

TRIGA Reactor Facility
Nuclear Engineering Teaching Laboratory
The University of Texas at Austin

SAFETY ANALYSIS REPORT

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The University of Texas TRIGA

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Chapter 1

INTRODUCTION AND SUMMARY

This report describes the TRIGA¹ reactor and The University of Texas facility, 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. A TRIGA type reactor was first operated between 1963 and 1988 on the main campus of The University of Texas at Austin. Subsequent operation experience included safe operation of the facility at steady state thermal power levels of 10 kW and 250 kW, and pulse powers of 250 Mw. Operation of a TRIGA reactor at the Balcones Research Center of The University of Texas is expected for steady state power levels of 1.1 MW(t) and pulse powers of 1700 MW(t).

Some values used in this report represent the latest design parameters, or maximum values as a means of evaluating the safety of the system. For this reason, these values may differ from those quoted in other documents or from those that will be measured in the operating reactor system. Safety analysis demonstrates safe operation at power levels as high as 1.5 MW steady state and 8400 MW peak pulse power.

Operation of the TRIGA reactor is one function of the Nuclear Engineering Teaching Laboratory. Laboratory functions consist of facilities and programs for the purpose of education, research, and technology development.

1.1 PRINCIPAL DESIGN CRITERIA

The reactor will be operated in two modes: pulse and steady-state. Reactor power levels in the steady-state mode will range up to and include 1.1 MW(t). Pulse mode operation will take place by step reactivity insertions with the reactor initially at a power level less than 1 kW. The maximum step reactivity insertion will be 2.2% $\delta k/k$ (53.14) which will produce a peak reactor power of approximately 1700 MW(t) with a prompt energy release of about 21 MW-sec. 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 reactor pool structure. Reactor cooling will be provided by natural circulation of pool water which is cooled and purified in external coolant circuits. Reactor experiment facilities will include a rotary specimen rack, a pneumatic transfer system, core irradiation tubes, and horizontal and vertical beam tubes.

¹ Manufactured by General Atomics, TRIGA Reactor Division

Table 1-1

PRINCIPAL DESIGN PARAMETERS

Reactor type	TRIGA Mark II
Steady state power (maximum)	1.1MW (1.5MW design)
Pulse power (maximum)	2.2% $\delta k/k$ (\$3.14)
Fuel element design	
Fuel-moderator material	U-ZrH
H/Zr ratio	1.6 (1.65 maximum)
Uranium content	8.5 wt %
Uranium enrichment	19.7% U-235
Shape	Cylindrical
Length of fuel	38 cm (15 in.) overall
Diameter of fuel	3.63 cm (1.43 in) o.d.
Cladding material	304 stainless steel
Cladding thickness	0.051 cm (0.020 in.)
Number of fuel elements	
Critical core	-64
Operational core	-90
Excess reactivity, maximum	4.9% $\delta k/k$
Number of control rods	4
transient (w/air follower)	1
regulating (w/fuel follower)	1
shim (w/fuel follower)	2
Total reactivity worth of rods	8.7% $\delta k/k$
Reactor cooling	Natural convection of pool water

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 and of these 31 are pulsing. TRIGA reactors have more than 450 reactor years of operating experience, over 30,000 pulses, 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-moderator 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 on 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 step reactivity insertions as large as 3.5% $\delta k/k$ with resulting peak reactor powers up to 8400 MW(t) on TRIGA cores containing similar fuel elements as are used in this TRIGA reactor.

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 [1-6].

1.3 CONCLUSIONS

Past experience has shown that TRIGA systems can be designed, constructed, and safely operated in the steady state and pulsing modes 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 such as is normally associated with reactors of like power levels.

Results of this safety analysis indicate that the TRIGA Mark II reactor system proposed for construction and operation 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 with the reactor at low and high power levels,
- b. Complete and instantaneous loss of coolant water in the reactor pool,
- c. And fission product release from a fuel element ruptured in air.

The insertion of excess reactivity may represent a normal reactor operating condition, while the loss of pool water is expected to be an abnormal condition. Conservative estimates of doses from fission product releases are made independent of accident scenarios.

In both these postulated 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 excessive radiation exposures.

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. "Hazards Report for TRIGA Mark II Pulsing Reactor", General Atomic Division Report GA-1998, February 1961.
2. "Hazards Summary Report for a TRIGA-I Nuclear Reactor", University of Texas Bureau of Engineering Research, October 1961.
3. "Safeguards Analysis Report for TRIGA Reactors using Aluminum-Clad Fuel", General Atomic Division Report GA-7860, March 1967.
4. "Safety Analysis Report for 250 Kilowatt Operation of a TRIGA Mark I Nuclear Reactor", University of Texas, College of Engineering August 1967.
5. "Safety Analysis Report for the TRIGA Mark II Reactor E-117-478, General Atomic Company, October, 1975.
6. "TRIGA Mark I Safety Analysis Report", University of Texas, January 1981.

Chapter 2

SITE DESCRIPTION

The site for the TRIGA reactor facility will be located on the east tract of the Balcones Research Center, a tract of land owned and operated by The University of Texas. The Research Center is located in northern Travis County and the City of Austin about 11.6 kilometers north-northwest of The University of Texas at Austin campus. Figures 2-1 thru 2-4 display the facility locations in relation to surrounding areas. Located near the transition line between hill country and rolling plains, the site is situated about 7.4 kilometers from where the flood controlled Colorado river crosses the transition region and Balcones fault zone. The Balcones Research Center east and west tracts span part of the inactive fault zone. The east tract is within the transition region to rolling plains.

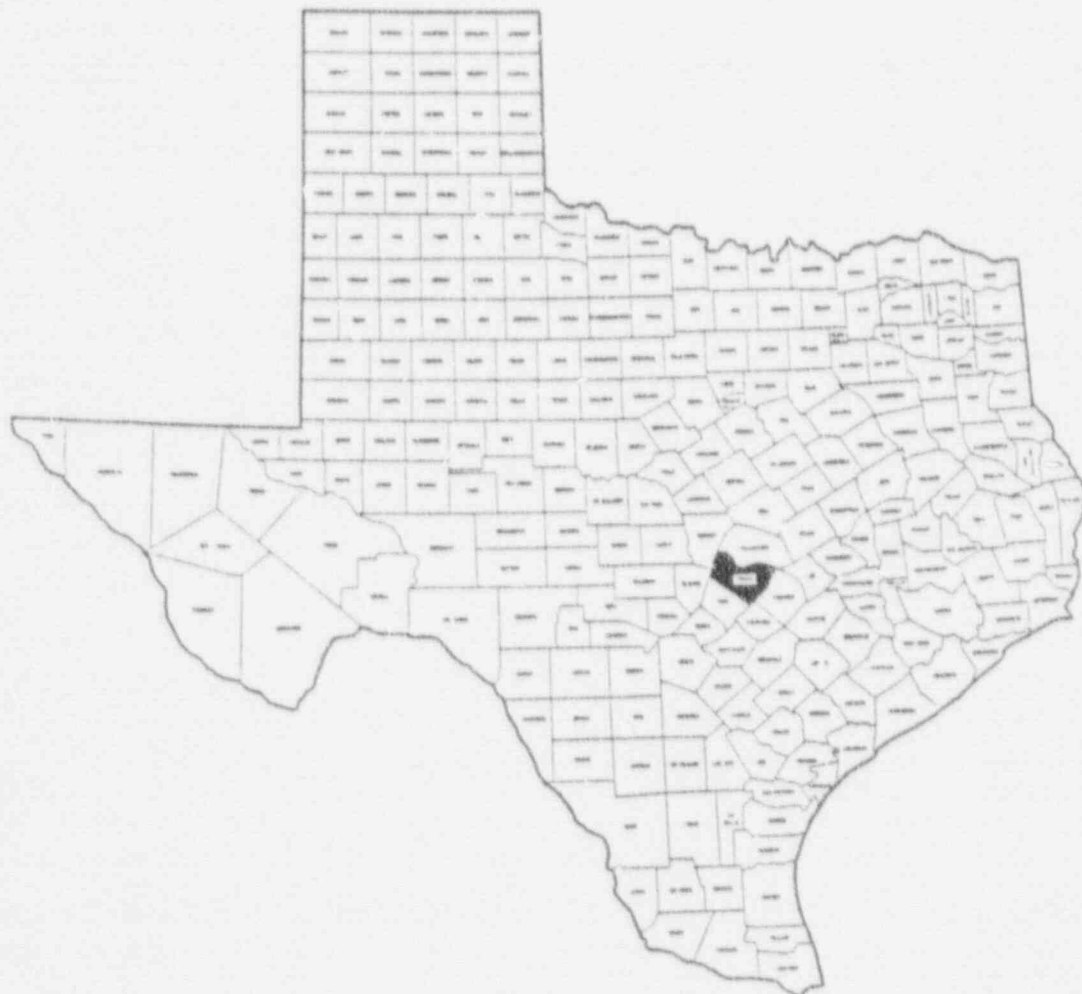
Site location of the TRIGA reactor is in the northeast region of the research center east tract. Adjacent to the north boundary of the research center and near to the eastern boundary, the site location is near the intersection of Braker Lane and Burnet Road. Figure 2-4 shows the site location within the Balcones Research Center. Reference guidance for site evaluation was ANS 15.7 [1].

2.1 GENERAL LOCATION AND AREA

Major activities of The University of Texas at Austin, State of Texas government, and City of Austin business district are centered at respective distances of 11.6, 12.6, and 12.9 kilometers to the south-southwest. Distances to air traffic landing facilities in the area are 6.2 kilometers for private aircraft, 9.7 kilometers for commercial aircraft and 20.8 kilometers for military aircraft.

A total area of 1.87 square kilometers is contained within the Research Center east and west tracts of land. The east side of the Center is bounded by a State highway, FM 1325, and the west side is bounded by a Federal highway, US 183. The two tracts are divided by a rail line, formerly the Missouri-Pacific, with 0.93 square kilometers in the east tract and 0.94 square kilometers in the west tract of land. Highway intersections of US 183 with FM 1325 and with Loops 1 and 360 are within two kilometers of the site. Both highway loops are planned for extension into the area associated with the west tract of land [2].

An area of about 9000 square meters in a rectangular shape of 120 meters by 75 meters will comprise the general site location. The 120 meter length is along the north research center boundary. Areas for parking, landscape and access roads are within the general site area. A buffer zone exists between the site area and activities or structures to the east and west. To the west the buffer zone is about 55 meters by 75 meters with parking also about 60 meters by 75 meters. The east buffer region is primarily open space that will provide the access to other development projects north of the general site area.



STATE OF TEXAS COUNTIES

Figure 2-1



TRAVIS COUNTY

Figure 2-2

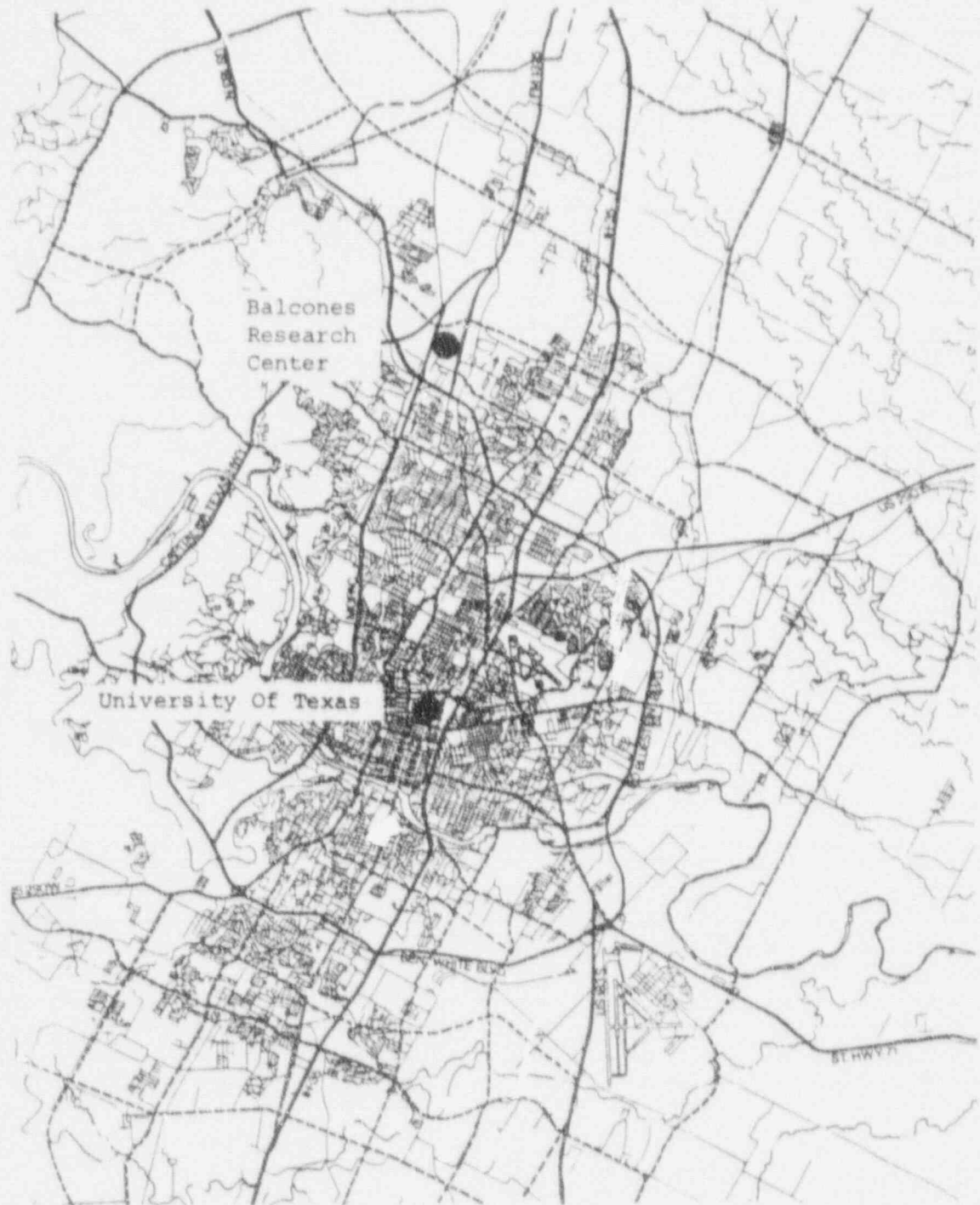
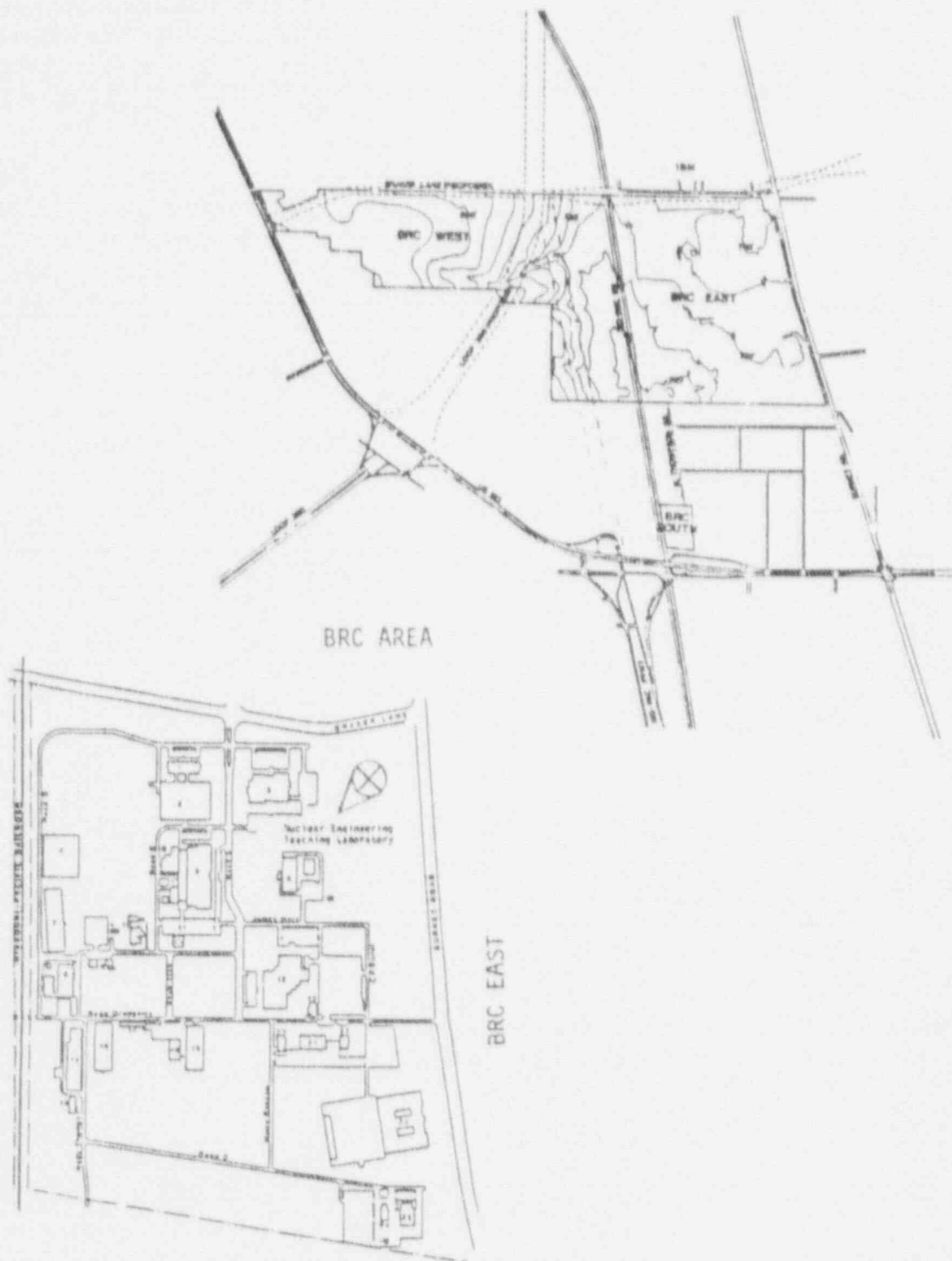


Figure 2-3



BALCONES RESEARCH CENTER

Figure 2-4

Most areas adjacent to the Research Center are developed for mixed commercial and industrial activities including warehouses, manufacturing facilities, small business parks and a few undeveloped areas. Mixed commercial and industrial areas south and east of the Research Center are bounded by highway US 183, highway FM 1325, and the Texas New Orleans Railroad to the east. Approximately 2.2 square kilometers of land are enclosed by the area. Much of the remaining area to the west of the Research Center is bounded by highway US 183 and the Missouri Pacific Railroad and is not developed. The area is planned for future road right-of-ways and includes the west tract area. Immediately north of the Balcones Research Center east tract is a 2.3 square kilometer complex operated by International Business Machines Corporation. Undeveloped areas around the Research Center are expected to develop as a mixed commercial, industrial, research or office park areas. Residential areas are located beyond adjoining areas around the Balcones Research Center with distances from the reactor facility site of 1.2 kilometers to 2.0 kilometers. Few residential structures, either multifamily or single family units are located within a radius of 1.2 kilometers of the reactor site.

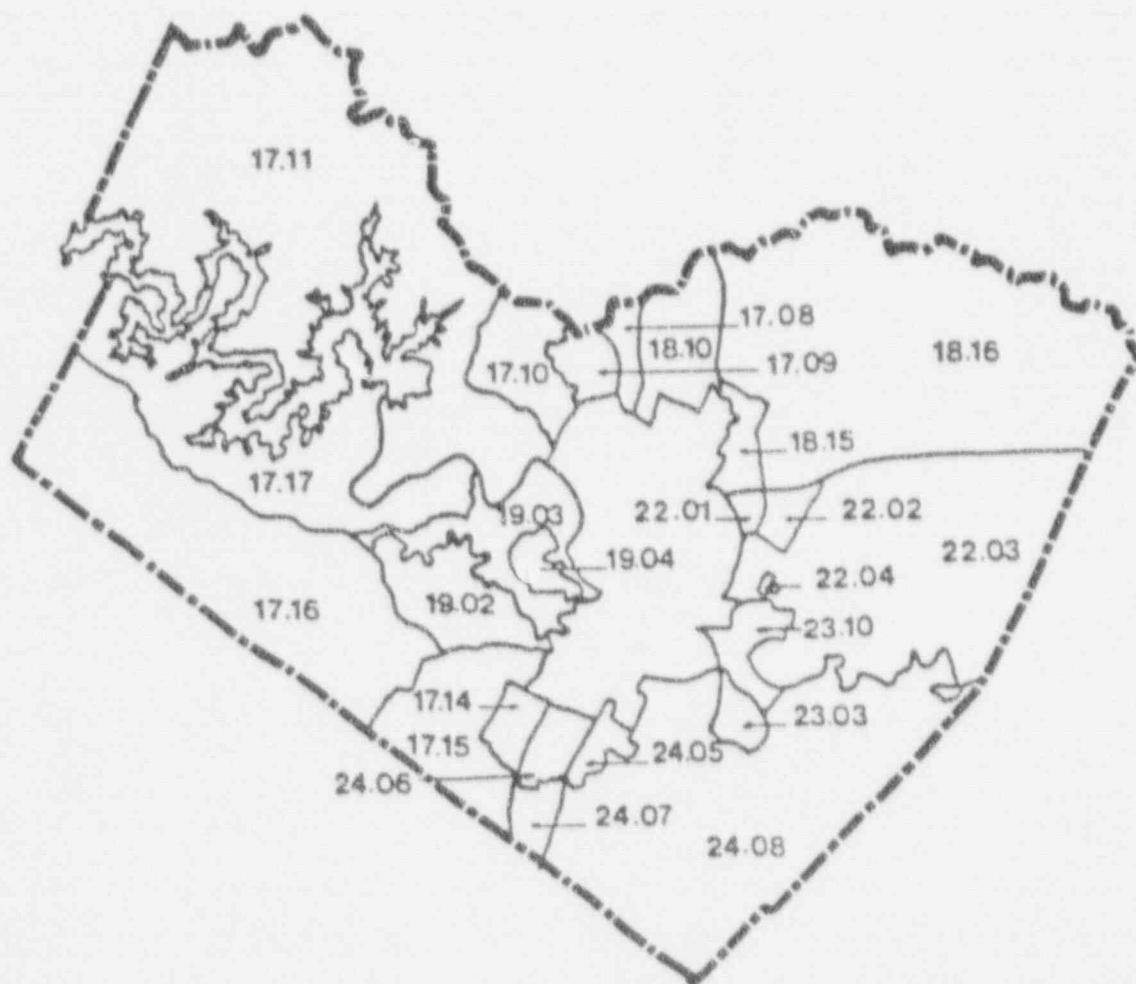
2.2 POPULATION AND EMPLOYMENT

Austin is composed primarily of governmental, business, and professional persons with their families. The city has substantial light industry but practically no heavy industry. Many of the persons in the local labor force are related to activities of the City and its role as a State Capitol or the University and its educational and research programs. A substantial population growth rate was in process prior to the time of project proposal and development. By the time of project completion a significant reduction of the local area growth rate was effective. These population trends were characteristic of the "Sunbelt" areas of the country. Population growth of Travis county between the 1970 and 1980 census was 42%. The substantial growth of Austin in particular, was the norm during the early 1980's with a significant reduction in the rate by the late 1980's. As one of the ten fastest growth areas, the Austin area growth was 35 percent between 1980-1986. By 1988 the annual rate of growth of the Austin metropolitan area was about one percent.

Population data of the Travis county region is presented in Table 2-1 with supportive data in Figure 2-5 and Figure 2-6. Estimates of the 1990 census project the 1990 Austin population to be 465,600. The 1980 population census listed the Austin city population at 345,496 and Travis county population at 419,573 with the Austin Standard Metropolitan Statistical Area population at 536,450 [3]. Three counties, Travis, Hays and Williamson, compose the Austin Standard Metropolitan Statistical Area. Population densities in Travis county range from 6.4 persons per 1000 square meters encompassing the main university campus to less than 0.2 persons per 1000 square meters in growth areas north of the research center site. Population census tracts adjacent to the site exhibit densities of 0.2 to 0.3 persons per 1000 square meters.

Table 2-1
 TRAVIS COUNTY
 1980 POPULATION DENSITY DISTRIBUTION

Census Tracts	Acres In City	Acres Out of City	Acres Total	Census Tracts	Acres In City	Acres Out of City	Acres Total	Population Density	Population
1.01	800	-	800	18.07	770	-	770	7.9	5,456
1.02	1,160	260	1,420	18.08	3,130	-	3,130	6.1	7,527
2.01	600	-	600	18.09	1,410	415	1,825	2.9	2,996
2.02	150	-	150	18.10	3,530	10,540	14,070	-	8,034
2.03	300	-	300	18.11	370	-	370	6.1	2,274
2.04	485	-	485	18.12	540	-	540	4.2	2,452
3.01	680	-	680	18.13	490	-	490	3.1	2,813
3.02	535	-	535	18.14	460	715	1,175	5.2	6,194
3.03	1,160	-	1,160	18.15	540	3,850	4,390	-	3,235
4.01	565	-	565	18.16	1,000	90,810	91,810	6.2	4,999
4.02	295	-	295	18.17	420	-	420	7.9	5,006
5.00	220	-	220	18.18	780	-	780	6.7	4,791
6.01	260	-	260	18.19	300	230	530	2.2	2,854
6.02	375	-	375	18.20	300	12,760	13,060	6.1	2,854
7.00	325	-	325	18.21	2,650	7,100	9,750	5.4	5,638
8.01	430	-	430	18.22	80	3,065	3,145	1.4	4,483
8.02	325	-	325	18.23	420	-	420	6.3	3,295
8.03	345	-	345	18.24	395	-	395	7.2	2,723
9.01	230	-	230	18.25	560	-	560	5.4	2,950
10.00	575	-	575	18.26	290	-	290	11.0	7,383
11.00	890	-	890	18.27	480	-	480	6.1	2,940
12.00	625	-	625	18.28	420	-	420	7.6	3,202
13.01	455	-	455	18.29	765	-	765	6.4	4,119
13.02	455	-	455	18.30	860	-	860	2.2	1,903
13.03	455	-	455	18.31	770	-	770	5.4	4,016
13.04	455	-	455	18.32	910	-	910	3.7	3,338
13.05	800	-	800	18.33	1,890	-	1,890	2.5	4,184
13.06	780	-	780	18.34	470	810	1,280	-	1,057
14.00	870	-	870	18.35	760	2,400	3,160	-	1,716
15.01	850	-	850	18.36	6,370	18,870	25,240	3.0	1,228
15.02	850	-	850	18.37	410	-	410	3.0	1,228
15.03	550	-	550	18.38	110	4,440	4,550	-	3,680
15.04	550	-	550	18.39	365	-	365	11.0	4,002
16.01	740	-	740	18.40	300	-	300	19.3	5,794
16.02	450	-	450	18.41	1,230	-	1,230	6.0	3,234
16.03	450	-	450	18.42	540	-	540	-	3,234
16.04	450	-	450	18.43	805	-	805	3.1	2,130
16.05	570	-	570	18.44	2,320	-	2,320	4.6	3,884
16.06	140	-	140	18.45	285	3,870	4,155	7.3	4,786
17.01	520	3,410	3,930	18.46	640	-	640	6.1	3,884
17.02	490	784	1,274	18.47	930	-	930	7.3	5,353
17.03	490	15	505	18.48	430	-	430	4.1	2,582
17.04	490	15	505	18.49	430	-	430	4.1	2,582
17.05	490	15	505	18.50	430	-	430	4.1	2,582
17.06	490	15	505	18.51	430	-	430	4.1	2,582
17.07	490	15	505	18.52	430	-	430	4.1	2,582
17.08	490	15	505	18.53	430	-	430	4.1	2,582
17.09	490	15	505	18.54	430	-	430	4.1	2,582
17.10	490	15	505	18.55	430	-	430	4.1	2,582
17.11	490	15	505	18.56	430	-	430	4.1	2,582
17.12	490	15	505	18.57	430	-	430	4.1	2,582
17.13	490	15	505	18.58	430	-	430	4.1	2,582
17.14	490	15	505	18.59	430	-	430	4.1	2,582
17.15	490	15	505	18.60	430	-	430	4.1	2,582
17.16	490	15	505	18.61	430	-	430	4.1	2,582
17.17	490	15	505	18.62	430	-	430	4.1	2,582
17.18	490	15	505	18.63	430	-	430	4.1	2,582
17.19	490	15	505	18.64	430	-	430	4.1	2,582
17.20	490	15	505	18.65	430	-	430	4.1	2,582
17.21	490	15	505	18.66	430	-	430	4.1	2,582
17.22	490	15	505	18.67	430	-	430	4.1	2,582
17.23	490	15	505	18.68	430	-	430	4.1	2,582
17.24	490	15	505	18.69	430	-	430	4.1	2,582
17.25	490	15	505	18.70	430	-	430	4.1	2,582
17.26	490	15	505	18.71	430	-	430	4.1	2,582
17.27	490	15	505	18.72	430	-	430	4.1	2,582
17.28	490	15	505	18.73	430	-	430	4.1	2,582
17.29	490	15	505	18.74	430	-	430	4.1	2,582
17.30	490	15	505	18.75	430	-	430	4.1	2,582
17.31	490	15	505	18.76	430	-	430	4.1	2,582
17.32	490	15	505	18.77	430	-	430	4.1	2,582
17.33	490	15	505	18.78	430	-	430	4.1	2,582
17.34	490	15	505	18.79	430	-	430	4.1	2,582
17.35	490	15	505	18.80	430	-	430	4.1	2,582
17.36	490	15	505	18.81	430	-	430	4.1	2,582
17.37	490	15	505	18.82	430	-	430	4.1	2,582
17.38	490	15	505	18.83	430	-	430	4.1	2,582
17.39	490	15	505	18.84	430	-	430	4.1	2,582
17.40	490	15	505	18.85	430	-	430	4.1	2,582
17.41	490	15	505	18.86	430	-	430	4.1	2,582
17.42	490	15	505	18.87	430	-	430	4.1	2,582
17.43	490	15	505	18.88	430	-	430	4.1	2,582
17.44	490	15	505	18.89	430	-	430	4.1	2,582
17.45	490	15	505	18.90	430	-	430	4.1	2,582
17.46	490	15	505	18.91	430	-	430	4.1	2,582
17.47	490	15	505	18.92	430	-	430	4.1	2,582
17.48	490	15	505	18.93	430	-	430	4.1	2,582
17.49	490	15	505	18.94	430	-	430	4.1	2,582
17.50	490	15	505	18.95	430	-	430	4.1	2,582
17.51	490	15	505	18.96	430	-	430	4.1	2,582
17.52	490	15	505	18.97	430	-	430	4.1	2,582
17.53	490	15	505	18.98	430	-	430	4.1	2,582
17.54	490	15	505	18.99	430	-	430	4.1	2,582
17.55	490	15	505	19.00	430	-	430	4.1	2,582
17.56	490	15	505	19.01	430	-	430	4.1	2,582
17.57	490	15	505	19.02	430	-	430	4.1	2,582
17.58	490	15	505	19.03	430	-	430	4.1	2,582
17.59	490	15	505	19.04	430	-	430	4.1	2,582
17.60	490	15	505	19.05	430	-	430	4.1	2,582
17.61	490	15	505	19.06	430	-	430	4.1	2,582
17.62	490	15	505	19.07	430	-	430	4.1	2,582
17.63	490	15	505	19.08	430	-	430	4.1	2,582
17.64	490	15	505	19.09	430	-	430	4.1	2,582
17.65	490	15	505	19.10	430	-	430	4.1	2,582
17.66	490	15	505	19.11	430	-	430	4.1	2,582
17.67	490	15	505	19.12	430	-	430	4.1	2,582
17.68	490	15	505	19.13	430	-	430	4.1	2,582
17.69	490	15	505	19.14	430	-	430	4.1	2,582
17.70	490	15	505	19.15	430	-	430	4.1	2,582
17.71	490	15	505	19.16	430	-	430	4.1	2,582
17.72	490	15	505	19.17	430	-	430	4.1	2,582
17.73	490	15	505	19.18	430	-	430	4.1	2,582
17.74	490	15	505	19.19	430	-	430	4.1	2,582
17.75	490	15	505	19.20	430	-	430	4.1	2,582
17.76	490	15	505	19.21	430	-	430	4.1	2,582
17.77	490	15	505	19.22	430	-	430	4.1	2,582
17.78	490	15	505	19.23	430	-	430	4.1	2,582
17.79	490	15	505	19.24	430	-	430	4.1	2,582
17.80	490	15	505	19.25	430	-	430	4.1	2,582
17.81	490	15	505	19.26	430	-	430	4.1	2,582
17.82	490	15	505	19.27	430	-	430	4.1	2,582
17.83	490	15	505	19.28	430	-	430	4.1	2,582
17.84	490	15	505	19.29	430	-	430	4.1	2,582
17.85	490	15	505	19.30	430	-	430	4.1	2,582
17.86	490	15	505	19.31	430	-	430	4.1	2,582
17.87	490	15	505	19.32	430	-	430	4.1	2,582
17.88	490	15	505	19.33	430	-	430	4.1	2,582
17.89	490	15	505	19.34	430	-	430	4.1	2,582
17.90	490	15	505	19.35	430	-	430	4.1	2,582
17.91	490	15	505	19.36	430	-	430	4.1	2,582
17.92	490	15	505	19.37	430	-	430	4.1	2,582
17.93	490	15	505	19.38	430	-	430	4.1	2,582
17.94	490	15	505	19.39	430	-	430	4.1	2,582
17.95	490	15	505	19.40	430	-	430	4.1	2,582
17.96	490	15	505	19.41	430	-	430	4.1	2,582
17.97	490	15	505	19.42	430	-	430	4.1	2,582
17.98	490	15	505	19.43	430	-	430	4.1	2,582
17.99	490	15	505	19.44	430	-	430	4.1	2,5



TRAVIS COUNTY
1980 CENSUS TRACT BOUNDARIES

Figure 2-5



1980 CENSUS TRACT BOUNDARIES

CITY OF AUSTIN CENSUS TRACT BOUNDARIES

Figure 2-6

Approximately 800 persons were involved in activities on the east tract of the Balcones Research Center in 1980 with projected activities at the site to add an estimated 900 to 1000 persons by the late 1980's. On the west tract the Microelectronics and Computer Technology Corporation setup operations with approximately 500 persons in 1985. Facilities north of the Research Center operated by International Business Machines Corporation employed an estimated 6500 persons in 1985 with approximately 500 additional persons by 1990.

Research activities at the Balcones Research Center are diverse, consisting of many different research organizations of the university science and engineering colleges. As of 1980, several research programs were already situated at the research center site of which the Applied Research Laboratories, Center for Research in Water Resources and Civil Engineering Structures Laboratory are examples. Between 1980 and 1985, additional facilities and 3 major additional programs, the Center for Energy Studies, Center for Electromechanics and Bureau of Economic Geology were added to the research center functions. Since 1985 program development, besides the Nuclear Engineering Teaching Laboratory (NETL), included a computer facility, the Center for High Performance Computing, and an engineering facility, for microelectronics and manufacturing research. Administrative functions of the Bureau of Economic Geology are situated adjacent to the NETL site. Expansion of other activities near the NETL site is possible in the future. However, as of 1988 there were no official plans for the adjacent site areas.

2.3 CLIMATOLOGY

Austin, capital of Texas, is located on the Colorado River where the stream crosses the Balcones Escarpment separating the Texas Hill Country from the Blackland Prairies to the east. Elevations within the City vary from 120 meters to 275 meters above sea level. Native trees include cedar, oak, walnut, mesquite, and pecan.

The climate [4] of Austin is humid subtropical with hot summers. Winters are mild, with below freezing temperatures occurring on an average of less than twenty-five days each year. Rather strong northerly winds, accompanied by sharp drops in temperature, occasionally occur during the winter months in connection with cold fronts, but cold periods are usually of short duration, rarely lasting more than two days. Daytime temperatures in summer are hot, but summer nights are usually pleasant with average daily minima in the low seventies.

Precipitation is fairly evenly distributed throughout the year, with heaviest amounts occurring in late spring. A secondary rainfall peak occurs in September. Precipitation from April through September usually results from thundershowers, with fairly large amounts falling within short periods of time. While thunderstorms and heavy rains have occurred in all months of the year, most of the winter precipitation occurs as light rain. Snow is insignificant as a source of moisture, and usually melts as rapidly as it falls. The City may experience several seasons in succession with no measurable rain fall.

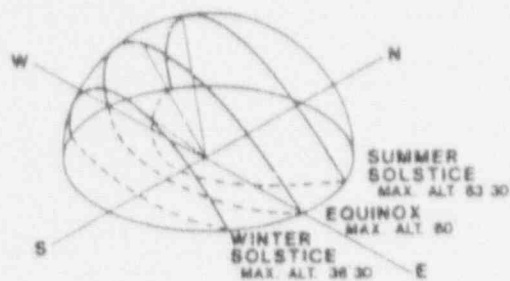
CLIMATOLOGICAL DATA

TEMPERATURE (DEGREES FAHRENHEIT)



TEMPERATURE EXTREMES: HIGH 109
LOW -2

SOLAR PATH DIAGRAM



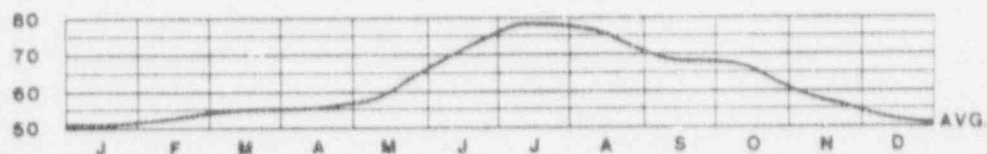
HUMIDITY (AVG. % RELATIVE)

MIDNIGHT: 76
6 A.M.: 84
NOON: 56
6 P.M.: 53

PREVAILING WINDS

SOUTHERLY
AVERAGING 9 MPH

SUNSHINE (% POSSIBLE)



AVERAGE ANNUAL: 61

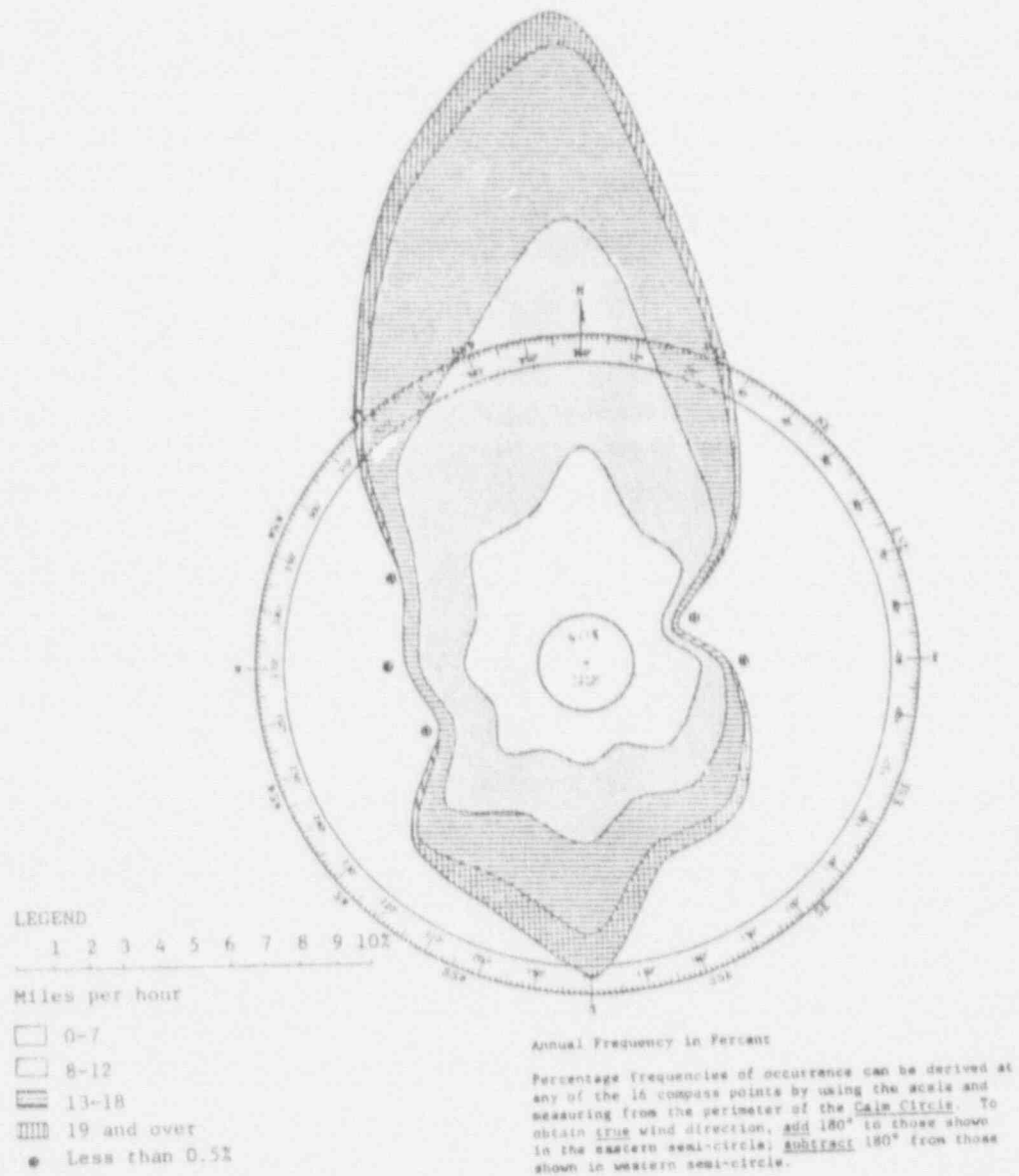
PRECIPITATION (INCHES)



AVERAGE ANNUAL: 38.1
24 HOUR EXTREME: 19.03

AUSTIN CLIMATOLOGY DATA

Figure 2-7



AUSTIN WIND ROSE DATA

Figure 2-8

Prevailing winds are southerly throughout the year. Northerly winds accompanying the colder air masses in winter soon shift to southerly as these air masses move out over the Gulf of Mexico.

Climatology data is summarized in Figure 2-7. Typical Austin wind data are presented in Figure 2-8 [5].

The average length of the warm season (freeze-free period) is 270 days. Based on data from 1943-1961, the average date of the last occurrence of freezing or below has occurred as late as April 13 (1940), and as early as October 26 (1924). Meteorological data is tabulated in Table 2-2 and Table 2-3 [4].

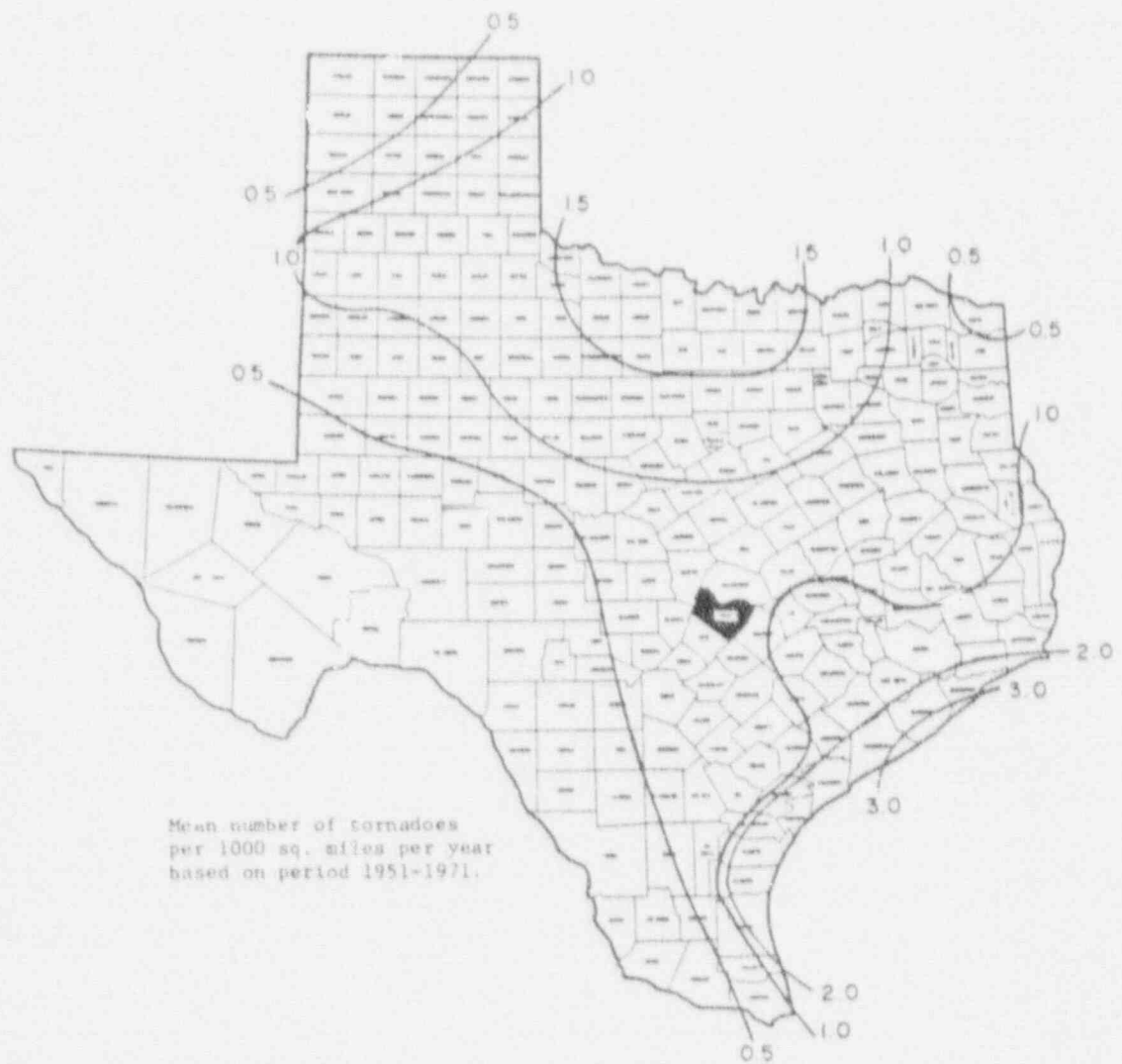
Destructive winds and damaging hailstorms are infrequent. On rare occasions, dissipating tropical storms effect the City with strong winds and heavy rains. The frequency of tornado type activity is illustrated in Figure 2-9 [6]. Recent tropic storm paths and local witing of tornadoes and funnel clouds are presented in Figure 2-10 and Figure 2-11 [7,8].

2.4 GEOLOGY

The northwestern half of Travis county is part of the physiographic province of Texas known as the Edwards Plateau. In Travis County, this is a highly dissected plateau with wooden hills rising in some places more than 150 meters above the drainage pathways. In marked contrast, the southeastern half of the county is gently rolling prairie land which is part of the physiographic province known as the Gulf Coastal Plain. These provinces are separated by the scarp of the Balcones fault zone, which rises 30 to 90 meters above the Coastal Plain. The scarp, however, is not a vertical cliff; it is an indented line of sloping hills leading up from the lower plain to the plateau summit.

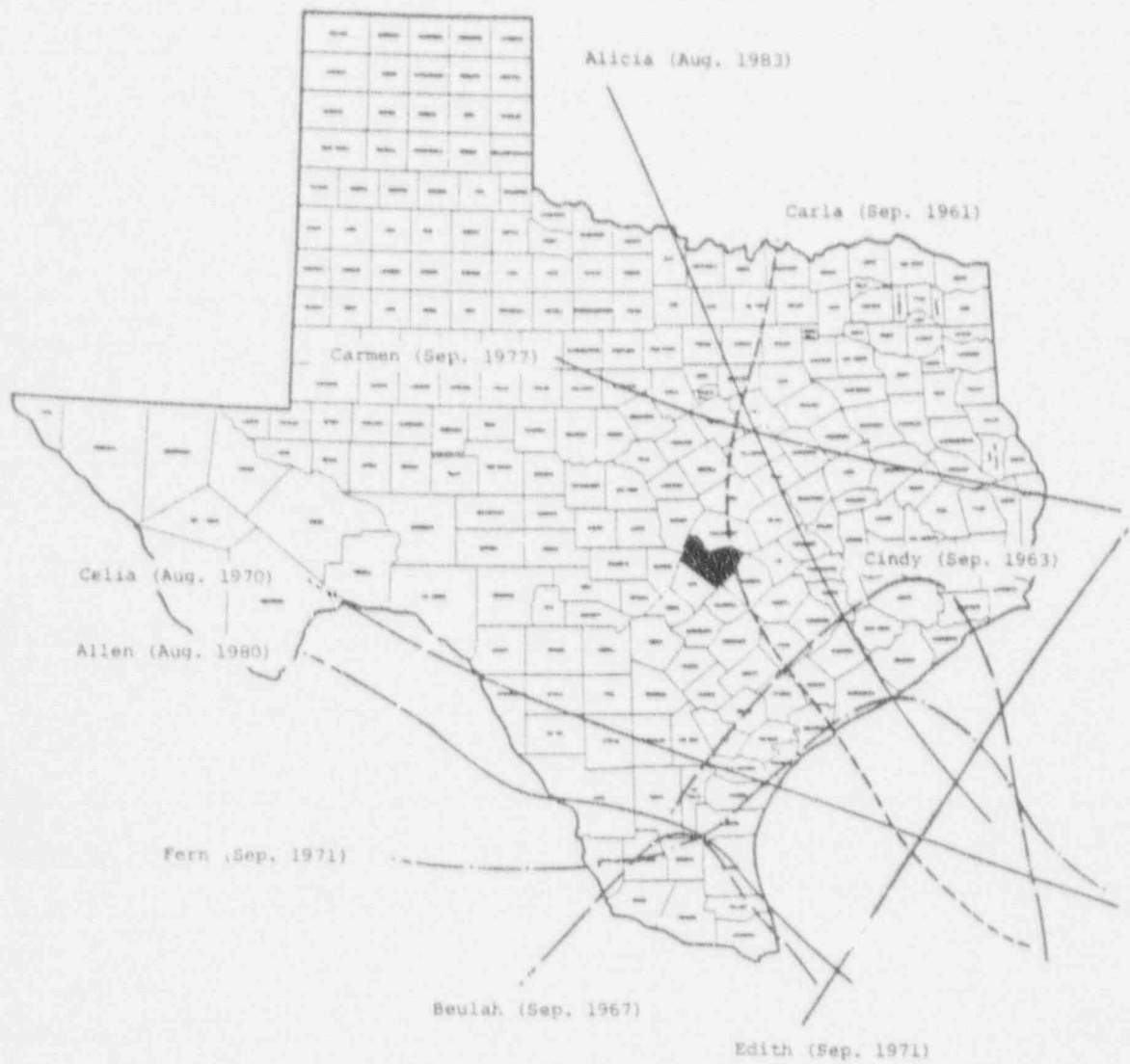
The rocks that outcrop in Travis County are primarily of sedimentary origin and of Mesozoic (Cretaceous) and Cenozoic age. They consist largely of limestone, clay, and sand strata which dip southeastward toward the Gulf of Mexico at an angle slightly greater than the slope of the land surface. Therefore, in going from southeast to northwest the outcrops of progressively older formations are encountered, and the rocks lowest in the geologic column have the highest topographic exposure.

At the reactor facility site on the east tract, the geology is of the Austin Group defined as chalk, marly limestone, and limestone with light gray, soft to hard, thin to thick bed, and massive to slightly nodular character. On the west tract, the geology changes to the Edwards Formation of limestone and dolomite with light gray to tan, hard to soft, thin to thick bed, and fine to medium grain character. The separate formations are, respectively, the up and down side of a segment of the Mount Bonnell Fault that passes approximately along the boundary of the east and west Balcones Research Center tracts. Distance to the fault is about 500 meters from the reactor facility site.



TEXAS TORNADO FREQUENCIES

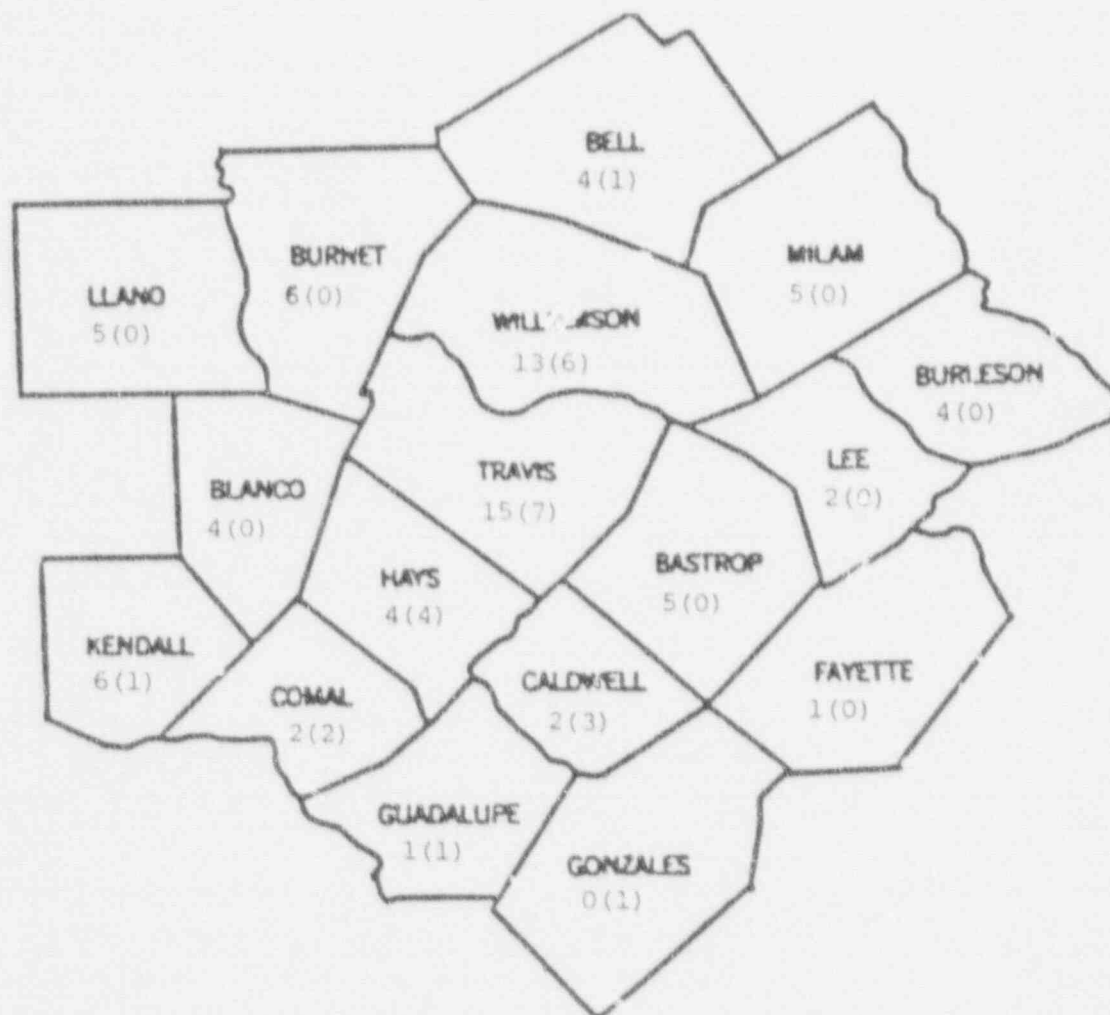
Figure 2-9



Hurricanes 1960-1983

TEXAS HURRICANE PATHS

Figure 2-10



Tornado and funnel cloud occurrences in a 50 mile radius of Austin for the years 1975-1983; # Tornadoes (#Funnel Clouds).

LOCAL FUNNEL CLOUD SITINGS

Figure 2-11

The Balcones fault zone, which extends from Williamson County to Uvalde County, extends the full length of Travis County on a line passing through Manchaca, Austin, and McNeil. Here the orderly sequence of formations is replaced by an outcrop pattern controlled by the faults, most of which are normal faults with the down-thrown side toward the coast. Most of the movement of the Balcones Fault zone occurred during the Miocene period. Since no movement has been detected during modern times, this fault is no longer considered active [9]. Location of the Balcones Fault zone and formations in the Austin area are depicted in Figure 2-12.

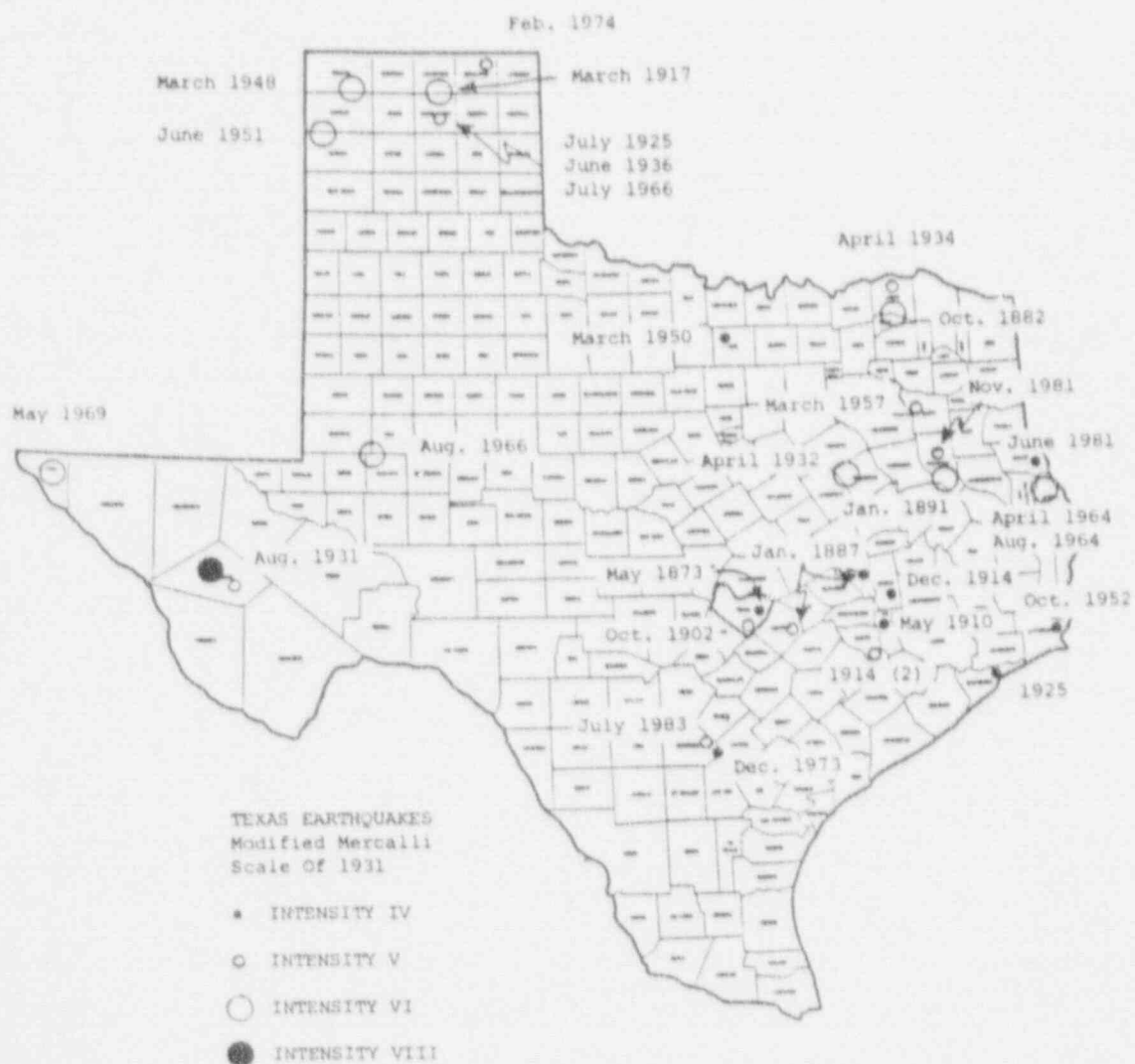
2.5 SEISMOLOGY

Thirty three earthquakes of intensity IV or greater have had epicenters in Texas since 1873 [10,11]. The earthquakes were characterized using the Modified Mercalli Scale of 1931. The scale has a range of I thru XII, on which an intensity of I is not felt, an intensity of III is a vibration similar to that due to the passing of lightly loaded trucks, and intensity of VII is noticed by all as shaking trees, waves on ponds, and quivering suspended objects but causes negligible damage to buildings of good design and construction, and an intensity of XII results in practically all works of construction being severely damaged or destroyed. The strongest earthquake, a maximum intensity of VIII, was in western Texas in 1931 and was felt over 1,165,000 square kilometers. Figure 2-13 shows the locations and intensities of all earthquakes in Texas since 1873. Of these, some are known to have been felt in Austin, but no damage has ever occurred to local buildings.

2.6 HYDROLOGY

Almost the entire county is drained by the Colorado River and its tributaries. Lake Travis, which is formed by the Mansfield Dam on the Colorado River, is part of the power, flood-control, water conservation, and recreation project of the Lower Colorado River Authority. Other lakes are also operated by the Authority, such as Town Lake and Lake Austin, and are created by Longhorn and Tom Miller dams, respectively. Low level alluvial deposits of the river are commonly saturated with water at relatively shallow depths. Recharge is primarily from the river and local surface contaminations are easily transmitted to this shallow water table.

Ground water from subsurface formation is found in basal Cretaceous sands referred to as the "Trinity" sands. Elevations of the Trinity aquifer range from depths commonly less than 300 meters east of the Balcones Fault Zone to greater than 450 meters to the west of the zone. East of the Mount Bonnell Fault, dolomite and dolomite limestones provide a source of ground water at shallower depths. Access to the Edwards aquifer ranges from 30 meters to 300 meters with natural springs occurring in areas near the Colorado River. Minor aquifers associated

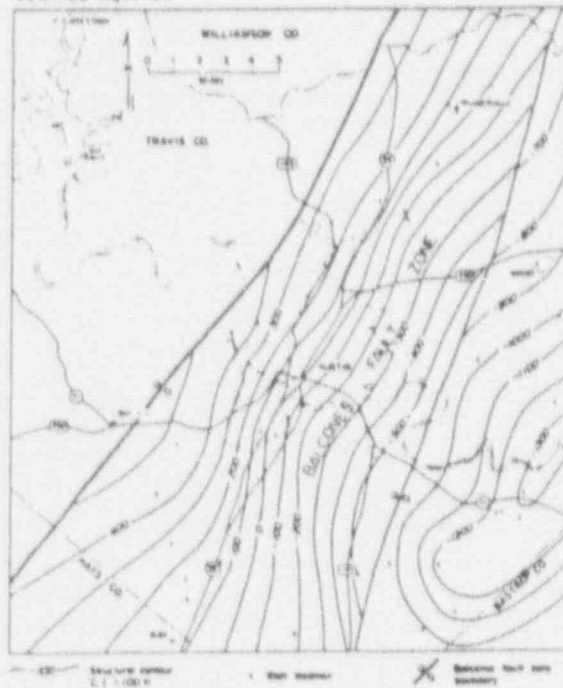


Texas Earthquakes, 1873-1983

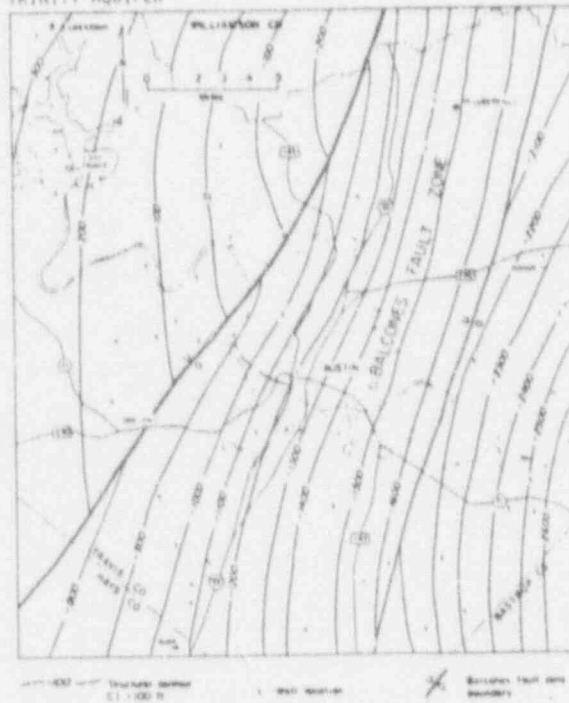
TEXAS EARTHQUAKE DATA

Figure 2-13

EDWARDS AQUIFER



TRINITY AQUIFER



LOCAL WATER AQUIFERS

Figure 2-14

with the Glenn Rose Formation supplies small quantities of water west of the Balcones Fault Zone. Water bearing areas in the formation are at varying depths and laterally discontinuous. On the Balcones Research Center east tract, wells drilled for environmental monitoring have produced ground water at depths of less than 15 meters. Figure 2-16 shows the location of the ground water aquifers.

Water supply for the research center and wastewater treatment is provided by the City of Austin. Although wells into the aquifers provide substantial water the city supply is filtered river water. Other area municipalities and organizations utilize aquifer water. Control of private wells is the function of county and state Health Departments. Gross beta radioactivity of city water has been measured and is reported in Table 2-4.

Table 2-4
GROUND WATER ACTIVITY
(gross beta)

Travis County	$6 \times 10^{-9} \mu\text{Ci/ml}$
Balcones Research Center	$8 \times 10^{-9} \mu\text{Ci/ml}$

2.7 HISTORICAL

Relocation of the UT TRIGA reactor and related facilities to the Balcones Research Center site is to help accommodate growth of programs both at the University main campus and at the Research Center site. The facility location at the Research Center is to be in the north-east corner of the research center site.

The original research center site area was operated as a magnesium manufacturing plant by the Federal government in the 1940's. Subsequent arrangements and acquisition by the University would determine activities of the site throughout the 1950's, 1960's and 1970's. Activities at the site were not fully developed prior to the 1980's. University functions or research activities were moved to the site when required accommodations were not available on the main campus. A few functions of the University at the site had resulted in the construction of major facilities suitable for long term use. Other activities at the site have utilized existing structures or other buildings not suited for long term use.

A major program [12] was established in the 1980's to develop the Balcones Research Center site activities. As part of the first phase of development, several major research programs associated with energy and engineering were moved to facilities constructed at the site. Features of the site, before the development activities by the University and after initial development in the 1980's, are illustrated in Figures 2-15 and 2-16.

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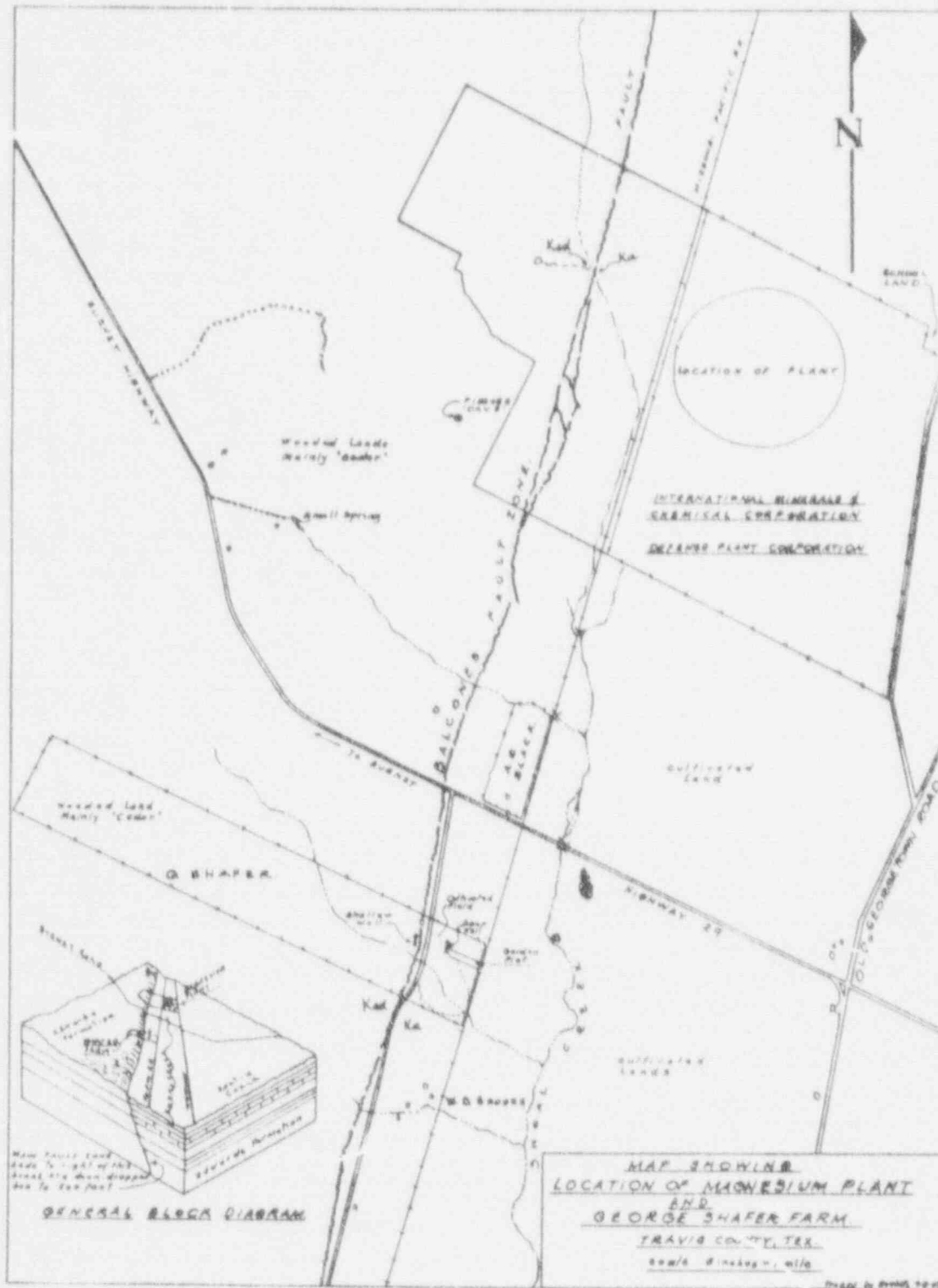
Several activities at the Research Center prior to 1980 had been associated with radioactive materials. These activities ranged from the burial of low level radioactive waste materials such as tritium and carbon-14 in the northwest corner of the site, to water transport studies performed in 30 meter diameter surface tanks. Isotopes of cesium-137, cesium-134, and cobalt-60 were present in sludge samples of one of the tanks, and are reported in Table 2-5. Gross beta activity in the samples of the west tank measured 22 microcuries per milliliter (1979).

Table 2-5

TANK SLUDGE SAMPLES

Total (1979)	22 $\mu\text{Ci/ml}$ (gross β)
Cs ¹³⁷	$1.3 \times 10^{-3} \mu\text{Ci/ml}$
Cs ¹³⁴	$2.5 \times 10^{-4} \mu\text{Ci/ml}$
Co ⁶⁰	$5.7 \times 10^{-5} \mu\text{Ci/ml}$

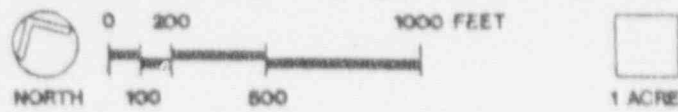
Radioactive waste and other materials at the Research Center site are part of the University broad license for radioactive materials which is managed by the University Safety Office and issued by the Texas Department of Health.



RESEARCH CENTER AREA 1960

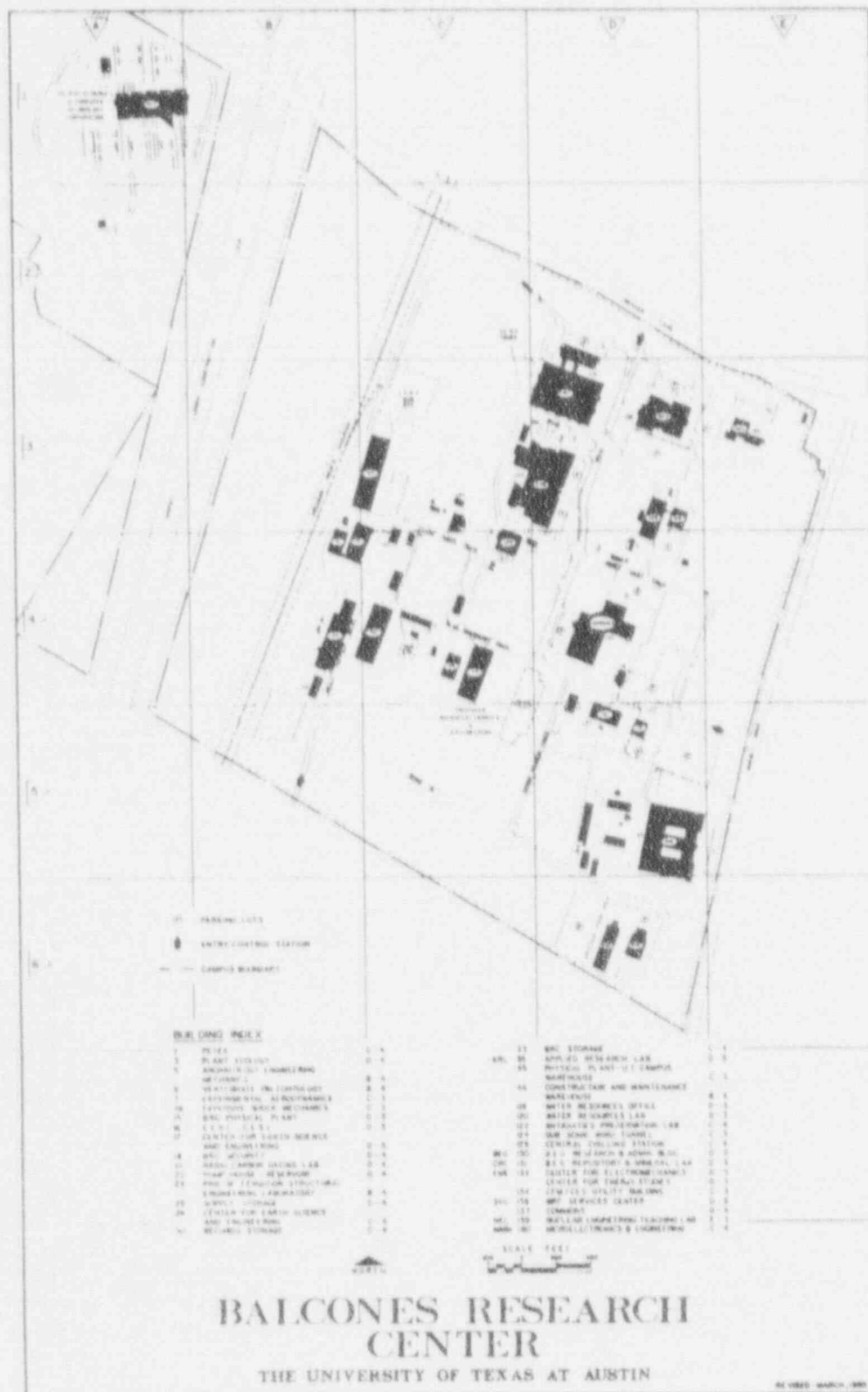
Figure 2-15a

EXISTING CONDITIONS



BALCONES RESEARCH CENTER 1960

Figure 2-15b



BALCONES RESEARCH CENTER 1990

Figure 2-16

Chapter 2 References

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11. Steven M. Caulson, "Investigations of Recent and Historical Seismicity in East Texas", Masters Thesis May 1984, University of Texas.
12. "Balcones Research Center Project Analysis", Volume I, The University of Texas, 1981.

Chapter 3

FACILITY DESIGN
STRUCTURES, SYSTEMS AND COMPONENTS

The facility in which the TRIGA reactor is to be located is a building with several special design features. Most of the building design specifications are for criteria that are independent of the reactor systems. However, several design features and specifications are to assure safe facility operation and effective utilization of facility equipment.

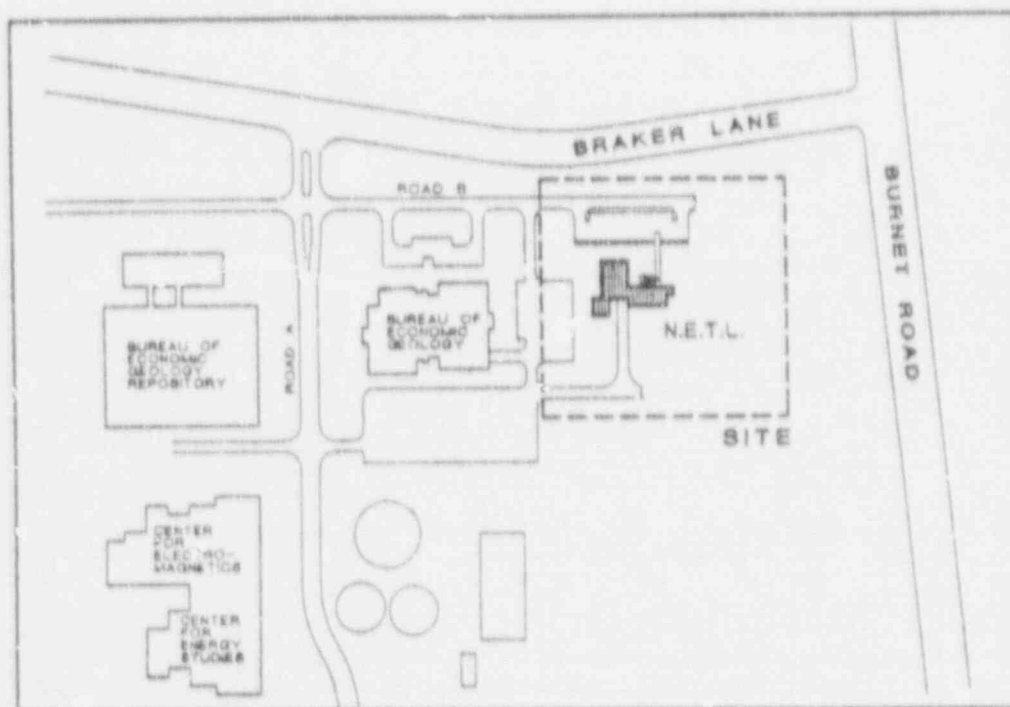
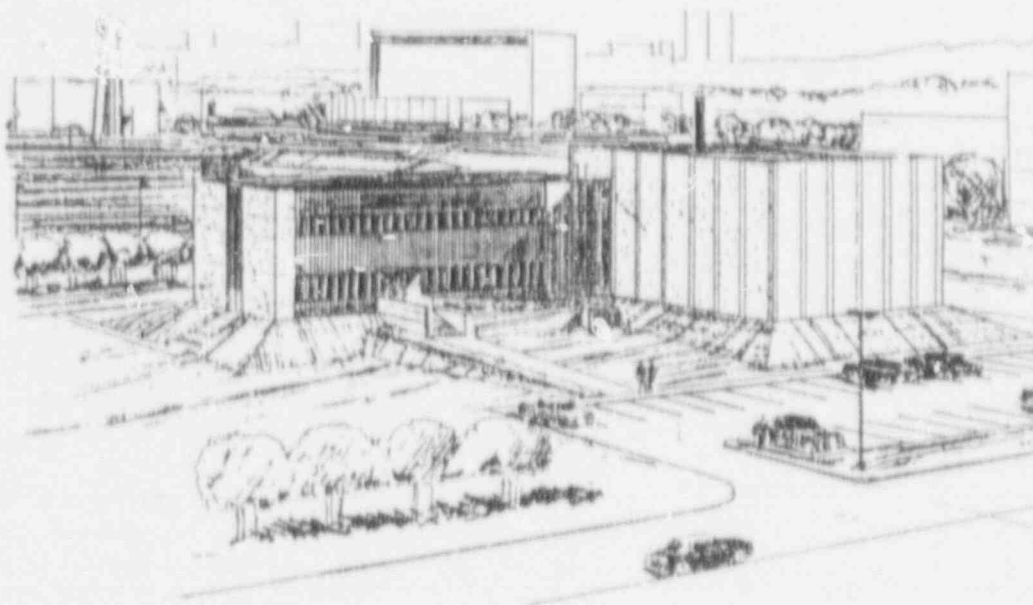
Building areas include office space and laboratory space. Space for administrative functions, library or conference rooms, maintenance shops and utility equipment will support all facility activities. The reactor, which is the primary piece of facility equipment, will establish the fundamental requirements for two basic areas of the building. One area is the reactor bay that contains the primary reactor structures and experiment facilities. The other area will contain the operation control center with space for the control console, training activities, and records maintenance.

Building orientation on the site plan is shown in Figure 3-1. Total gross floor space of the facility is about 1950 square meters (~21,000 sq. ft.).

Several design requirements to protect against the release of radioactive materials from reactor operation are set by the compliance requirements of the appropriate standards or regulations. Other potential releases that might occur from specific experiments or special materials or conditions are also subject to these standards and regulations. Facility design includes features for control of airborne release from reactor operation and releases of liquid waste or solid waste from other facility activities.

