Armed Forces Radiobiology Research Institute

Reactor Facility

Safety Analysis Report

July 2004

Docket 50-170

License R-84

Armed Forces Radiobiology Research Institute
Bethesda, Maryland

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This Safety Analysis Report is an update of the Final Safeguards Report (dated March 1962, with subsequent revisions). This report is submitted as part of the license renewal process for Facility License No. R-84 for the Armed Forces Radiobiology Research Institute (AFRRI) TRIGA Mark-F reactor in accordance with 10 CFR 50.

1.2 AFRRI

The Armed Forces Radiobiology Research Institute is located on the grounds of the National Naval Medical Center (NNMC), Bethesda, Maryland. AFRRI is a tri-service military organization under the Uniformed Services University of the Health Sciences. The mission of AFRRI is to conduct scientific research in the field of radiobiology and related matters essential to the support of the Department of Defense. To carry out this mission, AFRRI supports and utilizes the TRIGA Mark-F reactor, research laboratories, a 50-MeV electron linear accelerator, a hot cell, office space, an animal research facility, and a [redacted] Cobalt-60 facility, all of which are contained in the AFRRI complex.

1.3 AFRRI-TRIGA REACTOR

The AFRRI-TRIGA Mark-F reactor was originally developed and installed by the General Atomics Division of the General Dynamics Corporation. The AFRRI-TRIGA reactor first achieved criticality in 1962. The reactor is an open pool-type light water reactor which can operate in either the steady-state mode up to 1 megawatt (thermal) or pulse mode with a step reactivity insertion of up to 2.8% $\Delta k/k$ (Technical Specifications limit) and utilizes standard-design General Atomics fuel elements. The AFRRI-TRIGA Mark-F reactor has the unique capability of a horizontally movable core. The movable core allows utilization of a variety of experimental facilities, such as two separate exposure rooms and the pneumatic transfer system. The reactor and associated experimental facilities and equipment are contained in the reactor building located in the AFRRI complex. A cutaway view of the reactor is given in Figure 1-1.

TRIGA reactors designed by General Atomics provide facilities for training, research, and isotope production to universities and research institutions. Many TRIGA reactors, including the AFRRI-TRIGA reactor, have been in operation for over 30 years. The first TRIGA reactor was installed in 1958. The ongoing testing program of standard TRIGA fuel elements by General Atomics and the long experience with a large number of TRIGA reactors have demonstrated the inherent safety of the family of TRIGA reactors.

On the basis of the Safety Analysis (Section 6.0) presented in this Safety Analysis Report, it can be concluded that there is reasonable assurance that the AFRRI-TRIGA Mark-F reactor can be operated at its present location without undue risk to the health and safety of the AFRRI staff or the general public.

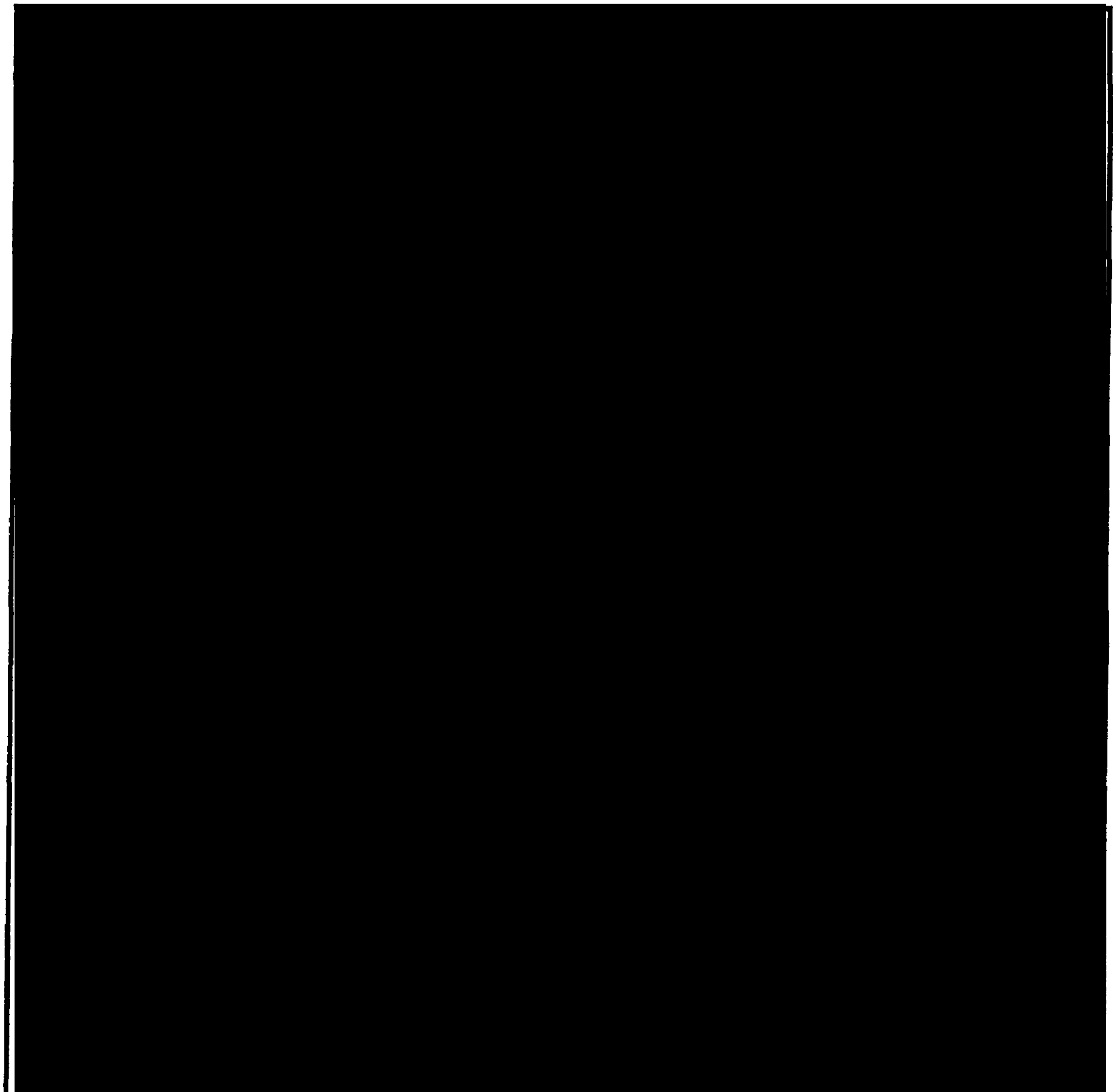

Figure 1-1.

CUT-AWAY VIEW OF THE AFRRI-TRIGA REACTOR

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2.0 SITE

2.1 LOCATION

The site of the Armed Forces Radiobiology Research Institute (AFRRI), which houses the TRIGA Mark-F reactor, is located on the grounds of the National Naval Medical Center (NNMC), Bethesda, Maryland. The coordinates of AFRRI are [redacted]. The National Naval Medical Center is approximately 3 miles north of the Washington, D.C.-Maryland line. The AFRRI site is located on a moderate slope, declining northward toward a narrow creek valley. The terrain surrounding AFRRI is rolling with elevations above sea level ranging from 230 to 320 feet.

The location of AFRRI within the National Naval Medical Center is shown on the Site Plan, Figure 2-1. The uses of each building at the NNMC are given in Table 2-1. Photographs of AFRRI from various perspectives are also provided in Figures 2-2, 2-3, and 2-4. The location of buildings at the NNMC complex, with respect to the reactor exhaust stack, is shown in Figure 2-5.

2.2 POPULATION

The National Naval Medical Center has a peak daytime population of approximately 6,798 persons which includes an estimated 1,900 visitors. The distribution of persons within the center is shown in Table 2-2 and is summarized below:

POPULATION DISTRIBUTION SUMMARY

Distance from Institute (yds)	Population
0-200	1,770
200-400	4,120
400-700	908
Total	6,798

Table 2-1

NATIONAL NAVAL MEDICAL CENTER
BUILDING INDEX

Building Number	Building Use
1	Hospital, Offices
2	Hospital, Offices
3	Hospital, Offices
4	Hospital, Offices
5	Hospital, Offices
6	Hospital, Offices
7	Hospital, Offices
8	Hospital, Offices
9	Outpatient Treatment
10	Inpatient Wards
11	Officer Quarters
12	Medical Holding Quarters
13	Laundry
14	Public Works
15	Maintenance Shops
16	Power Plant
17	Research (NMRI*)
18	Research (NMRI)
20	Fire Station
21	Animal Facilities (NMRI)
22	Flammable Storage
23	Recreation Building, Club
27	Medical Information Management Center
28	Research (NMRI)
29	Storage (NMRI)
30	Flagpole
34	Surgeon General's Quarters
35	Officers' Quarters
36	Officers' Quarters
37	Officers' Quarters
38	Officers' Quarters
39	Officers' Quarters
40	Officers' Quarters
41	Officers' Quarters
42	Reactor, Research (AFRRI)

43	Animal Research Facility (AFRRI)
44	LINAC Modulator (AFRRI)
45	Research, Tech. Support (AFRRI)
46	Research, Tech. Support (AFRRI)
47	Animal Building (AFRRI)
48	Waste Handling Facility
49	Storage
50	Enlisted Quarters
51	Gasoline Service Station
52	Navy Lodge
53	Research (NMRI)
54	Staff Parking Garage, Offices
55	Outpatient Parking Garage
56	Bowling Alley
57	Navy Exchange
58	Storage
59	Diving Lab (NMRI)
60	Enlisted Quarters
61	Enlisted Quarters
70	Uniformed Services University of the Health Sciences
71	Uniformed Services University of the Health Sciences
72	Uniformed Services University of the Health Sciences
73	Uniformed Services University of the Health Sciences
74	Flammable Storage (USUHS)
119	Research (NMRI)
125	Enlisted Transient Quarters
139	Machine Shop (NNMC), Research (NMRI)
141	Naval School of Health Sciences
143	Storage
146	Dog Run Area (NMRI)
147	Storage
148	Storage
149	Storage
152	Storage
153	Storage
154	Storage
155	Automotive Maintenance
159	Tennis Courts
160	Tennis Courts
161	Baseball Field
163	Gasoline Filling Station
165	Lunchroom (NMRI)
174	Storage (NMRI)
177	Underground Tunnel
180	Vehicle Bridge
181	Pedestrian Bridge
182	Pedestrian Bridge

188	Underground Shelter
189	Vehicle Bridge
190	Pedestrian Bridge
195	Fuel Storage Tank
196	Fuel Storage Tank
197	Heating Fuel Tank
198	Heating Fuel Tank
200	Helicopter Pad
202	Switching Station
203	Storage (AFRRI)
204	Pumphouse (AFRRI)
205	Radioactive Waste Tanks (AFRRI)
207	Gas Tanks
208	Gas Tanks
219	Dog Shelter (NMRI)
222	Switching Station (AFRRI)
225	Storage
239	Storage
241	Offices
242	Special Services
243	Electrical Transformer
244	Electrical Transformer
245	Utility Tunnel
246	Utility Tunnel
247	Bus Stop Shelter
248	Bus Stop Shelter
249	Heating Fuel Tank
250	Heating Fuel Tank
251	Environmental Health Lab
252	Cooling Tower
253	Oxygen Storage
254	Meter House
255	Meter House
256	Storage
257	Recreation Pavilion
259	Rest Rooms
260	Vehicle Bridge
Fisher Houses	Guest Quarters

* Naval Medical Research Institute

TABLE 2-2

NATIONAL NAVAL MEDICAL CENTER
POPULATION ESTIMATES

Building		Population
0-200 yards from AFRRI Stack		
Armed Forces Radiobiology Research Institute Complex (Bldgs. 42-47)		230
Naval Medical Research Institute Complex (Bldgs. 17, 18, 21, 59, 119, 139, 146, 165)		350
Uniformed Services University of the Health Sciences (Bldgs. 70-73)		500
15		15
16		25
23		510
27		80
39		10
Fisher Houses		50
	TOTAL	1,770
200-400 yards from AFRRI stack		
National Naval Medical Center Hospital Core (Bldgs. 1-10)		3,080
11		150
13		25
14		190
20		5
28		20
50		25
52		100
53		100
54		85
56		20
125		175
141		130
155		5
251		10
	TOTAL	4,120

Table 2-2 (cont)

Building	Population
400-700 yards from AFRRI stack	
Officers' Quarters complex (Bldgs. 34-38, 40-41)	45
12	85
51	15
57	310
60	208
61	240
241	5
TOTAL	908

2.3 METEOROLOGY

2.3.1 Regional Climatology

The Armed Forces Radiobiology Research Institute lies at the western edge of the Middle Atlantic coastal plain, approximately 50 miles east of the Blue Ridge Mountains and 35 miles west of the Chesapeake Bay. The site has a continental type A climate, moderated by the proximity of the Atlantic Ocean. Summers are warm and humid, and winters are relatively mild. During the winter and spring, the site lies near the principal track of storms that originate over the Gulf of Mexico and move northeast through the region. From October through June, the prevailing winds are from the northwest due to the preponderance of anticyclonic circulation over the northern portion of the country. Hence, continental polar air of Canadian origin is the predominant air mass throughout the winter. However, the Appalachian Mountains to the west, which act as a partial barrier to delay the advance of the cold air, and the relatively warm Atlantic Ocean to the east help to moderate the winter temperatures in the Washington area.

The coldest weather occurs in January, when the mean temperature is 35.2°F and the mean minimum daily temperature is 27.5°F. The lowest temperature ever recorded in the metropolitan area was -15°F, which occurred in February 1899. Temperatures of 32°F or lower normally occur on an average of 70 days annually, while temperatures of 0°F or below usually do not occur.[1]

During the summer, as the mean storm track moves far north of the Washington area, the semipermanent Atlantic High moves northward and eastward and dominates the circulation of air over the Eastern United States. Consequently, southerly winds prevail for much of the summer and transfer moist, tropical air from the Gulf of Mexico into the region.

The warmest weather occurs in July, when the mean temperature is 78.9°F and the mean maximum daily temperature is 87.9°F. The highest temperature ever recorded in the metropolitan area was 106°F, which occurred in July 1930. The temperature reaches or exceeds 90°F on an average of 38 days each year.[1]

Precipitation is spread out evenly throughout the year, averaging 39.0 inches of water equivalent. Precipitation in the summer mainly results from thunderstorms, and in the winter from rain or snow caused by the nearby passage of winter storm systems. Annual precipitation rates (in water equivalent) as low as 26.87 inches (1941) and as high as 57.54 inches (1948) have been recorded. In the entire Washington area, the maximum precipitation to occur in 1 month was 17.45 inches during September 1934, and in 24 hours was 7.31 inches during August 1928. Snow, which averages 16.4 inches a year, has occurred in measurable amounts as early as October (1940 and 1979) and as late as April (1940, 1957, 1964, and 1972). Annual snowfalls as low as 0.1 inch (1972-1973) and as high as 40.4 inches (1957-1958) have been recorded. In the entire Washington area, the maximum monthly and 24-hour snowfalls recorded were 35.2 inches during February 1899 and 25 inches during January 1922, respectively. Snowfalls of 1 inch or more occur (on the average) only 5 days each year, and significant accumulations (10 inches or more) are relatively rare due to rapid melting.[1]

Relative humidity normally averages 65 percent annually, and fog (visibility less than 1/4 mile) occurs on an average of 10 days each year.[1]

The joint wind frequency distributions presented in Appendix A were prepared using hourly National Weather Service observations taken at National Airport, Washington, D.C., for the period from January 1960 through December 1964. This period was utilized because 1964 was the last year that hourly observations were recorded and archived by the National Weather Service. The wind speed and direction were measured at a height of 7.0 meters. The stability is based on observations of wind speed, insolation, and cloud cover.[2]

The wind data show that the predominant wind direction is from the south with a frequency of 15.6 percent; winds from the northwest occur 9.4 percent of the time. This reflects the seasonal prevailing wind patterns described earlier. The most frequent stability is Pasquill type D stability, which represents neutral conditions.

2.3.2 Long-Term Diffusion Estimates

National Airport wind data from the calendar year of 1964 (Table 2, Appendix A) were used to calculate annual average relative concentrations (Table 1, Appendix B) using procedures outlined in Regulatory Guide 1.111.[3,4] The relative concentrations are required to assess possible doses that could be received by a hypothetical individual in an unrestricted area, primarily from

Argon-41, due to routine radioactive releases. Although the meteorological data was not measured on-site, it represents a conservative assessment of AFRRI diffusion conditions for the following reasons:

National Airport and the AFRRI complex are located in the same general vicinity and it would be expected that meteorological conditions measured at one location would be representative of the other; and the relative concentrations calculated are based on a full year's worth of data which includes nighttime observations. Since reactor operations at AFRRI primarily occur during daytime hours, the diffusion of radioactive gaseous effluents from AFRRI would be considerably better than the relative concentrations presented.

The relative concentrations are calculated assuming a ground level release with a building wake factor given by the 16.3-meter height of the tallest facility at the AFRRI complex.

The equation used to determine the annual average relative concentration caused by a ground level source is:

$$\overline{(X/Q)}_D = 2.032 \sum_{i=1}^{n} [NXU_i \Sigma_{z_i}]^{-1}$$

where

$$\Sigma_{z_i} = (\sigma_{z_i}^2 + 0.5H^2/\pi)^{1/2}$$

except when

$$\Sigma_{z_i} > \sqrt{3}\sigma_{z_i}$$

in which case

$$\Sigma_{z_i} = \sqrt{3}\sigma_{z_i}$$

and the variables are defined as follows:

$\overline{(\chi/Q)}_D$ is the average effluent concentration, χ, normalized by the source strength, Q, at distance, X, in a given downwind direction, D;

2.032 is $(2/\pi)^{1/2}$ divided by the width in radians of a 22.5° sector;

n is the number of hours of valid hourly data in which the wind direction occurs in the opposite sector from the sector containing the given downwind direction, D;

N is the total number of hours of valid hourly data (all wind directions);

X is the distance in meters downwind of the source;

U_i is the measured wind speed in meters per second during hour, i;

Σ_{z_i} is the vertical plume spread with the volumetric correction factor for a release within the building wake cavity, at a distance, X, for the stability class which occurred during hour, i;

σ_{z_i} is the vertical standard deviation of the materials in the plume at distance, X, for the stability class which occurred during hour, i; and

H is the maximum adjacent building height in meters either upwind or downwind from the release point.

The maximum annual average relative concentration occurs 100 yards north of the reactor stack and is 1.6×10^{-4} seconds per cubic meter. The minimum distance considered in the annual average relative concentration is 100 yards since this is the distance to the closest building which is not part of the AFRRI complex.

2.3.3 Short-Term Diffusion Estimates

The same set of wind data used for the annual average relative concentrations was used to calculate relative concentrations for potential accident consequences, using procedures outlined in Regulatory Guide 1.145, regulatory position 3.[5] These relative concentrations are required to assess possible doses, caused by Design Basis Accidents (such as fuel element cladding failures), that could be received by a hypothetical individual in an unrestricted area. The 5 percent overall site χ/Q relative concentration for a 25 meter circular exclusion area boundary is equivalent to dispersion conditions produced by Pasquill type F stability with a wind speed of 1.2 meters per second. Thus, the short-term atmospheric diffusion estimates used in Section 6.3.4, "Design Basis Accidents," represent a conservative value appropriate for evaluating dose consequences due to a potential accident at the AFRRI facility.

2.3.4 Severe Weather

Severe weather in the Washington area is mainly associated with thunderstorms, tornadoes, winter snowstorms, and hurricanes.

Thunderstorms, which occur on an average of 29 days per year, often bring sudden, heavy rains and may be accompanied by damaging winds, hail, or lightning. On June 9, 1929, a violent local thunderstorm occurred in the area with wind gusts up to 100 mph. Two hailstorms (April 1938 and May 1953) have occurred in the Washington area which have caused more than $100,000 damage.[1] There were 15 reports of hail 3/4 of an inch in diameter or larger within the 1-degree latitude-longitude square containing AFRRI during the period 1955 to 1967.[5]

Three rather destructive tornadoes have been recorded in the Washington area. Two tornadoes (April 1923 and November 1927) caused more than $100,000 in damage, and the third (April 1973) caused an estimated $15,000,000 in damage in the Fairfax, Virginia, area. Tornadoes are relatively rare. Eleven were reported within the 1-degree latitude-longitude square containing the AFRRI site during the period 1955 to 1967, giving a mean annual tornado frequency of 0.85 and a recurrence interval of 1,500 years for a tornado at the specific location of AFRRI.[6,7]

Tropical disturbances, during their northward passage, occasionally influence Washington's weather, mainly with high winds and heavy rainfall; extensive damage from wind and tidal flooding is rare. During the period from 1901 to 1963, 83 tropical storms, hurricanes, and depressions have passed close enough to the Middle Atlantic States (New York to Virginia) to affect various sections of the coast with strong winds, heavy rainfall, or high storm tides. However, of these 83 tropical disturbances, only three made landfall at the Middle Atlantic coast.[8] With the passage of Hurricane Hazel on October 15, 1954, the peak wind gust reached 98 mph, but only 1.73 inches of rainfall was recorded. Hurricane Connie, August 12-13, 1955 produced 6.60 inches of rainfall, but the peak wind gust was only 58 mph. During June 21-22, 1972, Hurricane Agnes produced 7.52 inches of rain, causing 16 deaths in the metropolitan area and $200,000,000 in damage in Virginia, Maryland, Delaware, and the District of Columbia.[1]

The "fastest mile" wind speed recorded at National Airport was 78 mph from the southeast at a 113-foot elevation on October 15, 1954 during the passage of Hurricane Hazel.[1] From 1955 to 1967 there were 29 reports of windstorms with wind speeds of 50 knots (58 mph) or greater within the 1-degree latitude-longitude square containing AFRRI.[6]

2.4 GEOLOGY

The AFRRI reactor site is "situated in the Piedmont Physiographic Province and is underlain by the Lower Pelitic schist of the Wissahickon Formation."[9] The Wissahickon schist is composed of quartz-mica schists, phyllites, and quartzites. Quartz forms the bulk of the schists, 30 to 60 percent, while sercite, biotite, and chlorite are also major constituents.

Test drillings at the AFRRI reactor site and the NNMC indicate that the saprolite consists of micaceous silt, clay, and fine-grained sand with dense to very dense consistency. The saprolite has a mottled light gray and brown or dark green appearance and extends to a depth of 35 to 45 feet. Unweathered bedrock varies from lenses and pods in the saprolite to a massive gneissic bedrock at about 45 feet.

The sorption of soil depends primarily on a large surface area of soil particles per unit volume. Since the percentage of small-sized particles (i.e., clays) is low in the weathered metamorphic rocks in the vicinity of Washington, the ion exchange capacity is also low (about 25 milliequivalents or less for 100 grams.)

In accordance with Navdocks DM-2 (December 1967) and the Uniform Building Code (1967), AFRRI is situated in an earthquake zone where the probability for seismic activity is very remote.[10] The fault line known to exist nearest AFRRI is approximately 19 miles away and extends from 30°07' N latitude, 77°26' W longitude in Montgomery County to 39°20' N latitude, 77°12' W longitude in Howard County.

2.5 HYDROLOGY

The main surface water feature at the NNMC is an unnamed second order stream which serves as storm drainage for NNMC. This stream transverses the complex from SW to NE and flows directly into Rock Creek. The watershed which drains into the unnamed stream has an area of 652 acres[11] and includes most of the NNMC grounds and nearly all its buildings. About three-fourths of the watershed is upstream of NNMC and includes commercial areas and suburban residential neighborhoods in Bethesda, Maryland.

The 100-year flood plain on-site, calculated for the level of urbanization in the area during 1975, is small due to the high slopes in close proximity to the stream. The flood plain encroaches on no buildings at NNMC but covers portions of two parking lots.[11]

Groundwater occurs in both the weathered and unweathered rock.

In weathered rock it occurs in the interstices between the soil or rock particles, and also in the openings along joints. Groundwater in the unweathered rock occurs in openings along joints or thinly laminated planes of schistosity. The complex contains no major aquifer.[11]

The silts and clays of the weathered rock have low permeability. However, openings along relief features in the weathered rock increase the permeability locally. The high permeability of the weathered rock near the water table is indicated by rock samples and loss of drilling water in this zone during test drilling. Permeability of the bedrock is low because of smaller openings along joints and lack of openings between the individual grains of the rock.

The recharge of the groundwater in the vicinity of the AFRRI reactor site is from local precipitation, which averages 39 inches a year,[1] but only a small fraction of the annual precipitation enters the zone of saturation.

The water table near the AFRRI reactor site is a subdued replica of the surface topography. The depth of the water table measured by test drillings varied from 38.8 to 41.4 feet. Seasonal variation of the water table in the Washington area is typically 5 to 7 feet.[9] Shallower water tables are associated with the channels of the on-site stream.

The movement of groundwater above the water table is through pores and openings along partings in the weathered rock and is probably in a near-vertical direction. After entering the zone of saturation, most of the groundwater probably moves along a permeable zone in the saprolite near the top of the water table, and is discharged to the on-site stream near the foot of the slope or to small wet weather seeps to the east or west of the site. Because the movement of groundwater in the bedrock is confined to fracture zones, the direction of groundwater movement may vary from the direction normal to the water table contours.

No determination of rate of movement of groundwater has been made. In general, the velocity of groundwater movement through fine-grained materials, such as clay or silt, is very low (on the order of a few feet per year). The velocity of groundwater movement in the saturation zones near the water table, where velocities are at a maximum, may range from less than a foot to several feet per day.

2.6 REFERENCES

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3. U.S. Nuclear Regulatory Commission, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors, Regulatory Guide 1.111, Revision 1 (July 1977).

4. Sagendorf, J.F., and J.T. Goll, XOQDOQ - Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations, U.S. Nuclear Regulatory Commission (September 1977).

5. U.S. Nuclear Regulatory Commission, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants, Regulatory Guide 1.145, Revision 1 (November 1982).

6. National Severe Storms Forecast Center, Severe Local Storm Occurrences, 1955-1957, ESSA Technical Memo-WBTM FCST 12, SEIS Unit Staff, Office of Meteorological Operations (1969).

7. Thom, H.C.S. "Tornado Probabilities," Monthly Weather Review, (October-December 1963).

8. Cry, G.W., Tropical Cyclones of The North Atlantic Ocean, Technical Paper No. 55, U.S. Weather Bureau (1965).

9. Woodward-Gardner & Associates, Inc., Preliminary Geotechnical Investigation - Uniformed Services University of the Health Sciences and Hospital Additions, National Naval Medical Center, Bethesda, Maryland (1974).

10. Letter from J.P. Mason, Project Director, Giffels & Rossetti, Inc., Detroit, Michigan, to H.B. Crawford, AFRRI, contained in the August 28, 1980, application for renewal of U.S. Nuclear Regulatory Commission License No. 19-08330-03.

11. Department of the Navy, Final Environmental Impact Statement for The Uniformed Services University of the Health Sciences, NNMC, Bethesda, Maryland (1975).

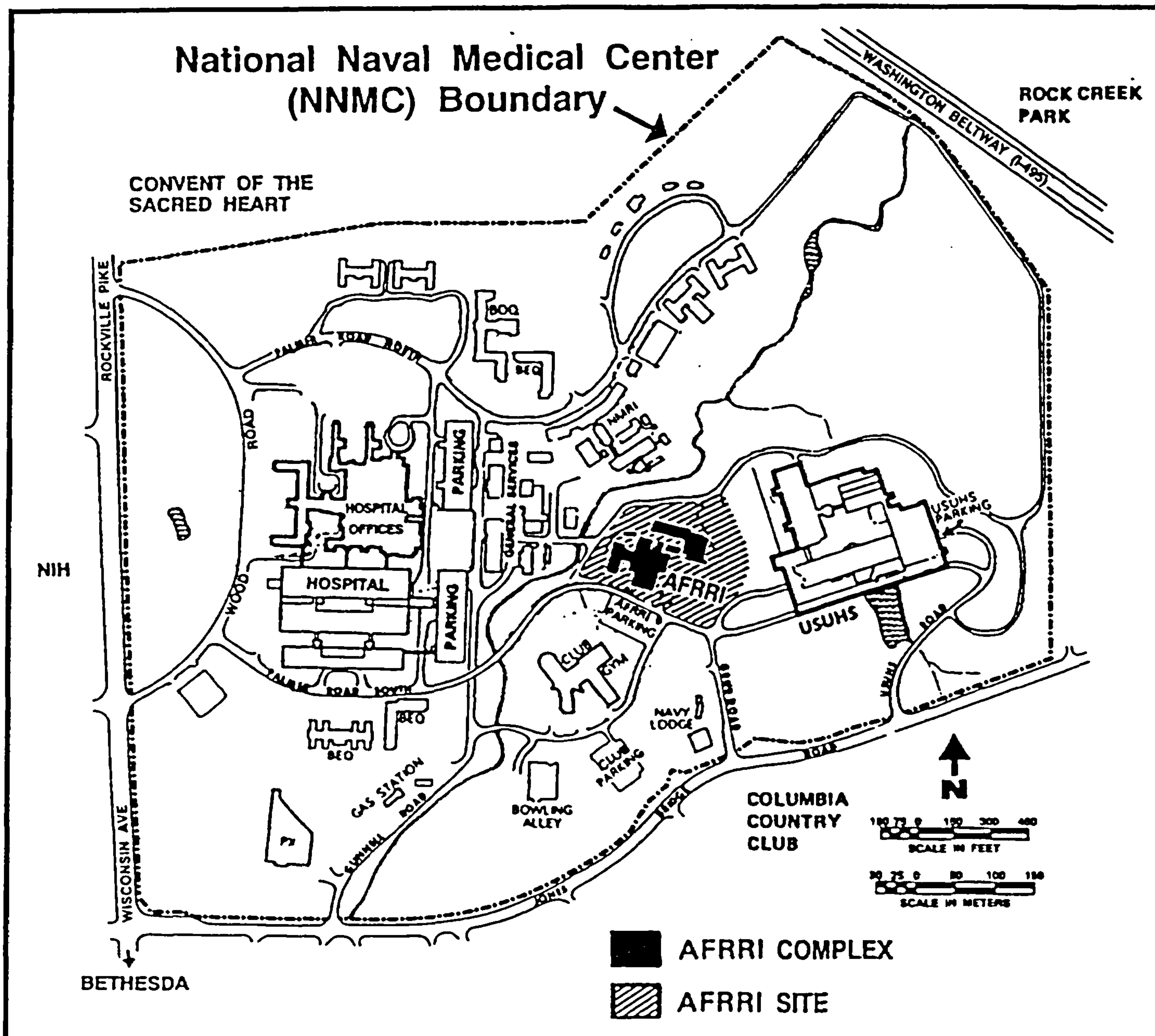

Figure 2-1.
SITE PLAN

RHS22

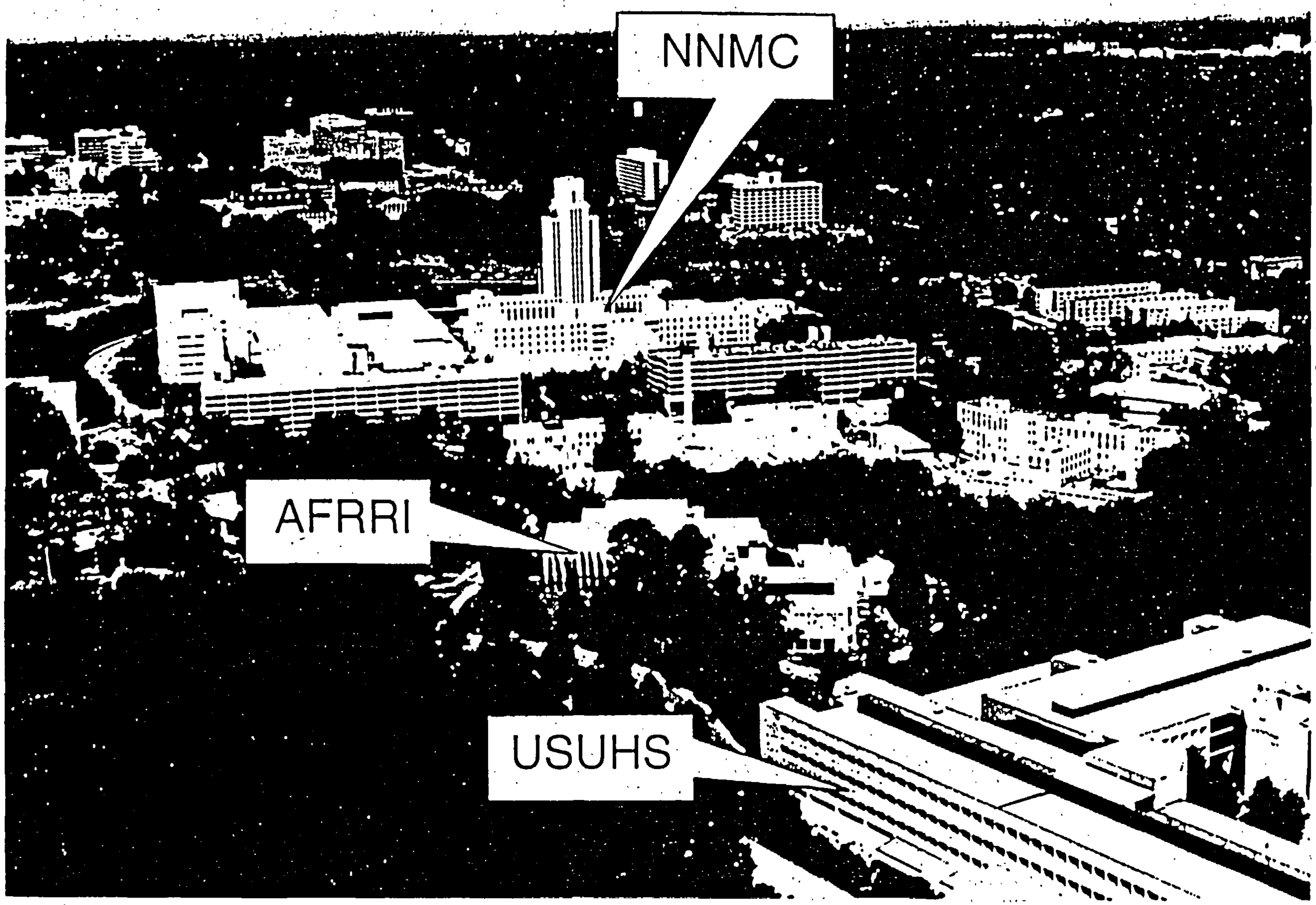

Figure 2-2

LOCATION OF THE AFRRI COMPLEX LOOKING WEST

Figure 2-3
AERIAL VIEW OF THE AFRRI COMPLEX

Figure 2-4
GROUND-LEVEL VIEW OF FRONT ENTRANCE

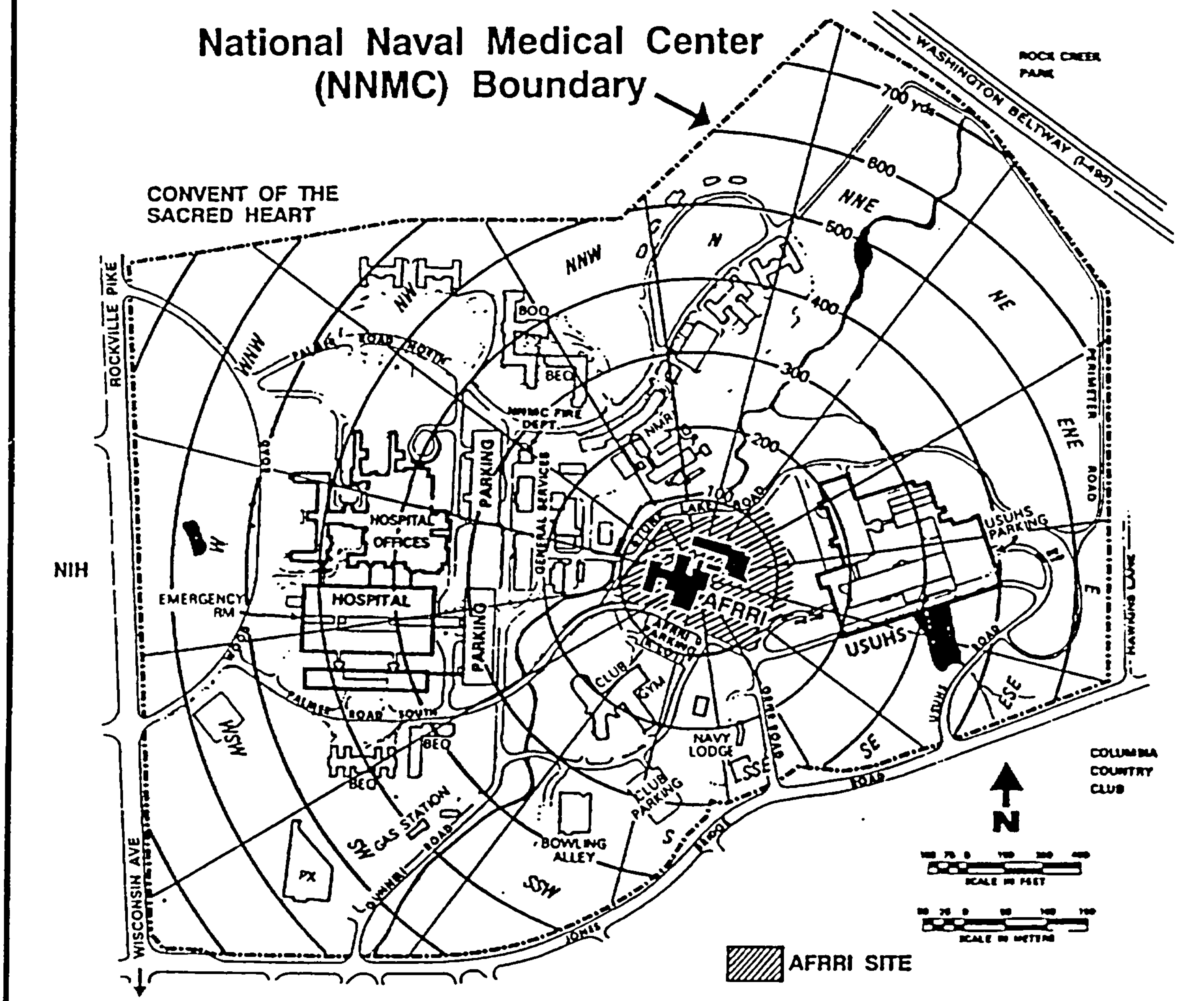

Figure 2-5.

ANNULI AND SECTORS WITH RESPECT TO THE REACTOR STACK

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3.0 FACILITY

3.1 GENERAL

The AFRRI-TRIGA Mark-F reactor is used to study the effects of neutron and gamma radiation on living organisms and instruments, and to produce radioisotopes. In addition to the reactor, the AFRRI complex also houses extensive research laboratories, a hot cell, a 50-MeV electron linear accelerator (LINAC), a [redacted] Cobalt-60 facility, office space, an animal clinical research facility, and related support areas. Various views of the AFRRI complex are shown in Figures 2-2 through 2-4. The AFRRI complex includes six separate primary buildings with several supporting facilities, as follows:

Primary Buildings	Support Structures
1. Reactor Building (#42)	7. LINAC Modulator Building (#44)
2. Laboratory Building (#42)	8. Radiological Waste Facility
3. Animal Clinical Research Facility (#43)	9. Pump House (#204)
4. Laboratory and Technical Support Building (#45)	10. Switching Station (#222)
5. Laboratory and Technical Support Building (#46)	
6. Animal Building (#47)	

The six primary buildings are arranged in an interconnected complex. Building #42, which appears from the outside to be one building, is actually two buildings: one building houses the TRIGA Mark-F reactor, and the other building contains office and laboratory space. The fact that there are six separate buildings and that the reactor is contained in its own building is important because the ventilation system in the reactor building is isolated from the ventilation systems for the rest of the AFRRI complex. The floor plan of the AFRRI complex, from the lowest level to the highest level, is shown in Figures 3-1 through 3-4, respectively. The building number designations are from Table 2-1.

Access to the entire AFRRI complex is controlled and all personnel are required to enter and exit the facility through the front and back entrances. [redacted]

An updated Physical Security Plan for the AFRRI-TRIGA Reactor Facility, prepared in accordance with 10 CFR 73 and on file with the U.S. Nuclear Regulatory Commission (Docket Number 50-170) as part of the relicensing process, details AFRRI's procedures and security measures for the physical protection of the AFRRI-TRIGA reactor and its associated equipment. As outlined in 10 CFR 2.790, the updated physical security plan contains classified information-DoD Instruction 5210.67 (Security Classification For Special Nuclear Material Information)-and is therefore deemed exempt from public disclosure.

3.2 REACTOR BUILDING

The reactor and its associated equipment are housed in a single building of reinforced concrete. This building is one of two separate buildings that are designated as Building #42. Figures 3-1 through 3-4 illustrate AFRRI's four levels, the respective working areas, and the exposure rooms.

3.2.1 Reactor Building Construction

The exterior walls of the reactor building are constructed entirely of reinforced ordinary (Portland) cement concrete with no voids in the concrete. There are no exterior windows in the reactor building. The interior of most of these walls are plastered and painted. Most interior partition walls in the reactor building are constructed of Portland cement concrete block in Portland cement mortar, and most of the walls are plastered on one side and painted.

A mat foundation under the building distributes floor and shielding loads and also provides shielding against potential soil activation. A concrete seal slab is provided under and over contained earth areas used for shielding, which might be subjected to possible activation, to prevent leaching and transport of contaminants to unrestricted areas.

The building roof construction consists of lightweight concrete poured over a corrugated steel form supported by steel roof trusses. The roof is sealed with a four-ply, hot-application, roofing compound over three to twelve inches of solid foam insulation. This insulation is sloped from the center of the roof to the corners, where the drains are located. The lines of contact between the poured light concrete roof and the reinforced concrete walls are further sealed by flashing, expansion joint material, and roofing compound.

All reactor building exterior doors are well fitted and weather-stripped with good-quality, commercial, interlocking-type stripping at the top and sides. [redacted]

3.2.2 <u>Reactor Building Ventilation</u>

The AFRRI ventilation system supplies filtered and conditioned air to the laboratories and offices of the Institute. The ventilation system is a standard industrial type that uses as supply air either 100% outside (new) air or outside air mixed with recirculating building air. The supply air is then conditioned to meet recognized standards for air quality. The air supplied to laboratories is 100% outside air; air supplied to offices is approximately 25% outside air. The supply air is filtered through a 25% roughing filter and an 85% final filter. The cleaned air is then heated or cooled to approximately 55°F. Then it is passed by air handlers through supply ducts to terminal devices, which are capable of changing the air to the temperature set by a thermostat in the area to be serviced. The entire system consists of several branches that supply and exhaust the different sections of AFRRI.

One branch is of special interest in the control of possible airborne radiation hazards that may arise from actions or operations involving the TRIGA reactor. That branch services select areas within the reactor building. Several smaller branches serve fume hoods and other sources of possible airborne radioactive contamination. The reactor branch and other branches are installed in a parallel manner, using the same supply and exhaust. However, the reactor related area is held negative with respect to other areas within the complex. The area of interest is shown in the block diagram (Figure 3-5).

The select portions of the reactor building included in the branch are the reactor room; Exposure Rooms 1, 2, and 3; the warm storage and decontamination room; and the reactor administration/control area. The branch is subdivided into three parallel legs that service these areas. These areas are supplied air from the building supply air. Air is then single-passed through each parallel leg, exhausted through a series of roughing filters, prefilters, and absolute filters, monitored, and then released via the reactor stack. All air from the reactor building branch is released through the reactor stack and is not recirculated through the other parts of the building.

Air from the building system is supplied to the legs of the reactor branch through a single supply damper. This damper (which allows air movement) is held open only as long as there is air flow in the exhaust ducts beyond the absolute filters. A flow-sensitive switch that senses a loss of exhaust flow would send a signal to close the supply damper, thus stopping the supply of air to the entire branch.

To provide isolation capabilities in the event of airborne contamination, the reactor room air ducts (both supply and exhaust) are provided with sealing dampers. When closed, these dampers restrict air movement into or out of the reactor room via the ventilation system.

These dampers may be opened or closed from two points: automatically from a radiation-sensing device sampling air in the reactor room or manually using pushbuttons in the control room. The dampers are located so that access to them can be achieved without entering the reactor room. The dampers will be a "close on fail" type to ensure a safe configuration in the event of system failure.

The location of the reactor administration/control area and its proximity to the reactor room requires air control primarily in the event of an airborne release. Since entry into the reactor room would be through this area and since action stations would be established in this area, the area is held negative with respect to the rest of AFRRI, but positive to the reactor room. During normal conditions, air is circulated parallel to the reactor room; however, during airborne radiation conditions, the exhaust damper closes with the reactor room dampers. This causes a buildup of pressure relative to the reactor room, causing leakage into that room either through the installed conduits or through the entrance door if entry becomes necessary. To allow any small overpressures from the reactor room to vent in a controlled manner after long-term shutdown of the system, a small vent line with an atmospheric relief damper runs from the reactor room to the reactor stack.

Another leg of this branch supplies air to the Exposure Rooms

(ER #1, 2, and 3) and the warm storage-decontamination area. The conditioned supply air is carried through ducts to these areas. The exhaust air is drawn through the filter bank and, along with the other legs, is exhausted through the reactor stack.

After being pulled from the various serviced areas by fan EF5 or EF6, the exhaust air is pulled through a set of absolute filters that ensure stoppage of particulate material 0.3 microns or larger with a 99.97% efficiency. The filter bank will nominally consist of prefilters, roughing filters, and absolute filters. Filters are tested upon installation to ensure that no leakage occurs. The filter buildup is measured by a differential pressure method. They are replaced when the pressure reaches the manufacturer's recommended levels. Isokinetic sampling is done on all air exiting the reactor facility, and radiological gas monitoring is then performed. Readouts and alarms for both flow and radiological levels are in the control room, and are set according to the current limits given in appropriate instructions. Normal air flow rate through the reactor stack is about 31,000 cfm.

3.2.3 Reactor Room

The reactor room (Room 3161) contains approximately 1,800 square feet of floor space, has a ceiling height of approximately 18 feet and contains the open surface of the TRIGA Mark-F reactor pool. Adjacent to the reactor room is the reactor control room (Room 3160), five offices (Rooms 3155 through 3159), and Hall 3106. The only access to these rooms is from Hall 3106 in the reactor administration/control area. Three large sealed windows in the east wall of the reactor control room allow visual observation by the reactor operator of operations within the reactor room. These windows are sealed and cannot be opened.

The only entrances to the reactor room are a pair of exterior doors in series, doors from Hall 3106 in the reactor administration/control area, a floor hatch, and a roof hatch.

The pair of exterior doors in series are sealed with compressible rubber gaskets. These doors cannot be opened from the outside, and at least one of these airlock doors is closed during reactor operation.

The double doors from the reactor room to the reactor administration/control area are sealed with a compressible rubber gasket. [redacted]

[redacted]

The reactor room floor hatch is sealed with a compressible rubber gasket and secured from within the reactor room during reactor operation.

