DEPARTMENT OF VETERANS AFFAIRS
4101 Woolworth Avenue
Omaha NE 68105

October 2, 2000

In Reply Refer To:

US Nuclear Regulatory Commission
ATTN: Alexander Adams, Jr., REXB
Mail Stop 012-D3
Washington, D.C. 20555-0001

RE: Docket 50-131

Dear Al:

Enclosed you will find two copies of the additional information provided within the Safety Analysis Report (SAR), requested for the renewal of our Facility Operating License R-57. The original has been sent to the Document Control Desk.

It is my hope that I have adequately addressed the questions and comments outlined in your August 3, 2000 letter as well as recent meetings. There were numerous changes and additions to the SAR. It has been given revision based upon discussions with you, consultants, and VA staff. The Technical Specifications will be forthcoming after additional review.

The SAR is enclosed as a complete document with the exception of Fig. 2.1 and Fig. 2.7.

Sincerely,

JOHN P. CLAASSEN
Reactor Manager

Enclosure: SAR

PART 4

A.J. Blotcky Reactor Facility

Omaha VA Medical Center
Omaha, NE

SAFETY ANALYSIS REPORT

Table of Contents

Safety Analysis Report
Omaha VA Medical Center - TRIGA

List of Figures

List of Tables

Table of Contents
Appendices

List of Figures
Appendices

List of Tables
Appendices

Appendix A

Appendix B

Appendix C

Appendix D

Appendix E

CHAPTER 1

INTRODUCTION AND SUMMARY

This report describes the Alan J. Blotcky Reactor Facility (AJBRF) and the Omaha Department of Veterans Affairs Medical Center, and provides a safety evaluation which shows that the reactor or facility does not cause undue risk to the health and safety of the public. The AJBRF reactor has been operated safely at the facility between June 1959 and April 12, 1991 at 18 kW and at 20 kW from April 12, 1991 to date. There has been no change in the method of operation since the issuance of the last facility license in 1983. Safety analysis demonstrates safe operation at power levels well above the requested licensed power. The reactor is used for the conduct of research, development and educational activities.

1.1 PRINCIPAL DESIGN CRITERIA

The reactor is operated only in the steady state mode. Reactor power levels range up to and include 20 kW. A summary of principal design parameters for the reactor is given in Table 1-1.

1.2 DESIGN HIGHLIGHTS

The reactor will be located in a below ground reactor pool structure. Reactor cooling is provided by natural circulation of pool water which is cooled and purified in an external coolant circuit. Reactor experiment facilities include a rotary specimen rack, a pneumatic transfer system, and a core irradiation tube.

The inherent safety of this TRIGA reactor has been demonstrated by the extensive experience acquired from similar TRIGA systems throughout the world. Forty-eight TRIGA reactors are now in operation throughout the world. TRIGA reactors have more than 450 reactor years of operating experience, and more than 15,000 fuel element years of operation. The safety arises from a large, prompt negative temperature coefficient that is characteristic of uranium zirconium hydride fuel-elements used in TRIGA systems. As the fuel temperature increases, this coefficient immediately compensates for reactivity insertions. The result is that reactor power excursions are terminated quickly and safely.

The prompt shutdown mechanism has been demonstrated extensively in many thousands of transient tests performed in two prototype TRIGA reactors at the GA Technologies laboratory in San Diego, California, as well as other pulsing TRIGA reactors in operation. These tests included reactivity insertions as large as $ 2.00 with resulting peak reactor power of 250 MW on TRIGA cores containing similar fuel elements as are used in this TRIGA reactor [1 & 2].

Table 1-1

PRINCIPAL DESIGN PARAMETERS

Core	
Fuel elements:	
Fuel-moderator material	8 wt-% uranium, 91 wt-% zirconium, 1 wt-% hydrogen.
Uranium enrichment	≤20%
Fuel-element dimensions	1.48 dia. by 28.4 in long
Cladding	0.030-in. thick aluminum or 0.02-in. thick Stainless
Active lattice dimension	≅14 in. dia. by 14 in. high
Reflector	
Material	Graphite
Cladding material	Aluminum
Radial Thickness	12 in.
Top and bottom thickness	4 in.
Nuclear Characteristics	
Thermal neutron flux at 20 kW:	
Average in core	$\cong 3.2 \times 10^{11}$ n/cm^2-s
At rotary specimen rack	$\cong 1.35 \times 10^{11}$ n/cm^2-s
At central thimble	$\cong 9.23 \times 10^{11}$ n/cm^2-s
At pneumatic transfer tube	$\cong 3.71 \times 10^{11}$ n/cm^2-s
Initial excess reactivity	≅0.7% δk/k
Reactivity loss per year (20-kw operation, 8 h/day)	≅0.14% δk/k/yr
Reactivity loss per year (averaged from 1980 to 1994)	= 0.02% δk/k/yr.
Core loading	≅2.0 kg U-235

Total reactivity worth of control rods.....	$\cong$5.0% δk/k
Void coefficient of reactivity in core.....	$\cong$-0.001 dk/k/1% void
Prompt neutron lifetime.....	8×10^{-5} s

Thermal Characteristics

Power.....	20 kW
Method of cooling.....	Natural convection of water through core; pool water circulated through 5-ton Freon water chiller with air-cooled condenser

Control

Boron carbide control rods.....	3
Drives.....	Rack and pinion
Maximum reactivity insertion rate.....	$\cong$0.03% δk/k/s

Instrumentation

NM-1000 microprocessor based neutron monitoring system with:

Count-rate channel Log n channel Period channel Linear channel.....	1
Independent % power channel.....	1
Servo amplifier for rod controller.....	1
Rod-position indicator.....	2
Magnet scram amplifier.....	1
Water radiation monitor.....	1
Water conductivity monitor.....	1
Water temperature monitor.....	1

Start-up Source	
2-Ci americium-beryllium source.....	1
Structures	
Reactor pool.....	6ft 6in ID by 21 ft deep
Shielding	
Vertical.....	Water, 16 ft above core
Irradiation Facilities	
Rotary specimen rack.....	40-position rack located in graphite reflector
Pneumatic tube.....	1
Central thimble.....	1

Because the reactor fuel is similar, the previously cited experience and tests apply to this TRIGA system. As a result it has been possible to use accepted safety analysis techniques applied to other TRIGA facilities to update evaluations with regard to the characteristics of this facility [3].

1.3 CONCLUSIONS

Past experience has shown that TRIGA systems can be designed constructed, and safely operated in the steady state mode of operation. This history of safety and the conservative design of the reactor have permitted TRIGA systems to be sited in urban areas using buildings without pressure type containment.

Results of this safety analysis indicate that the TRIGA Mark I reactor system will pose no health or safety problem to the public when operated in either normal or abnormal conditions.

Abnormal or accident conditions considered in this analysis include:

a. A step insertion of reactivity,

b. Complete and instantaneous loss of coolant water in the reactor pool,

c. And fission product release from a fuel element ruptured in air.

For both the postulated insertion of excess reactivity and the loss of cooling water accident conditions, fuel and clad temperatures remain at levels below those required to generate stress conditions which would cause loss of clad integrity. However, the results of a clad failure are analyzed and it is shown that such a failure will not cause excess radiation exposure.

The loss of pool water has been examined from the standpoint of direct radiation to operating personnel as well as in terms of maintaining fuel integrity.

The effects of argon-41 and nitrogen-16 production during normal operation of the reactor have also been evaluated. Results of these analyses show that production of these radioactive gases will present no hazard to persons in the reactor room or to the general public.

Chapter 1

References

1. "TRIGA Transient Experiments" General Atomic GA-0531, Sept. 1958

2. "Hazards Report for Torrey Pines TRIGA Reactor" GA-0722, 1959

3. "Safeguards Analysis Report for TRIGA Reactors using Aluminum-Clad Fuel", General Atomic Division Report GA-7860, March 1967

CHAPTER 2

SITE DESCRIPTION

2.1 GENERAL LOCATION

The A.J. Blotcky Reactor Facility (AJBRF) is located in the Department of Veterans Affairs Medical center in the City of Omaha, Douglas County, Nebraska (see Fig. 2.1). The reactor is housed in the basement of the southwest wing of the medical center building (see Fig. 2.2).

The medical center is built on the high point of a knoll. To the north is a County hospital, to the south a commercial district, to the west a residential area, and to the east a golf course. There are no industrial activities in the are that would have an impact on the facility. The reactor site is 2 miles northwest of a large railroad yard and 8 miles northwest of Offutt Air Force Base, headquarters for the Air Combat Command of the U.S. Air Force.

2.2 POPULATION DENSITY

The city of Omaha had a 1990 population of 335,795 with 618,262 people in the metropolitan statistical area (MSA). Omaha MSA population projections can be seen in Figure 2.3. Growth continues to grow to the southwest with the largest percentage increase expected in Sarpy County. Figure 2.4 shows the counties mentioned in the population projections.

2.3 METEOROLOGY

Omaha is situated on the west bank of the Missouri River; the river level at Omaha is normally about 965 ft above sea level. The rolling hills in and around Omaha rise to about 1300 ft above sea level. The climate is typical continental, with relatively warm summers and cold, dry winters. It is situated midway between two distinctive climatic zones--the humid east and the dry west. Fluctuations between these two zones produce periods of weather condition that are either zone or combinations of both. Omaha is also affected by most storms or "lows" that cross the country. This causes periodic and rapid changes in weather, especially during the winter.

Most of the precipitation falls during sharp showers or thunderstorms, and these occur mostly during the growing season, April to September. Of the total precipitation, about 75% falls during the 6-month period April to September, predominantly as evening or night showers and thunderstorms. Although winters are relatively cold, precipitation is light, with only 10% of the total annual precipitation falling during the winter.

Sunshine is fairly abundant, ranging from around 50% of the possible in the winter to 75% or the possible in the summer.

The mean date of the last killing freeze in spring is April 14, and the mean date of the first killing freeze in autumn is October 20. The longest freeze-free period on record is 219 days in 1924, and the shortest period, 152 days in 1885. The average length of the freeze-free period is 188 days. Figures 2.5 & 2.6 are a summary of the climatological data for Omaha.

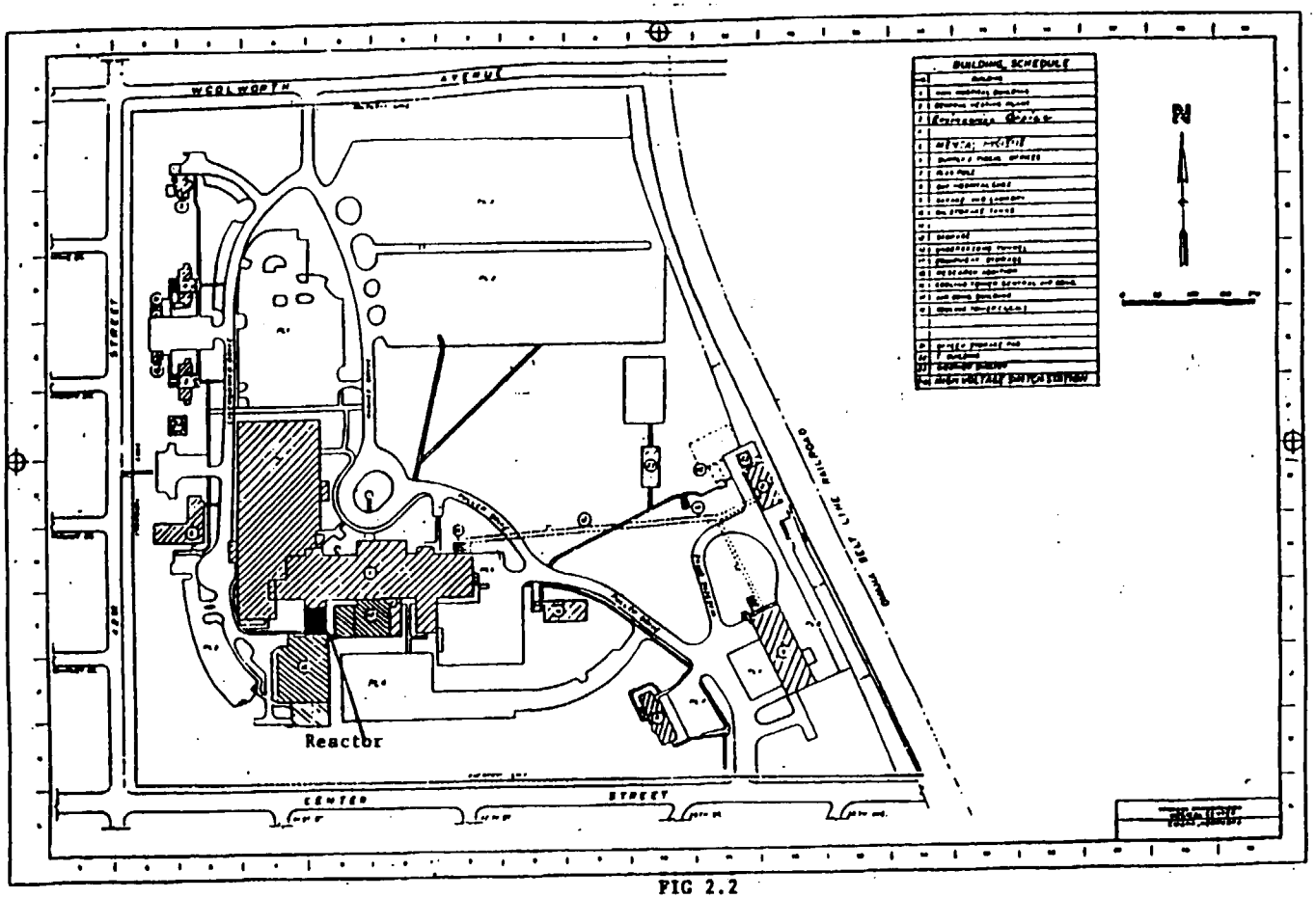


FIG 2.2

Trends

in the Omaha Economy 1984 - 1991

Greater Omaha Chamber of Commerce 1301 Harney St. Omaha, NE 68102 (402) 346-5000 FAX (402) 346-7050

The following report is a compilation of annual trends in the Omaha economy as reflected by the business indicator data collected each month by the Chamber research staff.

The bulk of the report concerns changes in employment in the Omaha metropolitan statistical area (MSA). Major sectors covered include construction and mining; finance, insurance and real estate (FIRE); government; manufacturing; trade; services; and transportation, communications and utilities (TCU).

Also briefly covered are population, age distribution, the unemployment rate, net taxable sales, airline passengers, the Omaha Help-Wanted Index, the change in the Consumer Price Index for All Urban Consumers (CPI), and building permits for the City of Omaha.

The annual averages for each category are in the appendix on the last page of this report.

Population

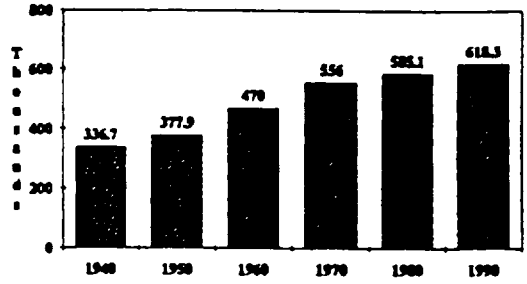
Omaha's economy can be characterized by slow steady growth over the past few decades. Between 1940 and 1990, the City of Omaha's population grew from 223,844 to 335,795, an increase of 50.0 percent. From 1940 through 1990, population of the Omaha MSA rose 83.6 percent, from 336,731 to 618,262. Since 1980, the metropolitan area's population has grown 5.7 percent.

The steady growth of the Omaha metropolitan area population is expected to continue. Preliminary population projections point to an estimated nine percent increase between 1990 and 2010 for the metro area. The largest percentage increase is expected in Sarpy County.

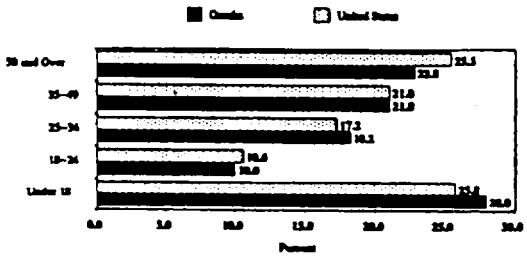
Omaha MSA Population Projections

County	1990 Census	Population Projections 1995	2000	2005	2010
Douglas	416,444	430,561	440,378	442,624	445,722
Sarpy	102,583	110,523	116,314	121,292	126,217
Washington	16,607	17,349	17,935	18,204	18,441
Pottawattamie	82,628	84,537	85,966	85,846	85,752
Omaha MSA	618,262	642,970	660,593	667,966	676,132

Population Growth
1940 - 1990



Age Distribution
1991



Non-Farm Wage and Salary Employment
Omaha MSA

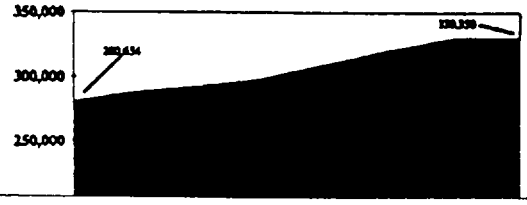


Fig 2-3

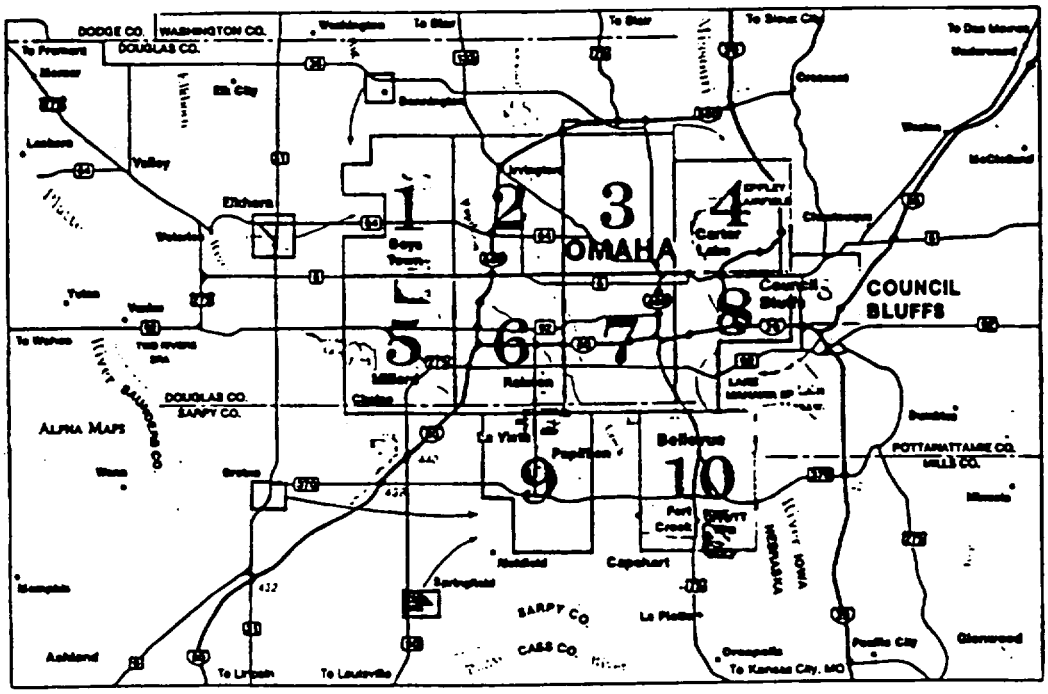

DODGE CO.
WASHINGTON CO.
DOUGLAS CO.
OMAHA
COUNCIL BLUFFS
1
2
3
4
5
6
7
8
9
10
DOUGLAS CO.
SARPY CO.
ALPHA MAPS
POTTAWATTAMIE CO.
MILLS CO.
SARPY CO.
CASS CO.
To Lincoln
To Louisville
To Kansas City, MO

Fig.2-4

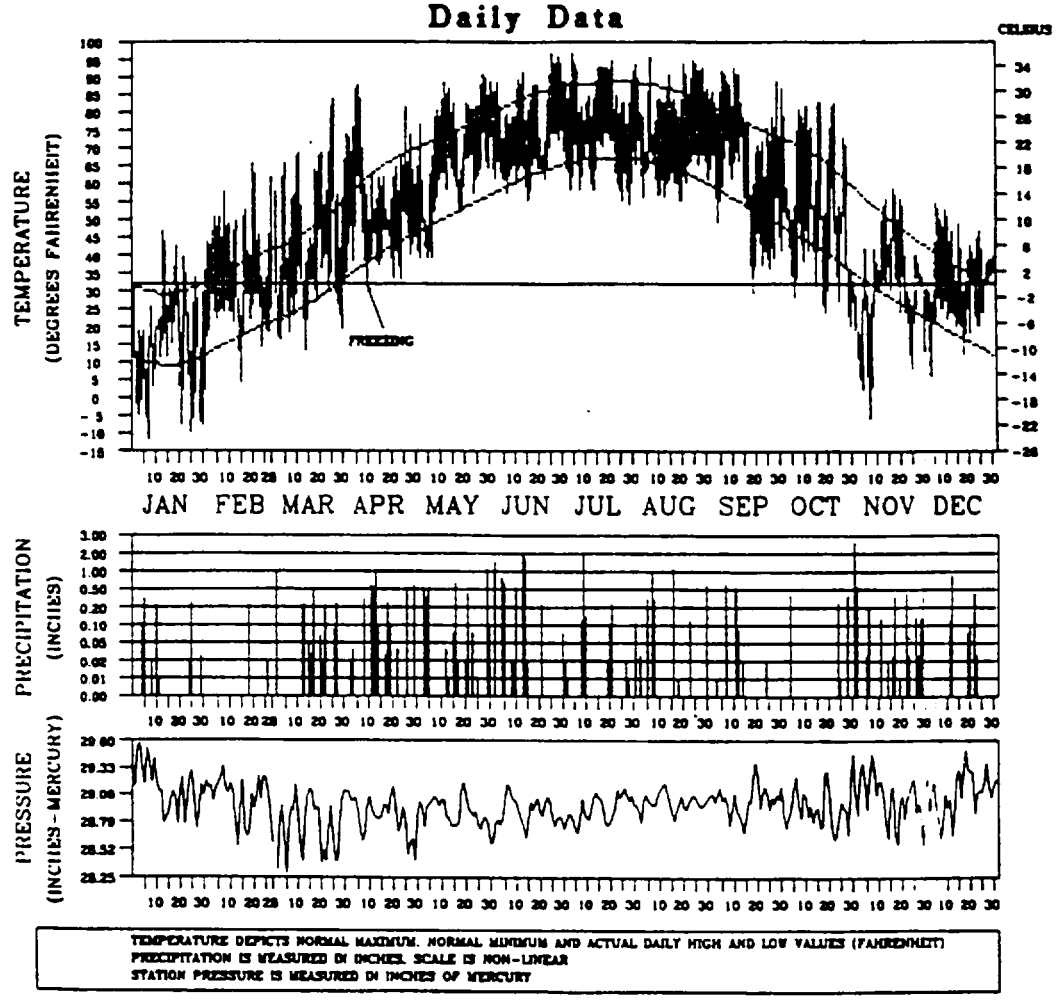
ISSN 0196-3172

991 LOCAL CLIMATOLOGICAL DATA

ANNUAL SUMMARY WITH COMPARATIVE DATA

OMAHA (EPPLEY AIRFIELD),

NEBRASKA



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DIRECTOR
NATIONAL CLIMATIC DATA CENTER

FIG 2.5

METEOROLOGICAL DATA FOR 1991

OMAHA (NORTH), NEBRASKA

LATITUDE: 41°22' N LONGITUDE: 96°01' W ELEVATION: FT. GRND 1309 BARO 1315 TIME ZONE: CENTRAL WBAN: 94918

	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEP	OCT	NOV	DEC	YEAR
TEMPERATURE °F:													
Averages													
-Daily Maximum	25.6	45.0	53.9	64.0	75.6	83.7	85.4	84.2	77.2	63.1	37.6	39.3	61.2
-Daily Minimum	8.5	25.6	30.8	44.3	58.1	65.9	66.2	65.0	55.3	40.7	23.4	25.2	42.4
-Monthly	17.1	35.3	42.4	54.2	66.9	74.8	75.8	74.6	66.3	51.9	30.5	32.3	51.8
-Monthly Dewpt.	8.6	24.1	29.8	41.2	55.9	63.6	63.1	63.0	51.4	36.7	23.9	24.5	40.5
Extremes													
-Highest	44	65	83	87	92	96	96	94	91	84	55	54	96
-Date	14	20	26	6	28	26	5	25	11	1	20	7	JUL 5
-Lowest	-8	2	15	33	38	58	57	58	31	18	-1	4	-8
-Date	7	15	2	10	6	7	29	19	19	30	7	4	JAN 7
DEGREE DAYS BASE 65 °F:													
Heating	1479	825	699	338	118	0	0	0	116	412	1027	1006	6020
Cooling	0	0	3	21	182	303	341	307	160	13	0	0	1330
% OF POSSIBLE SUNSHINE	48	62	54	53	50	70	78	76	76	65	37	59	62
AVG. SKY COVER (tenths)													
Sunrise - Sunset	6.0	5.4	6.5	7.3	7.4	6.5	4.6	4.7	5.3	5.5	7.2	4.9	5.9
Midnight - Midnight	5.6	4.9	5.8	6.7	7.4	6.2	4.5	4.3	4.4	4.7	6.4	4.5	5.5
NUMBER OF DAYS:													
Sunrise to Sunset													
-Clear	8	8	7	5	5	7	13	13	9	9	5	15	104
-Partly Cloudy	10	12	11	7	5	11	11	9	14	11	8	6	115
-Cloudy	13	8	13	18	21	12	7	9	7	11	17	10	146
Precipitation .01 inches or more	6	3	11	15	14	11	11	7	6	6	9	7	106
Snow, Ice pellets, hail 1.0 inches or more	5	0	1	0	0	0	0	0	0	1	3	0	10
Thunderstorms	0	0	6	7	9	9	7	6	3	3	2	0	52
Heavy Fog, visibility 1/4 mile or less	5	1	4	0	1	0	1	1	0	1	5	8	27
Temperature °F													
-Maximum													
90° and above	0	0	0	0	1	8	9	8	3	0	0	0	29
32° and below	23	4	2	0	0	0	0	0	0	1	12	8	50
-Minimum													
32° and below	31	24	18	0	0	0	0	0	1	9	23	27	133
0° and below	9	0	0	0	0	0	0	0	0	0	1	0	10
AVG. STATION PRESS. (mb)	972.9	969.9	961.4	964.4	965.8	966.8	967.8	969.9	970.5	967.8	969.9	972.2	968.2
RELATIVE HUMIDITY (%)													
Hour 00	71	70	73	72	78	75	70	76	69	66	81	82	74
Hour 06 (Local Time)	73	79	77	83	87	83	80	84	78	71	83	82	80
Hour 12	64	58	58	58	62	63	60	61	55	53	76	67	61
Hour 18	70	59	54	58	59	59	56	59	50	53	76	76	61
PRECIPITATION (inches):													
Water Equivalent													
-Total	0.92	0.46	4.24	5.74	3.42	6.52	3.49	3.87	2.06	3.02	1.67	1.87	37.28
-Greatest (24 hrs)	0.44	0.44	1.33	1.73	1.14	2.74	2.75	2.03	1.02	1.39	1.43	1.13	2.75
-Date	4-5	18	1	26	16-17	13-14	8-9	16	11	31	PM-1	11-12	JUL 8-9
Snow, Ice pellets, hail													
-Total	13.7	0.4	5.1	T	0.0	0.0	0.0	0.0	0.0	5.7	7.6	T	32.5
-Greatest (24 hrs)	5.4	0.4	3.6	T	0.0	0.0	0.0	0.0	0.0	5.7	5.9	T	5.9
-Date	24-25	26	12-13	29						31	PM-1	28	NOV PM-1
WIND:													
Resultant													
-Direction (!!!)	286	282	296	145	137	154	152	137	187	334	312	237	192
-Speed (mph)	2.0	1.8	1.8	1.8	2.7	5.3	1.4	2.2	2.2	1.5	2.6	1.4	0.8
Average Speed (mph)	8.9	9.1	10.5	9.4	8.4	8.6	7.9	7.1	8.9	9.8	10.1	8.4	8.9
Fastest Mile													
-Direction (!!!)	NW	NW	NW	SE	NW	SE	NW	W	NW	NW	NW	NW	NW
-Speed (mph)	30	34	35	26	21	23	17	27	26	24	26	28	35
-Date	20	14	27	6	5	14	28	7	25	17	23	13	MAR 27
Peak Gust													
-Direction (!!!)	NW	NW	NW	SE	NW	SE	NW	W	NW	NW	NW	NW	NW
-Speed (mph)	39	46	47	40	36	39	30	41	39	36	48	40	48
-Date	20	13	27	11	5	14	13	7	25	4	30	14	NOV 30

!!!! See Reference Notes on Page 68

Fig. 2-6

2.4 GEOLOGY

The area lies within the Dissected Till Plains of the Central Lowland Physiographic Province of the United States. The topography is gently rolling, and the ground surface at the medical center varies in elevation between 1200 and 1230 ft above MSL. These elevations represent some of the highest ground within the Omaha city limits, being approximately 275 ft above the level of the Missouri River. Figure 2.7 is a U.S. Geological Survey map of the Omaha area surrounding the medical center site.

The surface soils in the Omaha area are primarily loess and glacial drift deposits. Two stages of glaciation, the Nebraskan and the Kansan, left thick deposits of till overlying bedrock. It is believed that much of the glacial till has been eroded in the vicinity of the medical center and that not more than 100 ft remains. The till consists mainly of lean and gravelly clays with a few lenses of sand-gravel. The exact depth to bedrock directly below the medical center site is not known but is estimated to vary between MSL elevation 1000 and 1050, on the basis of the nearest top bedrock information.

The loess at the site is of Peorian and Loveland formations of the late Pleistocene period. The soil classification of the Peorian indicates that the material consists predominantly of clayey silts and lean clay. The soil of the Loveland formation varies from clayey silt to fat clay with minor amounts of sand and clayey sand in the basal part of the formation. At the Medical center site, the Peorian is from 30 to 45 ft thick and the Loveland is over 60 ft thick. This would mean that the total thickness of the overburden is approximately 200 ft. Bedrock in this area is limestone and shale of the Pennsylvania period. The surface of the bedrock is very irregular because of an extensive period of erosion that followed the uplift of the area in early Pennsylvania time and continued to the Pleistocene period. This uplift brought the granite to within 600 ft of the surface in certain areas, forming a ridge known as the Nemaha Ridge or Arch. Also, extensive faulting occurred that developed a major fault, known as the Humboldt fault, which has a throw of over 900 ft. There is no evidence of activity along this fault in recorded time.

No piezometers were installed or observation wells drilled at the site, so there is no definite information as to the exact depth of the water table. However, on the basis of logs of the borings drilled in 1946, the zone of saturation appears to be below 65 ft, although there is some indication of perched water levels in the soil strata as high as 15 ft.

The following report from the Omaha Testing Laboratories gives the results of a 30-ft test boring made at the center of the reactor location:

Field Work:

One auger boring was made on January 8 and 9, 1959 at the following location: Between Columns L and M, 10' from L and Between Columns 8 and 9, 10' from 9.

Soil samples were taken continuously for obtaining a Soil Log and in addition undisturbed samples were taken at 5' intervals with 2" O.D. Shelby Tubes for bearing index tests.

Laboratory Work:

All soils were classified as to texture, origin of deposit, and consistency [sic]. These classifications are shown in Table 2-1 attached hereto. The undisturbed samples were tested for moisture content, dry unit weight, and unconfined compressive strength. The shear strength was calculated as one-half the unconfined compressive strength. The test results are given in Table 2-2 attached hereto.

General Comments:

The soil boring shows the top soil layers for a depth of 26' to be Peorian Loess, a wind-blown deposit of clayey silt having low plasticity. The next 3' consists of Loveland Loess, a windblown deposit of silty clay having medium plasticity. The bottom 1' of the boring was in a glacial clay having a higher plasticity.

The test data show the soils to be strong near the surface then decreasing to medium strength at a depth just above the clay layer which is a very strong layer. No water table was encountered.

Summary:

Based on the test data, the soil should support itself in vertical walls of an 8' diameter excavation 20' deep even with the surcharge of 3000 psf from the building foundations on the adjacent soil.

The allowable bearing value at the 21' depth is calculated to be 4300 psf for a circular footing.

TABLE 2-1

LOG OF BORING

Depth	Soil Type	Origin	Consistency
0-5-3/4"	Concrete Floor Slab		
5-3/4"-8"	Sand Cushion		
8"-3'	Brown Clayey Silt	Peorian Loess	Stiff (-)
3'-13'	Light Yellow Clayey Silt	Peorian Loess	Stiff (-)
13'-21'	Light Grey Clayey Silt	Peorian Loess	Medium Stiff (+)
21'-26'	Tan Clayey Silt	Peorian Loess	Medium Stiff
26'-29'	Reddish Silty Clay	Loveland Loess	Stiff (-)
29'-30'	Reddish Clay	Glacial	Stiff

TABLE 2-2

LABORATORY TESTS

Sample (ft)	Depth Moisture Content (%)	Dry Unit Weight (pcf)	Unconfined Compressive Strength (psf)	Strength (psf)
3-3-1/2	20.1	*	*	*
8-8-1/2	19.3	104.9	9910	4955
13-13-1/2	21.8	101.0	6780	3390
18-18-1/2	20.8	100.6	6780	3390
23-23-1/2	23.6	98.1	3490	1745
28-28-1/2	22.6	99.6	3490	1745
29-1/2-30	22.0	101.6	11900	5950

2.4.1 Groundwater Hydrology

Groundwater generally moves in horizontal and lateral directions. In determining the subsurface movement of water, the actual trails are assumed to travel smooth pathways known as streamlines. Thus, the water molecules are taken to travel directly through matter. The laminar flow rate v of underground water is given by Darcy's equation [1&2].

$$v = K \times \frac{(H_1 - H_2)}{L}$$

where:

v	= Laminar flow rate
K	= Hydraulic conductivity
H1	= Total head measured by a piezometer at site 1
H2	= Total head measured by a piezometer at site 2
L	= Distance between the ends of the piezometers

The hydraulic conductivity of soil can be affected by temperature, ionic composition of the water, and the presence of entrapped air. The density and viscosity of water changes with temperature. K values are normally expressed at 20°C. The ionic composition of water can change the K value via ion exchange when exposed to porous material containing clay. In addition, the pores of these clays can be so small as to produce size exclusion for some of the larger ions. Entrapped air in the soil generally causes the K value to be less. Air can become trapped within the soil by a rise in the water table. It may also occur when colder outside water enters an aquifer. For the purposes of our calculations the hydraulic gradient of flow will be assumed to be unity. Therefore,

$$v = K$$

From the site of the reactor the ground water will flow to the south-west. Traveling downward by gravity through the relatively impermeable loess until it reaches the level of impermeable glacial till. As seen in the area map (Fig. 2-1), the Big Papillion Creek runs in an south-easterly direction approximately two miles west and four miles south-west of the site. With the water table troughing along this creek the underground water would migrate along the creek until it returns to the Missouri river south of Offut Airforce Base. Once within the Missouri river the water would be readily available to members of the public for ingestion. The closest wells in the immediate area are located 1.6 miles due west (on the Aksarben grounds [see Fig. 2-1), and in an area near 84th and L streets. Regarding the Aksarben site; the well would not be subject to contamination since the water flow occurs south-west. The site at 84th and L streets is further than the creek used in the below calculation. Thus, the path below is the quickest route. Glacial Till has a hydraulic conductivity in the range of 1×10^{-12} to 2×10^{-6} meters per second. The time for radioactive isotopes to be carried from the site to the Big Papillion Creek is as follows:

$6435\text{m} \div 2 \times 10^{-6}$ m/s = 102.03 years.

For comparison, the slowest rate is

$6435m \div 1x10^{-12}$ m/s = 204,052,511 years.

Thus, any radioactive isotopes produced via soil activation, discussed in SAR, Appendix D, would decay out well before it was made available to the public for consumption. In addition, there is no surface water present which could have an impact on reactor operation.

2.5 EARTHQUAKES

The site is located in Seismic Risk Zone 1 of the United States (Fig. 2.8) which is defined on the Modified Mercalli (MM) intensity scale as "Minor damage, distant earthquakes may cause damage to structures with fundamental periods greater than 1.0 seconds; corresponding to intensities V and VI of the MM scale"

These earthquake intensities are defined in qualitative terms by the MM intensity scale (Fig 2.9). The site is subject to an earthquake risk estimate characteristic of half the area of the 48 contiguous states. There is a risk of slight damage, principally to poorly built or designed structures.

There has been no reports or physical evidence of earthquakes at the site. The five largest earthquakes which have effected the region, as determined from searches of local historical newspaper accounts and other published materials, are indicated in Table 2-3. A complete listing of earthquake occurrences is given in Fig. 2.10. Based upon analysis of newspaper accounts since 1967, the seismic events experienced by the region have led to the following:

-no significant building damage
-no loss or threat of loss of life, and
-no livestock or crops affected.

Low level as well as low frequency of earthquakes characterizes this technically stable region. The lack of severity of the experienced events has led to their being treated as infrequent curiosities in the regional history.

A plot of epicenters of all earthquake occurrences is superimposed on the regional seismotectonic pattern in Fig. 2.11 [3], (The figure is from Fort Calhoun Station Environmental Report and consequently the site designation is that of Ft. Calhoun. Our site would be Omaha). The long time history of no significant damage over so broad a region of essentially similar geologic conditions further supports the contention that seismic or fault induced hazards are minimal.

Despite the extremely unlike nature of an earthquake sufficient to affect the building, it is possible that an earthquake could result in creating a breach in the reactor tank. This scenario has already been addressed in Appendix C in terms of a loss of coolant accident. In the case that the earthquake is sufficient to create significant damage to the building structure, concrete could fall in on the reactor core and cause significant fuel damage. However, given the basement location, this would also mitigate the hazard as the rubble would also provide significant shielding and limit the airborne transport of radionuclides.

2.6 TORNADOES

The solution development follows that presented in NUREG CR-4461 [4], assuming a log-normal distribution of events. Since damage area estimates are not available for the Nebraska data; the WASH-1300[6] average area of destruction of 2.82 mi^2 was used. It is assumed that the area of interest is small compared to the region over which the data was collected. This should be a valid assumption considering the data was for the entire state.

The probability that a tornado produces winds able to damage a nuclear power plant has been estimated as 10^{-7} y^{-1}, as found from WASH-1300/Reg. Guide 1.76. In general, research reactors are not constructed to power reactor standards. However, in the case of the Omaha reactor, the reactor is located below ground with substantial concrete directly above, and should offer a similar amount of protection. Despite this, a factor of 10 will be applied, increasing the value to 10^{-6} y^{-1}. This should yield conservatively larger probabilities.

The probability of a tornado striking a point or small area is given by:

$$P_s = \frac{A_t}{A_r N_y}$$

where P_s is the strike probability, A_t is the total area affected by the tornadoes, A_r is the area of the region and N_y is the number of years in which the tornado data was collected. As mentioned earlier, the total area affected by tornadoes is estimated as $A_t = 2.82\ n_e$, where n_e is the number of tornado events. Based on the data supplied in Table 2.4, the total number of tornadoes in the state of Nebraska was 1471 for the period 1950-1991. The area of the state of Nebraska is 76,639 mi^2. This gives a strike probability of $P_s = 1.3E10^{-3}$ y^{-1}.

The probability that such a tornado exceeds the design speed is the product of the limited intensity probability P_i and the strike probability P_s. The critical value for the intensity P_{ic} is then $10^{-6} \div P_s = 7.6E10^{-3}$.

Tornadoes have occurred in the general area of the site and may be expected each year. From 1953 through 1962 a total of 20 tornadoes occurred in a 1 degree square centered near the site. Since 1962 there have been 10 tornadoes in Douglas County. In one case a tornado that hit Omaha in May 6, 1975, was considered to be the most destructive one ever to hit a major American City. The one-quarter-mile wide and 8.7-mile-long path taken by the tornado resulted in three deaths and caused property damage estimated at $150 million to $500 million. The annual average number of tornadoes for Nebraska during the 41-year period from 1950 to 1991 was 35. Table 2-4 [5] presents the tornado statistics for Nebraska from 1950 through 1991.

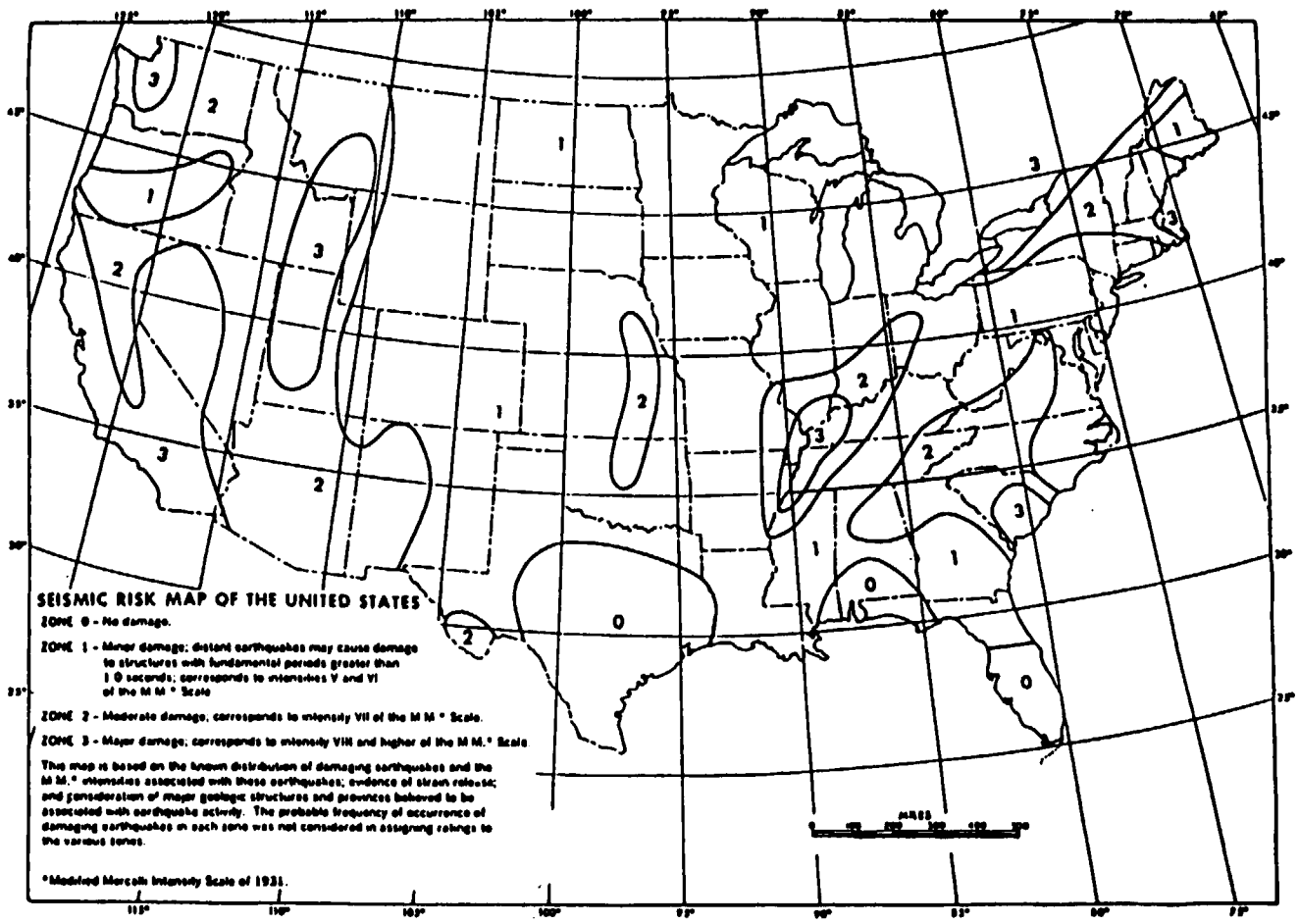


FIG 2.8 Information from OEP Report to the Congress, 1972

MODIFIED MERCALLI INTENSITY SCALE

ABRIDGED
MODIFIED MERCALLI INTENSITY SCALE

I. Not felt except by a very few under especially favourable circumstances.

II. Felt only be a few persons at rest, especially in upper floors of buildings. Delicately suspended objects may swing.

III. Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing of truck. Duration estimated.

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make creaking sound. Sensation like heavy truck stricking building. Stnading motor cars rocked noticeably.

V. Felt by nearly everyone; many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbance of trees, poles and other tall objects sometimes noticed. Pendulum clocks may stop.

VI. Felt by all; many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight.

VII. Everybody runs outdoors. Damage nigligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motor cars.

VIII. Damage slight in specially designed structures; considerable in ordinary substantial buildings with partial collapse; great in poorly built structures. Panel walls thrown out of frame structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Persons driving motor cars disturbed.

IX. Damage considerable in specially designed structures; well designed frame structures thrown out of plumb; great in substantial buildings, with partial collapse. Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken.

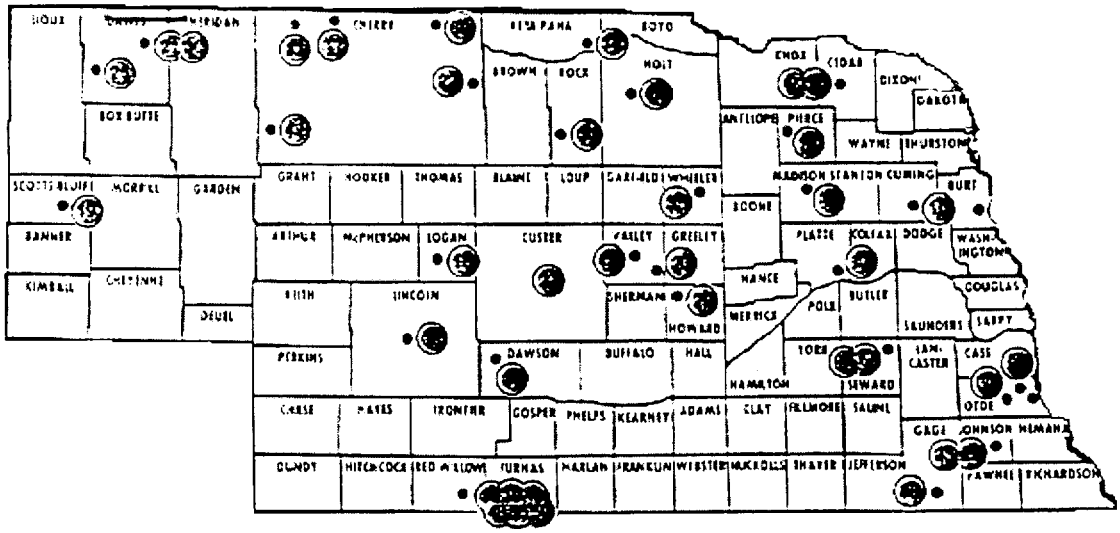
X. Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations, ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks.

FIG 2.9 Modified Mercalli Intensity Scale

Table 2-3 Five Largest Earthquake Events in the Region

Date	Nearest Town	Coordinates	Epicentral Intensities[1] (MM)
24 April 1867	Manhattan, Kansas	39.5N 96.7W	VII
15 November 1877	Garland, Nebraska	41.0N 97.0W	VII
28 July 1902	Battle Creek, Nebraska	42.0N 97.6W	V
7 January 1906	Manhattan, Kansas	39.3N 96.6W	VII
1 March 1935	Tecumseh, Nebraska	40.35N 96.15W	VI

[1]Final evaluation of epicentral intensities from newspaper accounts.



Location	Date	Time	Intensity
1. Nebraska City	April 28, 1867	na	IV
2. Maskell	Oct. 9, 1872	4 p.m.	V
3. Nebraska City	Dec. 9, 1875	9 a.m.	III
4. Garland	Nov. 15, 1877	5:45 p.m.	VII
5. Garland	Nov. 15, 1877	6:30 p.m.	na
6. North Platte	March 17, 1884	8 p.m.	IV
7. Hartington	Feb. 4, 1896	11:45 a.m.	III
8. Hartington	Sept. 16, 1898	9:59 a.m.	IV
9. Battle Creek	July 28, 1902	6 p.m.	VI
10. West Point	Dec. 1, 1904	9 a.m.	III
11. Neb.-S.D. border	May 10, 1906	12:27 a.m.	IV
12. Plainview	Jan. 26, 1909	8:15 p.m.	IV
13. Columbus	Feb. 26, 1910	8 a.m.	V
14. Kirkwood	Sept. 16, 1915	7 p.m.	IV
15. Stapleton	December 1916	na	III
16. Tekamah	Sept. 10, 1923	6:30 a.m.	IV
17. Gothenburg	Sept. 24, 1924	11 a.m.	IV
18. Ord	Oct. 14, 1927	4:10 p.m.	IV
19. Scottsbluff	Aug. 8, 1933	na	V
20. North Loup	May 11, 1934	10:40 a.m.	IV
21. Chadron	July 30, 1934	7:20 a.m.	VI
22. Wood Lake	Nov. 8, 1934	4:45 a.m.	III
23. Tecumseh	March 1, 1935	11 a.m.	VII
25. Tecumseh	March 22, 1935	10:45 p.m.	IV
26. Fort Robinson	March 24, 1938	1:11 p.m.	IV
27. Broken Bow	April 7, 1948	na	III
28. Atkinson	May 13, 1948	4:15 a.m.	IV
29. Cotesfield	Feb. 25, 1955	1:45 a.m.	IV
30. Chadron	March 9, 1963	3:25 p.m.	III
31. Syracuse	June 6, 1963	2:47 a.m.	III
32. Merriman	March 28, 1964	10:08 a.m.	VII
34. Bassett	Oct. 16, 1972	5:47 a.m.	V
35. Bartlett	May 13, 1975	7:53 a.m.	VI
38. SE Red Willow	Dec. 1, 1977	1:04 p.m.	III
39. SE Red Willow	Dec. 1, 1977	1:22 p.m.	III
42. SW Cherry	May 7, 1978	4:06 p.m.	V
46. SE Red Willow	June 6, 1979	4:16 p.m.	III
48. SE Red Willow	July 16, 1979	12:03 a.m.	III
55. SE Red Willow	Aug. 2, 1979	4:16 a.m.	III
59. SE Red Willow	Aug. 31, 1979	8 a.m.	IV
69. NE Cherry	Sept. 7, 1981	12:38 a.m.	IV
88. Wymore	June 3, 1982	2:20 p.m.	II
94. Dunbar	June 26, 1983	12:42 p.m.	IV

FIG 2.10 World-Herald

Location of 43 quakes felt by humans . . . Intensity is measured on Modified Mercalli scale, which ranges from I to XII (as opposed to the Richter Scale, which runs from 1 to 8). A quake measuring I is felt only by a few people under especially favorable conditions. A quake measuring VI causes slight damage. A quake measuring XII causes total damage. Numbers between I and 100 not on the map apply to quakes detected only by monitors, not by humans.