3.1 GENERAL CONDITIONS

The design of a structure to contain the TRIGA reactor facility depends at least partly on the protection requirements for the fuel elements. Among these requirements are the control of personnel access to special nuclear material and the control of personnel exposure to radioactive fuel material. Design of the fuel elements as a physical containment system is the basic design assumption. This fuel design controls the release of radioactive material during routine reactor operation and potential accident conditions. Facility design will control the exposure to the radiation levels that the fuel will create during operation. Other facility design requirements will control the release of operation effluents such as radioactive gases from either normal operation or accident conditions. Rupture of a fuel element will be the design basis accident. Design of the reactor bay as an air confinement system will protect the operation personnel and the general public against the operation hazards of the reactor. These hazards are release of air activation products during normal operation and the



NETL SITE PLAN FOR
BALCONES RESEARCH CENTER

Figure 3-1

release of fuel fission products during accident conditions. Release criterion are based on Title 10 Chapter 20 of the U.S. Code of Federal Regulations [1].

Engineering Design, specifications, and construction for the building are set by the State of Texas Uniform General Conditions and The University of Texas at Austin Supplementing Conditions [2,3,4,5]. Provisions of the Uniform Building Code [5] and other national codes for mechanical, electrical, and plumbing are applicable to this project. Equipment requirements will apply Underwriter's Laboratories standards or labels, when appropriate, to a piece, type, class, or group of equipment. Other specifications will conform to the standards of the American Society for Testing and Materials (ASTM). Provisions of the Life Safety Code are applicable. One code of importance, the National Fire Protection Code, will determine requirements that relate to fire safety for significant facility operation hazards.

Exploration logs for the site consist of 4 borings of depths to 30 feet (~9 meters). Approximate ground surface elevation is 791 feet (241 meters). At an elevation of about 787 feet (240 meters), each log records a consistent subsurface structure typical of the region. Installation of a piezometer in one bore hole allows a check of ground water. Evidence and experience indicate a periodic presence of some ground water, although seasonal variation of the level is probable. Water surface, depth at the time of piezometer installation was 29.1 feet (8.87 meters) after 24 hours and 9.3 feet (2.83 meters) after 3 months.

The building site is located on a rock subsurface of limestone. Soil tests of the subsurface set the load capacity at 1690 kg/m^2 (2.4 psi). Concrete piers and footings will provide building foundations.

Seismic design specifications will meet the requirements of the Uniform Building Code for zone 0. The zone designation is a reference that determines the appropriate earthquake activities in the continental United States. These specifications require no special provisions beyond those of standard building load requirements. Normal building loads from gravity and wind forces will exceed the seismic accelerations for buildings in zone 0. A building of good construction should withstand an earthquake acceleration of about .75 g. Ground accelerations that exceed this would be rare events in a region in which earthquakes are already infrequent.

Wind load designs will meet requirements of the Uniform Building Code for 70 mph (31.3 m/sec) winds. Normal wind and storm conditions will be within these design factors. Those specifications include factors for gusts in excess of the wind load criteria. Hurricanes are not likely to be a direct threat because of the natural dissipation of energy on land. However, tornados are a concern with their extreme wind velocities. Tornado type activity is roughly one event per year per 1000 square miles (2590 sq. kilometers) in the general site area. This activity represents a frequency of one per 2.5×10^5 yr for an area of a square with sides of 333 feet (31 meters) typical of the building site.

Design for water runoff in the project vicinity will provide for dispersal of water from local rainfall rates that are frequently sporadic but sometimes torrential. Drainage provisions for the building roof, site landscape, access roadways and subsurface will control local runoff. Local flood control will include gravity flow drainage and collection sumps with dual operation pumps. Roof drainage and site runoff will be by gravity flow. Separate sumps with pumps will control subsurface drainage at the building perimeter and beneath the reactor shield foundation.

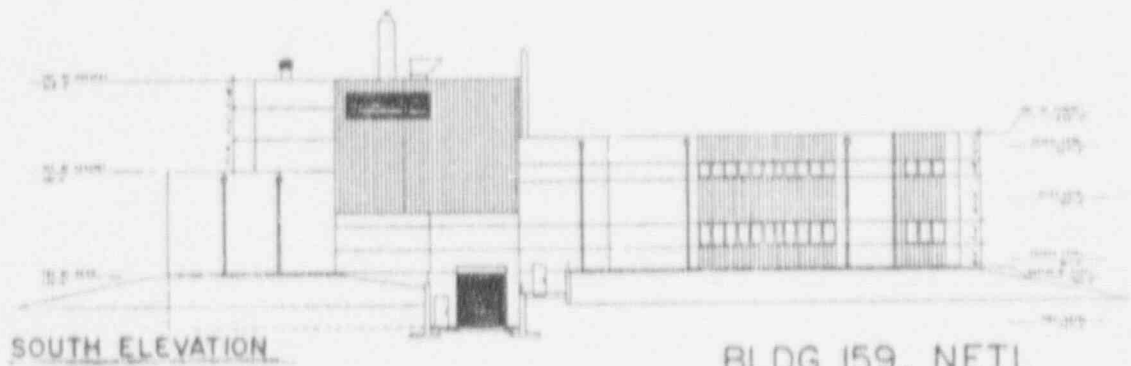
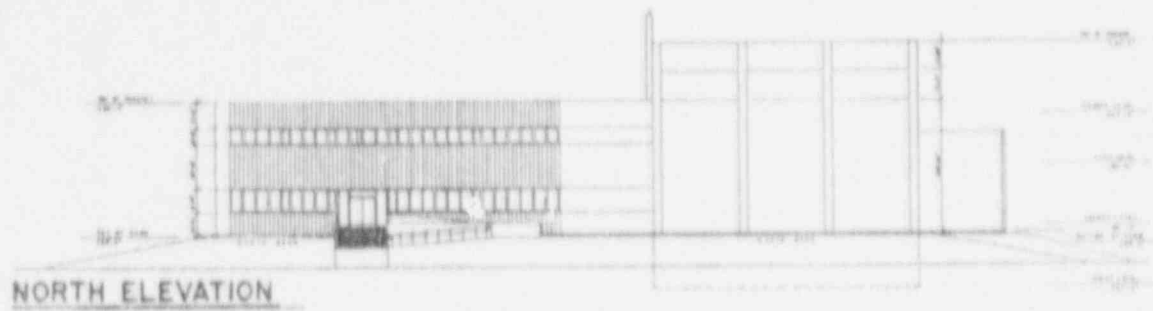
Gentle slope characteristics in the immediate site vicinity provide an ample gradient of about 3 feet (1 meter) for surface water runoff. Mean elevation at the local site is 791 feet (241 meters). Data from the National Flood Insurance Program indicates that no portion of the research center site is within the 100 or 500 year flood zone. Thus, the only flooding likely will be as a result of local runoff conditions.

3.2 ARCHITECTURAL AND STRUCTURAL ENGINEERING

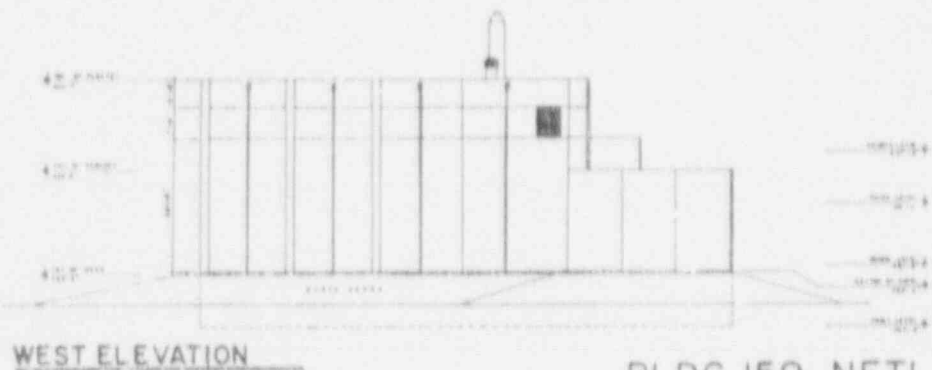
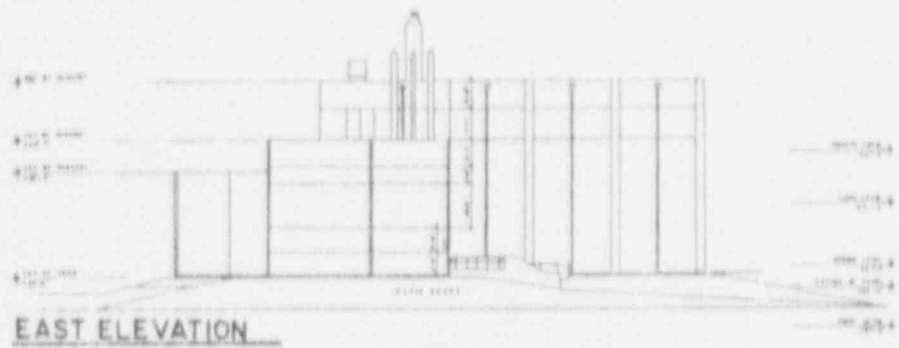
Architectural design of the building will develop two separate functional sections, one the reactor bay wing and the other an academic and laboratory wing. Structural design of the building sections is of concrete columns and beams with steel reinforcement. The first level of the reactor bay wing is 7 feet (2.1 meters) below the mean grade, while the academic wing entry level is 7 feet (2.1 meters) above the mean grade. Figures 3.2 and 3-3 illustrate the basic building design and arrangement.

The reactor bay wing will consist of three basic parts. One is the reactor bay with a floor to roof level of 56.5 feet (17.2 meters). The second is a four level section adjacent to the reactor bay. Third is the radiation experiment room with 4.25 feet (1.3 meters) thick shield walls.

The reactor bay wing construction consists of several types of concrete construction. The floor is a slab and beam design of reinforced concrete on compacted fill material. All building columns and first level walls are concrete, cast in place with steel reinforcement. Exterior walls of the reactor bay are concrete and steel construction with tilt panels and attachment columns. The combination of panels and columns set on top of the first level structure forms an integral unit by placement of the panels, then placement of the columns. Adjacent to the reactor bay, structural concrete and steel columns support slab and beam floors. Interior walls are primarily concrete blocks with a few plaster board type walls. Both concrete and metal panels complete the exterior construction of the reactor bay wing. Roof structure is a steel joist system with metal deck, concrete slab, and built-up composition roof that includes fire barrier and thermal insulation. A room of four walls and a roof of standard density concrete 4.25 feet thick forms a radiation shield room to complete the reactor bay wing. The room is cast in place with key joints between concrete placements. Tilt panels and composition roof finish the structure. All doors are of hollow metal construction.



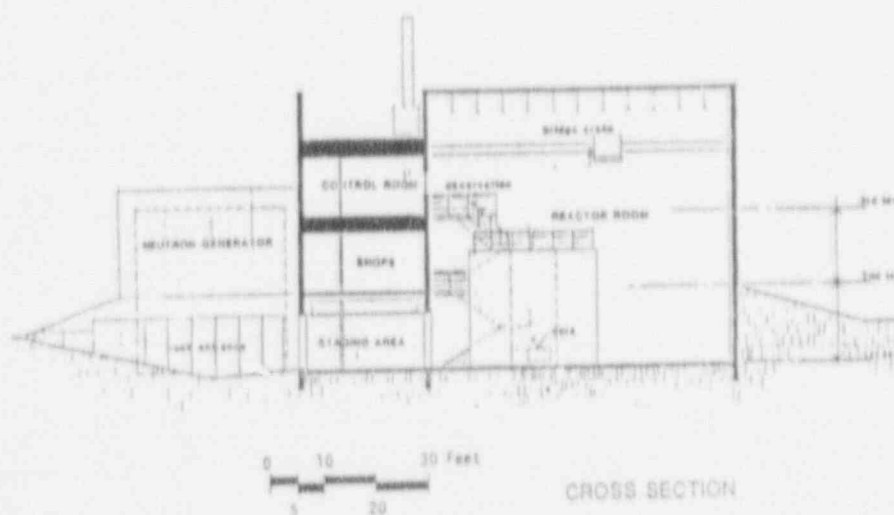
BLDG 159, NETL



BLDG 159, NETL

ELEVATION PLANS

Figure 3-2



BUILDING SECTION PLANS

Figure 3-3

Two floor levels will comprise the academic and laboratory wing. The entry floor level (second level) is an administrative and office section. Laboratories will be on the next level (third level). Construction of this wing is reinforced concrete pier and columns with poured beam and slab floors and roof. Exterior walls will consist of concrete tilt panel, metal siding and window units. Interior walls are metal stud frames with gypsum board panels. Doors are solid core wood. Entry way area and door is glass and metal frame. Stairwells at each end of the building wing will provide access to each building level.

Standards and specifications of mechanical, electrical, and other systems will be typical of university construction projects. In some cases, facility requirements may also supplement the standard requirements.

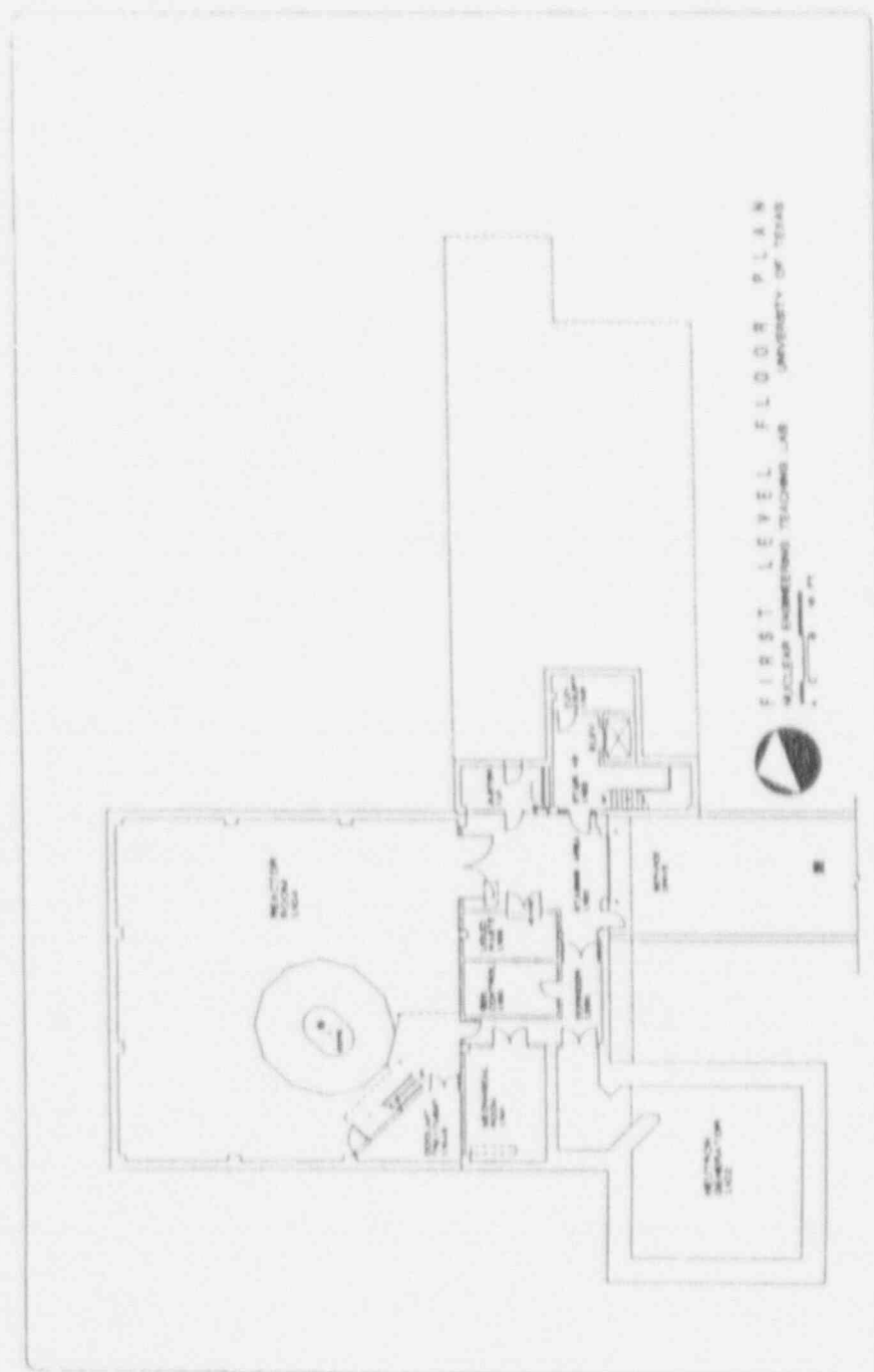
3.3 SPACE ALLOCATION

Net assignable space for administration, laboratory, and supplemental areas is about 50% of the total floor space. Figures 3-4 through 3-7 show the floor plan arrangements. Space in the building will consist of two major experiment facilities, a reactor bay room and a radiation experiment room. The reactor bay area consists of 350 square meters (3600 sq. ft.) of space for reactor structures, systems, components, and experiments. The radiation experiment room with about 85 square meters (900 sq. ft.) is a free space cube roughly 9 meters (30 ft.) on a side with thick shield walls. Besides the major experiment areas, 8 general laboratory areas and 6 supplemental support areas will be available. The respective spatial areas are 635 and 135 square meters (6840 and 1430 sq. ft.). Office, conference, and administrative space will contribute another 230 square meters (2510 sq. ft.) of assignable space in ten to twelve different rooms. These space allocations represent the result of planning estimates for a typical configuration of facility activities.

Functional arrangement of facility activities is one result of the two section building design. One of these sections, the reactor bay, wing, contains the primary experiment facilities, support facilities, such as shops and building mechanical and electrical equipment. The academic and laboratory wing is the other section, which includes administrative, office, communication, conference, and laboratory facilities.

3.4 REACTOR BAY AND OPERATION CONTROL

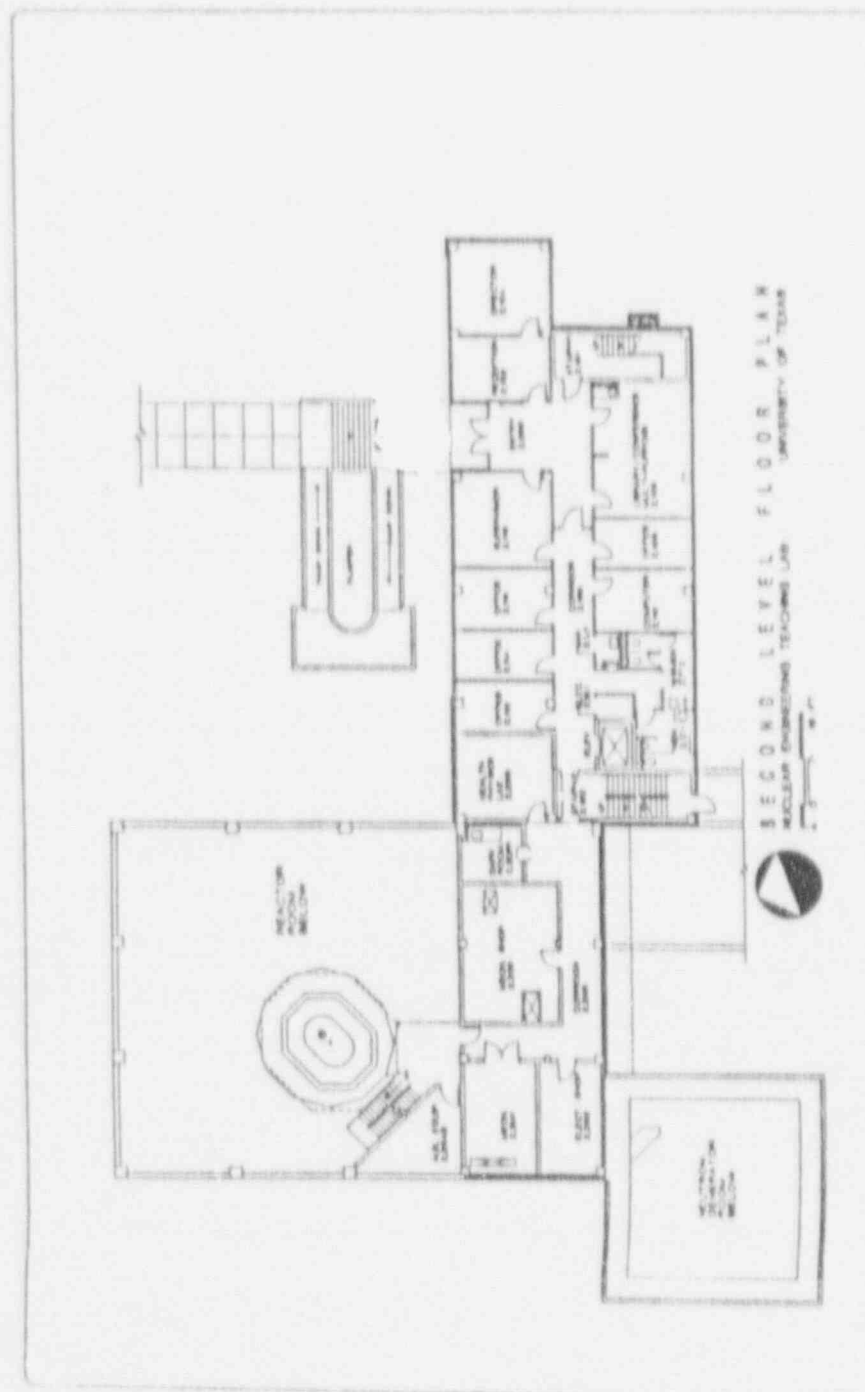
Reactor tank, shield and primary experiment facilities are located in a reactor bay area that is about 18.3 meters on each side. A total of 4575 cubic meters of volume is enclosed in the reactor bay above the 335 square meters of floor space. Operation control of reactor and of reactor experiment activities is provided by an area located adjacent to the reactor bay. Space in the operation control area is divided into control room, conference room, office, and entry way. Total operation control area (7.3 by 18.3 m) is 134 square meter of floor space and roughly 489 cubic meters of air space. The stairwell in the academic wing provides access to the reactor bay and operation control areas.



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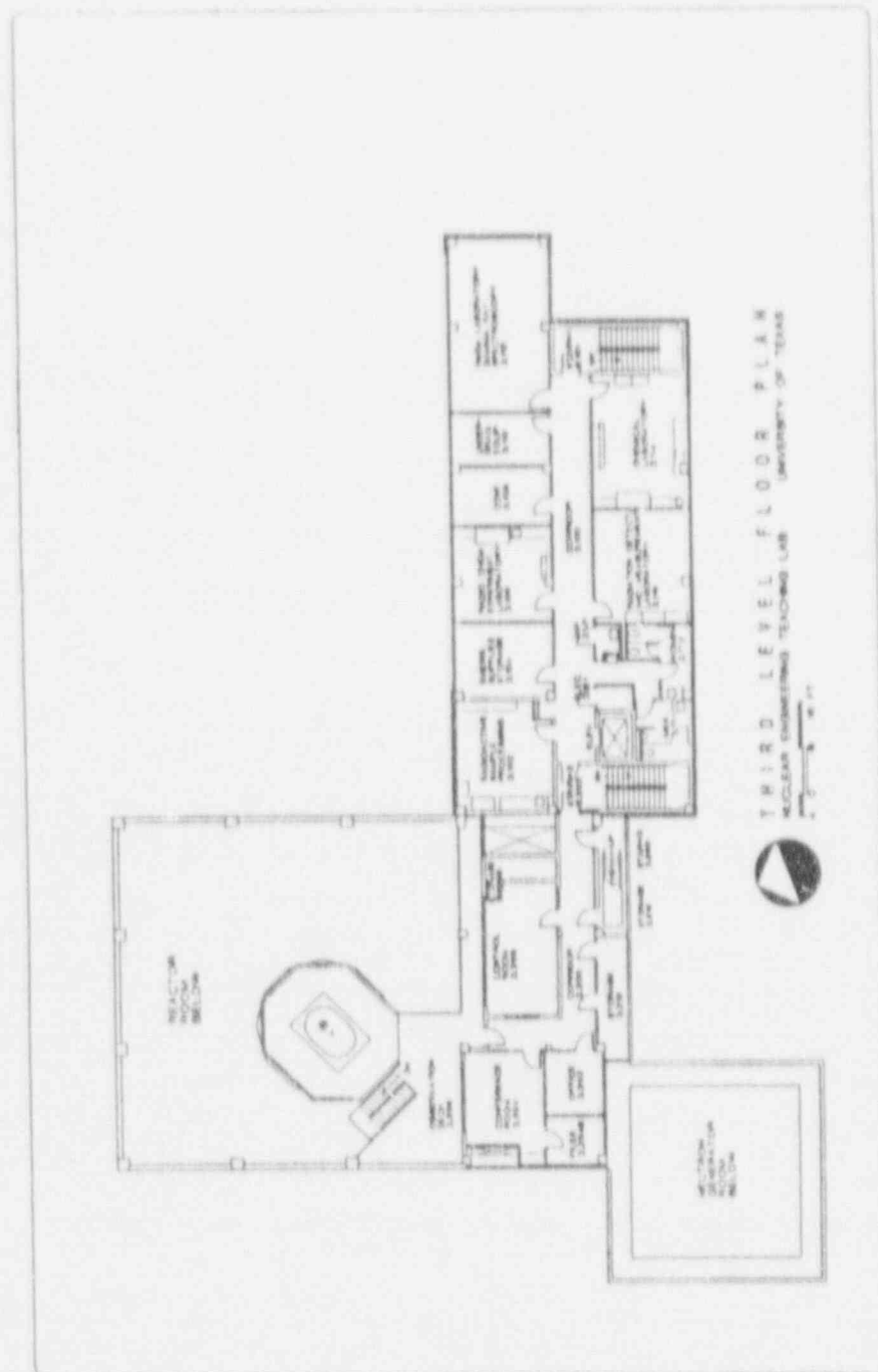
FIRST LEVEL FLOOR PLAN

Figure 3-4



SECOND LEVEL FLOOR PLAN

Figure 3-5



THIRD LEVEL FLOOR PLAN

Figure 3-6

Design of the reactor bay is specified by constraints on the function of the architecture design, access control for physical security, radiation protection for personnel safety, and applicable building code standards.

All access points to the reactor bay are located inside the engineering building and enter from the operation control area side. The remaining three sides of the reactor bay area are enclosed by exterior walls. Both emergency exits and equipment bay doors on the first level open into the adjacent area within the building from which building exits are accessible. Adjacent areas on the operation control side of the reactor bay provide some laboratory support functions in conjunction with other building support functions. On the first level adjacent to the reactor bay are staging areas for the equipment bay doors, a mechanical equipment room, liquid waste storage area, radiation experiment control center, and other areas.

Adjacent to the reactor bay on the second level is the building electrical distribution, a mechanical equipment shop, electronics equipment shop, and darkroom facility. The third level area adjacent to the reactor bay functions as the reactor operation control center. All building air handling systems and hot water systems are on the fourth level adjacent to the reactor bay.

Access from the reactor operation control center level to the reactor bay is at the top of the reactor shield at the point of access to the reactor pool. A stairway next to the reactor shield structure provides access to the first, second, and third levels of the building adjacent to the reactor bay.

Two rooms within the reactor bay will enclose reactor support systems. Pool water treatment systems for purification and cooling equipment are on the first level. Auxiliary equipment for experiment systems, such as pneumatic systems, will be in the second level room. Other features of the reactor bay include a five-ton bridge crane and six fuel storage pits. The storage pits and reactor shield structure are important systems to safely operate and store the reactor fuel materials. However, only the ventilation design for the reactor bay will be an engineering safety feature.

The operation control level consists of the reactor control room, routine entry hallway, operations office, conference room, and file storage. Windows between the operation control level and the reactor bay allow visual observation of operation activities. Routine access to the operation control area is by way of the academic wing stairway.

Design of all access points and barriers between the reactor bay or operation control area and adjacent areas will depend on requirements for access, security, fire and ventilation. Design will place limits on the size of penetrations into the area and require sealants to control air leakage.

On the third level from the reactor floor the adjacent area to the reactor bay is supplemented by the control room area, conference area, operation office, and routine entry point. The third level entry way is provided for access to the control area from the laboratory building and access to the reactor bay from the control area. Access at the third level is to the top of the reactor shield structure. A stair structure is attached to the reactor shield with a supplementary access point to the reactor bay on the second level.

Design of access points and interior walls are specified for security control, fire control, and ventilation control. Penetrations, besides the doors, into the reactor bay and control areas are limited in size and are sealed to limit air leakage. Details of the reactor bay area are presented in Figure 3-8. A 5-ton bridge crane is installed in the reactor bay for movement of shield structures, heavy equipment, and fuel transport loading.

3.5 SUPPORT FACILITIES

Support facilities for reactor operation include functions of the academic and laboratory wing of the building, two shop areas for mechanical and electronic work, and support laboratories.

3.5.1 Health Physics Laboratory

A Health Physics Laboratory is situated adjacent to the reactor facility on the second level. Radiation counting systems for evaluation of radiation exposure or contamination are maintained in the Health Physics Laboratory. Equipment such as a thermoluminescent reader and an alpha-beta proportional counter are maintained in the laboratory. Other equipment and supplies operated or stored in the laboratory include portable radiation monitors, coveralls, gloves and related items.

3.5.2 Sample Handling Laboratory

A sample handling laboratory is situated adjacent to the reactor facility on the third level for the processing of radioactive samples and materials. Access ports via air or gas transfer tubes are installed to move samples between the reactor area and the Sample Handling Laboratory. Two separate sample transfer systems are provided, one for the pneumatic tube irradiation facility and one for loading the rotary specimen rack facility.

A hood for handling radioactive materials and a sink for disposal of radioactive liquids are installed in the Sample Handling Laboratory. A safety shower is adjacent the laboratory doorway.

3.5.3 Effluent Control

Effluent pathways for air, liquid, or solid releases of radioactive material provide control of material releases. Control pathways for air and liquid effluents are by way of two rooms, room 4.1M3 and room 1.108.

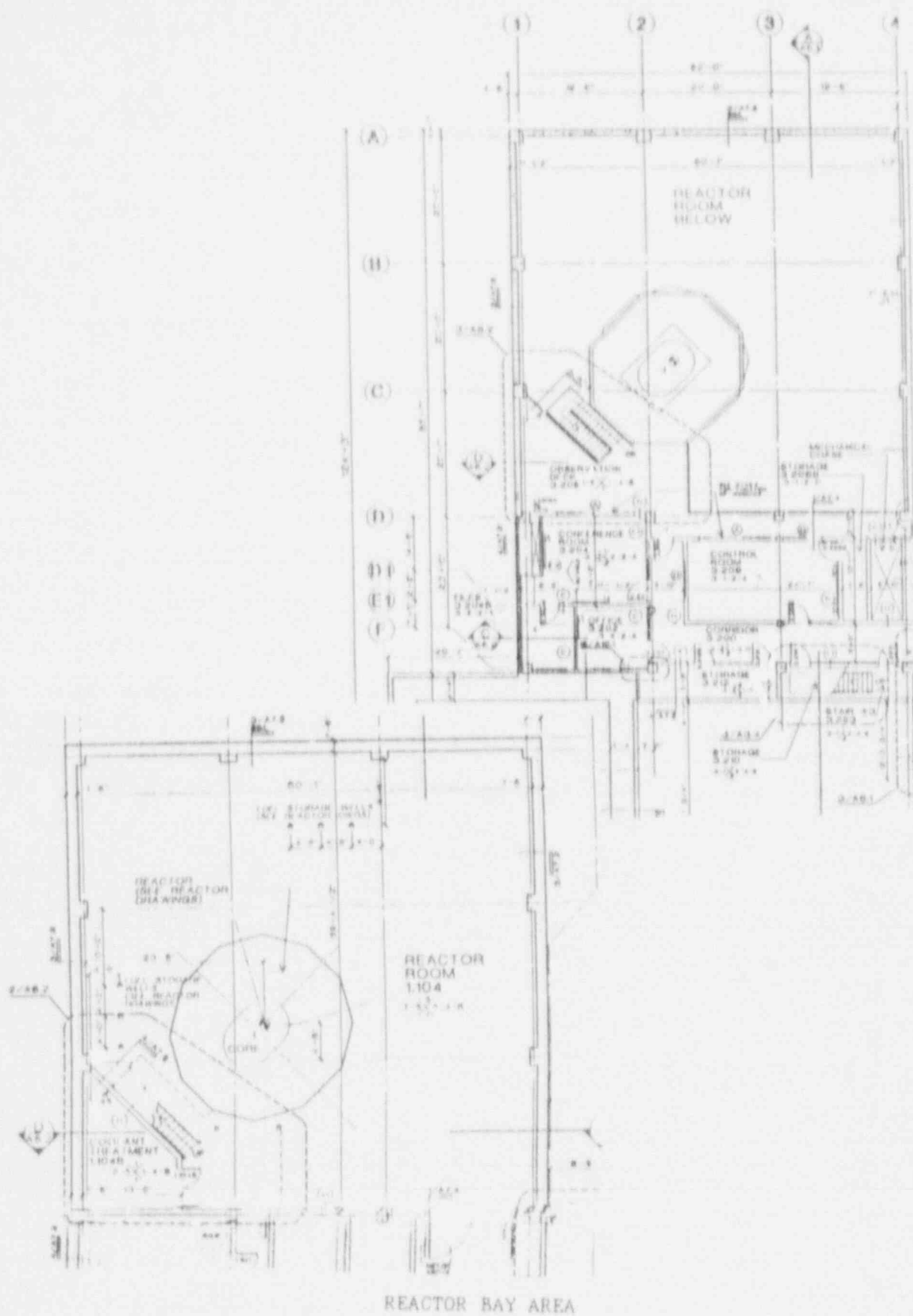


Figure 3-8

Control of air releases from reactor experiment areas is provided in room 4.1M3, which contains the air purge system isolation valve and filter bank. The filter bank normally contains prefilters and one high efficiency particulate filter. Space will be available for expansion that allows addition of a charcoal filter and additional high efficiency particulate filter.

Control of liquid releases that contain radioactive material is provided in room 1.108, which contains storage tanks for collection, processing, storage, or release of liquid effluents. The reactor pool will not release liquid effluents as a part of normal operation.

Solid waste control will typically consist of canisters, such as plastic bags and plastic or metal barrels. Location of these canisters for solid waste control will be adjacent to experiment areas that produce the waste.

3.6 DESIGN EVALUATION

Building design, construction, and inspection will provide a facility for safe operation of the TRIGA reactor. The facility and its features apply standard engineering practices to control quality of building systems. The features are typical of other similar installations and should provide a reasonable margin of safety for the operation of the TRIGA reactor. No unusual site conditions exist to threaten the building. Seismic and wind loads determine building structural design. These structural conditions protect against building failure from common site conditions of geological or meteorological origin.

Several special building features provide for control of effluents and support of operation and experiment activities. One specific engineering design feature, the reactor bay ventilation system, will protect against uncontrollable releases of airborne radioactive materials. Other building design features, such as the reactor shield structure, fuel storage wells and liquid waste system, supplement the ventilation system to provide protective or control functions that prevent the uncontrollable release of radiation or radioactive material.

Chapter 3 References

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2. "Specifications for Nuclear Engineering Teaching Laboratory", The University of Texas at Austin, Project No. 102-568, Sept. 15, 1986.
3. "Construction Administration Manual for Nuclear Engineering Teaching Laboratory", The University of Texas at Austin, Project No. 102-568, December 1986.
4. "NETL Project Nos. 1, 2, & 3", The University of Texas at Austin, Project 102-568 Amendments, December 1986.
5. "Uniform Building Code", International Conference of Building Officials, May 1, 1985.

Chapter 4

TRIGA REACTOR

The reactor design bases are predicted on the maximum operational capability for the fuel elements and configuration described in this report. The TRIGA reactor system has three major areas which are used to define the reactor design bases:

- a. Fuel temperature,
- b. Prompt negative temperature coefficient,
- c. Reactor power.

Of these three only one, fuel temperature, is a real limitation. A summary is presented below of the conclusions obtained from the reactor design bases described in this section.

4.1. DESIGN BASES

The fuel temperature is a limit in both steady-state and pulse mode operation. This limit stems from the out-gassing of hydrogen from U-ZrH (H/Zr : x) fuel and the subsequent stress produced in the fuel element clad material. The strength of the clad as a function of temperature can set the upper limit on the fuel temperature. Fuel temperature limits of 1150°C (with clad \leq 500°C) and 970°C (with clad \geq 500°C) for U-ZrH (H/Zr : 1.65) have been set to preclude the loss of clad integrity. Simnad [34] summarizes the properties of U-ZrH_x fuel materials for TRIGA reactors, including the limiting design bases and parameters.

The basic parameter which provides the TRIGA system with a large safety factor in steady-state operation and under transient conditions is the prompt negative temperature coefficient which is rather constant with temperature ($\sim 0.01\% \delta k/k^{\circ}C$), as described later. This coefficient is a function of the fuel composition and core geometry.

Fuel and clad temperature limit the operation of the reactor. However, it is more convenient to set a power level limit which is based on temperature. The design bases analysis indicates that operation at up to 1900 kW (with an 85 element core and 120°F inlet water temperature) with natural convective flow will not allow film boiling, and therefore high fuel and clad temperatures which could cause loss of clad integrity could not occur.

4.1.1 Reactor Fuel Temperature

The basic safety limit for the TRIGA reactor system is the fuel temperature; this applies for both the steady-state and pulse mode of operation.

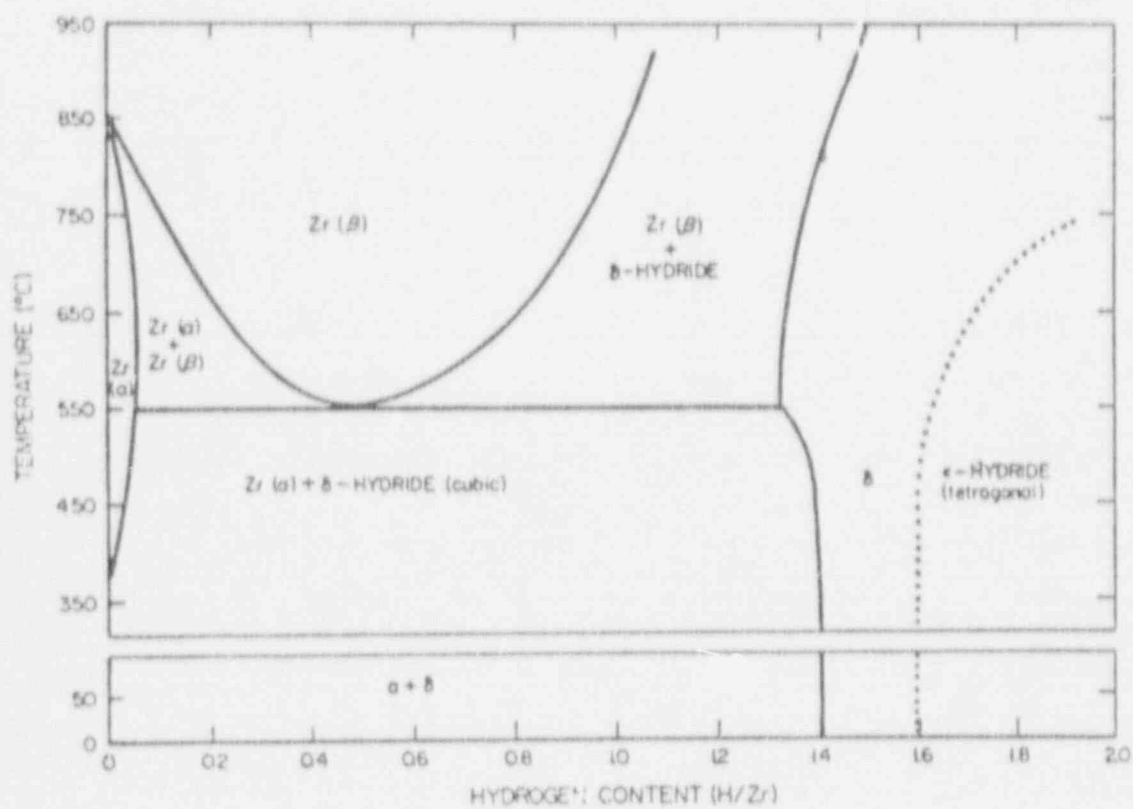
Two limiting temperatures are of interest, depending on the type of TRIGA fuel used. The TRIGA fuel which is considered low hydride, that with an H/Zr ratio of less than 1.5, has a lower temperature limit than fuel with a higher H/Zr ratio. Figure 4-1 indicates that the higher hydride compositions are single phase and are not subject to the large volume changes associated with the phase transformations at approximately 530°C in the lower hydrides. Also, it has been noted [1] that the higher hydrides lack any significant thermal diffusion of hydrogen. These two facts preclude concomitant volume changes. The important properties of delta phase U-ZrH are given in Table 4-1.

Table 4-1

PHYSICAL PROPERTIES OF DELTA PHASE U-ZrH

Thermal conductivity (93°C - 650°C)	13 Btu/hr-ft ² -°F
Elastic modulus: 20°C	9.1 x 10 ⁶ psi
650°C	6.0 x 10 ⁶ psi
Ultimate tensile strength (to 650°C)	24,000 psi
Compressive strength (20°C)	60,000 psi
Compressive yield (20°C)	35,000 psi
Heat of formation (ΔH_f^0 298°C)	37.72 kcal/g-mole

Among the chemical properties of U-ZrH and ZrH, the reaction rate of the hydride with water is of particular interest. Since the hydriding reaction is exothermic, water will react more readily with zirconium than with zirconium hydride systems. Zirconium is frequently used in contact with water in reactors, and the zirconium-water reaction is not a safety hazard. Experiments carried out at GA Technologies show that the zirconium hydride systems have a relatively low chemical reactivity with respect to water and air. These tests have involved the quenching with water of both powders and solid specimens of U-ZrH after heating to as high as 850°C, and of solid U-Zr alloy after heating to as high as 1200°C. Tests have also been made to determine the extent to which fission products are removed from the surfaces of the fuel elements at room temperature. Results prove that, because of the high resistance to leaching, a large fraction of the fission products is retained in even completely unclad U-ZrH fuel.



PHASE DIAGRAM OF THE
ZIRCONIUM-HYDROGEN SYSTEM

Figure 4-1

For the rest of the discussion of fuel temperatures, we will concern ourselves with the higher hydride ($H/Zr \geq 1.5$) TRIGA fuel clad with 304 stainless steel 0.020 in. (0.508 mm) thick, or a cladding material equivalent in strength at the temperatures discussed.

At room temperature the hydride is like ceramic and shows little ductility. However, at the elevated temperatures of interest for pulsing, the material is found to be more ductile. The effect of very large thermal stress on hydride fuel bodies has been observed in hot cell observations to cause relatively widely spaced cracks which tend to be either radial or normal to the central axis and do not interfere with radial heat flow. Since the segments tend to be orthogonal, their relative positions appear to be quite stable.

The limiting effect of fuel temperature then is the hydrogen gas over pressure. Figure 4-2 relates equilibrium hydrogen pressure over the fuel as a function of temperature for material with three different H/Zr ratios.

The hydrogen gas over pressure is not in itself detrimental but if the stress produced by the gas pressure within the fuel can exceed the ultimate strength of the clad material, a rupture of the fuel clad could occur. While the final conditions of fuel temperature and hydrogen pressure in which such an occurrence could come about are of interest, the mechanisms in obtaining temperatures and pressures of concern are different in the pulsing and steady-state mode of operation, and each mechanism will be discussed independently of the other.

In this discussion it will be assumed that the fuel consists of U-ZrH ($H/Zr : 1.65$) with the uranium being 8.5 wt. % and further that the cladding can is 304 stainless steel. The clad thickness is 0.020 in. (0.508 mm) with an inside clad diameter of 1.43 in. (3.63 cm). These fuel parameters have been chosen since they represent the nominal specifications for TRIGA fuel elements. Figure 4-3 shows the characteristic of 304 stainless steel with regard to yield and ultimate strengths as a function of temperature.

In determining the stress applied to the cladding from the internal hydrogen gas pressure the equation

$$S = P r / t , \quad (1)$$

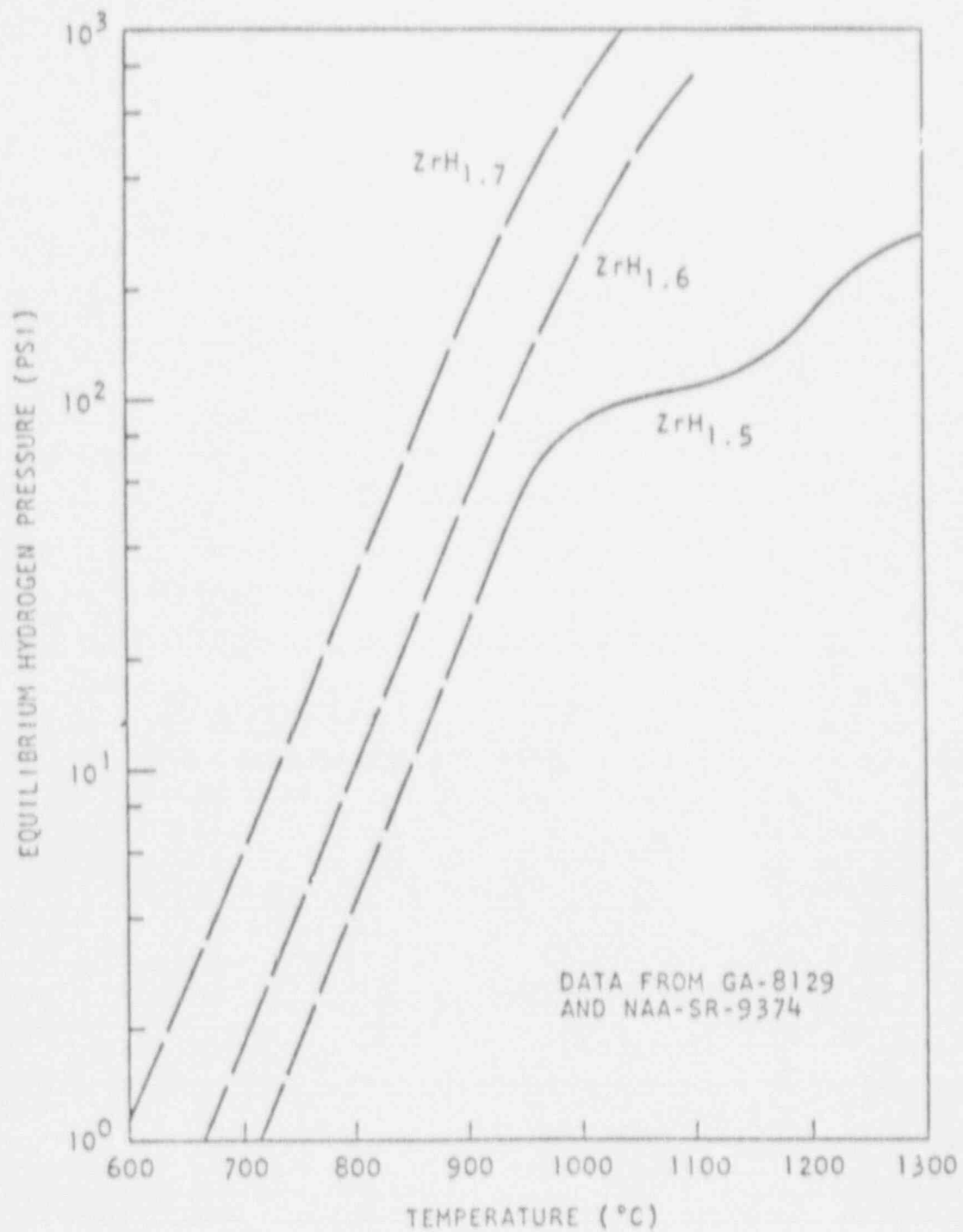
applies where

S = stress in psi,

P = internal pressure in psi,

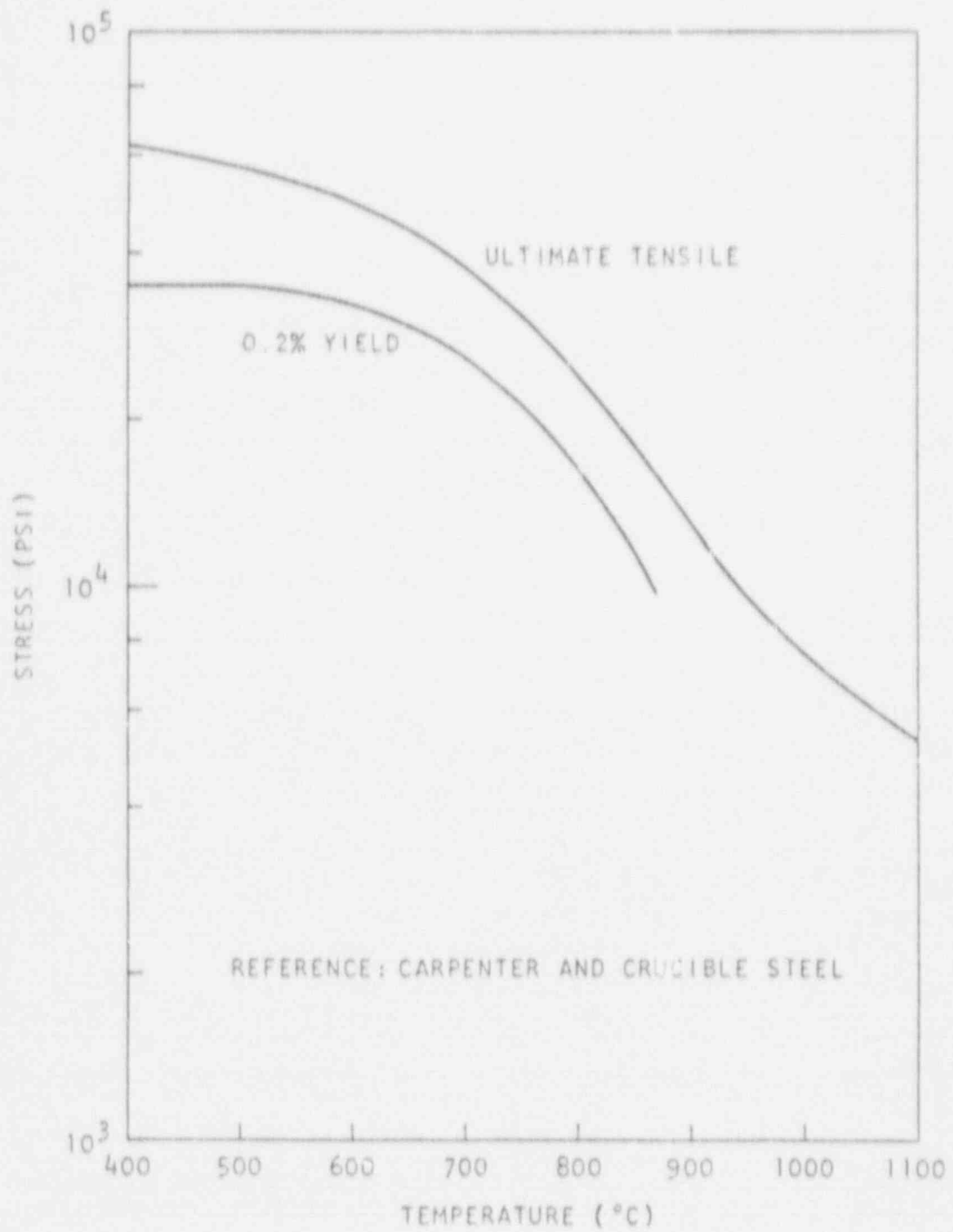
r = radius of the cladding can,

t = wall thickness of the clad.



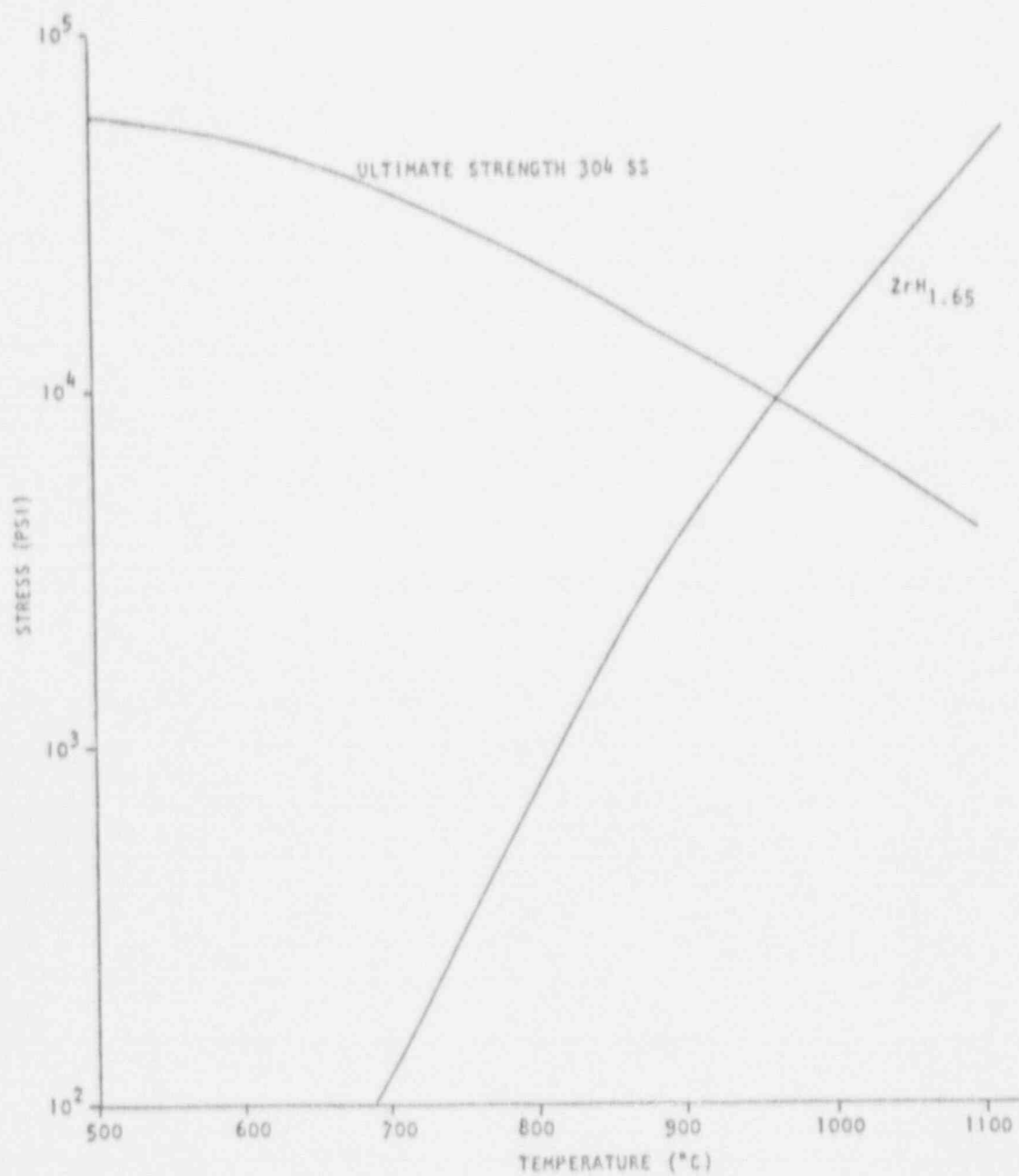
EQUILIBRIUM HYDROGEN PRESSURE
VERSUS TEMPERATURE FOR ZIRCONIUM-HYDROGEN

Figure 4-2



STRENGTH OF TYPE 304 STAINLESS STEEL
AS A FUNCTION OF TEMPERATURE

Figure 4-3



STRENGTH AND APPLIED STRESS AS A FUNCTION OF
TEMPERATURE, EQUILIBRIUM HYDROGEN DISSOCIATION PRESSURE

Figure 4-4

Then for the cladding we have approximately

$$S = 36.7 P, \quad (2)$$

or the stress applied to the clad is approximately 36.7 times the internal pressure.

It is of interest to relate the strength of the clad material at its operating temperature to the stress applied to the clad from the internal gas pressure associated with the fuel temperature. Figure 4-4 gives information as to the ultimate clad strength as a function of temperature and also describes the stress applied to the clad as a result of hydrogen dissociation for fuel having a H/Zr ratio of 1.65 as a function of temperature.

There are several reasons why the gas pressure should be less for the transient conditions than the equilibrium condition values would predict. For example, the gas diffusion rates are finite; surface cooling is believed to be caused by endothermic gas emission which tends to lower the diffusion constant at the surface; reabsorption takes place concurrently on the cooler hydride surfaces away from the hot spot; there is evidence for a low permeability oxide film on the fuel surface; and some local heat transfer does take place during the pulse time to cause a less than adiabatic true surface temperature.

4.1.1.1 Fuel and Clad Temperature. The following discussion relates the element clad temperature and the maximum fuel temperature during a short time after a pulse.

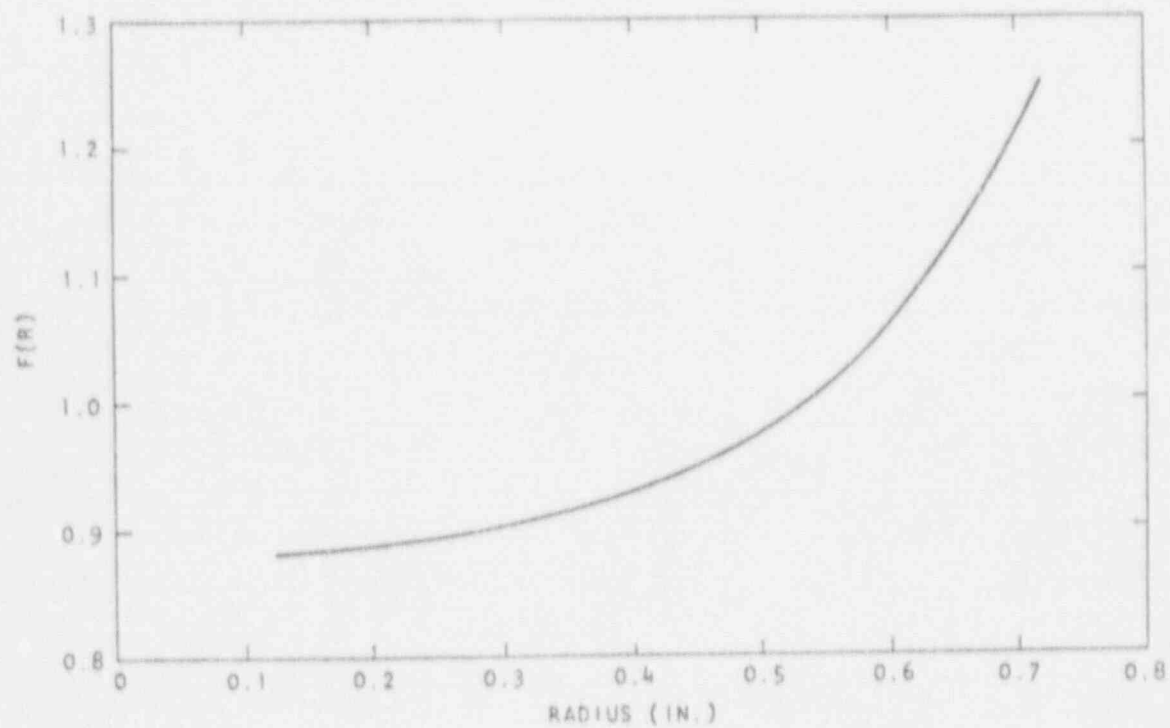
The radial temperature distribution in the fuel element immediately following a pulse is very similar to the power distribution shown in Figure 4-5. This initial steep thermal gradient at the fuel surface results in some heat transfer during the time of the pulse so that the true peak temperature does not quite reach the adiabatic peak temperature. A large temperature gradient is also impressed upon the clad which can result in a high heat flux from the clad into the water. If the heat flux is sufficiently high, film boiling may occur and form an insulating jacket of steam around the fuel elements permitting the clad temperature to tend to approach the fuel temperature. Evidence has been obtained experimentally which shows that film boiling has occurred occasionally for some fuel elements in the Advanced TRIGA Prototype Reactor located at GA Technologies [2]. The consequence of this film boiling was discoloration of the clad surface.

Thermal transient calculations were made using the RAT computer code. RAT is a 2-D transient heat transport code developed to account for fluid flow and temperature dependent material properties. Calculations show that if film boiling occurs after a pulse it may take place either at the time of maximum heat flux from the clad, before the bulk temperature of the coolant has changed appreciably, or it may take place at a much later time when the bulk temperature of the coolant has approached the saturation temperature, resulting in a markedly reduced threshold for film boiling. Data obtained by Johnson *et al.* [3] for

transient heating of ribbons in 100°F water, showed burnout fluxes of 0.9 to 2.0 MBtu/ft²-hr for e-folding periods from 5 to 90 milliseconds. On the other hand, sufficient bulk heating of the coolant channeled between fuel elements can take place in several tenths of a second to lower the departure from nucleate boiling (DNB) point to approximately 0.4 MBtu/ft²-hr. It is shown, on the basis of the following analysis, that the second mode is the most likely; i.e., when film boiling occurs it takes place under essentially steady-state conditions at local water temperatures near saturation.

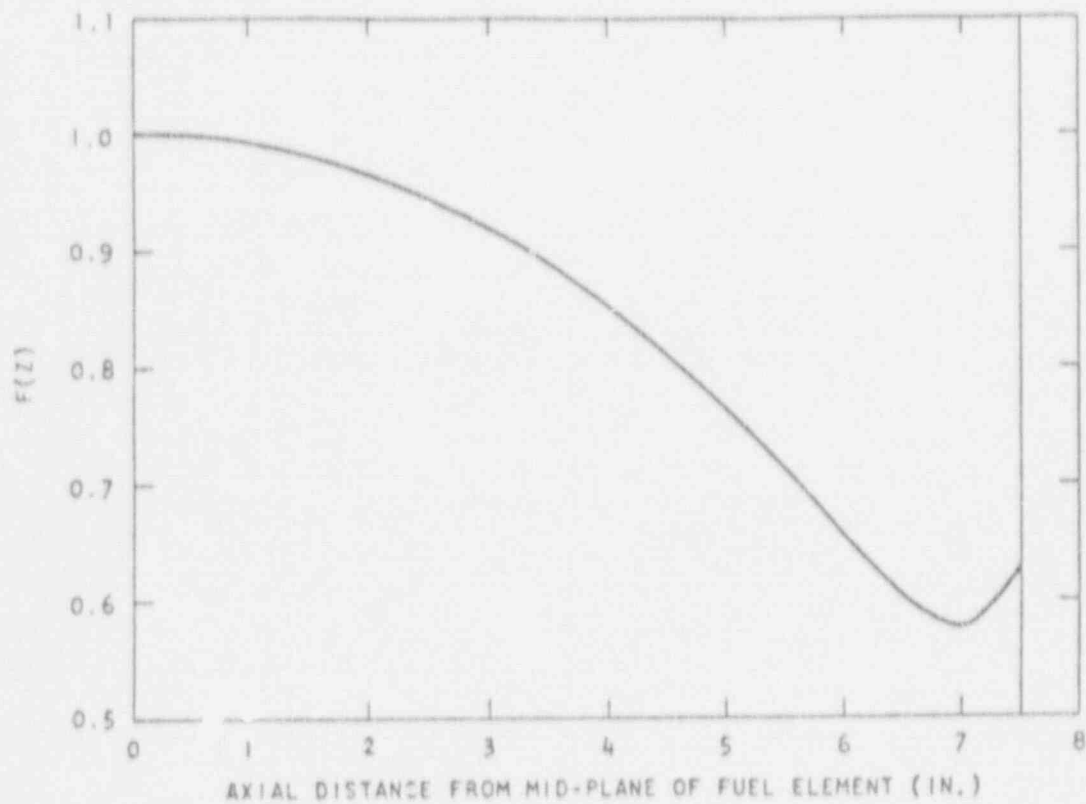
A value for the temperature that may be reached by the clad if film boiling occurs was obtained in the following manner. A transient thermal calculation was performed using the radial and axial power distributions in Figure 4-5 and Figure 4-6, respectively, under the assumption that the thermal resistance at the fuel-clad interface was nonexistent. A boiling heat transfer model, as shown in Figure 4-7, was used in order to obtain an upper limit for the clad temperature rise. The model used the data of McAdams [4] for the subcooled boiling and the work of Sparrow and Cess [5] for the film boiling regime. A conservative estimate was obtained for the minimum heat flux in film boiling by using the correlations of Speigler *et al.* [6], Zuber [7], and Rohsenow and Choi [8] to find the minimum temperature point at which film boiling could occur. This calculation gave an upper limit of 760°C clad temperature for a peak initial fuel temperature of 1000°C, as shown in Figure 4-8. Fuel temperature distributions for this case are shown in Figure 4-9 and the heat flux into the water from the clad is shown in Figure 4-10. In this limiting case, DNB occurred only 13 milliseconds after the pulse, conservatively calculated assuming a steady-state DNB correlation. Subsequently, experimental transition and film boiling data were found to have been reported by Ellion [9] for water conditions similar to those for the TRIGA system. The Ellion data show the minimum heat flux, used in the limiting calculation described above, was conservative by a factor of 5. An appropriate correction was made which resulted in a more realistic estimate of 470°C as the maximum clad temperature expected if film boiling occurs. This result is in agreement with experimental evidence obtained for clad temperatures of 400°C to 500°C for TRIGA Mark F fuel elements which have been operated under film boiling conditions [10].

The preceding analysis assessing the maximum clad temperatures associated with film boiling assumed no thermal resistance at fuel-clad interface. Measurements of fuel temperatures as a function of steady-state power level provide evidence that after operating at high fuel temperatures, a permanent gap is produced between the fuel body and the clad by fuel expansion. This gap exists at all temperatures below the maximum operating temperature. (See, for example, Figure 16 in Reference 10.) The gap thickness varies with fuel temperature and clad temperature so that cooling of the fuel or overheating of the clad tends to widen the gap and decrease the heat transfer rate. Additional thermal resistance due to oxide and other films on the fuel and clad surfaces is expected. Experimental and theoretical studies of thermal contact resistance have been reported [11-13] which provide insight into the mechanisms involved. They do not, however, permit quantitative prediction of this application because the basic data required for input



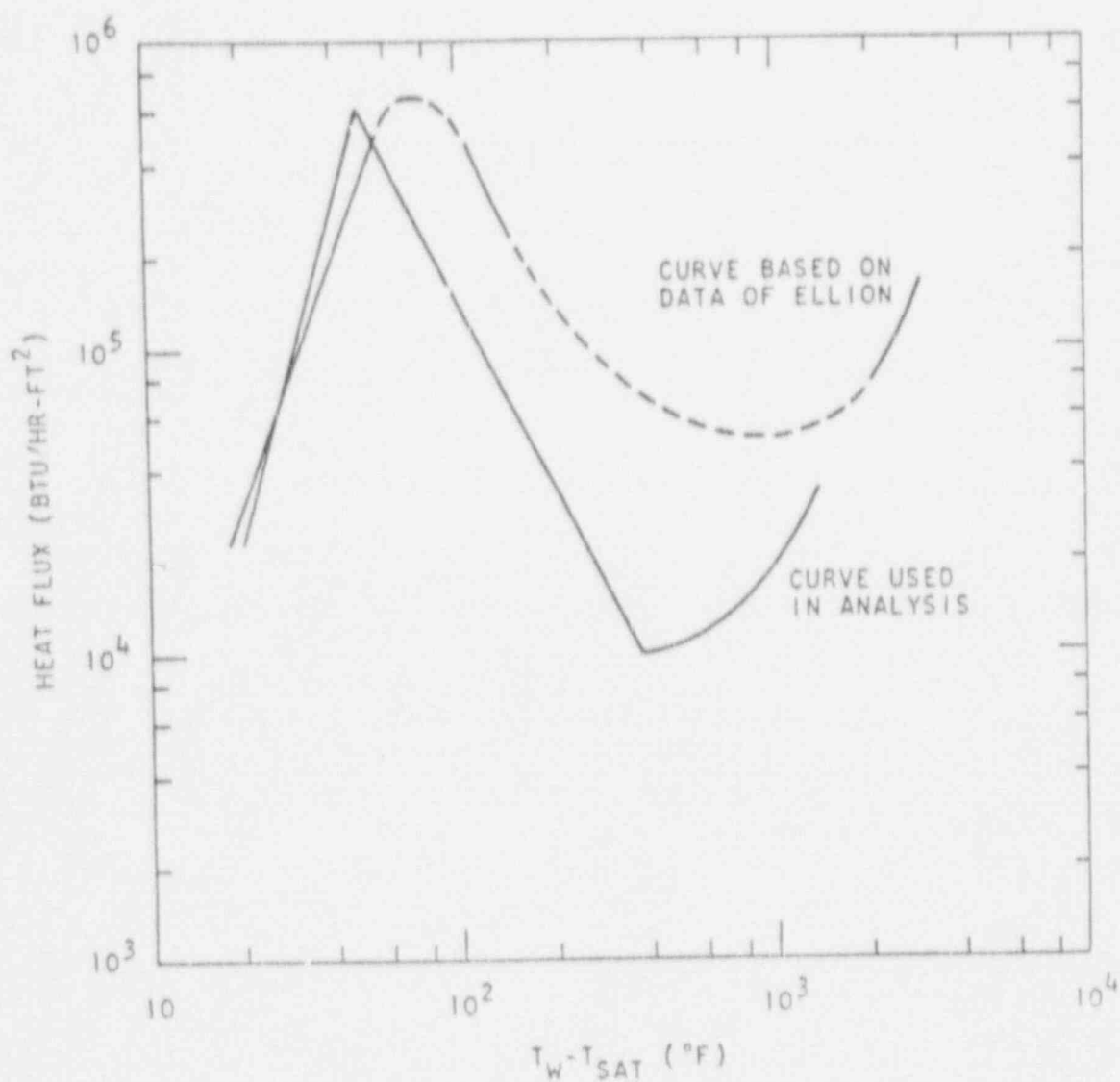
RADIAL POWER DISTRIBUTION IN
THE U-2rH FUEL ELEMENT

Figure 4-5



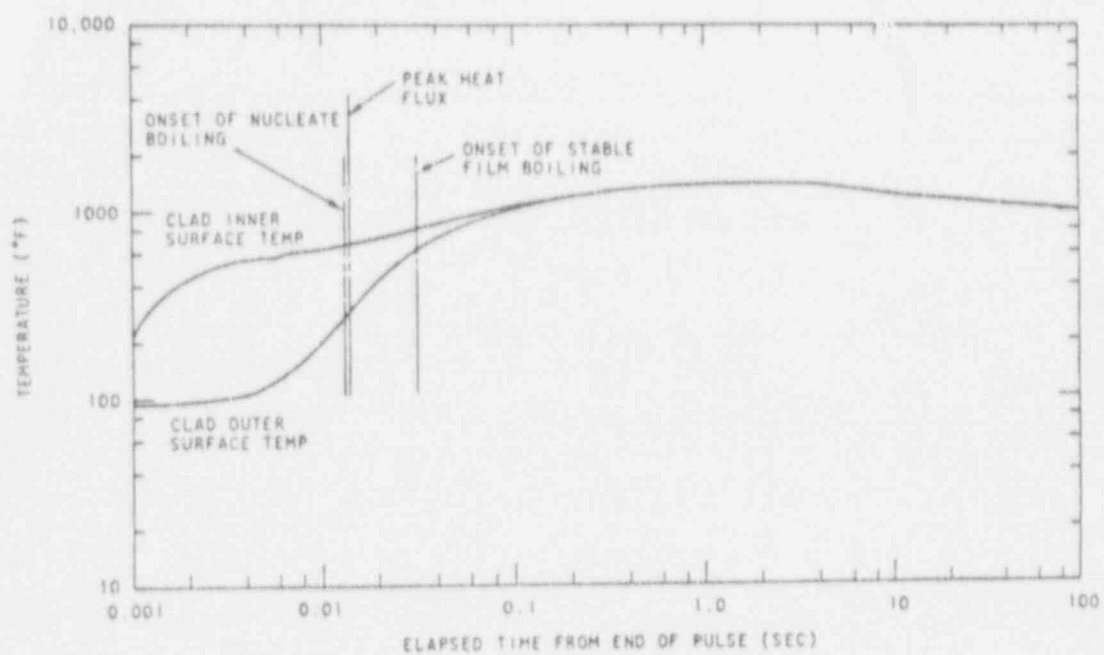
AXIAL POWER DISTRIBUTION IN
THE U-ZrH FUEL ELEMENT

Figure 4-6



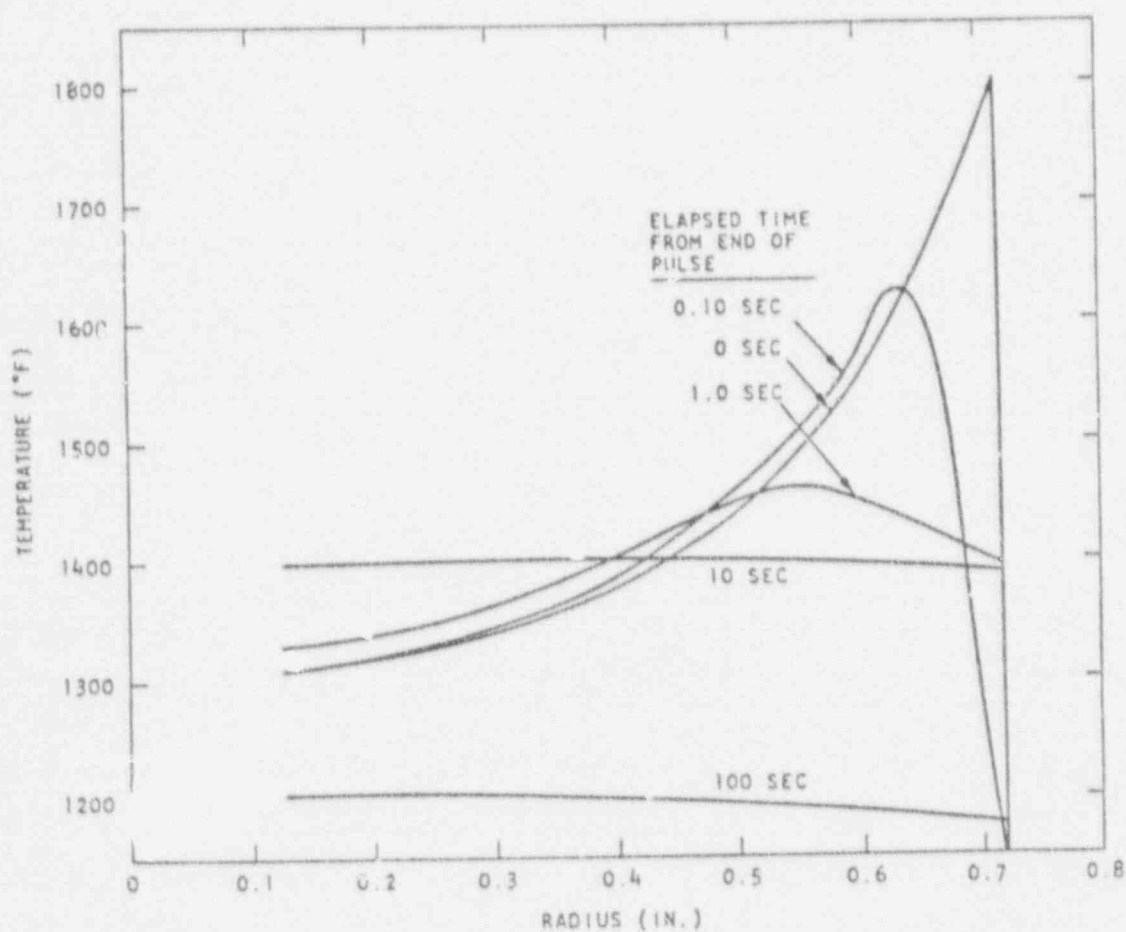
SUBCOOLED BOILING HEAT TRANSFER FOR WATER

Figure 4-7



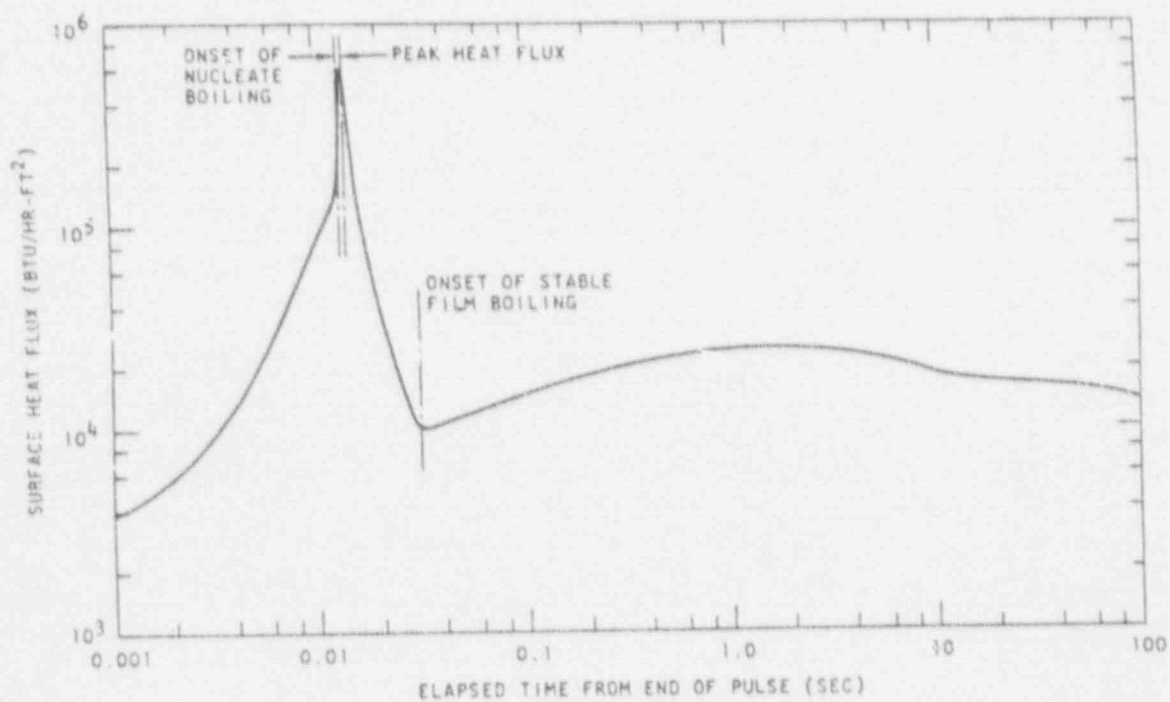
CLAD TEMPERATURE AT MIDPOINT OF
WELL-BONDED FUEL ELEMENT

Figure 4-8



FUEL BODY TEMPERATURES AT MIDPLANE OF WELL-BONDED
FUEL ELEMENT AFTER A PULSE

Figure 4-9



SURFACE HEAT FLUX AT MIDPLANE OF WELL-BONDED
FUEL ELEMENT AFTER A PULSE

Figure 4-10

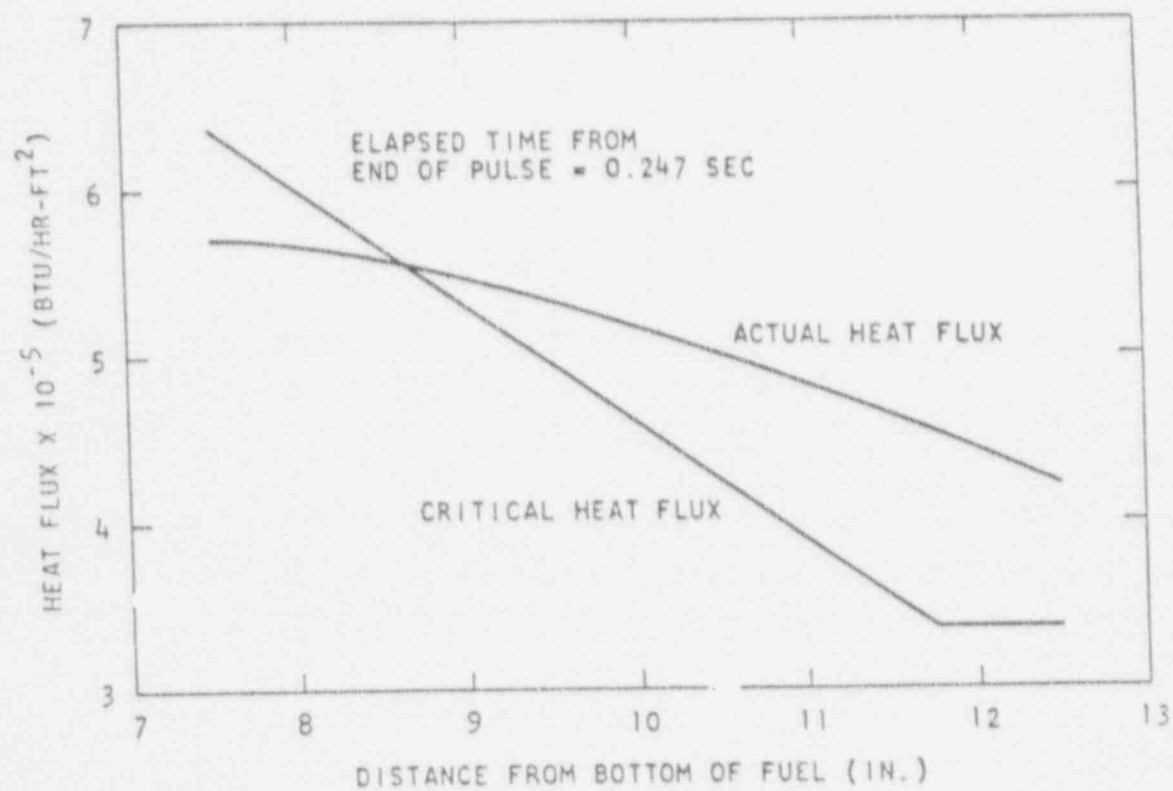
are presently not fully known. Instead, several transient thermal computations were made using the RAT code. Each of these was made with an assumed value for the effective gap conductance, in order to determine the effective gap coefficient for which departure from nucleate boiling is incipient. These results were then compared with the incipient film boiling conditions of the 1000°C peak fuel temperature case.