The first set of double doors adjacent to the stairwell door, in Hall 3105, serves as the normal entrance to the controlled access area. These doors, from Hall 3105 to Hall 3106A, are [redacted] mounted near the doors. The second set of double doors, from Hall 3106A beside the third floor equipment room (3152) to Hall 3106, are sealed with a compressible rubber gasket. These doors are not left open during reactor operation. To a degree, these doors serve as backup protection in confining the reactor room air in the event that the reactor deck double doors are inadvertently opened. Additionally, during nonduty hours they are [redacted]

3.2.4 <u>Reactor Room Confinement</u>

The reactor room contains approximately 1,800 square feet of floor space, has a ceiling height of approximately 18 feet, and a volume of about 32,400 cubic feet. The reactor building construction and ventilation system design allow confinement of the reactor room air volume. The external walls of the reactor room are constructed of reinforced ordinary (Portland) concrete. The roof of the reactor room is made of ordinary concrete poured over a corrugated steel form supported by steel roof trusses. The roof is sealed with a four-ply, hot-application, roofing compound over three to twelve inches of solid foam insulation. This insulation is sloped from the center of the roof to the corners, where the drains are located. The contact surfaces between the concrete roof and the reinforced concrete external walls are further sealed by flashing, expansion joint material, and roofing compound. The interior walls of the reactor room are formed from ordinary concrete blocks in Portland cement mortar and most of the walls are plastered and painted on one side. The doors and hatches leading to the reactor room are kept closed and locked and are sealed with a compressible rubber gasket. The windows between the reactor room and the reactor control room and adjacent offices are sealed and cannot be opened. All penetrations in the walls and floor for conduits and piping are also sealed. An air conditioner has been installed in the east wall of the reactor control room. Its use is optional. In accordance with the requirement that all penetrations into the reactor room for conduits and piping be sealed, a cover for the air conditioner that seals the entire opening has been installed. This cover will only be removed if emergency cooling is required in the reactor control room.

In the event of the release of airborne radioactivity within the reactor room, radioactivity will be detected by the reactor room continuous air monitors (CAMs). The high radiation alarm set point of the reactor room CAMs will initiate automatic closure of the reactor room dampers. The reactor room is then confined. The four dampers, upon closing, actuate microswitches which complete light circuits for four indicator lights in the reactor control room. The dampers may also be closed or reset (opened) manually from the reactor control room.

In order to accommodate significant changes in barometric pressure or a change in the reactor room air temperature that could appreciably alter the pressure in the reactor room, a special normally-closed atmospheric relief valve connects the reactor room directly to the atmosphere through a pipe connected directly into the base of the reactor stack. A rise or fall in pressure sufficient to cause structural damage to the reactor room would open the atmospheric relief valve, allowing air to be expelled from the room through a pipe into the base of the reactor stack.

3.3 REACTOR WATER PURIFICATION AND COOLANT SYSTEMS

The systems of the AFRRI-TRIGA Mark-F reactor associated with the reactor coolant are the primary cooling system, the secondary cooling system, the primary water purification system, the makeup water system for the primary coolant, and the reactor pool Nitrogen-16 (N^{16}) diffuser.

The open, nonpressurized reactor pool contains approximately 15,000 gallons of light, demineralized water. Natural convection of the water in the reactor pool disperses the heat generated by the reactor core. The reactor tank water level is monitored by a float-activated switch. A drop in the reactor tank water level of ~6 inches causes an immediate reactor scram, and activates several alarms.

3.3.1 Primary Cooling System

The primary cooling system consists of the reactor tank, the primary pump, the primary side of the plate and frame heat exchanger, and associated piping, valves, and fittings, all situated at elevations above the top of the core. [redacted]

The primary pump passes the water through the primary side of the heat exchanger at a rate of approximately 350 gallons per minute (gpm). [redacted]

[redacted] In the event of significant coolant depletion (below core height), due to a reactor tank leak or rupture, an emergency fill line will be connected from an outside, adjacent fire hydrant to the primary coolant loop in order to maintain the reactor tank coolant level above the reactor core.

3.3.2 Secondary Cooling System

The secondary cooling system consists of an enclosed forced-airflow wet tower with sump, a secondary pump, the secondary side of the plate and frame heat exchanger, and associated piping, valves, and fittings. The secondary pump draws raw industrial water from the sump of the cooling tower, passes the water through the secondary side of the heat exchanger at a rate of about 700 gpm, and returns the water to the top of the cooling tower, providing the heat sink to cool the primary water. The water cascades down through the tower, where it is cooled by direct contact with the outside air, and returns to the sump at the base of the cooling tower. The cooling tower is heated in winter only enough to prevent freezing.

3.3.3 Primary Water Purification System

The water purification system removes particulates and ions from the primary cooling water. This minimizes corrosion of all reactor components, reduces activation in the primary cooling water, and maintains the optical clarity of the reactor pool. The primary water purification system consists of a purification pump; water monitor box; prefilter; two mixed-bed demineralizers; two ultraviolet lamps; and associated piping, valves, and fittings. The purification pump draws a small portion (≤20 gpm) of primary water from the return line in the primary cooling system. From the water purification pump, the water passes through the water monitor box, the prefilter, and the mixed-bed demineralizers. The water monitor box measures the conductivity, water temperature (inlet), and gamma activity of the primary water before it enters the prefilter and demineralizers. Readout of these variables is provided in the reactor control room. The prefilter removes particles larger than 5 microns in diameter from the primary water. Pressure gauges, measuring the pressure drop across the prefilter, determine when the filters need to be changed. From the prefilter, the primary water enters two parallel, mixed-bed demineralizers. The demineralizers maintain the water conductivity level at better than 1 micromho per centimeter. Based on experience, activation of the primary water at this conductivity level is not significant. Each demineralizer has a capacity of 12 gpm and contains about 3 cubic feet of nuclear-grade ion exchange resin. The conductivity, measured at the outlet from each demineralizer, is also read out in the control room. Each ultraviolet lamp has a capacity of 30 gallons/minute. The ultraviolet lamps kill any bacteria that may pass through the system. The flow rate is also measured at the outlet from each ultraviolet lamp with only local readout provided. The water is then passed to the primary cooling system return line.

A schematic of the primary and secondary cooling systems and the primary water purification system is shown in Figure 3-6.

3.3.4 Primary Water Makeup System

The primary water makeup system replaces any primary coolant water lost due to evaporation from the pool surface. The evaporation of 67 gallons of primary water results in an approximate one inch decrease in the pool water level. Depending on the season of the year, up to 80 gallons of primary water are lost each week due to evaporation. The primary water makeup system consists of a water feed line from the AFRRI water purification system and a 100 gallon holding tank connected via valving and piping to the primary water makeup line. In the event of a primary makeup water system failure, the replenishing of the evaporated reactor pool water is achieved via a straight-feed system (with valving and coupling) which contains a mixed-bed demineralizer and filter system. This system is also capable of using raw industrial water as feed water. The discharge line from the holdup tank is above the reactor pool, which allows the makeup water to be gravity-fed to the reactor pool. This makeup water line is located along the east wall of the reactor pool and discharges the makeup water above the surface of the pool.

3.3.5 N^{16} Diffuser System

The N^{16} diffuser system reduces the gamma radiation level due to N^{16} at the top of the reactor pool immediately above the core. N^{16} is a short-lived, gamma-emitting gas (~7.14 second half-life) produced by the fast neutron irradiation of oxygen in the pool water [$O^{16}(n,p)N^{16}$].

The N^{16} diffuser system consists of a pump and motor mounted on the core dolly structure and two sections of 1-1/2 inch aluminum piping. Power for the motor is controlled from the reactor control room. The two sections of aluminum piping, located inside the cylindrical core support structure, serve as inlet and outlet lines to the pump. The diffuser pump discharges water through the outlet line at approximately 70 gpm. The outlet line is designed to discharge water in a tangential direction inside the cylindrical core support structure. This imparts a swirling motion to the water in the reactor pool which breaks the large gas bubbles into smaller ones, thereby decreasing the buoyancy vector and increasing the travel time to escape the reactor pool. The N^{16} diffuser system is not required and is utilized at the discretion of the reactor operator.

3.3.6 Water Systems Instrumentation

Instrumentation in the reactor tank, primary cooling water system, and primary water purification system permits the measurement of parameters important to the safe operation of the reactor and the associated cooling system.

The level of the reactor tank water is monitored by a float-activated switch which causes an automatic reactor scram if ≈6 or more inches of water are lost below the normal pool full water level. When activated, the switch also causes a visual alarm on the reactor console and an audible (during nonduty hours) and visual alarm on the annunciator panel in Hallway 3101. That panel informs [redacted] is present so that appropriate action may be taken.

The primary cooling water temperature is measured at three locations: just above the reactor core inside the core shroud (outlet), six inches below the pool surface (bulk), and in the water monitor box of the primary water purification system (inlet). These temperatures are measured by a resistance-temperature sensing element in a bridge circuit. A water temperature readout for these three probes is provided as part of the reactor status window screen on the reactor console. When the inlet water temperature reaches or exceeds 60°C (Technical Specifications limit), the inlet water temperature rod withdrawal prevent interlock (described further in Section 4.12) prevents the further addition of positive reactivity by locking out all control rod withdrawals. This rod withdrawal prevent interlock does not prevent control rod insertion or reactor scram, however.

Primary cooling water conductivities are measured at several points by conductivity cells containing titanium electrodes in microprocessor-based circuitry with a range of 0.2 megohm-cm to 20 megohm-cm. Water conductivity is measured at three places in the primary water purification system: the water monitor box (upstream from the mixed-bed demineralizers), and at the output from each demineralizer. The monitors provide measurements to ensure the efficient operation of the mixed-bed demineralizers, as well as the overall conductivity of the pool water. Readout for the conductivity monitors is located in the control room. An audible alarm connected to the water monitor box cell readout in the control room is activated if the bulk water resistivity falls below 0.5 megohm-cm. Under the Technical Specifications, the reactor shall not be operated if the water conductivity exceeds 2.0 micromhos per centimeter (or less than 0.5 megohm-cm resistivity) at the output of the purification system averaged over a week.

The gamma activity of the water in the purification system is measured by a G-M detector in the water monitor box. The readout and visual alarm for the detector are located in the control room.

3.4 RADIOACTIVE WASTES

Radioactive wastes generated by normal reactor operation include radioactive particulates, gases, and solid wastes which consist of routine laboratory wastes and animal carcasses. There are no liquid radioactive wastes generated as a result of normal reactor operation.

The handling of radioactive waste materials is conducted under the supervision of the Safety and Health Department to ensure compliance with NRC regulations, specifically 10 CFR 20. Utilizing procedures as set forth in the AFRRI Safety and Health Department's Health Physics Procedures, the handling and disposal of radioactive waste materials is conducted without endangering the health and safety of AFRRI personnel or the general public.

3.4.1 Gaseous Radioactive Wastes

Gaseous wastes generated by the AFRRI-TRIGA Mark-F reactor are released to the environment from the reactor building ventilation system through the reactor stack. The ventilation system (Section 3.2.2) contains a filter system which has an efficiency of 99.97 percent for 0.3-micron or larger diameter particles. The filters may be removed and analyzed for release inventory calculations. Stack monitor systems (Section 3.6) measure the stack flow rate and radioactive effluents which are released to the environment to ensure that the release of Ar^{41} and other radioactive effluents meets the requirements of 10 CFR 20.

Gaseous radioactive wastes associated with AFRRI reactor operation are generally limited to N^{16} and Ar^{41}. The N^{16} is generated by fast neutron activation of oxygen [$O^{16}(n,p)N^{16}$] in the coolant water in the vicinity of the core. The Ar^{41} is generated by neutron activation of naturally-occurring Ar^{40}, a constituent of air, dissolved in the coolant water and in the air of the exposure rooms, the pneumatic transfer system, and the CET. Other gaseous wastes exhausted from the AFRRI complex but not generated by the TRIGA reactor are O^{15} and N^{13} primarily resulting from linear accelerator operations.

3.4.1.1 Nitrogen-16

The amount of gamma radiation caused by N^{16} released to the environment is insignificant due to the rapid decay (~7.14 second half-life) of N^{16} and the time involved from its production to release. During 1 MW(t) steady-state operation, the gamma radiation levels directly above the reactor pool and at the reactor pool chain were measured to be 200 mrem/hr and 14 mrem/hr, respectively. Because of the amount of gamma radiation present near the pool surface from N^{16} during large steady-state power levels, access to the chained area around the reactor pool is limited during operations above 100 kw(t). The amount of time necessary for the N^{16} to be exhausted from the reactor room into the reactor stack is sufficiently long in comparison with the 7.14 second N^{16} half-life to ensure that N^{16} gamma radiation levels will be insignificant at the release point from the reactor stack.

3.4.1.2 Argon-41

The gaseous waste of primary concern generated by the AFRRI-TRIGA reactor is Ar^{41}. The total amount of Ar^{41} released through the reactor stack to the environment was measured to be slightly less than 19 curies during all of calendar year 1979. This amount of Ar^{41} (released during 1979) was larger than the total Ar^{41} release during any year since 1972. This relative maximum in the annual Ar^{41} release totals is due to the number of reactor operations and the types of experiments performed during 1979. Based on the present experimental facility operational configuration and anticipated reactor use, it is expected that the total amount of Ar^{41} released during any year will be no more than that released during 1979 (~20 curies) and certainly no more than twice that released during 1979 (~40 curies).

To calculate the annual gamma air dose (due to the release of Ar^{41}) to an individual in an unrestricted area outside the AFRRI complex, the following equation is utilized:[1]

D^{γ} = $3.17\times10^{4}(\overline{x/Q})(Q)(DF^{\gamma})$ where:

D^{γ} = annual gamma air dose at a point outside the

AFRRI complex due to Ar^{41} (mrad/yr)

$3.17x10^4$ = number of pCi per Ci divided by the number of seconds per year

$\overline{(\chi/Q)}$ = annual average relative concentration[2] (sec/m^3)

Q = annual release rate of Ar^{41} (Ci/yr)

DF = gamma air dose factor for a uniform semi-infinite cloud of Ar^{41} (mrad-m^3/pCi-yr).

The total activity released is conservatively assumed to be 40 curies during the year and the gamma air dose factor for Ar^{41} is 9.30×10^{-3} mrad-m^3/pCi-yr[1]. The maximum annual average relative concentration is calculated in Section 2.3.2 to be 1.6×10^{-4} seconds per cubic meter at 100 yards north of the reactor stack.

The maximum integrated air dose in an unrestricted area is thus calculated to be 1.9 millirads for the year due to Ar^{41}. This value is conservative because the calculation does not consider: a reduction in the gamma dose for the actual finite cloud size at that distance; the occupancy factor for normal working hours; and the degree of shielding provided for occupants in buildings located at that distance.

The maximum allowable total effective dose equivalent for gaseous effluents released to unrestricted areas occupied by the public is 100 millirem for gamma radiation as specified in 10 CFR 20. However, every reasonable effort should be made to maintain radiation exposures and releases of radioactive materials in effluents to unrestricted areas to levels that are as low as reasonably achievable (ALARA). Design objectives for nuclear power reactors have been established in 10 CFR 50, Appendix I, to meet

ALARA. The calculated maximum annual air dose due to Ar^{41}, which makes up the majority of AFRRI gaseous radioactive wastes, is a factor of five smaller than the design objective of 10 millirads for gamma radiation given in Appendix I to 10 CFR 50. The conservative nature of both the relative concentration calculations (described in Section 2.3.2) and the gamma dose calculations provides assurance that the ALARA concept is being implemented effectively.

3.4.2 Liquid Radioactive Wastes

There are no liquid radioactive wastes generated by normal reactor operations. The liquid waste disposal system is designed for temporary holding, primarily to provide a means of controlling the release of radioactive liquid wastes from the AFRRI complex to the public sanitary sewer system. Before being released to the sewer system, the liquid wastes are sampled and analyzed to ensure compliance with 10 CFR 20. The liquid radioactive waste is divided into three subsystems or categories: "very hot," "hot," and "warm." The liquid radioactive waste system consists of seven separate above ground tanks and associated drains, piping, and valving.

The very hot liquid radioactive waste subsystem consists of one 500-gallon tank which is filled manually. The radioactive waste placed in this tank is usually too highly radioactive to be diluted and discharged into the sanitary sewer and therefore must be shipped out in a tank truck for disposal with an NRC-authorized recipient. This subsystem has rarely been utilized at AFRRI.

The hot liquid radioactive waste subsystem consists of one

5,000-gallon tank with associated drains, piping, and valving. This subsystem collects liquid effluents from areas in AFRRI in which high-level radioactive activation or contamination may occur. The drains within AFRRI from which this subsystem collects effluents are shown in Figures 3-7 through 3-10. Raw industrial water can be added to the 5,000-gallon tank to dilute the effluent. Compressed air can be applied to the tank either to enhance the mixing of the raw water and the effluent or to stir the tank prior to sampling.

The warm liquid radioactive waste subsystem consists of five 5,000-gallon tanks with associated drains, grinders, piping, and valving. This subsystem collects liquid effluents from those areas in AFRRI in which low-level radioactive activation or contamination may occur. Prior to the effluent's entry into the waste tanks, the effluent passes though grinders to ensure that all of the waste effluent is broken up into small pieces which can be thoroughly mixed with the water in the tanks. The drains within AFRRI from which this subsystem collects effluents are shown in Figures 3-7 through 3-10. Raw industrial water can be added to the tanks to dilute the effluent. Compressed air can be applied to the tanks either to enhance mixing of the raw water and the effluent or to stir the tank prior to sampling. All of these five tanks are interconnected with overflow lines so that these tanks backup one another, reducing the likelihood of a spill condition.

3.4.3 Solid Radioactive Wastes

Solid radioactive wastes consist of routine laboratory wastes, such as glassware, paper, plastics, scintillation vials, disposable gloves, and radioactive animal carcasses. As a result of reactor operations, the same type of solid radioactive wastes are generated due to the handling of experiments. The radioactive solid wastes are temporarily stored in clearly labeled containers located throughout AFRRI and are periodically removed to the Radiological Waste Facility for long-term storage and preparation for shipment.

The Safety and Health Department is responsible for the preparation of solid radioactive wastes for shipment to an NRC-authorized recipient either directly or through other military installations in accordance with 10 CFR 71 and other NRC and DOT regulations.

3.5 SPECIAL NUCLEAR MATERIAL

All special nuclear material utilized by the AFRRI-TRIGA reactor is contained in fuel elements (Section 4.9), fuel-follower control rods (Section 4.10), and small quantities in fission foils and fission chambers. The special nuclear material is used by AFRRI to support the research mission of the AFRRI-TRIGA reactor in accordance with the requirements of 10 CFR.

The majority of the fuel elements for the AFRRI-TRIGA reactor are located in the reactor core. The reactor pool provides a minimum of 14 feet of pool water for shielding these fuel elements. [redacted] The reactor pool provides a minimum of 9 feet of pool water for shielding the fuel elements in the storage racks. The storage racks permit sufficient natural convective cooling by the pool water and assure that the fuel elements will not achieve criticality. The handling of fuel elements at AFRRI is supervised by a licensed senior reactor operator.

The special nuclear material is the responsibility of the Director of AFRRI. Access to the fuel elements is possible only from the reactor room. Access to the reactor room is controlled by the reactor staff and described in Section 3.2.3. Procedures for safeguarding and ensuring the accountability of the special nuclear material are detailed in the updated Physical Security Plan for the AFRRI-TRIGA Reactor Facility and in internal procedures.

Monitoring of the special nuclear material in the reactor room is accomplished by two (i.e., one primary and one backup) gamma-sensitive radiation monitors. [redacted] These monitors will actuate audible and visual alarms at appropriate locations within the AFRRI complex. A detailed description of these radiation monitors is provided in Section 3.6. [redacted] Because of the small amounts, an accidental criticality of the material is not possible.

3.6 RADIATION MONITORING SYSTEMS

The radiation monitoring systems associated with reactor operations at AFRRI are maintained as a means of ensuring compliance with radiation limits established under 10 CFR 20. These systems consist of remote area monitors, continuous air monitors, cooling water monitors, and AFRRI perimeter monitoring. Detailed information (such as alarm setpoints for the various monitors, appropriate reactor operator responses to radiation alarms, and procedures involving monitor data evaluation and archiving) can be found in References 3 and 4.

The radiation monitoring systems associated with AFRRI reactor operations provide readouts and radiation alarms at key locations in the AFRRI complex. These locations are:

- o Reactor Room (Room 3161)
- o Reactor Control Room (Room 3160)
- o Emergency Response Center (Room 3430)
- o Annunciator panel in Hallway 3101

The radiation alarms in the reactor room and the radiation alarm readouts in the reactor control room provide the reactor operators with information necessary for the safe operation of the AFRRI-TRIGA reactor. [redacted] When reactor personnel are present in the reactor administration/control area, the audible alarm on the annunciator panel in Hallway 3101 is turned off.

3.6.1 Remote Area Monitors

The remote area monitors (RAMs) in the remote area monitoring system of primary concern to the reactor are R-1, R-2, E-3, and E-6. These units are placed in various areas of the reactor building where potential radiation hazards may exist due to reactor operation.

The monitors utilize scintillation detectors which measure gamma radiation with energies greater than 80 keV. The units have a range of 1 mrem/hr to 10^5 mrem/hr and a nominal accuracy of ±15 percent at all levels. The units have a time constant of 2 seconds and a meter and alarm response time of less than 1 second. The monitors activate radiation alarms at various locations within AFRRI; the alarm set points are variable. The monitors also activate visual alarms in the control room and the Emergency Response Center (room 3430). The RAMs are calibrated at regular intervals using a radiation source of known intensity. The locations of the RAMs, the readouts, and the audible and visual radiation alarms are given in Table 3-1 and Figures 3-11 through 3-13. The alarm setpoints can be found in AFRRI internal documents.[3,4]

3.6.2 Continuous Air Monitors

The continuous air monitors (CAMs) of primary importance to the reactor are two CAMs located in the reactor room. Three additional CAMs, which monitor the exposure rooms and the prep area, are discussed in Section 5. The CAMs provide continuous air sampling and monitoring (gross beta-gamma activity) primarily of airborne particulate matter.

The CAMs draw air with an air pump (~7 cfm) through a shielded filter assembly, which traps any particulate matter greater than

TABLE 3-1

REACTOR REMOTE AREA MONITORS

RAM	LOCATION	READOUT	RADIATION ALARM
R-1	Approximately 7 feet above the floor on the reactor room east wall	Meter in reactor control room and Emergency Response Center (Room 3430)	Activates audible and visual alarm in the reactor room and in the reactor control room; activates visual alarm in the Emergency Response Center (Room 3430); activates visual and optional audible alarm on annunciator panel in Hallway 3101
R-2	Approximately 7 feet above the floor on the reactor room west wall	Same as R-1	Activates visual alarm in the reactor control room and in the Emergency Response Center (Room 3430)
E-3	6 feet above the floor on the west wall prep area opposite ER #1 plug door	Same as R-1	Same as R-2. In addition, a visual and audible local alarm exists in the prep area near ER #1 and a red light at the front desk
E-6	6 feet above the floor on the west wall prep area opposite ER #2 plug door	Same as R-1	Same as E-3 except the visual and audible local alarm exists in the prep area near ER #2

0.3 microns in diameter. A G-M detector measures any radioactive particulates trapped by the filter. The count rate (counts per minute) is recorded by a three-cycle, logarithmic, strip-chart recorder mounted on the CAM itself. The units have a sensitivity range of 50 cpm to 50 x 10^3 cpm and a nominal accuracy of ±10 percent. The units have a time constant which is inversely proportional to the count rate, being 200 seconds at 50 cpm and 1 second at 50,000 cpm. The units have the capability of actuating alarms at two adjustable radiation levels.

The primary reactor room CAM is located in the southwest corner of the reactor room and is visible from Room 3156. The air sampled by this CAM is taken from approximately 36 inches above the reactor pool surface inside the core support structure. The air is passed through a hose to the CAM. The air is exhausted by the CAM back to the reactor room. The reactor room CAMs form an [redacted] in that when either CAM's high-level alarm is activated, the supply and exhaust dampers to the reactor room in the ventilation system are automatically closed to isolate the reactor room air volume.

The backup reactor room CAM is located along the west wall of the reactor room and its alarms are visible from the control room and Room 3158. The air sampled by this CAM is taken from a point near the warm drain located along the west side of the reactor pool. The air is exhausted by the backup CAM back to the reactor room.

A description of the CAMs' alarms, locations and read-out is

TABLE 3-2

REACTOR ROOM CONTINUOUS AIR MONITORS

CAM	LOCATION OF AIR INTAKE	READOUTS	HIGH-LEVEL ALARM	LOW-LEVEL ALARM[1]
Primary	Approximately 36 inches above reactor pool inside core carriage	Meter in reactor control room	Activates audible and visual alarm on unit itself	Activates visual alarm on the unit itself
		Strip chart recorder located on the unit itself	Activates audible and visual alarm on reactor control room annunciator panel	
			Activates visual alarm on reactor room wall panel	
			Activates audible and visual alarm on annunciator panel in Hallway 3101	
			Causes solenoid valves to vent and close reactor room ventilation dampers	
Alternate (Backup)	Near the warm drain along the west side of the reactor tank	Strip chart recorder located on the unit itself and meter in reactor control room	Identical to primary CAM alarm indications, when connected	Activates visual alarm on the unit itself

[1] If the low level alarm is being utilized

given in Table 3-2 and Figures 3-11 through 3-13. The alarm setpoints can be found in the appropriate AFRRI internal documents.[4] Additionally a flashing visual light on thereactor auxiliary instrumentation console in the reactor control room will be illuminated when either reactor room CAM is set in the TEST mode during testing.

3.6.3 Stack Monitoring Systems

The stack monitoring systems consist of the stack flow monitor and the stack gas monitor. These systems provide data about the radioactive effluents discharged through the reactor stack. The stack flow monitor measurements are recorded by a strip chart recorder. Stack gas monitor measurements of Ar^{41} emissions are recorded on a strip chart recorder and can be viewed at the end of each day by an operator to verify that no unusual Ar^{41} releases have occurred.

3.6.3.1 Stack Flow Monitoring System

The stack flow monitoring system measures the average flow rate of air exhausted through the reactor stack. The system consists of a pair of pitot tubes and Magnehelic pressure gauges which mechanically measure the dynamic pressure in the stack and produce a proportional electrical signal. A strip chart recorder located in the reactor control room records the stack flow. There are no level alarms associated with this system, except when exhaust fan EF5 fails, in which case an audible and visual alarm is activated in the reactor control room.

3.6.3.2 Stack Gas Monitoring System

The stack gas monitor (SGM) system is a NaI scintillation detection system which samples exhaust air from the reactor stack. The air is passed through a filter to remove particulates before being analyzed. This system will detect those effluents which have been released into the reactor stack, and is set to alarm at the limit currently specified in the AFRRI Reactor Emergency Plan.

The stack gas monitor system is capable of activating alarms at two levels. Additionally, a flashing visual light on the reactor auxiliary instrumentation console in the reactor control room will be illuminated when the stack gas monitoring system pump motor is turned off. The location of the system readouts and alarms are illustrated in Table 3-3. The setpoints for the radiation alarms can be found in the appropriate AFRRI internal documents.

3.6.4 Perimeter Monitoring

An environmental monitoring program is conducted by AFRRI primarily to measure environmental doses received from radionuclides produced by the AFRRI-TRIGA reactor, particularly Ar^{41}. The environmental monitoring program shall consist of an NRC/EPA approved reporting method.

3.7 SUPPORT FACILITIES

The support facilities at AFRRI consist of those systems which service the AFRRI-TRIGA Mark-F reactor and its associated facilities.

3.7.1 Electrical Power

Electrical power is supplied to AFRRI from the NNMC power system. [redacted]

[redacted]. The electrical power is supplied to the reactor building through two transformers located in the reactor building.

One transformer supplies the low voltage requirements of the reactor building, such as lighting, receptacles, and the reactor console. A voltage regulator contained within an uninterruptible power supply prevents transient power surges overloading the circuits in the reactor console.

The second transformer supplies the high voltage requirements of the reactor building, such as pump motors and the air compressor (Section 3.7.4).

3.7.2 AFRRI and Reactor Stacks

The reactor stack, located on top of Building #45 (Table 2-1), releases the air exhausted from the reactor building ventilation system (Section 3.2.2). Other ventilation systems in separate AFRRI buildings are exhausted through the main AFRRI stack. The stack release points are a minimum of 18 feet above the highest building in the AFRRI complex. The total flow rate through the reactor stack is generally about 31,000 cfm during operations.

3.7.3 Meteorological Monitoring

A wind monitor measures wind speed and direction at the AFRRI complex. The wind sensor is mounted on a tower located on the roof of Building #42. The wind direction and speed are recorded on a strip chart located in the Emergency Response Center.

3.7.4 Pressurized Air Supply System

The air supply system consists of a compressor, a refrigerated air dryer, pressure regulators, and associated piping, valves, and fittings. The air dryer removes moisture in the pressurized air supplied from the compressor. The air is then passed through pressure regulators which reduce the air pressure to the desired levels.

The air supply system provides pressurized air (~9 psi) to the reactor tank lead shield door bearings. The bearings are supplied with pressured air to minimize the ingress of water into the shield door bearing housings should the bearing seals rupture or leak. The air supply system also supplies air (~80 psi) to the transient rod system. Compressed air for the reactor room ventilation system isolation dampers (~20 psi) is supplied by the air compressor in Penthouse A on top of AFRRI building 42. To remain open, the pneumatically operated dampers need a supply of pressurized air. When the reactor room is confined, either manually by a reactor operator or automatically by a high level reactor room CAM alarm, the solenoid valve vents the supply of pressurized air to each damper, all of which then spring closed automatically.

3.7.5 Water, Sewer, and Telephone

Water, sewer, and telephone requirements are supplied to AFRRI by existing systems at the NNMC.

TABLE 3-3

STACK MONITORING SYSTEMS

System	Readout	Radiation Alarm
Stack Flow Monitoring System	Strip chart recorder in reactor control room	(Not applicable) However, EF5 failure gives audible and visual alarm in reactor control room
Stack Gas Monitoring System	Meter in reactor control room	Activates visual alarm in reactor control room

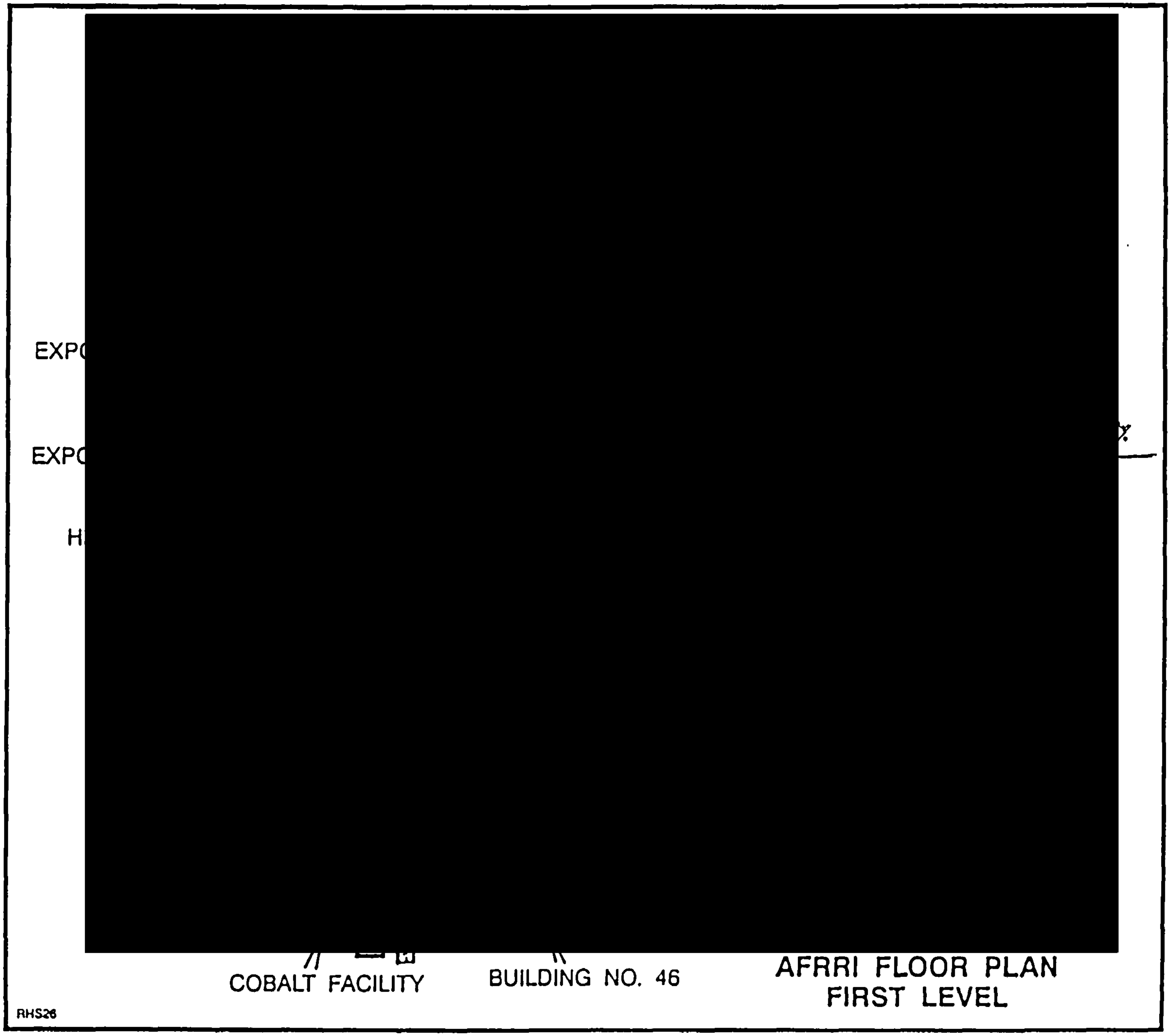
3.8 REFERENCES

1. U.S. Nuclear Regulatory Commission, Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Regulatory Guide 1.109, Revision 1 (October 1977).

2. U.S. Nuclear Regulatory Commission, Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors, Regulatory Guide 1.111, Revision 1 (July 1977).

3. Armed Forces Radiobiology Research Institute, Health Physics Procedures (HPPs), Safety and Health Department.

4. Armed Forces Radiobiology Research Institute, Reactor Operational and Administrative Procedures, Radiation Sciences Department, Reactor Division.



AFRRI FLOOR PLAN
FIRST LEVEL

[redacted]

AFRRI FLOOR PLAN
SECOND LEVEL

RHS27

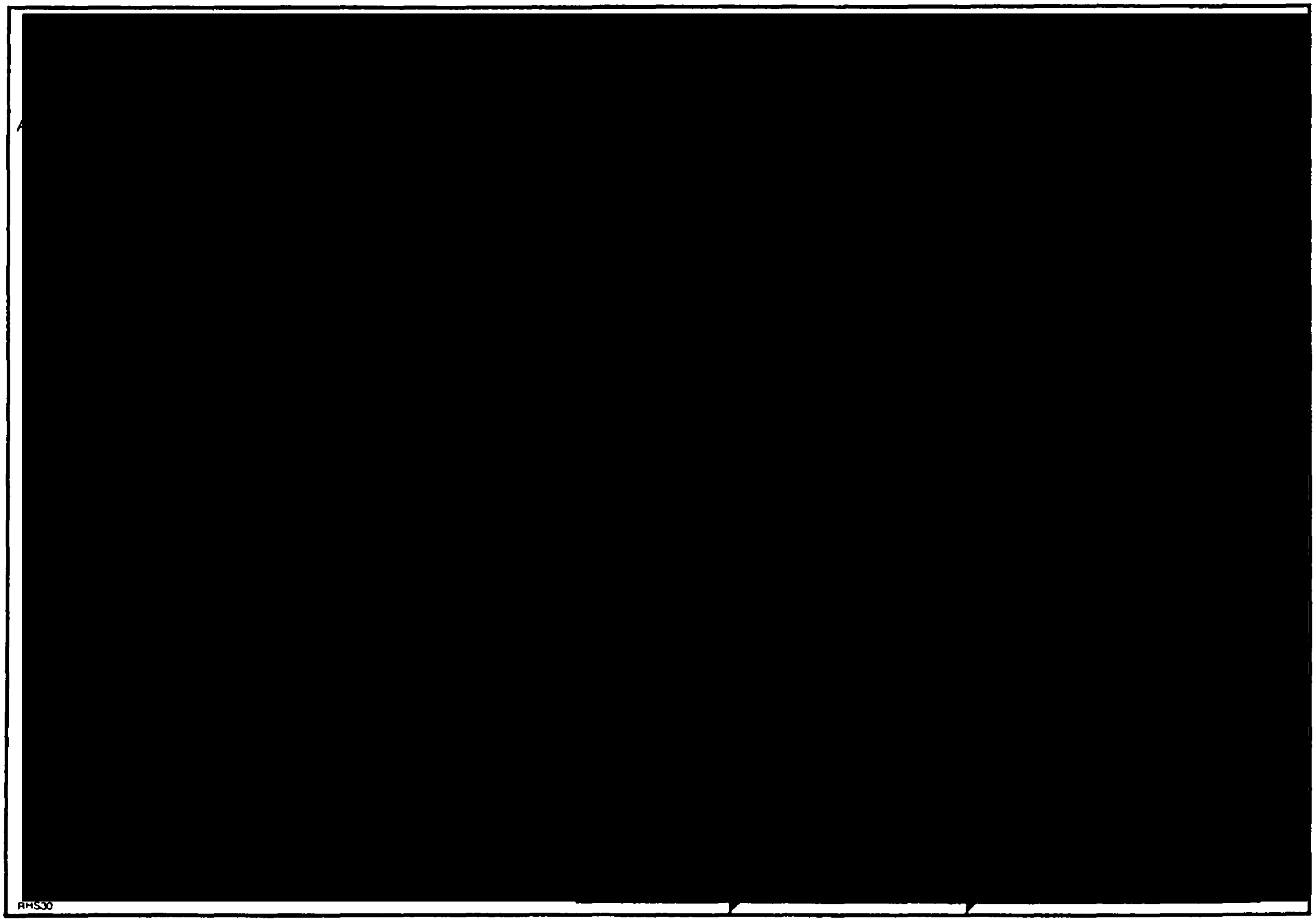
[redacted]

FRONT ENTRANCE

THIRD LEVEL

RHS28

RHS29


AHS30

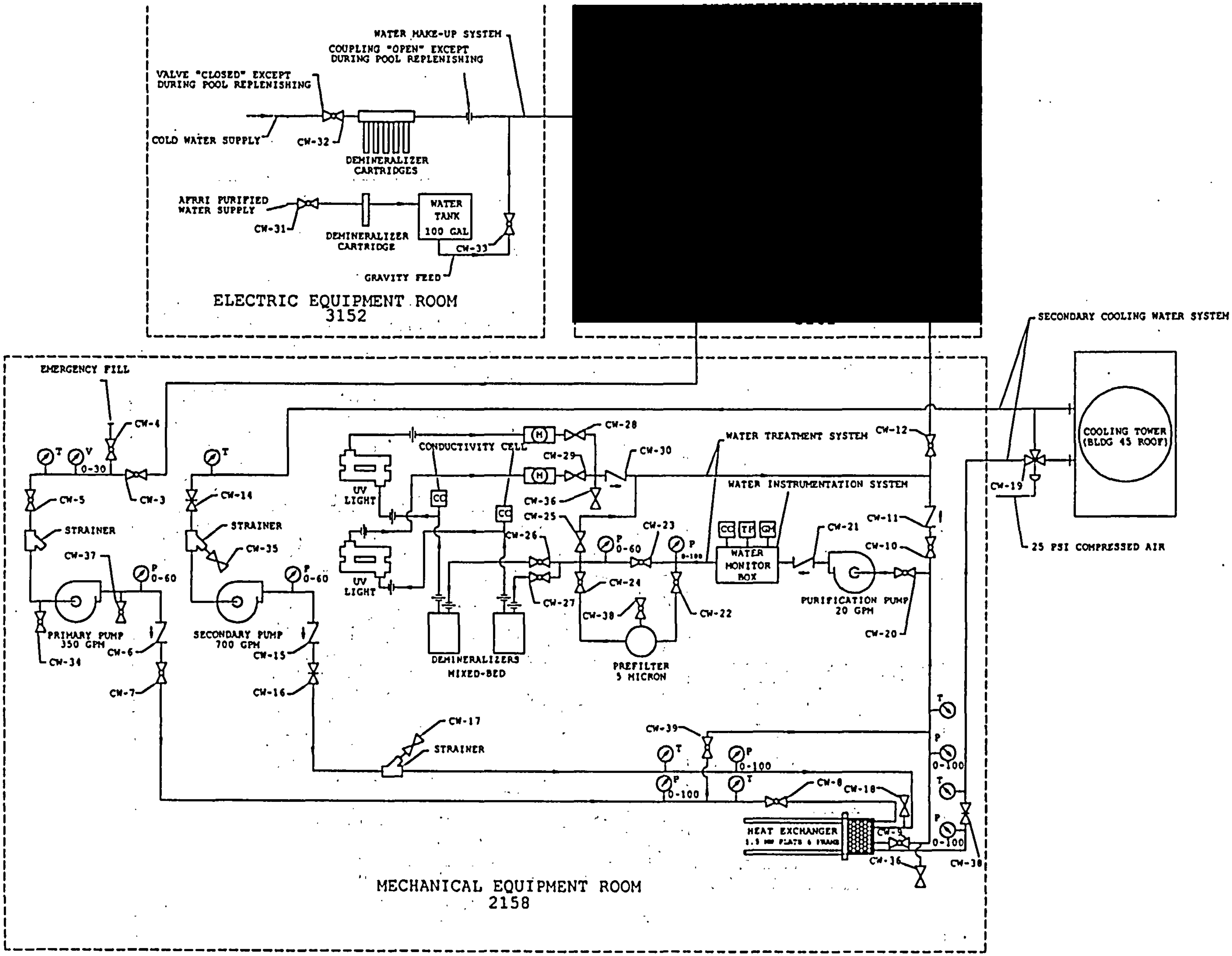


Figure 3-6

WATER PURIFICATION AND COOLING SYSTEMS

[redacted]

Figure 3-7.

LIQUID RADIOACTIVE WASTE DRAINS LOCATED ON FIRST LEVEL

RHS2

[redacted]

Figure 3-8.

LIQUID RADIOACTIVE WASTE DRAINS LOCATED ON SECOND LEVEL

RHS3

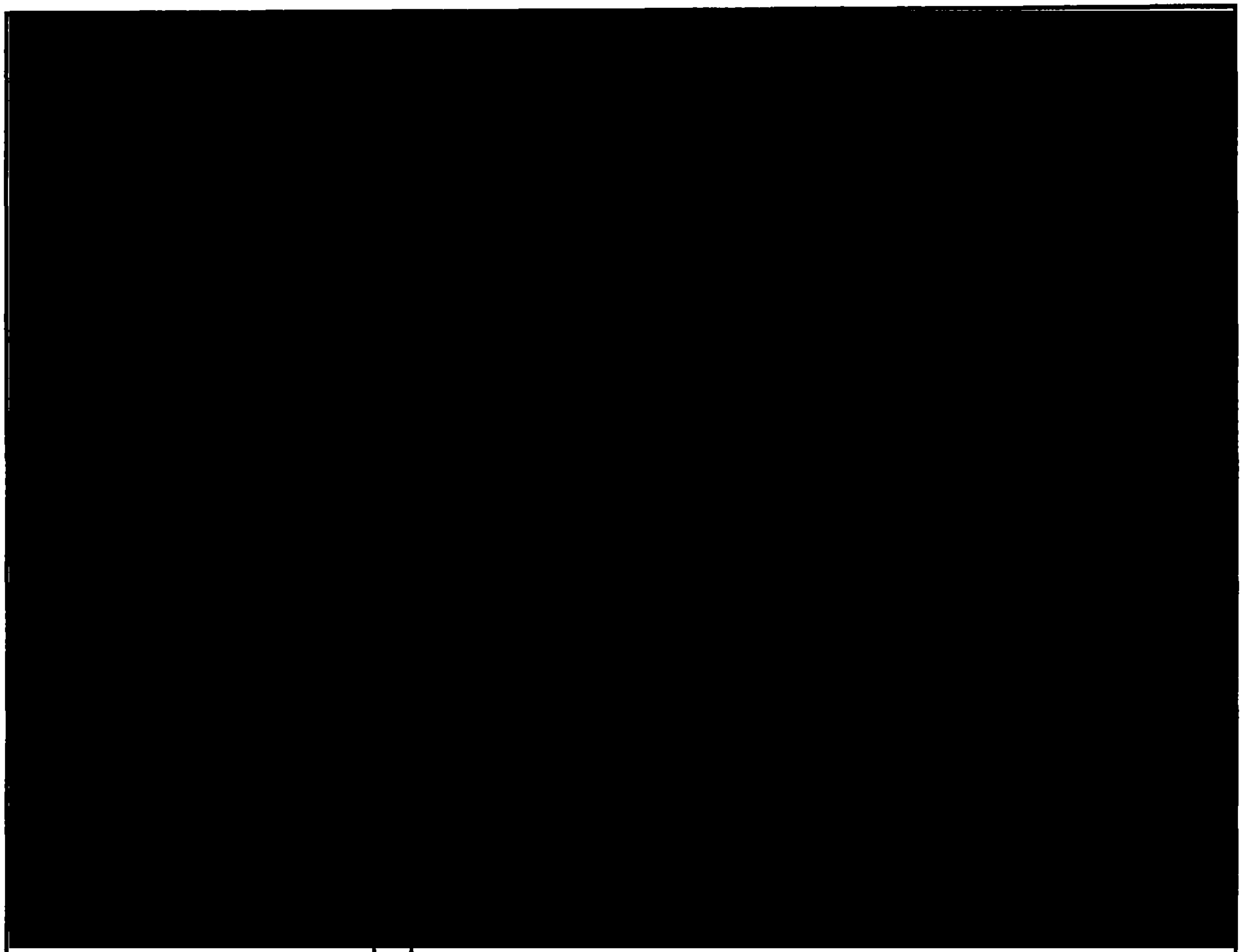


Figure 3-9.

LIQUID RADIOACTIVE WASTE DRAINS LOCATED ON
THIRD LEVEL

RHS4

Figure 3-10.

LIQUID RADIOACTIVE WASTE DRAINS LOCATED ON FOURTH LEVEL

RHSS

Figure 3-11.

AFRRI RADIATION MONITORS ASSOCIATED WITH AFRRI-TRIGA REACTOR FIRST LEVEL

RHS7

[redacted]

Figure 3-12.