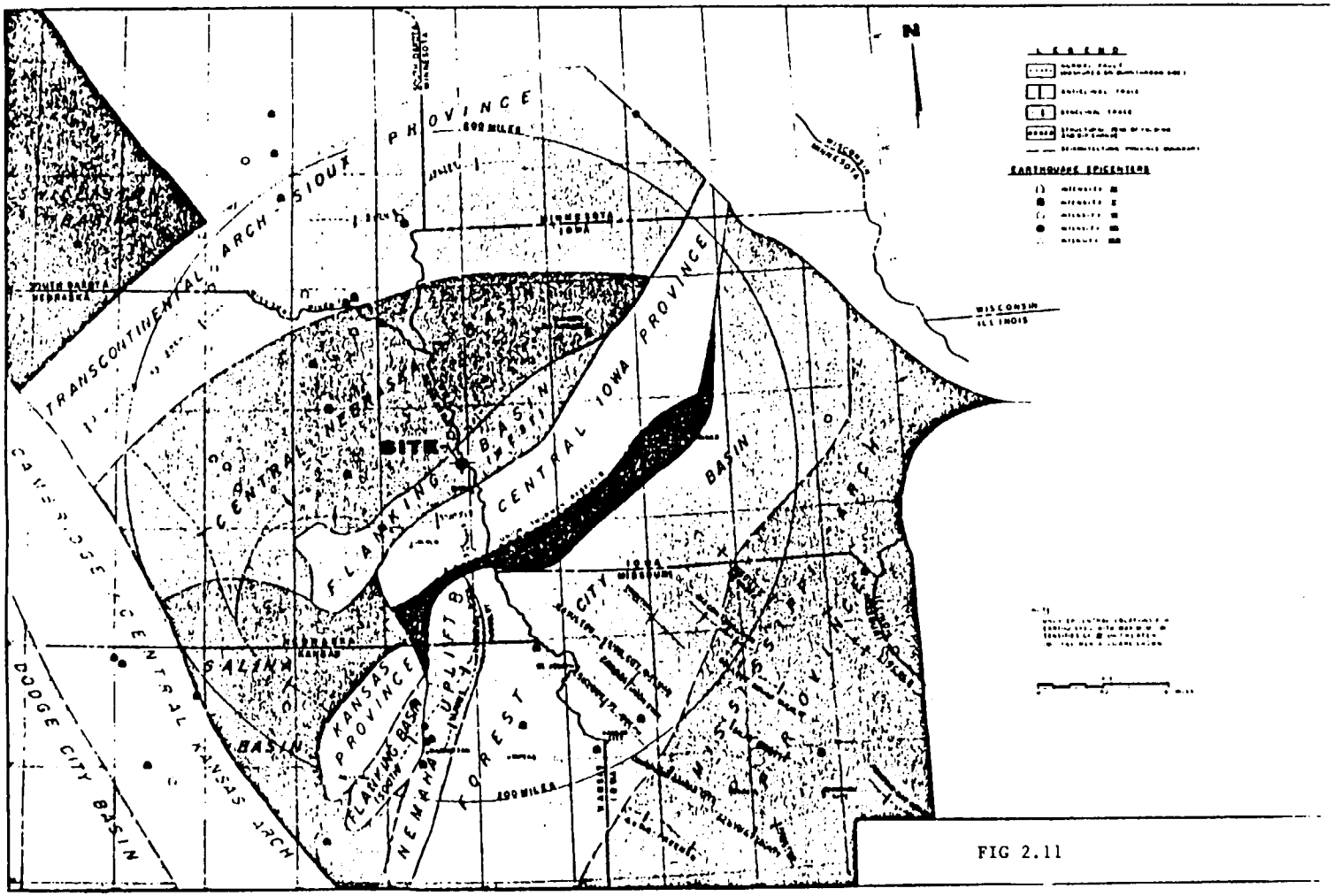

N
SITE
TRANSCONTINENTAL ARCH
SIOUX
PROVINCE
CENTRAL NEBRASKA BASIN
CENTRAL IOWA BASIN
FLANKING
CENTRAL IOWA PROVINCE
CAMBRIDGE
DODGE CITY BASIN
SALINA
BASIN
KANSAS PROVINCE
NEMAHA
UPLIFT
FOREST
CITY
STRUCTURE
100 MILES

FIG 2.11

The AJBRF is in the basement of the Medical Center surrounded by poured concrete walls with no windows and with 3-4" of concrete overhead makes tornado damage improbable. In fact, the area in and around the reactor room is also designated as the Medical Centers tornado shelter.

2.7 ATMOSPHERIC STABILITY

Wind direction and speed data are presented in Tables 2-5 and 2-6 [7].

Table 2.4

NEBRASKA TORNADO FREQUENCY BY MONTH

NEBRASKA TORNADO FREQUENCY BY MONTH AND YEAR SINCE 1950

	JAN	FEB	MAR	APR	MAY	JUN	JUL	AUG	SEP	OCT	NOV	DEC	TOTAL
1950	0	0	0	0	2	2	1	0	1	0	0	0	6
1951	0	0	0	1	3	2	2	1	0	0	0	0	9
1952	0	0	0	1	3	2	1	2	0	0	0	0	9
1953	0	0	0	2	9	24	7	2	0	0	0	0	44
1954	0	0	0	1	2	8	3	2	0	0	0	0	16
1955	0	0	0	1	4	9	10	3	4	0	0	0	31
1956	0	0	0	2	6	10	8	2	1	4	1	0	34
1957	0	0	0	9	16	15	11	2	0	0	0	0	53
1958	0	0	0	2	4	12	25	10	1	0	0	0	54
1959	0	0	0	0	26	9	3	5	1	0	0	0	44
1960	0	0	0	1	10	24	1	7	0	0	0	0	43
1961	0	0	0	0	6	3	2	0	0	1	0	0	12
1962	0	0	0	1	28	9	1	2	0	0	0	0	41
1963	0	0	0	5	3	7	1	1	0	0	0	0	17
1964	0	0	0	8	13	13	5	2	0	0	0	0	41
1965	0	0	0	0	32	8	6	0	0	0	0	0	46
1966	0	0	0	0	5	4	1	0	0	0	0	0	10
1967	0	0	1	0	1	35	0	1	1	0	0	0	39
1968	0	0	0	2	2	9	7	1	0	0	0	0	21
1969	0	0	0	0	3	8	7	2	0	0	0	0	20
1970	0	0	0	0	1	8	2	2	1	0	0	0	14
1971	0	0	0	1	18	22	8	0	0	3	0	0	52
1972	0	0	0	0	15	4	10	1	0	0	0	0	30
1973	0	0	0	3	4	2	5	0	3	2	0	0	19
1974	0	0	0	14	7	1	2	6	0	2	0	0	32
1975	0	0	2	10	22	35	5	2	1	0	0	2	79
1976	0	0	1	5	3	5	10	1	1	0	0	0	26
1977	0	0	0	4	35	8	8	8	5	0	0	0	68
1978	0	0	0	8	16	7	7	2	2	0	0	0	42
1979	0	0	0	3	0	8	6	2	0	1	0	0	20
1980	0	0	0	0	10	16	1	3	0	8	0	0	38
1981	0	0	0	3	0	5	6	5	0	0	0	0	19
1982	0	0	0	1	15	13	1	2	0	1	0	0	33
1983	0	0	0	0	2	12	1	0	0	0	0	0	15
1984	0	0	0	4	6	36	4	0	0	0	0	0	50
1985	0	0	4	15	14	4	2	6	7	0	0	0	52
1986	0	0	0	17	6	10	12	3	6	0	0	0	54
1987	0	0	3	0	6	8	5	4	0	0	0	0	26
1988	0	0	0	1	7	5	4	3	0	0	0	0	20
1989	0	0	0	8	4	21	4	4	0	0	0	0	41
1990	0	0	15	1	23	39	6	4	0	0	0	0	88
1991	0	0	0	9	29	23	1	0	1	0	0	0	63
Total	0	0	26	143	421	504	212	103	36	22	1	2	1471
Mean	0	0	0.6	3.4	10.0	12.0	5.0	2.5	0.9	0.5	0.0	0.0	35.0
Most	0	0	15	17	35	39	25	10	7	8	1	2	88
Year			1990	1986	1977	1990	1958	1958	1985	1980	1956	1975	1990

TABLE 2.5

CLIMATOLOGICAL NORMALS

COMPARISON OF NORTH OMAHA NWS WITH FORT CALHOUN STATION

	WIND DIRECTION (PERCENT)		WIND SPEED (MPH)	
Wind Direction	North Omaha NWS (1965-1969)	Fort Calhoun (1965-1969)	North Omaha NWS (1965-1969)	Fort Calhoun (1965-1969)
NNE	3.8	2.6	8.2	4.9
NE	3.3	2.4	6.9	4.8
ENE	3.2	2.4	6.5	4.5
E	3.1	3.1	6.7	5.0
ESE	5.4	5.7	6.8	5.4
SE	7.1	9.0	7.8	6.8
SSE	10.0	10.2	9.9	8.9
S	10.8	10.1	10.4	9.5
SSW	9.1	7.2	9.5	9.3
SW	4.3	3.7	8.5	7.5
WSW	2.4	3.0	7.2	5.5
W	3.9	4.5	7.7	4.4
WNW	5.3	7.6	10.1	4.6
NW	8.6	10.7	12.9	6.6
NNW	6.8	9.4	12.8	6.5
N	13.0	5.5	8.0	5.9
Missing	---	2.9	---	2.9
Average	---	---	8.7	6.3

NOTE: The wind speeds at the North Omaha NWS were recorded 20 feet above ground level, and the wind speeds at Fort Calhoun Station were recorded at 10 meters, above ground level. Data obtained from the Local Climatological Data; see References 4 and 5.

TABLE 2.6

MAXIMUM RECORDED AND MEAN WIND SPEEDS (MPH)

	EPPLEY AIRFIELD				NORTH OMAHA NWS			
Period	Fastest Wind Speed (1949-1990)	Direction (Degrees)	Year	Mean (1936-1990)	Fastest Wind Speed (1979-1990)	Direction (Degrees)	Year	Mean (1985-1990)
January	57	NW	1938	10.9	41	NW	1978	10.4
February	57	NW	1947	11.1	38	NW	1978	9.6
March	73	NW	1950	12.3	38	NW	1982	10.9
April	65	NW	1937	12.7	46	NW	1982	10.6
May	73	NW	1936	10.9	34	N	1983	8.9
June	72	N	1942	10.1	34	NW	1983	8.4
July	109	N	1936	8.9	46	NW	1980	7.5
August	66	N	1944	8.9	39	NW	1980	7.7
September	47	E	1948	9.5	35	NW	1980	8.4
October	62	NW	1966	9.8	34	NW	1979	8.9
November	56	NW	1951	10.9	38	NW	1982	9.9
December	52	NW	1938	10.7	37	NW	1981	9.9
Year	109	N	1936	10.6	46	NW	1982	9.3

NOTE: The wind speeds at Eppley Airfield were recorded at 70 feet above ground level (agl) until 1974; 20 feet agl since that time. The wind speeds at the North Omaha NWS were recorded at 20 feet agl. Data obtained from the Local Climatological Data; see References 4 and 5.

Chapter 2

References

1. Bouwer, "Groundwater Hydrology", Mcgraw-Hill. 1978.

2. Todd, "Groundwater Hydrology", John Wiley and sons. 1980.

3. Nuttie, O.W., "State-of-the Art for Assessing Earthquake Hazards in the U.S.", Misc. Paper 5-73-1, U.S. Army Engineer Waterways Experimental Station. 1973.

4. NUREG CR-4461

5. Regulatory Guide 1.76

6. WASH-1300

7. National Weather Service, Omaha, NE

CHAPTER 3

FACILITY DESIGN
STRUCTURES, SYSTEM AND COMPONENTS

3.1 REACTOR LABORATORY

The AJBRF will be located in the basement of the 11-story Medical Center building (Fig. 3.1), which was erected in 1951. The Medical Center is of brick and reinforced concrete construction, including floors and ceiling, except that the walls between the rooms in the reactor area are of wood-stud plaster construction. Entrance to the reactor laboratory is normally be through the door marked SW-2 (Fig. 3.2). The area to the left of the access door serves as a health physics control point where pocket dosimeters are issued as required by appropriate regulations and procedures. A log is also kept of all persons entering the area together with the exposure they received while in the area.

Samples to be irradiated are typically prepared in either room SW-2C or SW-2E. Isotopes are stored in the isotope-storage area SW-2F. The pneumatic transfer system is located at position PT as indicated in Fig. 3.2. Gamma counting is done in the area marked shield as shown in Fig 3.2.

The reactor room ventilation supply provides heated or cooled 100% outside air to the reactor laboratory at the rate of 1,520 CFM through six ceiling outlet ducts. The exhaust effluent of 2,970 CFM exits the reactor room into the outside air by means of an exhaust fan installed in the outside wall of the building. In addition, two laboratory fume hoods (Fig. 3.2) exhaust a total of 919 CFM by means of fans installed on the roof of the medical center. Thus, the reactor area is kept at a slight negative pressure. The reactor area exhaust fan is operated continuously and has a starter switch mounted on the reactor console so that it can be manually started or stopped. The fan is equipped with an automatic damper on the exhaust side, so that when the fan is off the exhaust portal will be closed. In addition, when the fan is stopped a duct pressure control closes an absolute damper in the air supply duct and simultaneously causes an alarm to be initiated on the medical center Honeywell Delta-2000 control system which is continually manned. Thus a single switch on the reactor console can stop air from entering or leaving the reactor laboratory and if the exhaust fan stops, the medical center ventilation engineers are immediately notified by the Honeywell computer. The two fume hoods as shown in Fig. 3-2 are operated continuously and are exhausted by means of independent exhaust motors on the roof of the medical center as shown in the attached Fig. 3.2a. The output of the pneumatic tube is piped from the blower in the reactor water treatment pit outside and adjacent to the basement reactor room to the duct of the fume hood shown in the room labeled Radioisotope Reactor Research Lab (Fig 3.2). This hood has a flow switch alarm that emits an audible signal if the hood blower stops.

The principle use of the AJBRF reactor is for neutron activation analysis of biological samples and since they must be counted in a well-Ge detector shortly after they are activated, the radioactivity of the samples must be low. Typical irradiation times are 3 minutes in the pneumatic transfer tube. Samples are irradiated in 5 mL vials and the vial is opened in the fume

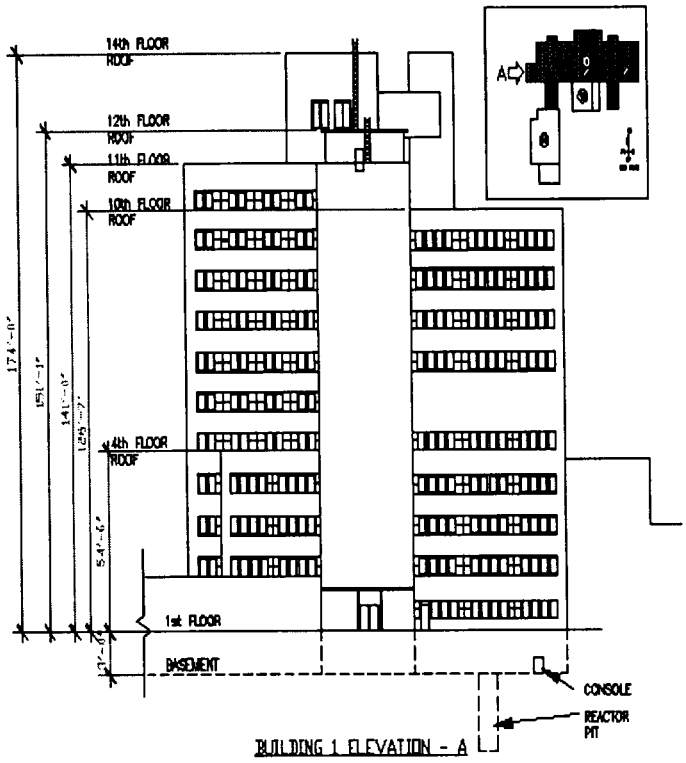

14th FLOOR
ROOF
12th FLOOR
ROOF
11th FLOOR
ROOF
10th FLOOR
ROOF
4th FLOOR
ROOF
1st FLOOR
BASEMENT
CONSOLE
REACTOR
PIT
A
BUILDING 1 ELEVATION - A

Fig 3-1

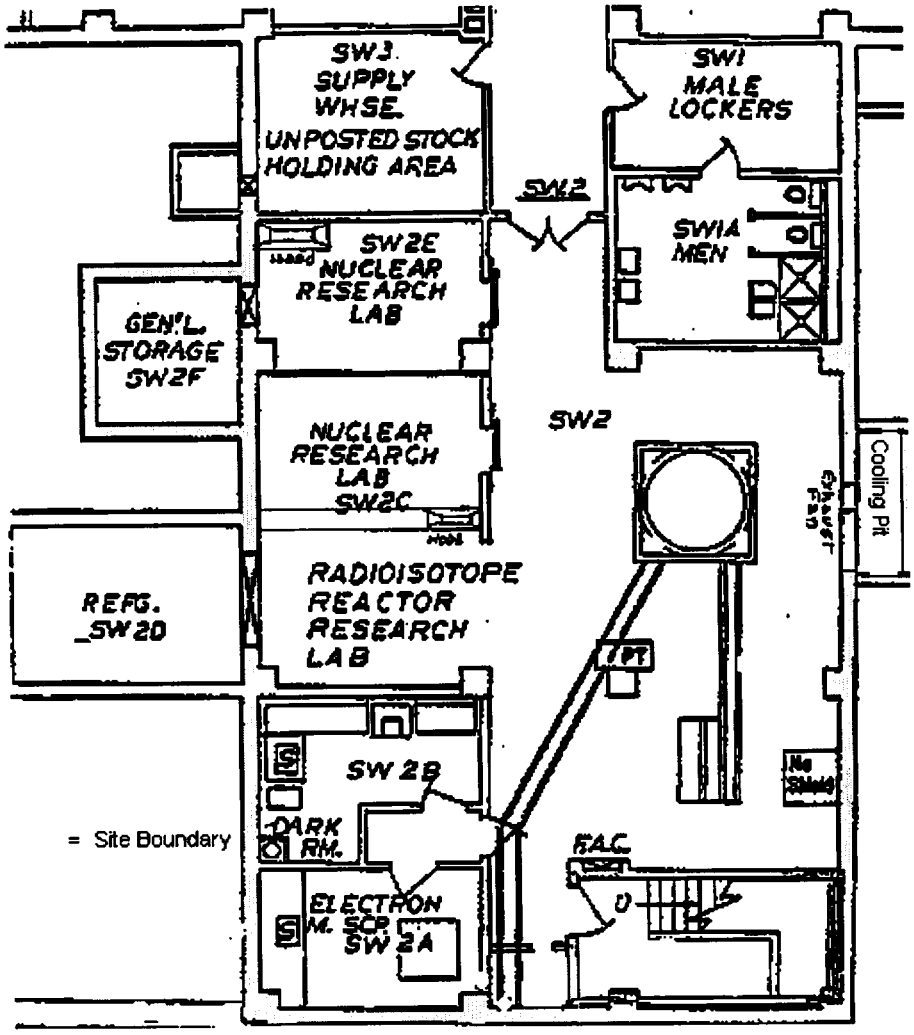


Fig 3-2 REACTOR LABORATORY BASEMENT

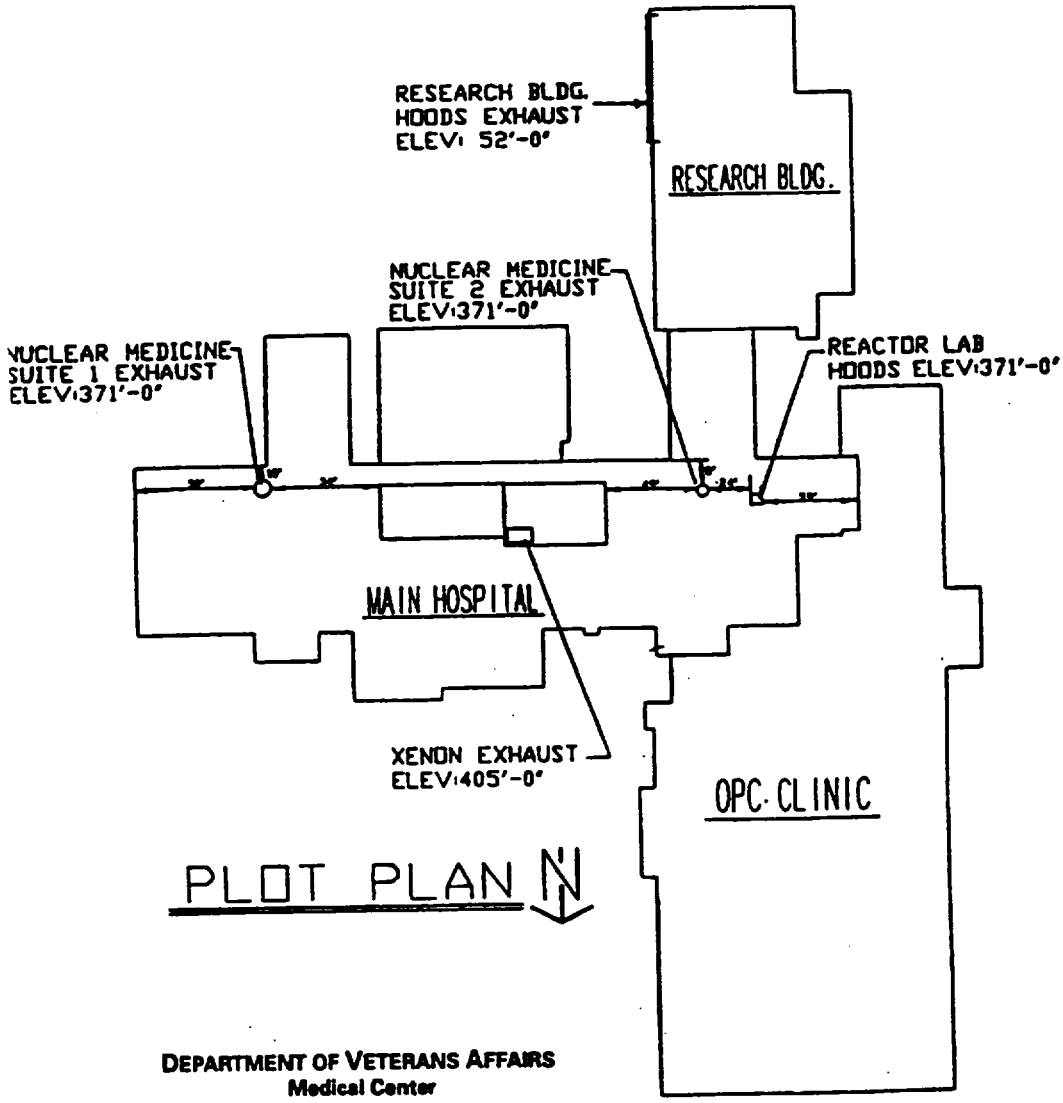

RESEARCH BLDG.
HOODS EXHAUST
ELEV: 52'-0"
RESEARCH BLDG.
NUCLEAR MEDICINE
SUITE 2 EXHAUST
ELEV:371'-0"
NUCLEAR MEDICINE
SUITE 1 EXHAUST
ELEV:371'-0"
REACTOR LAB
HOODS ELEV:371'-0"
MAIN HOSPITAL
XENON EXHAUST
ELEV:405'-0"
OPC. CLINIC
PLOT PLAN
N
DEPARTMENT OF VETERANS AFFAIRS
Medical Center
4101 Woolworth Avenue
Omaha NE 68105

Fig. 3.2a

hood to allow the ^{41}Ar to vent to the atmosphere. Assuming the irradiation vial was empty, a 3 min irradiation of 5 mL of air would emit 0.127 μCi of ^{41}Ar and since the sample occupies at least 2 mL of the vial, each sample would contain 0.05 μCi of ^{41}Ar.

All experiments irradiated in the reactor must be approved by the Reactor Supervisor or his delegate and the irradiation of individual samples must be approved by the reactor operator or SRO in control of the reactor. Approval of irradiation takes into consideration exposures in the fume hoods. The maximum potential dose vented to the public from the fume hood comes primarily from the pneumatic tube. This has been calculated in SAR, Appendix A. The fact that the pneumatic tube is within 6 ft of the operator and the rotary specimen rack is within 14 ft of the operator allows direct control of the experimental facilities. Since the blower for the hood exhaust is on the roof resulting in the entire duct having a negative pressure, any leakage would be into the duct. Consequently, there is no potential exposure within the medical center. This is further monitored by two area monitors (one GM and one scintillation detector) and two continuous air monitors.

If the exhaust fan stops while the reactor is operating the medical center HVAC engineer manning the central system will notify the maintenance man on duty to repair the fan. During off hours the medical center maintenance crew will also respond. If the SRO on duty determines that there is a potential hazard; the reactor will be scrammed immediately.

The areas of potential air exchange are predominantly at the doorways (the doors are normally closed). There are no gaskets, packing or other materials to prevent or inhibit air exchange between the reactor room and other spaces within the medical center to which the public has access. Breeches in the walls due to conduit, pipes, and other structures are sealed with concrete. With the ventfan off there is still a slight negative pressure in the reactor room caused by the two fume hoods.

3.2 REACTOR AND REACTOR SYSTEM

3.2.1 Reactor Pit

The reactor is located near the bottom of a cylindrical pit 6.1 m below ground level, as shown in Fig. 3.3. The pit contains a steel tank of 208 cm ID and 0.64 cm wall thickness; the tank rests on an 28 cm concrete slab. Approximately 25 cm of poured concrete will surround the outside of the tank. This steel-and-concrete structure was fabricated in 1.2 m sections at basement level above the pit, and was installed by simultaneously excavating the earth and lowering the tank sections into place without disturbing the adjacent soil. When all circumferential sections were installed, the bottom concrete slab was poured and the bottom of the steel tank welded in place. All tank welds were Zyglo tested to ensure leak tightness.

The inside of the steel tank is covered on the sides by a layer of gunite approximately 5 cm thick and on the bottom by poured concrete approximately 10 cm thick. The entire inner surface is coated with two applications of a waterproof epoxy resin coating.

Shielding above the reactor core is provided by 4.9 m of water. The reactor pit has been designed to ensure against leakage of the water: The gunite and its waterproof coating protect the steel tank against corrosion by water, and if a small defect in the coating should occur, the steel tank will provide a secondary containment vessel.

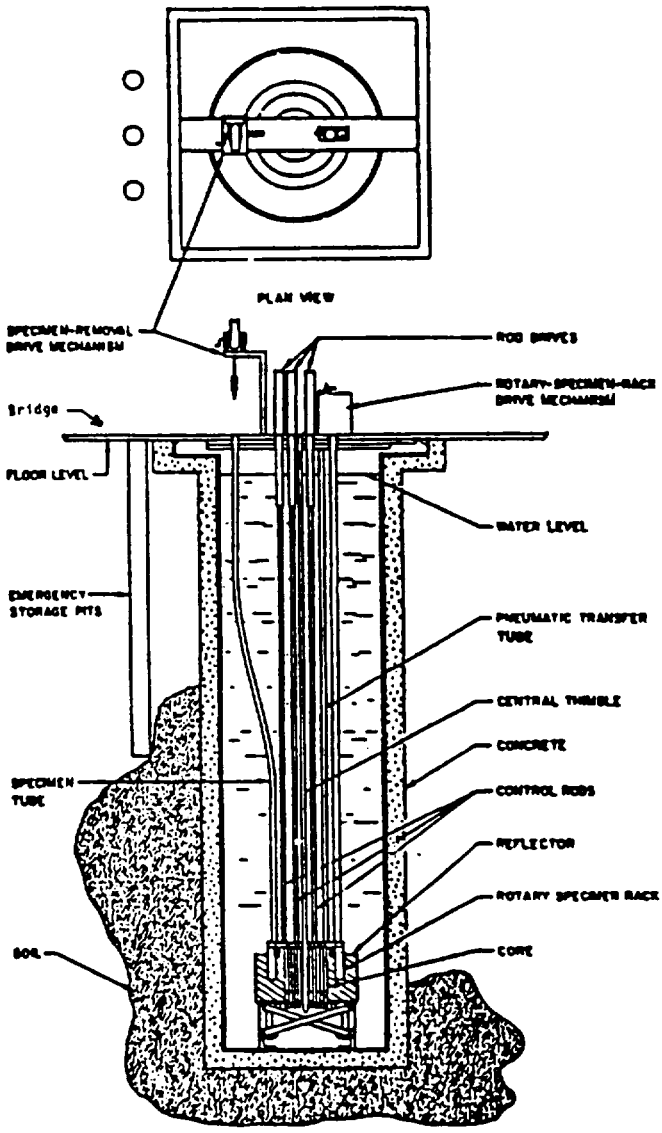


Figure 3.3 Reactor and pit

Three emergency storage pits are located immediately adjacent to the reactor tank. The pits are vertical steel pipes 25 cm in diameter and 305 cm long, and are lined with an organic coating. The pits may be filled with water and used for the temporary storage of irradiated specimens or failed fuel elements prior to their ultimate disposal.

The storage pits have been kept dry and have never been used. If used, all fuel elements will be readily supported during storage in a safe geometry (k_{eff} less than 0.8 under all conditions of moderation). Irradiated fuel elements will be stored in an array, which will permit sufficient natural convection cooling by water or air such that fuel element temperature will not exceed design values. The storage pits have a nominal inside diameter of 10 inches and the fuel elements have a nominal outside diameter of 1.48 inches. Calculations indicate that if the dimensions fall on the high and low limits, respectively, it might be just possible to wedge 37 fuel elements into a single layer in one of the pits. From the experimental data obtained during the initial loading of our reactor (6/59), the most conservative reciprocal multiplication factor (1/M) is 0.55 for 37 fuel elements. This gives a k_{eff} of 0.45, which is safely, sub critical. From experimental tests conducted by General Atomic, it is known that the standard core spacing of elements in the TRIGA is the optimum configuration. Therefore the close-packed spacing in the fuel storage pit would represent a somewhat less-than-optimum condition, and would give an even lower value of k_{eff}.

For practical reasons, if the reactor core had to be unloaded, the fuel elements would be divided between the three storage pits and consequently, the maximum number of elements that would ever be placed in a storage pit would be 20. From the loading curve, the 1/M value for 20 elements was 0.89 giving a $k_{eff} = 0.11$ which is safely subcritical. If the pits were ever used for storage both the water and surface of the pit would be would be monitored for radiation.

The spent fuel storage pits are designed with sufficient spacing to ensure that the array, when fully loaded, will be substantially subcritical. For comparison, actual measured multiplication in an array of five fully loaded (19 elements each) storage pits of similar design yields a k_{eff} of 0.45 (dry). Calculations were performed with MCNP for aluminum fuel for both dry and wet conditions. The results are provided as Appendix E and show $k_{eff} < 0.8$ for all conditions up to 25 elements per pit.

3.2.2 Core

The core forms a right circular cylinder and consists of a lattice of cylindrical fuel-moderator elements and graphite dummy elements immersed in water. Figure 3.4 shows the reactor core and reflector assembly.

The active part of each fuel element (Fig. 3.5) is approximately 3.6 cm in diameter by 0.36 m long and is a solid, homogeneous mixture of hydrided uranium-zirconium alloy containing 8 wt-% uranium enriched to less than 20% in U-235. The hydrogen-to-zirconium atomic ratio is approximately 1.0. A thin aluminum wafer at each end of the active fuel contains samarium oxide, a burnable poison. Each element is jacketed with a 0.076-cm thick aluminum can. Ten centimeter sections of graphite are inserted in the can above and below the fuel to serve as top

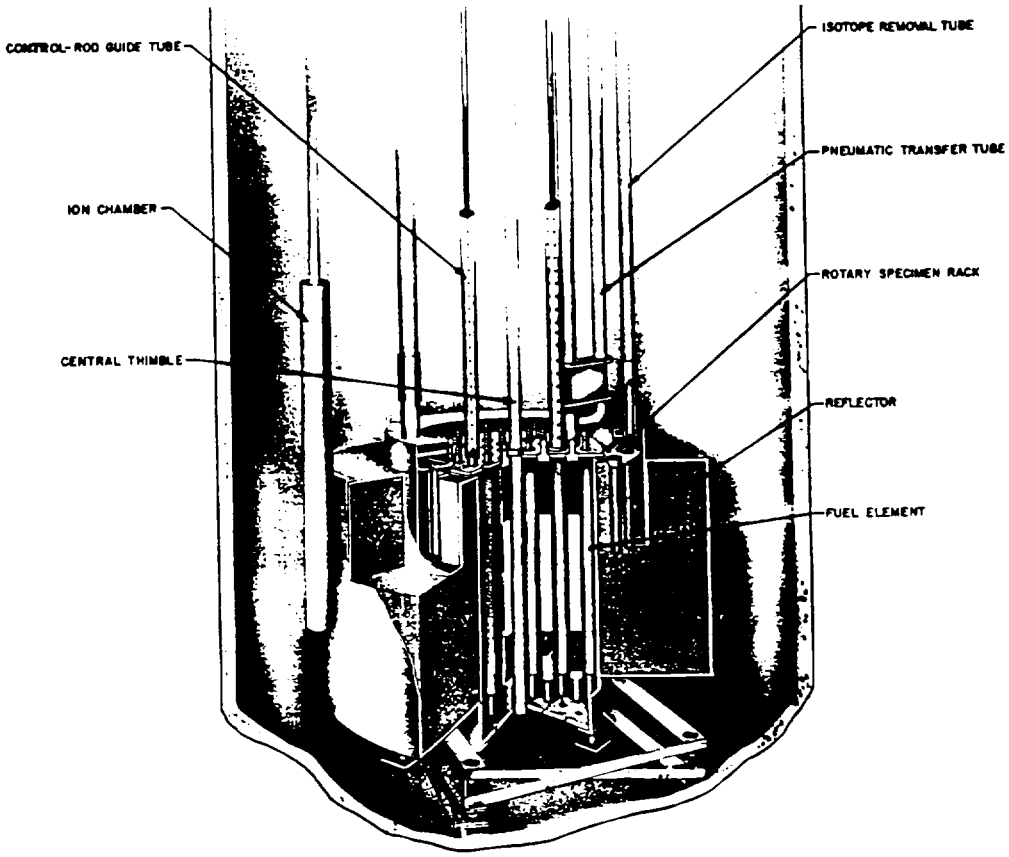


Fig. 3.4 Core and Reflector Assembly

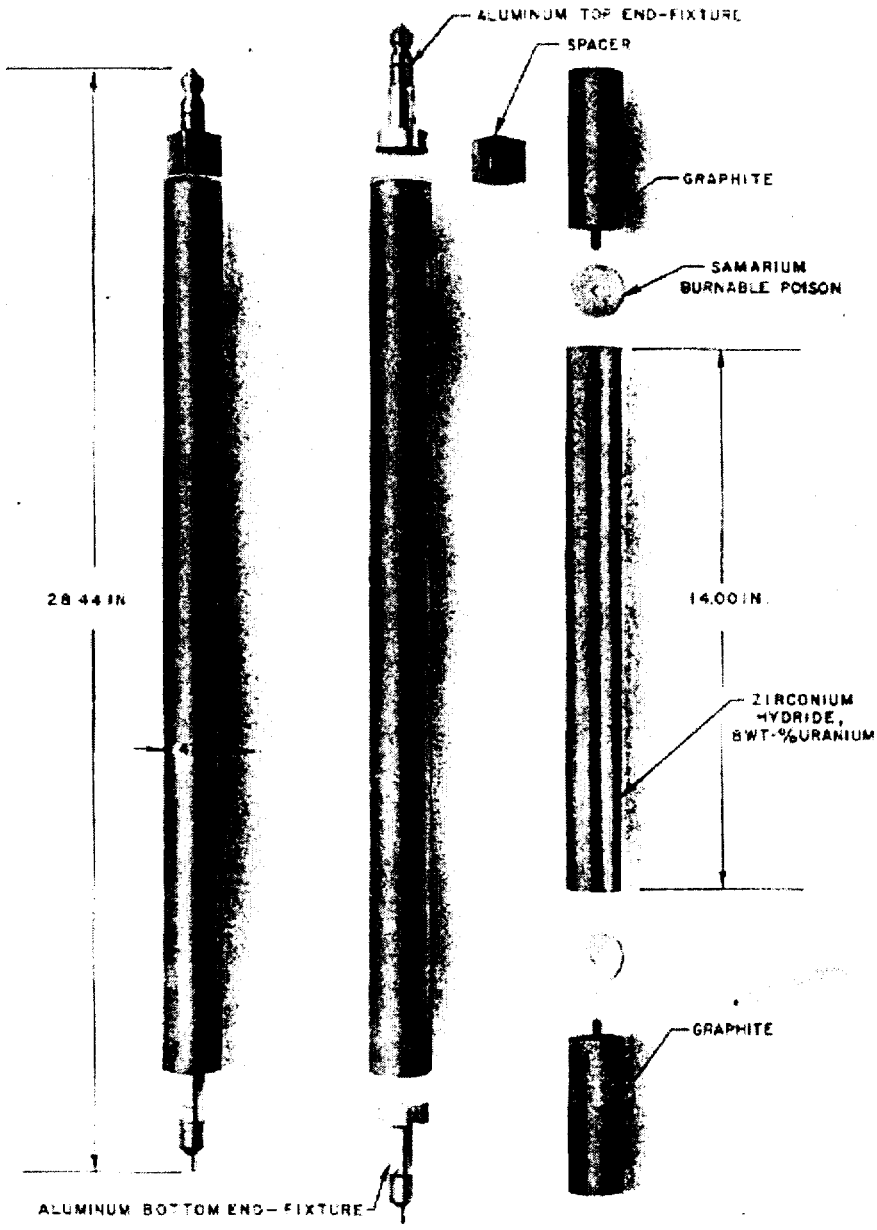


Fig. 3.5 Fuel-moderator-element assembly

and bottom reflectors for the core. Aluminum end fixtures are attached to both ends of the can. The over-all length of the fuel-element is approximately 72 cm.

An alternative TRIGA fuel element uses stainless-steel cladding and is the current standard element. Like the aluminum-clad elements, the stainless-steel-clad fuel elements are homogenous mixture of U-ZrH_x alloy containing approximately 8.5 weigh percent uranium enriched to less than 20% in U-235. The nominal weight of U-235 in each fuel element is 38 g. The hydrogen-to-zirconium ratio is approximately 1.65 to 1.7. The active part of each fuel element is approximately 3.6 cm in diameter by 0.38 m long. Aluminum-samarium wafers are located at each end of the active fuel as a burnable poison. Each element is jacketed with a 0.05 cm thick stainless-steel can. Graphite reflector plugs ($\cong$9 cm long) are located above and below the fuel and serve as neutron reflectors. Stainless-steel end-fixtures are attached to both ends of the can.

General Atomic has been using a mixed core of stainless steel and aluminum clad fuel since 1960 when they were first authorized to use a limited number of stainless steel clad together with aluminum clad elements, as long as fuel temperature in the mixed Al and SS core did not exceed 550°C. This was authorized by Amendment 9 to License No. R-38 in Oct., 1960. Change #1 to License No. R-38 dated Sept. 1965 authorized General Atomic to use stainless steel, aluminum, Hasteloy X or Incoloy 800 up to a full core loading. In addition Amendment No. 31 to Section 4.0 of GA TRIGA Mark I (R-38) Technical Specifications (dated March 1994) authorize various cladding materials and thicknesses, including a mixture of Al and SS clads. Consequently, since a mixed core of Al & SS has been used in the Mark I reactor for 34 years at a thermal power greater than the AJBRF reactor, it is concluded that the health and safety of the public will not be endangered by operating with mixed SS and Al fuel.

The elements are spaced so that about 33% of the core volume is occupied by water. This fuel-to-water ratio in the core was selected because calculations show that it gives very nearly the minimum critical mass. At the present time, the reactor contains 57 active fuel elements. The fuel inventory consists of 56 Al clad elements and 1 stainless steel clad element. The SS element was added to the core on Oct. 2, 1995. All elements in our present inventory were purchased new from General Atomic. Eighty-five fuel-element positions are available in the lattice; the unused positions will be occupied by graphite dummy elements, i.e., elements in which the uranium-zirconium-hydride fuel is replaced by graphite.

The elements are supported and spaced by top and bottom grid plates of 6061 aluminum. The bottom grid plate is 1.9 cm thick, with holes drilled in it to receive the lower end-fixtures of the elements. These lower end fixtures are 0.64 cm diameter cylindrical projections on the bottoms of the fuel cans. A 1.6 cm shoulder is provided on the end-fixture, and the hole in the bottom grid plate is countersunk by a corresponding amount. The weight of the element rests on this shoulder, not on the bottom of the end-fixture, which is used only to position the element as it is being put into place.

The top grid plate is also 1.9 cm thick and has 3.8 cm diameter holes. The top grid plate does not support any of the weight of the elements. The holes serve only to determine the lateral position of the elements and to permit their withdrawal from the core.

The core is cooled by natural circulation of water, which flows through the core from bottom to top. Space for the passage of the cooling water through the bottom grid plate is provided by 36

special holes, and through the top grid plate by the gap between the triangular section of the fuel elements and the round grid hole.

3.2.3 Reflector

The core is surrounded by a cylindrical reflector 30.5 cm thick, 43 cm ID, 107 cm OD, and 56 cm high. This reflector is completely encased in a welded aluminum can, and it is anticipated that flooding of the graphite, in the event that the can should leak, will decrease reactivity. The top and bottom reflectors are the 10 cm graphite sections encased in the fuel element cans, so that the reflector in this region is approximately 67 % graphite and 33 % water, by volume. The reflector assembly is supported at the bottom by an aluminum structure, as indicated in Fig. 3.4.

If water were to flood the reflector housing, then the safety margin would actually be increased by raising neutron absorption and thereby reducing k_{eff}. There is no direct way to verify leakage short of detecting a decrease in neutron flux. In a paper given by B. Dodd, A.G. Johnson and T.V. Anderson at the 11th TRIGA User's Conference at Bethesda, Maryland in 1988, they discuss evidence of possible flooding of the reflector at the Oregon State University TRIGA Reactor. They experienced a 20% drop in neutron flux and after evaluating measurements taken by a number of different individuals, concluded that the reflector was probably flooded. One of the reasons for keeping the reflector dry is to avoid the possibility of galvanic corrosion between the graphite and the aluminum; however, corrosion will only occur in the presence of an electrolyte. The water in the AJBRF reactor has always been kept at a very low conductivity level. Discussion with personnel from General Atomic indicate that impurities in the graphite may be Fe, Si, Ti, Zr, and Ca. However, according to GA these are all well bound and therefore will not change the conductivity of any water inside the reflector. In view of the above, galvanic corrosion appears very unlikely; however, if there were any corrosion inside the aluminum housing around the reflector it would not become a problem unless sufficient corrosion occurred to significantly reduce the structural integrity of the reflector housing. Galvanic corrosion produces pits and holes rather than an overall thinning of the material. Therefore, breakthrough corrosion would be easily detected by white powdery spots on the surface long before it has progressed far enough to weaken the reflector housing.

3.2.4 Center Channel Assembly and Reactor Tank Covers

The center channel assembly across the top of the reactor tank provides support for the drive-and-indicator assembly for the isotope production facility, control rod drives assemblies, and tank covers.

The top, or a portion of the top, of the reactor tank is closed by aluminum grating tank covers that are hinged and installed flush with the floor. A sheet of Lucite plastic attached to the bottom of each grating section prevents foreign matter from entering the tank but still permits visual observation.

3.2.5 Neutron Source

The neutron source consists of a mixture of americium-beryllium, double encapsulated to ensure leak-tightness. Its initial strength at manufacture (1-29-68) was 2 Ci. This source has a nominal outside diameter of approximately 2.5 cm and a height of 1.9 cm. The neutron source holder (Fig 3.6) is the same general size and shape as a fuel element; thus, it can be placed in any vacant fuel or graphite element location. The upper and lower portions of the

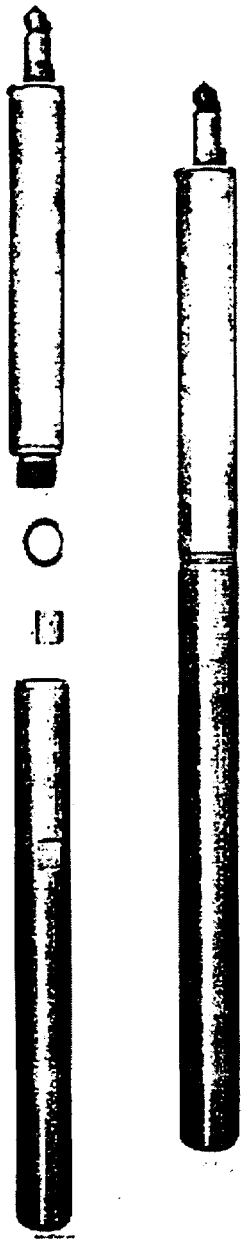


Fig. 3-6 SOURCE HOLDER

holder are screwed together to enclose a cavity that contains the source. A shoulder at the upper end of the neutron source holder supports the assembly on the upper plate.

3.2.6 Irradiation Facilities

Special irradiation facilities are provided for the production of radioisotopes. These include a rotary specimen rack located in the well in the reflector can, a pneumatic transfer tube, and a central thimble (Fig. 3.4). In addition, odd-shaped specimens may be irradiated in the water outside the reflector.

3.2.6.1 Rotary Specimen Rack

The rotary specimen rack consists basically of an aluminum ring which can be rotated around the core. Forty aluminum cups, evenly spaced, are hung from the ring and serve as irradiation specimen holders. The ring can be rotated manually from the top of the reactor pit, so that any one of these cups can be aligned with the single isotope-removal tube which runs up to the top of the reactor pit. This tube is used for removing and replacing irradiation specimens. An indexing and keying device is provided to ensure positive positioning of the cups.

The rotary specimen rack is completely enclosed in a welded aluminum box. The aluminum ring is located at approximately the level of the top grid plate, with the specimen cups extending from the ring down to about 10 cm below the top of the active lattice. In the radial direction, the centers of the cups are about 10 cm from the inner edge of the reflector assembly. The box enclosing the rotary specimen rack has been designed to ensure that it will remain watertight. Flooding of this box will decrease the reactivity of the reactor. The decrease in reactivity is due to the fact that there is a greater absorption of neutrons due to the presence of water. Four of the aluminum sample cups, spaced 90 degrees apart, have perforations in the walls. One of the four perforated tubes has a 0.625 inch (15.9 mm) diameter hole in the bottom. The hole permits testing of the rotary-specimen-rack housing to determine the extent of any accumulation of condensation or leaking water. If condensation occurs, as a result of high humidity in the reactor area and low operating temperature, the four perforated tubes can each be loaded (when the reactor is shut down) with a suitable porous container filled with a water absorbing agent.

3.2.6.2 Pneumatic Transfer Tube

The pneumatic transfer tube is provided for the production of isotopes with short half lives. It consists, in essence, of two tubes leading down through the water tank to a position at the outer edge of the core, where the tubes are joined. The specimen is fed in and out through one of the tubes and a blower connected to the other tube provides the pressure difference required to inject or eject the specimen. Specimens are inserted into and removed from the pneumatic system in the reactor laboratory. All samples inserted into the pneumatic transfer system (PTS) must be approved by the Reactor Supervisor or his designate. Investigator must have irradiations approved daily by the SRO who signs a posted checklist. There is only one PTS and it is within 5.5 ft of the reactor operator so control is direct. The same criteria is used for evaluating samples inserted into the PTS as is used for samples inserted into the other experimental facilities, [ANS 15.1, (1990) and AJBRF Tech.Specs.]. Samples are irradiated in an 11.5 cc carrier. The normal sample vial used is 7 cc plastic. The PTS is used mainly for neutron activation analysis where the radionuclides produced are short lived and analyzed

typically within one hour of irradiation. Samples are monitored upon removal and ALARA principles are followed.

3.2.6.3 Central Thimble

A central thimble is provided to permit irradiations or experiments in the region of maximum flux and maximum statistical weight. It consists of a vertical aluminum tube 3.4 cm ID leading from the top of the reactor pit through the exact center of the reactor core and terminating below the bottom of the core. The bottom of the tube is capped, but holes drilled in the wall of the tube ensure that the position in the active lattice will be filled with water during reactor operation.

The shield water can be removed from the portion of the central thimble above the upper grid plate using air pressure to force the water out of the tube through the holes in the tube wall. This provides a highly collimated beam of neutron and gamma radiation for experiments. The radiation dose on the next floor directly above the reactor is 1.5 mrem per hour with the thimble, which has 5 cm of lead shielding above it, operating
as a beam tube.

Lead bricks (2 in thick) are stacked around the central thimble before the shield water is removed and the radiation dose in the reactor room, when the thimble is used as a beam tube is less than 2 mrem per hour. The central thimble has only been used once since 1959 for a beam to determine radiation dose levels during such use and it was demonstrated that with minimal additional shielding (2 to 4 in. of lead) above the thimble it can be used with no significant hazard to the medical center staff or to the public.

As mentioned in this section, the waterless central thimble has been used only once, July 1, 1969 to determine radiation levels. At the present time we have no plans to utilize the facility to extract a beam since it is too collimated. However, if it is ever used we will assure that the radiation dose rates in occupational areas will comply with AJBRF's Radiation Protection Program, 10CFR20, and other applicable regulations. Since the floor above is occupied by an ear and eye clinic, we will shield the beam so no radiation above background will be received. The central thimble filled with water is used to irradiate samples by placing them in a water tight aluminum tube that is completely filled with samples and polyethylene so no extraneous air is within the device. The irradiation device is then is then lowered into the water filled tube. The dose rate at the top of the tube is no different than that at the top of the reactor.

Unauthorized use or inadvertent operation of the central thimble is prevented by the fact that the Reactor Operator is only 10 ft away and in direct visual contact with the top of the central thimble. The central thimble irradiation device can only be inserted or removed by the reactor operator or his designee.

3.2.7 Control Rods and Guide Tubes

The three boron carbide control rods operate in perforated aluminum guide tubes. The guide tubes are attached to the bottom grid, and the upper grid provides lateral support. The control rod has an extension tube which connects to the control-rod drive mechanism. The safety rod, which during normal operation is completely out of the core, and the shim-safety rod are each worth approximately $ 2.25. The regulating rod is worth approximately $ 0.85.

3.2.8 Control-rod Drives

The control-rod drive mechanisms, located on the bridge at the top of the reactor pool structure, consist of a motor and reduction gear that drive a rack and pinion, and a potentiometer for position indication. The control-rod extension tube and dashpot are connected to the rack through an electromagnet and armature. In the event of power failure or a scram signal, all three of the control-rod magnets are de-energized and the rods fall into the core. All control rods are scrammable. There is an interlock to prevent any two control rods, including the regulating rod from being withdrawn simultaneously. The interlock function of the source neutron count rate is above 2 counts/s. The drive is nonsynchronous, single phase, and instantly reversible. Electrical dynamic and static braking on the motor are used for fast stops. Limit switches mounted on the drive assembly indicate the up and down positions of the magnet, the down position of the rod, and magnet contact. The complete drive assembly is enclosed in an aluminum can. The control-rod drive mechanisms have a stroke of approximately 38 cm. The maximum rod withdrawal rate is 30.5 cm/min, and the maximum rate of reactivity insertion is about $ 0.05 per second. Rod-position indicators are provided on the regulating rod and on the shim rod. Fig. 3-7 shows the control-rod drive mechanism.

Interlocks are provided to assure minimum neutron countrate of 2 cps before control rods can be withdrawn and to prevent withdrawal of any two control rods, including the Reg rod, simultaneously.

3.2.9 Reactor Water Cooling and Purification System

The reactor is cooled by natural convection of the pool water. A 5-ton freon vapor-compression chiller with an air cooled condenser is used as the heat sink. Water from the reactor tank goes to the water monitor, where the temperature, gamma activity, and conductivity of the water are measured. It then goes to the suction end of a pump and from there to the chiller unit. From the chiller it goes through a filter and then through a mixed-bed type demineralizer. A bypass line is provided from the outlet of the chiller to the inlet of a rotometer, where flow rate is measure. The water is then returned to the tank. The flow inlet pipe is 13 ft above the top of the core. In the event of a rupture in the cooling system the maximum amount of water lost would be to this level. However, the water level would most likely lose only a few inches before the skimmer began to suck air. This would effectively cause the pumping system to lose its prime. Therefore, approximately 17 cu ft could be released within the concrete enclosure housing the cooling system or within the reactor room. There is little radiological significance to this as the water activity is well below any of the quantities specified in 10CFR20 Appendix B. Figure 3.8 shows a schematic of the cooling and water-treatment system. The water system serves four functions; it:

1. Maintains low conductivity of the water to minimize corrosion of all reactor components, particularly the fuel elements.
2. Reduces radioactivity in the water by removing nearly all particulate and soluble impurities.
3. Maintains optical clarity of the water.
4. Provides a means of dissipating the heat generated in the reactor.