For convenience, the calculations were made using the same initial temperature distribution as was used for the preceding calculation. The calculations assumed a coolant flow velocity of 1 ft per second, which is within the range of flow velocities computed for natural convection under various steady-state conditions for these reactors. The calculations did not use a complete boiling curve heat transfer model, but instead, included a convection cooled region (no boiling) and a subcooled nucleate boiling region without employing an upper DNB limit. The results were analyzed by inspection using the extended steady-state correlation of Bernath [14] which has been reported by Spano [15] to give agreement with SPERT II burnout results within the experimental uncertainties in flow rate.

The transient thermal calculations were performed using effective gap conductances of 500, 375, and 250 Btu/hr-ft²-°F. The resulting wall temperature distributions were inspected to determine the axial wall position and time after the pulse which gave the closest approach between the local computed surface heat flux and the DNB heat flux according to Bernath. The axial distribution of the computed and critical heat fluxes for each of the three cases at the time of closest approach is given in Figures 4-11 thru 4-13. If the minimum approach to DNB is corrected to TRIGA Mark F conditions and cross-plotted, an estimate of the effective gap conductance of 450 Btu/hr-ft²-°F is obtained for incipient burnout so that the case using 500 is thought to be representative of standard TRIGA fuel.

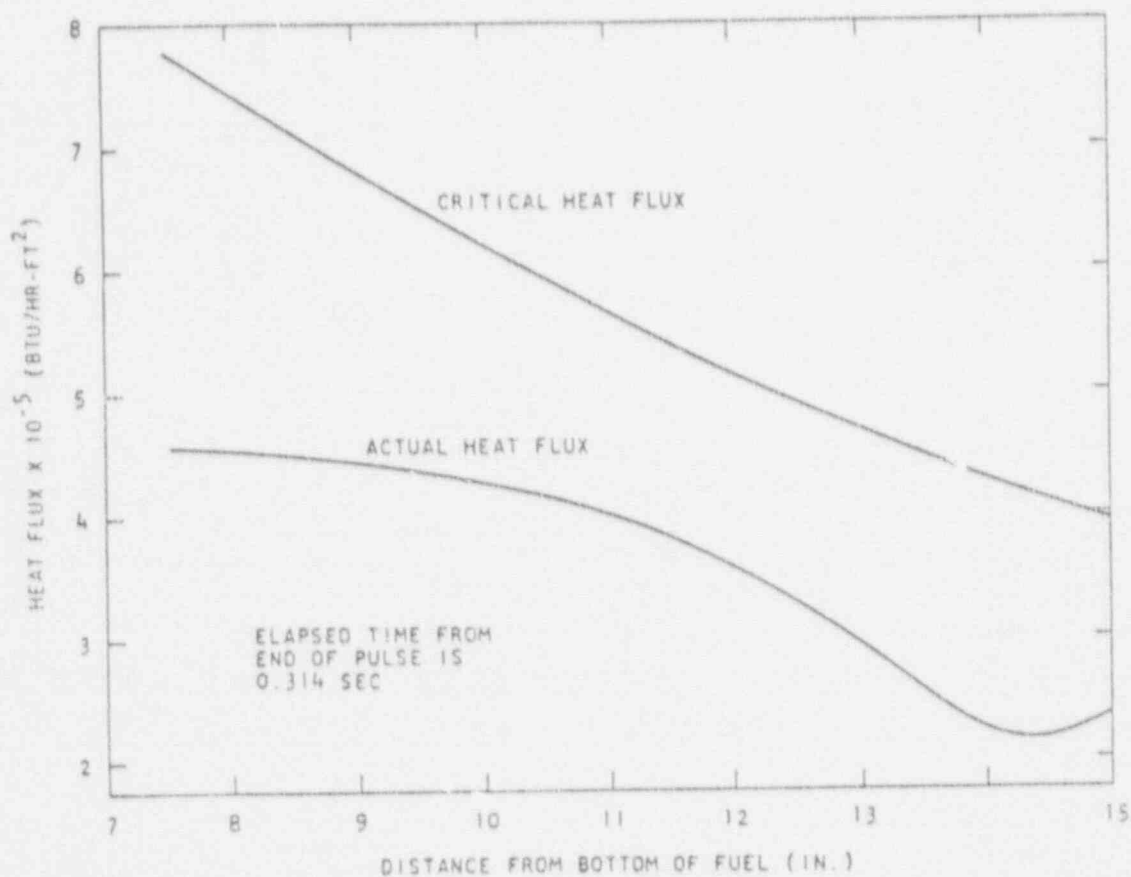
The surface heat flux at the midplane of the element is shown in Figure 4-14 with gap conductance as a parameter. It may be observed that the maximum heat flux is approximately proportional to the heat transfer coefficient of the gap, and the time lag after the pulse for which the peak occurs is also increased by about the same factor. The closest approach to DNB in these calculations did not necessarily occur at these times and places, however, as indicated on the curves of Figures 4-11 thru 4-13. The initial DNB point occurred near the core outlet for a local heat flux of about 340 kBtu/hr-ft²-°F according to the more conservative Bernath correlations at a local water temperature approaching saturation.

This analysis indicates that after operation of the reactor at steady-state power levels of 1 MW(t), or after pulsing to equivalent fuel temperatures, the heat flux through the clad is reduced and therefore reduces the likelihood of reaching a regime where there is a departure from nucleate boiling. From the foregoing analysis, a maximum temperature for the clad during a pulse which gives a peak adiabatic fuel temperature of 1000°C is conservatively estimated to be 470°C.



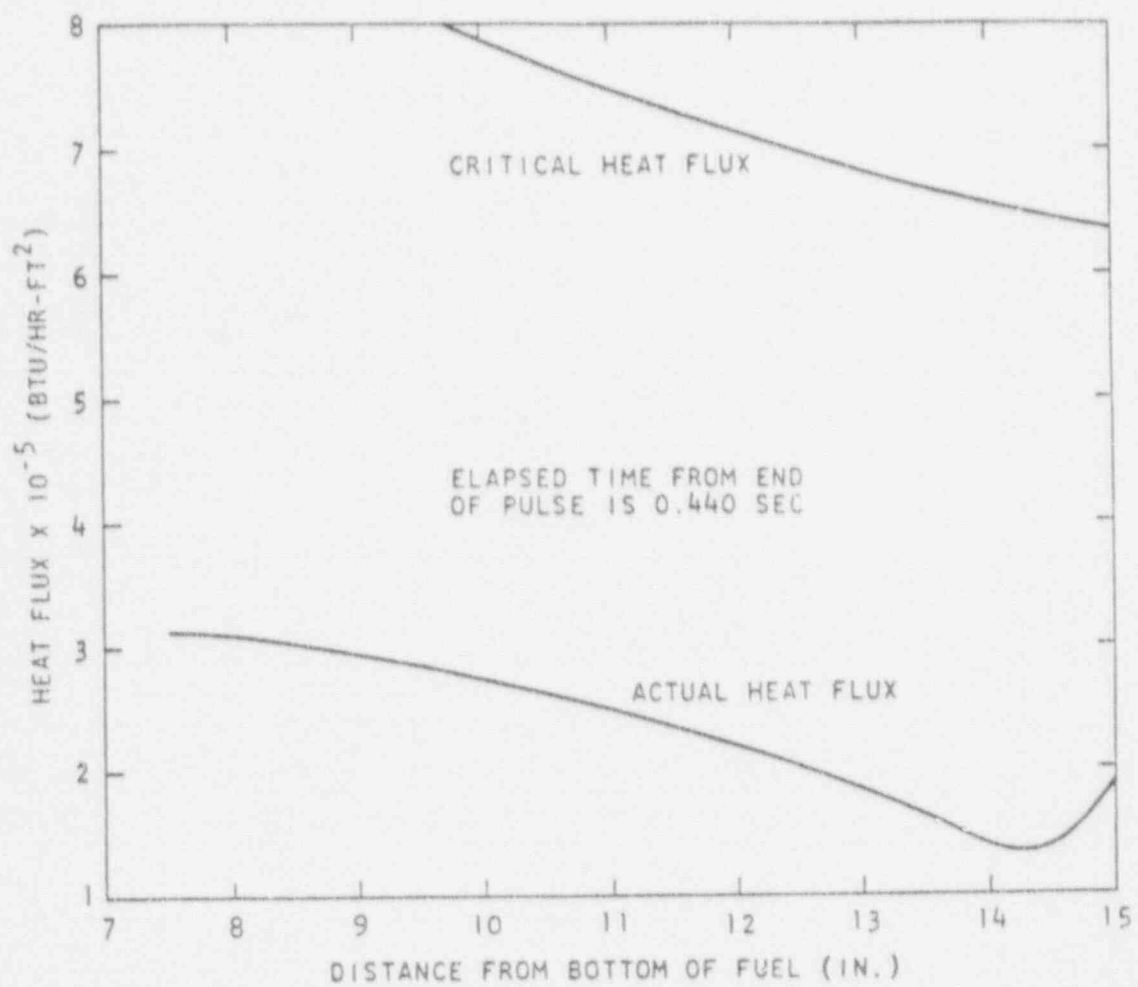
SURFACE HEAT FLUX DISTRIBUTION FOR STANDARD
NON-GAPPED ($h_{gap}=500$) FUEL ELEMENT AFTER A PULSE

Figure 4-11



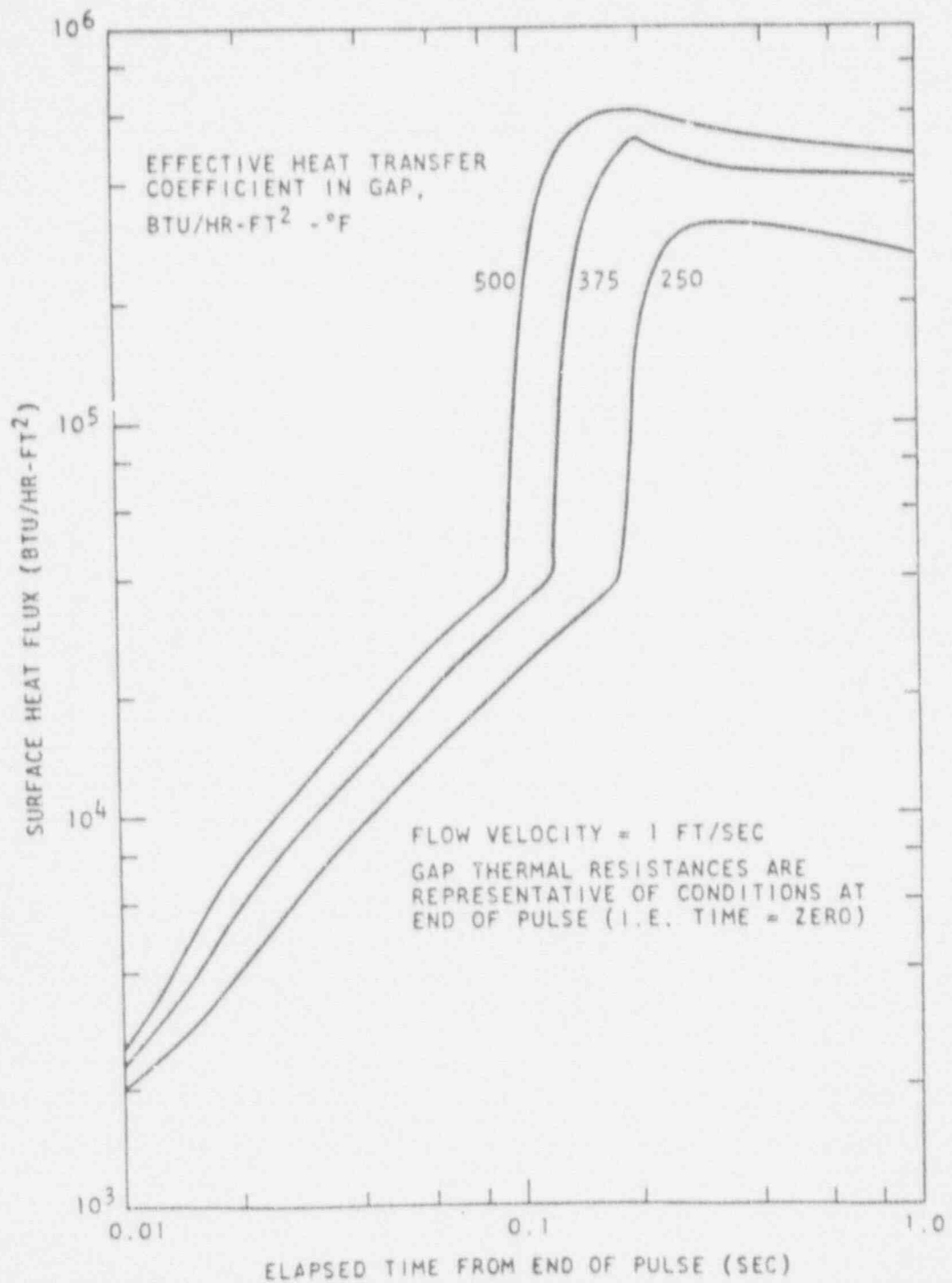
SURFACE HEAT FLUX DISTRIBUTION FOR STANDARD
NON-GAPPED ($h_{gap}=375$) FUEL ELEMENT AFTER A PULSE

Figure 4-12



SURFACE HEAT FLUX DISTRIBUTION FOR STANDARD
NON-GAPPED ($h_{gap}=250$) FUEL ELEMENT AFTER A PULSE

Figure 4-13



SURFACE HEAT FLUX AT MIDPOINT VERSUS TIME
FOR STANDARD NON-GAPPED FUEL ELEMENT AFTER A PULSE

Figure 4-14

As can be seen from Figure 4-3, the ultimate strength of the clad at a temperature of 470°C is 59,000 psi. If the stress produced by the hydrogen over pressure in the can is less than 59,000 psi, the fuel element will not undergo loss of containment. Referring to Figure 4-4, and considering U-ZrH fuel with a peak temperature of 1000°C, one finds the stress on the clad to be 12,600 psi. Further studies show that the hydrogen pressure which would result from a transient for which the peak fuel temperature is 1150°C would not produce a stress in the clad in excess of its ultimate strength. TRIGA fuel with a hydrogen to zirconium ratio of at least 1.65 has been pulsed to temperatures of about 1150°C without damage to the clad [16].

4.1.1.2 Finite Diffusion Rate. To assess the effect of the finite diffusion rate and the rehydrogening at the cooler surfaces, the following analysis is presented.

As hydrogen is released from the hot fuel regions, it is taken up in the cooler regions and the equilibrium that is obtained is characteristic of some temperature lower than the maximum. To evaluate this reduced pressure, we will use diffusion theory to calculate the rate at which hydrogen is evolved and reabsorbed at the fuel surface.

Ordinary diffusion theory provides an expression for describing the time dependent loss of gas from a cylinder:

$$\frac{\bar{c} - c_f}{c_i - c_f} = \sum_{n=1}^{\infty} \frac{4}{\xi_n^2} \exp \left[-\frac{\xi_n^2 Dt}{r_0^2} \right] \quad (3)$$

where

\bar{c} , c_f , c_i = the average, the initial, and the final gas concentration in the cylinder, respectively,

ξ_n = the roots of the Equation $J_0(x) = 0$,

D = the diffusion coefficient for the gas in the cylinder,

r_0 = the radius of the cylinder,

t = time.

Setting the term on the right-hand side of Equation 3 equal to κ , one can rewrite Equation 3 as:

$$\bar{c}/c_i = c_f/c_i + (1 - c_f/c_i) \kappa \quad (4)$$

and the derivative in time is given by

$$\frac{d(\bar{c}/c_i)}{dt} = (1 - c_f/c_i) \frac{d\kappa}{dt} \quad (5)$$

This represents the fractional release rate of hydrogen from the cylinder, $f(t)$. The derivative of the series in the right-hand side of Equation 3 was approximated by

$$\frac{dx}{dt} = - (7.339e^{-8.34\epsilon} + 29.88e^{-249\epsilon}) \frac{d\epsilon}{dt} \quad (6)$$

where $\epsilon = Dt/r_0^2$.

The diffusion coefficient for hydrogen in zirconium hydride in which the H/Zr ratio is between 1.56 and 1.86 is given by

$$D = 0.25 e^{-17800/R(T+273)} \quad (7)$$

where R = the gas constant and,

T = the zirconium hydride temperature in °C.

Equation 3 describes the escape of gas from a cylinder through diffusion until some final concentration is achieved. Actually, in the closed system considered here, not only does the hydrogen diffuse into the fuel-clad gap, but also it diffuses back into the fuel in the regions of lower fuel temperature. The gas also diffuses through the clad at a rate dependent on the clad temperature. Although this tends to reduce the hydrogen pressure, it is not considered in this analysis. When the diffusion rates are equal, an equilibrium condition will exist. To account for this, Equation 5 was modified by substituting for the concentration ratios the ratio of the hydrogen pressure in the gap to the equilibrium hydrogen pressure, P_h/P_e . Thus,

$$f(t) = \frac{d(\bar{c}/c_i)}{dt} = (1 - P_h(t)/P_e) \frac{dx}{dt} \quad (8)$$

where $P_h(t)$ = the hydrogen pressure, a function of time and

P_e = the equilibrium hydrogen pressure over the zirconium hydride which is a function of the fuel temperature.

The rate of change of the internal hydrogen pressure, in psi, inside the fuel element cladding is

$$\frac{dP_h}{dt} = \frac{14.7 f(t) N_h}{6.02 \times 10^{23}} \frac{22.4}{V_g} \frac{T+273}{273} \quad (9)$$

where N_h = the number of molecules of H_2 in the fuel,

T = the gas temperature (°C),

$f(t)$ = the fractional loss rate from Equation 8,

V_g = the free volume inside the fuel clad (liters).

As the atom density of hydrogen in ZrH (H/Zr ; 1.65) is about 0.1 moles and the fuel volume is 400 cubic cm., N_h is 19.9 moles (H_2). The free volume is assumed to consist of a cylindrical volume, at the top of the element, 1/8-in. high with a diameter of 1.43 in. for a total of 3.3 cubic cm. Also, the temperature of the hydrogen in the gap was assumed to be the temperature of the clad. The effect of changing these two assumptions was tested by calculations in which the gap volume was decreased by 90% and the temperature of the hydrogen in the gap was set up equal to the maximum fuel temperature. Neither of these changes resulted in maximum pressures different from those based on the original assumptions although the initial rate of pressure increase was greater. For these conditions

$$P_h = 7.29 \times 10^3 (T + 273) \int f(t) dt \quad (10)$$

The fuel temperature used in Equation 7 to evaluate the diffusion coefficient is expressed as

$$\begin{aligned} T(z) &= T_0 \quad ; \quad t < 0 \\ T(z) &= T_0 + (T_m - T_0) \cos [2.45(z-0.5)] \quad ; \quad t \geq 0 \end{aligned} \quad (11)$$

where T_m = the peak fuel temperature ($^{\circ}\text{C}$),

T_0 = the clad temperature ($^{\circ}\text{C}$),

z = the axial distance expressed as a fraction of the fuel length,

t = the time after step increase in power.

It was assumed that the fuel temperature was invariant with radius. The hydrogen pressure over the zirconium hydride surface when equilibrium prevails is strongly temperature dependent as shown in Figure 4-2 and, for ZrH (H/Zr ; 1.65), can be expressed by

$$P_e = 2.07 \times 10^9 e^{-1.974 \times 10^4 / (T+273)} \quad (12)$$

The coefficients have been derived from data developed by Johnson. The rate at which hydrogen is released or reabsorbed takes the form

$$g(t,z) = \frac{[P_e(z) - P_h(t)]}{P_e(z)} f(t,z) \quad (13)$$

where $f(t,z)$ = the derivative given in Equation 8 with respect to time evaluated at the axial position z ,

$P_h(t)$ = the hydrogen pressure in the gap at time t ,

$P_e(z)$ = the equilibrium hydrogen pressure at the ZrH temperature at position z .

The internal hydrogen pressure is then

$$P_H(t) = 7.29 \times 10^3 (T_0 + 273) \int_0^t \int_0^1 g(t,z) dz dt \quad (14)$$

This Equation was approximated by

$$P_H(t_1) = 7.29 \times 10^3 (T_0 + 273) \times \sum_{i=1}^n \sum_{j=1}^m \left[1 - \frac{P_H(t_{i-1})}{P_E(z_j)} \right] \times f(t_i, z_j) \delta z \delta t \quad (15)$$

where the internal summation is over the fuel element length increments and the external summation is over time.

For the case in which the maximum fuel temperature is 1150°C, the equilibrium hydrogen pressure in ZrH (H/Zr ; 1.65) is 2000 psi. Calculations indicate, however, that the internal pressure increases to a peak at about 0.3 sec, at which time the pressure is about one-fifth of the equilibrium value or about 400 psi. After this time, the pressure slowly decreases as the hydrogen continues to be redistributed along the length of the element from the hot regions to the cooler regions.

Calculations have also been made for step increases in power to peak fuel temperatures greater than 1150°C. Over a 200°C range, the time to the peak pressure and the fraction of the equilibrium pressure value achieved were approximately the same as for the 1150°C case. Thus, if the clad remains below about 500°C, the internal pressure that would produce the yield stress in the clad (35,000 psi) is about 1000 psi and the corresponding equilibrium hydrogen pressure corresponds to a maximum fuel temperature of about 1250°C in ZrH (H/Zr ; 1.65). Similarly, an internal pressure of 1600 psi would produce a stress equal to the ultimate clad strength (over 59,000 psi). This corresponds to an equilibrium hydrogen pressure of 5×1600 or 8000 psi and a fuel temperature of about 1300°C.

Measurements of hydrogen pressure in TRIGA fuel elements during steady-state operation have not been made. However, measurements have been made during transient operations and compared with the results of an analysis similar to that described here. These measurements indicated that in a pulse in which the maximum temperature in the fuel was greater than 1000°C the maximum pressure was only about 6% of the equilibrium value evaluated at the peak temperature. Calculations of the pressure resulting from such a pulse using the methods described above gave calculated pressure values about three times greater than the measured values.

An instantaneous increase in fuel temperature will produce the most severe pressure conditions. When a peak fuel temperature of 1150°C is reached by increasing the power over a finite period of time, the resulting pressure will be no greater than that for the step change in

power analyzed above. As the temperature rise times become long compared with the diffusion time of hydrogen, the pressure will become increasingly less than for the case of a step change in power. The reason for this is that the pressure in the clad element results from the hot fuel dehydriding faster than the cooler fuel rehydrides (takes up the excess hydrogen to reach an equilibrium with the hydrogen over pressure in the can). The slower the rise to peak temperature, the lower the pressure because of the additional time available for rehydriding.

4.1.1.3 Summary The foregoing analysis gives a strong indication that the clad will not be ruptured if fuel temperatures are never greater than in the range of 1200°C to 1250°C, providing that the clad temperature is less than about 500°C. However, a conservative safety limit of 1150°C has been chosen for this condition. As a result, at this safety limit temperature the pressure is about a factor of 4 lower than would be necessary for clad failure. This factor of 4 is more than adequate to account for uncertainties in clad strength and manufacturing tolerances.

Under any condition in which the clad temperature increases above 500°C, the temperature safety limit must be decreased as the clad material loses strength at elevated temperatures. To establish this limit, it is assumed that the fuel and the clad are at the same temperature. There are no conceivable circumstances that could give rise to a situation in which the clad temperatures was higher than the fuel temperature.

In Figure 4-4 there is plotted the stress imposed on the clad by the equilibrium hydrogen pressure as a function of the fuel temperature, again assuming a clad radius of 0.73 in. and a thickness of 0.02 in. Also shown is the ultimate strength of 304 stainless steel at the same temperatures. The use of these data for establishing the safety limit is justified as

- a. the method used to measure ultimate strength requires the imposition of the stress over a longer time than would be imposed for accident conditions,
- b. the stress is not applied biaxially in the ultimate strength measurements as it is in the fuel clad.

The point at which the two curves in Figure 4-4 intersect is the safety limit, that is, 970°C. At that temperature the equilibrium hydrogen pressure would impose a stress on the clad equal to the ultimate strength of the clad.

The same argument about the redistribution of the hydrogen within the fuel presented earlier is valid for this case also. In addition, at elevated temperatures the clad becomes quite permeable to hydrogen. Thus, not only will hydrogen redistribute itself within the fuel to reduce the pressure, but also some hydrogen will escape from the system entirely.

The use of the ultimate strength of the clad material in the establishment of the safety limit under these conditions is justified because of the transient nature of such accidents. Although the high clad temperatures imply sharply reduced heat transfer rates to the surroundings (and consequently longer cooling times), only slight reductions in the fuel temperature are necessary to reduce the stress sharply. A 50°C decrease in temperature from 970°C to 920°C will reduce the stress by a factor of 2.

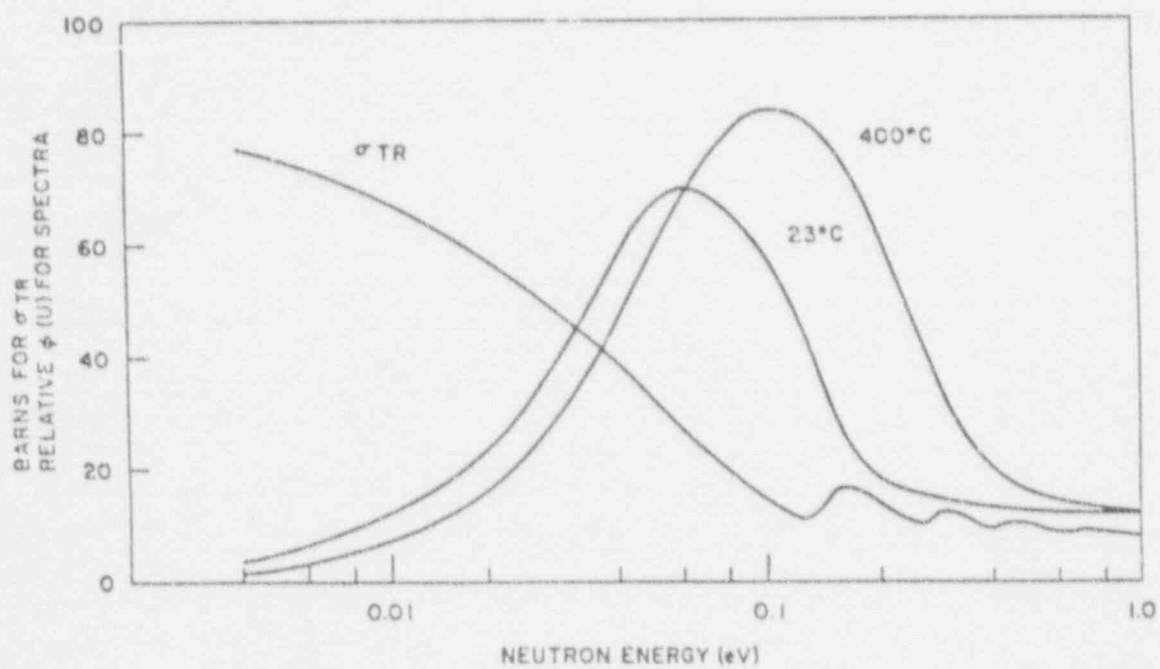
As a safety limit, the peak adiabatic fuel temperature to be allowed during transient conditions is considered to be 1150°C for U-ZrH_{1.65}.

4.1.2. Prompt Negative Temperature Coefficient

The basic parameter which allows the TRIGA reactor system to operate safely with large step insertions of reactivity is the prompt negative temperature coefficient associated with the TRIGA fuel and core design. This temperature coefficient (α) also allows a greater freedom in steady-state operation as the effect of accidental reactivity changes occurring from the experimental devices in the core is greatly reduced.

CA Technologies, the designer of the reactor, has developed techniques to calculate the temperature coefficient accurately and therefore predict the transient behavior of the reactor. This temperature coefficient arises primarily from a change in the disadvantage factor resulting from the heating of the uranium zirconium hydride fuel-moderator elements. The coefficient is prompt because the fuel is intimately mixed with a large portion of the moderator and thus fuel and solid moderator temperatures rise simultaneously. A quantitative calculation of the temperature coefficient requires a knowledge of the energy dependent distribution of thermal neutron flux in the reactor.

The basic physical processes which occur when the fuel-moderator elements are heated can be described as follows: the rise in temperature of the hydride increases the probability that a thermal neutron in the fuel element will gain energy from an excited state of an oscillating hydrogen atom in the lattice. As the neutrons gain energy from the ZrH, their mean free path is increased appreciably. This is shown qualitatively in Figure 4-15. Since the average chord length in the fuel element is comparable with a mean free path, the probability of escape from the fuel element before capture is increased. In the water the neutrons are rapidly rethermalized so that the capture and escape probabilities are relatively insensitive to the energy with which the neutron enters the water. The heating of the moderator mixed with the fuel thus causes the spectrum to harden more in the fuel than in the water. As a result, there is a temperature dependent disadvantage factor for the unit cell in the core which decreases the ratio of absorptions in the fuel to total cell absorptions as the fuel element temperature is increased. This brings about a shift in the core neutron balance, giving a loss of reactivity.



TRANSPORT CROSS SECTION FOR HYDROGEN
IN ZrH AND AVERAGE NEUTRON SPECTRA IN FUEL ELEMENT

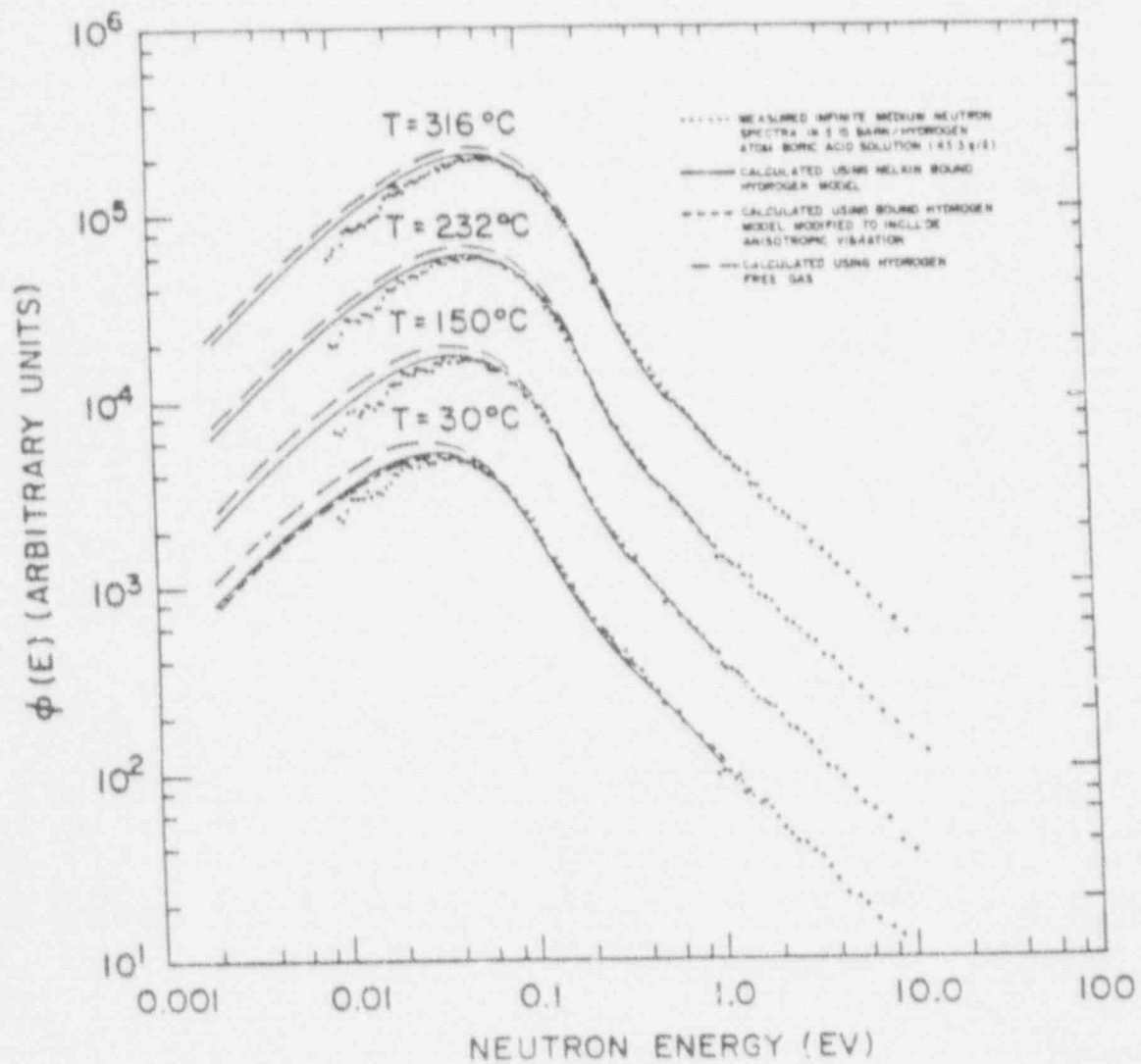
Figure 4-15

The temperature coefficient then, depends on spatial variations of the thermal neutron spectrum over distances of the order of a mean free path with large changes of mean free path occurring because of the energy change in a single collision. A quantitative description of these processes requires a knowledge of the differential slow neutron energy transfer cross section in water and zirconium hydride, the energy dependence of the transport cross section of hydrogen as bound in water and zirconium hydride, the energy dependence of the capture and fission cross sections of all relevant materials, and a multigroup transport theory reactor description which allows for the coupling of groups by speeding up as well as by slowing down.

4.1.2.1. Codes Used for Calculations. Computational work on the temperature coefficient made use of a group of codes developed by GA Technologies: GGC-3 [17], GAZE-2 [18], and GAMBLE-5 [19], as well as DTF-IV [20], an S_n multigroup transport code written at Los Alamos. Neutron cross sections for energies above thermal (≥ 1 eV) were generated by the GGC-3 code. In this code, fine group cross sections (~ 100 groups), stored on tape for all commonly used isotopes, are averaged over a space independent flux derived by solution of the B_1 equations for each discrete reactor region composition. This code and its related cross-section library predict the age of each of the common moderating materials to within a few percent of the experimentally determined values and use the resonance integral work of Adler, Hinman, and Nordheim [21] to generate cross sections for resonance materials which are properly averaged over the region spectrum.

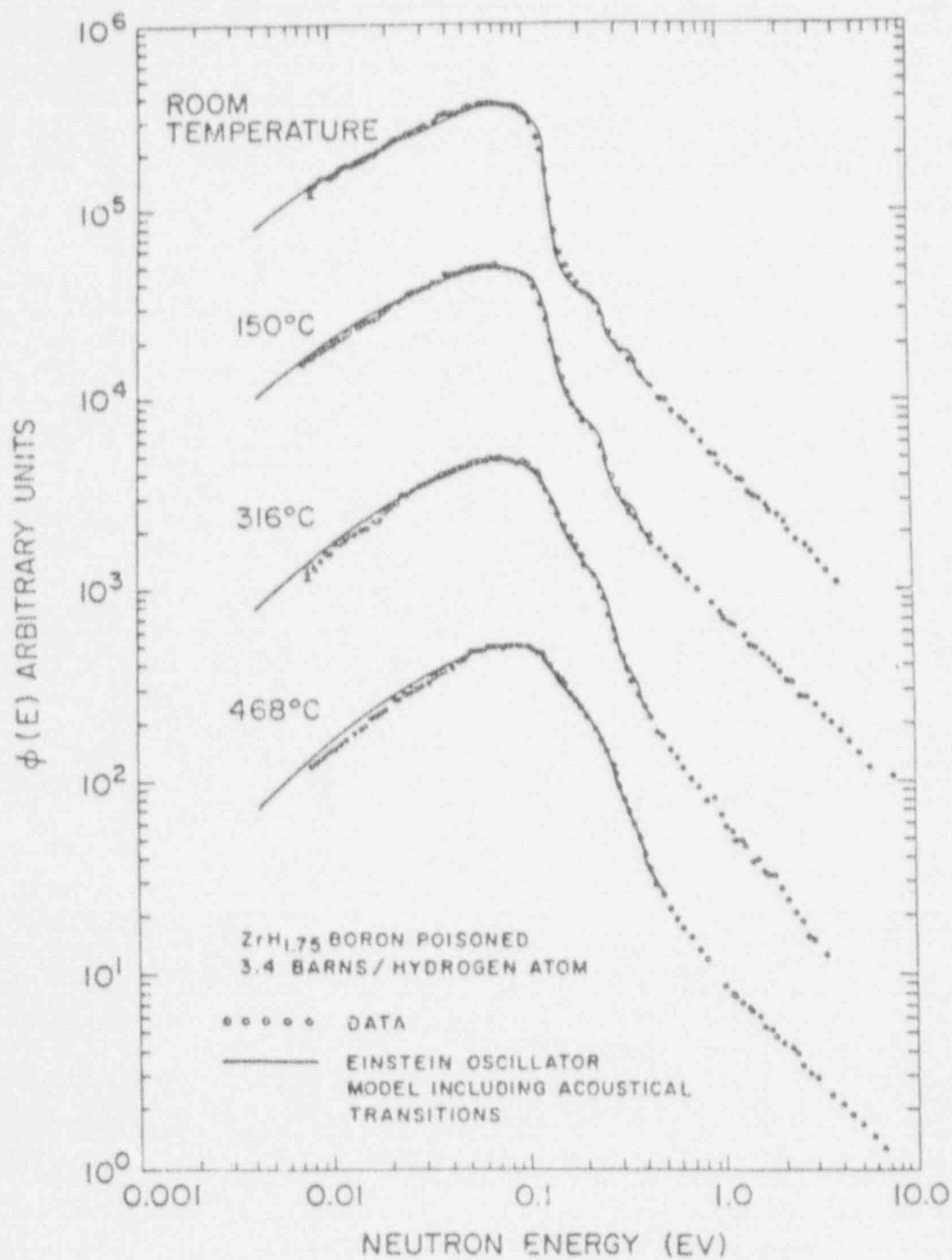
Thermal cross sections were obtained in essentially the same manner using the GGC-3 code. However, scattering kernels were used to describe properly the interactions of the neutrons with the chemically bound moderator atoms. The bound hydrogen kernels used for hydrogen in the water were generated by the THERMIDOR code [22] using thermalization work of Nelkin [23]. Early thermalization work by McReynolds *et al* [24] on zirconium hydride has been greatly extended at GA Technologies [25], and work by Parks resulted in the SUMMIT [26] code, which was used to generate the kernels for hydrogen as bound in ZrH. These scattering models have been used to predict adequately the water and hydride (temperature dependent) spectra as measured at the GA Technologies linear accelerator as shown in Figure 4-16 and Figure 4-17 [27].

4.1.2.2. ZrH Model. Qualitatively, the scattering of slow neutrons by zirconium hydride can be described by a model in which the hydrogen atom motion is treated as an isotropic harmonic oscillator with energy transfer quantized in multiples of ~ 0.14 eV. More precisely, the SUMMIT model uses a frequency spectrum with two branches, one for the optical modes for energy transfer with the bound proton, and the other for the acoustical modes for energy transfer with the lattice as a whole. The optical modes are represented as a broad frequency band centered at 0.14 eV, and whose width is adjusted to fit the cross section data of Woods *et al.* [28]. The low frequency acoustical modes are assumed to have a Debye spectrum with a cutoff of 0.02 eV and a weight determined by an effective mass of 360.



A COMPARISON OF NEUTRON SPECTRA
BETWEEN EXPERIMENTS AND SEVERAL HYDROGEN MODELS

Figure 4-16



EFFECT OF TEMPERATURE VARIATION ON
ZIRCONIUM HYDRIDE NEUTRON SPECTRA

Figure 4-17

This structure then allows a neutron to slow down by the transition in energy units of ~ 0.14 eV as long as its energy is above 0.14 eV. Below 0.14 eV the neutron can still lose energy by the inefficient process of exciting acoustic Debye type modes in which the hydrogen atoms move in phase with the zirconium atoms, which in turn move in phase with one another. These modes therefore, correspond to the motion of a group of atoms whose mass is much greater than that of hydrogen, and indeed even greater than the mass of zirconium. Because of the large effective mass, these modes are very inefficient for thermalizing neutrons, but for neutron energies below 0.14 eV they provide the only mechanism for neutron slowing down within the ZrH. (In a TRIGA core, the water also provides for neutron thermalization below 0.14 eV.) In addition, in the ZrH it is possible for a neutron to gain one or more energy units of ~ 0.14 eV in one or several scatterings, from excited Einstein oscillators. Since the number of excited oscillators present in a ZrH lattice increases with temperature, this process of neutron speeding up is strongly temperature dependent and plays an important role in the behavior of ZrH moderated reactors.

4.1.2.3. Calculations. Calculations of the temperature coefficient were done in the following steps:

- a. Multigroup cross sections were generated by the GGC-3 code for a homogenized unit cell. Separate cross-section sets were generated for each fuel element temperature by use of the temperature dependent hydride kernels and Doppler broadening of the U-238 resonance integral to reflect the proper temperature. Water at room temperature was used for all prompt coefficient calculations.
- b. A value for k_{∞} was computed for each fuel element temperature by transport cell calculations, using the P_1 approximation. Comparisons have shown S_4 and S_8 results to be nearly identical. Group dependent disadvantage factors were calculated for each cell region (fuel, clad, and water) where the disadvantage factor is defined as the ratio: Φ_g^r / Φ_g^c (region/cell).
- c. The thermal group disadvantage factors were used as input for a second GGC-3 calculation where cross sections for a homogenized core were generated which gave the same neutron balance as the thermal group portion of the discrete cell calculation.
- d. The cross sections for an equivalent homogenized core were used in a full reactor calculation to determine the contribution to the temperature coefficient due to the increased leakage of thermal neutrons into the reflector with increasing hydride temperature. This calculation still requires several thermal groups, but transport effects are no longer of major concern. Thus, reactivity calculations as a function of fuel element temperature have been done on the entire reactor with the use of diffusion theory codes.

Results from the above calculations indicate that more than 50% of the temperature coefficient for a standard TRIGA core comes from the temperature-dependent disadvantage factor or "cell effect", and ~20% each from Doppler broadening of the U-238 resonances and temperature dependent leakage from the core. These effects produce a temperature coefficient of $\sim -0.01\%/^{\circ}\text{C}$, which is rather constant with temperature. The temperature coefficient is shown in Figure 4-18 for the high-hydride core of this TRIGA.

4.1.3. Steady-State Reactor Power

The following evaluation has been made for a TRIGA system operating with cooling from natural convection flow around the fuel elements. This analysis investigates the limits to which such a system may be operated.

The analysis was conducted by considering the hydraulic characteristics of the flow channel from which the heat rejection rate is maximum. The geometrical data from this channel are given in Table 4-2. All symbols in Equation 16 through 45 are defined in the list of nomenclature in Section 4.1.3.9.

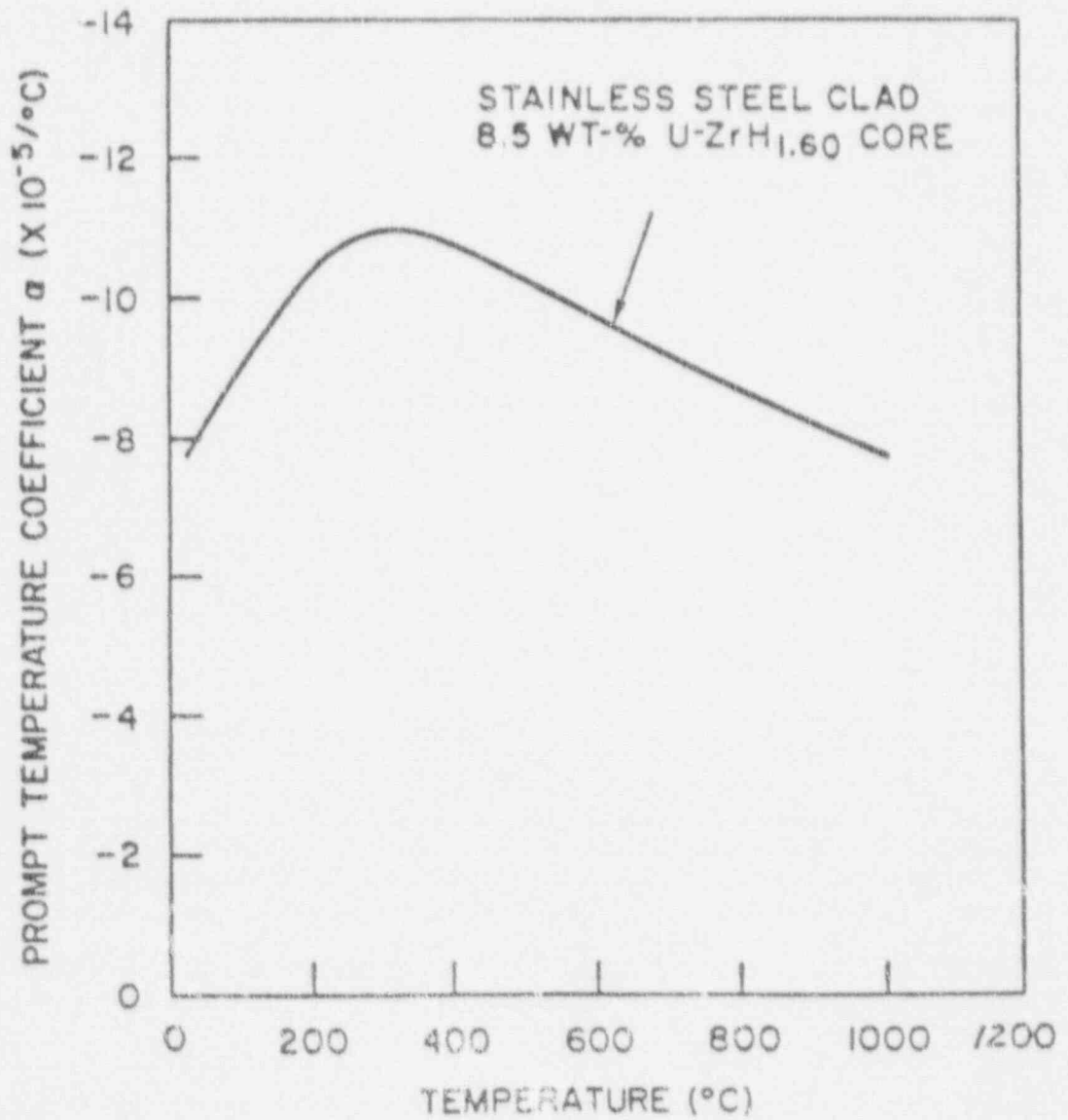
Table 4-2

HYDRAULIC FLOW PARAMETERS

Flow area ($\text{ft}^2/\text{element}$)	0.00580
Wetted perimeter ($\text{ft}/\text{element}$)	0.3861
Hydraulic diameter (ft)	0.0601
Fuel element diameter (ft)	0.1229
Fuel surface area (ft^2)	0.4826

The heat generation rate in the fuel element is distributed axially in a cosine distribution chopped at the end such that the peak-to-average ratio is 1.25. The number of fuel elements in the core is assumed for 1 MW operation, but the departure from nucleate boiling (DNB) ratio is conservatively evaluated on the basis of 85 elements.

The driving force is supplied by the buoyance of the heated water in the core. Countering this force are the contraction and expansion losses at the entrance and exits to the channel, and the acceleration and potential energy losses and friction losses in the cooling channel itself.



PROMPT NEGATIVE TEMPERATURE COEFFICIENT
VERSUS AVERAGE FUEL TEMPERATURE FOR TRIGA

Figure 4-18

Figure 4-19 illustrates schematically the natural convection system established by the fuel elements bounding one flow channel in the core. The system shown is general and does not represent any specific configuration. Steady-state flow is governed by the Equation

$$\delta p_i + \delta p_e + \delta p_f + \delta p_a + \sum_{j=1}^n \delta p_j = z_t/v_0 \quad (16)$$

where the left-hand member represents the pressure drops through the flow channel due to entrance, exit, friction, acceleration, and gravity losses and the right-hand member represents the driving pressure due to the static head in the pool. The pressure drops through the flow channel are dependent on the flow rate while the available static driving pressure is fixed for a known core height and pool temperature. The analysis, therefore, becomes an iterative one in which the left-hand side of Equation 16 is evaluated on the basis of an assumed flow rate and compared with the known right-hand side until equality is achieved. The method has been programmed for digital computer solution. The methods of evaluating each of the δp terms in Equation 16 for known power distribution and flow geometry and assumed flow rates are discussed below.

4.1.3.1. Entrance Loss, δp_i .

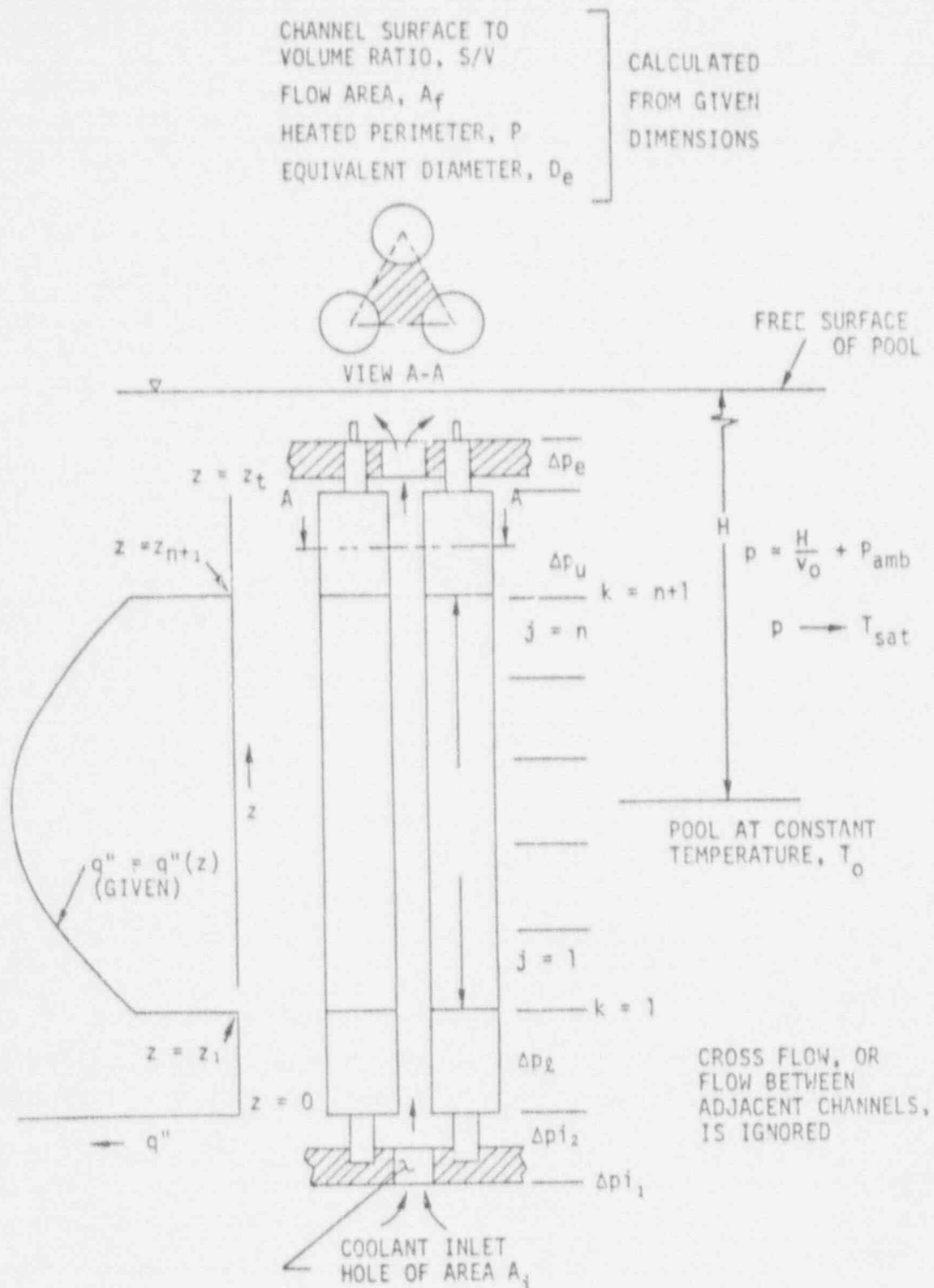
The entrance loss, δp_i , may be evaluated in the usual way as a fraction of the velocity head in the lower grid plate hole:

$$\delta p_i = \frac{(k_{i1} + k_{i2}) v_0}{2g A_1^2} (NW)^2 \quad (17)$$

where N = the number of channels which receive their flow from a single hole in the lower grid plate,

k_{i1} = the loss factor for the entrance to the hole in the lower grid plate. For even slight rounding of the entrance, k_{i1} will be no greater than 0.30,

k_{i2} = the loss factor covering transfer of the flow from the hole in the lower grid plate to the coolant channels. In most cases this can be satisfactorily approximated as a sudden expansion using $k_{i2} = 1.0$.



GENERAL FUEL ELEMENT CONFIGURATION FOR
 SINGLE COOLANT CHANNEL IN THE TRIGA

Figure 4-19

4.1.3.2. Exit Loss, δp_e .

The exit loss is expressed in terms of a coefficient K_e which is the fraction of the velocity head in the flow channel which is not recovered:

$$\delta p_e = \frac{K_e v_{n+1}}{2g A_f^2} W^2 \quad (18)$$

The term v_{n+1} is the specific volume at the highest axial station along the heated length of the core. It is evaluated from the temperature T_{n+1} which is obtained from an overall heat balance:

$$T_{n+1} = q_t / WC + T_o \quad (19)$$

$$\text{where } q_t = P \int_{z_1}^{z_{n+1}} q''(z) dz$$

4.1.3.3. Loss Through Portion of Channel Adjacent to Lower Reactor δp_l .

The flow is isothermal at the bulk pool temperature so that

$$\delta p_l = \frac{f_m v_o \delta z_l}{2g D_e A_f^2} W^2 + \frac{\delta z_l}{v_o} \quad (20)$$

f_m is evaluated from the Moody chart (assuming smooth surface) on the basis of a Reynolds number which is

$$Re = \frac{D_e v_o}{A_f \nu_o} W \quad (21)$$

4.1.3.4. Loss Through Portion of Channel Adjacent to Upper Reactor δp_u .

The flow is isothermal at T_{n+1} where T_{n+1} is determined by Equation 19 :

$$\delta p_u = \frac{f_m v_n \delta z_u}{2g D_e A_f^2} W^2 + \frac{\delta z_u}{v_n} \quad (22)$$

f_m is again evaluated from the Moody chart, assuming smooth surface, on the basis of a Reynolds number which is

$$Re = \frac{D_e v_n}{A_f \nu_n} W \quad (23)$$

4.1.3.5. Loss Through Each Increment of the Channel Adjacent to the Fueled Portion of the Elements, δp_j .

For the present, let us assume that the entire heated portion of the channel is in subcooled boiling. This implies that the wall temperatures calculated from subcooled boiling correlations are lower than those calculated for convection alone and that the liquid is below its saturation temperature at all locations. The pressure drop through an increment is given by

$$\delta p_n = (n+1) = \frac{v_{m_{k+1}} - v_{m_k}}{g A_f^2} W^2 + \frac{f_{bj} v_{mj} \delta z}{2g A_f^2 D_e} W^2 + \frac{\delta z}{v_{mj}} \quad (24)$$

(acceleration) (friction) (gravity)

4.1.3.6. Acceleration Term.

v_m denotes the mean specific volume and is larger than the liquid specific volume, because of the vapor voidage:

$$v_m = v/(1-\alpha) \quad (25)$$

α is the void fraction or the fraction of a channel cross section which is occupied by vapor. α may be calculated from the vapor volume (cubic in. vapor/square in. heating surface) and the flow channel geometry. Denoting the vapor volume as ξ ,

$$\alpha = \xi (S/V) \quad (26)$$

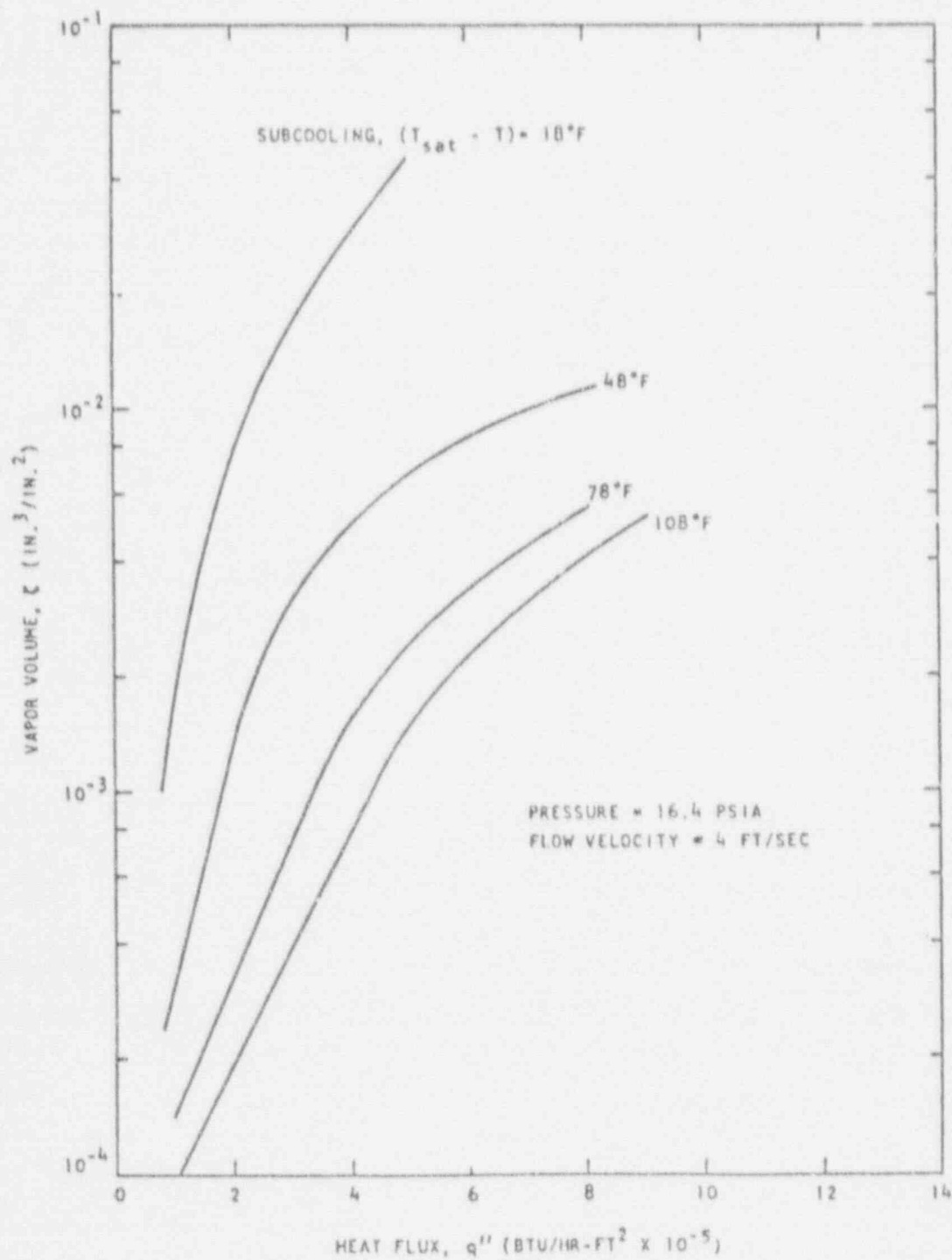
where S/V is the surface to volume ratio of the coolant channel. The parameter ξ , is dependent on the surface heat flux, the subcooling of liquid and the velocity of the liquid. It can be evaluated only by experiment. Data given by Jordan and Leppert [29] were used to estimate ξ ; these data are plotted in Figures 4-20 and 4-21. Most of this represents a flow velocity of 4 ft/sec and appears to be the only available data applicable under the thermal conditions encountered in TRIGA type reactors. Extrapolations from these data are made for flow velocities different from 4 ft/sec. The extrapolations were based on a small amount of data given for flow velocities other than 4 ft/sec. The liquid temperature at a station, T_k , may be calculated from

$$T_k = \frac{P \int_{z_1}^{z_k} q''(z) dz}{WC} + T_o \quad (27)$$

Therefore, one finds ξ (Figure 4-21) from $T_{sat} - T_k$ and q_k'' , where T_{sat} and q_k'' are known.

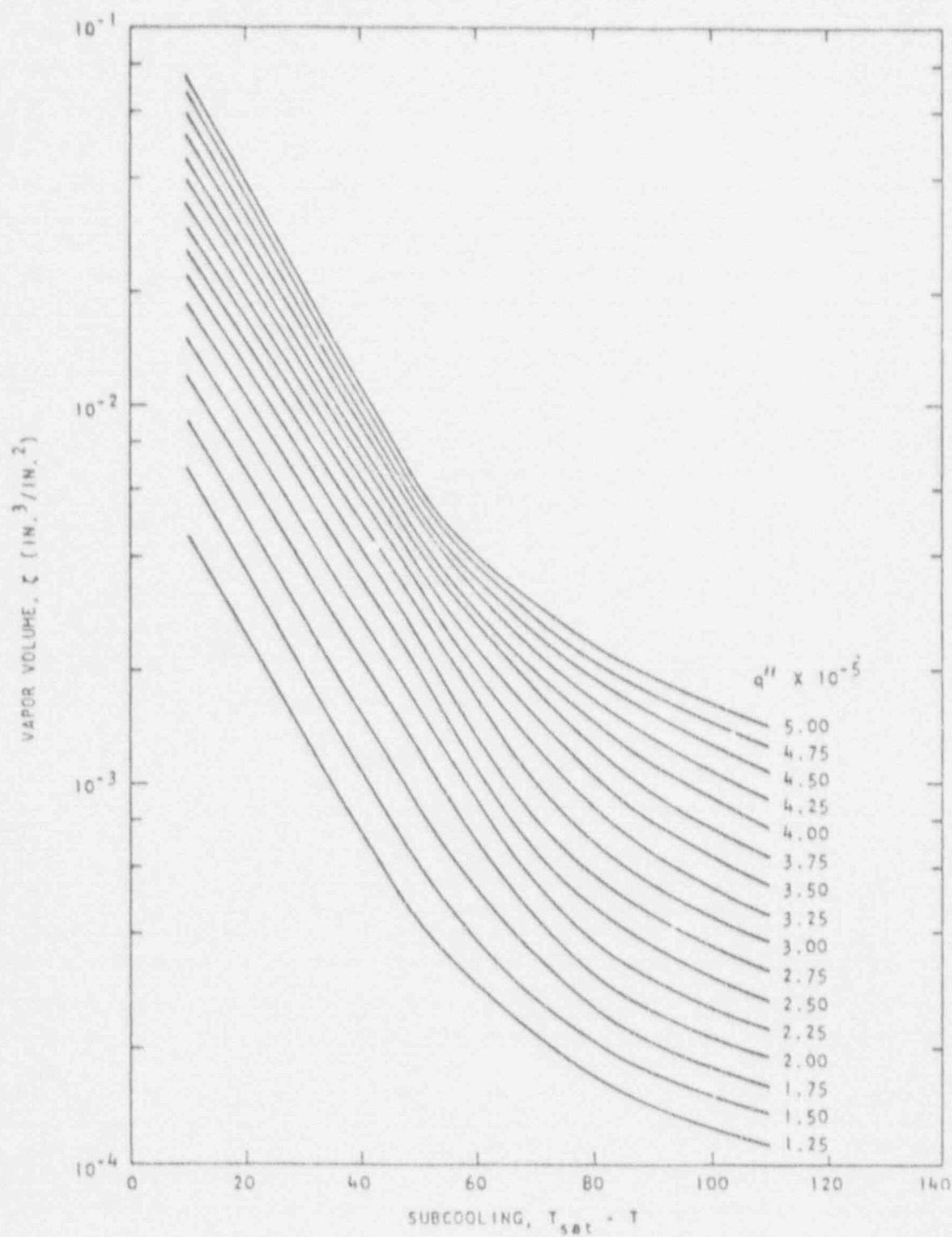
Since $\alpha_k = \xi_k (S/V)$

and v_k is a function of T_k , v_m may be evaluated from Equation 25.



EXPERIMENTALLY DETERMINED VAPOR VOLUMES FOR
SUBCOOLED BOILING IN A NARROW VERTICAL ANNULUS

Figure 4-20



CROSS PLOT OF Figure 4-20 USED IN CALCULATIONS

Figure 4-21

4.1.3.7. Friction Term.

v_{mj} denotes a linear average of the mean specific volumes at the upper and lower boundaries of an increment. The approximate mean value is assumed to apply over the entire increment so that

$$v_{mj} = \frac{v_{mk} + v_{mk+1}}{2} \quad (28)$$

A friction factor f_{bj} is applied locally to calculate the friction pressure drop over the increment in subcooled boiling. Jordan and Leppert develop the correlation

$$f_b = 8 S_t = \frac{8 h_b}{\rho CV} = \frac{8 q''}{\rho CV (T_w - T)} \quad (29)$$

and provide experimental verification near atmospheric pressure in the range $0.0015 < S_t < 0.0050$. This is simply an extension of Reynolds' analogy to the case of subcooled boiling. The Equation of continuity is used to write Equation 29 as

$$f_b = \frac{8 q'' A_f}{WC (T_w - T)} \quad (30)$$

which may be evaluated if T_w is known. For subcooled boiling, the heat transfer is usually defined by an experimentally determined correlation of q'' vs $(T_w - T_{sat})$ which has been obtained over a given range of flow velocity and pressure. McAdams [30] gives such a correlation for pressures between 2 and 6 atmospheres and flow velocities between 1 and 12 ft/sec. This correlation will be used to determine T_w for use in Equation 30.

Approximate mean values are assumed to apply over the entire increment so that

$$f_{bj} = 1/2 \frac{8 A_f}{WC} \left[\frac{q''_k}{T_{w,k} - T_k} + \frac{q''_{k+1}}{T_{w,k+1} - T_{k+1}} \right] \quad (31)$$

$$\text{and } T_w - T_{sat,j} = \frac{\phi(q''_k) + \phi(q''_{k+1})}{2}$$

where $\phi(q'')$ is the correlation of McAdams previously cited.

4.1.3.8. Gravity Term.

The gravity term is evaluated from v_j calculated from Equation 28.

As implied in Section 4.1.3.5., each increment must be checked to determine whether heat is being transferred by subcooled boiling or by convection. T_w is evaluated at the lower boundary of the increment on the basis of both the correlation from McAdams for subcooled boiling and a standard correlation for convection (Dittus-Boelter). If the T_w calculated from convection correlations is less than that obtained for subcooled boiling, boiling is assumed not to be present in the increment. Equation 24 still applies, but since there is no boiling and hence no vapor void, v_m becomes v and f_b becomes f_m .

In the foregoing analysis an assumption was made that all of the vapor formed on the surface of the fuel element detaches and adds to the fluid buoyancy. This is not a conservative assumption. The position where vapor bubbles first leave the heated surface is obtained from two considerations; first, the balance of the forces exerted on the vapor bubble while it is in contact with the wall (buoyancy, surface tension, and friction), and, second, the temperature distribution in the single phase liquid away from the walls.

Determination of the buoyance forces resulting from the formation and subsequent detachment of vapor bubbles is complicated by the difficulty in predicting the point at which the vapor detaches, and the fraction of that vapor which subsequently condenses. The problem was simplified by making use of an analysis performed by Levy [31] to determine the position at which the vapor detaches from the wall, assuming that at that point all of the vapor detaches and, finally, that there is no recombination of the vapor with subcooled fluid.

According to Levy the position at which the vapor leaves the surface is obtained from considering the balance of forces exerted on the vapor bubble while it is in contact with the wall, and the temperature distribution in the single phase liquid away from the wall.

The forces acting on the bubble in the vertical direction consist of a buoyant force, F_B ; a frictional force, F_F , exerted by the liquid on the bubble; and a surface tension force, F_S , vertical component.

The buoyant force, F_B , is given by

$$F_B = \frac{C_B r_B^3 (\rho_L - \rho_V)g}{g_c} \quad (32)$$

where r_B is the bubble radius, C_B is a proportionality constant, ρ_L and ρ_V are the liquid and vapor density, g is the acceleration due to gravity and g_c is a conversion ratio from lb-force to lb-mass. The frictional force, F_F , is related to the liquid frictional pressure drop per unit length, $(-dp/dz)_F$. The pressure differential across the bubble is proportional to the pressure differential times the bubble radius and it acts across an area proportional to the square of the bubble radius.

Relating the pressure differential to the wall shear stress τ_w by

$$-(dp/dz)_F = 4 \tau_w / D_H \quad (33)$$

there results for F_F :

$$F_F = C_F (\tau_w / D_H) r_B^3 \quad (34)$$

where C_F is a constant of proportionality and D_H is the hydraulic diameter (four times the cross-sectional area divided by the wetted perimeter). The surface tension force, F_S , is given by

$$F_S = C_S r_B \sigma \quad (35)$$

where C_S is a proportionality constant and σ is the surface tension. Assuming upward flow the balance of these forces results in the following solutions for the bubble radius:

$$r_B = \left[\frac{C_S \sigma}{C_B (g/g_c) (\rho_L - \rho_V) + C_F (\tau_w / D_H)} \right]^{1/2} \quad (36)$$

Assuming that the distance from the wall to the tip of the bubble is proportional to the bubble radius, a non-dimensional distance corresponding to this real distance can be given by

$$Y_B = \frac{(\sigma g_c D_H - \rho_L)^{1/2}}{\mu_L} \left[1 + C' \frac{g}{g_c} \frac{(\rho_L - \rho_V) D_H}{\tau_w} \right]^{-1/2} \quad (37)$$

where C and C' are appropriate constants. For those cases where the fluid forces are considerably greater than the buoyant forces this expression reduces to

$$Y_B = C (\sigma g_c D_H - \rho_L)^{1/2} / \mu_L \quad (38)$$

For the bubble to detach, the fluid temperature at the tip of the bubble must exceed the saturation temperature by an amount such that the pressure differential acting across the interface at the tip of the bubble balances the surface tension forces at the same position. By using the Clausius-Clapeyron solution of this pressure differential one finds that the fluid temperature-saturation temperature difference can be assumed to be zero.

The temperature at the tip of the bubble can also be specified from existing solutions for the fluid temperature distribution. Thus, if the flow is assumed to be turbulent, and using the solution proposed by Martinelli, we have

$$\begin{aligned} T_w - T_B &= \theta Pr Y_B ; & 0 \leq Y_B \leq 5 & \quad (39) \\ &= 5\theta (Pr + \ln [1 + (Y_B/5 - 1)]) ; & 5 \leq Y_B \leq 30 \\ &= 5\theta (Pr + \ln [1 + 5 Pr] + 0.5 \ln [Y_B/30]) ; & Y_B \geq 30. \end{aligned}$$

The parameter θ is a non-dimensional term defined through the heat flux and liquid specific heat, that is,

$$\theta = \frac{q / A}{\rho_L C_{pL} (\tau_w E_c / \rho_L)^{1/2}} \quad (40)$$

Levy obtained values for the constants C and C' by correlation with available experimental data. Using the accepted heat-transfer relation from Dittus-Boelter, one obtains

$$hD_H/k_L = 0.023 (WD_H/\rho_L)^{0.8} (Pr)^{0.4} \quad (41)$$

Calculating the friction factor from

$$f = 0.0055(1 + [20,000(e/D_H) + 10^6/(WD_H/\mu_L)]^{1/3}), \quad (42)$$

we are able to find the wall shear stress from

$$\tau_w = (f/8) (W^2/\rho_L E_c) \quad (43)$$

The correlation with experiment yielded values for the constants of

$$C = 0.015 \quad (44)$$

$$C' = 0$$

Finally, from the definition of the heat transfer coefficient, one obtains

$$T_w - T = q''/h \quad (45)$$

and setting the bubble tip temperature, T_B , equal to the saturation temperature, T_{sat} , we can express the relationship between the saturation temperature, the wall temperature, and the fluid temperature at which the bubble would detach from the wall by

$$(T_w - T_{sat})/(T_w - T) = 0.023 (WD_H/\mu_L)^{-0.2} (Pr)^{-0.6} (f/8)^{-0.5} \Omega, \quad (46)$$

$$\text{where } \Omega = Pr Y_B ; \quad 0 \leq Y_B \leq 5$$

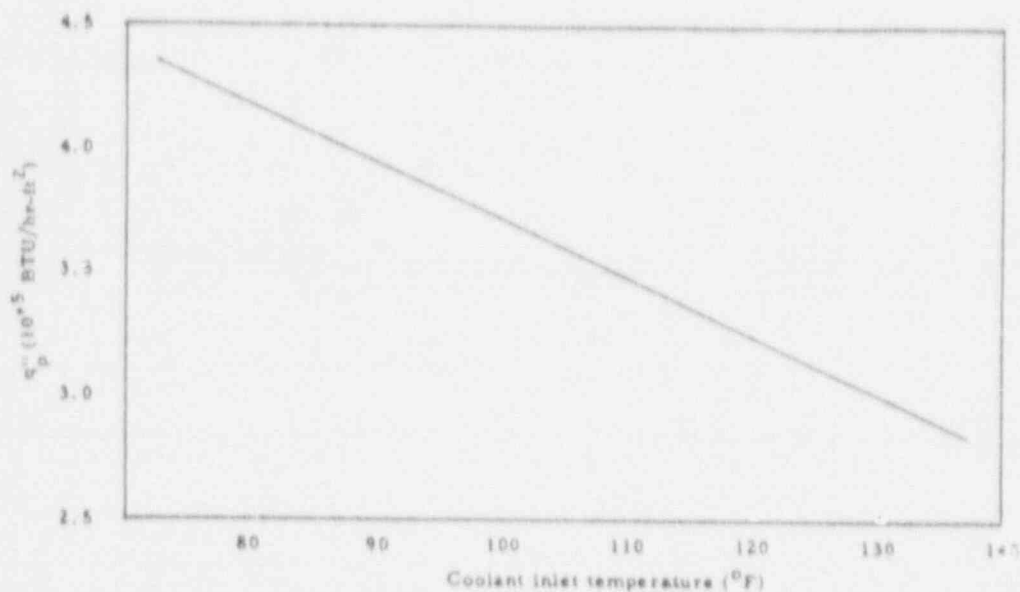
$$= 5 (Pr + \ln [1 + Pr (0.2 Y_B - 1)]) ; \quad 5 \leq Y_B \leq 30$$

$$= 5 (Pr + \ln (1 + 5Pr) + 0.5 \ln (Y_B/30)) ; \quad Y_B \geq 30.$$

The solution of the force balance equation with void detachment was accomplished by iterating on the void detachment point to find where the right and left sides of Equation 46 were equal. The point at which the void was assumed to separate from the surface was taken as the point at which equality obtained.

The peak heat flux, that is, the heat flux at which there is a departure from nucleate boiling and the transition to film boiling begins, was determined by two correlations. The first, given by McAdams [32], indicates that the peak heat flux is a function of the fluid velocity and the fluid only. The second correlation is due to Bernath [33]. It encompasses a wider range of variables over which the correlation was made and it takes into account the effect of different flow geometries. It generally gives a lower value for the peak heat flux and is the value used here for determining the minimum DNB ratio, that is, the minimum ratio of the local allowable heat flux to the actual heat flux. In general, the McAdams correlation gives a DNB ratio 50% to 80% higher than the Bernath correlation.

Figure 4-22 shows the results of this analysis. Here we have plotted the maximum channel heat flux for which the DNB ratio is 1, with bulk pool water temperature as a parameter. It is assumed that all the vapor above the detachment point separates from the heated surface. From the figure it can be seen that with the design cooling water temperature at the core inlet (120°F) the maximum heat flux is 325 kBTU/hr-ft². For a 85 element core with an overall peak-to-average power density ratio of 2.0, this heat flux corresponds to a maximum reactor power of 1900 kW.



PLOT FOR WHICH DNB RATIO IS 1.0 OF
MAXIMUM HEAT FLUX VERSUS COOLANT TEMPERATURE

Figure 4-22

4.1.3.9. Nomenclature

A	cross-sectional area, ft^2
A_f	channel free flow area, ft^2
C	coolant specific heat, $\text{Btu/lb}\cdot^\circ\text{F}$
d	diameter, in.
D_e	channel equivalent diameter, ft
D_H	hydraulic diameter, ft
f_b	friction factor with subcooled boiling, dimensionless
f_m	friction factor without boiling, dimensionless
F	forces acting on vapor bubble
g	constant, $4.18 \times 10^8 \text{ ft/hr}^2$
h_b	heat transfer coefficient with subcooled boiling, $\text{Btu/hr}\cdot\text{ft}^2\cdot^\circ\text{F}$
H	distance from midplane of heated channel to free surface of pool, ft
K	pressure loss factor at channel inlet or exit, dimensionless
n	number of equal axial increments into which heated length of core is subdivided
N	Number of channels which receive their flow from a single opening in the lower grid plate
p	absolute pressure, lb/ft^2
P	heated perimeter of channel, ft
Pr	Prandtl number
Δp	pressure loss, lb/ft^2
q	heat load, Btu/hr
q_t	total heat load to channel, Btu/hr
q''	heat flux, $\text{Btu/hr}\cdot\text{ft}^2$
q_p''	peak or "burnout" heat flux, $\text{Btu/hr}\cdot\text{ft}^2$
r_b	bubble radius
Re	Reynolds number, dimensionless
S/V	channel surface to volume ratio, in.^{-1}
T	coolant temperature, $^\circ\text{F}$
T_{sat}	coolant saturation temperature, $^\circ\text{F}$
v	specific volume, ft^3/lb
V	flow velocity, ft/hr
W	mass flow rate, lb/hr

Y	non-dimensional radius
z	axial coordinate in channel, ft
z _t	total length of channel, ft
δz	length of a calculation increment in the channel, ft
μ	dynamic viscosity, ft-lb/hr
α	void fraction or fraction of a channel cross section which is occupied by vapor, dimensionless
σ	surface tension, lb/ft
ξ	vapor volume, or volume of vapor produced per unit area of heated surface, cubic in./square in.
ν	kinematic viscosity, ft ² /hr
τ	shear stress, lb/ft ²
ρ	density, lb/ft ³
ε/D _e	relative roughness

Subscripts

e	conditions at channel exit
i	conditions at channel entrance or inlet
l	conditions in portion of channel adjacent to lower reflector
m	conditions averaged over the liquid and vapor phases
o	bulk pool conditions
u	conditions in portions of channel adjacent to upper reflector
j	axial increment index
k	axial station index
w	conditions at cladding outer surface
v	vapor
L	liquid

4.2. NUCLEAR DESIGN AND EVALUATION

The characteristics and operating parameters of this reactor have been calculated and extrapolated using experience and data obtained from existing TRIGA reactors as bench marks in evaluating the calculated data. There are several TRIGA systems with essentially the same core and reflector relationship as this TRIGA so the values presented here are felt to be accurate to within 5% but, of course, are influenced by specific core configuration details as well as operational details.

An operational core of 86 fuel elements, 3 fuel followed control rods, and one air followed control rod is to be arranged in 5 rings with a central, water filled hole. Dimension of the active fueled core, a cylinder, is 15 inches in height with an average radius of 8.6 inches. The cylinder radius is calculated as the average radius of a hexagonal fuel array with 91 unit cells (6 filled rings including the center hole) and a unit cell area of 2.55 square inches.

Table 4-3 summarizes the typical Mark II TRIGA reactor parameters for a core containing high-hydride, stainless steel clad fuel elements.

Table 4-3

TYPICAL TRIGA CORE NUCLEAR PARAMETERS

Fuel elements	SS-clad U-ZrH _{1.6}
Cold clean critical loading	-64 elements -2.5 kg U-235
Operational loading	-90 elements -3.4 kg U-235
ℓ , Prompt neutron lifetime	41 μ sec
β , Effective delayed neutron fraction	0.0070
α , Prompt negative temperature coefficient	-1.0 x 10 ⁻⁴ $\delta k/k^\circ C$
T _f Average fuel temperature (1.1 MW)	265°C
T _w Average water temperature (1.1 MW)	65°C
Water coolant volume to cell volume ratio	-1/3

4.2.1. Reactivity Effects

The reactivity associated with the control rod is of interest both in the shutdown margin and in calculations of possible abnormal conditions related to reactivity accidents. Table 4-4 gives approximate reactivity values associated with a total control rod travel of 15 in. (38.1 cm), the full travel in the core.

Table 4-4

ESTIMATED CONTROL ROD NET WORTH

Control Rod	diameter in. (cm)	$\delta k/k$ %
C ring - transient	1.25 (3.18)	2.1
C ring - regulating	1.35 (3.43)	2.6
D ring - shim 1	1.35 (3.43)	2.0
D ring - shim 2	1.35 (3.43)	2.0

The maximum reactivity insertion rate is that associated with the transient rod which can be fully removed from the core in 0.1 sec producing an average reactivity insertion rate of 21% $\delta k/k$ -sec.