AFRRI RADIATION MONITORS ASSOCIATED WITH AFRRI-TRIGA REACTOR SECOND LEVEL

RHS8

[redacted]

Figure 3-13.
AFRRI RADIATION MONITORS ASSOCIATED WITH AFRRI-TRIGA REACTOR
THIRD LEVEL

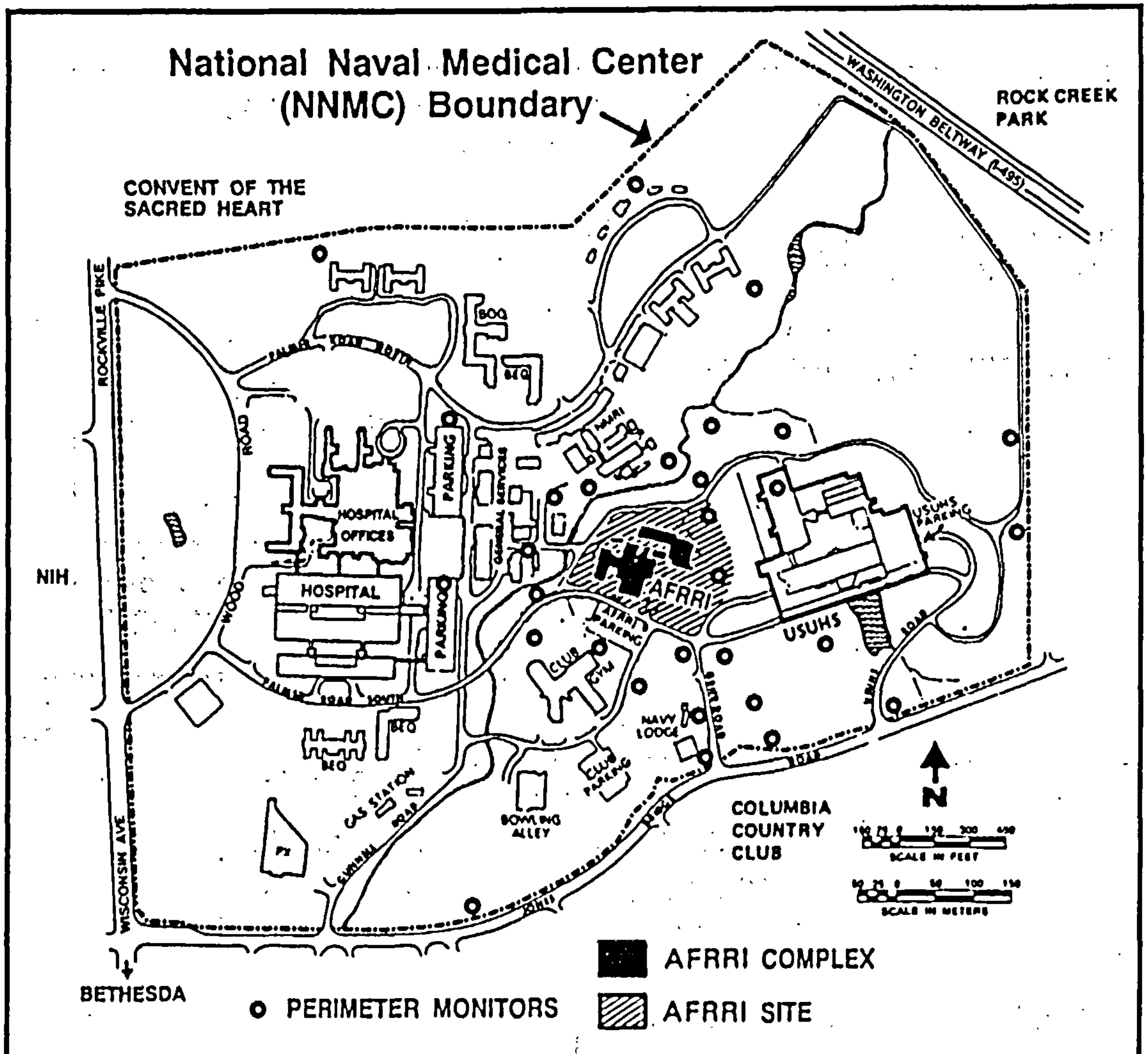


Figure 3-14.

RHS23 PERIMETER MONITORING SYSTEM ON NNMC GROUNDS

Figure 3-15.

PERIMETER MONITORING SYSTEM AT AFRRI COMPLEX

RHS25

4.0 REACTOR

4.1 GENERAL

The AFRRI-TRIGA Mark-F reactor was designed and manufactured by General Atomics and installed at AFRRI in 1962. The reactor is a tank-type light water reactor, with a horizontally movable core. The reactor tank is embedded in ordinary concrete. The cylindrical core consists of up to 87 standard TRIGA stainless steel clad cylindrical fuel elements with $U\text{-}ZrH_{1.7}$ as the fuel matrix material enriched to less than 20 percent in Uranium-235; four aluminum or stainless steel-clad borated graphite control rods with air, fuel, aluminum, or poison followers; and a startup neutron source and guide tube. The core moderator consists of both water and the zirconium hydride in the fuel. The reactor core is reflected on the top and bottom by graphite end plugs in each fuel element, and at the periphery by water. The cylindrical fuel elements and the control rods are positioned in the core in five concentric rings surrounding the centrally located transient control rod. The fuel elements and control rod guide tubes are held in place with top and bottom grid plates. The grid plates are attached to a cylindrical shroud which surrounds the core.

There are three principal and one optional experimental facilities associated with the AFRRI-TRIGA reactor. These are Exposure Room #1, Exposure Room #2, the removable in-core experiment tube (CET), and the Pneumatic Transfer System (installed as required). Experiments can also be placed either in the reactor tank or in the reactor core between the fuel elements, but these facilities are not normally used.

The AFRRI-TRIGA reactor operates in two basic modes: steady-state power levels up to 1 megawatt (thermal), and pulse operation with a step insertion of up to 2.8% $\Delta k/k$ reactivity, resulting in an average maximum pulse of 35 megawatt-seconds (integrated power).[1]

4.2 REACTOR TANK

The reactor core is positioned in the reactor tank under approximately 16 feet of light, demineralized water. The reactor tank water serves as radiation shielding, neutron moderator and reflector, and reactor coolant. The water purification and coolant systems which service the reactor pool water are discussed in Section 3.3.

The AFRRI-TRIGA reactor tank is constructed of aluminum and is embedded in ordinary concrete. The core is shielded in the radial directions by the reactor tank water and a minimum of approximately 9 feet of ordinary concrete (with the exception of the exposure rooms). The vertical shielding consists of approximately 16 feet of reactor tank water above the core and approximately 8 feet of ordinary concrete below the core separating the reactor tank from the subsoil underlying the reactor building. Aluminum was selected as the tank material to improve long-term reliability and to minimize problems of corrosion and neutron activation. The reactor tank, which is cloverleaf-shaped, is approximately 19-1/2 feet deep with a distance across the tank lobes of approximately 13 feet (Figure 4-1). The basic wall thickness of the tank is 3/8 inch, except for two cloverleaf projections that extend into the exposure rooms; this wall thickness is 1/4 inch. The cloverleaf projections which extend into the exposure rooms allow the core to face the exposure rooms over a 210° arc in the horizontal plane. The tank bottom and projection shelf thickness is ½ inch. The exposure rooms are described in Section 5.2.

Since the core can be positioned at numerous horizontal locations within the reactor tank, it is possible to create a variety of radial reflector conditions. The radial reflector materials, made up of the following combinations of water, aluminum, and lead, with the core adjacent to Exposure Rooms #1 and #2 and at the middle of the reactor tank are (respectively):

- Effectively infinite thickness of water and 3/8 inch of aluminum for 180°; 1 inch of water, 3/8 inch of aluminum, 1 inch of water, 1/4 inch of aluminum, a cadmium-gadolinium shield (Section 5.2.1), and from 0 to 6 inches of lead for 180°
- Effectively infinite water and 3/8 inch of aluminum for 180°; 1 inch of water, 3/8 inch of aluminum, 1 inch of water, and 1/4 inch of aluminum for 180°
- Effectively infinite thickness of water and 3/8 inch of aluminum for 360°.

The axial reflector, both above and below the active core, remains unchanged for each of the above cases. It consists of the graphite end plugs in the fuel elements and the pool water above and below the core region.

The reactor tank water level is monitored by a float-activated switch. A drop of approximately 6 inches in the reactor tank water level causes an immediate reactor scram and activates the following alarms:

- o Visual alarm on the reactor console
- o Audible (during nonduty hours) and visual alarm on the annunciator panel in Hallway 3101 (Section 3.6).

The reactor tank is illuminated by waterproof lights positioned along the wall of the reactor tank. Each lamp consists of a sealed-beam light enclosed in a waterproof aluminum housing and tube, suspended from the top edge of the reactor tank. As in the case of reactor piping, the sealed aluminum tubing supporting the underwater lights is bent to prevent radiation streaming from the reactor tank.

4.3 LEAD SHIELD DOORS

Two lead-filled radiation shield doors, shown in Figure 4-1, are located in the reactor pool and divide the reactor tank into two equal sections. The shield doors, when fully closed, allow access to one exposure room without significant radiation exposures while experiments are taking place in the other exposure room. The interlocking doors are constructed of ½ inch aluminum plate and 8 inch aluminum Z-sections. The doors are approximately 19 inches thick, 5 feet high, and 6 feet wide. Each watertight door is filled with approximately 18,000 pounds of #6 lead shot and approximately 90 gallons of transformer oil to fill the gaps between the lead shot. The interlocking end pieces (Z-sections) of the shield doors are stepped to prevent radiation streaming between the doors when closed. Each door is supported on a low-friction thrust bearing mounted at the bottom of the tank. Pressurized air is supplied to the shield door bearings to minimize leakage of water into the shield door bearing housing.

The shield doors may be rotated 90° to permit the core support carriage to move from one end of the reactor tank to the other. Rotation of the doors is accomplished using a fractional horsepower drive motor with slip clutch located in a small pit at the top of the reactor tank. Power for door rotation is transmitted through a set of reduction gears. Each shield door is connected to a reduction gear mounted on the side of the carriage track by a vertical shaft extending from the top of each door. Approximately three minutes are required to fully open or close the lead shield doors.

Limit switches are used to indicate the fully opened or closed positions of the shield doors. These limit switches, located on top of the reduction gears, are part of the Facility Interlock System (Section 4.13) which prevents movement of the core support carriage into the mid-pool region unless the shield doors are fully opened and also denies power to the control rod magnets unless the shield doors are either fully closed or fully opened. The position of the shield doors is indicated on the reactor console and a ceiling-mounted TV camera in the reactor room provides an overhead view of the reactor tank to the operator in the reactor control room.

4.4 SUPPORT STRUCTURES

A four-wheeled carriage, traveling on two tracks that span the reactor tank, is used to move the reactor core laterally from one operating position within the tank to another. In addition to supporting the core, the carriage also serves as a support for the four control rod drives, the N^{16} diffuser system (Section 3.5), and various electronic devices.

The carriage consists of a structural steel framework enclosed by removable aluminum covers. The carriage is approximately 64 inches square by 24 inches high. Four control rod drives are attached to a mounting plate elevated above the carriage. Elevation of the control rod drives assures adequate clearance above the pool water surface and permits direct-line access to the core components and fuel elements. The wheels on one side of the carriage are grooved to match a double-beveled track. Engagement of the wheels and track, therefore, restrains any lateral displacement of the carriage. The two wheels on the opposite side of the carriage are flat faced and roll on a flat track.

The carriage is propelled using a two-speed electric motor and a rack and pinion gear system. The gear rack is attached to the inside of the double-beveled track. The carriage is driven at two speeds, nominally 1-1/2 feet per minute and 2-1/4 feet per minute. Low speed is used during the first and last foot of travel as the carriage moves from one extreme limit of travel to the other. The intervening distance is traversed at the higher speed. Microswitches are employed to automatically change the drive motor's speed, depending on the carriage's position on the track, and to stop the drive motor when the carriage has reached either of its two extreme limits of travel. These switches also form part of the Facility Interlock System (Section 4.13). As a safety measure, mechanical stops are mounted at both ends of the carriage track to prevent carriage overtravel at the extreme limits. The carriage position is indicated on the reactor console. Precise repositioning of the carriage may require the use of reference marks on the carriage tracks. The variety of core carriage positions is broken down into three general regions. These regions are:

Region 1 - The range of positions within 12 inches of the maximum travel distance of the core dolly carriage at ER #1

Region 2 - The range of positions between Region 1 and Region 3 in which interference between the core shroud and the rotating reactor tank lead shield doors could occur

Region 3 - The range of positions within 12 inches of the maximum travel distance of the core dolly carriage at ER #2.

Movement of the core carriage is initiated from the reactor console. Travel time for the carriage from one extreme position in the reactor tank to the other is approximately 5 minutes.

Power, control wiring, and compressed air are supplied to the carriage through a trail cable and flexible hose. The cable and hose are attached to the carriage transient rod drive mounting pedestal. The trail cable and hose are supported by a wall-mounted swinging boom. The swinging boom also supports the reactor room primary continuous air monitor (CAM) inlet hose and relieves stress on wiring connections and cables.

The core support structure consists of an aluminum cylinder approximately 36 inches in diameter and 12 feet high, and an aluminum adapter 5 feet high and 19-1/2 inches in diameter (Figure 4-2). Both the cylinder and adapter are formed from 5/16 inch thick aluminum plate. The cylinder connects the adapter to the core carriage. A vertical slot 16 inches wide extends the full height of the aluminum cylinder at its north side. This slot provides access to the inside of the support structure, permitting the installation and removal of core components without raising them above the pool water level.

4.5 CORE

The core forms a right cylinder consisting of a relatively compact array of up to 87 standard TRIGA stainless steel-clad cylindrical fuel elements, four control rods and control rod guides, and a startup neutron source and guide tube. The core is enclosed in a 3/16 inch thick aluminum shroud attached to the bottom of the core support adapter. Grid plates bolted to the top and bottom of the shroud hold the fuel elements, the control rod guides, and the neutron source guide tube in place. The active (i.e., fueled) reactor core is positioned within the shroud so that its horizontal center line is approximately 29 inches above the bottom of the reactor tank. Serial numbers inscribed on the core components identify individual fuel elements and control rods.

4.6 GRID PLATES

The AFRRI-TRIGA fuel elements and other in-core assemblies are held in place in the core by grid plates (Figure 4-3) which also define the core matrix. Each grid plate is bolted to the shroud.

The upper grid plate is 18-3/4 inches in diameter, and contains ninety-one (91) 1-1/2 inch diameter holes. Four of these holes receive the guide tubes for the three standard control rods and the transient rod. The remaining 87 holes accept fuel elements and the CET. These holes, or grid spaces, are located in concentric rings. The rings are lettered A through F radially from the center. Spaces in each ring are identified numerically in a clockwise direction from a reference radial line which points in the direction of Exposure Room #1. Thus, a particular space in the grid array is identified by a letter-number combination, e.g., C-10. Eighteen (18) 1/4 inch diameter holes between the grid spaces are used for small in-core experiments and dosimetry. The upper grid plate also contains one hole near the edge of the grid array (between F-22 and F-23) for receiving the neutron source guide tube.

The lower grid plate is 16-5/8 inches in diameter and is gold-anodized to reduce grid plate wear and to aid light reflection in the core. The lower grid plate contains eighty-seven (87) 1/4 inch diameter holes to accept the fuel element bottom end fixtures, four (4) 1-1/2 inch diameter holes to accept the control rod guide tubes, and thirty (30) 5/8 inch diameter holes to permit flow of cooling water. Both grid plates are made from 3/4 inch thick aluminum plates. Each plate is bolted to the shroud by four captive screws. Correct positioning of the grid plates is ensured by two positioning dowels on the grid plate support pads.

4.7 NEUTRON STARTUP SOURCE AND HOLDER

An Americium-Beryllium (Am-Be) neutron startup source [redacted] curies) is used in the AFRRI-TRIGA reactor. The Am-Be source produces neutrons by the (α,n) reaction:

$$_{95}Am^{241} \rightarrow {}_{93}Np^{237} + {}_{2}\alpha^{4}(5.5\ \text{MeV})$$

$$_{4}Be^{9} + {}_{2}\alpha^{4} \rightarrow [{}_{6}C^{13}]^{*} \rightarrow {}_{6}C^{12} + {}_{0}n^{1}$$

*Excited state; i.e. unstable

Americium-Beryllium neutron sources are used for applications where small size and constant neutron strength over a relatively long useful life are desired. The Am-Be neutron startup source utilized in the AFRRI reactor yields approximately 2.1×10^{6} neutrons per second per curie and consists of pelletized intimate mixtures of AmO_2Be in a weight ratio of 1 to 10 and is doubly encapsulated in type 304L stainless steel.

The neutron source is inserted into the reactor core inside a 7/8 inch diameter tubular source guide tube (Figure 4-4). A hole in the upper grid plate (between F-22 and F-23) near the edge of the grid array receives the neutron source guide tube. When bolted to the upper grid plate, the guide tube projects downward alongside and just outside the fuel array. When the neutron source is inserted into the guide tube, the source material is positioned at the horizontal centerline of the core. The neutron source can be withdrawn from the guide tube for tests or storage by means of a wire attached to the top of the source holder.

4.8 IN-CORE EXPERIMENT TUBE (CET)

The CET provides users an exposure facility with a characteristic high thermal neutron flux level. The CET is primarily used for the production of radioisotopes and the activation of samples for subsequent analysis.

The CET is an air-filled guide tube of aluminum with a 1-5/16 inch inner diameter, into which sealed containers, called "rabbits," are lowered. The guide tube is formed from aluminum tubing with a nipple sealed to the lower end, which fits into the lower grid plate in a core fuel element location. The tube extends upward through the upper grid plate at the same grid array position and terminates at the core carriage. The CET has an S-bend above the upper grid plate to prevent radiation streaming. The CET may be positioned in any core fuel element location. The actual use of the CET is subject to limitations under the Technical Specifications. For example, at no time shall experiments be performed in the CET which would introduce more excess reactivity into the core than the Technical Specifications allow.

4.9 FUEL ELEMENTS

The AFRRI-TRIGA Mark-F reactor utilizes standard TRIGA stainless steel clad, cylindrical fuel elements (Figure 4-5) in which the zirconium hydride moderator is homogeneously mixed with the enriched uranium fuel. The active part of each fuel element consists of a cylindrical rod of uranium-zirconium hydride containing 8.5 weight-percent uranium with less than 20 percent U^{235} enrichment. The hydrogen-to-zirconium atom ratio of the fuel-moderator material is approximately 1.7 to 1. The nominal weight of U^{235} in each standard fuel element is [redacted]. The uranium-zirconium hydride section is approximately [redacted] in diameter. A solid zirconium rod 0.18 inch in diameter and 15 inches in length is centered in the fuel region of each fuel element to provide structural support. Graphite end plugs, 3.44 inches in length, are located above and below the fuel-moderator section as top and bottom axial reflectors.

Burnable poison (samarium) is included in each fuel element to minimize reactivity changes resulting from fission-product buildup and fuel burnup. The burnable poison is mixed with aluminum to form wafers approximately 0.015 inch thick. These wafers are located between the fuel-moderator section and the graphite reflector end plugs.

The fuel-moderator section, the two graphite end plugs, and the burnable poison wafers are contained in a Type 304 stainless steel 0.020 inch thick can. The rod is sealed at the top and bottom with stainless steel end fittings. An AFRRI-TRIGA fuel element is approximately 28.3 inches in length and weighs nearly [redacted]. Up to 87 fuel elements (86 fuel elements if the CET is in place) may be loaded into the core lattice.

In order to monitor the fuel temperature, two instrumented fuel elements (Figure 4-6) are placed in the core, one each in the

B- and C-rings. An instrumented fuel element has three chromel-alumel thermocouples embedded in the fuel-moderator section at different axial locations. The tips of the thermocouples are located near the vertical axis of the fuel section. The center thermocouple is located at the midplane of the fuel section and the other two thermocouples are located 1 inch above and below the center thermocouple. The thermocouple lead wires pass through a soft solder seal contained in the ½ inch outer diameter stainless steel tube welded to the element's top end fixture. The instrumented fuel elements are identical to the standard fuel elements in all other respects.

4.10 REACTOR CONTROL COMPONENTS

Reactor power in the AFRRI-TRIGA reactor is regulated by using three standard control rods and one transient control rod, all four of which contain neutron-absorbing material. Control rod movement within the core is accomplished using rack and pinion electromechanical drives for the standard control rods and a pneumatic-electromechanical drive for the transient control rod.

4.10.1 Standard Control Rods and Guide Tubes

The shim rod (SHIM), safety rod (SAFE), and regulating rod (REG) constitute the three standard control rods and are located in core positions D-1, D-7, and D-13, respectively (Figure 4-3). A standard control rod consists of a sealed stainless steel tube (0.020 inch thick) approximately 38 inches long and 1-1/8 inches in diameter (Figure 4-7). The upper 4.75 inches of the rod is an air filled extension section. The middle section (15 inches) of the rod contains a boron compound (25 percent free boron or boron compounds) as the neutron absorber, or poison. The lower 14.875 inches of the rod (the follower) can contain a 12 wt% U-ZrH fuel mixture with a solid zirconium rod (0.225 inch diameter) in the center, an air follower, or a solid aluminum follower. The control rod guide tubes provide space for inserting and withdrawing the control rods, pass through the upper and lower grid plates and are attached to the lower grid plate.

4.10.2 Standard Control Rod Drives

Rack and pinion electromechanical drive mechanisms (Figure 4-8), referred to as standard control rod drives, are used to change the positions of the shim, safety, and regulating rods. The standard drive consists of a stepping motor, a magnetic coupler, a rack and pinion gear system, and a potentiometer used to provide an indication of rod position, which is displayed on the reactor console. The pinion gear engages a rack attached to a draw tube supporting an electromagnet. The magnet engages an iron armature attached to the end of a long connecting rod. The connecting rod is attached at its lower end to the control rod. The magnet, its draw tube, the armature, and the upper portion of the connecting rod are housed in a tubular barrel. The barrel extends below the reactor pool water surface. The lower end of the barrel acts as a mechanical stop limiting the downward travel of the control rod assembly. A piston with a teflon seal is attached to the upper portion of the connecting rod just below the armature. Since the upper portion of the barrel is well ventilated by slotted vents, the piston moves freely in this range, but when the piston is within two inches of the bottom of its travel, its movement is dampened by a dashpot action of the graded vents in the lower end of the barrel. This dashpot action reduces bottoming impact during a scram.

When the stepping motor is energized, the pinion gear shaft rotates, thus raising the magnet draw tube. When the electromagnet, attached to the draw tube, is in contact with the armature and energized, the armature and connecting rod rise with the draw tube, withdrawing the control rod from the reactor core. When the reactor is shut down (scrammed), the electromagnet is de-energized, releasing the armature. The armature, connecting rod, and the control rod then drop by gravitational force, reinserting the neutron poison (absorber material) into the core.

A ROD DOWN microswitch indicates when the control rod is at its lower limit of travel, i.e., fully inserted into the core. When a standard control rod is fully inserted into the reactor core, a stainless steel washer beneath the armature bears against a rigid adjustable foot plate. The foot plate, which projects through a slot in the barrel, is attached to a spring-loaded pull rod that extends vertically up through the drive mount. An adjustable fixture on the upper end of the push rod engages the actuating lever of the ROD DOWN microswitch.

A second microswitch, the MAGNET UP microswitch, is used to indicate the full out position and stop the movement of the electromagnet (drive) when it reaches its upper limit of travel.

When the magnet draw tube is raised to its upper limit, the upper surface of the magnet contacts a push rod which extends vertically up through the drive mount. An adjustable fixture on the upper end of the push rod engages the actuating lever of the MAGNET UP microswitch.

A third microswitch, the MAGNET DOWN microswitch, is actuated when the electromagnet (drive) is at its lower limit. When the magnet reaches its lower limit, a rigid adjustable fixture at the upper end of the draw tube engages the actuating lever of the MAGNET DOWN microswitch which stops the movement of the electromagnet (drive).

The reactor interlock system prevents the simultaneous manual withdrawal of two or more standard control rods during steady-state modes of operation and prevents the withdrawal of any standard control rod during pulse operation.

4.10.3 Transient Control Rod

A fourth control rod, the transient rod, consists of a sealed aluminum tube of slightly larger diameter than a standard control rod (Figure 4-10). It is located in core position A-1. The upper section (15 inches) of the rod is filled with a boron compound (25 percent free boron or boron compounds).

The lower portion of the rod contains either a machined solid aluminum, combined poison and aluminum (4 percent total boron to total atom ratio), or air follower. The transient rod operates in a guide tube identical to those used for the standard control rods. Both the standard control rods and the transient rod have a maximum travel of 15 inches. When the control rods are at their upper limits of travel, the center neutron poison (absorber material) section of each rod is slightly above the fueled or active region of the core.

4.10.4 <u>Transient Control Rod Drive</u>

The AFRRI-TRIGA reactor is equipped with a pneumatic-electromechanical drive system for the transient control rod (Figure 4-9). The pneumatic-electromechanical drive, referred to as the transient rod drive, is basically a single-acting pneumatic cylinder. A piston within the cylinder is attached to the transient control rod by means of a connecting rod. The piston rod passes through an air seal at the lower end of the cylinder. For pneumatic operation, compressed air, admitted at the lower end of the cylinder, is used to drive the piston upward. As the piston rises, the air being compressed above the piston is forced out through vents at the upper end of the cylinder. At the end of its stroke, the piston strikes the anvil of a shock absorber; the piston is decelerated at a controlled rate during its final inch of travel. This action minimizes rod vibration when the piston reaches its upper limit stop. Adjustment of the anvil's position, i.e., the volume of the cylinder, controls the piston's stroke length and hence the amount of reactivity inserted during a pulse.

An accumulator tank mounted on the core support carriage stores compressed air required to operate the pneumatic portion of the transient rod drive. A solenoid valve, located in the piping between the accumulator tank and the cylinder, acts as an on/off switch controlling whether or not air is supplied to the pneumatic cylinder. De-energizing the solenoid valve vents the compressed air supply and relieves the pressure in the cylinder, allowing the piston to drop by gravity to its lower limit thereby fully inserting the transient rod into the core. This design ensures that the transient rod is fully inserted into the core except when compressed air is supplied to the cylinder and the anvil is raised above its lower limit.

The electromechanical portion of the transient rod drive consists of an electric motor, a ball-nut drive assembly, and the externally threaded air cylinder. During electromechanical operation of the transient rod, the threaded section of the air cylinder acts as the screw in the ball-nut drive assembly. These threads engage a series of balls contained in the ball-nut drive assembly of the drive housing. The ball-nut assembly is then connected through a worm gear drive to an electric motor. Therefore, the cylinder and the anvil at its upper end may be raised or lowered independently of the piston and the transient control rod by using the electric drive when compressed air is not supplied to the cylinder. Conversely, when compressed air is supplied and the electromechanical drive is raised from its lower extreme, the transient rod operates like a standard control rod.

A system of microswitches is used to indicate the position of the air cylinder (anvil) and the transient rod. Two of these switches, the (anvil) DRIVE UP and (anvil) DRIVE DOWN microswitches, are actuated by a small bar attached to the bottom of the air cylinder. This bar extends through the rod support guide to prevent rotation of the cylinder. A third microswitch, the ROD DOWN microswitch, is actuated when the piston reaches its lower limit of travel. The transient rod anvil position, measured by a potentiometer on the drive motor, is displayed on the reactor console. If the reactivity worth of the total transient rod exceeds $4.00, a mechanical stop is installed on the anvil to prevent withdrawal of the piston past the Technical Specifications pulse limit of $4.00. The mechanical stop may be removed for testing or calibration, but no pulses will be fired with the stop removed.

4.11 REACTOR INSTRUMENTATION

The AFRRI-TRIGA reactor core is monitored by a variety of detectors. One thermocouple from each of the two instrumented fuel elements comprise two of the detectors. Fission detectors, two or more ion chambers, gamma chambers, and Cerenkov detectors or additional detector devices comprise the remaining reactor detectors. These detectors are utilized to provide at least five independent "channels" which monitor the power level and fuel temperature of the core during steady-state operation and at least three independent "channels" which monitor the power level and fuel temperature of the core during pulse operations. A "channel" is the combination of a detector, interconnecting cables or lines, amplifiers, and output device(s) which measure a specific variable. The six channels utilized in the AFRRI-TRIGA reactor include: the multirange linear channel, the wide-range log channel, high flux safety channel one, high flux safety channel two, and fuel temperature safety channels one and two. Some of these channels, in addition to having readouts on the reactor console, form part of the reactor scram logic circuitry and the system of rod withdrawal prevent interlocks.

The NM-1000 system, which includes the multirange linear channel and the wide-range log channel, is contained in two National Electrical Manufacturers Association (NEMA) enclosures, one for the amplifier and one for the processor assemblies. The amplifier assembly contains modular plug-in subassemblies for pulse preamplifier electronics, bandpass filter and RMS electronics, signal conditioning circuits, low-voltage power supplies, detector high-voltage power supply, and digital diagnostics and communication electronics. The processor assembly is made up of modular plug-in subassemblies for communication electronics between amplifier and processor, the microprocessor, a control/display module, low-voltage power supplies, isolated 4-20 mA outputs, and isolated alarm outputs. Communication between the amplifier and processor assemblies is by two twisted shielded-pair cables.

4.11.1 Multirange Linear Channel

The multirange linear channel reports reactor power from source level (10^{-3} thermal watts) to full steady-state power (1.1 MW(t)). The output of the fission detector, fed through a preamplifier, serves as the channel input. The multirange linear channel consists of two circuits: the count rate circuit, and the campbelling circuit. For low power levels, the count rate circuit is utilized. The count rate circuit generates an output voltage proportional to the number of pulses or counts received from the fission detector. Hence, the output is proportional to the neutron population and the reactor power level. For higher steady-state power levels, the campbelling circuit is utilized. The campbelling circuit generates an output voltage proportional to the reactor power level by a verified technique of noise envelope amplitude detection and measurement known as campbelling. The NM-1000's microprocessor converts the signal from these circuits into ten linear power ranges. The multirange linear channel output is displayed in two formats. These are a bargraph indicator on the reactor console CRT display and a strip chart recorder located on the left-hand vertical panel of the control console. The power level as displayed on the CRT bargraph and the strip chart recorder is scaled between 0 and 100 percent for each of the ten linear power ranges. The multirange function is auto-ranged via the NM-1000 control system computer. The multirange linear output on the CRT bargraph is displayed for all steady-state modes of operation but not during pulse operation.

4.11.2 <u>Wide-Range Log Channel</u>

The wide-range log channel also measures reactor power from source level (10^{-3} thermal watts) to full steady-state power (1.1 MW(t)). The circuitry of this channel is very similar to that of the multirange linear channel. The preamplified output of the same fission detector feeding the multirange linear channel is also fed into the count rate and campbelling circuits of the wide-range log channel. The outputs of these two circuits are digitally combined and processed to provide the power rate of change (period) and the power level indicated on a ten decade log scale (10^{-3} watts(t) to 1.1 MW(t)). The wide-range log and period are both displayed on bargraph indicators on the reactor control CRT and on hardwired vertical LED bargraphs on the left side of the reactor control console. The outputs on the CRT bargraphs are displayed for all steady-state modes of operation, but not during pulse operation. The wide-range log channel output is also displayed on a strip chart recorder on the left side of the control console.

The wide-range log channel forms part of the rod withdrawal prevent interlock system. The channel activates variable set point bistable trips in the rod withdrawal prevent interlock system (Section 4.12) if source level neutrons (10^{-3} watts(t)) are not present, if the reactor power level is above 1 kilowatt(t) when switching to the pulse mode, if a steady-state power increase has a period of 3 seconds or faster during certain steady-state modes, or if high voltage is not supplied to the fission detector.

4.11.3 High Flux Safety Channels One and Two

High flux safety channels one and two report the reactor power level as measured by independent power monitors (ion chambers or Cerenkov detector) placed above the core.

High flux safety channels one and two are independent of one another but operate in identical manners during steady-state operation. Each channel consists of a detector placed above the core and the associated electronic circuitry. The steady-state power level, as measured by the two high flux safety channels, is displayed on two separate bargraphs located on the reactor console.

During pulse operation, high flux safety channel one is shunted and the sensor for high flux safety channel two is switched to a third, independent detector. High flux safety channel two measures the peak power level during the pulse (NV channel). The NV and NVT channel outputs are displayed on two separate bargraph indicators located on the console and on the pulse output display screen.

The scram function of each safety channel is checked automatically when the operator initiates the prestart checks by activation of the PRESTART CHECKS button on the control console. Any failure detected during the prestart checks will be automatically reported to the operator via the reactor status CRT screen. These prestart checks can only be performed while the reactor is SCRAMMED. Loss of high voltage to any of the ion chambers also causes an immediate reactor scram.

The high flux safety channels form part of the scram logic circuitry. When the steady-state reactor power level, as measured by either high flux safety channel, reaches maximum power levels specified in the Technical Specifications, a scram logic circuit is activated causing an immediate reactor scram. Similarly, when the reactor power level during pulse operation, as measured by high flux safety channel two, reaches the maximum pulse power level specified in the Technical Specifications, a scram logic circuit is activated which causes an immediate reactor scram. A trip test switch for each safety channel, located on the reactor console, adds additional current to the channel and provides a means of testing the scram logic circuit without actually exceeding 110 percent of authorized power. Checks and calibrations of these channels are performed periodically in accordance with the Technical Specifications.

4.11.4 Fuel Temperature Safety Channels

Fuel temperature safety channels one and two are independent of one another but operate in identical manners. One thermocouple from each of the two instrumented fuel elements, one in the B-ring and one in the C-ring, provides input to fuel temperature safety channels one and two, respectively. The two fuel temperature signals are amplified and displayed on both the reactor console CRT screen and on two separate bargraphs located on the reactor console. The fuel temperature safety channels have internal compensation for the chromel-alumel thermocouples and high noise rejection. The channels also have a scram test button for checking channel operation. Engaging this scram test results in an immediate reactor scram.

In addition to providing information to the reactor operator on fuel temperature, the fuel temperature safety channels also form part of the scram logic circuitry. When the fuel temperature, as measured by either fuel temperature safety channel, reaches the maximum allowable fuel temperature specified in the Technical Specifications, the scram logic circuit is activated causing an immediate reactor scram. The operational fuel temperature limit is usually set below the Technical Specifications limit of 600°C to assure an adequate degree of reactor protection.

The scram function of the fuel temperature channels is checked automatically when the reactor operator initiates the prestart checks by activation of the PRESTART CHECKS button on the reactor console. Any failure detected during the prestart checks will be automatically reported to the operator via the reactor status CRT screen. Checks and calibrations of these channels are performed periodically in accordance with the Technical Specifications.

4.12 <u>ROD WITHDRAWAL PREVENT INTERLOCKS</u>

A Rod Withdrawal Prevent (RWP) interlock stops any upward motion of the standard control rods and prevents air from being supplied to the transient control rod unless specific operating conditions are met. An RWP interlock, however, does not prevent a control rod from being lowered or scrammed. Therefore, any RWP interlock prevents any further positive reactivity from being inserted into the core until specific conditions are satisfied.

The system of RWP interlocks prevents control rod withdrawals under the following circumstances:

- o RWP prevents air from being applied to the transient rod unless the reactor power level is under 1 kilowatt(t)
- o RWP prevents any control rod withdrawal unless, as a minimum, source level neutrons (10^{-3} thermal watts) are present
- o RWP prevents any further control rod withdrawal unless the power level is changing on a 3-second or longer period as measured by the wide-range log channel during

certain steady-state operations

- RWP prevents any control rod withdrawal unless high voltage is being supplied to the fission detector for the multirange linear and wide-range log channel
- RWP prevents any control rod withdrawal unless the inlet water temperature is less than 60°C (Technical Specifications limit).

4.13 FACILITY INTERLOCK SYSTEM

The facility interlock system is designed to eliminate the possibility of accidental radiation exposure of personnel working in the exposure rooms or the preparation area, and to prevent interference (i.e., contact or impact) between the reactor tank lead shield doors and reactor core shroud. These interlocks prevent rotation (i.e., opening or closing) of the reactor tank shield doors and the operation and movement of the reactor core between different regions unless specific operating conditions are satisfied.

The system logic depends on the positions of the reactor core, the reactor tank shield doors, and the plug doors to the exposure rooms. The reactor core positions are classified into three regions:

Region 1 - The range of positions within 12 inches of the maximum travel distance of the core dolly carriage at ER #1

Region 2 - The range of positions between Region 1 and Region 3, in which interference between the core shroud and the rotating reactor tank shield doors could occur

Region 3 - The range of positions within 12 inches of the maximum travel distance of the core dolly carriage at ER #2.

The facility interlock system prevents the following operations unless the specific conditions given are satisfied:

- o The reactor tank shield doors cannot be closed unless the reactor core is in either Region 1 or Region 3
- o The reactor tank shield doors cannot be opened unless the reactor core is in either Region 1 or Region 3
- o The reactor tank shield doors cannot be opened to allow movement of the reactor core through Region 2 to an exposure room unless a warning horn has been sounded in that exposure room or unless two licensed reactor operators have visually inspected the exposure room to ensure that, prior to closing the exposure room plug door, no condition exists that is hazardous to personnel
- o The reactor cannot be operated unless the reactor tank shield doors are either fully opened or fully closed
- o The reactor cannot be operated within Region 1 unless the plug door to Exposure Room #1 is fully closed and the reactor tank shield doors are fully closed or, if the reactor tank shield doors are fully opened, then both exposure room plug doors must be fully closed
- o The reactor cannot be operated within Region 2 unless the

plug doors to both exposure rooms are fully closed and the reactor tank shield doors are fully opened

- The reactor cannot be operated within Region 3 unless the plug door to Exposure Room #2 is fully closed and the reactor tank shield doors are fully closed or, if the reactor tank shield doors are fully opened, then both exposure room plug doors must be closed
- The reactor cannot be moved into Region 2 unless the reactor tank shield doors are fully open.

Both exposure rooms are equipped with "emergency stop" buttons. An emergency stop button is also located on the reactor console. Pressing any of these emergency stop buttons causes an immediate reactor scram and gives a scram indication to the reactor operator at the console. Magnetic power to the standard control rods and the air supply to the transient rod cylinder cannot be obtained in an emergency stop scram condition without resetting the reactor console key which, when performed, automatically initiates a time delay with horns sounding in both exposure rooms. It also reinitiates the requirements for opening the lead shield doors. The emergency stop circuit, therefore, provides an independent means for an individual accidentally trapped in an exposure room to prevent an unsafe condition involving operation from occurring, while also providing a positive indication to the reactor operator that someone could be trapped in an exposure room.

4.14 SCRAM LOGIC CIRCUITRY

The scram logic circuitry assures that a set of reactor core and operational conditions must be satisfied for reactor operation to occur or continue in accordance with the Technical Specifications. The scram logic circuitry involves a set of open-on-failure logic relay switches in series. Any scram signal or component failure in the scram logic, therefore, results in a loss of standard control rod magnet power and a loss of air to the transient rod cylinder, resulting in a reactor scram. The time between activation of the scram logic and the total insertion of the control rods is limited by the Technical Specifications to assure the safety of the reactor and the fuel elements for the range of anticipated transients for the AFRRI-TRIGA reactor. The scram logic circuitry causes an automatic reactor scram under the following circumstances:

- The steady-state timer causes a reactor scram after a given elapsed time, as set on the timer, when utilized during steady-state power operations
- The pulse timer causes a reactor scram in pulse mode at the time set on the timer (less than 15 seconds) or an automatic software scram timeout at 15 seconds
- The manual scram button, located on the reactor console, allows the reactor operator to manually scram the reactor
- Movement of the console key to the OFF position causes a scram
- The reactor tank shield doors in any position other than fully open or fully closed will cause a reactor scram
- Activation of any of the emergency stop buttons in the

exposure rooms and on the reactor console causes a reactor scram

- A loss of AC power to the reactor causes a reactor scram
- High flux safety channel one causes a reactor scram at a reactor power level specified in the Technical Specifications for steady-state modes of operation
- High flux safety channel two causes a reactor scram at a reactor power level specified in the Technical Specifications for steady-state and pulse operations
- A loss of high voltage to any of the detectors for high flux safety channels one and two causes a reactor scram
- Fuel temperature safety channels one and two will each initiate a reactor scram if the fuel temperature, as measured independently by either channel, reaches 600°C (Technical Specifications limit). This assures that the AFRRI safety limit (core temperature) of 1,000°C for AFRRI stainless steel-clad cylindrical TRIGA fuel elements, as stated in the AFRRI Technical Specifications and testing conducted by General Atomics, is never approached or exceeded. The actual operational limit for the fuel temperature safety channels may be set lower than the Technical Specifications limit of 600°C
- A loss of reactor pool water which leaves less than or equal to 14 feet of pool water above the core (Technical Specifications limit) causes a reactor scram. The actual operational limits for the pool water level may be set

more conservatively

o Activation of either the DAC or CSC watchdog scram circuits causes a reactor scram.

4.15 FUEL HANDLING EQUIPMENT

[redacted]

The fuel element inspection tool is used to measure overall length and bow of a fuel element. The tool is designed to permit inspection of a fuel element while the element is submerged below 9 feet or more of water. The inspection tool consists of a tubular "go/no-go" gauge for detecting fuel element lateral bending (bow) and a device with a dial indicator for measuring fuel element length. The tool is held in place and positioned in the reactor tank by a supporting aluminum structure. The fuel element inspection tool may be placed within the core support structure or at other convenient locations within the reactor tank. By using an adapter, the tool can also be used to measure fuel-follower control rods.

Fuel storage racks permit temporary storage of up to [redacted] fuel elements per storage rack within the reactor tank, submerged a minimum of 9 feet beneath the pool water surface. Each rack is supported by two aluminum rods, which are attached by brackets to the aluminum channels at the top of the reactor tank. Use of the fuel storage racks allows core fuel loading changes to be made without the use of fuel handling casks. The fuel storage racks assure that stored fuel will not become critical and will not reach unsafe temperatures. Conservative calculations[2] show that in the event of a fully loaded storage rack failure where all [redacted] fuel elements fall to the bottom of the reactor tank in an optimal (i.e., worst case) neutronic geometrical configuration, a criticality excursion would not result.

4.16 REACTOR PARAMETERS

4.16.1 TRIGA Fuel Elements

The AFRRI-TRIGA Mark-F reactor core can accommodate up to 87 standard design TRIGA stainless steel clad, cylindrical fuel elements. The fuel elements are positioned in concentric rings to form a cylindrical, active (i.e., fueled) core region approximately [redacted]. The TRIGA fuel elements, described in Section 4.9, are summarized below:

Maximum fuel elements in core	87
Fuel-moderator material	8.5 wt-percent uranium homogeneously combined with 91.5 wt-percent zirconium-hydride
Hydrogen to zirconium atom ratio	Nominal 1.7 H to 1.0 Zr
Uranium enrichment	Less than 20% U^{235}
U^{235} loading per fuel element	Approximately [redacted]
Fuel element cladding	304 stainless steel, nominally 0.020 inch thick
Burnable poison	Samarium-aluminum wafers which form an integral part of the manufactured fuel element
Axial reflector	Graphite end slugs which form an integral part of the manufactured fuel element

4.16.2 Excess Reactivity

The core excess reactivity is measured each day a reactor operation involving the withdrawal of a control rod is planned. The reactor loading is limited by the Technical Specifications to a maximum of 3.5% Δk/k ($5.00) excess reactivity above cold critical, with or without all experiments in place. Single fuel element worths are approximately 0.79% Δk/k and 0.26% Δk/k in the B-ring and the F-ring, respectively. The sum of the absolute reactivity worths of all experiments in the reactor and the associated experiment and exposure facilities is limited by the Technical Specifications to 2.1% Δk/k ($3.00).

The net change in reactivity caused by the maximum operational withdrawal of a standard control rod is approximately 1.7% $\Delta k/k$ ($2.36). This amount of reactivity is due to the complete withdrawal of the borated graphite section of the standard control rod from the core and the consequent insertion of the follower into the core.

The net change in reactivity caused by the maximum operational withdrawal of the transient control rod is approximately 1.8% $\Delta k/k$ ($2.55). This amount of reactivity is due to the complete withdrawal of the borated graphite section of the transient control rod and the consequent insertion of the aluminum, air, or poison follower into the core.

The reactivity worths of each control rod and the shutdown margin are determined at periodic intervals specified in the Technical Specifications.

4.16.3 <u>Negative Temperature Coefficients of Reactivity</u>

The safe operation of the AFRRI-TRIGA Mark-F reactor during high power steady-state and pulse operations is ensured by the reactor's inherent large steady-state and prompt negative temperature coefficients of reactivity, respectively, due to intrinsic characteristics of the uranium zirconium-hydride fuel during heatup. Because of the large prompt negative temperature coefficient, any large (≥$1.00) step insertion of excess reactivity above cold critical will be immediately and automatically compensated for by the U-ZrH, terminating the excursion automatically without any dependence on the electronic or mechanical reactor systems or the reactor operator. Similarly, because of the large steady-state negative temperature coefficient of reactivity, any large insertion of excess reactivity during steady-state operation will be automatically compensated for by the U-ZrH, thus limiting the reactor steady-state power level.

These intrinsic negative temperature coefficients of reactivity (i.e., prompt and steady-state) are primarily due to the hardening of the neutron spectrum caused by heating the U-ZrH fuel. As the power level of the reactor is increased, the fuel temperature also increases. This increases the probability that thermal neutrons will gain energy from excited hydrogen atoms in the ZrH. Thus, the relative number of neutrons escaping from the fuel or being absorbed by parasitic capture before fission can occur increases, which introduces negative reactivity and, thus, limits the reactor power level actually achieved. Hence, any introduction of excess reactivity which increases the reactor power level and fuel temperature will be automatically compensated for.

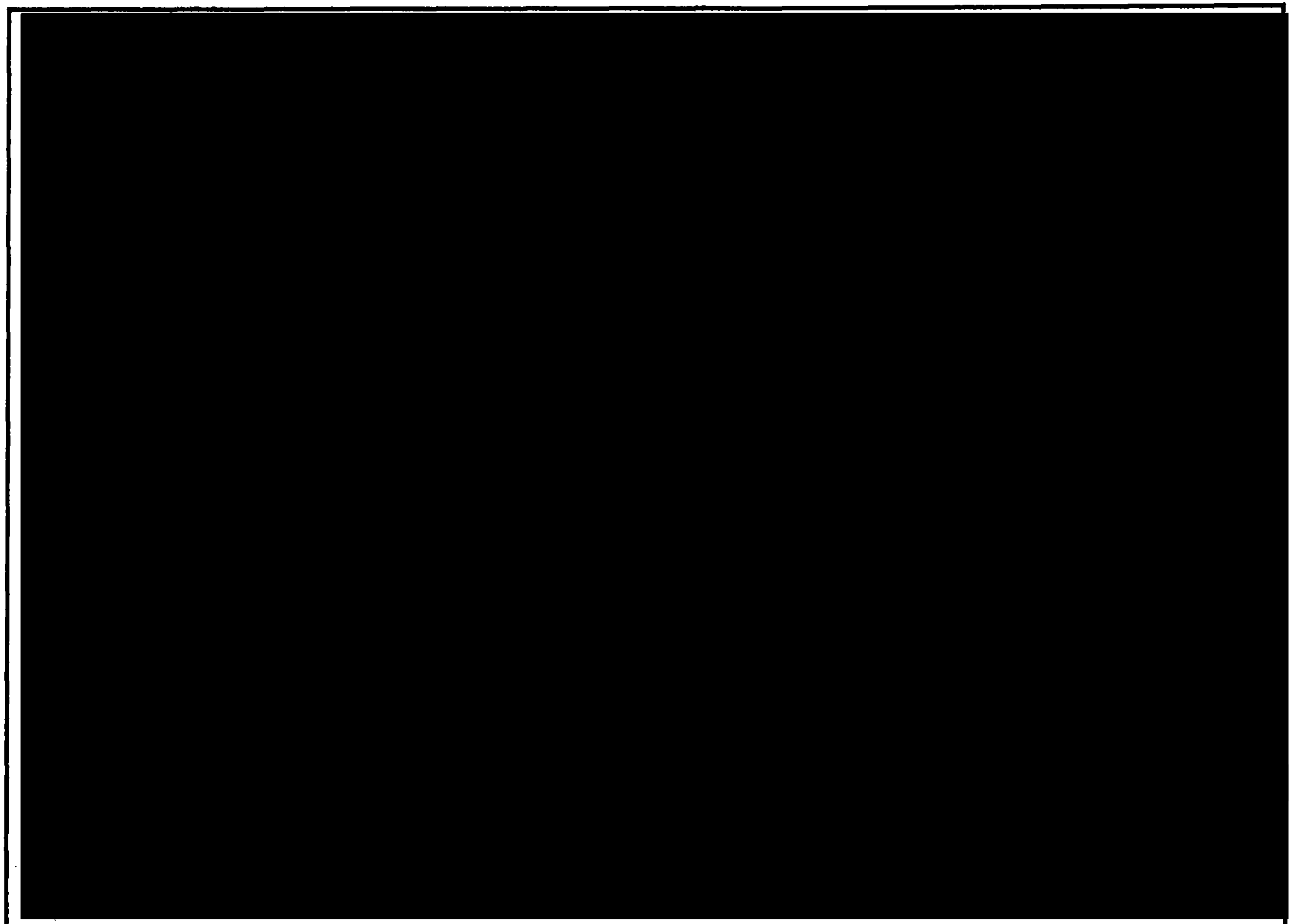
A second important but smaller contribution to the negative temperature coefficients of reactivity is due to Doppler broadening of the U^{238} neutron resonance absorption peaks at increased fuel temperatures. This mechanism decreases the resonance escape probability, and thus k_{eff} decreases. A third factor, the decrease in fuel and moderator density, also makes a small contribution to the total negative temperature coefficient.

