient.

For the radiation dose immediately outside the cell wall to exceed 25 rem, a power transient with an energy release greater than 1100 Mw-sec would be required.

(2) Core Fission Product Inventories

Since, from the standpoint of facility pressure design limitations, an energy release of 600 Mw-sec may not be exceeded, the core fission product inventories were assumed to correspond to this maximum. The critical assemblies contemplated for this facility will, because of the experimental program, have a limited fission product buildup from intermittent periods of operation at low power levels (several watts), but these will be disregarded in the dose calculations, because their hazard would be insignificant compared to that from the radioactivity created in the transient. The gross gamma activity in a core 1 second after a 600 Mw-sec excursion would correspond to 1.1×10^9 curies.

*A. W. Thiele, ed., "SNAP Critical Facility (Building 012) Summary Hazards Report," NAA-SR-MEMO-7205, April 6, 1962.

(3) Mode and Degree of Fission Product Release from Core

Since it has been postulated earlier in this section that combustion of reactor fuel releases 100% of the contained halogens, the case of burning uranium carbide fuel is more severe than the combustion of hydrogen from a zirconium hydride core. For this reason, the fission product release in a 600 Mw-sec transient was based on a uranium carbide core. On the basis of the assumptions made for the burning of reactor fuel, as described earlier in this section, the fission product release from the core was assumed to correspond to 100% of the volatiles and 1% of all other fission products. A subsequent deposition of one-half of the iodine within the building was postulated. The core fission product release was calculated to be 2.1×10^8 curies.

(4) Release from Building

The design leak rate for the cell is 1% of the cell volume in 24 hours at a differential pressure of 3 inches of H_2O. The cell has solid, 4-foot thick concrete walls on all sides and a roof of the same thickness. All cell penetrations terminate in the fuel storage and equipment room. This room also has a low leak rate, and has a volume of 6500 cubic feet. However, in the calculations which were performed, the conservative assumption was made that, in spite of the room leaktightness, all fission products released into this room would leak out to the atmosphere in about 1 hour. The calculated activity released to the atmosphere at the end of 1 hour, corrected for decay, is 100 curies. The release is postulated to occur through a 40-foot stack with which the facility is provided.

b. Consequences of Maximum Credible Accident

(1) Doses at the Point of Maximum Concentration

The point at which the concentration of the released fission products reaches a maximum lies 390 feet downwind from the facility. The doses calculated at that distance on-site are 6.0×10^{-2} rem to the thyroid, 5.4×10^{-3} rem whole body gamma from cloud immersion, and 5.7×10^{-4} rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 250 feet to the northwest of the facility. Since this distance is less than that to the point of maximum ground concentration, the maximum dose at the nearest site boundary will be somewhat smaller. However, at points on the site boundary located 390 feet from the facility (i. e., at the point of maximum ground concentration), the doses would be the same as in Section b. (1) above, except that the dose from ground deposition would be increased by a factor of 5.

(3) <u>Doses at the Nearest Community</u>

All doses at the nearest community (3 miles) from this accident will be negligible.

(4) <u>Direct Dose from Cell</u>

A person standing outside the cell wall during this accident would receive a direct dose from gammas and fast neutrons of 13.4 rem. The direct dose at the nearest facility, which is 80 feet away, would be about 0.4 rem.

4. SNAP 8 Experimental Reactor (Building 010)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The maximum credible accident postulates a power transient corresponding to an energy release of 76 Mw-sec. The accident was assumed to result from an unchecked ramp reactivity insertion of 2¢/sec, due to an uncontrolled inward rotation of the control drums, with simultaneous failures of all automatic scram channels and failure of the reactor operator to take corrective action. It was further assumed that the accident occurred immediately following completion of 10,000 hours of continuous reactor operation at 600 kw.

Under these conditions, ultimate reactor shutdown will occur about 74 sec after the initiation of the transient due to hydrogen loss from the core. The peak core temperature reached during the transient was calculated to be 2050°F.

The release of hydrogen from the fuel elements, plus the rise in the coolant temperature, will cause the rupture disc located in the reactor coolant inlet line to rupture, releasing NaK into the reactor containment vessel. It has been calculated that the combined effect of the released hydrogen and NaK will raise the temperature of the reactor vessel atmosphere to 1790°F and will create an overpressure reaching a peak of 167 psig. This overpressure will cause a portion of the reactor vessel atmosphere, which is contaminated with fission products, to leak into the high-bay area.

(2) Core Fission Product Inventories

Since the major hazard in this case results from fission products generated as a result of the extended period of steady-state reactor power operation, the activity produced in the transient was neglected. The fission product activity resulting from 10,000 hours operation at 600 kw was calculated to be 5.35×10^6 curies immediately after reactor shutdown.

(3) Mode and Degree of Fission Product Release from Core

The maximum fuel temperature in the core was calculated to reach about 2050°F, which is well below the fuel melting temperature of 3350°F.

*A. R. Piccot, ed., "SNAP 8 Experimental Reactor (S8ER) Final Safeguards Summary Report," NAA-SR-6958, February 28, 1962 (Classified).

Nevertheless, because of the assumptions outlined earlier in this section, a fission product release, comparable to a core meltdown, was assumed. The core release was taken as 100% of the noble gases, 26% of the halogens, 10% of the cesiums, 1% of the strontiums, and 0.5% of the bariums. The activity released from the core, based on these percentages, was calculated to be about 4.15×10^5 curies.

(4) Release from Building

The calculated average leakage from the containment vessel into the high-bay area for this accident was found to correspond to about 1% of the free containment vessel volume per 24 hours. Radioactivity released from the containment vessel was assumed to be carried away from the reactor room through the ventilation system and then released to the atmosphere through a 50-foot high stack. The ventilation system was assumed to operate without interruption both during and after the accident, releasing to the atmosphere an equivalent of 4.0×10^3 curies of gross gamma activity, uncorrected for decay. During the first 2 hours of the release, 330 curies are released.

b. Consequences of Maximum Credible Accident

(1) Doses at the Point of Maximum Concentration

Although the stack discharge of radioactive effluent will continue for 24 hours, the thyroid dose and the whole body gamma dose are based on a 2-hour exposure; the dose from ground contamination is based on iodine deposition for 24 hours.

The point where the ground concentration of the diffusing radioactivity reaches a maximum was calculated to be at 670 feet from the facility. The doses at that distance on-site were calculated to be 1.8 rem to the thyroid, 4.2×10^{-3} rem whole body gamma from cloud immersion, and 0.18 rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 300 feet to the northwest of the facility. Since this distance is less than the distance to the point where the concentration of radioactivity reaches a maximum, the related doses will be accordingly smaller. However, the dose at points on the site boundary which are 670 feet from the facility would be the same as in Section b.(1) above, except that the dose from ground deposition would be increased by a factor of 5.

(3) Doses at the Nearest Community

All doses at the nearest community are based on a 24-hour passage of the radioactive cloud. The doses at the nearest community were calculated to be 0.4 rem to the thyroid, 5.0×10^{-4} rem whole body gamma from cloud immersion, and 2.2×10^{-2} rem from ground deposition.

5. SNAP 8 Ground Prototype Test Facility (Building 059)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The postulated maximum credible accident is a power transient occurring immediately after the completion of a 10,000-hour run at 600 kw. The transient is assumed to be initiated by a reactivity ramp insertion of 2¢/sec, resulting from an unchecked inward rotation of the reactor control drums. It is also assumed that, simultaneously with the failure of the drum drive mechanism controls, all reactor scram channels fail to respond and the reactor operator does not take any corrective action. Under these conditions, ultimate reactor shutdown will occur 200 seconds after the initiation of the transient, through the combined effect of the negative temperature coefficient and the loss of hydrogen from the core. The peak power level reached will be 1.3 Mw and the total energy release will correspond to 163 Mw-sec.

(2) Core Fission Product Inventories

For the purpose of this report, the fission products on which dose calculations are based are those which were produced in the 10,000 hours operation at 600 kw. The radiological hazard from the inventory produced in the transient is negligible in comparison. The core inventory was found to contain a gross gamma activity of 5.35×10^6 curies.

(3) Mode and Degree of Fission Product Release from Core

The fission product release is based on the model assumed for zirconium hydride-uranium cores which are subjected to long operation (exposure) at the design power level. Thus, the assumed release is 100% of the noble gases, 26% of the halogens, 10% of the cesiums, 1% of the strontiums, and 0.5% of the bariums. These percentages correspond to a core release of 4.15×10^5 curies.

(4) Release from Building

The reactor will be operated in a vault which has a specified design leakage not exceeding 10% of the vault volume per day at 3 inches of water

*A. R. Piccot, ed., "The SNAP 8 Development System (S8DS) Test Facility Preliminary Safeguards Study," NAA-SR-6181, September 1, 1961 (Classified).

pressure differential. The postulated accident was estimated to create an average overpressure of 1.2 psig in the vault for 10 minutes. Due to this overpressure, 0.11% of the vault atmosphere will leak out of the vault into the high bay area. Assuming uniform mixing in the vault atmosphere of the fission products released from the core, 458 curies of fission products will be released into the high bay area.

For the purpose of this study it was assumed that the release from the vault into the high bay area is instantaneous, while that from the high bay area to the outside will depend on the ventilation rate. The ventilation system, which handles 10,000 cfm, was assumed to be functioning during and after the accident. The free volume of the high bay area is 45,000 ft^3. Therefore, the release time from the building, based on three air changes (as discussed earlier in this section), was assumed to be equivalent to 13 minutes. The high efficiency filters in the ventilation system were assumed to remove 99.95% of the nonvolatile activity prior to release to the atmosphere; thus the total release comprises 403 curies of gross gamma activity, uncorrected for decay. The building effluent is discharged from a 75-foot-high stack.

b. Consequences of Maximum Credible Accident

(1) Doses at the Point of Maximum Concentration

The point of maximum downwind ground concentration of the diffusing fission products was calculated to occur at a distance of 1110 feet from the facility. Doses at that distance on-site are 1.3 rem to the thyroid, 4.8×10^{-3} rem whole body gamma from cloud immersion, and 1.1×10^{-2} rem from ground deposition.

(2) Dose at the Site Boundary

The nearest site boundary is located 250 feet to the northwest. Since the nearest site boundary is closer to Building 059 than the point of maximum ground concentration, all doses at the nearest site boundary will be accordingly smaller.

However, doses at points on the site boundary located 1110 feet from the facility would be the same as in Section b.(1), except that the dose from ground deposition would be increased by a factor of 5.

(3) <u>Doses at the Nearest Community</u>

The calculated doses at the nearest community are 4.8×10^{-2} rem to the thyroid, 5.7×10^{-5} rem whole body gamma from cloud immersion, and 2.3×10^{-3} rem from ground deposition.

6. SNAP 8 Flight System Test Facility (Building 056)

Since the hazards and safeguards summary for Building 056 will not have been issued prior to the issuance of this report, the maximum credible accident will be discussed here in more detail than was the case with the other facilities.

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The maximum credible accident postulated for Building 056 is one in which a SNAP 8 reactor undergoes a power transient occurring immediately after the completion of a 10,000-hour test run at 600 kw.

Events leading to the accident assume that, owing to a failure of the controls on the safety drum drive mechanism, an unchecked ramp reactivity insertion is initiated. At the same time, all automatic scram channels with which the reactor is provided fail to respond and, in addition, the reactor operator fails to take any corrective action. Under these conditions the reactor power will rise until ultimate shutdown occurs through the combined effects of the negative fuel temperature coefficient and hydrogen loss from the core.

The estimated amount of released hydrogen and its temperature are 1.2 pounds and 1900°F, respectively. The bursting of the fuel cladding and the subsequent hydrogen release are assumed to generate a pressure surge sufficient to breach the NaK coolant system, spilling 200 pounds of the NaK, which is at a temperature of approximately 1700°F, onto the vault floor. About 15 pounds of the NaK is assumed to flash upon release and condense in the vault atmosphere.

The mercury vapor in the power conversion unit is also assumed to overheat and breaches its containment, releasing 100 pounds of vapor at 300 psi and 1200°F into the vault atmosphere. The released mercury vapor is assumed to condense immediately, imparting all its heat to the vault atmosphere.

(2) Core Fission Product Inventories

Since fission products generated during the long-term power run would have considerably more biological significance in this case than those

*R. S. Lubomirski, ed., "SNAP 8 Flight System (S8FS) Test Facility Safeguards Report," NAA-SR-MEMO-7359, to be published as classified document.

produced in the transient, only the radioactivity resulting from the power run will be analyzed. The gross gamma activity of the core due to reactor operation for 10,000 hours at 600 kw is calculated to be 5.35×10^6 curies.

(3) Mode and Degree of Fission Product Release from Core

In accordance with the postulated fission product release model for zirconium hydride-uranium fuel with significant burnup, the released activities for this accident are assumed to comprise 100% of the noble gases, 26% of the halogens, 10% of the cesiums, 1% of the strontiums, and 0.5% of the bariums. Based on these percentages, the gross gamma activity of the released fission products is 4.15×10^5 curies.

(4) Release of Fission Products from Vault

The reactor and the associated flight system will be tested in a low-leak-rate vault, with varying degrees of vacuum being applied during the tests. Calculations indicate that the maximum credible accident will not increase the vault pressure above atmospheric, unless the vault pressure at the time of the accident exceeds 0.94 atmosphere. Although the vault design pressure specification is 10^{-2} atmosphere, it is not anticipated that the operating pressure of the vault will exceed 0.6 atmosphere.

If the assumption is made, however, that, just before completion of the 10,000-hour reactor power run, the vault pressure had been raised to atmospheric (through instrument failures and operator inattention), then the heat from the transient would force some of the vault atmosphere into the high bay. The maximum credible accident was further developed considering this to have occurred. It is further assumed that all heat in the condensing mercury and NaK vapors and in the released hydrogen is transferred to the vault atmosphere. Although the vault liner cooling system is assumed to be in operation during this accident, the heat release rate will be sufficiently rapid to heat the vault atmosphere under essentially adiabatic conditions. Based on these assumptions, a peak overpressure of 0.88 psig will be reached in the vault 25 seconds after the rupture of the NaK and mercury systems. Cooling of the vault atmosphere would follow slowly, due, for the most part, to convective heat transfer to the vault liner. The vault pressure would return to atmospheric about 1 hour later. The average overpressure for this accident was calculated to correspond to 0.33 psig over the 1-hour period. For the purpose of this calculation, the vault leak rate was assumed to be 10% of the vault volume in 24 hours at a pressure differential of 0.6 psig.

The tentative design leak rate for the vault is 1% of the vault volume in 24 hours at a pressure differential of 4 psig. However, since the actual leak rate to be determined in the facility acceptance tests may differ from the design value, it is believed that the assumed value is sufficiently pessimistic to cover the worst case. Under these conditions, then, 0.17% of the vault atmosphere will be released to the high bay as a result of the assumed accident. If it is assumed that the core-released fission products are dispersed uniformly and instantaneously in the vault atmosphere, 723 curies of the gross gamma activity, uncorrected for decay, will leak into the high bay over a period of 1 hour.

(5) Release from Building

The ventilation system of the high bay is assumed to be functioning both during and after the accident. Therefore, the fission products released to the high bay will be carried with the building exhaust which must pass through roughing and high efficiency filters before being discharged to the atmosphere through a 100-foot-high stack. The high efficiency filters are assumed to remove 99.95% of all the nonvolatile fission products. The duration of the stack discharge was conservatively assumed to be 1 hour. The gross gamma activity released from the building, uncorrected for decay, was calculated to be equivalent to 693 curies.

b. Consequences of Maximum Credible Accident

(1) Doses at the Point of Maximum Concentration

The point of maximum ground concentration of the radioactive stack effluent occurs at a distance of 1550 feet downwind from the facility. The calculated doses at that distance on-site are 1.4 rem to the thyroid, 4.4×10^{-3} rem whole body gamma from cloud immersion, and 1.4×10^{-2} rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is about 325 feet to the northwest. Since this boundary is much closer to the facility than the point of maximum ground concentration, doses will be accordingly smaller. However, the doses at points on the site boundary which are 1550 feet from the facility will be the same as in Section b.(1) above, except that the dose from ground deposition would be increased by a factor of 5.

(3) Doses at the Nearest Community

Calculated doses at the nearest community are 0.1 rem to the thyroid, 2.7×10^{-4} rem whole body gamma from cloud immersion, and 5.0×10^{-3} rem from ground deposition.

7. SNAP Environmental Test Facility (Building 024) Containing the S2DS

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The maximum credible accident for the SETF when it contains the SNAP 2 Developmental Reactor is a ramp startup accident, based on the following sequence of highly improbable events:

a) A mechanism is postulated by which reactivity is inserted into the core at startup in an uncontrolled manner at a maximum rate of 2¢/sec.

b) The operator fails to take any corrective action despite visual and audible warnings.

c) All automatic scram channels fail to respond; i.e., one period, three power, the high fuel-temperature, and the high coolant-temperature scrams.

Ultimate reactor shutdown will occur about 80 seconds after initiation of the transient by loss of hydrogen from the core. Under these conditions, the peak power reached would be 1.3 Mw, the total energy release being 42 Mw-sec. The peak fuel temperature reached would be 1900°F. The heat generated in the transient is estimated to release about 17 pounds of NaK vapor. The released hydrogen and the NaK vapors will raise the pressure in the cell to 0.58 psig above normal. It was further assumed that, prior to the power transient, the reactor had been operated for 1 year at a power level of 50 kw.

(2) Core Fission Product Inventory

The fission product inventory in the core immediately after the accident corresponds to 4.5×10^5 curies of gross gamma activity from the 1 year of reactor operation at a power level of 50 kw. Fission products generated in the transient are neglected in this calculation, since their hazard is negligible in comparison to that from the inventory from steady-state operation.

*H. N. Rosenberg, ed., "Summary Safeguards Report for SNAP 2 Developmental System (S-2-DS)," NAA-SR-5483, November 23, 1960.
See also, G. H. Anno, ed., "SNAP II Environmental Test Facility Hazards Report," NAA-SR-3513, May 1, 1959 (Classified).

(3) Mode and Degree of Fission Product Release from Core

The release of fission products from the core is assumed to be comparable to a release from an entire core meltdown and corresponds to 100% of the noble gases, 26% of the halogens, 10% of the cesium, 1% of the strontiums, and 0.5% of the bariums.

(4) Release from Building

The fission product release from the cell into the high bay area is proportional to the pressure differential. Since the peak overpressure reached is 0.58 psig, the S2DR accident will result in the release of 0.27% of the vault atmosphere to the high-bay area.

The ventilation system is assumed to be functioning both during and after the accident. An instantaneous release from the cell into the high bay is assumed. The atmospheric release, based on three air changes, was assumed to be completed in about 20 minutes. Due to the presence of the high efficiency filters in the ventilation system, only volatiles are assumed to be released from the 85-foot stack. The atmospheric release of fission products, uncorrected for decay, is equivalent to 90 curies of gross gamma activity.

b. Consequences of Maximum Credible Accident

(1) Doses at the Point of Maximum Concentration

The point of maximum ground concentration occurs at a distance of 1260 feet from the building. The doses at that point on-site are 0.16 rem to the thyroid, 5.4×10^{-4} rem whole body gamma from cloud immersion, and 1.6×10^{-3} rem from ground deposition.

(2) Doses at the Site Boundary

The distance from Building 024 to the nearest site boundary is about 500 feet. Therefore, although the doses at the nearest site boundary will be less than those at the point of maximum ground concentration, the doses at points on the site boundary which are 1260 feet from the facility will be the same as in Section b. (1) above, except that doses from ground deposition will be increased by a factor of 5.

(3) Doses at the Nearest Community

Doses at the nearest community will be 1.1×10^{-2} rem to the thyroid, 2.7×10^{-5} rem whole body from cloud immersion, and 4.9×10^{-4} rem from ground deposition.

(4) Interaction Within the Building

The S2DR reactor and the SCA-4B critical assembly are both located in Building 024 in separate test cells. However, the cell penetrations end in a common working area.

On the basis of the core fission product release and the cell leak rate postulated for the S2DR maximum credible accident, the doses calculated for a 10-minute exposure in the common working area are 60 rem to the thyroid and 0.07 rem whole body due to immersion in the contaminated atmosphere.

One additional aspect of this problem was also investigated — the extent of residual contamination of the building by iodine deposition in the high-bay area. Considering that all of the iodine activity which passes through the high bay deposits there instead of being discharged from the stack, the total dose which would be accumulated if the activity remained undisturbed would amount to less than 1 rem over the life of the activity, taking into account the occupancy factor. This points out that the radiation level would not be so great as to impair immediate steps at facility decontamination. Therefore, the interruption of other programs in Building 024, due to the S2DR maximum credible accident, will not be excessively long.

