UNIVERSITY OF WISCONSIN
NUCLEAR REACTOR
LICENSE NO. R-74
DOCKET NO. 50-156

LICENSE RENEWAL APPLICATION

SAFETY ANALYSIS REPORT,
AND
TECHNICAL SPECIFICATIONS

(UPDATED SAR CHAPTERS 5 & 7)

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

UWNR

Nuclear Reactor Laboratory

University of Wisconsin-Madison

1513 University Avenue, Room 141 ME, Madison, WI 53706-1687, Tel: (608) 262-3392, FAX: (608) 262-8590

email: reactor@engr.wisc.edu, http://www.engr.wisc.edu/groups/rxtr.lab/

September 7, 2004 RSC 822

Director, Nuclear Reactor Regulations
ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

RE: License Renewal of the University of Wisconsin Nuclear Reactor
License R-74, Docket 50-156

Dear Sir:

The following information is submitted in support of our application for license renewal dated May 9, 2000.

1. Chapter 5 of the Safety Analysis Report for renewal of license R-74 for the University of Wisconsin Nuclear Reactor, hereinafter referred to as UWNR SAR, is replaced in its entirety. This replacement reflects revisions to the UWNR SAR as a result of a facility modification to the Reactor Cooling System.
2. Chapter 7 of the UWNR SAR is replaced in its entirety. This replacement reflects changes to the UWNR SAR as a result of facility modifications to the Instrumentation and Controls System. In specific, the insturmentation and controls system was modified to support a new pulse channel recorder and new cooling system status lights.
3. The Table of Contents of the UWNR SAR was updated to reflect correct cross references as a result of the changes to Chapters 5 and 7.
4. The index of the UWNR SAR was updated to reflect correct cross references as a result of the changes to Chapters 5 and 7.

All changes are indicated by a vertical line in the right hand margin.

In accordance with 10 CFR 50.30(b), I declare under penalty of perjury that the foregoing is true and correct.

If you have any further questions, please contact me at (608) 262-3392.

Sincerely,

Robert J. Agasie

Robert J. Agasie
Reactor Director

A035
A020

UNIVERSITY OF
WISCONSIN
MADISON

Director, Nuclear Reactor Regulation
ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555

May 9, 2000

RE: License Renewal of the University of Wisconsin Nuclear Reactor, License R74, Docket 50-156

Dear Sir:

This letter accompanies submission of the revised Safety Analysis Report (SAR) and proposed Technical Specifications (TS) for the University of Wisconsin Nuclear Reactor. The purpose of this submittal is to renew License R-74, a class 104 license, for a twenty year period until June 30, 2020.

The SAR and TS (chapter 14 of the SAR) have been reconfigured into the format of NUREG-1537, although the safety analysis is the same as in previous versions of the SAR. The changes and revisions to the documents were reviewed and approved by the Reactor Safety Committee at their May 5, 2000 meeting.

The applicant, the University of Wisconsin, is a non-profit educational institution and an agency of the State of Wisconsin. There is no foreign control of the University. Chapter 15 of the SAR details the financial qualifications of the University to operate and decommission the facility. The University reiterates the intent to provide decommissioning funding when needed.

Correspondence relating to this application should be directed to:

Reactor Director
Nuclear Reactor Laboratory
Room 130 Mechanical Engineering Building
1513 University Avenue
Madison, WI 53706-1572

Very truly yours,

John D. Wiley
Provost

Subscribed and sworn before me this 9th day of May 2000.

Maria Justiliano
Maria Justiliano, Notary Public

A020

1/22

SAFETY ANALYSIS REPORT

FOR RENEWAL OF LICENSE R-74

FOR THE

UNIVERSITY OF WISCONSIN

NUCLEAR REACTOR

April 2000

SAFETY ANALYSIS REPORT

FOR RENEWAL OF LICENSE R-74

FOR THE

UNIVERSITY OF WISCONSIN NUCLEAR REACTOR

May 2000

Index

1 THE FACILITY

1.1 Introduction

This document is prepared as part of an application for renewal of License R-74.

The University of Wisconsin has operated a teaching and research reactor, licensed as R-74 under Docket 50-156 since 1961. The reactor supports teaching as a facility of the Engineering Physics Department, other departments of the university, and other educational institutions. Research use of the reactor supports the department, other University of Wisconsin departments, numerous other educational institutions, and some non-educational groups.

The reactor is located on the campus of the university in a building located at 1513 University Avenue in Madison, Wisconsin. It currently operates at a 1000 kW steady-state power with pulsing capability to 1000 MW.

The original Hazards Summary Report has been amended a number of times over the operating history, and the present version of the Safety Analysis Report has been kept up to date by issuance of replacement pages at each annual report submission. As a request for license renewal progressed , however, it was determined that the Safety Analysis Report should be replaced with this completely new version structured in accordance with the guidance in NUREG 1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" dated February 1996.

1.2 Summary and Conclusions on Principal Safety Considerations

Analysis of possible accident scenarios is included in Chapter 13. As a TRIGA-type reactor, the primary safety features stem from the use of a fuel with a strong negative prompt temperature coefficient of reactivity which limits excursions from reactivity insertions, thus preventing fuel damage from credible reactivity accidents. Ejection of the transient from the core when the core is operating at the power level scram point will result in no fuel damage. Since experiments are limited to the same reactivity worth as the transient rod, experiment failure cannot result in more severe transients.

In addition, the operating power level of 1000 kW results in a decay heat potential in the fuel small enough that loss of reactor coolant does not result in fuel damage or release of fission products.

In extreme accident conditions in which operation is taking place with already damaged fuel, releases to the public are shown to be nominal except for a combination of loss of pool water and loss of the ventilation system concurrent with the fuel damage. Even with these extremely conservative assumptions, analysis of this accident shows the event will result in exposure to the public that would be classified as an Alert. More realistic assumptions used in the accident analysis indicates that the maximum hypothetical accident would result in emissions and radiation exposure within those allowed by 10CFR Part 20.

1.3 General Description of the Facility

The University of Wisconsin Nuclear Reactor is located in the Mechanical Engineering Building on the Madison campus of the University with the city of Madison, WI. The building also contains classrooms, laboratories, shops, and staff offices for the departments of Mechanical Engineering, Industrial Engineering, and Engineering Physics departments.

Figure 1-1 is a pictorial view of the reactor. The reactor is a heterogeneous pool-type nuclear reactor currently fueled with TRIGA-FLIP fuel modified to adapt to 4-element bundle assemblies. The coolant is light water which circulates through the core by natural convection. The core is reflected by water and graphite. Maximum steady-state power level is 1000 KW.

A 7 by 9 grid, surrounded by a core box, positions fuel, reflector, and control elements. Three shim-safety blades, a transient control rod, and a regulating blade control core reactivity. The transient control rod is guided by a tube replacing a fuel element in a central fuel bundle, while the control blades move vertically in two shrouds extending the length of the core. The grid box and control element drive mechanisms are supported by a suspension frame from the reactor bridge.

Cold, clean core excess reactivity is about 4.9 per cent reactivity. Control elements having a scram capability provide a shut-down margin of 4 per cent ΔK_{eff}.

Proposed technical specifications for the facility are included in this report as Chapter 14.

SUMMARY OF REACTOR DATA

Responsible Organization	The University of Wisconsin
Location	Madison, Wisconsin
Purpose	Teaching and Research
Fuel	
Type	TRIGA and TRIGA-FLIP High Hydride in 4 element clusters
Number of elements in standard 1000 KW Core	[redacted]

Control

Safety elements	Three vertical blades
Regulating-servo element	One vertical blade
Transient control	One rod

Experimental Facilities

Thermal column	One, 40-inch square graphite, φ th = 2×10^{8}
Beam Ports	Four, 6-inch diameter φ th = $1 - 3 \times 10^{10}$ nv at shield side of shutter; about 8×10^{11} at core end of port
Pneumatic Tube	One, 2-inch (sample size 1 1/4 inch diameter by 5-1/2 inches long), φ th = 4×10^{12} nv.
Hydraulic in-pool irradiation facilities	Presently three, 2.5 inch (sample size up to 1 7/8 diameter by 4 inches long) φ th = $8 \times 10^{12} - 2.4 \times 10^{13}$ nv, depending upon location

Thermal neutron fluxes for isotope production include the above, plus large irradiation spaces outside the core thermal neutron fluxes of around 1.3×10^{13} nv.

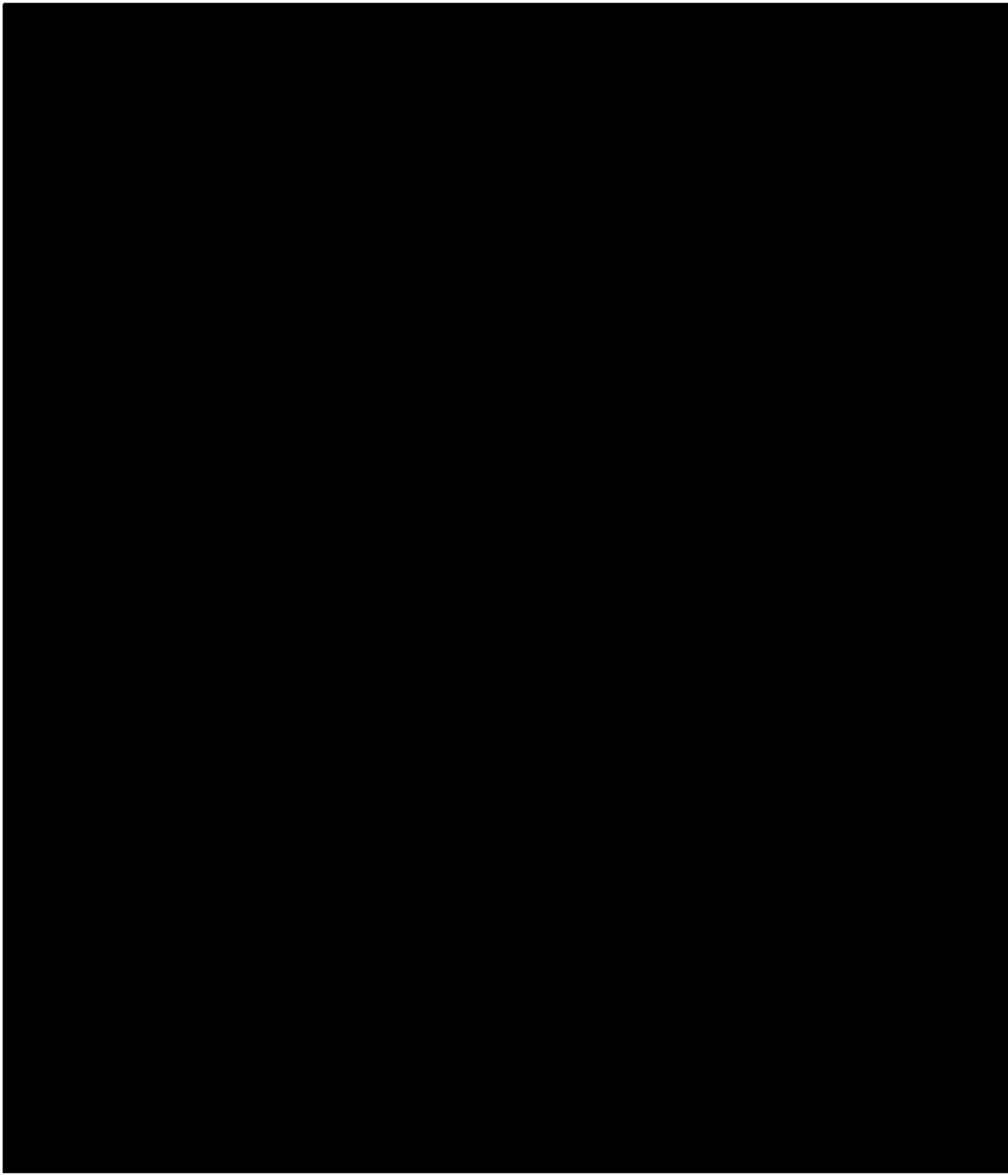

Figure 1-1 UWNR Open Pool Reactor

Dimensions

Pool	8 x 12 x 27-1/2 ft. deep
Standard 1000 KW core	15 x 17 x 15 inches high
Grid box	9 x 7 array of 3-inch modules
Control blades	10-1/2 inches wide

Fuel Element

Diameter	[redacted]
Length	[redacted]
Fueled length	[redacted]

Nuclear Characteristics

1 MW Steady state:

Maximum thermal neutron flux	3.2×10^{13} nv
Maximum fast neutron flux	3.0×10^{13} nv

1000 MW Pulse

Maximum thermal neutron flux	3.2×10^{16} nv
Maximum fast neutron flux	3.0×10^{16} nv

Prompt temperature coefficient of reactivity	-1.26×10^{-4} $\Delta K/^{\circ}C$
Void coefficient of reactivity	$-.2 \times 10^{-4}$ $\Delta K/\%$ void

Prompt neutron lifetime	42 μsec STD fuel, 24 μsec FLIP
Effective delayed neutron fraction	0.007

1.4 Shared Facilities and Equipment

The Reactor Laboratory and supporting laboratories are an integral part of the Mechanical Engineering Building, and thus share walls, heating, water supplies, sewage, and main electrical distribution with the remainder of the building. Ventilation systems and air conditioning systems are dedicated to non-shared use except for those air conditioning systems in office spaces that depend upon the campus chilled water system for cooling. [redacted]

1.5 Comparison With Similar Facilities

The best indication of reactor characteristics is the performance of the facility itself, which has been in routine operation with the present operational core since August, 1979.