The total prompt negative temperature coefficient of reactivity for the AFRRI-TRIGA Mark-F reactor is measured as approximately -0.0126% Δk/k per °C. The total steady-state negative temperature coefficient of reactivity for the AFRRI-TRIGA Mark-F reactor is measured as approximately -0.0051% Δk/k per °C.

4.17 REFERENCES

1. Amendment #22 to the Technical Specifications, Facility License No. R-84 for AFRRI-TRIGA Mark-F Reactor, Docket No. 50-170.

2. Sholtis, Joseph A., Jr., Capt., USAF, Nuclear Criticality Safety Analysis of Hypothetical AFRRI-TRIGA Fuel Element Storage Rack Accidents, AFRRI/SSD Memorandum for Record, January 19, 1981



RHS12

Figure 4-1.
AFRRI-TRIGA REACTOR TANK PLAN

[redacted]

Figure 4-2.

SUPPORT STRUCTURE FOR AFRRI-TRIGA CORE

RHS20

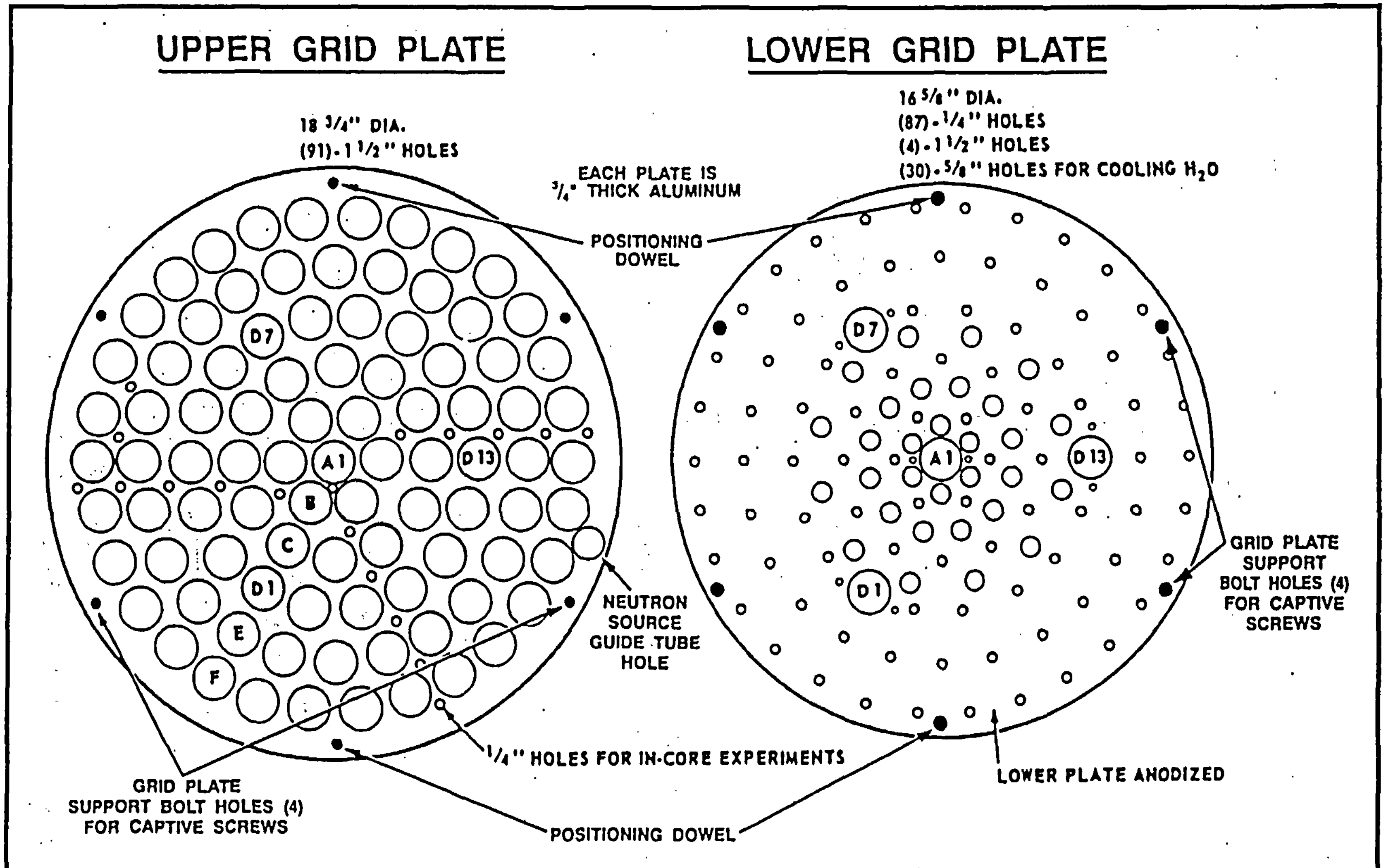


Figure 4-3.
UPPER AND LOWER GRID PLATES

RHS13

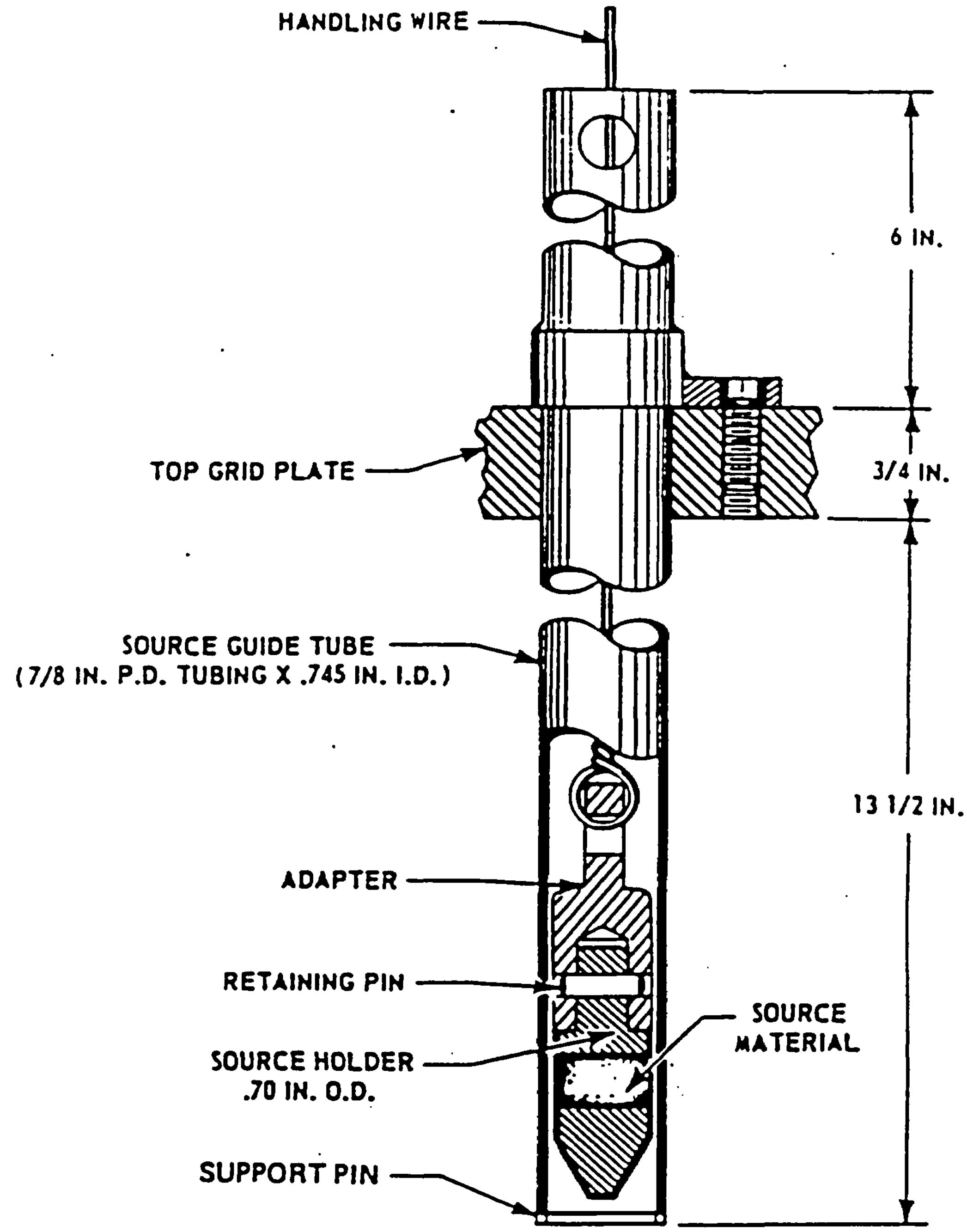


Figure 4-4.
NEUTRON SOURCE AND GUIDE TUBE

[redacted]

RHS17

Figure 4-5.
STANDARD FUEL ELEMENT

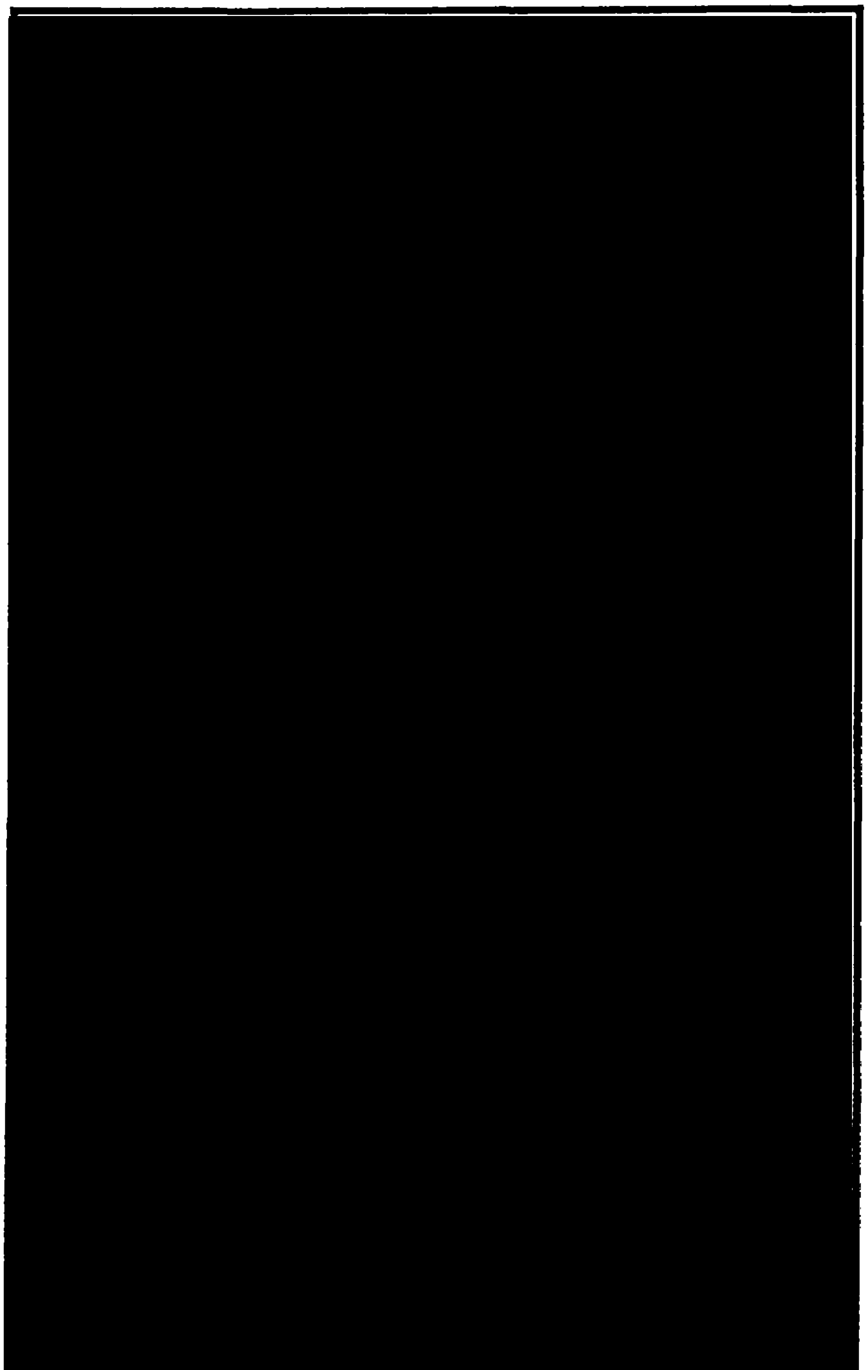


Figure 4-6.

INSTRUMENTED FUEL ELEMENT

RHS16

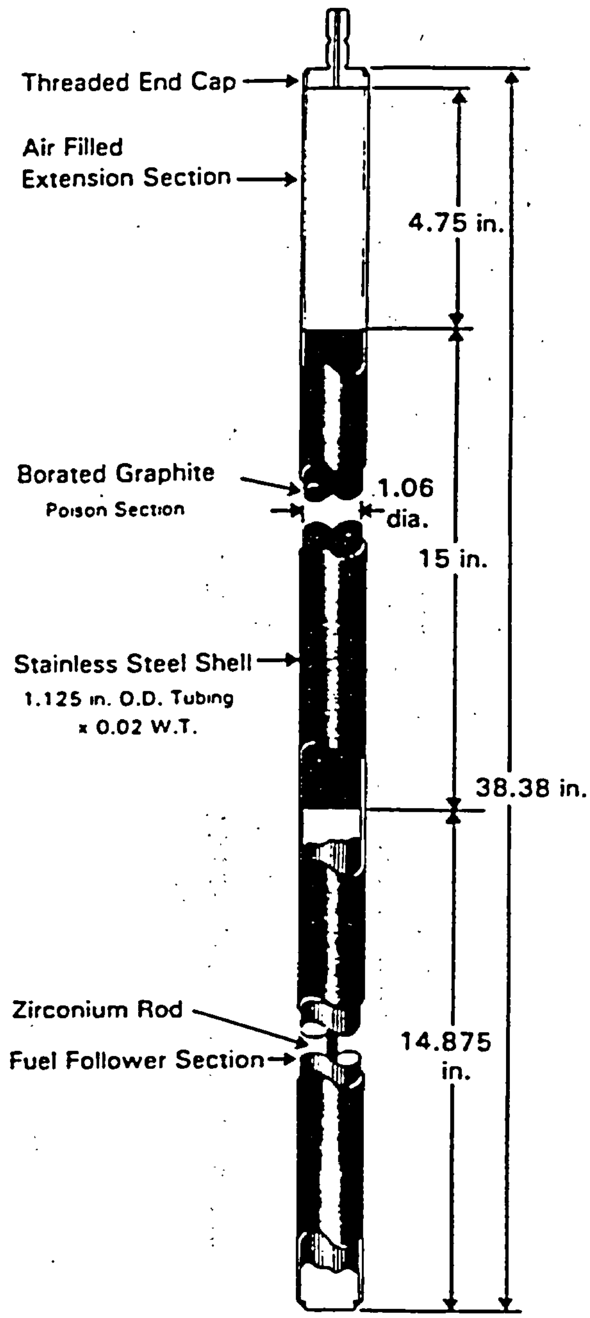


Figure 4-7.

STANDARD CONTROL ROD

Note: Dimensions given are nominal - refer to blueprints for details

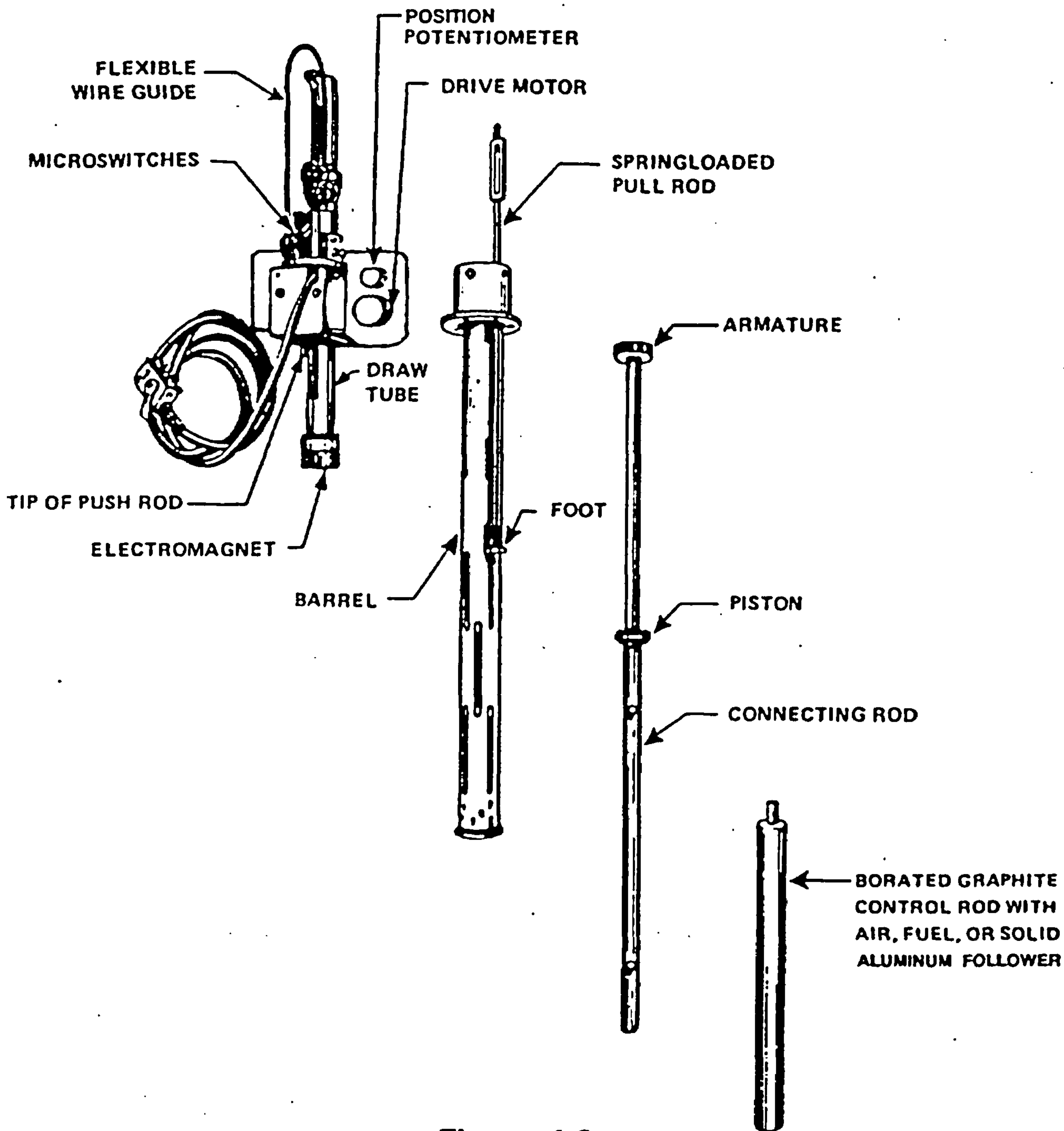


Figure 4-8.
STANDARD CONTROL ROD DRIVE

RHS15

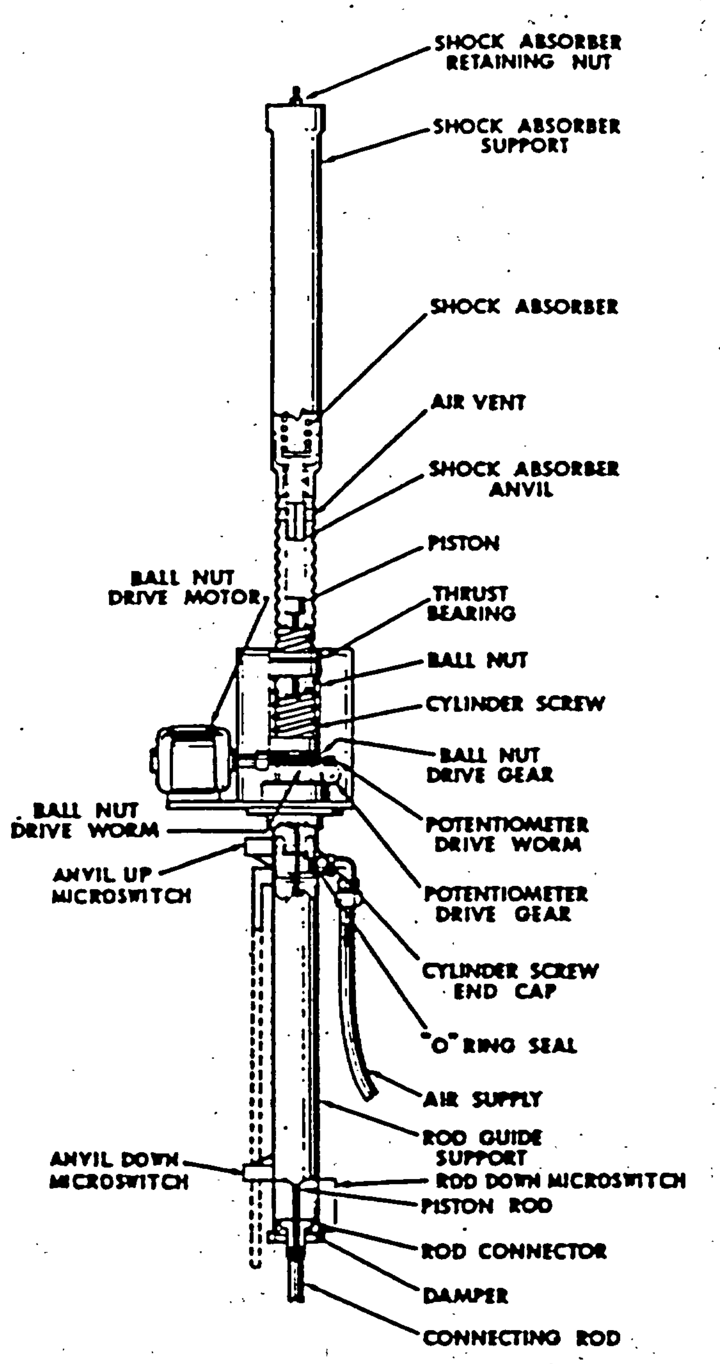


Figure 4-9.

TRANSIENT ROD DRIVE MECHANISM

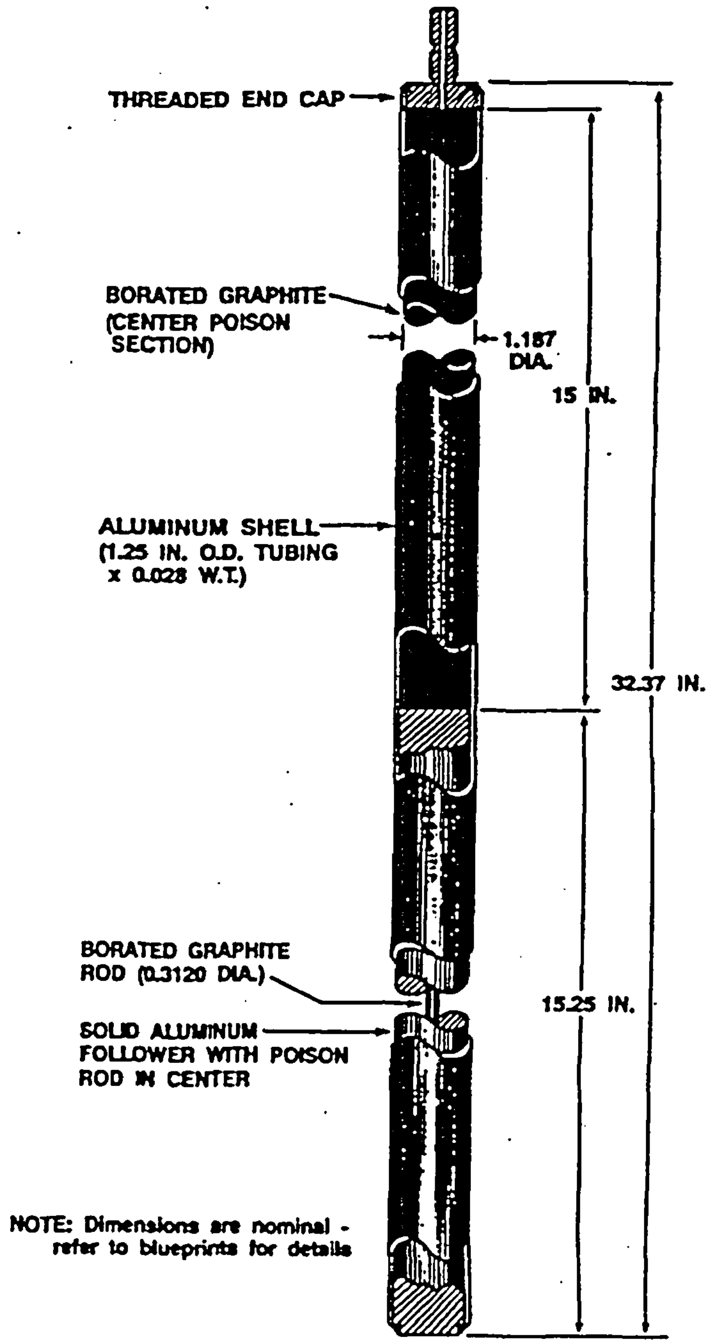


Figure 4-10.

TRANSIENT CONTROL ROD

RHS18

5.0 EXPERIMENTAL FACILITIES

5.1 GENERAL

The AFRRI-TRIGA Mark-F reactor serves as a source of both gamma and neutron radiation for research and radioisotope production. The unique flexibility of the AFRRI-TRIGA reactor is achieved by the horizontally movable core which can traverse from one irradiation position to another. The major irradiation facilities in which experiments can be carried out using the reactor are:

- o Exposure Room #1 and its extractor system
- o Exposure Room #2
- o Pneumatic Transfer System (Optional)
- o Pool Irradiation
- o In-Core Experiment Tube (CET)

The In-Core Experiment Tube was described in Section 4.8 and thus will not be addressed further here. A description of the other irradiation facilities is presented below.

5.2 EXPOSURE ROOMS #1 AND #2

The exposure rooms are located on the first level of the AFRRI reactor building at opposite ends of the reactor tank (Figure 5-1). Exposure Rooms #1 (ER #1) and #2 (ER #2) are located at the north and south ends of the reactor tank, respectively. A semi-cylindrical portion of the aluminum reactor tank (1/4 inch thick) extends into each exposure room. When the reactor core is moved to the extreme position adjacent to either exposure room, a water clearance of about 1 inch exists between the core shroud and the inside surface of the reactor tank.

5.2.1 Exposure Room Construction

The physical dimensions of ER #1 are approximately 20 feet by 20 feet by 10-1/2 feet high. Except for the aluminum reactor tank projection into each exposure room, all of the walls, the ceiling, and the floor of each exposure room are constructed of concrete which serves as the primary biological shield. The walls are constructed of high-density concrete while the ceilings and floors are ordinary (Portland) concrete. Approximately 6-1/2 feet of concrete separate the exposure rooms from the subsoil underlying the building. At least 12 feet of concrete or concrete and soil separate ER #1 from the occupied areas of AFRRI, and at least 9 feet of concrete or concrete and soil separate ER #2 from the occupied areas of AFRRI.

All six surfaces of both exposure rooms are covered with a 1 foot thick wood lining. By thermalizing a portion of the neutron population leaving the exposure room, the wood lining minimizes fast neutron activation of the concrete biological shield and, thereby, reduces the effects of secondary gamma radiation.

The walls and ceilings of the exposure rooms are also covered with 1/8 inch thick masonite panels painted with gadolinium oxide paint (~11 mg/cm^2). The gadolinium, which has a large absorption cross section for thermal neutrons, reduces the thermal neutron population in the exposure rooms. This, in turn, reduces the amount of Ar^{41} produced in the exposure rooms due to neutron activation of Ar^{40}, a normal constituent of air.

Exposure Room #1 contains three lead curtains, suspended from the exposure room ceiling by aluminum tie rods, which surround the top of the reactor tank projection. The lead curtains reduce scattered gamma radiation above or not attenuated by the reactor tank shield doors when the reactor is operating adjacent to ER #2 with the reactor shield doors closed. The lead curtains are 22 inches high and are constructed of stacked 1 inch lead sheets mounted in aluminum frames. The front curtain is 4 inches thick and 32 inches wide, and the two side curtains are 2 inches thick and 23 inches wide.

Exposure Room #1 has a cadmium-gadolinium (Cd-Gd) shield welded to the tank projection to reduce the thermal neutron leakage from the reactor core into ER #1. The Cd-Gd sheet is approximately 4 feet square and is centered on the reactor tank projection with respect to the core midplane. The shield consists of a 0.016 inch thick aluminum sheet sprayed with gadolinium oxide paint (~22 mg/cm^2) and covered with a 0.040 inch thick cadmium sheet. By reducing the thermal neutron population in ER #1, the Cd-Gd sheet helps to reduce the production of Ar^{41} in ER #1. The cadmium sheet also reduces secondary beta radiation emitted from the gadolinium.

A portable lead shield is also provided in ER #1 to provide a means of enhancing the neutron-to-gamma ratio for experimental purposes. The shield consists of an aluminum frame about 30 inches square by 12 inches thick that can be filled with lead bricks to obtain a lead shielding thickness of from 1 to 6 inches, as desired. The shield is mounted on a frame with wheels, which rolls on tracks secured to the floor and positions the shield approximately at the core midplane. When not needed, the portable shield can be rolled away from the tank projection against one of the exposure room walls. When positioned in front of the tank projection, the portable lead shield is secured to tie rods which support the lead curtains to ensure that the shield will not move during an experiment. ER #1 has two aluminum tracks screwed to the floor and extending back from the core projection to minimize wear to the wood floor by the wheels on movable tables.

A cable-driven experiment extractor system, which passes through the wall of ER #1, allows an operator in the prep area to place and retrieve small experiments without entering the exposure room (Figure 5-3), thus minimizing personnel exposure. As with the CET, the tube through the wall is bent to prevent radiation streaming (Section 4.8).

5.2.2 Exposure Room Access

Access to each exposure room is gained from the prep area through separate reinforced concrete rolling plug shield doors. Both plug doors are mounted on four wheels and can be moved, either by an electric motor or manually, along two recessed tracks anchored in the concrete floor. Both doors are stepped on the sides and top to prevent radiation streaming from the exposure rooms and are approximately 7 feet wide by 9 feet high. The ER #1 door is about 12 feet thick and weighs approximately 160,000 pounds. The ER #2 door is about 9 feet thick and weighs approximately 106,000 pounds. The Facility Interlock System (Section 4.13) acts to minimize any radiation exposure to personnel either in the prep area or entering either exposure room, while AFRRI's health physics procedures administratively govern the opening of the exposure room plug doors.

The plug doors may be opened or closed with electrical motors. The electric motor (~1 horsepower) for each plug door is controlled by a locked control power box in the prep area. The keys to the control power box are under the administrative control of the reactor operator. Once the control power box is opened, the plug door can be opened, closed, or stopped by pushing the appropriate button, given that the Facility Interlock System is satisfied. Microswitches interrupt the electrical power to the motor when the door is either fully opened or fully closed. A knife switch for each plug door motor, installed beside each plug door, provides a means of manually interrupting the electric power supply to the plug door motor in the event of a malfunction of the control power box or one of the microswitches.

The plug doors may also be opened and closed manually. This is accomplished by disengaging the padlocked coupler from the electric motor drive and then engaging the coupler to the manual drive. The handwheel is then turned to close or open the plug door. Manual operation of the exposure room plug doors overrides one part of the Facility Interlock System but does not affect its remaining logic. For example, if ER #1 is opened manually while the core is in Region 1 with the reactor tank lead shield doors closed or while the lead shield doors are open with the core in any region, an immediate reactor scram will occur. [redacted] is under the administrative control of the reactor operator. A picture of the ER #1 plug door in the closed position is provided in Figure 5-2. The ER #2 plug door has identical characteristics with respect to the electrical components and the type of drive mechanism.

5.2.3 Exposure Room Monitoring

Each exposure room is monitored by a separate continuous air monitor (CAM) located in the prep area close to the appropriate exposure room plug door. The CAMs provide continuous sampling and monitoring (gross beta-gamma activity) of airborne radioactive particulate matter present in ER #1 and ER #2. A general description of the CAMs is given in Section 3.6.2. The ER #1 CAM samples air taken directly from ER #1 and discharges the air back into ER #1. The ER #2 CAM samples air taken directly from ER #2 and discharges the air back into ER #2. Local readout with audible and visual alarms associated with certain preset airborne activity levels provide personnel in the prep area with information concerning airborne activity levels in the exposure rooms.

Before opening either exposure room plug door, the exposure room airborne activity levels, as measured by the appropriate exposure room CAM, must be below a certain level. If the CAM indicates that the plug door may be opened, access to the exposure room is permitted only after extensive radiation surveys are taken (with portable radiation monitors) of the exposure room, both from the prep area past the opened plug door and in the exposure room itself. If high radiation levels exist, the plug door must be closed and a certain amount of time allowed to elapse before exposure room entry is again attempted. All personnel in either the prep area or exposure rooms will have appropriate personnel radiation monitors and be assigned limited stay times, if significant radiation levels exist in either area, to assure compliance with both 10 CFR 20 and internal AFRRI occupational dose limits.

In addition to airborne radiation monitoring of the exposure rooms, radiation monitoring is also provided in the prep area. A CAM located north of the prep area near ER #1 samples air taken directly from, and discharged back to, the local area. The prep area CAM alarms, both audible and visual, are local (as with the exposure room CAMs). Remote Area Monitors (RAMs) are also provided in the prep area. Remote Area Monitors E-3 and E-6, which provide readouts in the reactor control room, are discussed in Section 3.6.1. RAMs E-4 and E-5, also located in the prep area, consist of a scintillation detector which measures gamma radiation with energies greater than 80 keV. The units have a range of 0.05 mrem/hr to 50 mrem/hr and a nominal accuracy of ±15 percent at all levels. The E-4 and E-5 RAM readouts are provided locally on the units themselves. RAMs E-4 and E-5 activate radiation alarms within the AFRRI complex at an adjustable, preset radiation level. RAMs E-4 and E-5 also activate visual alarms in the Emergency Response Center (Room 3430) and on the units themselves when a high-level alarm occurs or when loss of high voltage to the detector or a loss of signal to the remote readout occurs (failure alarm). The RAMs are calibrated at regular intervals using a radiation source of known intensity. The location of the prep area RAMs, the radiation readouts and alarms are shown in Table 5-1.

5.3 PNEUMATIC TRANSFER SYSTEM

The Pneumatic Transfer System (PTS) associated with the AFRRI-TRIGA reactor permits samples to be transported to a position in the reactor pool near the core shroud when the core is moved to the appropriate pool position. The PTS is an optional experimental facility as it is normally not in-place but can be installed as required.

The Pneumatic Transfer System is divided into two separate tube banks, designated as A-Tube Bank and B-Tube Bank (Figures 5-4 and 5-5). The A-Tube Bank and B-Tube Bank are capable of rapidly transporting samples between the Radiochemistry Laboratory and the reactor pool. The A-Tube Bank has the additional capability to divert the rabbits, containing irradiated samples, from the sender-receiver stations in the Radiochemistry Laboratory to the adjacent Hot Cell. Basically, each tube bank consists of a blower, an absolute filter, four sender-receiver stations, four rabbit tubes, a common air line, four solenoid (rabbit) valves, one four-way control valve, and a terminus. The samples are transported by small polyethylene or aluminum rabbits having nominal internal dimensions of <1 inch in diameter and 4 inches in length. The rabbits are transported within aluminum pneumatic tubes (rabbit tubes) that have a 1-1/4 inch inside diameter and a 0.083 inch wall thickness.

The rabbits are transported to and from the pool irradiating position by applying a partial vacuum to the appropriate rabbit tube volume.

The partial vacuum is created by a local blower in the Radiochemistry Lab pulling air through the absolute filter from the rabbit tubes. From the absolute filter, the air is exhausted to the reactor stack by way of the reactor building ventilation system. The absolute filter prevents radioactive particulate matter in the rabbit tubes from being discharged into the Radiochemistry Laboratory and subsequently into the reactor building ventilation system without first being filtered. The partial vacuum is applied to the UP-DOWN (4-way) valve. When the valve is in the UP position, the partial vacuum is applied to the manifold at the sender-receiver station. When the rabbit valves are opened, rabbits in the pool termini are drawn through the rabbit tubes to the sender-receiver stations, and air is drawn into the system through the common air tube. When the UP-DOWN valve is placed in the DOWN position, the partial vacuum is applied to the termini through the common air tube. When the rabbit valves are opened, the rabbits in the sender-receiver stations are drawn through the rabbit tubes into the pool termini and the atmospheric air is drawn into the system through the manifold. The relief valves allow air to flow to the blower when the rabbit valves are closed, preventing the blower from overheating.

The control circuit for the operation of the Pneumatic Transfer System is contained in the control unit which is located on the west wall of the Radiochemistry Laboratory next to the sender-receiver stations.

The types of experiments conducted with the Pneumatic Transfer System primarily involve radioisotope production and activation of small samples for subsequent neutron activation analysis. AFRRI's health physics procedures and internal reactor procedures administratively govern the use of the pneumatic transfer system in accordance with the Technical Specifications.

5.4 POOL IRRADIATION

The pool-type tank of the AFRRI-TRIGA Mark-F reactor permits the irradiation of waterproofed experiments submerged in the reactor pool. Eighteen (18) holes in the upper grid plate above the core also allow small samples to be inserted into the core for irradiation. The decision to perform an experiment in the reactor pool as opposed to the Pneumatic Transfer System or the CET is dictated by specimen size and the required radiation flux and intensity. The actual placement of experiments in the reactor core or the reactor pool will be limited by the Technical Specifications. At no time shall experiments be performed in the reactor pool which would introduce more excess reactivity into the core than the Technical Specifications allow. The use of the reactor pool as an experimental facility also includes the use of portable beam tubes placed in the reactor pool. The types of experiments performed in the reactor pool are similar to those performed in the Pneumatic Transfer System.

5.5 ARGON ACTIVATION

Argon-41 is produced in the exposure rooms, CET, reactor pool, and the Pneumatic Transfer System due to the leakage of thermal neutrons from the reactor core into these exposure facilities. Ar^{41} is produced by thermal neutron activation of naturally occurring Ar^{40} in the air of these experimental facilities (note that some air is always dissolved in the pool water). Since Ar^{41} is a noble gas, it is not retained in the human body and radiation exposures are a result of direct exposure (immersion) to the Ar^{41} gamma rays given off during its natural decay.

The exposure rooms and Pneumatic Transfer System are designed to minimize Ar^{41} exposures to the AFRRI staff. The exposure rooms contain design features, described in Section 5.2.1, which minimize the production of Ar^{41}. Air from Exposure Room 1, Exposure Room 2, and the Pneumatic Transfer System is exhausted to the reactor stack by way of the reactor building ventilation system. This limits egress of Ar^{41} to occupiable areas inside the AFRRI complex. The stack monitoring systems provide a means of ensuring that releases of Ar^{41} to unrestricted areas meet the requirements of 10 CFR 20.

Typical Ar^{41} production rates in these experimental facilities, based on measurements of Ar^{41} releases through the reactor stack during reactor operations, are:

o Exposure Room #1 with reactor core at extreme position closest to ER #1	0.40 mCi/kw-hr
o Exposure Room #2 with reactor core at extreme position closest to ER #2	1.89 mCi/kw-hr
o Pneumatic Transfer System with reactor core at position adjacent to pneumatic tubes	0.57 mCi/kw-hr

5.6 PROCEDURES

All experiments associated with the AFRRI-TRIGA reactor are performed in accordance with the Technical Specifications and regulations promulgated by 10 CFR Part 50, especially Section 50.59. The administrative procedures practiced at AFRRI provide a means of compliance with the appropriate regulations and are detailed in a set of internal written procedures approved by the Reactor Facility Director and reviewed by the Reactor and Radiation Facility Safety Committee (RRFSC).

Before performing any proposed first-time experiment (with the exception of routine operations outlined in the Technical Specifications and reactor operational procedures, such as Routine Reactor Authorizations and reactor parameter measurements, instrument checks, and calibrations), the experiment must be reviewed for radiological safety and approved by the Reactor Facility Director, the Reactor and Radiation Facility Safety Committee (RRFSC), and the Safety and Health Department. A Reactor Use Request (RUR) must also be completed, approved, and signed by the AFRRI Reactor Facility Director (RFD) or his designee before any experiment can be performed. Safety evaluations for proposed tests or experiments conducted without prior NRC approval under provisions of 10 CFR 50.59 are reviewed by the RRFSC and other appropriate groups or individuals as necessary to determine whether the proposed test or experiment meets any of the criteria in 10 CFR 50.59(c)(2) or requires a change to the Technical Specifications. The review and approval process allows senior staff personnel specifically trained in radiological safety to consider and recommend alternative operational profiles, such as different core positions, power levels, and irradiation times, etc., which might reduce staff exposures, the release of radioactive materials to the environment, or both. A record of tests and experiments performed, including written safety evaluations, is maintained by AFRRI in accordance with 10 CFR 50.59.

5.7 EXPERIMENTS

The limitations on experiments specified in the Technical Specifications for the AFRRI-TRIGA reactor state that:

- The sum of the absolute reactivity worths of all experiments in the reactor and in the associated experimental facilities shall not exceed \$3.00 (2.1% $\Delta k/k$). This value includes the total potential reactivity insertion which might result from experiment malfunction, accidental experiment flooding or voiding, and accidental removal or insertion of experiments.
- Each fueled experiment shall be limited so that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.3 curies, and the maximum strontium-90 inventory is not greater than 5 millicuries.
- Known explosive materials shall not be irradiated in the reactor in quantities greater than 25 milligrams.
- Samples shall be doubly contained when release of the contained material could cause corrosion of the

experimental facility.

Before any experiment can be conducted using the reactor or experimental facilities, the experiment must be reviewed by the Reactor and Radiation Facility Safety Committee. A Reactor Use Request (RUR) must also be completed, approved, and signed by the AFRRI Reactor Facility Director (RFD) or his designee before an experiment can be performed. If it is anticipated that an experiment will cause a reactivity change of approximately ±0.5% $\Delta k/k$, k-excess measurements must be made at the core position in which the experiment will be performed, both with and without the experiment inserted, to determine the actual reactivity worth of the experiment.

Typical experiments that have been performed at AFRRI using the AFRRI-TRIGA reactor and its associated experimental facilities, within the conditions stated above and as authorized by Facility License No. R-84, include:

- Irradiation of up to 500 pounds of animals for up to 1 MW-hr with or without shielding.
- Irradiation of various materials in the pneumatic tube system or CET. These materials included inorganic compounds of 70 different elements limited on the basis of the chemical or nuclear characteristics of the material. The alloys of 10 metals were also included in these experiments, limited on the basis of their nuclear characteristics.
- Preparation of Argon-41 calibration standards in the

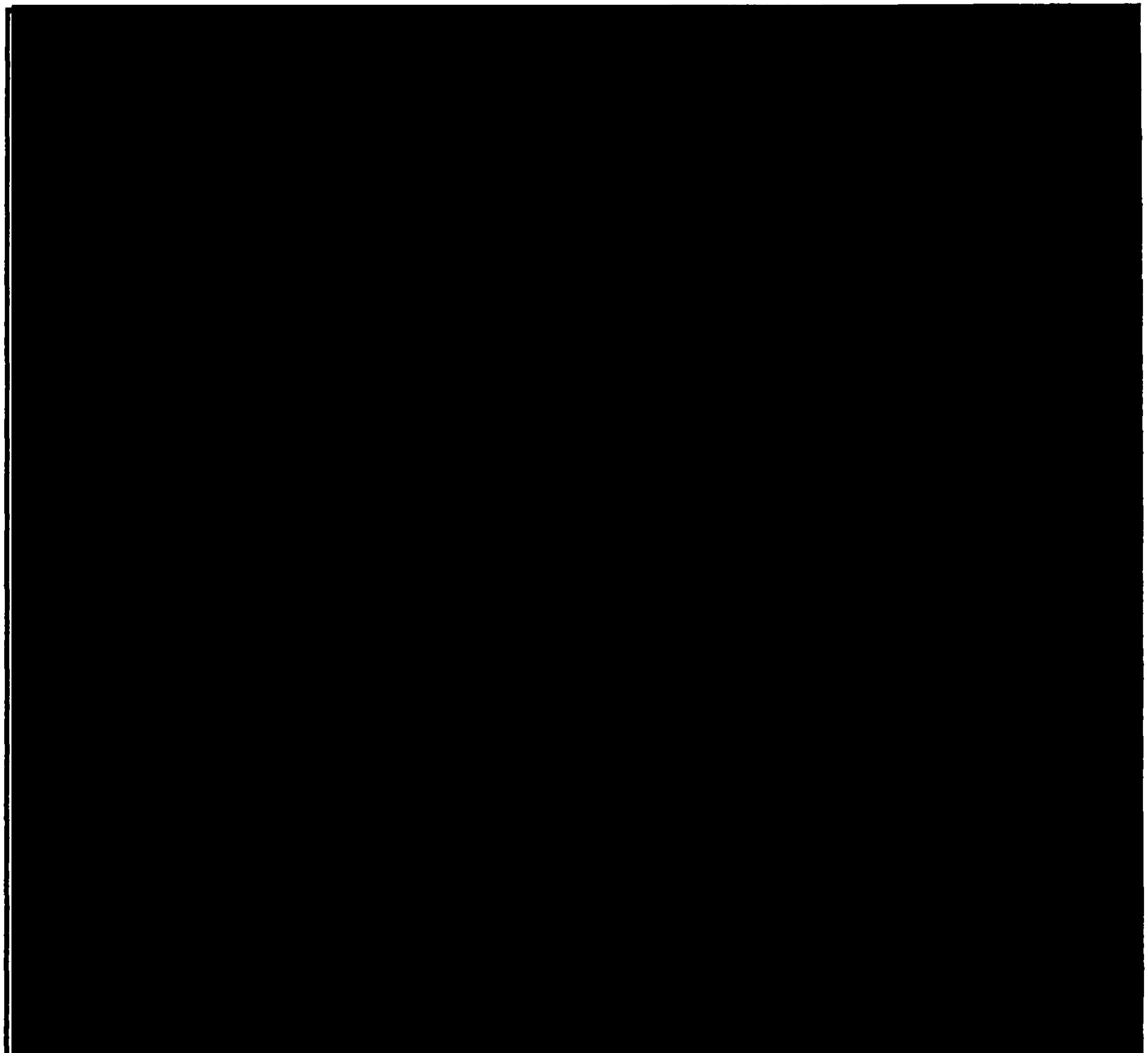
exposure rooms and irradiation of up to 5 grams of sodium carbonate in the pneumatic transfer system.