8. SNAP Environmental Test Facility (Building 024) Containing the SCA-4B Critical Assembly

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The SCA-4B critical assembly is enclosed in a sealed core vessel, which is in turn surrounded by a removable water reflector. The entire system is located in one of the test cells of Building 024. The maximum credible accident assumes that, at a time when the reflector is positioned for maximum reactivity, a sudden and rapid flooding of the core vessel occurs, introducing a ramp insertion of reactivity of $1.0/sec. A further assumption is made that all scram channels fail simultaneously, i.e., 3 period and 3 power channels, and that the reactor operator fails to take corrective action. Under these conditions, ultimate reactor shutdown will occur through loss of hydrogen from the core. The energy release attributed to this accident is 50 Mw-sec.

(2) Core Fission Product Inventories

The critical assembly will be operated only intermittently at relatively low power levels. For this reason, the core fission product inventories used for dose calculations will be based on that generated in the 50-Mw-sec excursion, the hazard from other fission products being negligible in comparison. The gamma activity in the core was calculated to be 0.9×10^8 curies at 1 second after termination of the transient.

(3) Mode and Degree of Fission Product Release from Core

It has been estimated that about 17.5% of the total hydrogen in the core would be released in this accident. Since it has been assumed that, for the case of fuel cladding rupture without fuel melting, the fission product release is related to the hydrogen release by the inverse square root of the molecular weight of the diffusing species, the volatile fission product activity released in this accident was calculated to be 1.8% of the volatile core inventory.

*L. Moss, ed., "The SNAP Critical Assembly-4B (SCA-4B) Water Immersion Summary Hazards Report and Operations Manual," NAA-SR-MEMO-6877, November 20, 1961.
See also H. N. Rosenberg, ed., "Summary Safeguards Report for SNAP 2 Development Systems (S2DS)," NAA-SR-5483, November 23, 1960 (Classified).

(4) Release from Building

The critical assembly will be located and operated in one of the test cells of Building 024. The design leakage of the cell for this experiment is 2.0% per hour, with a differential pressure of 3 inches of H_2O.

If it is assumed pessimistically that the released hydrogen will burn, a temperature increase of 150°F and a corresponding overpressure of 4 psig will be attained in the cell. Because the cell design pressure is 4 psig, a breach of the cell integrity is not expected, but an estimated 27% of the initially released fission products will be forced into the high bay.

It has been assumed conservatively, for the purpose of this study, that the cell releases the fission products instantaneously into the high-bay area. Because the accident in the cell will not affect the ventilation system in the high bay, the system will continue to operate without interruption. Thus, the release from the high bay to the atmosphere was based on the time for three air changes to take place, or about 20 minutes, at which time essentially all of the activity is considered to have been released. The radioactive effluent discharge occurs through an 85-foot-high stack at a rate of 12,000 cfm. The atmospheric release of fission products, uncorrected for decay, corresponds to 8.7×10^4 curies of gross gamma activity.

b. Consequences of Maximum Credible Accident

(1) Doses at the Point of Maximum Concentration

The distance to the point of maximum ground concentration for the release from the 85-foot stack was found to be 1260 feet. At that distance on-site, the calculated doses are 7.8×10^{-3} rem to the thyroid, 1.4×10^{-4} rem whole body gamma from cloud immersion, and 6.6×10^{-5} rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located at a distance of about 500 feet to the northwest. Since the site boundary at that point lies closer to the facility than the point of maximum ground concentration, all doses will be accordingly smaller. However, the maximum doses at points on the site boundary located 1260 feet from the facility will be the same as in Section b.(1), except that the dose from ground deposition would increase by a factor of 5.

(3) Doses at the Nearest Community

Doses at the nearest community were calculated to be quite small: 5.0×10^{-4} rem to the thyroid, 3.4×10^{-6} rem whole body gamma from cloud immersion, and 2.4×10^{-5} from ground deposition.

(4) Interaction Within the Building

This aspect of the maximum credible accident in Building 024 was discussed earlier for the S2DR accident. However, since the on-site and off-site doses from the postulated accident from the SCA-4B critical assembly are generally lower than those from the S2DR accident, the doses in the high bay will follow the same pattern. It is anticipated that, as a result of the postulated maximum credible accident for SCA-4B critical assembly, Building 024 will be evacuated, but, as in the case of S2DR accident, the interruption of the experimental program should be of short duration.

9. SNAP Flight Systems Nuclear Qualifications Test Facility (Building 019)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The postulated maximum credible accident is a power transient occurring during a wet critical test. The accident assumes a ramp reactivity insertion of 2¢/sec, resulting from an unchecked inward rotation of one control drum. All scram channels are assumed to fail simultaneously and, in addition, it is assumed that the reactor operator fails to take any corrective action. Under these conditions, the total energy release in the transient will be equivalent to 10 Mw-sec. The reactor reaches a peak power at 1.3 Mw and will be permanently shut down 60 seconds after initiation of the transient, through a combination of the negative fuel temperature coefficient and hydrogen loss from the core. Assuming, conservatively, an initial chamber pressure of 10^{-3} mm Hg and a temperature of 1000°F for the coolant, the resulting coolant temperature will approach 2900°F, resulting in a peak chamber pressure of 0.78 atmosphere. As a result of this accident about 50% of the hydrogen in the zirconium hydride matrix will dissociate and will be liberated.

(2) Core Fission Product Inventories

The activity in the core resulting from a transient of 10 Mw-sec was calculated to be 1.8×10^7 curies. Because the reactor will be operated only intermittently at power levels not exceeding approximately 10 watts, the hazard from the fission products generated in these low power operations is significantly less than that from those generated in the transient. Therefore, fission products produced by low power operation were neglected.

(3) Mode and Degree of Fission Product Release from Core

The original hazards and safeguards report for a SNAP 2 reactor in Building 024 assumed a core release of 50% of the volatiles and 5% of all other fission products. This core release model was modified for the purpose of this report, to conform with that postulated here for ruptured elements in critical assemblies where no fuel melting occurs (see Section VI-B). Using

*Compact Systems Division, "Preliminary Safeguards Report - Acceptance Test Building (019)," NAA-SR-6733, November 15, 1961.

the diffusion model postulated in this study, the release of fission products corresponding to a core loss of 50% of hydrogen, would be 5% of the volatiles, equivalent to 1.8×10^5 curies of gross gamma activity 1 second after the accident.

(4) Release from Building

The vacuum chamber enclosing the reactor has a volume of about 1000 cubic feet. The vacuum pumping rate is estimated to be 130 cubic feet per minute. As soon as the vent monitor senses the increase in radioactivity due to the released fission products, it will activate circuits to shut the quick-closing valves located on the exhaust line. Since a 1-second lag is expected for these valves, about 2 cubic feet of the 1000-cubic-foot chamber volume will escape prior to valve closing. Assuming uniform distribution of fission products in the vacuum chamber atmosphere, 360 curies of activity would be released.

This activity is assumed to be released in a 2-cubic-foot puff at ground level.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 059, located about 60 feet to the southwest. The calculated doses at that facility are 2.9 rem to the thyroid, 2.7×10^{-2} rem whole body gamma from the radioactive cloud volume, and 1.6×10^{-2} rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 250 feet to the northwest. The calculated doses are 0.39 rem to the thyroid, 1.0×10^{-3} rem whole-body gamma from the radioactive cloud voluem, and 0.13 rem from ground deposition.

(3) Doses at the Nearest Community

The additional reduction in the doses due to cloud transport to the nearest community was considered to be so great that the resulting doses would be insignificant and hence they were not calculated.

10. Shield Test Experiment (Building 028)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The STE reactor is a pool-type reactor which operates at a power level of 50 kw. The top of the reactor vessel is covered with a 1/2-inch-thick Lucite cover. The maximum credible accident assumes that, at a time when the reactor is already critical and the vessel cover has been removed, a fuel element is dropped into the reactor pool water. In such case a fast ramp insertion of reactivity would occur. For the purpose of accident evaluation, it was assumed that this fast ramp insertion can be approximated by a step reactivity insertion. It was further assumed that, simultaneously with this reactivity insertion, a failure of all reactor scram channels occurs, so that ultimate reactor shutdown will occur through void formation and ejection of water from the core. It was also assumed that the operator failed to take any corrective action.

For the most reactive core condition, the accidental introduction of one fuel element into a critical core was calculated to be equivalent to the addition of 86¢. The energy release was found to be about 1 Mw-sec, and the average peak temperature of the fuel was calculated to reach 990°F, well below the melting point of the zirconium hydride-uranium fuel alloy (3350°F).

(2) Core Fission Product Inventories

The core fission product inventory was assumed to correspond to that resultant from long-term steady-state operation at a power level of 50 kw and was calculated to be 4.5×10^5 curies. Inventories from the 1-Mw-sec transient were neglected, their hazard being insignificant in comparison to that from the power-produced fission product inventory.

(3) Mode and Degree of Fission Product Release from Core

Since the burnout heat flux is not reached in this accident, the probability of releasing nonvolatile fission products is quite remote. Nevertheless, it has been conservatively assumed, for the purpose of this study,

*R. L. Tomlinson, ed., "SNAP Shield Test Experiment Final Hazards Summary," NAA-SR-5896, March 17, 1961.

that, by virtue of cladding failure, the fission products from the equivalent of one fuel element will be released from the core. Since the fuel material is zirconium hydride-uranium alloy, the fission product release upon melting, as discussed earlier in this section, will consist of 100% of the noble gases, 26% of the halogens, 10% of the cesiums, 1% of the strontiums, and 0.5% of the bariums. Since it is assumed that there are 45 fuel elements in the core, the approximate activity of one fuel element is $4.5 \times 10^5/45 = 1 \times 10^4$ curies.

(4) Release from Building

The fission products released from the core must travel through 16 feet of water before reaching the reactor room atmosphere. Therefore, it was assumed that all nonvolatile fission products will be retained in the coolant. In addition, it was also conservatively assumed that 50% of the halogens will dissolve and not escape from the reactor vessel. However, it was assumed that the noble gases did not dissolve to any significant degree and were released instantaneously to the reactor room atmosphere.

The reactor room is ventilated at a rate of 10 air changes per hour. Therefore, it was assumed that all activity in the reactor room will be exhausted through the ventilation system to the atmosphere in about 20 minutes. The atmospheric diffusion of the released fission products is assumed to proceed at ground level. The gross gamma activity released to the atmosphere, uncorrected for decay, corresponds to 742 curies.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 024, located about 150 feet to the southeast. The calculated doses at that distance are 210 rem to the thyroid, 1.72 rem whole-body gamma from cloud immersion, and 1.2 rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 250 feet to the northwest. The doses calculated at that distance are 200 rem to the thyroid, 1.56 rem whole-body gamma from cloud immersion, and 5.7 rem from ground deposition.

(3) Doses at the Nearest Community

The doses at the nearest community were calculated to be 1.8×10^{-2} rem to the thyroid, 5.6×10^{-5} rem whole-body gamma from cloud immersion, and 8.4×10^{-4} rem from ground deposition.

11. Kinetic Experiment Water Boiler (Building 073)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The maximum credible accident was postulated to occur through an error in predicting the amount of control rod withdrawal necessary to result in the reactivity input desired for an experiment. As a result of this error, a transient was assumed to occur involving all the reactivity worth of the poison control rod (5.3% $\Delta k/k$). If the transient were controlled only by the temperature and void mechanism, and the reflector effects were ignored, a highly conservative dynamic energy coefficient of reactivity would result. The total energy generated under these conditions would be 12 Mw-sec. For the purpose of calculating the consequences of this maximum credible accident, it has been conservatively assumed that the transient energy release would be 20 Mw-sec, providing contingency for transient assymetry, etc. It was further postulated that the pressure generated within the core as a result of the transient, combined with that from a hydrogen-oxygen reaction, would rupture both the core vessel and secondary enclosure, releasing fission products into the test building.

(2) Core Fission Product Inventories

The fission product inventory in the core has been estimated by considering that the operational history of the KEWB experimental program can be approximated by assuming continuous operation at 50 watts for 3 years. However, a steady-state long-term operation at 100 watts was conservatively assigned for the maximum credible accident. The activity resulting from this period of continuous operation was calculated to be 910 curies. The amount of fission product activity generated in the transient was calculated to be 3.6×10^7 curies.

*R. S. Hart et al., "Change of KEWB Reactor Cores - Evaluation of Significance with Regard to Associated Hazards," NAA-SR-Memo-4928, February 4, 1960.

(3) Mode and Degree of Fission Product Release from Core

An instantaneous release to the reactor building of 100% of the noble gases and 50% of the halogens, equivalent to 6×10^6 curies, was assumed. The large halogen release was based on the release which would be associated with a violent dispersion of the core solution, since the resultant hydrogen-oxygen reaction was postulated to destroy the containment. Therefore, it has been conservatively assumed that one-half of the halogen content of the core would be released to the test room, with 50% of that amount depositing on colder surfaces within the room.

(4) Fission Product Release to the Atmosphere

The design of the reactor building is such that it will effectively confine the release. The hazards report assumes, however, that at the time of the accident, if the barometer were dropping steadily at a maximum rate observed at the Santa Susana site, a pressure differential of 0.41 inch would be created. Assuming a uniform dispersion of the volatile fission products within the building, this would result in the release of 2.22 cubic feet of the building atmosphere in 10 minutes, constituting 0.08% of the total building volume. Since this leakage is postulated to occur through various cracks in the building structure, a ground level release is assumed. The gross gamma activity released to the atmosphere, uncorrected for decay, was calculated to be equivalent to 5.7×10^4 curies.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 037, the SNAP Office Building No. 2, located about 150 feet to the southwest. Doses at that distance were calculated to be 2.0 rem to the thyroid, 1.1 rem whole-body from the radioactive cloud volume, and 5.6×10^{-3} rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 625 feet to the northwest. The doses at the site boundary were calculated to be 0.27 rem to the thyroid, 4.2×10^{-3} rem whole-body gamma from cloud immersion, and 4.0×10^{-3} rem from ground deposition.

c. Doses at the Nearest Population Center

All doses at a distance of 3 miles were assumed to be negligible, and therefore were not calculated.

12. AE-6 Reactor (Building 093)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

It has been assumed that the maximum credible accident for the AE-6 reactor is a ramp insertion of reactivity equivalent to 1.2% $\Delta k/k$. This accident can be initiated only as a result of gross negligence by operating personnel and a deliberate disregard of operating procedures. A reactivity insertion of this magnitude would occur as a result of an uncontrolled withdrawal of the coarse control rod or by malfunction of the automatic rod drive control system with a simultaneous failure of all the automatic scram circuitry. A similar ramp insertion of reactivity could be introduced if, due to a failure of the rod interlock system and prevalence of the conditions stated above, all the rods were withdrawn simultaneously.

A reactivity insertion of 1.2% $\Delta k/k$ would result in a 1-Mw-sec excursion. If the recombiner has not recombined any of the radiolytic gases generated in the excursion, the uncombined gas, upon ignition, could generate an overpressure of 600 psig, which is below the yield point of the reactor vessel by a factor of 2. The maximum credible accident assumes, however, that the vessel is breached, thus releasing the fission product inventory into the reactor room.

(2) Core Fission Product Inventories

In estimating the fission product inventory, the experimental program carried out with the AE-6 reactor was conservatively approximated by assuming continuous operation at 120 watts for one year, followed by 6 hours of operation at 2 kw. The 1-Mw-sec transient was assumed to occur immediately after cessation of 2-kw operation. The corresponding fission product inventories were calculated to consist of 1200 curies due to steady-state operation, and 1.8×10^6 curies from the excursion.

(3) Mode and Degree of Fission Product Release from Core

An instantaneous core release of 100% of xenon and krypton and 50% of iodine and bromine was assumed. This is a conservative assumption,

*G. L. Blackshaw and C. H. Skeen, "Safeguards Summary for the AE-6 Reactor," NAA-SR-Memo-5304, July 7, 1960.

since experiments carried out with the KEWB reactor have demonstrated that a preponderant amount of halogens will remain in solution, with very little being released to the atmosphere.* Since, however, the maximum credible accident postulates that the core vessel integrity will be destroyed as a result of a hydrogen-oxygen reaction, complete dispersion of the solution is assumed in this study.

It was, therefore, conservatively assumed that the dispersed droplets will release 50% of the core inventory of iodine and bromine to the reactor room atmosphere, with a subsequent deposition of one-half of the released material within the reactor building.

(4) Fission Product Release to the Atmosphere

A ground level release through open building windows was assumed for the AE-6 building. The hazards report assumed that the leakage of contaminated air from the building to the outside is based on wind-induced ventilation of the building. Therefore, the ventilation rate was taken as the wind velocity blowing through the open building windows. Based on the three air change criteria discussed in Appendix A, and considering a building volume of 985 m^3, a wind velocity of 0.5 m/sec, and assuming that one half of the open window area is 4.55 m^2, the building release time was assumed to be 20 minutes. The gross gamma activity released to the atmosphere, uncorrected for decay, was calculated to be equivalent to 2.8×10^5 curies.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 083, the KEWB control room, located 50 feet to the northwest of the AE-6 reactor.

Doses at that point for a ground level release were calculated to be 25 rem to the thyroid, 1.0 rem whole-body gamma from cloud immersion, and 0.17 rem from ground deposition.

(3) Doses at the Nearest Community

The doses at the nearest community were calculated to be 1.4×10^{-3} rem to the thyroid, 2.4×10^{-6} rem whole-body gamma from cloud immersion, and 6.6×10^{-5} rem from ground deposition.

* R. S. Hart et al., "Change of KEWB Reactor Cores - Evaluation of Significance with Regard to Associated Hazards," NAA-SR-Memo-4928, February 4, 1960.

13. Organic Moderated Reactor Critical Facility (Building 009)

a. Maximum Credible Accident*

The maximum credible accident postulated in the original hazards analysis for this facility is no longer applicable. This accident assumed an instantaneous flooding of a control rod thimble, which had been calculated to result in an instantaneous reactivity insertion of $1\% \Delta\rho$. In the present OMR critical assembly, the thimbles have been removed, so that the control rods are immersed directly in the organic moderator. Referring back to thehazards report, the accident which yielded the next largest energy release was chosen, for the purpose of this study, to represent the maximum credible accident. This accident results from a continued withdrawal of either a shim or control rod, and is caused by either human error or failure of the control rod drive system, combined with a simultaneous failure of the period, power level, and full-scale reading scram channels. The combination of these events would result in a continuous ramp reactivity insertion of $2 \times 10^{-4} \Delta\rho/\text{sec}$. Ultimate reactor shutdown would occur by void formation in the organic moderator, with a corresponding energy release of 547 Mw-sec. The temperature of the central fuel element resulting from this transient would be 1395°F, well below the 2072°F melting temperature of the fuel. Although melting of fuel will not occur, it is conceivable that the cladding on a number of fuel elements may be damaged. Since it is not possible to predict either the extent of this damage nor of the associated fission product release with any degree of accuracy, it is felt that the associated fission product release would be relatively small.

However, for the purpose of this study, it was conservatively assumed that a fission product release equivalent to that from the meltdown of one fuel element would occur. Because the core vessel is not designed for extreme pressures, it is conceivable that, under the postulated conditions, it might fail, thus releasing the fission products into the reactor room.

(2) Core Fission Product Inventories

Under the normal reactor experimental program, operation at a peak power of 100 watts and at average powers of 1 to 10 watts is expected.

*G. B. Zwetzig, ed., "Organic Moderated Reactor Critical Experiment Hazards Summary," NAA-SR-3220, December 15, 1958.

However, the hazard from the fission product inventories resultant from this type of power operation is relatively small compared to that resultant from the transient, and hence only the fission products produced in the transient will be considered. Based on the 547 Mw-sec energy release in the transient, the core inventory is equivalent to 1×10^9 curies of gamma activity.

(3) Mode and Degree of Fission Product Release from Core

The OMR critical assembly core loading may vary from 27 to 63 fuel elements. If it is assumed that, at the time of the accident, the core contained only 27 fuel elements, only one of which will melt as a result of the power transient, then the released activity will correspond to 6.1×10^6 curies.

In the melting of this type of fuel element, it was assumed that 100% of the noble gases and 50% of the halogens will be released, with a subsequent deposition of one half of the released iodine in the reactor room. In view of the fact that the iodine combines readily with the organic moderator, a release of 50% of the iodines is highly conservative.

(4) Fission Product Release to the Atmosphere

The reactor room volume is 54,000 cubic feet and the normal ventilation rate during reactor operation is 3,500 cubic feet per minute. The ventilation system is assumed to continue operating during this accident. Based on 3 air changes, the assumed atmospheric discharge of radioactivity will continue for 45 minutes, with the atmospheric release occurring through the facility stack. However, since the stack height is only 10 feet above the building, and because of the possible down wash and eddy effects from such a short stack, atmospheric diffusion of the released radioactivity is postulated to occur at ground level. Assuming that one-half of the halogens will deposit within the reactor building, the gross gamma activity released to the atmosphere, uncorrected for decay, was calculated to be equivalent to 5.5×10^6 curies.