The reactor at Washington State University is very similar to UWNR, having also originally been a General Electric Open Pool Reactor which was converted to TRIGA fuel, and eventually partially converted to FLIP fuel. The reactor at Texas A & M University is also a converted core, though the original reactor was not built by GE. The pool size and experimental facility configuration differs on the three reactors, but basic reactor behavior and accident analysis are quite similar. In addition, the nuclear characteristics of UWNR are quite similar to the TRIGA Mark III prototype and other FLIP fueled reactors . In chapter 4 of this report the similarity between the UWNR and the prototype is detailed.

1.6 Summary of Operations

Present plans and previous usage involve use of the reactor in performance of the following experiments:

1. Reactor Start-up and Operation;
2. Radiation Survey of the Reactor and Surroundings;
3. Control and Regulating Blade Calibration;
4. Measurement of Reactor Power and Calibration of Reactor Instruments;
5. Measurement of Shutdown Power Level;

6. Measurement of Reactor Period,

7. Measurement of Temperature Coefficient of Reactivity;

8. Measurement of Void Coefficient of Reactivity;

9. Experiments Involving the Danger Coefficient Method;

10. Experiments to Measure the Disadvantage Factor;

11. Studies of Reactor Buckling and Delta K/K.

12. Critical Mass Experiments;

13. Measurement of Thermal Neutron Cross Sections;

14. Delayed Neutron Emission;

15. Activation Analysis;

16. Experiments Utilizing Pile Oscillator Techniques;

17. Flux Distributions in Reactor and Effect of Absorbers on Flux Patterns;

18. Shielding Experiments;

19. Experiments on the Production of Radioisotopes;

20. Neutron Diffractometer Measurements.

21. Neutron Radiography

The above represents the experiments planned at present, but it is anticipated that further experiments (both for training and research) will be added.

Fueled Experiments

Fueled experiments will be so controlled that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies and the maximum Sr90 inventory is not greater than 5 millicuries.

1.7 Compliance With the Nuclear Waste Policy Act of 1982

In accordance with a letter from the U. S. Department of Energy (R. L. Morgan) to the U. S. Nuclear Regulatory Commission (H. Denton) dated May 3, 1983, it has been determined that all universities operating non-power reactors have entered into a contract with DOE that provides that DOE retain title to the fuel and DOE is obligated to take the spent fuel and/or high level waste for storage or reprocessing. Because the University of Wisconsin has entered into such a contract with DOE, the applicable requirements of the Nuclear Waste Policy Act of 1982 have been satisfied by the University of Wisconsin Nuclear Reactor Facility. A copy of the current Task Order under Master Task Agreement (which is the successor to contract DE-AC07-76ER01560) is included in Appendix B.

1.8 Facility Modifications and History

Construction permit CPRR-55, authorizing construction of the University of Wisconsin Nuclear Reactor (UWNR), was issued on June 7, 1960. License R-74 was issued November 23, 1960. The expiration date of the license was set at 40 years after the issuance of the construction permit (June 6, 2000). The University of Wisconsin Nuclear Reactor achieved initial criticality on January 5, 1961 as a 10 KW teaching and research reactor. After a license amendment dated October 22, 1964, the power level was increased to 250 KW on December 7, 1964, using the original flat-plate aluminum clad fuel. Operations with the original core ended October 13, 1967, after 2268.5 critical hours and 105, 650 kWhr core exposure.

A cooling system was installed and the reactor was converted to a 1000 KW, TRIGA reactor with pulsing capability in 1967. Construction permit CPRR-97 authorizing the changes was issued on June 7, 1967. Amendment No.8 to the operating license for the conversion was issued on November 13, 1967. Initial criticality with the TRIGA core occurred on November 14, 1967. After over 3 million kilowatt hours of operation with the TRIGA core a partial refueling was necessary. FLIP fuel was available to afford significantly improved core lifetimes, so a new Safety Analysis Report was submitted in April, 1973 describing facility characteristics and safeguards using standard fuel, FLIP fuel, and mixtures of the two fuel types in defined compositions. License amendment No. 10 was issued in response. The initial partial refuelling with 9 fuel bundles replaced with FLIP (Fuel Life Improvement Program) fuel in March 1974. Additional fuel replacements in January 1978 and August 1979 resulted in the present operating core, consisting entirely of FLIP fuel. The total fuel exposure since converting to TRIGA fuel is almost 17 million kilowatt hours.

A number of other license amendments were issued during the term of the license, involving inclusion of security, training, and emergency plans. Several other amendments changed the amount and types of Special Nuclear Material used in connection with the license. None of these changes had any effect on the operating characteristics of the reactor, and are therefore not detailed here. The most recent amendment was No. 15.

Two changes in the facility description (as would be reported in our annual report) are included in this Safety Analysis Report. The substantive changes are; elimination of the reactor trip on short period and elimination of the electronic scram capability of the safety amplifier. These changes have been approved by our Reactor Safety Committee based on a 10 CFR 50.59 analysis. Neither of these items have been required by Technical Specifications, since they do not provide significant protection for a pulsing TRIGA reactor. Sections 7.2.3 and 7.4 describe the instrumentation as it would be changed.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The University of Wisconsin Nuclear Reactor Laboratory is located within the Mechanical Engineering Building at 1513 University Avenue on the University of Wisconsin-Madison (UW) campus. The reactor is located a[redacted] On the United States Geological Survey (USGS) Madison West, WIS 15' Quadrangle topographical map, the Universal Transverse Mercator Coordinates are; [redacted] [redacted]

The UW campus is surrounded by the city of Madison in Dane county, Wisconsin. **Figure 2-1** shows the location of Dane count within Wisconsin. Madison, a city of approximately 191,000 residents (1990 Census statistics), is built around two lakes in the center of Dane county, **Figure 2-2** . Lake Mendota (15 square miles) lies northwest of Lake Monona (5 square miles) and the two lakes are only 2/3 of a mile apart at one point. The UW campus is set on this narrow neck of land between the two lakes, known as the isthmus, and on the southern shore of Lake Mendota. The Mechanical Engineering Building is near the southwestern border of the UW campus, where the nearest non UW owned property is 425 ft (130 m) from the reactor site. The reactor is 2300 ft (700 m) south of the shore of Lake Mendota.

2.1.1.2 Boundary and Zone Area Maps

A map of the City of Madison detailing the general topography and the surrounding urban and rural zones up to a distance of 8 km is reproduced in **Figure 2-3** . The UW campus, which is located on the southern shore of Lake Mendota, is shown in **Figure 2-4** . The Mechanical Engineering Building is located on the engineering campus which is the southwest corner of the UW campus as shown in **Figure 2-4** .

The operations boundary is defined as the Reactor Laboratory, Room 130, of the Mechanical Engineering Building. The site boundary is defined as that portion of the center and east wings of the Mechanical Engineering Building south of the lobby, including the basement area at the south end of the central and west wings of the building, plus the portion of Engineering Drive (formally Johnson Drive) south of the designated areas of the building. **Figure 2-6** , **Figure 2-7** , **Figure 2-8** and **Figure 2-9** depict the floor plans of the Mechanical Engineering Building's basement, first floor, second floor and third floor respectively. The emergency preparedness zone is entirely within the operations boundary, as defined above.

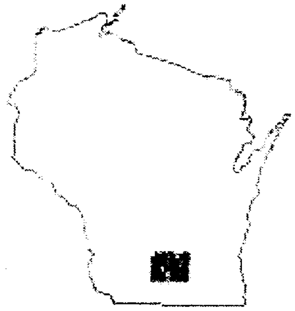

Figure 2-1 Wisconsin Map

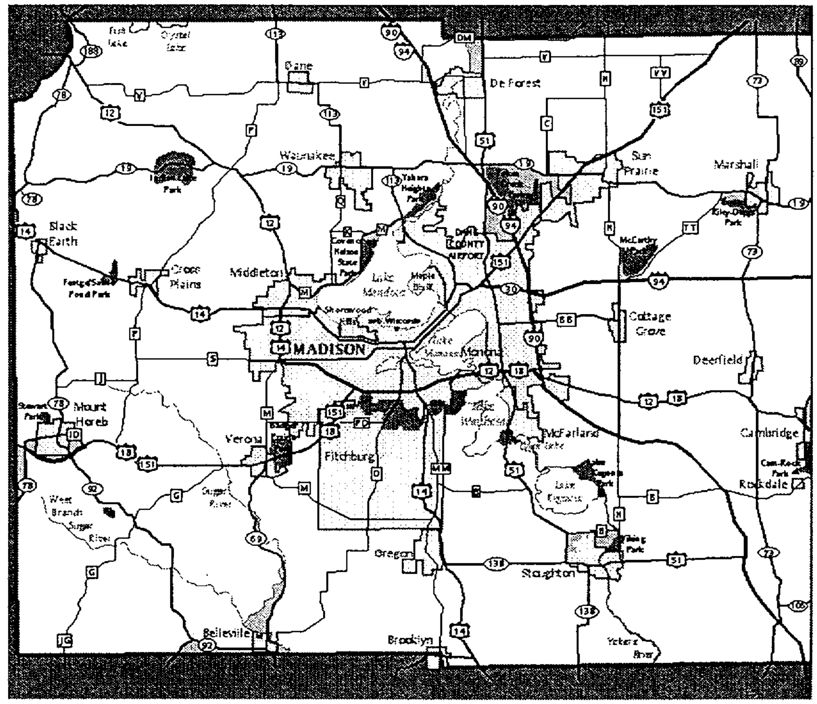

Figure 2-2 Dane County, Wisconsin

2.1.2 Population Distribution

Population distributions estimated by the uniform density model and the 1990 Census[1] are shown Table 2-1. The area around the campus is mature residential, and the central business district is only a short distance away. Population is therefore quite stable in the immediate surrounding areas of the UW campus. The 8 kilometer radius includes much more sparsely-settled regions to the north of Lake Mendota, and population in this region is likely to increase markedly in the future. The nearest permanent residence is approximately 150 m west of the reactor site.

Transient population around the reactor include students present in classrooms and offices in the Mechanical Engineering Building during the months of September through May as well as spectators attending sporting events at Camp Randall Stadium which is approximately 250 m due south of the reactor. The maximum number of students attending classes in the east wing of the Mechanical Engineering Building at any one time is estimated at 300. The maximum number of students present in the central wing of the Mechanical Engineering Building at any one time is estimated at 50. The maximum number of spectators contained by Camp Randall Stadium is 76,129.