The total reactivity worth of the control system is about 8.7%. With a core excess reactivity of 4.9%, the shutdown margin with all rods down is about 3.8% and with the most reactive rod stuck out is about 1%.

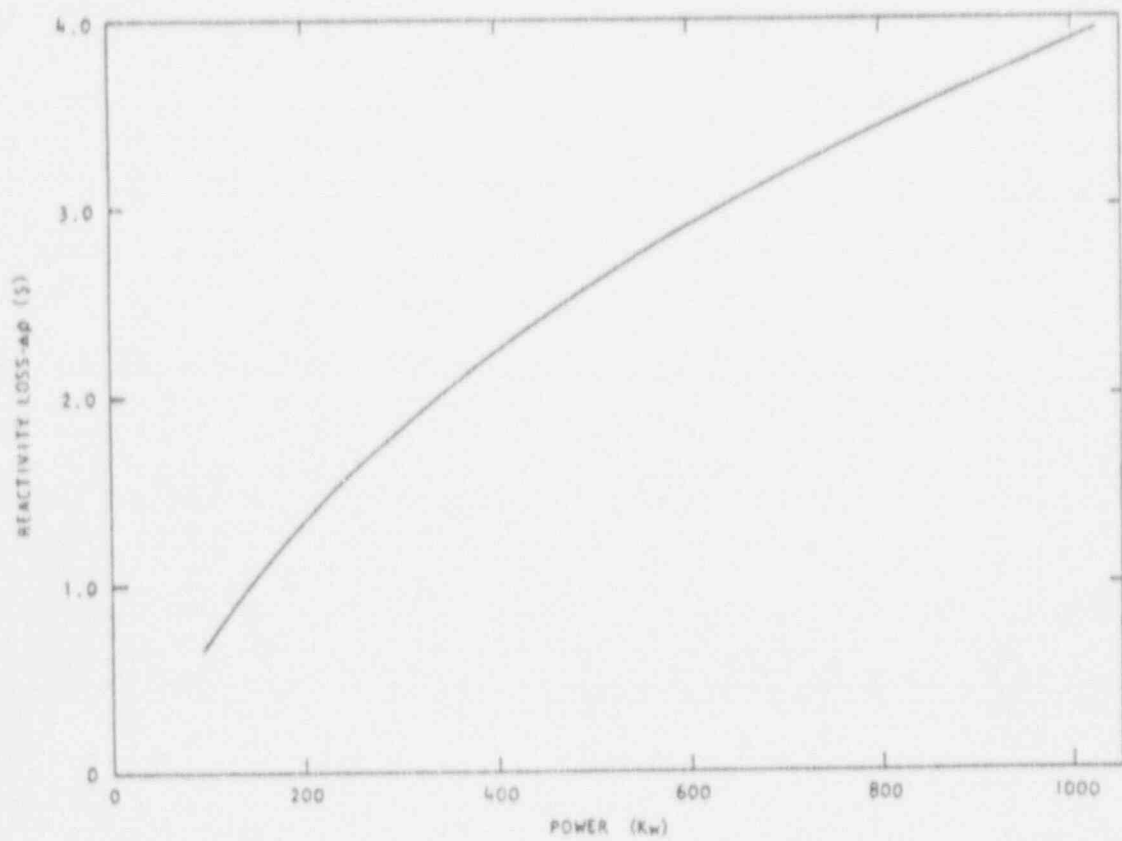
The reactivity worth of the fuel elements is dependent on their position within the core. Table 4-5 indicates the values that are expected in this installation.

Table 4-5

ESTIMATED FUEL ELEMENT REACTIVITY WORTH COMPARED WITH WATER AS A FUNCTION OF POSITION IN CORE

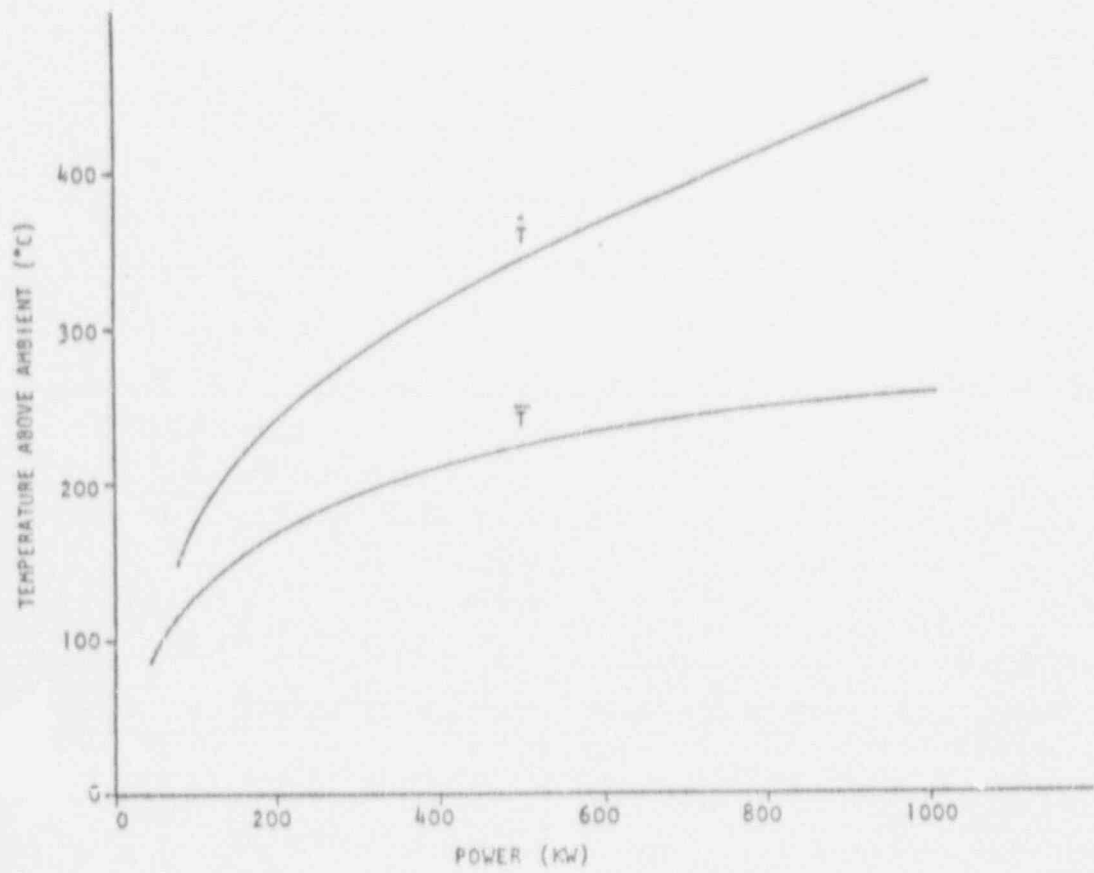
Core Position	Worth (% $\delta k/k$) SS Clad U-ZrH _{1.6}	Number of Fuel Positions
B ring	1.07	6
C ring	.85	12
D ring	0.54	18
E ring	0.36	24
F ring	0.25	30
G ring	0.19	36

Because of the prompt negative temperature coefficient a significant amount of reactivity is needed to overcome temperature and allow the reactor to operate at the higher power levels in steady-state operation. Figure 4-23 shows the relationship of reactor power level and associated reactivity loss to achieve a given power level. Figure 4-24 relates fuel temperature to a given steady-state reactor power level.



ESTIMATED REACTIVITY LOSS VERSUS POWER

Figure 4-23



ESTIMATED MAXIMUM B RING AND
AVERAGE CORE TEMPERATURE VERSUS POWER

Figure 4-24

The reactivity effects associated with the insertion and removal of experiments in or around the core are difficult to predict; however, Table 4-6 is supplied to provide a guide to the magnitude of the reactivity effects associated with the introduction of experiments in the reactor core.

Table 4-6
EXPECTED REACTIVITY EFFECTS ASSOCIATED WITH EXPERIMENTAL FACILITIES

	Worth (% $\delta k/k$)
Central thimble, fuel vs H ₂ O	+0.90
Central thimble, void vs H ₂ O	-0.15
Pneumatic transfer tube, (C ring) void vs H ₂ O	-0.10
Rotary specimen rack, void vs H ₂ O	-0.20

4.2.2. Evaluation of Nuclear Design

The TRIGA reactor system is well-known for its conservative design. The stability of this reactor type has been proven both through calculations as well as through tests performed with the many TRIGA reactors in operation throughout the world. The stability of the TRIGA type reactor stems from the prompt negative temperature coefficient associated with the U-ZrH_x fuel in conjunction with a suitable neutron thermalizing material. This TRIGA will have the stability that has been demonstrated on other TRIGA systems over the years.

A review of the reactivity worths associated with the reactor core indicates that no single item listed can produce a step reactivity insertion greater than that offered by routine pulse operation. In the pulsed mode of operation the results of a step insertion of \$3.00 are far below those attributed to test pulses on the advanced TRIGA prototype reactor in which 3.5% $\delta k/k$ was inserted in a step as is shown in Table 4-7.

A total reactivity limit for experiments is set at \$3.00. This limit constrains the worst case transient accident to less than the design pulse insertion.

The possibility of a reactivity accident which could produce reactor powers and fuel temperatures attributed to a \$4.00 step insertion has been considered and evaluated in the accident analysis section of this report. It is concluded from this analysis that the peak and average fuel temperatures resulting from this accident are well below the temperatures indicated as safety limits described in the reactor design bases of this document. It is further concluded that the integrity of the fuel containment will not be jeopardized and no adverse effects to the reactor system or personnel will arise from the advent of such an accident.

Table 4-7

COMPARISON OF REACTIVITY INSERTION EFFECTS

	Pulse Resulting from Insertion of Maximum Excess Reactivity in This TRIGA	Max Pulse Tested on SS-Clad, High Hydride Fueled TRIGA types
Reactivity insertion, \$	3.00	5.00
Steady-State power before pulse, kW	< 1	< 1
Peak power, MW	~1400	~8400
Total energy release, MW-sec	~18	~54
Period, msec	~3.1	~1.4
Maximum fuel temperature, °C	~540	~1050
Pulse width, msec	~11	~5.5

4.3 THERMAL AND HYDRAULIC DESIGN

This TRIGA reactor will be operated with natural convective cooling by reactor pool water. This method of heat dissipation is more than adequate for the power level of the reactor; i.e. 1100 kW(t). The thermal and hydraulic design of the reactor is well within the safety limits required to assure fuel integrity.

4.3.1. Design Bases

The thermal and hydraulic design for this TRIGA is based on assuring that fuel integrity is maintained during steady-state and pulsed mode operation as well as during those abnormal conditions which might be postulated for reactor operation. During steady-state operation fuel integrity is maintained by limiting reactor powers to values which assure that the fuel cladding can transfer heat from the fuel to the reactor coolant without reaching fuel-clad temperatures that could result in clad rupture. If these temperature conditions were exceeded, the maximum local heat flux in the core would be greater than the heat flux at which there is a departure from the nucleate boiling regime and consequently film blanketing of the fuel. This heat flux safety limit is a function of the inlet coolant temperature.

Figure 4-22 summarizes the results of the thermal and hydraulic analysis for steady-state operation of the TRIGA. In the figure critical heat flux for departure from nucleate boiling is plotted as a function of the coolant inlet temperature. The maximum power density in a TRIGA core is found by multiplying the average power density by a radial peak-to-average power generation ratio of 1.6 and an axial value of 1.25.

The correlation used to determine the heat flux at which there is a departure from nucleate boiling is from Bernath [33]. This correlation encompasses a wider range of experimental data than the usual correlations, e.g., the correlation due to McAdams, and, generally gives a lower value for the DNB ratio than the other correlations.

Table 4-8

1000 kW(t) TRIGA HEAT TRANSFER AND HYDRAULIC PARAMETERS

Number of fuel elements	90
Diameter	1.475 in.
Length (heated)	15.0 in.
Flow area	0.522 ft ²
Wetted perimeter	34.75 ft
Hydraulic diameter	0.0601 ft
Heat transfer surface	43.44 ft ²
Inlet coolant temperature	120°F (48.9°C)
Exit coolant temperature (average)	174°F (78.9°C)
Coolant mass flow	63,700 lb/hr
Average flow velocity	0.55 ft/sec
Average fuel temperature	500°F (260°C)
Maximum wall temperature	280°F (138°C)
Maximum fuel temperature	842°F (450°C)
Average heat flux	78,600 Btu/hr-ft ²
Maximum heat flux	157,100 Btu/hr-ft ²
Minimum DNB ratio	2.0

Table 4-8 lists the pertinent heat transfer and hydraulic parameters for the TRIGA operating at a nominal power level of 1000 kW. These data were taken from the results of calculations described in Section 4.1.

During pulsing operation the limiting thermal-hydraulic condition is the fuel temperature and the corresponding H_2 excess pressure beyond which clad rupture may occur. As indicated in Section 4.1, coolant temperature is not a limiting condition in pulsing since heating conditions are essentially adiabatic and significant transfer of heat energy to the coolant does not occur until after peak fuel-clad temperatures occur.

The safety limit on fuel temperature occurring in the pulse mode of operation is 1150°C . This temperature will give an internal equilibrium hydrogen pressure (U-ZrH fuel H/Zr; 1.6) less than that which would produce a stress equivalent to the ultimate strength of the clad at a temperature of 680°C . This clad temperature is higher than would actually occur and therefore conservative even in the case of a pulse producing a peak adiabatic fuel temperature of 1150°C .

4.3.2. Thermal and Hydraulic Design Evaluation

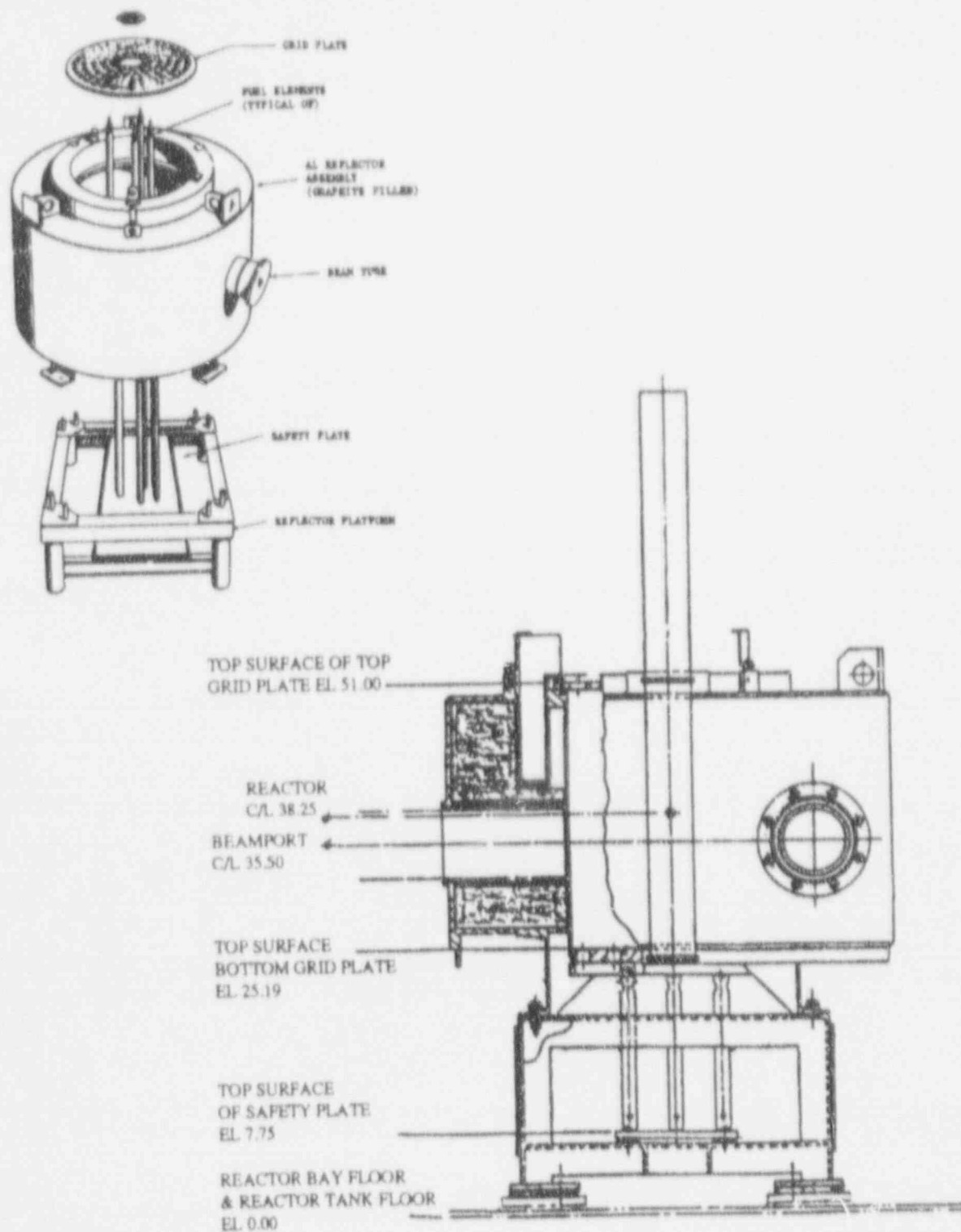
The validity and safety of the TRIGA thermal-hydraulic design is established in Section 4.1. In that section it is shown that design-basis conditions evaluated for TRIGA reactors using stainless steel clad U-ZrH (H/Zr; 1.6) fuel elements provide a generous safety margin for this TRIGA. These general evaluations are supported by extensive experience in operation of TRIGA cores at equivalent fuel temperatures and reactor power levels. No adverse results are reported from other similar TRIGA reactors with comparable fuel temperatures and power levels.

4.4. MECHANICAL DESIGN AND EVALUATION

4.4.1. General Description

The TRIGA Mark II reactor core assembly is located near the bottom of an elongated cylindrical aluminum tank surrounded by a reinforced concrete structure. A typical installation is shown in Figure 4-25. The standard reactor tank is a welded aluminum vessel with 1/4 in. thick (0.64 cm.) walls, a diameter of approximately 6.5 feet (2 meters), and a depth of at least 25 feet (7.6 m.). The tank is air-welded for water tightness. The integrity of the weld joints is verified by radiographic testing, dye penetrant checking, and helium leak testing. The outside of the tank is coated for corrosion protection.

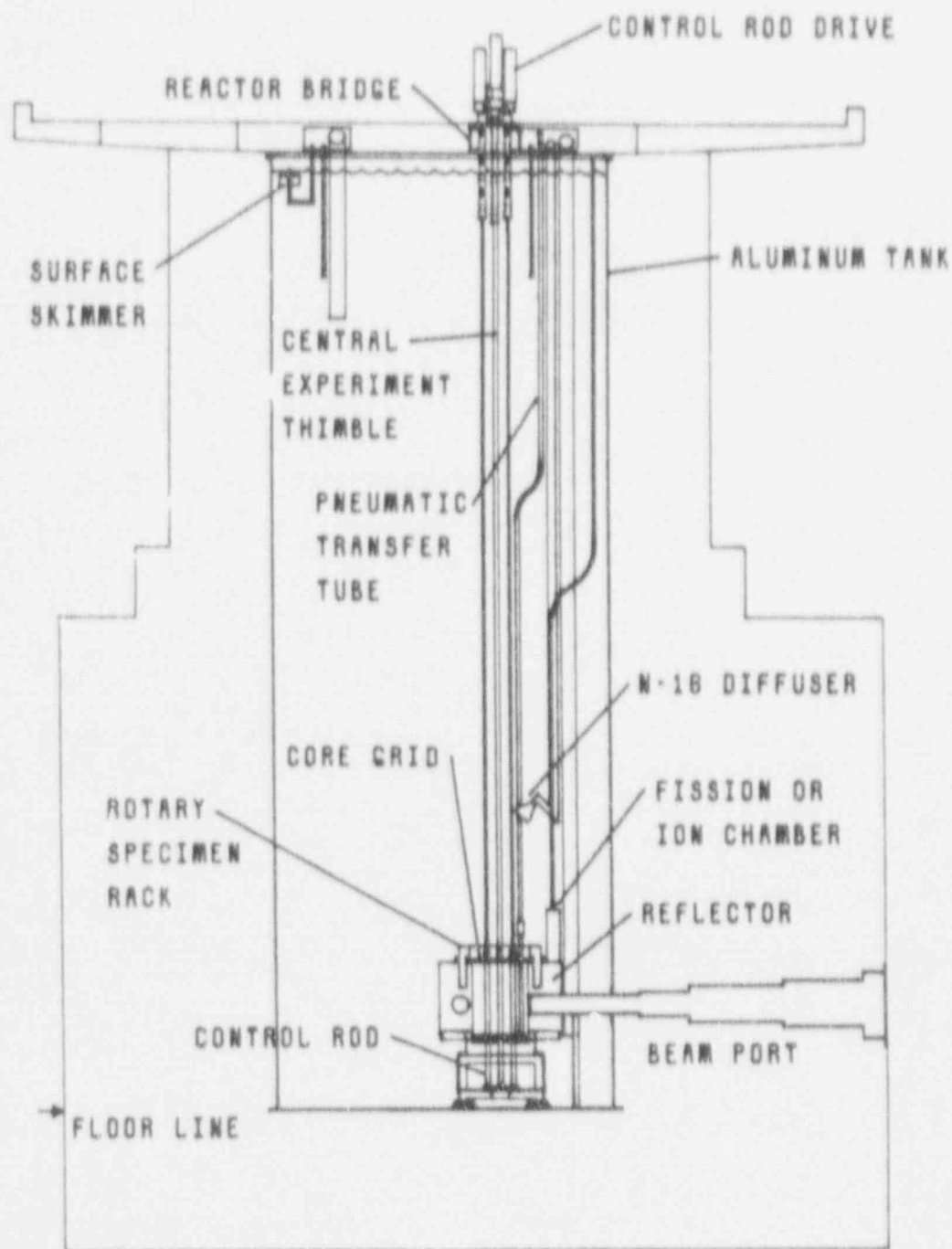
An aluminum angle used for mounting the ion chambers, fuel storage racks, underwater lights, and other equipment, is located around the top of the tank. Demineralized water in the tank is provided such that at least 21 feet (6.4 m.) of shielding water is above the core. The core is shielded radially by a minimum of 7.97 feet (2.43 m) of concrete with a density of 2.88 gm/cc, 1.5 feet (~45 cm.) of water, and 10.2 inches (25.9 cm.) of graphite reflector (see Figure 4-26).



TRIGA reactor - elevation view

REACTOR, REFLECTOR, AND SHIELDING

Figure 4-25



TRIGA MARK II REACTOR

Figure 4-26

4.4.2. Reflector Assembly

The reflector is a ring-shaped block of graphite that surrounds the core radially. The graphite is 10.2 inches (25.91 cm.) thick radially, with an inside diameter of 21-5/8 inches (54.93 cm.) and a height of 21-13/16 inches (54.40 cm.) The graphite is protected from water penetration by a leak-tight welded aluminum can.

A "well" in the top of the graphite reflector is provided for the rotary specimen rack. This well is also aluminum-lined, the lining being an integral part of the aluminum reflector can. The rotary specimen rack is a self-contained unit and does not penetrate the sealed reflector at any point.

The graphite, and outer surface of the aluminum can are pierced by an aluminum tube which forms the inner section of beam ports that penetrate the reflector. The reflector penetrating beam tubes are connected by aluminum couplings to the corresponding beam tube section fabricated as part of the reactor tank structure.

The reflector assembly rests on an aluminum platform at the bottom of the tank, and provides support for the two grid plates and the safety plate. Three lugs are provided for lifting the assembly.

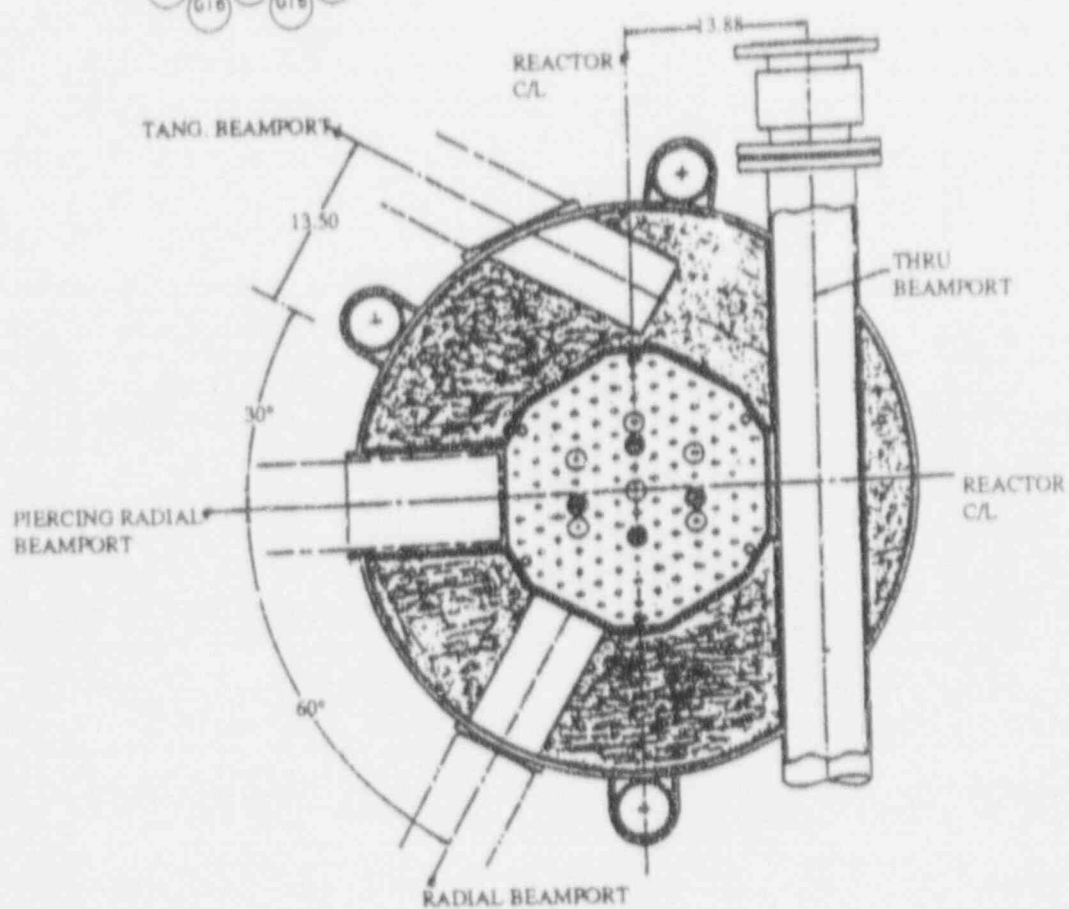
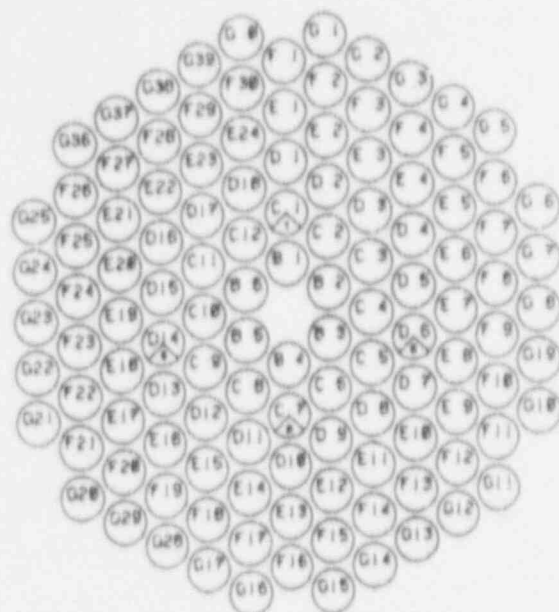
4.4.3. Grid Plates

The top grid plate is an aluminum plate 5/8 inches (1.59 cm.) thick (3/8 inches, 0.95 cm., thick in the central region) that provides accurate lateral positioning for the core components. The plate is supported by a ring welded to the top inside surface of the reflector container and is anodized to resist wear and corrosion.

One hundred twenty one (121) holes, 1.505 inches (3.823 cm.) diameter, are drilled into the top grid plate in a modified hexagonal pattern (the vertexes of the hex are omitted) around a central hole. The holes are to locate the fuel-moderator and graphite dummy elements, the control rods and guide tubes, and the pneumatic transfer tube. (See Figure 4-27.) An equivalent diameter center hole accommodates the central thimble. Small holes at various positions in the top grid plate permit insertion of wires or foils into the core to obtain flux data.

A hexagonal section can be removed from the center of the upper grid plate for the insertion of specimens up to 4.4 inches (11.18 cm.) in diameter into the region of highest flux; this requires prior relocation of the six fuel elements from the B ring to the outer portion of the core and removal of the central thimble. This removable section will not be used initially; a license amendment will be obtained prior to its use.

Two generally triangular-shaped sections are cut out of the upper grid plate. Each cut out encompasses two E and one D ring holes. Fuel elements placed in these locations, require lateral support by a special fixture. When the fuel elements and support are removed, there is room for inserting specimens up to 2.4 inches (6.1 cm.) in diameter.



TRIGA reactor - cross section through beamports

CORE ARRANGEMENT

Figure 4-27

The bottom grid plate is an aluminum plate $3/4$ inch (1.91 cm.) thick which supports the entire weight of the core and provides accurate spacing between the fuel-moderator elements. Six pads around a hexagonal ring which is welded to the reflector container support the bottom grid plate.

Holes in the bottom grid plate are aligned with fuel element holes in the top grid plate. They are countersunk to align the adaptor end of the fuel-moderator elements and the adaptor end of other in core experiment systems, such as the pneumatic transfer tube.

The differential area between the fitting at the top of the fuel elements and the round holes in the top grid plate permits passage of cooling water through the plate.

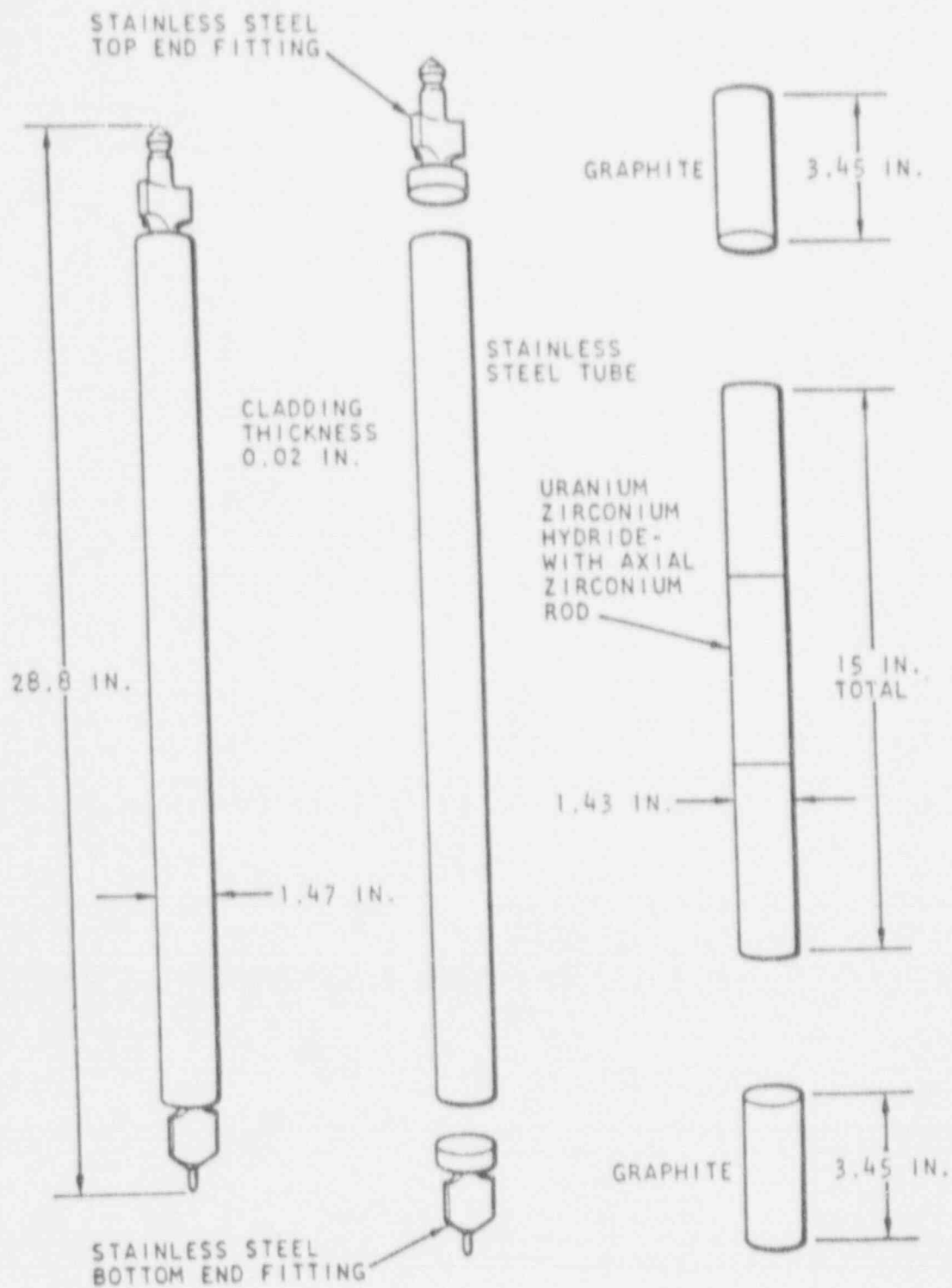
4.4.4. Safety Plate

The safety plate is provided to preclude the possibility of control rods falling out of the core. It is a $1/2$ inch (1.27 cm.) thick plate of aluminum set on the core support structure below the reflector. The plate is placed about 16 inches (40.6 cm.) below the bottom grid plate.

A central hole of 1.505 inches (3.823 cm.) in diameter in the lower grid serves as a clearance hole for the central thimble. Eight additional 1.505-inch (3.823 cm.) diameter holes are aligned with upper grid plate holes to provide passage of fuel-follower control rods. Those holes in the bottom grid plate not occupied by control rod followers are plugged with removable fuel element adaptors that rest on the safety plate. These fuel element adaptors are solid aluminum cylinders 1.5 inches (3.81 cm.) in diameter by 17 inches (43.18 cm.) long. At the lower end is a fitting that is accommodated by a hole in the safety plate. The upper end of the cylinder is flush with the upper surface of the bottom grid plate when the adaptor is in place. This end of the adaptor has a hole similar to that in the bottom grid plate for accepting the fuel element lower end fitting. With the adaptor in place, a position formerly occupied by a control rod with a fuel follower will now accept a standard fuel element. The adaptor can be removed with a special handling tool.

4.4.5. Fuel-Moderator Elements

The active part of each fuel-moderator element, shown in Figure 4-28, is approximately 1.43 in. (3.63 cm.) in diameter and 15 in. long (38.1 cm.). The fuel is a solid, homogeneous mixture of uranium-zirconium hydride alloy containing about 8.5% by weight of uranium enriched to 20% U-235. The hydrogen-to-zirconium atom ratio is about 1.6. To facilitate hydriding, a small hole is drilled through the center of the active fuel section and a zirconium rod is inserted in this hole after hydriding is complete. The hydriding hole and zirconium rod are not shown in Figure 4-28. Several types of end fittings exist, therefore, those shown are typical.



TRIGA STAINLESS STEEL CLAD
FUEL ELEMENT WITH END FITTINGS

Figure 4-28

Each element is clad with a 0.020 in. thick (.508 mm.) stainless steel can, and all closures are made by heliarc welding. Two sections of graphite are inserted in the can, one above and one below the fuel, to serve as top and bottom reflectors for the core. Stainless steel end fixtures are attached to both ends of the can, making the overall length of the fuel-moderator element 28.8 in. (73.2 cm.).

The lower end fixture supports the fuel-moderator element on the bottom grid plate. The upper end fixture consists of a knob for attachment of the fuel-handling tool and a triangular spacer, which permits cooling water to flow through the upper grid plate. The total weight of a fully-loaded fuel element is about 3.18 kg. (7 lb.).

4.4.5.1 Instrument Fuel Elements

An instrumented fuel-moderator element will have three thermocouples embedded in the fuel. As shown in Figure 4-29, the sensing tips of the fuel element thermocouples are located about 0.3 in. (0.76 cm.) from the vertical centerline. Thermocouple specifications are listed in Table 4-9.

The thermocouple lead wires pass through a seal in the upper end fixture. A lead tube provides a watertight conduit carrying the lead wires above the water surface in the reactor pool. Thermocouple specifications are listed in Table 4-9. In other respects the instrumented fuel-moderator element is identical to the standard element.

Table 4-9

THERMOCOUPLE SPECIFICATIONS

Type	Chromel-alumel
Wire diameter	0.005 in.
Resistance	24.08 ohms/double foot at 68°F
Junction	Grounded
Sheath material	Stainless steel
Sheath diameter	0.040 in.
Insulation	MgO
Lead-out wire	
Material	Chromel-alumel
Size	20 AWG
Color code	Chromel - yellow (positive) Alumel - red (negative)
Resistance	0.59 ohms/double foot at 75°F

4.4.5.2. Evaluation of Fuel Element Design

General Atomic has acquired extensive experience in the fabrication and operation of high hydride, stainless steel clad fuel elements. As shown in other sections of this report, the stresses associated with the fuel and cladding temperatures in all modes of operation, normal and abnormal, are within the safety limits described in the Reactor Design Bases.

Dimensional stability of the overall fuel element has been excellent in the TRIGA reactors in operation. Analysis of the heat removal from elements that touch owing to transverse bending shows that the contact will not result in hot spots that damage the fuel.

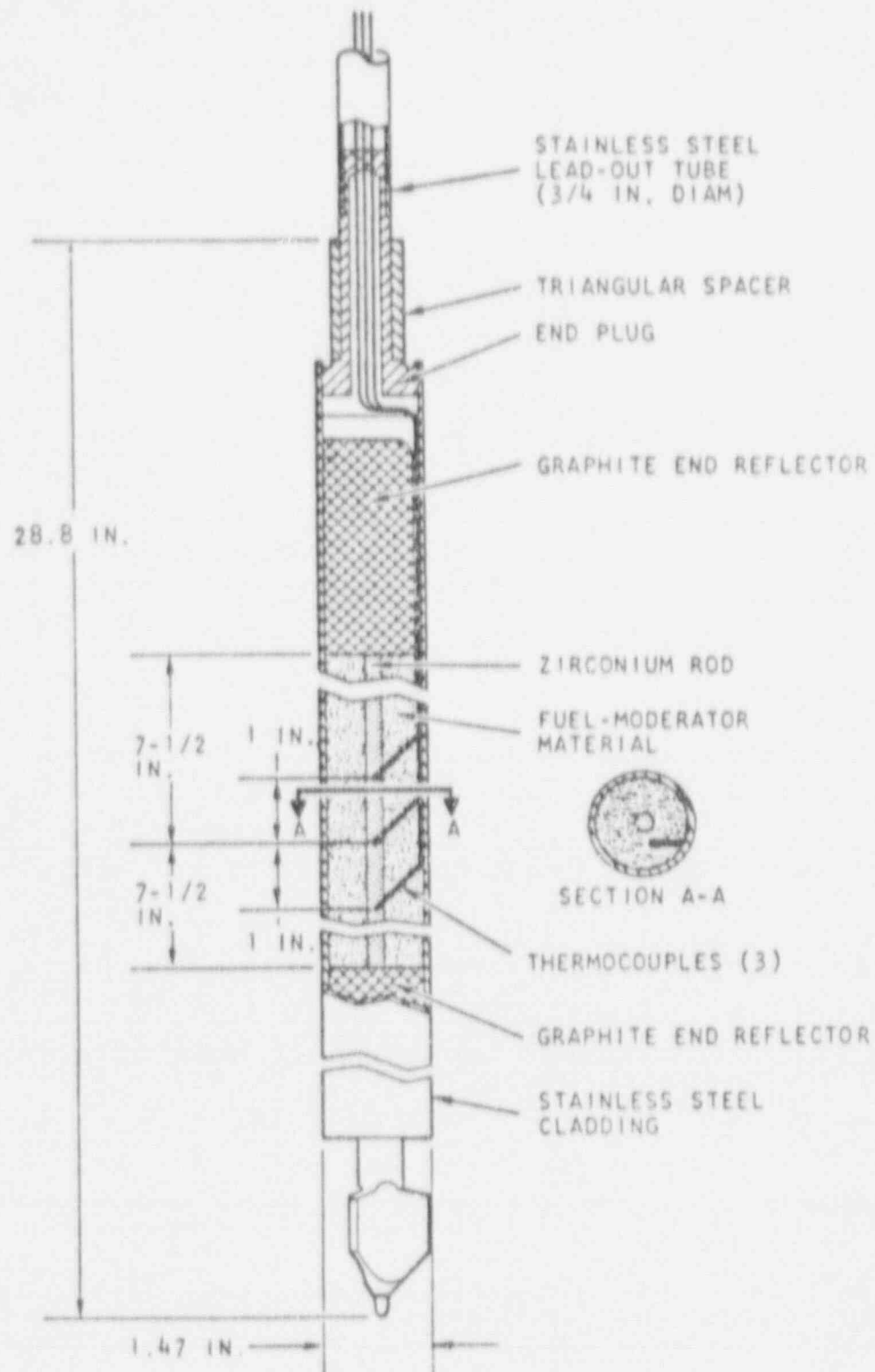
Tests have been conducted on TRIGA fuel elements to determine the strength of the fuel element clad when subjected to internal pressure. At room temperature the clads ruptured at about 2050 psig. This corresponds to a hoop stress at rupture of about 72,000 psi which compares favorably with the minimum expected value for 304 stainless steel.

It is concluded that the chemical stability of U-ZrH_{1.6} fuel-moderator material does not impose a safety limit on reactor operation (see Section 4.1.1). Table 4-10 gives a summary of the fuel element specifications.

Table 4-10

SUMMARY OF FUEL ELEMENT SPECIFICATIONS

	<u>Nominal Value</u>	
Fuel-Moderator Material		
H/Zr ratio	1.6	
Uranium content	8.5 wt %	
Enrichment (U-235)	19.7 #0.2	
Diameter	1.43 in.	
Length	15 in.	
Graphite End Reflectors	<u>Upper</u>	<u>Lower</u>
Porosity	20%	20%
Diameter	1.43 in.	1.43 in.
Length	3.44 in.	3.47 in.
Cladding		
Material	Type 304 SS	
Wall thickness	0.020 in.	
Length	22.10 in.	
End Fixtures and Spacer	Type 304 SS	
Overall Element		
Outside diameter	1.47 in.	(3.73 cm)
Length	28.37 in.	(72.06 cm)
Weight	7 lb.	(3.18 kg)



INSTRUMENTED FUEL ELEMENT

Figure 4-29

Most of the fuel for the initial core loading will consist of elements with burnups of a fraction of a MW-day to several MW-days. It is anticipated that an initial core loading of about 94 fuel elements, including instrumented elements, and fuel followed control rods, will produce a cold, clean excess reactivity of $\sim 4.9\%$ $\delta k/k$. The operational core configuration will contain two instrumented fuel elements with at least one located in the inner most reactor ring.

4.4.6. Neutron Source and Holder

A 2 or 3 curie americium beryllium neutron source will be used for startup. The neutron source holder is made of aluminum, is cylindrical in shape, and has a cavity to hold the source. The source holder can be installed in a vacant fuel or graphite element location. A shoulder at the upper end of the holder supports the assembly on the upper grid plate, the rod itself, which contains the source, extending down into the core region. The neutron source is contained in a cavity in the lower portion of the rod assembly at the horizontal centerline of the core. This cylindrical cavity is 0.981 in. (2.492 cm.) in diameter and approximately 3 in. (7.62 cm.) deep. The upper and lower portions are screwed together. A soft aluminum ring provides sealing against water leakage into the cavity. Since the upper end fixture of the source holder is similar to that of the fuel element, the source holder can be installed or removed with the fuel handling tool. In addition, the upper end fixture has a small hole through which one end of a stainless steel wire may be inserted to facilitate handling operation from the top of the tank.

4.4.7. Graphite Dummy Elements

Graphite dummy elements may be used to fill grid positions not filled by the fuel-moderator elements or other core compounds. They are of the same general dimensions and construction as the fuel-moderator elements, but are filled entirely with graphite and are clad with aluminum.

4.4.8. Control System Design

The reactor uses four control rods:

- a. Shim rod 1
- b. Shim rod 2
- c. A transient rod
- d. A regulatory rod

The regulating and shim rods are sealed 304 stainless steel tubes approximately 43 in. (109 cm) long by 1.35 in. (3.43 cm) in diameter in which the uppermost 6.5 in. (16.5 cm) section is an air void and the next 15 in. (38.1 cm) is the neutron absorber (boron carbide in solid form). Immediately below the neutron absorber is a fuel follower section consisting of 15 in. (38.1 cm) of U-ZrH_{1.6} fuel. The bottom section of the rod is 6.5 in. (16.5 cm) air void.

The regulating and shim rods pass through and are guided by 1.5 in. (3.81 cm) diameter holes in the top and bottom grid plates. A typical control rod with fuel follower is shown in the withdrawn and inserted positions in Figure 4-30.

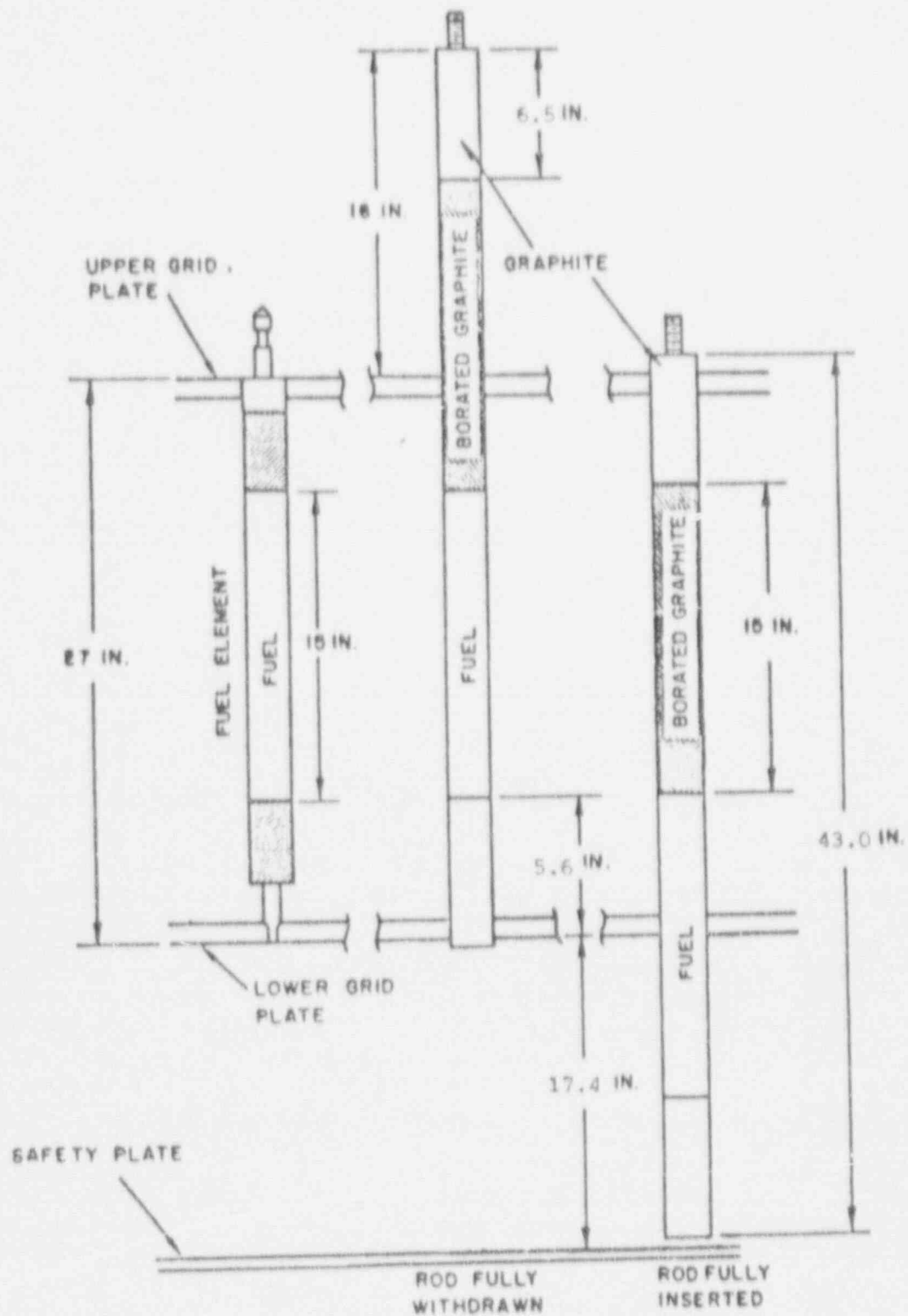
The safety-transient rod is a sealed, 36.75 in. (93.35 cm) long by 1.25 in. (3.18 cm) diameter tube containing solid boron carbide as a neutron absorber. Below the absorber is an air filled follower section. The absorber section is 15 in. (38.1 cm) long and the follower is 20.88 in. (53.02 cm) long. The transient rod passes through the core in a perforated aluminum guide tube. The tube receives its support from the safety plate and its lateral positioning from both grid plates. It extends approximately 10 in. (25.4 cm) above the top grid plate. Water passage through the tube is provided by a large number of holes distributed evenly over its length. A locking device is built into the lower end of the assembly.

The control rods are connected to their individual drive units by screwing the upper end of the rod into a control rod drive assembly connecting rod. Vertical travel of each rod is approximately 15 in. (38.1 cm). Reactivity worths and core positions for each rod are summarized in the section on nuclear design. A summary of other control rod design parameters is given in Table 4-11.

Table 4-11

SUMMARY OF CONTROL ROD DESIGN PARAMETERS

	Transient	Shim and Regulating
<u>Cladding</u>		
Material	Al	Type 304 SS
OD	1.25 in. (3.18 cm)	1.35 in. (3.43 cm)
Length	36.75 in. (93.35 cm)	43.13 in. (109.5 cm)
Wall thickness	0.028 in. (0.071 cm)	0.020 in. (0.051 cm)
<u>Absorber</u>		
Material	Boron Carbide (solid form)	
OD	1.19 in. (3.02 cm)	1.31 in. (3.32 cm)
Length	15 in. (38.1 cm)	14.25 in. (36.20 cm)
<u>Follower</u>		
Material	Air	U-ZrH _{1.6}
OD	1.25 in. (3.18 cm)	1.31 in. (3.34 cm)
Length	20.88 in. (53.02 cm)	15 in. (38.1 cm)



FUEL FOLLOWED CONTROL ROD

Figure 4-30

4.4.8.1. Control Rod Drive Assemblies

The control rod drive assemblies for the shim rods are mounted on a bridge assembly over the pool and consist of a motor and reduction gear driving a rack-and-pinion as indicated in Figure 4-31. A helipot connected to the pinion generates the position indication. Each control rod drive has a tube that extends to a dashpot below the surface of the water. The control rod assembly is connected to the rack through an electromagnet and armature. In the event of a power failure or scram signals, the control rod magnets are de-energized and the rods fall into the core. The time required for a rod to drop into the core from the full-out position is about 1 second.

The rod drive motor is non-synchronous, single-phase, and instantly reversible, and will insert or withdraw the control rod at a rate of approximately 18 in./min. (0.75 cm/sec) for the shim 1 and shim 2 rods. The regulating rod design is similar to the shim rods except for the type of rod drive.

A key-locked switch on the control console power supply prevents unauthorized operation of all control rod drives. Electrical dynamic and static braking on the motors are used for fast stops.

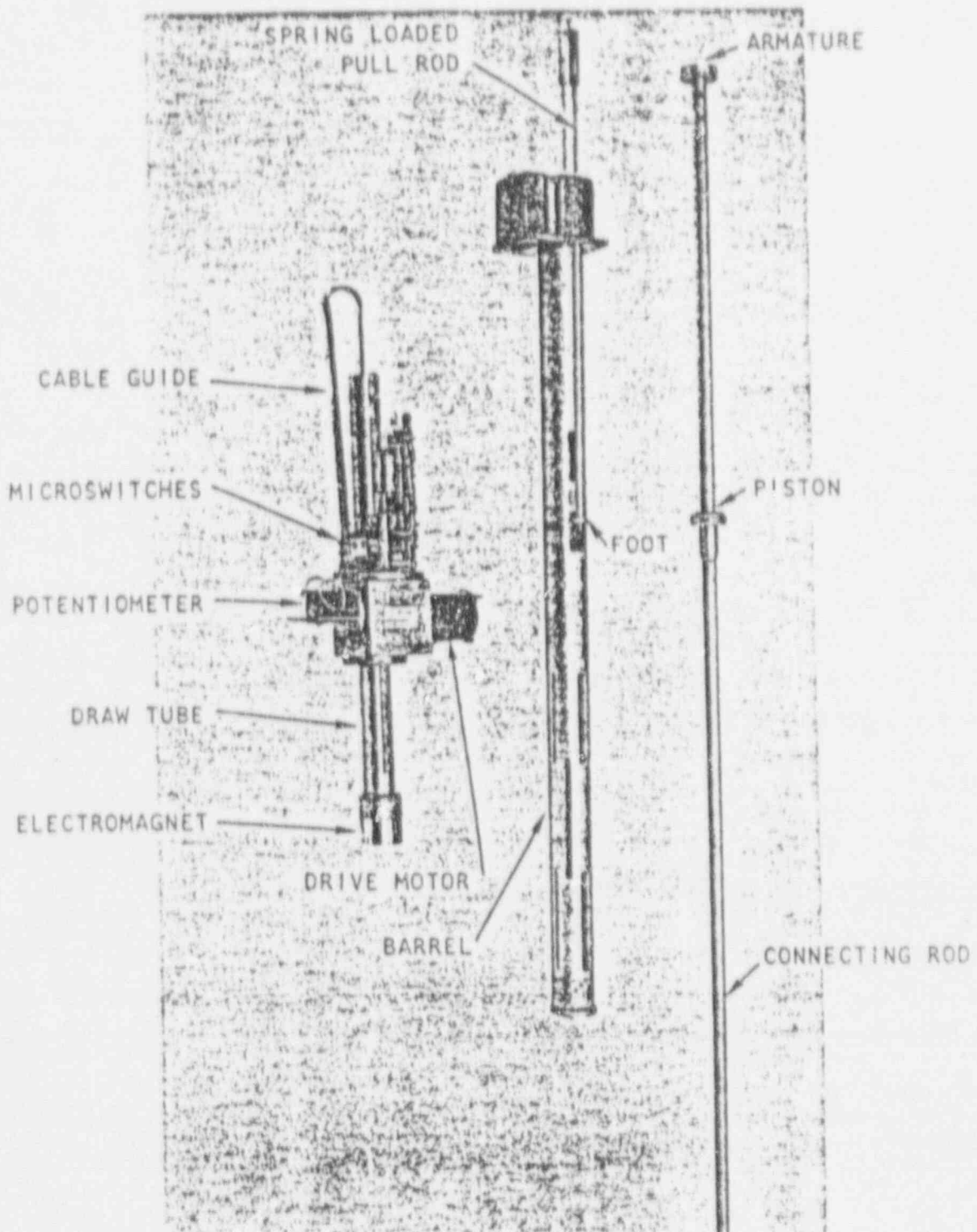
Limit switches mounted on the drive assembly actuate circuits which indicate the following:

- a. The "up" and "down" positions of the magnet
- b. The "down" position of the rod
- c. The magnet in contact with the rod

4.4.8.2. Regulating Rod and Stepping Motor Drive

The rod drive mechanism for the regulating rod will be an electric stepping-motor-actuated linear drive equipped with a magnetic coupler and a positive feedback potentiometer.

A stepping motor drives a pinion gear and a 10-turn potentiometer via a chain and pulley gear mechanism. The potentiometer is used to provide rod position information. The pinion gear engages a rack attached to the magnet draw tube. An electromagnet, attached to the lower end of the draw tube, engages an iron armature. The armature is screwed and pinned into the upper end of a connecting rod that terminates at its lower end in the control rod. When the stepping motor is energized (via the rod control UP/DOWN switch on the operator's console), the pinion gear shaft rotates, thus raising the magnet draw tube, the armature and the connecting rod will raise with the draw tube so that the control rod is withdrawn from the reactor core. In the event of a reactor scram, the magnet is de-energized and the armature will be released. The connecting rod, the piston, and the control rod will then drop, thus reinserting the control rod.



RACK AND PINION CONTROL ROD DRIVE

Figure 4-31

Stepping motors operate on phase-switched direct current power. The motor shaft advances 200 steps per revolution (1.8 degrees per step). Since current is maintained on the motor windings when the motor is not being stepped, a high holding torque is maintained.

The torque vs speed characteristic of a stepping motor is greatly dependent on the drive circuit used to step the motor. To optimize the torque characteristic vs motor frame size, a Translator Module was selected to drive the stepping motor. This combination of stepping motor and translator module produces the optimum torque at the operating speeds of the control rod drives.

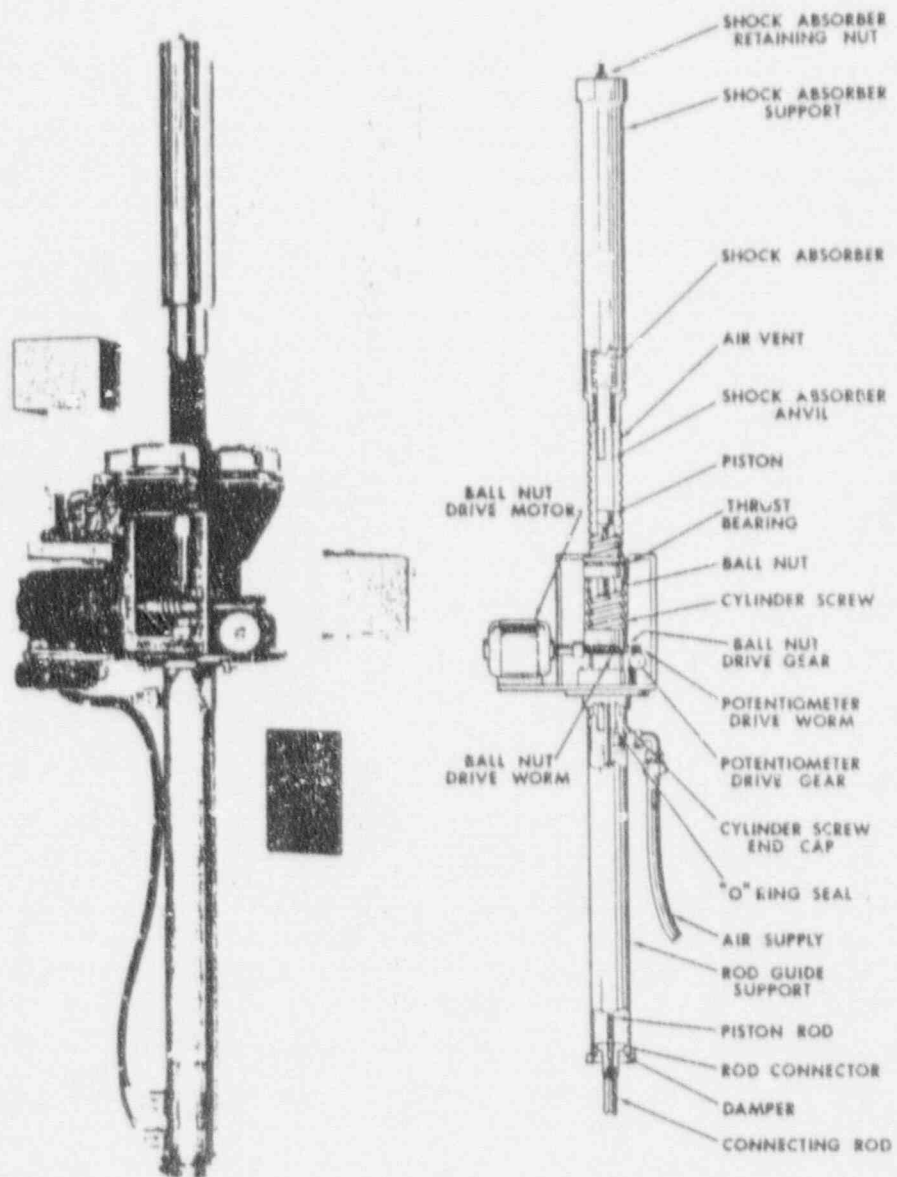
4.4.8.3. Transient Rod Drive Assembly.

The safety transient control rod on pulsing TRIGA Mark II reactors is operated with a pneumatic rod drive (see Figures 4-32 and 4-33). Operation of the transient rod drive is controlled from the reactor console.

The transient rod is a scrammable rod operated in both pulse and steady-state modes of reactor operation. During non pulse operation, the transient rod will function as an alternate safety rod with air continuously supplied to the rod. Rod position is thus controlled by operation of an electric motor that positions the air drive cylinder. The position of the transient control rod and its associated reactivity worth will generally dictate removal of the rod as the undid step of a startup for steady-state operation.

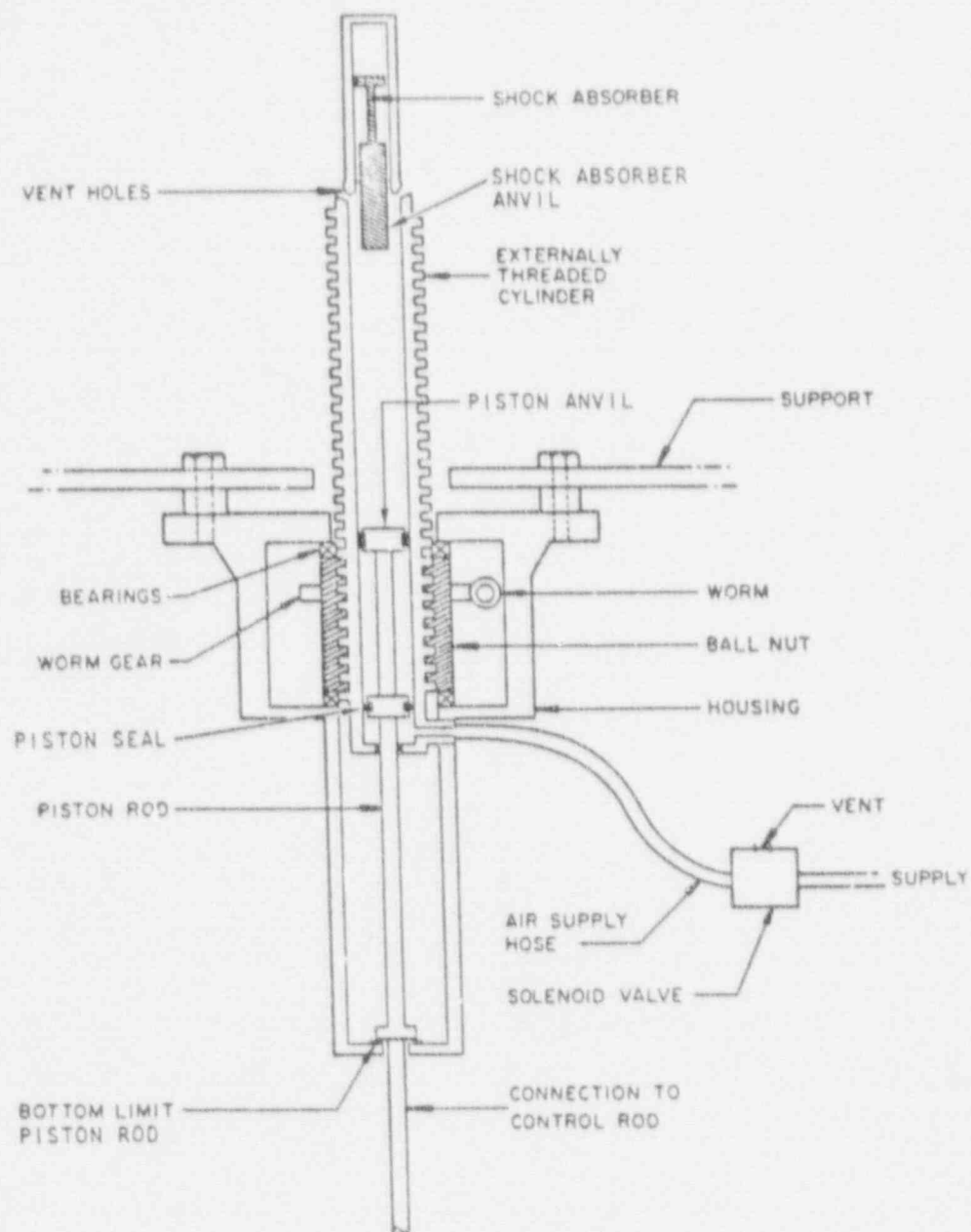
The transient rod drive is mounted on a steel frame that bolts to the bridge. Any value from zero to a maximum of 15 in. (38.1 cm.) of rod may be withdrawn from the core; administrative control is exercised to restrict its travel so as not to exceed the maximum licensed step insertion of reactivity ($\$3.14$ or $2.2\% \delta k/k$).

The transient rod drive is a single-acting pneumatic cylinder with its piston attached to the transient rod through a connecting rod assembly. The piston rod passes through an air seal at the lower end of the cylinder. Compressed air is supplied to the lower end of the cylinder from an accumulator tank when a three-way solenoid valve located in the piping between the accumulator and cylinder is energized. The compressed air drives the piston upward in the cylinder and causes the rapid withdrawal of the transient rod from the core. As the piston rises, the air trapped above it is pushed out through vents at the upper end of the cylinder. At the end of its travel, the piston strikes the anvil of an oil-filled hydraulic shock absorber, which has a spring return, and which decelerates the piston at a controlled rate over its last 2 in. (5 cm.) of travel. When the solenoid is de-energized, the valve cuts off the compressed air supply and exhausts the pressure in the cylinder, thus allowing the piston to drop by gravity to its original position and restore the transient rod to its fully inserted position in the reactor core.



ADJUSTABLE TRANSIENT ROD

Figure 4-32



TRANSIENT ROD OPERATIONAL SCHEMATIC

Figure 4-33

The extent of transient rod withdrawal from the core during a pulse is determined by raising or lowering the cylinder, thereby controlling the distance the piston travels.

The cylinder has external threads running most of its length, which engage a series of ball bearings contained in a ball-nut mounted in the drive housing. As the ball-nut is rotated by a worm gear, the cylinder moves up or down depending on the direction of worm gear rotation.

A ten-turn potentiometer driven by the worm shaft provides a signal indicating the position of the cylinder and the distance the transient rod will be ejected from the core. Motor operation for pneumatic cylinder positioning is controlled by a switch on the reactor control console. The magnet power key switch on the control console power supply prevents unauthorized firing of the transient rod drive.

Attached to and extending downward from the transient rod drive housing is the rod guide support, which serves several purposes. The air inlet connection near the bottom of the cylinder projects through a slot in the rod guide and prevents the cylinder from rotating. Attached to the lower end of the piston rod is a flanged connector that is attached to the connecting rod assembly that moves the transient rod. The flanged connector stops the downward movement of the transient rod when the connector strikes the damp pad at the bottom of the rod guide support. A microswitch is mounted on the outside of the guide tube with its actuating lever extending inward through a slot. When the transient rod is fully inserted in the reactor core, the flange connector engages the actuating lever of the microswitch and indicates on the instrument console that the rod is in the core.

In the case of the transient rod a scram signal de-energizes the solenoid valve which supplies the air required to hold the rod in a withdrawn position and the rod drops into the core from the full out position in about 1 second.

4.4.8.4. Evaluation of Control Rod System.

The reactivity worth and speed of travel for the control rods are adequate to allow complete control of the reactor system during operation from a shutdown condition to full power. The scram times for the rods are quite adequate since the TRIGA system does not rely on speed of control as being paramount to the safety of the reactor. The inherent shutdown mechanism of the TRIGA prevents unsafe excursions and the control system is used only for the planned shutdown of the reactor and to control the power level in steady-state operation.

4.5 SAFETY SETTINGS IN RELATION TO SAFETY LIMITS

As has been indicated, fuel temperatures are the main safety considerations in the operation of the TRIGA system. The temperature of the fuel may be controlled by setting limits on other operating parameters. The operating parameters of interest for TRIGA are:

- a. Maximum licensed steady-state power level
- b. Fuel temperature measured by thermocouple
- c. Maximum reactivity worth of transient rod
- d. Core inlet coolant water temperature

The safety settings as listed in Table 4-12 are such that in all operation, normal and abnormal, the safety limits indicated in the reactor design bases will not be exceeded.

Table 4-12

TRIGA SAFETY SETTINGS

Parameter Limited	Safety Setting	Function
Maximum steady-state power level	1100 kW (t)	Reactor scram
Maximum measurement of fuel temperature	500°C	Reactor scram

Administrative limitations are imposed for the excess reactivity, transient conditions and coolant water temperature as follows:

- a. Maximum core excess reactivity of 4.9% $\delta k/k$ with a shutdown margin of at least 0.2% $\delta k/k$, assuming one rod withdrawn.
- b. Maximum transient control rod worth of 2.8% $\delta k/k$ with a limit of 2.2% $\delta k/k$, for any transient insertion.
- c. Core inlet water temperature of 48.9°C.

These safety settings are conservative in the sense that if they are adhered to, the consequence of normal or abnormal operation would be fuel and clad temperatures well below the safety limits indicated in the reactor design bases. Because of the conservatism in these safety settings, it is reasonable that at some later date less restrictive safety system settings could be assigned in conjunction with upgrading of the reactor to operate at higher steady-state power levels or in the pulsing mode, while still using the same fuel and core configuration.

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Chapter 5

REACTOR COOLANT SYSTEM

The TRIGA is designed for operation with cooling provided by natural convective flow of demineralized water in the reactor pool. The suitability of this type of cooling at the power levels for this TRIGA has been demonstrated by numerous TRIGA installations throughout the world.

The primary functions of the coolant system are:

- a. to dissipate heat generated in the reactor,
- b. to provide vertical shielding of radiation from the reactor and allow access to the reactor core.

Heat dissipation is satisfied by natural convective flow of pool water through the reactor core and forced circulation of the pool water through an external heat exchanger. The pool coolant volume is composed of approximately 41.0 cubic meters in a two by three meter oval pool with a vertical depth of 8.1 meters. A vertical shield is provided by about 6.8 meters of water above the reactor core.

Other functions provided by the coolant system are:

- a. minimize corrosion of all reactor components, particularly the fuel elements,
- b. maintain a minimal level of radioactivity in the reactor pool water and
- c. maintain optical clarity of pool water.

These three functions are accomplished by a purification system that is included as a part of the coolant system.

5.1 DESIGN BASES

The design basis for the reactor coolant system is predicated on its primary function, reactor cooling. Other coolant system functions establish the design bases for the purification circuit.

5.1.1. Reactor Core Heat Removal

To assure adequate reactor cooling, the effectiveness of natural convective cooling has been evaluated with respect to the peak heat flux which may be achieved in the reactor. This evaluation then establishes the maximum heat flux beyond which fuel element cladding integrity cannot be assured.

Based on these evaluations, it is concluded that for steady-state operation the coolant inlet temperature and maximum heat flux at which fuel clad integrity is no longer assured is determined by the curve relating heat flux and coolant temperature for the hottest coolant channel. The maximum design temperature of the coolant system, coolant inlet temperature, is 120°F (48.9°C). The maximum allowable peak heat flux at this temperature is 325 kBtu/hr-ft² (103 watts/cm²) corresponding to a power level of 1900 kW for an 85 element core. Since the maximum licensed power level is 1100 kW, the resulting maximum heat flux will be 188 kBtu/hr-ft² (59.4 watts/cm²) which is well below the value at which clad integrity may be questioned.

5.1.2. Reactor Pool Heat Removal

Supplemental cooling of the reactor pool is required for continuous operation at the rated power level. A heat rate of 20.7°C/hour is expected with the reactor operated at 1000 kW. Heat removal from the pool is provided by heat exchange with a chilled water supply. The chilled water supply is operated by the University for cooling of Research Center buildings and equipment. Chilling capacity is provided by multiple 1200 ton (4200 kW) units. At reactor rated power the heat removal capacity required is represented by about 25% of the chilling system capacity of one unit. A tube and shell heat exchanger is installed for heat removal from the reactor pool to the available chilled water system.

5.1.3. Heat Exchanger Design Bases

Heat exchanger capacity is designed for a stable operating temperature of the reactor pool at or below the coolant design temperature, 120°F (48.9°C) for convective reactor core cooling. The stable temperature is maintained by a heat exchanger capacity equivalent to the reactor core thermal output capacity. Other heat losses such as evaporation, or heat gains from the pump, are considered negligible.

Heat transfer is defined by

$$q = U A \delta T_m \quad (1)$$

where U = overall heat transfer coefficient (watt/m² · °C)

A = surface area for heat transfer (m²)

δT_m = true mean temperature difference (°C)

For a tube and shell heat exchanger the overall heat transfer coefficient is composed of three terms, the convective heat transfer from the fluid in the tubes to the tube walls, the conductive heat transfer thru the tube wall, and the convective heat transfer from the outside tube wall to the fluid in the shell of the heat exchanger. Based on the outside tube area for heat transfer, the overall heat transfer coefficient is defined as [1],

$$U_c = \left[\frac{A_o}{A_i h_i} + \frac{A_o \ln (r_o/r_i)}{2\pi k l} + \frac{1}{h_o} \right]^{-1}, \quad (2)$$

where

A_o = total outside tube area (m^2)

A_i = total inside tube area (m^2)

r_i = tube inside radius (m)

r_o = tube outside radius (m)

h_i = convective heat transfer coefficient between fluid in tubes and tube wall ($W/m^2 \cdot ^\circ C$)

h_o = convective heat transfer coefficient between fluid in shell and tube wall ($W/m^2 \cdot ^\circ C$)

k = conductive heat transfer coefficient in the tube wall ($W/m^2 \cdot ^\circ C$)

l = total tube length in heat exchanger (m)

A correction is applied for fouling of heat exchanger caused by buildup of various deposits. The overall heat transfer coefficient for a fouled heat exchanger is determined by

$$U_f = \frac{1}{R_f + 1/U_c} \quad (3)$$

where R_f is the fouling factor, (non-dimensional). The convective heat transfer coefficient is defined as

$$h = \frac{Nu \, k}{d} \quad (4)$$

where

Nu = Nusselt Number

k = thermal conductivity of the fluid evaluated at the appropriate average temperature ($W/m \cdot ^\circ C$)

d = tube diameter or applicable hydraulic diameter (m)

The complicated nature of turbulent flow heat transfer is described by a Nusselt number determined by experimental correlation with the Reynolds and Prandtl Numbers. Dittus and Boelter [2] recommend the following relation for fully developed turbulent flow in tubes:

$$Nu_t = 0.023 Re^{0.8} Pr^n \quad , \quad (5)$$

where parameters are measured inside the tubes

Re = Reynolds Number based on tube diameter,

Pr = Prandtl Number at average fluid temperature,

n = 0.4 for heating,

n = 0.3 for cooling.

The relation for the shell side of a baffled cross flow heat exchanger is suggested by Colburn [3] as follows:

$$Nu_s = 0.33 Re^{0.6} Pr^{0.33} \quad , \quad (6)$$

where parameters are measured outside the tubes and

Re = Reynolds Number based on tube outside diameter and velocity at minimum shell cross sectional area,

Pr = Prandtl Number at average fluid temperature.

The product terms, $A \delta T_m$, are defined consistent with the definition of U and heat exchanger design. The total cross sectional area of the tubes is represented by the heat transfer area, A, as specified by the heat transfer coefficient, U. The true mean temperature difference, δT_m , is related to the heat exchanger type by a correction factor, F, and a log mean temperature difference, LMTD [4]. The correlation relates a simple single pass heat exchanger with more complex multiple pass baffled units. A relation is defined by

$$\delta T_m = F * LMTD \quad , \quad (7)$$

where

F = correction factor [5,6],

$$LMTD = (T_a - T_b) / \ln(T_a/T_b) \quad , \quad (8)$$

For a counter flow heat exchanger

$$T_a = (T \text{ hot fluid in} - T \text{ cold fluid out}) \quad , \quad (9)$$

$$T_b = (T \text{ hot fluid out} - T \text{ cold fluid in}) \quad , \quad (10)$$

Actual heat exchanger capacity is calculated using an energy balance on either the shell or tube fluid. The heat transfer is defined as:

$$q = C (T_{in} - T_{out}) \quad , \quad (11)$$

where

$$C = \dot{m} c_p$$

\dot{m} = mass flow rate,

c_p = fluid specific heat,

T_{in} = Temp of fluid entering heat exchanger,

T_{out} = Temp of fluid exiting heat exchanger.

In the current case T_{out} of either fluid is not known. Only T_{in} (100°F pool water, 48°F coolant water) and the mass flow rate of both fluids are known. To determine T_{out} the effectiveness/NTU method [4,7] is used. The dimensionless parameter called the heat exchanger effectiveness ϵ is defined as

$$\epsilon = \frac{\text{Actual Heat Transfer}}{\text{Maximum Possible Heat Transfer}} \quad (12)$$

where the maximum possible heat transfer is

$$q_{max} = C_{min} (T_{hot\ in} - T_{cold\ in}) \quad (13)$$

Substituting (11) for each fluid and (13) into (12) results in

$$\epsilon = \frac{C_{hot} (T_{hot\ in} - T_{hot\ out})}{C_{min} (T_{hot\ in} - T_{cold\ in})} \quad (14)$$

for the hot fluid and

$$\epsilon = \frac{C_{cold} (T_{cold\ out} - T_{cold\ in})}{C_{min} (T_{hot\ in} - T_{cold\ in})} \quad (15)$$

for the cold fluid. The heat exchange effectiveness determined by [7] for a shell and tube heat exchanger with one shell pass and any multiple of tube passes is

$$\epsilon = z \left[1 + r + B \left(\frac{1 + \exp(-NB)}{1 - \exp(-NB)} \right) \right]^{-1} \quad (16)$$

where

$$r = C_{min} / C_{max} \quad (17)$$

$$N = UA / C_{min} \quad (18)$$

U = overall heat transfer defined in (2)

A = surface area for heat transfer

$$B = (1 + r^2)^{1/2} \quad (19)$$

Once the effectiveness is calculated, (14) and (15) are used to determine $T_{hot\ out}$ and $T_{cold\ out}$. These may then be used in (11) to determine the capacity of the heat exchanger.

5.1.4. Water Purification Bases

The functions of corrosion control, radioactivity control, and optical clarity of the coolant water are provided by filtration and ion exchange. Control of the water purity is performed by analysis of the water conductivity. Measurements of water conductivity as low as 2.0 micromho per centimeter (or resistance of 1 megohm per centimeter) are maintained by filtration and ion exchange. The conductivity is reduced further by control of materials exposed to the reactor coolant, minimizing dust settling to the pool surface, and occasional cleaning of pool surfaces. Experience has shown that conductivities of 5.0 $\mu\text{mho/cm}$ are sufficient to maintain acceptable limits on corrosion plus good water optical quality and removal of activation products in the water.

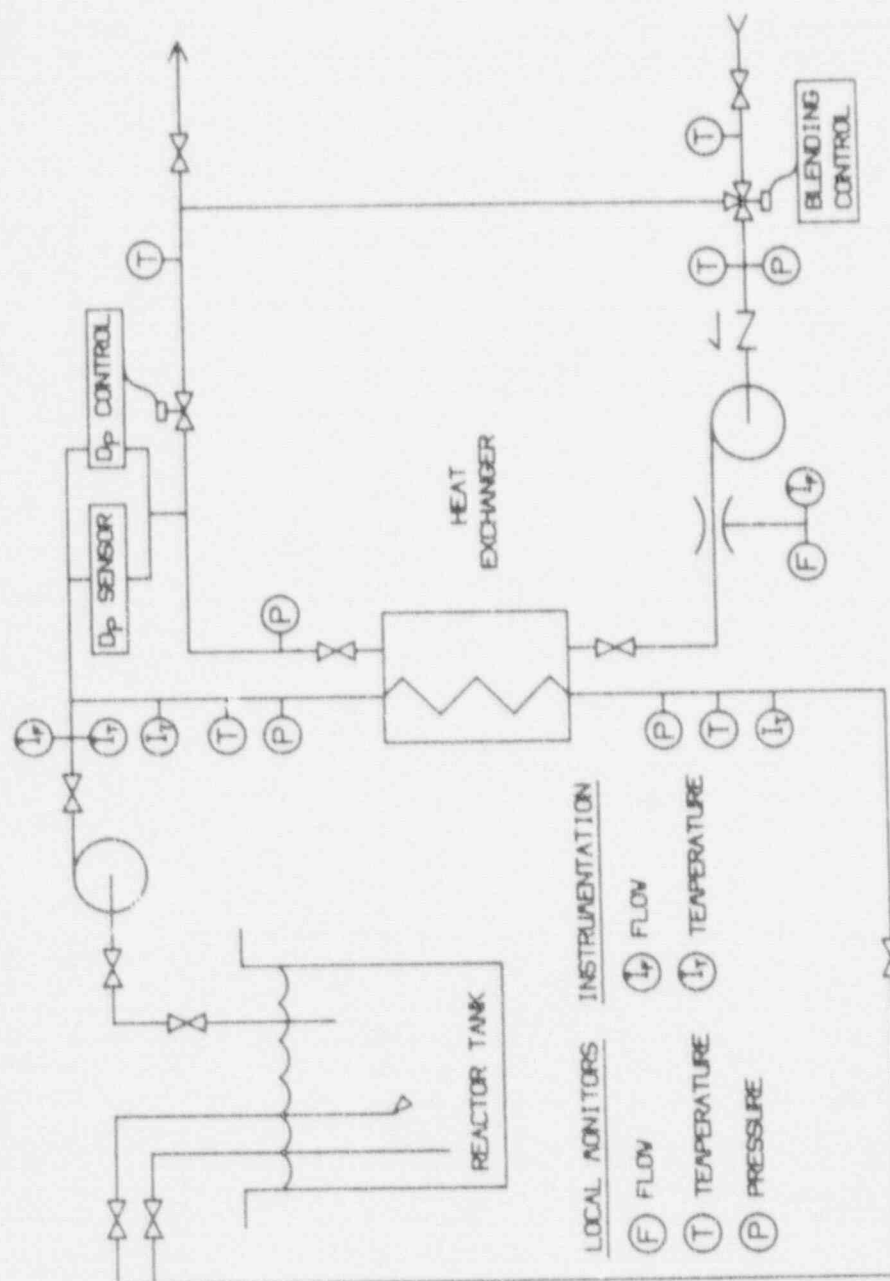
5.2 SYSTEM DESIGN

Principle components of the coolant system are the aluminum reactor pool tank, the external cooling loop consisting of heat exchanger and pump, and the purification loop consisting of filter, resin bed and pump. Most of the total coolant volume is represented by the approximately 41.0 cubic meters of water in the reactor pool. Typical flow diagrams for the systems are shown in Figures 5-1 and 5-2.