Since the volume of the building is relatively large, a 10-second delay time was allowed between the core release and the initial atmospheric release.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 100, the ECEL, located about 200 feet to the northeast. The calculated doses at that facility from the postulated accident are 30 rem to the thyroid, 4.1 rem whole-body gamma from cloud immersion, and 0.17 rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 650 feet to the northwest. The doses at this point are 26 rem to the thyroid, 2.0 rem whole-body gamma from cloud immersion, and 0.73 rem from ground deposition.

(3) Doses at the Nearest Community

The doses at the nearest community were calculated to be 2.0×10^{-2} rem to the thyroid, 1.4×10^{-4} rem whole-body gamma from cloud immersion, and 9.7×10^{-4} rem from ground deposition.

(4) Interaction Within the Facility

The SGR and OMR critical assemblies, which are both located in Building 009, have completely separate ventilation systems and hence the operation of one does not affect the other. The critical assemblies have separate control rooms which are shielded from the cells by 4 feet of ordinary concrete. The wall separating the two cells is also concrete, 4 feet thick. The prompt dose in the control room resulting from this accident would be about 11 rem. The residual dose would be negligible. Therefore, the postulated accident occurring in the OMR cell should not interfere with routine operations being carried out in the SGR portion of the building.

14. Sodium Graphite Reactor Critical Facility (Building 009)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

Should an excess amount of reactivity be inserted into the core due either to disregard of operating procedures or a failure of interlocks, a power excursion would result. If all of the reactor's four scram circuits failed to respond, ultimate shutdown of the reactor would occur through melting of the core.

The hazards report indicates that the core may be operated in either a 6.9 or 14.4-inch lattice with 25 and 14 fuel elements, respectively. The 6.9-inch lattice is characterized by a shorter neutron lifetime. The hazards report also states that a step insertion of reactivity above 3.1 dollars in the 6.9-inch lattice or 0.8 dollars in the 14.4-inch lattice, or an unchecked ramp insertion of 0.078 dollars per second in the 6.9-inch lattice (corresponding to a simultaneous withdrawal of all control rods) would raise the temperature of the fuel to its melting point. The approximate integrated power of the transients, when the peak fuel temperature reaches the melting point, is 126 and 70 Mw-sec for the 6.9 and 14.4-inch lattices, respectively.

The hazards report pessimistically assumes a 150 Mw-sec power transient for the maximum credible accident. This amount of energy would be necessary to completely melt one fuel element in the 6.9-inch lattice and would melt almost all the fuel in the 14.4-inch lattice. It was also estimated that, in the latter case, 50% of the fuel would oxidize.

(2) Core Fission Product Inventories

Because of the low power level (0.1 watt) and intermittent steady-state operation of the critical assembly, the hazard from the fission product accumulation prior to the postulated power excursion is negligible in comparison to that from the transient. Thus, only the fission products resulting from the excursion were considered. The fission product activity produced in the postulated accident was calculated to be 2.7×10^8 curies.

* D. E. Fletchall, ed., "Sodium Graphite Reactor Critical Experiment Hazards Summary," NAA-SR-3404, April, 1959.

(3) Mode and Degree of Fission Product Release from Core

It has been assumed that, because of oxidation of one-half of the core, 100% of the associated noble gases and halogens, and 1% of other fission products will be liberated, with one-half of the iodine depositing subsequently within the building. Since the oxidation of the fuel is not instantaneous, a conservative delay time of 10 seconds was assumed before the stack release.

(4) Fission Product Release to the Atmosphere

The volume of the reactor cell is 60,000 cubic feet. Therefore, it is extremely unlikely that the postulated accident, occurring in such a large volume, could impair the ventilation system. As a result, it has been assumed that the ventilation system will remain in operation both during and subsequent to the accident. The ventilation rate during reactor operation is 6,000 cfm, providing one air change every 10 minutes. Hence, in accordance with previous assumptions, the building release time was assumed to be 30 minutes.

The ventilation system is provided with high efficiency filters which will remove 99.95% of the particulates. However, the halogens are assumed to pass through the ventilation system without depositing either on the filters or in the ventilation ducts.

The point of actual discharge to the atmosphere is from a stack located 10 feet above the reactor building, which itself is 30 feet high. However, because of possible down wash and eddy effects associated with such a short stack, the conservative assumption was made that the atmospheric dissemination of radioactivity will take place as if the release occurred at ground level. The fission products released to the atmosphere correspond to 2.2×10^7 curies of gross gamma activity, uncorrected for decay.

Since the volume of the building is relatively large, a 10-second delay time was allowed between the core release and the initial atmospheric release.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 100, the ECEL, located about 200 feet to the northeast. The calculated doses at this location were 223 rem to the thyroid, 21.5 rem whole-body gamma from cloud immersion, and 1.3 rem from ground deposition.

(2) Doses at the Site Boundary

The nearest site boundary is located 650 feet to the northwest. Doses at the site boundary were calculated to be 194 rem to the thyroid, 10.6 rem whole-body gamma from cloud immersion, and 5.5 rem from ground deposition.

(3) Doses at the Nearest Community

The doses at the nearest community were calculated to be 0.15 rem to the thyroid, 1.1×10^{-3} rem whole-body gamma from cloud immersion, and 7.3×10^{-3} rem from ground deposition.

(4) Interaction Within the Facility

The SGR and OMR critical assemblies, which are both located in Building 009, have completely separate ventilation systems and hence the operation of one does not affect the other. The critical assemblies have separate control rooms which are shielded from the cells by 4 feet of ordinary concrete. The wall separating the two cells also contains 4 feet of ordinary concrete. The prompt dose in the control room, resulting from the postulated accident, would be 3 rem. The residual dose would be negligible. Therefore, the postulated accident occurring in the SGR cell should not interfere with routine operations being carried out in the OMR portion of the building.

15. Sodium Reactor Experiment (Building 143)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

The hazards report discusses two accidents, each of which starts with an extended, rapid, reactivity insertion at a time of low coolant flow (approximately 10 percent). In the first accident, this is postulated to occur simultaneously with leakage of the reactor helium cover gas atmosphere through a breach in the reactor top shield seal (although this latter is an unrelated event). The second accident postulates a subsequent release of primary sodium coolant followed by combustion. Results for either accident (helium atmosphere leak or sodium release and subsequent combustion) are not completely described in the hazards report. Accordingly, the radiological consequences of both accidents were re-evaluated for the purpose of this study. It should be emphasized that the accidents considered above, i.e., helium leaking through the top shield and the primary sodium release, are completely independent accidents (essentially, they are alternate maximum credible accidents).

A reactivity increase of 0.78 percent, inserted at a rate of 0.039%/sec, ensues if, through various system malfunctions or operator error, one control rod is switched to the "fast withdraw" condition and is so maintained for the maximum time (20 seconds) permitted by the electrical and mechanical interlocks. The accident requires failure of the period setback, and also rather pessimistically assumes that the high coolant outlet temperature scram will be delayed by various system properties for approximately 50 seconds from the time rod withdrawal begins. This delay is sufficient to permit the reactor power level to reach a peak of 70 percent of normal full power (which is 20 Mwt), the result being the total melting of one fuel element or the equivalent partial melting of several elements.

* A. I. Staff, "Hazards Summary for Thorium-Uranium Fuel in the Sodium Reactor Experiment," NAA-SR-3175 (Revised), July 1, 1959.
See also NAA-SR-3175 (Rev.) Supplement by D. H. Johnson, April 8, 1960.

This transient does not produce any measurable increase in radiation intensities around the plant, nor does it result in damage to any of the reactor system components (except for the damaged fuel elements). However, the fission products which are released from the melted fuel element(s) increase the quantity of radioactive material available for release in any coincident or subsequent abnormal condition. It is for this reason that the transient has been described and included as part of the description of the maximum credible accident.

(a) Helium Release Through Top Shield

In the case where the helium cover gas leaks through the top shield assembly, it was postulated* that an undetected 0.25-cfm leak rate (the maximum conceivable) existed. Since the transient does not result in damage to the containment vessel, the leak in the reactor top shield seal is assumed to have existed prior to the transient. However, it is worthwhile to note that, although operation of the reactor with a 0.25-cfm leak in the top shield seal is possible, it would require imposing restrictions upon entry into the high bay area, due to the magnitude of the airborne radioactivity concentrations which would exist at high power levels.

(b) Primary Sodium Release

In this accident it is postulated that, by some unspecified means, the primary coolant system piping is breached on the discharge side of the main primary sodium pump. Such a circumstance would result in the discharge of up to 3,000 gallons of liquid sodium (the maximum volume of sodium above the reactor discharge nozzle and attainable under very low flow, controlled conditions) onto the gallery floor. It is further assumed that the nitrogen atmosphere of the primary sodium gallery had been replaced with air without detection and that combustion ensued. It should be noted here that the gallery oxygen content is normally monitored continuously and maintained below a 3% concentration. Therefore, it must be assumed that the instrument was either inoperative or malfunctioning, and that, at least in the former case, the operator took no corrective action. Maximum doses would be produced if the sodium discharge

* A. I. Staff, NAA-SR-3175 (Revised), op. cit.

and combustion occurred when the reactor had operated sufficiently long that the specific activity of the sodium was at its saturation value.

(2) Radioactivity Inventories

Since fuel melting is involved, it is postulated that 100% of the noble gases, 50% of the halogens, and 1% of the remaining fission product activity are released from the melted fuel element. Analysis of experimental data obtained subsequent to the SRE fuel element damage indicates that the halogen and nonvolatile fission products would be entrained in the coolant, with only the noble gas fission products able to reach the helium atmosphere above the core,* where they then can leak into the reactor high bay area.

The Na^{24} activity in the helium atmosphere is expected to produce a concentration of approximately 9×10^{-7} curie/cm^3.† The xenon and krypton activity released into this volume would amount to 4.2×10^{-2} curie/cm^3.† It should be noted here that the airborne MPC values for Na^{24} and the noble gas isotopes are of similar magnitude,** although the Na^{24} presents a more significant gamma emitting source on a curie-for-curie basis. However, as Na^{24} represents less than 0.002% of the total activity available for release, its contribution to the downwind dose is insignificant by comparison, and therefore it will be ignored in the accident wherein the helium cover gas atmosphere leaks through the top shield.

The saturation, specific activity of the primary sodium coolant is approximately 0.3 curie/gram.† The specific activity of the halogen and nonvolatile fission products (released from the melted fuel element) entrained in the coolant is 0.015 curie/gram of sodium. It should be noted here that the hazards report† indicates this value to be 0.033 curie/gram of sodium; however, this included the activity from the assumed release considering subsequent entrainment in the coolant of 10% of the nonvolatile fission products. A

*R. S. Hart, "Distribution of Fission Product Contamination in the SRE," NAA-SR-6890, March 1, 1962.
†AI Staff, NAA-SR-3175 (Revised), op. cit.
**International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation, Pergamon Press, New York, 1959.

more realistic value of 1% of the nonvolatiles is used in this report, which results in the lower specific activity. As a result, for the case of the primary sodium release, only the Na^{24} activity was used in computing the downwind doses for the following reasons:

a) The initial Na^{24} activity per gram of coolant is twenty times that of the fission products.

b) The decay rate of the fission products is faster than that of Na^{24} for the first few hours, thus resulting in an increased Na^{24}-to-fission-product-activity ratio during that period.

c) The high energy photons emitted by Na^{24} (2.75 and 1.37 Mev), when compared to the average value generally used for mixed fission products (0.7 Mev), makes the Na^{24} more hazardous, curie-for-curie, when considering the external dose.

d) While the MPC for this mixture of beta, gamma emitting fission product isotopes in air is $1 \times 10^{-9} \mu c/cc$, almost all of the individual isotopes involved have airborne MPC values which are 10 to 100 times greater than $10^{-9} \mu c/cc$. The MPC for Na^{24} in air is $4 \times 10^{-7} \mu c/cc$, which compares favorably with the individual fission product values.*

(3) <u>Mode and Degree of Radioactive Material Release to High Bay</u>

(a) <u>Helium Release Through Top Shield</u>

All of the radioactivity contained within the helium cover gas atmosphere is assumed to be available for leakage into the high bay area through the 0.25-cfm leak in the reactor top shield seal. To define the duration of the leakage and consequent exposure, it is stipulated herein that the leakage can be stopped (or significantly reduced) two hours after starting. This could be accomplished, for example, by simply venting the helium atmosphere to the gas decay tanks. From the calculations which have already been performed,† it does not appear that the dose rates in the high-bay area will unduly restrict such actions.

*International Commission on Radiological Protection, op.cit.
†A. I. Staff, NAA-SR-3175 (Revised), op.cit.

(b) Primary Sodium Release

The 3, 000 gallons of sodium pumped into the primary gallery will cover the gallery floor (area $\approx$ 580 ft^2) to a depth of approximately eight inches. This depth is assumed to be too shallow to permit self-quenching by oxide smothering and too deep for reduction of the sodium temperature below the ignition point in air; therefore, combustion will continue as long as there is sufficient oxygen and moisture to sustain the reaction.* It would be well to note at this point that operation of the primary gallery dehumidification system, up to the time that the nitrogen atmosphere is replaced by air, ensures that the gallery wall, floor, and ceiling surfaces will be dry. Therefore, any moisture must be contained in the air that replaced the nitrogen.

Considering the temperature of the sodium discharging into the gallery, it will ignite almost immediately upon exposure to the assumed air atmosphere. The heat liberated will increase the pressure within the gallery and thereby deny entry to additional air. Here it has been conservatively assumed that neither the wall surfaces nor the tetralin cooling coil located in the gallery would remove any significant portion of the heat (mostly due to the relatively short duration of the reaction -- see below). The combustion of the sodium is therefore assumed to proceed at the maximum rate until all oxygen within one primary gallery volume has been either consumed or displaced by leakage around the gallery shield blocks. This analysis does not assume that the gallery shield blocks would be removed as a result of the pressure rise in the gallery. In addition it is conservatively assumed that 50% of the air originally contained within the gallery would be displaced because of expansion due to heat absorption. It should be noted that assuming all the available oxygen is to be consumed is quite conservative, as previous experience has indicated that the concentration of oxygen in the gallery would be about 11.5% when combustion terminated.† Consumption of all of the available oxygen will result in the combustion of 149 pounds of sodium.

*R. J. Begley and J. J. Droher (private communication, Atomics International, February 1962).

†J. D. Gracie and J. J. Droher, "A Study of Sodium Fires," NAA-SR-4383, October 15, 1960.

As in the Hallam Final Summary Safeguards Report,* it has been assumed that (a) the burning rate of the sodium is 1.3 $lb/hr\text{-}ft^2$ (combustion will, therefore, endure for 12 minutes) and (b) 50% of the combustion products become airborne. Since the gallery shield block design was not intended to provide any significant degree of gallery atmosphere confinement, little resistance to leakage from the gallery is available. However, due to the path which the leakage must follow in escaping from the gallery, it has been assumed that only 50% of the airborne sodium oxide will be released into the high bay, since at least some confinement ability is offered by the shield blocks.

(4) Activity Released to the Atmosphere

(a) Helium Release Through Top Shield

The radioactive noble gases contained in the helium atmosphere will leak into the high-bay area, which has a volume of about 300,000 cubic feet. No damage to the high-bay exhaust system can reasonably be expected to result from this accident, and it is therefore assumed to exhaust the activity into the outside atmosphere for the duration of the leakage from the reactor. The exhaust system consists of two fans, each mounted in separate cubicles atop the high bay roof. Each fan removes 12,000 cfm of high bay air through 16 CWS-type filters. Since the fan cubicles rise only about four feet above the roof, the release is considered as though it originated at ground level.

(b) Primary Sodium Release

Considering the large mass of material released by sodium combustion and that each of the 32 filters in the high-bay exhaust system will plug at a sodium oxide loading of 1/2 pound, the exhaust system will become inoperative within a very short period of time. Therefore, the high-bay exhaust system is not considered to be operating during the primary sodium release accident. (The direct radiation dose from the 16 pounds of sodium oxide which would plug the filters will be included in the tabulation of the radiation

*A. I. Staff, "Final Summary Safeguards Report for the Hallam Nuclear Power Facility," NAA-SR-5700, September 1961.

exposure). In addition, the fact that most of the nearby terrain rises to heights which are equal to or greater than the roof lends further credence to the assumption of a ground level release. As a result, all leakage to the outside atmosphere will be assumed to take place through openings in the high bay walls.

In calculating the leakage rate from the building during the combustion period, it was assumed that all the heat liberated in the combustion process was released to the high bay area. It was further assumed that the resulting overpressure attained its maximum value as soon as combustion started and dropped back to near atmospheric with the end of combustion. This overpressure would result in a leak rate of 4.15×10^5 cfh, which would continue for the 0.2 hour of combustion. From the time combustion terminated until the end of the exposure period, the high-bay leakage rate was computed from the pressure differential created within the high bay by the pressure exerted against the building walls by the external wind. The after-combustion leakage rates are 157 cfh for the on-site cases (wind speed = 0.5 meters/second) and 5650 cfh for the off-site cases (wind speed = 3.0 meters/second).

b. Consequences of Maximum Credible Accident

The internal doses computed on-site represent the TID received as a result of inhalation for two hours. The on-site, external, whole-body doses are the result of a 2-hour exposure within the release cloud, exposure to ground deposition, and the direct exposure from the sodium contained within (1) the high bay immediately after combustion terminates and (2) the high-bay exhaust system filters. The on-site dose from ground deposition considers exposure to the contaminated ground for the 2-hour release period and for the 90 days starting 24 hours after the accident. The dose received in the 90-day period is obtained by calculating the dose received in each 8-hour work period, correcting for an occupancy factor which effectively reduces the exposure received by 25%. (See Appendix B for further detail.) It should be noted that exposure to direct radiation is only possible at the nearest facility. Since all other locations are highly shielded by the terrain which surrounds the SRE on three sides, the dose from direct radiation exposure would be significantly reduced.

There is still another major conservatism implicit in the calculations of the doses received at on-site locations in the primary sodium release accident.

This stems from the fact that the maximum acceptable concentration (MAC) for sodium in air is 2.0 mgm/m^3,* and severe constriction of and burning in the throat occurs at concentrations even lower than this value. Calculations of the airborne concentrations at the nearest facility and at the site boundary indicate that the value will exceed the MAC; hence it is highly improbable that personnel would remain at these locations throughout the exposure period.

Doses at the nearest community are computed similarly except that the period of exposure to the cloud is 24 hours. Also, no occupancy correction is made to the dose from ground deposition. In addition, the direct dose is not computed due to the large distance involved.

(1) Doses at the Nearest Facility

The nearest facility is the Engineering Test Building, Building 003, which is located about 280 feet to the east. The whole body dose at this location for the helium atmosphere leak is calculated to be 7.4 rem. Since only noble gas fission products are contained within the release plume, no internal or fallout exposures will occur.

At this same location, the accident involving the primary sodium release would produce a whole-body gamma dose of 9.9 rem due to cloud immersion and 1.4 rem from direct radiation. For this accident, since it was assumed that the sodium oxide would be soluble, the whole body was used as the critical organ; thus, inhalation of the sodium oxide was calculated to result in a whole body dose of 6.1 rem. The dose from ground deposition is broken down into the exposure received during two different periods; the dose received in the first 2 hours after the accident was calculated to be 5.2 rem, whereas that received in the 90-day period starting 24 hours after the accident, would be 8.3 rem.

(2) Doses at the Site Boundary

The nearest site boundary lies approximately 330 feet northwest of the SRE. The whole-body dose at this point for the helium atmosphere leak was calculated to be 5.9 rem.

*American Conference of Governmental Industrial Hygienists, Threshold Limit Values for 1961, adopted at the 23rd annual meeting in Detroit, Michigan. April 9-12, 1961.

At this same location, the primary sodium release accident would result in a whole-body dose of 9.6 rem due to cloud immersion, and an additional 0.78 rem from direct radiation. The whole body dose from inhalation would be 6.0 rem. As in the case of the dose at the nearest facility, the dose from ground deposition is broken down into that received during two different periods; the dose received in the first 2 hours after the accident was calculated to be 3.1 rem, whereas that received in the 90-day period starting 24 hours after the accident would be 12.0 rem. The exposure during the 90-day period assumes continuous occupancy.

(3) Doses at the Nearest Community

The whole-body dose at the nearest community from the helium atmosphere leak was calculated to be 4.2×10^{-3} rem.

The primary sodium release accident would result in whole-body doses of 5.9×10^{-3} rem due to cloud immersion, 2.8×10^{-2} rem due to ground deposition, and 2.9×10^{-3} rem from inhalation.

16. Building 100 - Epithermal Critical Experiments Laboratory (ECEL)

a. Maximum Credible Accident*

(1) Events Leading to the Accident

In the extremely remote event that a core loading error is committed or that the critical assembly is flooded with water, a step insertion of 0.06% $\Delta k/k$ will result. The hazards report points out that the safety rods are effective for step reactivity insertions not exceeding 0.02% $\Delta k/k$, and that the shutdown fuse is effective for step insertions of up to 0.03% $\Delta k/k$. Under these conditions, ultimate shutdown will occur through the evaporation of fuel. The integrated energy release to produce evaporation in this case is assumed to be 150 Mw-sec.