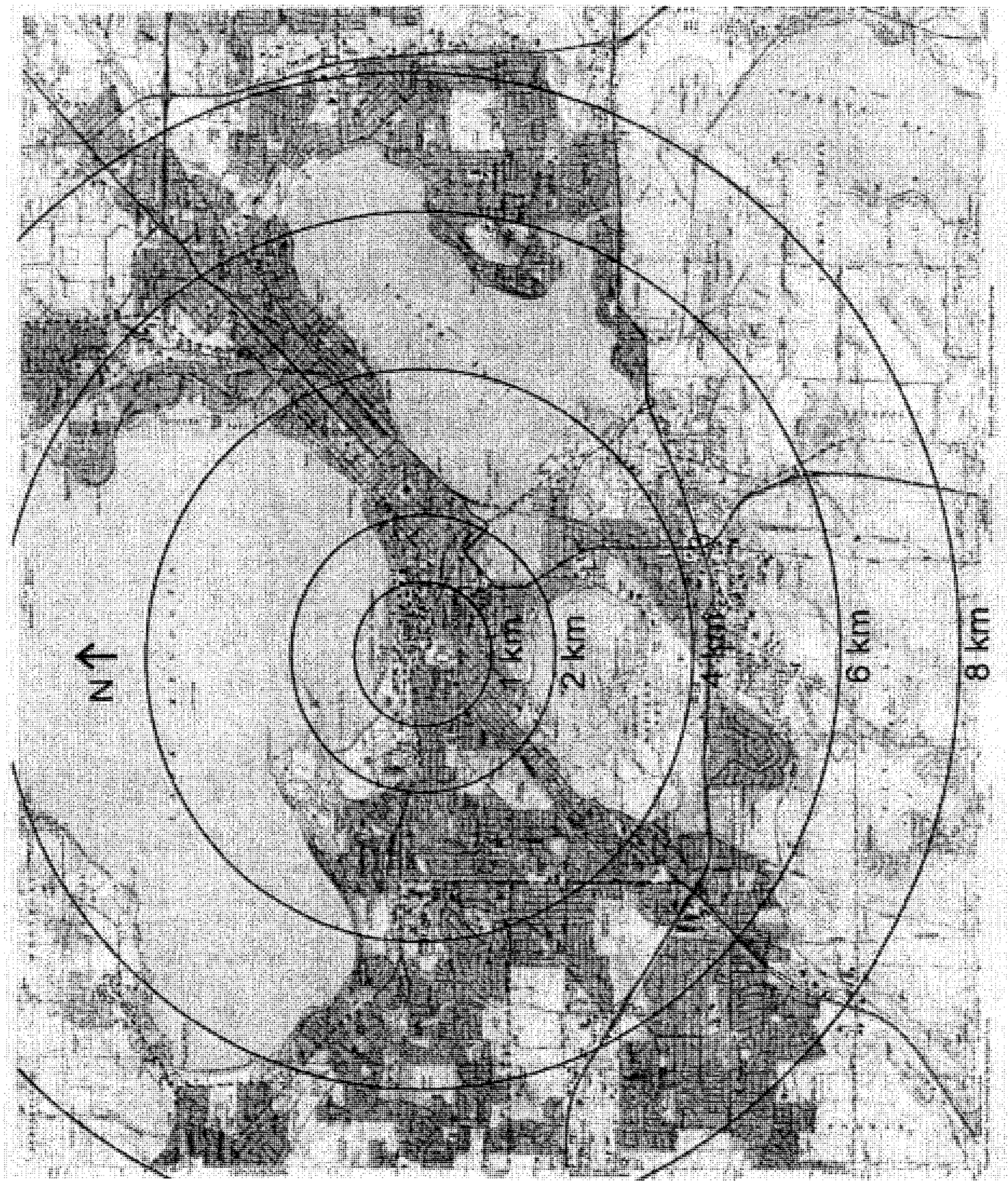

Figure 2-3 City of Madison, Wisconsin

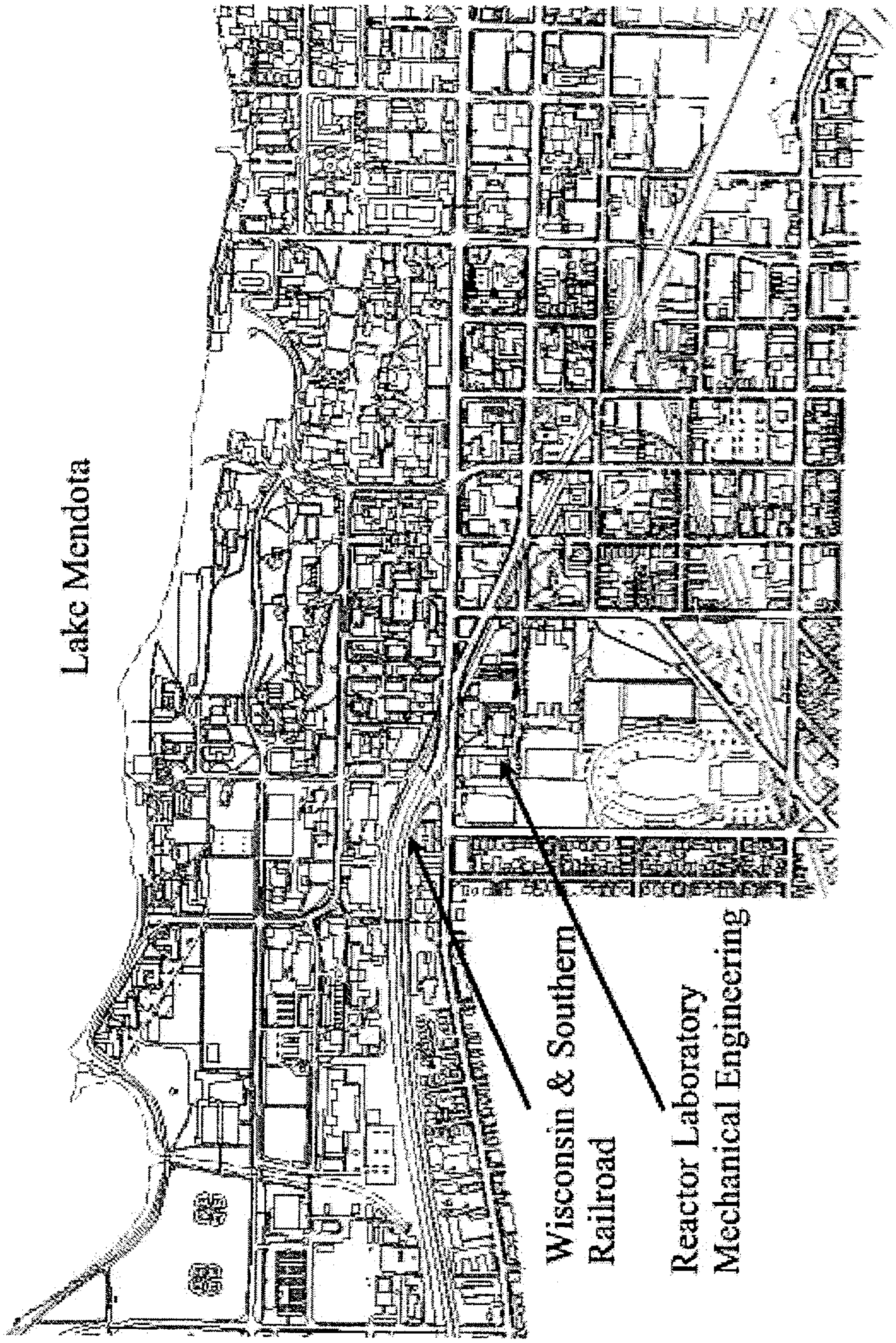

Figure 2-4 University of Wisconsin Campus

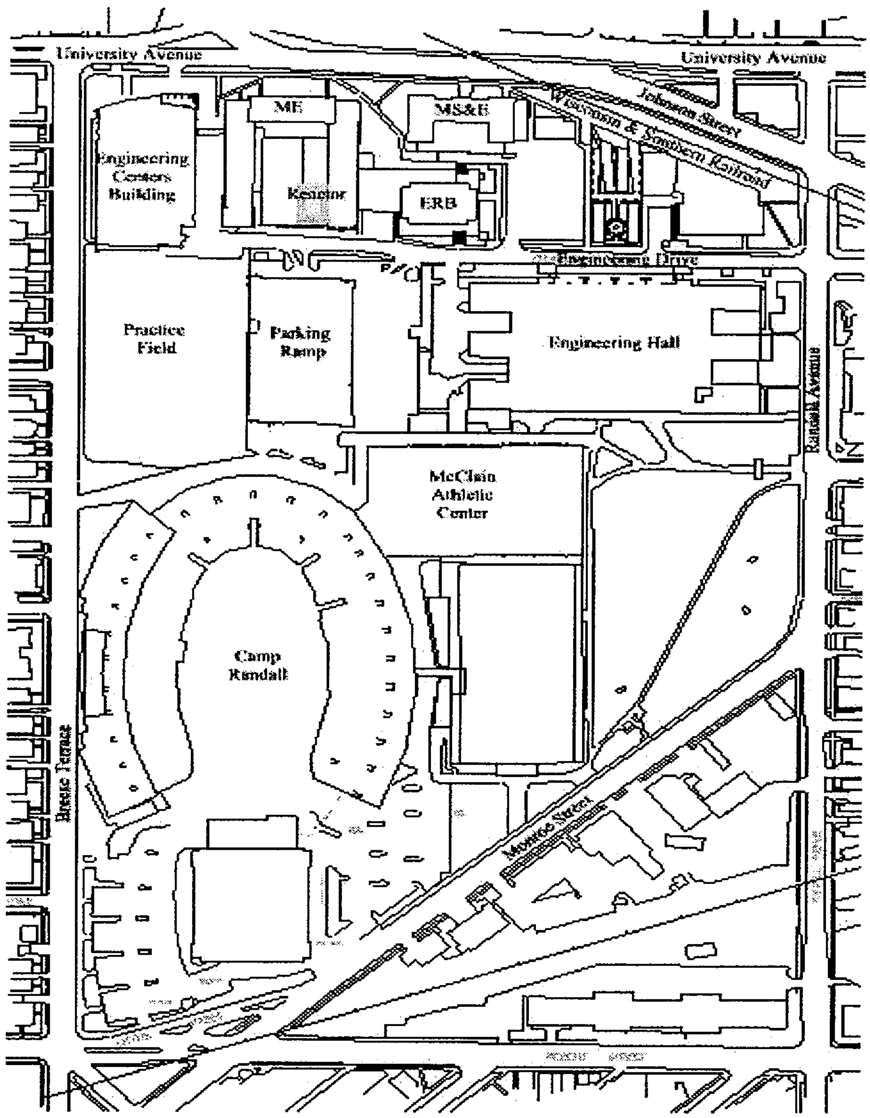

Figure 2-5 UW Engineering Campus

[redacted]

Figure 2-6 Mechanical Engineering - Basement

[redacted]

Figure 2-7 Mechanical Engineering - First Floor

[redacted]

Figure 2-8 Mechanical Engineering - Second Floor

[redacted]

Figure 2-9 Mechanical Engineering - Third Floor

TABLE 2-1
Population Distribution

Distance from Facility(kilometers)	Estimated 1990 Population
1	13,218
2	32,126
4	72,505
6	115,073
8	146,093

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

The UW campus is surrounded, mainly, by a residential district to the south and west, while to the east is primarily a commercial business district with city and state government office buildings. No industrial facilities are in the vicinity of the reactor.

A railroad spur of the Wisconsin & Southern Railroad Company runs through campus and is 100 m to the Reactor Laboratory at its closest approach. A rail car holding yard which is a part of this spur is approximately 300 m northwest. The primary commodity transported over this spur or resident at the rail car holding yard is coal[2].

The reactor is located approximately 4.5 km north of a bypass highway, known as the Beltline, for US highways 12, 14, 18 and 151. Interstates 90 and 94 are located approximately 10 km to the east and northeast of the reactor site.

There are no military facilities in the Madison area with the exception of the Wisconsin Air National Guard which is located on a military ramp of the Dane County Regional Airport. While the Wisconsin Air National Guard flies approximately 3000 missions annually, the flight patten of these missions typically are north of Lake Mendota and the City of Madison. At no time do any of these mission flights carry live ammunition[3]. More information about the Dane County Regional Airport facility is reported in section 2.2.2 Air Traffic.

2.2.2 Air Traffic

The Dane County Regional Airport is approximately 8 km to the north east of the reactor site. This is the only commercial airport near the reactor. While there are three smaller air fields within 16 km from the reactor in the communities surrounding Madison, these air fields are for general aviation only.

The Dane County Regional Airport has 3 runways with the following outbound headings; 360°(north)/180°, 310°/130°, and 210°/30°. None of these headings have trajectories that take commercial traffic directly over the reactor immediately before arrival or after departure. The airport is serviced by several commercial express carriers and is utilized by the Wisconsin Air National Guard as well as general aviation aircraft[4]. Of the 141,783 events (arrival and departure are counted as two separate events) in 1999, 80% were classified as general aviation, 16% commercial carriers/taxi and 4% military. Due to the infrequent arrival of commercial traffic, the air traffic control tower does not place inbound traffic in holding patterns around the city of Madison[5].