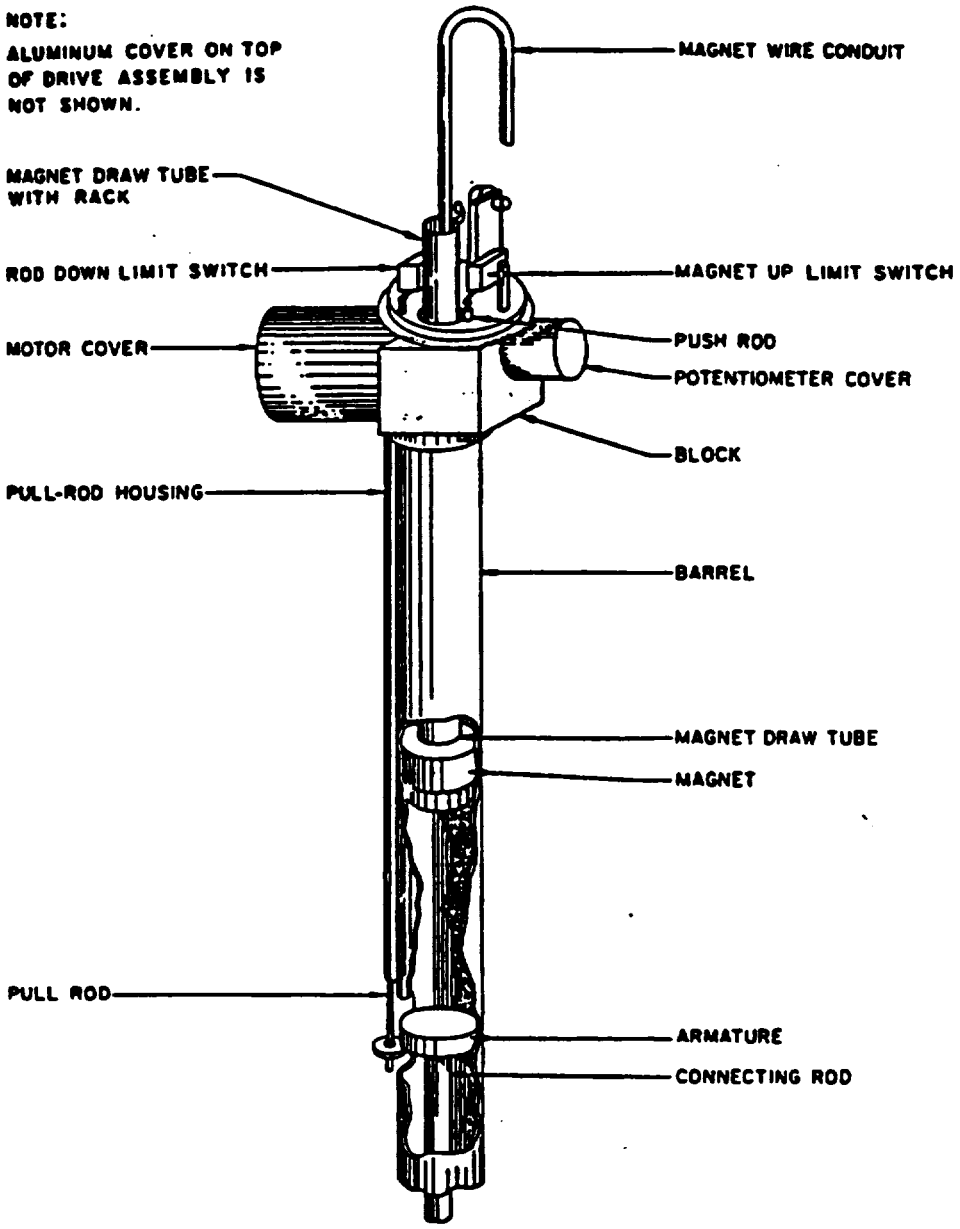


Figure 3.7 Control-rod drive mechanism

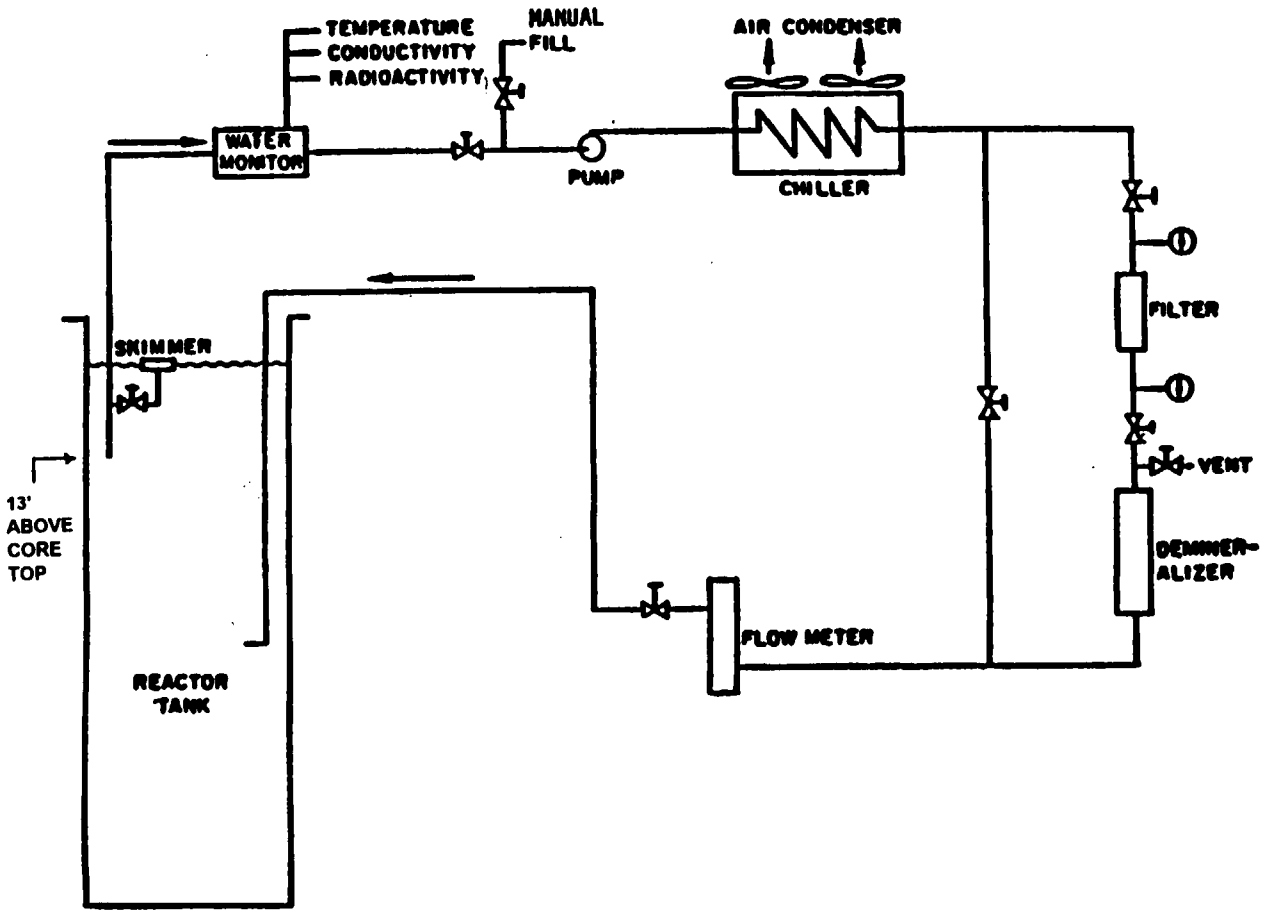


Figure 3.8 Reactor cooling system

The cooling system is situated above the reactor tank. Thus, there is no possibility of loss of primary coolant by siphoning. Analysis of the primary coolant has determined extremely minute amounts of radioactivity. As a result, the amounts determined in the primary coolant would meet the specifications required by 10 CFR 20.2003. As a result, there would be little environmental consequences from release of reactor coolant.

Makeup water is supplied to directly to the reactor pool. A transfer tank is used in this process which has no connection to the city water supply. Typically this amounts to 15 gallons per year.

3.2.10 Ability of Reactor Facility Structure, Systems and Components to Function Properly and Safely for the Term of the License

In the history of the facility there has not been any observed changes in the strength or integrity of the fuel element components, the tank or the lining material due to neutron or gamma radiation damage. Four fuel elements have been examined each quarter by removing each separately and placing it in a device that allows us to examine it in detail underwater with a 20 power telescope each element has failed to show any significant change. Likewise we have not observed any change in the reactor tank. Consequently, there is no reason to believe that there will be a breach of integrity of the components during the requested license extension.

3.2.10.1 Reactor Tank

3.2.10.1.1 The reactor tank is 0.64 cm steel conforming to ASTM standard A 7-56T. Welding at perimeters and joints of all pieces of plate being a minimum of 3 mm fillet. All welds were zyglo tested to insure leak tightness. The tank rests on a 28 cm concrete slab and approximately 25 cm of poured concrete surrounds the outside of the tank. All concrete had a minimum allowable compressive strength at 23 days of 3,000 psi. The inside of the steel tank is covered on the sides by a layer of gunite approximately 5 cm thick and on the bottom by poured concrete approximately 10 cm thick. The entire inner surface is coated with two applications of a waterproof epoxy resin coating. The reactor tank has never overflowed. Visual observation of the tank with binoculars shows absolutely no evidence of deterioration of the tank and consequently, it is reasonable to expect that operation for an additional 20 years will have no adverse effect on the tank.

3.2.10.2 Core

3.2.10.2.1 Fuel elements are jacketed with 0.076 cm thick aluminum, and visual inspection of 4 elements each quarter have shown no indication of any deterioration or swelling.

3.2.10.2.2 Grid plates are 1.9 cm thick 6061 aluminum and visual observation of the top plate with binoculars shows no evidence of deterioration.

3.2.10.2.3 All Control rods are visually inspected annually, at intervals not to exceed 15 months. Rods are physically removed from the core and visually inspected for signs of pitting or deterioration. To date we have replaced the following control rods for the reason shown:

Safety rod replaced 5/18/64 - Pitting
Shim and Regulating rods replaced 2/28/66 - Pitting
Regulating Rod replaced 11/18/73 - Pit thru cladding

The control rods are calibrated annually to measure reactivity changes in the core. Core excess reactivity and shutdown margin are also calculated in this process.

3.2.10.2.3.1 We were informed by General Atomic, after analyzing the 1966 rod, that the pits were probably due to iron particles becoming embedded in the surface of the rod during the manufacturing process. After being alerted of this process General Atomic revised their manufacturing and inspection procedure to minimize the possibility of iron being embedded in the aluminum. Since 1973 visual and contact inspection has revealed no evidence of pitting.

3.2.10.2.4 In summary, visual observation of all parts of the core reveals no indication that the facility cannot operate safely for the requested term of the renewal license.

3.2.10.2.5 Effects of Fuel Aging - There is some evidence that the U-ZrH_x fuel tends to fragment with use, probably as a result of the stresses caused by high temperature gradients and high rate of heating during pulsing [1-2]. Some of the possible consequences of fragmentation are (1) a decrease in thermal conductivity across cracks, leading to higher central fuel temperatures during steady-state operation (temperature distribution during pulsing would not be affected significantly by changes in conductivity because a pulse is completed before significant heat redistribution by conduction occurs), and (2) more fission products would be released into the cracks in the fuel.

With regard to the first item above, hot cell examination of thermally stressed hydride fuel bodies have shown relatively widely spaced cracks that would cause minimal interference with radial heat flow [3]. However, after pulsing, TRIGA-type reactors have exhibited an increase in both steady-state fuel temperatures and power reactivity coefficients. At power levels of 500 kW, temperatures have increased by approximately 20°C and power reactivity coefficients by approximately 20% [4]. General Atomic has attribute these changes to an increased gap between the fuel material and cladding caused by rapid fuel expansion during pulse heating, which reduces the heat transfer coefficient. Experience has shown that the observed changes occur mostly during the first several pulses and have essentially saturated after 100 pulses. Because these effects are small and have been observed in many TRIGA-type reactors operated at pulses up to $ 5.00 and power levels as high as 1.5 MW and because the AJBRF reactor is not operated in the pulse mode, they are not considered to pose any hazard during continued operation of the AJBRF reactor.

Two mechanisms for fission product release from TRIGA fuel meat have been proposed [3,6]. The first mechanism is fission fragment recoil into gaps within the fuel cladding. This effect predominates up to about 400°C and is independent of fuel temperature. AJBRF operating fuel temperatures have never exceeded 400°C; thus, this will be the main effect. General Atomic has postulated that in a

closed system such as exists in a TRIGA fuel element, fragmentation of the fuel material within the cladding will not cause an increase in the fission product release fraction [6]. The reason for this is that the total free volume available for fission products remains constant within the confines of the cladding. Under these conditions, the formation of a new gap or widening of an existing gap must cause a corresponding narrowing of an existing gap at some other location. Such a narrowing allows more fission fragments to traverse the gap and become embedded in the fuel or cladding material on the other side. In a closed system in which the density of the fuel meat is constant, the average gap size and therefore the fission product release rate remains constant, independent of the degree to which fuel material is broken up.

Above approximately 400°C, the controlling mechanism for fission product release is diffusion, and the amount accumulated in the gap is dependent on fuel temperature and fuel surface-to-volume ratios. In the AJBRF fuel this mechanism is not significant because of the low fuel temperature and low utilization factor.

The fuel inventory currently in use were acquired as new elements. Thus, the core element irradiation history is known. Therefore it is concluded that the likely process of aging of the U-ZrH_x moderator under low-power, steady-state, nonpulsing operation would not cause significant changes in the operating temperature of the fuel or affect the accumulation of gaseous fission products within the cladding. Therefore, there is reasonable assurance that fuel aging will not significantly increase the likelihood of fuel-cladding failure, or the quantity of gaseous fission products available for release in the event of loss of cladding integrity.

3.2.10.3 Electronics and Mechanical

3.2.10.3.1 Electronics - Routine maintenance is performed on all of our electronics systems by the VA Research Service electronics technician.

3.2.10.3.1.1 The original neutron monitoring system was replaced with a state-of-the art microprocessor system in 1991.

3.2.10.3.2 Mechanical - Routine maintenance is performed on all mechanical systems by Medical Center electrical, air-conditioning, refrigeration and plumbing personnel.

3.2.10.3.3 In summary, all systems are operating in excellent condition and there is no indication that their lifetime would affect the safe operation of the facility.

3.2.10.4 Conclusion - On the basis of the above evaluation there is a reasonable basis to expect the total reactor facility to remain safely operable for the requested period of license renewal.

3.2.11 Confinement Design Evaluation

3.2.11.1 Activation of the Soil Surrounding the Reactor Pit

The soil adjacent to the reactor pit can capture fast and thermal neutrons, which escape from the pit. The magnitude of the radioactivity induced has been approximated for a typical soil in order to determine whether leaching of activity in the soil might constitute a potential environmental hazard to the ground water. We have been unable to obtain the depth of the water table, but when the new clinic addition to the medical center was built (west of the reactor as indicated in Fig 2.2) pilings were sunk 150 ft with no trace of water. In addition, a test bore at the center of the reactor location (SAR, Sec. 2.4) indicated that no water table was encountered. Calculations as shown in SAR, Appendix D show that at saturation the activity that would be carried away from a soil volume under conservative conditions would decay well below applicable levels specified in 10CFR20, Appendix B, Table 2, Column 2. Due to the short half lives of most of the radioisotopes analyzed, the volume of ground water and the fact that the reactor is typically operated for a maximum of 7.5 hours per day; the activation of the soil should have no environmental impact

3.2.11.2 Production of Radioactive Gases by the Reactor

Routine radioactive releases from the reactor occur principally in the form of Ar-41 and N-16. Both isotopes are generated via activation of the pool water due to specific nuclear processes. Additionally, Ar-41 is also released as a result of activation of air in irradiation facilities. Specifically, the pneumatic transfer tube and the lazy susan. Determinations of these releases are summarized below:

Table 3 - 1
Summary of results
from ^{41}Ar (2000 hours)

	Ci/yr	Release Point (outside)	Wholebody DOSE mrem/yr MEW	Wholebody DOSE mrem/vr MMP	Wholebody DOSE mrem/vr NPR
^{41}Ar pool	$2.6x10^{-1}$	Ventfan	$3.9x10^{-1}$ [a]	$1.0x10^{-1}$ [b]	$4.4x10^{-2}$ [c]
^{41}Ar PT	$1.34x10^{-1}$	Hood exhaust	na	$4.3x10^{-2}$ [c]	$1.8x10^{-2}$ [c]
^{41}Ar RSR	$1.1x10^{-1}$	Ventfan	$1.8x10^{-1}$ [d]	$3.55x10^{-2}$ [c]	$1.5x10^{-2}$ [c]
Total ^{41}Ar	$5.1x10^{-1}$		$5.1x10^{-1}$	$1.49x10^{-1}$	$6.4x10^{-2}$

[a] Value determined in Appendix A, Section A.1.2

[b] Value determined at the end of Appendix A , Section A.2

[c] Value determined similarly as the calculation Appendix A , at the end of Section A.2 using the appropriate variables noted in Table #1

[d] Summary, Appendix A, at the end of Section A.4

Routine radioactive releases as well as the potential exposure to radiation workers and to members of the public are further calculated and summarized in SAR, Appendix A of this part.

3.2.12 Limiting Design Basis

The limit for TRIGA fuel is dictated by temperature. This limit is dependent on the type of TRIGA fuel used. The AJBRF has both Al clad low hydride (H/Zr ratio less than 1.5) fuel and stainless steel high hydride (H/Zr ratio greater than 1.5) fuel. The majority of which is Al clad.

The core currently consists of 56 Al clad and 1 stainless steel clad elements. The TRIGA fuel with low hydride ratio has a lower temperature limit than the high hydride fuel. Figure 3.10 indicates that the higher hydride compositions are single phase an are not subject to large volume changes associated with the phase transformations at 530°C in the lower hydrides. The higher hydride limit stems from the out-gassing of hydrogen from U-ZrH fuel and the subsequent stress produced in the fuel element clad material. It should be noted, however, that the higher hydrides lack any significant thermal diffusion of hydrogen [5].

The results of General Atomic's experimental and theoretical determinations [2,3], show that fuel element integrity is not compromised for cladding temperatures at or less than 500 °C. Reviews of these experiments and determinations can be found in NUREG-1282 [6] and NUREG 0988 [7].

3.2.13 Dynamic Behavior of Reactor

This section will consider the behavior of the reactor as a result of the sudden insertion of a large amount of excess reactivity into the core. General Atomic has continued the testing and evaluation of TRIGA by undertaking a high-power transient test program under controlled experimental conditions on the prototype reactor. A special license was obtained from the AEC for this series of tests. Some of the salient features of the tests are summarized here [1]. The test was performed using the Torrey Pines TRIGA Mark I reactor identical in construction to the AJBRF TRIGA. The principal design parameters are shown in Fig. 1-1 of our SAR. The only difference is that the Torrey Pines reactor had two safety rods worth $2.50 and $2.00, a pneumatically driven regulating rod worth $2.50 and a shim rod worth $4.50.

A 2-dollar step reactivity insertion has been demonstrated, without deleterious effects either to the reactor or to operating personnel in the immediate vicinity of the reactor. This 2-dollar insertion yielded a reactor period of 10 ms and a peak power of approximately 250 MW. This excess reactivity was rapidly compensated by the large prompt negative temperature coefficient, which is an inherent characteristic of this reactor core. Within 30 seconds after initiation of the transient, the reactor power level had returned to an equilibrium of 200 kW. The total energy release in the prompt burst was approximately 10 MW-s. The maximum transient fuel temperature was about 360° C.

Curves of the transient power level and of the fuel temperature during this transient are shown in Fig. 3.9. No boiling was observed in the reactor tank and no disturbance of the shielding-water surface was noted during the 2-dollar transient. The integrated radiation dose that an individual would have received had he stood immediately over the reactor tank during this power transient would have been 21 milliroentgen equivalent, man (mrem).

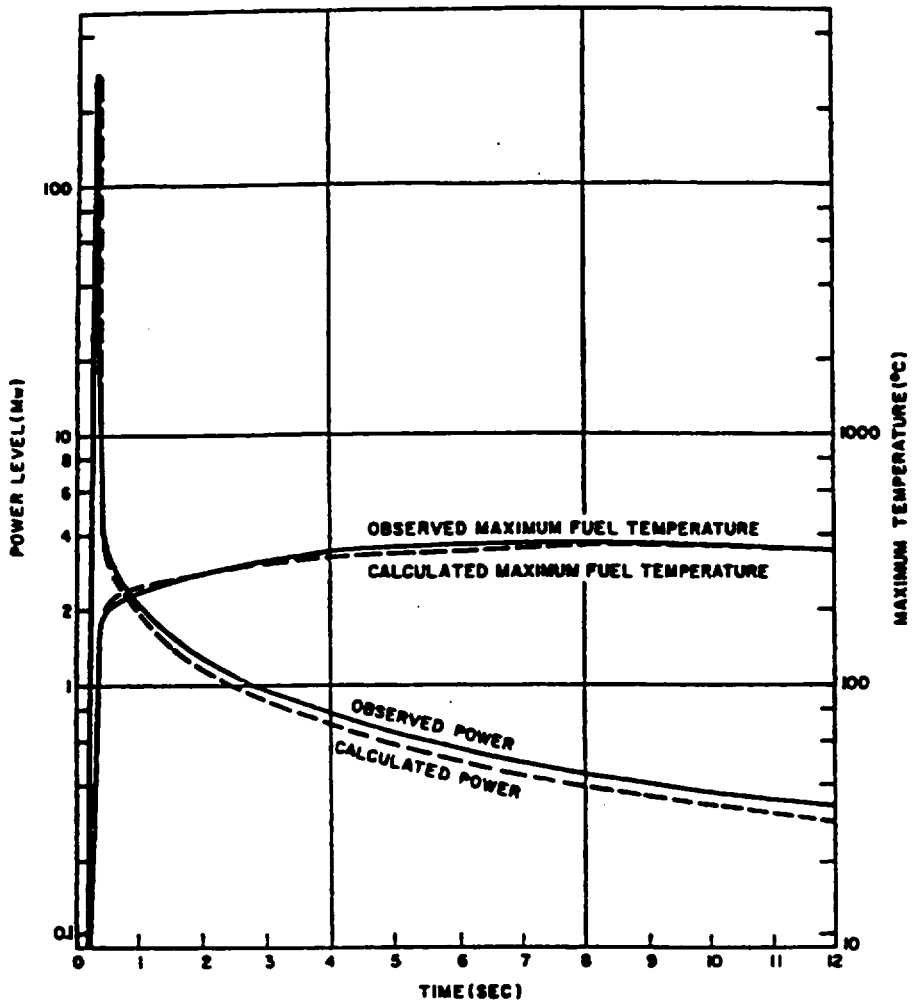


FIG 3.9 --Two-dollar reactivity transient

During the quasi-equilibrium experiments on the prototype TRIGA, the reactor was operated at a power of 330 kW for a period of approximately one hour with no indication of instability or bulk boiling in the reactor core. The data obtained in these experiments provide an experimental value of 80 ± 5 μsec for the effective neutron lifetime for this reactor. The temperature coefficient measured in the quasi-equilibrium experiments can be fitted to good approximation by a constant over the experimental temperature range. This temperature coefficient has been measured to be 0.016 dollar reactivity loss per degree centigrade rise in fuel temperature.

The core consists mostly of a Al-clad fuel with a H:Zr ratio of 1.0. Zirconium occurs in two crystalline forms: alpha (stable below 860 °C) and Beta (stable above 860 °C) (refer. Fig. 3.10). The alpha phase is close-packed hexagonal and does not absorb any large amount of hydrogen. The small amount of hydrogen it does take up forms a solid solution with it. Absorption of more hydrogen at elevated temperatures (>560 °C) cause a transition of part solid to the beta phase, which is body-centered cubic and in which hydrogen is added can go into solid solution up to an H:Zr ratio of 1.0. If more hydrogen is added than is required to saturate the beta phase, the precipitation of the gamma hydride, which has a ratio of H:Zr $\leq\sim$ 1.5, begins.

A loss in the integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel and the cladding if the fuel temperature exceeds the safety limit. The heating of air, fission product gases, and hydrogen causes the pressure from the dissociation of the fuel-moderator. The magnitude of this pressure is determined by the temperature of the fuel element and by the hydrogen content. Experience with operation of TRIGA-fueled reactors at power levels up to 1500 kW shows no damage to the fuel due to thermally induced pressures.

Thermal cycling tests have been performed to verify fuel matrix stability with resect to swelling or elongation. Simnad [3] has described these tests with temperatures in the range 500 °C to 725 °C. He has explained why there are no important changes in length or diameter of the test samples even though a small phase transition did occur at 653 °C (orthorhombic to tetragonal). For a TRIGA fuel with fuel temperatures $\leq$ 200 °C, there is no phase change or other transition to produce elongation or swelling in the fuel matrix.

Under long term, high burnup conditions of irradiation, the possibility would exist for hydrogen migration and accumulation of fission products in the fuel. Simnad has treated these features at length and demonstrated that none of these effects is important for fuel temperatures below 500 °C especially if the reactor is not pulsed. A temperature of 500 °C is well above the fuel temperatures characteristic of a TRIGA reactor operating at 20 kW

On the basis of the evidence presented above, it is concluded that there is no hazard associated with a rapid insertion of as much as 2 dollars excess reactivity (1.6 δk/k) in this reactor. From the above experiment the following reactivity limits can be justified:

a) **Excess Reactivity:**
The objective of limiting excess reactivity is to prevent the fuel element temperature safety limit from being reached by limiting the potential reactivity available to the reactor for any condition of operation. The maximum power excursion that could occur would be one resulting from inadvertent rapid insertion of the total available excess reactivity. Limiting the fuel loading of the AJBRF reactor to $ 1.00 excess reactivity under clean-cold critical conditions will assure that the fuel temperature will not reach the maximum

fuel temperature of 560°C where a phase change resulting in great enough internal pressure to cause cladding failure occurs [2,3].

b) **Shutdown Margin:**
Requiring a minimum shutdown margin of $ 0.51 with the highest worth control rod fully withdrawn, the highest worth non secured experiment in its most reactive state, and the reactor in the cold critical condition without xenon, assures that the reactor can be shut down from any operating condition.

c) **Reactivity limits on experiments:**
Limiting the worth of a single experiment to $ 1.00 assures that sudden removal of the experiment will not cause the fuel temperature to rise above the critical temperature level of 560°C. Limiting the worth of all experiments in the reactor and in the associated experimental facilities at one time to $ 1.00 will also assure that removal of the total worth of all experiments will not exceed the fuel element temperature safety limit of 500°C.

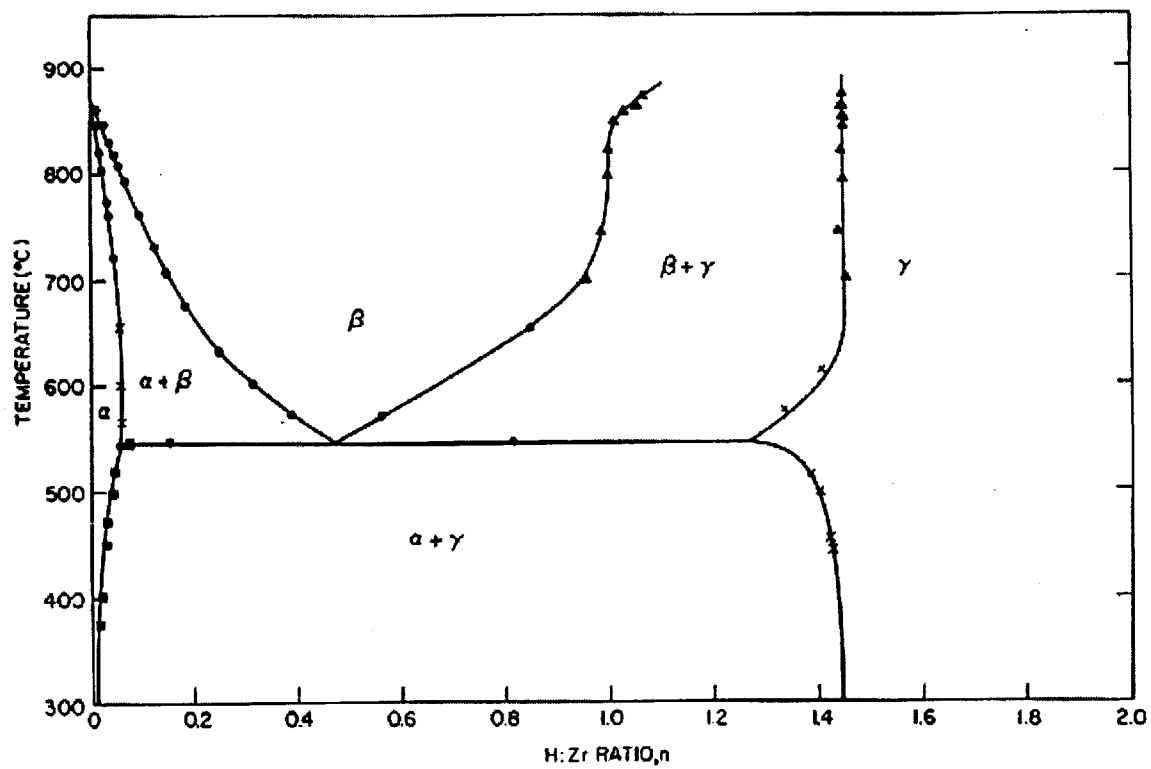


Fig 3-10 Zirconium-Hydride Phase Diagram

▲ R. K. Edwards, et al., J. Am. Chem. Soc. 77, 1307 (1955)
■ Gulbransen and Andrew, Trans. Amer. Inst. Min. Met. Eng. 203, 136 (1955)
● C. E. Ells and A. D. McQuillan, J. Inst. of Metals 85, 89 (1956)
X U. Merten and L. D. La Grange, unpublished data

Chapter 3

References

1. R.S. Stone and H.P. Sleeper, "TRIGA Transient Experiments", Interim Report General Atomic GA 531, September 19, 1958.

2. Simnad, M.T., Foushee, F.C. and West, G.B., "Fuel Elements for Pulsed TRIGA Research Reactors", Nuclear Technology, 28:31. 1976.

3. Simnad, M.T., "The U-Zrh_x Alloy: Its Properties and Use in TRIGA Fuel", GA-4314, E-117-833, GA Technologies, Inc., San Diego, CA, 1980.

4. "Thermionic Research TRIGA Description and Analysis", GA-5400, Rec. C., Nov. 1, 1965. [transmitted by letter dated Feb. 28, 1966 (Docket No. 50-277)].

5. Marten, U., et al., "Thermal Migration of Hydrogen in Uranium-Zirconium Alloys," General Dynamics, General Atomic Division Report GA-3618, 1962.

6. "Release of Rare Gas Fission Products from U-ZrH_x Fuel Material", F.S. Fushee, GA-8597, Mar. 1968.

7. "Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors", Docket No. 50-163GA Technologies, Inc., U.S. Nuclear Regulatory Commission, August 1987.

8. Safety Evaluation Report related to the renewal of the operating license for the research reactor at the Omaha Veterans Administration Medical Center, Docket No. 50-131. U.S. Nuclear Regulatory Commission, July, 1983.

CHAPTER 4

INSTRUMENTATION AND CONTROL SYSTEMS

4.1 SYSTEMS SUMMARY

The control and instrumentation systems for the AJBRF TRIGA are similar to those used in other research reactors in the United States. The nuclear fission process is controlled by using three control rods. The control and instrument systems are interlocked to provide the means for operating the various components in a manner consistent with design objectives. A block diagram of the AJBRF instrumentation and scram system is shown in Figure 4.1 and the minimum required reactor safety channels, functions and set points are shown in Table 4.1.

4.2 CONTROL CONSOLE

The reactor control console contains the control, indicating and recording instrumentation required for operation of the reactor. All of the reactor's essential functions are controlled from the console. On the control panel are:

(1) rod control switches for raising and lowering the control rods;
(2) rod-position indicators to show the position of the shim and regulating rods to within 0.2%; e.g. the exact linear rod positions can be reproduced by 0.2% of full travel. There is no position indicator for the safety rod. It must be completely up or down or an interlock will not allow the shim or regulating rod to be moved.
(3) enunciator lights to indicate the up or down position of each rod and rod-magnet contact;
(4) linear and log-N power recorders;
(5) period, power level, pool temperature, and start-up channel meters;
(6) monitor alarm lights; and
(7) additional pilot lights to indicate power on, cooling system on, and startup source countrate. Other enunciator lights on the console indicate the source of a scram signal.

Automatic scram is initiated by:

(1) an excessive reactor power level as indicated by a signal from either a wide range fission counter or an uncompensated ion chamber:
(2) a wide range fission counter or uncompensated ion chamber power supply failure (loss of high voltage): and
(3) an electrical power failure, or (4) a signal from the watchdog timer. Manual scram can be initiated by the operator by means of the console scram button or the magnet current key switch. The magnet current key switch breaks the rod-magnet circuit so that the console may be operated without rod withdrawal if the switch is off. After the rod(s) drop by loss of magnetic current, the drive mechanism automatically follows the rod down to reestablish contact.

For steady-state operation, the control rods are withdrawn slowly by manual control until the desired power is reached. A servo loop may be used to hold the power constant at the desired level by movement of the regulating rod. The desired power level, expressed as a percent of

Table 4.1 Minimum reactor safety channels		
Safety Channel	**Function**	**Set Point**
Percent power*	Scram	100% Licensed power
Linear power level**	Scram	100% Licensed power
Log N (period)	Prevents withdrawal of all control rods	Minimum period of 3 seconds
Startup (Neutron count rate)	Prevents withdrawal of all control rods	Less than 2 counts per second
Console scram button	Scram	Manual
Fission counter power supply	Scram	Loss of high voltage
Ion chamber power supply	Scram	Loss of high voltage
Watchdog Timer	Scram	Key software tasks take longer than 1.5 seconds
Magnet current key switch	Scram	Manual
Simultaneous manual withdrawal of two rods$^{\Psi}$	Prevents withdrawal	----------
Withdrawal of shim or regulating rod with safety rod completely withdrawn (or out) $^{\Psi}$	Prevents withdrawal	----------
Withdrawal of safety rod with shim or regulating rod not seated$^{\Psi}$	Prevents withdrawal	----------
Pool level	----------	Alarm when water level is less than 12 ft above top of core
Pool water temperature	Meter indication (Administratively controlled)	35°C
* Uncompensated ion chamber analog system **Fission counter digital system $^{\Psi}$ May be defeated for control rod calibration		

the full scale power calibration is set on the % DEMAND dial of the Regulating Rod Servo panel and the mode switch is turned to the AUTOMATIC position.

The purpose of the Reg rod drive servo is to regulate reactor power to a value set by the operator. The servo controller compares the reactor power with the power demand as set by the operator, develops an error signal, and adjusts the regulating rod position in accordance with the error.

During servo operation, the reactor is period limited to either 30 s or 60 s as determined by the position of the servo-manual switch on the front of the console. A change in power caused either by a demand change or by recovery from a transient can take place on a period no shorter than the value indicated by the switch. The rod drive servo receives the following three types of information:

1. Power demand, from the demand control
2. Reactor power information, from the linear recorder
3. Reactor period information, from the NM-1000.

The reactor power information comes from a retransmitting slide wire on the linear recorder. This signal feeds one leg of a bridge while the power demand signal feeds the other. Bridge output, representing the difference between reactor power and power demand then feeds the servo. The period information from the NM-1000 also feeds the circuit and limits the period that the reactor can be automatically enabled to 30 or 60 seconds. The retransmitting slide wire on the linear recorder is designed to operate from 0 to 100% of the recorder scale. If the recorder is noisy while in auto mode it will cause the reg rod to oscillate. If the recorder goes out the servo will drive the rod up being limited by the preset period and the per cent power scram setting.

Failure of the recorder could lead to inadvertent withdrawal of the control rod. Assuming no reactor operator intervention, the reactor would simply Scram once it reached the facility's percent scram power set point. However, an analysis of such an event is presented in Section 8.1.4.

4.3 CONTROL SYSTEM

The control system is composed of both nuclear and process control equipment and is designed for redundant operation in case of failure or malfunction of components essential to the safe operation of the reactor.

4.3.1 Nuclear Control System

The operation of the reactor is monitored by two separate detector channels. A wide-range fission chamber and a boron-lined uncompensated ion chamber constitute the reactor core monitoring system. These detectors monitor the neutron-flux density of the core and provide trip signals to the safety circuits.

4.3.1.1 Nuclear Instrumentation

This instrumentation provides the operator with the necessary information for proper manipulation of the nuclear controls (Figure 4.1 & 4.2).

(1) The General Atomic NM-1000 Monitoring and Safety Channel is an industrial neutron monitoring system which is used both in research reactors and in nuclear power plants. It utilizes a fission chamber for the neutron detector, pulse processing electronics and a microcomputer to process instrument readings. Output from the microcomputer is routed to an alphanumeric display terminal with date entry and control capabilities. Log and Linear Power can be read on the display terminal and are also displayed on a chart recorder. Reactor period can be read on both the display terminal and also on a bar graph. The linear power recorder is auto ranging and the range is indicated on a bar graph.

(2) The NM-1000 uses a 1.3 counts/s-nV encased fission chamber to provide 10 decades of power indication - from shut down (source) level to full power - hence it is also referred to as a wide-range power monitor. A count rate circuit is used to monitor power for six decades up from source level; the top four decades are monitored by a Cambelling circuit. When neutron flux levels become high enough so that the detector cannot be operated in the count rate mode (power proportional to the pulses from the detector) without excessive pulse pile-up problems, the Cambelling technique is used. This technique consists of electronically deriving a signal which is proportional to the mean square root of the current fluctuations present in the fission chamber.

The amplifier/processor circuit employs designs which perform automatic on-line diagnostics and calibration verification. Detection of unacceptable circuit performance is automatically alarmed. The system is calibrated and appropriate scrams checked prior to operation during the prestart checks. Examples of "unacceptable" circuit performance are listed in Appendix 4A under Stack Errors. These errors can effect the operation of the NM-1000. Failure of stack 1-9 to take longer than 1.5 s. will cause the watchdog timer to be tripped. Internal diagnostics and self tests are performed continuously in the NM-1000, whether the reactor is secured or at power, to insure operation integrity. RAM, ROM and battery backup RAM are continually monitored and tested. The accuracy of the channels is ±3% of full scale; period and high power trip settings are repeatable within 1% of full-scale input. The following are the Performance Specifications of the NM-1000:

Function	Sensitivity (cps/nV)	Range	Linearity (%)
Log/linear	0.2	$2x10^{-8}$ % to 120%	2
Percent power	0.2	1% to 110%	2

(3) A minimum source-neutron count rate interlock from the NM-1000 prevents rod withdrawal unless the measure source level exceeds a predetermined value.

(4) Power level and scram channel no. 2 comprises a separate uncompensated ion chamber, power supply, and power-range adjustment control and meter to indicate power level from

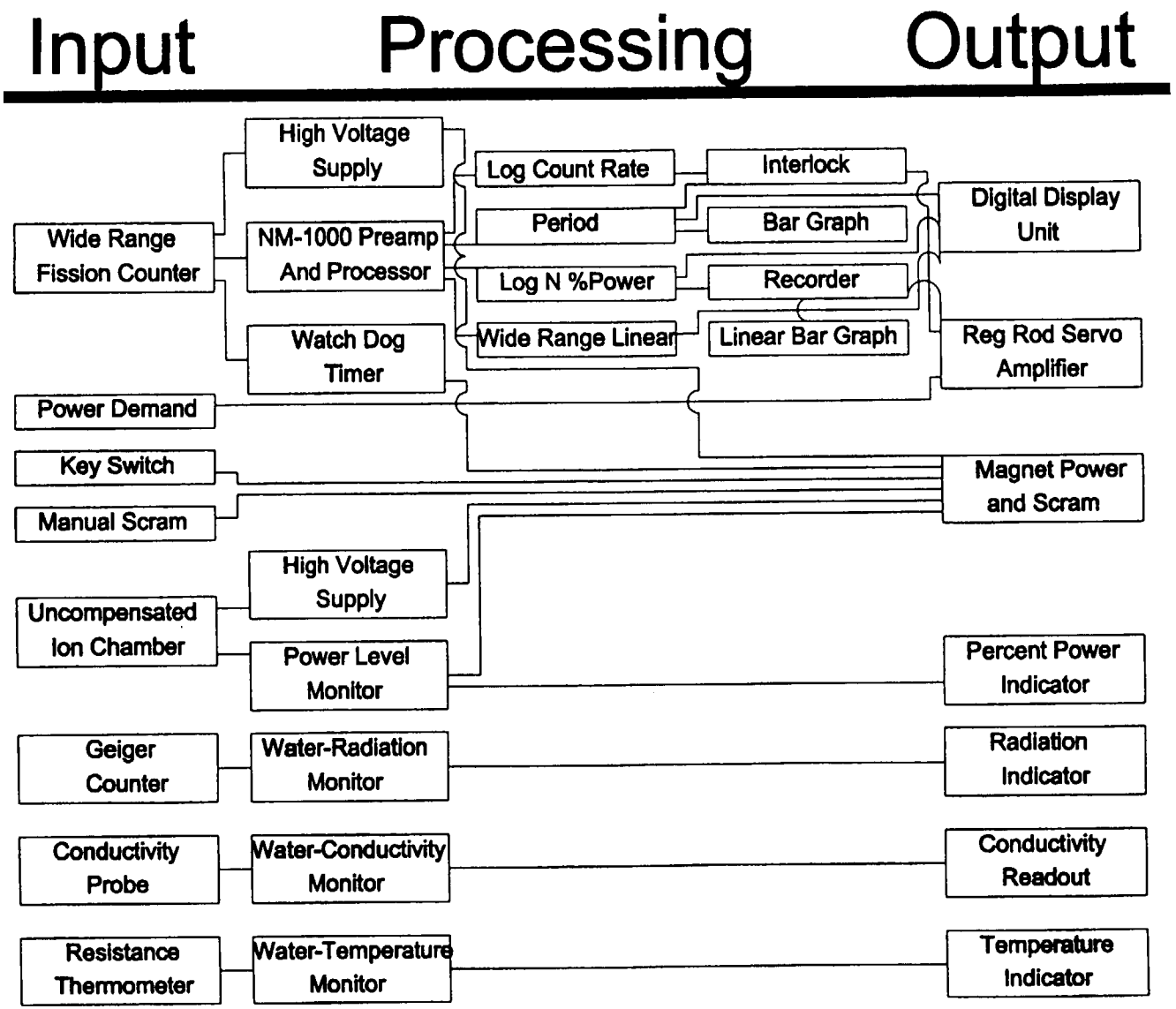
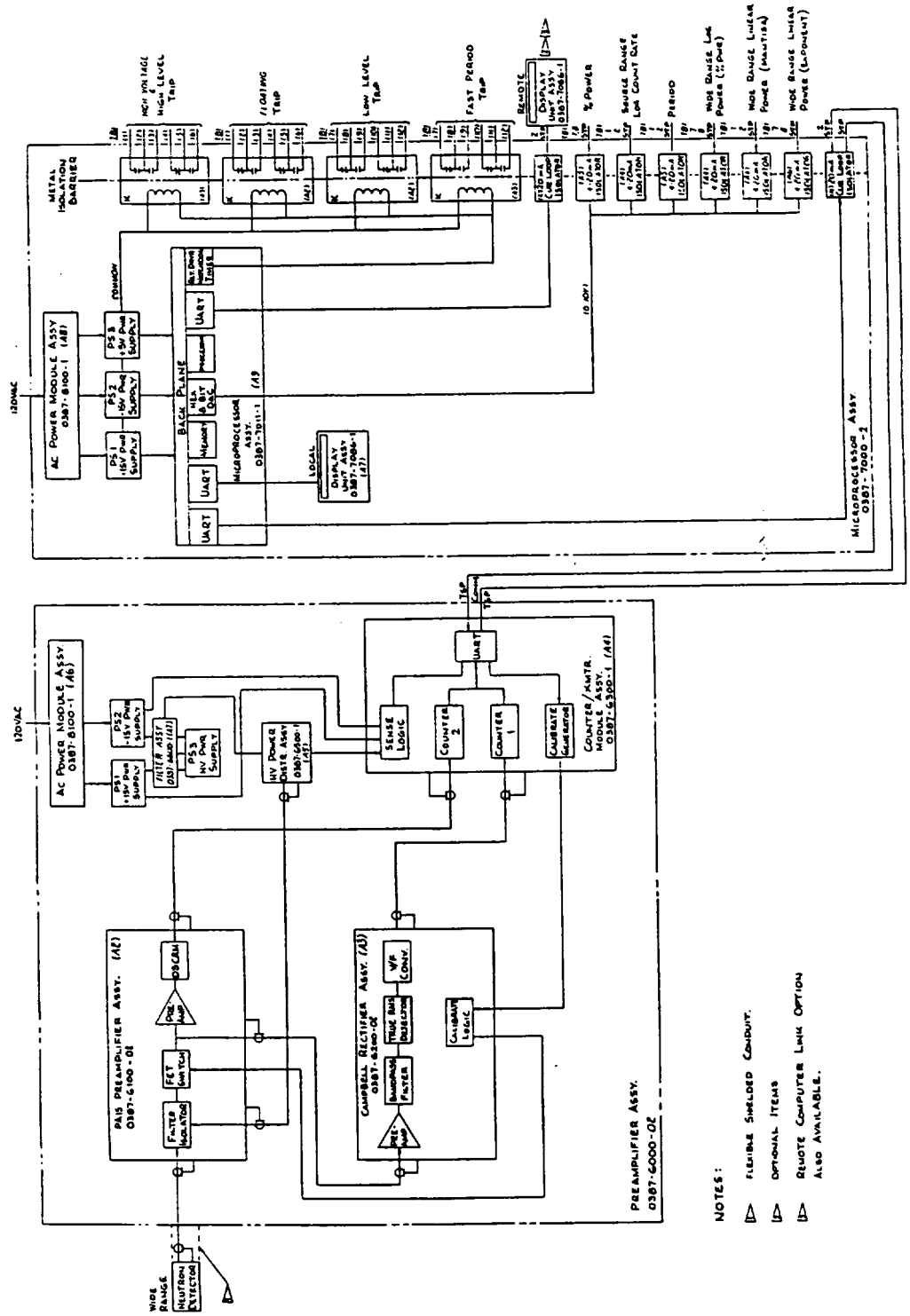


Fig. 4-1 BLOCK DIAGRAM OF INSTRUMENTATION



FUNCTIONAL DIAGRAM NM-1000
Fig. 4-2

0 to 110% of licensed power. Scram level on this channel may be adjusted from 20 to 110% of full power.

(5) The automatic regulating channel consists of a servoamplifier that controls the regulating rod and thus keeps the reactor power level constant. The servoamplifier is activated by an error signal that is governed by the setting of the power-demand control in relation to the actual reactor power level. Because period information also is employed, the servo amplifier may be used to automatically bring the reactor up to power level, within the limits of the worth of the regulating rod, on a preset period of either 30 or 60 s. Automatic changes in power level on these periods are possible. The servo amplifier will allow quick recovery to bring the power level back to within 1% of the original value, even when step changes in reactivity of up to several tenths of 1% of $\delta k/k$ are made.

(6) The two neutron-sensing chambers are hermetically sealed in aluminum or stainless steel cans and mounted on the outside of the reflector so that their positions are vertically adjustable in order to change sensitivity. There is no apparent indication of any rod shadowing or flux density shifts that effect the response of the two neutron detectors. The detectors are individually calibrated on a yearly basis.

4.3.1.2 Reactor Power Safety Channels.

The TRIGA Mark I power safety system is designed to comply with IEEE Standard 379-1977 [1] for single failure and common mode failures. A two-channel system is provided in a one-out-of two trip logic configuration.

One of the two power channels uses the output of an independent uncompensated ion chamber (Westinghouse 6937 or equivalent) and indicates percentage of power in the upper two decades of the power range. This channel is part of the original TRIGA Mark I system and is housed in its own independent enclosure with separate power supply. When a preset power level is reached on the meter a relay is activated in the control chassis causing the scram loop to open.

The second power safety channel is provided by the digital wide range power monitor (NM1000). This channel has been designed to operate as a Class 1E system as a nuclear safety channel for the nuclear power industry [1-4]. The NM1000 neutron monitor design utilizes high speed counting circuits, shielded signal and data communications cables, high speed digital (microprocessor) processing of the signal, and optically isolated output buffers for processing of power data from the fission chamber. To test its response to rapid power changes, the response time of the NM1000 to a sudden change in power (step changes in reactivity) has been measured and compared to the existing analog safety channels on the Mark I Torry Pines reactor by General Atomic [6]. They also directly compared, by measurement, the time required for detection of signals and low level, high level and period scrams of the NM-1000 with the TRIGA analog system. The times were found to be equivalent. Similar to the analog channel of the Per Cent Power channel, the NM1000 trip output is also hardwired (analog) into the scram loop; Thus any overpower condition in the NM1000 will also interrupt magnet current. The NM1000, therefore, also provides complete redundancy for operation as a safety channel with the analog per cent power channel.

The digital power monitor and safety channel (NM1000) uses the standard, well established technique of wide range power monitoring by the use of count rate and Cambelling techniques

to monitor power from source range to full power [7]. However, the processing of the data from the amplifiers is performed digitally, using state-of-the-art, high-speed data processors. The response time of the digitally processed signal for performance of the required safety function has been shown through direct parallel testing by General Atomic to be equivalent, as regards TRIGA safety, to that from the older analog safety system.

A schematic representation of conditions leading to a scram on the TRIGA Mark I reactor is shown in Fig. 4-3.

4.3.1.3 Internal Diagnostics

Internal diagnostics and self-tests within the NM-1000 are performed to ensure NM-1000 operation integrity. RAM, ROM and battery backed-up RAM (BBRAM) are continually monitored and tested. The NM-1000 hardware is also equipped with a watchdog timer that will reset the NM-1000 software if it is not reset periodically. In other words, the timer is in the software loop and receives a "keep alive" signal from the software. If the timer does not receive its "keep alive" signal within its preset time the computer is rebooted. Its purpose is to prevent the NM-1000 software from failing and not performing its power monitoring function while giving false results. It is reset every 16.7 ms while the NM-1000 communication is in sync with the counter/transmitter and the task level software is executing. Any failure will be indicated on the microterminal and a failure of the watchdog timer will initiate a scram.

The watchdog system will cause a scram if the Mini-Executive Overrun timer takes longer than 1.5 s. The timer ensures that the Mini-Executive software has been fully executed. The Mini-Executive Processing software performs the following functions: (1) Local terminal I/O processing; (2) remote console I/O processing; (3) Diagnostics/alarm processing; and (4) the executive overrun timer. The 1.5 s was selected because when the NM1000 is being used in the TRIGA reactor pulsing system the computer is dedicated to gathering other data and the 1.5 seconds is a designated value. During development, when all the code had been completed this value was used to allow all subroutines to update the programmed data and thus prevent spurious trips during pulsing.

4.3.1.4 Additional Information

Additional information requested by the NRC with reference to our request for Facility License Amendment on October 15, 1990 may be found in Appendix 4A. This information pertains to the NM-1000 system and covers the following topics:

(1) Verification and validation plan for GA Model NM 1000 Neutron Monitoring System.
(2) Loss of high voltage to neutron detector scram function.
(3) Location and installation configuration for the new instrumentation and control system.
(4) Maintenance and surveillance program.
(5) Operator Training for the NM 1000.