- Irradiation of old paintings for radiographic analysis of age and authenticity.
- Irradiation of electronic components in the exposure rooms and the pneumatic transfer system. These components include integrated circuits.

TABLE 5-1

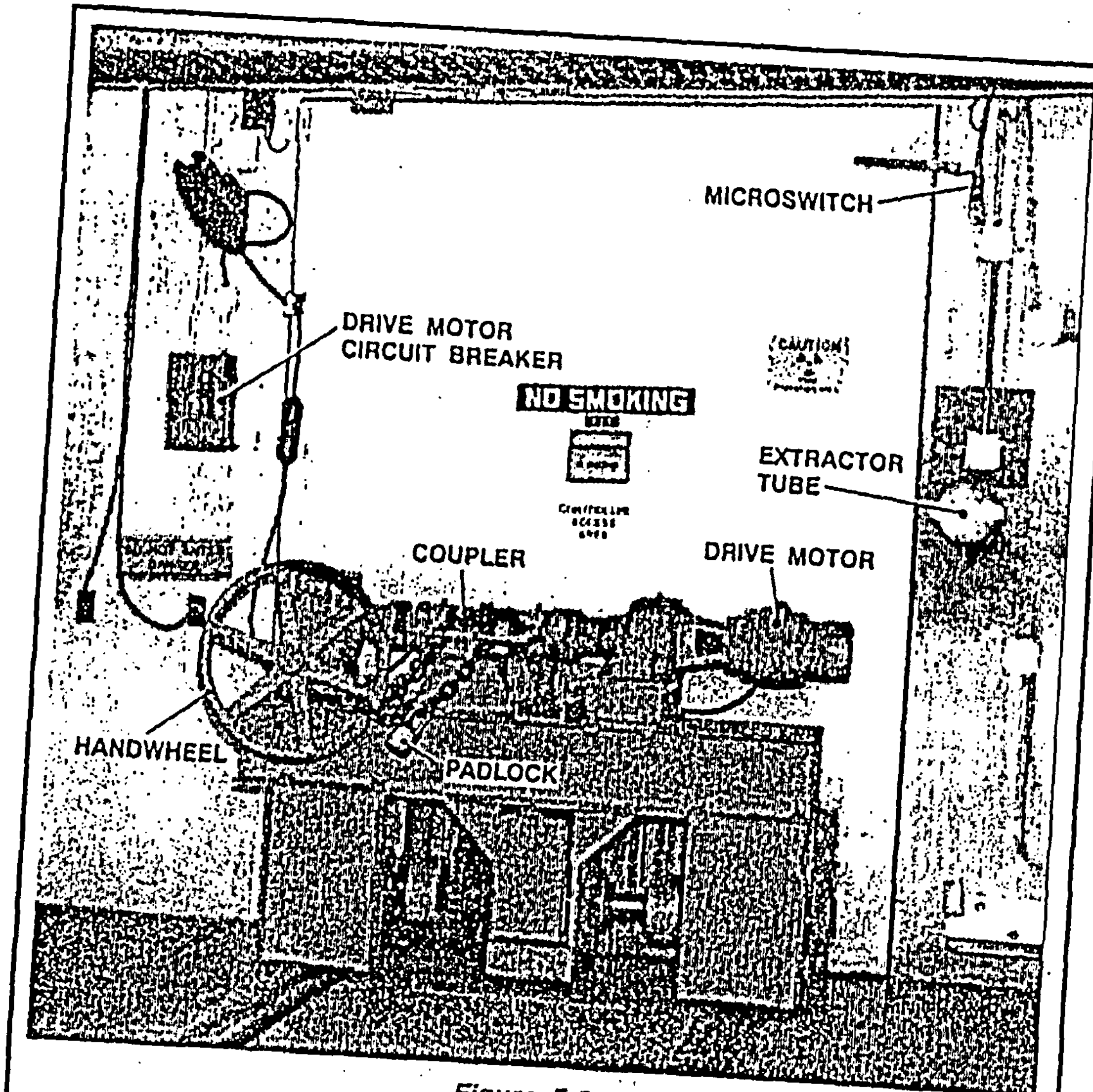
PREP AREA REMOTE AREA MONITORS

REMOTE AREA MONITOR	LOCATION	READOUT	RADIATION ALARMS
E-3	Approximately 6 feet above the floor on the prep area west wall opposite ER #1 plug door	Meter in the reactor control room, the Emergency Response Center and locally	Activates visual alarm in the reactor control room and in the Emergency Response Center; activates local visual and audible alarm on the prep area east wall near ER #1 and a red light at the front desk guard station
E-6	Approximately 6 feet above the floor on the prep area west wall opposite ER #2	Same as E-3	Same as E-3 (except that the alarms activated in the prep area are located on the east wall near ER #2)
E-4	Approximately 6 feet above the floor on the prep area east wall north of ER #1 plug door	Meter on unit itself and meter in Emergency Response Center	Activates audible and visual alarm on unit itself; activates visual alarm in the Emergency Response Center and a red light at the front desk guard station
E-5	Approximately 6 feet above the floor on the east wall of the prep area between the plug doors to ERs #1 and #2	Same as E-4	Same as E-4



RHS6

Figure 5-1.
REACTOR FACILITY CUTAWAY



RHS14

Figure 5-2.
EXPOSURE ROOM #1 DOOR IN CLOSED POSITION

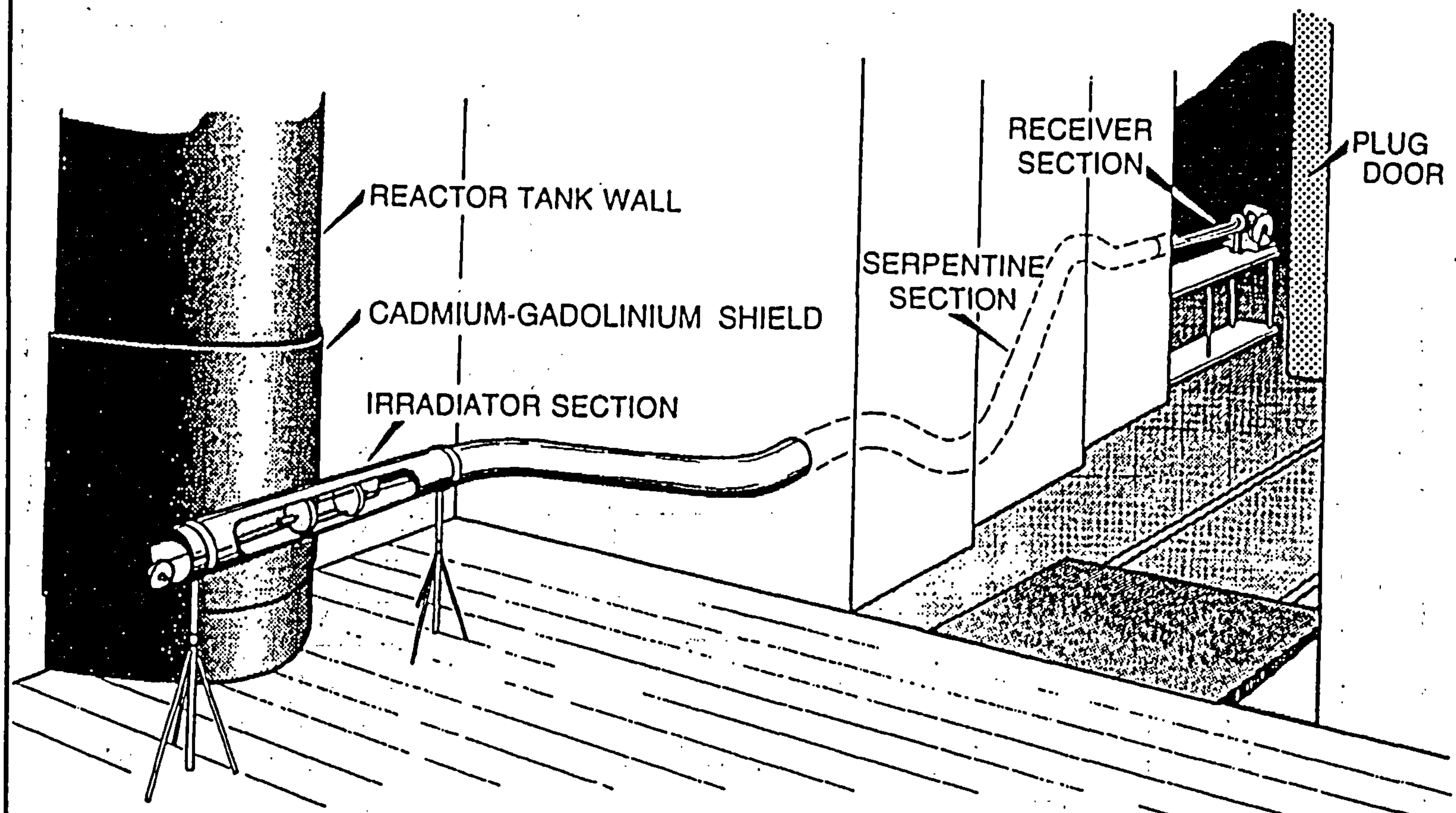


Figure 5-3.
EXPOSURE ROOM #1 EXTRACTOR SYSTEM

RHS11

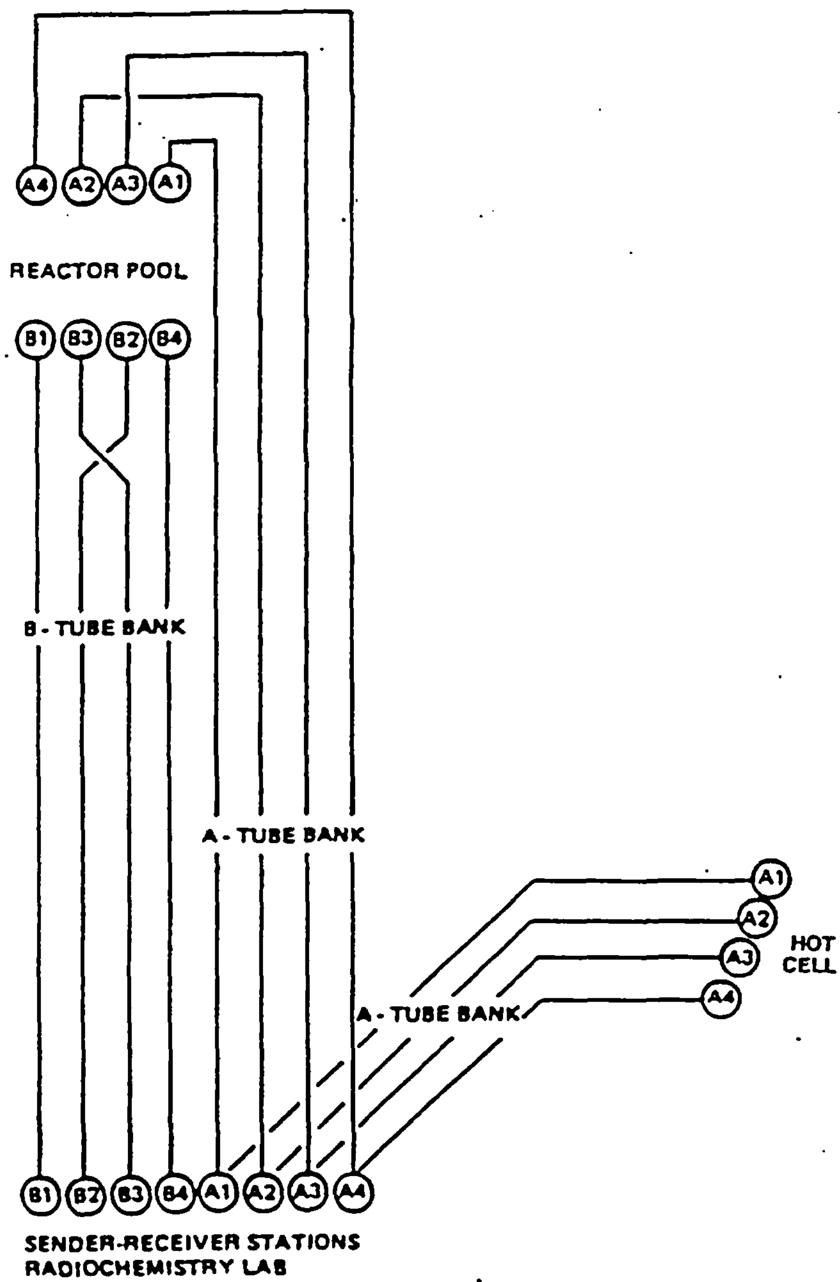


Figure 5-4.

PNEUMATIC TRANSFER SYSTEM ROUTING DIAGRAM

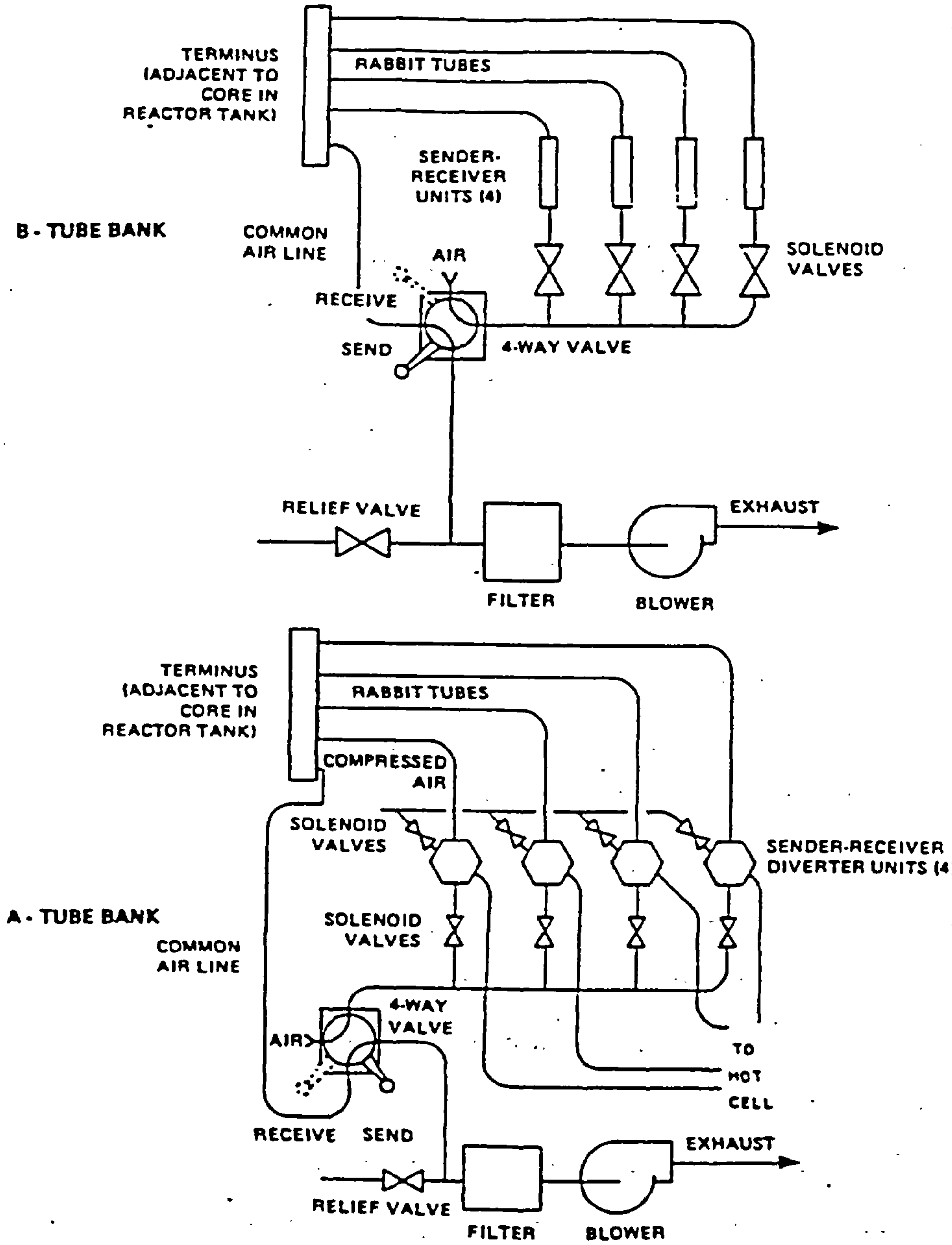


Figure 5-5.
PNEUMATIC TRANSFER SYSTEM

6.0 SAFETY ANALYSIS

6.1 POTENTIAL CONSEQUENCES ASSOCIATED WITH REACTOR OPERATIONS

The potential radiological consequences associated with routine and nonroutine operation of the AFRRI-TRIGA Mark-F reactor have been reviewed. This section presents the results of this study.

Routine operational risks are present in the manipulation of any radioactive material because of its potential for biological and physical damage. This study investigated potential consequences which may be associated with routine operation, including the handling of radioactive materials, reactor power transients, experiments associated with reactor operations, and production of radioactive gases (primarily through activation of argon).

Nonroutine operational risks may arise from the improper handling or from malfunctions of materials and equipment. The specific problems investigated were fuel element cladding failure, radioactive contamination of the reactor pool water, loss of shielding and cooling water, and Design Basis Accidents.

In order to evaluate the potential consequences associated with the operation of this reactor the following assumptions have been made:

- The reactor will normally have two modes of operation: Steady-State Operation at power levels not to exceed 1.1 megawatt (MW[1]) for testing/maintenance purposes and 1.0 megawatt (MW[1]) for normal operations. For calculations, the assumption of operation in the steady-state mode at a power level of 1.0 MW was utilized. The total reactor power generated has been roughly 811 MW-hrs from June 1964 through December 1993.

<u>Pulse Operation</u> achieved by intentionally placing the reactor on a prompt critical excursion by making a step insertion of reactivity above critical utilizing the transient rod and appropriate scrams and interlocks. The maximum step insertion is limited by the Technical Specifications to 2.8% $\Delta k/k$ ($4.00) reactivity in the pulse mode.[1] The average "maximum-pulse" used during AFRRI's TRIGA pulse operation has been 28 MW-sec. For calculations, the assumption of a maximum excursion pulse of 40 MW-sec was utilized as an upper limit for pulse operations or an inadvertent transient.

- The reactor operations will be supervised by individuals who are qualified operators trained in the detection and evaluation of radiological consequences.

In summary, the analysis indicated that the consequences from the Design Basis Accidents of a fuel element drop accident or a fuel element clad failure accident were insignificant. The maximum calculated whole-body dose was less than 2 mrad beyond 35 meters from the reactor stack. The maximum calculated thyroid dose was less than 57 mrad beyond 35 meters from the reactor stack. Doses from these postulated accidents to individuals beyond the boundary of the National Naval Medical Center site would be significantly less than 1 mrad. Therefore, it is concluded that the operation of the AFRRI reactor in the manner authorized by Facility License No. R-84 does not represent an undue risk to the health and safety of the operational personnel or the general public.

6.2 ROUTINE OPERATIONAL RISKS

6.2.1 Handling Radioactive Material

Because of the potential biological effects of radioactive material, special precautions should be taken in the handling of these materials. Reactor operations must be supervised by responsible individuals who are trained in the detection and evaluation of radiological consequences. Administrative, operational, and health physics procedures will be followed, and special equipment and procedures which are needed to maintain the ALARA (as low as reasonably achievable) concept of radiation protection will be used.

The radiological consequences associated with fuel elements are of the same nature as those associated with isotope production. Because of their high radiation levels, fuel elements should be kept under water for shielding purposes. In keeping with ALARA policies, a fuel element would generally not be removed from the reactor pool for at least two weeks following its use in the reactor core. When a fuel element is removed from the reactor tank, a conventional fuel element transfer cask may be used to reduce radiation levels to within acceptable limits. A Design Basis Accident (Section 6.3.4) is a fuel element drop during such a transfer in which a cladding failure occurs in air.

When proper administrative, operational, and health physics procedures are utilized, the handling of radioactive materials does not represent a significant risk to the health and safety of operating personnel or the general public.

6.2.2 Reactor Power Transients

The following discussion is based on experimentation and testing performed by General Atomics and presented in References 2 and 13. Theoretical estimates are also reviewed. The U-ZrH fuel elements used in the TRIGA reactor are capable of operating under conditions of transient experiments for delivery of high intensity bursts of neutrons. Fuel elements with 8.45 wt% U have been pulsed repeatedly in General Atomics' Advanced Triga Prototype Reactor (ATPR) to peak power levels of over 8,000 MW providing a neutron fluence per pulse of approximately 1.0×10^{15} nvt (neutrons/cm^2).

The ATPR fuel elements were subjected to thousands of pulses of 2,000 MW or more. The inherent safety of the fuel element stems from its large prompt negative temperature coefficient of reactivity, which causes the automatic termination of a power excursion before any core damage results. This temperature coefficient has been measured to be approximately $0.018 reactivity loss per 1°C rise in fuel temperature, i.e., -1.26×10^{-4} $\Delta k/k/°C$.

The reactor loading is limited by the Technical Specifications to a maximum of 3.5% $\Delta k/k$ ($5.00) excess reactivity above cold critical, with or without all experiments in place.[1] Thus, the maximum reactivity transient that could possibly occur would be that produced by the rapid insertion of the entire available amount of excess reactivity. In regard to TRIGA fuel performance, experiments at General Atomics' ATPR have been performed for step insertions of up to 3.5% $\Delta k/k$ (\$5.00) reactivity. The fuel elements were subjected to thousands of pulses of 2,000 MW or more and attained temperatures of up to 1,000°C. For example, the annular core TRIGA reactor at JAERI (Japan) has operated since 1975, with over 1,000 pulses of all sizes at fuel temperatures up to 1,000°C. From the results of the tests performed by General Atomics, there was no external evidence of change in any of the five special test elements in the first 200 pulses.

Theoretical estimates based on the Fuchs-Nordheim mathematical model of the AFRRI-TRIGA reactor have also been made. These calculations indicate that a step insertion of 2.8% $\Delta k/k$, that is a \$4.00 pulse, (AFRRI-TRIGA Technical Specification limit) would result in a fuel temperature rise of less than 550°C.

Therefore, based on operating experience of the ATPR and the Fuchs-Nordheim mathematical model, it can be concluded that the rapid insertion of the total authorized excess reactivity of 3.5% $\Delta k/k$ (\$5.00) would not represent an undue risk to the operating personnel or the general public.

6.2.3 Improper Fuel Loading

Fuel loading of the reactor is always supervised by trained, licensed, supervisory personnel. All reactor monitoring and shutdown devices will be operational during loading. The worst possible case of improper fuel loading would be for an operator to mistakenly insert a fuel element in a core that is already critical at a low power level, i.e., ≤1.0 w(t). In a core near its critical point, all of the inner fuel positions would be occupied so that the extra fuel elements could be added only in a peripheral position, where a fuel element worth is approximately 0.26% Δk/k ($0.37). Since step additions of 2.1% Δk/k ($3.00) excess reactivity are made on a routine basis for pulsing the reactor, the addition of 0.26% Δk/k ($0.37) would not present a danger of damaging the reactor or fuel. The reactor would undergo a mild transient and then operate at a steady-state power level of about 70 kw.

Even in the extremely unlikely event that a fuel element in the "B" ring should be improperly handled, its rapid insertion would result in an addition of about 0.79% Δk/k ($1.13). As indicated above, such an addition would not result in any damage to the reactor or the fuel and would not represent an undue risk to the health and safety of the operating personnel or the general public.

6.2.4 Production of Radioactive Gases in the Reactor Coolant

The production of radioactive gases by the reactor in its associated facilities originates through neutron activation of elements in the air or water. One of the most important of these activation products is radioactive argon (Ar^{41}) with a half-life of 1.83 hours.[4] Calculations are based on a temperature of 70°F (21°C), and Ar^{40} content of air of 0.94 percent by volume.

In the calculation to determine the amount of argon dissolved in the reactor pool water, it is assumed that the argon follows Henry's Law. At a water temperature of 70°F (21°C), the corresponding water vapor pressure is 19 mm Hg. The partial pressure for air would therefore be 760-19 = 741 mm Hg. Using an argon content of air as 0.94 percent by volume, the resulting partial pressure of argon is 741(9.4×10^{-3}) = 7 mm Hg. Applying Henry's Law, a saturation concentration of argon in water of 1.367 x 10^{-8} gram mole of Ar^{40} per 1 cm^3 H_2O is obtained. Using a microscopic thermal neutron absorption cross section[3] of 0.61 x 10^{-24} cm^2 for Ar^{40}, the macroscopic absorption cross section of argon in water becomes 5 x 10^{-9} cm^{-1}.

Activity production, assuming an Ar^{41} saturation condition has been achieved, can be given as:

$$A_p = \frac{\Phi_n(1 - e^{-\lambda t_p})\Sigma_a}{3.7 \times 10^4}$$

where:

A_p	=	activity in reactor water (μCi/cm^3)
Φ_n	=	thermal neutron flux (n/cm^2-sec)
t_p	=	core circulation time (sec)
Σ_a	=	macroscopic absorption cross section (cm^{-1})
3.7×10^4	=	constant to convert disintegrations per second to microcuries
λ	=	decay constant (sec^{-1})

A conservative value for the average thermal neutron flux (Φ_n) in the reactor core is estimated to be 1.0 x 10^{13} n/cm^2-sec at the

1.0 MW power level. The water circulates through the core by natural convection and is estimated to change completely in approximately 4 seconds. The Ar^{41} decay constant is 1.05×10^{-4} sec^{-1}. The core of the reactor holds 3.4×10^4 cm^3 of water and the rate of flow of water through the core is 0.9×10^4 cm^3/sec.[4] Substituting the appropriate values in equation (1) yields an Ar^{41} activity in the reactor water of:

$$A_p = 5.7 \times 10^{-4}\ \mu Ci/cm^3,$$

and a total Ar^{41} production rate in the core of:

$$Q = 5.1\ \mu Ci/sec$$

The travel time of Ar^{41} from the core to the water surface, a minimum distance of 457.2 cm (15 ft), has been estimated to be 24 seconds. Assuming the decay to be negligible, the maximum rate of activity reaching the surface is 5.1 μCi/sec.

Under saturated, steady-state conditions (a conservative assumption for this facility), the maximum rate at which Ar^{41} can escape from the water surface will be 5.1 μCi/sec, and an equivalent amount of Ar^{40} will dissolve in the water to replace the Ar^{41} depletion. As water temperature increases, the water vapor pressure will increase and the amount of dissolved Ar^{40} will decrease, resulting in a lower generation of Ar^{41} escaping to the reactor room. The radioactive argon (Ar^{41}) will escape from the reactor pool, dissipate in the reactor room air, and circulate through the ventilation system. Estimates are that the reactor room exhausts 9.64×10^7 cm^3/min (3430 ft^3/min) of the total 8.78×10^8 cm^3/min (31,000 ft^3/min) through the reactor stack. The added exhaust is from other radiological areas (Figure 3-5).

Since the air is exhausted from the system at a high rate, it is assumed that the Ar^{41} is distributed throughout the areas handled by the system and that the Ar^{41} decay is negligible. If it is assumed that there is continuous reactor operating time, equilibrium conditions can be assumed. Since equilibrium conditions are assumed, the same concentration of Ar^{41} will exist in the reactor room as in the exhaust air. The concentration in the reactor room is:

$$\frac{5.1\ \mu\text{Ci/sec}}{1.6 \times 10^{6}\ \text{cm}^3\text{/sec}} = 3.2 \times 10^{-6}\ \mu\text{Ci/cm}^3$$

The concentration in the stack is 3.5×10^{-7} μCi/cm^3 and is further dispersed in the air before reaching the surrounding population. The gamma dose rate in the reactor room can be estimated for the Ar^{41} concentration of 3.2×10^{-6} μCi/cm^3 by assuming submersion in a spherical source equivalent to the reactor room volume. The exposure rate can be given as:

$$D = \frac{S_v}{K\mu}(1-e^{-\mu r}) \qquad (2)$$

where: D = exposure rate (rem/hr)

S_v = gamma source strength (MeV/cm^3-sec)

μ = linear attenuation coefficient for air (cm^{-1})

K = flux-to-dose conversion factor (MeV/cm^2-sec per rem/hr)

r = radius of equivalent sphere (cm).

The volume of the reactor room is approximately 9.2×10^8 cm^3, which is equivalent to a sphere with a radius (r) of 600 cm.[5] An Ar^{41} concentration of 3.2×10^{-6} $\mu Ci/cm^3$ has a gamma source strength (S_v) equal to 0.18 MeV/cm^3-sec. For the Ar^{41} gamma photon of 1.29 MeV, the linear attenuation coefficient for air (μ) is 7.2×10^{-5} cm^{-1} and the flux-to-dose conversion factor (K) is 5.44×10^5 MeV/cm^2-sec per rem/hr. Substituting these values into equation (2) results in an exposure rate from the calculated Ar^{41} concentration of 0.18 mrem/hr. That exposure rate, if continuous over a 2000 hour year, is more than an order of magnitude below the yearly occupational dose limit of 5 rems (TEDE).

Another important activation product is radioactive nitrogen (N^{16}), with a half-life of ~7.14 seconds. As a result of its short half-life, N^{16} contributes to occupational exposures of individuals in the reactor room during operation, particularly high power operations, but poses no danger to the health and safety of the general public. The activation occurs as a result of the oxygen content in the pool water from the $O^{16}(n,p)N^{16}$ production process which is exclusively a fast neutron (i.e. $\geq$0.1 MeV) induced activation reaction.

Since N^{16} is produced by fast neutron activation of oxygen in the pool water within the core region, the activity production and exposure rate equations, cited above as equations (1) and (2), were used to calculate radioactive nitrogen (N^{16}) releases from the reactor pool to the reactor room air and the associated exposure dose rates that conservatively would be expected above the pool water surface directly over the core.

The concentration of O^{16} in water is approximately 0.0554 gm-mole O^{16} per ml of H_2O. Using a microscopic (n,p) cross section for O^{16} of 1.9×10^{-29} cm^2 which is averaged over the fission spectrum[6], the macroscopic (n,p) cross section for oxygen-16 in water becomes 6.34×10^{-7} cm^{-1}.

Using a conservative average fast flux value for the AFRRI-TRIGA reactor at 1.0 MW(t) of 5.0×10^{12} neutrons/cm^2/sec, and substituting the appropriate values for λ, t_p, and $\sum_a$ (i.e., $\sum_{n,p}$ for O^{16}) into equation (1) yields:

A_p = 27.6 μCi/cm^3,

and a total N^{16} production rate in the core at 1.0 MW(t) of:

Q = 0.25 Ci/sec.

Using a measured travel time for N^{16} bubbles to rise from the core to the water surface of ~24 seconds, the maximum rate at which activity from N^{16} reaches the pool surface is 0.024 Ci/sec.

The radioactive nitrogen will escape from the reactor pool, dissipate, and decay rapidly (~7.14 second half-life) in the reactor room air. The volume of air above the surface of the reactor pool in which the N^{16} activity per second is distributed was assumed to be a right circular cylinder with a diameter of 91.5 cm and height of 100 cm for negligible decay. This assumed volume, therefore, is 6.6×10^5 cm^3. The concentration of N^{16} in this volume per second would therefore be 3.64×10^{-2} μCi/cm^3 with a gamma source strength (S_v) equal to 6.2×10^3 MeV/cm^3/sec (E_γ = 4.6

MeV/disintegration). For the N^{16} primary gamma photon of 6.13 MeV, the linear attenuation coefficient for air (μ) is 3.2×10^{-5} cm^{-1} and the flux-to-dose conversion factor (K) is 8.4×10^{5} $MeV/cm^2/sec$ per rem/hr. Substituting these values into equation (2) for an equivalent volume sphere having a radius (r) of 54 cm results in a conservative estimate of the exposure dose rate due to N^{16} at 1.0 MW(t) of 398 mrem/hr immediately above the pool water surface directly over the core. It should be noted that the N^{16} activity evaluated by this analysis is due only to the amount of N^{16} that may be released from the pool. To reiterate, it should also be noted that due to the short half-life of N^{16} and the amount of time required to circulate air from the reactor room to the top of the reactor stack, essentially no N^{16} is released from the reactor stack to the environment. As a result, N^{16} only presents an occupational exposure potential to individuals in the reactor room near the pool surface during high power operation.

Based upon actual measurements made during reactor operations at 1.0 MW(t), the typical Ar^{41} release rate from the pool surface has been approximately 0.5 $\mu Ci/sec$ (September 22, 1980, Ar^{41} Report). Comparing this to the calculated value of 5.1 $\mu Ci/sec$ indicates a factor of ten conservatism in the calculation. A typical Ar^{41} concentration at the reactor stack has been approximately 3.0×10^{-8} $\mu Ci/cm^3$ as compared to the calculated value of 3.5×10^{-7} $\mu Ci/cm^3$, also indicating a factor of ten conservatism in the calculation.

Gamma radiation levels measured directly over the core just above the pool surface range from 75 mrem/hr initially to a maximum of about 200 mrem/hr. Gamma radiation levels measured around the reactor pool chain range up to 14 mrem/hr. It should be noted that all measured gamma exposure rates (from N^{16}) include N^{16} in both the pool water and the reactor room air. Even though the calculated exposure rate for N^{16} of approximately 400 mrem/hr only takes into account N^{16} in the reactor room air just above the pool surface, it still overestimates the actual dose rate by a factor of two.

6.2.5 Experiments

All experiments performed as part of the TRIGA reactor operations are reviewed by the Reactor and Radiation Facility Safety Committee and must be authorized prior to their performance. The Technical Specifications contain requirements that must be met before such experiments can be performed using the AFRRI-TRIGA reactor. Experiments are always supervised by trained, licensed, supervisory personnel. However, failure of an experiment is possible and worst-case conditions can be calculated to determine the postulated consequences.

The two worst-case conditions for failure of an experiment could result in instantaneous insertion of reactivity or the release of radioactive material from an experiment undergoing activation in the reactor. For an experiment failure in which reactivity could be added, the worst possible case would be the prompt addition of less than 0.36% $\Delta k/k$ (~\$0.51) in either Exposure Room #1 or #2. As discussed for the case of improper fuel loading (Section 6.2.3), the addition of 0.36% $\Delta k/k$ (~\$0.51) would be within the range of an improper fuel loading condition. Such an addition would not result in any damage to the reactor or the fuel. For an experiment failure in which radioactive material could be released from the experiment, i.e., activation products, the worst case would be the prompt release of the radioactive material to the atmosphere. An authorized experiment involves the irradiation of 20 liters of argon gas for one hour at a power level of 1 MW. The resulting activation would result in a total Ar^{41} activity of 5.6 Ci in the sealed container. If the container should fail and release all of the Ar^{41} activity, the resulting total whole body dose would be less than 2.7 mrad to an individual more than 35 meters from the reactor stack (Equation 3, Section 6.3.4.1). The failure of this authorized experiment represents the worst case for radiological consequences from an experiment failure in the AFRRI-TRIGA reactor. Such a whole body exposure would not represent an undue risk to the health and safety of the general public.

6.3 NONROUTINE OPERATIONAL RISKS

6.3.1 Radioactive Contamination of Reactor Shielding Water

Contaminant material susceptible to neutron irradiation in the shield water is maintained at low concentrations by the water purification system and an in-line set of particulate filters which remove particulates of 5 microns or larger. The consequences associated with a failure of the fuel-element cladding and subsequent fission product contamination of the water have been calculated and studied experimentally, as described in Section 6.3.2.1. The results (1962 FSAR)[5] show that in the improbable event of fuel element cladding failure, the water can be decontaminated by the resins of the purification system. Manufacturing inspection and quality controls assure that the possibility of cladding failure is minimal.

Experiments conducted over a period of 11 years by General Atomics[2] on 8.5 wt% U-ZrH (8.5 weight percent uranium) fuel elements under various conditions have shown that a small fraction of fission products are released from U-ZrH fuel into the gap between fuel and cladding. This release fraction varies from less than 1.5×10^{-5} for an irradiation temperature of 350°C to a theoretical maximum of 1.0×10^{-2} at 800°C.

Three mechanisms (recoil, diffusion, and dissolution) have been shown to be involved in fission product migration. The first mechanism is the fission fragment recoil into the gap between the fuel and cladding. This effect predominates at temperatures up to approximately 400°C. In this range, the recoil release rate is dependent on the fuel surface-to-volume ratio but is independent of temperature. This is the most important mechanism in this system and is considered in the calculations dealing with fission product release following fuel element cladding failure (Section 6.3.2). At temperatures above approximately 400°C, the diffusion-like process predominates. The amount released is dependent on the fuel temperature, the fuel surface-to-volume ratio, the length of time of irradiation, and the isotope half-life. This mechanism is of little significance to the fission-product release fraction considered in this system. The U-ZrH fuel is relatively chemically inactive in water, steam, and air at temperatures up to about 850°C [2,13]. Massive zirconium hydride has been heated in air for extended periods of time at temperatures up to 600°C with negligible loss of hydrogen.[2] This negligible loss of hydrogen is due to oxide film formation which inhibits hydrogen loss.

6.3.2 Fuel Element Cladding Failure

6.3.2.1 Summary of Previous Experience

Since the original development of the TRIGA reactor, General Atomics has maintained a program of research and development oriented toward continuing improvement in the design of the fuel elements so that TRIGA reactors could be operated in both steady-state and pulsing modes without undue concern regarding the integrity of the fuel.

Extensive operational testing of TRIGA fuel elements has been conducted by General Atomics in the Torrey Pines TRIGA and the prototype TRIGA Mark-F since the early 1960s. Experiments on fuel with 8.5 wt% uranium were conducted over a period of 11 years under a variety of conditions. These experiments included:[2]

- o 1960 - The measurement of the quantity of a single fission product isotope release from a full-size TRIGA fuel element during irradiation
- o 1966 - The measurement of the fractional release of several isotopes from small specimens of TRIGA fuel material during and after irradiation at temperatures ranging from approximately 25°C to 1100°C
- o 1971 - The measurement of the quantities of several

fission product isotopes released from a full-size TRIGA fuel element during irradiation in a duplication of the 1960 experiment.

In addition, quench tests were performed on TRIGA low-enriched uranium (LEU) fuel samples for temperatures ranging from 800° to 1200°C. The subsequent fuel samples quenched from 800°, 1000°, 1100°, and 1200°C showed remarkably benign response to the test conditions.[3]

In the course of developing a reactor for pulse operation, General Atomics has experienced three fuel element cladding failures in their Torrey Pines TRIGA reactor. These cladding failures have been of two types:

- Two aluminum cladding failures resulted from mechanical ratcheting of the cladding material induced by thermal cycling
- A cladding failure associated with an aluminum-clad element instrumented with internal thermocouples.

In the original TRIGA fuel element design, aluminum cladding was used and mechanical ratcheting occurred during pulse operations. Modification of the standard TRIGA fuel element design by replacing aluminum with stainless steel as the cladding material during the mid-1960s eliminated this cause of failure. The present stainless steel cladding has greater durability than the original aluminum which experienced the metal cladding failures. The other cladding failure involving an instrumented element was apparently the result of either internal pressure buildup or water seepage.

The consequences of these cladding failures at General Atomics were all minor and in the worst case experience (which involved the cladding failure of an element instrumented with internal thermocouples and which had experienced a steady-state burnup of ~22 MW-hrs and a pulse burnup of ~3 MW-hrs involving pulses as high as \$4.00) may be summarized as follows:

- The activity in the reactor water reached a maximum of 0.2 $\mu Ci/cm^3$. It decayed very rapidly and was measured 24 hours after the cladding failure to be 5 x 10^{-5} $\mu Ci/cm^3$, a 4,000-fold reduction
- The activity in the air of the reactor room reached about 10 times the maximum permissible concentration for fission products and then decayed rapidly. The faulty fuel element was removed and experiments were resumed 2 hours after the activity release. The maximum integrated exposure to operating personnel resulting from this release was 1 mrem
- The noble gases were not collected on the air monitor filters used, but it may be inferred from the nature of the particulates collected that only noble gas fission products escaped from the TRIGA pool in significant proportions when the fuel cladding failed.

It was concluded from these General Atomics experiments and actual cladding failure experiences that a cladding failure or even the simultaneous failure of the cladding of several fuel elements would not constitute an undue risk to the health and safety of the operating personnel or the general public.

6.3.2.2 Calculation of Maximum Fission Product Release After a Fuel Element Cladding Failure

Calculations and a related experiment have been made to determine the maximum concentration of fission products that might be present in the reactor room air following a fuel element cladding failure.

The calculations are based on the fact that as the reactor operates, the fission products will build in the uranium-zirconium fuel mixture until an equilibrium concentration is reached for each nuclide. The resulting equilibrium nuclide concentration of fission products depends upon the total energy release in the reactor, the decay process for each nuclide, and the yield of the species from fission. Only the kryptons, the xenons, and the iodines will migrate into the gap between the fuel material and fuel element cladding. A Design Basis Accident (Section 6.3.4) is a fuel element cladding failure during a pulse operation or inadvertent transient following steady-state operation.

Calculated Fission Product Inventory

To determine the various inventories of fission products produced in the core, data were used from Reference 7 for infinite steady-state power operation at 1 MW(t). The resultant full power steady-state inventories for the kryptons, xenons, and iodines were calculated and are shown in Table 1, Appendix C. The associated fission product inventory for an assumed 40 MW-sec maximum pulse was also calculated using the data from Reference 7 with the buildup and decay of activities from Reference 8 considering only prompt fission events in the pulse. The resultant 40 MW-sec pulse inventories for kryptons, xenons, and iodines are shown in Table 2, Appendix C. For those isotopes with relatively long half-lives, low fission yields, or complex decay chains, the activity present from pulse operation will not be significant. Such isotopes include Kr-83m, Kr-85, Xe-131m, Xe-133, Xe-133m, and I-131.

These inventories are utilized in the calculation of radiological consequences associated with the various fuel element cladding failures which may result during the operation of the AFRRI-TRIGA reactor. The gamma source strengths were derived using gamma energy decay data from Reference 4 and are shown in Tables 1 and 2, Appendix C.

<u>Determination of Fission Products in Gap</u>[2]

In order to determine the actual percentage of fission-product gases that escape from the fuel material and collect in the gap between the cladding and the fuel material, experiments were conducted in the TRIGA reactor at General Atomics. A fuel element was fabricated with a sealed tube that vented the gap to a charcoal-filled cold trap at the surface of the reactor tank. All of the fission-product gases that accumulated in the gap were collected in the liquid-air-cooled charcoal trap by purging the system with helium, and the trap was analyzed. This measured amount of radioactive noble gases enabled the determination of the fraction of the fission products that diffused through the uranium-zirconium hydride material into the gap.

Although the measured amount of radioactive noble gases for the operating conditions in the AFRRI reactor fuel would indicate a gap activity percentage of less than 0.01 percent, the theoretical limit of 0.1 percent gap activity for fission product gases of noble gases and iodines, as stated in Reference 2, will be used in the consequence analysis for the Design Basis Accidents (Section 6.3.4).

Air and Water Activity Following Cladding Failure

As shown in Table 1, Appendix C, the total quantity of all gaseous fission products in a TRIGA core for 1 MW operation at equilibrium is [redacted]. For the purposes of this calculation, it is conservatively assumed that the fractions of the iodine, krypton, and xenon isotopes produced that collect in the gap between the fuel material and fuel cladding and, therefore, available for release are 0.1 percent, corresponding to a fuel temperature of 600°C.[2] Thus, the total core activity in the gap is calculated to be [redacted]. The maximum amount of fission products that could be released in the event of a cladding failure of a single average fuel element for an average 85 fuel element AFRRI-TRIGA core is less than [redacted] during steady-state operation. As shown in Table 2, Appendix C, the total quantity of all gaseous fission products from a 40 MW-sec pulse operation is [redacted] The additional release in the event of a cladding failure in an average fuel element is less than [redacted] The total release from a cladding failure event is approximately [redacted] from an average fuel element during pulse operation following steady-state operation.

As given in Reference 5, the volume of water in the TRIGA reactor tank is approximately 5.7×10^7 cm^3 and the volume of air in the reactor room is approximately 9.2×10^8 cm^3. For the purpose of calculation, it is assumed that all the gaseous fission products in the gap are available for release. As concluded from measurements of the worst cladding failure experienced (Section 6.3.2.1), only noble gas fission products would be expected to escape from the TRIGA pool when the fuel cladding fails. Due to the low pressure in the gap and the small gap size in a TRIGA fuel element, any iodine released from the fuel element due to a cladding failure is expected to be completely dissolved in the pool water. However, for calculation purposes, it is conservatively assumed that as much as 0.2 percent of the iodine released from the fuel element due to a cladding failure and all of the kryptons and xenons could be released into the reactor room atmosphere. The release assumptions are used to determine the radiological consequences due to the Design Basis Accident of a Fuel Element Cladding Failure (Section 6.3.4.2). Of the [redacted] that could be released from the gap in the cladding failure of an average fuel element, less than [redacted] would be radioiodine while the remaining [redacted] would be krypton and xenon. Therefore, the concentration in the water would be less than 0.05 $\mu Ci/cm^3$, while the air concentration would be less than 0.005 $\mu Ci/cm^3$.

Calculation of Doses Following Cladding Failure

Since krypton and xenon are inert gases, the exposure due to their presence in air is from submersion. To calculate the exposure rate, Equation 2 for submersion exposure from a spherical source is used. The gamma source strengths for the kryptons, xenons, and iodines are given in Table 1, Appendix C, for the steady-state operation and Table 2, Appendix C, for the pulse operation. As previously determined, the volume of the reactor room is approximately 9.2×10^8 cm^3, which is equivalent to a sphere with a radius (R) of 600 cm. The gamma source strength from kryptons and xenons for steady-state operation is calculated as 1.04×10^{16} MeV/sec, while for pulse operation it is calculated as 0.04×10^{16} MeV/sec, or a total of 1.1×10^{16} MeV/sec. Using the gap activity as 0.1 percent and an 85 fuel element core, the cladding failure of an average fuel element will release 1.3×10^{11} MeV/sec of kryptons and xenons into the reactor room volume. The air concentration of 0.005 $\mu Ci/cm^3$ previously calculated is equivalent to a gamma volume source strength (S_v) of 1.6×10^2 MeV/cm^3-sec.

For the noble gas mixture, the linear attenuation for air (μ) is 9.7×10^{-5} cm^{-1} and the flux-to-dose conversion factor (K) is 5×10^5 MeV/cm^2-sec per rem/hr assuming a 0.7 MeV gamma photon. Substituting these values into equation (2) results in an exposure rate in the reactor room from the calculated noble gas concentration of about 0.2 rem/hr. Within an hour, the exposure rate in the reactor room will be reduced to less than 0.1 rem/hr.

Based upon these conservative calculations for a fuel element cladding failure during pulse operation following steady-state operation, a person could remain in the reactor room for more than 50 hours before exceeding the allowable yearly occupational exposure limit of 5 rems (TEDE).

Standard operational procedures require prompt evacuation of personnel following an indication of excessive airborne radioactivity in the reactor room. Reentry to the reactor room would be determined by the Reactor Facility Director (RFD) when the airborne concentration was safe for normal operations to resume.

6.3.3 Loss of Shielding and Cooling Water

It is highly improbable that the reactor will lose the pool shielding and cooling water. Loss of this water can occur by only two means: the tank may be pumped dry, or a tank rupture may allow the water to escape in a relatively short period of time. However, should such an event occur during the operation of the reactor, the reactor is designed to shut down if the pool water level is 14 feet or less above the core floor; and in any case, the loss of moderation will stop the fission process.

In the event that the only mechanism for heat removal after the loss of coolant is natural convection of air through the core, the fuel element cladding would not fail and the fission products would be retained within the fuel element.[2]

In the event that a rupture of the reactor tank should occur and repair of the reactor tank wall is required, the core and any stored fuel elements that are present would be moved behind the lead shield doors, so repairs can be made safely.