5.2.1. Coolant System

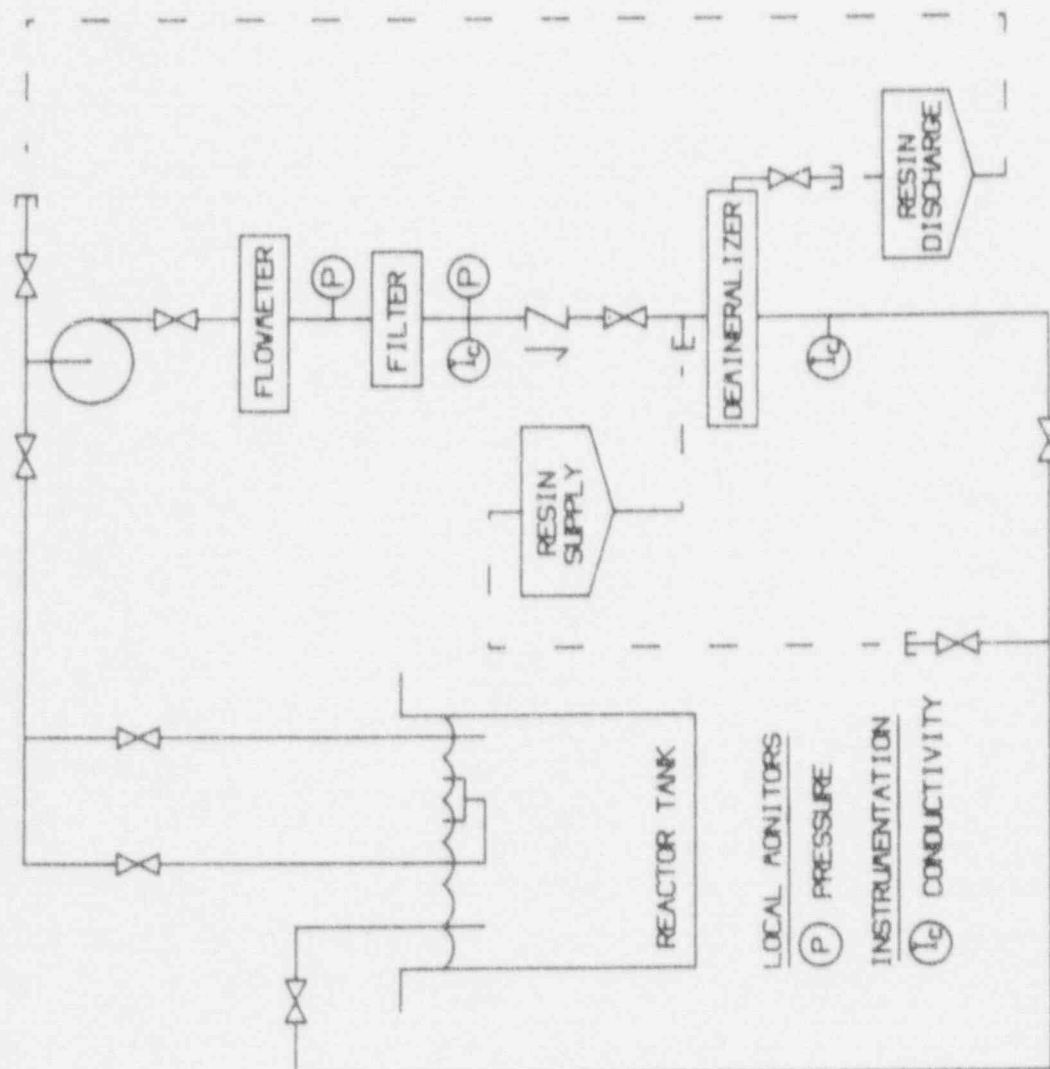
Suction of water from the pool is provided by an inlet which extends no more than 2 meters below the top of the reactor tank. The coolant water is drawn through the coolant pump and forced through the heat exchanger. Return of cooled water to the reactor pool is provided by single or multiple discharge outlets above the reactor core or an outlet near the tank bottom. A diffused water jet is created at the outlets above the reactor core by a nozzle. Delay and diffusion of the reactor core convective coolant column is enhanced by the action of the coolant discharge nozzle.

Accidental siphoning of reactor pool water is prevented by the presence of suction breaks on both suction and discharge lines of the coolant system. Siphon breaks are created by holes located in the lines approximately half a meter below the normal water level.



COOLANT SYSTEM LAYOUT

Figure 5-1



PURIFICATION SYSTEM LAYOUT

Figure 5-2

The heat exchanger and pump, the major components of the cooling system, are located in room 1.104b at about the same vertical level as the reactor core. Valves are provided in the coolant loop for control and isolation of the cooling system function. Specifications of cooling system components are listed in Table 5-1. A positive pressure difference of 1 psi (7 kilopascals) between the shell side outlet and tube side inlet of the heat exchanger is designed to prevent leakage of primary pool coolant into the secondary chilled water system. The pressure difference is maintained under varying flow conditions by a differential pressure controller which regulates the position of a throttle valve in the heat exchanger shell side outlet pipe. Coolant water supply temperature is regulated by a temperature controller coupled to a mixing valve in the chilled water supply line.

Table 5-1

REACTOR COOLANT SYSTEM DESIGN
SUMMARY

Reactor Tank		
Material	Al plate (6061)	
Thickness	1/4 in. (0.635 cm)	
Volume (maximum)	11000 gal. (41.64 m ³)	
Coolant Lines		
Pipe	Aluminum (6061)	
Valves	Iron-Plastic liner, 316 ss. Ball and Stem	
Fittings	Aluminum (Victaulic)	
Coolant Pump		
Type	Centrifugal	
Material	Stainless steel	
Capacity	250 gpm (15.8 liter/sec)	
Heat Exchanger		
Type	Shell and tube	
Materials: shell	Carbon steel	
tubes	304 stainless steel	
Heat Duty	1000 kW	
Flowrate: tubes	250 gpm (15.8 liters/sec)	
shell	400 gpm (25.2 liters/sec)	
Typical Parameters:		
Tube inlet	100°F	42 psia
Tube outlet	69°F	27 psia
Shell inlet	48°F	55 psia
Shell outlet	67°F	48 psia

5.2.2. Purification System

Suction of water from the pool is provided by two inlets, neither of which extend more than 2 meters below the top of the reactor tank. Valves at the pool surface allow suction from either a subsurface inlet or from a surface skimmer designed to collect and remove floating debris. Accidental siphoning of reactor pool water is prevented by siphon breaks similar to those on the coolant piping.

The purification skid is located in room 1.104b at about the same level as the reactor core. The skid consists of a pump, flowmeter, filter, resin bed, and instrumentation. Normally the purification system is operated continuously to provide removal of suspended particles and soluble ions in the coolant water. The system flow rate is about 10 gpm (0.6 lps).

Purification functions of the loop are generated by two components, a filter for removal of suspended materials and a resin bed for removal of soluble elements. Typical filtration is provided with 25 micron filters. Typical ion exchange is provided by .085 cubic meters of mixed cation and anion resin. Water purity is measured by conductivity cells at the inlet and outlet of the resin beds.

Return flow to the pool is thru a subsurface discharge pipe. Valves are provided for isolation of suction or return lines, and for isolation of system components for maintenance or resin replacement.

5.2.3. Water System Instrumentation

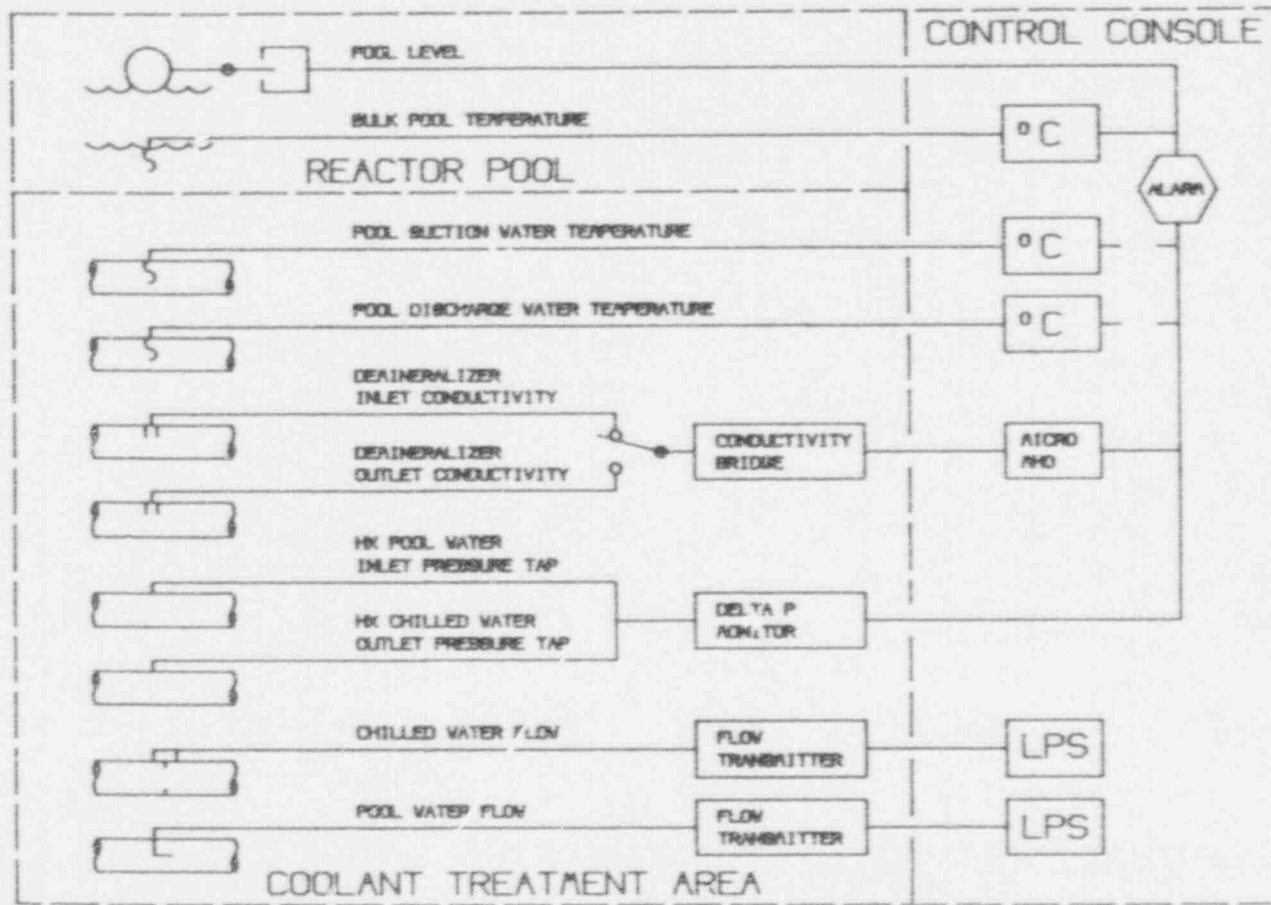
Several monitoring sensors are installed to allow remote readout of water system parameters in the reactor control room. Other system parameters are indicated by local monitoring devices. Parameter monitoring points are illustrated in Figure 5-1 and 5-2. The parameters that are considered part of the water system instrumentation system are presented in Figure 5-3.

Indication of the reactor pool status is determined by two sensors located in the pool. Pool level and bulk pool temperature sensors in the pool are monitored in the control room. An annunciator alarm indication is generated in the control room by abnormal pool levels and by high pool temperatures.

The cooling system parameters normally available in the control room include coolant temperatures, flowrates, and differential pressure status. Two temperature probes, one in the pool suction line and one in the pool discharge line, allow monitoring of heat exchanger cooling function. Typical temperature probes used are resistance temperature detectors (RTD's). Two flow meters, one in the chilled water line and one in the pool water line provide information on system flow rates. A differential pressure monitor provides an alarm if the pressure at the high pressure point on the heat exchanger tube side is not less than the low pressure point on the shell side. The differential pressure is designed for a difference substantially greater than 7 kilopascals (1 lb/sq. in.).

WATER SYSTEM INSTRUMENTATION

Figure 5-3



Numerous water system parameters are measured by local pressure or temperature sensors in the system lines. Both temperature and pressure probe are located on the inlet and outlet lines of the pool water side and chilled water side of the heat exchanger. A local indication of flow in the coolant loop is provided by the pressure drop across a venturi in the flow path. Purification loop flow is measured by an in line flow meter. Water pressure before and after the filter in the purification loop is measured for indication of filter condition.

Water quality is measured by two conductivity cells in the purification loop. The cells are located on inlet and outlet lines of the demineralizer that readout locally in the control room. Typical conductivity cells are composed of two parts, titanium electrodes shielded by ryton for conductivity measurement, and a thermister for temperature compensation. A Wheatstone bridge circuit on the purification skid is connected to the cells. A switch allows selection of either inlet or outlet conductivity.

5.3 WATER SYSTEM DESIGN EVALUATION

The water system including the reactor pool and the external cooling and purification loops have similar design features as used in many other operating TRIGA facilities. The demonstrated capability and integrity of this system provides assurance that the coolant system will perform its function properly and safely.

Availability of pool water for cooling and vertical shielding is assured by designing the system with siphon breaks on suction lines and discharge lines within 2 meters of the normal pool level. Greater losses of pool water are extremely improbable, although they could conceivably be initiated by rupture of the reactor tank. As shown in the loss of pool water accident analysis, even with complete loss of pool water fuel clad integrity is not threatened.

Adequacy of reactor cooling is assured by the large amount of cooling capacity inherent in the reactor pool volume as well as the capacity of the external cooling circuit which can dissipate heat at a rate equivalent to 1000 kW steady-state operation. If available heat exchanger capacity is diminished to 900 kW and initial pool temperature is 100°F, the reactor can be operated for more than 5 hours before the bulk pool temperature reaches 120°F. The actual time would be considerably longer since as bulk pool temperature increases heat exchanger heat removal capacity increases. Without external cooling or other heat loss the bulk pool temperature will rise about 20.7°C after one hour of operation at a steady-state power level of 1000 kW(t).

Heat removal capacity and thus pool heat rate is specified by analysis of a tube and shell heat exchanger. At a flow rate of 400 gal/min (25.2 liters/sec) of chilled water at 48°F (8.89°C) a heat removal rate of 1140 kW is expected. The presence of fouling in the heat exchanger is considered minimal based on the purity of the two heat exchanger fluids. Capacity is reduced to 1070 kW for a fouling factor of .0004. The heat transfer and hydraulic parameters are shown in Table 5-2.

Table 5-2

HEAT EXCHANGER: HEAT TRANSFER AND HYDRAULIC PARAMETERS

Tubes:	
Outside Diameter	0.750 inch (1.91 cm)
Wall Thickness	0.049 inch (0.124 cm.)
Thermal Conductivity	8.21 Btu/hr-ft-°F
Flow Area:	
Tube Side	8.1 in ² (52.3 cm ²)
Shell Side	33.8 in ² (218 cm ²)
Heat Transfer Surface	346 ft ² (32.1 m ²)
Average Prandtl Number	
Tube	5.38
Shell	8.41
Average Kinematic Viscosity	
Tube	8.63 x 10 ⁻⁶ ft ² /sec (8.02 x 10 ⁻⁷ m ² /sec)
Shell	1.28 x 10 ⁻⁵ ft ² /sec (1.19 x 10 ⁻⁶ m ² /sec)
Reynolds Number	
Tube	6.19 x 10 ⁴
Shell	2.02 x 10 ⁴
Corrective Heat Transfer Coefficients	
Inside Tubes	1710 Btu/hr-ft ² -°F (9701 W/m ² -°C)
Outside Tubes	1395 Btu/hr-ft ² -°F (7922 W/m ² -°C)
Overall Heat Transfer Coefficient	
Clean	520 Btu/hr-ft ² -°F (2953 W/m ² -°C)
Fouled	430 Btu/hr-ft ² -°C (2442 W/m ² -°C)
Effectiveness (ε)	
Clean	0.60
Fouled	0.56
LMTD	26.1°F (14.5°C)
Corrective Factor F	0.83
Capacity	
Clean	1140 kW
Fouled	1070 kW

Experience with this purification equipment in other TRIGA systems has shown that coolant conductivity can be easily maintained at levels of less than five micromhos per centimeter using the materials contained in the coolant system design. Furthermore, this experience has shown that no apparent corrosion of fuel clad or other components will occur if the conductivity of the water does not exceed five micromhos per centimeter when averaged over a 30-day period.

Control of radioactivity in the coolant is provided by the purification system. Should radioactivity be released from a clad leak or rupture of an experiment, detection of the release would be signaled by the continuous air monitor or by the reactor room area monitors. Based on coolant transport time calculations in the safety analysis section, these monitors should register an increase in coolant radioactivity within approximately 60 seconds of the time of radioactivity release. The transport time is estimated from the time for the coolant exposed in the core to reach the surface of the water where the continuous air monitor will detect a release of radioactivity from the pool water. An alternate indication of radioactive release is provided if a water activity monitor is installed or by a GM detector area monitor.

Chapter 5 References

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Chapter 6

INSTRUMENTATION AND CONTROL SYSTEM

Design of the instrumentation and control system is intended for new TRIGA reactor facilities and replacement of old reactor consoles. Initial verification and testing of the design by the manufacturer is a requirement prior to installation at The University of Texas at Austin. An evaluation by the University of the instrument and control console for the TRIGA is part of the initial installation of the console by the vendor. The system development, installation and initial testing is the responsibility of the vendor, General Atomics.

Control of research reactors, including the TRIGA family of reactors, has been accomplished using instrumentation and control systems based on solid-state, hardwired, analog circuitry. The logic for this control circuitry has been developed through many years of experience and includes attention to matters such as (1) necessary redundancy in the single inputs, signal processing, and controls; and (2) prevention of single-mode failure mechanisms.

Present day developments of personal computers (PC), such as the family of IBM PCs, provide many opportunities to replace the earlier analog circuitry by compact and powerful computer-based control systems. The system described in this document is a microprocessor-based instrumentation and control system developed by the General Atomics (GA) TRIGA Reactor Division. This system incorporates (1) a digital wide-range neutron power monitor, (2) analog power safety channels, (3) a variety of state-of-the-art signal conditioners and process controllers, plus (4) a digital data acquisition and control system incorporating IBM PC/AT or compatible computers in industrial versions.

6.1 DESIGN BASES

The design and manufacture of this system complies with the guidance given in American Nuclear Society and the American National Standards Institute Guide Criteria for the Reactor Safety Systems of Research Reactors (ANSI/ANS 15.15-1978) [1,2]. This standard serves the research reactor community in lieu of the ad hoc application of similar standards for power reactors. Even if single-failure criteria for plant protective actions - not deemed mandatory by ANSI/ANS 15.15 for negligible risk reactors - were applied, the standard allows the use of simple redundancy, i.e., the monitoring of the same reactor parameter using independent, redundant equipment, to satisfy the single-failure criteria for the reactor safety system.

There are several advantages in a microprocessor-based system which enhance system safety, reliability, and maintainability:

1. The use of powerful microcomputers allows data (operator input as well as output) to be more efficiently and systematically processed and recorded than ever before.

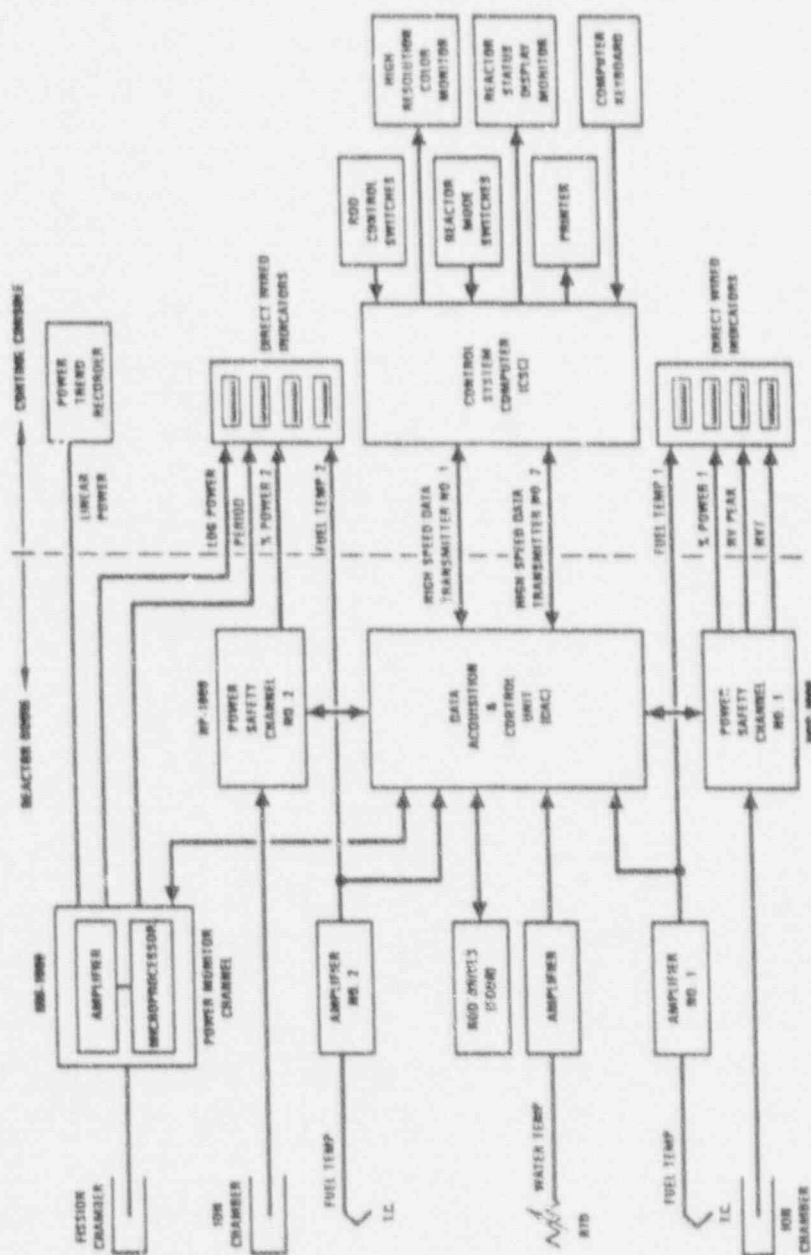
2. Several data reductions not previously possible (such as on-line calculation of the prompt period during a pulse) can be done in near-real-time.
3. On-line self-diagnostics can be performed which determines the state of the system at all times.
4. Operational surveillance and operations data are accommodated as never before with all information gathering and processing done routinely and regularly by the console computers.

The Instrumentation and Control System for the TRIGA reactor [3] is a computer-based design incorporating the use of one multifunction, NM-1000 microprocessor neutron flux monitoring channel and two companion NP-1000 current mode neutron monitoring safety channels. The combination of these two systems provides an independent operating channel and the redundant safety function of percent power with scram. The NM-1000 provides wide range log power and multi-range linear power from source level to full power. The control system logic is contained in a separate control system computer (CSC) with graphics and text displays which are the interface between the operator and the reactor. Another system for data acquisition and control (DAC) functions as the interface point for interface circuitry, process signals and communications. The multifunction NM-1000, two NP-1000 units, and two system microprocessors, the control system computer (CSC) and data acquisition and control system (DAC) are development products of General Atomics. The basic system configuration is shown in Figure 6-1.

Information from the NM-1000 channel is processed and displayed by the CSC. The two NP-1000 are independent channels that deliver power level data to the safety system scram circuit, hardwired analog indicators, and to the CSC for processing and display. Operating ranges for the neutron channels are shown in Figure 6-2.

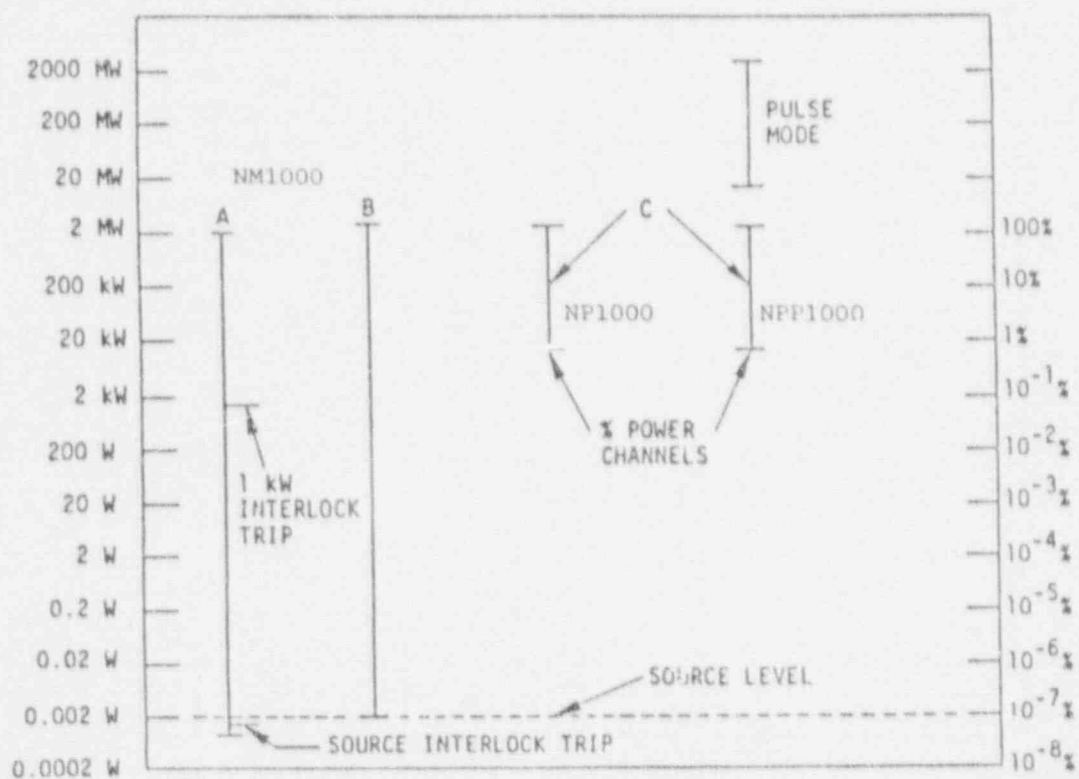
The NM-1000 digital neutron monitor channel was developed for the nuclear power industry and is fully qualified for use in the demanding and restrictive conditions of a nuclear power generating plant. Its design is based on a special GA-designed fission chamber, and low noise ultra-fast pulse amplifier. The NP-1000 was developed specifically for use with research reactor safety systems and includes several features not usually found in this type of application.

The CSC and its acquisition system, the DAC, manages all control rod movements, accounting for such things as interlocks and choice of particular operating modes. It also processes and displays information on control rod position, power level, fuel and water temperature, and can display pulse characteristics. The CSC also performs many other functions, such as calibrating control rods, monitoring reactor usage, and historical operating data can be saved for replay at a later date. A computer-based control system has many advantages over an analog system: speed, accuracy, reliability, and the ability for graphic displays self-calibration, improved diagnostics, and logging of vital information.



CONTROL SYSTEM BLOCK DIAGRAM

FIGURE 6-1



- A = Wide Range Log Channel, NM1000
 B = Wide Range Linear Channel, NM1000
 C = Manual, Automatic, and Squarewave Modes, NP1000

NEUTRON CHANNEL OPERATING RANGES

Figure 6-2

6.1.1. NM-1000 Neutron Channel

The NM-1000 nuclear channel has multifunction capability to provide neutron monitoring over a wide power range from a single detector. The selectable functions are any or all of the following:

- a. Percent power.
- b. Wide-range log power.
- c. Power rate of change.
- d. Multi-range linear power.

For the TRIGA ICS, one NM-1000 system is designated to provide the wide-range log power function and the multirange linear power function. The wide-range log power function is a digital version of the patented GA 10-decade log power system to cover the reactor power range from below surface level to 150% power and provide a period signal. For the log power function, the chamber signal from startup (pulse counting) range through the Campbelling (root mean square [RMS] signal processing) range covers in excess of 10-decades of power level. The self-contained microprocessor combines these signals and derives the power rate of change (period) through the full range of power. The microprocessor automatically tests the system to ensure that the upper decades are operable while the reactor is operating in the lower decades and vice versa when the reactor is at high power.

For the multirange function, the NM-1000 uses the same signal source as for the log function. However, instead of the microprocessor converting the signal into a log function, it converts it into 10 linear power ranges. This feature provides for a more precise reading of linear power level over the entire range of reactor power. The same self-checking features are included for the log function. The multi range function is either auto-range or slave to a position switch on the operator's console via the control system computer. A linear power level signal is available for the percent power safety function for 1 to 125%.

The NM-1000 system is contained in two National Electrical Manufacturers Association (NEMA) enclosures, one for the amplifier and one for the processor assemblies. The amplifier assembly contains modular plug-in subassemblies for pulse preamplifier electronics, bandpass filter and RMS electronics, signal conditioning circuits, low voltage power supplies, detector high-voltage power supply, and digital diagnostics and communication electronics. The processor assembly is made up of modular plug-in subassemblies for communication electronics (between amplifier and processor), the microprocessor, a control/display module, low-voltage power supplies, isolated 4 to 20 mA outputs, and isolated alarm outputs. Outputs are Class 1E as specified by IEEE. Communication between the amplifier and processor assemblies is via two twisted-pair shielded cables.

The amplifier/microprocessor circuit design employs the latest concepts in automatic on-line self diagnostics and calibration verification. Detection of unacceptable circuit performance is automatically alarmed. The system is automatically calibrated and checked (including the testing of trip levels) prior to operation. The checkout data is recorded for future use, and operation cannot proceed without a satisfactorily completed checkout. The accuracy of the channels is $\pm 3\%$ of full scale, and trip settings are repeatable within 1% of full-scale input.

The neutron detector uses the standard 0.2 counts per "nv" fission chamber that has provided reliable service in the past. It has, however, been improved by additional shielding to provide a greater signal-to-noise ratio. The low noise construction of the chamber assembly allows the system to respond to a low reactor shutdown level which is subject to being masked by noise.

6.1.2. NP-1000 Power Safety Channel

The NP-1000 Power Safety Channel is a complete linear percent power monitoring system mounted within one compact enclosure which contains current to voltage conversion signal conditioning, power supplies, trip circuits, isolation devices, and computer interface circuitry. The power level trip circuit is normally hardwired into the scram system and the isolated analog outputs are monitored by the CSC as well as being hardwired to a bargraph indicator.

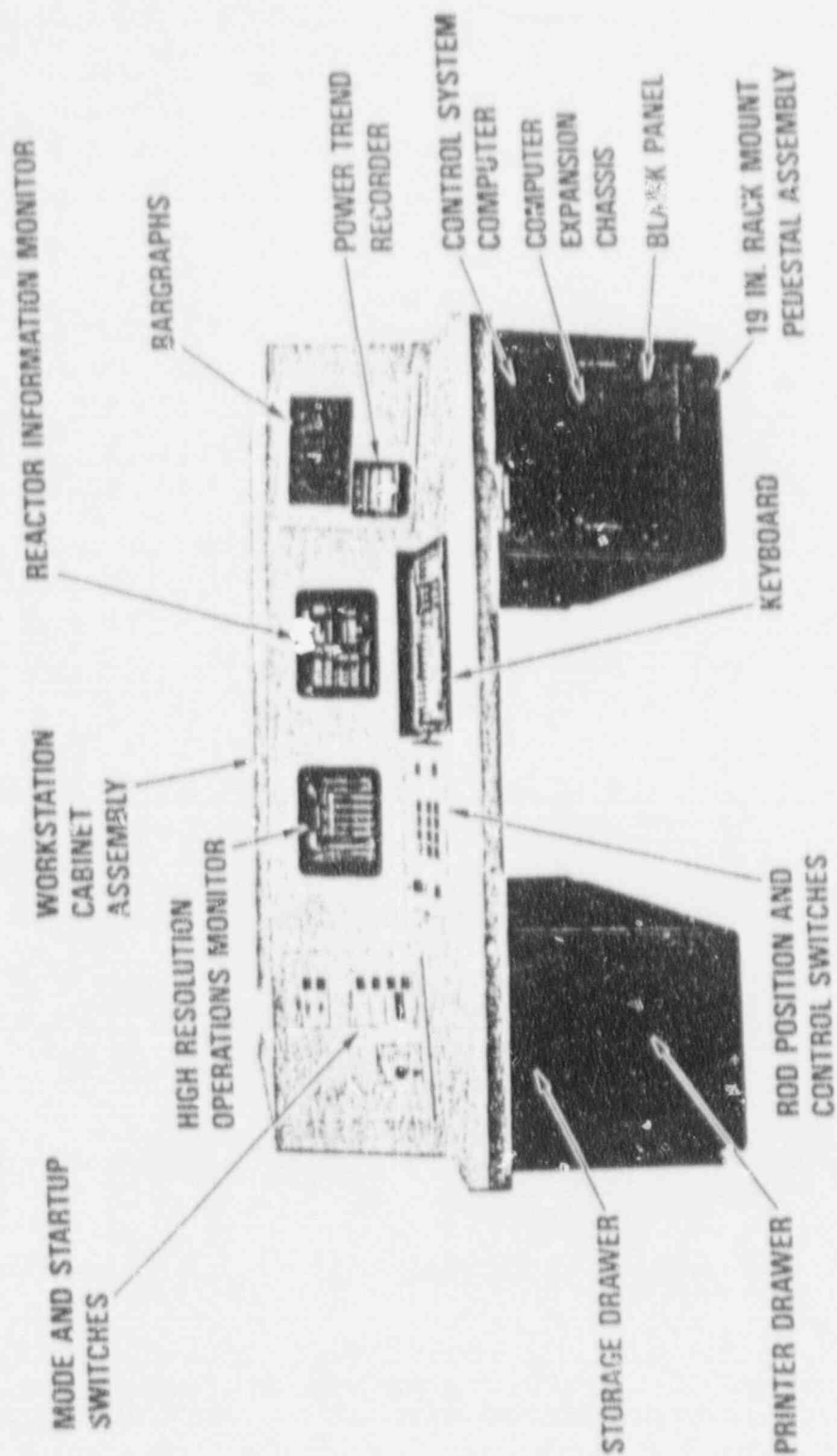
A special version of the safety channel, the NPP-1000, provides measurement functions for peak pulse power, total pulse energy, automatic gain change and related trip points. The control system automatically selects proper gain setting for steady-state or pulse mode when the operator determines the reactor operating mode. Peak pulse power and total pulse energy are also set by the pulse operation mode.

Both safety channels, the NP-1000 and the NPP-1000, are identical except for the peak and energy circuits. The detector for each safety channel is either an ionization chamber or self-powered in-core detector.

6.1.3. Reactor Control Console

A conceptual layout of the control console is shown in Figure 6-3. The reactor control console contains several components needed by the operator for reactor control. Included are the following:

- a. Reactor control panels.
- b. Control System computer (CSC).
- c. Two color graphics CRT monitors (one is high resolution).
- d. Power and temperature meter panels.
- e. Two disc drives and a graphics printer.



LAYOUT OF THE REACTOR CONTROL CONSOLE

Figure 6-3

A keyboard interface to the system computer is provided for operator control of several system functions. As previously mentioned, the power and period information from the NM-1000 channel and power level from the NP-1000 channels are processed and displayed by the CSC. However, several wide-range channel parameters are also present on linear bargraph meter displays at the console. Each NP-1000 safety system is independent, has its own output displays, and connects directly to the control system scram circuit. Thus, wide-range log power, period, multirange linear power and both percent power channels, have their output displayed on meters as well as on the color CRT. This is also true of fuel temperature. Typical layouts of the console panels and video displays are shown in Figures 6-4 and 6-5.

Functions of the rod control panel are represented in Figure 6-6 and are presented as:

- a. Key switch for rod magnet power (also operates "Reactor On" lights).
- b. Rod control switches and Annunciators.
- c. SCRAM-switch for safety function.
- d. Annunciation is also provided for reset of the audio channel, as well as for reset of the alarm indicator following alarm clearance.

The CSC provides all of the logic functions needed to control the reactor and augments the safety system by monitoring for undesirable operating characteristics. It displays reactor operational information in a color format on a high-resolution CRT monitor for ease of comprehension. Essentially all of the control systems logic contained in previous TRIGA reactor control systems is incorporated into the CSC. However, instead of using electronic circuits and electrical relay circuits, the logic programmed into the computer.

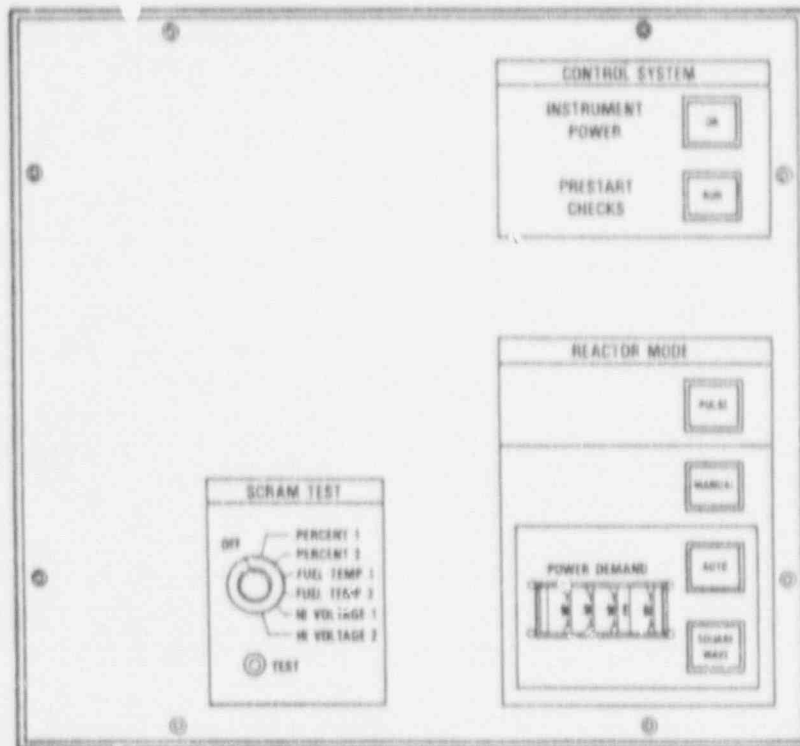
The availability of the computer allows great versatility and flexibility in operationally related activities aside from the direct control of rod movements. Many other functions can also be performed by the CSC, such as monitoring reactor usage, storing pulse data, reactor operating history and logging operator usage.

Two auxiliary cabinets can be provided to the console for the addition of process instrument readout.

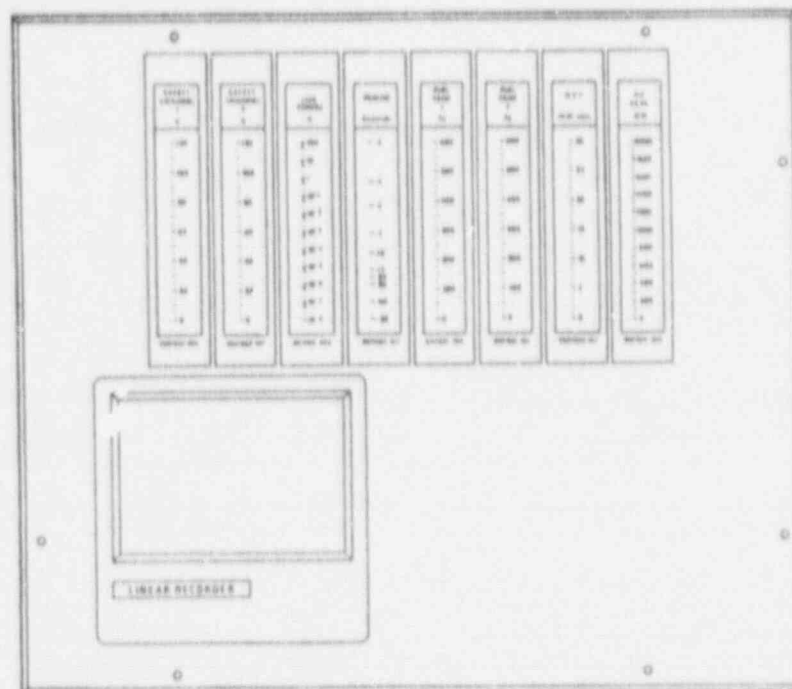
6.1.4. Reactor Operating Modes

There are four standard operating modes: manual and automatic, pulse, and square-wave.

The manual and automatic modes apply to the steady-state reactor condition; the pulse and square-wave modes are the conditions implied by their names and require a pulse rod drive.



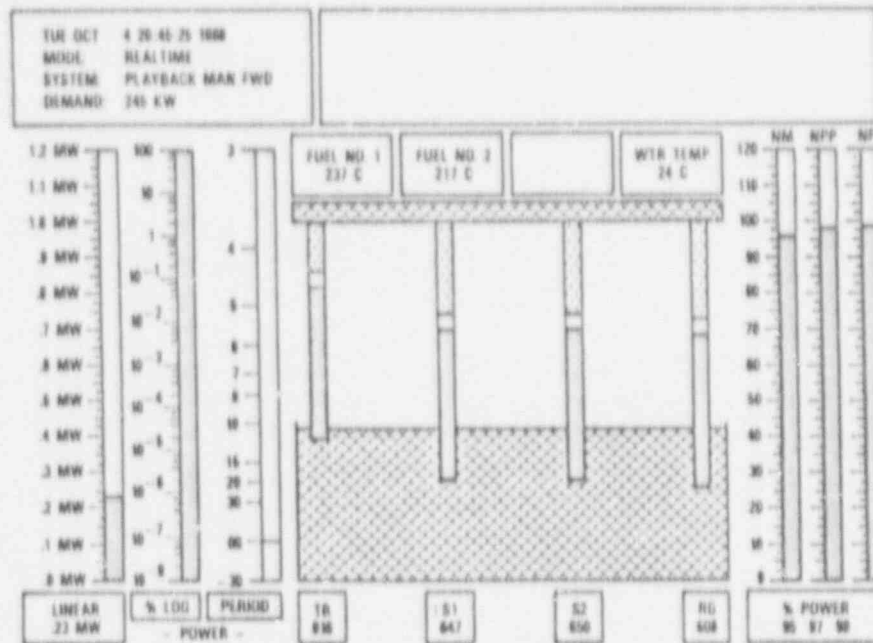
Mode Control Panel



Analog Display Panel

CONSOLE CONTROL PANELS

Figure 6-4



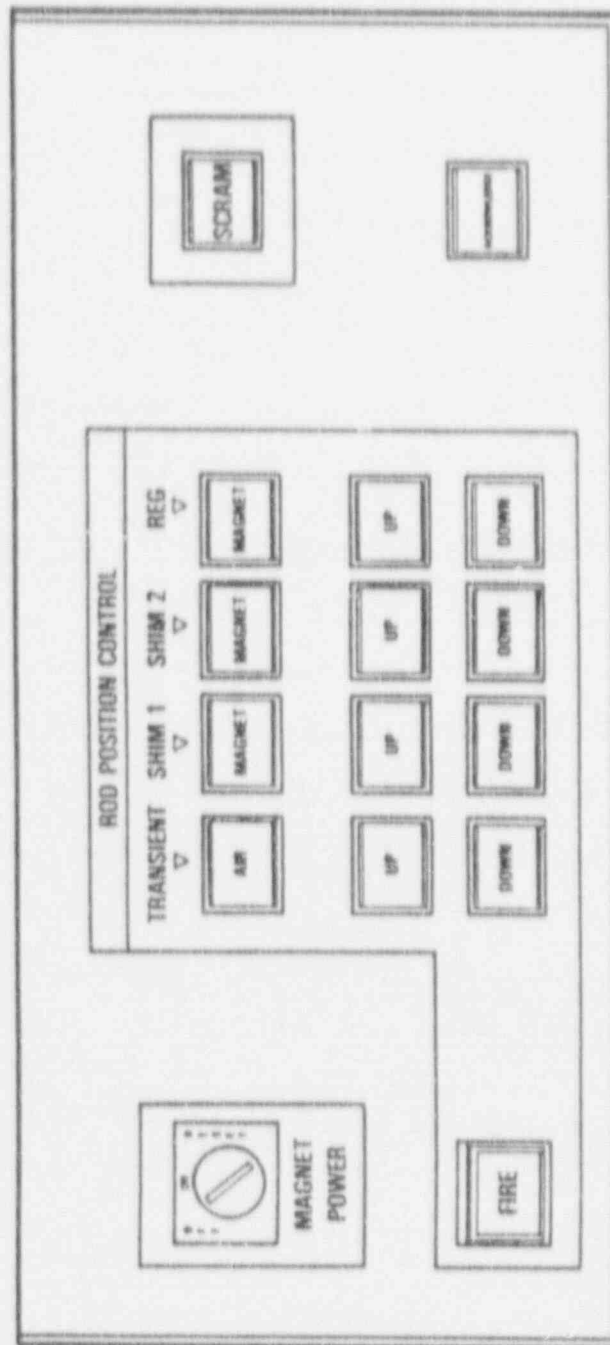
High-resolution graphic display

TEXAS STATUS WINDOW			
Linear Power	1.0e+0	Date	
% Power #1	0	Time	
% Power #2	0	Reactor Mode	SCRAM
Fuel Temp #1	25	Current Pulse Number	1
Fuel Temp #2	26		
Shim #1 Position	0	Control Room Arm	1.0e-1 mR
Shim #2 Position	0	Pool Surface Arm	2.0e-1 mR
REG Position	50	Area 1 Arm	2.0e-1 mR
Transient Position	600	Area 2-3 Arm	2.0e-1 mR
		Area 4-5 Arm	2.0e-1 mR
		Portable Arm	1.0e-1 mR
Min Source Interlock	OK	Particulate Monitor	200 cpm
Power >1, KW Interlock	OK	Stack AR-41 Monitor	20 cpm
Pool Level Lo/Hi	OK	Kx Bay Doors	OK
Primary Coolant Flow	0 gpm	Kx Bay Neg Air Pressure	OK
Secondary Coolant Flow	0 gpm		
HX Pool Water Inlet Temp	24.6°C	Beam Port 1	OK
HX Pool Water Outlet Temp	26.3°C	Beam Port 2	OK
Pressure Difference HX	OK	Beam Port 3	OK
Pool Temperature	23.4°C	Beam Port 4	OK
DEMIN Conductivity	2.02µmho	Beam Port 5	OK

Color text display

VIDEO DISPLAY DATA

Figure 6-5



ROD CONTROL PANEL

Figure 6-6

The manual and automatic reactor control modes are used for reactor operation from source level to 100% power. These two modes are used for manual reactor startup, change in power level, and steady-state operation. The pulse mode generates high-power levels for very short periods of time. High-power and low-power pulse mode options are available. The square-wave operation allows the power level to be raised quickly to a desired power level.

Manual rod control is accomplished by the lighted push buttons on the rod control panel. The top row of annunciators, when illuminated, indicates magnet contact with the armature and magnet current. Depressing any one of the AIR-MAGNET push buttons will interrupt the current to that magnet and extinguish the magnet current on indication. If the rod is above the down limit, the rod will fall back into the core and the AIR-MAGNET light will remain extinguished until the magnet is driven to the down limit where it again contacts the armature.

The middle row of pushbuttons (UP) and the bottom row (DOWN) are used to position the control rods. Depressing the pushbuttons causes the control rod to move in the direction indicated. Several interlocks prevent the movement of the rods in the up direction under conditions such as the following:

- a. Scrams not reset.
- b. Magnet not coupled to armature.
- c. Source level below minimum count.
- d. Two UP switches depressed at the same time.
- e. Mode switch in one of the pulse positions.
- f. Mode switch in AUTO position (regulating rod only).

There is no interlock inhibiting the down direction of the control rods except in the case of the regulating rod while in the AUTO mode.

Automatic (servo) power control can be obtained by switching from manual operation to automatic operation. All the instrumentation, safety, and interlock circuitry described above applies and is in operation in this mode. However, the regulating rod is now controlled automatically in response to a power level and period signal. The reactor power level is compared with the demand level set by the operator and is used to bring the reactor power to the demand level on a fixed preset period. Logic for the automatic control operation by proportional and integral control is contained within the digital algorithms of the control system. The purpose of this feature is to automatically maintain the preset power level during long-term power runs. The function of automatic control is provided by the regulating rod with a stepping motor drive.

Reactor control in the pulsing mode consists of establishing criticality at a flux level below 1 kW in the MANUAL mode. This is accomplished by the use of the motor-driven control rods, leaving the transient rod either fully or partially inserted. The mode selector switch is then depressed. The MODE selection switches automatically connect the pulsing chamber to monitor and record peak flux (nv) and energy release (nvt). Pulsing can be initiated from either the critical or subcritical reactor state.

In a square-wave operation, the reactor is first brought to criticality below 1 kW, leaving the transient rod partially in the core. All of the manual instrumentation is in operation. The transient rod is ejected from the core by means of the transient rod FIRE pushbutton. When the power level reaches the demand level, it is maintained much the same as in the automatic mode except that two rods are used to maintain power after the pulse rod is ejected.

6.1.5. Reactor Scram and Shutdown System

A reactor protective action [4] interrupts the magnet current and results in the immediate insertion of all rods under any of the following:

- a. High neutron fluxes from either NP-1000 or NPP1000.
- b. High-voltage failure on the NM-1000, NP-1000, or NP1000.
- c. High fuel temperature (one out of two).
- d. Manual scram.
- e. Peak neutron flux or energy (pulse mode).
- f. Minimum period (available for use as desired).
- g. External safety switches (for experiments).
- h. Loss of electrical power to the control console
- i. Watchdog circuits for each computer to monitor computer status by updating timers.

All scram conditions are automatically indicated on the CRT monitor. A manual scram will also insert the control rods and may be used for a normal fast shutdown of the reactor. The scram circuit safety function is an independent system that depends on wiring independent of the digital control system functions.

Several conditions of the digital processing system will cause the scram mode condition. Among these are the loss of communication between the two computers, a database timeout condition or failure of a digital input scanner. By updating dual programmable timers, watchdog circuits at periodic intervals, determine the execution status of key elements of the computer digital program.

Two options for reliable operation performance may be installed as necessary. One option for conditions requiring long-term, high power steady-state operation, is configuration of the three safety channels with 2 out of 3 logic, allowing one channel to be out of service without requiring reactor shutdown. Another option is an uninterruptable power supply as auxiliary power for the reactor control and monitoring systems for intermittent power failures of periods upto 15 minutes.

6.1.6. Logic Functions

A simplified control system logic diagram is shown in Figure 6-7. The two separate flux monitoring safety channels ensure safe operation of the reactor by monitoring the power level and act independently to shut the reactor down if a potentially dangerous condition exists. They provide information to the control system, which consists of three major parts: a reactor control console (RCC), Control System Computer (CSC) and Data Acquisition Computer (DAC). In addition, there are two color graphics CRT monitors and a graphics printer. The high resolution display monitor contains basic reactor operation control data. A medium resolution display monitor provides information on annunciators and special control features. Data from both displays may be sent to the printer for a log record.

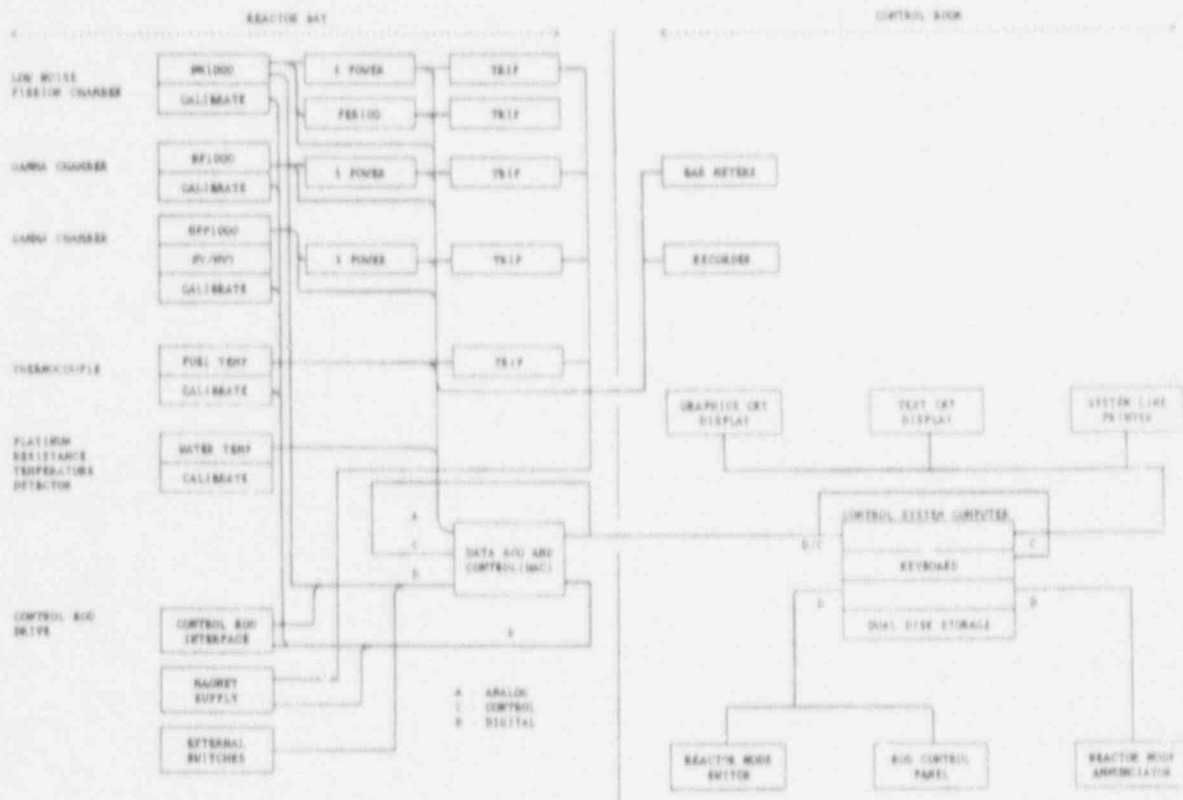
The CSC provides the operator with immediate information concerning reactor conditions visually on the monitors. At the same time, the DAC is collecting data from the reactor system and concentrating it into a permanent data base, which is transmitted to the CSC on request and maintained for historical purposes.

During operation of the reactor, the operator's commands to adjust control rod positions are transmitted from the CSC to the DAC to the drive mechanisms. In the automatic mode the DAC controls the position of the rods. The rod control program for automatic operation applies proportional-integral-differential control logic. Digital rod position indication is shown in inches, with a resolution of ≤ 0.1 in. and an accuracy equal to or better than $\pm 0.2\%$ of indicated position.

The control rod interface accepts the digital commands from the data acquisition and control system (DAC) to operate the control rod motors. It contains the opto-isolation circuits which send the up-down limits and loss of contact signals to the control rod logic system. An excitation power supply provides a stable reference voltage for the rod position indicator system.

The magnet supply furnishes the required 200 mA needed for the rod magnets to hold control rods in contact with the armature. An opto-isolator detects the absence of magnet current to each drive magnet.

A gamma chamber provides the signal for peak power (nv) and energy release (nvt) in the pulse mode. The nv/nvt amplifier provides the high impedance interface, high voltage and calibration circuits for the pulsing detector.



LOGIC DIAGRAM FOR CONTROL SYSTEM

Figure #7

All of the analog signals and digital signals are routed to the DAC chassis. However, the prime reactor operating signals are also sent directly to the control room. These signals include log power, period, percent power (2), fuel temperature (2), and pulse mode signals for peak and energy.

The DAC system converts the analog signals to a digital equivalent for transmitting along with the digital signals to the CSC in the control room. The DAC chassis receives control instructions from the CSC, via the communication link, which in turn moves the control rods as requested by the operator and causes the individual subsystems to go to the calibrate mode when commanded by the system or operator.

The fuel temperature transmitters are accurate, highly stable units which convert the 0-600°C fuel temperature into a 4-10 mA output signal. A level comparator is included which provides scram capability through an isolated contact state change when the preset level is exceeded.

The water temperature transmitters are standard Resistance Temperature Detector (RTD) transmitters which convert the 0 to 100°C temperature into a 4-20 mA signal. The transmitters have a self-contained power supply.

External switches are provided with terminal strips to terminate and connect various switches to the DAC chassis (beam port open-close, etc.)

6.1.7. Mechanical Hardware

Typical reactor installation will be contained in two NEMA enclosure junction boxes, one electronic equipment cabinet, separate stepping motor power supplies installed in the reactor bay, and reactor operator console components installed in the reactor control room.

The control console consists of the components needed by the operator for reactor control. These components include rod control switches and annunciators, the digital rod position indicators, on-line reactor status meters (power and temperature), the control system computer (CSC), reactor operating mode switch panel, color CRT monitor, printer, disc drives (2) and external switch annunciators (beam port open-close, reactor access, etc.).

Enclosure 1 contains NM-1000 high and low voltage power supplies, a pulse pre-amp with discriminator, an RMS Campbell convertor and a communications module.

Enclosure 2 contains the NM-1000 microprocessor selected to provide the 10-decade log signal and the multi-range linear function from the information provided by the circuits in enclosure 1. The information processed by the microprocessor is 10-decades of log power, rate of power change (period), multi-range linear function, linear percent power from 1 to 125%, level trips from the log and linear percent power, calibrate and failure signals.

Enclosure 3 is a standard rack type equipment enclosure for electronic components. Space in the enclosure provides the terminal strips for connections to the various signal detection systems and the communications to the RCC. The cabinet enclosure includes eight shelves with functional separation between shelves. Power supplies for subsystems are on shelf 1. Shelves 2 and 3 contain, respectively, ac digital and dc digital circuits for processing input or output circuits. Shelf 4 provides several special modules for signal processing. The two power safety channels are positioned on shelf 5. Shelf 6 contains the regulating rod drive control signal module and signal terminations. The computer and its expansion chassis comprise the final two shelves 7 and 8. The regulating rod drive translator for the stepping motor drives is contained in a separate, fourth enclosure.

6.2 DESIGN EVALUATION

The TRIGA reactor console [6,7,8] has developed through the successful operation of many installed facilities throughout the world. Design of the new ICS unit incorporates similar basic logic functions proven effective in prior designs. Incorporation of digital electronic techniques in the design to replace analogue circuits is justified by improved performance. Functional self-checks, circuit calibrations, and automated data logging are implemented effectively and efficiently.

Installation and verification of the original design is planned for a reactor operated by the ICS manufacturer, GA Technologies. Subsequent installation of the ICS unit at The University of Texas facility is planned with appropriate design changes, inclusion of facility specific parameters, and completion of an acceptance evaluation.

A multiphase design, development and installation program by the system manufacturer provides the initial demonstration of the system acceptance by analysis and review. No license modification was found necessary to implement the new digital system in place of the old analog system. An evaluation of the system relative to the facility safety analysis report was the basis for system evaluation. The final test phase and subsequent operation will provide information on system performance and reliability. The analysis and test program determined 1) that there was no increase of the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety, 2) that the system does not create the possibility of an accident or malfunction of a different type and 3) no reduction occurs in the margin of safety as defined in the basis for any technical specification.

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8. "Operation and Maintenance Manual NP1000/NPP1000 Percent Power Channel", E117-1010, General Atomics 1989.

Chapter 7

DESIGN FEATURES AND AUXILIARY SYSTEMS

Features and systems of the facility design provide several functions that determine safety conditions and auxiliary support of operations. Engineering features include the reactor shield system and the air confinement system for the reactor bay room and fuel storage systems. Auxiliary systems consist of standard systems for conditioned air, fire protection, communication equipment, and other systems.

Engineering safety features of the TRIGA reactor are a property of the fuel material and element design. These engineering safety features include two features of the reactor core and two features of the instrumentation control and safety system. Engineering safety features of the reactor core are the fuel material type and the fuel element cladding. Instrumentation, control, and safety system features are the design of the "scram" safety circuit and the design of the actuators for rod drives and control rods. The description of each of these design features are found in other chapters of this report.

Building design features important to safety of the reactor are the reactor pool and shield structure, and the air confinement system for the reactor room. These systems are not complete engineering safety features but may require specification of specific conditions or parameters to assure the appropriate safety function.

7.1 DESIGN BASES

The design of a structure to contain the TRIGA reactor depends on the protection requirements for the fuel elements and the control of exposures to radioactive materials. Fuel elements and other special nuclear materials are protected by physical confinement and surveillance. The physical confinement will also control the release of radioactive materials during routine operation or potential accident conditions. Release of airborne radioactivity consists mostly of air activation products from routine operation or fission product materials from a non-routine fuel element failure. Liquid and solid radioactive material are also controlled to assure compliance with appropriate release criteria standards. Other potential releases may be associated with specific types of experiments that require special equipment to provide sufficient control of material releases. The reactor room confinement is designed to control the exposure of operation personnel and the public from radioactive material or its release caused by reactor operation. Release criterion are based on Title 10 Chapter 20 of the U.S. Code of Federal Regulations [1].

7.2 DESIGN FEATURES

Two design conditions provide the primary safety features of the facility. One feature is the design of the reactor pool and shield system. The other feature is an air confinement system. These facility design features supplement safety features of the fuel element design, reactor core assembly and fuel storage systems.

7.2.1. Reactor Pool and Shield Structure

Pool water system and shield structure design combine to control the effective radiation levels from the operation of the reactor. One goal of the design is a radiological exposure constraint of 1 mrem/hour for accessible areas of the pool and shield system. Dose levels assume a full power operation level of 1.500 megawatts (thermal). Radiation doses above the pool and at specific penetrations into or through the shield may exceed the design goal. Figure 7-1 displays the basic design dimensions of the pool and shield system with some of its features. Representative 1 mrem/hour dose curves for a reference case design are shown in Figure 7-2. The reference case design is a solid structure without any system penetrations. Design of the reactor pool was of 1/2 inch (1.27 cm) base plate and 1/4 inch (9.64 cm) wall plate of 6061 aluminum alloy. Tank assembly is by shop fabrication. A protective layer of epoxy paint and bitumen coal tar pitch with paper provide a barrier between the aluminum pool tank and the reactor shield concrete.

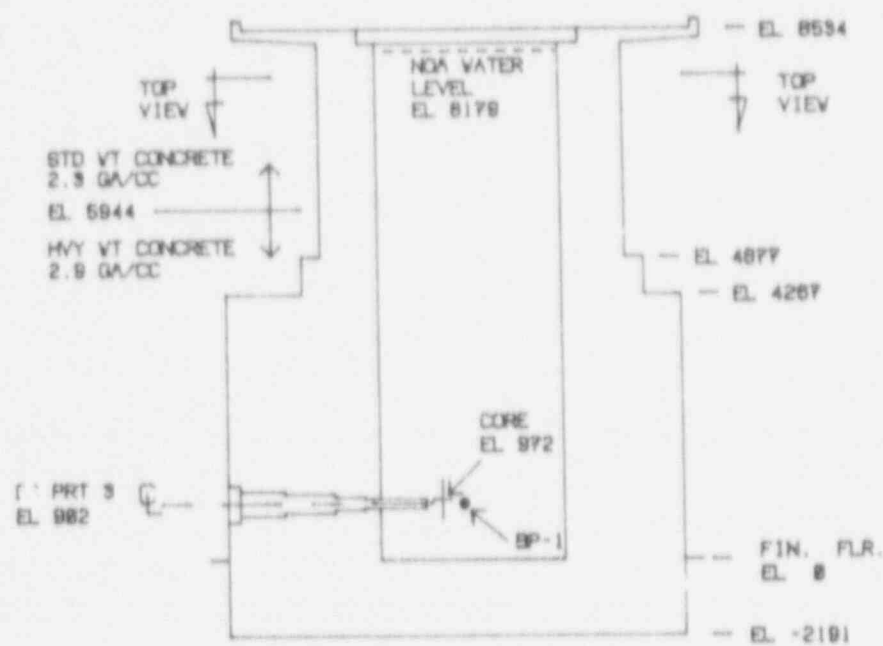
A four foot (1.22 meter) thick foundation pad supports the reactor pool and shield structure. Standard weight concrete, 150 lb/ft³ (2.33 g/cm³), comprises the foundation pad. High density concrete, 180 lb/ft³ (2.89 g/cm³), with a magnetite aggregate is the shield material of the first level of the shield structure. A transition from high density to standard density concrete is present about 4.5 feet (1.4 m) above the mid level platform of the shield. The top part of the shield stem and the top level platform are standard density concrete. The total shield weight is 2.03×10^6 lbs. (920 metric tons). Approximately 24,400 lbs. (11,100 kgs.) of structural steel, 56 conduits for signal and electrical lines with diameters of 1/2 inch to 3 inch, three central junction boxes and numerous local junction boxes are part of the shield system.

Five beam tubes at the level of the reactor provide experimental access to reactor neutron and gamma radiations. Two of the tubes combine to penetrate the complete reactor pool and shield structure from one side to the other side.

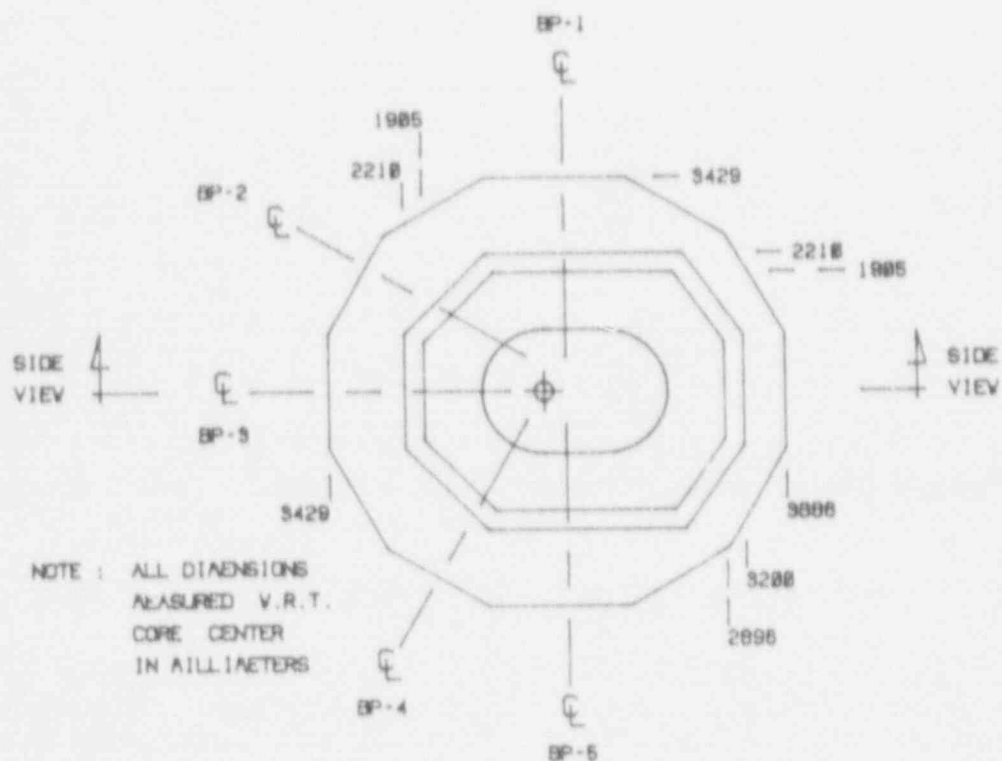
Special design features of the beam tubes are beam plugs, sliding lead shutters, bolted cover plates, and gasket seal for protection against reactor radiation and coolant leakage when the tubes are not in use.

7.2.2. Reactor Bay Ventilation Design

Ventilation design is specified to control air confinement and to isolate the reactor bay in the event a radioactive release is detected in the reactor areas. The ventilation system is designed to maintain a negative pressure within the reactor bay with respect to the building exterior and other building areas. Confinement and isolation is achieved by air control dampers and leakage prevention material at doors and other room penetration points. A separate system is designed to exhaust air from several locations within the reactor bay that could contain airborne radionuclides such as argon-41. Manual operation of start/stop controls of both main and purge air systems will be available in the reactor control room.

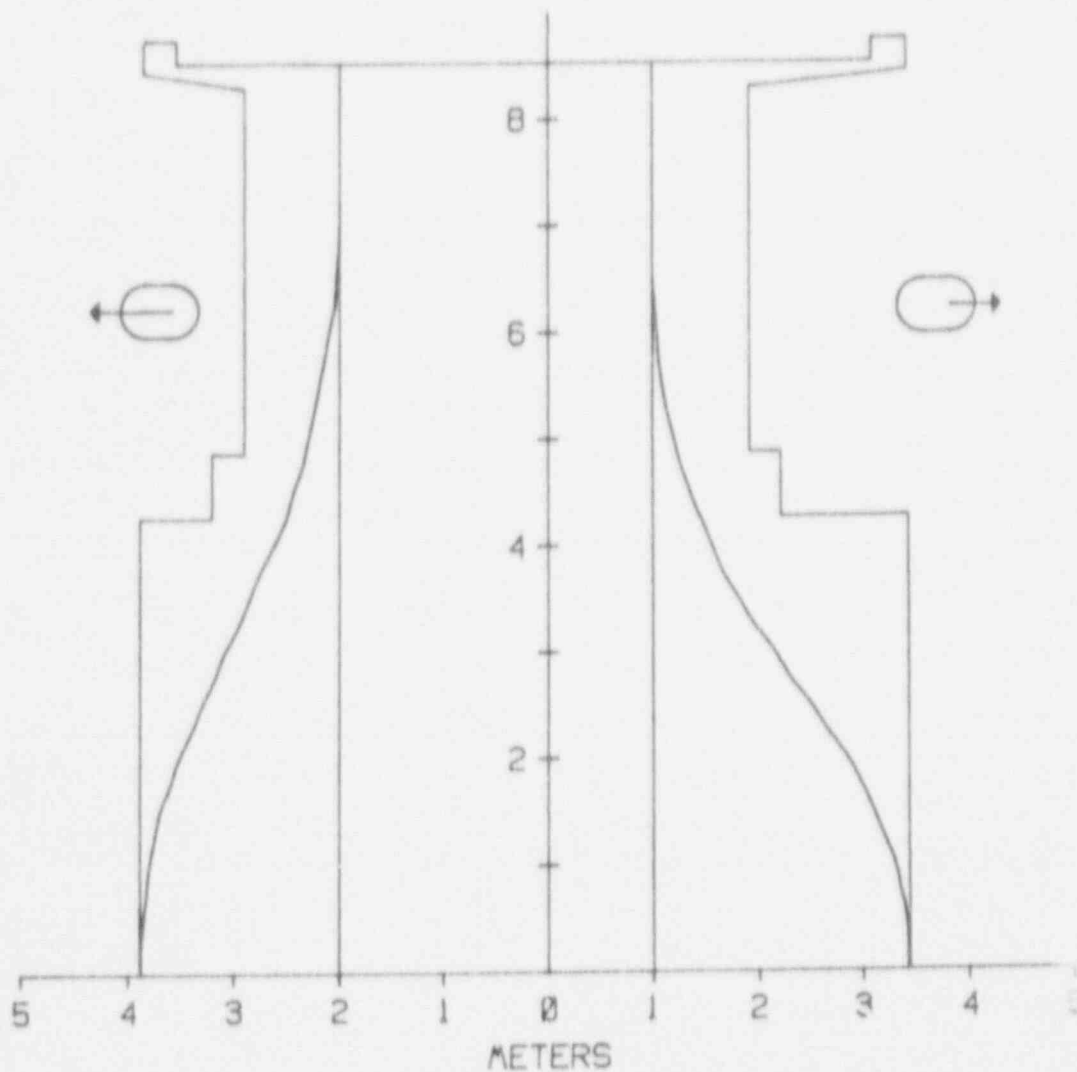


NOTE: DIMENSIONS IN MILLIMETERS



REACTOR SHIELD STRUCTURE

Figure 7-1



ISODOSE CURVES FOR
1 MILLIRAD / HOUR
AT 1.5 MEGAWATTS

ISODOSE CURVES FOR SHIELD STRUCTURE

Figure 7-2

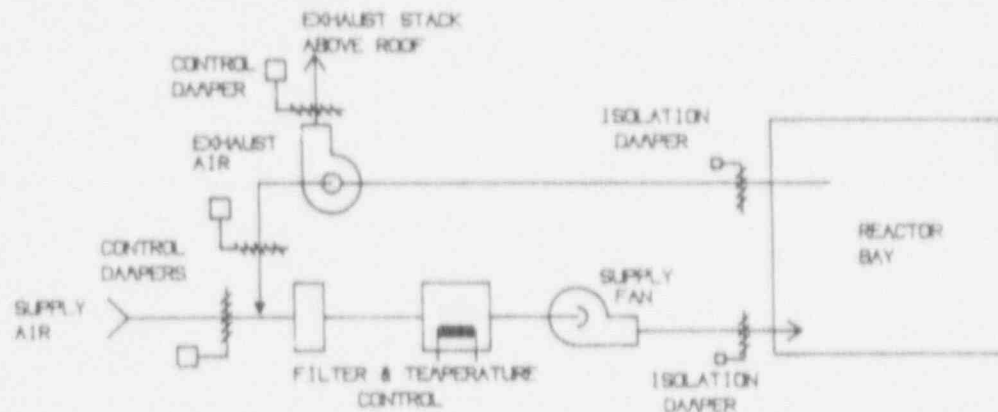
Ventilation of the reactor bay is provided by two modes of system operation. One mode is for standard operation with recirculation of air. The other mode is an exhaust operation with high volume flow that has no air recirculation. Design during exhaust mode operation is a rate of air exchange in excess of two per hour. Total volume for the room exceeds 4120 cubic meters. Normal operation of the ventilation system uses a roof stack for the exhaust of air from the reactor bay. Air filtration in the ventilation system is to be of typical design for normal HVAC operation with no special provisions. Schematics of the ventilation system for the reactor bay area and a logic diagram of the ventilation control system sensors and controls are provided in Figure 7-3.

Control of air confinement within the reactor bay is provided by differential pressure control between the reactor bay and a representative ambient external measurement point. Additional measurement points in ventilation zones adjacent to the reactor bay maintain the differential pressure between the reactor bay and adjacent access areas. The differential pressure control is intended to function in both standard and exhaust operation modes of the ventilation system.

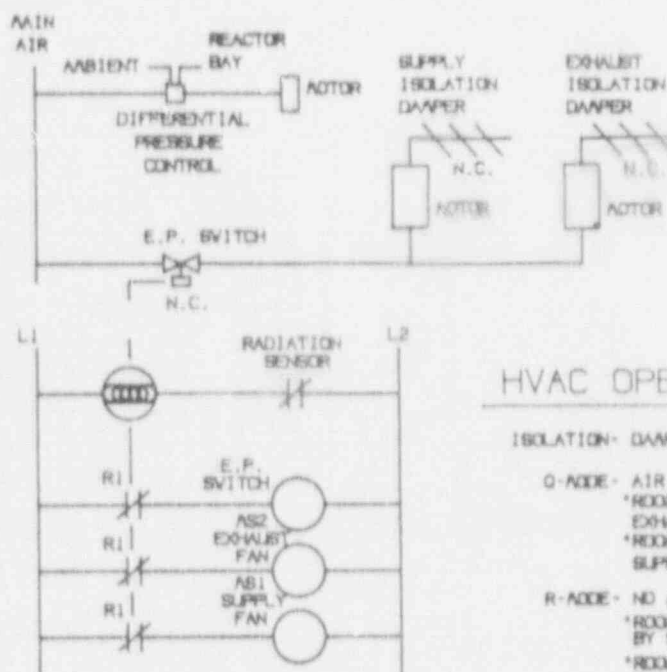
Isolation of the reactor bay is provided by ventilation dampers. These dampers will shut in response to either manual or automatic signal actuation. An automatic signal will initiate shutdown of the ventilation system by closure of the dampers if a set point for airborne particulate radioactivity exceeds a setpoint. Protective switches within the ventilation system will cause the air fans to respond to the position change of the dampers. Damper design is for fail-safe operation so that loss of control power will isolate the reactor bay. Dampers locations are in the vicinity of the duct penetrations into the reactor bay. An isolation damper is in each of two supply air ducts. One return air duct with two sections contains two isolation dampers, one in each section. A pair of return air ducts also contain a damper in each duct.

The separate air purge system is designed to exhaust air that may contain radionuclide products by a low volume system. The primary nuclide of interest is argon-41. Figure 7-4 shows a schematic of the argon purge system and its control logic. Air from potential sources of neutron activation such as beam tubes, sample transfer systems, and releases from exchanges at the pool water surface are subject to confinement and isolation by the system. Filtration of air in the system will include prefilter and high efficiency particulate filter. Design provisions allow for the addition of charcoal filters if experiment conditions should require the additional protection. Sample ports in the turbulent flow stream of the purge system exhaust provide for measurement of exhaust activities. Actuation of the isolation damper in the argon purge system is by manual operation of the fan control switch.

A schematic of each ventilation system is shown in Figure 7-5.



MAIN AIR SUPPLY SYSTEM



HVAC OPERATION MODES

ISOLATION- DAMPERS SHUT, FANS OFF

Q-MODE- AIR RECYCLE

- *ROOM AIR PRESSURE CONTROLS EXHAUST FAN SPEED
- *ROOM SPACE COMFORT CONTROLS SUPPLY FAN SPEED

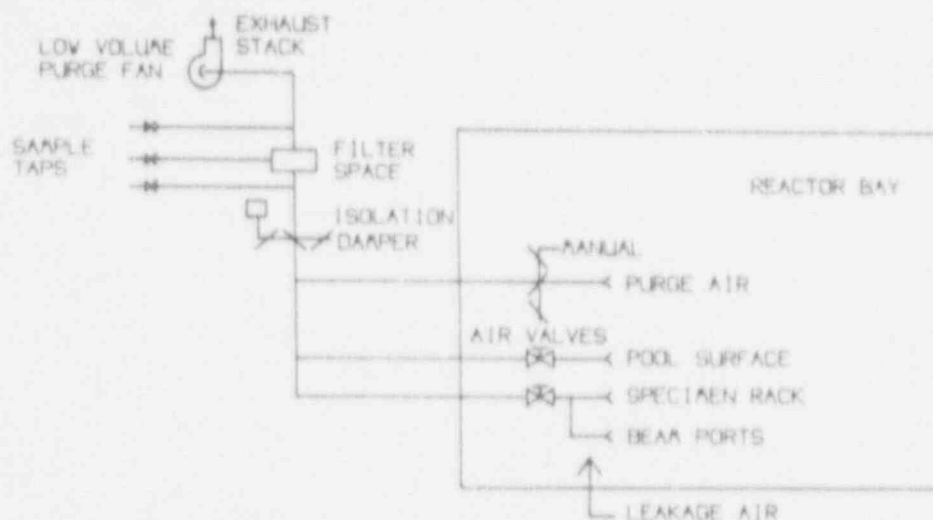
R-MODE- NO AIR RECYCLE

- *ROOM AIR PRESSURE CONTROLLED BY SUPPLY FAN SPEED
- *ROOM AIR CHANGE CONTROLLED BY STACK EXHAUST VELOCITY

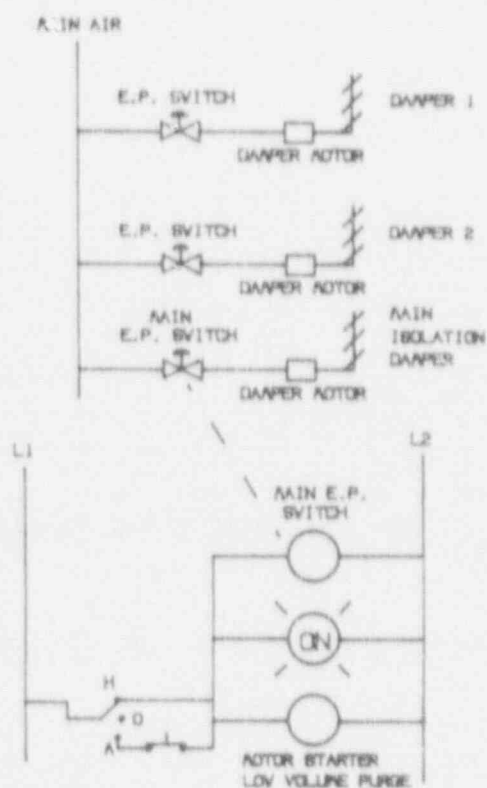
MAIN AIR SCHEMATICS

REACTOR BAY AIR VENTILATION SYSTEM

Figure 7-3



AUXILIARY AIR EXHAUST SYSTEM



AUXILIARY AIR SCHEMATICS

REACTOR BAY AUXILIARY EXHAUST SYSTEM

Figure 7-4

An exhaust stack on the roof combines the ventilation exhaust of both the main and purge systems. Mixing occurs at the exhaust exit point. The exhaust stack will extend 14 feet (4.24 meters) above the roof level. The effective release point above the exhaust stack can be calculated from the equation

$$\Delta h = D \frac{(V_g)^{1.4}}{\bar{\mu}} \quad (\text{Bryant-Davidson formula}), \quad (1)$$

where

Δ = height of plume rise above release point, m,

D = diameter of stack, m,

$\bar{\mu}$ = mean wind speed at stack height, m/s,

V_g = effluent vertical efflux velocity, m/s.