2.2.3 Analysis of Potential Accidents at Facilities

There are no industrial, transportation or military facilities within the vicinity of the reactor site that have the potential for accidents with consequences significant to impact the Reactor Laboratory. While a railroad spur passes within 100 m of the reactor facility, this spur transports non hazardous cargo and other major ground transportation routes are located at great distances from the Reactor Laboratory. Due to the frequency and flight paths of commercial and military air traffic the probability of occurrence of an accident is considered extremely low.

2.3 Meteorology

2.3.1 General and Local Climate

"Madison has the typical continental climate of interior North America with a large annual temperature range and with frequent short period temperature changes. The range of extreme temperatures is from about 110 to -40 degrees. Winter temperatures (December - February) average near 20 degrees and the summer (June - August) average temperature is in the upper 60s. Daily temperatures average below 32 degrees about 120 days and above 40 degrees for about 210 days of the year.

Madison lies in the path of the frequent cyclones and anticyclones which move eastward over this area during fall, winter and spring. In summer, the cyclones have diminished intensity and tend to pass farther north. The most frequent air masses are of polar origin. Occasional outbreaks of arctic air affect this area during the winter months. Although northward moving tropical air masses contribute considerable cloudiness and precipitation, the true Gulf air mass does not reach this area in winter, and only occasionally at other seasons. Summers are pleasant, with only occasional periods of extreme heat or high humidity.

There are no dry and wet seasons, but about 60 percent of the annual precipitation falls in the five months of May through September. Cold season precipitation is lighter, but lasts longer. During July, August, and September rainfall is mostly from thunderstorms and tends to be erratic and variable. Average occurrence of thunderstorms is just under 7 days per month during this period. Tornadoes are infrequent. Dane County has about one tornado in every three to five years.

The ground is covered with 1 inch or more of snow about 60 percent of the time from about December 10 to near February 25 in an average winter. The soil is usually frozen from the first of December through most of March with an average frost penetration of 25 to 30 inches."[6]

2.3.2 Site Meteorology

The summary of meteorological conditions for Madison is based on the records obtained from the International Station Meteorological Climate Summary[7] jointly produced by the National Oceanic and Atmospheric Administration (NOAA), the United States Air Force (USAF) and United States Navy. The data specifically compiled for Madison was obtained from the National Weather Service and unless specifically noted, is for the period of record from 1948 to 1995.

The Reactor Laboratory does not have a continuing onsite meteorological data measurements program. All future meteorological data will be obtained from the National Weather Service station in Madison.

2.3.2.1 Temperature

The monthly average and daily average extreme temperatures for the Madison area are shown in Table 2-2. The record extreme temperatures in the Madison area, as reported by the National Weather Service, have ranged from a low of -37 °F in January 1951 to a high of 107 °F in July 1936.

2.3.2.2 Precipitation

The Madison area normally receives an annual average of 31.6 inches of precipitation. The monthly average precipitation data is reported in Table 2-3. The record maximum level of precipitation to fall in Madison in one year, is reported by the National Weather Service to be 52.9 inches in 1881. The greatest 24-hour rain fall total for Madison was 5.25 inches on July 15-16, 1950. The 48-hour, 100-yr. return period rainfall for south central Wisconsin is estimated to be 7.82 inches[8].

The annual average snow fall for Dane County is 43.7 inches. The monthly average snow fall data is reported in Table 2-4. The record maximum snow to fall during the winter season is reported by the National Weather Service as 76.1 inches during the winter of 1978-79. The 24 hours state record for heaviest snow occurred December 27-28, 1904 in Neillsville, Wisconsin in which 26 inches of snow fell.

TABLE 2-2
Average Temperatures for the Madison Area

	Average Monthly Temperature (°F)	Average Daily Maximum Temperature (°F)	Average Daily Minimum Temperature (°F)
January	16.8	25.7	8.0
February	21.3	30.6	12.0
March	32.3	41.9	22.7
April	46.1	57.6	34.7
May	57.5	70.0	44.9
June	67.0	79.4	54.5
July	71.4	83.3	59.5
August	69.2	81.0	57.4
September	60.4	72.3	48.6
October	49.5	60.9	38.2
November	35.3	44.0	26.5
December	22.6	30.7	14.4
Year	45.8	56.5	35.1

TABLE 2-3
Monthly Precipitation Data for the Madison Area

	Average Monthly Total Precipitation (inches)	Average Monthly Maximum Precipitation (inches)
January	1.14	2.45
February	1.14	2.77
March	2.17	5.04
April	3.02	7.11
May	3.14	6.26
June	3.83	9.95
July	4.05	10.93
August	3.90	9.49
September	3.12	9.22
October	2.29	5.63
November	2.15	5.13
December	1.66	4.09

TABLE 2-4
Monthly Snowfall Data for the Madison Area

	Monthly Average Total Snowfall (inches)	Monthly Maximum Snowfall (inches)	Year
January	10.3	31.8	1929
February	7.7	37.0	1994
March	8.5	28.4	1923
April	2.3	17.4	1973
May	0.1	5.0	1935
June	0		
July	0		
August	0		
September	Trace		
October	0.2	5.0	1917
November	3.8	18.3	1985
December	10.9	32.8	1987

2.3.2.3 Winds

The average annual wind speed in the Madison area is 9.8 mph. The prevailing winds during the months of November through March are from the west-northwest direction, the remaining months of April though October the prevailing winds are from the south[9]. Table 2-5 reports the frequency of surface wind direction versus wind speed. The record gust in the Madison area occurred in June 1975 when wind gusts were reported at 83 mph from the west.

TABLE 2-5
Frequency of Surface Wind Direction versus Wind Speed

	Speed (knots)											
Direction	1 - 3	4 - 6	7-10	11-16	17-21	22-27	28-33	34-40	41-47	>=48	Percent	Wind Speed (knots)
N	0.5	1.4	1.6	1.3	0.2	*	*	*	0	0	4.7	8.2
NNE	0.3	0.7	1.1	0.8	0.2	*	*	0	0	0	3.3	9.3
NE	0.6	1.1	1.4	1.1	0.2	*	*	0	0	0	4.6	9
ENE	0.6	1.7	1.7	1.2	0.2	*	*	0	0	0	5.5	8.2
E	0.5	1.1	1.4	0.9	0.1	*	*	*	0	0	3.8	7.8
ESE	0.3	0.9	1.2	0.7	0.1	*	0	0	0	0	3.2	8.5
SE	0.4	1.1	1.5	0.7	0.1	*	0	0	*	0	3.8	8.5
SSE	0.4	1.4	1.7	1	0.1	*	0	0	0	0	4.9	8.4
S	0.6	2.8	4.1	2.9	0.5	0.1	*	*	*	0	10.4	8.8
SSW	0.3	1.1	2.4	2.3	0.4	0.1	*	*	0	0	6.8	10.1
SW	0.3	1.1	2.4	2	0.4	0.1	*	*	*	0	6.5	10.3
WSW	0.3	1.1	2	1.5	0.3	0.1	*	*	*	0	5.5	10
W	0.4	1.6	2.5	1.9	0.5	0.1	*	*	0	0	6.7	9.5
WNW	0.4	1.9	2.9	2.6	0.6	0.1	*	*	0	0	8.6	9.7
NW	0.6	1.6	2.4	2.5	0.6	0.1	*	*	*	0	8.3	10.1
NNW	0.5	1.6	1.6	1.6	0.3	0.1	*	*	0	0	5.8	9
VAR	0	0	0	0	0	0	0	0	0	0	0	0
CLM	0	0	0	0	0	0	0	0	0	0	7.6	0
ALL	6.9	22.2	32.1	25.2	4.7	1	0.1	*	*	0	100	8.5

* = PERCENT < .05

2.4 Hydrology

Madison is located just northeast of the "Driftless Area" of southwestern Wisconsin. The glacially shaped topography of Madison and the surrounding area in central Dane County is irregular, ranging from flat or gently rolling to hilly. The most prominent geomorphic features were glacially formed. Among these are Lakes Mendota, Monona, Waubesa, Kegonsa and Wingra. Glacial drift covers the entire area except for local areas of bedrock outcrop[10].

In the vicinity of the reactor, a glacial deposit exists which contains clay and large boulders. Although this deposit may be as much as 100 feet thick, it is probably less than twenty. The reason for the variation in thickness is that the bed-rock sandstone which underlies the deposit is very uneven. The bed-rock consists of Cambrian sandstones which are 700 to 800 feet thick and which are permeable to water.

The Yahara River is the main drainage feature in the Madison area. The Yahara River gradient is essentially flat where it flows through the lakes in Madison. Water surface elevations are controlled mostly by dams at the outlets of lakes Mendota and Waubesa. The ground water flow from the reactor site is generally toward the Lake Mendota - Yahara River - Lake Monona system. Thus, the general flow is toward the east and south. Historical data obtained from the USGS Wisconsin Geological and Natural History Survey[11] indicated the annual average water table in the vicinity of the reactor is approximately 60 feet below the surface.

Madison obtains its drinking water supply from several deep well aquifers drilled, typically several hundred feet, into the Cambrian sandstone described above. The location of these wells is shown on **Figure 2-10** , and they supply the University as well as the city. All of these wells are cased from ground level into the sandstone so as to keep out ground water from the glacial deposit. The closest well to the reactor is Madison City Well 27 located 2,000 feet southeast.

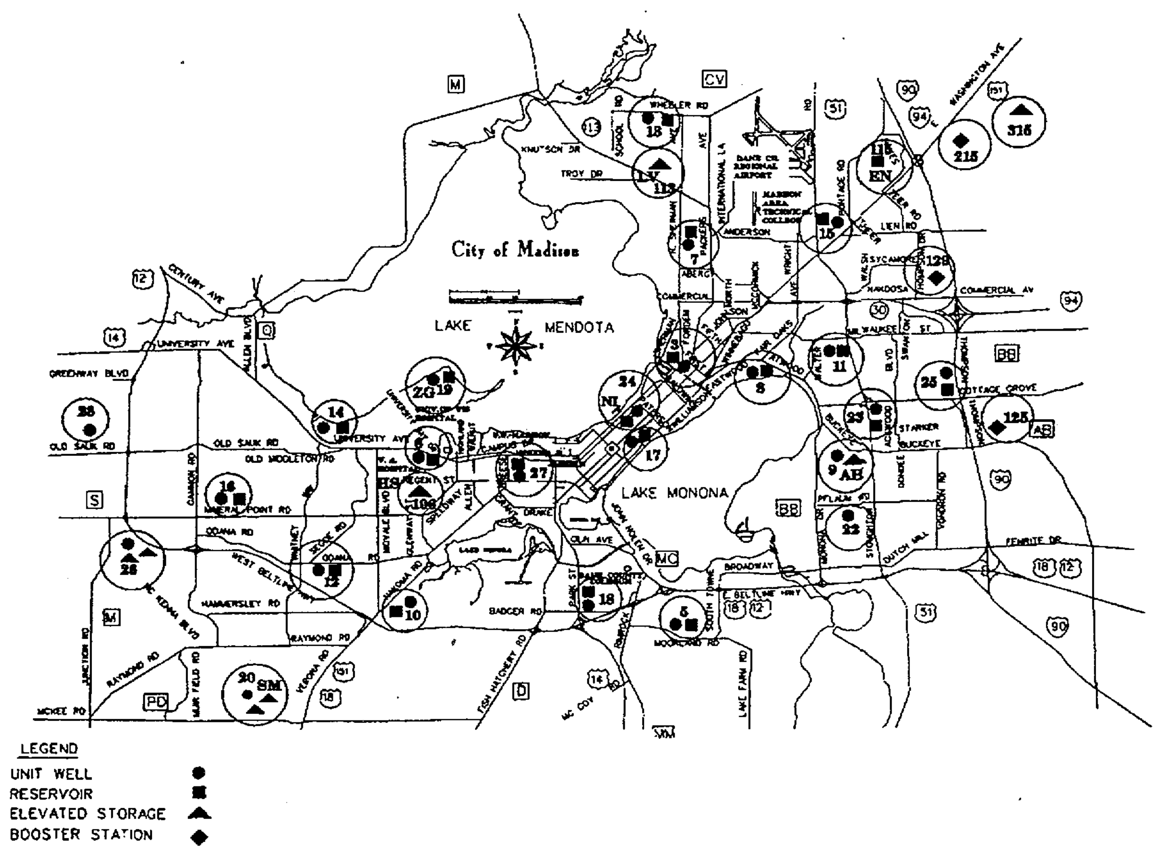

Figure 2-10 City of Madison Well Water Supply

Due to the large drainage capacity of the Yahara river and the outlet dams of lakes Mendota and Wausbesa flooding is not a serious problem in most of Madison. The 100 year flood is estimated to increase the surface level of Lake Mendota approximately 3 feet causing the lake to over flow its existing banks by about 30 feet[12].