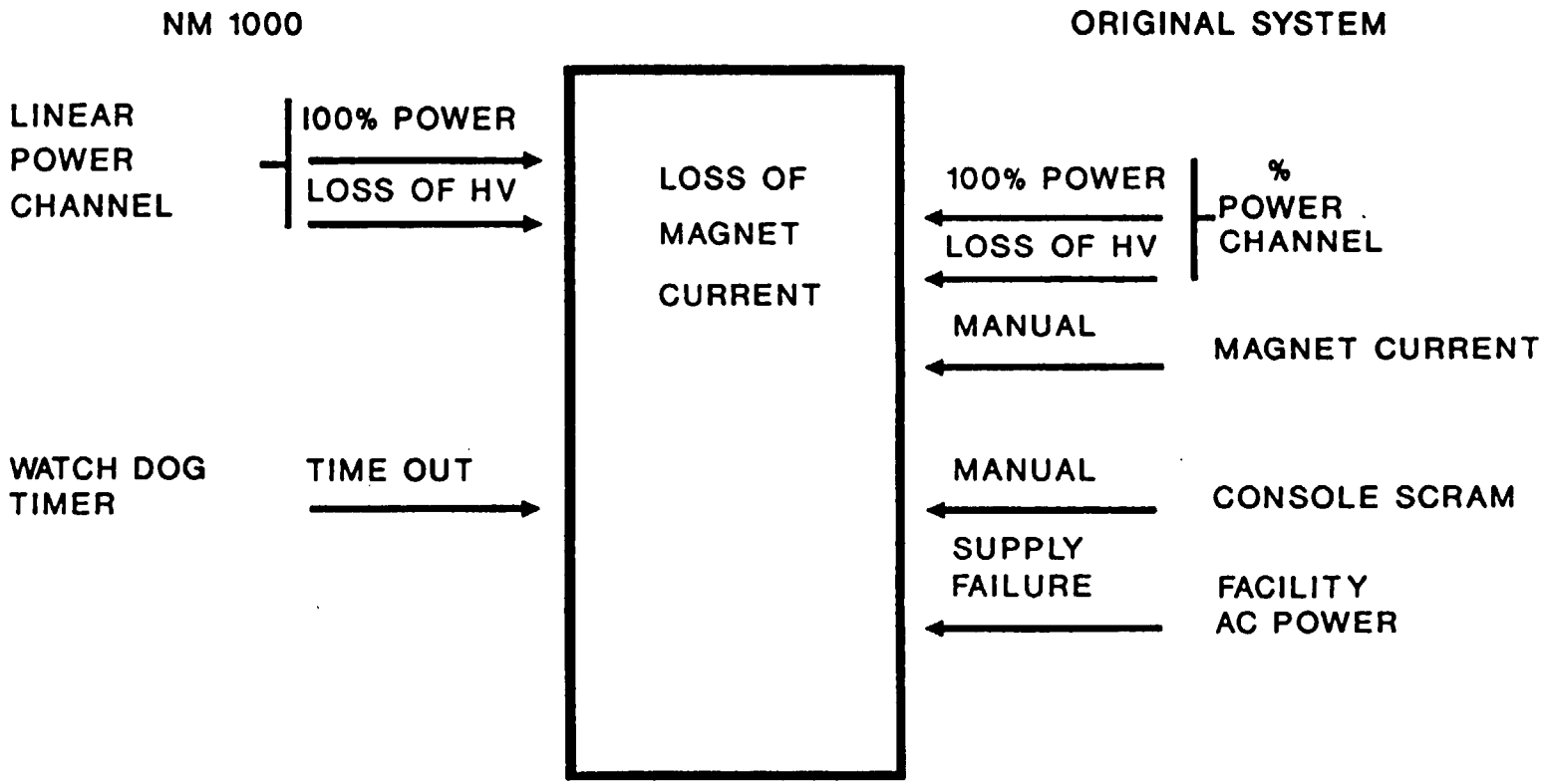


Fig. 4.3 SCHEMATIC REPRESENTATION OF CONDITIONS LEADING TO A SCRAM ON THE TRIGA MARK I REACTOR

(6) Hardwiring of NM 1000 trip output to TRIGA Control Unit.
(7) Description of Calibration Procedures

4.3.2 Process Instrumentation

This instrumentation is used for (1) sensing and monitoring parameters associated with the pool water and (2) radiation monitoring.

(1) The water-radioactivity monitor comprises a gamma-radiation detector and a count-rate-meter circuit that gives both audible and visible alarms if the gamma activity in the pool water reaches a preset value. The water monitor is calibrated so that when it is removed from the box (Fig. 3.8) and exposed to a 100 mrem/hr field, a full scale reading on the meter is equivalent to 0.1 $\mu Ci/cm^3$ of 10-min-old fission products in the water system. The alarm is set for 100% of this value. Based on calculations provided in Appendix B, the maximum activity available for release to the coolant per element is 0.07 Ci. Since General Atomic sized the water box to give 100 mrem/h for a 0.1 $\mu Ci/cm^3$ solution of 5 minute old fission products, the alarm would only sound if more than 25 elements failed. The water box monitor will not alarm for a single fuel failure. However, the increased readings should be noticeable. Since initial startup (1959) the meter has not indicated over 2% of full scale. The water monitor reading is recorded on the daily check list before each startup.

(2) The water-conductivity monitor consists of a conductivity probe and Wheatstone bridge circuit. Daily check list measurements of the conductivity are made to ensure that neutron activation of pool water impurities will be small and that chemical corrosion of fuel cladding is limited. Since initial startup in 1959, maintaining conductivity below 5 μmhos per cm, no significant radiological hazards in the coolant have been observed. In fact, the primary coolant has always remained within 10CFR20 Appendix B limits for discharge to sanitary sewers.

(3) The water-temperature monitor consists of a resistance-bulb thermometer that senses the bulk pool temperature. Temperature indication is provided on the control console while the thermistor is within the water monitor (see fig. 3.8). This system is required to be operational whenever the reactor is in operation. The reactor is administratively shut down if the temperature exceeds 35°C which is the recommended upper limit for the ion exchanger.

(4) The water-level monitor consists of a float-switch and associated circuitry. This provides both an audible and visual alarm if the water level is less than 3.6 m above the top of the core. An audible and visual alarm is also triggered at the medical center switchboard which is occupied 24 hours a day.

(5) The facility also has the following radiation monitors which are in operation whenever the reactor is operating:

a. A calibrated, nonjamming gamma-ray monitor with an audible alarm. This monitor is position a short distance from the isotope removable tube and has an alarm set point of 2 mrem per hour.

b. A calibrated, continuous airborne radiation monitor (CAM) located in the reactor laboratory near the top of the reactor. There are currently 2 CAM's (NMC AM2D & Eberline AMS 3A) available for operation. The monitor can detect radioactive airborne particulates but is unable to detect noble gasses, argon-41, or iodine-131 directly. As seen in Appendix A and B, the levels of these nuclides during normal operation

and during a Maximum Hypothetical Accident (MHA) are below levels specified in 10CFR20 Appendix B. Gaseous releases from failed experiments are minimized by up front calculations and Technical Specification restrictions. Experiments run in the reactor are limited as described in SAR, Management Surveillance, Section 7.2.3. Thus 10CFR20 Appendix B levels will not be approached from experiments because they are limited administratively ahead of time.

The NMC monitor also contains a charcoal filter to provide the capability of collection of radioiodine. The use of a charcoal filter is not required for reactor operation. Assay for radioiodine can be accomplished by removing the filter and counting on a Ge well detector and a multichannel analyzer.

4.3.3 Electrical Power System

The electrical power requirements of the reactor facility are supplied by three circuits from the medical center electrical distribution system. The reactor facility has no emergency electrical power system except for two battery-powered lanterns that activate when the building power fails. In the event of loss of electrical power, the control rods are released to fall into the core by gravity, causing safe shutdown of the reactor.

Chapter 4

References

1. "Application of the Single Failure Criterion to Nuclear Power Generating Class 1E Systems", Institute of Electrical and Electronic Engineers Standard IEEE 379-1977.

2. "Criteria for Protection Systems for Nuclear Power Generating Stations", Institute of Electrical and Electronic Engineers Standard IEEE 279-1971.

3. "Digital Neutron Monitor NM-1000", GA Technologies Inc. Document INS-25, GA Proprietary Information. 1983.

4. "Qualification Test Report: NM-1000 Digital Neutron Monitoring System", GA Document E-269-1239, GA Technologies Proprietary Information. May 1984.

5. J. Razvi and W.K. Hyde "Results of Timing Tests on NM-1000 High Power Scrams", General Atomics Internal Correspondence. May 1988.

6. "Radiation Detection and Measurement", Chapter 14, John Wiley Sons. 1979.

Additional Information Submitted for the Amendment
of Facility License No. R-57
Omaha Department of Veterans Affairs Medical Center

I. Verification and validation plan for GA Model NM 1000 Neutron Monitoring System.

A. The complete reactor control console will not be replaced, only the neutron monitoring system for the linear, log and period. The power supply and pre-amplified (A) and computer modules (B) are mounted on the wall behind the console (see Fig 1). The Burr-Brown TM 76 Microterminal © and the Period (D) and Power (E) Bargraphs are located on the original console (Fig. 1). When changeover occurs the new Westronic Recorder (F) will replace the old recorder presently installed in the console. Terminal Display items for the Microterminal are described in Attachment 1.

B. Scram Verification - "Item" notation refers to pressing key on microterminal (See Attachment 1 for Reference)

1. Individually test the trip relays in the NM 1000 with a meter while they are isolated from the TRIGA console to assure that they are operating properly while they are put through the tests outlined below. Relay connections shown in Fig. 2. Location of cards shown in Figures 3 & 4.

a. Power Level Trip. Relay-Board A3, Pins 8&9.
   (1) Hi level - push "Power Scram Test" button on console.
      (a) Hi Level Trip.
      (b) Item 41, Press F8 key (this puts you in Data Entry Mode)
      (c) Enter 1.0E+02 and then press enter key. This sets scram at 100%.
      (d) Verity that A2 light is on.
      (e) Item 15 - Verify that read out shows H for High.

b. Period Trip. Relay-Board A2, Pins 8&9.
   (1) Item 43, Press F8 key (this puts you into Data Entry Mode).
   (2) Enter 7 for 7 seconds and then press enter key. Once this value is entered it should not be changed.
   (3) Item 50, Press F8 key, Enter 5 (Campbelling High Test).
   (4) The above should cause a momentary high positive period and activate the period trip.
   (5) Verify that light A2 is on and Item 15 show a read out of R for rate.

c. High Voltage Trip. (Loss of High Voltage) Relay Board A3, Pins 8&9.
   (1) Push "High Voltage Test" button on console.
   (2) Verify that A2 light is on.
   (3) Item 60, Verify that Burr-Brown readout shows 10/ CXHIV.
   (4) Item 61-69 until readout shows empty (this assures that there are no other errors.

d. Startup Channel (Low Level Trip) Relay Board A3, Pins 2&3.
   (1) Item 40, Press F8 key (this puts you into "Data Entry Mode")
   (2) Enter 3.74E-07. This corresponds to 2 counts/sec (1.87E-7 = 1 sec).
   (3) Item 10.

(4) Remove neutron source.

(5) Verify that light A2 goes on when Burr-Brown readout shows 3.7E-7 and Item 15 shows a read out of L for low level trip.

e. Watchdog Timer Relay Board A3, Pins 8&9.

(1) Disconnect plug labeled A3 CTX (card in position 4 in microprocessor card box).

(2) Green light on IO and Memory Card (Card in position 9 in microprocessor card box) should go off and yellow light on.

(3) Verify that light A1 is on and Item 60 shows O2 CXFAIL.

2. When the operation of all scrams have been verified with the unit disconnected from the existing system, connect the relays in the NM 1000 microprocessor assembly to pug 3B in the TRIGA Console as show in Fig. 2 and raise control rod for each test.

a. Repeat the procedures outlined in Paragraph I.B.1 above without voltmeter, since control rod will drop signifying operation of relay.

b. Do not proceed with installation unless all scrams are operating properly.

C. The NM 1000 Neutron Monitoring System has been installed in parallel with our existing system since October 1989, (scram relays not connected and reactor completely controlled with our existing licensed system) and the only major problem that we have had was a noisy fission chamber resulting in too high of a count rate when the neutron source was removed. This problem was resolved by replacing the high voltage power supply and assuring that the system was adequately grounded. Critical calibration values as described in paragraph IV below have been recorded each time the reactor is operated since October 1989, and all values have varied less than 5% of the configured values.

D. Scram Response Time

1. Since the scram relays in the NM 1000 are connected to the original TRIGA Mark I control console, the response time testing will utilize the same procedure as previously used.

2. Raise a control rod and measure the scram time with a stop watch. Compare the scram time with the scram time previously determined before the NM 1000 was connected.

3. In accordance with Paragraph 3.3.1 of our Technical Specifications, "The maximum scram time for any fully withdrawn rod shall be 2 seconds from the time of initiation of scram signal to full insertion of the rod."

E. Sensitivity of Detector (Calibration)

1. Before replacing the neutron monitoring system in the old console the following procedure will be followed: (The complete procedure described below was done on August 13, 1990, and all values have agreed since then. However, the procedure will be repeated prior to switching over systems with the exception that the thermal calibration will be done after we have removed the linear ion chamber from its aluminum tube guide that is attached to the ion chamber mounting ring and inserted the new fission counter).

a. Place the new Reuter Stokes fission chamber as close to the original linear compensated ion chamber as possible.

b. Align the fission chamber as outlined in Attachment 2.

c. Thermally power calibrate the reactor as per existing SOP and axially move the linear, log and per cent power ion chambers so that their output devices read the calibrated power (linear and log on old recorder).

d. Axially move the new fission counter so that the new linear recorder reads the calibrated per cent power.
e. Verify that the new log recorder is also reading the calibrated value.
f. Verify that the ion chamber readings compare with the fission chamber readings.

II. Loss of high voltage to neutron detector scram function.
A. Original System - Per Cent Power Chamber
1. The original per cent power ion chamber and scram circuit will be used as our second detector and consequently there has been no change made in the loss of high voltage scram function.
B. New NM 1000 system.
1. Loss of High Voltage to the new fission counter activates Relay A3 in the NM 1000 assembly which is hard-wired to the per cent power scram in the TRIGA console (Fig. 2). The system then scrams as in the original system.

III. Location and installation configuration for the new instrumentation and control system - See Fig.1.
A. Specification of the temperature and humidity conditions of the system.
1. The NM 1000 was tested to the following extreme conditions and found to operate satisfactorily.
a. Temperature: 0-60oC.
b. Relative humidity: 0-98%
2. The Omaha V.A. TRIGA is installed in an air-conditioned humidity controlled room.
B. Evaluation of enclosures, cabinets and connections to building structures for general ruggedness under potential dynamic conditions.
1. System was tested to meet the requirements of IEEE 344-1975.
2. Preamplifier and microprocessor assembly enclosures are mounted on basement cement wall (Fig. 1, A & B) and hard wiring to TRIGA console is done through conduit (Fig. 1, G).
C. Evaluation of potential contact chatter during dynamic conditions.
1. All relays are normally energized and do not chatter under postulated seismic acceleration.
2. The system is designed to scram on potential contact chatter conditions.
D. Evaluation of cable and component shielding, configuration and/or isolation to mitigate the consequences of electro-magnetic interference (EMI).
1. Signal outputs are either OPTO or transformer isolated, inter-connection is via twisted shielded pairs.
2. EMI levels sufficient to cause a response would cause a transient upscale response. If transient upscale response exceeded 100%, scram would occur.
E. Evaluation of power supply buffers to mitigate power transient effects.
1. Power supplies are electronically regulated and input power is buffered by a shielded passive line filter followed by a shielded active tracking filter. All are enclosed within a steel NEMA 1 enclosure.
2. After a power loss the surveillance program described in paragraph 4 of this document will be repeated. This is the same program used each day before starting up the reactor.
F. Evaluation of instrument isolation devices.

1. For the analog outputs isolation is provided by an "Analog Devices" isolation converter. Analog Devices rates the input to output isolation at 1500 V RMS. The device meets the IEEE Standard for Transient Voltage Protection (472-1974: Surge Withstand Capability) and offers reliable operation over -25o to +85oC temperature range.
2. Trip outputs are provided by relay contacts. The isolation ratings are not supplied by the manufacturer.
3. Isolation for communication is provided by optical isolation. General Atomic has tested the isolation of these optical isolators to 120V A.C.
4. Mechanical isolation is provided at the field termination points for all safety and non safety inputs and outputs

IV. Maintenance and surveillance program.

A. At the start of each working day or after each major interruption of operation, the reactor electrical and mechanical systems shall be checked out and certified to being proper working order, in accordance with the check list shown in our existing SOP.

B. When the NM 1000 is installed the Chamber and Instrument Sensitivity section of the SOP will be replaced with the checks described in Attachment 3.

C. Maintenance will be performed on any of the items in the Daily Checklist which cannot be verified; so that the facility is in compliance with the current Technical Specifications.

V. Operator training for the new system (outline listed below).

A. Description and theory of fission counter.
   1. Discrete neutron counting techniques.
   2. Campbelling techniques.

B. General Description of NM 1000.
   1. Physical Description.
   2. Performance specifications.
   3. Amplifier Assembly.
   4. Signal Process Assembly.
   5. Installation and Setup - Calibration.

C. Functional Description.
   1. General.
   2. Source-Range Log Count Rate.
   3. Wide-Range Log Power.
   4. Power Range.
   5. Multirange Linear Power.

D. System Description.
   1. Hardware.
   2. Software.

E. NM 1000 Software Description.
   1. Hardware/Software Description.
   2. NM 1000 System Function.
   3. Software Organization:
      a. CPU Reset.
      b. Counter/Transmitter Message Character Received.
      c. Local Display Input Character Received.
      d. Local Display Output Character Sent.
   4. Database Organization.

a. Database Item Description.
b. Error Description.
c. Daily Checklist.

VI. Hardwiring of NM 100 trip output to TRIGA Control Unit.

A. The relay outputs of the NM 1000 as described in paragraph I B above are connected to plug 3B of the Control Unit of the TRIGA Console. Connections are shown in Fig. 2 and Fig. 7-2 attached. (Please substitute the enclosed Fig. 7-2 for our original submission.)

B. Connection is through conduit G, Fig. 1.

VII. Figure 7.2 (see revised Fig. attached) Replace Fig. 7-2 submitted October 1989.

A. Connections of V/F out on J2A1 to counter transmitter J1A1 (drawing 0387 60820) to E3 to U1-4 to U1-9 to U1-2 of counter 1 (drawing 0387 60820). Drawings are in GA Operations and Maintenance Manual E117-1000.

1. The connection between the V/F Convertor and Counter 1 has been made on revised Fig. 7-2.

B. We do not have remote display unit so it has been deleted on the diagram.

C. The calibration generator uses the summation of the outputs of a multi-frequency digital clock to produce a pseudo-square wave in the Campbelling region. The calibration values are adjustable and stability is determined by power supply and passive component drift. In count rate mode discrete frequencies are counted and stability is determined by the clock crystal. Function switching is performed by transistor switches controlled by the counter transmitter which is in turn software controlled. If a calibration function is selected when at power a rod withdrawal prohibit function operates, except in high Campbell calibration which causes a scram.

VIII. Proposed Technical Specification Changes - Watchdog Timer.

A. See revision to Table 3-1 (page 7) and page 8 (Attachment 4 & 5).

B. See revision to page 7-6 (Table 7.1) of proposed Amendment No. 1, SER (Attachment 6).

C. Please substitute the enclosed Attachments 4, 5 and 6 for the corresponding pages submitted in October 1990.

IX. Minimum count rate rod withdrawal interlock.

A. The minimum count rate withdrawal interlock was set for 10 counts per second in our October 1990 request due to the fact that the noise level of the detector was high. However, since the original request for change was submitted the noise has been eliminated. Consequently, we request that 10 counts per second be deleted and replaced with the original licensed limit of 2 counts per second (Attachments 4 and 6).

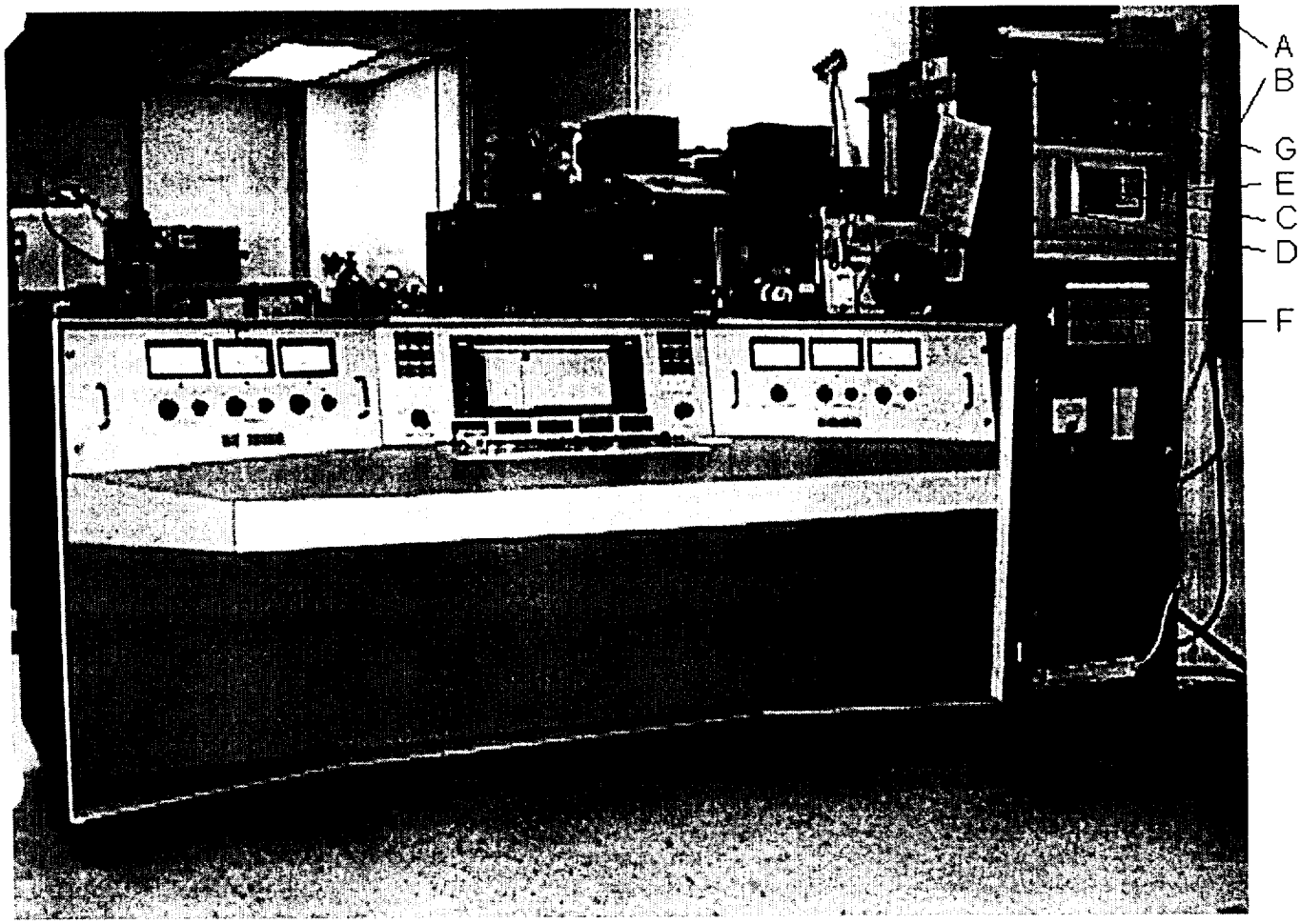


Fig 4-4 GA REACTOR CONSOLE

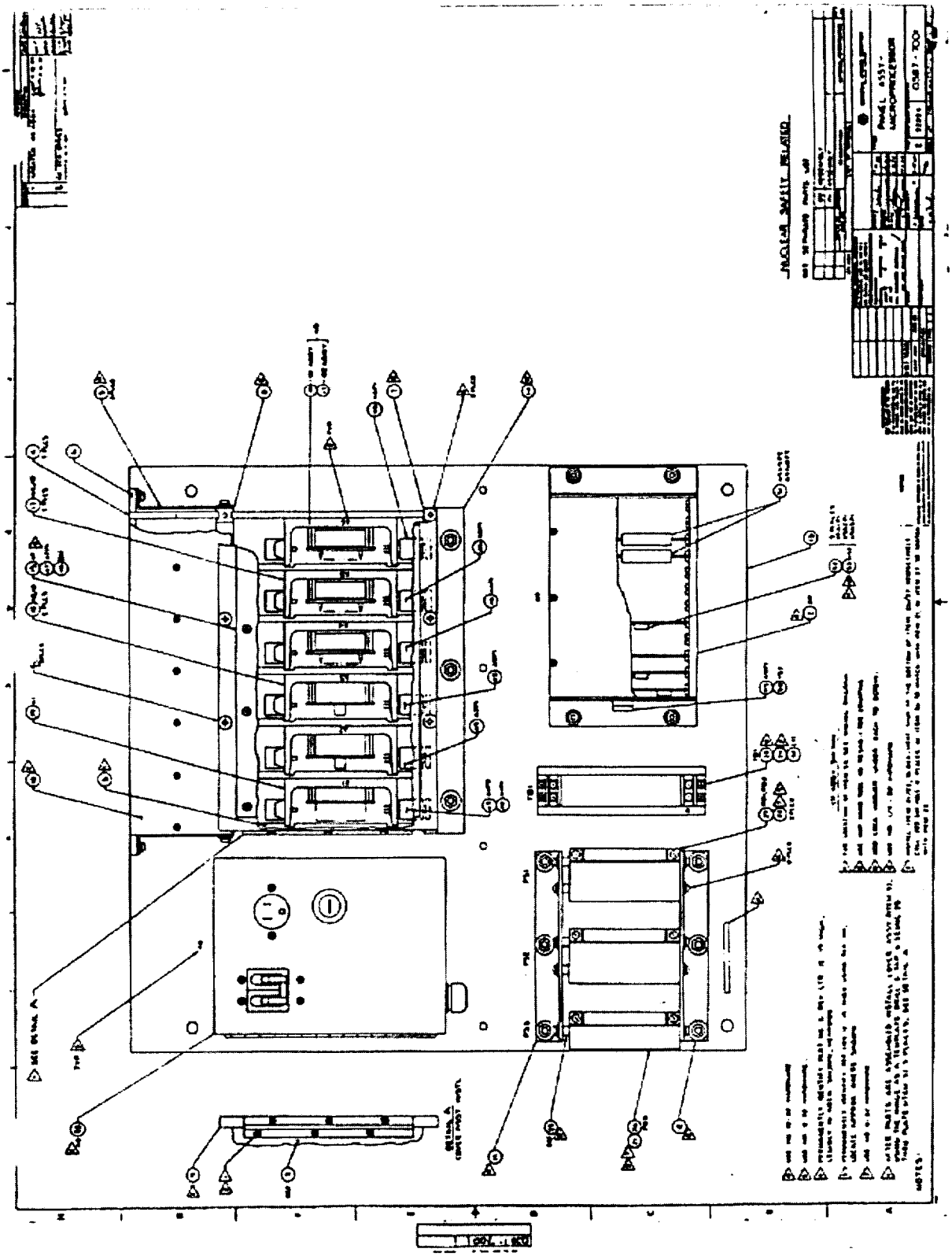


Fig 4-5 NM-1000 SCHEMATIC

TERMINAL DISPLAY ITEM

GROUP 1 (F1 KEY) — COMPUTER VALUES

	10= PERCENT POWER	(S)
	11= PERCENT POWER	(F)
	12= PERIOD	(S)
	13= PERIOD	(F)
	14= MODE	(I)
@	15= RELAY STATUSES	(I)
	16=	
@	17= 200MS MESSAGE	(A)
	18= %POWERL - MANTAS	(S)
	19= %POWER - EXPONT	(S)

GROUP 2 (F2 KEY) — SINGLE DETECTOR

	20= DET COUNTS	(S)
	21= ALPHA OFFSET	(S)
@	22= (20) - (21)	(S)
	23=	
	24=	
	25= DET PPCONST	(S)
	26=	
	27=	(S)
@	28= DET XOVR VAL	(S)
	29= DET XOVR SETP	(S)

GROUP 3 (F3 KEY) — CAMPBELL DETECTOR

	30= CMB COUNTS	(S)
	31= NOISE OFFSET	(S)
@	32= ((30)-(31))**2	(S)
	33= CMB MULTIPLIER	(S)
@	34= (33) * (32)	(S)
	35= CMB DET PPCONST	(S)
	36=	
	37=	
@	38= CMB XOVR VAL	(S)
	39= CMB XOVR SETP	(S)

GROUP 4 (F4 KEY) — TRIP SETPOINTS

	40= LOW LVL TRIP	(S)
	41= HI LVL TRIP	(S)
	42= FLOAT LVL TRIP	(S)
	43=RATE TRIP	(S)
	44=	
	45=	
	46=	
	47=	
	48=	
	49=	

GROUP 5 (F5 KEY) — MODES OF OPERATION

	50= OPERATIO MODE	(B)
	51= FLT TRIP MODE	(B)
*	52= M-LINEAR MODE	(B)
*	53= LOCKED EXP	(I)
*	54= PERIOD DAC HI	(B)
	55=	
	56=	
	57= MEMORY ADDRESS	(I)
*	58= MEMORY VALUE	(B)
	59= VERSION NUMBER	(A)

GROUP 6 (F6 KEY) — ERROR STACK

	60= ----	
	61= STK POSITION 1	(B)
	62=STK POSITION 2	(B)
	63= STK POSITION 3	(B)
	64= STK POSITION 4	(B)
	65= STK POSITION 5	(B)
	66= STK POSITION 6	(B)
	67= STK POSITION 7	(B)
	68= STK POSITION 8	(B)
	69= STK POSITION 9	(B)

A1 LIGHT DEFINITIONS

ON = EQUIPMENT FAILURE (ERR IN STACK)
OFF = NO EQUIPMENT FAILURES

A2 LIGHT DEFINITIONS

ON = RATE OF CHANGE TRIP, HIGH LVL TRIP, OR LOSS OF HIGH VOLTAGE
OFF = NONE OF THE ABOVE STATUSES

STEPS TO READ OUT A TERMINAL DISPLAY ITEM:

1. SELECT GROUP BY PRESSING THE FOLLOWING:
   F1 KEY = GROUP 1 (ITEM 1 WILL ALWAYS BE DISPLAYED)

F2 KEY = GROUP 2
.....ETC.....

2. GO TO ANY ITEM IN GROUP BY PRESSING THE NUMBER KEYS (0-9)
   OR -----> PRESS "." TO STEP FORWARD TO NEXT ITEM
   OR -----> PRESS "-" TO STEP BACKWARDS TO LAST ITEM.

TO CLEAR ALL ALARMS, PRESS F7 KEY, THEN ENTER CODE 90.

DATA ENTRY STEPS:
(*=LOCAL ONLY)
(%=REMOTE ONLY)
(@=DISPLAY ONLY)

1. SELECT TERMINAL DISPLAY ITEM TO BE CHANGED.
2. PRESS F8 KEY (THIS PUTS YOU INTO DATA ENTRY MODE).
3. ENTER NEW VALUE FOR ITEM, THEN PRESS "ENTER" KEY.

		in single range		in campbelling range
Power Equation:	POWER =	(ITEM22)*ITEM25	=	(ITEM34)*ITEM35

(A) = ASCII (B) = BYTE DATA (F) = FIXED (I) = INTEGER (S) = SCIENTIFIC

OPERATING MODES	DESCRIPTION (Control byte bits 7 6 5 4 3 2 1 0)	
0 = NORMAL	Normal Operation, No self test	1 1 X X X X 0 0
1 = CTR LOW	Counter Low Test	1 0 X X X X 0 1
2 = CTR MID	Counter Middle Test	1 0 X X X X 1 0
3 = CTR HI	Counter High Test	1 0 X X X X 1 1
4 = CMB LOW	Campbelling Low Test	0 1 X X X X 0 1
5 = CMB HI	Campbelling High Test	0 0 X X X X 0 1
6 = SQUARE WAVE	Square Wave	* Automatically *
7 = PULSE	Reactor Pulse	**** Cycled ****
		(X means don't care)

Operation Mode 6 causes Rate of Change trip to be inhibited for 10 seconds. After 10 seconds, the NM-1000 automatically switches to Operation Mode 0.

Operation Mode 7 causes the NM-1000 to internally sequence through Operation Mode 4 for 10 seconds, Operation Mode 0 for 10 seconds, followed by a switch to Operation Mode 0. All trip statuses are latched upon entry to Operation Mode 7 and unlatched upon completion of the timed Operation Mode 0.

STACK ERRORS	DESCRIPTION
00 = EMPTY	Stack position is empty (i.e., no error)
01 = BADER	BAD Error (posting procedure detected a bad error code)
02 = CXFAIL	Ctr/Xmt Failure (No input to Micro from Ctr/Xmt Assembly)
03 = CXSYN1	Ctr/Xmt Out-of-Sync (detected by the SCAN program)
04 = CXSYN2	Ctr/Xmt Out-of-Sync (detected by the PROCES program)
05 = CXBUSY	Ctr/Xmt Busy Error (Output Uart is not ready)
06 = CXCBE	Ctr/Xmt Rcv Uart Error (No Control Byte Received)
07 = CXCOMM	Ctr/Xmt Rcv Uart Error (Parity, Framing or Overrun)
08 = CX-15V	Ctr/Xmt Assembly; Failure in -15V power supply
09 = CX+15V	Ctr/Xmt Assembly; Failure in +15V power supply
10 = CMHIV	Ctr/Xmt Assembly; Failure in High Voltage power supply
11 = CXHARD	Ctr/Xmt Hardware Error (status bits 7, 6 or 5 is low)
12 = MI-15V	Microprocessor Assembly -15V failure
13 = MV+15V	Microprocessor Assembly +15V failure
14 = SDXOVR	(Item 28 > Item 29) and (Item 38 < Item 39)
15 = SOERR	Single Detector Offset Err (Item 20 - Item 21) < 0
16 = COERR	Camb Detector Offset Err (Item 30 - Item 31) < 0
17 = BADRAM	BAD RAM error (hardware error, replace RAM chips)
18 = BADROM	BAD ROM error (hardware error, replace ROM board)
19 = CORUN	Computing Task Overrun (TASK 1 took more than 0.184 seconds)
20 = EORUN	Executive Overrun (all TASKS took longer than 1.5 seconds)
21 = IORUN	Interrupt Overrun (TASK 1 & 9 not run within 1.5 seconds)
22 = T1FAIL	TASK 1 FAILURE (took more than 20 milliseconds)
23 = DBCHG	Data Base Change Error (change in battery bakced up RAM)

POWER LEVELS			DESCRIPTION - ITEM 14	
0	=	CAMP	Campbelling Operating Range	(using only Camp. Signal)
1	=	SINGLE	Single Detector Range	(using Single Detector)

POWER RANGE SWITCH OVER POINTS

1)	SINGLE	--> CAMP	ITEM28 > ITEM29
	CAMP	--> SINGLE	ITEM38 < ITEM39

NM-1000 NEUTRON MONITOR COMPUTED VALUE EQUATIONS

1) ITEM10 & ITEM 11 - PERCENT POWER

ITEM14 = 1, SINGLE DETECTOR

ITEM10 = ITEM22 (COUNTS PER SECOND) * ITEM 25 (SDET PERCENT POWER CONSTANT)

ITEM14 = 0, CAMPBELLING DETECTOR

ITEM10 = ITEM34 (COUNTS PER SECOND) * ITEM35 (CDET PERCENT POWER CONSTANT)

2) ITEM12 & ITEM 13 - RATE OF CHANGE

ITEM12 = LOG (ITEM10 [CURRENT PERCENT POWER] - ITEM10 [PERCENT POWER 1 SECOND AGO]) * 60.0 / 26.05767

3) ITEM28 - SINGLE DETECTOR CROSSOVER VALUE

ITEM28 = ITEM20

4) ITEM38 - CAMPBELLING CROSSOVER VALUE

ITEM38 = ITEM 30

MULTI-LINEAR MODES	DESCRIPTION
0 = AUTO	Continuous tracking over all decades.
1 = MANUAL	Locked onto a specified decade.

Note: Use display item 52 to set the above multi-linear mode.
Also, display item 50 is used to set the operation modes.

RELAY	DEFINITION	WHEN ACTIVE
1	FLOATING PERCENT POWER TRIP	FOLLOWS CONDITION
2	RATE-OF-CHANGE TRIP	FOLLOWS CONDITION
3	LOW LEVEL PERCENT POWER TRIP	FOLLOWS CONDITION
4	HIGH LEVEL PERCENT POWER TRIP (OR) HIGH VOLTAGE FAILURE	FOLLOWS EITHER CONDITION
5*	ANY PROCESS FAILURE (TURN ON A2 LIGHT)	PROCESS FAILURES ARE: 1. RATE-OF-CHANGE 2. LOW LEVEL PERCENT POWER 3. HIGH LEVEL PERCENT POWER
6*	UNDEFINED	NEVER

*NOTE: CURRENTLY, THE TRIGA HARDWARE DOES NOT SUPPORT THESE RELAYS.

THE A1 LIGHT WILL REMAIN ON UNTIL A CLEAR CODE 90 IS ENTERED.
THE A2 LIGHT WILL REMAIN ON UNTIL THE F7 KEY IS PRESSED.

Alignment of NM-1000 Neutron Monitors

Theory: The NM-1000 neutron monitor is capable of measuring ten decades of neutron flux with a single fission chamber. Alignment of channel requires a basic understanding of the software operation of the NM - 1000, which is detailed below.

The NM-1000 uses two techniques to calculate reactor power. For low power operation the channel calculates reactor power utilizing counting techniques where discrete neutron counts from the fission chamber are directly proportional to reactor power. For high power operation, the channel calculates reactor power utilizing Campbelling techniques where the reactor power is proportional to the square of the rms value of the a.c. signal from the fission chamber. Combining these techniques, with sufficient overlap, allows the NM-1000 to cover a full ten decades.

To calculate the reactor power, the following two equations are used by the NM-1000.

Count Rate Region: (Equation 1)

Percent Power - [Counts/Sec}*{Count Rate Power Constant}
ITEM 10 - ITEM 20 * ITEM 25

Campbell Region:

Campbell Linearizing Campbell
Percent power - [Counts/Sec]2 * [Factor] * [Power Constant]
ITEM 10 - [ITEM 30]2 * ITEM 33 * ITEM 35

The following procedure details the complete calibration of the NM-1000 channel. For routine recalibration, follow steps 3.2, 4.2, 5.1 and 6.1

Method: The first step in the NM-1000 alignment is to properly position the fission chamber in the reactor core. The detector is a standard General Atomics supplied RSN-314 Reuter Stokes fission chamber.

1.1) The detector should be positioned to draw 1.0 mA from the high voltage power supply (800 volts nominal) at 100% power. Next, the PA-I5 preamp discriminator should be adjusted. The number of shutdown counts (with the start-up source in a cold core) will depend on the reactors license power, and will be a function of the crossover from count rate to Campbell.

The cross over point from count rate to Campbell should be set about three decades down from the full power flux (about 0.1% power). This gives three full decades of Campbell signal with adequate hysteresis for the crossover from Campbell to count rate.

2.1) To set the discriminator, change the count rate to Campbell crossover in the NM-1000 software to 8X106 (ITEM 29). Bring the reactor to 0.1% power. Adjust the discriminator (R304 in the PA-15) for 1.2X106 counts per second as displayed on the NM-1000 display terminal (ITEM 20). This setting will give 12 counts per second at 1X10-6 percent power (typical shutdown power), and 1.2 counts per second at 1X10-7 percent power (typical rod withdrawal permit point). Change the count rate percent power constant

(ITEM 25) to 1.87X10-7 for a power indication of 0.1 percent power as read on the NM-1000 display terminal (ITEM 10).

3.1) Change the count rate to Campbell cross over to 1.2X106 (ITEM 29).
3.2) Increase reactor power to full power and allow the reactor to stabilize (several minutes).
3.3) Adjust the Campbell amplifier gain (R27 in the Campbell amp) for 8X104 counts per second as displayed on the NM-1000 terminal (ITEM 30).

4.1) Set the Campbell linearizing factor to 0.370 (ITEM 33), the Campbell to count rate crossover to 1950 (ITEM 39), and the Campbell noise constant (ITEM 31) to 65.
4.2) To set the Campbell percent power constant, take ten consecutive readings of the Campbell signal (ITEM 30) at full power, and find the average. Using equation (3), calculate the Campbell percent power constant and enter as ITEM 35.

$$\text{ITEM 35} = \frac{\text{ITEM 10}}{[\text{ITEM 30}]2 * [\text{ITEM 33}]} \qquad \text{(Equation 3)}$$

Verify that the power indicated on the NM-1000 is 100% (ITEM 10).

5.1) To check the crossover alignment from Campbell to countrate, turn on the log chart recorder and scram the reactor from full power. Examine the trace in the crossover region (about 0.1 percent power) and note any discontinuity. If a discontinuity is evident, observe whether the Campbell signal is too high or too low at the crossover.

6.1) To precisely align the Campbell to countrate crossover if a discontinuity is evident, use the following procedure.

Campbell signal too high at crossover:
Increase the Campbell detector noise constant (ITEM 31) by about five to ten percent of the current Campbell detector noise constant and repeat step 5.1 above. Note any discontinuity at the crossover, and make appropriate adjustments to the Campbell detector noise constant. Repeat steps 5.1 and 6.1 as necessary.

Campbell signal too low at crossover:
Decrease the Campbell detector noise constant (ITEM 31) by about five to ten percent of the current Campbell detector noise constant and repeat step 5.1 above. Note any discontinuity at the crossover, and make appropriate adjustments to the Campbell detector noise constant. Repeat steps 5.1 and 6.1 as necessary.

ADDENDUM TO DAILY CHECKLIST

CHAMBER AND INSTRUMENT SENSITIVITY

I. NM-1000 Calibration Constants

A. Verify that all calibration constants entered into the NM-1000 agree with the values posted on the control console (see sample label below).

NM-1000
Calibration Constants

21 =
25 =
29 =

31 =
33 =
35 =
39 =

40 =
41 =
42 =
43 =

51 =
52 =
53 =

PA-15 DISC =

DATE:
BY:

II. Calibration modes 1,3,4 &5 are sequentially tested for correct power level outputs. Item 50 of the NM-1000 is programmed to the appropriate mode and the corresponding power level is read from Item 10. The power level is then compared with the configured test levels and is deemed OK if it falls between 95% and 105% of the configured vales.

A. The configured values are stored in the following configuration channels. (Attachment 1, Page 2)

1.	Item 1	Counter Low Test	2.24E-5 % Power
2.	Item 3	Counter High Test	5.74E-2 % Power
3.	Item 4	Campb. Low Test	1.08E+1 % Power
4.	Item 5	Campb. High Test	1.09E+2 % Power

B. The procedure is as follows: (Item refers to pressing key on Burr-Brown Microterminal).
   1. Item F5, Item F8, Item 1, Enter, (Reading=CTR LOW)
   2. Item F1, Read % Power and linear recorder and record in log
   3. Item F5, Item F8, Item 3, Enter, (Reading=CTR HI)
   4. Item F1, Read % Power and linear recorder and record in log
   5. Item F5, Item F8, Item 4, Enter, (Reading=CMB LOW)
   6. Item F1, Read % Power and linear recorder and record in log
   7. Item F5, Item F8, Item 5, Enter, (Reading=CMB HI)
   8. Item F1, Read % Power and linear recorder and record in log
   9. At the end of the calibration test reset the NM-1000 to the normal mode. Item F5, Item F8, Item 0, Enter (Reading = Normal)

III. High Power Level Trip
   A. Raise control rod.
   B. Push "Power Scram Test" button on console.
   C. Verify that control rod scrams.
   D. Verify that A2 on Burr-Brown is on.
   E. Item F1, Item 5, Verify that read out shows H for High.

IV. Period Trip
   A. Raise control rod (do not use same rod used for III above).
   B. Item F5, Item F8, Item 5, Enter (Reading=CMB HI).
   C. The above should cause a momentary high positive period and activate the period trip.
   D. Verify that control rod scrams.
   E. Verify that A2 light is on.
   F. Verify that Item F1, Item 5, shows a read out of R for rate.

V. Loss of High Voltage Trip
   A. Raise control rod.
   B. Push "High Voltage Test" button on console.
   C. Verify that control rod scrams.
   D. Verify that A2 light is on.
   E. Verify that Item F15 shows a read out of 32/V.

VI. Startup Channel (Low Level Trip)
   F. Remove Neutron Source.
   G. Allow enough time for the NM-1000 power to drop below the source trip limit.
   H. Verify that light A2 goes on with Burr-Brown readout shows 3.7E-7.
   I. Verify that Item F1, Item 5 shows a readout of "L" for low level trip.
   J. Try to raise control rod.

VII. Watchdog Timer (Do first run day of each month)
   A. Raise control rod.
   B. Disconnect plug labeled A3 CTX (card in position 4 in microprocessor card box).
   C. Green light on IO & Memory Card (card in position 9 in microprocessor card box) should go off and yellow on.

D. Verify that light A1 is on.
E. Verify that Item F6(60) shows CXFAIL.
F. Go through Item F6, 1-9 until read out shows empty.
G. Verify that control rod scrams.

CHAPTER 5

CORE PHYSICS

The physics of the reactor has been studied in considerable detail on a critical-assembly mock-up as well as on the operating prototype. In the succeeding sections, the more important features of the core physics are discussed.

5.1 CRITICAL MASS

The prototype TRIGA reactor attained criticality with 54 fuel elements, or about 1.9 kg of U-235. The AJBRF reactor attained criticality on June 26. 1959 with 54 fuel elements.

5.2 VOID COEFFICIENT

The void coefficient is very similar to the values obtained on the critical-assembly, which were measured to be -0.15% δk per 1% water void at 23° C in the central region of the core and +0.04% δk per 1% water void at 23° C at the core-reflector interface, where there is a region of graphite-loaded dummy elements. The core average value is approximately -0.14% δk per 1% water void, which agrees with calculations.

Since the core is approximately 33% water, loss of coolant would reduce reactivity by 4.6% delta-k. This reactivity loss is significantly larger than the excess reactivity; thus, the core will be subcritical when the water moderator is removed.

5.3 MODERATING PROPERTIES OF ZIRCONIUM HYDRIDE

Experiments performed by General Atomic personnel at the Brookhaven National Laboratory have shown that zirconium hydride has very unusual moderating properties for slow neutrons [1]. The results of these experiments can be explained by assuming that the hydrogen-atom lattice vibrations can be described by an Einstein model with a characteristic energy $h\nu = 0.130$ eV. This description is consistent with the theory that the hydrogen atom occupies a lattice site at the center of a regular tetrahedron of zirconium atoms. The basic consequences of this model, which have been experimentally verified, are that:

1. Neutrons with energies of less than $h\nu$ cannot lose energy in collisions with zirconium hydride.
2. A slow neutron can gain an energy $h\nu$ in a collision with zirconium hydride with a probability proportional to $\exp(-h\nu/kT)$, which increases very rapidly with temperature.

Since $h\nu > kT$, it has been found that zirconium hydride is not effective in thermalizing neutrons but that it can speed up neutrons already thermalized by water by transferring to them a quantum of energy $h\nu$.

5.4. TEMPERATURE COEFFICIENTS

A particularly large effort has gone into designing the reactor in such a way that an increase in the temperature of the fuel elements will result in a relatively large decrease in reactivity. This large prompt negative fuel-temperature coefficient results from the following effects:

1. Cell and inhomogeneities
2. Doppler
2. Core leakage

Cell an Inhomogeneities. Although a TRIGA reactor is frequently referred to as a homogenous reactor, a large part of the prompt negative temperature coefficient arises because of the inhomogeneity associated with the 3.8 cm diameter fuel-moderator elements and the interstitial cooling water. This cooling water provides a part of the neutron moderator (metal:water ratio of $\cong$ 2.1) and is instrumental in producing the large negative temperature coefficient by the process noted below and first identified by F. Dyson. When the fuel temperature increases, the zirconium hydride temperature essentially follows it instantaneously, thus increasing exponentially (Boltzman equation) the number of bound hydrogen atoms in excited levels. This increases the probability of speeding up the neutrons within the fuel element when they collide with the bound hydrogen and gain energy (h_ν from the lattice vibrations. This results in the hardening of the neutron spectrum, a decrease in the fission probability, and an increase in the fraction of neutrons lost from the fuel element because of leakage from the element. When the neutrons leave the hot fuel element, they rethermalized in the cooling water and undergo increased parasitic capture in the interstitial core water and the cladding material, especially when steel clads are used. CELL EFFECT is an important contributor to the prompt negative coefficient; note that it is almost entirely dependent on the heterogeneous appearance of the core to thermal neutrons. If the core had no water and were constructed entirely of U-ZrH_x, the prompt negative coefficient would of course still exist but would be much smaller because the harder neutron spectrum would not permit as much parasitic capture in hydrogen and cladding material.

Doppler Effects. The uranium in the LEU is approximately 20% U-235 and 80% U-238. The capture resonances in U-238 are Doppler-broadened by an increase in the fuel temperature causing a decrease in the resonance escape probability, p.

Core Leakage . The core leakage contribution derives basically from the same mechanism which produces the cell effect. The core can be envisaged as a large super-cell with reflector acting as a moderator. When the core heats up, leakage is increased and relatively more captures occur outside the fuel.

Experiments at General Atomic have shown that the cell effect is the dominant contributor to the temperature coefficient. The fuel temperature coefficient of the TRIGA reactor has been experimentally demonstrated to be -0.01% δk/k per °C rise in average fuel temperature. The temperature coefficient associated with heating the water and the fuel in the TRIGA reactor core is extremely small. The total reactivity contribution do to this latter coefficient over the range of 10°to 60°C is less then 0.08% δk/k. The operational characteristics of the reactor are therefore primarily determined by the extremely large prompt negative temperature coefficient

within the TRIGA fuel. The experiments performed to determine this temperature coefficient demonstrate that it is a prompt coefficient and it is nearly constant over the power range from 0 - 1.4 MW.

Reactivity effects associated with water temperature will have essentially no effect on either the normal operating characteristics or the transient behavior of the reactor for the following reasons:

1. Under normal operating conditions, if the reactor were operated at a power level of 18 kW even with the water-cooling system turned off, the average temperature of the $\cong$ 4000 gal of water in the core tank would be increased at a rate of less than 0.78°C/h. this would certainly cause a negligible reactivity perturbation.
2. The temperature of the core water does not change materially during a reactor transient. The transient behavior of the reactor is determined primarily by changes in fuel temperature.

The transient behavior of the reactor has been studied in detail in the test program of the prototype TRIGA [2] (see also Section 3.2.13 Dynamic Behavior of Reactor).

5.5 REACTIVITY PERTURBATIONS

Perturbations of the reactivity resulting from physical changes in the core and reflector can be of importance to the safety of some reactors. It is possible for water to be introduced accidentally into the reflector region of TRIGA by flooding the reflector graphite, the specimen rack, and the pneumatic transfer tube. It is expected that water in any of these regions will cause a decrease in reactivity. This decrease in reactivity is due to the fact that there is a greater absorption of neutrons because of the presence of water.

The effect on reactivity of interchanging fuel elements and graphite dummy elements must also be considered. Experiments on the General Atomic subcritical assembly have shown that the cylindrically symmetrical loading of fuel elements surrounded by graphite dummy elements is the maximum reactivity configuration. Any rearrangement will result in a decrease in reactivity.

Chapter 5

References

1. A.W. McReynolds, M. Nelkin, M.N. Rosenbluth, and W. Whittemore, "Neutron Thermalization by Chemically Bound Hydrogen and Carbon", Proceedings of the Second United Nations International Conference on Peaceful Uses of Atomic Energy, Geneva. Paper UN/1540. 1958.

2. "TRIGA Transient Experiments: Interim Report", General Atomic Report GA-531, September 1958.

CHAPTER 6

CONDUCT OF OPERATIONS

6.1 FACILITY ADMINISTRATION

6.1.1 Overall Organization

Figure 6-1 illustrates the organizational structure that is applied to the management and operation of the reactor facility. These responsibilities include safeguarding the public and staff from undue radiation exposure and adherence to license or other operation constraints. The Reactor Supervisor is delegated responsibility for overall facility operation.

Facility operation staff is an organization of a Reactor Manager/Reactor Supervisor and at least one equivalent person (Reactor Operator). This staff of two provides for basic operation requirements. A staff of one may occur during transitional periods (e.g. to replace a vacant position). Students and researchers supplement the organization. Titles for staff position are descriptive and may vary from actual designations. Description of key components of the organization are outlined below.

6.1.1.1 Director, Veterans Administration Medical Center

Has overall responsibility for all functions of the Medical Center and has delegated his authority to the Reactor Supervisor to assure the integrity and security of the special nuclear material and the safe operation of the reactor facility.

6.1.1.2 Chief of Staff

Is responsible for the professional staff of the Medical Center and is Chairman of the Reactor Safeguards Committee (RSC).

6.1.1.3 Associate Chief of Staff Research

Is the Director of the Research Service to which the AJBRF is attached. All administrative functions such as personnel matters, payroll, purchasing; secretarial, etc. are supplied by the Research Service.