The following values were used for this calculation

$D = 0.46$ m,

$\bar{\mu} = 4$ m/s,

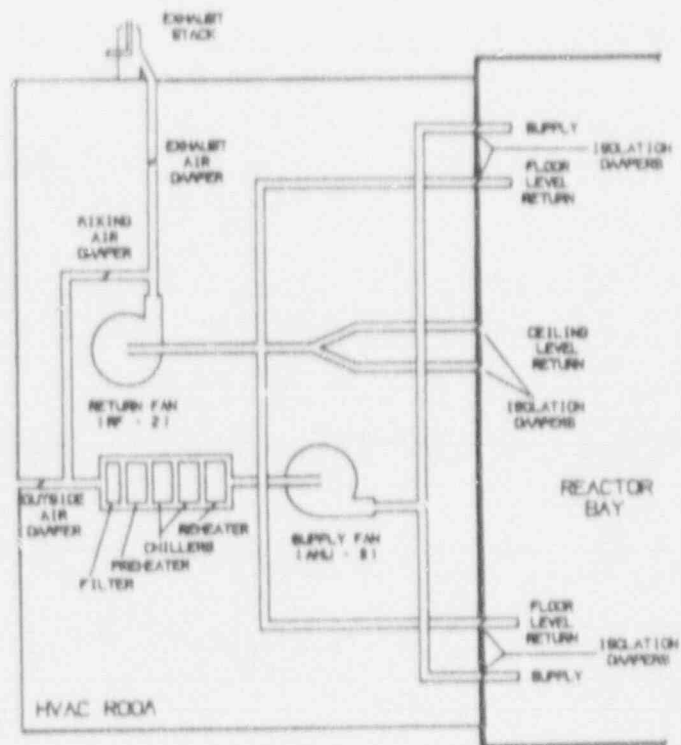
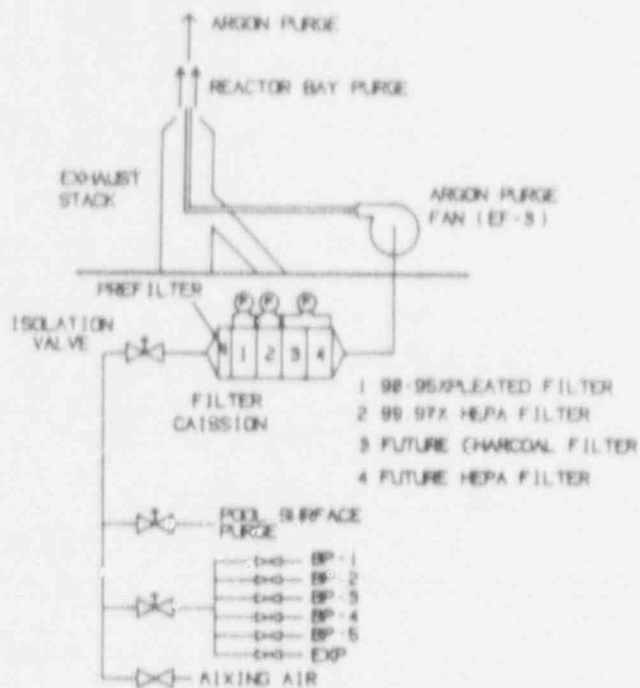
$V_g = 12.2$ m/s,

then

$$\Delta h = 0.46 \frac{(12.2 \text{ m/s})}{4 \text{ m/s}},$$

$\Delta h = 2.2$ m.

Ground elevation in the area is 794 feet. Roof elevation at the stack is 843 feet. Therefore, the effective release point is at least 60 feet above the maximum ground elevation at the building.



SCHEMATIC OF VENTILATION SYSTEMS

Figure 7-5

7.2.3. Fuel Materials

Design of reactor fuel elements and reactor core assembly are specified for conditions that exceed the normal operation of the system. Special provisions are necessary for the storage of fuel elements that are not in the core assembly. The design of fuel storage systems requires consideration of the geometry, cooling, shielding, and the ability to account for each of the fuel elements. These storage systems will be racks for use both inside the reactor pool and outside the reactor shield.

Inside the reactor pool several racks will determine the total capacity of the storage. The racks are aluminum, suspended from the pool edge by connecting rods. Elements are stored six per rack in a linear array. Each rack is 24 inches long (60.96 cm.) by 12 inches wide (30.48 cm.) by 3 inches (7.62 cm.) deep and is generally located below more than 8 feet of water shielding. To facilitate extra storage, 2 racks may be attached to the same connecting rods by locating one rack at a different vertical level and offsetting the horizontal position slightly. If a storage requirement of 80% of the core grid capacity is specified, then 16 racks with a total of 96 positions would be necessary.

Outside the reactor pool, rack design is intended to fit in special storage wells. The wells are pits in the reactor floor that are fabricated of 10 inches (25.4 cm.) diameter stainless steel pipe. Six wells are designed each 15 feet (4.57 m.) deep and located 3 feet (0.91 m.) from the adjacent well. Each storage well pit will contain a shield plug and features to control access and to circulate water. Nineteen elements may be stored in each well and water added for shielding or cooling. Spaces in the rack will provide a storage array for the fuel equivalent to the innermost 3 rings of the reactor core. This includes the placement of one element in the center.

Most routine fuel storage is intended to be inside the reactor pool. Outside the reactor pool, supplemental fuel storage is planned for temporary storage of elements transferred to or from the facility, for isolation of fuel elements with clad damage, emergency storage of elements from the reactor pool and core assembly and routine storage of other radioactive materials. Temporary storage for some reactor components or experiments may also use the fuel storage racks in the reactor pool. Other locations not in the pool will also provide storage for radioactive non-fuel materials.

7.2.4. Safety Feature Evaluation

Design and features of the reactor pool and shield structure systems, air confinement system, and fuel storage systems are similar to those of other TRIGA installations. Operation at these installations demonstrates the performance of the system. Evaluation of the performance indicates that the basic and typical design characteristics provide safe and reliable performance.

7.3 AUXILIARY SYSTEMS

Several systems that supplement the purpose of reactor safety are standard applications of the applicable construction codes or typical of university specifications. In a few cases, specifications particular to this installation or its systems will be applicable. A few of the systems of importance will be those for fire protection, user comfort, communications, and utility services. These systems are building systems and not specific to the reactor bay.

7.3.1. Life Safety and Fire Protection

Design specifications are to meet life-safety requirements appropriate for the conditions. These specifications will set requirements for emergency lighting, stairwells and railings, exit doors, and other building safety features. An emergency shower and eye wash are available in the hallway adjacent to laboratory areas.

The goal of fire protection shall be to provide reasonable assurance that safety-related systems perform as intended and that other defined loss criteria are met [1,2]. For the purpose of fire protection, loss criteria should include protection of safety-related systems, prevention of radioactive releases, personnel protection, minimization of property damage, and maintenance of operation continuity. Three components shall be applied to the fire protection objective. The three components are passive and active fire protection, and fire prevention.

Each of the three components of the fire protection program shall be applied to the design, operation and modification of the reactor facility and its components. Fire prevention is primarily a function of operation rather than design.

7.3.2. Passive Fire Protection Elements

Passive fire protection should provide a potential for fire safety that does not require physical operation or personal response to achieve the intended function. Passive elements to be considered should include inherent design features, building physical layout, safety-related systems layout, fire barriers, and construction or component materials. Other passive elements that may be considered for special conditions will be frangible walls for overpressure relief, curbs for containment of hazardous liquids, and drainage for control of fire protection runoff water. Penetrations in fire barriers shall have fire resistant ratings compatible with the purpose of the fire barrier.

Safety-related systems shall incorporate passive fire protection features that provide protection necessary for the design functions of the system. Separation of redundant system components, if applicable, and protection of distribution systems should be examined. Materials of noncombustible or limited combustion properties should be used when practical.

Materials such as sealing materials, electrical insulation, structural finishes, adhesives, and linings should be selected so as to minimize fire hazards. Material selection may consider characteristics such as calorie content, ignition properties, flame retardation, and rate of heat release.

Elements of the passive fire protection include the structural construction system and the architectural separation into two separate buildings. Building structural materials are concrete cast in place for foundation, support columns and roof. Steel beam, metal and concrete deck comprise the reactor bay roof. A built-up composition roof with fire barrier materials complete the roof system. Pre-cast panels that are cast at the construction site cover 75% of the external perimeter. Metal paneling covers the other 25% of the perimeter. Design and installation of systems and components are subject to the applicable building codes.

The common wall between the academic wing and the reactor bay wing of the building is to be a fire barrier. Doors between these two building sections and other penetrations such as HVAC chases will conform to applicable codes. Although a few metal stud and plaster board walls will be in the reactor bay wing, the typical wall system is of concrete block construction.

Basic design features of the reactor assembly, pool and shield system, and the instrumentation, control, and safety system represent passive fire protection elements. These basic features are sufficient passive protection to protect safety-related systems.

7.3.3. Active Fire Protection Elements

Active fire protection elements differ from passive elements in that active features require automatic operation, manual response, or personnel action for the intended function. Active elements to be considered should be automatic fire detection, automatic fire suppression, fire information transmission, manual fire suppression and other manual fire control or loss control measures.

Manual protection shall consist of manual fire fighting actions and the systems necessary to support those actions such as extinguishers, pumps, valves, hoses, and the inspection, maintenance and testing of equipment to assure reliability and proper operation. Other manual actions that are elements of active fire protection shall be utility control, personnel control, and evacuation. Preplanning shall be an additional element applied to active protection by training facility personnel and emergency personnel and the appropriate actions in response of fire and the possible hazards involved.

Automatic and manual protection systems in the building include several different type systems. In the academic wing of the building automatic protective actions are provided by a sprinkler system with heat sensitive discharge nozzles, detectors for heat and smoke, and ventilation systems dampers. The reactor bay wing will use detectors for smoke and heat for areas that exclude the reactor bay. The reactor bay ventilation system will have smoke detectors that provide a warning of problems within the reactor bay. Although not a strict safety requirement, a gaseous extinguisher system may be functional to protect the reactor instrumentation, control and safety system. Manual protection equipment will include a dry stand pipe system in each building stairwell. Portable extinguishers such as CO₂, halon and dry chemical will be in specific locations throughout the building.

7.3.4. HVAC System

Personnel comfort conditions in the building use air handling units that have cold and hot water coils for temperature control. Two separate HVAC systems consisting of a total of three air handling units provide all the air ventilation and comfort. One unit with both cold and hot water coils is a single duct system that serves the reactor bay. The other two units are the cold and hot deck components of a double duct system that conditions air in all building zones which exclude the reactor bay. Both air handling systems are adjacent to the reactor bay on the fourth level of the reactor bay wing.

Water temperatures of the heating and cooling coils in the air handling units are controlled by both on-site and off-site systems. The heating system is an on-site boiler unit with a design capacity set by local building (HVAC) requirements. The cooling system is an off-site treatment plant with design capacity set by overall research center requirements. Thermostats control zone or room temperatures. An instrument air system will provide control air for HVAC systems. Controls and air balancing of the two air handling systems are to provide user comfort and pressure differentials between the reactor bay and adjacent zones, and between the adjacent zones and the academic wing of the building.

Relative to ambient atmospheric pressure the design goals for the reactor bay, adjacent zones and academic wing of the building is a negative pressure difference. The differential pressures are 0.06: 0.04: 0.03 inches (0.15: 0.10: 0.80 cms.) of water. Air balance values relate typical or standard values except in the case of the reactor bay.

7.3.5. Communications and Security

A communication system of typical telephone equipment will provide basic services between the building and other off-site points. Supplemental features to this system, such as intercom lines between terminals or points within the building and zone speakers for general announcements are to provide additional communication within the building. The telephone system is installed and maintained by the university. Connection of the main university telephone system is to standard commercial telephone network.

Telephones with intercom features are to be located at several locations in the building. Among these locations will be the receptionist office, the reactor control room and the reactor bay. By use of the intercom feature, each of these locations will be able to access public address speakers in one of several building zones.

A video camera system and a separate intercom system may supplement the building telecommunication network. These two systems are intended to contribute to safe operation by enhancement of visual and audio communication between the operator and an experimenter. Each system should have a central station in the control room with other remote stations in experiment areas.

7.3.6. Compressed Air and Deionized Water

The compressed air system will provide air for general applications and the operation of the transient control rod system. Design of the system will consist of a dual compressor with filter, dryer and regulator for supply air. One air supply line with valves, regulator and surge tank will supply air to the transient control rod drive.

The deionized water system will provide water for laboratory applications and replacement water for the reactor pool system. A system with replaceable resin containers, pipes, valves, and conductivity indication performs treatment of the domestic water supply. At periodic intervals either batch quantities or continuous replenishment of the pool water will be made from the deionized water system.

7.3.7. Utilities

Design of systems for electric power, cold water, hot water, chilling water, waste water, and sanitary sewer are similar to other university installations. Specifications for these systems are the responsibility of physical plant organization for operation, maintenance and repairs. Other systems in the building include a natural gas supply and vacuum system. Computer networks may be available for experiment systems.

7.3.8. Hazard Liquid Waste

Drains from several sinks and the emergency shower will be part of a liquid hazardous waste system. The system includes drains, traps, stainless steel pipe, valves, vent, and three waste storage tanks. Design will allow isolation of each tank, a double valve to drain any tank and a sample outlet line. Rupture of any tank will be contained by a sump that is part of the floor design. To drain any tank or the sump requires a pump (portable) and a connection to the sanitary drain. The design intent is to prevent accidental releases by elimination of the hazard waste to sanitary waste pathway. Only if analysis and authorization allows, would the connection and transfer be made.

7.4 CONFINEMENT DESIGN EVALUATION

Confinement evaluation depends on the quantity of airborne radioactivity release possible from the air and water that are in the region of the reactor during operation. Calculation, measurement, and experience of similar research reactors support the evaluation. Evaluation is limited to routine effluents and should be supplemented for experiment conditions that present specific release problems. Analysis of fission product releases are treated in another chapter. The most significant radiological effluents of the reactor are argon-41 and nitrogen-16.

Measurement and experience of other facilities have shown that for a facility of this type the most significant routine radiological contributions are caused by argon-41 generated by the exposure of air in experimental facilities and by nitrogen-16 transported in the coolant from the reactor core region. Argon-41, a noble gas, is contained for decay or eventually released to the atmosphere for decay. Nitrogen-16 a dissolved gas is contained in the coolant and is dissipated by radioactive decay.

7.4.1. Release of Nitrogen-16 and Argon-41

The nitrogen-16 produced through the (n,p) reaction with the oxygen in the water molecule has a very short half-life (7 sec) so only a very small fraction of that produced in the core will find its way to the pool surface. The principal radiological effect of the nitrogen-16 is as a contributor to the radiation level at the pool surface. Calculations of the nitrogen-16 transported to the pool surface estimate radiation dose rates of between 16 to 400 mrem/hour with the reactor operating at 1000 kW.

Argon-41 is produced in the reactor pool as a result of the (n, γ) reaction with argon-40 dissolved in the pool water. Most of this argon-41 remains in solution but some of it is transferred to the reactor room air at the pool surface. Calculations of the argon released from the pool surface estimate a concentration in the room of 1.6×10^{-6} $\mu\text{Ci/cc}$ with the reactor operating at 1000 kW.

7.4.1.1. Nitrogen-16 Activity in Reactor Room

The cross-section threshold for the oxygen-16 (n,p) nitrogen-16 reactions 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 effective cross-section of oxygen-16 (n,p) nitrogen-16 reaction averaged over the TRIGA spectrum is 0.021 millibarns. This value agrees well with the value obtained from integrating the effective cross section over the fission spectrum.

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

$$N_2 = \frac{N_1 \sigma_1 \phi_v}{\lambda_2} (1 - e^{-\lambda_2 t}) \quad (2)$$

where

N_2 = nitrogen-16 atoms per cm^3 of water,

ϕ_v = neutron flux = 1.0×10^{13} n/ cm^2 -sec,
(0.6 - 15 MeV),

N_1 = oxygen atoms per cm^3 of water = 3.3×10^{22} #/ cm^3 ,

σ_1 = (n,p) cross section of oxygen = 2.1×10^{-29} cm^2
(averaged over 0.6 - 15 MeV),

λ_2 = nitrogen-16 decay constant = 9.35×10^{-2} sec^{-1} ,

t = average time of exposure in reactor.

The average exposure time in the core (2.3 sec), was derived in the discussion on argon activity. Solving for N_2 in the equation above, one obtains

$$\begin{aligned} N_2 &= 7.41 \times 10^7 (1 - e^{-9.35 \times 10^{-2} \times 2.3}) \\ &= 1.43 \times 10^7 \text{ atoms/cm}^3 \end{aligned}$$

as the density of nitrogen-16 in the water leaving the core.

If it is assumed that the water continues to flow at the same velocity to the surface, a distance of -640 cm, the transit time from core to surface is

$$t_{\text{rise}} = 640/16.8 = 38.1 \text{ sec} \quad (3)$$

where the flow velocity, 16.8 cm/sec (Table 4-7), was given in the discussion on heat transfer.

This assumption is quite conservative as energy losses from the fluid stream resulting from turbulent mixing will reduce the velocity significantly. Furthermore, delays in transit time resulting from operation of the diffuser pump are sizeable. Measurements made of the dose rates at the pool surface of several TRIGA reactors show that the operation of the diffuser pump reduces the nitrogen-16 contribution to the surface dose rate by an order of magnitude of more depending on the size of the pool.

In 38 seconds the nitrogen-16 decays to 2.86×10^{-2} times the value of the activity leaving the core. Thus the concentration of nitrogen-16 atoms that reach the region near the surface of the pool is estimated at about 410,000 atoms/cm³ per cubic centimeter.

Only a small proportion of the nitrogen-16 atoms present near the pool surface are 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 (approximately 2 μ mho) 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 [4]. It is assumed that at least one-half of all ions formed are anions. Because of its 7.1-sec half-life, the nitrogen-16 decays before reaching a uniform concentration in the tank water. The activity will be dispersed over the surface area of the pool and much of it will decay during the lateral movement.

For the purpose of the analysis it is postulated that the water-bearing nitrogen-16 rises from the core to the surface and then spreads across a disk source with an equivalent radius of 125 cm. For a constant velocity of 16.8 cm/sec the cycle time for distributing the nitrogen-16 over the pool surface would be

$$t_s = 125 \text{ cm} / 16.8 \text{ cm/sec} = 7.4 \text{ sec}$$

The average concentration during this time is

$$\bar{N} = 1/t_s \int_0^{t_s} N_0 e^{-\lambda t} dt \quad (4)$$

$$\begin{aligned} &= \frac{N_0}{\lambda t_s} (1 - e^{-\lambda t_s}) = \frac{4.10 \times 10^5}{9.35 \times 10^{-2}} (1 - e^{-0.69}) / 7.4 \quad (5) \\ &= 3.0 \times 10^5 \text{ atoms/cm}^3 \end{aligned}$$

The interest from the point of safety is then the number of nitrogen-16 atoms escaping into the air from the diffusing source above the core. The number escaping to the air would be about is estimated from the escape velocity, 0.009 cm/sec, from Dorsey [5] as

$$\begin{aligned} &(3.0 \times 10^5 \text{ atoms/cc}) (0.9 \times 10^{-2} \text{ cm/sec}) \\ &= 2700 \text{ atoms/cm}^2 \cdot \text{sec} \end{aligned}$$

In the room, the activity is affected by dilution, ventilation, and decay. Thus the rate of accumulation of nitrogen-16 in the room as a whole is given by

$$\frac{d(VN^{16})}{dt} = S - (\lambda + q/V) VN^{16} \quad (6)$$

where S = number of nitrogen-16 atoms entering the room from the pool per second,

$$(2700) (5.05 \times 10^4) = 1.36 \times 10^8 \text{ atoms/sec,}$$

V = volume of the reactor room, $4.12 \times 10^9 \text{ cm}^3$,
(effective),

q = volume flow rate, $2.29 \times 10^6 \text{ cm}^3/\text{sec}$,
(reactor room exhaust).

For saturation conditions

$$\begin{aligned} VN^{16} &= \frac{S}{(\lambda N + q/V)} = \frac{1.36 \times 10^8}{9.35 \times 10^{-2} + 2.8 \times 10^{-5}} \\ &= 1.4 \times 10^9 \text{ nuclei.} \end{aligned} \quad (7)$$

This corresponds to an activity concentration of $8.8 \times 10^{-7} \mu\text{Ci/cc}$.

The gamma dose rate from nitrogen-16 of this concentration in the air is

$$\begin{aligned} D &= \frac{3.7 \times 10^4 \text{ photons/sec-}\mu\text{Ci} \times 8.8 \times 10^{-7} \mu\text{Ci}^3/\text{cm} \times 1000 \text{ cm}}{2 \times 1.6 \times 10^5 \text{ (photons/sec-cm}^2/\text{rad-hr)}} \\ &= 1.0 \times 10^{-4} \text{ rad/hr} = 100 \mu\text{rad/hr} \end{aligned}$$

when the effective radius of the room, taken to be a hemisphere with a volume of 4120 cubic meters is 10 m.

The thickness of the layer of nitrogen-16 bearing water is

$$h = \frac{v_l t_s}{A_s} = \frac{8.0 \times 10^3 \times 7.4}{5.05 \times 10^4} = 1.17 \text{ cm} \quad (8)$$

where the volume flow rate $8.0 \times 10^3 \text{ cm}^3/\text{sec}$ was given in the discussion on heat transfer.

The dose rate at the pool surface arising from the nitrogen-16 near the surface is

$$D = \frac{\lambda \bar{N}}{2\mu K} [1 - E_2(\mu h)] \quad (9)$$

where

μ = attenuation coefficient for 6 MeV photons in water
(0.0275 cm^{-1}),

$K = 1.6 \times 10^5$ photons/cm²-sec per rad/hr
flux to dose rate conversion factor,

E_2 = second exponential integral.

This yields, approximately,

$D = 400$ mr/hr

This value is larger than those extrapolated from measurements made on other TRIGA reactors. Transport times from the reactor core to the pool surface in excess of those estimated will lower the calculated dose substantially. A delay time twice as long as 38 sec. will generate a calculated dose rate twenty-five times less.

7.4.1.2. Release of Argon-41 from Reactor Pool 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 at which argon-41 escapes from the reactor pool water into the reactor room. The calculations show that most of the argon-41 decays while in the water. The changes in argon-41 concentration in the core region, in the pool water external to the reactor, and in the air of the reactor room, are calculated using the variables as follows:

N atomic density (atoms/cm³)

λ decay constant (sec⁻¹), 1.06×10^{-4}

σ absorption cross section (cm²), 0.61 b

q volume flow rate, reactor room exhaust (cm³/sec)

V volume of region (cm³)

ρ density (gm/cm³)

A_f channel free flow area (cm²)

ℓ_c channel length (cm)

w mass flow rate (gm/sec)

\bar{v} volume flow rate through the core (cm³/sec)

$\bar{\phi}$ average thermal neutron flux in the core (n/cm²-sec)

The volume flow rate through the core is

$$\dot{V} = \frac{w}{\rho} = \frac{8000 \text{ g/sec}}{1 \text{ g/cm}^3} = 8.0 \times 10^3 \text{ cm}^3/\text{sec}$$

From the flow channel volume, $A_f l_c$, the exposure time in the core is

$$t = V/\dot{V} = A_f l_c / \dot{V} = \frac{485.0 \times 38.1}{8.0 \times 10^3} = 2.3 \text{ sec}$$

It remains to find the atom density N for dissolved argon-40 in the reactor pool water.

Dissolved gases in the reactor coolant contain the radioactive noble gas argon-41. The release of argon-41 activity from the coolant depends on the gaseous exchange rate at the air-water interface and the change in gas solubility as a function of temperature.

According to Henry's law for gases in contact with liquids the equilibrium concentration in the liquid is proportional to the partial pressure of the gas. The saturated concentration of argon in water at one atmosphere of standard air is given in Table 7-1. Concentrations at equilibrium conditions for argon in air that is in contact with the water depends on the air volume, air exchange rate, water surface volume and the water-air exchange rate.

Table 7-1

SATURATED ARGON CONCENTRATION IN WATER [2]

Temperature (°C)	S(atoms Ar-40)/cm ³ H ₂ O
10	1.14×10^{16}
20	0.94×10^{16}
30	0.79×10^{16}
40	0.69×10^{16}
50	0.62×10^{16}
60	0.56×10^{16}
70	0.52×10^{16}
80	0.48×10^{16}

Argon-41 activities at conditions of equilibrium concentration are functions of the pool water volume, reactor core volume, water flow rates through the core, and the production and decay rates for the radioisotope.

The argon-40 concentration in the water at the core inlet temperature (38°C) is

$$N_1^{40} = 7.1 \times 10^{15} \text{ atoms/cm}^3$$

and the concentration of argon-40 in the water at the core exit temperature (68°C) is

$$N_1^{40} = 5.3 \times 10^{15} \text{ atoms/cm}^3$$

From the reactor power the cycle times for argon exposed in the core and circulated in the pool are estimated. The volume flow rate is given by

$$\bar{v} = Q / (P_W C_p \Delta T) \quad (10)$$

where:

\bar{v} = Volume flow rate of water through the core.

Q = Reactor power level (10^6 watts).

P_W = Pool water density (0.96 gm/cm^3).

C_p = Specific heat of water ($4.2 \text{ watt sec/gm}^\circ\text{C}$).

ΔT = Temperature rise in core (30°C).

Thus:

$$\bar{v} = 10^6 / (0.96 \times 4.2 \times 30) = 8.3 \times 10^3 \text{ cm}^3/\text{sec}.$$

The exposure time, t , and cycle time, T , in the core are calculated from

$$t = V_c / \bar{v} = A_f L_c / \bar{v} \quad (11)$$

$$\text{and } T = V_p / \bar{v} = [\pi h (w/2)^2 + h(1-w)w] / \bar{v} \quad (12)$$

where:

A_f = flow channel area (485 cm^2),

L_c = length of flow channel (38.1 cm),

V_c = volume of water in the core,

w = width of reactor pool (200 cm),

l = length of reactor pool (300 cm),

h = height of pool water (750 cm).

Exposure time, t, is about 2.2 seconds and cycle time, T, is about 4.6×10^3 seconds.

Argon atoms exchanged at the water-air interface depend on a water thickness depth that is small relative to the pool dimensions and, therefore, a small fraction of the available saturated argon is exchanged with the air. During the time required for the pool water to circulate once through the reactor core, about one hour and twenty minutes, the argon equilibrium concentration should deplete to the lowest solubility value for equilibrium concentration. The argon release as a function of temperature and solubility thus approaches zero. This depletion occurs as the activity of the argon radioisotope increases but is substantially complete as the argon-41 activity reaches half the equilibrium value at about 110 minutes.

Evaluation of the water-air interface exchange rate for argon is related to an air and water thickness depth that depends on the argon atom diffusion coefficient. The total exchange rate then is a function of the pool surface area, A_s , and an effective release volume V_i' . The two terms are related by

$$\beta_i A_s = f_{i \rightarrow j} V_i' \quad (13)$$

where

β_i is a surface exchange coefficient (cm/sec), and $f_{i \rightarrow j}$ is the fraction of atoms exchanged from volume i to j (sec^{-1}).

Estimates of the surface exchange coefficient (i.e., the gas in a unit volume that is exchanged at the surface per unit time per unit surface area) for argon vary considerably.

One method of arriving at a value for this parameter is through the diffusion coefficient of the gas in water. The mean square distance traversed by a molecule is

$$\langle \Delta X \rangle^2 = 2Dt \quad (14)$$

where D = diffusion coefficient (cm^2/sec),

t = time (sec).

The exchange coefficient is assumed to be evaluated for 1 sec as

$$\begin{aligned} \beta &= (\langle \Delta X \rangle^2)^{1/2} / t \\ &= (2D/t)^{1/2} \end{aligned}$$

The diffusion coefficient at 40°C is about 1.1×10^{-5} cm²/sec, and, if one assumes that only one-half of the argon atoms within one diffusion length of the surface escape,

$$\beta = 1/2 (2 \times 1.1 \times 10^{-5})^{1/2} = 2.35 \times 10^{-3} \text{ cm/sec}$$

Values for the surface exchange coefficient have been reported by Dorsey [3] for air, O₂, and N₂. The values for these three gases are all about equal. Assuming argon behaves as do these gases, a value is obtained of

$$\beta = 5.7 \times 10^{-3} \text{ cm/sec.}$$

Measurements have been made of the argon-41 activity in a TRIGA Mark III reactor pool and from the data acquired from these measurements it was possible to construct a value for the surface exchange coefficient. This value at 40°C is about 2.9×10^{-4} cm/sec.

During equilibrium conditions and assuming no difference in the rates of escape fractions for Ar-40 and Ar-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.,

$$f_{1 \rightarrow j} V_1' N_1 = f_{j \rightarrow 1} V_j' N_j \quad (15)$$

where $N_j = 2.1 \times 10^{17}$ argon atoms/cm³ of air -N⁴⁰,

$N_1 = 7.1 \times 10^{15}$ argon atoms/cm³ of water -N⁴⁰.

Solving for $f_{j \rightarrow 1} V'$ gives

$$f_{j \rightarrow 1} V_j' = f_{j \rightarrow j} V_j' (N_1/N_j) \quad (16)$$

The following calculations were performed to evaluate the rate of Ar-41 escaping from the reactor pool water into the room enclosure. The calculations show that the Ar-41 decays while in the water, and most of the radiation is safely absorbed in the water. The changes in Ar-41 concentration in the reactor, in the pool water external to the reactor, and in the air of the room enclosure are given by

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \phi N_1^{40} \sigma^{40} - N_1^{41} (v_1 + V_1 \phi \sigma^{41} + \lambda^{41} V_1) + N_2^{41} v_1 \quad (17)$$

$$V_2 \frac{dN_2^{41}}{dt} = -\lambda^{41} N_2^{41} V_2 + v_1^{41} (N_1^{41} - N_2^{41}) - (f_{2 \rightarrow 3} N_2^{41} V_2' - f_{3 \rightarrow 2} N_3^{41} V_3') \quad (18)$$

$$V_3 \frac{dN_3^{41}}{dt} = (f_{2 \rightarrow 3} N_2^{41} V_2' - f_{3 \rightarrow 2} N_3^{41} V_3') - N_3^{41} (\lambda^{41} V_3 + q) \quad (19)$$

where

subscript 1 = Reactor region (water region in core)

subscript 2 = Reactor tank water region external to the reactor

subscript 3 = Room enclosure region

superscript 40 = Ar-40

superscript 41 = Ar-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 room enclosure exhaust (cm^3/s)

v_1 = Volume flow rate through region No.1 (cm^3/s)

ϕ = Average thermal neutron flux in region No.1 ($\text{n}/\text{cm}^2 \times \text{s}$)

$f_{i \rightarrow j} V_i$ = Fraction of Ar-41 atoms in region i that escape to region j per unit time (cm^3/sec)

The values of constants in equations (17), (18), and (19) are

$$\begin{aligned} \phi &= 1.2 \times 10^{13} \text{ n}/\text{cm}^2\text{-s} , & V_1 &= 1.85 \times 10^4 \text{ cm}^3 , \\ \sigma^{40} &= 0.47 \times 10^{-24} \text{ cm}^2 , & V_2 &= 4.00 \times 10^7 \text{ cm}^3 , \\ \sigma^{41} &= 0.060 \times 10^{-24} \text{ cm}^2 , & V_3 &= 4.12 \times 10^9 \text{ cm}^3 , \\ \lambda^{41} &= 1.06 \times 10^{-4} \text{ sec}^{-1} , & \beta &= 2.35 \times 10^{-3} \text{ cm}/\text{sec} , \\ q &= 2.29 \times 10^6 \text{ cm}^3/\text{s} , & N_1^{40} &= 7.1 \times 10^{-15} \text{ \#}/\text{cm}^3 , \\ v_1 &= 8.13 \times 10^3 \text{ cm}^3/\text{s} . \end{aligned}$$

Equation (1) can be reduced to

$$V_1 \frac{dN_1^{41}}{dt} = V_1 \phi N_1^{40} \sigma^{40} - (N_1^{41} - N_2^{41}) v_1 \quad (20)$$

by

$$v_1 + V_1 \phi \sigma^{41} + \lambda^{41} V_1 = v_1$$

During equilibrium conditions the three equations reduce to:

$$V_1 \phi N_1^{40} \sigma^{40} = (N_1^{41} - N_2^{41}) v_1 \quad , \quad (21)$$

$$N_2^{41} [\lambda^{41} v_2 + f_{2 \rightarrow 3} v_2'] = (N_1^{41} - N_2^{41}) v_1 + f_{3 \rightarrow 2} N_3^{41} v_3' \quad , \quad (22)$$

$$N_3^{41} [\lambda^{41} v_3 + q + f_{3 \rightarrow 2} v_3'] = f_{2 \rightarrow 3} N_2^{41} v_2 \quad . \quad (23)$$

Combining equations (21) and (22) gives

$$N_2^{41} = \frac{V_1 \phi N_1^{40} \sigma^{40}}{\lambda^{41} v_2 + f_{2 \rightarrow 3} v_2'} + \frac{f_{3 \rightarrow 2} N_3^{41} v_3'}{\lambda^{41} v_2 + f_{2 \rightarrow 3} v_2'} \quad , \quad (24)$$

which inserted into equation (23) for N_2^{41} yields (25)

$$N_3^{41} \left[\frac{\lambda^{41} v_3 + q + f_{3 \rightarrow 2} v_3'}{f_{2 \rightarrow 3} v_2'} - \frac{f_{3 \rightarrow 2} v_3'}{\lambda^{41} v_2 + f_{2 \rightarrow 3} v_2'} \right] = \frac{V_1 \phi N_1^{40} \sigma^{40}}{\lambda^{41} v_2 + f_{2 \rightarrow 3} v_2'} \quad ,$$

Solving for N_3^{41} yields 7.4 atoms/cm³. This corresponds to a concentration of Ar-41 activity of

$$A^{41} = \frac{\lambda^{41} N_3^{41}}{C} = \frac{1.06 \times 10^{-4} \times 7.4}{3.7 \times 10^4} = 2.12 \times 10^{-8} \text{ } \mu\text{Ci/cm}^3 \quad ,$$

where

$$A^{41} = \text{Ar-41 concentration, } \mu\text{Ci/cm}^3 \quad ,$$

C = Conversion factor from disintegrations to μCi .

This is below the reference level limit recommended by 10 CFR part 20 for unrestricted areas ($4 \times 10^{-8} \text{ } \mu\text{Ci/cm}^3$) and takes into account no dilution, which is conservative.

7.4.1.3. Activation of Air in the Experimental Facilities

In the TRIGA reactor installation, the following experimental facilities contain air: rotary specimen rack, pneumatic transfer tube, and neutron beam tubes. Of the radioisotopes produced in these air cavities, argon-41 (with 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. Estimated releases of argon-41 from reactor operations indicate an upper limit for the release exposure as 190 $\mu\text{rad/hr}$. Actual values are expected to be less than 1/50 of the estimated value.

The saturated activity of argon-41 in an experimental cavity is calculated from

$$A = N^{41} \lambda^{41} = \frac{\lambda^{41} S}{C(\lambda^{41} + q/V)} \text{ } \mu\text{Ci/cm}^3 \quad , \quad (26)$$

where

$$C = 3.7 \times 10^4 \text{ disintegrations/sec per } \mu\text{Ci},$$

$$S = \phi \Sigma_a \text{ n/cm}^3\text{-sec.}$$

$$\Sigma_a = 1.59 \times 10^{-7} \text{ cm}^{-1}.$$

The effective air volumes of several experimental facilities are listed in Table 7-2. Also given are conservative estimates of the average thermal neutron fluxes for 1100 kW operation.

Table 7-2

VOLUMES AND THERMAL FLUXES OF FACILITIES

Region	Effective Air Volume (cm ³)	Average Thermal Flux at 1100 kW (n/cm ² -sec) x 10 ¹²
Central thimble	4.3×10^2	23.8
Rotary specimen rack	3.3×10^4	7.2
Pneumatic tube	1.6×10^3	7.2
1&5 Thru beam port	2.7×10^5	0.1
2 Tangential beam port	7.2×10^4	0.1
3 Radial beam port	1.8×10^5	0.1
4 Radial beam port	7.0×10^4	0.1

For volume exhaust rates where the decay constant is negligible, such that

$$\lambda^{41} \ll q/V, \quad (27)$$

the activity released from each volume is given by

$$A^{41} q_i = \lambda^{41} (\phi \Sigma_a)_i V_i / C \text{ } \mu\text{Ci/sec} \quad (28)$$

Total volume of all air cavities without any experiments in place is about 0.6 cubic meters. A flow rate of $6.4 \times 10^2 \text{ cm}^3/\text{sec}$ is necessary to neglect the decay. Actual flow rates are about $3.15 \text{ cm}^3/\text{sec}$ (4 cfh) for each beam port. At this rate the fraction of air removal from the cavities is only 9 percent per hour. The total activity calculated for the air leaving the beam port experiment facilities is therefore, $6 \text{ } \mu\text{Ci/sec}$.

It should be emphasized that the air activation and subsequent release activity are predicted for vacant beam ports and conservative neutron fluxes. Actual release rates depend on the particular configuration of experiments and the air exchange rate in each facility.

7.4.2. Evaluation of Argon 41 Release

The release of argon-41 from the facility is diluted by the ventilation exhaust rate, assumed to represent two air changes per hour, and averaged for a 5 day, 8 hour operation schedule at full power. The release concentration from the pool averaged for one year is,

$$.24 (1.65 \times 10^{-6}) = 3.96 \times 10^{-7} \mu\text{Ci}/\text{cm}^3.$$

Only 20% of the experiment facility argon-41 is assumed to exhaust since experiments will replace some or most of the exposed air,

$$6 (.20) (.24)/2.29 \times 10^6 = 1.3 \times 10^{-7} \mu\text{Ci}/\text{cm}^3.$$

Total estimated release is $4.1 \times 10^{-7} \mu\text{Ci}/\text{cm}^3$.

The whole body gamma ray dose rate to a person immersed in a semi-infinite cloud of radioactive gases can be approximated by

$$D = 900 E A_D \quad (29)$$

where E = the photon energy, 1.3 Mev

A_D = effective exposure concentration, Ci/m^3 .

The concentration downwind from the point at which the activity is discharged from the building is

$$A_D = A q \psi(x), \quad (30)$$

where ψ = the dilution factor at the distance x , (sec/m^3) ,

A = activity concentration in the discharge (Ci/m^3) ,

q = the building exhaust rates (m^3/sec) .

If it is assumed that the discharge is at the roof line, the dilution factor in the lee of the building ($x = 0$), is given [6] by:

$$\psi(0) = 1/csu, \quad (31)$$

where c = a constant (0.5),

s = building cross-sectional area normal to the wind direction (m^2) ,

u = wind velocity (m/sec) .

A minimum cross-sectional area is assumed of 234 m^2 ($60 \times 42 \text{ ft}$) and, for a wind velocity of $4 \text{ m}/\text{sec}$,

$$\psi(0) = 1/(0.5 \times 4 \times 234) = 2.1 \times 10^{-3} \text{ sec}/\text{m}^3 \quad (32)$$

The averaged dose rate at the exhaust stack is

$$D = 900 \times 1.3 (4.1 \times 10^{-7}) = 4.8 \times 10^{-4} \text{ rads/hr,}$$

an average of 0.48 mrad/hr in the stack or

$$D = 4.8 \times 10^{-4} (2.3 \times 2.1 \times 10^{-3}) = 2.3 \times 10^{-6} \text{ rads/hr,}$$

an average of 2.3 μ rad/hr at ground level.

At the limiting exhaust rate, 640 cm³/sec, to ignore the argon-41 decay, the source term is 27 μ Ci/sec for the beam ports which would increase the dose rate to 10 μ rad/hr. If core experiment facilities, such as the center tube and rotary specimen rack, are vented at the same exhaust rate the releases would increase by 102 μ Ci/sec. However, these exhaust rate conditions represent limiting conditions, not actual release rates. For the exhaust manifold rate of 3.15 cm³/sec the release rate would be 50 μ Ci/sec for the two core experiment facilities. The rotary specimen rack is the primary source of the activity.

Venting of experiment facilities, especially the rotary specimen rack, will require monitoring of the release rate or replacement of the air with gases such as nitrogen or carbon dioxide to control release concentrations of air activity. However, normal operating conditions do not vent the rotary specimen rack, although this is an optional operating condition. The pneumatic transfer system, by comparison, routinely contains nitrogen or carbon dioxide gas to limit the releases within the room that contains the access terminal.

Actual dose values for the argon-41 release may vary. The beam port release estimate is less than 4.8 μ rad/hr which is equivalent to 10 mrad/yr. Lower neutron fluxes, smaller air volumes, shorter operation times and larger dilution factors will assure that releases do not exceed annual release constraints. Monitoring the exhaust will verify that other release points such as the core experiment facilities do not cause the total to exceed preset limits.

Chapter 7 References

1. Code of Federal Regulations, Chapter 10 part 20, U.S. Government Printing Office, 1982.
2. Dorsey, N.E., Properties of Ordinary Water-Substance, Reinhold Publishing Corp., New York p. 537.
3. Ibid., p. 554.
4. Mittl, R.L., and M.H. Theys, "N-16 Concentration in EBWR," Nucleonics p. 81 (1961).
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Chapter 8

EXPERIMENT AND IRRADIATION FACILITIES

The experimental and irradiation facilities of the TRIGA Mark II reactor are extensive and versatile. Physical access and observation of the core are possible at all times through the vertical water shield. Experimental tubes can easily be installed in the core region to provide facilities for high-level irradiations or in-core experiments. Areas outside the core and reflector are available for larger experiment equipment or facilities.

8.1. STANDARD EXPERIMENT FACILITIES

8.1.1. Central Thimble

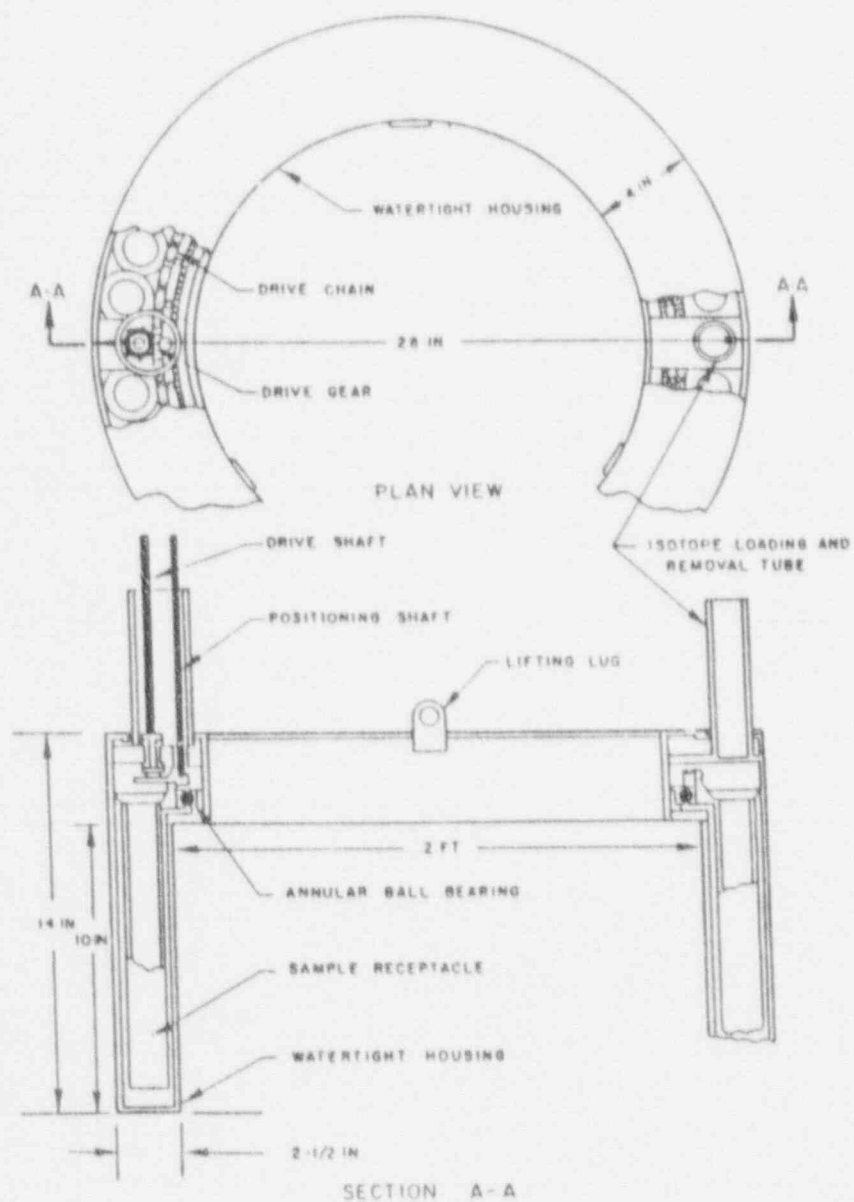
The reactor is equipped with a central thimble for access to the point of maximum flux in the core. The central thimble consists of an aluminum tube that fits through the center hole of the top and bottom grid plates. Dimensions of the tube are 1.5 in. o.d. (3.81 cm.) and 1.33 in. i.d. (3.38 cm.). Holes in the tube assure that it is normally filled with water. Water is expelled from the tube by compressed air. Experiments with the central thimble include irradiations of small samples and the exposure of materials to a collimated beam of neutrons or gamma rays.

8.1.2. Rotary Specimen Rack

A rotary, multiple-position (40) specimen rack located in a well in the top of the graphite reflector provides for the large scale production of radioisotopes and for the activation and irradiation of multiple samples. All positions in this rack are exposed to neutron fluxes of comparable intensity. Specimen positions are 1.23 in. (3.18 cm.) in diameter by 10.80 in. (27.4 cm.) in depth. Samples are loaded from the top of the reactor through a water-tight tube into the rotary rack using a specimen lifting device or pneumatic pressure for insertion and removal of samples from the sample rack positions. The rotary specimen rack can be turned from the top of the reactor by manual operation or by a motor drive. Figure 8-1 shows the rack.

8.1.3. Pneumatic Specimen Tube

A pneumatic transfer system permits applications with short-lived radioisotopes. The in-core terminus of this system is normally located in the outer ring of fuel element positions, a region of high neutron flux. The sample capsule (rabbit) is conveyed to a receiver-sender station via 1.25 in. o.d. (3.18 cm.) aluminum tubing. Effective space in the specimen transfer capsules is 0.68 in. (1.7 cm.) diameter by 4.5 in. (11.4 cm.) height. An optional transfer box may be employed to permit the sample to be sent and received from up to three different receiver-sender stations. A schematic of the pneumatic irradiation terminal and air flow control valves is shown in Figure 8-2.



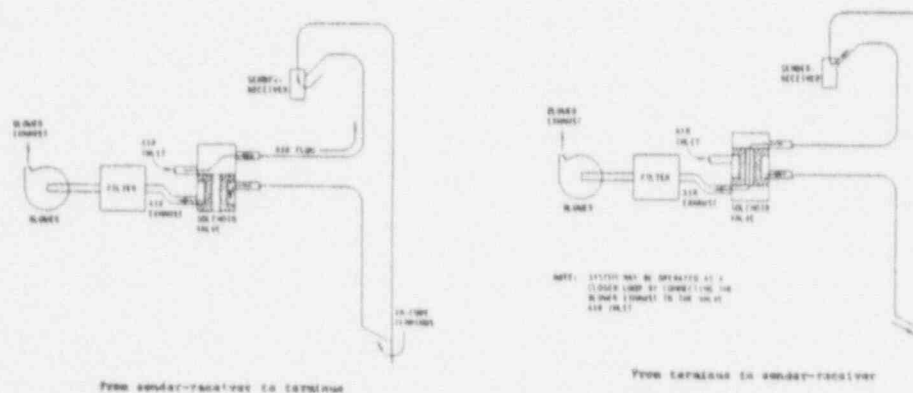
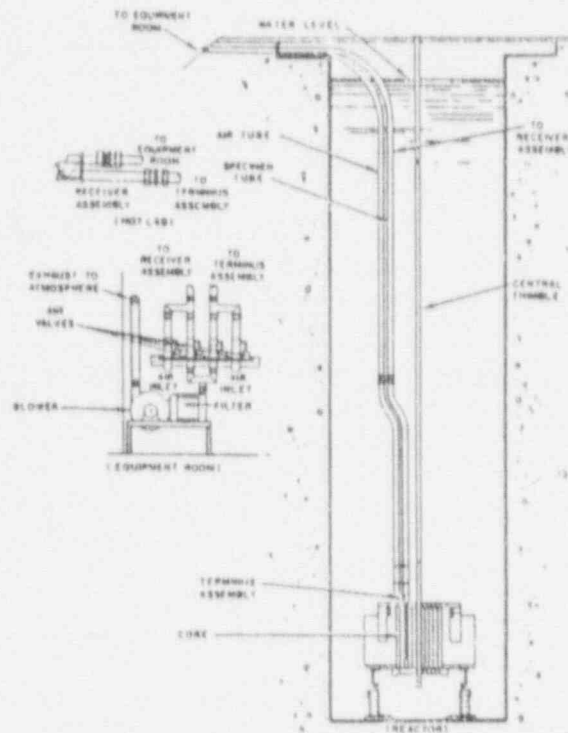
SECTION A-A

MI-6

Rotary specimen rack assembly (schematic)

ROTARY SPECIMEN FACILITY

Figure 8-1



PNEUMATIC TRANSFER FACILITY

Figure 8-2

8.1.4. Beam Tube Facilities

Access to horizontal neutron beams is created by five beam tubes penetrating the reactor shield structure. All beam tubes are 6 inch diameter tubes originating at or in the reactor reflector. One tangential beam tube is composed of a penetration in the reactor reflector assembly with extensions through both sides of the reactor shield. A second tangential beam tube penetrates and terminates in the reactor reflector. The two remaining tubes are oriented radial to the reactor core.

The beam ports (Figure 8-3) provide tubular penetrations through the concrete shield and reactor tank water, making beams of neutrons (or gamma radiation) available for experiments. The beam ports also provide an irradiation facility for large sample specimens in a region close to the core. Beam port diameters near the core are 6 in. (15.2 cm.).

There are five beam ports divided into two categories as follows:

8.1.4.1. Tangential beam ports

Two beam ports are oriented tangential to the reactor core, penetrate the graphite reflector, the coolant water, and the concrete shield. A hole is drilled in the graphite tangential to the outer edge of the core. One beam port terminates at the tangential point to the core. The other beam tubes extend both directions from the reflector and out opposite sides of the reactor shield.

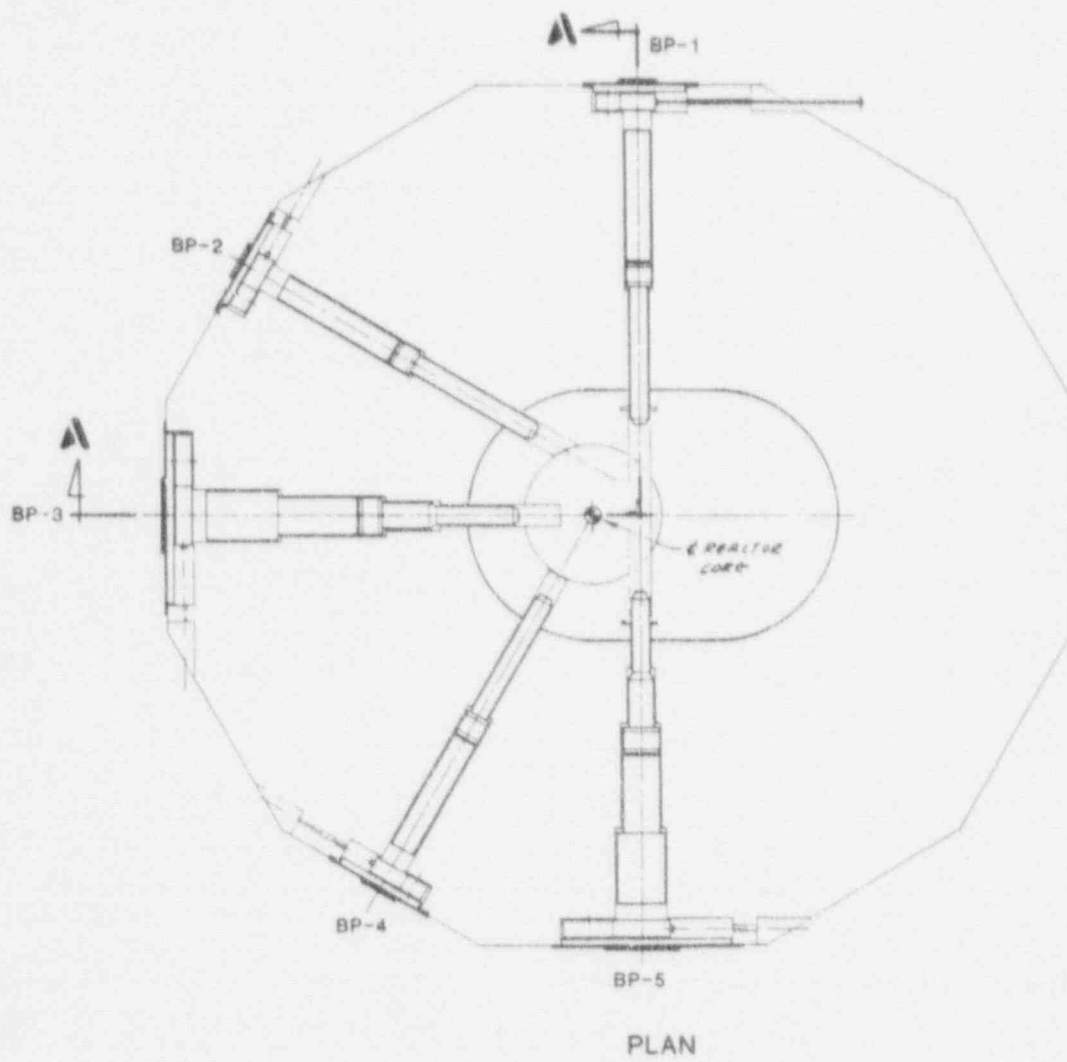
8.1.4.2. Radial beam ports

There are two radial beam ports, each which penetrate the concrete shield structure and the coolant water. One radial port terminates at the outer edge of the reflector. The second radial port also terminates at the outer edge of the reflector. However, a hole drilled in the graphite reflector extends the effective source of the radiations to the reactor core region.

8.1.4.3. Beam Tube Plugs

A step is incorporated into each beam port to prevent radiation streaming through the gap between the beam tube and shielding plug. The inner section of each beam port is an aluminum pipe 6 in. (15.2 cm.) in diameter. The outer section of beam ports 1, 2 and 4 is a steel pipe 8 in. (20.3 cm.) in diameter. Beam ports 3 and 5 have three outer sections with 8 in., 12 in., and 15.25 in. diameters. A lead shield ring in the shield structure provides a "shadow" shield for the 15.25 in. beam port section. Special shielding reduces the radiation outside the concrete to a safe level when the beam port is not in use. The shielding is provided in four sections as follows:

- a. An inner shield plug.
- b. An outer shield plug.
- c. A lead-filled shutter, and door.



BEAM TUBE CONFIGURATION

Figure 8-3

The inner shield plug consists of graphite cylinder, backed with a 0.125-in. (0.32-cm) sheet of boral and 5 in. (12.7 cm) of lead, sandwiched between two 1.25 in. (3.2 cm) thick steel plates. Beam ports 1, 2, and 4 have a section of graphite 6 in. (15.2 cm) in diameter. Beam ports 3 and 5 have the same configuration as the other beam ports, except that the graphite portion is 6 in. (15.2 cm) in diameter, with a change to 8 in. (20.3 cm) in diameter to provide graphite shielding in the 6 in. and 8 in. portions of the tube. Two rollers are provided to facilitate the insertion and removal of the inner shield plugs. To help guide the shield plug over the steps in the beam tube during insertion, the inner end of the plug is cone-shaped. A threaded hole is provided in the outer end of the plug for attaching the beam tube plug-handling tool. The graphite sections are encased in an aluminum cannister.

The outer shield plug is wooden and is 8 in. (20.3 cm) in diameter and 42 in. (1.07 m) long for beam ports 1, 2, and 4. Beam ports 3 and 5 have a wooden shield plug for the outer portion of the tube that has a length of 48 in. (1.22 m) and diameter of 15 in. (38.1 cm) for the outer portion of the tube. A handle on the outer end of this plug is provided for manual handling. The plug is equipped with an electrical circuit consisting of a position switch mounted in the front of the plug and an electrical connector at the rear of the plug. The switch can be actuated only by the inner plug when the inner plug is installed in the beam tube.

A physical contact between the inner and outer shield plug, and an electrical connection between the outer plug and the beam tube are part of an installation status circuit. The circuit monitors the plug configuration or other experiment shield conditions. Information on the console for each beam tube indicates the plug or beam tube status.

The lead-filled shutter and lead-lined door provide limited gamma shielding when the plugs are removed. The shutter is contained in a rectangular steel housing recessed in the outer surface of the concrete shield. The shutter is ~10 in. (25.4 cm) in diameter and 9.5 in. (24.1 cm) thick for beam ports 1, 2, and 4. Beam ports 3 and 5 have a shutter that is 15.25 in. (38.7 cm) in diameter and 9.5 in. (24.1 cm) thick. The shutter is operated by a removeable push rod on the face of the shield structure and can be moved even with the shutter housing door closed. In the open position, a section of the shutter consisting of pipe of equal diameter to the outer portion of the beam tube is aligned with the beam port and the outer shield center plug to facilitate insertion or removal of the beam plugs. The shutter housing is equipped with a steel cover plate lined with 1.25 in. (3.2 cm) of lead for additional shielding.

A removeable cover plate provides easy access to the beam port. The plate can be bolted shut so that the seal would prevent loss of shielding water if the beam tube should develop a serious leak.

8.1.5. Evaluation of Materials in Experiment Facilities

The following information is a guide for evaluating experiment materials in order to prevent the introduction of materials that could damage the reactor or its components. A careful evaluation [1] of proposed experiment materials shall be performed to classify the experiment as an approved experiment. Guidelines for the following types of experiment materials are provided: materials which require double encapsulation, explosive materials and their confinement, fueled experiments, and materials which could be sources of airborne radioactivity. The limits referenced in this section are technical specification requirements to prevent the occurrence of a serious safety hazard.

8.1.5.1. Double Encapsulation

Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials, and liquid fissionable materials shall be doubly encapsulated. Chemical Hazard Information in the "Handbook of Laboratory Safety," [2] shall be used as a guide for classifying the first three categories of materials. A table of isotopes shall be used as a guide for the last category.

The Chemical Hazard Information lists several categories of information about hazardous materials. One category of significant interest has the title "Relative Hazard to Health from Concentrated Short-Term Exposure." This category identifies materials that are highly corrosive. From a conservative standpoint, those materials which are corrosive to human tissue shall be considered to be corrosive for the purposes of double encapsulation. References also exist to other toxic or hazardous material listings for details about a particular material.

Two categories determine the explosive and flammability properties of materials. One titled "N.F.P.A. Hazard Identification Signals" and another titled "Flammable Limits in Air," specify materials that can represent safety hazards. The table identifies degree of explosive potential, chemical reactivity and flammability limits.

Material ratings for "N.F.P.A. Hazard Identification" consist of a scale of 0 (low) to 4 (high), regarding their health, fire, and reactivity hazards while under fire conditions. In this table, fire ratings indicate the degree of susceptibility for a particular material to burn. Further, a reactivity rating indicates the degree of susceptibility for a particular material to release energy. The code rating scale for flammability specifies limits as a function of the air volume percent.

Materials having fire ratings greater than four or reactivity ratings greater than 2 should be doubly encapsulated. The encapsulation is to protect against the energy release and corrosion properties. Materials having a reactivity rating of 1 or a flammability rating greater than 1 should be evaluated individually to determine if double encapsulation is warranted.

The primary hazard attributed to liquid fissionable experiment materials is leakage of the material from the experiment container. To minimize this risk, all experiment materials containing more than 100 ppm of thorium, uranium, or plutonium shall be doubly encapsulated.

The type of encapsulation material should be compatible with the encapsulation contents. Although double encapsulation of hazardous experiment materials greatly reduces the likelihood of their release, it is possible for these materials to escape their experiment containers. If an experiment capsule fails and releases material which could damage the reactor fuel, structure, or systems, the capsule shall be promptly removed and inspected in order to determine the consequences and the need for corrective action.

8.1.5.2. Explosive Materials

A 25 milligram explosive releases approximately 25 calories (104.2 joules) calories of energy with the creation of 25 cm³ of gas. For the explosive TNT, the density is 1.654 gm/cm³ so that 25 mg represents a volume of 0.015 cm³. If the assumption is made that the energy release occurs as an instantaneous change in pressure, the total force on the encapsulation material is the sum of two pressures. For a one cm³ volume the energy release of 104.2 joules represents a pressure of 1032 atmospheres. The instantaneous change in pressure due to gas production in the same volume adds another 25 atmospheres. The total pressure within a 1 cm³ capsule is then 1057 atmospheres for the complete reaction of 25 mg of explosive.

Typical construction materials of capsules are stainless steel, aluminum and polyethylene. Table 8-1 lists the mechanical properties of these encapsulation materials.

Table 8-1

Material Strengths

Material	Yield	Ultimate	Density
Stainless Steel (type 304)	35 ksi	85 ksi	7.98 g/cm ³ (500 lb/ft ³)
Aluminum (alloy 6061)	40 ksi	45ksi	2.739 g/cm ³ (171 lb/ft ³)
Polyethylene	1.7 ksi	1.4ksi	0.923 g/cm ³

Analysis of the encapsulation material determines the material stress limits that must exist to confine the reactive equivalent of 25 mg of explosive. The stress limit in a cylindrical container with thin walls is one half the pressure times the ratio of the capsule diameter to wall thickness,

$$\sigma_{\max} = \frac{pd}{2t} \quad (1)$$

where

- σ_{\max} - maximum stress in container wall,
- p - total pressure within the container,
- d - diameter of the container,
- t - wall thickness.

When evaluating an encapsulation material's ability to confine the reactive equivalent of 25 mg of explosive, the maximum stress in the container wall is required to be less than or equal to the yield strength of the material:

$$\frac{pd}{2t} \leq \sigma_{\text{yield}} \quad (2)$$

where σ_{yield} is the yield strength. Solving this equation for d/t provides an easy method of evaluating an encapsulation material:

$$\frac{d}{t} \leq \frac{2}{p} \sigma_{\text{yield}} \quad (3)$$

Table 8-2

Container Diameter to Thickness Ratio

Material	d/t
Stainless Steel (type 304)	4.5
Aluminum (alloy 6061)	5.1
polyethylene (low density)	0.22

Assuming an internal pressure of 1057 atm (15,538 psi), maximum values of d/t are displayed in Table 8-2 for the encapsulation materials of Table 8-1. The figures indicate that a polyethylene vial is not a practical container since its wall thickness must be 4.5 times the diameter. Both the aluminum and the stainless steel made satisfactory containers.

8.1.5.3. Fueled Experiments

Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 750 millicuries and the maximum strontium inventory is no greater than 2.5 millicuries. These restrictions are used to limit the fission product release in the event of an experiment container rupture. In terms of hazard to humans, the iodine isotopes represent the most harmful short-lived isotopes. The values are justified by making the following assumptions:

- a. Half of the total iodine and strontium inventories are comprised of I-131 and Sr-90, the isotopes of each element which have the lowest permitted concentration levels.
- b. The iodine/strontium inventories are evenly distributed throughout the reactor room, which has a volume of $4.83 \times 10^3 \text{ m}^3$.
- c. At two air changes per hour, the isotope inventory is removed from the room in one-half hour.
- d. Average indoor concentration over an occupational year.

Calculations based on these assumptions result in indoor concentrations below the occupational concentration limits of each isotope, as shown in Table 8-3. Assuming the building wake dilution factor to be $5.3 \times 10^5 \text{ sec/ft}^3$. Calculations indicate that ground level concentrations outside the building are below the reference levels [4] of each isotope.

Table 8-3

Calculated Isotope Release Values

	Units	I ¹³¹	Sr ⁹⁰
indoor	$\mu\text{Ci/ml}$	1.9×10^{-8}	6.5×10^{-11}
occupational limit	$\mu\text{Ci/ml}$	2.0×10^{-8}	2.0×10^{-9}
outdoor	$\mu\text{Ci/ml}$	2.2×10^{-11}	7.5×10^{-14}
reference level	$\mu\text{Ci/ml}$	2.0×10^{-10}	5.0×10^{-12}

8.1.5.4. Airborne Experiment Releases

Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under one of the following conditions shall be limited to occupational levels for airborne radioactivity concentration, as specified in 10CFR20, when averaging over a year:

- a. Normal operating conditions of the experiment or reactor.
- b. Credible accident conditions in the reactor.
- c. Possible accident conditions in the experiment.

The radioactivity release is based on the assumption that 100% of the gaseous activity or radioactive aerosols produced escape to the reactor room or the atmosphere.

When considering materials for experimentation, the type of radiation emitted by the airborne radioactive products should be noted. For example, alpha-emitters, while not considerably hazardous outside the body, may cause significant damage to human tissue if they are inhaled. Although concentration limits in 10CFR20 reflect these considerations, the experimenter has a responsibility to know what hazards may exist during and after the experiment.

The following assumptions shall be used to calculating the airborne reactivity concentration :

- a. If the effluent from an experimental facility exhausts through a holdup tank which closes automatically on high radiation level, at least 10% of the gaseous activity or aerosols produced will escape.
- b. If the effluent from an experimental facility exhaust through a filter installation designed for greater than 99% efficiency for 0.25 micron particles, at least 10% of these vapors will escape.
- c. For materials whose boiling point is above 55°C and where vapors formed by boiling this material can escape only through an undisturbed column of water above the core, at least 10% of these vapors will escape.

These three assumptions contain phrases stating that at least 10% of the vapors escape from the specified, engineered safety systems that are part of the experiment design. For the purposes of calculation, the assumptions are intended to provide a conservative estimate of the amount of radioactivity that will be released.

Calculations of the airborne radioactivity concentration depend on the reactor room effective volume, building wake dilution factor, and the ventilation system flow rates. The reactor room effective volume, $4.83 \times 10^3 \text{ m}^3$, the building wake dilution factor, $5.31 \times 10^5 \text{ sec/ft}^3$, and the ventilation system flow rates are facility design features. Calculations have been done to demonstrate the fraction of airborne release that will occur interior and exterior to the building for different ventilation flow rates.

8.2. SPECIAL EXPERIMENTAL FACILITIES

8.2.1. Reactor Core Facilities

Reactor core experiment facilities are designed for replacement of either single fuel element positions or a special multi-element position. Access to the core peak flux is provided by a central thimble. The wet central thimble is designed to allow insertion of an encapsulated sample into the core center or extraction of a vertical neutron beam from the core center. Experiment facilities are located in single element positions in the reactor core for insertion of samples into the reactor neutron flux. A pneumatic terminal is one type of single element experiment facility. More than one such terminal may be in place for experiment use.

Experiments with reactor characteristics may be placed in single element positions. One example of a single position experiment is the measurement of void or reactivity coefficients of various materials. Another characteristic is demonstrated by a reactivity oscillator apparatus. The oscillator is fabricated with rotating absorbers that are inserted in a single element position.

Two types of experiment locations in the core structure provide for removal of multiple element sections. One of these types is a three element section at two different positions of the D and E rings of the core array. The second type is a six element removeable section that removes the elements of the B ring and changes the size of the center hole position. Fabrication of experiments for the multi-element positions are projected for future facility development. Reactor core limits on reactivity for experiments will apply to installation of any experiment facility.

8.2.1.1. Three Element Feature

The three element feature of the core grid structure is a 2.062 in. (5.24 cm) diameter hole. Location of the hole center coincides with the center of a three element subarray. Total area of the hole is 3.34 in.² (21.5 cm²), but also contains additional area for each of the three elements of the original subarray. If the total available area of the cutout in the subarray were used, the area would exceed the primary hole, but be less than the sum of three element holes and the primary hole. Three single element holes have a total area of 5.34 in.² (34.4 cm²). One three element subarray cutout in the subarray is one D ring and two E ring locations. The other three element subarray is one E ring and two D ring locations.

8.2.1.2. Six Element Feature

A six element feature at the center of the core grid structure controls the position of the center hole and adjacent B ring elements. Removal of the A and B ring element array will create a 4.005 in. (10.02 cm) diameter hole. This hole has an area of 12.6 in.² (81.3 cm²) with its center at the reactor core center.

Additional space is available in the partial element holes that remain in the B ring. This additional space is the result of the experiment hole diameter, which exceeds the diameter of the center hole, but is less than a diameter that would include all the B ring holes. A limit shall exist on the use of this space such that the total area available to the experiment remains less than 15.8 in.² (102 cm²).

8.2.2. Gamma Irradiation Facility

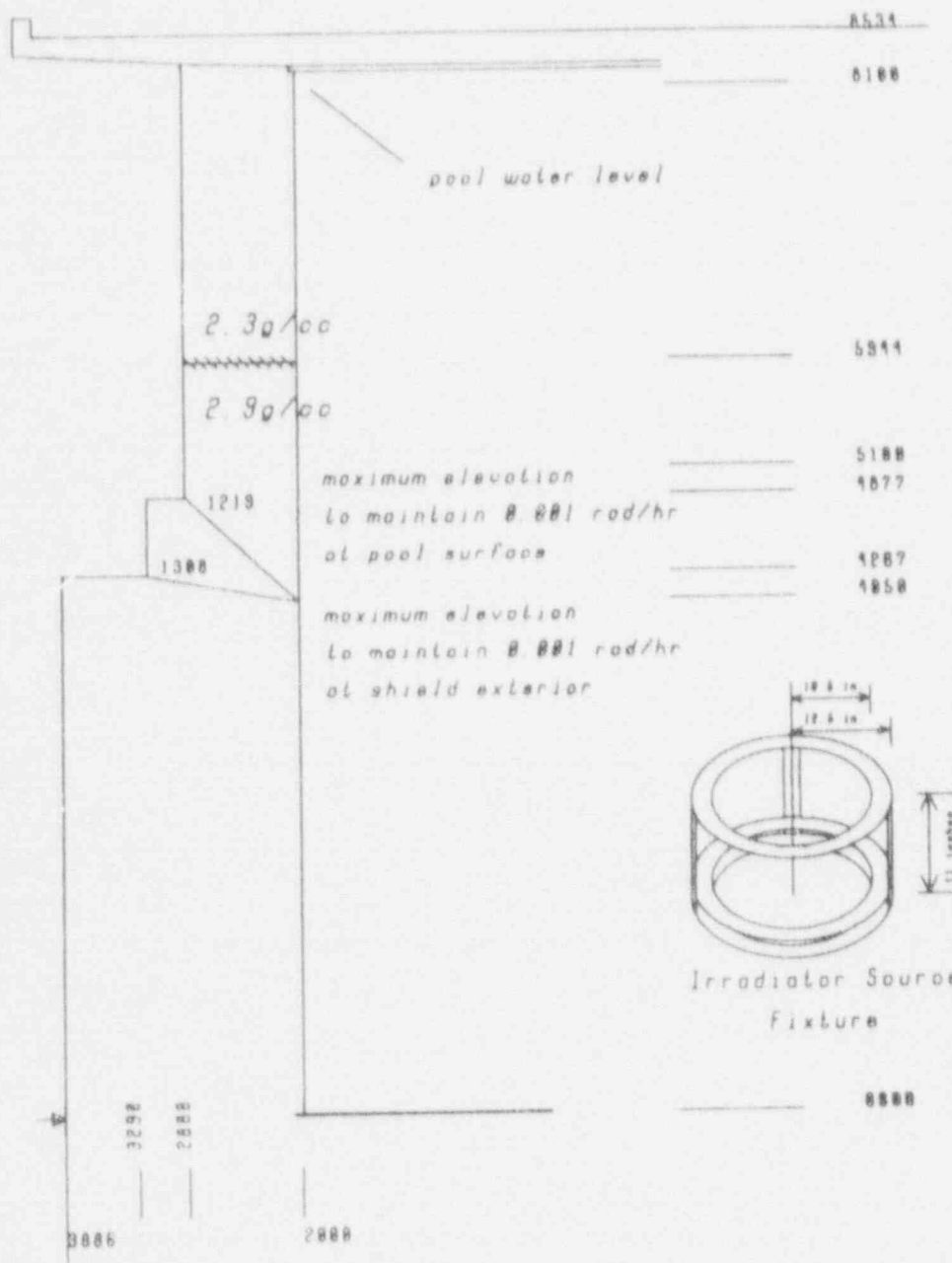
Gamma irradiation with cobalt-60 is provided by an irradiation facility in the reactor pool. The irradiator consists of 156 pencil size sources. Each source is contained in a space 0.5 inches in diameter and 11.25 inches in length. Sources are doubly encapsulated with inner aluminum clad and outer stainless steel clad. The source pencils are arranged in a double staggered, circumferential array, with an inside radius of 10.5 inches and outside radius of 12.5 inches (Figure 8-4a). The total source strength when installed was 9260 Curies. A shelf suspended in the pool at the end opposite to the reactor location holds the cobalt-60 irradiation facility below 10 feet of water. Access to the facility is provided by water tight canisters or dry irradiation tubes. Two canisters with 0.625 inch wall thickness, 5.0 inch radius and 11.5 inch height, and several 2 inch diameter vertical beam tubes, provide exposure volumes for routine experimentation.

8.2.2.1. Hazard to the Pool Water System

Hazards to the pool water system consider three conditions. The conditions are the physical location with reference to the reactor system, the possible leakage of a facility source and the radiation in the areas of the pool water system.

Separation of the gamma irradiation facility and the reactor system is necessary to avoid system interactions. Separation of 0.5 meters provides sufficient distance to assure that the irradiator or its experiments do not represent a potential reactivity source to the reactor. The distance also controls the radiation impact of one system on the other. Two examples of possible radiation impacts consider both irradiator and reactor radiations. One example is irradiation of the gamma source by neutrons that induce activation reactions within the source or its experiments. Another example is the irradiation of reactor instrumentation detectors by gamma rays that could cause undesirable background levels at the detectors.

Double encapsulation makes the probability of cobalt-60 escaping into the pool water system extremely remote. Further, the cobalt slugs within the pencils are quite insoluble in water, so that any leakage would occur slowly, allowing time to locate and remove the leaky source. Detection of a source leak will be done by monitoring the concentration of beta-gamma activity in the pool water. Measurements each two months will monitor the pool water activity. In the event a leak does occur the pool water purification system, which operates continuously, will become a factor controlling cobalt-60 accumulation in the deionizing resins and concentration in the pool water.



GAMMA IRRADIATOR

Figure 8-4

An action level for the cobalt-60 concentration in the water is set at $2.5 \times 10^{-5} \mu\text{Ci}/\text{m}^3$, equivalent to the allowable concentration for the monthly release of sewage waste ($3 \times 10^{-5} \text{Ci}/\text{m}^3$). The pool water system contains approximately 10,500 gallons (40,000 liters) with a water treatment rate of 8 gallons per minute (30 liters/min). This treatment rate processes all the water in about 22 hours. However, no discharges to a waste system occur as a part of normal operation.

Equilibrium concentrations in the water balance the source leakage with the cleanup rate,

$$S = (\lambda + q/v)N_0, \quad (1)$$

where

S = is the production rate (leakage) of the source (uCi/hr)

N_0 = equilibrium activity in the pool water (Ci)

λ = radioactive decay constant (hr^{-1})

q = flow rate for the purification system (m^3/hr)

v = volume of the pool water system (40 m^3)

At equilibrium the reference concentration or action level represents a source leak rate of

$$\begin{aligned} S &= (1.5 \times 10^{-5} + 1.8/40) 40 (2.5 \times 10^{-5}) \\ &= 4.5 \times 10^{-5} \text{ Ci/hr.} \end{aligned}$$

Assumptions of the analysis are complete dispersion of the activity in the pool for a uniform concentration and 100% efficiency for cleanup function of the pool water purification system.

Although the assumptions are ideal, none-the-less, the results demonstrate the bounds of possible detectable events. Two situations are of interest:

- a. If a pulse release occurs with no water cleanup or monitoring occurs immediately following the event the detectable leak is one millicurie.
- b. As the recirculation of the water introduces a cleanup rate to balance leakage rate the monthly equilibrium detectable activity is thirty millicuries.

Pulse release:

$$(40m^3) * 10^{-5} \text{ Ci/m}^3 = 1 \times 10^{-3} \text{ Ci} \quad (2)$$

Equilibrium release

$$(24 \text{ hr/dy}) (30 \text{ dy/au}) 4.5 \times 10^{-5} \text{ Ci/hr} = 3 \times 10^{-2} \text{ Ci} \quad (3)$$

Projection of the dose from the cobalt-60 isotope in the deionizer resin has been done by calculation with Microshield [2] (version 3.12).

Table 8-4
Microshield Data [2]

CASE: Dose at deionizer, 30 millicuries cobalt-60, one meter
GEOMETRY: Cylindrical source from side - cylindrical shields

Distance to detector.....	100.	cm
Source length.....	86.	"
Source cylinder radius.....	23.	"
Dose point height from base.....	42.	"
Thickness of shield.....	0.480	"
Air Cap.....	76.520	"
Source Volume: 142924. cubic centimeters		

MATERIAL DENSITIES (g/cc):

Material	Source	Shield	Air gap
Air			0.00120
Iron		7.860	
Water	1.0		

BUILDUP FACTOR: based on TAYLOR method.
Using the Characteristics of the materials in the source region.

INTEGRATION PARAMETERS:

Number of lateral angle segments (Ntheta).....	5
Number of azimuthal angle segments (Mpsi).....	5
Number of radial segments (Nradius).....	5

SOURCE NUCLIDES:

Co-60: 3.0000e-02 curies

RESULTS:

Group #	Energy (MeV)	Activity (photons/sec)	Dose point flux MeV/(sq cm)/sec	Dose rate (mr/hr)
1	1.3359	1.110e+09	7.591e+03	1.370e+01
2	1.1797	1.110e+09	6.736e+03	1.252e+01
3	.6953	1.811e+05	7.741e-01	1.594e-03
TOTALS:		2.220e+09	1.433e+04	2.622e+01

At 30 days the accumulation of 30 millicuries in the deionizer resin represents a dose of 26 millirem/hour at one meter from the tank axis. Table 8-4 contains the data for the analysis. Measurements each two months will monitor the pool water activity.

8.2.2.2. Hazard to Laboratory Personnel

The cobalt-60 facility location is deep enough in the pool so that dose rates at the surface of the pool water and at other points of the pool water system shield will be less than 1 mrem/hr. The location of the source and the geometry of the shield used for these calculations are shown in Figure 8-4.

Calculations assume a point to obtain the dose contributions from the irradiator at the surface of the pool water and outside the concrete reactor shield. Calculations for the irradiation facility at the position shown indicate that the dose rate outside the reactor pool shield structure is just under 1.0 mrem/hr. However, the dose rate at the surface of the pool will be less than .01 mrem/hr. Due to the results of these calculations, the Co⁶⁰ irradiator shall be no less than 4.48 m (14.75 ft) from the pool surface deck. At the normal water elevation of 8.10 meters this will assure 3.00 meters of water shielding. Details of the calculations are given in the following paragraphs.

8.2.2.3. Point Source Shielding Calculations

A conservative estimate of the dose at the surface of the pool water due to the presence of a 10,000 curie Co⁶⁰ source assumes a point source. Calculations are done for distances in pool water and shield concrete. Location of the point source assumes a water depth of 4.05 meters and a shield thickness of 1.2 meters. The dose rate from this source is given by the expression

$$D = \frac{S_0 B e^{-\mu x}}{4\pi r^2 K} \quad (4)$$

where:

- D = dose rate,
- S₀ = source strength (photons/sec),
- B = build-up factor,
- μ = linear absorption coefficient (cm⁻¹),
- x = thickness of shield (cm),
- r = distance from source to reception (cm),
- K = conversion from gamma-ray flux to dose rate.