Therefore, the maximum credible accident is postulated to be a transient associated with a 0.06% $\Delta k/k$ step reactivity insertion, resulting in a 150 Mw-sec energy release. As a result of this accident, approximately 2 kg (25%) of uranium is assumed to reach the molten state, and, being exposed to the atmosphere, is assumed to burn.

(2) Core Fission Product Inventory

Since the critical assembly is operated intermittently and only at low power levels, the accumulation of fission products from these operations may be disregarded in the analysis of radiological consequences of the fission product release. This fission product activity formed in the 150 Mw-sec transient is equivalent to 2.7×10^8 curies of gamma activity 1 second after the termination of the transient.

(3) Mode and Degree of Fission Product Release from Core

Since about 63% of the fissions are calculated to occur in the driver region and 25% of the fuel material is assumed to burn as a result of the postulated accident, about 16% of the total fission product inventory is considered to be available for release from the core.

Because burning of the fuel is assumed, the associated fission product release is assumed to be comprised of 100% of the noble gases, 100%

*D. T. Eggen et al., "Epithermal Critical Experiments Preliminary Safeguards Report," AI-4120, August 12, 1959.

of the halogens, and 1% of the remaining activity. The activity released from the core was calculated to be equivalent to 9×10^6 curies of gross gamma activity.

(4) Fission Product Release to the Atmosphere

Since fuel combustion is not instantaneous and the cell volume is relatively large, it does not appear credible that the accident would create an overpressure within the cell capable of destroying the ventilation system. Therefore, the ventilation system was assumed to be functioning both during and after the accident. The volume of the reactor cell is 54,000 cubic feet and the ventilation rate is 6000 cfm. If the ventilation system is switched off, all fission products would be confined within the cell and the total amount of radioactivity released to the atmosphere would be significantly smaller than in the release assumed here.

Since the facility has only a short stack, the atmospheric release was assumed to propagate as in a ground level release. The fission product release time from the building, based on three air changes, was assumed to be 30 minutes. Assuming that one-half of the airborne halogens deposits within the building and that the high-efficiency filters in the ventilation system will remove 99.95% of the nonvolatiles, the gross gamma activity released from building, uncorrected for decay, corresponds to 7.1×10^6 curies.

Since the volume of the building is relatively large, a 10-second delay time was allowed between the core release and the initial atmospheric release.

(5) Release of Uranium and Its Daughter Products to the Atmosphere

Since the assumption was made that, during this accident, the ventilation system will remain in operation, the high efficiency filters, which will remove 99.95% of the nonvolatile fission products, will also eliminate a corresponding percentage of any airborne uranium or daughter products. Therefore, only minute quantities of uranium or its daughter products will reach the atmosphere, resulting in insignificant exposures.

b. Consequences of Maximum Credible Accident

(1) Doses at the Nearest Facility

The nearest facility is Building 009, the SGR and OMR Critical Facility, located about 200 feet to the southwest. For the postulated 2-hour exposure at that distance, the dose to the thyroid was calculated to be 75 rem and the whole-body dose from cloud immersion to be 7.5 rem. The dose from ground deposition was calculated to be 0.43 rem.

(2) Doses at the Site Boundary

The nearest site boundary is located 650 feet to the northwest. The doses at that distance were calculated to be 66 rem to the thyroid and 3.7 rem whole body from cloud immersion. The dose from ground deposition was calculated to be 1.9 rem.

(3) Doses at the Nearest Community

The doses at the nearest community were calculated to be 4.8×10^{-2} rem to the thyroid, 3.4×10^{-4} rem whole body gamma from cloud immersion, and 2.3×10^{-3} rem from ground deposition.

F. SUMMARY AND DISCUSSION OF RE-EVALUATION OF RADIOLOGICAL CONSEQUENCES OF MAXIMUM CREDIBLE ACCIDENTS

The results of the re-evaluation of the radiological consequences of the maximum credible accidents are listed in Table VI-2. Inspection of the table indicates that, although there are some doses which approach the dose criteria established for on-site or for the site boundary, there is only one case in which the criteria were exceeded. This occurs as a result of the SRE accident which postulates a release of primary sodium coolant to the gallery and subsequent combustion. The on-site ground deposition dose from the sodium oxide was calculated to result in a dose of 8.3 rem in the 90-day period commencing 24 hours after the start of the accident, which exceeds the criteria established. Actually, however, the dose received in the first 8 hours of the 90-day period would be 5.6 rem (which considers that for 1/4 of the exposure period the person is inside a building where he is protected from the fallout). The dose received in the second day would be 1.9 rem, which would indicate that the total remaining exposure would be about 0.8 rem (i.e., 8.3 - 5.6 - 1.9). Therefore, because of the short half-life of the Na^{24}, it is evident that if access to the nearest facility were delayed until a total of about 2-1/2 days had elapsed after the accident occurred, the 1.7 rem dose criteria for exposure in the 90-day period following the accident would not be exceeded. This short interruption of the program being conducted in the facilities which would be affected is not considered significant so that it can be said that no interaction actually exists for this case.

In addition, calculations indicate that the total dose in this same 90-day period at the nearest perimeter of the SNAP complex, due to the ground deposition of sodium oxide, would be less than the 1.7 rem value; hence normal operations in this area could continue uninterrupted commencing 24 hours after the accident.

As was expected, considering the results of Section III, the doses at the nearest community were quite low for all cases considered. In no case was the dose to the thyroid greater than 0.4 rem and, similarly, the whole body dose did not exceed 3.7×10^{-2} rem, considering exposure from immersion in the cloud and the dose due to ground deposition, the latter being for the full 90-day period starting with the arrival of the cloud.

TABLE VI-2

SUMMARY OF DOSES RECEIVED FROM RE-EVALUATION OF MAXIMUM CREDIBLE ACCIDENTS

Facility (Building)	Reference Report	Maximum Credible Accident	Core Inventory Emitted: Steady-State Operation	Core Inventory Emitted: Transient	Release to Atmosphere: Mode	Release to Atmosphere: Activity[4] (curies)	Release Height	Nearest Building or Point of Max. Concentration	Distance (ft)	Exposure at Nearest Building or Point of Max. Concentration (rem): Inhaled	Whole Body	Ground	Distance to Nearest Site Boundary (ft)	Exposure at Nearest Site Boundary (rem): Inhaled	Whole Body	Ground	Exposure at Nearest Community (rem): Inhaled	Whole Body	Ground
[illegible] (373)	NAA-SR-MEMO-[illegible]	50 Mw-sec excursion	Negligible	9.0×10^6	Puff	5.5×10^5	Ground level	OMR Control Shelter (373)	350	127	10.5	0.73	120	166	7.1	0.57	5.8×10^{-2}	1.1×10^{-2}	[illegible]
S8ER (373)	NAA-SR-MEMO-7027	60 Mw-sec excursion	Negligible	1.1×10^4	Puff	2.6×10^5	Ground level	OMR Control Shelter (373)	350	61	5.2	0.35	120	18	5.5	0.27	1.8×10^{-2}	2.1×10^{-3}	1.1×10^{-1}
(012)	NAA-SR-MEMO-7200	600 Mw-sec transient	Negligible	2.1×10^7	Continuous 1 hr	4.6×10^5 (Continuous, corrected for decay)	Stack, 10 ft	Point of max concentration	500	6.0×10^{-1}	0.1[7]	5.7×10^{-1}	250	5.0×10^{-1}	0.05[8]	1.6×10^{-5}	Negligible	Negligible	Negligible
SER (010)	NAA-SR-6954	10,000-hr operation at 600 kw followed by transient of 72 Mw-sec	1.0×10^6	Negligible	Continuous 2 hr	1.0×10^3 (100 in last 2 hr)	Stack, 50 ft	Point of max concentration	670	1.65	1.2×10^{-3}	0.18	300	1.35	$< 1.2 \times 10^{-3}$	< 0.9	3.1	5×10^{-4}	2.2×10^{-2}
[illegible] (055)	NAA-SR-6383	10,000-hr operation at 600 kw followed by saturation of 12.5 Mw-sec	5.5×10^6	Negligible	Continuous 15 min	305	Stack, 75 ft	Point of max concentration	1150	1.1	1.8×10^{-4}	1.1×10^{-2}	250	< 1.1	$< 4.8 \times 10^{-3}$	$< 3.5 \times 10^{-2}$	1.8×10^{-2}	5.7×10^{-5}	2.1×10^{-3}
S8FDTF (056)	NAA-SR-MEMO-7359	10,000-hr operation at 600 kw followed by transient	5.35×10^6	Negligible	Continuous 1 hr	691	Stack, 100 ft	Point of max concentration	1550	1.1	4.1×10^{-5}	3.1×10^{-4}	125	< 1.1	$< 1.3 \times 10^{-3}$	$< 7.0 \times 10^{-2}$	3.1	2.1×10^{-4}	1.6×10^{-3}
S2DS (024)	NAA-SR-5383	1 year of operation at 50 kw followed by excursion of 12 Mw-sec	1.5×10^5	Negligible	Continuous 20 min	90	Stack, 85 ft	Point of max concentration	1200	0.36	5.3×10^{-1}	1.0×10^{-3}	500	< 0.36	$< 3.1 \times 10^{-1}$	$< 8.0 \times 10^{-1}$	1.1×10^{-2}	4.7×10^{-9}	1.7×10^{-3}
SCA-4B (024)	NAA-SR-MEMO-6877	40 Mw-sec excursion	Negligible	9×10^4	Continuous 20 min	6.2×10^1	Stack, 85 ft	Point of max concentration	1200	7.8×10^{-1}	1.1×10^{-1}	6.6×10^{-5}	500	$< 7.6 \times 10^{-4}$	$< 1.1 \times 10^{-1}$	$< 3.3 \times 10^{-4}$	5.0×10^{-4}	3.3×10^{-6}	2.1×10^{-3}
S8DR, ATF (059)	NAA-SR-6753	16 Mw-sec excursion	Negligible	1.4×10^4	Puff	310	Ground level	S8DRTF (059)	60	2.9	2.7×10^{-2}	1.2×10^{-4}	250	[illegible]	1×10^{-3}	0.13	Negligible	Negligible	Negligible
STE (028)	NAA-SR-5898	Long operation at 50 kw followed by [illegible]	1.5×10^5	Negligible	Continuous 30 min	712	Ground level	SETF (021)	150	210	1.72	1.2	250	300	1.56	5.7	1.8×10^{-2}	8.6×10^{-2}	5.1×10^{-1}
KEWB (073)	NAA-SR-MEMO-1938	Long operation at 1 kw followed by 20 Mw-sec excursion	910	3.7×10^7	Continuous 10 min	5.7×10^3	Ground level	SNAP Critical Bldg No. 2 (037)	150	2.0	1.1	5.0×10^{-3}	625	0.27	1.2×10^{-3}	1.0×10^{-1}	Negligible	Negligible	Negligible
AE-6 (093)	NAA-SR-5502	1 year of operation at 120 w and 2 hr at 2 kw followed by 1 Mw-sec excursion	6.00	1.6×10^5	Continuous 20 min	7.6×10^5	Ground level	AI W? Control Bldg (085)	50	25	1.0	0.17	250	20	0.16	0.37	1.2×10^{-5}	2.3×10^{-5}	[illegible]
SNAP Critical (009)	NAA-SR-5103	400 Mw-sec excursion	Negligible	2.7×10^8	Continuous 30 min	2.2×10^7	Ground level	[illegible] (100)	200	321	21.5	1.1	650	190	10.6	5.5	4.15	1.1×10^{-3}	7.3×10^{-1}
OMR Critical (009)	NAA-SR-1220	547 Mw-sec excursion	Negligible	1.2×10^7	Continuous 45 min	5.5×10^6	Ground Level	ECSA (100)	200	30	4.1	0.17	650	26	2.0	0.73	2.4×10^{-2}	1.1×10^{-4}	9.1×10^{-4}
SRE (143)	NAA-SR-5175 (Rev)	Melting of one fuel element and subsequent release of helium cover gas at 0.25 cfm to high bay	5.5×10^5	Negligible	Continuous 4 hr	1.05×10^4	Ground Level	Engineering Test Bldg (003)	280	None	7.1	None	340	None	5.9	None	None	1.2×10^{-3}	None
SRE (143)	NAA-SR-5175 (Rev)	Release of primary sodium to gallery and subsequent combustion	$6.6 \times 10^{?}$[1]	Negligible	Continuous 24 hr	210[2] 1360[3]	Ground level	Engineering Test Bldg (003)	200	6.1	11.4 (includes 1.4 from direct dose)	5.2 in first 2 hr, 3.3 in the 90 days starting at 24 hr after the accident	130	6.1[4]	10.1 (includes 0.6 from direct dose)	3.1 in first 2 hr, 12.0 in the 30 days starting at 24 hr after the accident	[illegible][5]	[illegible]	[illegible][10]
LCEL (100)	AI-1120	450 Mw-sec excursion	Negligible	2.3×10^6	Continuous 30 min	7.1×10^6	Ground level	SCR-OMR Cent. Exp. Facility (009)	200	75	7.5	0.11	650	66	1.7	1.9	4.8×10^{-2}	3.7×10^{-4}	2.3×10^{-3}

(1) From Na^{24} only
(2) For wind velocity of [illegible]
(3) For wind velocity of [illegible]
(4) Not corrected for decay
(5) Dose to whole body from inhaled radioactive sodium oxide
(6) Dose from deposited radioactive sodium oxide
(7) Direct dose at nearest facility [illegible] at a distance of 80 ft. Dose at point of maximum ground concentration is negligible in comparison
(8) Compared to [illegible]

The analysis for the two facilities wherein intra-facility interaction is possible, i.e., Building 009 (SGR-OMR Critical Facility) and Building 024 (containing the two test vaults of the SNAP Environmental Test Facility) indicated that the doses received by personnel would be less than the values adopted for such emergencies. More specifically, in Building 009, the only radiation exposure would be due to prompt radiation directly from the core, because the ventilation systems for both parts of the facility are completely independent. The doses would be 11 and 3 rem in the SGR and OMR portions of the facility, respectively, from the accident occurring in the other portion.

In Building 024, prompt radiation is not a problem, the principal source of radiation exposure being that portion of the test cell atmosphere which leaks into the high-bay area (common to both test cells). In this case the thyroid dose is controlling but it is well below the permissible emergency level, e.g., the dose is calculated to be 60 rem TID due to a 10-minute exposure. Since the facility would be evacuated long before 10 minutes had lapsed, there should be no problem. The extent of contamination in the high-bay area was also investigated and found to be of such low magnitude that the radiation levels would not prevent immediate steps at facility decontamination.

It can be concluded, therefore, that insofar as conformance to the dose criteria established in Section IV for the site is concerned, considering the location of multiple independent reactor facilities, there would be no detrimental effects on program schedules as a result of any of the maximum credible accidents.

VII. POTENTIAL EXPANSION OF EXISTING FACILITIES AND ADDITION OF NEW FACILITIES AT THE SITE

New construction at the site, both actual and proposed, is concerned principally with the SNAP Program, and the extent of this construction in the future is largely dependent upon the degree to which the concepts embodied in the present program will be developed and expanded in the next few years. Thus, any discussion of proposed facilities or additions to existing facilities must of necessity be considered tentative, and should be accepted only as representative of future plans, rather than as the actual plans themselves.

Figure VII-1 is a site plan showing existing buildings, buildings under construction, and proposed nuclear buildings, including expansion of existing facilities. As indicated above, the size, location, or even the existence of the proposed building must be considered subject to change, depending upon the direction in which the related programs develop in the next few years.

Table VII-1 presents a description of proposed nuclear facilities or additions to existing facilities, and includes information on expected funding year, proposed experimental program, and containment and shielding criteria, where known at this time. In general, three types of facilities are proposed:

1) Subcritical, with containment provided by once-through ventilation and "absolute" filters, and portable shielding used as required, together with radiation monitoring and strict administrative controls to assure subcriticality.

2) Critical ("zero" power), with permanent shielding but no pressure containment except as required for non-nuclear incidents.

3) Operational (power), with the reactor located below ground level and enclosed by massive shielding, and with the enclosure designed for maximum overpressures and minimum leak rates.

All reactor facilities to be constructed will be required to meet the criteria developed in this report. Therefore, the design specifications for the facilities will reflect those containment requirements necessary to assure compliance.

TABLE VII-1

PROPOSED NUCLEAR FACILITIES

Bldg. No.	Description (Anticipated Funding Year)	Experimental Program	Containment
012	Expansion of Bldg. 012 for zero power tests on advanced reactor concepts (No definite date)	Zero power, high temperature experiments with the SCA-3 for application to higher power space systems. Experiments on other metal hydride assemblies. SCA-6 studies on neutron thermalization.	Above-ground cell with 4 ft concrete walls and 1/4-in. steel liner, designed for a maximum overpressure of 110 psig and a leak rate of 1%/24 hr at 1 psig.
016	New building for tests on a SNAP 4 2-loop system operating at 12 Mwt and rejecting heat to a cooling tower. (1963)	Power tests to provide basic proof of the feasibility of the SNAP 4 concept, to support the development program, and to develop final components.	Below-ground vault, nonhermetically sealed, enclosed in gas-tight steel-lined concrete secondary containment shell, and with overhead concrete shield plugs. Containment shell designed to contain overpressure of approximately 70 psi.
017	New building for tests on SNAP 4 prototypes (1964)	Power tests (12 Mwt) on prototypes contained within integral containment vessel Tests may be underwater in flooded vault, or above ground, using integral heat rejection system.	Below-ground vault, provided for shielding only. Containment provided by integral containment vessel.
019	Expansion of Bldg. 019 for final flight calibration of SNAP 2 and SNAP 10 systems. (1964)	Subcritical tests of flight systems for final flight calibration.	Concrete cell similar to cell now under construction. No pressure containment provided. Administrative controls used to prevent criticality. Shielding installed when approaching critical Once-through ventilation system with "absolute" filters.
024	Expansion of Bldg. 024 consisting of one additional power cell and transfer cell, for life and operating tests of SNAP 2 and SNAP 10 systems at 50 kwt. (1965)	Power tests (50 kwt) on SNAP 2 and SNAP 10 systems, to be part of a follow-up program designed to improve system performance. Objectives are higher power output with lower weight. Tests will be run at approximately 1300°F.	Below-ground vaults with 9 ft thick walls and 8 ft ceiling, lined with 3/16 in. aluminum. Designed for a maximum pressure of 4 psig and a leak rate of 2%/hr at 3 in. water.
028	Additional shielding to be added to Bldg. 028, to permit higher fission-plate power levels (by removal of internal shielding). (1965)	Power experiments to determine shielding information.	No changes required.
050 051	New buildings for closed fuel cycle tests on all SNAP systems. (1965)	Closed fuel cycle pilot plant for subcritical tests, and facility for mockup tests.	No pressure containment. Once-through ventilation with "absolute" filters. Radiation monitoring provided. Administrative controls and subcritical arrays prevent criticality. Storage vaults are to be provided. Blast walls and roofing used where needed.
067 068	New buildings for U^{233} fuel development. (1963)	Subcritical processes involving development of U^{233} fuel	Same as for Bldgs. 050 and 051.
070	New building for SNAP 10A fuel fabrication and calibration. (1965)	Subcritical processes involving fabrication and calibration of SNAP 10A fuel.	Same as for Bldgs. 050 and 051.
082	New building for final assembly and acceptance testing of SNAP 8 systems. May be addition to Bldg. 019. (1966)	Critical experiments at operating temperature on SNAP 8 systems designed for terrestrial use.	Unit lowered into underground vault which is sealed by placing shield plug on top. Containment provided by vacuum shell placed around unit.