6.1.1.4 Reactor Safeguards Committee

The Reactor Safeguards Committee (RSC) has broad responsibilities to provide independent review of facility activities for safe operation. The RSC provides the following functions: (1) Review experiments, procedures, facility changes to determine if there are any Unreviewed Safety Questions as defined in 10CFR50.59. These would be issues that haven't been previously considered in the facility SAR; and (2) Audit and oversight of all pertinent reactor operations.

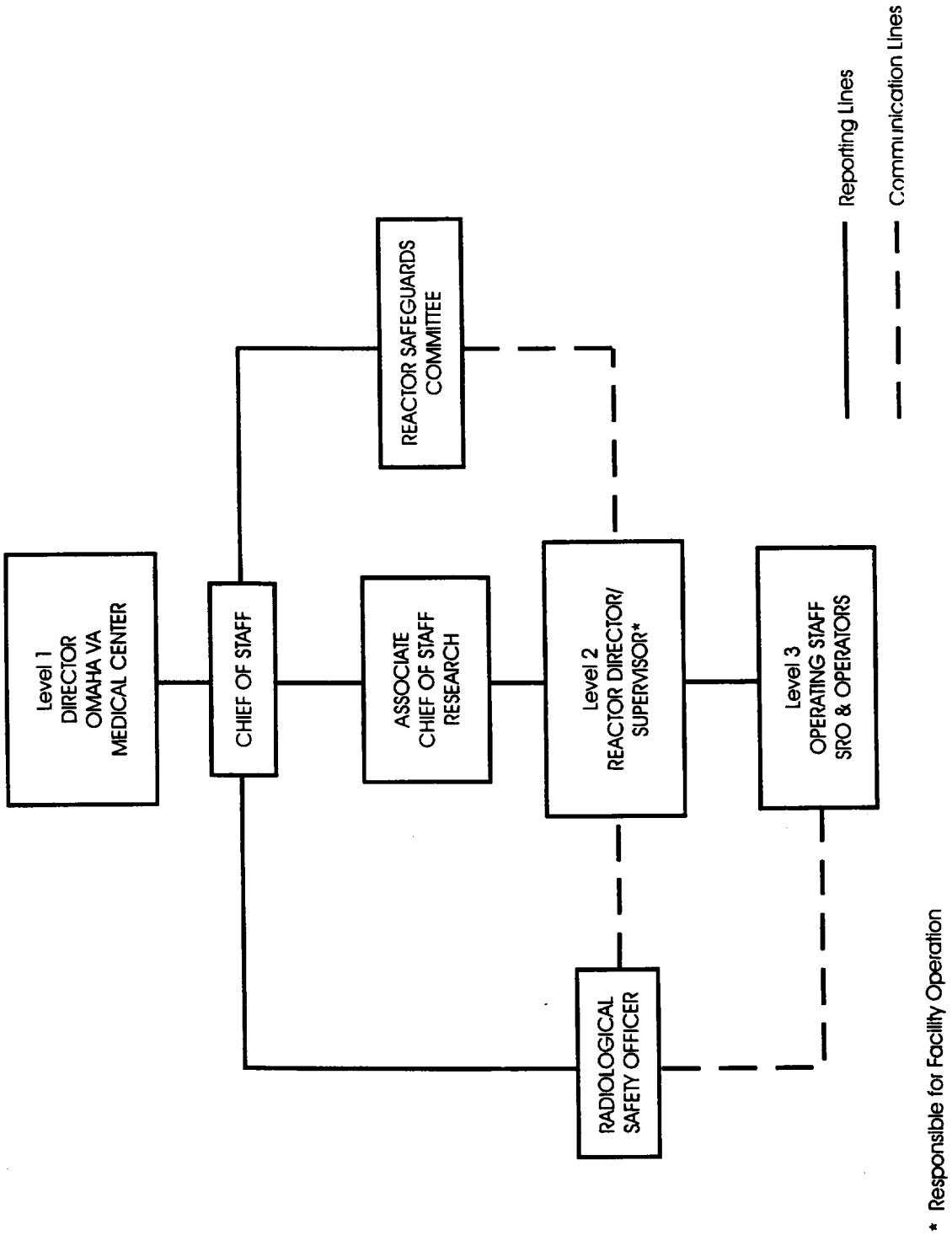


Fig 6-1 FACILITY ORGANIZATION

The Reactor Safeguards Committee (RSC) functions to provide independent review and audit of facility activities. This autonomous oversight is essential for safe operation of the facility and the protection of the health and safety of the public. Independent professionals will provide the ability to examine activities of the facility, which improves the overall performance of the program. Independent audits allow the facility to correct problems in advance of periodic NRC inspections.

The committee is composed of a minimum of 6 members, 3 of which include the following:

1. Chairman	Chief of Staff
2. Member (ex-officio)	Radiation Safety Officer
3. Member	Reactor Director/Supervisor

A minimum qualification for persons on the RSC shall be 5 years of professional work experience in the discipline or specific field he represents. A baccalaureate degree may fulfill 4 years of experience. Members should have a fundamental knowledge of radiation and its use in the conduct of experiments. Committee members should have a variety of backgrounds and thus provide for both additional expertise that the reactor staff may not have and to allow a second opinion in areas in which the reactor staff has expertise.

Qualified and approved alternates may serve in the absence of regular members. No more than two alternates shall participate on a voting basis in RSC activities at any one time.

Members and alternates shall be appointed by and report to Level 1 management.

6.1.1.4.1 Charter and Rules

The operations of the Reactor Safeguards Committee will accomplish its agenda utilizing an established charter or directive including provisions for:

(1) Meeting frequency - at least on an average semiannually.
(2) Quorums - not less than one-half of the membership where the operating staff does not constitute a majority.
(3) Use of subgroups
(4) Dissemination, review, and approval of minutes in a timely manner (within a month following the meeting).

6.1.1.4.2 Review Function

The review function of the committee includes facility operation related to reactor and radiological safety. The following items are subject to review:

(1) Determinations that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question as required by 10 CFR 50.59.

(2) All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance
(3) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity
(4) Proposed change in technical specifications, license, or charter
(5) Violations of technical specifications, license, or charter. Violations of internal procedures or instructions having safety significance.
(6) Operating abnormalities having safety significance.
(7) Reportable occurrences listed in Technical Specifications, Section 6.7.2
(8) Audit reports

A written report of minutes of the findings and recommendations of the review group are submitted to Level 1 management and the review and audit group members within a month after the review has been completed.

6.1.1.4.3 Audit Function

The audit function includes selective (but comprehensive) examination of operating records, logs, and other documents. Discussions with personnel and observation of operations should be used also as appropriate. In no case shall the individual immediately responsible for the area perform an audit in that area. The following items are to be audited:

(1) Facility operations for compliance to the technical specifications and applicable license or charter conditions annually.

(2) The retraining and requalification program for the operating staff, on an average of at least once every other calendar year (intervals between audits not to exceed 30 months)

(3) The results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operation that affect reactor safety, on an average of at least once per calendar year (intervals between audits not to exceed 15 months)

(4) The reactor facility emergency plan, and implementing procedures on an average of at least once every other calendar year (at intervals between audits not to exceed 30 months)

Deficiencies uncovered that affect reactor safety shall immediately be reported to Level 1 management. A written report of the findings of the audit are to be submitted to Level 1 management and the review and audit group within three months after the audit has been completed.

6.1.1.5 Radiation Safety Officer

The Radiation Safety Officer (RSO) acts as the delegated authority of the Radiation Safety Committee in the daily implementation of policies and practices regarding the safe use of radioisotopes and sources of radiation that involve the Medical Centers Broad Scope License. He or his delegate will be knowledgeable of the facility radiological hazards. Responsibilities will include calibration of radiation detection instruments, measurement of radiation levels, control of radioactive contamination, maintenance of radiation records, and assistance with other Broad Scope License monitoring activities. The RSO is a member of the Reactor Safeguards Committee (ex-officio).

The Radiation Safety Committee has no oversight regarding reactor operations or functions. The committee does, however, review all dosimetry reports including those of reactor personnel. Historically, the chairman of both Radiation Safety Committee and the Reactor Safeguards Committee has been chaired by the Chief of Staff for the Medical Center.

6.1.1.6 Reactor Supervisor

Reactor operation at the AJBRF is directed by a reactor supervisor. Responsibilities of the reactor supervisor include control of license documentation, reactor operation, equipment maintenance, experiment operation, instruction of persons with access to laboratory areas, and development of research activities.

Activities of reactor operators with USNRC licenses will be subject to the direction of a person with a USNRC senior operator permit. The reactor supervisor shall be qualified as a senior operator. This person is to be knowledgeable of regulatory requirements, license conditions, and standard operating practices.

6.1.1.7 Professional and Classified Staff

Professional and classified staff, such as research scientists, reactor operators, technicians and secretaries, will supplement the organization as necessary to support facility programs. Personnel associated with the research reactor facility [1] shall have a combination of academic training, experience, skills, and health commensurate with the responsibility to provide reasonable assurance that decisions and actions during normal and abnormal conditions will be such that the facility and reactor are operated in a safe manner.

6.1.1.8 Facility Staff Qualifications

The necessary level of expertise of staff directly involved with nuclear operations will be maintained. Classifications and specific duty requirements of operations personnel will be subject to approval by the Reactor Safeguards Committee on an individual basis.

6.1.1.8.1 Reactor Supervisor

At the time of appointment to the position the Reactor Supervisor shall have a minimum of 5 years of nuclear experience. He shall have a baccalaureate or higher degree in engineering or other scientific field. The degree will fulfill 4 years of experience on a one-for-one time basis. Equivalent education or experience may be substituted for a degree.

6.1.1.8.2 Senior Reactor Operator

At the time of appointment to the position a Senior Reactor Operator shall have minimum of a high school diploma or equivalent and should have 4 years of nuclear experience. A maximum of 2 years of experience may be fulfilled by related academic or technical training on a one-for-one time basis.

6.1.1.8.3 Reactor Operator

At the time of appointment to the active position, operators shall have a high school diploma or equivalent.

6.2 REACTOR OPERATIONS

The TRIGA reactor for the Veterans Administration Medical Center in Omaha, Nebraska is designed to operate continuously at a power level of 20 kW, however, normal operation is usually 7-8 hours per day. The maximum available excess reactivity, for any temperature conditions, is limited in the facilities Technical Specifications. Limiting excess reactivity capability reduces the effects cause by unexpected power excursions. Experiments conducted with General Atomic's prototype TRIGA show that the reactor power level would be limited to safe values even if all of this available excess reactivity should be suddenly introduced into the reactor.

Staffing requirements for the facility are outlined in the Technical Specifications. Operation of the reactor and activities associated with the reactor control system, instrument systems radiation monitoring system, and engineered safety features will be the function of staff personnel with appropriate license certifications [1]. Operation will include the implementation of required procedures, execution of appropriate experiments, actions related to safety, and the preparation of required reports and records. Thus, by requiring specific levels of expertise among staff members supplemented by safety tests and reports; the potential for increased risk to the health and safety of the public is lessened.

Reportable events, reporting time requirements, and record retention are documented in detail in the facility Technical Specifications. A report containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each will be submitted on an annual basis.

Conduct of licensed activities are contained within but are not limited to applicable regulations in the Federal Code of Regulations; and the facility's Technical Specifications, Emergency Plan,

Requalification Plan, and Security procedures. These plans are specifically implemented to describe in more detail how the facility will address the issues of requalification, emergency preparedness, and security as well as technical aspects of the reactor.

6.2.1 Procedures

Written procedures shall govern many of the activities associated with reactor operation. Preparation of the procedures and minor modifications of the procedures will be by licensed operator. Substantive changes or major modifications to procedures, and prepared procedures will be submitted to the Reactor Safeguards Committee for review and approval. Temporary deviations from the procedures may be made by the reactor supervisor or designated senior operator provided changes of substance are reported for review and approval. Written procedures provide for a uniform understanding of various processes to key reactor tasks. In addition, they provide continuity of processes during staffing changes.

6.2.2 Routine Operation Procedures

The Daily Checklist will be completed prior to each daily start-up, at the completion of each day's operation and before start-up after any maintenance on the reactor. This checklist allows the checking of each electronic and mechanical component to assure that it is functioning properly. The checklist also verifies the settings of the microprocessor constants.

Written procedures shall be reviewed and approved by the Reactor Director/Supervisor and reviewed by the Reactor Safeguards Committee prior to initiation of the following activities.

(1) Startup, operation, and shutdown of the reactor
(2) Fuel loading, unloading, and movement within the reactor
(3) Maintenance of major components of systems that could have an effect on reactor safety
(4) Surveillance checks, calibrations, and inspections required by the technical specifications or those that may have an effect on reactor safety
(5) Personnel radiation protection, consistent with applicable regulations or guidelines. The procedures shall include managements commitments and programs to maintain exposures and releases as low as reasonably achievable (ALARA) in accordance with the guidelines of ANSI/ANS-15.11-1993, "Radiation Protection at Research Reactor Facilities".
(6) Administrative controls for operation and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.
(8) Implementation of required plans such as emergency or security plans.
(9) Any additional plans that may be deemed necessary for operation of the facility.

Substantive changes to the procedures shall be made effective only after documented review by the Reactor Safeguards Committee and approval by the Reactor Director/Supervisor. Minor modifications that do not change their original intent may be made by the Reactor Director/Supervisor (Level 3), but the modifications must be approved by the Reactor Safeguards Committee (Level 2) within 14 days if not a unreviewed safety question. Thus,

changes to established procedures are subject to independent review to ensure these changes stay within the context of their necessary function.

Chapter 6

References

1. "Selection and Training of Personnel for Research Reactors", ANSI/ANS - 15.4 - 1988.

CHAPTER 7

RADIOACTIVE MATERIALS AND RADIATION MEASUREMENT

Radioactive materials and radiation control within the Reactor Laboratory will be subject to industry standards [1,2], license conditions, and 10 CFR 20.1001 - 20.2401 and appendices.

7.1 RADIOACTIVE MATERIAL CONTROL

Physical control of radioactive materials shall be provided as an essential part of the radiological safety program. Control shall include identification of items or storage in identified locations. Controls such as shielding, isolation, containment and ventilation will be provided, as necessary, to control radiation exposure to the inventory of radioactive materials. Since most of the liquid waste from our neutron activation procedures involve short half-lived isotopes the waste is stored until it reaches background. If release into sanitary sewerage is indicated it will be done so as to comply with 10 CFR 20.2003 and the values recorded in the Omaha VA Medical Center sewerage disposal records. If assay of the waste is required it will be done with calibrated radiation survey meters or a well Ge detector coupled to a multichannel analyzer.

7.1.1 Reactor Fuel

A maximum of 3.3 kilograms of ^{235}U at enrichments less than 20% will be needed for future upgrades, replacement of failed fuel elements, and account for core burnup. Irradiated reactor fuel shall be maintained in the reactor core, reactor pool storage racks, or the three emergency storage pits located immediately adjacent to the reactor tank as described in Section 3.2.1, of this part.

7.1.2 Reactor Components

Each reactor component removed from the reactor pool shall be measured for activation levels and removable contamination. All components remaining in the pool shall be assumed to be radioactive. Components removed from the pool will be cleaned or covered as necessary to control radioactive contamination. Components that contain radioactive material will be labeled and stored in the isotope storage cell as shown in Fig 3.2 (SW 2F).

No more than 20 g of ^{235}U at enrichments greater than 20% would be needed to allow the facility additional fission chambers for use and extra chambers for spares.

7.1.3 Isotope Storage Cell

The isotope storage cell, SW 2F (Fig. 3.2), is underground and adjacent to the reactor laboratory. The storage cell contains 10 holes in the floor 20 ft deep and 64 holes in the wall 10 inches deep. Both types of holes have lead plugs. No part of the Medical Center is directly above, below, or adjacent to this storage with the exception of a single side which is the entrance into this storage area.

7.1.4 Experiment Facilities

Experiment facilities shall consist of the rotary specimen rack, vertical tubes, pneumatic transfer systems, central thimble, and in-pool irradiation facilities. Removal of experiment facilities from the pool or the beam originating from the reactor shall be subject to the same controls as those for reactor components.

7.1.5 Activated Samples

Materials that are inserted into the reactor experiment facilities or reactor beam shall be controlled as radioactive materials until disposed as radioactive waste, transferred to an authorized user, or decayed to releasable levels for non-radioactive materials.

7.1.6 Radioactive Waste

Canisters shall be available and labeled for radioactive waste at locations where contamination from sample processing or other activities with contamination occur. Locations shall be designated for storage of solid wastes. Liquid and solid waste will be stored in the isotope storage cell in the reactor room until release criteria are determined such as decay, dilution, or processing. Specific sinks in the facility that are designated for radioactive materials shall be identified. Radioactive liquids will be determined to be soluble or readily dispersible in water as referenced in 10 CFR 20.2003 before release into the sanitary sewers. Solubility will be determined based on various reference guides and consultation with knowledgeable professionals e.g. chemists, etc. Historically, since most of our liquid samples have been mostly short lived; the encapsulated liquids are usually allowed to decay out before release.

Gaseous wastes from experiments are doubly encapsulated. The radioactive material content, including fission products of any experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation will not result in dose in excess of the annual limits stated in 10 CFR 20. The appropriate radiation detection equipment or theoretical calculations, as appropriate, will determine these activities. Typically, these samples are allowed to decay to background levels before disposal.

All procedures involving radioactive waste will follow the criteria as described in the AJBRF Radiation Protection Program (RPP).

7.1.7 Other Radioactive Material

Radioactive reactor components, contaminated tools and fixtures and other radioactive materials shall be included in the Radiation Protection Program. These materials shall be maintained in a restricted area or be under the control of authorized individuals. They may be released by authorized individuals for unrestricted use upon decontamination, e.g. they are no longer radioactive, using the criterion specified in the Radiation Protection Program, 10 CFR 20 subpart K, and other applicable regulations.

The following materials are needed for reactor operation and instrument calibration:

(1) 8 Ci of sealed polonium – beryllium;
(2) 4 Ci of americium – beryllium;
(3) 1.5 Ci of cesium – 137; and
(4) 10 mCi of iodine - 129, simulated iodine - 131, lead - 210, cobalt - 60, and technetium – 99.

Either of items (1) or (2) will be used as a sealed neutron source for reactor start-up. Byproduct material requirements specified in items (3) and (4) are sealed sources necessary to check and calibrate radiation detection equipment used in conjunction with reactor operation.

7.2 Radiation Monitoring

Radiation monitoring consists of fixed, portable, or sampling type systems. Monitoring systems are applied to measurement of radiation areas and high radiation areas around the reactor facility, significant contamination within and adjacent to the facility, and radioactive materials and their concentrations in effluents. Monitoring shall be considered for routine operations, abnormal conditions, and emergency situations. Thus minimizing personal radiation exposures and following ALARA procedures.

7.2.1 Minimum Procedures

Zone identification, access control, and protective equipment are designated. Zone identification for radioactive materials and radiation areas are designated as specified by 10 CFR part 20. Access control for zones shall be to control radiation exposures and physical security of the reactor facility and it's material as specified by 10 CFR parts 19 and 73, (Notices, Instructions, and Reports to Workers; Inspections and Physical Protection of Plants and Materials). Protective equipment for routine abnormal and emergency conditions shall include at least tape, plastic bags, absorbent paper, gloves, shoe covers, coveralls.

Contamination areas or areas that are routinely subject to contamination shall be marked clearly and control points established to monitor for contamination of personnel or equipment that leave the designated area. Measurements shall provide action levels for removable activities. Restrictions for unconditional release of radioactive materials as specified in the facility RPP will be observed. Periodic monitoring of areas in which contamination is probable will be of an adequate frequency to reveal significant changes in contamination levels. Decontamination of personnel, equipment, and surfaces shall be appropriate to requirements for control of radiation exposure and control of radioactive material containment. Release of radioactive components, etc. for unrestricted use may only be done by the Reactor Manager or a delegated Senior Reactor Operator as referenced in the facility RPP which is approved by the Reactor Safeguards Committee.

In order to evaluate various conditions of airborne radioactivity, monitoring for airborne radioactivity consists of continuous sampling of air particulate activity in the reactor area. Monitoring will occur during reactor operation or activities involving fuel, core, or experimental

facilities, and will provide measurements for routine, abnormal, and emergency conditions. Additional airborne monitoring equipment will be provided for special experiment needs.

Personnel dosimetry will be required for access to reactor areas and some other facility activities. Monitoring devices will typically be film badges with pocket dosimeters for supplemental measurements. Other personnel monitoring, such as bioassays will be applied as determined by the activity and conditions or radiation exposure situations. Personnel shall use supplemental dosimetry during activities that deviate substantially from routine operations. Dosimetry will be provided for persons visiting areas with potential radiation exposure under the criteria set forth in 10CFR20 and the facility RPP.

7.2.2 Monitoring Techniques

Implementation of radiation monitoring to maintain the goal of "As Low As Reasonably Achievable" should consist of: (a) preoperation planning, (b) operations techniques, and (c) post operation analysis.

7.2.3 Management Surveillance

Management oversight is conducted through the Reactor Safeguards Committee (RSC). All new experiments are reviewed by the committee to ensure the safety of personnel and protection of reactor components and structures. The review will be applied to determine whether a "unreviewed safety question" is involved as specified in 10CFR50.59 and whether facility modifications or procedures should be implemented to maintain radiation exposures "As Low as Reasonably Achievable".

Material limits on experiments and failure and malfunctions of experiments are outlined in detail within the facility Technical Specifications. Specifically, individual experiments are limited to one dollar of reactivity and sum of all associated experimental facilities at one time shall not exceed two dollars. The reactor is always in shut down condition when changing or moving a secured experiment. Actual determination of an experiment's reactivity is performed if the estimated worth is greater than forty cents.

Additional material restrictions prevent release of radioactive material or subsequent damage to the reactor in the event of an experiment failure. Experiments containing liquid, gas and potentially corrosive material will be doubly encapsulated. Compounds highly reactive with water, potentially explosive materials, and liquid fissionable materials shall not be irradiated in the reactor. Guidance for classification of materials shall be "Dangerous Properties of Industrial Materials" by N.I. Sax (Reinhold Publishing) or equivalent.

The radioactive material content, including fission products of any experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components from encapsulation, will not result in doses in excess of the limits stated in 10 CFR 20 and other applicable guidance [2,4]. With regard to these limitations, calculations will determine if an experiment fails and releases radioactive gases to the reactor room or atmosphere. The following is assumed:

1. 100% of the radioactive gases or aerosols escape.

2. If the effluent exhausts through a filter with 99% efficiency for 0.3-micron particles, at least 10% of these vapors escape.
3. For materials whose boiling point is above 130°F (54°C); the vapors of at least 10% of the materials escape through an undisturbed column of water above the core.

7.3 INSTRUMENTATION

Instrumentation for the evaluation of radiation exposures from routine, abnormal and emergency situations shall consist of fixed area monitors, portable survey monitors, and appropriate sampling methods. The minimum instrumentation available during reactor operation shall consist of fixed area gamma dose rate monitors, continuous air particulate monitor, portable thin window GM tube survey meter and pocket dosimeters with charger. Other detecting equipment that may be available includes an alpha-beta proportional counter, a multichannel gamma pulse height analyzer with Ge detector, a liquid scintillation detector, low range beta-gamma dose rate meters, a low energy ionization chamber type meter, and GM tube or equivalent friskers.

7.3.1 Fixed Area Monitors

Fixed area monitors have audible and visual alarms. One monitor is permanently mounted approximately 1.4 meters from the isotope removal port and the other is near the pneumatic tube.

7.3.2 Airborne Radioactivity Monitors

A continuous air particulate fixed filter monitor with audible and visual alarm shall be functional in the reactor vicinity during reactor operation. A thin window GM detector or equivalent will also monitor the activity and provide alert and alarm conditions with an audible enunciator. Typically, the count rate of the instrument includes the range of 50 to 50,000 counts /minute

The continuous air monitor also is provided with a charcoal filter for sampling the air for presence of iodine radioactivity.

7.3.3 Survey and Laboratory Instruments

Portable survey monitors for alpha, beta, or gamma radiation shall be maintained for area surveys of laboratory and experiment areas. Survey instruments will consist of the following instruments or equivalents: (1) a pancake style GM or low energy scintillation detector and (2) ionization chamber.

Supplemental measurements can be made with an alpha beta proportional counter, or a gamma ray pulse height analyzer. A liquid scintillation counter is also available in the adjacent research building.

7.3.4 Liquid Effluents

The reactor generates no radioactive effluents. Radioactive liquid waste generated in the research program are governed by the requirements of the AJBRF Radiation Protection Plan (RPP). Reactor coolant water may be monitored for radioactivity as a supplemental indicator of water activity.

7.3.5 Calibrations

Calibration methods, accuracy, frequency and fundamental checks are established for radiation monitors following the procedures specified in reactor operation specifications and other applicable documents.

7.3.6 Records

Records are specified for maintenance of radiological data that relate to reactor operation. These records shall include:

a. Personnel dosimetry including bioassays or other special measurements made.
b. Radiological control surveys required by facility specifications.
c. Gaseous and liquid radioactive effluents released to the environment.
d. Radiation Surveys.
e. Instrument calibration records.
f. Radioactive material receipt and transfer records.
g. Solid radioactive waste disposal records.
i. Data on radiological incidents.

7.4 EVALUATION OF MONITORING SYSTEMS

The radiation monitors provide information to operating personnel of impending or existing hazards from radiation so that there will be sufficient time to take the necessary steps to control the exposure of personnel and the release of radioactivity or to evacuate the facility. Two types of radiation monitors are used: a continuous air particulate for determining radiation levels due to particulate radioisotopes suspended in the reactor room and area monitors for determining the gamma field at several locations in the facility.

Each type of radiation monitor has a specific radiological purpose. The particulate air monitor is used to measure airborne particulate radioisotopes while the area radiation monitors are used to assess radiation intensities. Thus, personal exposures can be limited to ensure that regulatory limits are not exceeded and that ALARA principals are met. The radiation monitors described below are typical instruments possessed at the time this application was written. Replacements may have slightly different characteristics but will be at least equivalent.

7.4.1 Particulate Air Monitor

Currently, there are two particulate CAMs. Either of the two existing CAMs or its equivalent shall be available for use during reactor operation. Both contain audible and visual alarms with a fixed filter beta particulate monitor. As previously mentioned in SAR section 4.3.2, these CAMs are capable of monitoring for airborne radiation in the form of particulates.

One air monitor in use is a Nuclear Measurements Corp. Model AM-2d gross beta-gamma air monitor configured for continuous sampling of airborne beta-emitters on a fixed filter. It uses an end-on coplanar GM type detector with a window diameter of 1 7/8" and a thickness of 5.6 mg/cm^2. The detector and a particulate collector (filter paper, etc.) are housed within a lead shield assembly. After air enters the shield, it passes through the collector filter paper deposit the particulates (microns in size and up) on it. From there the air travels through a calibrated orifice where the pressure drop can be read by the Magnehelic gauge. The Magnehelic reads directly in CFM thus facilitating any sensitivity computations.

The other air monitor is an Eberline Model AMS-3A. It contains 2 pancake GM tube detectors, which are 1 ¾ inches in diameter with a 1.4-2.0 mg/cm^2 mica window. A shield with equivalency of 2 inches of lead is used in conjunction with one of the GM detectors for background subtraction.

Both detectors are calibrated with a ^{99}Tc standard of 0.005 μCi that emits a 0.29 MeV beta or equivalent. Since the monitors use a thin end-window GM tube, a calibrated source of an E_β greater than 0.29 MeV would have a greater efficiency and consequently result in a larger cpm/cm^3-h. Using the efficiency determined from the ^{99}Tc standard the calibration of the CAMs are based on detection of particulate activity concentrations below the occupational DAC values of Appendix B of 10 CFR 20.1001-20.2401 for the relevant isotopes in the ranges 84-105 and 129-149. The alarm set point is set at 70% of these DAC values or 2000 pCi/ml.

7.4.2 Area Radiation Monitors

Several area radiation monitors which observe the gamma field are part of the permanent installation. Some locations are experiment areas in which shield configurations determine the levels of radiation during reactor operation. Alarm set points for all area radiation monitors will be at either 2 mrem/h or 5 mrem/h or lower. The first number is obtained by dividing the maximum desired dose each week by the number of working hours each week. The second number is obtained from the definition of a radiation area in 10CFR20.

7.5. Radiation Hazards from Experiments or Fuel

Because there can be intense radiation fields from radioactive isotopes produced by this reactor, it is considered necessary that reactor operations be supervised by individuals who are trained in the detection and evaluation of radiological hazards. It should be noted that these hazards do not differ from those encountered with any reactor operating at comparable power levels.

Calculations based on ^{60m}Co indicate that the reactor is capable of producing an equilibrium concentration of radioisotopes of approximately 160 Ci in the rotary specimen rack if the reactor is operated at 20 kW [3]. This production of of radioisotopes is distributed in the 40 sample positions each containing two sample containers. The maximum amount of activity which can be withdrawn at one time is therefore approximately 2 Ci. This constitutes an intense source of radiation; however, it is a source intensity that is routinely handled by competent, technically trained personnel. Because the isotope-Production capabilities of this reactor are so large, shielded isotope-handling equipment will be used to reduce radiation dosage to reasonable levels in accordance with 10 CFR 20.1201 and ALARA. Detailed specifications of Experiments are listed in the Technical Specifications.

The radioactivity hazards associated with fuel elements are of the same nature as those associated with isotope production. The calculated dose from a single fuel element after irradiation for 8 h at 20 kW and at a distance of 6 ft is 90 R/h at the time of shutdown [SAR, Appendix F].

It should be realized that the hazard associated with direct exposure of personnel to highly radioactive fuel is not unique with the TRIGA reactor. This hazard will be encountered to the same extent with any reactor operating at similar power levels for similar periods of time.

Because of the significant radiation level associated with the reactor fuel-moderator elements, it will normally be necessary to keep the elements under water for shielding. If an element is to be removed from the reactor shield tank, a conventional fuel-element transfer cask will be used to reduce the radiation level to within tolerable limits.

Chapter 7

References

1. "Radiological Protection at Research Reactor Facilities", ANSI/ANS-15.11. 1993.

2. "Development of Technical Specifications for Experiments in Research Reactors", Regulatory Guide 2.2, U.S. Atomic Energy Commission. Nov 1993.

3. "Safeguards Analysis Report for TRIGA Reactors Using Aluminum-Clad Fuel", GA-7860, General Atomic. March 16, 1967

4. "Review of Experiments for Research Reactors", Standard ANSI N401-1974, ANSI-15.6, 1974.

CHAPTER 8

ACCIDENT ANALYSIS

8.1. HAZARDS ASSOCIATED WITH THE OPERATION OF THE REACTOR

Certain potential hazards associated with the operation of the reactor system have been studied and have been found to cause no serious environmental hazard, nor any hazard to operating personnel. However, in the handling of all radioactive material, because of its very nature, it is necessary to observe accepted safety precautions. In the normal use of a TRIGA reactor it will be necessary to handle irradiated samples, and standard health-physics procedures will be followed. On the rare occasions when it is necessary to remove highly radioactive fuel elements from the water shield, special equipment and safety procedures will be utilized.

Specifically, the following problems were investigated:

1. fuel cladding rupture and subsequent release of fission product gases, their release in the reactor room, and the environment;
2. fuel cladding rupture and subsequent release of fission product gases in water and their release in the reactor room and the environment; and radioactive contamination of the shielding water;
3. the possibility of loss of shielding water and resultant projected doses; and
4. failure of the recorder resulting in an inadvertent withdrawal of the regulating rod.

Among the potential accidents considered to be credible, the one with the greatest potential effect on the environment and the unrestricted area outside of AJBRF is the loss of the cladding integrity of an irradiated fuel rod in air in the reactor laboratory. This has been designated as the "Maximum Hypothetical Accident", MHA.

A MHA is defined as a postulated accident with potential consequences greater than those from any event that can be mechanistically postulated. We have evaluated other possible accident sequences that originate in the intact reactor core and none pose a significant risk of cladding failure. However, it is possible that an operator, when removing a fuel element from the core or relocating one previously removed following irradiation, could experience an accident that would break the integrity of the fuel cladding. It is assumed that the accident occurs but no attempt is made to describe or evaluate deterministically the mechanical details of the accident or the probability of its occurrence. Only the consequences are considered. Although less significant than the MHA, a fuel cladding failure in water is also reviewed.

8.1.1 Handling Irradiated Fuel – Maximum Hypothetical Accident

Guidelines for preparation of the SAR require analysis of a maximum hypothetical accident involving failure of the cladding of one fuel element and the escape of radioactive noble gases and iodine. The analysis can be found in Appendix B, Section B.1, source quantities of radioactive noble gases and iodine are computed and tabulated for a MHA involving cladding failure of a single TRIGA fuel element and the escape of the radionuclides into the environment.

Following are assumptions and approximations applied to calculations:

1. Calculations of radionuclide inventory in fuel are based on continuous operation prior to fuel failure for 40 years at the average thermal power experienced by the reactor during its first 40 years of operation, namely, 1.50 kW. This is followed by 20 years of operation at full licensed power of 20 kW.
2. Radionuclide inventory in one "worst-case" fuel element is based on 57 elements in the core, 36 grams of ^{235}U per element [B3, B4] and a value of 2.0 as a very conservative value of the ratio of the maximum power in the core to the average power.
3. The principal fission products arise only from ^{235}U. Since the fuel is enriched to 20% ^{235}U and has very low relative burnup in a thermal spectrum, fission products from the ingrowth and fission of ^{239}Pu and fast fission of ^{238}U occur in considerably less quantities.
4. The fraction of noble gases and iodine contained within the fuel that is actually released is 1.5×10^{-5}, a value measured at General Atomics [B5] and used in SARs for other non-pulsing TRIGA reactor facilities [B6].
5. No release of particulate (radionuclides other than noble gases and iodine) is considered as no credible release mechanism exists.

As the fuel has been at the AJBRF since new, no residual sources from other irradiations are examined. Potential consequences of radiological releases are examined. Even in the MHA, no workers or members of the public are at risk of receiving radiation doses in excess of limits prescribed in federal regulations.

Fission product inventories in TRIGA fuel elements were calculated with the ORIGEN code, using very conservative approximations. Then, potential radionuclide releases from worst-case fuel elements were computed, again using very conservative approximations. Even if it were assumed that releases took place immediately after reactor operation, and that radionuclides were immediately dispersed inside the reactor room workplace, few radionuclide concentrations would be in excess of occupational derived air concentrations, and then only for a matter of hours or days. Data from SAR Appendix B, Sub-Appendix B for the worst case TRIGA fuel element are compared and the greater values for any one isotope are selected as reference case source terms for the MHA. Data are presented in Table 8.1 (Apd-B, Table I) for halogens and noble gases.

The raw data of Table 8.1 are activities potentially released from a single worst-case fuel element that has experienced a cladding failure. This activity may itself be compared to the annual limit of intake (ALI) to gauge the potential risk to an individual worker. By dividing the activity by the 7.075×10^8 cm^3 free volume of the reactor room, one obtains an air concentration (specific activity) that may be compared to the derived air concentration (DAC) for occupational exposure as given 10CFR20 or in EPA federal guidance [Eckerman et al., 1988].

TABLE I

MHA, AVAILABLE ACTIVITY AFTER REACTOR SHUTDOWN

Reference case iodine and noble gas source terms for the maximum hypothetical accident at the AJBRF. Available activity (μCi) of iodine, krypton, and xenon radionuclides from a single worst-case fuel element as a function of time after reactor operation. Data are derived from ORIGEN 2.1 calculations as summarized in SAR, Appendix B, Sub-Appendix B. Only nuclides with half-lives in excess of 2 seconds are presented.

		Available activity (μCi) at time in days after reactor shutdown					
Element	Nuclide	0	1	2	3	7	14
I	129	0.3	0.3	0.3	0.3	0.3	0.3
I	131	249	231	214	197	140	77
I	132	374	311	251	203	87	19
I	133	597	276	124	56	2.3	0.0
I	135	558	45	3.6	0.3	0	0
Kr	83	0.2	0.2	0.2	0.2	0.2	0.2
Kr	83m	47	0.2	0	0	0	0
Kr	84	0.3	0.3	0.3	0.3	0.3	0.3
Kr	85m	112	2.8	0.1	0	0	0
Kr	85	18	18	18	18	18	18
Kr	86	0.7	0.7	0.7	0.7	0.7	0.7
Kr	87	228	0	0	0	0	0
Kr	88	321	0.9	0	0	0	0
Kr	89	411	0	0	0	0	0
Kr	90	406	0	0	0	0	0
Kr	91	308	0	0	0	0	0
Xe	131	1.5	1.5	1.5	1.5	1.5	1.5
Xe	131m	2.8	2.8	2.7	2.7	2.5	2.1
Xe	132	2.2	2.2	2.2	2.2	2.2	2.2
Xe	133m	17	16	13	10	3.1	0.3
Xe	133	598	576	529	475	288	115
Xe	134	4.1	4.1	4.1	4.1	4.1	4.1
Xe	134m	2.1	0	0	0	0	0
Xe	135m	100	7.2	0.6	0	0	0
Xe	135	580	212	44	7.8	0	0
Xe	137	530	0	0	0	0	0
Xe	138	554	0	0	0	0	0
Xe	139	455	0	0	0	0	0
Xe	140	321	0	0	0	0	0

The whole body dose rate for the Most Exposed Worker (MEW) can be estimated by assuming the room is a hemisphere with an equivalent volume and an individual is positioned at the center of the hemisphere. Using this assumption, the dose rate is determined by

$$D = BS_u \frac{\left(1 - e^{-\mu_s R_o}\right)}{2\mu_s g} \qquad (1)$$

where:

D= Dose in mrem/hr
B = Dose buildup factor
R_o = Radius of hemisphere = 696cm
S_u = Source strength (dis/s-cm^3)
μ_s = Linear absorption coefficient (1/cm)
g = dose conversion factor $\left(\frac{\gamma/s - cm^2}{mR/hr}\right)$

It should be noted that values for g and μ_s can differ significantly for each isotope since they are energy dependent. For our calculations, it is assumed each disintegration event produced a gamma whose average energy is shown in SAR, Appendix B, Table II. Using equation 1, it has been determined that the whole body dose for an individual in the room with the initial room concentration for one hour is 6.23×10^{-2} mrem for noble gases. It should be reiterated that this value is conservatively large. The room's air concentration would decrease do to the multiple air changeovers (greater than 3) during this time.

Another mitigating circumstance involves the reactor ventilation system (SAR, Section 3.1) which can be shut down and thus isolate the airflow through the facility. The ventilation is not a designed safety feature and as a result is assumed to be running during the accident scenario. Public doses would however be significantly reduced as expense of occupational exposure.

The air from the reactor room is exhausted by a ventilation fan and two fume hoods. The exhaust fan typically releases 85% of the reactor room air at ground level while the fume hoods provide an elevated release of the remaining 15% on the 12th floor of the Medical Center. In order to consider the worst case of an element release it should be assumed that entire fission product cloud release is at ground level. Referring to SAR Appendix A, figure A-1, the distance to the MMP is from the vent fan to the sidewalk (102 meters) and the distance to the NPR is a commercial building further south (158 meters).

It is assumed that a individual is exposed to a cloud for one hour. Even in very stable atmospheric conditions this would be a conservative assumption. Dose determinations are performed based on Regulatory Guides 1.109 and 1.145. It is recognized that the doses calculated are based on annual dose assessments; however, correcting these totals for a one hour exposure leads to the desired result. Table 8.2 (SAR, Apd-B, Table V) provides a summary of doses associated with the MHA. Based on the calculations and results summarized here and presented in detail in SAR, Appendix B, it is extremely unlikely that 10CFR20 occupational dose limits would be exceeded from a fuel failure. Similarly, radionuclides immediately released from a damaged fuel element to the outside atmosphere

are unlikely to produce doses in excess of 10CFR20 public dose limits when atmospheric dispersion is taken into consideration.

Table 8.2
Dose Summary Maximum Hypothetical Accident

	WHOLE BODY (noble gases) mrem	CDE[a] THYROID mrem	CEDE[b] mrem
MEW	6.23E-02	7.20E2	2.3E1
	WHOLE BODY mrem	**THYROID mrem**	
MMP	3.51E-03	6.26E-01	
NPR	1.49E-03	2.68E-01	

[a] Committed Dose Equivalent
[b] Committed Effective Dose Equivalent

8.1.2. Fission Product Release in Water

The hazards associated with a failure of the fuel-element cladding and consequent fission-product contamination has been studied experimentally [1] and theoretically in SAR Appendix B, Section B.2. The results show that in the improbable event of a cladding failure,

There would be no doses in excess of regulatory limits to workers or members of the public. Release of noble gases xenon and krypton will, by their chemical nature, be only slightly mitigated by the water. Iodine, however, will be retained in the water along with other fission products.

With regard to the water itself, the activity of contaminants other than fission products in the shield water is kept at low levels by equipping the reactor with a demineralizing system. The demineralizer is in a pit 6' 10" below ground adjacent to the medical center basement (see Fig. 3-2, area adjacent to the exhaust fan). Consequently, if the fission products were collected in the demineralizer they would be shielded from the public and ALARA principles would be applied to limit exposure. In the event of a cladding failure, the leaking element would be removed from the core as soon as it is discovered.

8 1.3 Loss of Shielding Water

Because there are many floors in the medical center building immediately above the reactor that are normally occupied, the possibility of loss of shielding water has been considered. This loss of water can occur by only two means: (1) the tank may be pumped dry, or (2) a tank failure may allow the water to drain into the soil.

The tank outlet water line extends only 3 ft below the normal water level. Therefore, even if the water system is operated carelessly if, for example, it is operated when the pump discharge line

has been disconnected for repairs the tank cannot be accidentally pumped dry. This can only be done by deliberate action on the part of the operating crew. In the unlikely event that it is necessary to drain the tank for repairs, the fuel will first be removed in shielded casks. Since the recirculating pump does not have sufficient suction head to drain the tank, another more powerful pump must be installed with its suction line inlet below the core.

Severe earthquake or major settling of the building foundation could possibly cause tank failure. The tank has been designed by Holmes and Narver, a firm experienced in the design of earthquake-proof structures, and there is no record of a structure of this type sustaining damage through earthquake. No earthquake damage has been reported in the Omaha area. The tank has been designed to withstand the existing bearing loads from the building foundation. As described in Section 3.2.1., "Reactor Pit", the reactor tank has been carefully installed so as not to disturb the soil under this foundation. The building has been in existence since 1950 and has exhibited no evidence of foundation failure. There are five barriers, which will prevent water leakage from the tank. Two of these barriers are the waterproof epoxy resin coating and the welded steel tank. The other three barriers would present a very high resistance to water leakage. The gunite, the reinforced concrete, and the adjacent soil itself. The core drilling made at the reactor location shows the soil to be clayey silt and glacial clay, both of which are essentially impervious [3].

Even though the possibility of a loss of shielding water is believed to be exceedingly remote, a calculation has been performed to evaluate the radiological hazard associated with this type of accident. Calculations for this scenario are found in SAR, Appendix C. The exposure summaries are provided in the Table 8.3 (Apd. C, Table I). The radiation from the unshielded core would be highly columnated, so that if an individual did not expose himself directly to the core, he could work in the immediate vicinity of the tank for several hours. He would fill the tank with water from a fire hose and view the interior of the tank with a mirror while making the necessary emergency repairs.

Table 8.3-Waste Level Exposures

Decay Time	Waist-Level Exposure in Air ($R.h^{-1}$)					
	Directly Above Tank (10 - 100cm)		Reactor Room (150 - 175 cm)		Upstairs Room (10 – 100 cm)	
	10 h	1000 h	10 h	1000 h	10 h	1000h
1 minute	97.9	113	3.39	4.05	12.5	14.3
1 hour	25.3	39.6	0.936	1.53	3.12	4.63
1 day	0.141	10.9	0.006	0.466	0.015	1.16
1 week	0.169	5.32	0.007	0.209	0.018	0.589
1 month	0.034	1.69	0.001	0.066	0.004	0.183

To ensure that this accident would not go unnoticed, a float switch is installed in the reactor tank to actuate an audible alarm located at the switchboard. The operator will then notify the Reactor Supervisor or designated alternates in the medical center building, who will take immediate remedial action.

Because the water is required for adequate neutron moderation its removal would terminate any significant neutron chain reaction. However, the residual radioactivity would continue to deposit heat energy within the fuel. Calculations have been produced by General Atomic [4] to determine the maximum fuel temperature rise resulting from a loss of coolant after operation for an infinite time at 250 kW. Results indicate that if the water loss in the core occurs immediately after the reactor has been shut down, the maximum temperature of the fuel, and consequently the aluminum cladding, is less than 150°C. This temperature is such that the pressure exerted by the trapped air and fission product gases is less than 30 psi. This pressure produces a stress of about 660 psi, whereas, the yield stress for the aluminum cladding is greater than 5000 psi at 150°C. Consequently, it is concluded that, subsequent to loss of cooling water after infinite operation at 20 kW, the release of hydrogen from the fuel and the expansion of air and fission gasses in the space between fuel and graphite end pieces will not result in the rupture of the fuel element cladding.

A loss of cooling accident was also analyzed for the Reed College TRIGA reactor, a typical Mark I model fueled with aluminum clad elements using $ZrH^{1.1}$ fuel [5]. The postulated loss-of cooling accident showed that the maximum fuel temperature would be less than 150°C after the infinite operation at 250 kW was terminated by the instantaneous loss of water. At this temperature the equilibrium pressure from fission gases, entrapped air and dissociated hydrogen was reported to produce a stress of only 660 psi which is well below the yield stress of greater than 5000 psi for aluminum cladding at 150°C.

Consequently, it is concluded that after-heat in this reactor following a water-loss accident would be such that the system temperatures would be far below that required to melt the aluminum fuel-element cladding. Therefore, no dispersal of fission products would take place.

It is also concluded that the possibility of loss of shielding water is extremely remote, that the consequences would be unlikely to cause severe injury to personnel or damage to the reactor, and, therefore, that this type of accident does not present a significant hazard to the public.

8.1.4 Failure of the Recorder

Failure of the recorder could lead to inadvertent withdrawal of the regulating rod. Assuming no reactor operator intervention, the reactor would simply Scram once it reached the facility's Scram set points. Assuming failure of the recorder and failure of both %power Scrams, the servo system would only able to manipulate the regulating rod drive, the reactivity insertion is limited to the full worth of the rod. Given a worth less than one dollar, the reactor will remain in a delayed critical configuration. Since the excess reactivity limit for our facility is one dollar and the typical worth of our reg rod is 50 cents; this is a reasonable assumption.

For the prompt critical configuration, the reactor period will be equal to the neutron lifetime divided by the delayed neutron fraction, or approximately 10 ms. From pulsing data at the Kansas State University (KSU) reactor, where the high power scram function is disabled, maximum power has been shown to be less than 2 MW and peak fuel temperatures less than 120°C, both of which are well within the limits of the aluminum fuel. The aluminum fuel has

been shown to withstand pulsing up to $2.00, with peak temperatures to 550°C and peak powers of 250 MW.

Of course, our regulating rod will not add reactivity as quickly as a pulse. Typically, the withdrawal time for the regulating rod from its seated position to full out is approximately 80 seconds. The recorder has 30 and 60 second period controls, which would allow the power to increase no faster than a 30-second period. Under these circumstances, power would reach approximately 21.3 kW before the reactor would scram assuming a percent power scram set point at 20 kW.

8.2. HAZARDS NOT ASSOCIATED WITH THE OPERATION OF THE REACTOR

8.2.1 Mechanical Damage to the Reactor

It is conceivable that a heavy weight, such as a lead transfer cask, could be dropped on the reactor core from above and could smash the core in such a way as to change the fuel-to-water ratio. The designed fuel-to water ratio in the core was selected because this ratio was calculated to give very nearly the minimum critical mass. Consequently, smashing the core is likely to decrease the reactivity, and at worst cannot increase it appreciably. Mechanical damage to the reactor core could cause a fuel-clad failure within the reactor tank and consequently a release of fission products into the water. This type of accident has been analyzed in Section 8.1.1 and Appendix B of the SAR.

8.2.2 Failure of Electric Power

The reactor control system is fail-safe in the event of power failure; i.e., loss of power will de-energize the magnets and release the control rods.

8.2.3 Fire

The medical center building is constructed almost entirely of fireproof materials. The load-bearing walls, the ceilings, and the floors are of reinforced concrete. Carbon dioxide fire extinguishers are located in all halls and laboratories. The reactor room also is equipped with a sprinkler system that is dry until it is charged by heat sensors. The system has heat sensors in the ceiling which activate at 135°C to fill the system and activate an audible alarm. The sprinkler heads do not activate until the temperature reaches 165°C. The sprinkler system is independent of the Medical Center system and can be turned off in the room designated 5W1A on Fig 3.2. If activated, the system will release about 50 gal/min. Aid from the Omaha Fire Department is available in less than ten minutes. The sprinkler was installed because of the requirements of the Veterans Administration that the entire Medical Center have automatic sprinklers. The VA would not allow a dry sprinkler to be installed due to the possibility of toxic vapors. If the sprinkler system was activated it could possibly flood the reactor tank causing the water to overflow. In that case all of the water, the area, and the outer clothing of all individuals involved in the fire would have to be monitored for radioactivity. The Fire Department is briefed yearly on this possibility and is instructed not to leave the area until they are monitored. Through their HAZMAT training they are familiar with radioactivity. The radioactivity of the water, if measurable, would depend on when and for how long the reactor was operated. Six

ml samples of reactor water taken at shut-down and analyzed for 10 hours in a 70 cc well type Ge detector have not shown any significant gamma peaks. Because of the proximity of the fire extinguisher, most fires would be extinguished before the sprinklers are activated. Consequently, the possibility of fire will not contribute significantly to any radiological hazard.

8.2.4 Air Traffic

Eppley Airfield is a commercial airport that lies 5.5 nautical miles north east of the reactor site. Figure 8.1 shows area chart A2 for Kansas City. It contains the Low Altitude (Victor) Airways. The navigational aid, OMAHA (OVR), has twelve different radials which define Low Altitude Airways. Only the low altitude airway V181 on the 295 degree radial would be in proximity to the reactor site. The VA Medical Center would lie in the vicinity of the number "2" in the 295 that is highlighted in yellow. The V181 Airway is from 2000 ft above ground level, 3000 ft Mean Sea Level (MSL), up to 17,000 ft MSL.

The High Altitude chart (Figure 8.2) defines jet airways 18,000 ft MSL and above. The two yellow highlighted "J" routes do not directly overlie and portion of the Omaha downtown area. J10-144 is approximately 10 miles north of Epply Airport and J60-146 is approximately 20 miles south of Eppley Airport. Neither of the "J" routes would be in proximity to the VA Medical Center.

Figure 8.3 shows the airports listed in terms of relative traffic around our facility. Table 8.4 summarizes the operations at airports in the local area. The reactor facility lies within Eppley's Class C airspace; thus, all traffic will be in radar contact and that principal aircraft will have FAA standard type ratings to operate in this airspace.

Table 8.4 summarizes the operations at airports in the local area:

Airport*	# Operations per Day	% Transient General Aviation	% Commercial	% Local	% Air Taxi	% Military
Eppley (OMA)	455	30	30	20	20	<1
Offutt (OFF)	168					100
Millard (MLE)	181	53		45	2	

*Data taken from AirNAV data server on 23 August 1998.

It is concluded that there is very little probability for air traffic to cause harm to the reactor facility. Since the reactor is housed in the basement of a multilevel concrete building is unlikely that an air traffic accident would impact the facility.

8.3 Conclusions

In accordance with the discussion and analysis above we conclude that even in the case of the MHA, the radiation doses to both occupational personnel and to the public in unrestricted areas would be far below the guidelines and limits of 10 CFR 20.1001-20.2401. This fact was also stated in the previous Safety Evaluation Report related to the renewal of the operating license for the research reactor at the Omaha Veterans Administration Medical Center, NUREG-0988, dated July 1983.