The linear absorption coefficient depends on the mass attenuation coefficient and the density,

$$\mu = (\mu/\rho) \rho$$

For water and concrete the respective mass attenuation coefficients and densities are as follows:

	μ/ρ	ρ
water	0.060 cm ² /g	1.0 g/cm ³
concrete	0.0567 cm ² /g	2.9 g/cm ³

Calculating μx for the distances in water and concrete determines the shield attenuation and build up effects.

In water,

$$\begin{aligned} \mu x &= (0.0600 \text{ cm}^2/\text{g}) (1.0 \text{ g/cm}^3) (405 \text{ cm}) \\ &= 24.3 \end{aligned}$$

In concrete,

$$\begin{aligned} \mu x &= (0.0567 \text{ cm}^2/\text{g}) (2.9 \text{ g/cm}^3) (122 \text{ cm}) \\ &= 20.1 \end{aligned}$$

The product, μx , not only determines the shield attenuation but also relates to the scattering buildup within a shield thick enough to cause multiple scattering.

The build-up factor, B, is calculated from the expression

$$B = A_1 e^{-\alpha_1 \mu x} + A_2 e^{-\alpha_2 \mu x}$$

where A_1 , A_2 , α_1 , and α_2 are constants [3].

For water the buildup factor constants have the following values (Eγ 1.35):

$$\begin{aligned} A_1 &= 8.5 & A_2 &= -7.5 \\ \alpha_1 &= -0.093 & \alpha_2 &= 0.064 \end{aligned}$$

Thus, the build-up factor may be expressed as

$$\begin{aligned} B &= 8.5 e^{+0.093(24.3)} - 7.3 e^{-0.064(24.3)} \\ &= 81.5 - 1.5 \\ &= 80 \end{aligned}$$

For concrete, the build-up factor constants have the following values (E_γ 1.35 MeV):

$$\begin{aligned} A_1 &= 9.9 & A_2 &= -8.9 \\ \alpha_1 &= -0.088 & \alpha_2 &= 0.029 \end{aligned}$$

Thus, the build-up factor may be expressed as

$$\begin{aligned} B &= 9.9e^{+0.088(20)} - 8.9e^{-0.029(20)} \\ &= 57.5 - 5.0 \\ &= 52.5 \end{aligned}$$

Conversion from gamma-flux to dose rate is

$$K = 4.17 \times 10^5 \text{ (photons/cm}^2\text{-sec)/(rad/hr)}$$

and the source strength is

$$S_0 = (10,000 \text{ curies}) (3.7 \times 10^{10} \text{ dis/sec-Ci}) \times (2 \text{ photons/dis})$$

$$S_0 = 7.4 \times 10^{14} \text{ (photons/sec)}$$

Dose Rate at Pool Surface

$$D = \frac{(7.4 \times 10^{14}) (80) e^{-24.3}}{4 \pi (4.17 \times 10^5) (412)^2}$$

$$D = 1.9 \times 10^{-6} \text{ rad/hr}$$

Dose Rate at the Shield Surface

$$D = \frac{(7.4 \times 10^{14}) (52.5) e^{-20.0}}{4 \pi (4.17 \times 10^5) (120)^2}$$

$$D = 1.1 \times 10^{-3} \text{ rad/hr}$$

In the event of no pool water shielding the dose rate for a 10,000 Curie source at the example location will be equivalent to the calculation of a loss of pool water for the reactor core one hour after operation. The dose rate which will be about 850 rad/hour will require special precautions.

8.3. OTHER EXPERIMENT FACILITIES

8.3.1. Subcritical Reactor and Moderators

Cylindrical assemblies of graphite and polyethylene are utilized for student laboratory experiments with neutron sources and a subcritical uranium-235 reactor assembly. The plutonium-beryllium neutron sources and uranium dioxide used in the polyethylene subcritical assembly may be stored and used in the room containing the reactor, but are licensed separately from the reactor. The subcritical core and moderator assemblies are products of Lockheed Nuclear Products (Figure 8-5).

The subcritical polyethylene core is a cylinder 10 inches in diameter and 14 inches long. Reflector assemblies can be assembled with or without the fueled core. Dimensions of the cylindrical reflector assemblies are 30 inch diameter by 34 inch length for the graphite moderator and 22 inch diameter by 25 inch length for the polyethylene moderator. An additional graphite moderator cylinder 30.5 inches high by 24 inch diameter is available for neutron source moderation.

8.3.2. 14 MeV Neutron Generator

An accelerator is maintained for the production of D-T reaction neutrons for research measurements and activation experiments. The accelerator is a Cockcroft-Walton type with an acceleration voltage of 150 kilovolts and beam current of 50 microamps. Application areas of the source of neutrons are planned in neutron dosimetry, neutron activation, and neutron interactions for analysis of related research problems.

Chapter 8 References

1. "Review of Experiments for Research Reactors", ANSI/ANS 15.6 - 1974 (N401).
2. MICROSHIELD Users Manual, Version 2, Grove Engineering, Rockdale, Maryland, 1985.
3. A. B. Chilton, J. K. Shultis, R. E. Faw, Principles of Radiation Shielding, Prentice Hall Inc., 1984.
4. "Standards for Protection Against Radiation," Part 20, Chapter 10 U.S. Code of Federal Regulations, 1991.

Chapter 9

RADIOACTIVE MATERIALS AND RADIATION MEASUREMENT

Radioactive materials and radiation control within the Nuclear Engineering Teaching Laboratory will be subject to industry standards [1,2] and license conditions [3,4] of state and federal agencies. The U.S. Nuclear Regulatory Commission will regulate the TRIGA reactor, use of special nuclear materials, and related activities. Other materials and activities in the facility will be regulated by the State of Texas Department of Health, Division for Radiation Control. Monitoring and sample programs will control release of effluents and waste. Effluent pathways from the reactor bay consist of the HVAC exhaust air and the purge system exhaust air. No liquid effluent will occur from the reactor bay as a result of normal reactor operation. Effluent pathways from the building consists of fume hoods, exhaust vents of vacuum pumps, and liquid waste discharge from storage tanks. Solid waste for the reactor and facility is packaged as necessary for shipment and disposal. Figures 9-1,2 show material use areas and release pathways.

Features of the building design provide two monitoring points for ground water. One is a sample well in the reactor bay floor. The other point is a sump for water drainage from the reactor shield foundation. Both of these points are for evaluation of environmental conditions. No conditions of normal operation will release effluents to the ground water. Some areas are likely to contain concentrations of radioactive materials for extended periods of time.

9.1. RADIOACTIVE MATERIALS CONTROL

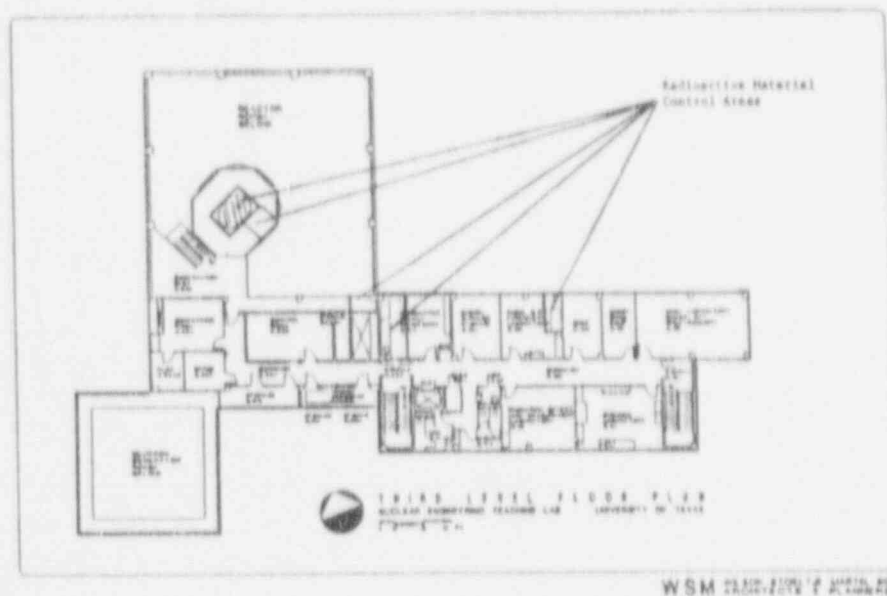
Physical control of radioactive materials shall be provided as an essential part of the radiological safety program. Control shall include identification of items or storage in identified locations. Controls such as shielding, isolation, containment and ventilation will be provided, as necessary, to control radiation exposure to the inventory of radioactive materials.

9.1.1. Reactor Fuel

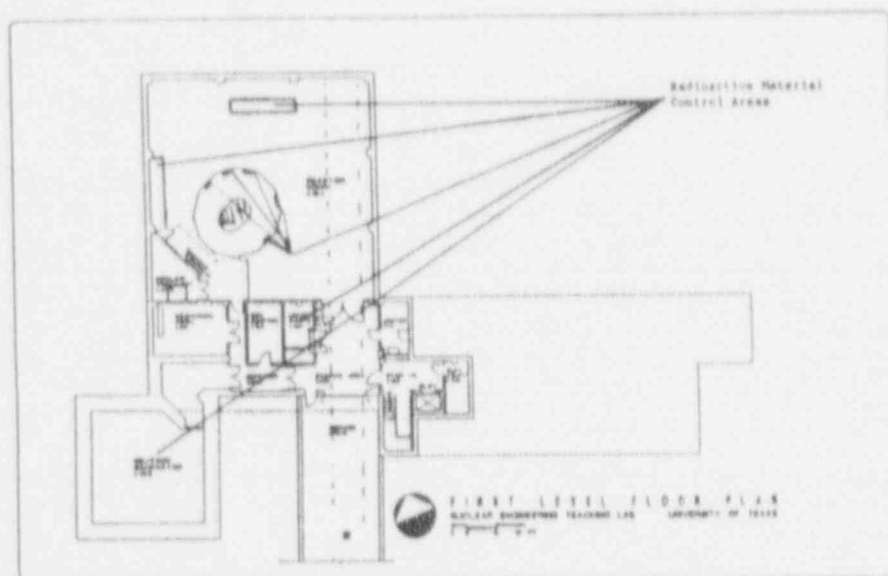
Irradiated reactor fuel shall be maintained in the reactor core, reactor pool storage racks, or reactor bay storage pits. Fuel elements will be removed from these facilities only for transport, measurement, or experimentation. An area of the reactor facility will be designated for the storage of a few single fuel elements of fresh fuel prior to irradiation in the reactor.

9.1.2. 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 an area designated for such components.



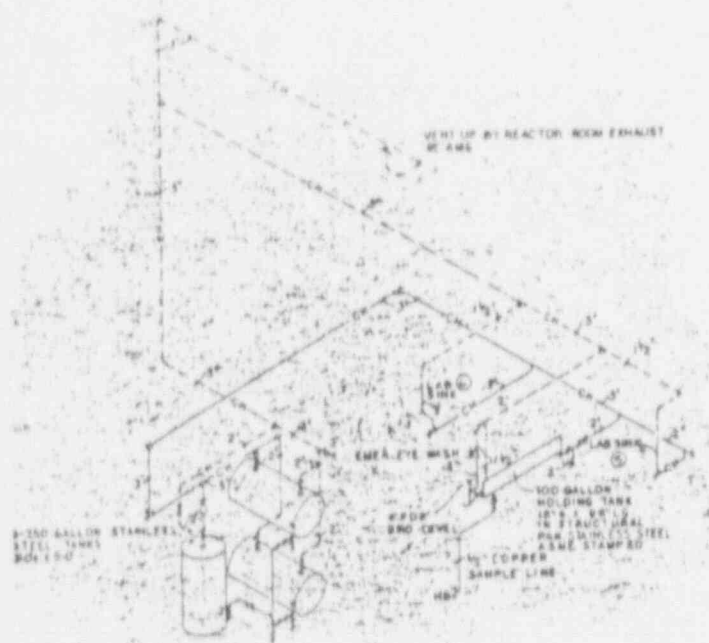
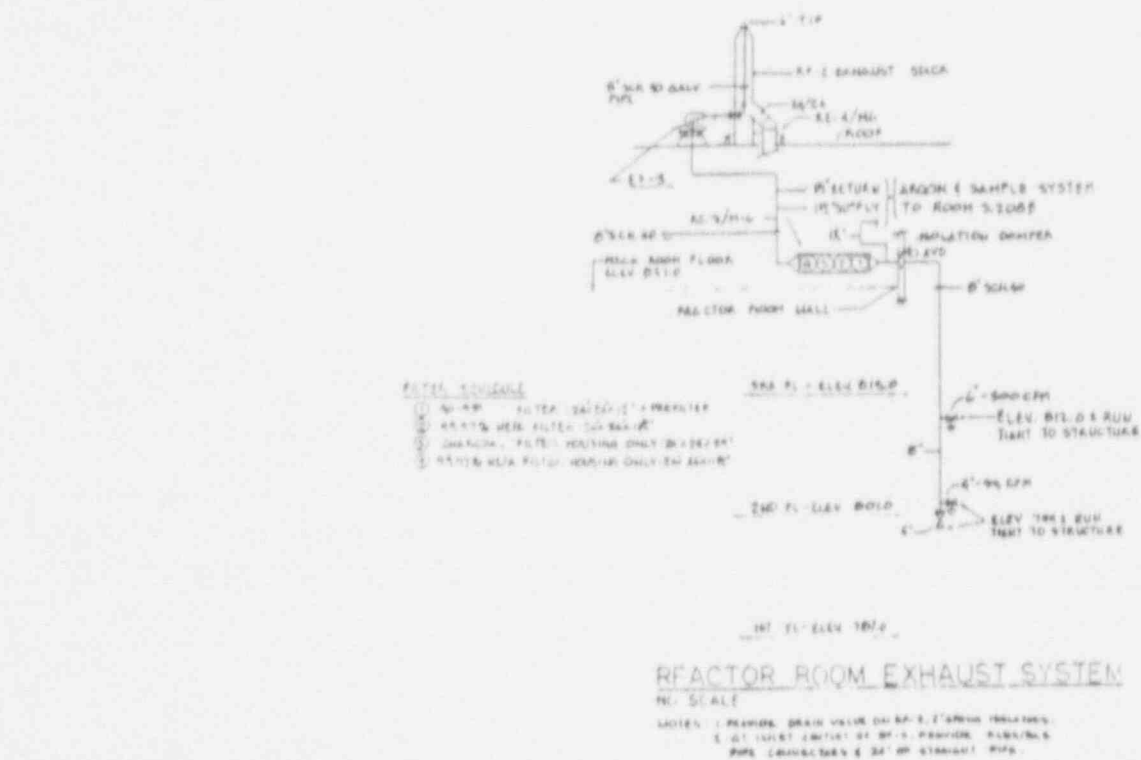
WSM CONTROLS F. 10/10/85



WSM CONTROLS F. 10/10/85

RADIOACTIVE MATERIAL USE AND STORAGE AREAS

Figure 9-1



RADIOACTIVE MATERIAL
EFFLUENT CONTROL SYSTEMS

Figure 9-2

9.1.3. Experiment Facilities

Experiment facilities shall consist of all tubes or penetrations into the reactor core or reflector that provide access to the reactor neutron flux for an experiment application and shall include facilities in which materials are exposed to beams originating from the reactor core. The cobalt-60 irradiator shall also be considered an experiment facility. Removal of experiment facilities from the pool or the beams originating from the reactor shall be subject to the same controls as those for reactor components.

9.1.4. Activated Samples

Materials that are inserted into reactor experiment facilities or reactor beams 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. Samples exposed in a gamma irradiator (cobalt 60) will not be considered activated as radioactive materials. Specific locations for the storage of samples may depend on the analytic status. Locations shall be designated and labeled for storage of samples and sample encapsulations, before and after analysis. Locations should be designated for storage of sample or encapsulation materials that are decaying and that are to be:

- a. analyzed
- b. disposed
- c. released to an authorized user
- d. released as non-radioactive material
- e. and retrieved for subsequent use.

9.1.5. Radioactive Waste

Cannisters shall be available and labeled for radioactive waste at locations where contamination from sample processing or other activities with contaminated materials occur. Locations shall be designated for storage of solid wastes that are to be released for disposal. Liquid wastes shall be maintained in a designated storage location until release criteria are determined, such as decay, dilution, or processing. Specific sinks and drains in the facility that are designated for radioactive materials shall be identified. Gaseous wastes are to be vented through low volume facility hoods according to allowable release criteria. Appropriate monitoring will be applied as required.

9.1.6. Other Materials

Other materials that are to be identified and controlled by identification and location are encapsulated isotopic radiation sources, radiochemical source materials and process equipment, or tools possibly contaminated with radioactive materials. Activity levels of encapsulated and radiochemical sources are expected to vary widely as will the handling and storage precautions. Activity levels associated with process equipment or tools shall be identified so that appropriate handling and storage precautions can be instituted.

9.2. RADIATION MONITORING

Radiation monitoring shall consist of fixed, portable, or sampling type systems. Monitoring systems will be applied to measurement of radiation areas and high radiation areas around the reactor facility, significant contamination within and adjacent to the reactor facility, and radioactive materials and their concentrations in effluents. Monitoring shall be considered for routine operations, abnormal conditions, and emergency situations.

9.2.1 Minimum Procedures

Zone identification, access control, and protective equipment shall be designated. Zone identification for radioactive materials and radiation areas are designated as specified by 10 CFR, part 20 (Standards for Protection Against Radiation). 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, and half mask air purifying respirators.

Continuous monitoring or control of radiation fields in the restricted area around the reactor shall occur whenever levels greater than 100 mrem/hr are produced in accessible areas. The radiation levels may be caused by normal operation of the reactor or an experiment, deviations from normal operation, or easily changed shield configurations. Periodic measurement of accessible areas should occur in locations with significant radiation levels that do not require continuous monitoring. Personnel shall be informed of high radiation levels and care taken to prevent inadvertent increases in the levels. Continuous monitoring may be replaced by periodic monitoring for temporary conditions that do not violate applicable regulations or license constraints.

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 leaves the designated area. Measurements shall provide action levels for removable activities of 500 disintegrations per minute. Periodic monitoring of areas in which contamination is probable shall be of 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.

Airborne radioactive monitoring shall consist of continuous sampling of air particulate activity in the reactor area. Warning levels and action levels will be determined relative to allowable maximum permissible concentrations. Measurements should be sensitive to one maximum permissible concentration change in one hour. Monitoring will occur during reactor operation or activities involving fuel, core,

or experiment facilities, and will provide measurements for routine, abnormal, and emergency conditions. Additional airborne monitoring equipment should be provided for special experiment needs or locations remote to the reactor area particulate monitor.

Effluent monitoring shall be provided for the discharge of the radioactive noble gas argon-41. Monitoring will consist of either the use of integrating dosimeters at a location of interest or sampling of a point in the release path. Measurements shall determine that the dose at a location of interest is either less than ten mrem per year above natural background or two percent of the allowable maximum permissible concentration for the year. Liquid effluents shall be monitored before release by sampling of gross beta-gamma activity. Specific isotopes should be identified and dilutions calculated such that released concentrations averaged over one year do not exceed 1% of the allowable maximum permissible concentrations. Other gaseous or radioactive effluents are to be examined on a case to case basis.

Personnel dosimetry shall be required for access to reactor areas and some other facility activities. Monitoring devices will typically be film badges with pocket dosimeters and thermoluminescent detectors for supplemental measurements. Other personnel monitoring, such as bioassays or whole body counting, will be applied as determined by the activity and conditions of radiation exposure situations. Personnel shall use supplemental dosimetry during activities that deviate substantially from routine operations with supplemental dosimetry also provided for persons visiting areas with potential radiation exposure.

9.2.2. 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, (c) and post operation analysis.

9.2.3. Management Surveillance

A review by management of radiation exposures related to operations that cause significant radiation exposures compared to routine operations will be performed. The review should be applied to determine whether facility modifications or procedures should be implemented to maintain radiation exposures "as low as reasonably achievable."

9.2.4. Frequency and Accuracy

Monitoring frequency and accuracy of activities will be determined by several factors related to personnel access, requirements, probability, and consequences of equipment failure, contamination potential, periodicity of modifications, and adequacy of current monitoring. Accepted standards for measurement sensitivity and accuracy should be appropriate to maintain radiation exposures "as low as reasonably achievable." Frequency and accuracy specifications should be specified by procedures or other documents when appropriate.

9.3. 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, portable neutron sensitive counter, and pocket dosimeters with charger. Other detecting equipment that should be available includes alpha-beta proportional counter, multichannel gamma pulse height analyzer, thermoluminescent detector with reader, alpha scintillation detector, high and low range beta-gamma dose rate meters, and GM tube or equivalent friskers.

9.3.1. Fixed Area Monitors

Fixed area gamma monitors shall have remote readouts with audible and visual alarms at the reactor control console. Local readouts should be provided in areas with significant radiation levels and routine personnel access. A multiple channel area monitoring system with GM type detector probes will be installed. The system should have at least four channels functional. Measurement should include the dose range of 1 millirem/hour to 1 rem/hour.

The fixed area monitors are designated for six general areas. The exact location within an area may vary depending on the presence of experiments or equipment. Three of the fixed locations, that are not likely to change are the control room (room 3.208), the level three access point to the reactor pool, and the level two access point to the reactor shield structure. The three remaining fixed locations in beam port experiment areas are intended to change if the arrangement of experimental equipment requires a more appropriate monitoring configuration. Locations of the three experiment area monitors are at beam port 1 and between beam ports 2 and 3, and ports 4 and 5.

9.3.2. Airborne Radioactivity Monitors

A continuous air particulate monitor with audible and visual alarms shall be functional in the reactor vicinity during reactor operations. A fixed filter beta particulate monitor with 70 lpm flow rate capacity or equivalent system will provide air particulate monitoring. Detectors such as thin window GM detectors will monitor activity and provide alert and alarm conditions with visual and audible annunciators. Count rate of the instrument should include the range of 50 to 50,000 counts/minute.

A gas monitor system for the noble gas effluent, argon-41, shall also be operable during operation, or sufficient data available to demonstrate a calculated release quantity. Design goal for the argon-41 monitor is a sensitivity of 50% of 1 mpc (maximum permissible concentration) for unrestricted areas. Test measurements indicate a sensitivity for a ten minute count of 2×10^{-8} $\mu\text{Ci/cc}$.

The airborne monitors are located at two separate locations. For air particulates the sample point and instrument location are both in the vicinity of the reactor pool observation deck, room 3.206. In contrast the argon-41 gaseous monitor is located in the control room area, room 3.208b, with sampling pipes that allow sampling up stream or down stream of the argon purge system filter bank. The sample return line returns the air to the purge system exhaust. By alignment of the purge system source valves the gaseous monitor has the ability to sample air from the reactor bay room, the reactor pool access area, or the experiment systems manifold. A filter in the sample line is available for analysis of particulates, if abnormal conditions should occur.

9.3.3. Survey and Laboratory Instrumentation

Portable survey monitors for alpha, beta, gamma, or neutron 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 probe or alpha-beta scintillator to detect contamination, (2) ionization chamber for radiation field of 0 to 50 R/hr, and (3) neutron detector with spherical moderator to monitor neutron radiations.

Supplemental measurements should be available with alpha beta proportional counters or gamma ray pulse height analyzers. A TLD (thermoluminescent) detector should be available for measurement and evaluation of doses.

Survey instruments and an alpha-beta proportional counter are located in room 2.208, the Health Physics Laboratory. Gamma ray spectroscopy systems are located in room 3.112.

9.3.4. Liquid Effluent Sampling

Liquid effluents shall be monitored by sampling methods to determine gross alpha-beta activity. Gamma spectral analysis should be applied for identification of isotope mixtures that require substantial dilution for disposal. Liquid effluents shall be released in batches after storage for decay and dilution determinations. Reactor coolant may be monitored for radioactivity in the coolant or purification loops as a supplemental indicator of water activity.

9.3.5. Range and Spectral Response

Instruments shall be available to measure the various types of radiation and the presence of low and high levels of radiation. Several types of detectors should be available for measurement determinations.

9.3.6. Calibrations

Calibration methods, accuracy, frequency, and functional checks shall be established for radiation monitors. Two classes of monitor calibration will be applied. One class of calibration will consist of monitors applied to routine facility operation and surveys. Maintenance, calibration, and functional checks will be subject to reactor operation specifications. The second class of instruments should have functional checks at annual intervals, but may be calibrated infrequently or at the time of application.

9.4 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 survey records,
- e. Instrument calibration records,
- f. Radioactive material receipt and transfer records
- g. Solid radioactive waste disposal records,
- h. Leak tests of sealed sources,
- i. Data on radiological incidents.

9.5. 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. Three types of radiation monitors are used: a continuous air particulate monitor for determining radiation levels due to particulate radioisotopes suspended in the reactor room air, a continuous air gaseous monitor for determining radiation levels due to argon-41 in the room air, and area radiation 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 detect radioisotopes released due to fuel element failure (a design basis accident). The gaseous air monitor is used to determine the effluent radiation release of argon-41. Argon, a component of air (.04% by volume) may be activated to produce argon-41 in potentially significant quantities. Finally, the area radiation monitors are used to minimize personnel radiation exposures. The radiation monitors in section 9.5.1., 9.5.2., and 9.5.3. are typical instruments at the time of original installation. Replacements may have slightly different characteristics.

9.5.1. Particulate Air Monitor

Set points for the particulate continuous air monitor warn of the presence of particulate fission product nuclides. Since gaseous and volatile elements such as krypton, xenon, bromine, and iodine have particulate decay products, the presence of some of their radioisotopes should also be detected. An alarm set point at 2000 picocuries/milliliter detects particulate activity concentrations at the occupational values of 10 CFR 20 for 70% of the relevant isotopes in the ranges 84-105 and 129-149. These ranges of isotopes represent the one percent yield for fission products of uranium 235. Significant fission products as a percent of total release are shown in Table 9-1.

The air monitor in use is a Ludlum Model 333-2 beta air monitor, configured for continuous sampling of airborne beta-emitters. It uses two standard pancake G-M tubes, each having a 1.75 inch effective diameter. The two detectors are arranged in line so that gamma background subtraction is performed. This increases the accuracy of the beta count. The 333-2 will accept air flow rates ranging from 10-100 liters per minute.

Particle accumulation on a fixed filter continues at a constant rate. Activity on the filter, however, is a function of the air flow rate, filter collection efficiency, and the decay rates for nuclides that are present on the filter. If one assumes that the source in the room is constant the activity at the filter will be the accumulation term minus the decay term and will have the same functional form as the activation equation.

$$\frac{dN_f}{dt} = \frac{q}{V} \epsilon N_r \quad (1)$$

with the solution,

$$N_f(t) = \frac{q}{V} \frac{\epsilon N_r}{\lambda} (1 - e^{-\lambda t}) \quad (2)$$

Figure 9-3 represents a plot of equation 2 for the particulate monitor with typical background conditions and the following assumptions:

$$\begin{aligned} q/V &= \frac{6.5 \times 10^4 \text{ cm}^3/\text{min}}{4.83 \times 10^9 \text{ cm}^3} \\ &= 1.35 \times 10^{-6}/\text{min} \\ \epsilon &= 98 \text{ percent} \\ \lambda &= (\ln 2)/(2 \text{ hours}) \\ &= .346 \end{aligned}$$

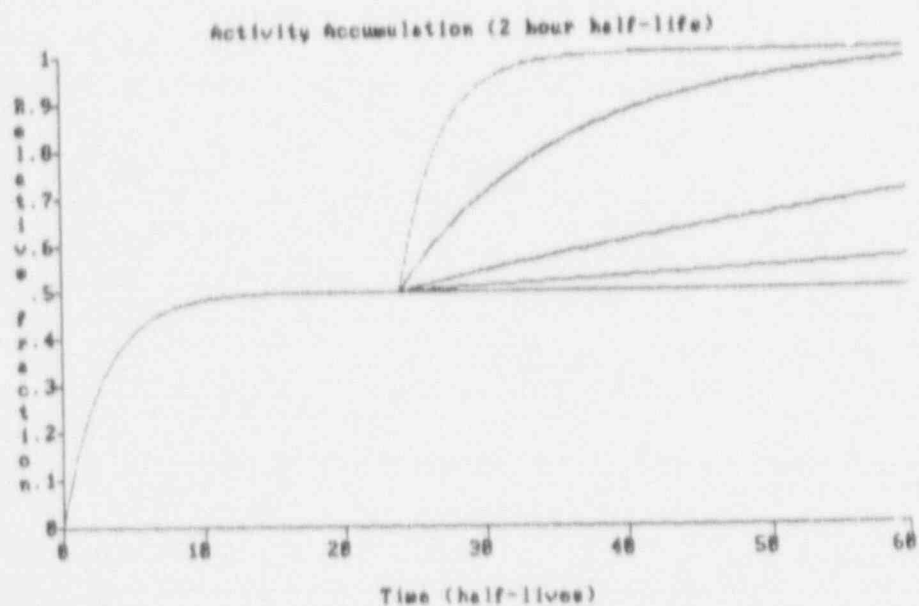
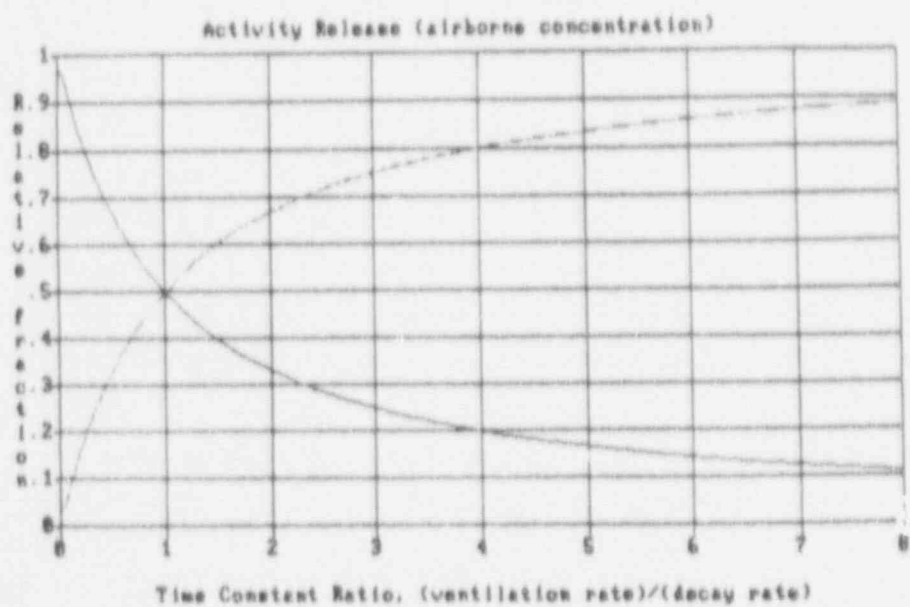
At equilibrium for a 2 hour half-life isotope, the saturation condition is

$$\begin{aligned} N_f(t \rightarrow \infty) &= \frac{q}{V} \frac{\epsilon N_r}{\lambda} \\ N_f(t \rightarrow \infty) &= \frac{2000 \text{ cpm}}{.35} \\ &= 5700 \text{ dpm} \\ N_{f\infty} &= \frac{q}{V} \frac{\epsilon N_r}{\lambda} \\ N_r &= \frac{N_{f\infty}/\epsilon}{(q/V)\lambda} \\ &= \frac{5700/.98}{(1.35 \times 10^{-6}/\text{min})(.346/\text{hr})(1\text{hr}/60 \text{ min})} \\ N_r &= 7.5 \times 10^{11} \text{ #/cm}^3 \end{aligned}$$

Table 9-1

Significant Fission Products
Contribution to Total Activity, Percent

Element	Isotope	1 Day	10 Days	30 Days	90 Days	1 Year	β -energy (MeV)
Strontium	Sr ⁸⁹		2.8	6.7	10.5	2.7	1.49
	Sr ⁹⁰					1.8	.546
Yttrium	Y ⁹⁰						2.29
	Y ⁹¹		3.4	7.6	12.5	3.9	1.55
Strontium	Sr ⁹¹	6.7					1.1, 1.38, 2.6
Yttrium	Y ⁹²	4.2					3.62
	Y ⁹³	7.6					2.89
Zirconium	Zr ⁹⁵		3.7	8.2	14.7	7.3	.366, .398, .688
Niobium	Nb ⁹⁵			4.1	18.2	15	.159
Zirconium	Zr ⁹⁷	9.0					1.93
Niobium	Nb ⁹⁷	9.6					1.28
Molybdenum	Mo ⁹⁹	4.6	6.8				1.23
Rhodium	Rh ^{103m}		2.55	5.5	7.0		
Ruthenium	Ru ¹⁰³		2.65	5.7	7.2		.225
Rhodium	Rh ¹⁰⁵	1.35					.566, .25, .26
	Rh ¹⁰⁶						3.54
Ruthenium	Ru ¹⁰⁶					2.4	.0394
Iodine	I ¹³¹		6.8	3.7			.606, .25, .81
	I ¹³²	2.7	5.3				.80
	I ¹³³	7.3					1.27
Tellurium	Te ¹³²	2.6	5.1				.23
Xenon	Xe ¹³³	1.23	11.4	2.6			.346
Iodine	I ¹³⁵	4.7					1.0, .5, 1.5
Xenon	Xe ¹³⁵	12.5					.91
Barium	Ba ^{137m}						
Cesium	Cs ¹³⁷					1.5	.512, 1.173
Barium	Ba ¹⁴⁰	1.25	10.6	10.8	1.6		1.01, .47
Lanthanum	La ¹⁴⁰		12.0	12.5	2.4		1.36, 1.25, 1.68
Cerium	Ce ¹⁴¹		6.3	11.2	8.5		.436, .581
Lanthanum	La ¹⁴¹	1.4					2.43
Praseodymium	Pr ¹⁴³		10.0	11.2	1.9		.932
Cerium	Ce ¹⁴³	6.8					1.09, .1.39
	Ce ¹⁴⁴			2.0	6.0	26.5	.316, .182
Praseodymium	Pr ¹⁴⁴						3.00
Neodymium	Nd ¹⁴⁷		4.8	4.1			.804, .364
Promethium	Pm ¹⁴⁹	1.45					1.072



ACTIVITY ACCUMULATION ON PARTICULATE FILTER

Figure 9-3

If a sudden appearance of an additional nuclide or group of nuclides of similar half-life occurs, an additional curve similar to the background condition will occur. Cases are shown in Figure 9-3 for 2 hr, 8 hr, 2 dy and 8 dy half-lives.

To determine the count rate set point, a , on the particulate air monitor, the following equation is used:

$$a = \alpha V t \eta, \quad (3)$$

where,

- a = alarm level, cpm
- α = reference activity concentration, $\text{dis}/\text{min}\cdot\text{cm}^3$
- q = air sampling rate, cm^3/min
- t = particulate accumulation time, min
- η = efficiency of the detector, $\text{cpm}/(\text{dis}/\text{min})$

The reference activity concentration may be expressed as follows:

$$\begin{aligned} \alpha &= (2 \times 10^{-9} \mu\text{Ci}/\text{cm}^3) (3.7 \times 10^4 \frac{\text{dis}/\text{sec}}{\mu\text{Ci}}) (60 \text{ sec}/\text{min}) , \\ &= 4.44 \times 10^{-3} \frac{\text{dis}/\text{min}}{\text{cm}^3} . \end{aligned}$$

The following data pertains to the Ludlum Model 333-2 particulate air monitor.

$$\begin{aligned} q &= 6.5 \times 10^4 \text{ cm}^3/\text{min} , \\ \eta &= 0.3 . \end{aligned}$$

An electronic circuit monitors the detector chamber flow rate to assure adequate flow rates. Particulates in the air flow path accumulate on a filter with 1% efficiency for 0.25 micron particles. The value of the detector efficiency is a conservative estimate based on the beta energies of interest. Using a Tc-99 source, the detector efficiency is rated at 36% for a beta spectrum with 0.3 MeV energy range. Since most of the isotopes of interest have higher beta energies than Tc-99, values of detector efficiency will exceed 30% in most cases. In fact, efficiencies of 50% may be applicable for many isotopes, thus assuring a conservative detection limit for all except a few fission product isotopes.

Monitor position to sample reactor room air is within 5 meters of the pool at the pool access level. The location will sample air activity in the vicinity of the reactor pool. Leakage of fission products from the fuel into the water then into the room would occur at the room air to pool water interface. Background measurements of air particulate activities between Sept. 90 and Sept. 91 provide a record of the naturally occurring count rate levels for the Ludlum Model 333-2 in room 1.104 of the NETL facility. These data indicate that count rates of 4000 cpm to 6000 cpm will occur several times each year as a result of weather conditions that effect vertical air stability such as, frontal lines, temperature inversions and storm systems. A set point at 5000 cpm will provide an alert level with an occasional alert for a natural occurring condition.

The count rate alarm set point will assume beta energies of 0.3 and detector efficiency of 30%. The set point may now be calculated:

$$a = (4.44 \times 10^{-3} \frac{\text{dis/min}}{\text{cm}^3}) (6.5 \times 10^4 \text{ cm}^3/\text{min}) (120 \text{ min}) (0.3) \\ = 10,400 \text{ counts/min.}$$

Refer to Table 9-1. Most of the isotopes listed in the 1 day column have beta energies greater than 0.5 MeV. Since the detector efficiency increases with the incident beta energy, a more representative estimate for the detector efficiency may be 50%. The set point may be calculated as follows:

$$a = (4.44 \times 10^{-3} \frac{\text{dis/min}}{\text{cm}^3}) (6.5 \times 10^4 \text{ cm}^3/\text{min}) (120 \text{ min}) (0.5) \\ = 17,300 \text{ counts/min.}$$

A particle accumulation time, t , of two hours may be considered, as shown in Figure 9-3.

9.5.2. Argon-41 Monitor

Set points for the argon-41 continuous air monitor should warn of excessive radiation levels for effluent release and occupational exposure. This radiation monitor will operate whenever the reactor system and the auxiliary air purge system are operating.

As specified in 10 CFR 20, the reference concentration of argon-41 is $1 \times 10^{-8} \text{ } \mu\text{Ci/cm}^3$. Dividing this number by the purge exhaust system flow rate and by the building wake dilution factor yields the average annual concentration limit for release at the stack, which is $2 \times 10^{-6} \text{ mCi/cm}^3$. An alarm set point at ten times this level, $2 \times 10^{-5} \text{ mCi/cm}^3$, will warn of an excessive daily release. In the event of a gaseous fission product release in interference will occur in the argon-41 count due to betas emitted by isotopes of krypton and xenon, refer to Table 9-2.

Monitor position to sample reactor room air is within 5 meters of the pool at the pool access level. The location will sample air activity in the vicinity of the reactor pool. Leakage of fission products from the fuel into the water then into the room would occur at the room air to pool water interface. Background measurements of air particulate activities between Sept. 90 and Sept. 91 provide a record of the naturally occurring count rate levels for the Ludlum Model 333-2 in room 1.104 of the NETL facility. These data indicate that count rates of 4000 cpm to 6000 cpm will occur several times each year as a result of weather conditions that effect vertical air stability such as, frontal lines, temperature inversions and storm systems. A set point at 5000 cpm will provide an alert level with an occasional alert for a natural occurring condition.

The count rate alarm set point will assume beta energies of 0.3 and detector efficiency of 30%. The set point may now be calculated:

$$a = (4.44 \times 10^{-3} \frac{\text{dis/min}}{\text{cm}^3}) (6.5 \times 10^4 \text{ cm}^3/\text{min}) (120 \text{ min}) (0.3)$$

$$\approx 10,400 \text{ counts/min.}$$

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$$\approx 17,300 \text{ counts/min.}$$

A particle accumulation time, t , of two hours may be considered, as shown in Figure 9-3.

9.5.2. Argon-41 Monitor

Set points for the argon-41 continuous air monitor should warn of excessive radiation levels for effluent release and occupational exposure. This radiation monitor will operate whenever the reactor system and the auxiliary air purge system are operating.

As specified in 10 CFR 20, the reference concentration of argon-41 is $1 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$. Dividing this number by the purge exhaust system flow rate and by the building wake dilution factor yields the average annual concentration limit for release at the stack, which is $2 \times 10^{-6} \text{ mCi}/\text{cm}^3$. An alarm set point at ten times this level, $2 \times 10^{-5} \text{ mCi}/\text{cm}^3$, will warn of an excessive daily release. In the event of a gaseous fission product release in interference will occur in the argon-41 count due to betas emitted by isotopes of krypton and xenon, refer to Table 9-2.

Table 9-2

Beta-Emitting, Gaseous Radionuclides of Interest

Isotope	Yield (%)	# isotopes of mass N	Half-life	Beta Energies	%	Reference Level Concentrations in Air ($\mu\text{Ci/ml}$)
^{41}Ar	0	1	109m	2.49 1.20	1% 99%	1×10^{-8}
^{85}Kr	1.33	5	10.7y	0.687	99.57%	7×10^{-8}
$^{85\text{m}}\text{Kr}$	1.33	5	4.68h	0.841	78.8%	1×10^{-8}
^{87}Kr	2.56	5	76m	3.889 3.486 1.335 3.044 1.475	31% 40% 9.2% 7.1% 5.7%	2×10^{-8}
^{88}Kr	3.7	5	2.8h	0.521 2.913 0.681	67% 14% 9.1%	9×10^{-9}
^{89}Kr	4.8	5	3.16m	4.93 2.33 3.24 2.53	23% 15% 10% 5.6%	-
^{133}Xe	6.77	6	5.29d	0.346	99.2%	5×10^{-7}
^{135}Xe	6.7	6	9.17h	0.909	96%	7×10^{-6}
^{138}Xe	6.7	4	14.2m	0.803 2.82 2.38 2.418 0.567	34% 22% 18% 12% 10%	2×10^{-6}

The radioactive gas monitor in use is the P.R.M. Model AR-1000 argon-41 monitor. It uses a 50mm X 0.4mm CaFl scintillator to detect the betas emitted by argon-41. Detection chamber volume is 11.4 liters, and it accepts a nominal gas flow rate of 30 liters per minute. The system automatically performs background subtraction.

To determine the count rate set point on the argon-41 air monitor, the following equation is used:

$$B = \frac{a\eta}{\psi Vf} \quad (4)$$

where

- a = alert level on the AR-1000, cpm
- α = reference concentration level, $\mu\text{Ci}/\text{cm}^3$
- η = detector response, $\text{cpm}/(\mu\text{Ci}/\text{cm}^3)$
- ψ = building wake dilution factor, sec/cm^3
- V = argon purge system flow rate, cm^3/min
- f = fraction of 24 hour day the reactor actually operates

Reference concentration level, α , at ground level outside the building is $1 \times 10^{-8} \mu\text{Ci}/\text{cm}^3$.

For the AR-1000, the detector response, η , has been determined to be

$$\eta = 1.5 \times 10^8 \frac{\text{count/min}}{\mu\text{Ci}/\text{cm}^3}$$

The building wake dilution, ψ , may be calculated from the following equation:

$$\psi = \frac{1}{0.5Av} \quad (5)$$

where

- A = building cross-sectional area, m^2
- v = wind speed, m/sec

The building cross-sectional area, A, is conservatively determined from the smallest side of the reactor building:

$$A \approx 234 \text{ m}^2 = 2.34 \times 10^6 \text{ cm}^2$$

The wind speed, v, is assumed to be 1 meter per second, also a conservative value.

The building wake dilution may now be calculated:

$$\psi = \frac{1}{0.5 (2.34 \times 10^6 \text{ cm}) (100 \text{ cm/sec})} = 8.55 \times 10^{-9} \text{ sec/cm}^3$$

The argon purge system flow rate, V, is $6.14 \times 10^5 \text{ cm}^3/\text{sec}$ (1300 cfm).

Assuming the reactor operates for 8 hours each day, calculate f:

$$f = 8/24 = 0.333$$

The count rate set point on the AR-1000 for an alert may now be calculated:

$$a = \frac{(1 \times 10^{-8} \text{ } \mu\text{Ci/cm}^3) (1.5 \times 10^8 \frac{\text{counts/min}}{\mu\text{Ci/cm}^3})}{(8.55 \times 10^{-9} \text{ sec/cm}^3) (6.14 \times 10^5 \text{ cm}^3/\text{sec}) (0.333)}$$

$$= 858 \text{ counts/min.}$$

Since the alarm concentration level is ten times the value of the alert concentration level, the count rate set point for an alarm would be at 8580 cpm.

9.5.3. 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. When possible, alarm set points for all area radiation monitors will be at either 2 mr/hr or 5 mr/hr. 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 10 CFR 20.

A high radiation area, defined in 10 CFR 20 as having a radiation level > 100 mr/hr, may exist above the pool access area during some operations. The area radiation monitor located above the pool access area will have an alarm set point of 100 mr/hr. Although the doses within the pool protection railings may exceed doses of 100 mrem/hr, the dose exists only in the immediate area of the pool surface. At other locations of the pool shield platform level, the doses are significantly less than 100 mrem/hr, but may exceed the 2 to 5 mrem/hr range. While the reactor is operating, one area radiation monitor will operate above the pool access area, as well as at least two additional area radiation monitors located at other positions around the reactor shield and at the beam port facilities. The radiation monitor system consists of six units with GM tubes that detect dose rates from 1 mr/hr to 10 kr/hr.

9.5.4. Monitor Availability Conditions

Several factors apply to the requirements for availability of the reactor radiation monitoring systems. Among these factors are the types of conditions each monitoring system detects. If one of the continuous air monitors (CAM) is out of service, reactor operations may continue for a limited period of time, provided the other CAM is operating. Reactor operation is permitted for up to one week when the particulate air monitor is inoperable, provided a filter evaluation is performed daily, and a signal from the argon-41 air monitor is available to provide information for manual shutdown of the HVAC. This is necessary to detect the design basis fuel element leak. When the argon-41 monitor is not available, operating the reactor with auxiliary air purge system shall be limited to a period of ten days. This constraint restricts any effluent release from the reactor building.

The particulate air monitor may be inoperable due to either an electronics failure or a pump failure. If an electronics failure occurs, the filter will accumulate particles as usual. Since the flow rate is known, the radioactive particles in the filter may be evaluated daily using a portable detector. The expected drop in detector efficiency may be offset by the eight hour accumulation time. If a pump failure occurs, particles will not accumulate on the filter of the particulate CAM. Instead, a daily evaluation is performed of the radioactive particles accumulated on the filter of the argon CAM. Evaluation of the radioactive particles on the CAM filters should occur near the end of daily reactor operation. This technique will detect the presence of those particulate fission products having half-lives greater than a few hours.

While daily evaluations of the CAM filter will detect a minor, persistent fuel element leak, the signal from the AR-1000 must be monitored for changes resulting from a major sudden fuel element failure. This method detects gaseous fission products, such as xenon and krypton, which have rather short half-lives, as shown in Table 2. The reference level concentrations for most of the xenon and krypton isotopes of interest meet or exceed the reference level concentration of argon-41 (1×10^{-8} mCi/ml); however, the argon-41 alert concentration at the detector is 2×10^{-6} mCi/ml. Because occupational level concentrations for the isotopes of interest are typically 300 times

their reference values and because their betas should be readily detected by the argon continuous air monitor, the argon CAM alert set point can identify concentrations below the occupational levels for these xenon and krypton isotopes.

When the argon-41 monitor is inoperable, argon production and release may be calculated. Provided the shielding configuration (including beam ports) is not altered, argon production and release should not change. The release rates can be calculated from measurement data and design flow rates for air through the stack. A 10 day limit is set for the inoperable period to limit the amount of release without direct measurement. This period represents the maximum averaged over a year. The effluent release during a 10 day period would be about 4% of the average annual concentration limit.

If the reactor is operating, at least half of the six radiation monitors must be operating, one of which must be located near the pool access area. This number of monitoring points including the pool area detector is sufficient to warn of unusual operating conditions.

However, some consideration would be made to assure that monitors are operable within areas of experiment and personnel activity.

Chapter 9 References

1. "Radiological Control at Research Reactor Facilities", ANSI/ANS-15.11 1977 (N628).
2. "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents", ANSI/ANS - 15.12 1977 (N647)
3. "Nuclear Regulatory Commission", Chapter 10, U.S. Code of Federal Regulations, Part 20.
4. "Texas Health Department", Texas Regulations for Radiation Control, Bureau of Radiation Control.

Chapter 10

CONDUCT OF OPERATIONS

10.1 FACILITY ADMINISTRATION

10.1.1. Organization10.1.1.1. Structure

Figure 10-1 illustrates the organizational structure that is applied to the management and operation of the reactor facility. Responsibility for the safe operation of the reactor facility is a function of the management structure of Figure 10-1 [1]. These responsibilities include safeguarding the public and staff from undue radiation exposures and adherence to license or other operation constraints. Functional organization separates the responsibilities of academic functions and business functions. The office of the President administers these activities and other activities through several vice presidents.

Facility operation staff is an organization of a director and at least four full time equivalent persons. This staff of four provides for basic operation requirements. Four typical staff positions consist of a reactor supervisor, reactor operator, health physicist, and research scientist. The reactor supervisor, health physicist, and one other position are to be full time. One full time equivalent position may consist of several part-time persons such as assistants, technicians and secretaries. Faculty, students, and researchers supplement the organization. Titles for staff positions are descriptive and may vary from actual designations. Descriptions of key components of the organization follow.

10.1.1.2. Executive Vice President and Provost.

Research and educational programs are administered through the Office of the Executive Vice President and Provost. Separate officers assist with the administration of research activities and academic affairs with functions delegated to the Dean of the College of Engineering and Chairman of the Mechanical Engineering Department.

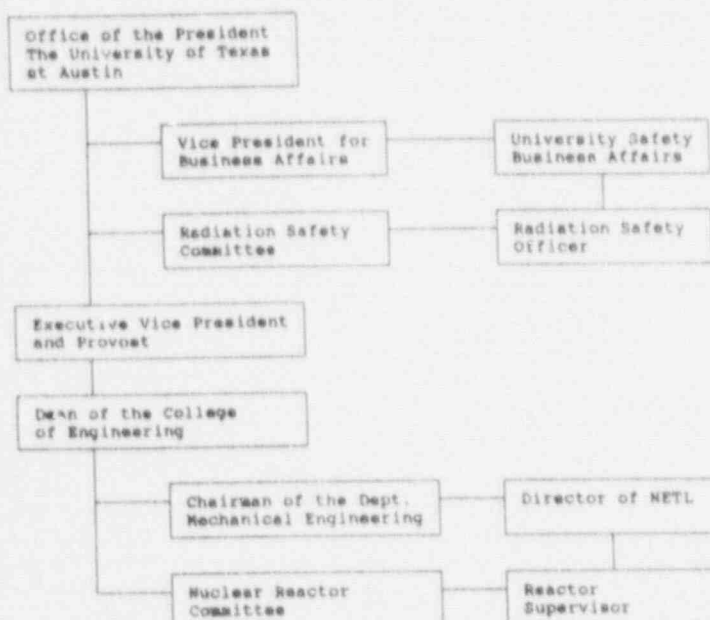
10.1.1.3. Vice President for Business Affairs.

Business activities are administered through the Office of the Vice President for Business Affairs. One responsibility of the office is the administration and operation of safety programs.

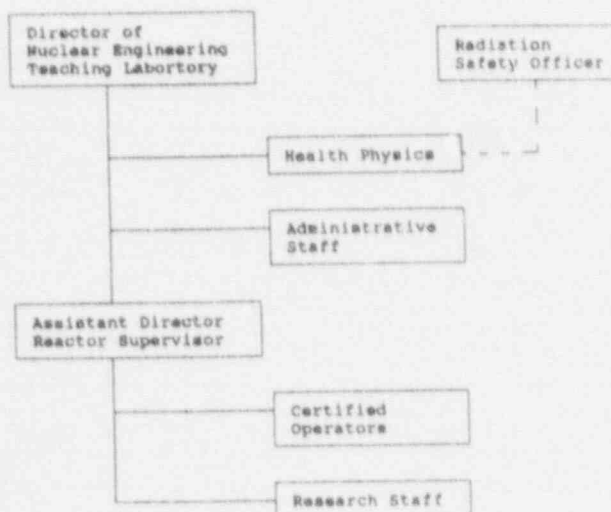
10.1.1.4. Director of Nuclear Engineering Teaching Laboratory

Nuclear Engineering Teaching Laboratory programs are directed by an engineering faculty member that teaches courses in nuclear engineering and performs research related to nuclear applications. The Director is a member of the College of Engineering and Department of Mechanical Engineering.

The University of Texas at Austin
Organization



Nuclear Engineering Teaching Laboratory
Organization



ADMINISTRATION

Figure 10-1

10.1.1.5. Nuclear Reactor Committee

The Nuclear Reactor Committee is established through the Office of the Dean of the College of Engineering of The University of Texas at Austin. Broad responsibilities of the committee include the evaluation, review, and approval of facility standards for safe operation.

The Dean shall appoint at least three members to the Committee that represent a broad spectrum of expertise appropriate to reactor technology. The committee will meet at least twice each calendar year or more frequently as circumstances warrant. The Nuclear Reactor Committee shall be consulted by the Nuclear Engineering Teaching Laboratory concerning unusual or exceptional actions that affect administration of the reactor program.

10.1.1.6. Radiation Safety Officer

A Radiation Safety Officer 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 as determined by the Radiation Safety Committee. The Radiation Safety Program is administered through the University Safety Office and a University Safety Engineer. The responsibilities of the Radiation Safety Officer are outlined in The University of Texas at Austin Manual of Radiation Safety.

10.1.1.7. Radiation Safety Committee

The Radiation Safety Committee is established through the Office of the President of The University of Texas at Austin. Responsibilities of the committee are broad and include all policies and practices regarding the license, purchase, shipment, use, monitoring, disposal, and transfer of radioisotopes or sources of ionizing radiation at The University of Texas at Austin.

The President shall appoint at least three members to the Committee and appoint one as Chairperson. The Committee will meet at least once each year on a called basis or as required to approve formally applications to use radioactive materials. The Radiation Safety Committee shall be consulted by the University Safety Office concerning any unusual or exceptional action that affects the administration of the Radiation Safety Program.

10.1.1.8. Reactor Supervisor

Reactor operation at the Nuclear Engineering Teaching Laboratory 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. A UT TRIGA Operations Manual will be maintained by the reactor supervisor.

10.1.1.9. Health Physicist

Radiological safety of the Nuclear Engineering Teaching Laboratory is monitored by a health physicist, who will be knowledgeable of the facility radiological hazards. Responsibilities of the health physicist will include calibration of radiation detection instruments, measurements of radiation levels, control of radioactive contamination, maintenance of radiation records, and assistance with other facility monitoring activities.

Activities of the health physicist will depend on two conditions. One condition will be the normal operation responsibilities determined by the director of the facility. A second condition will be communications specified by the radiation safety officer. This combination of responsibility and communication provides for safety program implementation by the director, but establishes independent review. Health physicist's activities will meet the requirements of the director and the policies of an independent university safety organization.

10.1.1.10. Professional and Classified Staff

Professional and classified staff, such as research scientists, research engineers, reactor operators, technicians and secretaries, will supplement the organization as necessary to support facility programs.

10.1.2. Qualifications

Personnel associated with the research reactor facility [2] shall have a combination of academic training, experience, skills, and health commensurate with the responsibility to provide reasonable assurance that decisions and actions during all normal and abnormal conditions will be such that the facility and reactor are operated in a safe manner.

10.1.2.1. Job Descriptions

Qualifications for university employment positions are subject to job descriptions that summarize the job function and scope. The typical description includes title, duties, supervision, education, experience, equipment, working conditions, and other special requirements for the job position. University job positions are separated into three generic classification types. The three types are academic faculty, professional staff, and classified staff. Typical staff appointments will include personnel from each of these categories.

10.1.2.2. Facility Director

A combination of academic training and nuclear experience will fulfill the qualifications for the individual identified as the facility director. A total of six years experience will be required. Academic training in engineering or science, with completion of a baccalaureate degree, may account for up to four of the six years experience. The director is generally a faculty member with a Ph.D. in nuclear engineering or a related field.

10.1.2.3. Reactor Supervisor

A person with special training to supervise reactor operation and related functions will be designated as the reactor supervisor. The reactor supervisor will be qualified by certification as a senior operator as determined by the licensing agency. Additional academic or nuclear experience will be required as necessary for the supervisor to perform adequately the duties associated with facility activities. The supervisor is typically a person with at least one graduate degree in nuclear engineering or a related field.

10.1.2.4. Health Physicist

A person with a degree related to health, safety, or engineering, or sufficient experience that is appropriate to the job requirements will be assigned the position of health physicist. A degree in health physics or similar field of study and some experience is preferred. Certification is not a qualification, but work towards certification should be considered a requirement.

10.1.2.5. Professional and Classified Staff

Qualifications vary substantially for other staff positions. Technical skills and requirements range from low to high. Education, training and experience may vary in a similar manner. Qualifications for operator certification by senior operator permit or reactor operator permit will require special consideration. This consideration will be necessary to determine that the knowledge and skills are sufficient to expect successful completion of training and certification by the licensing agency.

10.1.3. Reactor Operations

Operation of the reactor and activities associated with the reactor, control system, instrument system, radiation monitoring system, and engineered safety features will be the function of staff personnel with the appropriate license certifications [2]. Operation will include the implementation of required procedures, execution of appropriate experiments, actions related to safety, and the preparation of required reports and records.

10.1.3.1. Staffing

All activities that require the presence of license certified operators will also require the presence in the facility complex of a second person capable of performing prescribed written instructions. Unexpected absence of a second person for greater than two hours will be acceptable if immediate action is taken to obtain a replacement. A designated license certified senior operator will be readily available on call during all periods in which activities requiring a certified operator are being performed. The person on call will be considered available if the time to initiate a call request and respond on site is less than 1 hour.

Movement of fuel or control rods and relocation of experiments with greater than one dollar reactivity worth will require the presence of a license certified senior operator. Other activities, such as initial startup, recovery from unscheduled shutdowns and modifications to instrument systems, control systems, safety systems, radiation measurement equipment or engineered safety features, will require concurrence and documentation by a license certified senior operator.

Operation of reactor controls, movement of reactor experiments, maintenance of instrument control, safety, and radiation measurement systems will require the presence of a license certified operator. A license certified operator will be present in the control room whenever the reactor is not shut down by more than one dollar of reactivity or the control and system console panel is not secured.

The staff required for performing experiments with the reactor will be determined by a classification system specified for the experiments. Requirements will range from the presence of a certified operator for some routine experiments to the presence of a senior operator and the experimenter for other less routine experiments. Some other activities that occur in the area of the reactor will require knowledge of a license certified operator, but not necessarily the presence of the operator. Such activities will include maintenance, handling of radioactive materials and experiment preparation.

10.1.3.2. 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 certified operators. Substantive changes or major modifications to procedures, and prepared procedures will be submitted to the Nuclear Reactor 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.

Activities subject to written procedures will include routine startup, shutdown and operation of the reactor; fuel loading, unloading and movement within the reactor; and routine maintenance of major components of systems that could have an effect on reactor safety.

Activities subject to written procedures will also include surveillance tests and calibrations that may effect reactor safety; administrative controls for operation and maintenance that could effect core reactivity or reactor safety; personnel radiation protection and implementation of the emergency plan.

10.1.3.3. Experiments

Proposed experiments will be submitted to the reactor committee for review and approval of the experiment and its safety analysis [3]. Substantive changes to approved experiments will require reapproval while insignificant changes that do not alter experiment safety may be approved by the reactor supervisor or designated senior operator. Experiments will be approved first as proposed experiments for one time application, and subsequently, as approved experiments for repeated applications following a review of the results and experience of the initial experiment implementation.

Each experiment will be designated as one of three classes. One class will consist of experiments such as routine reactor operation for calibration or instruction, and routine irradiations such as neutron activation analysis. This class of experiment will require only the reactor operator during the reactor operation or experiment set up. A few experiments may require the presence of both a certified operator and the experimenter, and will be designated as a separate class of experiment. Another class of experiments will be specified for experiments that require large reactivity changes, such as experiment facility movement, fuel or control rod movement, or significant changes to shielding of core radiation. This class will require the supervision of a senior operator.

10.1.4. Actions and Reports

10.1.4.1. Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to licensing authorities within three months following the end of each prescribed year. Each annual operating report shall include the following information:

- a. A narrative summary of reactor operating experience including the energy produced by the reactor or the hours the reactor was critical, or both.
- b. The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.
- c. Tabulation of major preventive and corrective maintenance operations having safety significance.

- d. Tabulation of major changes in the reactor facility and procedures, and tabulation of new tests or experiments, or both, that are significantly different from those performed previously, including conclusions that no unreviewed safety questions were involved.
- e. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the owner-operator as determined at or before the point of such release or discharge. The summary shall include, to the extent practicable, an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed or recommended, a statement to this effect is sufficient.
- f. A summarized result of environmental surveys performed outside the facility.
- g. A summary of exposures received by facility personnel and visitors where such exposures are greater than 25% of that allowed or recommended.

10.1.4.2. Safety Limit Violation

Actions to be taken in the case of safety limit violation shall include cessation of reactor operations until a resumption is authorized by the licensing authority, a prompt report of violation to license authorities and management, and a subsequent follow-up report reviewed by the reactor committee and submitted to the license authority. The follow-up report shall describe applicable circumstances leading to the violation including causes and contributing factors that are known, effect of the violation upon reactor facility components, systems or structures, health and safety of personnel and the public, and corrective action to prevent recurrence. Prompt reporting of the event shall be by telephone and confirmed by written correspondence within 24 hours. A written report is to be submitted within 14 days.

10.1.4.3. Release of Radioactivity

Actions to be taken in the case of release of radioactivity from the site above allowable limits shall include a return to normal operation or reactor shutdown until authorized by management if necessary to correct the occurrence a report to management and license authority, and a review of the event by the reactor committee at the next scheduled meeting. Prompt reporting of the event shall be by telephone and confirmed by written correspondence within 24 hours. A written report is to be submitted within 14 days.

10.1.4.4. Other Reportable Occurrences

Other events that will be considered reportable events are listed in this section. A return to normal operation or curtailed operation until authorized by management will occur. Appropriate reports shall be submitted to license authorities. (Note: Where components or systems are provided in addition to those required by the technical specifications, the failure of components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.)

- a. Operation with actual safety-system settings for required systems less conservative than the limiting safety system settings specified in the technical specifications.
- b. Operation in violation of limiting conditions for operation established in the technical specifications unless prompt remedial action is taken.
- c. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function, unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdowns.
- d. Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks) where applicable which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.
- e. An observed inadequacy in implementation of administrative or procedural controls such that the inadequacy causes, or could have caused, the existence or development of an unsafe condition with regard to reactor operations.

10.1.4.5. Other Reports

A written report within 30 days to the chartering or licensing authorities of:

- a. Permanent changes in the facility organization involving Director or Supervisor.
- b. Significant changes in the transient or accident analysis as described in the Safety Analysis Report.

10.1.5. Records

Records of the following activities shall be maintained and retained for the periods specified below [4]. The records may be in the form of logs, data sheets, or other suitable forms. The required information may be contained in single or multiple records, or a combination thereof.

10.1.5.1. Lifetime Records

Lifetime records are records to be retained for the lifetime of the reactor facility. (Note: Applicable annual reports, if they contain all of the required information, may be used as records in this section.)

- a. Gaseous and liquid radioactive effluents released to the environs.
- b. Offsite environmental monitoring surveys required by Technical Specifications.
- c. Radiation exposure for all personnel monitored.
- d. Updated drawings of the reactor facility.

10.1.5.2. Five Year Period

Records to be retained for a period of at least five years or for the life of the component involved whichever is shorter.

- a. Normal reactor facility operation (supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least one year).
- b. Principal maintenance operations.
- c. Reportable occurrences.
- d. Surveillance activities required by technical specifications.
- e. Reactor facility radiation and contamination surveys where required by applicable regulations.
- f. Experiments performed with the reactor.
- g. Fuel inventories, receipts, and shipments.
- h. Approved changes in operating procedures.
- i. Records of meeting and audit reports of the review and audit group.

10.1.5.3. One Training Cycle

Training cycle records to be retained for at least one training cycle are the requalification records of certified operations personnel. Records of the most recent complete cycle shall be maintained at all times the individual is employed.

10.2. OPERATOR REQUALIFICATION

10.2.1. Introduction

Reactor operator requalification applies to all the controls and some features of the TRIGA reactor at The University of Texas at Austin (UT), Balcones Research Center (BRC). The purpose of this plan is to provide training of each individual that is to qualify for a license to operate or direct the operation of the TRIGA reactor. There are two license classes, one is an operator and the other is a senior operator. License qualification by written and operating test, and license issuance or removal, are the responsibility of the U.S. Nuclear Regulatory Commission. No rights of the license may be assigned or otherwise transferred and the licensee is subject to and shall observe all rules, regulations and orders of the Commission. Requalification training maintains the skills and knowledge of operators and senior operators during the period of the license. Training also provides for the initial license qualification.

10.2.2. Operator License Status

Active status of any licensee shall require the performance of the functions of an operator or senior operator for a minimum of four hours each calendar quarter. If the condition of an active license status is not met, the Director of the facility shall certify, (1) that the qualifications and status of the licensee are current and valid, and (2) that for recertification a minimum of six hours of license functions have been done. The license functions shall be done with supervision of the appropriate operator or senior operator. Otherwise the license status shall be inactive and no functions of the license shall be done.

10.2.3. Requalification Program Bases

Regulatory requirements and standards provide guidance for requalification training. Specific regulatory requirements are found in 10CFR55 for the licensing of operators and senior operators with regulations for requalification set forth in section 55.59. Standards for the selection and training of facility personnel and reactor operators are available in ANS 15-4. Specific regulations in the form of two sets of license conditions also apply to the facility personnel and reactor operators. One set of conditions for the facility license, 10CFR 50.54, applies to facility personnel. The other set of conditions for individual licenses, 10CFR 55.53 applies to operators and senior operators. The following plan documents the requalification of operators and senior operators for the UT BRC TRIGA reactor facility.

10.2.4. Requalification Program

The requalification program consists of training personnel by lectures, instruction, discussion and self-study. At times the number of operators with licenses may be as few as 1 or 2. In these circumstances the application of discussion and self-study methods are necessary to accomplish the training process.

10.2.5. Schedule

Lectures from the requalification program topics and on-the-job training will be done on a two year cycle for the completion of all requirements. The part of the program done each year will consist of six lectures, two on-the-job training activities, and the performance of sixteen hours of license functions.

Lectures or instruction on the topics of the requalification program consist of eight topics shown in section 10.2.6. Three lectures will be given each six months so that during the year there is an average of one topic presentation every two months. Each of the eight topics will occur during the two year cycle with four topics each year. The other two lectures each year are available for special subjects, repeat subjects or review.

On-the-job training relies on two specific reactor control manipulations to be done each year. These control manipulations will consist of startup, shutdown, operation, coolant loss, loss of control rod pneumatic air or electrical power events, and other system malfunctions. The two control manipulations will require a change in system reactivity and will use events from each of the two lists in section 10.2.7. One event from the list of section 10.2.7.1 must be done each year and one event from the list of section 10.2.7.2 must be done each two years. A program of less than two years duration may accelerate the training of persons for new operator certification.

10.2.6. List of Subjects

- a. Theory and principles of operation.
- b. General and specific plant operating characteristics.
- c. Plant instrumentation and control systems.
- d. Plant protection systems and Engineered safety systems.
- e. Normal, abnormal, and emergency operating procedures.
- f. Radiation control and safety.
- g. Technical specifications.
- h. Applicable portions of Title 10, Chapter I, Code of Federal Regulations.

10.2.7. On-the-job training:

10.2.7.1. List of annual training tasks; (one must be done each year):

- a. Plant or reactor startups to include a range that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established.

- b. Plant shutdown.
- c. Significant (>10 percent) power changes in manual rod control.
- d. Loss of coolant inside or outside primary confinement.
- e. Loss of coolant, large or small, including leakrate estimate.
- f. Loss of control air (or inadequate pressure).
- g. Loss of electrical power (or degraded power sources).
- h. Loss of core coolant flow/natural circulation.

10.2.7.2. List of training tasks; system malfunctions (one must be done each two years):

- a. Reactor trip.
- b. A nuclear instrumentation failure.
- c. Loss of protective system channel.
- d. Control rod or drive failure such as rod position error, rod drop or stuck drive, mispositioned control rod or rods (or rod drops), inability to drive control rods.
- e. Fuel cladding failure or high activity in reactor coolant or offgas.
- f. Malfunction of an automatic control system that affects reactivity.

10.2.7.3. On-the-job training will perform the following periodic training checks or functions.

- a. Observation at least once each year of a satisfactory understanding of the reactivity control system and knowledge of operating procedures.
- b. Each operator or senior operator will review facility design changes, procedure changes and license changes as they occur or once each 6 to 8 months.
- c. A review of the contents of abnormal and emergency procedures will be done by each operator or senior operator at 6 to 8 month intervals so that at least 3 reviews occur during the two year training cycle.

10.2.8. Evaluation

Evaluation of license personnel depends on annual examination and periodic observation. The evaluation by annual written examination determines the knowledge level and requirements for retraining by a percentage test score. Other evaluations by visual observation assess the performance and competency with routine procedures and the skill at manipulating the controls of the reactor.

The written annual examination will assess operator or senior operator knowledge of current training subjects and review requirements. A five part test with objective questions will assess the knowledge of four of the eight program subjects, and the areas of section 7.3 b and c. These two sections pertain to changes in facility design, normal procedures, reactor license, and abnormal and emergency procedures.

Each of the five parts of the exam will have a 100 point basis with an average of 80% as the acceptance criteria. An overall score of less than 65% shall require an immediate evaluation of license duties. Proficiency by retraining shall demonstrate acceptance within 4 months or license duties shall suspend until proficiency is acceptable. A person that scores between 65%-80% shall retrain as necessary in those areas that written or oral exams indicate a deficiency.

A systematic observation of license activities, by a supervisory senior operator or a level of the facility management to which a supervisory operator is responsible will evaluate operator and senior operator performance. Visual observation of the performance in response to the conditions of sections 7.2.1 and 7.2.2 will provide the basis for judgement of the operator's skill. In the case of a senior operator the performance may be either direct actions or the direction of a response by another operator. Judgements of a person's skill or competency is subjective and may include general observations of performance at any time the person is responsible for license functions.

10.2.9. Records

Records for each operator or senior operator will consist of the documentation for the requalification activities within the two year training cycle. The records will be kept until the completion of the next training cycle. A record for each operator includes at least the following information:

- a. Attendance at training lecture or acceptable review of the material, including topic and date.
- b. Completion, satisfactorily, of two on-the-job training events with performance evaluation; recording date and performance as excellent, average or poor.
- c. Total # of reactor control system hours and energy production in each calendar quarter.
- d. Scores of the written examination and copies of the exam questions, answers, and responses by personnel.

10.3 RADIOLOGICAL PROTECTION PROGRAM

Protection of personnel and the general public against hazards of radioactivity and fire is established through the safety programs of the University Safety Office. Safety programs at the reactor facility supplement the university programs so that appropriate safety measures are established for the special characteristics of the facility [5,6].

10.3.1. Management and Policy

Radiological management policy shall include a commitment to keep occupational exposures as low as is reasonably achievable to facility personnel and the general public. Other elements of the radiological management will include:

- a. instruction of personnel in awareness of the low as reasonably achievable commitment,
- b. identification of radiation protection personnel and their responsibilities,
- c. authority of personnel to communicate with management and modify or suspend activities for reasons of radiation protection,
- d. assurance of sufficient and appropriate training of personnel in radiological safety,
- e. periodic evaluations of the program to determine possibilities for lower radiation exposures.

Suggestions and recommendations for modifications to operating and maintenance procedures and to reactor equipment and facilities shall be considered by management to reduce exposure to radiation. Implementation of modifications will occur if substantial exposure reductions are possible at acceptable cost.

10.3.2. Responsibilities

Radiation protection at the reactor facility is the responsibility of the Reactor Supervisor, Health Physicist, or a designated senior operator in charge of operation activities. Responsibility shall include the authority to act on questions of radiation protection, the acquisition of appropriate training for radiation protection and the reporting to management of problems associated with radiation protection.

10.3.3. Organizational Access

The person responsible for radiation protection at the reactor facility will have access to other individuals or groups responsible for radiological safety at the University. Contact with the Radiation Safety officer will occur on an as needed basis and contact with the Reactor Committee will occur on a periodic basis.

10.3.4. Equipment and Supplies

Equipment and supplies maintained for radiological safety management shall include:

- a. fixed area radiation monitors,
- b. air particulate monitor,
- c. gaseous effluent monitor,
- d. portable radiation monitors,
- e. detectors for contamination measurement,
- f. maintenance and calibration capacity for equipment,
- g. laboratory counting and analysis equipment,
- h. supplies for storage of contaminated equipment,
- i. provisions for radioactive waste disposal,
- j. decontamination facilities,
- k. protective clothing,
- l. respiratory protection equipment,
- m. and emergency response equipment.

10.3.5. Training and Safety

Each person in the restricted area of the reactor facility shall have sufficient radiological safety training for the purpose of access to the area or be escorted by a person with the appropriate training. Training will be appropriate to the activities of persons admitted to the area and will range from simple instructions of emergency alarms and evacuation procedures to more complex implementation of the area emergency plan.

Training for facility personnel shall be specified by the Reactor Supervisor and shall provide sufficient training in radiation safety policies and procedures, and in the use of radiation safety equipment located in the facility to control exposure during normal, abnormal and emergency situations. Training will consist of:

- a. radiological safety policies, plans and procedures,
- b. Radiation hazards and health risks,
- c. use of protective clothing and equipment
- d. use of portable radiation monitoring equipment,
- e. and other documents such as the emergency plan and federal and state notices to workers.

An evaluation shall occur every two years to determine whether additional training of personnel is required and that the radiological safety program is functioning adequately.

Safety programs, with the exception of reactor operations, are operated as a function of the business administration of the University and include a radiation safety organization as presented in Figure 10-1.

10.4 FIRE PROTECTION PROGRAM

Fire protection consists of 3 factors, passive fire protection, active fire protection, and fire prevention. Management of the Nuclear Engineering Teaching Laboratory shall be knowledgeable of fire protection controls. The controls will consist of actions of equipment, actions of laboratory staff and interactions with University inspection personnel.

10.4.1. Facility Fire Protection Elements

Fire protection is recognized as an important element of the safe operation of the TRIGA reactor facility. Commitment by the University to fire protection is provided by the functions of the University Safety Office.

The organization for fire protection consists of the University Fire Marshall, a member of the University Safety Office and the Reactor Supervisor, a member of the Nuclear Engineering Teaching Laboratory. Responsibilities of the Fire Marshal are the maintenance of fire protection equipment and inspections for fire prevention. Responsibilities of the Reactor Supervisor are knowledge of potential hazards and implementation of fire protection recommendations.

Although fire protection is provided for the general safety of personnel and preservation of property, special considerations shall be provided for systems designated as safety related. Primarily special considerations are applied to protection of the reactor and shield structure, and fuel storage wells. Design features of these facility components provide a major factor of the fire protection. Fire protection for the instrumentation and control system, and radiation measurement systems are important for the initial reactor shutdown and the availability in emergency conditions. Fire protection of the reactor bay area boundary is of importance to the extent of limiting either internal conditions that would cause the release of hazardous materials or external conditions that would threaten the release of hazardous materials.

Loss criteria for decisions on fire protection at the reactor facility shall consist of preventing any injury to personnel, and minimizing the potential or actual release of radioactivity to the environment. No injury or exposure to the public should occur from the adverse effects of a fire.

Laboratory personnel, particularly certified operators, shall be instructed to continually observe conditions that might represent a risk to fire protection. Appropriate assessment of the risk should be provided by the Reactor Supervisor and will include consultation with the Fire Marshal when appropriate.

Passive fire protection elements effectively protect the reactor core, fuel elements and storage wells. Inherent design of the reactor bay and reactor tank structure, construction materials, building layout and fire barriers are all applied to the protection. Instrumentation and control systems and radiation measurement systems primarily are protected by fire detection and alarm information. These systems are important to safety only for the initial shutdown and removal of personnel. Protection of other equipment and the reactor bay boundary is accomplished in part by building design, but primarily by detection and alarm.

10.4.2. Facility Fire Protection Control

The Reactor Supervisor and the Nuclear Reactor Committee shall consider the impact of major facility modifications and experiment programs on facility fire protection. The University Fire Marshal will recommend fire protection requirements and provide for inspection and test of fire protection components.

Activities such as welding, cutting, open flames, electrical loads, or other equipment that effect fire protection shall be examined on a case by case basis by the Reactor Supervisor.

Laboratory staff shall be instructed in fire response actions and notification of response personnel. A program to familiarize response personnel with laboratory equipment, material hazards and physical layout is considered the major element for response of emergency response organizations.

10.4.3. Fire Safety Assurance

At intervals of two years the fire protection program should be examined actively by the Reactor Supervisor, University Fire Marshal and Nuclear Reactor Committee. Evaluations of past inspections, tests or incidents shall be incorporated into an assessment of the fire protection evaluation. Recommendations if any should be identified and appropriate actions taken.

10.5. SECURITY AND EMERGENCY PLANS

Plans for physical security and emergency response shall be established, maintained, and implemented by the Reactor Supervisor. These plans will be separate documents with application procedures. Each plan development will use available reference documents for guidance. Review of the plans will occur at two year intervals.

10.6. QUALITY ASSURANCE PROGRAM

Objectives of quality assurance (QA) may be divided into two major goals. First is the goal of safe operation of equipment and activities to prevent or mitigate an impact on public health and safety. Second is the reliable operation of equipment and activities associated with education and research functions of the University. The risk or potential release of radioactive materials is the primary impact on public health and safety, and may be divided into direct risks and indirect risks. Direct risks are activities such as waste disposal, fuel transport and decommissioning that introduce radioactive materials into the public domain. Indirect risks are accident conditions created by normal or abnormal operating conditions that generate the potential or actual release of radioactive materials from the controlled areas of a facility.

10.6.1 Introduction

Characteristics of uranium loaded zirconium hydride fuel used in the TRIGA reactor provide substantial benefits to safe reactor operation. Many accident situations are simulated by normal operation of the fuel in either pulse mode or steady state mode. Other features, such as fission product retention, stainless steel cladding design, facility engineered features, and periodic schedule of operation, combine with routine operation procedures to decrease the consequences of failure of any reactor components. The limited scope of application of formal quality assurance criteria is due to the fact that most parts and procedures associated with operation of the TRIGA type reactor are not relevant to public health and safety.

Safety-related identifications for quality assurance are determined from safety analyses. Although several systems such as the reactor safety and protection system, engineered safety features and radiation monitoring systems are important to safety, only one reactor component is identified as safety-related. The quality assurance program is not applied to routine reactor operations and surveillance activities but shall be implemented for non routine activities determined to be safety-related in nature or affecting safety-related components. Activities shall include design, construction, testing, modification and maintenance of safety-related items. Other components related to safety limits, limiting conditions for operation and design features, as identified in technical specifications, will apply only those elements of quality assurance necessary to establish reliable performance of the intended structure, system or component function. The following table lists the components subject to quality assurance program or selected sections of the program.

Two additional conditions remain, however, that are important to the application of at least portions of the quality assurance program. One is the safety to operation personnel and experimenters and the other is continuity of the operations programs. Each of these conditions must be examined objectively relative to operation procedures and program expectations. In general, the application of good industry quality assurance practices is sufficient to meet operational program goals.

Table 10-1
Q-List for 1MW UT TRIGA

Structures, systems or components	Safety*	QC/QA Specs
Fuel Element		
Cladding structure	1	manuf.
Shipping package	1	
Reactor Core		
Structural components	2	manuf.
Tank structure	2	design
Shield structure	2	constr.
Experiment Equipment (core reflector)		
Beam tube components	2	design
Rotary rack system	2	design
Experiment Equipment (core grid)		
Pneumatic tube components	2	design
Installed core system	2	design
Protective Systems		
Instrumentation system	2	manuf.
Control system	2	manuf.
Safety system	2	manuf.
Auxiliary Systems		
Pool coolant system	2	design
Water purification system	2	design
Room confinement components	2	const.
Area ventilation components	2	const.
Area radiation monitor system	2	manuf.
Air radiation monitor system	2	manuf.

- 1 - All sections of quality assurance program shall be considered applicable.
- 2 - Specific sections of quality assurance program should be applied as required to assure reliable performance.

The quality assurance program shall be commensurate with the TRIGA type reactor, The University of Texas administrative programs and the goals of quality assurance. This document provides requirements for establishing, managing, conducting and evaluating the QA Program. The QA Program applied to items or activities determined to be safety-related follows the guidelines of Reg. Guide 2.5 (77/05) [9,10].

10.6.1.1. Purpose

Quality assurance of certain activities associated with the University of Texas TRIGA reactor facility is important for the safe and efficient completion of tasks that are identified as safety-related. This document outlines the general elements of quality assurance applied to safety-related structures, systems or components, and activities. Requirements are documented for establishing, managing, conducting, and evaluating the QA Program. Although aspects of the QA Program may be routinely applied to many facility activities, the formal implementation of the program is limited to specific items or activities related to public health and safety. Table 10-1 lists the quality level and description of key systems and components.