The predicted gamma dose rates that would result from loss of pool water before the reactor core could be moved behind the lead shield doors would be approximately 300 mrem/hr near the reactor pool (10 feet) due to skyshine (air scatter), and 3.2 rem/hr on the roof above the reactor due to direct gamma exposure. Predicted whole body dose rates to individuals within the AFRRI facility, but outside the reactor room, should be less than 10 mrem/hr. Whole body dose rates to individuals more than 20 meters outside the east wall of the reactor room should be less than 1 mrem/hr. These dose rates would be reduced by a factor of 4 within 1 hour due to decay of the fission products in the reactor core. The assumptions, equations, and attendant calculations are given in Appendix C.

6.3.4 Design Basis Accidents

6.3.4.1 Fuel Element Drop Accident

A Design Basis Accident (DBA) for the AFRRI reactor is postulated to be the occurrence of a cladding failure of a fuel element after a 2-week period where the saturated fission product inventory of a 1 MW steady-state operation (100 hours to reach saturated fission product inventory) has been allowed to decay after being taken out of the operating core and placed in storage. The cladding failure could occur when the fuel element is withdrawn from the reactor pool. When the fuel element is exposed to air, a cladding failure could occur coincidentally, or due to a drop. The probability of such an accident is considered to be extremely remote. The probability of cladding failure has been further reduced under the postulated accident conditions by the substitution of stainless steel cladding for the former aluminum cladding on the fuel elements. The fission products released from the gap will depend upon the temperature of the fuel following 2 weeks decay. This temperature is expected to be less than 50°C. The temperature needed to volatilize iodine (183°C) is, therefore, not reached and gaseous iodine should not be released. The kryptons and xenons will normally remain in the gaseous state and would be released (100 percent). Although iodine will not be volatile under the assumed accident conditions, a release of 1 percent of the gap activity has been assumed for calculational purposes.

The diffusion factor (x/Q) and finite cloud correction factors (F) are dependent upon the distance from the source and are presented in Table 3, Appendix C. These parameters were based upon a Pasquill type F stability condition with a 1 m/sec wind speed and a cross sectional area for the AFRRI facility of 450 m^2. The methodology described in Reference 9 was used to determine the diffusion factor. The methodology described in Reference 10 for defining the ratio of gamma dose from a finite cloud to an infinite cloud with the same centerline concentration was used to determine the finite cloud correction factor. Table 4, appendix C presents the source terms of radioactivity and thyroid and gamma source strengths in the total core for 1 MW steady-state operation after 2 weeks of decay. These values are to be reduced to 0.1 percent for the gap activity and for the 85 fuel element core to determine the amount of fission products that might be released in an average fuel element drop accident resulting in cladding failure. The total release is less than 0.15 curies with a gamma source strength of

3.6×10^8 MeV/sec. The iodine release fraction is assumed to be 1 percent for a thyroid source strength of 1.3×10^3 rads. To calculate the whole body gamma dose to an individual outside the AFRRI facility, the following equation was utilized:[11]

$$D_\gamma = 0.25\,(E_\gamma)(Q)(\chi/Q)(F) \tag{3}$$

where:

D_γ = integrated whole body gamma dose (rad)

E_γ = gamma energy released per disintegration (MeV/dis)

Q = total activity released (Ci)

χ/Q = diffusion coefficient (sec/m^3)

F = ratio of gamma dose from finite cloud to infinite cloud with same centerline concentration

0.25 = combined conversion factor and constant for semi-infinite cloud dose (rad-m^3-dis/MeV-sec-Ci)

To calculate the thyroid dose to an individual outside the AFRRI facility, the following equation was utilized:[11]

$$D_{th} = (BR)(\chi/Q)(S_{th}) \tag{4}$$

where:

D_{th} = integrated thyroid dose (rad)

BR = standard man breathing rate = 3.47×10^{-4} m^3/sec

χ/Q = diffusion coefficient (sec/m^3)

S_{th} = (DCF)(Q), iodine inhalation source term (rad)

DCF = dose conversion factor (rad/Ci)

Q = iodine activity released (Ci).

The whole body and thyroid doses calculated for individuals downwind from the AFRRI facility in the event of a fuel element drop accident are given in Table 5, Appendix C. The conservative assumption is made that the radioactive gases released in the AFRRI reactor room will be released directly to the atmosphere without significant holdup within the facility. The current design of the AFRRI reactor room would cause isolation of the reactor room by automatic closure of the ventilation pathway to the reactor stack and would prevent excessive leakage to other parts of the AFRRI facility by the airtight access doors.

The calculated whole body dose is insignificant at all distances downwind from the AFRRI facility. The maximum calculated thyroid dose of 45 mrad is more than a factor of 100 below the Protective Action Guide (PAG) thyroid committed dose equivalent of 5 rem, below which no protective action is recommended.[12] The calculated thyroid dose for individuals beyond the boundary of the National Naval Medical Center site within which the AFRRI facility is located would be significantly less than 1 mrad.

6.3.4.2 Fuel Element Cladding Failure Accident

Another Design Basis Accident for the AFRRI reactor is postulated to be the cladding failure of a fuel element during a pulse operation or inadvertent transient following steady-state operation at 1 MW. The assumed fission product inventories in the core are those given previously in Table 1, Appendix C (1 MW operation) and Table 2, Appendix C (40 MW-sec pulse operation), and analyzed in Section 6.3.2. As discussed for the failure of cladding in a fuel element drop accident, the likelihood of occurrence of a fuel element cladding failure during normal operation or an inadvertent transient is considered to be extremely remote. The data given in Table 3, appendix C as well as equations (3) and (4) will be used in the evaluation of this DBA to calculate the whole body and thyroid doses to individuals downwind from the AFRRI facility.

Table 6, Appendix C, presents the source terms of radioactivity and thyroid and gamma source strengths postulated to be released to the atmosphere from the AFRRI reactor room. The resulting whole body and thyroid doses calculated for individuals downwind from the AFRRI facility are given in Table 7, Appendix C. The same conservative assumption made for the fuel element drop accident regarding the prompt release of radioactive material to the atmosphere has been made for this DBA..

The calculated whole body dose is insignificant at all distances downwind from the AFRRI facility. The maximum calculated whole body dose is less than 2 mrad or a factor of 500 below the PAG whole body dose of 1 rem, below which no protective action is recommended.[12] The maximum calculated on-site thyroid dose of 57 mrad is nearly a factor of 90 below the PAG thyroid committed dose equivalent of 5 rem, below which no protective action is recommended.[12] The calculated thyroid dose for individuals beyond the boundary of the NNMC site would be well below one mrad.

6.4 REFERENCES

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7.0 ADMINISTRATION

7.1 GENERAL

The Armed Forces Radiobiology Research Institute (AFRRI) is designated as a subordinate unit of the Uniformed Services University of the Health Sciences.

The Uniformed Services University of the Health Sciences (USUHS) is a subordinate command of the Department of Defense and, as such, is under the direction, authority, and control of the Secretary of Defense. The mission of the USUHS is to educate and train competent medical personnel qualified to serve the needs of the Uniformed Services of the United States through providing the highest quality education programs in the health sciences including conducting medical readiness training and continuing education programs.

The mission of AFRRI is to conduct scientific research in the field of radiobiology and related matters essential to the medical support of the Department of Defense.

The main organization of AFRRI consists of the Director, the Scientific Director, and the supporting staff. A Board of Governors is also established to advise the Director of AFRRI in matters of professional policy direction and related issues. The Board of Governors consists of the Director, Defense Nuclear Agency as Chairman, and the Surgeons General of the Army, Navy, and Air Force. The Board of Governors ensures equal support and participation by the three services in Institute activities and a balanced response by the Institute to the needs of each service.

The supporting staff of AFRRI consists of both military and civilian personnel. In accordance with the triservice nature of the Institute, military personnel are provided on an approximately equal basis by each military department. The cost of maintaining and operating AFRRI is borne by USUHS. The size of the staff is determined by a Joint Table of Distribution, which is developed by the President of USUHS and approved by the Joint Chiefs of Staff. The general organization of AFRRI is shown in Figure 7-1.

7.2 THE OFFICE OF THE DIRECTOR

The Office of the Director of AFRRI consists of the Director, the Scientific Director, and the Chief of Staff. Two committees, the Reactor and Radiation Facility Safety Committee (RRFSC) and the Radionuclide and X-Ray Safety Committee (RXSC), report directly to the Director concerning safety matters at the Institute. The RRFSC is responsible for advising the Director in matters concerning the safe operation of the AFRRI-TRIGA reactor and shall be discussed later. The RXSC exercises no control over the reactor and its operations and, therefore, will not be discussed further.

The Director of AFRRI must be a medical department officer of either the Army, Navy, or Air Force, who is technically qualified in the fields of medicine, radiobiology, and radiation effects research. He is appointed by the President of USUHS after consultation with the Board of Governors. He is directly responsible to the President of USUHS for the overall administration and supervision of the Institute. Since the Director has overall responsibility for the safe operation of the AFRRI-

TRIGA reactor, he is authorized to communicate directly with the Nuclear Regulatory Commission without prior knowledge or approval of any other office, including USUHS headquarters. The Scientific Director assumes the responsibilities of the Director in his absence.

The Scientific Director must be technically and administratively qualified in the fields of radiation, radiobiology, and radiation effects research. He must have recognized capabilities, such as demonstrated experience in managing research programs, high professional stature as evidenced by publications in technical journals and appointments or membership in professional societies, and additional education above the Bachelor's degree level. He is directly responsible to the Director of AFRRI for the overall development and coordination of the multidisciplined research programs conducted at AFRRI. These responsibilities include advising the Director on technical matters, providing technical direction to the diverse professional staff at AFRRI, and acting as liaison between AFRRI and other agencies.

7.3 REACTOR AND RADIATION FACILITY SAFETY COMMITTEE

The Reactor and Radiation Facility Safety Committee (RRFSC) is directly responsible to the Director of AFRRI. The committee oversees the safety aspects of the following facilities at the Institute:

- o AFRRI-TRIGA Reactor
- o LINAC Linear Accelerator

- Cobalt Facility
- X-Ray Facility
- Any other machine capable of producing radiation in excess of 1 rad/hr in an area which can be occupied at any time.

The safety aspects of these facilities include the physical facilities, the planned operations, and the qualifications of supervisory and operating personnel which relate to the safety of the Institute, its staff, the public, and the environment. With respect to the AFRRI-TRIGA reactor, the RRFSC reviews all radiological health and safety matters concerning the reactor and its associated equipment.

Regular (Permanent) RRFSC membership shall be composed of:

- Chairman, as appointed by the AFRRI Director
- AFRRI Radiation Protection Officer
- Reactor Facility Director, AFRRI
- One to three non-AFRRI members, as appointed by the AFRRI Director, who are knowledgeable in fields related to reactor safety. At least one shall be a reactor operator or a health physics specialist.

Special (temporary) RRFSC members may include:

- Other knowledgeable persons, as appointed by the AFRRI Director, to serve as alternates for the one to three non-AFRRI regular members listed above.
- Voting ad hoc members, invited by the Director of AFRRI, to assist in review of a particular problem.

The RRFSC or a subcommittee thereof meets at least four times each calendar year. The full RRFSC meets at least semiannually. A quorum of the RRFSC for review consists of the Chairman of RRFSC (or his designated alternate) and two other members (or alternate members), one of which must be a non-AFRRI member. A majority of those present must be regular members. All RRFSC decisions are binding upon being endorsed by the Director of AFRRI.

7.4 RADIATION SCIENCES DEPARTMENT

The Radiation Sciences Department (RSD) supports the AFRRI scientific research program by providing, operating, and maintaining radiation sources, which include the AFRRI-TRIGA reactor.

The Radiation Sciences Department (RSD) has sole responsibility for the operation and maintenance of the AFRRI-TRIGA reactor and other assigned radiation sources. RSD is composed of two divisions, each division being responsible for one or more radiation sources. The Reactor Division is the division within RSD responsible for the operation and maintenance of the AFRRI-TRIGA reactor.

The Reactor Division of the Radiation Sciences Department operates, calibrates, and maintains the AFRRI-TRIGA reactor and associated systems in compliance with appropriate regulations. The Reactor Division consists of the Reactor Facility Director, the Reactor Operations Supervisor, and the Reactor Staff. The chain of command of the Reactor Division is given in Figure 7-2.

The Reactor Facility Director (RFD) is directly responsible to the AFRRI Director for operational, technical, and safety matters pertaining to the utilization of the AFRRI-TRIGA reactor, and for ensuring compliance with NRC licenses and regulations as well as internal reactor operational procedures and AFRRI instructions. The Reactor Facility Director is responsible to the Chairman of the Radiation Sciences Department for administrative matters. The RFD shall possess a NRC Senior Reactor Operator License for the AFRRI-TRIGA Mark-F reactor.

The Reactor Operations Supervisor (ROS) is responsible to the RFD for the efficient and safe operation of the AFRRI-TRIGA reactor on a daily, routine basis. The ROS shall possess a NRC Senior Reactor Operator License for the AFRRI-TRIGA reactor.

Senior Reactor Operators (SROs) and Reactor Operators (ROs) are directly responsible to the Reactor Operations Supervisor for the safe and legal operation of the AFRRI-TRIGA reactor in compliance with NRC licenses and regulations, and internal reactor operational procedures and AFRRI instructions. SROs shall possess a NRC Senior Reactor Operator License; ROs shall possess a NRC Reactor Operator License for the AFRRI-TRIGA reactor.

7.5 SAFETY AND HEALTH DEPARTMENT

The Safety and Health Department (SHD) is responsible for developing and maintaining a comprehensive Health Physics Program, encompassing all sources of radiation within AFRRI. The AFRRI Radiation Protection Officer, also called the Radiation Safety Officer, must be academically and technically qualified in the fields of radiological safety and health physics as they relate to reactor operations. He is directly responsible to the AFRRI Director for radiation protection matters. If possible, he should be a Board Certified Health Physicist of the American Board of Health Physics. As Radiation Protection Officer, he serves as a permanent voting member of the Reactor and Radiation Facility Safety Committee.

The Safety and Health Department is responsible for developing and maintaining an adequate radiation protection program for the Institute. This includes maintaining complete personnel radiation exposure files, implementing and maintaining an ALARA program, i.e., reducing radiation exposures to the staff and the public to levels as low as reasonably achievable (ALARA), supervising adequate environmental monitoring programs, and ensuring that all applicable radiological safety regulations and directives of the Department of Defense, the Nuclear Regulatory Commission, and other agencies are followed. To ensure continuity in the area of radiation protection at AFRRI, particular attention is given to centering responsibility for radiological safety in experienced and highly qualified civilian personnel where possible.

7.6 NATIONAL NAVAL MEDICAL CENTER

The National Naval Medical Center (NNMC) is one of the major medical research centers in the greater Washington area. The close proximity of NNMC, the National Institutes of Health, and the National Library of Medicine to AFRRI gives the AFRRI staff the advantage of having ready access to a well-established and extensive research community.

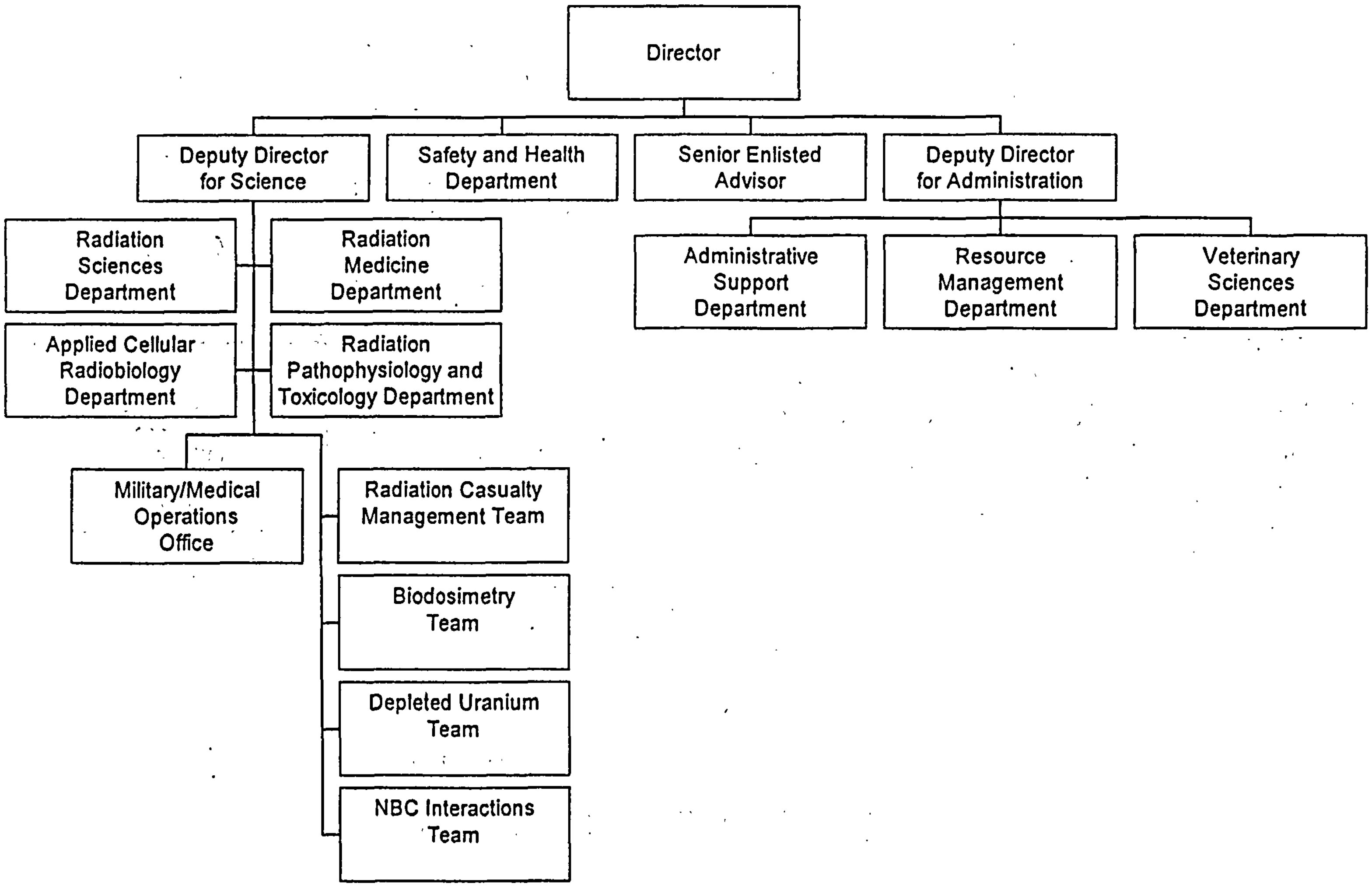
The National Naval Medical Center hosts several medical commands and organizations in the military. These may or may not be under the direct military command of NNMC. The separate military commands and units at NNMC are as follows:

- Armed Forces Radiobiology Research Institute (AFRRI)
- National Naval Dental Center (NNDC)
- Naval Health Sciences Education and Training Command (HSETC)
- Naval Medical Information Management Center
- Naval Medical Research and Development Command (NMRDC)
- Naval Medical Research Institute (NMRI)
- Naval School of Health Sciences (NSHS)
- Personnel Support Detachment (PSD)
- Uniformed Services University of the Health Sciences (USUHS)
- Naval Dosimetry Center

The National Naval Medical Center itself falls under the command and coordination control of the Commandant, Naval District, Washington, D.C.

AFRRI is housed at NNMC under the conditions of a host-tenant agreement. This agreement specifies that, although AFRRI is under the military command of the Uniformed Services University of the Health Sciences, NNMC provides certain logistical support to AFRRI on either a reimbursable or nonreimbursable basis. NNMC supplies to the military personnel of AFRRI certain services such as officer and enlisted quarters as available; medical and dental treatment; and access to the Center's stores, services, and recreation facilities. NNMC specifically supplies to AFRRI certain services such as road, parking, and ground maintenance; trash removal; maintenance of AFRRI facilities; utility services such as electricity, sewage, and water; security protection for AFRRI's perimeter and security personnel in response to emergency security situations at AFRRI; and fire protection which includes fire safety inspections, training of AFRRI personnel in fire fighting techniques, routine maintenance of AFRRI fire extinguishers, and fire fighting equipment and personnel in response to a fire at AFRRI.

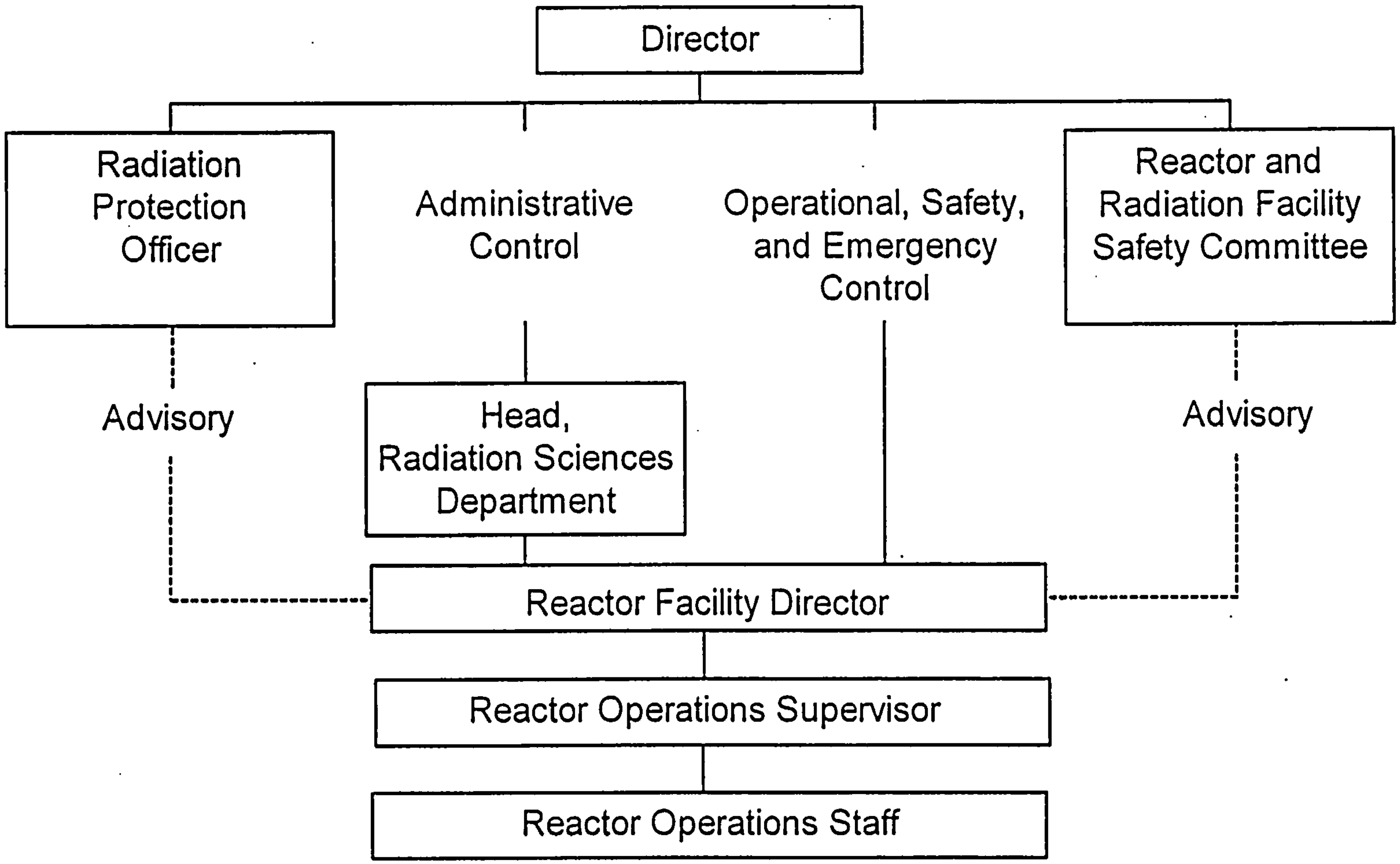
Armed Forces Radiobiology Research Institute
Organization Structure
Figure 7-1



Reactor Division

Organization Structure

Figure 7-2



APPENDIX A

Joint Wind Frequency Distribution for 1960-64 and 1964 Inclusive

TABLE 1

Joint Wind Frequency Distribution
for 1960-1964 Inclusive

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

STABILITY CLASS: PASQUILL A
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 10/09/80. 20.02.07.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	1 .30 .00	11 3.34 .03	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	12 3.65 .03	2.22
NE	5 1.52 .01	18 5.47 .04	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	23 6.99 .05	2.08
ENE	5 1.52 .01	20 6.08 .05	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	25 7.60 .06	2.10
E	4 1.22 .01	14 4.26 .03	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	18 5.47 .04	2.11
ESE	3 .91 .01	23 6.99 .05	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	26 7.90 .06	2.26
SE	5 1.52 .01	16 4.86 .04	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	21 6.38 .05	2.17
SSE	1 .30 .00	12 3.65 .03	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	13 3.95 .03	2.25
S	5 1.52 .01	24 7.29 .05	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	29 8.81 .07	2.15
SSW	0 0.00 0.00	9 2.74 .02	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	9 2.74 .02	2.38
SW	1 .30 .00	5 1.52 .01	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	6 1.82 .01	2.25
WSW	4 1.22 .01	4 1.22 .01	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	8 2.43 .02	1.92
W	1 .30 .00	14 4.26 .03	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	15 4.56 .03	2.36
WNW	5 1.52 .01	8 2.43 .02	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	13 3.95 .03	2.10
NW	3 .91 .01	10 3.04 .02	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	13 3.95 .03	2.23
NNW	4 1.22 .01	11 3.34 .03	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	15 4.56 .03	2.11
N	5 1.52 .01	20 6.08 .05	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	25 7.60 .06	2.28
CALM	58 17.63 .13						58 17.63 .13	CALM
TOTAL	110 33.43 .25	219 66.57 .50	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	329 100.00 .75	1.80

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 1 (cont'd)

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

STABILITY CLASS: PASQUILL B
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 10/09/80. 20.02.07.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	35	41	72	0	0	0	148	2.71
	1.28	1.50	2.64	0.00	0.00	0.00	5.42	
	.08	.09	.16	0.00	0.00	0.00	.34	
NE	44	58	76	0	0	0	178	2.60
	1.61	2.13	2.78	0.00	0.00	0.00	6.52	
	.10	.13	.17	0.00	0.00	0.00	.41	
ENE	61	60	57	0	0	0	178	2.37
	2.24	2.20	2.09	0.00	0.00	0.00	6.52	
	.14	.14	.13	0.00	0.00	0.00	.41	
E	39	61	33	0	0	0	133	2.33
	1.43	2.24	1.21	0.00	0.00	0.00	4.87	
	.09	.14	.08	0.00	0.00	0.00	.30	
ESE	53	50	39	0	0	0	142	2.26
	1.94	1.83	1.43	0.00	0.00	0.00	5.20	
	.12	.11	.09	0.00	0.00	0.00	.32	
SE	34	48	51	0	0	0	133	2.53
	1.25	1.76	1.87	0.00	0.00	0.00	4.87	
	.08	.11	.12	0.00	0.00	0.00	.30	
SSE	43	61	125	0	0	0	229	2.80
	1.58	2.24	4.58	0.00	0.00	0.00	8.39	
	.10	.14	.29	0.00	0.00	0.00	.52	
S	77	98	373	0	0	0	548	3.07
	2.82	3.59	13.67	0.00	0.00	0.00	20.08	
	.18	.22	.85	0.00	0.00	0.00	1.25	
SSW	24	24	47	0	0	0	95	2.70
	.88	.88	1.72	0.00	0.00	0.00	3.48	
	.05	.05	.11	0.00	0.00	0.00	.22	
SW	30	17	37	0	0	0	84	2.46
	1.10	.62	1.36	0.00	0.00	0.00	3.08	
	.07	.04	.08	0.00	0.00	0.00	.19	
WSW	24	20	58	0	0	0	102	2.79
	.88	.73	2.13	0.00	0.00	0.00	3.74	
	.05	.05	.13	0.00	0.00	0.00	.23	
W	19	13	67	0	0	0	99	2.99
	.70	.48	2.46	0.00	0.00	0.00	3.63	
	.04	.03	.15	0.00	0.00	0.00	.23	
WNW	18	24	67	0	0	0	109	3.07
	.66	.88	2.46	0.00	0.00	0.00	3.99	
	.04	.05	.15	0.00	0.00	0.00	.25	
NW	19	26	91	0	0	0	136	3.08
	.70	.95	3.33	0.00	0.00	0.00	4.98	
	.04	.06	.21	0.00	0.00	0.00	.31	
NNW	16	30	77	0	0	0	123	3.03
	.59	1.10	2.82	0.00	0.00	0.00	4.51	
	.04	.07	.18	0.00	0.00	0.00	.28	
N	31	48	91	0	0	0	170	2.81
	1.14	1.76	3.33	0.00	0.00	0.00	6.23	
	.07	.11	.21	0.00	0.00	0.00	.39	
CALM	122						122	CALM
	4.47						4.47	
	.28						.28	
TOTAL	689	679	1361	0	0	0	2729	2.65
	25.25	24.88	49.87	0.00	0.00	0.00	100.00	
	1.57	1.55	3.11	0.00	0.00	0.00	6.23	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 1 (cont'd)

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

STABILITY CLASS: PASQUILL C
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 10/09/80. 20.02.07.

AFRRI-TRIGA REACTOR
BETHESDA, MD.
AFRRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
	WIND SPEED CATEGORIES(METERS PER SECOND)							
NNE	12	38	104	38	0	0	192	3.73
	.23	.74	2.02	.74	0.00	0.00	3.73	
	.03	.09	.24	.09	0.00	0.00	.44	
NE	13	72	100	15	1	0	201	3.30
	.25	1.40	1.94	.29	.02	0.00	3.90	
	.03	.16	.23	.03	.00	0.00	.46	
ENE	17	68	83	26	0	0	194	3.26
	.33	1.32	1.61	.50	0.00	0.00	3.77	
	.04	.16	.19	.06	0.00	0.00	.44	
E	16	44	60	15	0	0	135	3.25
	.31	.85	1.17	.29	0.00	0.00	2.62	
	.04	.10	.14	.03	0.00	0.00	.31	
ESE	7	35	44	11	0	0	97	3.37
	.14	.68	.85	.21	0.00	0.00	1.88	
	.02	.08	.10	.03	0.00	0.00	.22	
SE	24	21	57	9	2	0	113	3.24
	.47	.41	1.11	.17	.04	0.00	2.19	
	.05	.05	.13	.02	.00	0.00	.26	
SSE	7	67	140	34	0	0	248	3.65
	.14	1.30	2.72	.66	0.00	0.00	4.82	
	.02	.15	.32	.08	0.00	0.00	.57	
S	43	210	706	252	12	0	1223	3.98
	.83	4.08	13.71	4.89	.23	0.00	23.75	
	.10	.48	1.61	.58	.03	0.00	2.79	
SSW	22	77	206	74	13	1	393	3.99
	.43	1.50	4.00	1.44	.25	.02	7.63	
	.05	.18	.47	.17	.03	.00	.90	
SW	25	45	127	40	3	1	241	3.65
	.49	.87	2.47	.78	.06	.02	4.68	
	.06	.10	.29	.09	.01	.00	.55	
WSW	23	37	115	43	0	0	218	3.75
	.45	.72	2.23	.83	0.00	0.00	4.23	
	.05	.08	.26	.10	0.00	0.00	.50	
W	11	37	113	49	8	2	220	4.10
	.21	.72	2.19	.95	.16	.04	4.27	
	.03	.08	.26	.11	.02	.00	.50	
WNW	7	30	185	87	10	0	319	4.39
	.14	.58	3.59	1.69	.19	0.00	6.19	
	.02	.07	.42	.20	.02	0.00	.73	
NW	11	45	226	139	27	3	451	4.63
	.21	.87	4.39	2.70	.52	.06	8.76	
	.03	.10	.52	.32	.06	.01	1.03	
NNW	12	38	260	99	11	1	421	4.27
	.23	.74	5.05	1.92	.21	.02	8.17	
	.03	.09	.59	.23	.03	.00	.96	
N	11	55	232	91	5	0	394	4.07
	.21	1.07	4.50	1.77	.10	0.00	7.65	
	.03	.13	.53	.21	.01	0.00	.90	
CALM	90						90	CALM
	1.75						1.75	
	.21						.21	
TOTAL	351	919	2758	1022	92	8	5150	3.88
	6.82	17.84	53.55	19.84	1.79	.16	100.00	
	.80	2.10	6.29	2.33	.21	.02	11.75	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 1 (cont'd)

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

STABILITY CLASS: PASQUILL D
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 10/09/80. 20.02.07.

AFRRI-TRIGA REACTOR
BETHESDA, MD.
AFRRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES (METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	42	125	350	445	78	8	1048	4.89
	.21	.61	1.71	2.18	.38	.04	5.13	
	.10	.29	.80	1.02	.18	.02	2.39	
NE	54	195	464	517	81	12	1323	4.70
	.26	.95	2.27	2.53	.40	.06	6.47	
	.12	.45	1.06	1.18	.19	.03	3.02	
ENE	75	218	527	447	77	21	1365	4.54
	.37	1.07	2.58	2.19	.38	.10	6.68	
	.17	.50	1.20	1.02	.18	.05	3.12	
E	66	153	237	142	20	3	621	3.81
	.32	.75	1.16	.69	.10	.01	3.04	
	.15	.35	.54	.32	.05	.01	1.42	
ESE	65	123	178	94	28	2	490	3.77
	.32	.60	.87	.46	.14	.01	2.40	
	.15	.28	.41	.21	.06	.00	1.12	
SE	64	129	205	113	11	1	523	3.68
	.31	.63	1.00	.55	.05	.00	2.56	
	.15	.29	.47	.26	.03	.00	1.19	
SSE	33	142	262	133	30	9	609	4.13
	.16	.69	1.28	.65	.15	.04	2.98	
	.08	.32	.60	.30	.07	.02	1.39	
S	95	368	1238	1026	142	13	2882	4.56
	.46	1.80	6.06	5.02	.69	.06	14.10	
	.22	.84	2.83	2.34	.32	.03	6.58	
SSW	41	166	552	669	191	28	1647	5.17
	.20	.81	2.70	3.27	.93	.14	8.06	
	.09	.38	1.26	1.53	.44	.06	3.76	
SW	46	123	282	307	90	20	868	4.90
	.23	.60	1.38	1.50	.44	.10	4.25	
	.10	.28	.64	.70	.21	.05	1.98	
WSW	43	117	189	144	34	14	541	4.33
	.21	.57	.92	.70	.17	.07	2.65	
	.10	.27	.43	.33	.08	.03	1.23	
W	38	67	140	231	99	24	599	5.39
	.19	.33	.68	1.13	.48	.12	2.93	
	.09	.15	.32	.53	.23	.05	1.37	
WNW	27	85	245	786	407	151	1701	6.67
	.13	.42	1.20	3.85	1.99	.74	8.32	
	.06	.19	.56	1.79	.93	.34	3.88	
NW	28	76	354	1157	630	209	2454	6.75
	.14	.37	1.73	5.66	3.08	1.02	12.00	
	.06	.17	.81	2.64	1.44	.48	5.60	
NNW	22	81	342	900	391	78	1814	6.22
	.11	.40	1.67	4.40	1.91	.38	8.87	
	.05	.18	.78	2.05	.89	.18	4.14	
N	76	147	429	816	242	27	1737	5.41
	.37	.72	2.10	3.99	1.18	.13	8.50	
	.17	.34	.98	1.86	.55	.06	3.96	
CALM	220						220	CALM
	1.08						1.08	
	.50						.50	
TOTAL	1035	2315	5994	7927	2551	620	20442	5.20
	5.06	11.32	29.32	38.78	12.48	3.03	100.00	
	2.36	5.28	13.68	18.09	5.82	1.42	46.66	

KEY
XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 1 (cont'd)

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

STABILITY CLASS: PASQUILL F
DATA SOURCE: NATIONAL AIRPORT. WASH.. D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 10/09/80. 20.02.07.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
	WIND SPEED CATEGORIES(METERS PER SECOND)							
NNE	0	63	140	27	0	0	230	3.60
	0.00	.99	2.20	.42	0.00	0.00	3.61	
	0.00	.14	.32	.06	0.00	0.00	.52	
NE	0	69	157	28	0	0	254	3.50
	0.00	1.08	2.47	.44	0.00	0.00	3.99	
	0.00	.16	.36	.06	0.00	0.00	.58	
ENE	0	76	140	26	0	0	242	3.45
	0.00	1.19	2.20	.41	0.00	0.00	3.80	
	0.00	.17	.32	.06	0.00	0.00	.55	
E	0	51	62	6	0	0	119	3.18
	0.00	.80	.97	.09	0.00	0.00	1.87	
	0.00	.12	.14	.01	0.00	0.00	.27	
ESE	0	42	81	13	0	0	136	3.45
	0.00	.66	1.27	.20	0.00	0.00	2.14	
	0.00	.10	.14	.03	0.00	0.00	.31	
SE	0	79	108	13	0	0	200	3.24
	0.00	1.24	1.70	.20	0.00	0.00	3.14	
	0.00	.18	.25	.03	0.00	0.00	.46	
SSE	0	64	134	19	0	0	217	3.42
	0.00	1.01	2.10	.30	0.00	0.00	3.41	
	0.00	.15	.31	.04	0.00	0.00	.50	
S	0	232	691	132	0	0	1055	3.67
	0.00	3.64	10.85	2.07	0.00	0.00	16.57	
	0.00	.53	1.58	.30	0.00	0.00	2.41	
SSW	0	149	517	82	0	0	748	3.67
	0.00	2.34	8.12	1.29	0.00	0.00	11.75	
	0.00	.34	1.18	.19	0.00	0.00	1.71	
SW	0	156	203	24	0	0	383	3.26
	0.00	2.45	3.19	.38	0.00	0.00	6.02	
	0.00	.36	.46	.05	0.00	0.00	.87	
WSW	0	131	125	11	0	0	267	3.09
	0.00	2.06	1.96	.17	0.00	0.00	4.19	
	0.00	.30	.29	.03	0.00	0.00	.61	
W	0	58	137	24	0	0	219	3.57
	0.00	.91	2.15	.38	0.00	0.00	3.44	
	0.00	.13	.31	.05	0.00	0.00	.50	
WNW	0	51	316	100	0	0	467	4.01
	0.00	.80	4.96	1.57	0.00	0.00	7.33	
	0.00	.12	.72	.23	0.00	0.00	1.07	
NW	0	80	474	127	0	0	681	4.02
	0.00	1.26	7.44	1.99	0.00	0.00	10.70	
	0.00	.18	1.08	.29	0.00	0.00	1.55	
NNW	0	54	461	111	0	0	626	4.05
	0.00	.85	7.24	1.74	0.00	0.00	9.83	
	0.00	.12	1.05	.25	0.00	0.00	1.43	
N	0	95	335	93	0	0	523	3.83
	0.00	1.49	5.26	1.46	0.00	0.00	8.21	
	0.00	.22	.76	.21	0.00	0.00	1.19	
CALM	0						0	CALM
	0.00						0.00	
	0.00						0.00	
TOTAL	0	1450	4081	836	0	0	6367	3.68
	0.00	22.77	64.10	13.13	0.00	0.00	100.00	
	0.00	3.31	9.31	1.91	0.00	0.00	14.53	

KEY
XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 1 (cont'd)

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

STABILITY CLASS: PASQUILL F — AFFRI-TRIGA REACTOR
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C. — BETHESDA, MD.
WIND SENSOR HEIGHT: 7.00 METERS — AFFRI
TABLE GENERATED: 10/09/80, 20.02.07. — DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES (METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	83	109	44	0	0	0	236	2.10
	.94	1.24	.50	0.00	0.00	0.00	2.68	
	.19	.25	.10	0.00	0.00	0.00	.54	
NE	89	123	53	0	0	0	265	2.14
	1.01	1.40	.60	0.00	0.00	0.00	3.01	
	.20	.28	.12	0.00	0.00	0.00	.60	
ENE	111	114	46	0	0	0	271	2.06
	1.26	1.30	.52	0.00	0.00	0.00	3.08	
	.25	.26	.10	0.00	0.00	0.00	.62	
E	63	56	13	0	0	0	132	1.87
	.72	.64	.15	0.00	0.00	0.00	1.50	
	.14	.13	.03	0.00	0.00	0.00	.30	
ESE	66	48	24	0	0	0	138	1.92
	.75	.55	.27	0.00	0.00	0.00	1.57	
	.15	.11	.05	0.00	0.00	0.00	.31	
SE	91	92	31	0	0	0	214	1.99
	1.03	1.05	.35	0.00	0.00	0.00	2.43	
	.21	.21	.07	0.00	0.00	0.00	.49	
SSE	61	79	43	0	0	0	183	2.19
	.69	.90	.49	0.00	0.00	0.00	2.08	
	.14	.18	.10	0.00	0.00	0.00	.42	
S	350	478	268	0	0	0	1096	2.21
	3.98	5.43	3.05	0.00	0.00	0.00	12.46	
	.80	1.09	.61	0.00	0.00	0.00	2.50	
SSW	189	453	253	0	0	0	895	2.36
	2.15	5.15	2.88	0.00	0.00	0.00	10.17	
	.43	1.03	.58	0.00	0.00	0.00	2.04	
SW	348	533	147	0	0	0	1028	2.08
	3.96	6.06	1.67	0.00	0.00	0.00	11.68	
	.79	1.22	.34	0.00	0.00	0.00	2.35	
WSW	431	530	95	0	0	0	1056	1.97
	4.90	6.02	1.08	0.00	0.00	0.00	12.00	
	.98	1.21	.22	0.00	0.00	0.00	2.41	
W	197	216	69	0	0	0	482	1.99
	2.24	2.46	.78	0.00	0.00	0.00	5.48	
	.45	.49	.16	0.00	0.00	0.00	1.10	
WNW	120	196	106	0	0	0	422	2.24
	1.36	2.23	1.20	0.00	0.00	0.00	4.80	
	.27	.45	.24	0.00	0.00	0.00	.96	
NW	63	190	141	0	0	0	394	2.47
	.72	2.16	1.60	0.00	0.00	0.00	4.48	
	.14	.43	.32	0.00	0.00	0.00	.90	
NNW	98	207	131	0	0	0	436	2.35
	1.11	2.35	1.49	0.00	0.00	0.00	4.96	
	.22	.47	.30	0.00	0.00	0.00	1.00	
N	189	333	148	0	0	0	670	2.23
	2.15	3.78	1.68	0.00	0.00	0.00	7.62	
	.43	.76	.34	0.00	0.00	0.00	1.53	
CALM	880						880	CALM
	10.00						10.00	
	2.01						2.01	
TOTAL	3429	3757	1612	0	0	0	8798	1.95
	38.97	42.70	18.32	0.00	0.00	0.00	100.00	
	7.83	8.57	3.68	0.00	0.00	0.00	20.08	

KEY
XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 1 (cont'd)

JOINT WIND FREQUENCY DISTRIBUTION BY STABILITY CLASS
DATA PERIOD: 1960 THRU 1964

ALL CLASSES
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 10/09/80. 20.02.07.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	173 .39	387 .88	710 1.62	510 1.16	78 .18	8 .02	1866 4.26	4.07
NE	205 .47	535 1.22	850 1.94	560 1.28	82 .19	12 .03	2244 5.12	3.94
ENE	269 .61	556 1.27	853 1.95	499 1.14	77 .18	21 .05	2275 5.19	3.82
E	188 .43	379 .87	405 .92	163 .37	20 .05	3 .01	1158 2.64	3.26
ESE	194 .44	321 .73	366 .84	118 .27	28 .06	2 .00	1029 2.35	3.19
SE	218 .50	385 .88	452 1.03	135 .31	13 .03	1 .00	1204 2.75	3.11
SSE	145 .33	425 .97	704 1.61	186 .42	30 .07	9 .02	1499 3.42	3.49
S	570 1.30	1410 3.22	3276 7.48	1410 3.22	154 .35	13 .03	6833 15.60	3.81
SSW	276 .63	878 2.00	1575 3.59	825 1.88	204 .47	29 .07	3787 8.64	4.02
SW	450 1.03	879 2.01	796 1.82	371 .85	93 .21	21 .05	2610 5.96	3.35
WSW	525 1.20	839 1.91	582 1.33	198 .45	34 .08	14 .03	2192 5.00	2.91
W	266 .61	405 .92	526 1.20	304 .69	107 .24	26 .06	1634 3.73	3.80
WNW	177 .40	394 .90	919 2.10	973 2.22	417 .95	151 .34	3031 6.92	5.25
NW	124 .28	427 .97	1286 2.94	1423 3.25	657 1.50	212 .48	4129 9.42	5.53
NNW	152 .35	421 .96	1271 2.90	1110 2.53	402 .92	79 .18	3435 7.84	4.96
N	312 .71	698 1.59	1235 2.82	1000 2.28	247 .56	27 .06	3519 8.03	4.27
CALM	1370 3.13						1370 3.13	CALM
TOTAL	5614 12.81	9339 21.31	15806 36.07	9785 22.33	2643 6.03	628 1.43	43815 100.00	3.98

NUMBER OF VALID OBSERVATIONS 43815 99.92 PCT.
NUMBER OF INVALID OBSERVATIONS 33 .08 PCT.
TOTAL NUMBER OF OBSERVATIONS 43848 100.00 PCT.