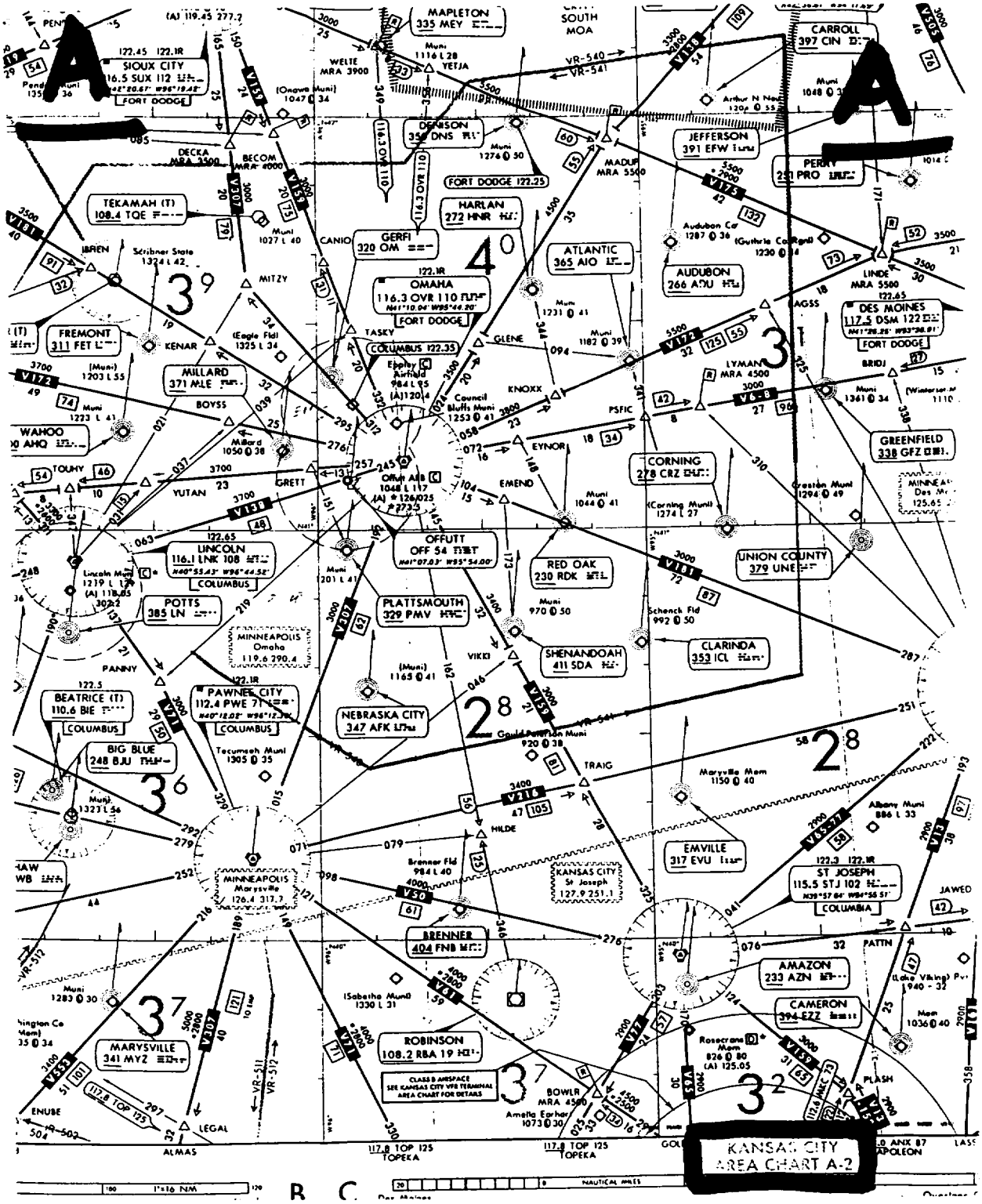


Fig. 8-1 LOW ALTITUDE AIRWAYS

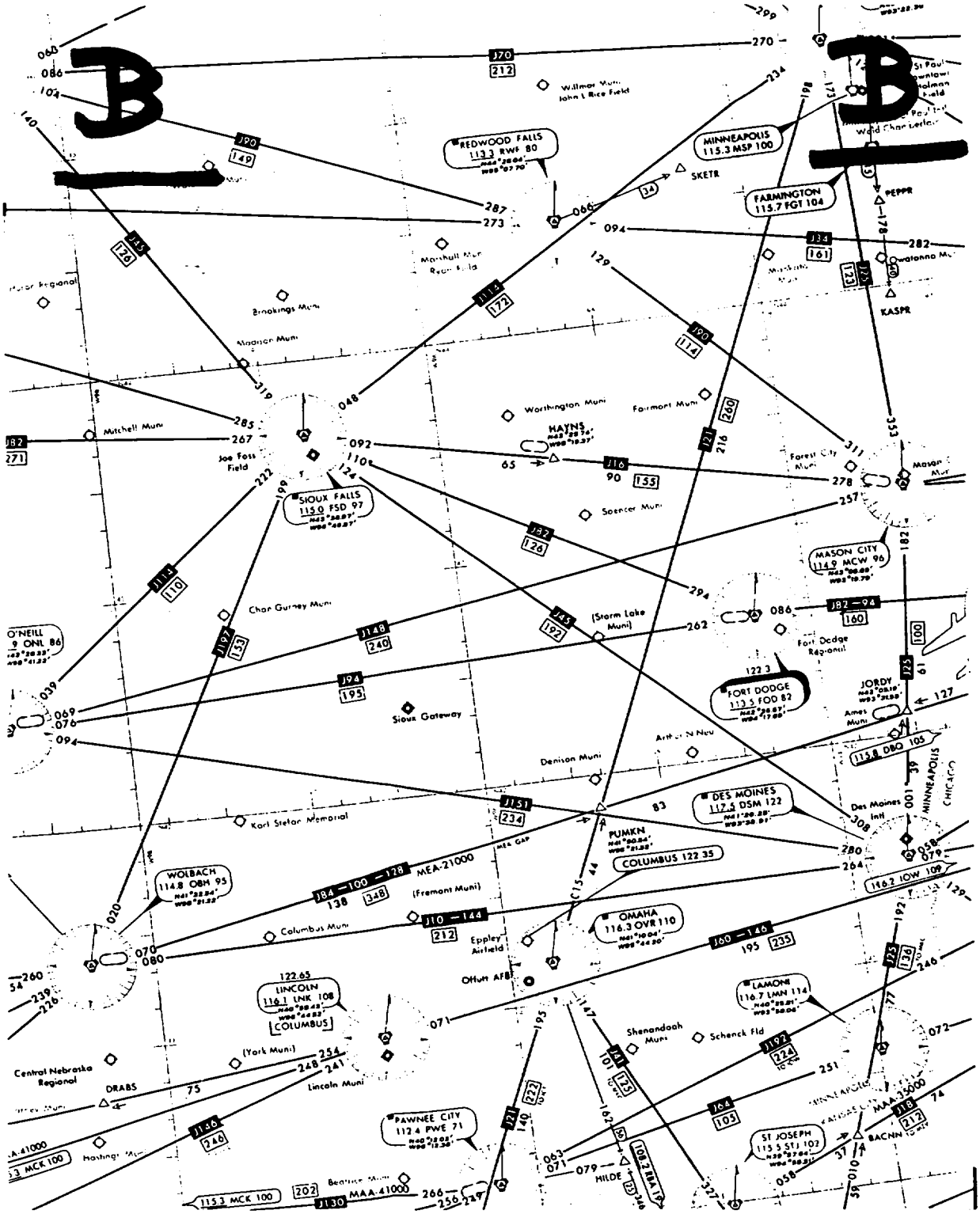


Fig. 8-2 HIGH ALTITUDE CHART

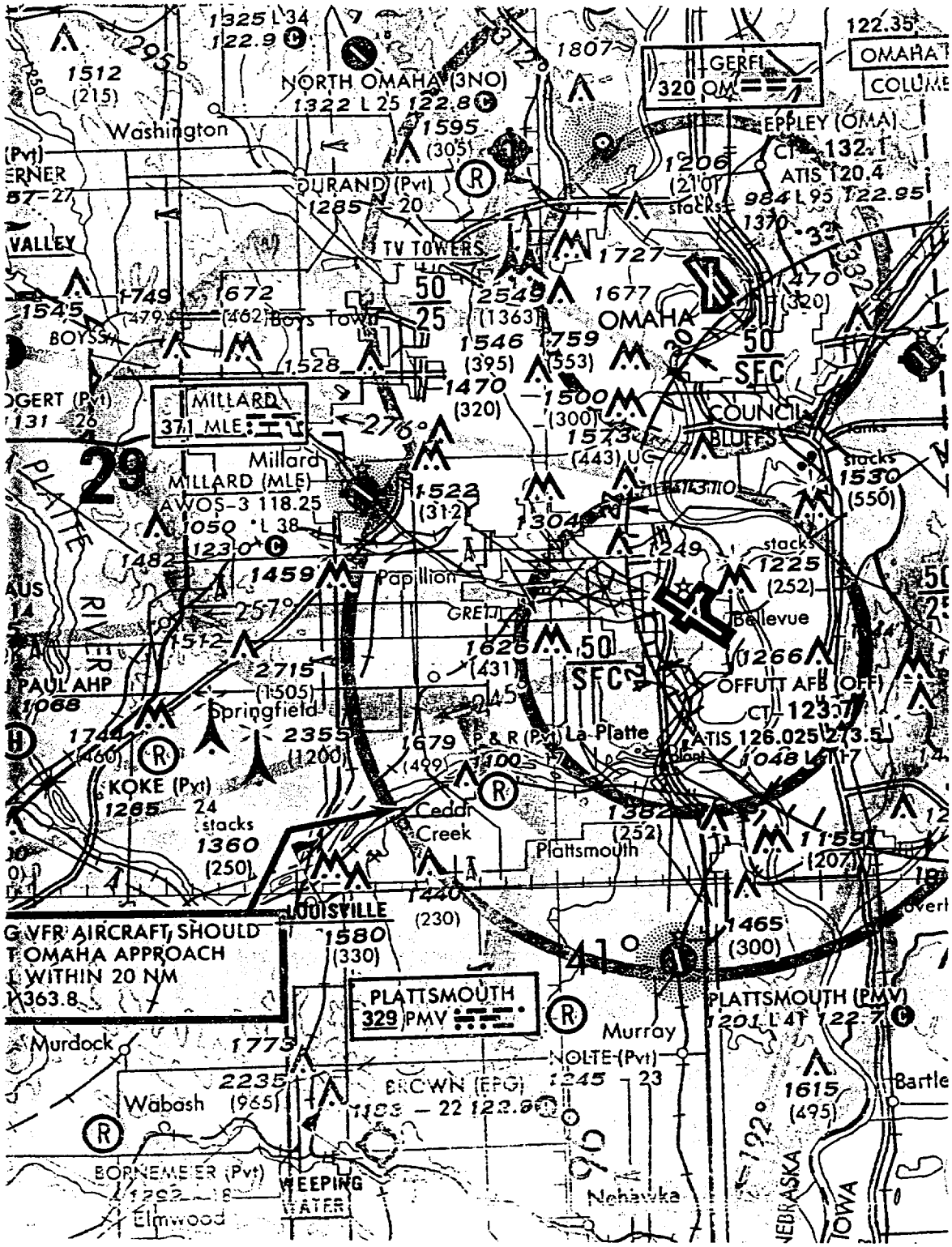


Fig. 8-3 AIRPORT TRAFFIC

Chapter 8

References

1. Hawley, S.S. and Kathren, R.L., "Credible Accident Analysis for TRIGA and TRIGA Fueled Reactors, NUREG/CR-2387, PNL-4028, April 1982.

2. "Technical Foundations of TRIGA", GA-471 (1956), pg 114-115.

3. Way, K. and Wigner, E.P., "Rate of Decay of Fission Products", Phy. Rev. , 73:1318, 1948.

4. Batch J.M. and Foushee, F.C., "Safeguard Analysis Report for TRIGA Reactors using Aluminum Clad Fuel", GA-7860, General Atomic, 1967.

5. "Safety Analysis Report", 1967, Reed College, Portland, Oregon, 1967.

CHAPTER 9

OCCUPATIONAL RADIATION EXPOSURES

9.1 PERSONNEL MONITORING PROGRAM

The AJBRF personnel exposures are measured by the use of film badges assigned to individuals who might be exposed to radiation. TLD neutron dosimeters (or equivalent) capable of detecting neutrons with an energy of 0.06 ev-4.4 MeV are issued to the two reactor operators. In addition, self reading pocket ion chambers or electronic dosimeters are used. Instrument dose rate and time measurements are used to ensure that administrative occupational exposure limits are not exceeded. These limits are in conformance with the limits specified in 10 CFR 20.1001-20.2401 and other applicable regulations.

9.1.1 Personnel Exposures

The AJBRF reactor facility personnel annual exposure history for the last few years is given in Table 9-1.

9.2 EFFLUENT MONITORING

9.2.1 Airborne Effluents

As discussed in Chapter 3, radioactive airborne effluents from the reactor facility consist principally of activated gases. The airborne radioactivity is monitored to provide prompt indication of any abnormal concentrations being discharged to the environment. This is accomplished by withdrawing a representative stream from a point near the top of the reactor through a continuous air monitor. This monitor also is provided with a charcoal filter for collection of iodine radioactivity. The output of the monitor is indicated on a meter having adjustable alarm set points, and a continuous record also is provided. The characteristics of the monitor are described in Section 7.4.1. As shown in Appendix A and Section 3.2.11.2, the total ^{41}Ar vented to the reactor room and then to the environs is well below 10CFR20 limits.

9.2.2 Liquid Effluents

The reactor generates no radioactive liquid effluents. Radioactive liquid waste generated in the research program is discussed in paragraph 7.1.6.

9.3 ENVIRONMENTAL MONITORING

Radioactive gas is the only potentially radioactive material released to the environment as a result of the routine operation of the AJBRF TRIGA. The routine gaseous effluent measurements consist of those recorded by the continuous air monitor, and the exposure date obtained from film badges located within the reactor room, at the exhaust port output, and at the water treatment pit output. The latter represents the airborne exhaust to the environment because the reactor room air is discharged through the water treatment pit. The net integrated

exposure at the water treatment pit output for 1992 was 21 mremad. The position of the pit output monitor was under the 1/32" corrugated steel roof of the pit, 8'7" from the mixed bed resin tank. Considering that the natural background levels in the Omaha area are about 80 mrem per year causing the film badge monitor to detect some background radiation together with some radiation from the resin tank, the exposure is within applicable levels.

A continuous air sampler was operated by the State of Nebraska Health Department's Division of Radiological Health on the roof of the Omaha-Douglas County Hospital (300 m from the AJBRF) for a number of years, primarily as a weapons testing fallout monitor. At no time was any activity detected that could be attributed to the AJBRF operation. This monitoring program was discontinued because there was no further need for the program.

9.3.1 Potential Dose Assessments

Recently the facility has placed a Landauer "Low Level Environmental" monitor outside of the medical center, and for the third quarter of 1994 obtained a net reading of 23.8 mrem. This would be consistent with the 80 mrem/year background level stated above.

Based on the calculations in this SAR it is felt the release rates from this facility fall well within the requirements of 10 CFR 20.1001-2401 and comply with ALARA concepts.

Table 9-1

Recent Exposure History of Reactor Facility Personnel

	Whole body exposure mrem/year					
Individual	1988	1989	1990	1991	1992	1993
1	20	30	10	40	≤15	≤15
2		≤15	≤15	30	≤15	≤15

APPENDIX A

Airborne Releases during Normal Operation

A.1 RELEASE OF ARGON-41 FROM REACTOR WATER

The argon-41 activity in the reactor pool water results from irradiation of the air dissolved in the water.

The following calculations were performed to evaluate the rate of argon-41 escaping from the reactor pool water into the reactor room. The calculations show that the argon-41 decays while in the water (note the $-\lambda N^{41}V$ terms in eq. 1-3), and most of the radiation is safely absorbed in the water. The changes in argon-41 concentration in the reactor, in the pool water external to the reactor, and in the air of the reactor room are given by:

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \Phi N_1^{40} \sigma^{40} - N_1^{41}(\nu_1 + V_1 \Phi \sigma^{41} + \lambda^{41} V_1) + N_2^{41} \nu_1, \quad (1)$$

$$V_2 \frac{dN_2^{41}}{dt} = -\lambda^{41} N_2^{41} V_2 + \nu_1 (N_1^{41} - N_2^{41}) - (f_{2\to3} N_2^{41} V_2 - f_{3\to2} N_s^{41} V_3), \quad (2)$$

$$V_3 \frac{dN_3^{41}}{dt} = (f_{2\to3} N_2^{41} V_2 - f_{3\to2} N_3^{41} V_3) - N_3^{41}(\lambda^{41} V_3 + q), \quad (3)$$

Where:

subscript 1	= Reactor region (water region in core),
subscript 2	= Reactor tank water region external to the reactor,
subscript 3	= Reactor room region,
Superscript 40	= Argon-40,
Superscript 41	= Argon 41,
Superscript A	= Argon-40 plus argon-41,
V	= Volume of region, cm^3,
N	= Atomic density, atoms/cm^3,
λ	= Decay constant, sec^{-1},
σ	= Absorption cross section, cm^2,
q	= Volume flow rate from reactor room exhaust (cm^3/sec),
ν_1	= Volume flow rate through region No. 1 (cm^3/sec),
Φ	= Average thermal neutron flux in Region No. 1, (n/cm^2-sec),
$f_{i\to j}$	= Fraction of argon-41 atoms in region i that escape to region j per unit time, sec^{-1}.

To estimate the volume flow rate of the water in the reactor, the following equation is used:

$$\nu_1 = \frac{Q}{C_p \delta T \rho} \tag{4}$$

Where:

ν_1 = Volume flow rate of the water through the core,
Q = Reactor power 20 X 10^3 watts,
C_p = Specific heat of water ≅ 4.19 watt-sec/g-°C,
δT = Temperature rise across the core ≅ 11°C,
r = Exit water density = 0.99862 g/cm^3.

Thus:

$$\nu_1 = \frac{20x10^3}{(4.19)(11)(0.99862)} \frac{cm^3}{\text{sec}} = 4.35x10^2 \ \frac{cm^3}{\text{sec}}$$

Equation (1) can be reduced to

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \Phi N_1^{40} \sigma^{40} - (N_1^{41} - N_2^{41}) \nu_1$$

by considering the following numbers:

ν_1 = 4.35 x 10^2 cm^3/sec,
V_1 = 1.45 x 10^4 cm^3,
Φ = 4.8 x 10^{11} n/cm^2-sec,
σ^{41} = 0.060 x 10^{-24} cm^2,
λ^{41} = 1.06 x 10^{-4} sec^{-1}.

to show that:

$$\nu_1 + V_1 \Phi \sigma^{41} + \lambda^{41} V_1 \cong \nu_1$$

During equilibrium conditions the three equations reduce to:

$$V_1 \Phi N_1^{40} \sigma^{40} = (N_1^{41} - N_2^{41}) \nu_1 \tag{5}$$

$$N_2^{41} \lambda^{41} V_2 + f_{2\to3} V_2 = \left(N_1^{41} - N_2^{41}\right) \nu_1 + f_{3\to2} N_3^{41} V_3 \tag{6}$$

Combining equations (5) and (6) gives

$$N_3^{41}\left[\lambda^{41}V_3 + q + f_{3\to 2}V_3\right] = f_{2\to 3}N_2^{41}V_2, \tag{7}$$

$$N_2^{41} = \frac{V_1\Phi N_1^{40}\sigma^{40}}{\lambda^{41}V_2 + f_{2\to 3}V_2} + \frac{f_{3\to 2}N_3^{41}V_3}{\lambda^{41}V_2 + f_{2\to 3}V_2}, \tag{8}$$

which inserted into equation (7) for N_2^{41} yields

$$N_3^{41}\left[\frac{\lambda^{41}V_3 + q + f_{3\to 2}V_3}{f_{2\to 3}V_2} - \frac{f_{3\to 2}V_3}{\lambda^{41}V_2 + f_{2\to 3}V_2}\right] = \frac{V_1\Phi N_1^{40}\sigma^{40}}{\lambda^{41}V_2 + f_{2\to 3}V_2} \tag{9}$$

The values of constants in equation (9) are

V_2 = 0.189×10^8 cm^3 (6.5 ft. dia. x 20.08 ft. high)
V_3 = 7.075×10^8 cm^3 (1951 ft^2 x 12.83 ft)
q = 1.12×10^6 cm^3/sec
σ^{40} = 0.47×10^{-24} cm^2

which leaves the following to be evaluated:

$$N_1^{40},\ f_{2\to 3},\ f_{3\to 2},\ N_3^{41}$$

The argon activity in the reactor pool water results from argon dissolved in water. In the calculation to determine the amount of argon dissolved in the pool water, the assumption was made that argon follows Henry's law. If the water temperature is taken to be 70°F, the corresponding water vapor pressure is 26 mm Hg. The partial pressure of air is then 760 - 26 = 734 mm Hg. The argon content of air is 0.94% by volume; hence, the partial pressure of argon is 734 x (9.4×10^{-3}) = 7 mm Hg. Henry's Law Constant for Argon in water is 0.00151 (moles/liter) per atmosphere at 25°C. Thus, the moles of argon per cm^3 is determined by:

$$\left(\frac{7mmHg}{760mmHg/atm}\right)\left(0.0015\ moles/L/atm\right)\left(0.001\ L/cm^3\right) = 1.39x10^{-8}\ \text{moles Ar/cm}^3\ \text{water}$$

This yields

$1.39 \times 10^{-8} \times 6.02 \times 10^{23} = 8.37 \times 10^{15}$ argon atoms/cm^3 H_2O.

The model used for the exchange rate is based on the compilation of data presented by Dorsey [A1]. The model presented gives the exchange rate across a liquid surface to be:

$$\frac{dm}{d\tau} = (\alpha P - \beta c)A$$

where dm is the amount of gas passing through area A in time τ, P is the partial pressure of the gas, c is the concentration of gas in the liquid, and α and β are the entrance and exit coefficients, respectively. For this analysis, it is assumed that the ^{41}Ar comprise only a small fraction of the total argon and hence the entrance coefficient α will be equal to zero. On the other hand, the concentration of gas in the liquid will be based on the argon concentration in the water, as the ^{41}Ar will behave chemically in a similar manner to other argon isotopes.

The temperature dependence of such a system produces a larger exit coefficient at higher temperatures. The A.J. Blotcky Reactor Facility (AJBRF) has a refrigerant cooling system with typical coolant temperatures around 7°C. In order to be on the conservatively high side, the values used will be for 20°C. Additionally, values for argon are not listed in the Dorsey's reference, so values for air will be used, which are some of the larger values on the chart. At 20°C, the exit coefficient of air in water is 0.534 cm/min = 8.90×10^{-3} cm/s.

As previously stated, the concentration of argon in the coolant is $8.37 \times_{\times} 10^{15}$ atoms Ar/cm^3 and in the air is $2.10 \times_{\times} 10^{17}$ atoms Ar/cm^3. Of course, this is a function of the water temperature relative to the air. The room volume is 7.075×10^8 cm^3 and the tank volume is 1.890×10^7 cm^3. The surface area of the tank is 3.083×10^4 cm^2.

A.1.1 Calculations

Using the fact that c is the concentration in the volume of the tank V_2 and that we want the fractional release from the water to the air $f_{2\rightarrow 3}$ (s^{-1}), the previous equation may be altered to read:

$$f_{2\rightarrow 3} = \frac{dm}{m_2}\frac{1}{d\tau} = \frac{\beta_2 A_s}{V_2}$$

where A_s is the surface area of the tank. Substituting numbers $f_{2\rightarrow 3} = 1.45 \times_{\times} 10^{-5}\ s^{-1}$. At equilibrium, the reciprocal transfer is related by the source volumes:

$$f_{3\rightarrow 2} = f_{2\rightarrow 3}\frac{N_2 V_2}{N_3 V_3} \qquad (10)$$

where $N_{3,2}$ are the concentrations in air and the tank, respectively and V_3 is the volume of air. Therefore, the fractional transfer is $f_{3\rightarrow 2} = 1.28 \times_{\times} 10^{-9}\ s^{-1}$. During equilibrium conditions and assuming no difference in the rates of escape fraction for argon-40 and argon-41, the number of

argon atoms that escape from the water into the air equals the number of argon atoms that enter the water from the air, i.e.,

$$f_{2\rightarrow 3} N_2^A V_2 = f_{3\rightarrow 2} N_3^A V_3, \quad \textbf{(11)}$$

where:

N_3^A = 2.10 x 10^{17} argon atoms/cm^3 of air $\cong N_3^{40}$,
N_2^A = 8.37 x 10^{15} argon atoms/cm^3 of water $\cong N_1^{40}$,

solving for $f_{3\rightarrow 2}$ gives

$$f_{3\rightarrow 2} = f_{2\rightarrow 3} \frac{N_2^A V_2}{N_3^A V_3} \quad \textbf{(12)}$$

$$= 1.54 \times 10^{-8} \text{ sec}^{-1}$$

Since $\lambda^{41} > f_{2\rightarrow 3} > f_{3\rightarrow 2}$, equation (9) reduces to

$$N_3^{41} = \frac{V_1 \Phi N_1^{40} \sigma^{40}}{(\lambda^{41} V_3 + q)} \frac{f_{2\rightarrow 3}}{\lambda^{41}} \quad \textbf{(13)}$$

where:

V_1 = 1.45 x 10^4 cm^3
Φ = 4.8 x 10^{11} n/cm^2-sec
N_1^{40} = 8.37 x 10^{15} argon atoms/cm^3
σ^{40} = 0.47 x 10^{-24} cm^2
$f_{2\rightarrow 3}$ = 1.45 x 10^{-5} atom/sec
λ^{41} = 1.06 x 10^{-4} sec^{-1}
q = 1.12 x 10^6 cm^3/sec
V_3 = 7.075x10^8 cm^3

Solving for N^{41} yields 3.13 atoms/cm^3. This corresponds to a concentration of argon-41 activity of

$$A^{41} = \frac{\lambda^{41} N^{41}}{C} = \frac{1.06x10^{-4} x3.13}{3.7x10^4} = 8.97x10^{-9} \mu Ci/cm^3 \quad \textbf{(14)}$$

where:

A^{41} = Argon-41 concentration, μc/cm^3
C = Conversion factor from disintegration/sec to μCi.

For a typical workday, applying a factor of 0.68 takes into account the buildup time of 8 hours will produce a value of 6.10×10^{-9} μCi/cm^3. This is below the occupational emersion DAC value of 3×10^{-6} μCi/ cm^3 as listed in Appendix B of 10CFR20.1001-20.2401 and complies with the provisions of 10CFR20.1302.

The air concentration value above would actually be less since the hatches to the reactor tank are kept closed and have 1/4 inch plastic sheets under the grates which allow only 2% of the tank to be uncovered.

A.1.2 Reactor Room Dose Calculations

The whole body dose rate for the Most Exposed Worker (MEW) can be estimated by assuming the room is a hemisphere with an equivalent volume and an individual is positioned at the center of the hemisphere [A2]. Using this assumption, the dose rate is determined by

$$D = BS_u \frac{\left(1 - e^{-\mu_s R_o}\right)}{2\mu_s g} \qquad \textbf{(15)}$$

where:

D= Dose in mR/hr
B = Dose buildup factor
R_o = Radius of hemisphere = 696cm
S_u = Source strength (dis/s-cm^3)
μ_s = Linear absorption coefficient (1/cm) = 8×10^{-5} for air
g = dose conversion factor $\left[\frac{\gamma/s - cm^2}{mrem/hr}\right]$ = 5.5×10^2 for Ar-41

Using equation 15, it has been determined, using conservative assumptions (no build up factor), that the whole body dose for a individual in the room with a concentration of 8.97×10^{-09} μCi/cm^3 for 2000 hours is 3.9×10^{-1} mR.

A.2 OFFSITE DOSE CALCULATIONS

The argon-41 activity discharge rate from the reactor room is obtained by multiplying the activity concentration by q, the value of air discharged per unit time, that is

A_q^{41} = 8.97×10^{-9} μCi/cm^3 x 1.12×10^6 cm^3/sec
= 1.00×10^{-2} μCi/sec

Calculations of dose to the Most Exposed Member of the Public (MMP) and the Nearest Permanent Resident (NPR) were determined following the criteria specified in Regulatory Guides 1.109 [A3] and 1.145 [A4]; calculations were made to determine the potential dose to the public outside the facility. Where as the infinite cloud approach is an appropriate methodology to use;

it cannot be stressed enough that its use is highly conservative. The total body dose for ground release of noble gas is given by

$$D_{\infty}^{T}(r,\theta) = S_F \sum_i x_i(r,\theta) DFB_i \qquad (16)$$

where:

DFB_i = the total body dose factor for a semi-infinite cloud of the radionuclide i which including the attenuation of 5 g/cm^2 of tissue in mR/pCi-yr;

$D_{\infty}^{T}(r,\theta)$ = the annual total body dose due to immersion in a semi-infinite cloud at the distance r in sector θ, in mR/yr;

S_f = a shielding factor that accounts for the dose reduction due to shielding provided by residential structures during occupancy, dimensionless; and

$\chi_i(r,\theta)$ = the annual average ground-level concentration of radionuclide i at the distance r in sector θ, in pCi/m^3

$\chi_i(r,\theta)$ is determined by

$$\chi_i(r,\theta) = 3.17 \times 10^4 \, Q_i \, [\chi/Q]^D(r,\theta) \qquad (17)$$

and

$$Q_{ni}^{D}\left(\chi/Q\right) = Q_i \exp\left(-\lambda_i r / U_n\right)\left(\chi/Q\right)(r,\theta)$$

$[\chi/Q]^D(r,\theta)$ is the gaseous dispersion factor (corrected for radioactive decay) in the sector at angle θ at the distance r from the release point, in s/m^3. 3.17 x 10^4 is the number of pCi/Ci divided by the number of seconds in a year. In addition, λ_i is the decay constant, U_n is the average windspeed, Q_i is the initial point of release in Ci/yr and χ/Q is further defined as below.

For atmospheric stability conditions when the windspeed at the 10-meter level is greater than 6 meters per second horizontal plume meander may not be considered. χ/Q values may be determined from the set of equations specified in Regulatory Guide 1.145. For this particular scenario the windspeed is presumed to be 10 m/s (typical annual average) with a Pasquill stability class of F to generate a conservative χ/Q. With respect to our specific parameters, the atmospheric diffusion can be described by

$$\chi/Q = \frac{1}{U(3\pi\sigma_y\sigma_z)} \qquad (18)$$

where:

χ = the short term average centerline value of the ground level concentration in Ci/m^3

Q = the amount of material released Ci/s

U = the average windspeed m/s

σ_y = is the lateral plume spread which is a function of the atmospheric stability and distance (Fig.1, Regulatory Guide 1.145)

σ_z = is vertical plume spread which is a function of the atmospheric stability and

distance (Fig.2, Regulatory Guide 1.145)

In using equation 16, the terms S_f, $\chi_i(r,\theta)$, and DFB_i must be resolved. In our particular conditions, there is no account due to shielding provided by residential structures. Thus, S_f is equal to 1. Values for DFB_i were obtained from Table B-1, p.21, of reg guide 1.109 (γ-Body). $\chi_i(r,\theta)$ is obtained from equation 17 and 18. Q_i is based on the release rate A_q[41] for a calendar year ($3.15x10^7$ sec.). Thus the exploded version is

$$\chi_i(r,\theta) = 3.17x10^{4} \frac{pCi/Ci}{s/yr} x3.2x10^{-1} \frac{Ci}{yr} x \exp\left(\frac{-1.103x10^{-4} s^{-1} x102m}{10\frac{m}{s}}\right)\left(\frac{1}{10\frac{m}{s}(3\pi x4mx2.3m)}\right)$$

$= 1.15x10^{1}$ pCi/m^3

Thus, the Dose for the Most Exposed Member of the Public (MMP) (102m, see fig. A-1) due to ^{41}Ar released from the pool during normal operation is

$$D_{\infty}^{T}(r,\theta) = 1x\left(11.5\frac{pCi}{m^3}\right)\left(8.84x10^{-3}\frac{mrem-m^3}{pCi-yr}\right)$$

$= 1.02x10^{-1}$ mrem/yr.

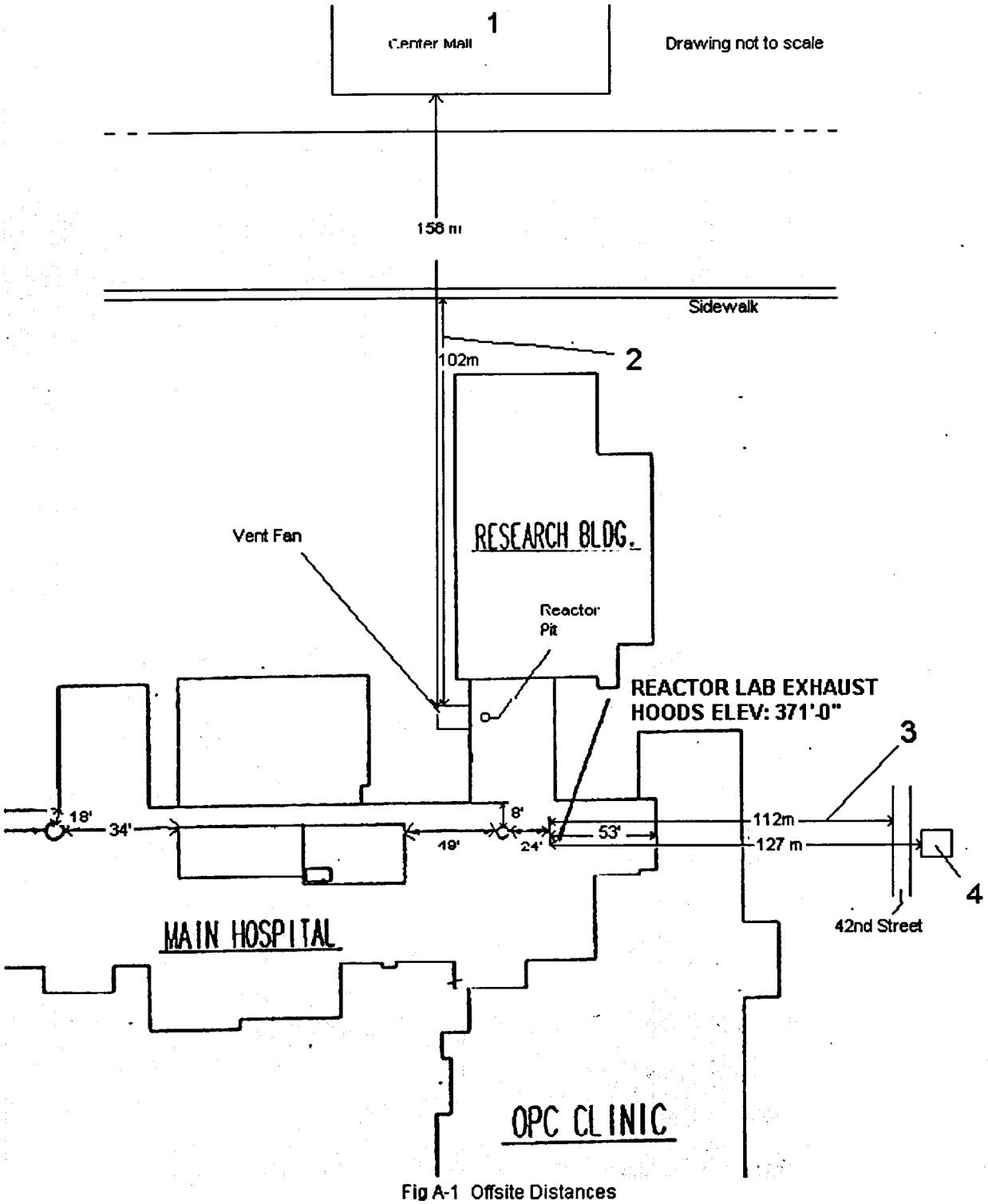


Fig A-1 Offsite Distances

1. NPR from cooling pit exhaust (Commercial Building)
2. MMP from cooling pit exhaust
3. MMP from the roof exhaust
4. NPR from the roof exhaust (Residential House)

Figure A-1

A.3 NITROGEN-16 ACTIVITY IN REACTOR ROOM

The cross section threshold for the oxygen-16 (n,p) nitrogen-16 reaction is 9.4 MeV; however, the minimum energy of the incident neutrons must be about 10.2 MeV because of center of mass corrections. This high threshold limits the production of nitrogen-16 since only about 0.1% of all fission neutrons have an energy in excess of 10 MeV. Moreover, a single hydrogen scattering event will reduce the energy of these high-energy neutrons to below the threshold.

The measured effective cross section for oxygen-16 (n,p) nitrogen-16 reactions regarding the fission neutron spectrum is 1.85×10^{-29} cm^2 [A5]. This value agrees well with the value obtained from integrating the effective cross section over the entire fission spectrum.

The concentration of nitrogen-16 atoms per cm^3 of water as it leaves the reactor core is given by

$$N^N = \frac{\Phi_v N^O \sigma^O}{\lambda}\left[1 - e^{-\lambda t}\right] \qquad \textbf{(19)}$$

Where:

- N^N = Nitrogen-16 atoms per cm^3 of water,
- Φ_v = Virgin fission neutron flux $\cong 4.8 \times 10^{11}$ n/cm^2-sec at 20 kW
- N^O = Oxygen atoms per cm^3 of water = 3.3×10^{22} atoms/cm^3
- σ^O = Absorption cross section of oxygen = 2×10^{-29} cm^2
- λ = Nitrogen-16 decay constant = 9.7×10^{-2} sec^{-1}
- t = Average time of exposure in reactor

The average exposure time in the reactor is given by

$$t = \frac{V_c}{v_1} \qquad \textbf{(20)}$$

where V_c is the core water column exposed to flux Φ_V. and v_1 is the volume flow rate through the core (see section A1). Thus,

$$t = \frac{1.45x10^4\, cm^3}{4.35x10^2\, cm^3 / \sec} = 33.3 \sec$$

Solving for N^N from equation (19), one obtains 3.14×10^6 nitrogen-16 atoms per cm^3 of water leaving the core. With a flow of 435 cm^3/sec, the rate of nitrogen-16 leaving the core is therefore 1.36×10^9 atoms/sec.

The relatively short half-life associated with N-16 decay requires a somewhat different approach that was used in Ar-41 determinations. Thus the transport time from the core is a much more relevant factor. In the TRIGA Mark I, the measured transport time for the water to travel the 16 feet from the reactor core to the surface of the tank is 42 seconds when the reactor is operating at

100 kW [A6]. To a first approximation the velocity of the rising water is proportional to the density difference between the pool water and the heated water from the core, that is,

$$v_2 = K(\rho_0 - \rho_{exit})$$

Thus the velocity of the rising water column for the AJBRF TRIGA Mark I can be estimated from

$$v_2 = \frac{v'(\rho_0 - \rho_{exit})}{(\rho'_0 - \rho'_{exit})}$$

Where,

v_2 = the velocity of the rising water column (AJBRF)
ρ_0 = initial density (AJBRF)
= 0.99987 g/cm^3
ρ_{exit} = exit density (AJBRF)
= 0.99862 g/cm^3
v' = the velocity of the rising water column density (reference reactor [A6])
= 16 feet/42 sec
ρ'_0 = initial density (reference reactor [A6])
= 0.99705 g/cm^3
ρ'_{exit} = exit density (reference reactor [A6])
= 0.99021 g/cm^3

The transport time for nitrogen-16 through the 16 ft. of water above the reactor core is then 230 seconds.

This assumes that the nitrogen-16 rises straight up toward the pool surface. In practice, however, the nitrogen-16 is slowed down by the interruption of the vertical convective currents from the discharge of water horizontally from the coolant flow return pipe. In 230 seconds, the nitrogen-16 decays to 1.95 x 10^{-10} of its initial value. Thus, the number of nitrogen-16 atoms that reach the water near the pool surface is about 0.27 atoms/second.

Only a small portion of the nitrogen -16 atoms present near the pool surface is transferred into the air of the reactor room. When a nitrogen-16 atom is formed, it appears as a recoil atom with various degrees of ionization. For high-purity water (~ 2 μmho), practically all of the nitrogen-16 combines with oxygen and hydrogen atoms of the water. Most of it combines in an anion form, which has a tendency to remain in the water [A5]. It is assumed that at least one-half of all ions formed are anions. Because of its 7.1-second half-life, the nitrogen-16 will not live long enough to attain a uniform concentration in the tank water. Assume that the nitrogen-16 atoms will be dispersed in the 1 ft. of water at the top of the pool directly above the core. In the area directly above the core, the dominant contribution to the dose rate is the direct radiation from the core. It should be noted that the extremely minimal airborne release of nitrogen –16 (0.27 atoms/s) is insignificant with regard to inhalation dose. Thus, there is no further calculation due to dose from inhalation.

However, it may be useful to calculate dose due to shine from the nitrogen –16. For the purposes of analysis, it is postulated that the water-bearing N-16 rises from the core to the surface and then spreads across a disk source with a radius of 100 cm and area A_s = 3.14 x 10^4 cm^2.

For a constant velocity of v_2; the cycle time for distributing the N-16 over the tank surface would be:

$$t_s = \frac{r}{v_2} = \frac{100cm}{2.16cm/s} = 46s$$

The average concentration at this time is

$$\overline{N} = \int_0^{t_s} \frac{[N_0 e^{-\lambda t}\, dt]}{t_s} = \frac{N_0}{\lambda t_s}(1 - e^{-\lambda t_s}) = \frac{9.01x10^{-4}}{9.7x10^{-2}x46}\left(1 - e^{-9.7x10^{-2}x46}\right) = 2.00x10^{-4}\,\frac{atom}{cm^3} \qquad \textbf{(21)}$$

The thickness of the layer of N-16 bearing water is:

$$h = \frac{v_1 t_s}{A_s} = \frac{4.35x10^2 x46}{3.14x10^4} = 6.37x10^{-1}\,cm \qquad \textbf{(22)}$$

The dose rate at the tank surface arising from the N-16 near surface is

$$D = \frac{\lambda \overline{N}}{2\mu K}\left[1 - E2(\mu h)\right] \qquad \textbf{(23)}$$

Where μ is the attenuation coefficient for 6 MeV photons in water with a value of (0.0277 cm^{-1}), K is the flux-to-dose-rate conversion:

$$1.6x10^5\,\frac{photons/cm^2 s}{r/hr}$$

and E2 is the second exponential function

$$E_2(\mu h) = \int_1^\infty \frac{e^{-\mu h T}}{T^2}\,dT$$

It may be worth noting that equations 15 and 23 are quite similar as they reflect related physical phenomenon. For x << 1 the interval of integration is taken to be from 1 to 10/x as a close approximation. It can also be seen that since

$$e^{-xt} = 1 - \mu ht + \frac{(\mu ht)^2}{2!} - \frac{(\mu ht)^3}{3!} + \cdots$$

then $e^{-\mu ht} \cong 1 - \mu ht$. Thus, the dose equation becomes

$$D = \frac{\lambda \bar{N}}{2\mu K}\left(1 - \int_{x}^{10} \frac{1-\mu hT}{T^2}\, dT\right). \qquad (24)$$

x, t, and T are variables used in approximating the E2 integral. This yields a dose rate of 1.71 x 10^{-7} mR/hr where the target individual is within the reactor room very close to the source. This negligible exposure precludes the necessity of any further dose calculations to individuals at greater distances. The production of N-16 is of no significance in terms of the dose received and poses no health hazard.

A.4 ACTIVATION OF AIR IN THE EXPERIMENTAL FACILITY

In the AJBRF, the rotary specimen rack and pneumatic transfer tube contain air. Of the radioisotopes produced in these air cavities, argon-41 (with a half-life of 110 min.) is the most significant with respect to airborne radioactivity hazards. Nitrogen-16 (7.11 sec half-life) and oxygen-19 (26.9 sec half-life) are considerably less significant.

Volumes and thermal neutron fluxes of facilities are as follows:

Facility	Effective air volume (cm^3)	Ave. Thermal Flux at 20 kW (n/cm^2-sec)x10^{11}
Rotary Specimen Rack	3.3 x 10^4	1.36
Pneumatic Tube	2 x 10^2	3.74

A.4.1 Pneumatic Tube (PT)

This facility exhausts into a fume hood which, in turn, exhausts to the roof of the hospital. Assuming a maximum of 2000 hours operation at 20kW full power, as this is a full working year cycle. During the last 3 years the following total number of samples were irradiated in all of the experimental facilities.

Year	Number of samples
1994 (6mo)	2,294 (2x1147)
1993	2,813
1992	1,995

Assuming all of the samples were irradiated in the pneumatic tube and that the blower for the pneumatic tube was used for 30 sec for each irradiation (sample transfer time = 2 sec). A check, with a volometer, of the input air ducts to the pneumatic tube showed that the only time air is circulated through the PT tube is when the blower is on for the 30 seconds per sample. The blower is turned off while the sample is being irradiated. Calculations will therefore be based on 3000 samples irradiated a year. As a further assumption, there is no elapsed time between samples. This will yield the largest production time, and hence the largest value for activity. This

leaves a production time of 40 minutes between samples. The pneumatic tube has an inner diameter of roughly 2.54 cm inner diameter. The tube extends 6 cm below the core centerline, with a solid aluminum foot connecting to the lower grid plate. The effective length of the tube in the core is 35 cm. Flux profile measurements at similar facilities indicate that the flux decreases by at least one order of magnitude from the center to the top core plate. Therefore taking the peak flux value of this location of 3.74×10^{11} n per $cm^{-2}\cdot s^{-1}$ as the average value, a conservatively large estimate will be reached. Based on this, the effective length will be chosen conservatively as 40 cm, which is still below the location where the return air pipe joins into the main tube. Therefore the effective volume of air will be 200 cm^3.

The ^{41}Ar activity (Ci) produced in the tube per irradiation is given by:

$$\lambda_{41}{}^{41}\mathrm{Ar} = \frac{\varphi \sum_a V}{3.7x10^{10}}(1 - e^{-\lambda_{41} t_{irr}}) \qquad \textbf{(25)}$$

Where:

λ_{41} = Decay constant (1.063×10^{-4} s^{-1})
φ = Thermal neutron flux density (3.74×10^{11} n/cm^2 s)
Σ_a = Macroscopic absorption cross section (9.86×10^{-8} cm^{-1})
V = Volume (200 cm^3)
t_{irr} = Irradiation time (2400 s)

Substituting these values, the total activity per discharge is 4.48×10^{-5} Ci. Multiplying by 3000 discharges per year gives a value of 0.134 Ci/y.

Since the fume hood is operated continuously, the annual volume of air vented is as follows:

Fume hood exhaust = $2.37x10^5$ cm^3/sec
Number of sec/yr = $31.536x10^6$ sec/yr

Therefore:

$2.37x10^5$ cm^3/sec x $31.536x10^6$ sec/yr = $7.47x10^{12}$ cm^3 air exhausted per year.

Thus the average concentration will be 0.134 Ci y $\div 7.47x10^{12}$ cm^3/y = 1.79×10^{-14} Ci/cm^3.

Results:

The total release of ^{41}Ar was estimated at 0.134 Ci/y, with the concentration in air leaving the building = 1.79×10^{-14} Ci/cm^3.

Calculations of dose to the MMP and the NPR were again determined as in section A.2 based on the more conservative ground level releases rather than elevated release in this case. Plumes of gaseous effluents are considered semi-infinite in the case of ground-level noble gas releases; while releases within a room are typically finite. The distances to the MMP and NPR vary slightly as the initial point of exhaust is assumed to be at the hood location (refer Fig. A-1).

A.4.2 Rotary Specimen Rack (RSR)

Two approaches are taken for ^{41}Ar determinations. One with convection flow plus samples (upper limit) and a second for samples alone which is more realistic as the sample cover is always covered when not in use. It must be understood that the upper limit determination uses parameters that are well beyond the scope of normal conditions. The sample hole is always covered during when not in use and annual operation times are usually much less than 2000 hours. Nonetheless it may serve as a near worse case scenario.

Following are assumptions and approximations applied to calculations.

1. The reactor operates 2000 h per year at full power (20 kW), creating a neutron flux Φ of 1.36×10^{11} n/cm^2 s in the RSR.
2. The maximum flow rate due to natural convection (sample hole uncovered) is 10 lfm. For an exit tube of 3.39 cm tube, this gives a volumetric flow rate q of 45.9 cm^3/s.
3. 100 samples are inserted from the RSR per year. However, for each sample the reactor is shutdown for an extended period either prior to insertion (most common) or after (to allow sample decay). Thus on average it is assumed that only 100 transitions of the loading tube by a sample vial will occur. This displaces an additional 4400 cm^3 of air per sample, or 440,000 cm^3/y.

For ^{41}Ar production and decay the time rate of change of ^{41}Ar atoms in the RSR is given by:

$$\frac{dN}{dt} = \Sigma_a \phi - \lambda N - \frac{Nq}{V}, \tag{26}$$

where N is the number of ^{41}Ar atoms per cm^3, Σ_a is the macroscopic absorption cross section = 0.986×10^{-7} cm^{-1}, λ is the decay constant = 1.06×10^{-4} s^{-1}, q is the volumetric flow rate, and V is the volume of the RSR = 3.3×10^4 cm^3. Solving:

$$\lambda N = \frac{\lambda \Sigma_a \phi}{(\lambda + \frac{q}{V})}\left(1 - e^{-(\lambda + q/V)t}\right)$$

The average of the buildup term is:

$$B = \frac{1}{T}\int_0^T (1 - e^{-(\lambda + q/V)t})dt, \tag{27}$$

where T is the average irradiation time = 8 h. The average value of the buildup term for an eight-hour run is 0.98, so buildup is neglected. (The effective half-life with the removal term is 0.2 h). The calculated equilibrium activity is 950 Bq/cm^3 = 0.026 μCi/cm^3. The total flow of air through the RSR is 45.9 cm^3/s × 3600 s/h × 2000 h/y + 440,000 cm^3 = 0.33 × 10^9 cm^3/y. This corresponds to 8.6 Ci/y. With a room flow rate of 1.12 × 10^6 cm^3/s × 3600 s/h x 2000 h/y = 8.1 × 10^{12} cm^3/y. This gives an average reactor room concentration of 1.1 × 10^{-6} μCi/cm^3. For release to the environment, 8760 hours can be used as the averaging time, and the undiluted (before dispersion) concentration will be 2.4 × 10^{-7} μCi/cm^3.

The second approach is by sample insertion alone (RSR sealed during operation)
Eliminating the q/V term in the aforementioned equation, the equilibrium concentration of ^{41}Ar in the RSR is 0.362 μCi/cm^3. Multiplying by the average buildup term of 0.69 and the volume of 440,000 cm^3 discharged, this gives a total activity release of 0.11 Ci/y. This gives an average reactor room concentration of 1.3 × 10^{-8} μCi cm^3. For release to the environment, 8760 hours can be used as the averaging time, and the undiluted (before dispersion) concentration will be 3.1 × 10^{-9} μCi cm^3.

Summary:
With the first approach the doses are as follows:

MEW	50.1 mrem/yr
MMP	2.8 mrem/yr
NPR	1.2 mrem/yr

With the second approach the doses are as follows:

MEW	0.2 mrem/yr
MMP	3.6E-2 mrem/yr
NPR	1.5E-2 mrem/yr

Table #1 displays the values for the various variables used within this appendix concerning offsite dose calculations. A summary of doses for ^{41}Ar releases is presented in Table #2. The value for the RSR determination is based on the second approach, as it is the most realistic. The table clearly shows no health hazard to workers or the public from normal operations of the reactor.