10.6.1.2. Responsibility

The University of Texas at Austin as owner and operator of the TRIGA reactor facility, shall be responsible for a quality assurance program. The owner-operator shall establish and implement a program consistent with the goals of quality assurance for safety-related activities, structures, systems and components. Identification of safety-related items shall be the responsibility of the owner-operator and will include a description of the item and the applicable elements of the quality assurance program. Special quality provisions, delegated functions of the program, and unresolved quality assurance problems shall also be identified by the owner-operator. The facility supervisor shall have the ultimate responsibility for both the specifications of quality related requirements and the functions of quality related activities. Table 10-2 lists the responsibilities and key personnel participating in the University TRIGA QA Program.

10.6.1.3. Organization

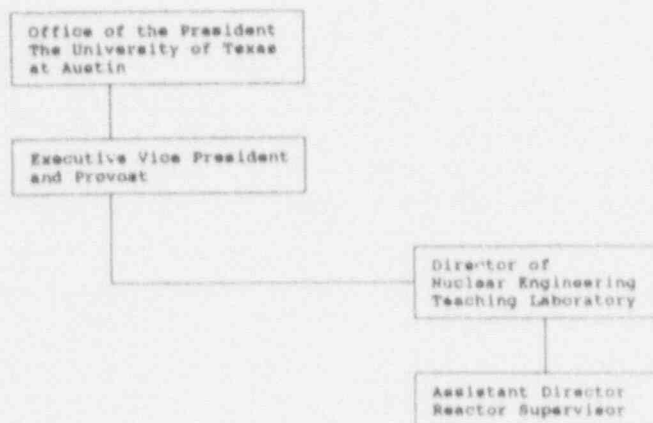
The organization applied to quality assurance activities shall be part of the normal university administrative structure. The facility Supervisor shall develop and implement the quality assurance program and identify safety related items. Unresolved issues of quality assurance shall be reported to the Director of the facility and the appropriate administrative vice president of the university. Execution of specific elements of the program may be delegated to persons in the University organization or other organizations as appropriate. University persons shall include committees, faculty, researchers or staff, as required for specific program applications. Non-university organizations or persons may supplement University personnel when specialized qualifications are necessary for specific quality assurance tasks. The University organization applied to reactor safety and quality assurance is the academic administration represented by Figure 10-2.

Table 10-2

RESPONSIBILITIES AND KEY PERSONNEL

<u>Responsibilities</u>	<u>Key University Personnel</u>
1. Establish program Implement program Identify Safety-related items	Director or Supervisor of TRIGA facility
2. Unresolved issues	President or Executive Vice President and Provost
3. Delegated functions	Faculty and staff
4. Specialized functions	Specified personnel

Quality Assurance



QUALITY ASSURANCE ORGANIZATION

Figure 10-2

10.6.1.4. Documentation

All activities affecting safety-related items subject to the quality assurance program shall be identified and documented formally. The format of Table 10-3 shall be used to identify applicable elements of the Quality Assurance Program and identify documents, procedures, reviews, inspections, tests, or other quality assurance features that are to be applied to a safety-related activity. The checklist or approval shall be incorporated in the table format for the acceptance of each specified quality assurance element by the facility supervisor.

10.6.2. Quality Assurance Controls

10.6.2.1. Design Controls

Design controls shall consist of design specifications, references to applicable codes, standards and regulations, design verifications and document approval. Applicable codes, standards, regulations or other quality requirements will be identified and requirements incorporated into the design documents. Design document approvals shall be part of the design document. Design approval will be by a person, other than the design originator, that is knowledgeable of the design criteria and is informed of the quality requirements.

Modifications of safety-related documents shall be subject to the same provisions as the original document. Approval of the design modification will be included with the design document and the modification identified.

Verification of design adequacy shall be provided by either design reviews, alternate calculation, test program or other method, determined to be appropriate. Verifications of the design shall check characteristics such as compatibility of materials; suitability of application of inspection, maintenance and repair; proper interfacing of sub-systems, and proper acceptance criteria. Method of verification will be identified and documented by approval of the design document.

10.6.2.2. Procurement Controls

Procurement controls shall consist of procurement specifications, references to applicable codes, standards and regulations, procurement acceptance and document approval. Applicable codes, standards, regulations or other quality requirements will be identified and references incorporated into the procurement documents. Procurement document approvals shall be part of the procurement document. Procurement approval will be by a person, other than procurement originator, that is knowledgeable of the procurement specifications and is informed of the quality requirements.

Changes to safety-related procurement documents shall be subject to the same provisions as the original document. Approval of procurement changes will be included with the procurement document and the change identified.

Table 10-3

FORMAT FOR SAFETY RELATED QA CHECKS

Each safety-related activity structure, system, or component will be given a letter symbol, such as A, B, C, and be appended with the following designations (for example, A1.0):

1.0 Title

Identification and description of safety-related item

1.1 Participation - supplemental organization and functions

1.2 Documents - applicable procedures or special measures

2.1 Design Control

2.1.1 Codes, standards and regulations

2.1.2 Method of verification

2.1.3 Modifications proposed

2.2 Procurement Control

2.2.1 Codes, standards and regulations

2.2.2 Quality assurance specifications

2.2.3 Proposed changes enacted

2.2.4 Procurement conformance method

2.3 Document Control

2.4 Material Control

2.4.1 Special procedures required

2.4.2 Equipment required

2.4.3 Personnel qualifications

2.5 Process Control

2.5.1 Special procedures

2.5.2 Special equipment

2.5.3 Personnel qualifications

3.1 Inspection Program Description

3.2 Test Program Description

3.3 Measurement Equipment

3.4 Nonconformance Item and Disposition

3.5 Corrective Actions Instituted

4.0 Records List

Acceptance of procured items or services shall consist of evidence provided by the contractor, evaluation of the procurement source, inspection at the source or inspection upon receipt. Acceptance of the procurement should require measures such as quality assurance by contractor, inspection and test functions, or controls on materials processes and nonconformances. The methods of acceptance will be identified and documented by approval of the procurement document.

10.6.2.3. Document Control

Document control consists of monitoring the development, revision, release and use of documents, drawings or specifications affecting safety-related activities. Document control shall include assurance that safety related documents are identified as such, and are completed and maintained properly. The laboratory Supervisor shall provide control of safety related documents that are specified according to the format of Table 10-2.

10.6.2.4. Material Control

Procedures shall be written to establish material control when special measures are necessary to assure material quality of safety-related items. Controls shall be applied to activities such as identification, handling, storage, shipping, cleaning and preservation. Procedures shall specify equipment and personnel required to accomplish the specified material control. Applicable codes, standards, specifications and personnel qualifications shall be documented.

10.6.2.5. Process Control

Procedures shall be written to establish process control when special measures are necessary to assure process quality of safety-related items. Controls shall be applied to activities such as crimping, soldering, welding, painting, cleaning and heat treating. Procedures shall specify qualifications of equipment and personnel required to perform the appropriate process control. Applicable codes, standards, specifications and personnel qualifications shall be documented.

10.6.3. Inspection and Corrective Actions

10.6.3.1. Inspection Program

An inspection program shall be established for safety related items or activities. The inspection program shall apply to construction, procurements, experiment equipment fabrication, and modifications that effect safety-related structures, systems, or components. Persons delegated to perform inspections shall not be the same person involved in the safety-related activity but may be from the same organization.

The inspection program will consist of written procedures that will include, as appropriate, procedures specifying characteristics to be inspected, acceptance criteria and inspection hold points.

Procedures should provide for identification of inspected and tested items. Provisions shall be made to clearly identify non conforming items from conforming items. In situations that inspections are not advantageous, a description shall be provided for monitoring actions.

Procedures shall be written for in service inspections of safety-related structures, systems or components.

10.6.3.2. Test Program

A test program shall be established for safety-related items or activities. The test program shall apply to prototype qualifications, installation proofs and functional tests. Testing shall be performed in accordance with acceptance criteria derived from design or procurement documents.

The test program will consist of written procedures that will include, as appropriate, procedures that specify acceptance criteria, monitoring requirements, equipment required, personnel qualifications, environmental conditions, data acquisition, and documentation of results.

10.6.3.3. Measuring and Test Equipment

Measurement tools, gages, instruments, and other measuring or test devices that measure critical parameters of safety-related items shall be identified. Provisions for identified measuring and test devices shall include availability, adjustment, calibration and accuracy as required for each application. Test equipment will be identified.

10.6.3.4. Non-Conforming Material and Parts

Non-conforming materials and parts associated with safety-related structures, systems or components shall be identified. The disposition such as acceptance, repair, rework or rejection of parts from safety-related functions will be determined by the person responsible for document control. Repair or reworked parts will be removed or labeled until accepted. Rejected parts will be removed and labeled. The disposition of non conforming materials will be documented.

10.6.3.5. Corrective Action

Documentation of specified quality control or assurance documents shall provide evidence of quality of safety-related items. Significant deviations from acceptable quality, repeated quality problems or unresolved quality issues shall be noted and reported in writing to administrative management personnel. It should be recognized that a determination of a quality problem may be subjective and should include evaluation of the documented quality requirements relative to the impact on the safety-related nature of the item.

10.6.3.6. Experimental Equipment

Design, construction, modification, inspection, testing and maintenance of experimental equipment shall be subject to this quality assurance program to the extent that these activities are safety-related.

10.6.3.7. Replacements, Modifications, or Changes

Insofar as possible, the replacement, modification, or change to structures, systems or components with a safety-related function shall be documented as meeting the requirements of the original structure, system or component. Evaluation should establish a performance and reliability equivalent or exceeding the original.

10.6.4. Records and Audits

10.6.4.1. Quality Assurance Records

Records that document quality of safety-related items or activities are identified according to Table 10-3. The records identified consist of inspection and test results, quality assurance reviews, quality assurance procedures and engineering analysis in support of design modifications or changes. The records shall be retained with as-built drawings, manuals and other records of important facility and system information. The retention period is to be the life of the facility or system for most, if not all, safety-related items. The retention period is indicative of the expectation that items which affect safety related to as TRIGA reactor are integrally related to the reactor, instrumentation and facility design and should persist for the system or facility life.

10.6.4.2. Audits

An audit shall be conducted to examine the records and function of the quality assurance program. Audits will occur within two years of the QA Program activities by designated persons that were not directly responsible for the audited functions. Written procedures, Table 10-4, for the audit will be considered part of the Quality Assurance Program. A report of the audit results, actions to resolve deficiencies and evaluation of the program will be made to a facility operations committee and university administrative management, and maintained with other Quality Assurance Program documents.

Table 10-4

QUALITY ASSURANCE PROGRAM
AUDIT PROCEDURES

-
1. Designate a person or persons responsible to perform the program audit.
 2. Determine the date of the previous audit.
 3. Review the Quality Assurance Program document.
 4. Examine the list of safety-related items.
 5. Note additions to the safety-related items.
 6. Identify records applicable to additional items.
 7. Determine the location of all indicated records.
 8. Review records for abnormalities and completeness.
 9. Prepare statement that evaluates functions of Quality Assurance Program.
 10. Report findings of audit and program functions to operations committee and management.
-

10.7. STARTUP PROGRAM

Startup and testing of the Balcones Research Center TRIGA facility shall be performed by personnel of The University of Texas with consultation of the reactor manufacturer, General Atomics (GA Technologies). The University of Texas has accumulated nearly 25 years operation experience with a TRIGA reactor prior to the new facility proposal. More than twenty TRIGA type reactors, eleven in the U.S., with power levels of one megawatt or more have been produced by General Atomics (GA Technologies).

Training of university personnel associated with startup activities at the new facility will consist of the relicensing of at least two operators from the current facility that have certified senior operator permits. Training of an additional operator or retraining of a current operator by the manufacturer should occur to provide an effective transfer of the manufacturer's experience to the owner-operator. One or more of the certified operators will have a bachelor's or advanced degree in a field of engineering.

A checkout and evaluation plan for the instrumentation, control, and safety system will assure a complete test of the system. Acceptance of the instrumentation, control, and safety system will also depend on the completion of test and acceptance programs at the site of the manufacturer. This test and acceptance at both sites is necessary for proper verification of installation, in this case of substantial technological changes in the system design. A substantial change of the system design from analog to digital provides for numerous improvements in performance, but will also be subject to design limitations that are characteristic of such changes.

The startup program is to consist of five phases, beginning with the storage of nuclear fuel on site to the reporting of observed reactor parameters. At each phase written procedures, check lists and other documents shall be developed for activities or measurements that will have significant importance to safety or operation. Documentation shall include information required by the various programs to be implemented at the facility, such as operator qualifications, radiological protection, fire protection, and quality assurance, plus operating procedures and other requirements of license authorizations. The startup program is to be divided into the following phases:

- a. Storage of fuel and acquisition of components,
- b. Tests of systems before core loading,
- c. Fuel loading and core criticality,
- d. Tests subsequent to core criticality and
- e. Acceptance of core operation.

10.7.1. Storage of Fuel and Acquisition of Components

Provisions for the storage fuel and components for the reactor facility at the completion of the facility construction shall require the limited implementation of administrative controls. A license authorization for the possession and storage of special nuclear materials and other radioactive components, source materials or by-product materials, will be obtained and materials relocated to the facility. Storage of non-radioactive components, storage of other reactor components and instrumentation, and assembly of facility systems will be performed in the initial startup phase.

10.7.2. Tests of Systems Before Core Loading

Facility systems, auxiliary systems, and reactor systems or physical parameters shall be tested for the appropriate operating conditions prior to fuel transfer into the reactor core. Fuel may be loaded into the pool during this phase. Systems shall be tested according to designated specifications, when applicable, and acceptable operation shall be established before core loading proceeds. Facility systems to be tested should include security, fire, communication, and ventilation systems. Auxiliary systems to be tested should include radiation monitoring, pool coolant, alarm, and interlock systems. Reactor systems to be tested will include the instrument and control system, and verification of physical specifications for assembly, and operation of reactor components. Some systems or components that do not meet specifications and are not required for operation may be deferred for acceptance to a later startup program phase.

10.7.3. Core Load For Initial Criticality

Continuous operation of coolant system, insertion of the neutron source, installation of the cobalt-60 irradiator, and movement of fuel into the core will begin the core load startup program phase. Certain verifications of instrumentation and control system functions will be completed before initialization of an approach to critical experiment by standard reciprocal source multiplication factor measurements. Rod worth values shall be estimated from the core loading procedures.

10.7.4. Tests Subsequent to Core Criticality

Rod calibration shall be determined by positive period measurements before reactor operation at power levels affected by the power coefficient. Next, an intermediate power calibration shall be made, along with an evaluation of the fuel temperature as measured by an instrumented fuel element. Last, the fuel loading of the core should be adjusted for full power operation and operation of the cooling system verified at power. Any variation of core parameters significantly different than predicted by calculations or experience shall be resolved during this startup program phase.

10.7.5. Acceptance for Operation

The final startup program phase shall consist of the resolution of all deviations from specifications. Deviations should be resolved as specified for quality assurance or other methods determined to be acceptable. Three months after completion of requisite initial startup and power-escalation testing of the reactor, or nine months after initial criticality, a written report shall be submitted to licensing authorities. The report shall include a summary of the following:

- a. Description of measured values of operating conditions or characteristics obtained and comparison of these values with design predictions or specifications.
- b. Description of major corrective actions taken to obtain satisfactory operation.
- c. Re-evaluation of safety analysis where measured values indicate substantial variance from those values used in the Safety Analysis Report.

Results of the startup program shall become a supplement to The University of Texas TRIGA Safety Analysis Report. Chapter 12 of the report will contain results of the startup program.

Chapter 10 References

1. "Standard for Administrative Controls" ANSI/ANS - 15.18 1979.
2. "Selection and Training of Personnel for Research Reactors", ANSI/ANS - 15.4 - 1970 (N380).
3. "Review of Experiments for Research Reactors", ANSI/ANS 15.6-1974 (N401).
4. "Records and Reports for Research Reactors", ANSI/ANS - 15.3-1974 (N399).
5. "Radiological Control at Research Reactor Facilities", ANSI/ANS-15.11 1977 (N628)
6. "Design Objectives for and Monitoring of Systems Controlling Research Reactor Effluents", ANSI/ANS - 15.12 1977 (N647)
7. "Standard for Fire Protection Program Criteria for Research Reactors", ANSI/ANS - 15.17 (1981).
8. Nuclear Research Reactors 1983, National Fire Protection Association, Inc., NFPA 802.
9. "Quality Assurance Requirements for Research Reactors", Nuclear Regulatory Guide 2.5 (77/05).
10. "Quality Assurance Program Requirements for Research Reactors", ANSI/ANS - 15.8 - 1976 (N402).
11. "Nuclear Regulatory Commission", Chapter 10 U.S. Code of Federal Regulations

Chapter 11

SAFETY ANALYSIS

In this section an analysis of abnormal operating conditions will be made with conclusions concerning the effects on safety to the reactor, the public, and the operations personnel, as a consequence of any abnormal operations.

The abnormal conditions that will be analyzed are:

- a. Reactivity accident.
- b. Loss of reactor coolant.
- c. Fission product release from clad rupture.

11.1. REACTIVITY ACCIDENT

11.1.1. Summary

Rapid insertion of reactivity into a TRIGA reactor is a designed feature of the fuel performance [1]. Thus, most plausible reactivity accidents do not subject the fuel to conditions more severe than normal operating situations. Postulated accidents for other undetermined scenarios also are predicted not to exceed fuel element safety conditions.

The standard TRIGA fuel element of U-ZrH (H/Zr: 1.6) is composed of a stable gamma phase ZrH that does not undergo a phase transition at temperatures less than about 1250°C [2]. Pulsing limits for fuel elements clad in stainless steel are set by the hydrogen equilibrium pressure within the fuel element. This pressure is a function of temperature and must not exceed the rupture stress of the fuel element cladding. For the stainless steel cladding (0.02 inch thick), the rupture pressure has been measured to be 1800 psi at 100°C. The fuel temperature at which the equilibrium hydrogen pressure will be 1800 psi is about 1150°C. The average and peak fuel temperatures at 1.5 Mw steady-state operation are about 220°C and 400°C, occurring well below the limit. The average and peak fuel temperatures occurring after a 2.8% reactivity insertion at respective power levels of 1 kW and 880 kW are also less than 1150°C. Values of 376°C and 843°C are predicted for a low power insertion and values of 460°C and 795°C are calculated for a high power insertion.

Two reactivity accident scenarios are presented. The first is the insertion of 2.8% reactivity at zero power by the sudden removal of the maximum worth control rod. The second is the sudden removal of the same 2.8% reactivity with the reactor operating at a power level equivalent to the balance of the core excess reactivity. It is unlikely that movement of reactor fuel or experiments would lead to the postulated accidents. Movements of control rods for the first case are controlled administratively while movements of control rods for the second case are prevented by control circuit design.

The analysis of a four dollar (2.8% $\delta k/k$) pulse insertion also provides information about accidents with experiment systems. Provided the total worth of reactor experiments are limited to \$3.00 no experiment movement could generate the postulated accident.

Pulse powers predicted from kinetics formulations based on the Fuchs-Nordheim-Scalletar model are displayed in Figure 11-1. Pulse shape, energy and temperature for \$3 and \$4 pulse insertions are shown.

11.1.2. Analysis of 2.8% Insertion at 1 kW.

A rapid insertion of excess reactivity in the reactor system is postulated. The method of inserting this reactivity is through the rapid removal of a control rod. This reactivity insertion is the most serious that could occur. It is also the normal pulsing condition and the analysis is presented here as a point of information since it is not actually an accident condition.

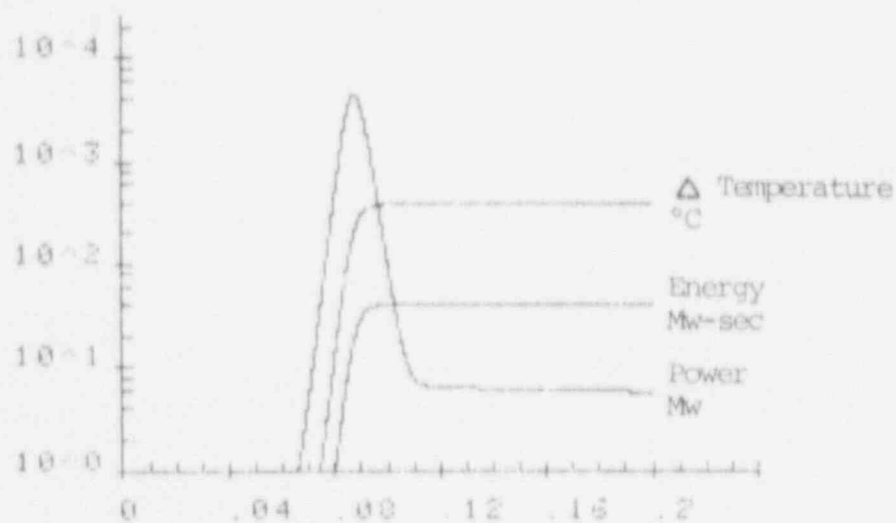
The sequence of events leading to the postulated reactivity accident is:

- a. The reactor is just critical at a low power level (less than 1 kW).
- b. Upward force is applied to a high worth control rod causing it to be ejected from the core and to introduce an excess reactivity of \$4.00.

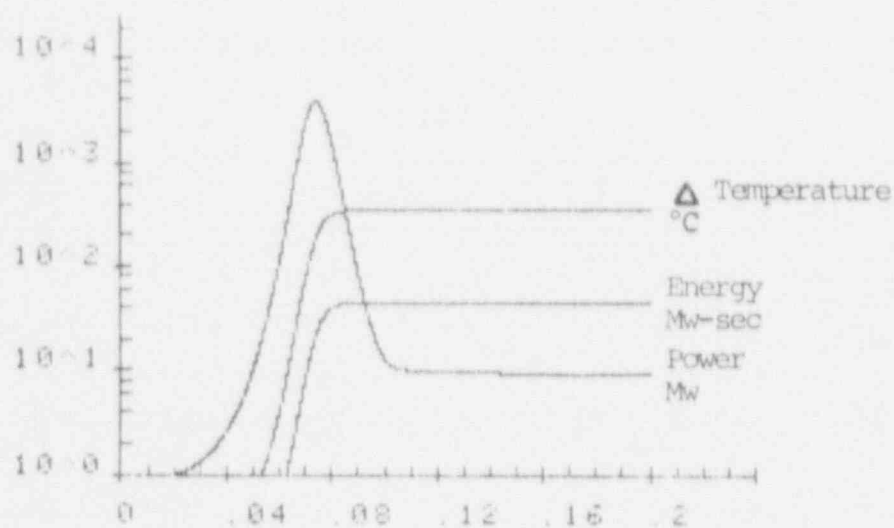
The consequences of the above sequence of events are:

- a. Reactor power is increased to a maximum power of approximately 4220 MW.
- b. A maximum energy release of approximately 36 MW-sec is reached when the maximum fuel temperature of 843°C is reached.
- c. Stresses are predicted in the stainless steel cladding of approximately 2940 psi. These pressures are caused by expansion of the air and fission product gases and the hydrogen release from the fuel material. Neither of the preceding stress values will cause cladding rupture.

The analysis of this accident is conservative in a number of ways, some of which have been indicated in the reactor design bases (Chapter 4). For example, the equilibrium pressure of hydrogen over the fuel is not achieved during a pulse or step insertion of reactivity.



\$4 Pulse
Initial Power 1 Kw



\$4 Pulse
Initial Power 880 Kw

CALCULATED PULSE SHAPE, ENERGY
AND TEMPERATURE

Figure 11-1

It was assumed that the reactor is just critical at a low power level with a fuel and coolant temperature of 25°C. Additional input parameters are summarized in Table 11.1.

Table 11-1

REACTIVITY TRANSIENT INPUT PARAMETERS

Reactivity insertion, $\$$	4.0
Temperature coefficient, prompt $(\delta k/k)/^{\circ}\text{C}$	-1.1×10^{-4}
Delayed neutron fraction β , %	0.70
Neutron lifetime ℓ , μsec	41
Heat capacity C_p , watt-sec/element	$817 + 1.6 T_{\text{fuel}}$

The computations leading to these conclusions are determined by the following lumped parameter analysis. The Fuchs-Nordheim model for reactor dynamics yields the coupled set of differential equations:

$$\ell \delta P / \delta t = (\delta k - \alpha T) P \quad (1)$$

$$C \delta T / \delta t = P - P_0 \quad (2)$$

$$\text{with } C = C_0 + C_1 T \quad (3)$$

where ℓ = Prompt neutron lifetime, sec.

P = Power level, (P_0 = initial power), watts.

δk = Reactivity above prompt critical, $\delta k/k - \beta$.

α = Magnitude of the negative temperature coefficient, $^{\circ}\text{C}^{-1}$.

T = Temperature (avg. over fuel) above the equilibrium temperature at P_0 , $^{\circ}\text{C}$.

C = Heat capacity of the fuel in the core, W-sec/ $^{\circ}\text{C}$.

C_0 = Heat capacity at the equilibrium temperature corresponding to P_0 , W-sec/ $^{\circ}\text{C}$.

C_1 = Rate of change of heat capacity with temperature, W-sec/ $^{\circ}\text{C}^2$.

The above lumped parameter system neglects heat transfer and delayed neutron effects and averages space and neutron energy variations so that all coefficients are assumed constant. Combining equations

$$\frac{dP}{dT} = \frac{(\delta k - \alpha T) (C_0 + C_1 T) P}{\ell (P - P_0)} \quad (4)$$

Integrating, using the condition that $T = 0$ when $P = P_0$, yields

$$\ell \{ (P - P_0) - P_0 \ln(P/P_0) \} = T \{ \delta k C_0 - (\alpha C_0 + C_1 \delta k) T/2 - \alpha C_1 T^2/3 \} \quad (5)$$

Maximum (or minimum) temperatures occur when the pulse initiates and after culmination of the pulse such that

$$P - P_0 = 0,$$

$$P = P_0,$$

and then,

$$T \{ \delta k C_0 - (\alpha C_0 + C_1 \delta k) T/2 - \alpha C_1 T^2/3 \} = 0 \quad (6)$$

The roots of this equation are

$$T_a = -3/8 (\sigma - 1) \{ 1 + [1 + 16/3 \sigma / (\sigma - 1)^2]^{1/2} \} T_f \quad (7)$$

where $\sigma = \alpha C_0 / \delta k C_1$,

$$T_f = 2 \delta k / \alpha,$$

and the positive sign is taken for the square root if $\sigma \leq 1$.

From the fuel heat capacity the core heat capacity with 90 elements is

$$\begin{aligned} C_0 &= 817 + 1.6 T_g \text{ W-sec/}^\circ\text{C-element} \\ &= 857 \text{ W-sec/}^\circ\text{C-element} \times 90 \text{ elements/core} \\ &= 7.71 \times 10^4 \text{ W-sec/}^\circ\text{C.} \end{aligned} \quad (8)$$

For the insertion of \$4.00 of reactivity a value of δk equal to $0.021 = (\$4.00 - \$1.00) (0.7\%/\$)$.

$$\sigma = \frac{1.1 \times 10^{-4}}{2.1 \times 10^{-2}} (857/1.6) = 2.81 \quad (9)$$

$$\text{and } T_f = 2 (2.1 \times 10^{-2}) / 1.1 \times 10^{-4} = 382^\circ\text{C} \quad (10)$$

Thus

$$\begin{aligned}
 T_a &= -3/8 (2.81 - 1) \\
 &\quad (1 - [1 + 16/3 \cdot 2.81/(2.81 - 1)^2]^{1/2}) T_f \\
 &= 0.92 T_f = 351^\circ\text{C}
 \end{aligned}
 \tag{11}$$

Therefore, at the conclusion of the pulse the average fuel temperature will be

$$T_{ss} = 352 + 25 = 376^\circ\text{C} \tag{12}$$

To determine the maximum temperature in the hottest fuel element, the average energy release is determined and then multiplied by the peak power ratio to obtain the maximum energy release in the center element. The peak power ratio includes radial, axial and element peaking factors. Then one returns to the energy versus temperature equation to determine the maximum temperature.

Let E equal the energy necessary to raise the average core temperature from the temperature at the initial power level to the temperature at the final power level. Then

$$E = \int_0^{T_a} C \, dT = \int_0^{T_a} [C_0 + C_1 T] \, dT \tag{13}$$

$$E = C_0 T_a + C_1 T_a^2/2 \tag{14}$$

$$E = 3.57 \times 10^7 \text{ watt-sec}$$

For a peak-to-average power ratio of 2.2 and an element peaking factor of 1.4 the energy release of the element producing the peak power is $3.1E$ or 110 Mw-sec . A peak temperature is calculated by substituting this energy into the previous equation and solving for the temperature.

$$T_p = -C_0/C_1 + [(C_0/C_1)^2 + 2(3.1E/C_1)]^{1/2} \tag{15}$$

$$T_p = 818^\circ\text{C}$$

with

$$T_{ss} = 818 + 25 = 843^\circ\text{C}$$

During the time of peak fuel temperature, the stress on the clad from the pressure produced by the expansion of air and fission product gases and the hydrogen released from the fuel is less than the strength of the clad material and therefore there is no loss of clad integrity.

The partial pressure exerted by gases is

$$P_t = N_t RT/V \tag{16}$$

where initially the volume, V , is taken as a 1/8-in. space between the fuel and reflector end piece. This result is conservative since the porosity of the graphite reflector of 20% is neglected.

The volume then is

$$V = \pi r^2 h = \pi (1.80)^2 0.317 = 3.23 \text{ cm}^3 \quad (17)$$

The partial pressure of the air in the element is

$$P_{\text{air}} = \frac{RT}{2.24} \times 10^3 = 4.46 \times 10^{-5} RT \quad (18)$$

Calculation of the fission product gases in a fuel element is determined by burnup. For an element operated at three times the 4.5 MW-days discussed in Section 11.2, a total of 0.016 moles of stable and radioactive gases are produced. If the release fraction is taken as .0015% as discussed in Section 11.3, then

$$N_{\text{fp}} = \frac{3.22 \times 10^{21}}{6.02 \times 10^{23}} (1.5 \times 10^{-5}) = 2.4 \times 10^{-7} \text{ moles.} \quad (19)$$

From this, one obtains,

$$P_{\text{fp}} = 7.45 \times 10^{-8} RT \quad (20)$$

The total pressure exerted by the air and fission products is

$$P_1 = (4.46 + 0.007) \times 10^{-5} RT = 4.47 \times 10^{-5} RT \quad (21)$$

$$P_1 = P_{\text{air}}$$

Also we have

$$P_{\text{air}} = 14.7 T/273 \text{ psi} \quad (22)$$

As an upper limit, assuming an air temperature equal to the peak fuel temperature of 843°C or 1116°K, one obtains

$$P_1 = (14.7) 1116/273 = 60.1 \text{ psi} \quad (23)$$

The equilibrium hydrogen pressure over ZrH (H/Zr; 1.6) at 843°C is 20 psi. The total internal pressure then is

$$P_t = P_h + P_1 = 80 \text{ psi} \quad (24)$$

Assuming no expansion of the clad, the stress produced in the clad by this pressure is

$$S = r P_t / t = 0.735 P_t / 0.020 = 36.75 P_t \quad (25)$$

$$= (36.75) (80) = 2940 \text{ psi.}$$

For a reactivity insertion of \$4.00, the clad surface temperature would be approximately equal to the saturation temperature of the water which is 113°C at a pressure of 23.4 psia. At this temperature, the ultimate tensile strength for type 304 stainless steel is greater than 60,000 psi with a yield stress of approximately 36,000 psi. Comparing this strength with the stress applied to the cladding during the reactivity insertion, it is seen that the strength of the material far exceeds the stress which would be produced. Therefore there would be no loss of clad integrity or damage to the fuel as a result of the reactivity accident.

11.1.3. Analysis of 2.8% Insertion at 880 kW.

The reactivity accident considered here would take place in the following manner. Initially, the reactor is cold clean with all control rods inserted. The reactor is loaded with 4.9% $\delta k/k$ excess reactivity and the pool coolant is at a temperature of 42°C. This accident requires someone deliberately violating the operating license and several interlocks and scrams.

The sequence of events leading to the postulated reactivity accident is:

- a. The operator slowly withdraws all the control rods except the maximum-worth rod, until all the rods are completely out and the reactor is operating at a high steady state power.
- b. Upward force is applied to the maximum worth rod ejecting it (by some means impossible to conceive) from the hot operating reactor.

The consequences of the above sequence of events are:

- a. Reactor power and fuel temperatures are increased by the compensated reactivity of \$3.00 (that is $4.9\% - 2.8\% = 2.1\% = \3.00) to levels of 880 kW with fuel temperatures of 380°C, peak, and 207°C, average.
- b. A prompt insertion of 2.8% $\delta k/k$ results in an average temperature in the core of 460°C and a peak temperature of 795°C.
- c. Stresses are predicted in the clad of about 2100 psi. Even if the clad were at the maximum fuel temperature this stress is a factor of ten below the ultimate strength of the clad.

The analysis of this accident is conservative as described in the previous accident case. The values do not exceed equilibrium element conditions and pressure. Calculations include finite reactivity insertion time, delayed neutrons and heat transfer.

It is assumed that the reactor power level is 880 kW (\$3.00 of power coefficient); the average fuel temperature is $(165 + 42)^{\circ}\text{C}$; the peak fuel temperature is $(338 + 42)^{\circ}\text{C}$; with a pool coolant temperature of 42°C assumed. Values of the input parameters are summarized in Table 11-2.

Table 11-2

REACTIVITY TRANSIENT INPUT PARAMETERS

Reactivity insertion, $\$$	4.0
Temperature coefficient,	
prompt $(\delta k/k)/^{\circ}\text{C}$	-1.1×10^{-4}
delayed neutron fraction β , %	0.70
neutron lifetime l , μsec	41
Heat capacity, watt-sec/element	
fuel C_p , at 0°C	817
fuel C_p , temperature dependence	$1.6 T_{\text{fuel}}$
water C_p , at 25°C	879
Thermal resistance, $^{\circ}\text{C}/\text{MW}$:	
fuel to cooling channel	5.29×10^4
cooling to pool	1.42×10^3

A computer program was used to calculate the energy release in the transient. The program is a one-dimensional combined reactor kinetics heat transfer program that works extremely well for reactor transients in which detailed heat transfer analysis is not required. Delayed neutrons and finite reactivity insertion time are included in the program.

Using the parameters given above it was found that the addition of \$4.00 reactivity ($2.8\% \delta k/k$) from an average fuel temperature of 207°C (the average fuel temperature at 880 kW) produced an energy release of 32 Mw-sec in the 90 element core.

The energy density at the axial midplane of the maximum power density element, E_m , is:

$$E_m = 1.1 P_r P_a E/n$$

where P_r = relative power in element,

P_a = axial peaking factor,

E = energy release in the transient,

n = number of elements,

and the factor of 1.1 accounts for uncertainties.

In Chapter 4 the radial power distribution within a fuel element is shown. The energy deposited at radius r per unit volume is:

$$E_r = f(r) E_m$$

where $f(r)$ is taken from data in Chapter 4. The after pulse temperature at the radial distance r is given by:

$$T_p(r) = -C_0/C_1 + ((C_0/C_1)^2 + 2E_r/C_1)^{1/2}$$

where

$$C_0 = 8.17 \times 10^{-4} \text{ Mw-sec/}^\circ\text{C}$$

and

$$C_1 = 1.6 \times 10^{-6} \text{ Mw-sec/} (^\circ\text{C})^2.$$

Calculation of the final element fuel temperature is accomplished by adding the temperature of the pulse deposited energy to the fuel element temperature prior to the pulse.

The radial temperature distribution in the fuel prior to the initiation of the transient is given by:

$$T(r) \cdot T_s = (T_m - T_s) [1 - r' - r' \ln(1/r')] / r_0'$$

where $T(r)$ = temperature at radial distance from center r ,

T_s = temperature at fuel surface at axial center, $^\circ\text{C}$

T_m = maximum temperature in element, $^\circ\text{C}$

r, r_0 = radial position and radius of fuel, cm,

$$r' = (r/r_0)^2$$

$$r_0' = [1 - r'' - r'' \ln(1/r'')]$$

$$r'' = (r_1/r_0)^2$$

and

r_1 = inside fuel radius, cm.

In Figure 11-2 there is shown the before pulse and after pulse temperature in the axial midplane of the maximum power density element. As can be seen the maximum temperature occurs at the periphery of the fuel. The adiabatic value is 795°C . In Chapter 4 a plot is shown of pulse temperature distribution as a function of time. This figure shows the typical dependence of temperature as heat flows quickly toward the fuel center and toward the clad.

For a fuel temperature of 800°C the equilibrium hydrogen pressure over the fuel would be less than 15 psi and the pressure exerted by air within the element would be less than 60 psi even if it were at the maximum fuel temperature. The stress imposed on the clad by 75 psi would be about 2800 psi. The ultimate strength of the clad is over 20,000 psi at 800°C. Therefore one can conclude that the clad integrity would not be compromised as a consequence of either of these events.

A similar analysis was made in which the reactor was assumed to be operating at 1.5 Mw. In this case only \$2.79 (1.95% $\delta k/k$) was available to be inserted. The peak before pulse temperature was 475°C and the reactivity insertion resulted in an energy release of almost 30 Mw-sec. The peak after pulse adiabatic temperature was 823°C which occurs at the inner fuel radius because of the high initial temperature at that point.

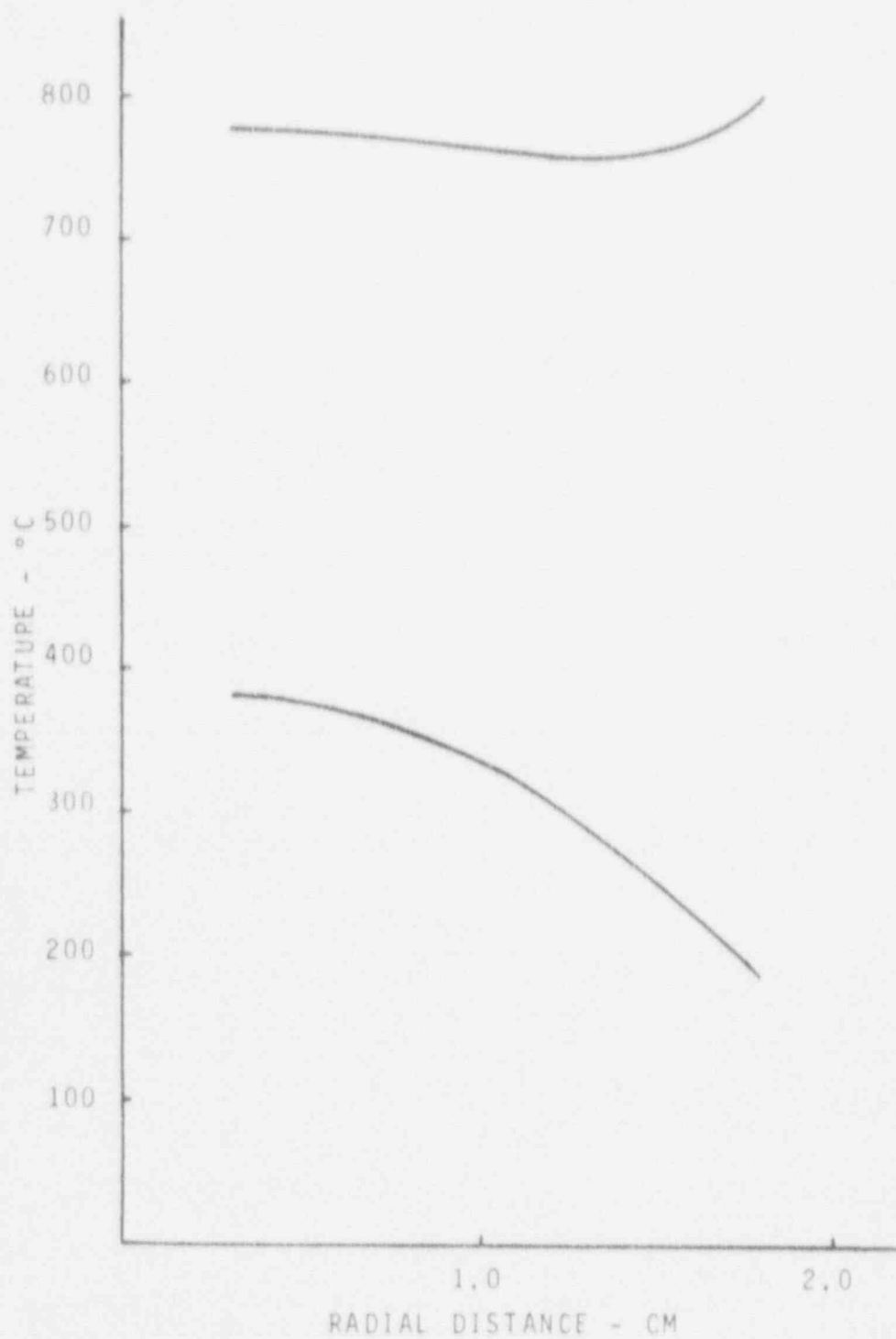
11.2. LOSS OF REACTOR COOLANT

11.2.1. Summary

The reactor will operate at a calculated maximum power density of 18 kW/element when the reactor power is 1000 kW and there are 90 elements in the core, all of which are standard TRIGA fuel. If the coolant is lost immediately after reactor shutdown, the fuel temperature, indicated in Figure 11-3, will rise to a maximum value of 750°C. The stress imposed on the fuel element clad by the internal gas pressure, presented in Figure 11-4 is about 2300 psi when the fuel and clad temperature is 750°C and the yield stress for the clad is about 19,500 psi. Therefore, it can be concluded that the postulated loss-of-coolant accident will not result in any damage to the fuel, will not result in release of fission products to the environment, and will not require emergency cooling.

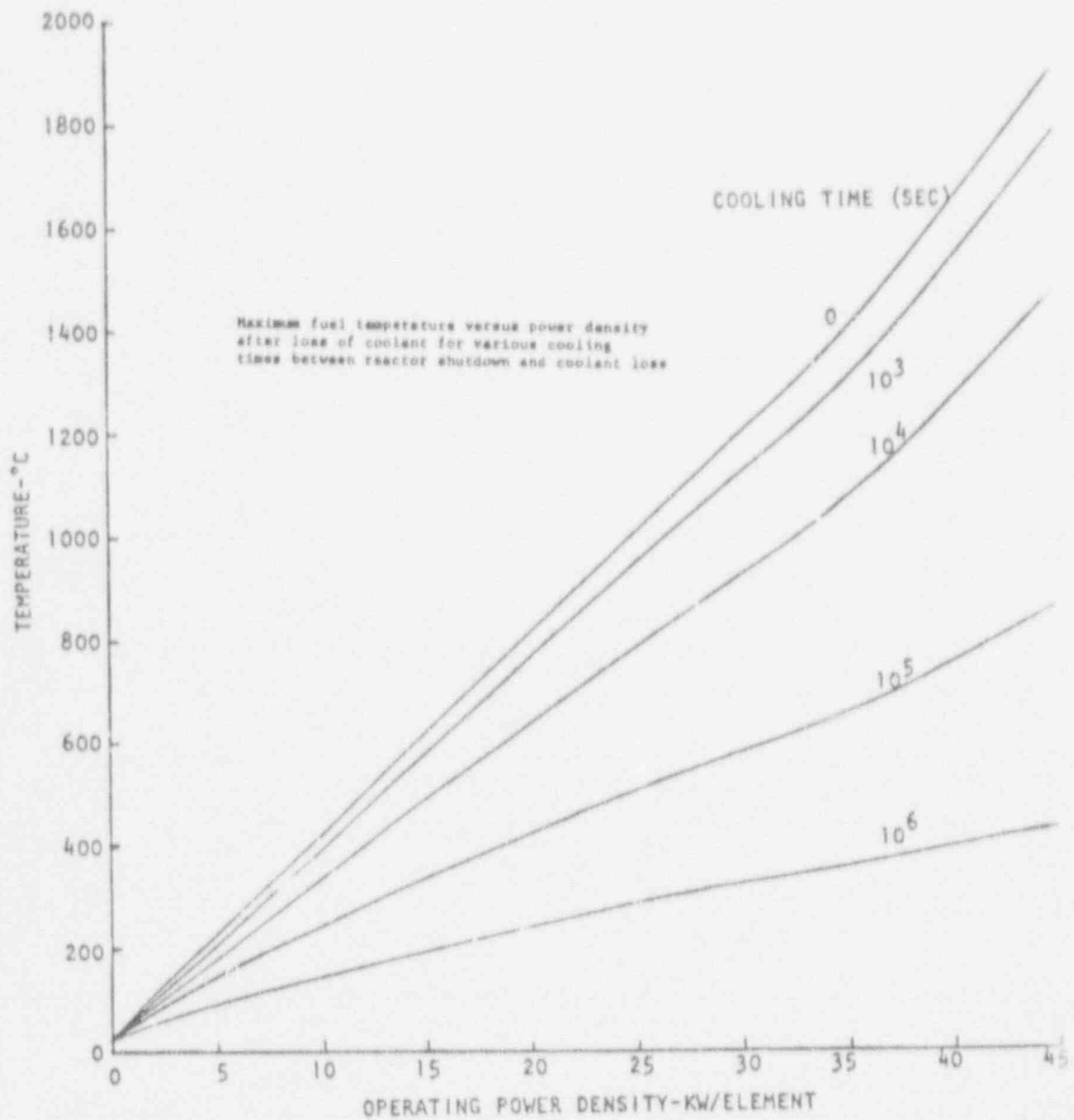
If the reactor tank is drained of water, the fission product decay heat will be removed through the natural convective flow of air up through the reactor core. If the decay-heat production is sufficiently low because of a low fission product inventory or a long interval between reactor shutdown and coolant loss, the flow of air will be enough to maintain the fuel at a temperature at which the fuel elements are undamaged. The following analysis shows that:

- a. The maximum temperature to which the fuel can increase is 900°C without substantial yielding of the clad or subsequent release of fission products.
- b. This temperature will never be exceeded under any conditions of coolant loss if the maximum operating power density is 22 kW/element or less.
- c. For maximum operating power densities greater than 22 kW/element, emergency cooling can be provided to ensure that the fuel element temperature does not exceed 900°C.



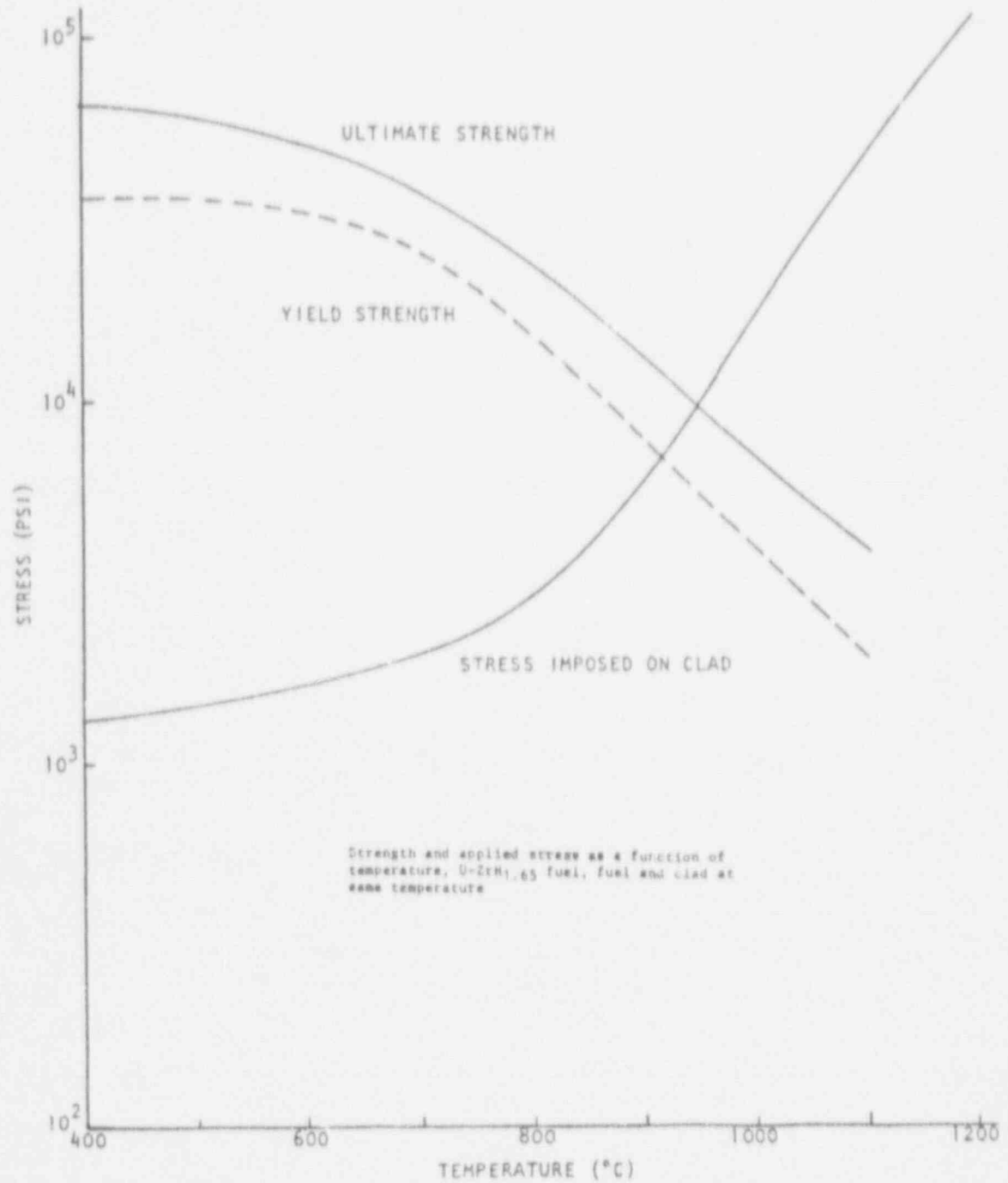
FUEL TEMPERATURE DISTRIBUTION
BEFORE AND AFTER PULSE

Figure 11-2



FUEL TEMPERATURE AND POWER DENSITY
FOR ELEMENT COOLING TIMES

Figure 11-3



U-ZRH (1.6) STRENGTH AND STRESS
VERSUS TEMPERATURE

Figure 11-4

If power densities exceed the limit for immediate coolant loss, emergency cooling will depend on the power density. The required emergency cooling time as a function of maximum operating power density is shown in Figure 11-5. Maximum operating power density is to be used to determine emergency cooling times.

11.2.2. Fuel Temperature and Clad Integrity

The strength of the fuel element clad is a function of its temperature. The stress imposed on the clad is a function of the fuel temperature as well as the hydrogen-to-zirconium ratio, the fuel burnup, and the free gas volume within the element. In the analysis of the stresses imposed on the clad and strength of the clad the following assumptions will be made:

- a. The fuel and clad are at the same temperature.
- b. The hydrogen-to-zirconium ratio is 1.65.
- c. The free volume within the element is represented by a space 1/8-in. high within the clad.
- d. The reactor contains fuel that has experienced burnup equivalent to only about 4.5 MW-days.

The fuel element internal pressure p is given by

$$P = P_h + P_{fp} + P_{air}$$

where

P_h = hydrogen pressure,

P_{fp} = pressure exerted by volatile fission products,

P_{air} = pressure exerted by trapped air.

For hydrogen-to-zirconium ratios greater than about 1.58 the equilibrium hydrogen pressure can be approximated by

$$P_h = \exp [1.767 + 10.3014x - 19740.37/(T_K)] \quad (26)$$

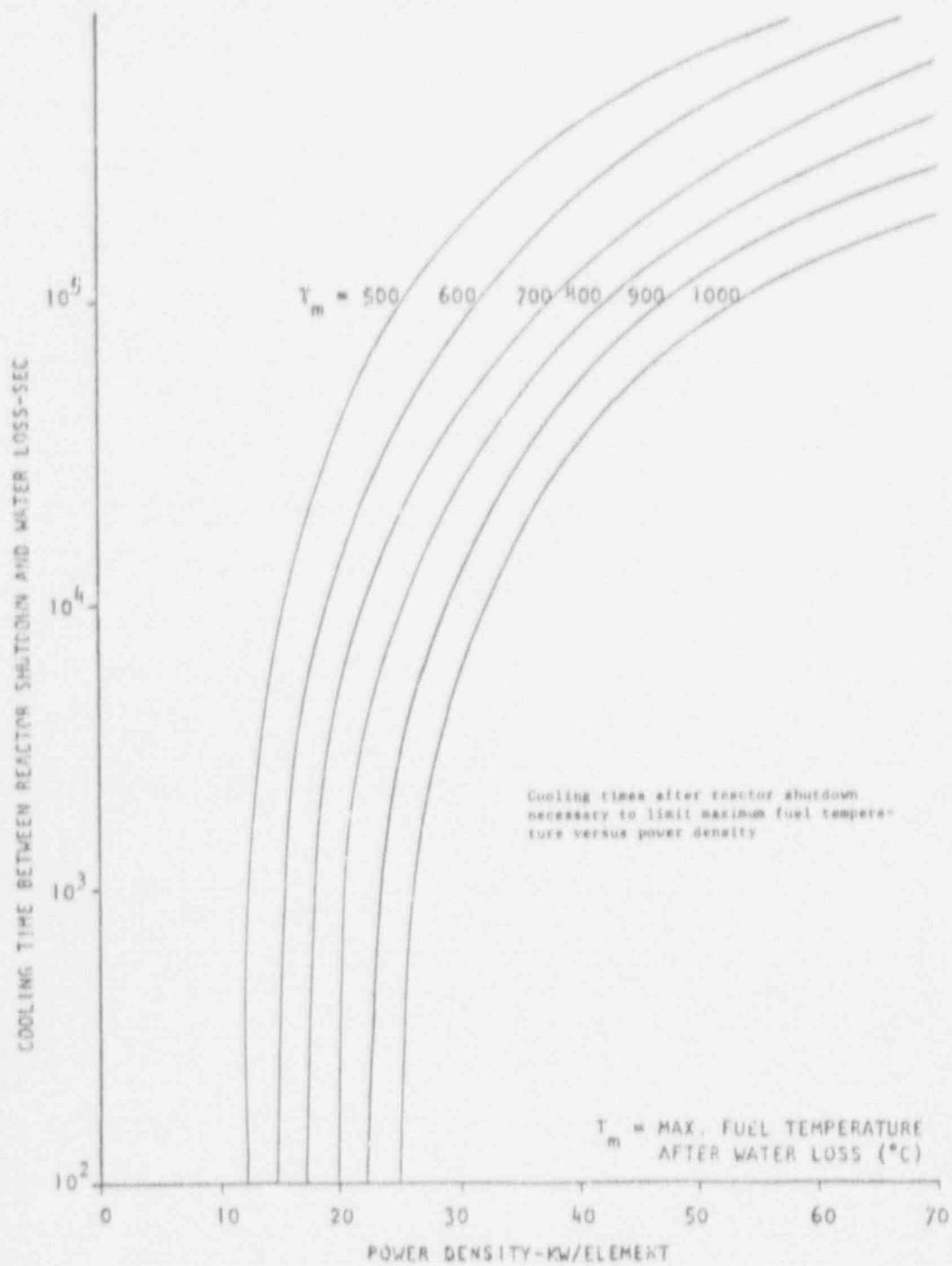
(atmospheres),

where x = ratio of hydrogen atoms to zirconium atoms, and

T_K = fuel temperature ($^{\circ}K$).

This expression was derived from least-square fits to the data of Dee and Simnad [3]. For ZrH (H/Zr; 1.65) the hydrogen pressure becomes

$$P_h = 1.410 \times 10^8 \exp [-19740.37/(T_K)] \text{ (atmospheres).}$$



COOLING TIMES AFTER REACTOR SHUTDOWN
TO LIMIT MAXIMUM FUEL TEMPERATURE VERSUS POWER DENSITY

Figure 11-5

The pressure exerted by the fission product gases is given by

$$P_{fp} = f \frac{n R}{E V} T_K E, \quad (27)$$

where f = fission product release fraction,

n/E = number of moles of gas evolved per unit of energy produced, moles/MW-day,

R = gas constant, 8.206×10^{-2} liters-atmospheres/mole-°K,

V = free volume occupied by the gases, liters, and

E = total energy produced in the element, MW-day.

The fission product release fraction [4] is given by

$$f = \int_n f_n dn \quad (28)$$

$$f_n = \{1.5 \times 10^{-5} + 3.6 \times 10^3 \exp [-1.34 \times 10^4/T_n]\}$$

where

T_n = fuel temperature in the differential volume of the element during normal operation, °K,

f_n = differential release fraction and

n = fuel volume normalized to 1.

The fission product gas production rate n/E is not independent of power density (neutron flux) but varies slightly with the power density. The value $n/E = 0.00119$ moles/MW-day is accurate to within a few percent over the range from, a few kilowatts per element to well over 40 kW/element. The free volume occupied by the gases is assumed to be a space 1/8-in. (0.3175-cm) high at the top of the fuel so that

$$V = 0.3175 \pi r_1^2, \quad (29)$$

where r_1 = inside radius of the clad (1.822 cm).

For standard TRIGA fuel the maximum burnup is about 4.5 MW-days/element. Pressure exerted by fission product gases is not significant.

The air trapped within the fuel element clad would exert a pressure

$$P_{air} = RT_K/22.4, \quad (30)$$

where it is assumed that the initial specific volume of the air (22.4 liters/moles) is present at the time of the loss of coolant. Actually, the air forms oxides and nitrides with the zirconium so that after relatively short operation the air is no longer present in the free volume inside the fuel element clad.

For ZrH (H/Zr: 1.6) fuel burned up to 4.5 MW-days/element, with a maximum operating temperature of 600°C, the internal pressure as a function of maximum fuel temperature T_K is

$$P = 1.410 \times 10^8 \exp(-19740.37/T_K) + 3.66 \times 10^{-3} T_K \quad (31)$$

(atmosphere)

or

$$P = 2.073 \times 10^9 \exp(-19740.37/T_K) + 5.38 \times 10^{-2} T_K \quad (\text{psi}).$$

The stress imposed on the clad by the gases within the free volume inside the clad is

$$S = (r_c/t) p \quad (32)$$

where r_c = clad outside radius (1.873 cm),

t = clad thickness (0.051 cm).

If the previous and initial equation are combined, the stress can be rewritten as

$$S = 36.75 p \quad (33)$$

$$= 7.61 \times 10^{10} \exp(-19740.37/T_K) + 1.97 T_K \quad (\text{psi}).$$

Figure 11-4 plots this imposed stress as a function of maximum fuel temperatures. Also plotted are the yield and ultimate strength of the type 304 stainless steel clad. The clad ultimate strength is not exceeded if the maximum fuel temperature is maintained below about 950°C and the yield strength is not exceeded for any fuel temperatures below about 920°C, slightly below the yield point and well below the rupture point.

11.2.3. After-Heat Removal Following Coolant Loss

It is assumed that the reactor operates continuously at a constant power density level P_0 so that the maximum inventory of fission products is available to produce heat after the reactor is shut down. The power density after reactor shutdown P is given by

$$P = 0.1 P_0 [(t + 10)^{-0.2} + 0.87 (t + 2 \times 10^7)^{-0.2}] \quad (34)$$

$$\star [1.3 \cos[2.45 (0.26x - 0.5)]]$$

where P_0 = operating power density, W/cm³.

t = time after reactor shutdown, sec,

x = distance from the bottom of the fuel region, cm.

At the time that the coolant is lost from the core the fuel and its surroundings are assumed to be at a temperature of 27°C. This is not necessarily true, for an accident can be postulated in which the coolant loss is the mechanism by which the reactor is shut down. (For the standard non-gapped fuel element, under normal conditions, the time to cool down from operating temperatures is a matter of one to two minutes.) Although such an accident does not appear to be conceivable, calculations indicate that: if it is assumed that the average fuel temperature at the time of coolant loss is equivalent to the operating average fuel temperature, the maximum temperature after the coolant loss is not appreciably different (2% - 4% higher) from that calculated assuming 27°C fuel initially.

The after-heat removal will be accomplished by the flow of air through the core. To determine the flow through the core the buoyant forces were equated to the friction, end, and acceleration losses in the channel as shown in the expression

$$\delta p_b = \delta p_f + \delta p_e + \delta p_i + \delta p_a \quad (35)$$

The buoyant forces are given by

$$\delta p_b = \rho_0 L + \int \rho dx = \rho_0 L + \rho_0 L_0 + \rho L_f + \rho_1 L_t \quad (36)$$

where

ρ_0, ρ, ρ_1 = the entrance, mean, and exit fluid densities, respectively,

L = the effective length of the channel
($L = L_0 + L_f + L_t$),

L_0, L_f, L_t = the length of the channel adjacent to the bottom end reflector, fuel, and top end reflector plus ten channel hydraulic diameters, respectively.

The friction losses in the flow channel are given by

$$\delta p_f = \sum^I f_{F1} \frac{4L_1}{D_e} \frac{W^2}{2g\rho_1 A_c^2} \quad (37)$$

where the summation is over the lower unheated length, the heated length, and the upper unheated length, and

D_e = the hydraulic diameter (0.0601 ft),

f_{F1} = the friction factor $(23.46/R_e)$ [5] .

A_c = the flow area through the core per element (0.0058 ft²).

$G = 4.17 \times 10^8$ ft/hr² .

The sum of the exit and inlet losses, using appropriate expansion coefficients, is given by

$$\delta p_e + \delta p_i = \frac{(\Sigma K) w^2}{2g\rho_0 A_c^2} \quad (38)$$

with

$$\Sigma K = \sum_{j=1}^J [k_j (A_c/A_j)^2] = 1.57$$

The acceleration losses are given by

$$\delta p_a = (1/\rho_1 - 1/\rho_0) (w^2/gA_c^2) \quad (39)$$

By substituting the appropriate expression in Equation 35, using the definition of the Reynolds number, and $L = 2.40$ ft, $L_o = 0.29$ ft, $L_f = 1.25$ ft, and $L_t = 0.87$ ft, one obtains

$$\begin{aligned} & (0.700/\rho_1 + 0.149/\rho_0) \times 10^{-4} w^2 \\ & + (0.153\mu_1/\rho_0 + 0.665\mu/\rho + 0.153\mu_2/\rho_1) \times 10^{-2} w \\ & + (1.25\rho + 0.889\rho_1 + 2.139\rho_0) = 0 \end{aligned} \quad (40)$$

with the flow w in units of lb/hr and μ the viscosity in units of lb/hr-ft.

The properties of air for use in Equation 40 are expressed as

$$\rho_1 = 40/T_1 \text{ (lb/ft}^3\text{)} \quad (41)$$

and

$$\mu_1 = 5.739 \times 10^{-3} + 7.601 \times 10^{-5} T_1 + 1.278 \times 10^{-8} T_1^2 \text{ (lb/hr}\cdot\text{ft)},$$

where T_1 is the appropriate temperature in °R.

The heat transfer coefficient was calculated through the relationship

$$\begin{aligned} N_u &= 6.3 & R_a &\leq 1000 \\ &= 0.806 R_a^{0.2976} & R_a &> 1000 \end{aligned} \quad (42)$$

where N_u = the Nusselt number = hD_e/k ,

R_a = the Rayleigh number $De^4 \rho^2 g \beta \delta T c_p / \mu k L$

h = the heat transfer coefficient, Btu/hr-ft²-°F,

k = the thermal conductivity of the laminar film,
Btu/hr-ft²-°F,

B = the volumetric expansion coefficient, °F⁻¹,

δT = the temperature rise over the channel length, L(°F),

c_p = the specific heat of air (Btu/lb-°F).

The expression for the Nusselt number was derived from the work of Sparrow, Loeffler, and Hubbard [6] for laminar flow between triangular arrays of heated cylinders.

The thermal conductivity and specific heat are given by

$$k = 2.377 \times 10^{-4} + 2.995 \times 10^{-5} T - 4.738 \times 10^{-9} T^2 \quad (43)$$

(Btu/hr-ft-°F)

and

$$c_p = 2.413 \times 10^{-1} + 1.780 \times 10^{-6} T + 1.018 \times 10^{-8} T^2 \quad (44)$$

(Btu/lb-°F),

where T is the appropriate temperature in °R.

These two expressions, as well as that given for the dynamic viscosity of air in Equation 41, are least-square fits to the data presented by Etherington [7].

TAC2D [8], a two-dimensional transient-heat transport computer code developed by GA Technologies, was used for calculating the system temperatures after the loss of tank water. The parameters derived above were programmed into the calculations.

The maximum temperatures reached by the fuel are plotted as a function of operating power density in Figure 11-3 for several cooling or delay times between reactor shutdown and loss of coolant from the core. For reactor operation with maximum power density of 18 kW/element, or less, loss of coolant water immediately upon reactor shutdown would not cause the maximum fuel temperature to exceed 750°C. Operation at maximum power densities greater than 18 kW/element will not result in fuel temperatures above 750°C, if the coolant loss occurs sometime after shutdown, or if emergency cooling is provided. (The time required between shutdown and the beginning of air cooling depends on power density.)

In Figure 11-5, the data presented in Figure 11-3 were replotted to show the time required for natural convective water cooling or emergency cooling, after reactor shutdown, to produce temperatures no greater than a given value. Thus, for example, for a reactor in which the maximum operating power density is 27 kW/element and to limit the temperature to 950°C, or less, there must be an interval of at least 3730 sec (or 1.04 hr) between reactor shutdown and either the loss of tank water from the core or the cessation of emergency cooling. The 65 minute delay time applies to the power density of a 90 element core operated at 1.5 MW but shrinks to a negligible value for the power density in a 100 element core.

11.2.4. Radiation Levels

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 this type of accident (see Table 11-3). Assuming that the reactor has been operating for 10 hours and 1000 hours at 1.5 MW prior to losing all of the shielding water, the radiation dose rates at two different locations are listed in the table.

Time is measured from the conclusion of operation at 1.5 MW. Dose rates assume no water in the tank. The first location (direct radiation) is 6.4 meters above the unshielded reactor core, near the top of the reactor tank. The second is at the top of the reactor shield; this location is shielded from direct radiation but is subject to scattered radiation from a thick concrete ceiling 4.6 m above the top of the reactor shield. The assumption that there is a thick concrete ceiling maximizes the reflected radiation dose. Normal roof structures would give considerably less backscattering.

Table 11-3

Calculated Radiation Dose Rates For Loss of Shield Water

Decay Time	Direct Radiation R/hr		Scattered Radiation R/hr	
	10 hour	1000 hour	10 hour	1000 hour
1 minute	3980	4920	3.7	4.6
1 hour	929	1820	.87	1.7
1 day	87	681	.08	.64
1 week	10	281	.009	.26
1 month	2	104	.002	.10

The tabulated data show that if an individual does not expose himself directly to the core he could work for approximately 2 hours (3 hours for 1 MW) at the top of the shield tank 1 day after shutdown without receiving a dose in excess of that permitted by regulations for a calendar quarter.

For persons outside the building, the radiation from the unshielded core would be collimated upward by the shield structure and, therefore, would not give rise to a public hazard. The method of calculation follows.

The core, shut down and drained of water, was treated as a bare cylindrical source of 1-MeV photons of uniform strength. Its dimensions were taken to be equal to those of the active core lattice. The source strength as a function of time was determined from Way and Wigner's [9] (Equation 45) data on fission product decay. No accounting was made of sources other than fission product decay gammas (i.e., activation gammas from the steel cladding and the aluminum grid plates) or of attenuation through the fuel element end pieces and the upper grid plate. The first of these assumptions is optimistic, the second conservative; the net effect is conservative. The conservative assumption of a uniformly distributed source of 1-MeV photons was balanced by not assuming any buildup in the core.

An approximation of the fission product energy release term is taken as:

$$\Gamma(t) = 1.26 t^{-1.2} \quad (45)$$

where

$\Gamma(t)$ = energy release in MeV/sec-fission,

t = is the time after fission in seconds.

By integration the total core source term is

$$S(t,T) = 3.1 \times 10^{10} P_0 \int_t^{t+T} \Gamma(\tau) d\tau \quad (46)$$

$$= 1.95 \times 10^{11} P_0 (1 - [1 + T/t]^{-0.2}) t^{-0.2} \quad (47)$$

where

$S(t,T)$ = energy release in MeV/sec-watt,

P_0 = reactor power, watts,

T = period of time at power.

The volumetric source of 1 MeV photons is

$$S_v = \frac{S(t_2, T)}{\pi r_c^2 x_c} \quad (48)$$

The direct dose rate at a point outside and on the axis of a cylindrical source is given by:

$$D_d = \frac{S_v}{K} \int_0^{x_c} \int_0^{r_c} e^{-\mu_c z} \frac{2\pi r dr dx}{4\pi R^2} \quad (49)$$

where

S_v = source strength in photons, 1 MeV/cm³-sec,

K = flux-to-dose conversion factor,
 5.77×10^5 photons/cm²-sec per rad/hr,

$2\pi r dr dx$ = cylindrical volume element, dV

r_c = core radius, 26 cm

x_c = core height, 38 cm

μ_c = core attenuation coefficient, 0.27 cm^{-1}

R = distance from volume element to receiver, cm

z = slant penetration in core = $xR/(a+x)$, cm

a = distance from top to core to receiver, 640 cm

For distances far from the core (i.e., for $a \gg r_c$ and x_c) the above expression reduces to

$$D_d = \frac{S_v r_c}{4\mu_c a^2 K} (1 - e^{-\mu_c x_c}) \quad (50)$$

The scattered dose rate was calculated from

$$D_s = 6.03 \times 10^{23} \rho \frac{Z}{A} \frac{I_0 C Q_a}{K(E) x^2} \quad (51)$$

where

ρ = Density of scattering material, concrete,
 2.3 g/cm^3

$\frac{Z}{A}$ = Ratio of average atomic number to atomic mass
of the scatterer, 0.5

and

$I_0 C$ = Incident current times cross section of beams,
photons/sec

K = Photon current to dose rate conversion,
 2.75×10^6 photons/cm²·sec per rad/hr

ϵ = Energy of scattered photon, Mev

x = Distance from scattering point to detector,
400 cm

$$Q_a = \frac{1}{\mu_0 + \mu_1 (\cos\theta_0/\cos\theta_1)} \frac{\delta\sigma}{\delta\Omega}$$

μ_0, μ_1 = Attenuation coefficient in scatter for incident
and scattered photons, cm⁻¹, .146, .292

θ_0, θ_1 = Incident and scattered angle (measured from the
normal to the scatterer), 0, 25 degree

$\delta\sigma, \delta\Omega$ = Differential Klein-Nishina scattering cross
section, cm²/electron-steradian

It was assumed that all of the source photons that exit the top of the reactor pool were incident normally to the concrete roof (i.e., $\theta_0 = 0$) at a point directly over the core, thus

$$I_0 C = S_0 W \quad (52)$$

where

$$S_0 = S_v \pi r_c^2 / \mu_c \quad (53)$$

$$W = \left\{ \sin^{-1} \left[\frac{y_0^2(r_0^2 - x_0^2) + r_0^2(r_0^2 + x_0^2)}{(r_0^2 + x_0^2)(r_0^2 + y_0^2)} \right] - \pi/2 \right\} \frac{1}{2\pi}$$

and

r_0 = Distance from the core to the top of the
pool, ~6.4 m

x_0 = Half width of the pool, ~1 m

y_0 = Half length of the pool, ~1.5 m

S_v, r_c, μ_c have already been defined.

The energy of the scattered photons is given by

$$E = \frac{E_0}{1 + E_0(1 - \cos\theta)/0.51} \quad (54)$$

where E_0 is the incident photon energy (1 Mev) and θ is the scattering angle = $\pi - (\theta_0 + \theta_1)$.

The differential scattering cross section is given by

$$\frac{d\sigma}{d\Omega} = \frac{r_e^2}{2} \left[\frac{E}{E_0} + \left(\frac{E}{E_0} \sin\theta \right)^2 + \left(\frac{E}{E_0} \right)^3 \right] \quad (55)$$

where r_e is the classical electron radius = 2.818×10^{-13} cm.

11.3. FISSION PRODUCT RELEASE

In the analysis of fission product releases under accident conditions, it is assumed that a fuel element in the region of highest power density fails. The failure is assumed to occur in air after operation at full power for an extended period.

11.3.1. Fission Product Inventory

Table 11-4 gives the inventory of radioactive noble gases and halogens in the TRIGA Mark II after continuous operation at 1.5 MW for four years (i.e. 6MW-yrs). The estimated inventory is conservative since actual operation after 4 years is expected to be less than 5% of 4 MW-yrs.

11.3.2. Fission Product Release Fractions

The release of fission products from U-ZrH fuel has been studied at some length. A summary report of these studies [4] indicates that the release from the U-ZrH (H/Zr; 1.6) fuel meat at the steady-state operating temperatures is principally through recoil into the fuel-clad gap. At high temperatures (above 400°C or 500°C), the release mechanism is through a diffusion process and is temperature dependent, unlike recoil.

For the accident considered here, it is assumed that a fuel element in the region of highest power density fails in water and that the peak fuel temperature in the element is less than 400°C. At this temperature, the long-term release fraction would be less than 0.0015%. For the purpose of this analysis it is also assumed that 100% of the noble gases and 100% of the halogens are released from the highest power density fuel element in which 2.22% of the total power is generated.

Table 11-4

NOBLE GAS AND HALOGENS IN THE
REACTOR

Isotope	Quantity (Ci)
Br-83	6,120
Br-84m	6,120
Br-84	12,360
Br-85	12,900
Kr-85m	12,900
Kr-85	678
Kr-87	32,400
Kr-88	46,200
Kr-89	58,500
Kr-90	65,100
Kr-91	44,100
I-131	35,700
Xe-131m	288
I-132	53,100
I-133	86,100
Xe-133m	2,100
Xe-133	86,100
I-134	96,600
I-135	80,400
Xe-135m	24,300
Xe-135	83,100
I-136	77,700
Xe-137	75,300
Xe-138	70,200
Xe-139	70,800
Xe-140	48,600

It is important to note that the release fraction in accident conditions is characteristic of the normal operating temperature and not the temperature during the accident conditions. This is because the fission products released as a result of a fuel clad failure are those that have collected in the fuel-clad gap during normal operation.

Other assumptions concerning estimated accident scenario doses are:

- a. Assume an element fails in air such that all (100%) noble gases and halogens in the gap are effectively released.
- b. There is no plate-out of any released fission products.
- c. After the failure a ventilation rate of 10 air changes per hour occurs with no air filtration.
- d. Doses are calculated assuming exposure to a semi-infinite cloud.

- e. Doses are calculated for release from the total core and a single fuel element (90 element core with peak to average flux of 2.0).
- f. Doses external to the building are calculated by assuming a minimum building dilution factor for releases (1.0 m/sec wind velocity with building cross section of 234 m²).
- g. Doses were also calculated for personnel in the reactor room by dumping rapidly a small fraction of the total inventory into the room such that the continuous release is equivalent to a constant concentration.

The net effect of these assumptions is that for the single element accident condition, the fraction of the noble gases released from the building is:

$$\begin{aligned} f_{NG} &= 2.0 \times 10^{-5} \times 1.0 \times 2.22 \times 10^{-2} \\ &= 4.44 \times 10^{-7} \end{aligned} \quad (56)$$

and of the halogens is:

$$\begin{aligned} f_H &= 2.0 \times 10^{-5} \times 1.0 \times 2.22 \times 10^{-2} \\ &= 4.44 \times 10^{-7} \end{aligned} \quad (57)$$

where a conservative release fraction of 0.002% is applied.

11.3.3. Downwind Dose Calculation

The minimum roof level dilution factor was calculated, assuming a building cross sectional area of 234 square meters. The factor is based on mixing in the lee of the building when the wind velocity is 1 m/sec. A dilution factor of 0.00854 seconds per cubic meter is applied.

The calculation of whole body gamma doses and thyroid doses downwind from the point of release was accomplished through the use of the computer code GADOSE [10]. In this code the set of differential equations describing the rate of production of an isotope through the decay of its precursors and the rate of removal through radioactive decay and removal by the ventilation system is integrated for each member of the chain. The release rate q_i to the environment for the i th isotope at time t_i in hours is:

$$q_i(t) = \epsilon_i Q(t) (\ell/V)/3600 \quad (58)$$

where $Q_i(t)$ = the release of the i th isotope in Ci,

ℓ/V = the building leakage rate in (m³/hr)/m³,

ϵ_i = the filter efficiency for the isotope,

$$g_i = 1 - \epsilon_i.$$

The quantity $Q_i(t)$ is the amount of the i th isotope in the discharged air at the time, t . This quantity is given by

$$Q_i(t) = f_i Q_i(0) e^{-(\lambda_i + l/V)t} \quad (59)$$

where $Q_i(0)$ = the inventory of the i th isotope as found in Table 11-4,

λ_i = the decay constant for the i th isotope, and

f_i = the release fraction to the reactor hall.

The concentration downwind at a distance x for the i th isotope is calculated from

$$Q_i(t, x) = q_i(t - \tau) \psi(x) e^{-\lambda_i \tau} \quad (60)$$

where τ = the transit time from the release point to the dose point, hr,

$\psi(x)$ = the dilution factor at the distance x , sec/m^3

The whole body gamma ray dose rate for the i th isotope, D_{wi} , at the distance x and time t is calculated, assuming a semi-infinite cloud, through the expression:

$$D_w(t, x) = 900 \sum_i E_i Q_i(t, x) \quad (61)$$

where E_i = the average gamma ray energy per disintegration, MeV, and the constant includes the attenuation coefficient for air as well as the conversion factors required. Dose rate is in units of rad/hr.

Internal dose rates, in this case the dose rate to the thyroid, are calculated by:

$$D_{th}(t, x) = 3600 \sum_i B K_i Q_i(t, x) \quad (62)$$

where B = the breathing rate, m^3/sec , and

K_i = the internal dose effectivity of the i th isotope, rem/Ci .

The values for the breathing rate are given in Table 11-5 and are taken from a published regulatory guide [11].

The average gamma ray energy per disintegration and the internal dose effectivity for each isotope considered are given in Table 11-6.

The decay products of these isotopes are also included in the calculation; however, their contribution to the dose rates are small and therefore the data for these isotopes were not included in the table.

11.3.4. Downwind Doses

The whole body gamma dose and thyroid dose in the lee of the building are shown in Table 11-7. These doses are acceptable relative to the conservative nature of the calculations and likelihood that an accident scenario would actually lead to these results.

Table 11-5

ASSUMED BREATHING RATES:

Time (hr)	Breathing Rate (m^3/sec)
0 to 8	3.47×10^{-4}
8 to 24	1.75×10^{-4}
Over 24	2.32×10^{-4}

Table 11-6

AVERAGE GAMMA RAY ENERGY AND INTERNAL DOSE
EFFECTIVITY FOR EACH FISSION PRODUCT ISOTOPE

Isotope	E_i (MeV)	K_i (rem/Ci)
Br-83	0.92×10^{-2}	
Br-84	1.87	
I-131	0.40	1.486×10^6
I-132	1.96	5.288×10^4
I-133	0.56	3.951×10^5
I-134	3.02	2.538×10^4
I-135	1.77	1.231×10^5
I-136	2.91	
Kr-83m	0.8×10^{-3}	
Kr-85m	0.16	
Kr-85	0.4×10^{-2}	
Kr-87	1.07	
Kr-88	2.05	
Kr-89	2.40	
Xe-131m	0.82×10^{-2}	
Xe-133m	0.37×10^{-1}	
Xe-133	0.29×10^{-1}	
Xe-135m	0.46	
Xe-135	0.25	
Xe-137	1.22	
Xe-138	1.57	

Table 11-7

DOSES FROM FISSION PRODUCT RELEASES

Accident Condition	Time After Release (Hr)	Dose (Rad) ¹			
		Halogens		Noble Gases	
		Whole Body	Internal	Whole Body	Internal
Core Inventory Release ² 1.5MW	0.1	0.3	5.9×10^1	0.19	1.0×10^{-4}
	1.0	2.1	5.8×10^2	1.14	5.0×10^{-3}
	8.0	8.3	4.2×10^3	3.57	5.7×10^{-2}
	24.0	13.8	1.1×10^4	4.51	1.8×10^{-1}
	720.00	31.4	8.6×10^4	6.37	4.3×10^0
Core Inventory Release ³ 1.5MW	0.1	0.019	3.8	0.013	5.6×10^{-6}
	1.0	0.028	6.0	0.017	1.7×10^{-5}
	8.0	0.028	6.0	0.017	1.7×10^{-5}
Single Element Release ² 1.0MW * 2.22% of energy	0.1	0.004	8.7×10^{-1}	0.002	1.5×10^{-6}
	1.0	0.031	8.6×10^0	0.016	7.4×10^{-5}
	8.0	0.123	6.2×10^1	0.053	8.4×10^{-4}
	24.0	0.204	1.6×10^2	0.067	2.7×10^{-3}
	720.0	0.465	1.3×10^3	0.094	6.4×10^{-2}
Single Element Release ³ 1.0MW * 2.22% of Energy	0.1	2.8×10^{-4}	0.056	1.9×10^{-4}	8.3×10^{-8}
	1.0	4.1×10^{-4}	0.089	2.5×10^{-4}	2.5×10^{-7}
	8.0	4.1×10^{-4}	0.089	2.5×10^{-4}	2.5×10^{-7}

1.) Doses calculated assuming 4 years full power operation and release fraction of 0.002%.

2.) Release calculated for semi-infinite cloud in room volume of 4290 m³ with no ventilation.

3.) Release calculated at zero distance with an air change rate of ten per hour and building dilution factor of 8540 sec/cc.

Chapter 11 References

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