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES

TABLE 2

Joint Wind Frequency Distribution
for 1964

STABILITY CLASS: PASQUILL A
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C.
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	0	4	0	0	0	0	4	2.47
	0.00	6.56	0.00	0.00	0.00	0.00	6.56	
	0.00	.05	0.00	0.00	0.00	0.00	.05	
NE	1	5	0	0	0	0	6	2.33
	1.64	8.20	0.00	0.00	0.00	0.00	9.84	
	.01	.06	0.00	0.00	0.00	0.00	.07	
ENE	1	3	0	0	0	0	4	2.07
	1.64	4.92	0.00	0.00	0.00	0.00	6.56	
	.01	.03	0.00	0.00	0.00	0.00	.05	
E	0	4	0	0	0	0	4	2.47
	0.00	6.56	0.00	0.00	0.00	0.00	6.56	
	0.00	.05	0.00	0.00	0.00	0.00	.05	
ESE	0	4	0	0	0	0	4	2.35
	0.00	6.56	0.00	0.00	0.00	0.00	6.56	
	0.00	.05	0.00	0.00	0.00	0.00	.05	
SE	0	3	0	0	0	0	3	2.43
	0.00	4.92	0.00	0.00	0.00	0.00	4.92	
	0.00	.03	0.00	0.00	0.00	0.00	.03	
SSE	0	5	0	0	0	0	5	2.30
	0.00	8.20	0.00	0.00	0.00	0.00	8.20	
	0.00	.06	0.00	0.00	0.00	0.00	.06	
S	1	3	0	0	0	0	4	2.20
	1.64	4.92	0.00	0.00	0.00	0.00	6.56	
	.01	.03	0.00	0.00	0.00	0.00	.05	
SSW	0	0	0	0	0	0	0	0.00
	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
SW	0	0	0	0	0	0	0	0.00
	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
WSW	1	1	0	0	0	0	2	2.05
	1.64	1.64	0.00	0.00	0.00	0.00	3.28	
	.01	.01	0.00	0.00	0.00	0.00	.02	
W	0	2	0	0	0	0	2	2.35
	0.00	3.28	0.00	0.00	0.00	0.00	3.28	
	0.00	.02	0.00	0.00	0.00	0.00	.02	
WNW	1	2	0	0	0	0	3	2.07
	1.64	3.28	0.00	0.00	0.00	0.00	4.92	
	.01	.02	0.00	0.00	0.00	0.00	.03	
NW	0	0	0	0	0	0	0	0.00
	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
	0.00	0.00	0.00	0.00	0.00	0.00	0.00	
NNW	0	2	0	0	0	0	2	2.35
	0.00	3.28	0.00	0.00	0.00	0.00	3.28	
	0.00	.02	0.00	0.00	0.00	0.00	.02	
N	0	7	0	0	0	0	7	2.46
	0.00	11.48	0.00	0.00	0.00	0.00	11.48	
	0.00	.08	0.00	0.00	0.00	0.00	.08	
CALM	11						11	CALM
	18.03						18.03	
	.13						.13	
TOTAL	16	45	0	0	0	0	61	1.90
	26.23	73.77	0.00	0.00	0.00	0.00	100.00	
	.18	.51	0.00	0.00	0.00	0.00	.69	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 2 (cont'd)

STABILITY CLASS: PASQUILL B
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	2	6	15	0	0	0	23	3.11
	.40	1.21	3.03	0.00	0.00	0.00	4.65	
	.02	.07	.17	0.00	0.00	0.00	.26	
NE	10	10	18	0	0	0	38	2.64
	2.02	2.02	3.64	0.00	0.00	0.00	7.68	
	.11	.11	.20	0.00	0.00	0.00	.43	
ENE	11	12	9	0	0	0	32	2.36
	2.22	2.42	1.82	0.00	0.00	0.00	6.46	
	.13	.14	.10	0.00	0.00	0.00	.36	
E	6	12	7	0	0	0	25	2.48
	1.21	2.42	1.41	0.00	0.00	0.00	5.05	
	.07	.14	.08	0.00	0.00	0.00	.28	
ESE	14	13	3	0	0	0	30	2.00
	2.83	2.63	.61	0.00	0.00	0.00	6.06	
	.16	.15	.03	0.00	0.00	0.00	.34	
SE	5	9	11	0	0	0	25	2.70
	1.01	1.82	2.22	0.00	0.00	0.00	5.05	
	.06	.10	.13	0.00	0.00	0.00	.28	
SSE	12	12	33	0	0	0	57	2.89
	2.42	2.42	6.67	0.00	0.00	0.00	11.52	
	.14	.14	.38	0.00	0.00	0.00	.65	
S	7	12	64	0	0	0	83	3.27
	1.41	2.42	12.93	0.00	0.00	0.00	16.77	
	.08	.14	.73	0.00	0.00	0.00	.95	
SSW	3	9	10	0	0	0	22	2.84
	.61	1.82	2.02	0.00	0.00	0.00	4.44	
	.03	.10	.11	0.00	0.00	0.00	.25	
SW	9	2	8	0	0	0	19	2.34
	1.82	.40	1.62	0.00	0.00	0.00	3.84	
	.10	.02	.09	0.00	0.00	0.00	.22	
WSW	1	0	8	0	0	0	9	3.26
	.20	0.00	1.62	0.00	0.00	0.00	1.82	
	.01	0.00	.09	0.00	0.00	0.00	.10	
W	4	6	16	0	0	0	26	2.99
	.81	1.21	3.23	0.00	0.00	0.00	5.25	
	.05	.07	.18	0.00	0.00	0.00	.30	
WNW	0	6	11	0	0	0	17	3.31
	0.00	1.21	2.22	0.00	0.00	0.00	3.43	
	0.00	.07	.13	0.00	0.00	0.00	.19	
NW	2	3	15	0	0	0	20	3.24
	.40	.61	3.03	0.00	0.00	0.00	4.04	
	.02	.03	.17	0.00	0.00	0.00	.23	
NNW	3	7	11	0	0	0	21	2.97
	.61	1.41	2.22	0.00	0.00	0.00	4.24	
	.03	.08	.13	0.00	0.00	0.00	.24	
N	1	4	11	0	0	0	16	3.19
	.20	.81	2.22	0.00	0.00	0.00	3.23	
	.01	.05	.13	0.00	0.00	0.00	.18	
CALM	32						32	CALM
	6.46						6.46	
	.36						.36	
TOTAL	122	123	250	0	0	0	495	2.67
	24.65	24.85	50.51	0.00	0.00	0.00	100.00	
	1.39	1.40	2.85	0.00	0.00	0.00	5.64	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 2 (cont'd)

STABILITY CLASS: PASQUILL C
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES (METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	0	9	12	8	0	0	29	3.81
	0.00	.89	1.19	.80	0.00	0.00	2.88	
	0.00	.10	.14	.09	0.00	0.00	.33	
NE	0	10	21	4	0	0	35	3.60
	0.00	.99	2.09	.40	0.00	0.00	3.48	
	0.00	.11	.24	.05	0.00	0.00	.40	
ENE	0	8	15	5	0	0	28	3.57
	0.00	.80	1.49	.50	0.00	0.00	2.78	
	0.00	.09	.17	.06	0.00	0.00	.32	
E	2	9	11	6	0	0	28	3.64
	.20	.89	1.09	.60	0.00	0.00	2.78	
	.02	.10	.13	.07	0.00	0.00	.32	
ESE	0	7	4	1	0	0	12	3.06
	0.00	.70	.40	.10	0.00	0.00	1.19	
	0.00	.08	.05	.01	0.00	0.00	.14	
SE	2	3	13	2	2	0	22	3.97
	.20	.30	1.29	.20	.20	0.00	2.19	
	.02	.03	.15	.02	.02	0.00	.25	
SSE	0	16	29	19	0	0	64	4.06
	0.00	1.59	2.88	1.89	0.00	0.00	6.36	
	0.00	.18	.33	.22	0.00	0.00	.73	
S	2	35	127	68	0	0	232	4.17
	.20	3.48	12.62	6.76	0.00	0.00	23.06	
	.02	.40	1.45	.77	0.00	0.00	2.64	
SSW	3	18	55	17	0	0	93	3.82
	.30	1.79	5.47	1.69	0.00	0.00	9.24	
	.03	.20	.63	.19	0.00	0.00	1.06	
SW	2	12	32	7	0	0	53	3.63
	.20	1.19	3.18	.70	0.00	0.00	5.27	
	.02	.14	.36	.08	0.00	0.00	.60	
WSW	2	7	23	10	0	0	42	4.06
	.20	.70	2.29	.99	0.00	0.00	4.17	
	.02	.08	.26	.11	0.00	0.00	.48	
W	1	7	28	23	0	0	59	4.34
	.10	.70	2.78	2.29	0.00	0.00	5.86	
	.01	.08	.32	.26	0.00	0.00	.67	
WNW	0	5	34	12	2	0	53	4.39
	0.00	.50	3.38	1.19	.20	0.00	5.27	
	0.00	.06	.39	.14	.02	0.00	.60	
NW	0	6	38	26	2	0	72	4.71
	0.00	.60	3.78	2.58	.20	0.00	7.16	
	0.00	.07	.43	.30	.02	0.00	.82	
NNW	2	5	53	19	2	0	81	4.34
	.20	.50	5.27	1.89	.20	0.00	8.05	
	.02	.06	.60	.22	.02	0.00	.92	
N	1	8	50	17	0	0	76	4.04
	.10	.80	4.97	1.69	0.00	0.00	7.55	
	.01	.09	.57	.19	0.00	0.00	.87	
CALM	27						27	CALM
	2.68						2.68	
	.31						.31	
TOTAL	44	165	545	244	8	0	1006	3.97
	4.37	16.40	54.17	24.25	.80	0.00	100.00	
	.50	1.88	6.21	2.78	.09	0.00	11.45	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 2 (cont'd)

STABILITY CLASS: PASQUILL D
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA. MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	7	20	54	117	14	0	212	5.13
	.16	.47	1.26	2.73	.33	0.00	4.94	
	.08	.23	.61	1.33	.16	0.00	2.41	
NE	4	30	100	136	26	0	296	5.06
	.09	.70	2.33	3.17	.61	0.00	6.89	
	.05	.34	1.14	1.55	.30	0.00	3.37	
ENE	3	26	106	113	12	4	264	4.88
	.07	.61	2.47	2.63	.28	.09	6.15	
	.03	.30	1.21	1.29	.14	.05	3.01	
E	6	25	71	46	4	0	152	4.12
	.14	.58	1.65	1.07	.09	0.00	3.54	
	.07	.28	.81	.52	.05	0.00	1.73	
ESE	6	17	22	13	6	0	64	4.07
	.14	.40	.51	.30	.14	0.00	1.49	
	.07	.19	.25	.15	.07	0.00	.73	
SE	4	16	43	25	4	0	92	4.18
	.09	.37	1.00	.58	.09	0.00	2.14	
	.05	.18	.49	.28	.05	0.00	1.05	
SSE	8	31	72	32	2	0	145	3.88
	.19	.72	1.68	.75	.05	0.00	3.38	
	.09	.35	.82	.36	.02	0.00	1.65	
S	8	75	301	303	28	0	715	4.71
	.19	1.75	7.01	7.06	.65	0.00	16.66	
	.09	.85	3.43	3.45	.32	0.00	8.14	
SSW	5	33	181	186	39	4	448	5.10
	.12	.77	4.22	4.33	.91	.09	10.44	
	.06	.38	2.06	2.12	.44	.05	5.10	
SW	4	13	59	85	18	6	185	5.39
	.09	.30	1.37	1.98	.42	.14	4.31	
	.05	.15	.67	.97	.20	.07	2.11	
WSW	3	16	37	25	9	9	99	5.13
	.07	.37	.86	.58	.21	.21	2.31	
	.03	.18	.42	.28	.10	.10	1.13	
W	7	17	38	51	17	4	134	5.17
	.16	.40	.89	1.19	.40	.09	3.12	
	.08	.19	.43	.58	.19	.05	1.53	
WNW	3	5	34	103	36	7	188	6.23
	.07	.12	.79	2.40	.84	.16	4.38	
	.03	.06	.39	1.17	.41	.08	2.14	
NW	2	7	37	251	131	44	472	7.12
	.05	.16	.86	5.85	3.05	1.02	10.99	
	.02	.08	.42	2.86	1.49	.50	5.37	
NNW	3	11	65	203	114	38	434	6.79
	.07	.26	1.51	4.73	2.66	.89	10.11	
	.03	.13	.74	2.31	1.30	.43	4.94	
N	3	20	63	177	53	9	325	5.93
	.07	.47	1.47	4.12	1.23	.21	7.57	
	.03	.23	.72	2.02	.60	.10	3.70	
CALM	68						68	CALM
	1.58						1.58	
	.77						.77	
TOTAL	144	362	1283	1866	513	125	4293	5.35
	3.35	8.43	29.89	43.47	11.95	2.91	100.00	
	1.64	4.12	14.61	21.25	5.84	1.42	48.88	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 2 (cont'd)

STABILITY CLASS: PASQUILL E
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	0	9	30	9	0	0	48	3.80
	0.00	.69	2.29	.69	0.00	0.00	3.66	
	0.00	.10	.34	.10	0.00	0.00	.55	
NE	0	3	27	5	0	0	35	3.90
	0.00	.23	2.06	.38	0.00	0.00	2.67	
	0.00	.03	.31	.06	0.00	0.00	.40	
ENE	0	13	27	3	0	0	43	3.31
	0.00	.99	2.06	.23	0.00	0.00	3.28	
	0.00	.15	.31	.03	0.00	0.00	.49	
E	0	14	15	1	0	0	30	3.12
	0.00	1.07	1.14	.08	0.00	0.00	2.29	
	0.00	.16	.17	.01	0.00	0.00	.34	
ESE	0	9	13	3	0	0	25	3.36
	0.00	.69	.99	.23	0.00	0.00	1.91	
	0.00	.10	.15	.03	0.00	0.00	.28	
SE	0	19	22	4	0	0	45	3.29
	0.00	1.45	1.68	.31	0.00	0.00	3.43	
	0.00	.22	.25	.05	0.00	0.00	.51	
SSE	0	16	39	2	0	0	57	3.46
	0.00	1.22	2.97	.15	0.00	0.00	4.35	
	0.00	.18	.44	.02	0.00	0.00	.65	
S	0	31	164	47	0	0	242	3.93
	0.00	2.36	12.51	3.59	0.00	0.00	18.46	
	0.00	.35	1.87	.54	0.00	0.00	2.76	
SSW	0	29	147	28	0	0	204	3.83
	0.00	2.21	11.21	2.14	0.00	0.00	15.56	
	0.00	.33	1.67	.32	0.00	0.00	2.32	
SW	0	25	41	7	0	0	73	3.47
	0.00	1.91	3.13	.53	0.00	0.00	5.57	
	0.00	.28	.47	.08	0.00	0.00	.83	
WSW	0	23	21	3	0	0	47	3.12
	0.00	1.75	1.60	.23	0.00	0.00	3.59	
	0.00	.26	.24	.03	0.00	0.00	.54	
W	0	12	22	4	0	0	38	3.52
	0.00	.92	1.68	.31	0.00	0.00	2.90	
	0.00	.14	.25	.05	0.00	0.00	.43	
WNW	0	2	41	20	0	0	63	4.23
	0.00	.15	3.13	1.53	0.00	0.00	4.81	
	0.00	.02	.47	.23	0.00	0.00	.72	
NW	0	6	80	29	0	0	115	4.20
	0.00	.46	6.10	2.21	0.00	0.00	8.77	
	0.00	.07	.91	.33	0.00	0.00	1.31	
NNW	0	8	107	36	0	0	151	4.22
	0.00	.61	8.16	2.75	0.00	0.00	11.52	
	0.00	.09	1.22	.41	0.00	0.00	1.72	
N	0	12	57	26	0	0	95	4.04
	0.00	.92	4.35	1.98	0.00	0.00	7.25	
	0.00	.14	.65	.30	0.00	0.00	1.08	
CALM	0						0	CALM
	0.00						0.00	
	0.00						0.00	
TOTAL	0	231	853	227	0	0	1311	3.83
	0.00	17.62	65.06	17.32	0.00	0.00	100.00	
	0.00	2.63	9.71	2.58	0.00	0.00	14.93	

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 2 (cont'd)

STABILITY CLASS: PASQUILL F
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	8 .49 .09	11 .68 .13	10 .62 .11	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	29 1.79 .33	2.31
NE	7 .43 .08	17 1.05 .19	8 .49 .09	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	32 1.98 .36	2.34
ENE	12 .74 .14	22 1.36 .25	15 .93 .17	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	49 3.03 .56	2.40
E	6 .37 .07	21 1.30 .24	1 .06 .01	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	28 1.73 .32	2.22
ESE	7 .43 .08	8 .49 .09	8 .49 .09	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	23 1.42 .26	2.37
SE	4 .25 .05	25 1.55 .28	10 .62 .11	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	39 2.41 .44	2.46
SSE	3 .19 .03	11 .68 .13	13 .80 .15	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	27 1.67 .31	2.68
S	23 1.42 .26	74 4.58 .84	59 3.65 .67	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	156 9.65 1.78	2.55
SSW	26 1.61 .30	90 5.57 1.02	58 3.59 .66	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	174 10.76 1.98	2.50
SW	51 3.15 .58	132 8.16 1.50	38 2.35 .43	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	221 13.67 2.52	2.25
WSW	74 4.58 .84	128 7.92 1.46	20 1.24 .23	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	222 13.73 2.53	2.11
W	28 1.73 .32	56 3.46 .64	18 1.11 .20	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	102 6.31 1.16	2.21
WNW	7 .43 .08	26 1.61 .30	20 1.24 .23	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	53 3.28 .60	2.58
NW	10 .62 .11	20 1.24 .23	26 1.61 .30	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	56 3.46 .64	2.57
NNW	9 .56 .10	40 2.47 .46	31 1.92 .35	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	80 4.95 .91	2.54
N	12 .74 .14	42 2.60 .48	31 1.92 .35	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	85 5.26 .97	2.54
CALM	241 14.90 2.74						241 14.90 2.74	CALM
TOTAL	528 32.65 6.01	723 44.71 8.23	366 22.63 4.17	0 0.00 0.00	0 0.00 0.00	0 0.00 0.00	1617 100.00 18.41	2.02

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES THIS CLASS
XXX PERCENT OCCURRENCES ALL CLASSES

TABLE 2 (cont'd)

ALL WINDS
DATA SOURCE: NATIONAL AIRPORT, WASH., D.C
WIND SENSOR HEIGHT: 7.00 METERS
TABLE GENERATED: 11/19/80. 20.00.32.

AFFRI-TRIGA REACTOR
BETHESDA, MD.
AFFRI
DAMES AND MOORE JOB NO: 11985-001-27

WIND SECTOR	WIND SPEED CATEGORIES(METERS PER SECOND) 0.0-1.5	1.5-3.0	3.0-5.0	5.0-7.5	7.5-10.0	>10.0	TOTAL	MEAN SPEED
NNE	17 .19	59 .67	121 1.38	134 1.53	14 .16	0 0.00	345 3.93	4.43
NE	22 .25	75 .85	174 1.98	145 1.65	26 .30	0 0.00	442 5.03	4.41
ENE	27 .31	84 .96	172 1.96	121 1.38	12 .14	4 .05	420 4.78	4.13
E	20 .23	85 .97	105 1.20	53 .60	4 .05	0 0.00	267 3.04	3.58
ESE	27 .31	58 .66	50 .57	17 .19	6 .07	0 0.00	158 1.80	3.20
SE	15 .17	75 .85	99 1.13	31 .35	6 .07	0 0.00	226 2.57	3.50
SSE	23 .26	91 1.04	186 2.12	53 .60	2 .02	0 0.00	355 4.04	3.57
S	41 .47	230 2.62	716 8.15	418 4.76	28 .32	0 0.00	1433 16.31	4.16
SSW	37 .42	179 2.04	451 5.13	231 2.63	39 .44	4 .05	941 10.71	4.16
SW	66 .75	184 2.09	178 2.03	99 1.13	18 .20	6 .07	551 6.27	3.60
WSW	81 .92	175 1.99	109 1.24	38 .43	9 .10	9 .10	421 4.79	3.15
W	40 .46	100 1.14	122 1.39	78 .89	17 .19	4 .05	361 4.11	3.85
WNW	11 .13	46 .52	140 1.59	135 1.54	38 .43	7 .08	377 4.29	4.96
NW	14 .16	42 .48	196 2.23	306 3.48	133 1.51	44 .50	735 8.37	5.97
NNW	17 .19	73 .83	267 3.04	258 2.94	116 1.32	38 .43	769 8.75	5.47
N	17 .19	93 1.06	212 2.41	220 2.50	53 .60	9 .10	604 6.88	4.81
CALM	379 4.31						379 4.31	CALM
TOTAL	854 9.72	1649 18.77	3298 37.55	2337 26.61	521 5.93	125 1.42	8784 100.00	4.18

NUMBER OF VALID OBSERVATIONS 8784 100.00 PCT.
NUMBER OF INVALID OBSERVATIONS 0 0.00 PCT.
TOTAL NUMBER OF OBSERVATIONS 8784 100.00 PCT.

KEY XXX NUMBER OF OCCURRENCES
XXX PERCENT OCCURRENCES

APPENDIX B

Annual Averaged Relative Concentrations (χ/Q)

TABLE 1

Annual Averaged Relative Concentrations (χ/Q)

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/64 TO 12/31/64

PROGRAM ANDIFF
GROUND RELEASE - SPECIAL PTS.
NATIONAL AIRPORT MET. DATA
DATE 10/17/80. TIME 12.11.20.

AFRRI-TRIGA REACTOR
AFRRI
BETHESDA, MARYLAND
DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
100 YARDS															
1.2E-04	1.1E-04	1.0E-04	5.9E-05	4.0E-05	6.1E-05	7.1E-05	6.6E-05	3.7E-05	4.4E-05	4.9E-05	3.4E-05	2.3E-05	3.2E-05	4.1E-05	1.6E-04
1.2E-04	1.1E-04	1.0E-04	5.9E-05	4.0E-05	6.1E-05	7.1E-05	6.6E-05	1.7E-05	4.4E-05	4.9E-05	3.4E-05	2.3E-05	3.2E-05	4.1E-05	1.6E-04
0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
1.2E-04	1.1E-04	1.0E-04	5.9E-05	4.0E-05	6.1E-05	7.1E-05	6.6E-05	3.7E-05	4.4E-05	4.9E-05	3.4E-05	2.3E-05	3.2E-05	4.1E-05	1.6E-04
1.2E-04	1.1E-04	1.0E-04	5.9E-05	4.0E-05	6.1E-05	7.1E-05	6.6E-05	3.7E-05	4.4E-05	4.9E-05	3.4E-05	2.3E-05	3.2E-05	4.1E-05	1.6E-04
1.2E-04	1.1E-04	1.0E-04	5.9E-05	4.0E-05	6.1E-05	7.1E-05	6.6E-05	3.7E-05	4.4E-05	4.9E-05	3.4E-05	2.3E-05	3.2E-05	4.1E-05	1.6E-04
1.2E-04	1.1E-04	1.0E-04	5.9E-05	4.0E-05	6.1E-05	7.1E-05	6.6E-05	1.7E-05	4.4E-05	4.9E-05	3.4E-05	2.3E-05	3.2E-05	4.1E-05	1.6E-04
91.	91.	91.	91.	91.	91.	91.	91.	91.	91.	91.	91.	91.	91.	91.	91.
150 YARDS															
6.1E-05	5.1E-05	4.8E-05	2.9E-05	2.0E-05	3.1E-05	3.7E-05	3.3E-05	1.9E-05	2.3E-05	2.6E-05	1.8E-05	1.2E-05	1.6E-05	2.1E-05	8.1E-05
6.0E-05	5.0E-05	4.7E-05	2.8E-05	1.9E-05	3.0E-05	3.6E-05	3.2E-05	1.9E-05	2.3E-05	2.5E-05	1.7E-05	1.2E-05	1.6E-05	2.1E-05	7.9E-05
2.9E-07	1.7E-07	1.3E-07	1.1E-07	1.2E-07	2.3E-07	2.4E-07	1.9E-07	1.1E-07	1.4E-07	1.1E-07	8.3E-08	4.9E-08	7.0E-08	1.1E-07	4.4E-07
6.1E-05	5.1E-05	4.8E-05	2.9E-05	2.0E-05	3.1E-05	3.7E-05	3.3E-05	1.9E-05	2.3E-05	2.6E-05	1.8E-05	1.2E-05	1.6E-05	2.1E-05	8.1E-05
6.1E-05	5.1E-05	4.8E-05	2.9E-05	2.0E-05	3.1E-05	3.7E-05	3.3E-05	1.9E-05	2.3E-05	2.6E-05	1.8E-05	1.2E-05	1.6E-05	2.1E-05	8.1E-05
6.0E-05	5.0E-05	4.7E-05	2.8E-05	1.9E-05	3.0E-05	3.6E-05	3.2E-05	1.9E-05	2.3E-05	2.5E-05	1.7E-05	1.2E-05	1.6E-05	2.1E-05	7.9E-05
6.0E-05	5.0E-05	4.7E-05	2.8E-05	1.9E-05	3.0E-05	3.6E-05	3.2E-05	1.9E-05	2.3E-05	2.5E-05	1.7E-05	1.2E-05	1.6E-05	2.1E-05	7.9E-05
137.	137.	137.	137.	137.	137.	137.	137.	137.	137.	137.	137.	137.	137.	137.	137.
200 YARDS															
1.8E-05	1.1E-05	2.9E-05	1.8E-05	1.2E-05	2.0E-05	2.3E-05	2.1E-05	1.2E-05	1.5E-05	1.6E-05	1.1E-05	7.4E-06	1.0E-05	1.3E-05	5.1E-05
1.7E-05	1.0E-05	2.8E-05	1.7E-05	1.2E-05	1.9E-05	2.3E-05	2.0E-05	1.2E-05	1.4E-05	1.6E-05	1.1E-05	7.2E-06	9.7E-06	1.3E-05	5.0E-05
1.9E-07	1.1E-07	8.5E-08	7.3E-08	7.6E-08	1.5E-07	1.6E-07	1.2E-07	7.0E-08	8.9E-08	8.5E-08	5.4E-08	3.2E-08	4.6E-08	7.2E-08	2.9E-07
1.8E-05	1.1E-05	2.9E-05	1.8E-05	1.2E-05	2.0E-05	2.3E-05	2.1E-05	1.2E-05	1.5E-05	1.6E-05	1.1E-05	7.4E-06	1.0E-05	1.3E-05	5.1E-05
1.8E-05	1.1E-05	2.9E-05	1.8E-05	1.2E-05	2.0E-05	2.3E-05	2.1E-05	1.2E-05	1.5E-05	1.6E-05	1.1E-05	7.4E-06	1.0E-05	1.3E-05	5.1E-05
1.7E-05	1.0E-05	2.8E-05	1.7E-05	1.2E-05	1.9E-05	2.3E-05	2.0E-05	1.2E-05	1.4E-05	1.6E-05	1.1E-05	7.2E-06	9.7E-06	1.3E-05	5.0E-05
1.7E-05	1.0E-05	2.8E-05	1.7E-05	1.2E-05	1.9E-05	2.3E-05	2.0E-05	1.2E-05	1.4E-05	1.6E-05	1.1E-05	7.2E-06	9.7E-06	1.3E-05	5.0E-05
183.	183.	183.	183.	183.	183.	183.	183.	183.	183.	183.	183.	183.	183.	183.	183.

TOTAL OBS - 8783   TOTAL INV OBS - 100   CALMS UPPER LEVEL - 0.00   CALMS LOWER LEV - 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3)   ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)   ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS   ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS   ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/64 TO 12/31/64

PROGRAM AXDIFF	AFRRI-TRIGA REACTOR
GROUND RELEASE - SPECIAL PTS.	AFRRI
NATIONAL AIRPORT MET. DATA	BETHESDA, MARYLAND
DATE 10/17/80. TIME 12.11.28.	DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
250 YARDS															
2.7E-05	2.1E-05	2.0E-05	1.2E-05	8.6E-06	1.4E-05	1.6E-05	1.4E-05	8.4E-06	1.0E-05	1.1E-05	7.7E-06	5.1E-06	7.0E-06	9.3E-06	3.6E-05
2.6E-05	2.0E-05	1.9E-05	1.2E-05	8.3E-06	1.3E-05	1.5E-05	1.4E-05	8.1E-06	9.9E-06	1.1E-05	7.4E-06	4.9E-06	6.7E-06	8.9E-06	3.4E-05
1.4E-07	8.0E-08	6.1E-08	5.2E-08	5.4E-08	1.1E-07	1.1E-07	8.7E-08	5.0E-08	6.4E-08	6.1E-08	3.9E-08	2.3E-08	3.3E-08	5.1E-08	2.1E-07
2.7E-05	2.1E-05	2.0E-05	1.2E-05	8.6E-06	1.4E-05	1.6E-05	1.4E-05	8.4E-06	1.0E-05	1.1E-05	7.7E-06	5.1E-06	7.0E-06	9.3E-06	3.6E-05
2.7E-05	2.1E-05	2.0E-05	1.2E-05	8.6E-06	1.4E-05	1.6E-05	1.4E-05	8.4E-06	1.0E-05	1.1E-05	7.7E-06	5.1E-06	7.0E-06	9.3E-06	3.6E-05
2.6E-05	2.0E-05	1.9E-05	1.2E-05	8.3E-06	1.3E-05	1.6E-05	1.4E-05	8.1E-06	9.9E-06	1.1E-05	7.4E-06	4.9E-06	6.7E-06	8.9E-06	3.4E-05
2.6E-05	2.0E-05	1.9E-05	1.2E-05	8.3E-06	1.3E-05	1.6E-05	1.4E-05	8.1E-06	9.9E-06	1.1E-05	7.4E-06	4.9E-06	6.7E-06	8.9E-06	3.4E-05
229.	229.	229.	229.	229.	229.	229.	229.	229.	229.	229.	229.	229.	229.	229.	229.
300 YARDS															
2.0E-05	1.6E-05	1.5E-05	9.2E-06	6.5E-06	1.0E-05	1.2E-05	1.1E-05	6.3E-06	7.7E-06	8.4E-06	5.8E-06	3.8E-06	5.3E-06	6.9E-06	2.7E-05
1.9E-05	1.6E-05	1.5E-05	8.8E-06	6.3E-06	1.0E-05	1.2E-05	1.0E-05	6.1E-06	7.4E-06	8.0E-06	5.6E-06	3.7E-06	5.0E-06	6.6E-06	2.6E-05
1.0E-07	6.0E-08	4.6E-08	4.0E-08	4.1E-08	8.1E-08	8.4E-08	6.6E-08	3.8E-08	4.8E-08	4.6E-08	2.9E-08	1.7E-08	2.5E-08	3.9E-08	1.6E-07
2.0E-05	1.6E-05	1.5E-05	9.2E-06	6.5E-06	1.0E-05	1.2E-05	1.1E-05	6.3E-06	7.7E-06	8.4E-06	5.8E-06	3.8E-06	5.3E-06	6.9E-06	2.7E-05
2.0E-05	1.6E-05	1.5E-05	9.2E-06	6.5E-06	1.0E-05	1.2E-05	1.1E-05	6.3E-06	7.7E-06	8.4E-06	5.8E-06	3.8E-06	5.3E-06	6.9E-06	2.7E-05
1.9E-05	1.6E-05	1.5E-05	8.8E-06	6.3E-06	1.0E-05	1.2E-05	1.0E-05	6.1E-06	7.4E-06	8.0E-06	5.6E-06	3.7E-06	5.0E-06	6.6E-06	2.6E-05
1.9E-05	1.6E-05	1.5E-05	8.8E-06	6.3E-06	1.0E-05	1.2E-05	1.0E-05	6.1E-06	7.4E-06	8.0E-06	5.6E-06	3.7E-06	5.0E-06	6.6E-06	2.6E-05
274.	274.	274.	274.	274.	274.	274.	274.	274.	274.	274.	274.	274.	274.	274.	274.
350 YARDS															
1.6E-05	1.3E-05	1.2E-05	7.3E-06	5.1E-06	8.2E-06	9.7E-06	8.6E-06	5.0E-06	6.0E-06	6.6E-06	4.5E-06	3.0E-06	4.1E-06	5.4E-06	2.1E-05
1.5E-05	1.2E-05	1.2E-05	7.0E-06	4.9E-06	7.8E-06	9.3E-06	8.2E-06	4.7E-06	5.7E-06	6.3E-06	4.3E-06	2.9E-06	3.9E-06	5.1E-06	2.0E-05
8.3E-08	4.8E-08	3.6E-08	3.1E-08	3.3E-08	6.4E-08	6.7E-08	5.2E-08	3.0E-08	3.8E-08	3.6E-08	2.3E-08	1.4E-08	2.0E-08	3.1E-08	1.2E-07
1.6E-05	1.3E-05	1.2E-05	7.3E-06	5.1E-06	8.2E-06	9.7E-06	8.6E-06	5.0E-06	6.0E-06	6.6E-06	4.5E-06	3.0E-06	4.1E-06	5.4E-06	2.1E-05
1.6E-05	1.3E-05	1.2E-05	7.3E-06	5.1E-06	8.2E-06	9.7E-06	8.6E-06	5.0E-06	6.0E-06	6.6E-06	4.5E-06	3.0E-06	4.1E-06	5.4E-06	2.1E-05
1.5E-05	1.2E-05	1.2E-05	7.0E-06	4.9E-06	7.8E-06	9.3E-06	8.2E-06	4.7E-06	5.7E-06	6.3E-06	4.3E-06	2.9E-06	3.9E-06	5.1E-06	2.0E-05
1.5E-05	1.2E-05	1.2E-05	7.0E-06	4.9E-06	7.8E-06	9.3E-06	8.2E-06	4.7E-06	5.7E-06	6.3E-06	4.3E-06	2.9E-06	3.9E-06	5.1E-06	2.0E-05
320.	320.	320.	320.	320.	320.	320.	320.	320.	320.	320.	320.	320.	320.	320.	320.
400 YARDS															
1.3E-05	1.1E-05	1.0E-05	6.0E-06	4.2E-06	6.6E-06	7.8E-06	7.0E-06	4.0E-06	4.8E-06	5.3E-06	3.6E-06	2.4E-06	3.3E-06	4.3E-06	1.7E-05
1.2E-05	1.0E-05	9.5E-06	5.7E-06	4.0E-06	6.3E-06	7.5E-06	6.6E-06	3.8E-06	4.6E-06	5.0E-06	3.5E-06	2.3E-06	3.2E-06	4.1E-06	1.6E-05
6.6E-08	3.9E-08	3.0E-08	2.5E-08	2.7E-08	5.2E-08	5.4E-08	4.3E-08	2.4E-08	3.1E-08	3.0E-08	1.9E-08	1.1E-08	1.6E-08	2.5E-08	1.0E-07
1.3E-05	1.1E-05	1.0E-05	6.0E-06	4.2E-06	6.6E-06	7.8E-06	7.0E-06	4.0E-06	4.8E-06	5.3E-06	3.6E-06	2.4E-06	3.3E-06	4.3E-06	1.7E-05
1.3E-05	1.1E-05	1.0E-05	6.0E-06	4.2E-06	6.5E-06	7.8E-06	7.0E-06	4.0E-06	4.8E-06	5.3E-06	3.6E-06	2.4E-06	3.3E-06	4.3E-06	1.7E-05
1.2E-05	1.0E-05	9.5E-06	5.7E-06	4.0E-06	6.3E-06	7.4E-06	6.6E-06	3.8E-06	4.6E-06	5.0E-06	3.5E-06	2.3E-06	3.2E-06	4.1E-06	1.6E-05
1.2E-05	1.0E-05	9.5E-06	5.7E-06	4.0E-06	6.3E-06	7.5E-06	6.6E-06	3.8E-06	4.6E-06	5.0E-06	3.5E-06	2.3E-06	3.2E-06	4.1E-06	1.6E-05
366.	366.	366.	366.	366.	366.	366.	366.	366.	366.	366.	366.	366.	366.	366.	366.

TOTAL OBS - 8783   TOTAL INV OBS - 300   CALMS UPPER LEVEL - 0.00   CALMS LOWER LEV - 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3)
ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)
ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/64 TO 12/31/64

PROGRAM ANDIFF — AFRRI-TRIGA REACTOR
GROUND RELEASE - SPECIAL PTS. — AFRRI
NATIONAL AIRPORT MET. DATA — BETHESDA, MARYLAND
DATE 10/17/80. TIME 12.11.28. — DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

450 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
1.1E-05	9.0E-06	8.5E-06	5.0E-06	3.4E-06	5.5E-06	6.5E-06	5.8E-06	3.3E-06	3.9E-06	4.4E-06	3.0E-06	2.0E-06	2.8E-06	3.5E-06	1.4E-05
1.0E-05	8.5E-06	8.0E-06	4.7E-06	3.3E-06	5.2E-06	6.1E-06	5.5E-06	3.1E-06	3.7E-06	4.1E-06	2.8E-06	1.9E-06	2.6E-06	3.3E-06	1.3E-05
5.5E-08	3.2E-08	2.5E-08	2.1E-08	2.2E-08	4.3E-08	4.5E-08	3.5E-08	2.0E-08	2.6E-08	2.5E-08	1.6E-08	9.3E-09	1.3E-08	2.1E-08	8.4E-08
1.1E-05	9.0E-06	8.4E-06	5.0E-06	3.4E-06	5.5E-06	6.5E-06	5.8E-06	3.3E-06	3.9E-06	4.4E-06	3.0E-06	2.0E-06	2.8E-06	3.5E-06	1.4E-05
1.1E-05	9.0E-06	8.5E-06	5.0E-06	3.4E-06	5.5E-06	6.5E-06	5.8E-06	3.3E-06	3.9E-06	4.4E-06	3.0E-06	2.0E-06	2.8E-06	3.5E-06	1.4E-05
1.0E-05	8.5E-06	8.0E-06	4.7E-06	3.3E-06	5.2E-06	6.1E-06	5.5E-06	3.1E-06	3.7E-06	4.1E-06	2.8E-06	1.9E-06	2.6E-06	3.3E-06	1.3E-05
1.0E-05	8.5E-06	8.0E-06	4.7E-06	3.3E-06	5.2E-06	6.1E-06	5.5E-06	3.1E-06	3.7E-06	4.1E-06	2.8E-06	1.9E-06	2.6E-06	3.3E-06	1.3E-05
411.	411.	411.	411.	411.	411.	411.	411.	411.	411.	411.	411.	411.	411.	411.	411.

500 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
9.1E-06	7.7E-06	7.2E-06	4.2E-06	2.9E-06	4.6E-06	5.5E-06	4.9E-06	2.8E-06	3.3E-06	3.7E-06	2.5E-06	1.7E-06	2.3E-06	2.9E-06	1.2E-05
8.6E-06	7.2E-06	6.8E-06	4.0E-06	2.7E-06	4.3E-06	5.2E-06	4.6E-06	2.6E-06	3.1E-06	3.5E-06	2.4E-06	1.6E-06	2.2E-06	2.8E-06	1.1E-05
4.7E-08	2.7E-08	2.1E-08	1.8E-08	1.9E-08	3.7E-08	3.8E-08	3.0E-08	1.7E-08	2.2E-08	2.1E-08	1.3E-08	7.9E-09	1.1E-08	1.8E-08	7.1E-08
9.1E-06	7.7E-06	7.2E-06	4.2E-06	2.9E-06	4.6E-06	5.5E-06	4.9E-06	2.8E-06	3.3E-06	3.7E-06	2.5E-06	1.7E-06	2.3E-06	2.9E-06	1.2E-05
9.1E-06	7.7E-06	7.2E-06	4.2E-06	2.9E-06	4.6E-06	5.5E-06	4.9E-06	2.8E-06	3.3E-06	3.7E-06	2.5E-06	1.7E-06	2.3E-06	2.9E-06	1.2E-05
8.6E-06	7.2E-06	6.8E-06	4.0E-06	2.7E-06	4.3E-06	5.1E-06	4.6E-06	2.6E-06	3.1E-06	3.5E-06	2.4E-06	1.6E-06	2.2E-06	2.8E-06	1.1E-05
8.6E-06	7.2E-06	6.8E-06	4.0E-06	2.7E-06	4.3E-06	5.2E-06	4.6E-06	2.6E-06	3.1E-06	3.5E-06	2.4E-06	1.6E-06	2.2E-06	2.8E-06	1.1E-05
457.	457.	457.	457.	457.	457.	457.	457.	457.	457.	457.	457.	457.	457.	457.	457.

550 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
7.8E-06	6.6E-06	6.2E-06	3.6E-06	2.5E-06	3.9E-06	4.7E-06	4.2E-06	2.4E-06	2.8E-06	3.1E-06	2.1E-06	1.4E-06	2.0E-06	2.5E-06	1.0E-05
7.3E-06	6.2E-06	5.9E-06	3.4E-06	2.3E-06	3.7E-06	4.4E-06	3.9E-06	2.2E-06	2.6E-06	2.9E-06	2.0E-06	1.3E-06	1.9E-06	2.3E-06	9.4E-06
4.0E-08	2.4E-08	1.8E-08	1.6E-08	1.6E-08	3.2E-08	3.3E-08	2.6E-08	1.5E-08	1.9E-08	1.8E-08	1.1E-08	6.8E-09	9.7E-09	1.5E-08	6.2E-08
7.8E-06	6.6E-06	6.2E-06	3.6E-06	2.5E-06	3.9E-06	4.7E-06	4.2E-06	2.4E-06	2.8E-06	3.1E-06	2.1E-06	1.4E-06	2.0E-06	2.5E-06	1.0E-05
7.8E-06	6.6E-06	6.2E-06	3.6E-06	2.5E-06	3.9E-06	4.7E-06	4.2E-06	2.4E-06	2.8E-06	3.1E-06	2.1E-06	1.4E-06	2.0E-06	2.5E-06	1.0E-05
7.3E-06	6.2E-06	5.9E-06	3.4E-06	2.3E-06	3.7E-06	4.4E-06	3.9E-06	2.2E-06	2.6E-06	2.9E-06	2.0E-06	1.3E-06	1.9E-06	2.3E-06	9.4E-06
7.3E-06	6.2E-06	5.9E-06	3.4E-06	2.3E-06	3.7E-06	4.4E-06	3.9E-06	2.2E-06	2.6E-06	2.9E-06	2.0E-06	1.3E-06	1.9E-06	2.3E-06	9.4E-06
503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.	503.

600 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
6.8E-06	5.8E-06	5.5E-06	3.2E-06	2.2E-06	3.4E-06	4.1E-06	3.6E-06	2.0E-06	2.4E-06	2.7E-06	1.9E-06	1.2E-06	1.7E-06	2.1E-06	8.7E-06
6.4E-06	5.4E-06	5.1E-06	2.9E-06	2.0E-06	3.2E-06	3.8E-06	3.4E-06	1.9E-06	2.3E-06	2.5E-06	1.7E-06	1.1E-06	1.6E-06	2.0E-06	8.1E-06
3.5E-08	2.1E-08	1.6E-08	1.4E-08	1.4E-08	2.8E-08	2.9E-08	2.3E-08	1.3E-08	1.7E-08	1.6E-08	1.0E-08	5.9E-09	8.5E-09	1.3E-08	5.4E-08
6.8E-06	5.8E-06	5.5E-06	3.2E-06	2.2E-06	3.4E-06	4.1E-06	3.6E-06	2.0E-06	2.4E-06	2.7E-06	1.9E-06	1.2E-06	1.7E-06	2.1E-06	8.7E-06
6.8E-06	5.8E-06	5.5E-06	3.2E-06	2.2E-06	3.4E-06	4.1E-06	3.6E-06	2.0E-06	2.4E-06	2.7E-06	1.9E-06	1.2E-06	1.7E-06	2.1E-08	8.7E-06
6.3E-06	5.4E-06	5.1E-06	2.9E-06	2.0E-06	3.2E-06	3.8E-06	3.4E-06	1.9E-06	2.3E-06	2.5E-06	1.7E-06	1.1E-06	1.6E-06	2.0E-06	8.1E-06
6.4E-06	5.4E-06	5.1E-06	2.9E-06	2.0E-06	3.2E-06	3.8E-06	3.4E-06	1.9E-06	2.3E-06	2.5E-06	1.7E-06	1.1E-06	1.6E-06	2.0E-06	8.1E-06
549.	549.	549.	549.	549.	549.	549.	549.	549.	549.	549.	549.	549.	549.	549.	549.

TOTAL OBS - 8783 TOTAL INV OBS - 380 CALMS UPPER LEVEL - 0.00 CALMS LOWER LEV - 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3)
ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)
ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/66 TO 12/31/66

PROGRAM AOQTT — AFRRI-TRIGA REACTOR
GROUND RELEASE - SPECIAL PTS. — AFRRI
NATIONAL AIRPORT MET. DATA — BETHESDA, MARYLAND
DATE 10/17/80. TIME 12.11.29. — DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

650 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
6.0E-06	5.1E-06	4.8E-06	2.8E-06	1.9E-06	3.0E-06	3.6E-06	3.2E-06	1.8E-06	2.1E-06	2.4E-06	1.6E-06	1.1E-06	1.5E-06	1.9E-06	7.6E-06
5.6E-06	4.7E-06	4.5E-06	2.6E-06	1.8E-06	2.8E-06	3.3E-06	3.0E-06	1.7E-06	2.0E-06	2.2E-06	1.5E-06	9.9E-07	1.4E-06	1.7E-06	7.1E-06
1.1E-08	1.0E-08	1.4E-08	1.2E-08	1.2E-08	2.4E-08	2.5E-08	2.0E-08	1.1E-08	1.5E-08	1.4E-08	8.8E-09	5.2E-09	7.5E-09	1.2E-08	4.7E-08
6.0E-06	5.1E-06	4.8E-06	2.8E-06	1.9E-06	3.0E-06	3.6E-06	3.2E-06	1.8E-06	2.1E-06	2.4E-06	1.6E-06	1.1E-06	1.5E-06	1.9E-08	7.0E-06
6.0E-06	5.1E-06	4.8E-06	2.8E-06	1.9E-06	3.0E-06	3.6E-06	3.2E-06	1.8E-06	2.1E-06	2.4E-06	1.6E-06	1.1E-06	1.5E-06	1.9E-06	7.6E-06
5.6E-06	4.7E-06	4.5E-06	2.6E-06	1.8E-06	2.8E-06	3.3E-06	3.0E-06	1.7E-06	2.0E-06	2.2E-06	1.5E-06	9.9E-07	1.4E-06	1.7E-06	7.1E-06
5.6E-06	4.7E-06	4.5E-06	2.6E-06	1.8E-06	2.8E-06	3.3E-06	3.0E-06	1.7E-06	2.0E-06	2.2E-06	1.5E-06	9.9E-07	1.4E-06	1.7E-06	7.1E-06
594.	594.	594.	594.	594.	594.	594.	594.	594.	594.	594.	594.	594.	594.	594.	594.

700 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
5.3E-06	4.5E-06	4.3E-06	2.5E-06	1.7E-06	2.6E-06	3.2E-06	2.8E-06	1.6E-06	1.9E-06	2.1E-06	1.4E-06	9.4E-07	1.3E-06	1.6E-06	6.7E-06
4.9E-06	4.2E-06	4.0E-06	2.3E-06	1.6E-06	2.4E-06	2.9E-06	2.6E-06	1.5E-06	1.7E-06	1.9E-06	1.3E-06	8.7E-07	1.2E-06	1.5E-06	6.2E-06
2.8E-08	1.6E-08	1.2E-08	1.1E-08	1.1E-08	2.2E-08	2.3E-08	1.9E-08	1.0E-08	1.3E-08	1.2E-08	7.9E-09	4.6E-09	6.6E-09	1.0E-08	4.2E-08
5.3E-06	4.5E-06	4.3E-06	2.5E-06	1.7E-06	2.6E-06	3.2E-06	2.8E-06	1.6E-06	1.9E-06	2.1E-06	1.4E-06	9.4E-07	1.3E-06	1.6E-06	6.7E-06
5.3E-06	4.5E-06	4.3E-06	2.5E-06	1.7E-06	2.6E-06	3.2E-06	2.8E-06	1.6E-06	1.9E-06	2.1E-06	1.4E-06	9.4E-07	1.3E-06	1.6E-06	6.7E-06
4.9E-06	4.2E-06	4.0E-06	2.3E-06	1.6E-06	2.4E-06	2.9E-06	2.6E-06	1.5E-06	1.7E-06	1.9E-06	1.3E-06	8.7E-07	1.2E-06	1.5E-06	6.2E-06
4.9E-06	4.2E-06	4.0E-06	2.3E-06	1.6E-06	2.4E-06	2.9E-06	2.6E-06	1.5E-06	1.7E-06	1.9E-06	1.3E-06	8.7E-07	1.2E-06	1.5E-06	6.2E-06
640.	640.	640.	640.	640.	640.	640.	640.	640.	640.	640.	640.	640.	640.	640.	640.