Table #1
Summary of values for various
Points of release

	Pool - water	RSR	PnT
U_{10}	10 m/s	10 m/s	10 m/s
r_{MMP}	102 m	102 m	112 m
r_{NPR}	158 m	158 m	127 m
Atmospheric Stability	F	F	F
σ_y^{MMP}	4 m	4 m	4.5 m
σ_y^{NPR}	6.5 m	6.5 m	5 m
σ_z^{MMP}	2.3 m	2.3 m	2.5 m
σ_z^{NPR}	3.3 m	3.3 m	2.8 m

Table #2
Summary of results
from ^{41}Ar (2000 hours)

	Ci/yr	Release Point (outside)	Wholebody DOSE mrem/yr MEW	Wholebody DOSE mrem/yr MMP	Wholebody DOSE mrem/yr NPR
^{41}Ar pool	$2.6x10^{-1}$	Ventfan	$3.9x10^{-1}$ [a]	$1.0x10^{-1}$ [b]	$4.4x10^{-2}$ [c]
^{41}Ar PT	$1.34x10^{-1}$	Hood exhaust	na	$4.3x10^{-2}$ [c]	$1.8x10^{-2}$ [c]
^{41}Ar RSR	$1.1x10^{-1}$	Ventfan	$1.8x10^{-1}$ [d]	$3.55x10^{-2}$ [c]	$1.5x10^{-2}$ [c]
Total ^{41}Ar	$5.1x10^{-1}$		$5.1x10^{-1}$	$1.49x10^{-1}$	$6.4x10^{-2}$

[a] Value determined in Appendix A, Section A.1.2

[b] Value determined at the end of Appendix A , Section A.2

[c] Value determined similarly as the calculation Appendix A , at the end of Section A.2 using the appropriate variables noted in Table #1

[d] Summary, Appendix A, at the end of Section A.4

Appendix A

References

A1 Dorsey, Noah E. Properties of Ordinary Water-Substance. Reinhold Publishing, New York; 1940.

A2 FSAR, Stationary neutron Radiography System, McClellan Air Force Base, p. A-9. Jan. 1992.

A3 Regulatory Guide 1.109, "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", USNRC, rev. 1, Oct. 1977.

A4 Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", USNRC, rev. 1, Nov. 1982.

A5 Henderson, W.J., and Tunnicliffe, P.R., "The production of N-16 and N-17 in the cooling water of the NRX Reactor", pp. 145-150. NSE 1958.

A6 "Hazard Report for Torrey Pines TRIGA Reactor", General Atomic GA-722. 1959.

A7 Mittll, R.L. and Theys, M. H., "N-16 Concentrations in EBWR", Nucleonics, p. 81. March 1961

APPENDIX B

Fission Product Release Calculations

B.1 Maximum Hypothetical Accident

B.1.1 Description

Guidelines for preparation of the SAR [B1] require analysis of a maximum hypothetical accident (MHA) involving failure of the cladding of one fuel element and the escape of radioactive noble gases and iodine. Section B.1.(1 – 4) of this appendix is based on similar calculations performed by Dr. Richard E. Faw for the Kansas State University TRIGA Mark II reactor [B2].

Source quantities of radioactive noble gases and iodine are computed and tabulated for a maximum hypothetical accident involving cladding failure of a single TRIGA fuel element and the escape of the radionuclides into the environment. This limiting case is based on 40 years of operation at an average power of 1.5 kW (established by previous operation) followed by 20 years continuous operation at 20 kW. As the fuel has been at the AJBRF since new, no residual sources from other irradiations are examined. Potential consequences of radiological releases are examined.

B.1.2 Problem Modeling and Assumptions:

Following are assumptions and approximations applied to calculations.

1. Calculations of radionuclide inventory in fuel are based on continuous operation prior to fuel failure for 40 years at the average thermal power experienced by the reactor during its first 40 years of operation, namely, 1.50 kW. This is followed by 20 years of operation at full licensed power of 20 kW.
2. Radionuclide inventory in one "worst-case" fuel element is based on 57 elements in the core, 36 grams of ^{235}U per element [B3, B4] and a value of 2.0 as a very conservative value of the ratio of the maximum power in the core to the average power.
3. The principal fission products arise only from ^{235}U. Since the fuel is enriched to 20% ^{235}U and has very low relative burnup in a thermal spectrum, fission products from the ingrowth and fission of ^{239}Pu and fast fission of ^{238}U occur in considerably less quantities.
4. The fraction of noble gases and iodine contained within the fuel that is actually released is 1.5×10^{-5}, a value measured at General Atomics [B5] and used in SARs for other non-pulsing TRIGA reactor facilities [B6].
5. No release of particulate (radionuclides other than noble gases and iodine) is considered as no credible release mechanism exists.

B.1.3 Radionuclide Inventory Buildup and Decay:

Consider a mass of ^{235}U yielding thermal power P (kW) due to thermal-neutron induced fission. The fission rate is related to the thermal power by the factor $k = 3.12 \times 10^{13}$ fissions per second per kW.[1] Consider also a fission product radionuclide, which is produced with yield Y, and which decays with rate constant λ. It is easily shown that the equilibrium activity A_∞ (Bq) of the fission product, which exists when the rate of creation by fission is equal to the rate of loss by

[1] Note that the product of k and yield Y may be stated as $3.12 \times 10^{13} \times Y$ Bq/kW or $843 \times Y$ Ci/kW.

decay, is given by $A_\infty = kPY$. Here it should be noted that the power must be small enough or the uranium mass large enough that the depletion of the ^{235}U is negligible.[2] Starting at time t = 0, the buildup of activity is given by

$$A(t) = A_\infty\left(1 - e^{-\lambda t}\right). \quad (1)$$

For times much greater than the half-life of the radionuclide, $A \approx A_\infty$, and for times much less than the half-life, $A(t) \approx A_\infty \times \lambda t$. If the fission process ceases at time t_1, the specific activity at later time t is given by

$$A(t) = A_\infty\left(1 - e^{-\lambda t_1}\right)e^{-\lambda(t-t_1)}. \quad (2)$$

Consider the fission product ^{131}I, which has a half-life of 8.04 days (λ = 0.00359 h^{-1}) and a chain (cumulative) fission product yield of 0.031. At a thermal power of 1 kW, the equilibrium activity is about A_∞ = 9.67 × 10^{11} Bq (26.1 Ci). After only eight hours of operation, though, the activity is only about 0.74 Ci. For equilibrium operation at 1.5 kW, distributed over 57 fuel elements, the average activity per element would be 26.1 × 1.5 ÷ 57 = 0.69 Ci per fuel element. The worst case element would contain twice this activity. With a release fraction of 1.5×10^{-5}, the activity available for release would be 0.69 × 2 × 1.5×10^{-5} = 1.04×10^{-5} Ci. For eight hours operation at 20 kW, the activity available for release would be (0.74 × 20 ÷ 57) × 2 × 1.5 × 10^{-5} = 7.79×10^{-6} Ci.

This type of calculation is performed by the ORIGEN code [B7] for hundreds of fission products and for arbitrary times and power levels of operation as well as arbitrary times of decay after conclusion of reactor operation. The code accounts for branched decay chains. It also may account for depletion of ^{235}U and ingrowth of ^{239}Pu, although those features were not invoked in the calculations reported here because of minimal depletion in TRIGA fuel elements.

B.1.4 ORIGEN 2.1 Calculations:

Data input files and abridged output files for the ORIGEN-2.1 calculations are included as Sub-Appendices A - C. Sub-Appendix A contains raw data for the buildup of long-lived radionuclides for the past and proposed operating history of the AJBRF. The problem was modeled as large amount of ^{235}U at a thermal power of 1.5 kW for 40 years, followed immediately by 20 years of operation at 20 kW. Tabulated results are Ci activities, by nuclide, immediately after reactor shutdown, and at 1, 2, 3, 7, and 14 days after shutdown. Data are provided only for those nuclides present at activities greater than 100 mCi in a single fuel element immediately after reactor shutdown.

Also presented in the appendices are tables of available activities for release from a single, worst-case fuel element. In Sub-Appendix B, raw data from Sub-Appendix A (Ci) are multiplied by the release fraction of 1.5 × 10^{-5}, divided by 57 to account for the power distributed over 57 fuel elements, and multiplied by 2 to account for the peak to average ratio for the power distribution in the core.

Reference Case Source Terms:

Data from Sub-Appendix B for the worst case TRIGA fuel element are compared and the greater values for any one isotope are selected as reference case source terms for the maximum hypothetical accident. Data are presented in Table I for halogens and noble gases.

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2 Negligible burnup is modeled in ORIGEN calculations by setting the fuel mass very large (1 tonne) and the thermal power very low (1 kW or less).

Derived quantities:

The raw data of Table I are activities potentially released from a single worst-case fuel element that has experienced a cladding failure. This activity may itself be compared to the annual limit of intake (ALI) to gauge the potential risk to an individual worker. By dividing the activity by the 7.075×10^8 cm^3 free volume of the reactor room, one obtains an air concentration (specific activity) that may be compared to the derived air concentration (DAC) for occupational exposure as given 10CFR20 or in EPA federal guidance [B8].

TABLE I

MHA, AVAILABLE ACTIVITY AFTER REACTOR SHUTDOWN

Reference case iodine and noble gas source terms for the maximum hypothetical accident at the AJBRF. Available activity (μCi) of iodine, krypton, and xenon radionuclides from a single worst-case fuel element as a function of time after reactor operation. Data are derived from ORIGEN 2.1 calculations [B7] as summarized in Sub-Appendix B. Only nuclides with half-lives in excess of 2 seconds are presented.

		Available activity (μCi) at time in days after reactor shutdown					
Element	Nuclide	0	1	2	3	7	14
I	129	0.3	0.3	0.3	0.3	0.3	0.3
I	131	249	231	214	197	140	77
I	132	374	311	251	203	87	19
I	133	597	276	124	56	2.3	0.0
I	135	558	45	3.6	0.3	0	0
Kr	83	0.2	0.2	0.2	0.2	0.2	0.2
Kr	83m	47	0.2	0	0	0	0
Kr	84	0.3	0.3	0.3	0.3	0.3	0.3
Kr	85m	112	2.8	0.1	0	0	0
Kr	85	18	18	18	18	18	18
Kr	86	0.7	0.7	0.7	0.7	0.7	0.7
Kr	87	228	0	0	0	0	0
Kr	88	321	0.9	0	0	0	0
Kr	89	411	0	0	0	0	0
Kr	90	406	0	0	0	0	0
Kr	91	308	0	0	0	0	0
Xe	131	1.5	1.5	1.5	1.5	1.5	1.5
Xe	131m	2.8	2.8	2.7	2.7	2.5	2.1
Xe	132	2.2	2.2	2.2	2.2	2.2	2.2
Xe	133m	17	16	13	10	3.1	0.3
Xe	133	598	576	529	475	288	115
Xe	134	4.1	4.1	4.1	4.1	4.1	4.1
Xe	134m	2.1	0	0	0	0	0
Xe	135m	100	7.2	0.6	0	0	0
Xe	135	580	212	44	7.8	0	0
Xe	137	530	0	0	0	0	0
Xe	138	554	0	0	0	0	0
Xe	139	455	0	0	0	0	0
Xe	140	321	0	0	0	0	0

B.1.5 Doses within the reactor room:

Air changeover in the facility takes approximately 0.17 hours. For purposes of dose to the MEW it will be assumed that personnel within the reactor room would be exposed to the initial effluent concentration for one hour. This is a conservative estimate considering that over 3 air changeovers occur during this time. Two separate calculations will be performed. The first calculation will be based on methodologies and values specified in 10 CFR 20. The second will be based on using a semihemispherical cloud with a radius such that the cloud is equivalent to the volume of the room.

The ALI is the activity that, if ingested or inhaled, would lead to either (a) the maximum permissible committed effective dose equivalent incurred annually in the workplace, nominally 5 rem, or (b) the maximum permissible dose to any one organ or tissue, nominally 50 rem. The DAC is the air concentration that, if breathed by reference man for one work year (2000 h), would result in the intake of the ALI.

The whole body dose rate due to the noble gases for the MEW can be estimated by assuming the room is a hemisphere with an equivalent volume and an individual is positioned at the center of the hemisphere. Using this assumption, the dose rate is determined by

$$D = BS_u \frac{\left(1 - e^{-\mu_s R_o}\right)}{2\mu_s g} \qquad (3)$$

where:

D= Dose in mrem/hr

B = Dose buildup factor

R_o = Radius of hemisphere = 696cm

S_u = Source strength (dis/s-cm^3)

μ_s = Linear absorption coefficient (1/cm)

g = dose conversion factor $\left(\frac{\gamma/s - cm^2}{mrem/hr}\right)$

It should be noted that values [B9] for g and μ_s can differ significantly for each isotope since they are energy dependent. For our calculations, it is assumed each disintegration event produced a gamma whose average energy [B10] is shown in Table II. Using equation 3, it has been determined that the whole body dose for an individual in the room with the initial room concentration for one hour is 6.23×10^{-2} mrem for noble gases. It should be reiterated that this value is conservatively large. The room's air concentration would decrease do to the multiple air changeovers (greater than 3) during this time.

Using Inhalation dose conversion factors taken from Federal Guidance Report No. 11 [B11] and the values specified in 10 CFR 20; the iodine dose was determined to be 23 mrem Committed Effective Dose Equivalent (CEDE) and a thyroid Committed Dose Equivalent (CDE) of 720 mR. These calculations are summarized in Table II and Table V. It may be noted that the total dose to the MEW is far below the annual dose limit as specified in 10CFR20.

TABLE II

Iodine and noble gas available activities immediately after reactor shutdown with reactor room concentrations with DACs for iodine, and MEW doses based on DAC for iodine and Eq. 3 for noble gases.

Element	Nuclide	Half-life	Available Activity (μCi)	Average gamma energy (MeV)	g γ/cm²-sec per rem/hr	μ_s (m^{-1})	Reactor room conc. (μCi/cm³)	DAC (μCi/cm³)	MEW (mrem)	Thyroid Dose[d] (mrem)
I	129	16 My	0.3				$5x10^{-10}$	4E-9	9.4 E-2[b]	3.1 E0
I	131	8.04 d	249				$4x10^{-7}$	2E-8	1.4 E1[b]	4.5 E2
I	132	2.30 h	374				$5x10^{-7}$	3E-6	2.3 E-1[b]	3.9 E0
I	133	20.8 h	597				$8x10^{-7}$	1E-7	7.0 E0[b]	2.2 E2
I	134	52.6 m	673				$1x10^{-6}$	2E-5	1.6 E-1[b]	1.3 E0
I	135	6.61 h	558				$8x10^{-7}$	7E-7	1.5 E0[b]	3.8 E1
Kr	83m	1.83 h	47	0.8E-3	3.4E5	480	$7x10^{-8}$		7.51E-7[c]	
Kr	85m	4.48 h	112	0.16	3.8E6	3.3E-3	$2x10^{-7}$		5.32E-4[c]	
Kr	85	10.7 y	18	0.4E-2	3.4E5	480	$3x10^{-8}$		2.85E-7[c]	
Kr	87	76.3 m	228	1.07	5.2E5	3.5E-3	$3x10^{-7}$		7.89E-3[c]	
Kr	88	2.84 h	321	2.05	3.1E5	3.0E-3	$5x10^{-7}$		1.86E-2[c]	
Xe	131m	12 d	2.8	0.82E-2	3.4E5	480	$4x10^{-9}$		4.43E-8[c]	
Xe	133m	2.19 d	17	0.37E-1	5.0E6	1.1E-2	$2x10^{-8}$		6.09E-5[c]	
Xe	133	5.25 d	598	0.29E-1	3.4E6	2.1E-2	$8x10^{-7}$		2.98E-3[c]	
Xe	135m	15.3 m	100	0.46	1.1E6	3.9E-3	$1x10^{-7}$		1.64E-3[c]	
Xe	135	9.09 h	580	0.25	2.4E6	3.6E-3	$8x10^{-7}$		4.35E-3[c]	
Xe	138	14.1 m	554	1.57	3.8E5	3.2E-3	$8x10^{-7}$		2.62E-2[c]	
Totals[a]										7.2E2

[a] Totals are in mR and based on one hour exposure to initial room concentration

[b] Values are as Committed Effective Dose Equivalent

[c] External dose

[d] Committed Dose Equivalent (inhalation)

B.1.6 Offsite Dose Calculations

The air from the reactor room is exhausted by a ventilation fan and two fume hoods. The exhaust fan typically releases 85% of the reactor room air at ground level while the fume hoods provide an elevated release of the remaining 15% on the 12th floor of the Medical Center. In order to consider the worst case of an element release it should be assumed that entire fission product cloud release is at ground level. Referring to SAR Appendix A, figure A-1, the distance to the MMP is from the vent fan to the sidewalk (102 meters) and the distance to the NPR is a commercial building further south (158 meters).

Again it is assumed that an individual is exposed to a cloud for one hour. Even in very stable atmospheric conditions this would be a conservative assumption. It is recognized that the doses calculated below are based on annual dose assessments; however, correcting these totals for a one hour exposure leads to the desired result.

For atmospheric stability conditions when the windspeed at the 10 meter level is less than 6 meters per second horizontal plume meander may be considered. χ/Q values may be determined from the set of equations specified in Regulatory Guide 1.145. For this particular MHA, the windspeed is presumed to be 2 m/s with a Pasquill stability class of F to generate a conservative χ/Q. With respect to the to our specific parameters, the atmospheric diffusion can be described by

$$\chi/Q = \frac{1}{U\pi\Sigma_y\sigma_z} \qquad (4)$$

where:

- χ = the short term average centerline value of the ground level concentration in Ci/m^3
- Q = the amount of material released Ci/s
- U = the average windspeed 2 m/s
- σ_z = is the vertical plume spread which is a function of the atmospheric stability and distance
- Σ_y = Is lateral plume spread with meander and building wake effects which function of atmospheric stability, windspeed and distance. For less than 800 meters distance $\Sigma_y = M\sigma_y = 4\sigma_y$ = where M is a correction factor based on stability class [B13].

For comparison, it may be noted that the correction factor M, for the Ar-41 release during normal operations (Apd. A, Eq. 18) is unity. However, the horizontal plume meander tends to dominate dispersion during light wind and stable or neutral conditions and building wake mixing becomes more effective in dispersing effluents than meander effects as the windspeed increases and the atmosphere becomes less stable. It may also be noted that vertical plume meander is shown to be virtually nonexistent during light wind, stable conditions.

Following the criteria specified in Regulatory Guides [B12] and [B13]; calculations were made to determine the potential dose to the public outside the facility. The total body dose for ground release of noble gas is given by

$$D_{\infty}^{T}(r,\theta) = S_F \sum_i x_i(r,\theta) DFB_i \qquad (5)$$

where:

DFB_i = the total body dose factor for a semi-infinite cloud of the radionuclide i which including the attenuation of 5 g/cm^2 of tissue in mrem/pCi-yr;

$D_{\infty}^{T}(r,\theta)$ = is the annual total body dose due to immersion in a semi-infinite cloud at the distance r in sector θ, in mrem/yr; and

$\chi_i(r,\theta)$ = the annual average ground-level concentration of radionuclide i at the distance r in sector θ, in pCi/m^3

$\chi_i(r,\theta)$ is determined by

$$\chi_i(r,\theta) = Q_i\,[\chi/Q]^D(r,\theta) \qquad (6)$$

$[\chi/Q]^D(r,\theta)$ is the gaseous dispersion factor (corrected for radioactive decay) in the sector at angle θ at the distance r from the release point, in s/m^3. Q_i is the value based on our one hour release model.

The annual organ dose from inhalation of radionuclides in air is given by

$$D_{ja}^{A}(r,\theta) = R_a \sum_i \chi_i(r,\theta) DFA_{ija} \qquad (7)$$

where:

$D_{ja}^{A}(r,\theta)$ = the annual dose to organ j of an individual in the age group a at location (r,θ) due to inhalation, in mR/yr;

DFA_{ija} = the inhalation dose factor for radionuclide I, organ j, and age group a, in mrem/pCi;

R_a = the annual air intake for individuals in the age group a, 8000 m^3/yr

all other factors are defined above. The dose specified for Eq. 7 can be used for radioiodines and other gases other than the Noble gases. In this case the organ with the highest dose will be the thyroid. Whole body doses are typically insignificant in this type of paradigm but nonetheless are provided. As a result, values for internal and external dose are not significantly different for a finite cloud approach. Postulated dose assessments are summarized in Tables (III-V).

Table III
Offsite Dose Calculation for MMP NPR from Noble Gas Plume

Element	Nuclide	Decay constant	MMP			NPR	
			$\chi(r,\theta)$ pCi/m^3	DFB	$D(r,\theta)$	$\chi(r,\theta)$ pCi/m^3	$D(r,\theta)$
Kr	83m	1.01E-04	5.62E+01	7.56E-08	4.25E-06	2.40E+01	1.82E-06
Kr	85m	4.38E-05	1.34E+02	1.17E-03	1.57E-01	5.75E+01	6.73E-02
Kr	85	2.04E-09	2.16E+01	1.61E-05	3.48E-04	9.27E+00	1.49E-04
Kr	87	1.48E-04	2.72E+02	5.92E-03	1.61E+00	1.16E+02	6.87E-01
Kr	88	6.88E-05	3.84E+02	1.47E-02	5.65E+00	1.65E+02	2.42E+00
Xe	131m	6.68E-07	3.36E+00	1.47E-02	4.94E-02	1.44E+00	2.12E-02
Xe	133m	3.49E-06	2.04E+01	2.51E-04	5.13E-03	8.76E+00	2.20E-03
Xe	133	1.51E-06	7.18E+02	2.94E-04	2.11E-01	3.08E+02	9.06E-02
Xe	135m	6.42E-04	1.16E+02	3.12E-03	3.63E-01	4.90E+01	1.53E-01
Xe	135	2.12E-05	6.96E+02	1.81E-03	1.26E+00	2.98E+02	5.40E-01
Xe	138	8.19E-04	6.38E+02	1.47E-02	9.38E+00	2.68E+02	3.93E+00
				TOTAL(mrem/y)	1.87E+01		7.91E+00
				TOTAL(mrem for one hour exposure)	**2.13E-03**		**9.03E-04**

Table IV
Offsite Dose Calculation for MMP NPR from Iodine Gas Plume

Element	Nuclide	Decay constant	MMP			NPR	
			DFA	$D(r,\theta)$	$\chi(r,\theta)$ pCi/m^3	$D(r,\theta)$	$\chi(r,\theta)$ pCi/m^3
I	129	1.38E-15	2.56E-06	7.38E-03	3.60E-01	3.17E-03	1.55E-01
I	131	9.90E-07	2.56E-06	6.13E+00	2.99E+02	2.63E+00	1.28E+02
I	132	8.37E-05	1.45E-07	5.19E-01	4.47E+02	2.22E-01	1.91E+02
I	133	9.48E-06	5.65E-07	3.24E+00	7.17E+02	1.39E+00	3.07E+02
I	134	2.14E-04	7.69E-08	4.92E-01	8.00E+02	2.10E-01	3.41E+02
I	135	1.92E-04	3.21E-07	1.70E+00	6.64E+02	7.27E-01	2.83E+02
				TOTAL(mrem/y) =	1.21E+01		5.18E+00
				WHOLE BODY TOTAL (mrem for one hour exposure) =	**1.38E-03**		**5.91E-04**
I	131	9.90E-07	1.49E-03	3.57E+03		1.53E+03	
I	132	8.37E-05	1.43E-05	5.12E+01		2.19E+01	
I	133	9.48E-06	2.69E-04	1.54E+03		6.61E+02	
I	134	2.14E-04	3.73E-06	2.39E+01		1.02E+01	
I	135	1.92E-04	5.60E-05	2.97E+02		1.27E+02	
				TOTAL(mrem/y)	5.48E+03		2.35E+03
				THYROID TOTAL (mrem for one hour exposure)	**6.26E-01**		**2.68E-01**

Table V
Dose Summary Maximum Hypothetical Accident

	WHOLE BODY (noble gases) mrem	**CDE[a] THYROID mrem**	**CEDE[b] mrem**
MEW	6.23E-02	7.20E2	2.3E1
	WHOLE BODY mrem	**THYROID mrem**	
MMP	3.51E-03	6.26E-01	
NPR	1.49E-03	2.68E-01	

[a] Committed Dose Equivalent

[b] Committed Effective Dose Equivalent

B.2 Fission Product release in water

It has been demonstrated that the available radionuclide inventory in TRIGA fuel is1.5×10^{-5} of the total activity contained in the fuel. Total activities were calculated using both historical and worst case data in Section B.1, with the aforementioned release fraction then applied to obtain projected releases. Three species were considered to be of primary importance: krypton, xenon, and iodine. As noble gases, both krypton and xenon are assumed to be instantly and uniformly distributed in the containment atmosphere. Of the iodine released from the fuel, it is assumed that all is absorbed in the water. The krypton and xenon gases are assumed to have a 98% and 95% release fraction respectively, the balance being held in solution in the primary coolant. The doses from such an event are as follows:

Table VI
Dose Summary Fuel Rupture in Water

	WHOLE BODY mrem
MEW	6.00E-02
MMP	2.05E-03
NPR	8.69E-04

Conclusions:

Fission product inventories in TRIGA fuel elements were calculated with the ORIGEN code, using very conservative approximations. Then, potential radionuclide releases from worst-case fuel elements were computed, again using very conservative approximations. Even if it were assumed that releases took place immediately after 20 years of continuous full power operations. Based on reactor operations history of 1.5 kW average power, this estimate is at least an order of magnitude high. Additionally, reactivity constraints would limit the prolonged operation.

With this conservatism in mind, it is extremely unlikely that 10CFR20 occupational dose limits would be exceeded from a fuel failure. Similarly, radionuclides immediately released from a damaged fuel element to the outside atmosphere are unlikely to produce doses in excess of 10CFR20 public dose limits when atmospheric dispersion is taken into consideration.

SUMMARY OF CONTENTS OF SUB-APPENDICES

A. ORIGEN 2.1 output file, listing activity (Ci) as a function of time after shutdown for a ^{235}U-fueled thermal reactor operating at 1.5 kW thermal power for 40 years, followed by 20 years at 20 kW.

B. Maximum activity (μCi) available for release from a single TRIGA fuel element as a function of time after shutdown for a ^{235}U-fueled thermal reactor operating at 1.5 kW thermal power for 40 years, followed by 20 years at 20 kW.

C. ORIGEN 2.1 input file.

Sub-Appendix A

One Tonne ^{235}U at 1.5 kw for 40 years and 20 kW for 20 years

Origen 2.1, thermal neutron cross sections

Activity (Ci) as function of time after shutdown

		Time after reactor shutdown (d)					
Z	A	0	1	2	3	7	14
I	129	6.35E-01	6.35E-01	6.35E-01	6.36E-01	6.36E-01	6.36E-01
I	131	4.74E+02	4.40E+02	4.06E+02	3.74E+02	2.66E+02	1.46E+02
I	132	7.11E+02	5.91E+02	4.78E+02	3.86E+02	1.65E+02	3.71E+01
I	133	1.14E+03	5.24E+02	2.36E+02	1.06E+02	4.32E+00	1.60E-02
I	133M	2.16E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	134	1.28E+03	3.24E-05	1.87E-13	1.08E-21	0.00E+00	0.00E+00
I	134M	7.13E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	135	1.06E+03	8.57E+01	6.92E+00	5.59E-01	2.38E-05	5.33E-13
I	136	5.06E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	136M	3.33E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	137	5.45E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	138	2.75E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	139	1.26E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	140	3.79E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	141	5.53E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I	142	3.70E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	83	3.30E-01	3.30E-01	3.30E-01	3.30E-01	3.30E-01	3.30E-01
KR	84	6.44E-01	6.44E-01	6.44E-01	6.44E-01	6.44E-01	6.44E-01
KR	86	1.30E+00	1.30E+00	1.30E+00	1.30E+00	1.30E+00	1.30E+00
KR	83M	8.91E+01	3.52E-01	3.64E-04	3.49E-07	2.83E-19	0.00E+00
KR	85	3.38E+01	3.38E+01	3.38E+01	3.38E+01	3.38E+01	3.37E+01
KR	85M	2.13E+02	5.27E+00	1.29E-01	3.14E-03	1.12E-09	5.75E-21
KR	87	4.34E+02	9.13E-04	1.90E-09	3.96E-15	0.00E+00	0.00E+00
KR	88	6.10E+02	1.74E+00	4.96E-03	1.41E-05	9.34E-16	1.43E-33
KR	89	7.81E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	90	7.72E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	91	5.86E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	92	2.58E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	93	8.97E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	94	3.95E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	95	1.35E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
KR	96	2.61E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	131	2.76E+00	2.76E+00	2.76E+00	2.76E+00	2.76E+00	2.76E+00
XE	132	4.18E+00	4.18E+00	4.18E+00	4.18E+00	4.19E+00	4.19E+00
XE	134	7.70E+00	7.70E+00	7.70E+00	7.70E+00	7.70E+00	7.70E+00
XE	136	6.54E+00	6.54E+00	6.54E+00	6.54E+00	6.54E+00	6.54E+00
XE	131M	5.26E+00	5.25E+00	5.22E+00	5.17E+00	4.82E+00	3.93E+00
XE	133	1.14E+03	1.10E+03	1.01E+03	9.03E+02	5.48E+02	2.19E+02
XE	133M	3.30E+01	3.02E+01	2.48E+01	1.93E+01	5.92E+00	6.55E-01
XE	134M	3.97E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	135	1.10E+03	4.03E+02	8.29E+01	1.48E+01	1.07E-02	2.94E-08
XE	135M	1.90E+02	1.37E+01	1.11E+00	8.96E-02	3.81E-06	8.53E-14
XE	137	1.01E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	138	1.05E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	139	8.64E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	140	6.10E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	141	2.04E+02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	142	6.25E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	143	8.55E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
XE	144	1.01E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL		**1.68E+04**	**3.25E+03**	**2.30E+03**	**1.87E+03**	**1.05E+03**	**4.64E+02**

One TRIGA element at 1.5 kW for 40 years and 20 kW for 20 years
(Power divided by 57 elements x Peak Factor of 2)
Origen 2.1, thermal neutron cross sections
Release Activity (μCi) as a function of time after shutdown

		Time after reactor shutdown (d)						Inhalation						Fractional Release		
								ALI	DAC	EC	Initial Room Concentration	Initial Conditions				
Z	A	0	1	2	3	7	14	(μCi)	(μCi/cm³)	(μCi/cm³)	(μCi/cm³)	C>DAC	A>ALI	C/DAC	O/EC	A/ALI
I	129	0.3	0.3	0.3	0.3	0.3	0.3	9.E+00	4.E-09	4.E-11	5.E-10			1.E-01	1.E+01	4.E-02
I	131	249.3	231.5	213.6	196.6	139.9	76.6	5.E+01	2.E-08	2.E-10	4.E-07	x	x	2.E+01	2.E+03	5.E+00
I	132	374.3	310.9	251.3	203.2	86.7	19.5	8.E+03	3.E-06	2.E-08	5.E-07			2.E-01	3.E+01	5.E-02
I	133	597.4	275.8	123.9	55.7	2.3	0.0	3.E+02	1.E-07	1.E-09	8.E-07	x	x	8.E+00	8.E+02	2.E+00
I	133M	11.4	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	2.E-08					
I	134	672.6	0.0	0.0	0.0	0.0	0.0	5.E+04	2.E-05	6.E-08	1.E-06			5.E-02	2.E+01	1.E-02
I	134M	37.5	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	5.E-08					
I	135	558.4	45.1	3.6	0.3	0.0	0.0	2.E+03	7.E-07	6.E-09	8.E-07	x		1.E+00	1.E+02	3.E-01
I	136	266.2	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	4.E-07					
I	136M	175.3	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	2.E-07					
I	137	287.0	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	4.E-07					
I	138	144.9	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	2.E-07					
I	139	66.5	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	9.E-08					
I	140	19.9	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	3.E-08					
I	141	2.9	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	4.E-09					
I	142	0.2	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	3.E-10					
KR	83	0.2	0.2	0.2	0.2	0.2	0.2	n.a.	n.a.	n.a.	2.E-10					
KR	84	0.3	0.3	0.3	0.3	0.3	0.3	n.a.	n.a.	n.a.	5.E-10					
KR	86	0.7	0.7	0.7	0.7	0.7	0.7	n.a.	n.a.	n.a.	1.E-09					
KR	83M	46.9	0.2	0.0	0.0	0.0	0.0	n.a.	1.E-02	5.E-05	7.E-08			7.E-06	1.E-03	
KR	85	17.8	17.8	17.8	17.8	17.8	17.8	n.a.	1.E-04	7.E-07	3.E-08			3.E-04	4.E-02	
KR	85M	112.3	2.8	0.1	0.0	0.0	0.0	n.a.	2.E-05	1.E-07	2.E-07			8.E-03	2.E+00	
KR	87	228.3	0.0	0.0	0.0	0.0	0.0	n.a.	5.E-06	2.E-08	3.E-07			6.E-02	2.E+01	
KR	88	320.9	0.9	0.0	0.0	0.0	0.0	n.a.	2.E-06	9.E-09	5.E-07			2.E-01	5.E+01	
KR	89	411.1	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	6.E-07					
KR	90	406.2	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	6.E-07					
KR	91	308.3	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	4.E-07					
KR	92	135.9	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	2.E-07					
KR	93	47.2	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	7.E-08					
KR	94	20.8	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	3.E-08					
KR	95	0.7	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	1.E-09					
KR	96	0.1	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	2.E-10					
XE	131	1.5	1.5	1.5	1.5	1.5	1.5	n.a.	n.a.	n.a.	2.E-09					
XE	132	2.2	2.2	2.2	2.2	2.2	2.2	n.a.	n.a.	n.a.	3.E-09					
XE	134	4.1	4.1	4.1	4.1	4.1	4.1	n.a.	n.a.	n.a.	6.E-09					
XE	136	3.4	3.4	3.4	3.4	3.4	3.4	n.a.	n.a.	n.a.	5.E-09					
XE	131M	2.8	2.8	2.7	2.7	2.5	2.1	n.a.	4.E-04	2.E-06	4.E-09			1.E-05	2.E-03	
XE	133	597.9	576.3	529.5	475.3	288.4	115.2	n.a.	1.E-04	5.E-07	8.E-07			8.E-03	2.E+00	
XE	133M	17.4	15.9	13.0	10.2	3.1	0.3	n.a.	1.E-04	6.E-07	2.E-08			2.E-04	4.E-02	
XE	134M	2.1	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	3.E-09					
XE	135	580.5	212.2	43.6	7.8	0.0	0.0	n.a.	1.E-05	7.E-08	8.E-07			8.E-02	1.E+01	
XE	135M	100.2	7.2	0.6	0.0	0.0	0.0	n.a.	9.E-06	4.E-08	1.E-07			2.E-02	4.E+00	
XE	137	530.5	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	7.E-07					
XE	138	554.2	0.0	0.0	0.0	0.0	0.0	n.a.	4.E-06	2.E-08	8.E-07			2.E-01	4.E+01	
XE	139	454.6	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	6.E-07					
XE	140	321.3	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	5.E-07					
XE	141	107.4	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	2.E-07					
XE	142	32.9	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	5.E-08					
XE	143	4.5	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	6.E-09					
XE	144	0.5	0.0	0.0	0.0	0.0	0.0	n.a.	n.a.	n.a.	8.E-10					
		8839.9	1712.1	1212.6	982.3	553.4	244.1				1.2E-05			28.1	2915.9	7.4

Sub appendix C

ORIGEN 2.1 input file.

```
Content-Type: application/octet-stream; name=ova2.inp
Content-Transfer-Encoding: quoted-printable
Content-MD5: mh7dMG1PUA8DCi76uM4IBw==
Content-Description: ova2.inp
Content-Disposition: attachment; filename=ova2.inp

-1=0D
-1=0D
-1=0D
RDA     ORIGEN2, Version 2.1 OVA Reactor Reference=0D
RDA     Case run by Brendan Ryan, Kansas State University=0D
RDA     Updated By: Richard Faw, Kansas State University=0D
BAS     1 tonne of U-235=0D
RDA     -1 =3D 8 hr per day for 5 days=0D
RDA     Warning: Vectors are often changed with respect to their=0D
RDA              content and will be noted on RDA cards.=0D
CUT     -1=0D
RDA     Library Print (1 print, 0 no print)=0D
LIP     0 0 0=0D
RDA     Decay Library Choices (0 print, 1 2 3 Decay Libraries; 601=0D
RDA     Cross Sections; etc.  see page 47)=0D
LIB     0  1 2 3  201  202  203   9   3   0  1   38=0D
PHO     101 102 103 10   <<< Photon Libraries, pg. 47=0D
TIT     Omaha VA Total Activity after five 8 h days=0D
TIT     Based on 1 tonne U-235=0D
RDA     Read fuel compositions including impurities=0D
INP     -1  1 -1 -1  1  1=0D
MOV     -1  1  0  1.0=0D
HED     1                                          Charge=0D
BUP=0D
IRP       40.0   0.0015   1    2    5 2  40 years at 1.5kW=0D
IRP       60.0   0.020    2    3    5 0  20 years at 20kW=0D
BUP=0D
OPTL     4*8    7 8 3   17*8    Activation Product Output p. 56=0D
OPTA     4*8    7 8 3   17*8    Actinide Output p. 59 =0D
OPTF     4*8    7 8 3   17*8    Fission Product Output p. 59=0D
RDA     Move composition vector from 3 to 1=0D
MOV     3     1   0  1.0=0D
RDA     Decay For Various Intervals=0D
DEC     1      1   2   4   2=0D
DEC     2      2   3   4   0=0D
DEC     3      3   4   4   0=0D
DEC     7      4   5   4   0=0D
DEC    14      5   6   4   0=0D
DEC    28      6   7   4   0=0D
OUT     -6     1     -1    0=0D
OUT      6     1     -1    0=0D
END=0D
2   922340    0.0   922350  1.E6 922380   0.     0     0.0    PURE U-235=0D=

0 =0D
--41f9_4250-7e60_58fd-5b2f_5de9--
```

Appendix B

References

B1 NUREG-1537, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content,* Report NUREG-1537 Part 1, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1996.

B2 Faw, R.E., and J.K. Shultis, *Radiological Assessment*, Prentice Hall, Englewood Cliffs, N.J., 1993.

B3 Hawley, S.C., and R.L. Kathren, *Credible Accident Analyses for TRIGA and TRIGA-Fueled Reactors*, Report NUREG/CR-2387 (PNL-4028), Pacific Northwest Laboratory, Richland Washington, 1982.

B4 NUREG-1282, *Safety Evaluation Report on High-Uranium Content, Low-Enriched Uranium-Zirconium Hydride Fuels for TRIGA Reactors*, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1987.

B5 NUREG-2382

B6 Simnad, M.T., F.C. Faushee, and G.B. West, "*Fuel Elements for Pulsed TRIGA Research Reactors*," Nuclear Technology **28**, 31-56 (1976).

B7 NUREG-1390, *Safety Evaluation Report Relating to the Renewal of the Operating License for the TRIGA Training and Research Reactor at the University of Arizona*, Report NUREG-1390, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, 1990.

B8 CCC-371, *ORIGEN 2.1 Isotope Generation and Depletion Code: Matrix Exponential Method*, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1991.

B9 Radiological Health Handbook, US Department of Health, Education, and Welfare (1970).

B10 Safety Analysis Report, University of Texas at Austin, (1991).

B11 Eckerman, K.F., A.B. Wolbarst, and A.C.B. Richardson, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report No. 11, Report EPA-5201/1-88-020, U.S. Environmental Protection Agency, Washington, DC, 1988.

B12 Regulatory Guide 1.109, Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents For The Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I, U.S. Nuclear Regulatory Commission, 1977.

B13 Regulatory Guide 1.145, Atmospheric Dispersion Models For Potential Accident Consequence Assessments at Nuclear Power Plants, U.S. Nuclear Regulatory Commission, 1982.

APPENDIX C

LOSS OF COOLANT WITHOUT FUEL ELEMENT CLADDING FAILURE

Even though the possibility of the loss of shielding water is believed to be exceedingly remote, a calculation has been performed to evaluate the radiological hazard associated with a LOCA scenario.

C.1 PROBLEM DESCRIPTION:

Potential radiation dose rates from an exposed reactor core during a LOCA scenario are calculated for the AJBRF Safety Analysis Report. Doses for both the reactor room and the room above are to be determined for the limiting conditions of 10 h and 1000 h of reactor operation prior to loss of coolant. It should be noted that since the reactor is located below ground in clay-type soil, the possibility of rapidly losing shielding water is extremely improbable.

C.2 PROBLEM MODELING:

The photon source strength was modeled as an energy-dependent source using values obtained from ORIGEN data, using the method developed by Faw [C1], described previously. Photon transport calculations were performed using MCNP [C2]. As shown in the attached figures, a detailed geometry was developed based on dimensions provided in the SAR. A sample-input file is also provided. The reactor core was modeled as three layers surrounded by a reflector region. Material compositions were homogenized in each region. This homogenization will neglect the line-of-sight contribution to dose from the space between the fuel elements. However in a tight-packed vertical array, this contribution is minimal. In the core regions, the effect of air void was accounted for by a reduction in density of the other materials, as the number of atoms in the air was trivial compared to that of fuel and structural materials. The top core region consists of the aluminum in the top plate, cladding, and graphite in the upper fuel elements. The center section consists of cladding and the UZr-H fuel. The lower region is similar to that of the upper region. The reflector consists of graphite and aluminum.

The reactor tank and rooms are filled with air. The tank is lined with epoxy, which was approximated by polyethylene at twice its normal density. The epoxy was followed by the steel liner and then by concrete. Because several mean free paths of concrete surround the tank, the soil was not modeled. To compensate, the concrete was extended out to the problem boundaries to simulate the presence of soil. The rooms were modeled by a concrete room of rectangular shape. The lower room has the reactor in one corner. A concrete ceiling separates the rooms. Room walls were taken to be very thick concrete, continuing out to the problem boundary. The horizontal boundary was taken as an optically thick cylinder around these structures. The lower boundary

was placed well below the reactor tank. The other boundary was placed at the ceiling of the upper room.

The photon source was evenly distributed within the fuelled region of the reactor core. The source position distribution was visually verified with VISED, an MCNP editor that allows the origin of multiple source particles to be viewed. The energy distribution was modeled as a histogram from the ORIGEN data. The probability of photon emission was normalized to unity, with the final result multiplied by the average number of photons released per fission. The final result was then multiplied by the number of fissions per second at 20kW, 6.24×10^{14} s^{-1}. This was then multiplied by 3600 s/h to give results in dose per hour.

The dose was calculated by flux tallies on four surfaces: reactor room floor, reactor room waist level, upper room floor, and upper room waist level (85 cm above the floor). Each surface was divided into several radial regions: < 10 cm, 10-100 cm, 100-125 cm, 125-150 cm, 150-175 cm, and > 175 cm from the center axis of the tank. These distances were chosen to provide a peak value at the center, an average value over the tank, a value at the tank edge, and several values away from the tank, respectively. The photon energy spectrum recorded by the tallies was modified by a response function for exposure in air, to give values in Roentgen.

Because of the extensive geometry, particle importance splitting was used to bias the result in the forward direction; however, this did not effect the overall result. Fifty million source particles were started for each run to yield acceptable statistics. With the exception of the 10-cm tally, all tallies passed their respective statistical checks. The 10-cm tally had larger errors in several cases due to its small area.

Results:

Two detailed tables are provided (Sub-Appendix C-2) with all of the locations listed. As a summary, the following table presents waist-level values for primary areas of interest.

Table I

Waist-Level Exposure in Air

Decay Time	Waist-Level Exposure in Air (R/h)					
	Directly Above Tank (10 - 100cm)		Reactor Room (150 - 175 cm)		Upstairs Room (10 – 100 cm)	
	10 h	1000 h	10 h	1000 h	10 h	1000h
1 minute	97.9	113	3.39	4.05	12.5	14.3
1 hour	25.3	39.6	0.936	1.53	3.12	4.63
1 day	0.141	10.9	0.006	0.466	0.015	1.16
1 week	0.169	5.32	0.007	0.209	0.018	0.589
1 month	0.034	1.69	0.001	0.066	0.004	0.183

Sub Appendix C-1

Sample Input File:

```
Loss of Coolant Gamma Dose Calculation
c Omaha VA Hospital
c Brendan Ryan
c September 1998
c **************
c CELL CARDS
c **************
1   3 -7.92   1 -2 5 -25          imp:p=2  $Steel side of tank
2   3 -7.92   -1 -4 5            imp:p=2  $Steel bottom of tank
3   2 -2.3    2 -3 6 -25         imp:p=1  $Concrete around side
4   2 -2.3    -2 -5 6            imp:p=1  $Concrete base
5   4 -2.0  7 -1 4 -25           imp:p=2  $Epoxy Coating on side of tank
6   4 -2.0  -7 -8 4              imp:p=2  $Epoxy on Bottom of tank
7   1 -0.0012 8 -25 -7 #11 #12 #13 #14 imp:p=3 $Air in tank
8   1 -0.0012 45 -26 -27 28 29 -30   imp:p=4  $Air in room
9   2 -2.3  -26 -27 28 29 30 -31     imp:p=3  $Concrete Ceiling
10  1  -0.0012 -26 -27 28 29 46 -32   imp:p=5  $Air in room above
11  5  -1.90 -15 -20 19              imp:p=3  $Homogenized Upper Core Region
12  5  -1.90 -15 -18 17              imp:p=3  $Homogenized Lower Core Region
13  6  -2.62 15 -16 -20 17           imp:p=3  $Homogenized Reflector Region
14  7  -3.22 -19 18 -15              imp:p=3  $Fuel Region
15  0 32:3:-6                       imp:p=0 $Boundary
16  2  -2.3 -32 25 -3 #8 #9 #10 #17 #18  imp:p=1 $Room Wall
17  1  -0.0012 25 -26 -27 28 29 -45   imp:p=4 $Waist Level Plane
18  1  -0.0012 -26 -27 28 29 31 -46   imp:p=5 $Waist Level Plane

c **************
c SURFACE CARDS
c **************
c Reactor Tank
1   cz 104.00  $ Steel Tank Inner radius
2   cz 104.64  $ Steel Tank Outer radius 2
3   cz 750.00  $ Concrete Outer radius - Boundary
4   pz -610.00 $ Bottom of steel tank
5   pz -610.64 $ Bottom of steel liner
6   pz -700.00 $ Bottom of concrete - Boundary
7   cz 99.00   $ Inside of epoxy coating
8   pz -600.00 $ Bottom epoxy coating
c Reactor Core
15  cz 21.50   $ Core outside radius
16  cz 53.50   $ Reflector outside diameter
17  pz -560.00 $ Bottom of core radius
18  pz -548.5  $ Bottom of fuel region
19  pz -513.00 $ Top of fuel region
20  pz -501.50 $ Top of core
c Reactor Room
25  pz 0.00    $ Floor
26  py 190     $ North Wall
27  px 190     $ East Wall
28  py -570    $ South Wall
29  px -240    $ West Wall
30  pz 460     $ Ceiling
31  pz 470     $ Room above
32  pz 900     $ Ceiling in room above - boundary
c Tally Surfaces
40  cz 10    $Central Dose Region
41  cz 100   $Remainder of Center Region
42  cz 125   $Includes some direct near lower level
43  cz 150   $No Direct on lower level
44  cz 175   $Scattered Only
45  pz 85    $Waist-level Reactor Room
46  pz 555   $Waist-level Upper room
```

```
c **************
c DATA CARDS
c **************
c Source is from ORIGEN Data  1000 h operation 1 minute cool
SDEF  pos=0 0 -530.75  axs=0 0 1  rad=d1  ext=d2  erg=d3  cel=14
SI1  0  21.50        $ range of radius sampling: 0 to Rmax
SP1  -21 1           $ radial distriubtion: here r^1
SI2 -17.75 17.75       $ range of axial sampling
SP2  -21  0          $ axial distribution: here z^0
SI3  H 0.01 1 2 3 4 5 7.5 $ORIGEN Group Average Energy
SP3 0.0 0.7331 0.2011 0.0498 0.0113 0.0044 0.0004 $Number of photons per fission
c ***************
c RUN CARDS
c ***************
MODE P
NPS 50000000
CTME 600
c ***************
c TALLY CARDS
c ***************
F2:P 25  $Floor Dose in Reactor Room
FS2 -40 -41 -42 -43 -44
SD2 (314.16 31101.77 17671.5 21598.5 25525.4  230588.7)
c ------------------------------------------------------------
c  Air-exposure  X=1.835E-08 E (mu_en/rho)  R-cm^2  (data from T-C7 S&F)
c ------------------------------------------------------------
de2  0.010 0.015 0.02 0.03 0.04 0.05 0.06 0.08 0.10 0.15 0.2 0.3 0.4
     0.5 0.6 0.8 1.0 1.25 1.5 2. 3. 4. 5. 6. 8. 10.
df2  0.8702E-09 0.3672E-09 0.1978E-09 0.8461E-10 0.5015E-10 0.3760E-10
     0.3348E-10 0.3533E-10 0.4266E-10 0.6870E-10 0.9806E-10 0.1581E-09
     0.2165E-09 0.2721E-09 0.3251E-09 0.4231E-09 0.5118E-09 0.6115E-09
     0.7011E-09 0.8606E-09 0.1132E-08 0.1373E-08 0.1596E-08 0.1813E-08
     0.2239E-08 0.2661E-08
FM2 9.145E18   $# fiss per s and # of gammas times 3600 s
F12:P 31 $Floor Dose in Room Above
FS12 -40 -41 -42 -43 -44
SD12 (314.16 31101.77 17671.5 21598.5 25525.4  230588.7)
c ------------------------------------------------------------
c  Air-exposure  X=1.835E-08 E (mu_en/rho)  R-cm^2  (data from T-C7 S&F)
c ------------------------------------------------------------
de12  0.010 0.015 0.02 0.03 0.04 0.05 0.06 0.08 0.10 0.15 0.2 0.3 0.4
     0.5 0.6 0.8 1.0 1.25 1.5 2. 3. 4. 5. 6. 8. 10.
df12  0.8702E-09 0.3672E-09 0.1978E-09 0.8461E-10 0.5015E-10 0.3760E-10
     0.3348E-10 0.3533E-10 0.4266E-10 0.6870E-10 0.9806E-10 0.1581E-09
     0.2165E-09 0.2721E-09 0.3251E-09 0.4231E-09 0.5118E-09 0.6115E-09
     0.7011E-09 0.8606E-09 0.1132E-08 0.1373E-08 0.1596E-08 0.1813E-08
     0.2239E-08 0.2661E-08
FM12 9.145E18   $# fiss per s and # of gammas times 3600 s
F32:P 45  $Waist-Level Dose in Reactor Room
FS32 -40 -41 -42 -43 -44
SD32 (314.16 31101.77 17671.5 21598.5 25525.4  230588.7)
c ------------------------------------------------------------
c  Air-exposure  X=1.835E-08 E (mu_en/rho)  R-cm^2  (data from T-C7 S&F)
c ------------------------------------------------------------
de32  0.010 0.015 0.02 0.03 0.04 0.05 0.06 0.08 0.10 0.15 0.2 0.3 0.4
     0.5 0.6 0.8 1.0 1.25 1.5 2. 3. 4. 5. 6. 8. 10.
df32  0.8702E-09 0.3672E-09 0.1978E-09 0.8461E-10 0.5015E-10 0.3760E-10
     0.3348E-10 0.3533E-10 0.4266E-10 0.6870E-10 0.9806E-10 0.1581E-09
     0.2165E-09 0.2721E-09 0.3251E-09 0.4231E-09 0.5118E-09 0.6115E-09
     0.7011E-09 0.8606E-09 0.1132E-08 0.1373E-08 0.1596E-08 0.1813E-08
     0.2239E-08 0.2661E-08
FM32 9.145E18   $# fiss per s and # of gammas times 3600 s
F42:P 46 $Waist-Level Dose in Room Above
FS42 -40 -41 -42 -43 -44
SD42 (314.16 31101.77 17671.5 21598.5 25525.4  230588.7)
c ------------------------------------------------------------
c  Air-exposure  X=1.835E-08 E (mu_en/rho)  R-cm^2  (data from T-C7 S&F)
c ------------------------------------------------------------
```

```
de42  0.010 0.015 0.02 0.03 0.04 0.05 0.06 0.08 0.10 0.15 0.2 0.3 0.4
     0.5 0.6 0.8 1.0 1.25 1.5 2. 3. 4. 5. 6. 8. 10.
df42  0.8702E-09 0.3672E-09 0.1978E-09 0.8461E-10 0.5015E-10 0.3760E-10
     0.3348E-10 0.3533E-10 0.4266E-10 0.6870E-10 0.9806E-10 0.1581E-09
     0.2165E-09 0.2721E-09 0.3251E-09 0.4231E-09 0.5118E-09 0.6115E-09
     0.7011E-09 0.8606E-09 0.1132E-08 0.1373E-08 0.1596E-08 0.1813E-08
     0.2239E-08 0.2661E-08
FM42 9.145E18   $# fiss per s and # of gammas times 3600 s
c ------------------------------ c
c Air rho=0.0012 g/cm^3
c ANSI/ANS 6.4.3 composition          c
c ENDF/B-V continuous data @300K      c
c ------------------------------ c
m1   7014  -0.75519  8016 -0.23179 &
     6012  -.00014   18000  -0.01288
c
c ------------------------------ c
c Standard concrete rho=2.3 g/cm^3    c
c ANSI/ANS-6.4.3                      c
c Mass fractions taken from "Radiation  c
c Shielding" J.K.S & R.E.F            c
c ENDF/B-V continuous data (300 K)    c
c ------------------------------ c
m2    1001  -0.005599   13027  -0.045595   8016  -0.498250 &
     14000  -0.315768   11023  -0.017098   16032  -0.001200 &
     12000  -0.002400   19000  -0.019198   20000  -0.082592 &
     26000  -0.012299
c ------------------------------c
c ------------------------------ c
c Stainless Steel rho=7.92 g/cm^3     c
c ENDF/B-V continuous data (300 K)    c
c No thermal treatment                c
c Composition taken from MCNP manual    c
c Apx. C-10                           c
c ------------------------------ c
m3   26000  -0.695   24000  -0.190 &
     28000  -0.095   25055  -0.020
c ------------------------------ c
c Epoxy - simulate as polyethelene at   c
c twice normal density = 2 g/cm^3     c
c assume CH2 composition              c
c ------------------------------c
m4   6000 0.3333 1001 0.6667
c ------------------------------c
c Top and bottom core regions         c
c 4.91 kg Al, 26.8 kg C, 0.0005 kg air  c
c The air will be neglected           c
c Average Density = 1.90 g/cm^3       c
c ------------------------------c
m5   6000 -0.845 13027 -0.155
c ------------------------------c
c Homogenized reflector region        c
c 17.3 kg Al, 1138 kg C               c
c Average Density 2.62 g/cm^3         c
c ------------------------------c
m6   6000 -0.985  13027 -0.015
c ------------------------------c
c Homogenized fuel region             c
c Neglect air, H in fuel, control rods,  c
c and water in central thimble        c
c Average Density = 3.22 g/cm^3       c
c ------------------------------ c
m7   6000 -0.1677 13027 -0.0475 &
     92235 -0.0133 92238 -0.0534 &
     40000 -0.7181
```