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

The Midwestern United States does not lie on, or anywhere near, a tectonic plate boundary. The region is in the middle of the North American Plate, hundreds of miles from both the eastern and western edges. The Midwest, however, has a series of faults around the Mississippi Valley, **Figure 2-11** , the most active of which is the New Madrid Fault System. These faults were formed by the tearing open of the ancient continental crust almost 5 million years ago. This region is known as the New Madrid Seismic Zone which includes the states of Missouri, Arkansas, Louisiana, Illinois, Indiana and parts of Kentucky and Tennessee[13]. Wisconsin is not associated with this region. The New Madrid Fault System, located near New Madrid, Missouri is greater than 500 miles from Madison, Wisconsin and poses little or no seismic hazard.

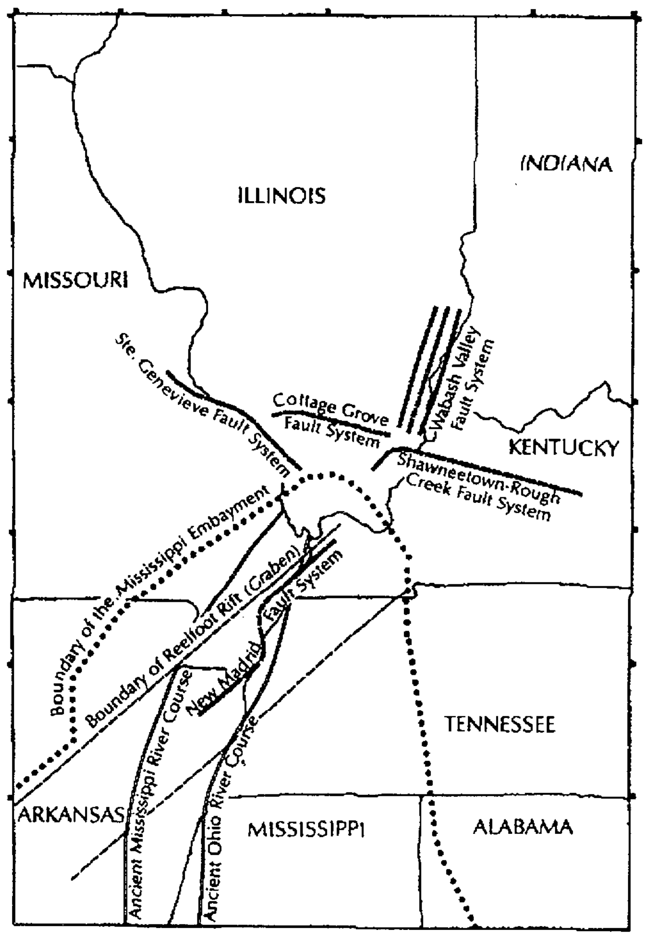

Figure 2-11 New Madrid Seismic Zone

2.5.2 Site Geology

As discussed in section 2.4 Hydrology, the local geology of the reactor site includes a layer of glacial deposits which rests on the Cambrian sandstone bedrock. The layer of the glacial deposit is variable, however, soil bores at the reactor site indicate this layer is approximately 16 feet deep, Table 2-6[14,15,16]. The Cambrian sandstone layer below the glacial drift is approximately 700 to 800 feet thick. Below the sandstone is impermeable basement rock. There are no geological structures of consequence in the vicinity of the reactor site.

TABLE 2-6
Depth of Glacial Deposit to Cambrian Sandstone

Distance from Reactor Site (Feet)	Depth of Glacial Deposit (Feet)
0	16
150	16.5
175	4
200	14
250	13.25
300	9.5
350	17.5
400	15
450	9.5

2.5.3 Seismicity

As discussed in section 2.5.1, Regional Geology, Wisconsin is located in a geologically stable region of the United States. The closest active fault is greater than 500 miles in the New Madrid Seismic Zone. While no seismic events have occurred at the reactor site, there are records of earthquakes occurring within 200 km of the reactor site. A review of the USGS earthquake database[17] for all earthquakes of modified Mercalli intensity greater then IV or magnitude greater than 3.0 which is within 200 km of the reactor site resulted in the data reported in Table 2-7.

TABLE 2-7
History of Seismic Events Within 200 km of Madison, Wisconsin.

Date	Magnitude (Richter)	Intensity (Modified Mercalli)	Distance (km)
August 20, 1804	4.4	VI	178
May 27, 1881	4.6	VI	198
May 26, 1909	5.1	VII	195
January 2, 1912	4.5	VI	190
November 12, 1934	4.0	VI	196
September 15, 1972	4.5	VI	158
September 9, 1985	3.0	V	178

2.5.4 Maximum Earthquake Potential

The determination of earthquake potential and frequency is based on data from previous events. Because there are few historic moderate to large earthquakes in the vicinity of the reactor, analysis is difficult. The maximum earthquake potential may be inferred from data supplied by the USGS for the 50 year peak ground acceleration estimate[18]. The estimated 50 year peak ground acceleration due to a seismic event in the vicinity at the reactor is less than 0.01 g.

2.5.5 Vibratory Ground Motion

Due to insufficient data from previous seismic events the vibratory ground motion can only be inferred from the peak ground acceleration data of section 2.5.4 of less than 0.01 g.

2.5.6 Surface Faulting

Based on the distance to any known active faults and the stable site geology, surface faulting is not considered to be a credible event in the vicinity of the reactor site..

2.5.7 Liquefaction Potential

Based on the distance to any known active faults and the stable site geology, the liquefaction potential is considered to be insignificant.

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3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Design Criteria

When the University of Wisconsin Nuclear Reactor was to be upgraded by increasing authorized power to 1000 kW the principal design criterion was to assure the facility could withstand loss of pool water and any other credible accident with no hazard to the public, without reliance on engineered safety features. This criterion was met by selecting stainless-steel-clad TRIGA fuel due to the well-documented characteristics of this fuel type[1]. Details of the physical mechanisms and characteristics that cause TRIGA and FLIP fuel to exhibit the prompt negative temperature coefficient responsible for the fuel characteristics are given in the reference and a number of other documents and are not included here. Detailed analysis for this facility (Chapter 13 of this report) agree with the conclusions in the reference. The design criteria that result in this negligible safety risk are the result of the fuel composition and cladding, not of specific features provided in the equipment and building that surrounds the reactor.

The Mechanical Engineering Building which houses the reactor laboratory was completed in 1930. Extensive remodeling of the room that became the Reactor Laboratory took place in 1960 when the reactor was installed. The original reactor installation used fuel and components manufactured by General Electric, and the specifications to which structures were built were those stated by General Electric. Specific design criteria were not stated. The ventilation system was designed in 1962 and installed in 1963. The design specifications stated only the desired flow rates and stack height. The last major upgrade was the addition of the cooling system which was designed in 1966 and installed in 1967. The only design criterion was the heat removal rate required. During this upgrade, the N^{16} diffuser system was also installed. This system was designed and fabricated by General Atomic to their design specifications. Conversion to TRIGA fuel took place in 1969, and the auxiliaries for pulsing operation (transient rod and drive) were designed and built by General Atomic to their specifications. All building modifications and equipment additions were in conformance with the building codes in existence at that time.

3.2 Meteorological Damage

There are no design criteria for the protection of facility structures from meteorological conditions except that all facility structures were constructed to applicable building codes in existence at the time. The Reactor Laboratory has endured approximately 40 years of Wisconsin weather with no meteorological damage. Furthermore, no facility structures are assumed to be operable in this SAR for the mitigation of any accident (see Chapter 13, Accident Analysis).

3.3 Water Damage

There are no design criteria for the protection of facility structures, systems and components from water damage. The possibility of flooding due to the lake system of Dane County is considered insignificant due to the distance of the Reactor Laboratory to the Lake Mendota flood plain as described in Chapter 2, section 2.4, Hydrology. Furthermore, no facility structures, systems and components susceptible to water damage are assumed to be operable in this SAR for the mitigation of any accident (see Chapter 13, Accident Analysis).

3.4 Seismic Damage

There are no design criteria for the protection of facility structures, systems and components from seismic damage except that all facility structures were constructed to applicable building codes in existence at the time. The probability of a seismic event in the vicinity of the reactor site is considered insignificant due to the stable regional geology (see Chapter 2, section 2.5, Geology, Seismology and Geotechnical Engineering). Furthermore, no facility structures, systems and components, including the reactor pool, susceptible to seismic damage are assumed to be operable in this SAR for the mitigation of any accident (see Chapter 13, Accident Analysis).

3.5 Systems and Components

At the time of original construction of the Reactor Laboratory, design bases were not provided by General Electric for facility systems and components. With the upgrade to TRIGA and TRIGA-FLIP fuel, accident analyses, including NUREG/CR 2387 and Chapter 13, show that by the design of TRIGA fuel, reliance upon other systems, structures and components are not necessary to ensure safety of the general public. Therefore, with the exception of the fuel, no other facility structure, system and component are assumed to be operable in this SAR for the mitigation of any accident.

Nevertheless, experience gained on facility systems and components over many years of operation have shown these systems to be highly reliable. Descriptions of system design and operation of these systems are discussed in the succeeding chapters of this SAR.

3.6 References

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4 REACTOR DESCRIPTION

4.1 Summary Description

4.1.1 Introduction

The reactor was constructed and installed by the Atomic Power Equipment Department of the General Electric Company. The present modification employs a core composed of TRIGA-FLIP fuel supplied by the General Atomic Company.

Initial criticality was achieved on 26 March 1961. The original maximum steady state power level was 10 kW. Power was increased to 250 kW on 7 December 1964 and again increased to the present maximum steady state power level of 1,000 kW on 14 November 1967. Operation with FLIP fuel began in March 1974 with a mixed core containing 9 FLIP bundles. In January 1978 an additional 6 FLIP bundles were added. In August 1979 the conversion to FLIP fuel core was completed.

Figure 4-1 is a pictorial cutaway view of the reactor. The reactor is a heterogeneous pool-type, fueled with TRIGA or TRIGA-FLIP fuel which is cooled by natural convection. The fuel is currently all 70% enriched in Uranium U^{235}, although 20% enriched fuel can also be used. Light water acts as both coolant and moderator as well as being a biological shield. The core is reflected on two sides by graphite and on two sides by water, the water-reflected areas being utilized as irradiation facility locations. The top and bottom reflector region is partially graphite and partially water.

A 7-by-9 grid, surrounded by a core box, positions fuel, reflectors, control elements, and irradiation facilities. Core reactivity is changed and controlled by three shim safety blades, a regulating blade, and a transient control rod. All control elements move vertically in shrouds positioned in the core box or inside a fixed tube as is the case of the transient rod. A suspension frame supports the grid box and control element drive mechanisms. The suspension frame, in turn, is supported by the reactor bridge.

Cold, clean core excess reactivity in the present operational core is about 4.2 % reactivity. Control elements (control blades and the transient rod) provide a shutdown margin of about 4 % reactivity.

[redacted]

Figure 4-1 University of Wisconsin Nuclear Reactor (UWNR)

4.1.2 Summary of Reactor Data

Responsible Organization	The University of Wisconsin
Location	Madison, Wisconsin
Purpose	Teaching and Research
Fuel	
Type	TRIGA High Hydride in 4 element clusters
Number of elements in standard 1000 kW core	[redacted]
Control	
Safety elements	Three vertical blades
Regulating-servo element	One vertical blade
Transient control	One rod
Experimental Facilities	
Thermal Column	One, 40-inch square graphite, φ th = 2×10^8
Beam Ports	Four, 6-inch diameter φ th = 1 - 3×10^{10} nv at shield side of shutter; about 8×10^{11} at core end of port
Pneumatic tube	One, 2-inch (sample size 1-1/4 inch diameter by 5-1/2 inches long), φ th = 5×10^{12} nv

Thermal neutron fluxes for isotope production include the above, plus large irradiation spaces outside the core with thermal neutron fluxes of around 1.3×10^{13} nv.