750 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
4.7E-06	4.1E-06	3.9E-06	2.2E-06	1.5E-06	2.4E-06	2.8E-06	2.5E-06	1.4E-06	1.7E-06	1.9E-06	1.3E-06	8.4E-07	1.2E-06	1.5E-06	6.0E-06
4.4E-06	3.8E-06	3.6E-06	2.0E-06	1.4E-06	2.2E-06	2.6E-06	2.1E-06	1.3E-06	1.5E-06	1.7E-06	1.2E-06	7.7E-07	1.1E-06	1.4E-06	5.5E-06
2.5E-08	1.5E-08	1.1E-08	9.5E-09	9.9E-09	1.9E-08	2.0E-08	1.6E-08	9.1E-09	1.2E-08	1.1E-08	7.0E-09	4.2E-09	6.0E-09	9.4E-09	3.8E-08
4.7E-06	4.1E-06	3.9E-06	2.2E-06	1.5E-06	2.4E-06	2.8E-06	2.5E-06	1.4E-06	1.7E-06	1.9E-06	1.3E-06	8.4E-07	1.2E-06	1.5E-06	6.0E-06
4.7E-06	4.1E-06	3.9E-06	2.2E-06	1.5E-06	2.4E-06	2.8E-06	2.5E-06	1.4E-06	1.7E-06	1.9E-06	1.3E-06	8.4E-07	1.2E-06	1.5E-06	6.0E-06
4.4E-06	3.8E-06	3.6E-06	2.0E-06	1.4E-06	2.2E-06	2.6E-06	2.1E-06	1.3E-06	1.5E-06	1.7E-06	1.2E-06	7.7E-07	1.1E-06	1.4E-06	5.5E-06
4.4E-06	3.8E-06	3.6E-06	2.0E-06	1.4E-06	2.2E-06	2.6E-06	2.1E-06	1.3E-06	1.5E-06	1.7E-06	1.2E-06	7.7E-07	1.1E-06	1.4E-06	5.5E-06
686.	686.	686.	686.	686.	686.	686.	686.	686.	686.	686.	686.	686.	686.	686.	686.

800 YARDS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
4.3E-06	3.7E-06	3.5E-06	2.0E-06	1.3E-06	2.1E-06	2.5E-06	2.3E-06	1.3E-06	1.5E-06	1.7E-06	1.1E-06	7.5E-07	1.1E-06	1.3E-06	5.4E-06
3.9E-06	3.4E-06	3.2E-06	1.8E-06	1.2E-06	1.9E-06	2.3E-06	2.1E-06	1.2E-06	1.4E-06	1.5E-06	1.1E-06	6.9E-07	9.9E-07	1.2E-06	4.9E-06
2.2E-08	1.3E-08	1.0E-08	8.6E-09	9.0E-09	1.6E-08	1.8E-08	1.4E-08	8.2E-09	1.1E-08	1.0E-08	6.4E-09	3.8E-09	5.4E-09	8.5E-09	3.4E-08
4.3E-06	3.7E-06	3.5E-06	2.0E-06	1.3E-06	2.1E-06	2.5E-06	2.3E-06	1.3E-06	1.5E-06	1.7E-06	1.1E-06	7.5E-07	1.1E-06	1.3E-06	5.4E-06
4.3E-06	3.7E-06	3.5E-06	2.0E-06	1.3E-06	2.1E-06	2.5E-06	2.3E-06	1.3E-06	1.5E-06	1.7E-06	1.1E-06	7.5E-07	1.1E-06	1.3E-06	5.4E-06
3.9E-06	3.4E-06	3.2E-06	1.8E-06	1.2E-06	1.9E-06	2.3E-06	2.1E-06	1.2E-06	1.4E-06	1.5E-06	1.1E-06	6.9E-07	9.9E-07	1.2E-06	4.9E-06
3.9E-06	3.4E-06	3.2E-06	1.8E-06	1.2E-06	1.9E-06	2.3E-06	2.1E-06	1.2E-06	1.4E-06	1.5E-06	1.1E-06	6.9E-07	9.9E-07	1.2E-06	4.9E-06
732.	732.	732.	732.	732.	732.	732.	732.	732.	732.	732.	732.	732.	732.	732.	732.

TOTAL OBS - 8783 TOTAL INV OBS - 380 CALMS UPPER LEVEL - 0.00 CALMS LOWER LEV - 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3)
ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)
ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/69 TO 12/31/69

PROGRAM ANDIFF — AFRRI-TRIGA REACTOR
GROUND RELEASE - STANDARD PTS. — AFRRI
NATIONAL AIRPORT MET. DATA — BETHESDA, MARYLAND
DATE 10/16/80. TIME 20.12.21. — DAMES AND MOORE JOB 11905-001-27

AFFECTED SECTORS

0.4 KM

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
1.1E-05	9.6E-06	8.8E-06	5.2E-06	3.6E-06	5.7E-06	6.8E-06	6.0E-06	3.4E-06	4.1E-06	4.6E-06	3.1E-06	2.1E-06	2.9E-06	3.7E-06	1.5E-05
1.1E-05	8.9E-06	8.3E-06	4.9E-06	3.4E-06	5.4E-06	6.4E-06	5.7E-06	3.3E-06	3.9E-06	4.3E-06	3.0E-06	2.0E-06	2.7E-06	3.5E-06	1.4E-05
5.8E-08	3.4E-08	2.6E-08	2.2E-08	2.3E-08	4.5E-08	4.7E-08	3.7E-08	2.1E-08	2.7E-08	2.6E-08	1.6E-08	9.7E-09	1.4E-08	2.2E-08	8.8E-08
1.1E-05	9.6E-06	8.8E-06	5.2E-06	3.6E-06	5.7E-06	6.8E-06	6.0E-06	3.4E-06	4.1E-06	4.6E-06	3.1E-06	2.1E-06	2.9E-06	3.7E-06	1.5E-05
1.1E-05	9.6E-06	8.8E-06	5.2E-06	3.6E-06	5.7E-06	6.8E-06	6.0E-06	3.4E-06	4.1E-06	4.6E-06	3.1E-06	2.1E-06	2.9E-06	3.7E-06	1.5E-05
1.1E-05	8.9E-06	8.3E-06	4.9E-06	3.4E-06	5.4E-06	6.4E-06	5.7E-06	3.3E-06	3.9E-06	4.3E-06	3.0E-06	2.0E-06	2.7E-06	3.5E-06	1.4E-05
1.1E-05	8.9E-06	8.3E-06	4.9E-06	3.4E-06	5.4E-06	6.4E-06	5.7E-06	3.3E-06	3.9E-06	4.3E-06	3.0E-06	2.0E-06	2.7E-06	3.5E-06	1.4E-05
400.	400.	400.	400.	400.	400.	400.	400.	400.	400.	400.	400.	400.	400.	400.	400.

0.8 KM

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
1.7E-06	3.2E-06	3.0E-06	1.7E-06	1.2E-06	1.8E-06	2.2E-06	2.0E-06	1.1E-06	1.3E-06	1.4E-06	9.8E-07	6.4E-07	9.3E-07	1.1E-06	4.6E-06
1.4E-06	2.9E-06	2.8E-06	1.6E-06	1.1E-06	1.7E-06	2.0E-06	1.8E-06	9.9E-07	1.2E-06	1.3E-06	9.0E-07	5.9E-07	8.5E-07	1.0E-06	4.2E-06
1.9E-08	1.1E-08	8.7E-09	7.5E-09	7.9E-09	1.5E-08	1.6E-08	1.2E-08	7.1E-09	9.1E-09	8.7E-09	5.5E-09	3.3E-09	4.7E-09	7.3E-09	3.0E-08
1.7E-06	3.2E-06	3.0E-06	1.7E-06	1.2E-06	1.8E-06	2.2E-06	2.0E-06	1.1E-06	1.3E-06	1.4E-06	9.8E-07	6.4E-07	9.3E-07	1.1E-06	4.6E-06
1.7E-06	3.2E-06	3.0E-06	1.7E-06	1.2E-06	1.8E-06	2.2E-06	2.0E-06	1.1E-06	1.3E-06	1.4E-06	9.8E-07	6.4E-07	9.3E-07	1.1E-06	4.6E-06
1.4E-06	2.9E-06	2.8E-06	1.6E-06	1.1E-06	1.7E-06	2.0E-06	1.8E-06	9.9E-07	1.2E-06	1.3E-06	9.0E-07	5.9E-07	8.5E-07	1.0E-06	4.2E-06
1.4E-06	2.9E-06	2.8E-06	1.6E-06	1.1E-06	1.7E-06	2.0E-06	1.8E-06	9.9E-07	1.2E-06	1.3E-06	9.0E-07	5.9E-07	8.5E-07	1.0E-06	4.2E-06
800.	800.	800.	800.	800.	800.	800.	800.	800.	800.	800.	800.	800.	800.	800.	800.

1.2 KM

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
1.9E-06	1.7E-06	1.6E-06	9.0E-07	6.0E-07	9.3E-07	1.1E-06	1.0E-06	5.5E-07	6.4E-07	7.3E-07	5.0E-07	3.2E-07	4.7E-07	5.6E-07	2.4E-06
1.7E-06	1.5E-06	1.4E-06	8.0E-07	5.3E-07	8.3E-07	1.0E-06	8.9E-07	4.9E-07	5.7E-07	6.5E-07	4.4E-07	2.9E-07	4.2E-07	5.0E-07	2.1E-06
1.0E-08	5.9E-09	4.5E-09	3.9E-09	4.1E-09	7.9E-09	8.3E-09	6.5E-09	3.7E-09	4.7E-09	4.5E-09	2.9E-09	1.7E-09	2.4E-09	3.8E-09	1.5E-08
1.9E-06	1.7E-06	1.6E-06	8.9E-07	6.0E-07	9.3E-07	1.1E-06	1.0E-06	5.5E-07	6.4E-07	7.3E-07	5.0E-07	3.2E-07	4.7E-07	5.6E-07	2.4E-06
1.9E-06	1.7E-06	1.6E-06	8.9E-07	6.0E-07	9.3E-07	1.1E-06	1.0E-06	5.5E-07	6.4E-07	7.3E-07	5.0E-07	3.2E-07	4.7E-07	5.6E-07	2.4E-06
1.7E-06	1.5E-06	1.4E-06	8.0E-07	5.3E-07	8.3E-07	1.0E-06	8.9E-07	4.9E-07	5.7E-07	6.5E-07	4.4E-07	2.9E-07	4.2E-07	5.0E-07	2.1E-06
1.7E-06	1.5E-06	1.4E-06	8.0E-07	5.3E-07	8.3E-07	1.0E-06	8.9E-07	4.9E-07	5.7E-07	6.5E-07	4.4E-07	2.9E-07	4.2E-07	5.0E-07	2.1E-06
1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.	1200.

1.6 KM

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
1.2E-06	1.1E-06	1.0E-06	5.6E-07	3.7E-07	5.8E-07	7.0E-07	6.3E-07	3.4E-07	4.0E-07	4.6E-07	3.1E-07	2.0E-07	2.9E-07	3.4E-07	1.5E-06
1.0E-06	9.3E-07	9.0E-07	4.9E-07	3.3E-07	5.0E-07	6.1E-07	5.5E-07	3.0E-07	3.5E-07	4.0E-07	2.7E-07	1.7E-07	2.6E-07	3.0E-07	1.3E-06
6.3E-09	3.7E-09	2.8E-09	2.4E-09	2.5E-09	4.9E-09	5.2E-09	4.1E-09	2.3E-09	3.0E-09	2.8E-09	1.8E-09	1.1E-09	1.5E-09	2.4E-09	9.6E-09
1.2E-06	1.1E-06	1.0E-06	5.6E-07	3.7E-07	5.8E-07	7.0E-07	6.2E-07	3.4E-07	4.0E-07	4.5E-07	3.1E-07	2.0E-07	2.9E-07	3.4E-07	1.5E-06
1.2E-06	1.1E-06	1.0E-06	5.6E-07	3.7E-07	5.8E-07	7.0E-07	6.3E-07	3.4E-07	4.0E-07	4.5E-07	3.1E-07	2.0E-07	2.9E-07	3.4E-07	1.5E-06
1.0E-06	9.2E-07	8.9E-07	4.9E-07	3.3E-07	5.0E-07	6.1E-07	5.5E-07	3.0E-07	3.5E-07	4.0E-07	2.7E-07	1.7E-07	2.6E-07	3.0E-07	1.3E-06
1.0E-06	9.3E-07	8.9E-07	4.9E-07	3.3E-07	5.0E-07	6.1E-07	5.5E-07	3.0E-07	3.5E-07	4.0E-07	2.7E-07	1.7E-07	2.6E-07	3.0E-07	1.3E-06
1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.	1600.

TOTAL OBS - 8703 TOTAL INV OBS - 180 CALMS UPPER LEVEL - 0.00 CALMS LOWER LEV - 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3)
ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)
ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/64 TO 12/31/64

PROGRAM ANDIFF — AFRRI-TRIGA REACTOR
GROUND RELEASE - STANDARD PTS. — AFRRI
NATIONAL AIRPORT MET. DATA — BETHESDA, MARYLAND
DATE 10/15/90. TIME 20.12.21. — DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
2.4 KM															
6.2E-07	5.6E-07	5.4E-07	3.0E-07	1.9E-07	3.0E-07	3.6E-07	3.3E-07	1.8E-07	2.0E-07	2.3E-07	1.6E-07	1.0E-07	1.5E-07	1.7E-07	7.6E-07
5.3E-07	4.7E-07	4.6E-07	2.5E-07	1.6E-07	2.5E-07	3.1E-07	2.8E-07	1.5E-07	1.7E-07	2.0E-07	1.3E-07	8.7E-08	1.3E-07	1.5E-07	6.4E-07
3.2E-09	1.9E-09	1.4E-09	1.2E-09	1.3E-09	2.5E-09	2.6E-09	2.1E-09	1.2E-09	1.5E-09	1.4E-09	9.2E-10	5.4E-10	7.8E-10	1.2E-09	4.9E-09
6.2E-07	5.6E-07	5.4E-07	2.9E-07	1.9E-07	3.0E-07	3.6E-07	3.2E-07	1.8E-07	2.0E-07	2.3E-07	1.6E-07	1.0E-07	1.5E-07	1.7E-07	7.5E-07
6.2E-07	5.6E-07	5.4E-07	3.0E-07	1.9E-07	3.0E-07	3.6E-07	3.3E-07	1.8E-07	2.0E-07	2.3E-07	1.6E-07	1.0E-07	1.5E-07	1.7E-07	7.5E-07
5.3E-07	4.7E-07	4.6E-07	2.5E-07	1.6E-07	2.5E-07	3.0E-07	2.7E-07	1.5E-07	1.7E-07	2.0E-07	1.3E-07	8.6E-08	1.3E-07	1.5E-07	6.4E-07
5.3E-07	4.7E-07	4.6E-07	2.5E-07	1.6E-07	2.5E-07	3.1E-07	2.7E-07	1.5E-07	1.7E-07	2.0E-07	1.3E-07	8.7E-08	1.3E-07	1.5E-07	6.4E-07
2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.	2400.
3.2 KM															
4.0E-07	3.6E-07	3.5E-07	1.9E-07	1.2E-07	1.9E-07	2.3E-07	2.1E-07	1.1E-07	1.3E-07	1.5E-07	9.9E-08	6.4E-08	9.7E-08	1.1E-07	4.8E-07
3.3E-07	3.0E-07	2.9E-07	1.6E-07	1.0E-07	1.5E-07	1.9E-07	1.7E-07	9.1E-08	1.0E-07	1.2E-07	8.2E-08	5.3E-08	8.0E-08	9.0E-08	3.9E-07
2.0E-09	1.2E-09	9.0E-10	7.7E-10	8.0E-10	1.6E-09	1.6E-09	1.3E-09	7.3E-10	9.4E-10	8.9E-10	5.7E-10	3.4E-10	4.8E-10	7.5E-10	3.0E-09
3.9E-07	3.6E-07	3.5E-07	1.9E-07	1.2E-07	1.9E-07	2.3E-07	2.1E-07	1.1E-07	1.3E-07	1.5E-07	9.9E-08	6.4E-08	9.6E-08	1.1E-07	4.7E-07
4.0E-07	3.6E-07	3.5E-07	1.9E-07	1.2E-07	1.9E-07	2.3E-07	2.1E-07	1.1E-07	1.3E-07	1.5E-07	9.9E-08	6.4E-08	9.7E-08	1.1E-07	4.8E-07
3.2E-07	2.9E-07	2.9E-07	1.5E-07	1.0E-07	1.5E-07	1.9E-07	1.7E-07	9.1E-08	1.0E-07	1.2E-07	8.1E-08	5.3E-08	7.9E-08	9.0E-08	3.9E-07
3.3E-07	2.9E-07	2.9E-07	1.5E-07	1.0E-07	1.5E-07	1.9E-07	1.7E-07	9.1E-08	1.0E-07	1.2E-07	8.2E-08	5.3E-08	8.0E-08	9.0E-08	3.9E-07
3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.	3200.
4.0 KM															
2.8E-07	2.6E-07	2.5E-07	1.3E-07	8.7E-08	1.3E-07	1.6E-07	1.5E-07	7.8E-08	8.9E-08	1.0E-07	7.0E-08	4.6E-08	6.8E-08	7.7E-08	3.3E-07
2.3E-07	2.1E-07	2.0E-07	1.1E-07	7.0E-08	1.1E-07	1.3E-07	1.2E-07	6.3E-08	7.2E-08	8.4E-08	5.6E-08	3.7E-08	5.5E-08	6.2E-08	2.7E-07
1.4E-09	8.0E-10	6.1E-10	5.3E-10	5.5E-10	1.1E-09	1.1E-09	8.8E-10	5.0E-10	6.4E-10	6.1E-10	3.9E-10	2.3E-10	3.3E-10	5.2E-10	2.1E-09
2.8E-07	2.5E-07	2.5E-07	1.3E-07	8.6E-08	1.3E-07	1.6E-07	1.4E-07	7.7E-08	8.9E-08	1.0E-07	7.0E-08	4.5E-08	6.8E-08	7.6E-08	3.3E-07
2.8E-07	2.5E-07	2.5E-07	1.3E-07	8.6E-08	1.3E-07	1.6E-07	1.5E-07	7.8E-08	8.9E-08	1.0E-07	7.0E-08	4.5E-08	6.8E-08	7.7E-08	3.3E-07
2.2E-07	2.0E-07	2.0E-07	1.1E-07	6.9E-08	1.1E-07	1.3E-07	1.2E-07	6.2E-08	7.1E-08	8.3E-08	5.6E-08	3.6E-08	5.5E-08	6.1E-08	2.7E-07
2.2E-07	2.1E-07	2.0E-07	1.1E-07	7.0E-08	1.1E-07	1.3E-07	1.2E-07	6.2E-08	7.2E-08	8.4E-08	5.6E-08	3.7E-08	5.5E-08	6.2E-08	2.7E-07
4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.	4000.
4.8 KM															
2.1E-07	1.9E-07	1.9E-07	1.0E-07	6.5E-08	1.0E-07	1.2E-07	1.1E-07	5.9E-08	6.7E-08	7.8E-08	5.3E-08	3.4E-08	5.2E-08	5.8E-08	2.5E-07
1.7E-07	1.5E-07	1.5E-07	8.0E-08	5.2E-08	7.9E-08	9.6E-08	8.6E-08	4.6E-08	5.3E-08	6.2E-08	4.1E-08	2.7E-08	4.1E-08	4.5E-08	2.0E-07
1.0E-09	5.9E-10	4.5E-10	3.9E-10	4.0E-10	7.8E-10	8.2E-10	6.4E-10	3.7E-10	4.7E-10	4.5E-10	2.9E-10	1.7E-10	2.4E-10	3.8E-10	1.5E-09
2.1E-07	1.9E-07	1.9E-07	1.0E-07	6.5E-08	9.9E-08	1.2E-07	1.1E-07	5.8E-08	6.7E-08	7.8E-08	5.2E-08	3.4E-08	5.1E-08	5.7E-08	2.5E-07
2.1E-07	1.9E-07	1.9E-07	1.0E-07	6.5E-08	9.9E-08	1.2E-07	1.1E-07	5.8E-08	6.7E-08	7.8E-08	5.3E-08	3.4E-08	5.2E-08	5.8E-08	2.5E-07
1.7E-07	1.5E-07	1.5E-07	7.9E-08	5.1E-08	7.8E-08	9.5E-08	8.6E-08	4.6E-08	5.3E-08	6.1E-08	4.1E-08	2.7E-08	4.0E-08	4.5E-08	2.0E-07
1.7E-07	1.5E-07	1.5E-07	8.0E-08	5.1E-08	7.9E-08	9.6E-08	8.6E-08	4.6E-08	5.3E-08	6.2E-08	4.1E-08	2.7E-08	4.1E-08	4.5E-08	2.0E-07
4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.	4800.

TOTAL OBS = 8783 TOTAL INV OBS = 340 CALMS UPPER LEVEL = 0.00 CALMS LOWER LEV = 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3)
ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)
ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/66 TO 12/31/66

PROGRAM AUDIFF
GROUND RELEASE - STANDARD PTS.
NATIONAL AIRPORT MET. DATA
DATE 10/16/80. TIME 20.12.21.

AFRRI-TRIGA REACTOR
AFRRI
BETHESDA, MARYLAND
DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
12.0 KM																
	5.4E-08	5.1E-08	5.0E-08	2.6E-08	1.7E-08	2.5E-08	3.1E-08	2.9E-08	1.5E-08	1.7E-08	2.0E-08	1.3E-08	8.9E-09	1.3E-08	1.4E-08	6.3E-08
	3.8E-08	3.6E-08	3.5E-08	1.8E-08	1.2E-08	1.7E-08	2.1E-08	1.9E-08	1.0E-08	1.2E-08	1.4E-08	9.2E-09	6.2E-09	9.3E-09	1.0E-08	4.4E-08
	2.0E-10	1.2E-10	9.1E-11	7.0E-11	8.2E-11	1.6E-10	1.7E-10	1.3E-10	7.5E-11	9.5E-11	9.1E-11	5.0E-11	3.4E-11	4.9E-11	7.7E-11	3.1E-10
	5.3E-08	5.0E-08	4.9E-08	2.6E-08	1.6E-08	2.5E-08	3.0E-08	2.9E-08	1.4E-08	1.6E-08	2.0E-08	1.3E-08	8.7E-09	1.3E-08	1.4E-08	6.3E-08
	5.4E-08	5.1E-08	5.0E-08	2.6E-08	1.7E-08	2.5E-08	3.1E-08	2.9E-08	1.5E-08	1.7E-08	2.0E-08	1.3E-08	8.8E-09	1.3E-08	1.4E-08	6.3E-08
	3.7E-08	3.5E-08	3.4E-08	1.8E-08	1.1E-08	1.7E-08	2.1E-08	1.9E-08	1.0E-08	1.1E-08	1.4E-08	9.1E-09	6.1E-09	9.1E-09	9.9E-09	4.4E-08
	3.8E-08	3.5E-08	3.5E-08	1.8E-08	1.2E-08	1.7E-08	2.1E-08	1.9E-08	1.0E-08	1.2E-08	1.4E-08	9.2E-09	6.2E-09	9.2E-09	1.0E-08	4.4E-08
	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.	12000.
16.0 KM																
	3.6E-08	3.4E-08	3.4E-08	1.8E-08	1.1E-08	1.6E-08	2.0E-08	1.8E-08	9.7E-09	1.1E-08	1.3E-08	8.7E-09	5.9E-09	8.8E-09	9.6E-09	4.2E-08
	2.4E-08	2.3E-08	2.2E-08	1.2E-08	7.3E-09	1.1E-08	1.4E-08	1.2E-08	6.4E-09	7.3E-09	8.7E-09	5.8E-09	3.9E-09	5.9E-09	6.3E-09	2.8E-08
	1.2E-10	7.1E-11	5.5E-11	4.7E-11	4.9E-11	9.5E-11	1.0E-10	7.8E-11	4.5E-11	5.7E-11	5.4E-11	3.5E-11	2.0E-11	2.9E-11	4.6E-11	1.9E-10
	3.5E-08	3.3E-08	3.3E-08	1.7E-08	1.1E-08	1.6E-08	2.0E-08	1.8E-08	9.5E-09	1.1E-08	1.3E-08	8.5E-09	5.8E-09	8.6E-09	9.4E-09	4.1E-08
	3.6E-08	3.4E-08	3.3E-08	1.7E-08	1.1E-08	1.6E-08	2.0E-08	1.8E-08	9.6E-09	1.1E-08	1.3E-08	8.7E-09	5.9E-09	8.8E-09	9.5E-09	4.2E-08
	2.3E-08	2.2E-08	2.2E-08	1.1E-08	7.2E-09	1.1E-08	1.3E-08	1.2E-08	6.3E-09	7.1E-09	8.5E-09	5.7E-09	3.8E-09	5.7E-09	6.2E-09	2.7E-08
	2.4E-08	2.3E-08	2.2E-08	1.2E-08	7.3E-09	1.1E-08	1.3E-08	1.2E-08	6.4E-09	7.3E-09	8.6E-09	5.8E-09	3.9E-09	5.8E-09	6.3E-09	2.8E-08
	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.	16000.
24.0 KM																
	2.1E-08	2.0E-08	1.9E-08	1.0E-08	6.1E-09	9.3E-09	1.2E-08	1.0E-08	5.4E-09	6.2E-09	7.4E-09	4.9E-09	3.4E-09	5.0E-09	5.4E-09	2.4E-08
	1.3E-08	1.2E-08	1.2E-08	6.2E-09	3.9E-09	5.7E-09	7.1E-09	6.5E-09	3.4E-09	3.8E-09	4.6E-09	3.0E-09	2.1E-09	3.1E-09	3.3E-09	1.5E-08
	5.9E-11	3.4E-11	2.6E-11	2.2E-11	2.3E-11	4.6E-11	4.8E-11	3.8E-11	2.1E-11	2.7E-11	2.6E-11	1.7E-11	9.8E-12	1.4E-11	2.2E-11	8.9E-11
	2.0E-08	1.9E-08	1.9E-08	9.7E-09	6.1E-09	9.1E-09	1.1E-08	1.0E-08	5.3E-09	6.0E-09	7.2E-09	4.8E-09	3.2E-09	4.9E-09	5.2E-09	2.3E-08
	2.0E-08	1.9E-08	1.9E-08	1.0E-08	6.2E-09	9.2E-09	1.1E-08	1.0E-08	5.4E-09	6.1E-09	7.3E-09	4.9E-09	3.3E-09	5.0E-09	5.3E-09	2.3E-08
	1.2E-08	1.2E-08	1.1E-08	6.0E-09	3.8E-09	5.6E-09	6.9E-09	6.3E-09	3.2E-09	3.7E-09	4.4E-09	2.9E-09	2.0E-09	3.0E-09	3.2E-09	1.4E-08
	1.3E-08	1.2E-08	1.2E-08	6.2E-09	3.8E-09	5.7E-09	7.0E-09	6.4E-09	3.3E-09	3.8E-09	4.5E-09	3.0E-09	2.1E-09	3.1E-09	3.3E-09	1.4E-08
	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.	24000.
32.0 KM																
	1.4E-08	1.3E-08	1.3E-08	6.9E-09	4.3E-09	6.3E-09	7.8E-09	7.1E-09	3.7E-09	4.1E-09	5.0E-09	3.3E-09	2.3E-09	3.4E-09	3.6E-09	1.6E-08
	8.1E-09	7.8E-09	7.7E-09	4.0E-09	2.5E-09	3.6E-09	4.5E-09	4.1E-09	2.1E-09	2.4E-09	2.9E-09	1.9E-09	1.3E-09	2.0E-09	2.1E-09	9.2E-09
	3.5E-11	2.0E-11	1.6E-11	1.3E-11	1.4E-11	2.7E-11	2.8E-11	2.2E-11	1.1E-11	1.6E-11	1.5E-11	9.8E-12	5.8E-12	8.3E-12	1.3E-11	5.3E-11
	1.3E-08	1.3E-08	1.3E-08	6.5E-09	4.1E-09	6.0E-09	7.5E-09	6.8E-09	3.5E-09	4.0E-09	4.8E-09	3.2E-09	2.2E-09	3.3E-09	3.5E-09	1.5E-08
	1.4E-08	1.3E-08	1.3E-08	6.8E-09	4.2E-09	6.2E-09	7.7E-09	7.0E-09	3.6E-09	4.1E-09	4.9E-09	3.3E-09	2.3E-09	3.4E-09	3.6E-09	1.6E-08
	7.7E-09	7.4E-09	7.3E-09	3.8E-09	2.4E-09	3.5E-09	4.3E-09	3.9E-09	2.0E-09	2.3E-09	2.8E-09	1.8E-09	1.3E-09	1.9E-09	2.0E-09	8.9E-09
	8.0E-09	7.7E-09	7.6E-09	3.9E-09	2.4E-09	3.6E-09	4.5E-09	4.1E-09	2.1E-09	2.4E-09	2.9E-09	1.9E-09	1.3E-09	2.0E-09	2.1E-09	9.1E-09
	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.	32000.

TOTAL OBS = 8784   TOTAL INV OBS = 300   CALMS UPPER LEVEL = 0.00   CALMS LOWER LEV = 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - X/Q (S/M**3)
ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2)
ENTRY 4 DECAYED X/Q (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED X/Q (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 6 DEC+DPL X/Q (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL X/Q (S/M**3) - HALF LIFE 8.00 DAYS
ENTRY 8 - DISTANCE IN METERS

TABLE 1

AVERAGE METEOROLOGICAL RELATIVE CONCENTRATION ANALYSIS
DATA PERIOD : 01/01/66 TO 12/31/66

PROGRAM AIRDIFF	AFRRI-TRIGA REACTOR
GROUND RELEASE - STANDARD PTS.	AFRRI
NATIONAL AIRPORT MET. DATA	BETHESDA, MARYLAND
DATE 10/16/80. TIME 20.12.21.	DAMES AND MOORE JOB 11985-001-27

AFFECTED SECTORS

NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	N
5.6 KM															
1.7E-07	1.5E-07	1.5E-07	8.0E-08	5.2E-08	7.0E-08	9.6E-08	8.7E-08	4.6E-08	5.3E-08	6.2E-08	4.2E-08	2.7E-08	4.1E-08	4.5E-08	2.0E-07
1.3E-07	1.2E-07	1.2E-07	6.2E-08	4.0E-08	6.1E-08	7.4E-08	6.7E-08	3.6E-08	4.1E-08	4.8E-08	3.2E-08	2.1E-08	3.2E-08	3.5E-08	1.5E-07
7.7E-10	4.5E-10	3.4E-10	3.0E-10	3.1E-10	6.9E-10	6.3E-10	4.9E-10	2.8E-10	3.6E-10	3.4E-10	2.2E-10	1.3E-10	1.9E-10	2.9E-10	1.2E-09
1.7E-07	1.5E-07	1.5E-07	8.0E-08	5.1E-08	7.0E-08	9.5E-08	8.6E-08	4.6E-08	5.2E-08	6.1E-08	4.1E-08	2.7E-08	4.1E-08	4.5E-08	2.0E-07
1.7E-07	1.5E-07	1.5E-07	8.0E-08	5.2E-08	7.0E-08	9.6E-08	8.7E-08	4.6E-08	5.3E-08	6.2E-08	4.1E-08	2.7E-08	4.1E-08	4.5E-08	2.0E-07
1.3E-07	1.2E-07	1.2E-07	6.2E-08	4.0E-08	6.0E-08	7.4E-08	6.7E-08	3.5E-08	4.1E-08	4.8E-08	3.2E-08	2.1E-08	3.1E-08	3.5E-08	1.5E-07
1.3E-07	1.2E-07	1.2E-07	6.2E-08	4.0E-08	6.1E-08	7.4E-08	6.7E-08	3.6E-08	4.1E-08	4.8E-08	3.2E-08	2.1E-08	3.2E-08	3.5E-08	1.5E-07
5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.	5600.
6.4 KM															
1.4E-07	1.3E-07	1.2E-07	6.6E-08	4.2E-08	6.4E-08	7.8E-08	7.1E-08	3.8E-08	4.3E-08	5.1E-08	3.4E-08	2.2E-08	3.3E-08	3.7E-08	1.6E-07
1.0E-07	9.6E-08	9.4E-08	5.0E-08	3.2E-08	4.9E-08	6.0E-08	5.4E-08	2.9E-08	3.3E-08	3.8E-08	2.6E-08	1.7E-08	2.5E-08	2.8E-08	1.2E-07
6.1E-10	3.6E-10	2.7E-10	2.3E-10	2.5E-10	4.4E-10	5.0E-10	3.9E-10	2.2E-10	2.9E-10	2.7E-10	1.7E-10	1.0E-10	1.5E-10	2.3E-10	9.3E-10
1.4E-07	1.3E-07	1.2E-07	6.5E-08	4.2E-08	6.4E-08	7.8E-08	7.0E-08	3.7E-08	4.3E-08	5.0E-08	3.4E-08	2.2E-08	3.3E-08	3.7E-08	1.6E-07
1.4E-07	1.3E-07	1.2E-07	6.6E-08	4.2E-08	6.4E-08	7.8E-08	7.1E-08	3.8E-08	4.3E-08	5.0E-08	3.4E-08	2.2E-08	3.3E-08	3.7E-08	1.6E-07
1.0E-07	9.5E-08	9.3E-08	5.0E-08	3.2E-08	4.8E-08	5.9E-08	5.3E-08	2.8E-08	3.2E-08	3.8E-08	2.6E-08	1.7E-08	2.5E-08	2.8E-08	1.2E-07
1.0E-07	9.6E-08	9.4E-08	5.0E-08	3.2E-08	4.9E-08	6.0E-08	5.4E-08	2.9E-08	3.3E-08	3.8E-08	2.6E-08	1.7E-08	2.5E-08	2.8E-08	1.2E-07
6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.	6400.
7.2 KM															
1.1E-07	1.1E-07	1.0E-07	5.5E-08	3.5E-08	5.4E-08	6.6E-08	5.9E-08	3.1E-08	3.6E-08	4.2E-08	2.8E-08	1.9E-08	2.8E-08	3.1E-08	1.4E-07
9.6E-08	8.0E-08	7.8E-08	4.1E-08	2.7E-08	4.0E-08	4.9E-08	4.4E-08	2.4E-08	2.7E-08	3.2E-08	2.1E-08	1.4E-08	2.1E-08	2.3E-08	1.0E-07
5.0E-10	2.9E-10	2.2E-10	1.8E-10	2.0E-10	3.9E-10	4.1E-10	3.2E-10	1.8E-10	2.3E-10	2.2E-10	1.4E-10	8.4E-11	1.2E-10	1.9E-10	7.6E-10
1.1E-07	1.1E-07	1.0E-07	5.5E-08	3.5E-08	5.3E-08	6.5E-08	5.9E-08	3.1E-08	3.6E-08	4.2E-08	2.8E-08	1.9E-08	2.8E-08	3.1E-08	1.3E-07
1.1E-07	1.1E-07	1.0E-07	5.5E-08	3.5E-08	5.3E-08	6.5E-08	5.9E-08	3.1E-08	3.6E-08	4.2E-08	2.8E-08	1.9E-08	2.8E-08	3.1E-08	1.4E-07
8.5E-08	7.9E-08	7.7E-08	4.1E-08	2.6E-08	4.0E-08	4.9E-08	4.4E-08	2.3E-08	2.7E-08	3.1E-08	2.1E-08	1.4E-08	2.1E-08	2.3E-08	1.0E-07
8.6E-08	8.0E-08	7.8E-08	4.1E-08	2.6E-08	4.0E-08	4.9E-08	4.4E-08	2.4E-08	2.7E-08	3.2E-08	2.1E-08	1.4E-08	2.1E-08	2.3E-08	1.0E-07
7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.	7200.
8.0 KM															
9.8E-08	9.1E-08	9.0E-08	4.7E-08	3.0E-08	4.6E-08	5.6E-08	5.1E-08	2.7E-08	3.1E-08	3.6E-08	2.4E-08	1.6E-08	2.4E-08	2.6E-08	1.2E-07
7.3E-08	6.7E-08	6.6E-08	3.5E-08	2.2E-08	3.4E-08	4.1E-08	3.7E-08	2.0E-08	2.3E-08	2.7E-08	1.8E-08	1.2E-08	1.8E-08	2.0E-08	8.6E-08
4.2E-10	2.4E-10	1.9E-10	1.6E-10	1.7E-10	3.2E-10	3.4E-10	2.7E-10	1.5E-10	1.9E-10	1.9E-10	1.2E-10	7.0E-11	1.0E-10	1.6E-10	6.3E-10
9.7E-08	9.0E-08	8.8E-08	4.7E-08	3.0E-08	4.5E-08	5.6E-08	5.0E-08	2.7E-08	3.0E-08	3.6E-08	2.4E-08	1.6E-08	2.4E-08	2.6E-08	1.1E-07
9.8E-08	9.1E-08	8.9E-08	4.7E-08	3.0E-08	4.6E-08	5.6E-08	5.1E-08	2.7E-08	3.1E-08	3.6E-08	2.4E-08	1.6E-08	2.4E-08	2.6E-08	1.2E-07
7.2E-08	6.7E-08	6.5E-08	3.5E-08	2.2E-08	3.3E-08	4.1E-08	3.7E-08	2.0E-08	2.2E-08	2.6E-08	1.8E-08	1.2E-08	1.8E-08	1.9E-08	8.5E-08
7.2E-08	6.7E-08	6.6E-08	3.5E-08	2.2E-08	3.4E-08	4.1E-08	3.7E-08	2.0E-08	2.3E-08	2.7E-08	1.8E-08	1.2E-08	1.8E-08	1.9E-08	8.5E-08
8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.	8000.

TOTAL OBS - 8783 TOTAL INV OBS - 180 CALMS UPPER LEVEL - 0.00 CALMS LOWER LEV - 0.00

KEY ENTRY 1 RELATIVE CONCENTRATION - XOQ (S/M**3) | ENTRY 2 DEPLETED RELATIVE CONCENTRATION (S/M**3)
ENTRY 3 RELATIVE DEPOSITION RATE (1/M**2) | ENTRY 4 DECAYED XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 5 DECAYED XOQ (S/M**3) - HALF LIFE 8.00 DAYS | ENTRY 6 DEC+DPL XOQ (S/M**3) - HALF LIFE 2.26 DAYS
ENTRY 7 DEC+DPL XOQ (S/M**3) - HALF LIFE 8.00 DAYS | ENTRY 8 - DISTANCE IN METERS

APPENDIX C

ANALYSIS OF LOSS-OF-COOLANT ACCIDENT

FOR

AFRRI-TRIGA REACTOR

APPENDIX C

ANALYSIS OF LOSS-OF-COOLANT ACCIDENT

FOR AFRRI-TRIGA REACTOR

The assumption is made that the water from the reactor tank is completely lost due to a tank rupture at or below the reactor core level. Rupture of the fuel cladding or fuel melt would not occur even if only convective air cooling was available following loss of the water. Within an hour, the reactor core would be moved behind the lead shield so that repair of the tank could be made without a radiation exposure risk to workers and other personnel.

If the water shield is lost, the possible exposure to AFRRI personnel or the general public would be due to direct or scattered gamma radiation from the exposed reactor fuel inside the reactor tank. The following assumptions have been made for the irradiated fuel in the analysis of postulated gamma doses within the reactor room and outside the AFRRI facility:

1. Power level = 1 MW(t) (steady-state operation) [1]
2. Number of fuel elements = 85
3. Mass of each fuel element = [redacted] [redacted]
4. Mass of uranium = [redacted] [redacted]
5. Mass of zirconium and stainless steel = 2.1 kg [2]
6. Distance from top of core to reactor tank top = 505 cm
7. Distance from reactor room floor to ceiling = 550 cm
8. Reactor tank diameter = 366 cm [3]
9. Ceiling thickness = 10 cm of ordinary concrete [3]
10. Wall thickness = 25 cm of ordinary concrete [3]
11. Total core activity = [redacted] [redacted]
12. Total gamma power = 1.75×10^{17} MeV/sec [5,6]
13. Density of fuel element = 7.9 gm/cm^3
14. Volume of fuel elements = [redacted] [redacted] [redacted]

The gamma dose rate on the end surface of a right cylinder is determined by the following equation.[7] The reactor core is assumed to be a right cylinder.

$$D_\gamma = \frac{S_v}{2\mu_s K}\left[1 - E_2(\mu_s t)\right]$$

where:

D_γ = source gamma dose rate (rem/hr)

S_v = volume source strength ($MeV/sec\text{-}cm^3$)

μ_s = attenuation coefficient for source (cm^{-1})

K = flux to dose rate conversion factor (MeV/cm^2-sec per rem/hr)

E_2 = exponential function

t = thickness of source (cm)

The source strength (S_v) for the reactor core is determined to be 7×10^{12} $MeV/sec\text{-}cm^3$. The average photon energy is assumed to be 1 MeV. Therefore, the attenuation coefficient for the core (μ_s) is 0.46 cm^{-1} (coefficient of iron) and the flux to dose conversion factor (K) is 5.2×10^5 MeV/cm^2-sec per rem/hr. The thickness of the source is 87 cm. The calculated gamma dose rate (D_γ) at the top surface (1 cm) of the core is 1.5×10^7 rem/hr. The measured gamma dose rate at the surface of TRIGA fuel operated at 250 kw(t) after 1 hour decay is 10^3 rem/hr.[2] Since the dose reduction factor for 1 hour decay is 4[5,6], and the power level is a factor of 4 lower, the expected surface dose rate of an AFRRI fuel element would be 1.6×10^4 rem/hr. For an 85 fuel element core, the maximum expected gamma dose rate above the core from the measured dose rate would be 1.4×10^6 rem/hr, or a factor of more than 10 less than the calculated value. The actual operating time for the fuel element which was measured would be another factor in reducing the difference between the calculated and measured gamma dose rate. The calculated value assumes an equilibrium or steady-state condition. For example, the gamma decay power level for a 1-day operation is only a factor of 1.5 less than the equilibrium value.[4]

To determine the gamma dose rate to an individual in the reactor room, the contribution from the air scatter and the ceiling scatter must be determined. From Reference 8, the reduction factors for a narrow beam at various distances for 1 MeV photons are 1.4 x 10^{-8} at 10 feet, 3.8 x 10^{-9} at 30 feet, and 1.5 x 10^{-9} at 60 feet. The other factor to be determined is the solid angle of the escaping cone from the reactor core (a point source). The solid angle for the reactor tank diameter opening is approximately 0.45 (40° solid angle). The following equation is used to determine air-scattered gamma dose rate within the reactor room:

$$D_A = (D)(RF)(A)$$

where:

D_A = air-scattered gamma dose rate (rem/hr)

D = source gamma dose rate at 1 cm (rem/hr)

RF = reduction factor for air scatter

A = solid angle of escaping beam (steradian).

With a source gamma dose rate of 1.5 x 10^{7} rem/hr, the air scatter gamma dose rates at the reactor room floor level are calculated to be 3 x 10^{-1} rem/hr at 10 feet, 8 x 10^{-2} rem/hr at 30 feet, and 3.2 x 10^{-2} rem/hr at 60 feet from the reactor tank.

The scattering from the ceiling can be determined by using the method given in Reference 8 for an extended slab. The following equation can be used to determine the contribution from the ceiling scatter of the escaping beam (reactor core):

$$D_s = (D)(r_i^{-2})(r_s^{-2})(RF_s)$$

where:

D_s = slab scattered gamma dose rate (rem/hr)

D = source gamma dose rate at 1 cm (rem/hr)

r_i^{-2} = inverse square factor for incidence beam

r_s^{-2} = inverse square factor for scattered beam

RF_s = reflection factor for extended slab.

The inverse square factor for the incidence beam is 9 x 10^{-7} (1055 cm) and for the scattered beam is 3.3 x 10^{-6} (550 cm). The reflection factor at normal incidence is 7.8 x 10^{-4}. With a source gamma dose rate of 1.5 x 10^{7} rem/hr, the ceiling scattered gamma dose rate is negligible (3.5 x 10^{-7} rem/hr) and would not contribute to the gamma dose rate to an individual in the reactor room or outside the AFRRI facility.

The gamma dose rate on the roof of the AFRRI facility over the reactor room can be determined by the following equation:[8]

$$D_d = D_\gamma r_i^{-2} e^{-\mu t}$$

where:

D_d = direct gamma dose rate (rem/hr)

D_γ = source gamma dose rate (rem/hr)

r_i^{-2} = inverse square factor for incidence beam

μ = attenuation coefficient for shield (cm^{-1})

t = thickness of shield (cm).

For the ordinary concrete ceiling, the thickness is 10 cm and the attenuation coefficient for 1 MeV photons is 0.145 cm^{-1}. The inverse square factor for the incidence beam is 9×10^{-7}. With a source gamma dose rate of 1.5×10^7 rem/hr, the gamma dose rate on the roof above the reactor core is 3.2 rem/hr. The air scatter to the ground from this source would be negligible. This gamma dose rate of 3.2 rem/hr should be reduced by at least the inverse square factor of the scattered beam, which would be 3.3×10^{-6} (550 cm).

To determine the maximum gamma dose rate to an individual outside the AFRRI facility, the calculated air-scatter gamma dose rate at 10 feet (nearest outside wall of reactor room) should be reduced by the shielding of the concrete wall. The calculated gamma dose rate at the outside surface of the nearest wall in reactor room would be 8 mrem/hr. Beyond 20 meters from the nearest wall, the dose rate would be less than 1 mrem/hr.

All of the calculated gamma dose rates assume a steady-state condition of 1 MW operation (at least 1 year operation) which probably results in a source term that is conservative by a factor of 10. These calculated gamma dose rates would be reduced by a factor of 4 within 1 hour and by a factor of 11 within 1 day.[5,6] Therefore, the integrated gamma dose for 1 hour would be 0.35 of the calculated hourly dose and the integrated gamma dose for 1 day would be only 3 times the calculated hourly dose.

TABLE 1

Gaseous Fission Product Inventory
1 MW Steady State Operation

[redacted]

* y = year, d = day, h = hour, m = minute

TABLE 2

Gaseous Fission Product Inventory for 40 MW-sec Pulse Operation

[redacted]

---

* -d = day, h = hour, m = minute

TABLE 3

Diffusion Factor and
Finite Cloud Correction Factor
Used for Dose Calculations

Distance (m)	χ/Q[8] (sec/m^3)	F*[9]
25	1.0 E-1	1.5 E-2
50	2.7 E-2	2.7 E-2
75	1.2 E-2	4.2 E-2
100	7.5 E-3	5.5 E-2
150	3.7 E-3	8.0 E-2
200	2.2 E-3	1.0 E-1

*F represents a ratio of gamma doses from a finite size cloud to that calculated from an infinite cloud with same center-line concentration.

TABLE 4

Gaseous Fission Product Inventory
for 1 MW Steady State Operation
with 2-Week Decay

[redacted]

TABLE 5

Calculated Doses to Individuals Standing Downwind of AFRRI Facility Following Fuel Element Drop Accident

Distance (m)	Whole Body Dose (mrads)	Thyroid Dose (mrads)
25	0.0004	45
50	--*	12
75	--	6
100	--	3
150	--	2
200	--	1

*Insignificant

TABLE 6

Gaseous Fission Product Inventory Released to Atmosphere
Following Fuel Element Clad Failure Accident

[redacted]

REFERENCES

1. Amendment #22 to the Technical Specifications, Facility License No. R-84 for AFRRI-TRIGA Mark-F Reactor, Docket No. 50-170.

2. Personal communications with AFRRI staff (Reactor Fuel Records).

3. AFRRI, Final Safeguard Report: Revised Edition, Docket No. 50-170, General Dynamics and Holmes & Narver, Inc. (March 1962).

4. Blomeke, J.O., and Todd, Mary F., Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time, Part 1, Volume 2, ORNL-2127 (December 1958).

5. Knabe, W.E., and Putnam, G.E., The Activity of the Fission Products of U^{235}, APEX-448 (January 1959).

6. Smith, M.R., The Activity of the Fission Products of U^{235} (Program 408), XDC-60-1-157 (December 10, 1959).

7. Foderaro, A., and Obenshain, F., Fluxes from Regular Geometric Sources, WAPD-TN-508 (June 1955).

8. Wende, C.W.J., The Computation of Radiation Hazards, AECD-3661, January 11, 1944.