UNDER CONSTRUCTION

- 012 SNAP GENERALIZED CRITICAL
- 019 FLIGHT SYSTEMS NUCLEAR QUAL TEST
- 013 NON-NUCLEAR COMP. ASSY & PERF TEST
- 023 LIQUID METALS BLDG
- 038 SNAP OFFICE BUILDING
- 039 SNAP TECHNICAL ADMINISTRATION
- 059 SNAP-8 GROUND PROTOTYPE SYS NUCL TEST
- 056 SNAP-8 FLIGHT PROTOTYPE SYS NUCL TEST
- 057 LAUNCH HANDLING & MOBILE EQUIP DEV

PROPOSED

- 016 SNAP-4 DEVELOPMENTAL SYSTEM
- 017 SNAP-4 PROTOTYPE TEST
- 050 CLOSED FUEL CYCLE PILOT PLANT
- 051 CLOSED FUEL CYCLE MOCK-UP
- 067 U-233 FUEL DEVELOPMENT SUPPORT
- 068 U-233 FUEL DEVELOPMENT (SNAP-8)
- 070 SNAP 10-A FUEL FABR, LOADING, & REACTOR CALIB
- 082 SNAP 8-A NUCLEAR QUALIFICATION TEST

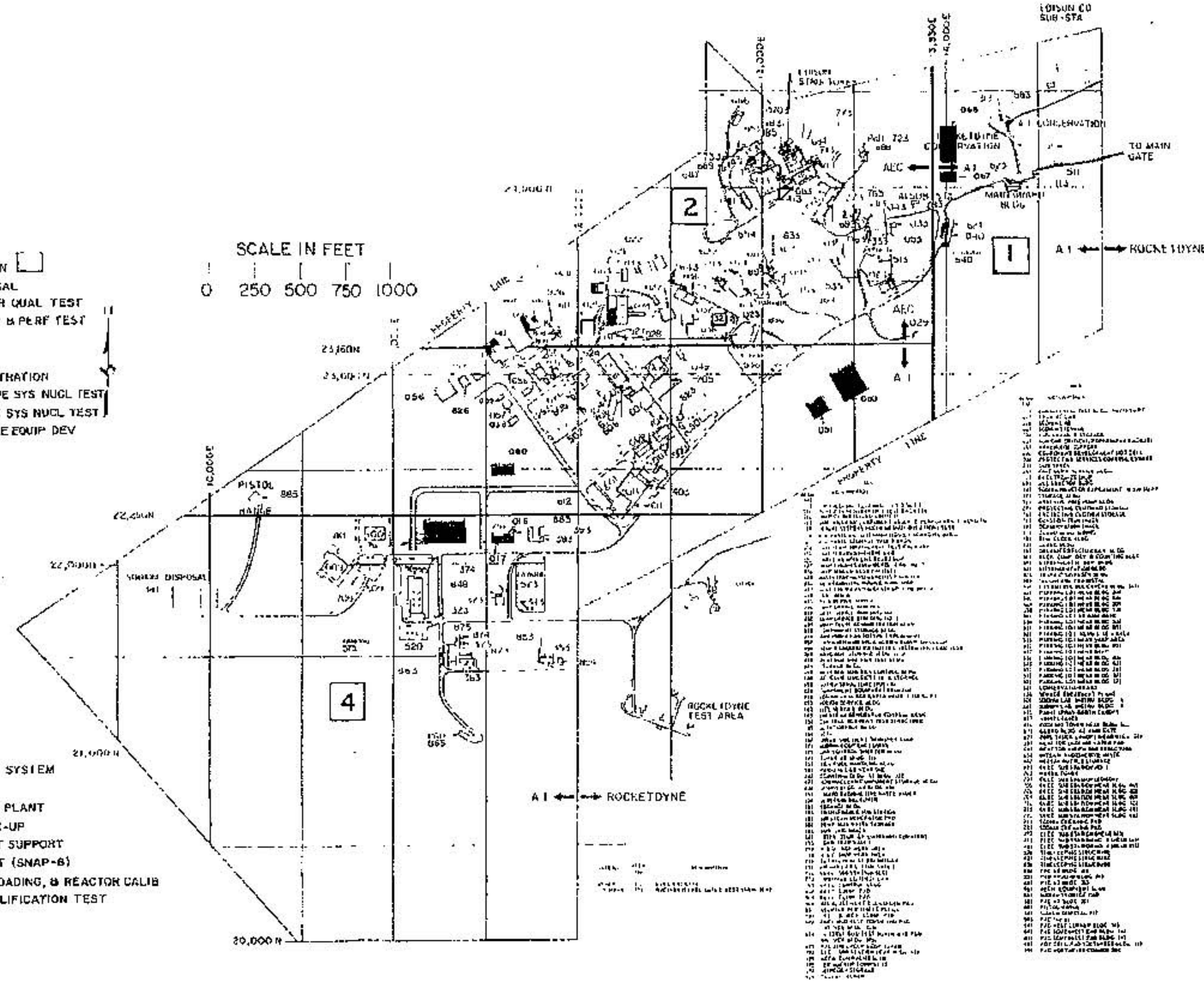

Figure VII-1. Layout of Future Facilities on Nuclear Development Field Laboratory

VIII. SITE EMERGENCY PLAN

A. OVERALL PHILOSOPHY

The Atomics International Emergency Plan for the Nuclear Development Field Laboratory is a broad and flexible plan for the protection of personnel, facilities, equipment, products, and records of the Company and the Government during any period of impending or actual emergency or disaster. It assures coordinated utilization of qualified personnel and equipment required during impending and actual emergencies and includes the training of key personnel to ensure that plans will be effective.

Since radioactive materials are present in many of the buildings on the site, emergency plans and procedures have been built around emergencies involving radioactive material. Thus, the first principle emergency action is the evacuation of personnel away from the hazardous area. Evacuation of personnel is planned in three steps, which are as follows:

1) Local evacuation from the area of direct exposure to a predesignated local assembly area.
2) Movement of personnel from the local assembly area to another local assembly area more remote from the scene of the radiological incident.
3) Movement of personnel from the local assembly areas to points off-site (i.e., site evacuation).

Steps 2 and 3 are utilized as warranted by the conditions existent at the time.

A Master Radiological Emergency Plan which encompasses department or project local radiological emergency action plans has been developed.

The following classifications of emergencies have been established and are applied to radiological incidents to define the various degrees of severity and to specify the action required to effectively control the emergency:

Class I

An emergency of a localized nature, but of such magnitude as to require cessation of normal operation of the facility. Control of the situation can be handled by personnel assigned to the facility with possible backup assistance from established emergency units.

Class II

An emergency of such magnitude and extent as to require evacuation of the area involved, and which constitutes an imminent threat to other facilities in the immediate area. Control of the situation requires active assistance from established emergency units.

Class III

An emergency of such magnitude and extent that other facilities and areas are definitely affected and major emergency procedures are involved. Control of the situation requires full activation of established emergency units.

B. RESPONSIBILITIES

The delineation of authority has been established and specific duties have been assigned in the Master Radiological Emergency Plan as follows:

1. Industrial Security

a. General

In the event of a radiological emergency, the Director, Industrial Security, or his designated representative, will be responsible for:

1) Notifying concerned AI personnel, and assuming primary jurisdiction over personnel, equipment, and supplies during the emergency.

2) Determining the extent of emergency action to be taken, establishing restrictions on access to emergency areas, providing area evacuation and re-entry notifications to all personnel, and coordinating and directing rescue and containment actions in an emergency area.

3) Providing protective services relative to security, fire prevention and suppression, light rescue, and communications.

4) Maintaining an operational and communications Control Center on a 24-hr basis. This center is established for the purpose of receiving emergency alarm signals, for initiation of necessary instructions and directions, dispatching of assistance, and for coordination of all activities being performed in an emergency situation.

In the absence of the Director of Industrial Security, control will be delegated to the Emergency Coordinator of Industrial Security, the Fire Chief, and the Police Chief, or their representatives, in that order.

In carrying out the above functions, Industrial Security will rely on the advice and assistance of the Health, Safety, and Radiation Services Department, and operating department supervision.

b. Master Radiological Emergency Plan

Industrial Security will be responsible for:

1) Preparing a Master Radiological Emergency Plan and furnishing copies of the plan to concerned operating and service departments.

2) Providing coverage, as part of the Master Plan, for emergencies which may occur on AI premises as well as those which occur off AI premises in which the company may be involved; e.g., incidents involving carriers of radioactive and fissionable materials on public streets and highways.

3) Providing guidance to operating departments in the preparation of local radiological emergency plans, and integrating such plans into an effective overall plan of action.

4) Conducting periodic tests of the Master Radiological Emergency Plan in coordination with the Health and Safety Section and the concerned operating supervision.

5) Seeing that all equipment installed in a facility for the continuous monitoring and detection of radiation emergency situations is properly tied in with the overall emergency alarms and communications system, and that the system is working satisfactorily.

In coordination with the concerned departments or units, Industrial Security will establish and include as a part of the Master Radiological Emergency Plan the following:

1) Plans for the type, location, and quantity of emergency supplies and equipment to be maintained, and for periodic checks to ensure that such items will be available as planned.

2) A Master Evacuation Plan; planned evacuation routes and emergency assembly areas will be made known to all personnel by means of markings, signs, and maps posted in various buildings and areas, periodic drills, and by other means deemed necessary by Industrial Security.

3) Notification lists and plans for the special services and equipment to be made available in the event of an emergency by emergency support groups.

2. Public Relations

The Director of Public Relations will coordinate all Public Relations matters, including the issuance of public statements, press releases, etc.; liaison will be maintained with the Control Center.

3. Material

The Director of Material and the Traffic Manager, or their representatives, will make available operators for the full utilization of vehicles and equipment under their control at the time of an emergency or disaster in order to transport the injured, evacuate employees, and transport critical materials. All communications, coordination, and liaison with other emergency services will be through the Control Center.

4. Plant Engineering

The Plant Engineer will organize, plan, and direct the activities of Plant Engineering and Maintenance personnel, utilizing equipment and facilities as required. He will supervise heavy rescue, safeguard structures against collapse, and afford protection for utilities, processes, machinery, and equipment. He will make safe, shut down, repair, and effect rehabilitation of the plant. All communications, coordination, and liaison with other emergency services will be through the Control Center.

5. Medical

The Medical Director will organize and direct emergency first aid and medical service, utilizing first aid stations, company doctors, nurses, and available supplies and equipment.

6. Health, Safety, and Radiation Services

In the event of a radiological emergency, the Health, Safety, and Radiation Services Department will be responsible for:

1) Assessing the nature and magnitude of radiological hazards.

2) Providing monitoring assistance and hazards evaluation advice for emergency team rescue and containment action.

3) Determining permissible radiation exposures for personnel involved.

4) Advising Industrial Security and operating department representatives as to:

   a) Emergency area perimeter zones

   b) Deviations to planned evacuation routes and emergency assembly areas

   c) Decontamination, sealing, and containment procedures pertaining to the minimization and confinement of radiological hazards

   d) Re-entry of personnel to emergency areas

   e) Personnel radiation protection methods and devices.

The Health, Safety, and Radiation Services Department, with the advice and assistance of the Atomics International Reactor Safeguards Review Panel and operating supervision, will determine when, where, and what type of radiation detection equipment (for the continuous monitoring and detection of radiation emergency situations) is required. They will, in coordination with Plant Engineering, see that such equipment is properly installed. In addition, they will coordinate with Industrial Security and operating supervision on all proposed installations of radiation detection devices which will activate emergency alarms.

7. Operating Departments

a. Local Radiological Emergency Action Plans

Each department or project responsible for areas of actual or potential criticality, high radiation, or contamination will prepare and maintain a local radiological emergency plan for each such area under its jurisdiction.

b. General

In the event of a radiological emergency, operating personnel in charge of a local area will be responsible for:

1) Taking immediate action to prevent or minimize injury or damage to personnel and property within or adjacent to their area of jurisdiction.

2) Providing instructions and advice to Industrial Security and Health and Safety representatives as to materials and equipment involved in radiological emergency areas and special hazards and problems that may exist.

3) Providing support for emergency team action for containment of radiological hazards in conjunction with Fire, Police, and Health and Safety units, as may be necessary.

C. DETECTION SYSTEMS

Accidental criticality monitors and alarm systems are installed to meet the specifications of Title 10, Code of Federal Regulation, Part 70.24 and AEC Manual Chapter 050B, which require radiation detection systems to be located in all areas of possible criticality and high radiation. If any of these probes detects a condition of criticality or high radiation, the electronic mechanism automatically:

1) Actuates alarm sirens located throughout the area involved.

2) Transmits an alarm to the Industrial Security Control Center to indicate the location of the emergency.

3) Actuates a flashing red light on the outside of the building that contains the probes which activated the alarm system.

The automatic siren alarm initiates evacuation of the area involved and also initiates Control Center emergency action.

The radiation alarm system is tested and calibrated weekly.

D. LOCAL EVACUATION

1. Route and Assembly Area Selection

a. Philosophy

Because of the number of facilities and the extent of the area included in the site, it has been necessary to establish a number of different emergency assembly areas. A total of five such areas have been provided, and the facilities which evacuate to each specifically delineated. (As mentioned in Section IV, each facility has only one emergency assembly area.) These areas are shown on Figure VIII-1.

The major factor in the choice of these five emergency assembly areas and the routes thereto lies in the requirement that they provide for rapid and safe evacuation from the affected facilities and/or areas. Because of the location of facilities on the site, it has not always been possible to choose assembly areas and routes which are upwind of the more prevalent wind directions. However, it should be pointed out that the results of the accident studies performed in Section VI of this report indicate that the radiation doses which could be received in the assembly areas, in the event winds are so oriented as to send the radioactivity in that direction, are not high. Therefore, although some additional exposure would be received, the total dose would not cause any of the dose criteria to be exceeded.

As a result, in choosing emergency assembly areas, greater emphasis has been placed on locating them closer to means of transportation, and to provide in the emergency plans for additional evacuation if radiological considerations require. Evacuation in such instances could be expedited by use of the cars already available in or near the assembly area. (Occupancy of the cars also can provide some additional degree of protection for the personnel in that the amount of internal exposure due to inhalation of airborne radioactivity would be reduced.) Other factors given consideration in the selection of emergency assembly areas, in addition to their being within a reasonable distance of the facilities which evacuate thereto, are: terrain surface between facilities and assembly areas, accessibility of assembly areas during inclement weather, the availability of hard surface arterial roads for vehicular evacuation and emergency vehicles, the proximity to auto parking areas (see above), and the proximity of other areas of potential hazard.

The general approach taken in evacuation of buildings is that whenever the situation warrants, only that portion of a building that might be affected by an accident is evacuated. This would result, e.g., from accidents of a highly localized nature wherein only a small portion of a building is affected. In accidents of greater magnitude, the entire building is evacuated. In such instances, in order to minimize radiation exposure to personnel in surrounding facilities, it was de-

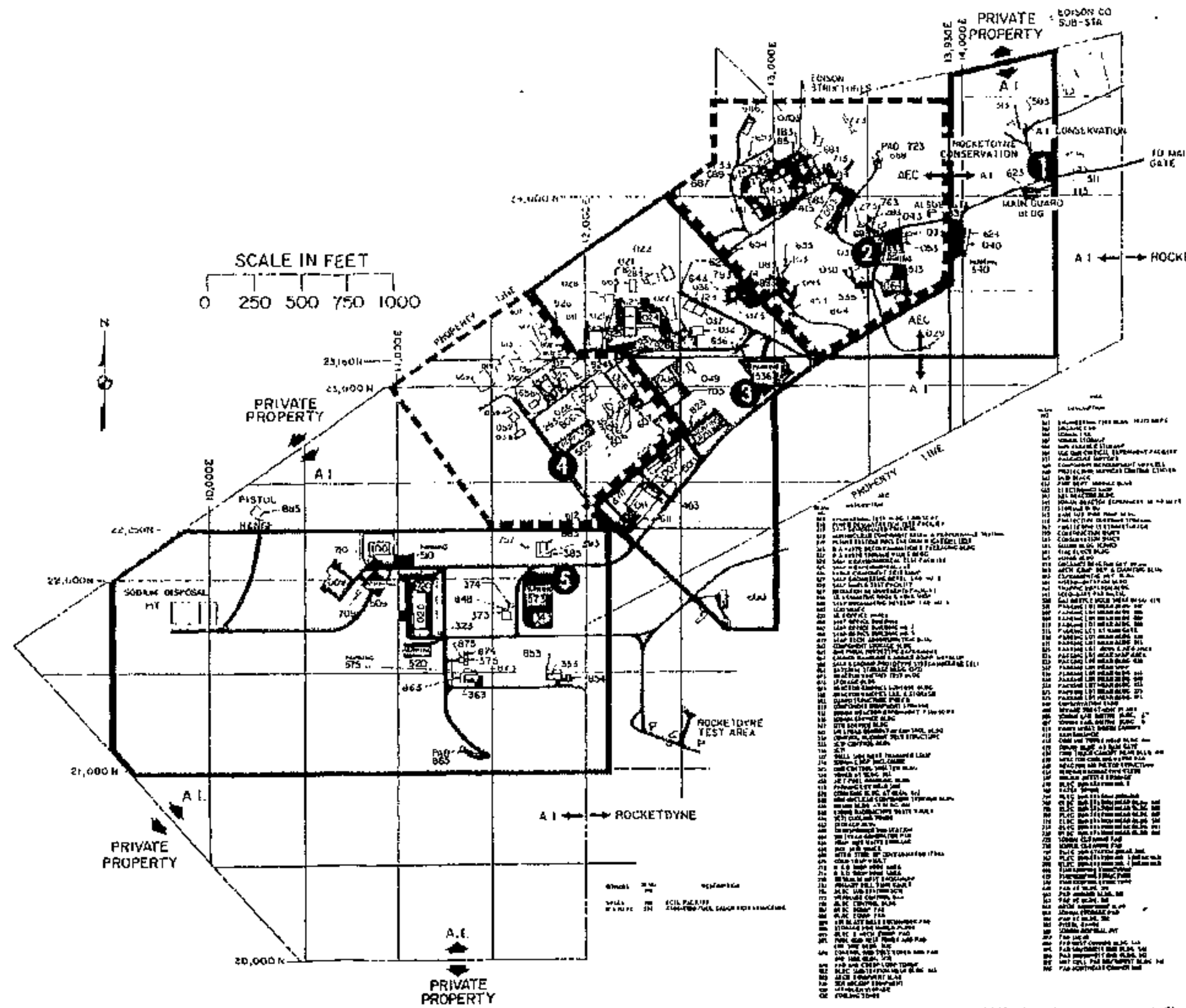

Figure VIII-1. Location of Emergency Assembly Areas cided that evacuation of certain facilities should automatically require similar action in others. Studies of the evaluation of potential hazards associated with certain of the facilities on the site have resulted in a procedure wherein, if an accident occurs in a given facility, certain others are ordered to evacuate and still others are alerted to the situation. The facilities which are so advised are always in close proximity to the facility having the accident. The evacuation warning system has been subdivided accordingly.

b. Locations

Internal and external evacuation routes and emergency assembly areas have been predetermined with the approval of the Health and Safety Section and are marked with red and white signs. Building floor plans and a site plan are also marked to reflect predetermined evacuation routes and assembly areas. Copies of the plans reflecting the evacuation routes and assembly areas are posted in all buildings.

2. Local Emergency Evacuation Plans

Local plans include assignment of group responsibilities; instructions for evacuation of the area; instructions for shutdown of tests and equipment, including reactor shutdowns, where necessary; provision for local area emergency supplies and equipment; and instructions for providing notification to Industrial Security, Health and Safety, and appropriate individuals within the concerned department.

Each Local Radiological Emergency Plan is reviewed by the appropriate Committee of the Reactor Safeguards Review Panel, approved by Health and Safety, and coordinated with Industrial Security to ensure its adequacy and compatibility with the Industrial Security Master Radiological Emergency Plan.

Each department is required to familiarize personnel covered by a local plan with its contents and application by means of reviews and periodic drills. These drills are accomplished in conjunction with tests of the Industrial Security Master Radiological Emergency Plan. A copy of the local plan is posted in each building, and copies are made available to personnel covered by the plan.

3. Communications

An emergency communications system composed of an emergency public address system, an emergency telephone network, and mobile radio communications is utilized to alert personnel in the event of an emergency; to give evacuation instructions; and to direct and control actions of emergency personnel.

In the event that evacuation of a building or area becomes necessary for personnel safety, notice to evacuate will generally be received by the personnel involved by one or more of the following:

1) Verbal local or "in area" notice from responsible operating personnel or from the health physicist near the scene of the incident, whether radiological or otherwise.

2) Public address announcement for various incidents.

3) Siren alarms signaling notice to evacuate.

4. Control

In the event of an emergency situation, a representative of Industrial Security is designated to function as an Area Control Officer, and an Emergency Area Control Point is established at each emergency assembly area. The Industrial Security Area Control Officer will be responsible for:

1) Establishing road blocks and boundaries of the danger areas by use of personnel barricades, rope, signs, flares, etc., in order to establish and maintain complete area control.

2) Maintaining communications and liaison with the Control Center.

3) Controlling movement of personnel, equipment, and vehicles in and out of the emergency assembly area.

4) Maintaining close liaison and coordinating area control and evacuation action with Health and Safety.

E. SITE EVACUATION

1. Location of Routes

The primary access road to the Field Test Area is a two-lane, well-maintained, hard surface road (see Figure II-22). Site feeder roads branch off the primary road to service facility buildings. The primary access road would be utilized as the primary evacuation road in the event of site evacuation. This road intersects with another primary road at a point approximately 4 miles northeast of the entrance to the Field Test Area. In the event of site evacuation, traffic leaving the site could be divided at this "Y" intersection to expedite traffic flow. Secondary evacuation routes leading southwest towards the community of Thousand Oaks, northwest towards the community of Simi Valley, and southeast towards Canoga Park have their origin at the extreme west end of the site.