Reactor Materials

Fuel - moderator material	U-Zr $H_{1.6}$

U^{235} enrichment	70%
U^{235} content/element (average)	[redacted]
Burnable poison	1.5wt.% Natural Erbium
Cladding	20 mill stainless steel
Reflector	Water and graphite
Coolant	Light water
Control	Boral & stainless steel; borated graphite for transient rod
Structural material	Aluminum
Shield	Concrete and water

Dimensions

Pool	8 x 12 x 27-1/2 ft. deep
Standard 1000 kW Core	15 x 17 x 15 inches high
Grid box	9 x 7 array of 3 inch modules
Fuel element	
Diameter	[redacted]
Length	[redacted]
Active Length	[redacted]

Nuclear characteristics

1 MW Steady State:

Maximum thermal neutron flux	3.2×10^{13} nv
Maximum fast neutron flux	3.0×10^{13} nv

1000 MW Pulse	
Maximum thermal neutron flux	3.2×10^{16} nv
Maximum fast neutron flux	3.0×10^{16} nv
Core Loading (Standard 1000 kW core)	[redacted]
Operating excess reactivity	~4% ΔK_{eff}
Minimum reactivity in safety blades	6.9% ΔK_{eff}
Average prompt temperature coefficient of reactivity	-1.26×10^{-4} ΔK/°C
Void coefficient of reactivity	$-.2 \times 10^{-4}$ ΔK/% void
Prompt neutron lifetime	2.4×10^{-5} second
Effective delayed neutron fraction	0.007

4.1.3 Experimental Facilities

Facilities are provided to permit use of radiations from the reactor in experimental work without endangering personnel. These facilities include three hydraulic irradiation facilities ("whales"), four beam ports, one thermal column, and a pneumatic transfer system ("rabbit").

Hydraulic Irradiation Facility (Whale)

Aluminum pipes of 2-7/16" internal diameter extend from approximately 18" below the pool surface to grid box positions on the periphery of the core. These pipes draw sample bottles made of polyethylene down and position them approximately at the center line of the fuel. Two sample containers can be loaded in each tube. The addition of a second sample bottle, however, causes the natural rotation of the first bottle to stop. Thermal Neutron Fluxes in these positions are approximately 10^{13} nv.

Irradiations are also conducted in the other reflector regions surrounding the core. Irradiation baskets may be inserted in any vacant grid position, and irradiations can also be conducted outside the grid box in specially fabricated enclosures.

Thermal Column

The thermal column is a graphite-filled horizontal penetration through the biological shield which provides neutrons in the thermal energy range (about 0.025 eV) for irradiation experiments. The column, which is about 8 feet long, is filled with about 6 feet of graphite. A small experimental air chamber (40" x 40" x 24") between the face of the graphite and the thermal column door has conduits for service connections (air, water, electricity) to the biological shield face. The compensated ion chambers for the safety and logN instrumentation channels are located in the thermal column.

Personnel in the building are protected against gamma radiation from the column by a dense concrete door which closes the column at the biological shield. The door moves on tracks set into the concrete floor perpendicular to the shield face.

Beam Ports

Four 6-inch beam ports penetrate the shield and provide fluxes of both fast and thermal neutrons for experimental use. The ports are air filled tubes, welded shut at the core ends and provided with water-tight covers on the outer ends. The portions of the ports within the pool are made of aluminum, while the portions within the shield are steel.

A shutter assembly, made of lead encased in aluminum, is opened for irradiations by a lifting device. When closed, the shutter shields against gamma rays from the shut-down core, allowing experiments to be loaded and unloaded without excessive radiation exposure to personnel.

Shielding plugs are installed in the outer end of each port. The plugs, made of dense concrete in aluminum casings, have spiral conduits for passage of instrument leads.

Pneumatic Tube

A pneumatic tube system conveys samples from a basement room to an irradiation position beside the core. The "rabbits" used in the system will convey samples up to 1-1/4 inches diameter an 5-1/2 inches long. The system operates as a closed loop with carbon dioxide cover gas limiting generation of Ar^{41} activity.

The reactivity effect from a sample in the pneumatic tube is restricted to less than 0.2% ρ. Tests run with water and cadmium samples indicate that sample reactivity effects will normally be less than 0.01% ρ. Static reactivity measurements will be run for samples of fissionable material or particularly strong absorbers such as some of the rare earths.

4.2 Reactor Core

The reactor may be operated with either standard (20% enriched) or FLIP (70% enriched) fuel as described in section 4.2.1. In addition, mixed cores containing fuel of both types may be loaded. Since funding is not presently available for replacing the FLIP fuel with LEU with burnable poison, the core that will usually be operated is composed only of FLIP fuel [redacted]. The timing of funding for LEU fuel is unknown at present. If funding for converting the entire core is received it may be necessary to revert to cores of mixed standard and FLIP cores before loading a new core [redacted] If funding for LEU replacement is received a few fuel bundles at a time, reversion to a mixed standard/FLIP core may not be required, but further analysis of cores of mixtures of FLIP and the new LEU fuel would be required. Such cores are not considered in this SAR.

The use of the reactor as a training and research tool requires flexibility of core arrangement. These arrangements are subject, however, to the following criteria:

a. A mixed core must contain at least 9 FLIP bundles.
b. Any FLIP fuel must be located in a central contiguous region.
c. The core must be a close packed array except for single fuel element (not fuel bundle) positions or grid positions on the core periphery.
4. Calculations indicate that operation will be within safety limits on power generation per element and fuel temperature.

4.2.1 Reactor Fuel

The fuel is of the TRIGA four-element bundle type developed to provide a simple means of converting reactors using flat-plate fuel to TRIGA reactors. A variant bundle, called a three-element bundle, has only three fuel elements installed; the fourth space is used for an aluminum control rod guide tube, an aluminum irradiation thimble, or an instrumented fuel element. **Figure 4-1** shows a four-element bundle, a three-element bundle containing a control rod guide tube, and a three-element bundle containing an instrumented fuel element (the conduit for the thermocouple leads is shown cut short)..

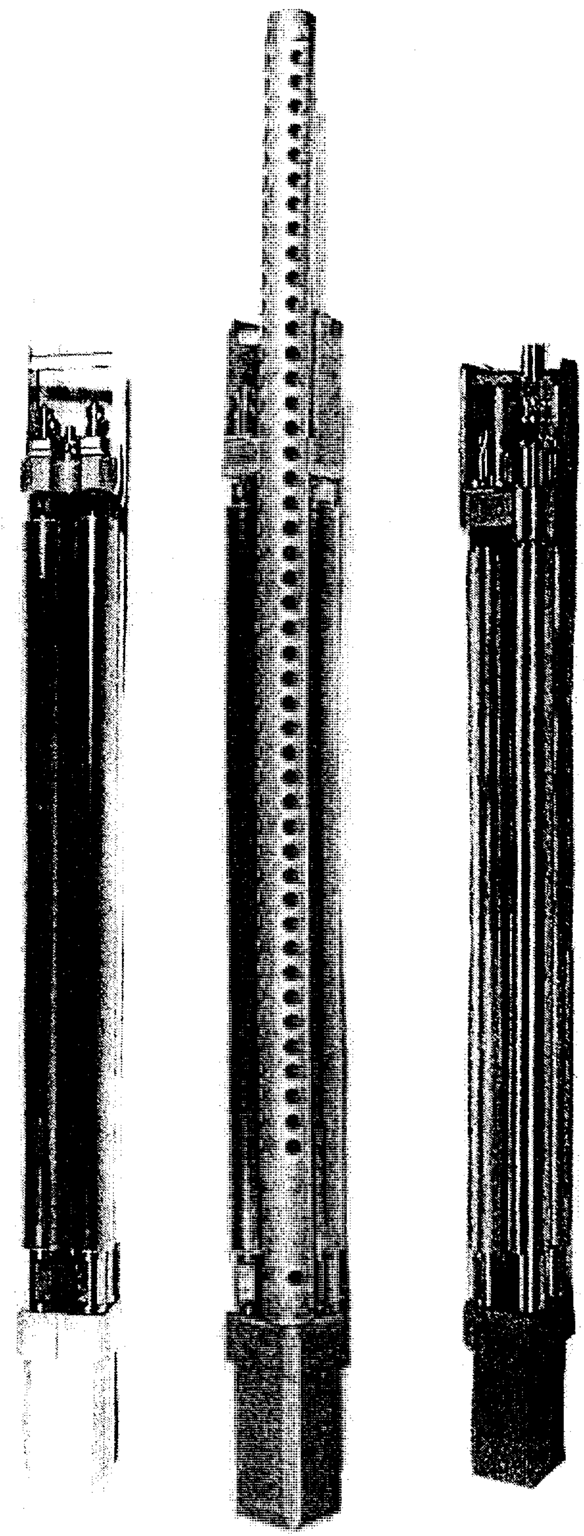


Figure 4-2 Fuel Bundles

The four-element bundle (See **Figure 4-3**) consists of bottom adapter, top adapter, and four TRIGA elements. The bottom adapter of the bundle fits the existing grid plate as did the original flat-plate fuel elements. The end fittings on individual TRIGA elements are threaded into the bottom adapter until a flange on the element seats firmly against the adapter, providing rigid cantilever-type support. The top adapter serves both as a handling fitting and as a spacer for the upper ends of the fuel elements. A sliding fit between this adapter and the fuel element end fittings allows for differential expansion of the elements. This top fitting can be removed with remote handling tools to disassemble the bundles for replacement of individual fuel elements or for shipping spent elements for reprocessing.

The individual fuel elements (**Figure 4-4**) are quite similar to the TRIGA elements used on TRIGA reactors using the standard TRIGA grid plates. The differences are (1) reduction of diameter fro[redacted] to maintain the proper metal-to-water ratio in this core; (2) the bottom end fixture is threaded; (3) flats on the stainless steel bottom end fixture provide wrench surfaces for disassembly without stressing the cladding; and (4) the top end fixture is modified to allow the top end fitting to be locked in place.

The TRIGA elements used at UWNR are of two types, standard and FLIP. Both have outside dimensions, clad thickness, and construction as shown in **Figure 4-4** . The two types differ as shown in the following table:

Design Parameter	Standard Fuel	Flip Fuel
Fuel moderator material	U-Zr $H_{1.7}$	U-Zr $H_{1.6}$
U^{235} enrichment	20%	70%
U^{235} content/element (average)	[redacted]	[redacted]
Burnable Poison	None	Natural erbium
Erbium content	---	1.5 wt %

The FLIP fuel was designed to extend the lifetime of TRIGA fuel, and was used in step-wise additions of fuel to the University of Wisconsin Nuclear Reactor. The reactor is currently operating with a core consisting entirely of FLIP fuel. Fuel bundles contain only one type of fuel, and the top adapters for FLIP fuel bundles are marked (by notches machined into the top of the top adapter) to facilitate identification during underwater fuel handling.

[redacted]

Figure 4-3 Four-element Bundle Assembly

[redacted]

Figure 4-5 shows an instrumented element. This element is fitted with three thermocouples. At least one such element (inserted into the vacant position of a three-element bundle) is included in every core. The sensing tips in the thermocouples are located at the vertical centerline of the fuel section and one inch above and below the centerline. The thermocouple leads pass through a seal in a stainless steel tube which provides a water-tight conduit carrying the lead-out wires above the surface of the pool water.

[redacted]

Figure 4-5 Instrumented Fuel Element

4.2.2 Control Elements

Both blade and rod shaped control elements are used.

Control Blade Shrouds and Guide Tubes

Each blade type control element, both safety and regulating, is guided throughout its travel by a shroud as shown in **Figure 4-6** . The shroud consists of two thin aluminum plates 38 inches high, separated by aluminum spacers to provide a 1/8-inch water annulus. The shrouds can be removed, if necessary, by use of a grapple hook. Small flow holes at the bottom of the shroud minimize the effect of viscous damping on the scram time.