Sub Appendix C-2

MCNP Calculated Data for Omaha VA Hypothetical Loss of Coolant Accident
10h run time data — Values in R/h — Error is MCNP-reported tally error

Run Time	10 hour		Cool		1 minute							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	151.462	4.2%	142.385	0.5%	7.226	3.6%	0.534	6.6%	0.523	6.5%	0.330	3.1%
Reactor Waist	103.861	5.4%	97.897	0.6%	60.963	2.3%	6.916	2.3%	3.380	2.8%	0.802	2.2%
Upper Room Floor	18.435	14.6%	15.803	1.8%	14.456	2.2%	14.526	2.0%	14.861	1.9%	1.448	2.1%
Upper Room Waist	15.227	15.4%	12.457	1.8%	11.291	2.4%	10.052	2.3%	10.547	2.1%	2.091	1.8%

Run Time	10 hour		Cool		1 hour							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	36.811	4.4%	36.906	0.5%	1.791	3.9%	0.152	7.5%	0.133	6.3%	0.091	3.1%
Reactor Waist	26.849	5.5%	25.272	0.6%	15.681	1.0%	1.790	2.3%	0.936	2.9%	0.220	2.3%
Upper Room Floor	4.142	15.7%	4.008	1.7%	3.572	2.3%	3.565	2.1%	3.578	1.9%	0.367	2.1%
Upper Room Waist	4.066	14.5%	3.123	1.8%	2.826	2.5%	2.844	2.4%	2.800	2.3%	0.527	1.8%

Run Time	10 hour		Cool		1 day							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	0.240	5.3%	0.210	0.5%	0.008	4.6%	0.001	7.2%	0.001	7.9%	0.001	3.1%
Reactor Waist	0.152	6.0%	0.141	0.6%	0.087	1.1%	0.011	2.3%	0.006	3.0%	0.002	2.3%
Upper Room Floor	0.024	20.8%	0.019	1.9%	0.018	2.5%	0.017	2.3%	0.017	2.1%	0.002	2.4%
Upper Room Waist	0.019	15.9%	0.015	1.9%	0.013	2.8%	0.013	2.3%	0.012	2.5%	0.002	1.9%

Run Time	10 hour		Cool		1 week							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	0.281	4.7%	0.250	1.1%	0.011	4.3%	0.001	8.1%	0.001	10.9%	0.001	3.7%
Reactor Waist	0.198	5.7%	0.169	0.6%	0.104	1.1%	0.013	2.4%	0.007	3.2%	0.002	2.5%
Upper Room Floor	0.022	17.9%	0.024	1.9%	0.022	2.6%	0.021	2.5%	0.021	2.2%	0.002	2.4%
Upper Room Waist	0.017	20.4%	0.018	2.1%	0.017	2.8%	0.016	2.6%	0.015	2.5%	0.003	2.1%

Run Time	10 hour		Cool		1 month							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	0.054	4.4%	0.050	0.5%	0.002	4.0%	0.000	6.8%	0.000	6.2%	0.000	3.6%
Reactor Waist	0.037	5.6%	0.034	0.6%	0.021	1.0%	0.003	2.3%	0.001	3.1%	0.000	2.3%
Upper Room Floor	0.004	16.1%	0.005	1.7%	0.005	2.4%	0.004	2.2%	0.005	2.0%	0.000	2.2%
Upper Room Waist	0.004	18.0%	0.004	1.9%	0.004	2.7%	0.003	2.4%	0.003	2.4%	0.001	1.8%

MCNP Calculated Data for Omaha VA Hypothetical Loss of Coolant Accident
1000 h run time data — Values in R/h — Error is MCNP-reported tally error

Run Time	1000 hour		Cool		1 minute							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	173.228	4.4%	164.590	0.5%	7.863	4.1%	0.682	7.1%	0.650	6.8%	0.388	3.1%
Reactor Waist	121.015	5.5%	112.884	0.6%	70.166	1.0%	7.949	2.4%	4.047	3.0%	0.966	2.2%
Upper Room Floor	19.441	16.2%	16.060	1.7%	16.457	2.3%	16.666	2.1%	16.039	1.9%	1.671	2.3%
Upper Room Waist	19.707	14.6%	14.297	1.9%	13.056	2.5%	12.035	2.4%	11.848	2.3%	2.381	1.8%

Run Time	1000 hour		Cool		1 hour							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	61.706	4.5%	58.099	0.5%	2.709	4.1%	0.308	8.3%	0.229	8.7%	0.151	3.4%
Reactor Waist	40.421	5.9%	38.837	0.6%	24.887	1.0%	2.988	2.5%	1.530	3.1%	0.357	2.3%
Upper Room Floor	5.549	17.2%	5.897	1.8%	5.451	2.4%	5.182	2.3%	5.487	2.1%	0.536	2.3%
Upper Room Waist	4.860	16.9%	4.831	2.0%	4.184	2.7%	3.967	2.5%	3.828	2.4%	0.788	1.8%

Run Time	1000 hour		Cool		1 day							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	16.889	4.8%	16.114	0.5%	0.872	4.8%	0.087	8.2%	0.071	8.2%	0.048	3.3%
Reactor Waist	11.304	5.8%	10.834	0.6%	6.615	1.0%	0.869	2.3%	0.466	2.8%	0.106	2.2%
Upper Room Floor	1.368	11.0%	1.515	1.3%	1.368	1.7%	1.368	1.6%	1.340	1.5%	0.137	1.7%
Upper Room Waist	1.062	9.0%	1.162	1.4%	1.065	1.6%	1.013	1.5%	0.940	1.5%	0.165	1.4%

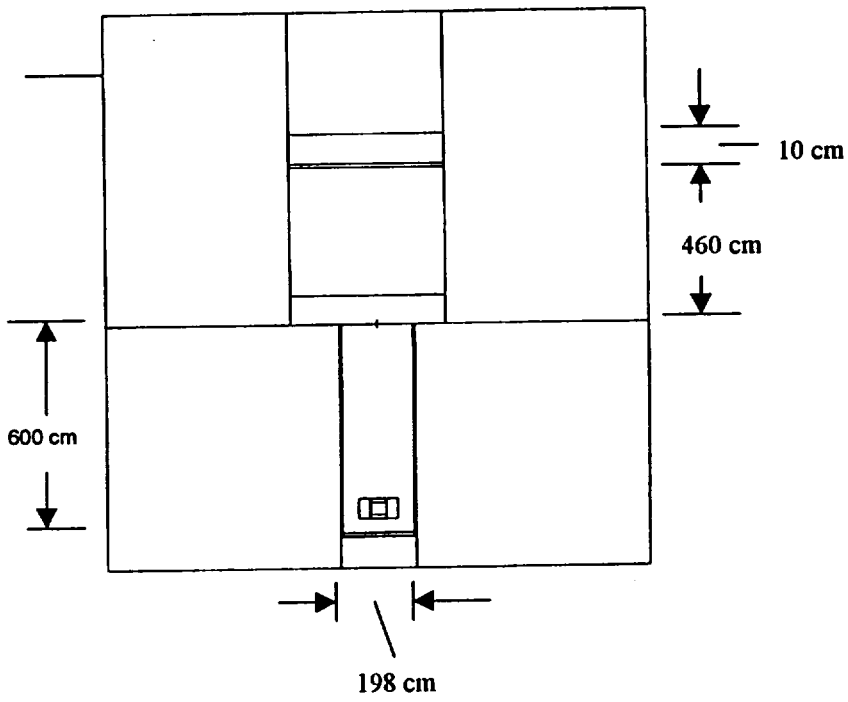
Run Time	1000 hour		Cool		1 week							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	8.527	4.9%	7.867	0.5%	0.363	4.1%	0.044	10.4%	0.032	8.8%	0.021	3.3%
Reactor Waist	5.791	5.8%	5.320	0.6%	3.288	1.0%	0.387	2.4%	0.209	3.1%	0.050	2.2%
Upper Room Floor	0.739	16.0%	0.767	1.8%	0.686	2.4%	0.682	2.3%	0.705	2.1%	0.076	2.3%
Upper Room Waist	0.603	19.0%	0.589	2.0%	0.552	2.7%	0.506	2.5%	0.487	2.4%	0.101	1.8%

Run Time	1000 hour		Cool		1 month							
Location	< 10 cm	error	10-100 c	error	100-125	error	125-150	error	150-175	error	>175	error
Reactor Floor	2.560	4.8%	2.491	0.5%	0.107	4.2%	0.013	8.0%	0.012	11.1%	0.007	3.7%
Reactor Waist	1.862	5.7%	1.867	0.6%	1.038	1.1%	0.131	2.4%	0.066	3.2%	0.016	2.4%
Upper Room Floor	0.204	17.6%	0.238	1.9%	0.219	2.5%	0.208	2.5%	0.210	2.2%	0.022	2.4%
Upper Room Waist	0.166	20.6%	0.183	2.0%	0.168	2.8%	0.156	2.6%	0.149	2.5%	0.031	2.1%

Sub Appendix C-3

```
Loss of Coolant Gamma Dose
 Calculation

probid =
basis:
( 1.000000, 0.000000, 0.000000)
( 0.000000, 0.000000, 1.000000)
origin:
(     0.00,      0.00,      0.00)
extent = (  1000.00,  1000.00)
```



For these calculations the room is taken to be 430 cm by 760 cm. This is conservatively small. The walls and floor are infinitely thick concrete to maximize reflection.

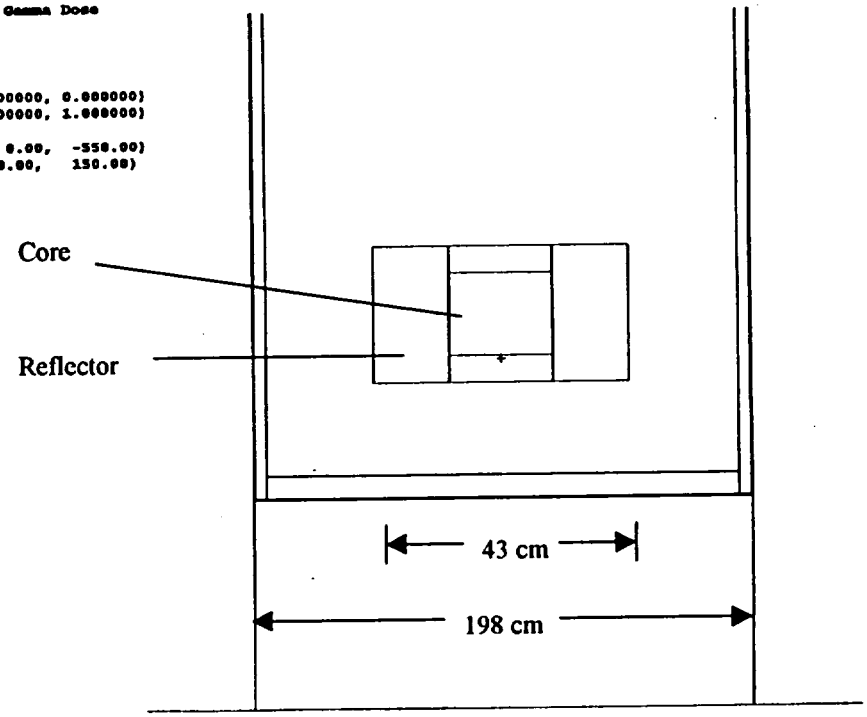
Distance from top of tank to the ceiling is 460 cm., which is 10 cm thick.

Drawings not to scale

Side View

```
Loss of Coolant Gamma Dose
 Calculation

probid =
basis:
( 1.000000, 0.000000, 0.000000)
( 0.000000, 0.000000, 1.000000)
origin:
(      0.00,      0.00,   -550.00)
extent = (   150.00,    150.00)
```



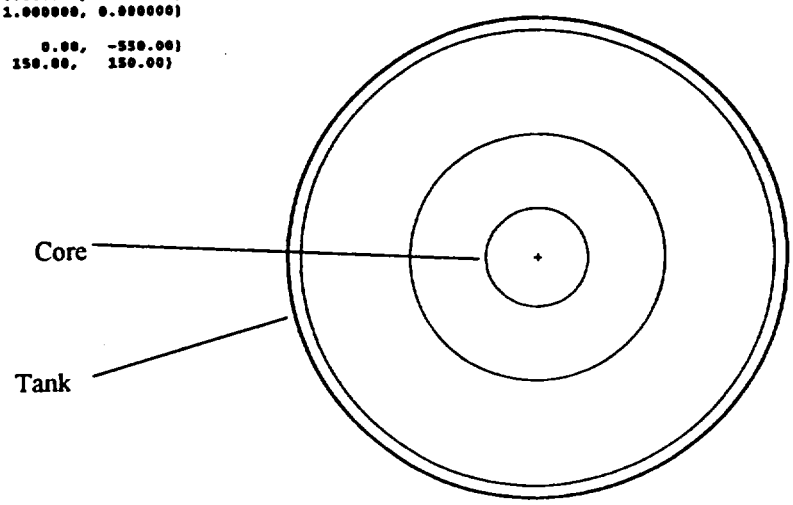
- Core is taken as 43 cm outer diameter.
- Tank is 198 cm outer diameter.
- Core is 501.5 cm deep in tank.

Drawings not to scale

```
Loss of Coolant Gamma Dose
 Calculation

probid =
basis:
( 1.000000, 0.000000, 0.000000)
( 0.000000, 1.000000, 0.000000)
origin:
(      0.00,      0.00,   -550.00)
extent = (    150.00,    150.00)
```

Top View



Drawings not to scale

Appendix C

References

C1 Chilton, A B., R. E. Faw, and J. K. Shultis. Principles of Radiation Shielding. Prentice Hall, New Jersey; 1984.

C2 Los Alamos National Laboratory MCNP 4A Monte Carlo N-Particle Transport Code System CCC-200, RSIC Collection, Oak Ridge, TN: 1993.

APPENDIX D

DETERMINATION OF SOIL ACTIVATION OUTSIDE OF REACTOR TANK

D.1 NEUTRON FLUX ATTENUATION

Purpose

Calculate the penetrating neutron flux for the Veterans Administration (VA) Medical Center nuclear reactor. The results of the flux calculation will be used to estimate soil and ground water activation.

Assumptions

1. The nonhydrogenous materials are less than 5 relaxation lengths thick. This assumption is reasonable since these shields have a relaxation length of $\mu_r t \cong 3.4$.
2. The hydrogenous materials are least than 50 centimeters (cm) between the source and the shield. Since there are 45.7 cm of water and 5 cm of Gunite concrete between the core and the side of the steel vessel, this assumption should be considered valid.
3. The fission neutron spectrum flux on the outside of the core reflector is 3×10^{10} n/cm^2-sec. [D1].
4. The fission neutron spectrum flux on the bottom of the core is 1.2×10^{11} n/cm^2-sec. [D1].
5. The reactor vessel shell is assumed to be made of carbon steel.
6. The thickness of Gunite was neglected in these calculations. This is a conservative assumption, because flux calculations will be slightly over estimated by not including the 5 cm thickness of Gunite.

Inputs

1. Technical information found in reference D6.
2. Attenuation coefficients found in reference D2.
3. Equations and methods found in reference D4.
4. Calculations where performed for a one use, using MathCad Version 5.0 plus by Mathsoft Corporation.

Method

The attenuation coefficients are based on an idealized fission source embedded in an infinite homogeneous hydrogenous medium into which a nonhydrogenous component in the form of a homogeneous slab of thickness t is inserted. According to with reference 3 experimental results have shown the attenuation is related to the flux with the slab in position (i.e., at distance x of hydrogenous medium plus a thickness t of the nonhydrogeneous component) to the flux at a distance x from the source, without the slab, by

$$\phi = \phi_o e^{-\mu_r \cdot t} \qquad 1$$

[4] where μ_r, is the removal coefficient and has a constant characteristic of the nonhydrogenous component for a given fission neutron energy spectrum. Two important requirements must be

met for the removal constant to be valid. First there must be at least 50 cm of hydrogenous material between the source and the nonhydrogenous component. Second, the thickness t must be such that $\mu_r t$ is less than 5.

To calculate the removal coefficients for various compounds, the following empirical formulas (in units of cm^2/g) have been developed to permit interpolation between measured values.

$$\frac{\mu_r}{\rho} = 0.206\, A^{-\frac{1}{3}} Z^{3.294} \qquad 2$$

[4] where A and Z are the atomic mass and atomic number, respectively, for the elements of concern.

Table 1
Removal Coefficients and Attenuation Lengths

Material	Removal Coefficient μ_r (cm^{-1})	Attenuation Length $1/\mu_r$ (cm)
Water	0.103	9.7
Paraffin	0.106	9.4
Iron	0.158	6.34
Concrete (6% H_20)	0.089	11.3
Graphite (density 1.54)	0.079	12.7

Taken from Protection Against Neutron Radiation, NCRP Report No. 38

The steel shell of the reactor vessel was assumed to be carbon steel. Therefore, using reference 2, a compound specific removal coefficient was calculated (see section D.2). According to reference 2, carbon steel is made of 1% Manganese, 0.9% Carbon, and 98.1% Iron. These weight percents were used to calculate the removal coefficient for carbon steel.

Results

The results of the calculation suggest the fission spectrum flux is reduced by over 99% at the vessel soil interface. Table 2 shows the relative geometry and thickness of materials used to calculate the reduced flux.

Table 2
Reactor Vessel Materials and Geometry

Material	Side (cm)	Bottom (cm)	Flux f_o n/cm^2-sec
Water	45.7	61.0	Side ($3x10^{10}$)
Gunite	5.0	5.0	Bottom ($2.1x10^{11}$)
Steel	0.64	0.64	
Concrete	25.0	38.0	

The calculated neutron flux at the surface of each side of the reactor vessel is estimated at $2.6x10^7$ n/cm^2-sec, and $6.9x10^6$ n/cm^2-sec at the bottom.

The following calculations are based on an idealized fission source embedded in an infinite homogeneous hydrogenous medium into which a nonhydrogenous component in the form of a homogeneous slab of thickness x is inserted.

Constants

$\rho_1 := 1 \cdot \frac{g}{cm^3}$ $\rho_2 := 2.35 \cdot \frac{g}{cm^3}$ $\rho_3 := 7.76 \cdot \frac{g}{cm^3}$ $f_1 := 3 \cdot 10^{10} \cdot cm^{-2} \cdot sec^{-1}$

$m_1 := 0.103 \cdot cm^{-1}$ $m_2 := 0.089 \cdot cm^{-1}$ $f_2 := 1.2 \cdot 10^{11} \cdot cm^{-2} \cdot sec^{-1}$

$x_1 := 45.7 \cdot cm$ $x_2 := 25 \cdot cm$ $x_3 := 0.64 \cdot cm$ $x_4 := 38 \cdot cm$ $x_5 := 61 \cdot cm$

Calculate Attenuation Coefficient for Carbon Steel

Carbon steel is made of 1% Mn, .9% C, and 98.1 Fe reference 2

$Z_{eff} := 0.01 \cdot 25 + 0.0009 \cdot 6 + 0.981 \cdot 26$

$Z_{eff} = 25.761$

$A_{eff} := 0.01 \cdot 55 + 0.0009 \cdot 12 + 0.981 \cdot 56$

$A_{eff} = 55.497$

$mper\rho := 0.206 \cdot A_{eff}^{-\frac{1}{3}} \cdot Z_{eff}^{0.294} \cdot \frac{cm^2}{g}$ reference 3

$m_3 := mper\rho \cdot \rho_3$

$m_3 = 0.161 \cdot cm^{-1}$

Calculation of Reactor Flux at Outside Surface of Reactor Vessel from the Side

$s_s := m_1 \cdot x_1 + m_2 \cdot x_2 + m_3 \cdot x_3$

$s_s = 7.035$

$f_s := f_1 \cdot e^{-s_s}$

$f_s = 2.641 \cdot 10^7 \cdot cm^{-2} \cdot sec^{-1}$

Calculation of Reactor Flux at Outside Surface of Reactor Vessel from the Bottom

$s_b := m_1 \cdot x_5 + m_2 \cdot x_4 + m_3 \cdot x_3$

$s_b = 9.768$

$f_s := f_2 \cdot e^{-s_b}$

$f = 6.90 \cdot 10^6$ n/cm² s

1

D.2 SOIL ACTIVATION

To determine the radioactivity of the soil induced by reactor neutrons, a soil target volume of one cubic centimeter was chosen. The target is exposed to the highest possible flux after attenuation through the tank structure. The highest flux that the soil would receive at the bottom and sides of the tank are 6.9×10^6 n/cm^2 s and 2.6×10^7 n/cm^2 s respectively.

The composition of the commoner chemical elements in the earth's crust was taken from the Handbook of Chemistry and Physics [D3]. From a book entitled "Soil Survey for Douglas and Sarpy Counties, Nebraska", U.S. Dept. of Agriculture Soil Conservation Service, dated December 1975, the soil was determined to be Monona silt loam with a permeability of 0.6-2.0 in/hr. Where permeability is defined as an estimate of the rate at which saturated soil transmits water in a vertical direction under a unit head of pressure. Groundwater hydrology is presented with more detail in SAR, Section 2.4.1.

For our model, the shortest distance of travel and the highest flux will be used. Thus, saturation activities will be used and the target volume will be transported by water of equal volume from the bottom of the tank to our defined water table (30 feet depth). Transport time is based on gravitational flow using the soil permeability value of 0.6-2.0 in/hr. Upon reaching our defined water table, the transport time to the site boundary will be based on methodologies used in SAR, Section 2.4.1. The distance from the reactor to the nearest site boundary is 124.5 meters (straight-line distance).

We have been unable to find the depth to the water table. The original 30 ft test boring at the center of the reactor location indicated no water table was encountered (SAR, Sec. 2.4). The total number of activity per cm^3 that reached the 30 foot depth after the soil was saturated with water was calculated. For the purpose of radioactive decay it was assumed that the time started when leached water carried the activity from the bottom of the tank to the water table (straight-line distance). It is not readily apparent that the activated products within the soil about the reactor tank would have any method of transport to the water table. The reactor tank is situated within a poured concrete basement floor. The closest area of exposed soil is a minimum of 28 feet radially from the top of the tank. It is assumed that the water removed all the soil activity and that the concentration of activity was neither diluted nor concentrated. The specific activity (at saturation of the radioisotopes selected) and the appropriate decay factor yield the final activities. Results of the calculations for the specific activities at our defined water table and at the site boundary are shown in Table 3 and 4. Table 3 provides the values used in the calculations. Table 4 summarizes the values of each isotope and compares them with 10 CFR 20 limits.

The above analysis is based on assuming that water reaches the area around the reactor core and flushes the entire induced radioactivity into the water table. How much of the element is mobilized and actually reaches the water depends upon the interaction between the phases of the soil i.e. liquid, solid and gas. For example, the general rules governing the mobilization and fixation of Fe are that oxidizing and alkaline conditions promote the precipitation of Fe, whereas acid and reducing conditions promote the solution of Fe compounds. The released Fe readily precipitates as oxides and hydroxides, but it substitutes for Mg and Al in the minerals and often complexes with organic ligands. The solubility of Fe in soils is extremely low in comparison with the total iron content [D7]. Consequently, even in the worst case scenario, after chemical

reactions, radioactive decay and dilution with the water table the amount of radioactivity that would reach the site boundary would be well below the effluent concentrations specified in 10CFR20, Appendix B, Table 2, Column 2.

TABLE 3

SOIL ACTIVATION (AT SATURATION) FROM OVAMC TRIGA MARK I NUCLEAR REACTOR

ELEMENT	WEIGHT%	ATOM%	VOLUME%	ATOMIC WEIGHT	TARGET ISOTOPE	%NATURAL ABUNDANCE	CROSSECTION BARNS	ATOMS/ g SOIL	ATOMS/ cm^3 SOIL
OXYGEN	46.60	65.55	91.92	15.9994	O-18	0.204	0.00016	3.58E+19	1.35E+19
SILICON	27.72	21.22	0.80	29.9738	Si-30	3.100	0.10700	1.73E+20	6.54E+19
ALUMINUM	8.13	6.47	0.77	26.9815	Al-27	100.000	0.23300	1.81E+21	6.87E+20
IRON	5.00	1.92	0.68	57.9332	Fe-58	0.280	1.20000	1.45E+18	5.51E+17
MAGNESIUM	2.09	1.84	0.56	25.9826	Mg-26	11.010	0.03600	5.33E+19	2.02E+19
CALCIUM	3.63	1.94	1.48	45.9537	Ca-46	0.004	0.70000	1.90E+16	7.21E+15
CALCIUM	3.63	1.94	1.48	47.9525	Ca-48	0.187	1.10000	8.52E+17	3.23E+17
SODIUM	2.83	2.64	1.60	22.9898	NA-23	100.000	0.40000	7.41E+20	2.81E+20
POTASSIUM	2.59	1.42	2.14	40.9618	K-41	6.730	1.46000	2.56E+19	9.70E+18

				CORRECTED FOR DECAY	
ELEMENT	RADIOACTIVE ISOTOPE	uCi/ cm^3 SOIL	HALFLIFE HOURS	water table TOTAL uCi	site boundary TOTAL uCi
OXYGEN	O-19	1.52E-06	7.47E-03	less than 1E-10	less than 1E-10
SILICON	Si-31	4.92E-03	2.62E+00	less than 1E-10	less than 1E-10
ALUMINUM	Al-28	1.12E-01	3.75E-02	less than 1E-10	less than 1E-10
IRON	Fe-59	4.65E-04	1.07E+03	4.49E-04	less than 1E-10
MAGNESIUM	Mg-27	5.11E-04	1.58E-01	less than 1E-10	less than 1E-10
CALCIUM	Ca-47	3.54E-06	1.09E+02	2.56E-06	less than 1E-10
CALCIUM	Ca-49	2.50E-04	1.45E-01	less than 1E-10	less than 1E-10
SODIUM	NA-24	7.89E-02	1.50E+01	7.36E-03	less than 1E-10
POTASSIUM	K-42	9.96E-03	1.24E+01	5.63E-04	less than 1E-10
			TOTAL uCi =	8.37E-03	less than 1E-10

NEUTRON FLUX (n/cm^2 sec) =	2.60E+07
TIME FOR WATER TO TRAVEL TO DEFINED WATER TABLE (hr) =	51.22
TIME FOR WATER TO TRAVEL TO SITE BOUNDARY (hr) =	1.73E+04

Table 4

	SOIL ACTIVITIES AT 30' DEPTH AND AT SITE BOUNDRY		
			10 CFR 20 (EC)
RADIOACTIVE ISOTOPE	Water Table TOTAL uCi	Site Boundary TOTAL uCi	Table 2 Col. 2 uCi/ml
O-19	less than 1E-10	less than 1E-10	na
Si-31	less than 1E-10	less than 1E-10	1.00E-04
Al-28	less than 1E-10	less than 1E-10	na
Fe-59	4.49E-04	less than 1E-10	1.00E-05
Mg-27	less than 1E-10	less than 1E-10	na
Ca-47	2.56E-06	less than 1E-10	na
Ca-49	less than 1E-10	less than 1E-10	na
NA-24	7.36E-03	less than 1E-10	5.00E-05
K-42	5.63E-04	less than 1E-10	6.00E-05
total	8.37E-03	less than 1E-10	

na = not applicable

Appendix D

References

D1 J.E. Larson, "Calculated Fluxes and Gross Sections for TRIGA Reactors", General Atomic GA-4361, Suppl. B. 1966.

D2 NCRP No. 38. "Protection Against Neutron Radiation". National Council on Radiation Protection and Measurements Washington DC. 1971.

D3 CRC Handbook of Chemistry and Physics 61st Edition. 1981.

D4 Arthur B. Chilton, J. Kenneth Shultis, Richard E. Faw. "Principles of Radiation Shielding". Prentice-Hall. 1984.

D5 James E. Tuner. "Atom, Radiation, and Radiation Protection". Pergamon Press. 1986.

D6 Section 3.2. "Reactor and Reactor System". Construction Permit for VA Medical Center.

D7 A.Kabata-Pendias and H. Pendias, "Trace Elements in Soils and Plants", CRC Press, Boca Raton, Fl. 1984.

APPENDIX E

REACTIVITY CALCULATIONS FOR FUEL STORAGE PIT

The criticality coefficient k_{eff} of the fuel storage pits was calculated using MCNP [E1], which is the Monte Carlo Neutral Particle code developed and maintained by Los Alamos. Evaluated were cases for 15, 19 and 25 Mark II element loading under dry and wet conditions. Twenty-five elements is the maximum that can reasonably fit into the pit, and 15 elements was used to bracket the estimated 19 elements that our facility would need to load into each pit to store their entire core inventory.

E.1 PROBLEM MODELING AND ASSUMPTIONS:

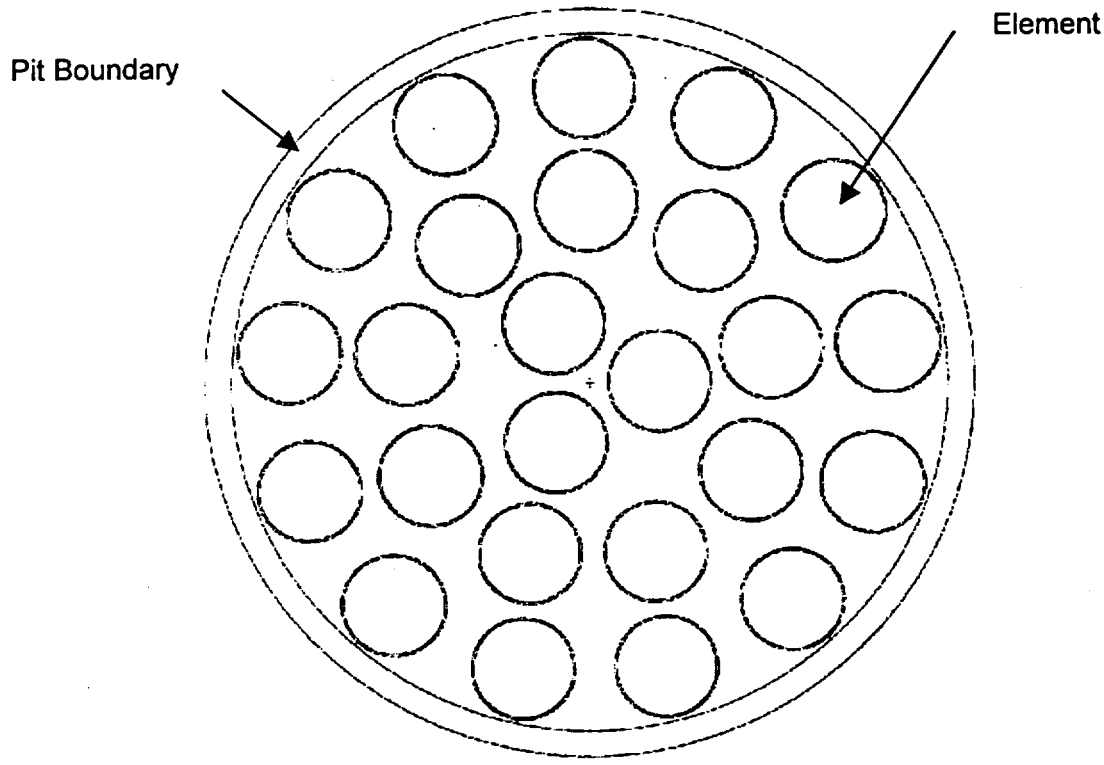
For this problem, detailed modeling was used instead of homogenization. The dimensions, material content, and composition of the fuel were taken from General Atomic specifications for TRIGA Mark II aluminum elements with $ZrH_{1.0}$, 8 wt-% U, and 20% enrichment. The pit was modeled from design drawings with steel surrounded by concrete. An additional 10 mean-free-paths of concrete was placed below the pit as well for scattered neutrons. The geometrical modeling was accurate, except that the aluminum triflute at the top of each element was modeled as a cylinder of similar volume. Gaps between the fuel and cladding, spacers, etc. were neglected, as most were less than 1 mm. Also stainless steel was used in place of the reported galvanized mild steel; however, the basic compositions are similar and unlike to influence the results. Fuel layout was selected to provide nearly equal distances between fuel elements to maximize moderation for the water-moderated cases. Sample plots from VISED, an MCNP geometry plotter, are shown [Figures 1 & 2]. Please bear in mind that each plot is only a two-dimensional slice of the corresponding three-dimensional model.

Neither the samarium wafers or uranium burnup were accounted for in this calculation, although both would reduce the actual value of k_{eff}. To account for neutronic coupling between the storage pits, reflective boundaries were chosen at the midpoints between the pits. Therefore an infinite square array of these pits is calculated. Since the reactor has just three pits aligned in a row, this will also tend to reduce the actual value. In the event that the fuel elements are shifted, the fuel is already close-packed and further reductions in inter-element spacing are likely to similarly reduce the values, since moderation is likely to be more a more dominant effect than leakage. This is due to a much greater decrease in water volume than the corresponding decrease in surface area. These assumptions allow for some conservatism in the answers presented.

E.2 CODE VALIDATION:

The code package used was MCNP-4B with the ENDF/I-V cross section library. The code was run on a Sun Enterprise 2, for which the validation runs were completed. The ZrH moderator was treated using the special ZAID for TRIGA fuel. The production run contained 500 cycles each with 1000 source particles. The material compositions were checked principally for the ^{235}U content. The MCNP output indicated 35g per element, which is the correct nominal weight (actual weights should be less due to burnup). Additionally, the code was run for different intervals (e.g. 100 cycles with 5000 particles) and different initial random numbers. Similar results were found for each case.

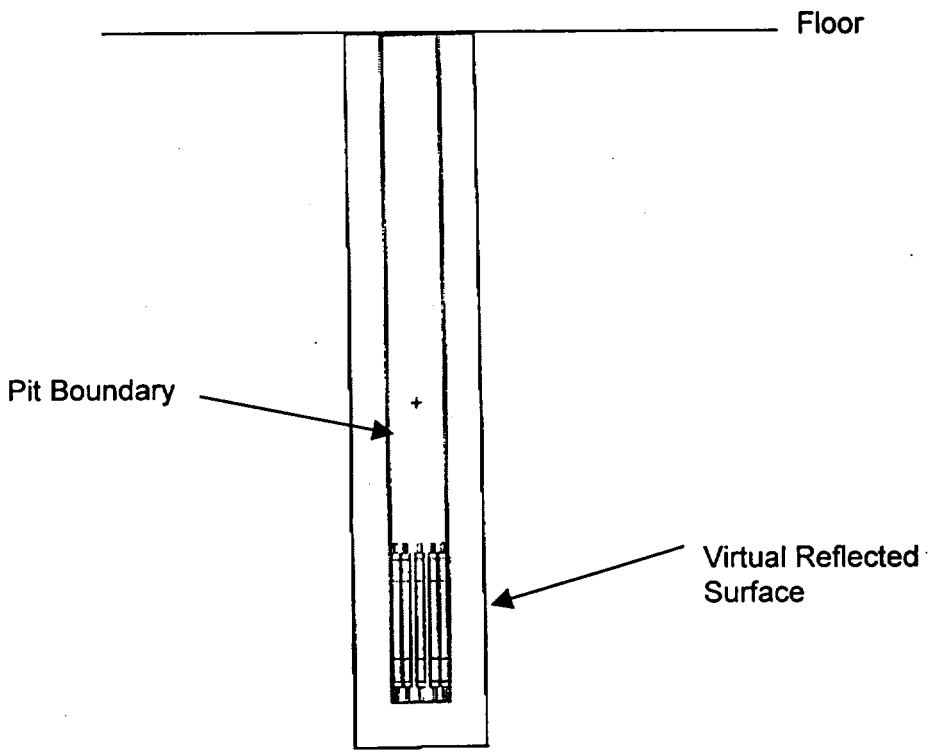
Top View




```
triga spent-fuel storage pit
 criticality analysis-unit cell
 w/ per. B.C.
probid =
basis:
( 1.000000, 0.000000, 0.000000)
( 0.000000, 1.000000, 0.000000)
origin:
(     0.00,     0.00,     0.00)
extent = (    15.00,    15.00)
```

Figure E.1

Side View




```
Triga spent-fuel storage pit
 criticality analysis-unit cell
 w/ per. B.C.
probid =
basis:
( 1.000000, 0.000000, 0.000000)
( 0.000000, 0.000000, 1.000000)
origin:
(     0.00,     0.00,   100.00)
extent = (   200.00,   200.00)
```

Figure E.2

E.3 RESULTS:

MCNP outputs seven different estimators for k_{eff}: collision, absorption, track length, and the respective combined averages of the three. In each of the cases all of the estimators were well within the standard deviation of the combined average (and also of one another), so the combined average value was used. MCNP bases its error estimates upon the convergence of the estimator and are given as one standard deviation, or 68% confidence interval.

An unexpected effect was observed that the value for 25 elements in air was slightly smaller than for the 19 elements in air. The most likely explanation is due to the greater number of elements at the outer periphery combined with the lack of an element in the direct center. Without the water in place, neutronic coupling between elements is essentially limited to line-of-sight. Thus the 25 element may have significantly more leakage.

Table E-1

Summary of K_{eff} Values for Various Conditions

Number of Elements	Water-Moderated	Air
15	0.643 ± 0.001	0.309 ± 0.003
19	0.765 ± 0.002	0.413 ± 0.002
25	0.790 ± 0.002	0.404 ± 0.001

It should be kept in mind that the values in Table E-1 are conservatively large. However, they remain below the 0.8 recommended maximum limit for stored fuel. In conclusion, these calculations indicate that 19 elements per storage pit (Section 3.2.1) will be sufficiently subcritical, with $k_{eff} < 0.8$, under all conditions of moderation. For validation and comparison, the analyzed reference for 19 elements in air in five pits to be 0.45 [E2], which is comparable to the calculated value of 0.41.

E.4 SAMPLE CALCULATIONS

Sample MCNP Input File For 15 Elements:

```
Triga spent-fuel storage pit criticality analysis-unit cell w/ per. B.C.
C ********************************************************************** C
C Geometric Model: Square unit cell with a pitch of 60.325 cm. Periodic
C                  B.C. are applied on cell boundries. Fuel and pit
C                  model are the same as in the infinite concrete
C                  medium case. Approximately 10 mfps of concrete is
C                  placed at the bottom of the pit. 15 fuel elements
C
C Variance Red. Used: none
C
C Special Physics Treatments: implicit capture--no phys:n card in the deck
C
C Last modified: 8/26/98
C ********************************************************************** C
c
c
C ************ C
C  CELL CARDS  C
C ************ C
c
c ###Definition of Universe 1--Fuel element slightly larger than actual F.E.###c
1     1    -5.8121  -2 -5 6      u=1       imp:n=1     $meat
3     3    -2.62    -7  5 -2       u=1       imp:n=1     $Top reflector
4     3    -2.62    -6  8 -2       u=1       imp:n=1     $Bottom reflector
5     7    -7.92    -9  7 -3       u=1       imp:n=1     $Top end cap
6     7    -7.92   -11  9 -4       u=1       imp:n=1     $Top end cap
7     7    -7.92    -8 10 -3       u=1       imp:n=1     $Bottom end cap
8     7    -7.92   -10 12 -4       u=1       imp:n=1     $Bottom end cap
9     7    -7.92    -3  2 -7 8     u=1       imp:n=1     $Cladding
c ###End definition of universe 1###c
c
c ###Window cell for the first F.E.###c
20    0          (-20 -23 24):(-22 -21 23):(-21 -24 25) fill=1 imp:n=1 $First F.
c
c ###Copies of the above F.E.###c
21    like 20 but trcl=(  3.8180   3.8180 0.0) $First ring
22    like 20 but trcl=( -3.8180   3.8180 0.0)
23    like 20 but trcl=( -3.8180  -3.8180 0.0)
24    like 20 but trcl=(  3.8180  -3.8180 0.0)
25    like 20 but trcl=( 10.700    0.0    0.0) $Second ring
26    like 20 but trcl=(  8.6465   6.2893 0.0)
27    like 20 but trcl=(  3.3065  10.1763 0.0)
28    like 20 but trcl=( -3.3065  10.1763 0.0)
29    like 20 but trcl=( -8.6465   6.2893 0.0)
30    like 20 but trcl=(-10.700    0.0    0.0)
31    like 20 but trcl=( -8.6565  -6.2893 0.0)
32    like 20 but trcl=( -3.3065 -10.1763 0.0)
33    like 20 but trcl=(  3.3065 -10.1763 0.0)
34    like 20 but trcl=(  8.6465  -6.2893 0.0)
c
c ###Medium surrounding the F.E.###c
50    2 -1.0 (-60 -63 25) #20 #21 #22 #23 #24 #25 #26 #27 #28 #29 #30 &
                          #31 #32 #33 #34  imp:n=1 $Water surrounding the F.E.s
c
c ###Storage Pit###c
51    4   -7.92     60 -61 -63 25          imp:n=1 $S.S. storage pit
52    4   -7.92    -61 -25 64              imp:n=1 $ "   "       "
53    4   -7.92    -61 -62 63              imp:n=1 $ "   "       "
53    4   -7.92    -61 -62 63              imp:n=1 $ "   "       "
c
c ###Concrete surrounding the pit
54    6   -2.3     (61 -62 64 -81 -83 82 80):(-64 84 -81 -83 82 80) imp:n=1
55    0             81:83:-80:-82:62:-84    imp:n=0 $Void outside periodic B.C.

C *************** C
C  SURFACE CARDS  C
```

```
C *************** C
c
c ###Fuel Element---Universe One###c
2    cz      1.7907    $Outer surface of the meat
3    cz      1.8679    $Outer surface of the element(20 mils cladding+)
4    cz      0.764   $Top and bottom (cylinder...?) + 0.01 (0.754)
5    pz      17.78    $Top of the meat/Bottom of axil reflector
6    pz     -17.78    $Bottom  of the meat/Top of axil reflector
7    pz      27.94    $Top of top axial reflector/Bottom of top S.S. cap
8    pz     -27.94    $Bottom of bottom axial reflector/Top of bottom S.S. cap
9    pz      30.65    $Top of the top S.S. cap +0.01 (30.64)
10   pz     -30.65    $Bottom of the bottom S.S. cap - 0.01 (-30.64)
11   pz      35.10    $Top of top fixture + 0.01 (35.09)
12   pz     -37.00    $Bottom of bottom fixture - 0.01 (-36.99)
c ###End of Fuel Element Cards###c
c
c ###Surfaces for first F.E. window###c
20   cz       1.8669 $Outer surface of the cladding
21   cz       0.754  $Top/Bottom cylinders
22   pz      35.09    $Top of top cylinder
23   pz      30.64    $Top of main cylinder
24   pz     -30.64    $Bottom of main cylinder
25   pz     -36.99    $Bottom of bottom cylinder
c
c ###Storage Pit###c
60   cz      12.7254 $Inside cylinder of the pit
61   cz      13.6525 $Outside cylinder of the pit
62   pz     267.81    $Top surface of the lid
63   pz     267.18    $Bottom surface of the lid
64   pz     -37.625   $Bottom surface of the bottom plate
c ###End of Storage Pit Cards###c
c
c ###Square Unit Cell with periodic B.C.###c
80  -81 px -30.1625  $16.51 cm(6.5") of concrete
81  -80 px  30.1625  $" "
82  -83 py -30.1625  $" "
83  -82 py  30.1625  $" "
84   pz     -57.625   $~10 mfp below bottom of pit
c

C ************ C
C  DATA CARDS  C
C ************ C
c
c ###Physics Cards###c
c phys:n 20 20
c
c ###Criticality Cards###c
kcode 1000 0.64 0 500
ksrc  0.4   0.0 0.0    4.1  4.1 0.0   -4.1  4.1 0.0   -4.1 -4.1 0.0 &
       4.1  -4.1 0.0   11.1  0.0 0.0    9.0  6.5 0.0    3.4 10.6 0.0 &
      -3.4  10.6 0.0   -9.0  6.5 0.0  -11.1  0.0 0.0   -9.0 -6.5 0.0 &
      -3.4 -10.6 0.0    3.4 10.6 0.0    9.0 -6.5 0.0
c
PRINT -128 -98 -70 -50
c
c ###Material Cards###c
c
c ------------------------------------ c
c U + Zr-H  rho=5.8121 g/cm^3          c
c ENDF/B-V continuous data (300 K)     c
c S(a,b)- Zr-H @ 300 K                 c
c ------------------------------------ c
m1     1001.50c  0.6121     40000.50c  0.3743 &
      92235.51c  0.002720   92238.50c  0.01088
mt1   h/zr.01t
c
c ------------------------------------ c
c Light water rho=1.00 g/cm^3          c
c ENDF/B-V continuous data (300 K)     c
c S(a,b)- H20 @ 300K                   c
```

```
c ------------------------------------ c
m2    1001.50c  0.66667    8016.50c  0.33333
mt2   LWTR.01T
c
c ------------------------------------ c
c Graphite rho=2.62   g/cm^3           c
c ENDF/B-V continuous data (300 K)     c
c S(a,b)- Carbon @ 300K                c
c ------------------------------------ c
m3    6000.50c  -1.0
mt3   GRPH.01T
c
c ------------------------------------ c
c Stainless Steel rho=7.92 g/cm^3      c
c ENDF/B-V continuous data (300 K)     c
c No thermal treatment                 c
c Composition taken from MCNP manual   c
c Apx. C-10                            c
c ------------------------------------ c
m4    26000.50c  -0.695   24000.50c  -0.190 &
      28000.50c  -0.095   25055.50c  -0.020
c ------------------------------------ c
c Air rho=0.0012 g/cm^3
c ANSI/ANS 6.4.3 composition           c
c ENDF/B-V continuous data @300K       c
c ------------------------------------ c
m5    7014.50c  -0.75519  8016.50c -0.23179 &
      6012.50c  -.00014   18000.35c  -0.01288
c
c ------------------------------------ c
c Standard concrete rho=2.3 g/cm^3     c
c ANSI/ANS-6.4.3                       c
c Mass fractions taken from "Radiation  c
c Shielding" J.K.S & R.E.F             c
c ENDF/B-V continuous data (300 K)     c
c ------------------------------------ c
m6     1001.50c  -0.005599   13027.50c  -0.045595    8016.50c  -0.498250 &
      14000.50c  -0.315768   11023.50c  -0.017098   16032.50c  -0.001200 &
      12000.50c  -0.002400   19000.50c  -0.019198   20000.50c  -0.082592 &
      26000.50c  -0.012299
c ---------------------------------------c
c Aluminum Clad                          c
c Density 2.7 g/cm^3                     c
c ENDF/B-V continuous data @ 300K        c
c ---------------------------------------c
m7    13027.50c -1.0
```

Appendix E

References

E1 Los Alamos National Laboratory MCNP 4A Monte Carlo N-Particle Transport Code System CCC-200, RSIC Collection, Oak Ridge, TN: 1993.

E2 FSAR, Stationary neutron Radiography System, McClellan Air Force Base, p. 9-5. Jan. 1992.

APPENDIX F

GAMMA FLUX FROM IRRADIATED FUEL

This section determines the gamma flux emanating from the irradiated fuel element upon shutdown. The mean weighted energy of emission and the dose rate from a single element are separately examined.

F.1 PROBLEM MODELING AND ASSUMPTIONS:

The gamma flux will be modeled as a group representation from 0 to 7.5 MeV. The source strengths per fission as a function of time were calculated using FPGCAL, a program written by Dr. R. Faw to evaluate the expressions contained in Appendix 6 of his book. These expressions were based on data obtained from ORIGEN, a fission product decay chain code. The average energy for the SAR, Appendix C calculation was evaluated by examining the extremes of short and long irradiations with short and long cooling times.

For the calculation of dose from a single element, the irradiation time was taken to be 8 h at 20 kW. The power was spread over 57 elements with a peaking factor of 2 for the most activated element.

F.2 CALCULATIONS:

Using the aforementioned FPGCAL program, the following gamma release rates were calculated:

Gamma Energy Release Rates

Energy Group	0-1	1-2	2-3	3-4	5-6	5-7.5	
Average Energy	0.5	1.5	2.5	3.5	4.5	6.25	Avg.
Irradiate/Cool			(MeV per fission)				(MeV)
10 h /1 month	5.90E-04	5.14E-04	3.53E-05	3.98E-07	1.09E-07	1.39E-08	1.01
1000 h / 1 m	1.49E+00	1.23E+00	5.06E-01	1.61E-01	8.06E-02	1.02E-02	1.39
1000 h / 1 month	3.80E-02	2.05E-02	1.43E-03	1.35E-05	2.87E-06	3.84E-07	0.89
8 h / 0	2.24E+00	1.84E+00	8.10E-01	3.05E-01	1.66E-01	4.28E-02	1.48

The overall energy-weighted average for the gamma flux was found to be 1.2 MeV for the first four cases listed.

The next calculation involves the gamma dose from a single fuel element. The number of fissions occurring per second in a fuel element:

$$\frac{\#\ fissions}{s} = \frac{20\times10^{3} J}{s}\,\frac{6.24\times10^{12} MeV}{J}\,\frac{fission}{200 MeV}\times\frac{2(peakfactor}{57 elements}) = 2.18\times10^{13}$$

The gamma dose rate from the above table for 8h or operation is 5.4 MeV·fission^{-1}. The dose rate in air X is given by

$$X = \frac{6CEN}{r^{2}}$$

where C is the activity in Ci, EN is the energy release per second, and r is the distance in feet from the source. Thus the exposure rate at 6 ft after 8 hours of operation is $X= 2.18\times10^{13}\div3.7\times10^{10}\times5.4\div36 = 90\ R\cdot h^{-1}$.

F.3 RESULTS SUMMARY:

In defense of the average value of 1.0 MeV photons used in Appendix C, calculations show a value of 1.2 MeV, which is comparable. Using this new estimate absorption values may be decreased by about 10% based on air data, and total dose rates will likely increase by about 10%. The total energy release rate in either case is unaffected by this the value.

With regard to immediately after irradiation for 8 h at full power, the estimated exposure rate at 6 ft is 90 $R\cdot h^{-1}$.

Appendix F

References

F1 Chilton, A B., R. E. Faw, and J. K. Shultis. *Principles of Radiation Shielding.* Prentice Hall, New Jersey; 1984.