2. Load That Routes Can Take

At present, on a normal work day at the peak time of shift start, the primary access road handles, through the AI entrance guard post, approximately 300 cars in 30 minutes, utilizing but one lane of the road. It should be noted that during this tabulation, all cars were required to stop, per normal procedures, for identification of the occupants prior to being permitted entry to the AI site.

Evacuation from AI facilities into adjoining Rocketdyne facilities could also be accomplished very rapidly, utilizing both the primary access road and the inter-site feeder roads. Therefore, egress from the site during an emergency condition could certainly proceed at a faster rate than that indicated in the tabulation. In the event evacuation of the entire site is required, evacuation from AI facilities to surrounding communities would be accomplished by utilizing, as required, controlled and coordinated movement of traffic along the primary evacuation road and secondary routes.

3. Communications

Mobile and portable radio communications are utilized to maintain communications between key control points in the area control system and between the area control point and the Industrial Security Control Center.

In the event site evacuation becomes necessary, the entire emergency communications system, public address, telephone, portable page, and radio would be utilized to give evacuation instructions and to direct and control actions of emergency personnel.

4. Control

A representative of Industrial Security is designated to function as an Area Control Officer. The Industrial Security Area Control Officer will be responsible for:

1) Issuing site evacuation instructions to personnel at emergency assembly areas upon receipt of directives from the Control Center.

2) Instituting emergency traffic control plans for orderly and rapid movement of personnel and vehicles.

3) Coordinating actions with Health and Safety and the Control Center.

F. SUMMARY

The Atomics International site emergency plan has been developed in an evolutionary manner by continually making those changes which are apparent from practice and site development progress as being required. The plan explicitly delineates authority and responsibility for action and also provides for proper procedural actions once an accident warning is received. An emergency evacuation plan has been established and is in operation, with practice drills being held at regular intervals.

The selection of evacuation routes and emergency assembly areas is based on many considerations, all of which have as their chief objective provision for the safe and rapid removal of personnel from the vicinity of an incident. Tabulation of traffic intensity on the primary AI site access road indicates that the entire AI site could be evacuated in a period of time which would not result in overexposure of personnel.

Consideration of these factors indicates that adequate provision has been made for institution of proper emergency actions to minimize the hazard to personnel located on the site, and for the handling of emergency situations. Assignment of staff responsibility, as has been done, will assure proper and continuing attention to this aspect of site management and the adequacy of emergency capabilities.

IX. SUMMARY AND CONCLUSIONS

A. ADEQUACY OF METEOROLOGICAL DATA

From the material presented in Section II, it is apparent that substantial meteorological knowledge of the Atomics International site exists. Previously unreported data have been included which provide an improved and more complete picture of this aspect of the site description.

A comparison between data obtained from Rocketdyne and Atomics International wind instrumentation indicates good agreement (see Section II. C. 2). Therefore, much of the data which have been accrued by Rocketdyne can be considered applicable to the Atomics International site.

The need for additional meteorological data, particularly experimental data on diffusion coefficients appropriate to different weather conditions prevalent at the site, has also been investigated. In this regard, evaluation of the Atomics International site indicates that, insofar as assessing the hazard to the general public is concerned, the doses received at the site boundary are controlling (see Section III.C.2). This is due to the fact that any radioactive cloud generated at the site would have to travel 3 miles to the nearest community. As a result, diffusion would reduce the dose to a small fraction of that at the site boundary, even considering the increased length of exposure permitted at the nearest community by the dose criteria (e.g., 10 CFR 100). Considering the relatively short distances from the reactors to the site boundary, and, on occasion the even shorter distances between facilities, it is apparent that changes in the diffusion coefficients associated with the specific unfavorable meteorological conditions used in this report would result in only minor changes in the doses received at these points. Although the magnitude of the change in the dose at the nearest community would be larger if different coefficients were used, the long transport distance would maintain the dose at a low value compared to that of the site boundary. It is apparent, then, that whereas the short distances between facilities or to the site boundary require small releases, they also relegate the role of diffusion to a relatively unimportant consideration (see Section VI. C).

This does not mean that meteorological considerations at the site are absent; rather, the information which is available can be used with confidence (a) in assessing the degree of hazard from operations, (b) in future site expansion planning, and (c) in locating emergency evacuation areas.

Therefore, it can be concluded that sufficient meteorological information exists for the Atomics International site and there is no need at present to develop more detailed information. However, in the event that the type of activity being carried out on the site were to change so that the amount of radioactivity released from routine operations were to increase substantially, the need for such additional information should be re-examined (see also Section VI.C).

B. ADEQUACY OF SITE TO ACCOMMODATE EXISTING AND PRESENTLY PLANNED FACILITIES

1. Normal Operations

Under normal operating conditions, substantially all routine releases of radioactivity are controlled so that the concentration at the stack exit is such that little credit need be taken for dilution. In this respect, the occurrence of adverse meteorological conditions, which would essentially negate the use of any dilution factor, is quite rare.

Non-routine releases resultant from normal operations are controlled by the Health and Safety Unit assigned to the site and, as a result, simultaneous releases from several facilities are prevented. The above reasoning can also be extended to cover the anticipated site expansion. It can be seen, therefore, that radioactivity releases from normal operations are carefully controlled so that no reduction of the applicable permissible release concentrations is warranted for the site (see Section IV.B).

2. Emergency Situations

The criteria developed in this report for establishing the adequacy of a site containing multiple independent reactor facilities are as follows: As a result of the maximum credible accident:

a) The dose received at the nearest facility or at the on-site point of maximum ground concentration in the first 2 hours after the accident should not exceed 25 rem whole body or 300 rem TID to the thyroid. In addition, considering 40 hour/week exposure in the 90-day period commencing 24 hours after the accident, the total dose due to ground deposition, assuming no reduction by decontamination or weathering, should not exceed 1.7 rem.

b) The dose received at the nearest site boundary, or that at the off-site point of maximum ground concentration, in the first 2 hours after the accident, should not exceed 25 rem whole body or 300 rem TID to the thyroid.

c) The total dose at the nearest community in the first 24 hours after the accident should not exceed 25 rem whole body or 300 rem TID to the thyroid.

With reference to item (a) above, it should be noted that although the criteria specify a value of 1.7 rem from ground deposition, for the 90-day period commencing 24 hours after the accident, the critical item for a multiple reactor site is the time interval between the occurrence of the accident and the time when routine reoccupancy of other facilities can be established so that subsequent exposure will not exceed the 1.7 rem value. This period of time establishes the extent of program interruption which could occur, assuming no reduction in ground deposition concentrations due to decontamination or weathering (which is indeed conservative). It is this period of time, then, which is the critical item; if it is excessive, an unacceptable situation can be considered to exist.

In addition to the criteria listed above, there are two others: first, that, in the event of an emergency in a reactor facility, adequate means are provided for safe and rapid evacuation of personnel from the affected facility as well as from its environs without causing unnecessary additional exposure; and second, that facilities be located so that the more prevalent winds will not carry released contaminants to other facilities. These latter criteria are certainly desirable, but the degree of conformance thereto in any given situation is subject to judgment.

These, then, are the criteria to be met (see Section IV. C). With regard to the dose criteria, examination of the results obtained from the re-evaluations of the maximum credible accidents at each reactor facility* (Table VI-2) indicates conformance in all cases except one, where the maximum on-site dose from

*Note that the doses from ground deposition as listed in Table VI-2 (with the exception of the SRE) do not separate the dose received in the first 2 hours from that received in the 90-day period which follows. The doses were calculated without this breakdown as a matter of expediency. However, if the result, i. e., the total dose, was less than 1.7 rem, the criteria would be satisfied.

ground deposition in the 90-day period commencing 24 hours after the accident exceeded the 1.7 rem criteria. However, since the radioactivity is essentially all due to Na^{24}, it was found that unrestricted access to the nearest facility could be accomplished in 2-1/2 days after the accident, and the exposure in the succeeding 90 days would not exceed the 1.7 rem value. Also, due to the additional distance for the radioactivity to travel in the event that the wind direction brought the cloud to the SNAP complex, the increased width of the cloud would reduce the fallout (ground) concentration and, as a result, access in this locale of the site would be possible 24 hours after the accident. In this case the 2-1/2 day delay was considered acceptable.

Also, it should be emphasized that none of the accidents, even in buildings where intra-facility interaction is possible, could directly cause a maximum credible accident in another facility. Therefore, insofar as existing reactor facilities are concerned, it can be concluded that they all meet the dose criteria (or intent thereof) which were established and, hence, the site can be considered adequate for them in this regard (see Section VI.F). With regard to facilities which are presently planned for construction, they, too, should present no problem since their design will also conform to the dose criteria established herein. These are and will continue to be a requirement in determining design criteria for the reactors and their containment structures.

With regard to the existence of an adequate emergency evacuation plan, such a plan is in effect at the site and has been for some time. As indicated in Section VIII, explicit responsibilities have been delineated, carefully planned evacuation routes and emergency assembly areas have been selected, the chief goal of which is to achieve the safe and rapid removal of personnel from the vicinity of an accident, and the plan is routinely practiced. A plan has also been developed outlining the re-occupancy phases of accident situations.

Because of the importance of this aspect of safety, Atomics International has a full-time emergency coordinator, whose function it is to continually review this aspect of the problem and the overall adequacy of emergency capabilities. It can be concluded, therefore, that Atomics International has taken an extremely aggressive attitude regarding site emergencies, and that an adequate plan has been developed to provide for the safety of site personnel and equipment in the event of any emergency situation (see Section VIII). As in the past, continuous reappraisal of the plan will be conducted in order that it be suitable for the site as it continues to expand.

Insofar as the location of facilities on-site is concerned, it is rather evident that the site cannot possibly achieve the perfect distribution with respect to prevalent wind directions that is the objective of one of the criteria. However, this does not mean that the site is unsuitable. The results of the accident studies indicate that, regardless of the wind direction, radiation exposures on-site are not excessive and in fact, are well within the dose criteria established — even considering the close proximity of facilities (see Section VI. F).

Similarly, the evacuation routes and assembly areas cannot always be so selected as to be upwind (insofar as the more prevalent directions are concerned) from all reactor facilities. In this case, the results of the accident studies show that the doses from the released clouds would not result in excessive additional exposure to personnel in the process of either evacuating to or from emergency assembly areas. Since these areas (generally they are parking lots) have been so selected that readily available transportation could provide evacuation to a more remote area in the event that the situation required, it would appear that adequate protection is being provided (see Section VIII. D. 1. a).

It can be concluded therefore, that, since the facilities presently located on the site meet all of the criteria which have been established, and do not produce any unacceptable interaction with other facilities and programs, the site can be considered adequate for these facilities. These same criteria will be used in siting facilities now in the planning stages. These new facilities will require continuing re-assessment of the site emergency plans, but means for implementing this effort are presently in existence.

C. ADEQUACY OF THE SITE TO ACCOMMODATE ADDITIONAL FACILITIES

From inspection of Figures VII-1 and II-4, as well as the frontispiece, it is apparent that whereas certain portions of the site are relatively saturated with buildings and ancillary items, other portions of the site are comparatively undeveloped. The major area in this category is that located at the southern portion of the site. In particular, the gentle slopes to the south and west of Buildings 009 and 020, and the plain to the south and east of Building 363 represent the area most suitable for additional expansion within the confines of the present site boundary. These areas represent those which would require the least amount of grading work for locating facilities. Other areas would require more extensive efforts along this line, with perhaps the greatest potential being in the area to the southeast of parking areas 501 and 536. Still other smaller areas suitable for locating one or two additional facilities are sprinkled throughout the site; some of these would not require any significant grading. It would appear then, that even considering the facilities which are already planned and whose locations have been selected (see Section VII), the site can accommodate a relatively large number of additional facilities before reaching saturation.

It should be noted that, for such additional reactor facilities as might eventually be contemplated, the data in Section III of this report will provide proper guidance in establishing siting, containment, and stack height requirements. In this manner, i.e., with early consideration of the safety aspects effecting reactor siting, compliance with the criteria set forth in this report can be assured.

D. SIGNIFICANT REQUIREMENTS AND CRITERIA FOR DESIGN OF NEW FACILITIES

As has already been mentioned, the criteria set forth in Section IV of this report will be used to determine the adequacy of any reactor design for location on the site. All presently existing facilities meet these requirements and, in fact, in almost all cases, the facilities are capable of releasing significantly greater quantities of activity than would result from the maximum credible accidents.

Perhaps the newest and most significant of the requirements to be met by any reactor to be located on the site is the 1.7 rem dose in the 90-day period starting 24 hours after the accident. This limit is quite conservative, but compliance thereto will prevent the unnecessary hampering of the performance of normal duties in facilities not directly affected by the accident. In a site as complex as the Nuclear Development Field Laboratory, wherein so many diversified programs are being conducted simultaneously, such a criteria is felt necessary in order to prevent the excessive interruption of programs unrelated to that affected by an accident. (See Section IV. C. 3.)

In addition, the data of Section III will do much to provide early design criteria for reactors to ensure that they meet the limitations imposed on the site by population and meteorological conditions. With early attention to these aspects, a significant step will have been taken to ensure that operation of reactors to be located on the site will not result in undue hazard to the general public. Application of the criteria dealing with on-site radiological considerations (Section IV) will extend this assurance to site personnel, property, and programs as well.

APPENDIX A

METHODS FOR CALCULATING THE RADIOLOGICAL CONSEQUENCES OF THE MAXIMUM CREDIBLE ACCIDENTS

1. Core Inventories

The equivalent gross gamma activity from a transient was calculated according to the method by J. J. Fitzgerald:†

$$A = 1.8 \times 10^{6} P_e t^{-1.2} \text{ curies,} \qquad \ldots (A\text{-}1)$$

where P_e is the released energy in Mw-sec, t is the time in seconds after the transient, and the average gamma-ray energy is assumed to be 0.7 Mev.

Equation A-1 is valid for decay times longer than 10 seconds after the accident. However, according to information in AECU 3066,§ extending this formula to decay times shorter than 10 seconds does not introduce a significant error. Therefore, core activities from the postulated accidents were calculated using Equation A-1 and considering a decay time of 1 second after the accident.

Gross gamma activities resultant from steady-state operation were calculated assuming the same average gamma-ray energy (0.7 Mev) and using the integrated form of the Way-Wigner** equation:

$$A = 8.9 P_s\left[t^{-0.21} - (t_o + t)^{-0.21}\right] \text{ curies,} \qquad \ldots (A\text{-}2)$$

where P_s is the reactor steady-state operating power in Mw, t_o is the duration of steady-state operation in seconds, and t is the time in seconds after shutdown.

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*It is not intended that the methods described in this appendix be considered as the best available — rather, they were chosen mainly as a matter of expediency but consistent with the desired degree of conservatism and the length of time available for their use. More accurate and in most cases, more time-consuming methods are available.

†J. J. Fitzgerald, "Reactor Safeguards and Environmental Hazard Evaluation," American Industrial Hygiene Association Quarterly, 1956.

§"Meteorology and Atomic Energy," AECU-3066, U. S. Department of Commerce Weather Bureau, July 1955.

**A. Brodsky and G. V. Beard, "A Compendium of Information for Use in Controlling Radiation Emergencies," TID-8206 (Revised), 1961.
J. W. Healy, "Calculations of Environmental Consequences of Reactor Accidents," HW-54128, December 1957.

The iodine inventories produced from steady-state operation and from the transients were obtained from the expression

$$A_i = 8.4 \times 10^5 P_s y\left(1 - e^{-\lambda_i t_o}\right) e^{-\lambda_i t} \quad , \qquad \ldots(A-3)$$

where P_s is the steady-state power in Mw, y is the cumulative isotopic fission yield, λ_i is the isotopic decay constant, t_o is the operating time of the reactor, and t is the decay time after shutdown; λ_i, t_o, and t_i have consistent units. Table A-1 lists the cumulative yields and half-lives of the iodine isotopes as used in Equation A-3. It should be noted that use of cumulative yields for calculating transient-produced iodine inventories is extremely conservative. As a result, the thyroid doses and the fallout doses from transients will be quite conservative.

2. Building Release to the Atmosphere

In all accidents which do not postulate violation of the building integrity, the fission products released from the core will be held up in the building for a period of time which is a function of the building ventilation rate. Assuming an instantaneous fission-product release from the core to the building and instantaneous, uniform mixing of fission products in the building air, a certain fission-product concentration (C_o) in the building will be obtained; the units of C_o will be taken as gm/m^3. Ignoring for the moment the effects of radioactive decay, this concentration will decrease as a function of time as $e^{-Rt/V}$, where R is the building ventilation rate in m^3/sec, V is the building free air volume in m^3, and t is the time in seconds after the fission product release. Therefore, the fission product concentration as a function of time, C(t), is given by

$$C(t) = C_o e^{-Rt/V} \quad , \ 2m/m^3. \qquad \ldots(A-4)$$

Table A-2 illustrates rounded-off fractions of the original fission product concentration which will be left in the building after 3, 4, 4.5 and 5 air changes.

For the purpose of this study, it shall be assumed that, in such cases, when three air changes have been completed, all contained fission products will then have been released. When the radioactive nature of the material is considered, it becomes evident that the three air-change criterion is conservative from the standpoint of the allowable decay time prior to the atmospheric release. The conservatism is even more evident in those cases involving fission products resulting from power

TABLE A-1

YIELDS AND DECAY SCHEMES OF IODINE ISOTOPES*

Nuclide	Half Life	Cumulative Fission Yield (%)	Decay Scheme			
			Beta Rays		Gamma Rays	
			Energy (Mev)	Number per Disintegration	Energy (Mev)	Number per Disintegration
I-131	8.05 days	2.9	0.25	0.028	0.722	0.03
			0.34	0.093	0.637	0.09
			0.61	0.872	0.364	0.80
			0.82	0.007	0.284	0.053
I-132	2.4 hr	4.4	0.7	0.15	2.2	0.02
			0.9	0.20	1.96	0.05
			1.16	0.23	1.40	0.11
			1.53	0.24	1.16	0.08
			2.12	0.18	0.96	0.20
					0.78	0.75
					0.673	1.00
					0.624	0.06
I-133	20.8 hr	6.1	0.5	0.06	1.4	0.01
			1.4	0.94	0.85	0.05
					0.53	0.94
I-134	52.5 min	7.6	1.5	0.70	1.78	~0.35
			2.5	0.30	1.10	~0.35
					0.86	~0.30
					0.20	~0
					0.18	~0
I-135	6.68 hr	3.1	3.6	0.35	1.8	~0.50
			5.0	0.40	1.27	~0.50
			6.4	0.25		

*J.O. Blomeke, "Nuclear Properties of U^{235} Fission Products," ORNL-1783, November 1957, and W.H. Sullivan, "Trilinear Chart of the Nuclides," (Revised) January 1957.

TABLE A-2

REDUCTION OF INITIAL CONCENTRATION WITH BUILDING AIR CHANGES

Number of Air Changes	Fraction of Initial Concentration Remaining
3	0.05
4	0.02
4.5	0.01
5	0.007

transients, where the radioactivity is decaying as $t^{-1.2}$, rather than as $t^{-0.21}$, which is the case for the decay of fission product inventories accumulated from extended steady-state power operation.

It should be noted that, in the following sections, since computations of integrated exposure at a point of reception deal with the radioactivity concentration and not the mass concentration, the decay characteristics of the material released are actually utilized.

3. Atmospheric Release

a. On-Site Exposures

(1) Elevated Source Release

The elevated source releases were in all cases continuous releases from stacks. It was postulated in Section VI that, during the time of any elevated source release, the unfavorable meteorological condition of the type known as fumigation would exist. Under fumigation conditions, no maximum ground level concentration is reached as, for example, in the case of the lapse condition; rather, the concentration increases with decreasing downwind distance until the fumigation formula becomes invalid because of proximity to the elevated source point. Concentrations at ground level up to the point where the fumigation formula begins to apply are assumed to be governed by mild lapse conditions postulated to exist under the inversion layer. In this model, the transition between the radioactivity concentrations obtained from the mild lapse and fumigation conditions is not well-defined; for the purpose of this report, however, the transition point is taken to exist at the downwind distance where the cloud first reaches the ground to a significant degree. This latter distance (d_{max}) is taken to correspond approximately to the distance to the point of maximum ground concentration as obtained from Holland's integrated and maximized formula for elevated sources:*

$$d_{max} = \left(\frac{h^2}{C^2}\right)^{\frac{2}{2-n}} \quad \text{meters,} \qquad \ldots(A-5)$$

where h is the stack height in meters, and C and n are meteorological parameters, values of which were defined in Table VI-1.

Based on these premises, the doses were calculated using Holland's fumigation formula,* evaluated at the distance of maximum ground concentration, obtained under mild lapse conditions. In this manner, the effect of cloud recoil from the inversion layer is included. Results obtained from this model are conservative since the dose at the point of maximum ground concentration, obtained under lapse conditions, will be significantly smaller than the dose calculated at the same distance but under fumigation conditions.