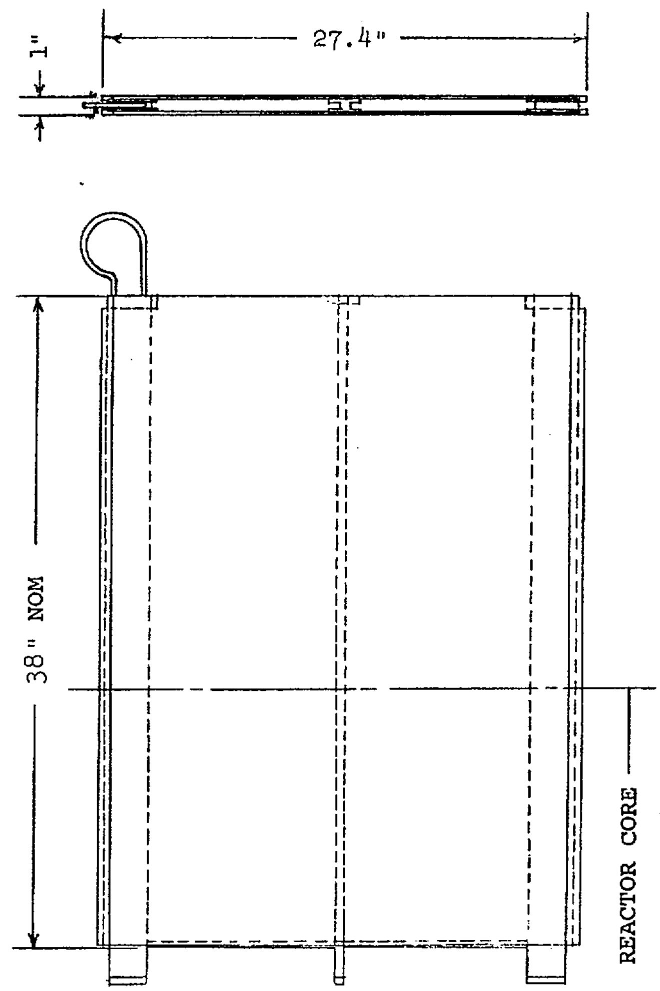


Figure 4-6 Control Blade Shroud

Rod shaped control elements are guided by a guide tube as shown in one of the three-element bundles of **Figure 4-1** . Holes drilled in the sides of the guide tube allow for water displacement when the control rod is fired out during pulsing operation or dropped in response to a scram condition.

Safety Blade

Reactor control for startup and shutdown is accomplished by three blade-type control elements, **Figure 4-7** , with a total shutdown worth between 6.9 and 11 per cent ΔK_{eff}. The poison section is boral sheet (boron carbide and aluminum sandwiched between aluminum side plates). It is 40.5 inches long. When the blade is full in the bottom of the blade overlaps the bottom of the active fuel by 1.5 inches.

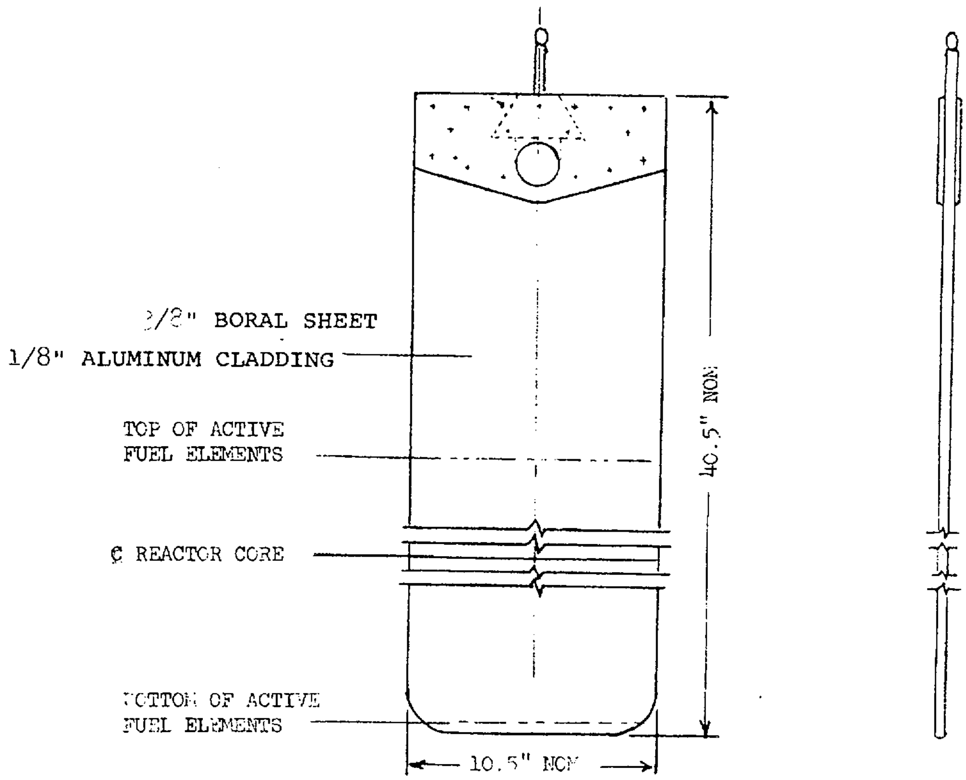


Figure 4-7 Shim/Safety Blade

Regulating Blade

The regulating blade, **Figure 4-8** (shown upside down for ease in reading the dimensions), is a stainless-steel sheet with curls on the vertical edges, about 11 inches wide and 40 inches long, supported and guided in the same manner as the safety blades. It is used to compensate for small changes of reactivity during normal reactor operation and may be actuated by a servo-control channel.

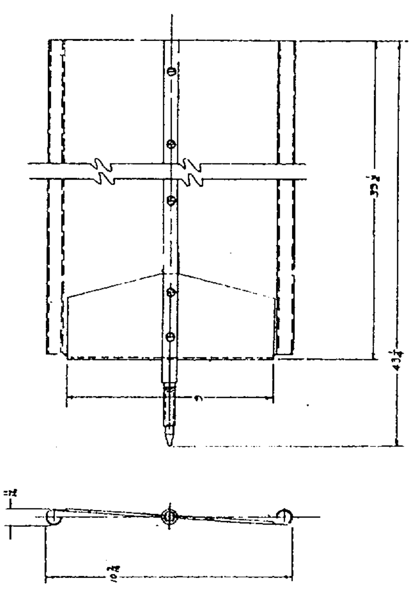


Figure 4-8 Regulating Blade

Transient Control Rod

The transient control rod is boron carbide or borated graphite contained in a 1.25 inch diameter stainless steel or aluminum tube (**Figure 4-9**). The poison section is approximately 15 inches long. This rod is guided laterally by the aluminum guide tube in a special three-element fuel bundle. A hold-down tube extends from this guide tube to the top of the reactor structure and holds the three-element bundle in place during transient rod movement.

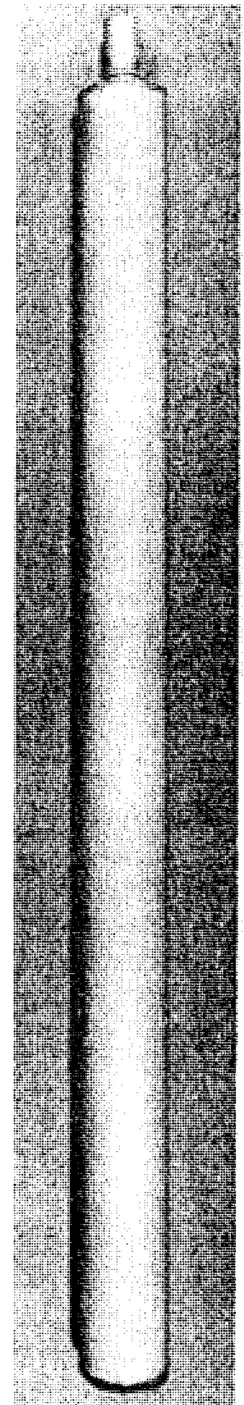


Figure 4-9
Transient Control Rod

4.2.3 Neutron Moderator and Reflector

Pool water serves as moderator for the core and as reflector above, below, and on those sides of the core not provided with graphite reflectors or special reflector elements designed to condition the quality of a beam being extracted from the core. (Individual fuel elements contain an internal 3.5 inch long graphite end reflector above and below the fueled portion, so the core top and bottom are actually reflected by a mixture of graphite and water.)

The reflectors are standard General Electric Company reflectors furnished at initial startup of UWNR. The nominal 3-inch square reflector elements are made of AGOT grade graphite clad with aluminum (**Figure 4-10**). Reflector element lifting handles are diagonal to facilitate identification when viewing the core and storage racks.

Special reflectors are the same size as the graphite reflectors, but may consist of solid aluminum, hollow aluminum, or combinations of graphite sections with the center portion replaced with solid aluminum, voids, or gamma absorbers such as lead or bismuth. Such special reflectors are used for irradiation facilities or to adjust the mix of thermal, epithermal, and fast neutrons transmitted to experimental facilities.

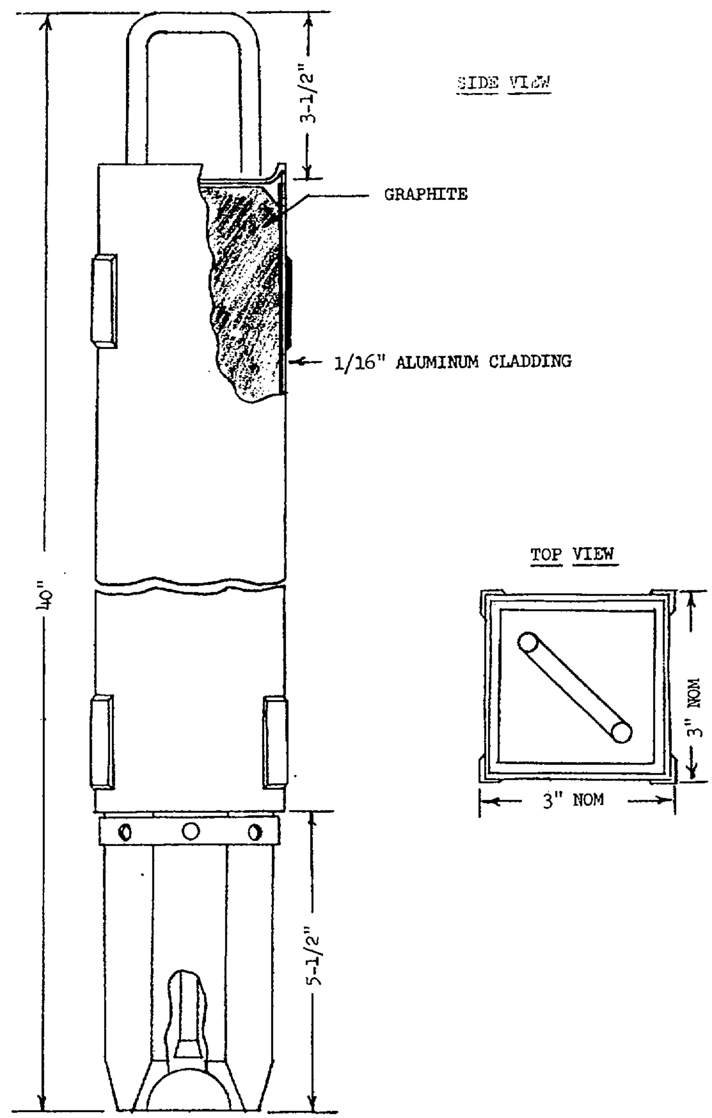


Figure 4-10 Graphite Reflector Element

4.2.4 Neutron Startup Source

The neutron source is a 100 mg radium-beryllium source irradiated to give an output greater than 10^7 neutrons/second. It is encapsulated in a 0.515 inch diameter by 3.10 inch long stainless steel welded cylindrical capsule, which in turn contains two 1.25 inch long welded stainless steel capsules.

The source fits into a source holder (**Figure 4-11**).