Holland's expression for obtaining the total integrated exposure (TIE) under fumigation conditions is given by*

$$TIE = \frac{Q(t)}{\pi^{1/2} C_y \bar{u} H \left(x_o + d\right)^{\frac{2-n}{2}}} \quad \text{curies-sec/m}^3, \qquad \ldots(A-6)$$

where Q(t) is the released activity reaching the point d at time t, corrected for decay at time t; C_y and n are meteorological parameters specified in Table VI-1; H is the assumed height of the inversion layer (= h + 5 in this study) in meters; d is the downwind distance to the point of maximum ground concentration in meters; and x_o is the upwind, virtual point source distance in meters.

The virtual point source distance is given by*

$$x_o = \left[\frac{V}{\pi^{3/2} C^3}\right]^{\frac{2}{3(2-n)}} \quad \text{meters.} \qquad \ldots(A-7)$$

where V is the initial cloud volume in m^3 and the remaining terms have already been defined.

It should be noted that Holland's virtual point source concept was originally developed for puff-type releases. Use of the concept in this study

*AECU-3066, op. cit.

essentially provides a factor by which credit can be taken for the initial dilution of the fission products in the released cloud. Since there is no reason why similar credit should not be taken in the case of a continuous release, where initial dilution due to ventilation is even larger, this concept was also utilized in such cases.

In puff-type releases, the initial cloud volume, V, is generally assumed to be approximately equivalent to the building volume. In the case of continuous elevated releases, it was assumed that V is the volume of air necessary to evacuate the fission products from the building (from Section 2 in this appendix, this volume is assumed to be equivalent to three times the building free-air volume). In extending the virtual point source concept used for puff-type releases to continuous releases, it is also necessary to consider the average fission-product concentration in the released cloud. Since this concentration decreases exponentially in continuous releases, the average concentration is not the amount released divided by the cloud volume, as was the case in a puff-type release. It can be demonstrated, however, that the integrated exponential average over the assumed time required to release the fission products from the facility, based on three building air changes, is not significantly lower than the average concentration, as obtained simply by dividing the amount released by the three building volumes. Therefore, in the calculations for continuous elevated source releases, the volume of air, V, used in Equation A-7 to calculate x_o, was approximated by three building air volumes.

(2) Ground Level Releases

To obtain the total integrated exposure (TIE) from a ground release, Holland's formula* for elevated source releases was used, except that the value of the exponent was taken to be unity, since h = 0. Thus, we may write

$$\mathrm{TIE} = \frac{2Q(t)}{\pi C^2 \bar{u}(x_o + d)^{2-n}} \quad \text{curies-sec/m}^3, \qquad \ldots(\text{A-8})$$

where d is the downwind distance in meters. If the accident were assumed to destroy the integrity of the facility, e.g., by rupturing the facility roof, then the air volume necessary to calculate x_o was assumed to equal the volume of the

*AECU-3066, op. cit.

reactor room. On the other hand, if the reactor room remained ventilated after the accident, the air volume was taken as three times the building free-air volume. In cases where the building did not have a ventilation system, or if the ventilation system were inoperative during and after the accident, V was based on the building leak rate.

b. Exposure at the Nearest Site Boundary

The meteorological parameters used to obtain the extent of radiation exposure at the site boundary are the same as for the on-site calculations. The total integrated exposure at the site boundary was actually calculated only for the ground level release model. For an elevated source release (when winds are assumed to be appropriately directed), the nearest site boundary was usually closer to the facility than the point of maximum downwind ground concentration, and hence the total exposure was accordingly smaller at the boundary, although the maximum dose would occur at some more remote point on the site boundary. For this reason, no attempt was made in such cases to calculate doses at the nearest site boundary and, hence, in these instances Table VI-2 indicates only the maximum dose which would exist at the site boundary (or, correspondingly, the maximum dose which would exist beyond this boundary).

c. Off-Site Exposure

The off-site releases, where the receptor is located in the valleys below the site, are assumed to be independent of the on-site type of release, i.e., whether it was elevated or not. The off-site release is calculated by assuming an effective release height of 200 feet relative to the valley floor below (for both ground and elevated releases at the site). To obtain the total integrated exposure at the nearest community, Holland's elevated source formula was used again:

$$\mathrm{TIE} = \frac{2Q(t)}{\pi C^2 \bar{u} d^{(2-n)}} \exp\left[-\frac{h^2}{C^2 d^{(2-n)}}\right] \text{ curies-sec/m}^3, \qquad \ldots(\text{A-9})$$

where d was taken equivalent to a distance of 3 miles (4830 meters) and h equivalent to an effective height of 200 feet (61 meters).

4. Total Integrated Doses

a. Thyroid Dose

Relative activities of transient- and steady-state-power-produced iodine isotopes were calculated by Equation A-3. The inhalation doses to the thyroid were calculated using values suggested in comments prepared by ORNL on the May 1961 version of 10 CFR Part 100 and more recently included in a report by DiNunno et al.* The values used in this study are listed in Table A-3. The assumption was made in the calculations that the isotopic ratios of the iodine isotopes at the receptor location would be the same as at the time of the core release. The values obtained were then normalized to 1 curie-sec/m^3 of mixed iodine isotopes. The total integrated dose to the thyroid from an exposure of 1 curie-sec/m^3 of mixed iodine isotopes was found to be 24.3 rem for transient-produced iodine inventories, and 147 rem for steady-state-power-produced iodine inventories. These values conservatively assume that no radioactive decay of the iodine isotopes will occur between the time of core release and the time of arrival at the receptor point.

TABLE A-3

CONVERSION FACTORS FOR DOSE FROM INHALATION OF 1 curie-sec/m^3 OF SPECIFIC IODINE ISOTOPES

Isotope	Thyroid Dose (rem)
I^{131}	742
I^{132}	26.5
I^{133}	200
I^{134}	12.5
I^{135}	61.5

b. Whole Body Gamma Dose

The whole-body gamma dose was calculated, in most cases, from the expression†

$$TID = 0.246\,E_{\gamma}(TIE) \text{ rem,} \qquad \ldots(A\text{-}10)$$

where E_{γ} is the average gamma energy (0.7 Mev), and TIE is the total integrated exposure in curie-sec/m^3. This equation assumes that the exposure is to an infinite hemispherical cloud, which is a conservative assumption, especially at distances close to the facility. However, the over-estimate is at least partially compensated for by the neglect of the dose received during the approach of the cloud.

*J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 23, 1962.
†AECU-3066, op. cit.

In the case where the size of the released cloud was very small, a direct dose rather than an immersion dose was calculated at on-site locations. In such instances (cf., Building 019), the cloud was represented by a point source, and the dose was obtained by integration over the cloud trajectory.

c. Dose From Ground Deposition

The rate of ground deposition of fission products from a cloud is related to the concentration of these fission products in the cloud by*

$$W = \overline{\chi} V_g \quad \text{curies/m}^2\text{-sec}, \qquad \ldots(\text{A-11})$$

where $\overline{\chi}$ is the average concentration of the species in the cloud in curies/m^3, and V_g is the settling velocity in m/sec. V_g is a function of wind velocity and was assumed to be 0.025 m/sec for a wind velocity of 3 m/sec.*

The total deposition was calculated from the relation

$$W_T = \overline{\chi} V_g t \quad \text{curies/m}^2, \qquad \ldots(\text{A-12})$$

where t was taken as the release time for continuous releases, or as the cloud diameter divided by the cloud velocity for puff-type releases. Since Equation A-12 ignores cloud depletion, it yields conservative results, particularly at large distances such as that to the nearest community.

Since $\overline{\chi}t$ is simply the TIE calculated for different conditions by Equations A-6, A-8, and A-9, Equation A-12 becomes

$$W_T = (\text{TIE}) V_g \quad \text{curies/m}^2. \qquad \ldots(\text{A-13})$$

The total integrated dose from a given ground contamination level was calculated from the following relationship:

$$\text{TID}_i = \frac{10}{0.7} \frac{\sum_i E_i A_{r_i}}{\sum_i A_{r_i}} \int_0^\infty e^{-\lambda_i t} dt \quad \text{rem}. \qquad \ldots(\text{A-14})$$

*J. C. Couchman, "Graphic and Tabular Aids for Reactor Hazards Evaluation," FZM-2277, June 9, 1961; AECU-3066, op. cit.; and A. C. Chamberlain, "Aspects of Travel and Deposition of Aerosol and Vapor Clouds," AERE HP/R-1261, 1955.

This equation is based on a calculated dose rate of 10 rem/hr at a distance of 1 meter from an infinite surface contaminated with 1 curie/m^2 of 0.7 Mev gammas. In Equation A-14, E_i is the isotopic (average) gamma energy calculated from the energies listed in Table A-1 and A_{r_i} is the relative isotopic abundance as determined from Equation A-3, using cumulative yields for y (Table A-1).

Integration of Equation A-14 and appropriate substitution for λ_i yields

$$TID_i = 20.65 \frac{\sum_i E_i A_{r_i}}{\sum_i A_{r_i}} \left(T_{1/2}\right)_i \quad \text{rem}, \qquad \ldots(\text{A-15})$$

where $\left(T_{1/2}\right)_i$ is the isotopic half life in hr^{-1}.

Using Equation A-15, it was found that the life-time dose from ground contaminated uniformly with 1 curie/m^2 of mixed iodine isotopes would be 46.2 rem for transient-produced inventories and 280 rem for steady-state-power-produced inventories. Therefore, the dose due to ground contamination from the postulated accidents is 46.2 W_T and 280 W_T for transient- and steady-state-power-produced iodine inventories, respectively.

d. Direct Dose

The direct dose from fission products remaining in the building was calculated for those cases where the released radioactivity (or a portion thereof) was effectively contained by the facility design and resulted in a significant radiation source above ground level. The calculations of the direct dose conservatively assumed no air or structural attentuation, except in those cases where substantial shielding material existed between the remaining radioactivity and the receptor.

APPENDIX B

METHODS FOR CALCULATING THE RADIOLOGICAL CONSEQUENCES OF THE SRE MAXIMUM CREDIBLE ACCIDENTS

1. Activity Inventories

a. Helium Release Through Top Shield

The activity inventory in curies, $Q_i(0)$, of each individual noble gas fission product isotope in the reactor helium atmosphere immediately after the fuel element melts was calculated from

$$Q_i(0) = \frac{Y_i\left(1 - e^{-\lambda_i t}\right)}{\sum_i \left[Y_i\left(1 - e^{-\lambda_i t}\right)\right]} CV \quad \text{curies,} \qquad \ldots(B-1)$$

where Y_i is the fission yield of the i^{th} isotope, λ_i is the decay constant of the i^{th} isotope, t is the operating time of the reactor, C is the concentration of noble gas fission products in the helium atmosphere immediately after the fuel element melts, and V is the volume of the helium atmosphere. Values for the fission yield, Y_i, were obtained from Katcoff.* Half-life values used to compute λ_i were obtained from Stehn† and Strominger et al.§ (Isotopes with half-lives less than two minutes were omitted.) The reactor operating time was assumed to be equivalent to one year. The value of C was taken to be 4.2×10^{-2} curies/cm^3 of helium as reported in the hazards summary.** The volume, V, of the helium atmosphere is 9.15×10^6 cm^3.

b. Primary Sodium Release

The heat generated from the combustion of sodium will cause the gallery atmosphere to expand. A conservative calculation indicates that one-half of the 5600 cubic foot primary sodium system gallery atmosphere will be released

*S. Katcoff, "Fission Product Yields From Neutron-Induced Fission," Nucleonics, 18, No. 11 (November 1960).

†J. F. Stehn, "Table of Radioactive Nuclides," Nucleonics, 18, No. 11 (November 1960).

§D. Strominger, J. M. Hollander, and G. T. Seaborg, "Table of Isotopes," Reviews of Modern Physics, 30, No. 2 (April 1958).

**The Atomics International Staff, "Hazards Summary for Thorium-Uranium Fuel in the Sodium Reactor Experiment," NAA-SR-3175 (Revised), July 1, 1959.

from the gallery due to the increased pressure resulting from absorption of the heat generated during combustion of the released sodium. This limits the oxygen available for reaction with the sodium to 736 moles. Assuming all of the oxygen to be consumed in the formation of the combustion by-product, Na_2O, 149 lb of sodium will react. This analysis implicitly assumes that the overpressure does not remove the shield blocks. Taking the saturation activity of Na^{24} in the primary coolant as 0.3 curies per gram,* this corresponds to 2.03×10^4 curies. (For further details, see the summary of the SRE maximum credible accident as discussed in Section VI.)

2. Release to the Atmosphere

a. Helium Release Through Top Shield

The reactor helium atmosphere is assumed to leak at a rate of 0.25 cfm through the reactor top shield seal into the 300,000 cubic foot high-bay volume and subsequently to the outside atmosphere. Since this constitutes a double-compartment type of release, the quantity of radioactivity escaping the high bay was obtained from

$$S_i(t) = \frac{Q_i(0)}{F - G} e^{-\lambda_i(t+d/\bar{u})}\left[F\left(1 - e^{-Gt}\right) - G\left(1 - e^{-Ft}\right)\right] \text{ curies, } \quad \ldots(B-2)$$

where

$S_i(t)$ = quantity of the i^{th} isotope released from the high bay at time t, curies

$Q_i(0)$ = quantity of the i^{th} isotope contained in the helium atmosphere at time t = 0 (as calculated by Equation B-1), curies

F = leakage rate of the helium atmosphere into the high bay, 1.29×10^{-5} sec^{-1}

G = leakage rate of the high-bay volume, 1.33×10^{-5} sec^{-1}

λ_i = decay constant of the i^{th} isotope, sec^{-1}

t = time after leakage commences (here it is assumed that corrective action can be taken after 2 hr), 7200 sec

*The Atomics International Staff, NAA-SR-3175 (Revised), op. cit.

d = distance from SRE to receptor, 86 meters for the nearest facility, 101 meters for the site boundary, and 4830 meters for the nearest community

$\overline{u}$ = average wind speed, 0.5 meters/sec for on-site calculations and 3.0 meters/sec for off-site calculations.

b. Primary Sodium Release

Of the 149 lb of sodium which react to produce Na_2O, it has been assumed that only 50% of the Na_2O becomes airborne* and that, due to adsorption on the gallery shield block and wall surfaces, only 50% of the airborne fraction is released subsequently into the high-bay area, corresponding to a total release of 37-1/4 lb of sodium. Of this amount, 12 lb are entrained in the high-bay exhaust system filters (12 lb of sodium corresponds to the 16 lb of Na_2O which would plug the 32 filters, assuming a loading of 1/2 lb per filter as the limiting case). The filters therefore contain 1.63×10^3 curies of Na^{24}, leaving 3.43×10^3 curies available for release to the outside atmosphere.

During the period of sodium combustion, it is assumed that the release rate from the high bay is determined by the pressure resulting from absorption by the air in the high bay of the heat generated by the sodium combustion. To simplify the determination of this pressure and to ensure the conservatism of the calculated value, it has been assumed that all of the heat generated by combustion of the sodium is so absorbed. It has also been assumed that the pressure thus created in the high bay increased in a step function to the maximum value as soon as combustion commenced, remained constant throughout the combustion period, and decreased in a step to near atmospheric when combustion ceased. The leak rate associated with this period of high-bay overpressure results in a constant high-bay contaminant release rate which is calculated from

$$R_v = \frac{HMV}{t_c m c_v T_o} \quad ft^3/hr, \qquad \ldots (B-3)$$

where

R_v = leak rate from the high bay maintained throughout the combustion period, cfh

*The Atomics International Staff, "Final Summary Safeguards Report for the Hallam Nuclear Power Facility," NAA-SR-5700, September 1961.

H = heat of combustion of sodium with Na_2O as the end product, 4100 Btu/lb*

M = mass of sodium reacted, 149 lb

V = volume of high bay, 3×10^5 ft^3

t_c = duration of combustion, 0.198 hr

m = mass of high-bay air, 2.42×10^4 lb

c_v = specific heat of air at constant volume, 0.174 Btu/lb-°F

T_o = original temperature of high bay air, 530°R.

Substitution of these values into Equation B-3 yields a leak rate from the high bay, R_v, of 4.15×10^5 cfh. At this leak rate, 8.2×10^4 ft^3 of contaminated air will be released from the high bay during the combustion period (approximately 12 minutes). Since the activity released to the high bay is contained in 3.82×10^5 ft^3 of air (i.e., the high-bay volume plus the overpressure volume), the total activity released during the combustion period will be $\left[(3.43 \times 10^3)(8.2 \times 10^4)\right]/(3.82 \times 10^5) = 7.36 \times 10^2$ curies, or 3.68×10^3 curies/hour.

Therefore, upon termination of combustion 2.70×10^3 curies of Na^{24} (i.e., $3.43 \times 10^3 - 7.36 \times 10^2$) would remain in the high bay. From this point on, the driving force for leakage from the high bay is assumed to be the pressure differential created by the wind blowing against the exterior walls of the high-bay area. Two equations were considered for calculating the pressure exerted on the walls by the wind. The first† is

$$P = \rho k\left[\left(\frac{h}{30}\right)^{1/7} v\right]^2 \text{ in. water,} \qquad \ldots(B\text{-}4)$$

*J. D. Gracie and J. J. Droher, "A Study of Sodium Fires," NAA-SR-4383, October 15, 1960.

†J. E. Dykins, "Navy Facilities for Evaluating Prefabricated Buildings," ASTM Special Technical Publication No. 210, presented in the Symposium on Full-Scale Tests on House Structures at the 2nd Pacific Area National Meeting of the American Society for Testing Materials in Los Angeles, California, September 18, 1956.

and the second* is

$$P = 4.5k\frac{\rho v^2}{2} \quad \text{in. water,} \qquad \ldots(B-5)$$

where

P = pressure exerted on building walls by the external wind, inches of water

ρ = density of air, 0.0025 slugs/ft^3

k = in. of water/lb-ft^2, 0.192 in. of water/psf

h = height of building, ft

v = wind speed, mph in Equation B-4 and fps in Equation B-5.

Equation B-4 is considered to yield more realistic results because the building height is taken into account, but, since both equations are valid and more conservative results (i.e., higher pressures and resultant leak rates) are obtained from Equation B-5, this latter equation was used in the calculations.

There are two wind speeds and, consequently, two building leak rates used in the analysis (see Section VI). The first, which is for the on-site calculations, assumes a wind speed of 0.5 meters per second, resulting in a pressure of 0.0029 in. water. The other, for the off-site (nearest community) calculations, assumes a wind speed of 3.0 meters per second, resulting in a pressure of 0.105 in. water. Normal construction methods result in a leak rate of 1-1/2 cfh per ft^2 of wall area at 1 in. water;† however, the SRE leakage is assumed to be 3 cfh per ft^2 of wall area at one inch of water.§ Using this value and a high-bay wall area of 1.8×10^4 ft^2, the leak rates subsequent to sodium combustion are 157 cfh and 5,650 cfh, respectively.

To summarize briefly, then, the release of Na^{24} from the SRE high bay occurs in two distinct ways: (1) throughout the period in which sodium in the gallery

*Ning Chien, Yin Feng, Hung-Ju Wang, and Tien-To Siao, "Wind Tunnel Studies of Pressure Distribution on Elementary Building Forms," Iowa Institute of Hydraulic Research, State University of Iowa (unpublished thesis).

†"NAAMM Specifications for Static Load Testing of Curtain Walls," Metal Curtain Wall Manual, Section 3.5 (Copyright 1960).

§R. L. Koontz (unpublished data, Reactor Housing Study Group, Atomics International, March 1962).

is reacting with oxygen, the release is constant at 3.73×10^3 curies/hour, and (2) from the time sodium combustion terminates through the end of the exposure period, the release occurs at a volume leak rate determined by the wind speed.

The total quantity of Na^{24} arriving at a downwind receptor during the entire exposure period was computed from

$$Q = qt_c e^{-\lambda_r d/\bar{u}} + Ae^{-\lambda_r (t_c + d/\bar{u})} \left(1 - e^{-Rt_a/V}\right) \text{ curies,} \qquad \ldots(B-6)$$

where

Q = total quantity of Na^{24} arriving at the receptor from time $t = 0$ to $t = t_c + t_a$, curies

q = release rate of Na^{24} during combustion, 8.95×10^4 curies/day

A = total quantity of Na^{24} remaining in high bay at time $t = t_c$, 2.70×10^3 curies

t_c = duration of combustion period, 0.00824 days

t_a = duration of release after combustion period, 0.0751 days for on-site calculations and 0.992 days for off-site calculations (see also discussion of the dose from fallout, section 3.b of this appendix).

λ_r = radioactive decay constant of Na^{24}, 1.109 day^{-1}

d = distance from SRE to receptor, 86 meters for the nearest facility, 101 meters for the site boundary, and 4,830 meters for the nearest community

$\bar{u}$ = average wind speed, 4.32×10^4 meters/day for on-site calculations and 2.59×10^5 meters/day for off-site calculations

R = high bay leak rate after the combustion period, 3.77×10^3 ft^3/day for on-site calculations and 1.36×10^5 ft^3/day for off-site calculations

V = volume of high-bay area, 3×10^5 ft^3.

3. Cloud Diffusion and Ground Deposition

a. Helium Release Through Top Shield

The helium leakage accident will release only noble gas fission products. Accordingly, there will be no deposition of activity on the ground and the only dose resulting from the accident will be the whole body immersion dose. Since this accident does not result in any damage to the high-bay ventilation system, the release is assumed to be from the high-bay roof ventilators. As a result, the initial cloud size will be quite small. In addition, since the extent of diffusion of the cloud after release will be limited at on-site locations because of the diffusion coefficients associated with the ground release accidents (see Table VI-1), the diameter of the cloud will remain relatively small as it passes over the site. Therefore, whole body immersion doses computed by assuming submersion in an infinite hemispherical cloud would provide values too conservative to permit realistic evaluation of the helium atmosphere leakage accident. As a result, values for the whole body immersion dose per curie of activity released were obtained from Holland's nomogram,* which takes credit for small cloud diameters. Values of 5×10^{-4} and 4×10^{-4} rem/curie at the nearest facility and site boundary, respectively, were obtained using meteorological and downwind distance parameters previously defined in this appendix or given in Table VI-1.

However, in the calculation of off-site doses, cloud diffusion was considered, since the cloud diameter, after traveling the three-mile distance to the nearest community, would be sufficiently large that the dose would be more accurately obtained by assuming immersion in an infinite hemispherical cloud. The total integrated exposure was calculated from Equation A-9 using the appropriate parameters defined in Appendix A, Table VI-1, and in this appendix.

b. Primary Sodium Release

The on-site cloud immersion and inhalation exposures were calculated from Equation A-8, no credit being taken for fallout. However, since two distinct models were used to describe the release of Na^{24} (during and after combustion), TIE/Q is used in the computation of internal dose, as discussed in section 4.b of this appendix.

The distance upwind of the SRE to the virtual point source was estimated from the relation

$$x_o = \left[\frac{V_o}{\pi C_y C_z \overline{u}}\right]^{1/(2-n)} , \qquad \ldots (B-7)$$

*"Meteorology and Atomic Energy," AECU 3066, United States Department of Commerce Weather Bureau (July 1955).

where values for all but V_o have been previously defined or are given in Table VI-1. The initial volume of the cloud, V_o, since this is a continuous and not a puff release, was taken as the product of the area of the smallest wall of the high bay (approximately 280 meters^2) and the distance traversed by the cloud in one second (0.5 meters).

The off-site cloud immersion and inhalation exposures were calculated from the expression

$$\text{TIE} = \frac{2Q}{\pi C^2 \bar{u} d^{2-n}} \exp\left[-\frac{h^2}{C^2 d^{2-n}}\right] \exp\left[-\frac{4V_g d^{n/2}}{\pi^{1/2} n \bar{u} C_z}\right] \text{curie-sec/m}^3, \quad \ldots(B\text{-}8)$$

where V_g is the deposition velocity in meter/sec, and all other parameters have been defined previously or given in Table VI-1.

Since a review of the literature revealed that no value for the deposition velocity for Na_2O was available, it was found necessary to choose a value. (It was not felt that this effect should be eliminated, due to the fact that, on travelling the 3-mile distance to the nearest community, certainly some significant amount of fallout of the relatively heavy Na_2O particles would result.) In order to determine the effect of this parameter, a series of calculations were performed in which V_g was varied over a range of values. These calculations indicated that, for the 3-mile distance, as the value of V_g increased, the fraction of the cloud radioactivity content falling out at that point would also increase; however, correspondingly, due to the 3-mile transport distance, the larger values for V_g substantially depleted the cloud of its radioactivity content, with the result that the amount of radioactivity remaining in the cloud became quite small. As a result of these calculations it was decided to use a value for V_g of 10^{-2} meters/sec. This value was chosen since it was not so large as to markedly deplete the cloud, yet not so small as to negate its effect.

In computing the ground concentration on-site, it was decided to assume maximum fallout from the cloud as defined by

$$\omega = \frac{nQ}{2e\pi^{1/2} C_y d^{2-(n/2)}}, \quad \ldots(B\text{-}9)$$

where all terms have been defined previously. This method of arriving at the fallout dose on-site was used since, as mentioned previously, no fallout correction was applied to the cloud transport equations for on-site dose evaluations.

In computing the ground concentration off-site, the method described by Equation A-13 was used, since it provides more realistic results when a value of V_g can be selected.

4. Total Integrated Doses

a. Helium Release Through Top Shield

As previously mentioned, the only dose resulting from the release of the reactor helium atmosphere is the whole body immersion dose. Since the plume is of very small diameter at the on-site locations investigated, the doses per curie released were obtained from Holland's nomogram.* Since nomogram values are based upon an average wind speed of one meter per second and a fission product gamma-ray energy of 0.7 Mev, the values obtained for B_o, i.e., 5×10^{-4} rem/curie released at the nearest facility and 4×10^{-4} rem/curie released at the site boundary, were corrected by the following relation:

$$B_{c_i} = B_o \sum_{ij} \left[\frac{E_{ij} k_{ij}}{0.7\bar{u}} e^{-\lambda_i d/\bar{u}} \right] \text{ rem/curie,} \qquad \ldots(B\text{-}10)$$

where B_c is the corrected value of the dose per curie and $(E_{ij} k_{ij})/0.7$, $1/\bar{u}$, and $e^{-\lambda_i d/\bar{u}}$ represent energy, wind speed, and in-transit decay corrections, respectively, for the j^{th} photon of the i^{th} isotope. Values for E_{ij} and k_{ij} were obtained from Stehn† and Strominger et al.§

Combining Equations B-2 and B-10, one obtains the following expression for the whole body immersion dose:

$$D = \sum_i B_{c_i} S_i(t) \text{ rem .} \qquad \ldots(B\text{-}11)$$

*AECU-3066, op. cit.
†J. F. Stehn, op. cit.
§D. Strominger, op. cit.

b. Primary Sodium Release

(1) Immersion Doses

The external whole body (immersion) dose either on-site or off-site, received by personnel standing at the cloud centerline during the appropriate portion of the release period, is calculated from Equation A-10, except that E_γ, the gamma energy, is 4.12 Mev per disintegration of Na^{24}. The method described in Section 4.a above was not used in this case since the size of the cloud at the release point was initially large (i.e. cloud width = building width).

(2) Ground Deposition Doses

(a) On-Site

The on-site whole body dose resultant from standing on contaminated ground is received in increments. The total dose is comprised of that received in the first 2 hours of the release plus that received in the 8-hour work periods starting 24 hours after the accident and continuing for 90 days. For the purpose of maximizing the dose from ground deposition, and considering that a 24-hour evacuation period would follow the first 2-hour exposure period, it has been assumed that the accident occurred at 8:00 A.M. on a Monday morning; should the accident occur at any time later than this, the total dose received from fallout will be less because of the short half-life of Na^{24} (15 hours).

In calculating the dose, it has also been assumed that the values for ω obtained from Equation B-9 represent the average level to which the ground area has been contaminated. Since these values actually represent the maximum deposition at a particular downwind distance, this assumption will lead to a significant over-estimate of the dose. The equation used for calculating the fallout dose is

$$D_F = \frac{2.4\Gamma}{\lambda_r} \ln\left(\frac{h_c^2 + a^2}{h_c^2}\right)\left[\omega_1\left(1 - e^{-\lambda_r t_{e_1}}\right) + \omega_2\left(1 - e^{-\lambda_r t_{e_2}}\right)\left(\sum_i e^{-\lambda_r t_i}\right)\right] \text{ rem,} \qquad \ldots\text{(B-12)}$$

where

D_F = whole body external dose from fallout, rem

Γ = gamma-ray dose rate conversion factor,* 12 r/mc-hr at 1 cm in air for 2.75 Mev photons plus 6.9 r/mc-hr at 1 cm in air for 1.37 Mev photons = 18.9 r/mc-hr at 1 cm in air

ω_1 = fallout on-site in first 2 hours, curies/meter^2

ω_2 = fallout on-site in first 24 hours, curies/meter^2

t_{e_1} = duration of initial exposure period, 0.0833 days

t_{e_2} = duration of exposure period after return to site, 0.333 days

h_c = height of critical organ above ground, 100 centimeters

a = radius of contaminated areas ($\approx$ half the width of the SRE high bay), 1.5×10^3 centimeters

t_i = waiting period before restarting exposure: 1, 2, 3, 4, 7, 8, 9, 10, 11, 14 . . . . . 90 days (weekends omitted),

and the other parameters are as previously defined.

While the contaminated area will, in all probability, be a narrow rectangular strip extending from the SRE, slight changes in wind direction could produce, instead, a wider area of contamination. In this case, the average ground contamination level would obviously be less than that predicted by Equation B-9. However, in order to be conservative, the values of ω determined from B-9 were assumed to exist in a wide area which was approximated, for purposes of computing dose, by a circle 15 meters in radius.

Although this calculation accounts for the fact that personnel are on the site for only 8 of every 24 hours, it assumes that the exposed individual will stand on the contaminated ground surface for the entire working day. Since relatively few persons at the site work out-doors for any extended period of time, the dose received from ω_2 has been reduced by 25% (i. e., to 6 hours exposure per work day). This reduction factor is probably too small, so that the result will still over-estimate the dose from fallout.

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*G. J. Hine and G. L. Brownell, Radiation Dosimetry (Academic Press, Inc., New York, 1956), Ch. 16.

including this reduction factor, Equation B-12 becomes

$$D_F = \frac{2.4\Gamma}{\lambda_r} \ln\left(\frac{h_c^2 + a^2}{h_c^2}\right)\left[\omega_1\left(1 - e^{-\lambda_r t_{e_1}}\right)\right.$$

$$\left. + 0.75\,\omega_2\left(1 - e^{-\lambda_r t_{e_2}}\right)\left(\sum_i e^{-\lambda_r t_i}\right)\right] \quad \text{rem,} \qquad \ldots(B\text{-}13)$$

where all terms are as defined previously.

It should be noted that additional conservatism is provided by the fact that no reduction of the ground concentration is taken due either to weathering or to decontamination efforts.

(b) Off-Site

The calculation of the dose from fallout for the off-site case is similar, but no reduction in the dose for occupancy period has been assumed. The off-site dose then becomes

$$D_F = \frac{2.4\Gamma\omega}{\lambda_r}\left(1 - e^{-\lambda_r t}\right)\ln\left(\frac{h_c^2 + a^2}{h_c^2}\right) \quad \text{rem,} \qquad \ldots(B\text{-}14)$$

where a, the radius of the contaminated area, is now taken as 10^5 cm because of the extent of diffusion of the cloud in travelling the distance to the nearest community; t, the exposure period, is taken as 90 continuous days; and the other parameters are as defined previously.

(3) Direct Radiation Doses

The portion of the external whole body dose contributed by direct radiation from the activity contained within the high-bay exhaust system filters and the high-bay air volume immediately after combustion terminates was estimated from

$$D_D = \frac{6(F + A)\,E_\gamma t_{e_1}}{d^2 r} \quad \text{rem,} \qquad \ldots(B\text{-}15)$$

where F is the activity contained in the filters (1.63×10^3 curies), r is the factor of reduction in the dose by the concrete walls of the high bay (two for Na^{24} photons shielded by 6 inches of concrete), and the other parameters are as defined previously. It should be noted that the rugged terrain surrounding the SRE would substantially reduce the direct dose received at other more distant facilities to much lower values than those computed from Equation B-15.

(4) Inhalation Doses

The inhalation dose, either on-site or off-site, from exposure to the cloud is obtained by integration over the appropriate exposure period and the subsequent 90 days. The critical body organ is the whole body, since the Na^{24} activity is assumed to be in the form of soluble oxides. The integrated whole body dose from internally distributed Na^{24} was calculated from the expression

$$D_B = \frac{5.12 \times 10^7 f \left[\Sigma EF(RBE)n\right] b}{m\lambda_e} \frac{TIE}{Q} e^{-\lambda_r d/\bar{u}} \left\{ q \left[\frac{1 - e^{-\lambda_r t_c}}{\lambda_r} + \frac{e^{-\lambda_e t_1}\left(1 - e^{\lambda_b t_c}\right)}{\lambda_b} \right] + \frac{A}{t_a} e^{-\lambda_r t_c}\left(1 - e^{-Rt_c/V}\right)\left[\frac{1 - e^{-\lambda_r t_a}}{\lambda_r} + \frac{e^{-\lambda t_2}\left(1 - e^{\lambda_b t_a}\right)}{\lambda_b}\right]\right\} \qquad \ldots (B-16)$$

where

D_B = inhalation dose in 90 days resultant from exposure to the cloud for a time $t = t_c + t_a$, rem

f = fraction of inhaled activity reaching the critical organ, 0.75*

$\Sigma EF(RBE)n$ = effective energy for Na^{24}, 2.7 Mev*

b = breathing rate, 5×10^{-4} cubic meters/sec

m = mass of critical organ, 7×10^4 grams*

*International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation (Pergamon Press, Inc., New York, 1959).

λ_e = effective decay constant of Na^{24}, 1.172 $days^{-1}$*

λ_b = biological decay constant of sodium, 0.063 $days^{-1}$*

t_1 = time of exposure from time $t = t_c$ to 90 days, 90 days

t_2 = time of exposure from time $t = t_a$ to 90 days, 90 days,

and the other parameters are as defined previously.

*International Commission on Radiological Protection, *op. cit.*

APPENDIX C

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INTER-OFFICE LETTERS ONLY

TO:	T. Tung	ADDRESS:	717-70 Fallbrook
FROM:	J. D. Moore	ADDRESS:	779-21 Headquarters #004
PHONE:	1148	DATE:	October 11, 1962

SUBJECT: Meteorological Data - Santa Susana

In response to your request for wind direction and velocity data recorded by the Aerovane at Santa Susana, the appended windroses covering August 1957 through July 1960 are presented. The Atomics International Santa Susana weather station is located at the SRE, the Aerovane transmitter being located due North of the reactor at an approximate elevation of 2000 feet above sea level. Compilation of Aerovane data has not been carried out since February 1961, although the original charts are available should more recent data be required. Data from July 1960 to February 1961 is unavailable due to equipment malfunction.

It is suggested that the Rocketdyne Meteorologist, Mr. H. Weiss Department 796-60, Santa Susana, DI 7-5651, Extension 5600, be contacted for additional data. Also NAA-SR-7300 may be reviewed for meteorological data relative to Santa Susana and the Nuclear Development Field Laboratory in particular.

Should further information be desired, the undersigned may be contacted at Extension 1148, Headquarters.

J. D. Moore

J. D. Moore
Laboratory Unit
Health and Safety Section

JDM:bi VXB

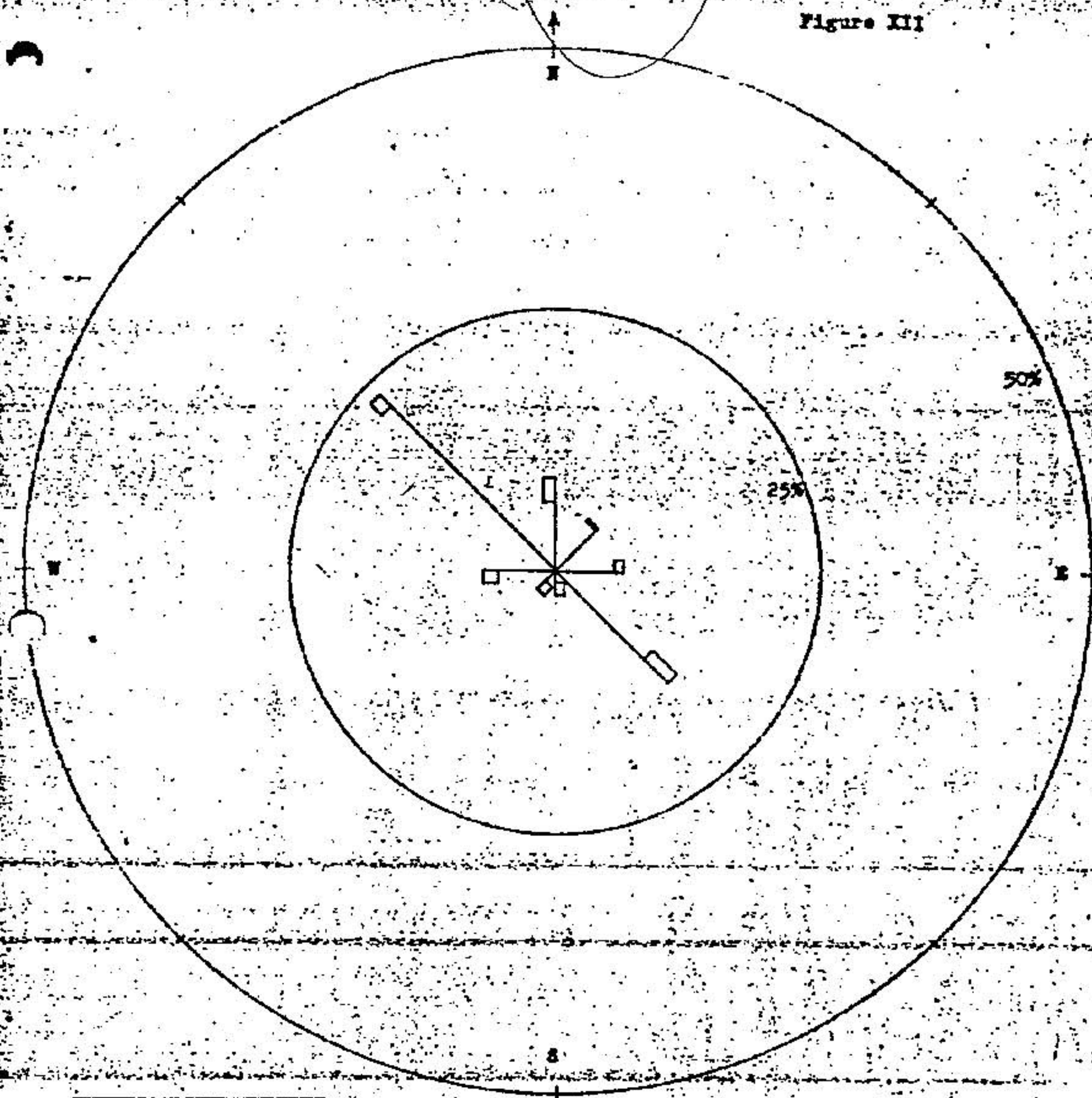
cc: W. L. Fisher w/o enclosures
P. A. Rowe " "
Central Files " "

Arc?

WIND ROSE

SRE - Annual 1959

Figure XII



LEGEND

3 -15 15-31 31+

Windspeed
in miles per hour

This data represents 43% of the total possible observations.

Calm 0-3 MPH 14%

2011776

POOR QUALITY

December 1959

HYGRO THERMOGRAPH SRE

Date	Max. Temp	Time of Max. Temp	Min. Temp	Time of Min. Temp	Average Daily Temp.	Hours Below Freezing	Precipitation (inches)
1	81	1300	64	0600	72.5	0	0
2	80	1400	60	0700	70.0	0	0
3	76	1500	56	0700	66.0	0	0
4	76	1500	56	0600	66.0	0	0
5	75	1400	58	0900	66.5	0	0
6	74	1500	57	0400	65.5	0	0
7	70	1400	58	0300	64.0	0	0
8	60	1600	55	1000	57.5	0	0
9	63	1200	52	2400	57.5	0	0
10	68	1400	49	0600	58.5	0	0
11	68	1400	48	0600	58.0	0	0
12	74	1400	52	2400	63.0	0	0
13	54	1200	42	2400	48.0	0	0
14	60	1400	42	0600	51.0	0	0
15	76	1300	54	0400	65.0	0	0
16	70	1400	50	2400	60.0	0	0
17	64	1400	48	0200	56.0	0	0
18	72	1200	52	0700	62.0	0	0
19	69	1500	60	0600	64.5	0	0
20	61	1200	51	2200	56.0	0	0
21	63	1400	49	0700	56.0	0	0
22	56	1400	49	0500	52.5	0	0
23	59	1200	50	0400	54.5	0	0
24	53	1600	50	1000	51.5	0	1.55
25	54	1400	44	2400	49.0	0	0
26	56	1400	43	0700	49.5	0	0
27	60	1200	46	0500	53.0	0	0
28	63	1200	48	0700	55.5	0	0
29	66	1200	49	0700	57.5	0	0
30	58	1300	42	2400	50.0	0	0
31	48	1200	36	2300	42.0	0	0
Ave:	65.3		50.6	58.0			
Total:						0	1.55