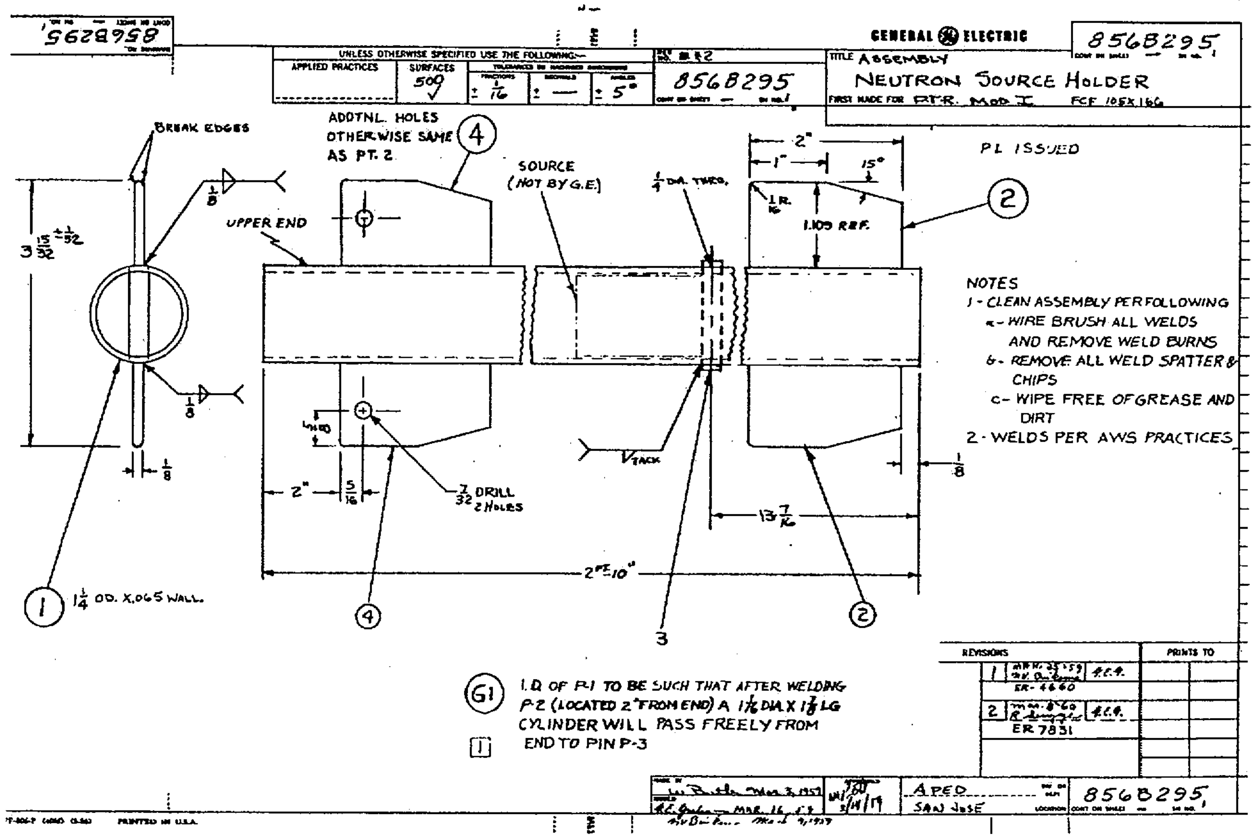


Figure 4-11 Neutron Source Holder

The source holder, in turn, fits into an irradiation basket (**Figure 4-12**) occupying one grid module adjacent to the active core. (The irradiation basket is shown upside down for ease in reading the lettering). The source is usually left in for full power operation, and will, with the normal operating cycle, maintain its output of about 10^7 neutrons/second. If the source is not left in during full power operation, neutron emission rate will decrease over a long time period to approximately 10^6 neutrons/second.

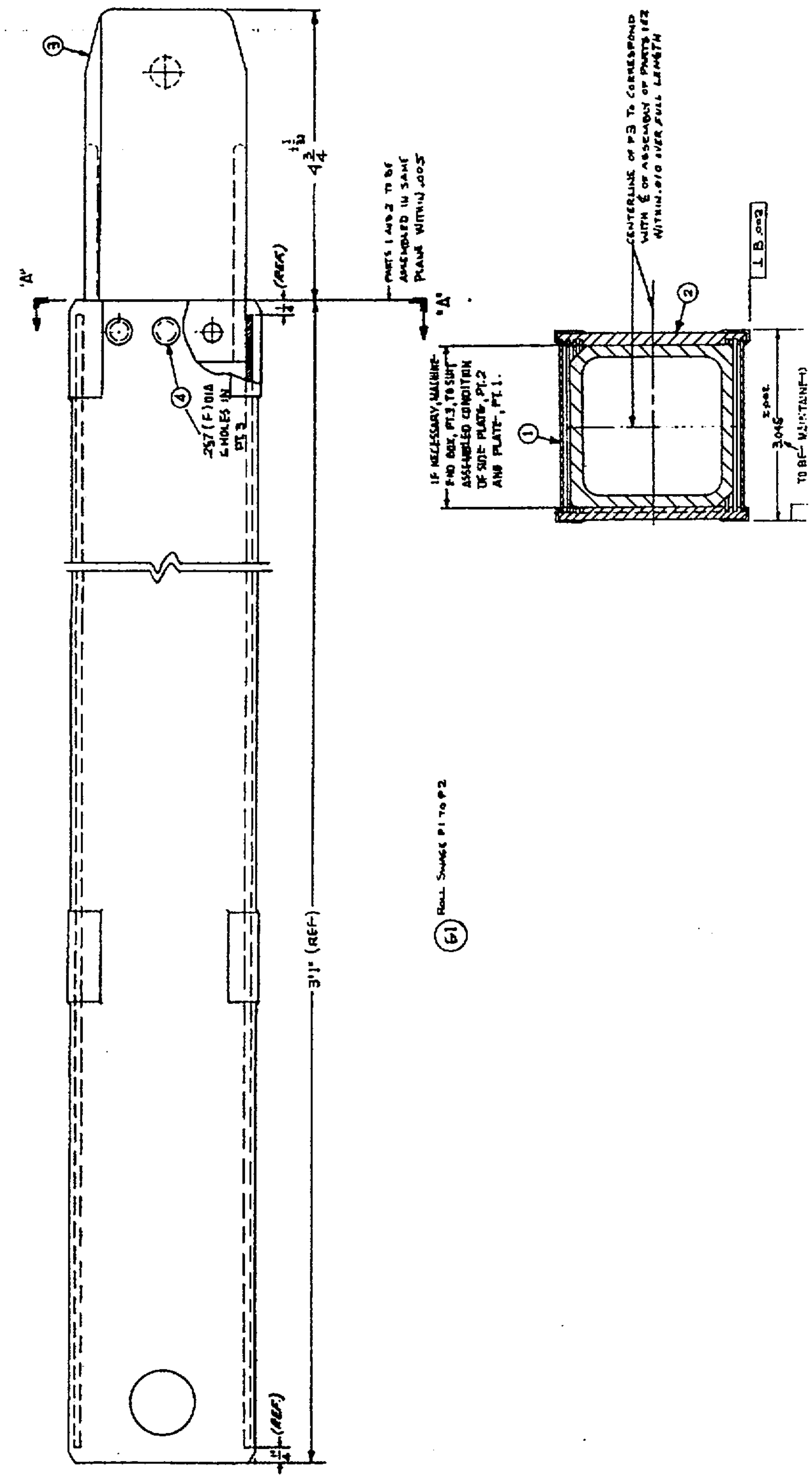


Figure 4-12 Irradiation Basket

4.2.5 Core Support Structure

The core is suspended from an all-aluminum frame, **Figure 4-13** , which extends from the grid box to a height of about one foot above the pool surface. One of the hollow corner posts of the suspension frame serves as a guide for the gamma chamber used in pulsing operation. The other three corner posts may also be used to position detectors in positions above the core.

The reactor bridge (mounted over the pool) supports the core suspension frame. The all-steel, prefabricated bridge was bolted together in the field and aligned with shims.

A locating plate, made of 0.5-inch steel, spans the upper end of the suspension frame. It is bolted to the bridge and aligns the four control blade drive mechanisms and the transient rod drive with the core. The five mechanisms work through individual clearance holes, each mechanism being secured to the locating plate. The plate and mechanisms are not removable as a unit to prevent accidental withdrawal of the control elements. The fission counter drive is mounted on a portion of the hand railing support structure.

Four 4-inch square 6061 aluminum suspension tubes (0.25 inch wall thickness) extend from the bridge to the grid box, and support the grid box by bolted connections. The aluminum grid box (**Figure 4-14**) encloses and supports the 6-inch thick cast aluminum grid plate which is machined to locate and support the control element shrouds and bottom end fittings of fuel bundles, reflectors and in-core experimental facilities such as hydraulic irradiation positions and irradiation baskets. **Figure 4-15** shows the grid position designation system, location of experimental facilities and radiation detectors relative to the grid box, and letter and number codes used in later descriptions to identify fuel and reflector reactivity worths.

An aluminum coolant header (not shown in) mates with the bottom of the grid box and forms a transition to the coolant piping originally provided (but not used) for future use with a forced convection cooling system. An opening in the side of the header, 24 inches wide by 12 inches high, allows cooling water flow for natural convection.

A diffuser pump and jet above the core deflects the cooling water streaming from the core to reduce N^{16} activity above the core.

[redacted]

Figure 4-13 Core Suspension

[redacted]

Figure 4-15 Grid Arrangement With Fuel and Reflector Codes

4.3 Reactor Pool

The aluminum-lined concrete pool, **Figure 4-16** , is 8 feet wide, 12 feet long, and 27 1/2 feet deep. The reinforced concrete pool walls also serve as the biological shield (further details on the pool walls are found in the next section). The pool is penetrated by experimental ports. Piping systems connecting with the pool are discussed in detail in Chapter 5. In summary, all piping connections are built to preclude accidental loss of pool water by failure of components located outside of the pool. When possible, pipes enter and leave the pool above the water surface (primary cooling system, diffuser system, and hydraulic irradiation tube water system) and are equipped with passive siphon breakers that prevent loss of more than a few inches of water even in the event of a pipe break or system mis-operation.

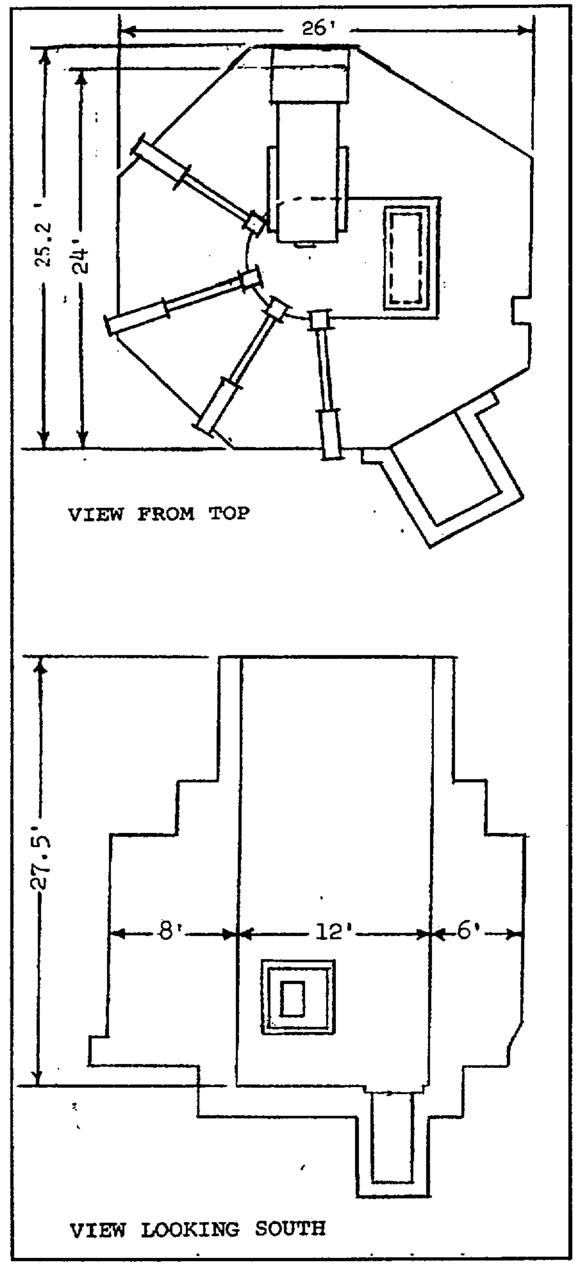


Figure 4-16 Reactor Pool and Shield

Two 8 inch aluminum pipes intended for use in a forced-convection cooling system were imbedded in the concrete at initial construction. (See **Figure 5-2**, especially the note concerning anti-siphon loop). One of these pipes penetrates the pool wall about 14 feet below the pool curb, but is closed with flanges on both the inside and outside ends, preventing loss of pool water unless both flanges fail. The other 8 inch pipe was looped inside the concrete and equipped with a siphon breaker that extends from the top of the loop to above the pool curb. One end of the loop is flanged closed outside the shield, while the other end vertically penetrates the bottom of the pool within the coolant header (below the grid box). If the outer flange of this pipe were to fail, this pipe could drain the pool to 14 feet below the pool curb. At original construction, a 2 inch pipe, also embedded in the concrete shield, was provided to serve as a pool drain. This pipe connects to the looped 8 inch pipe on the pool side of the loop, thus eliminating the loop and the siphon breaker protection. The return water flow from the pool makeup and purification system once entered this pipe. Because the potential for inadvertent loss of pool water through this 2 inch pipe was recognized many years ago, changes were made to eliminate this potential leak path. The return line for the pool makeup and purification system was re-routed to enter the looped 8 inch line outer flange, thus providing siphon breaker protection for this return line (this line is also equipped with a check valve that precludes loss of pool water for a break in the demineralizer piping). The 2 inch line was then sealed with two leak-tight expanded aluminum plugs located within the concrete shield, and capped where it exits the shield.

The pool makeup and purification system supply pipe penetrates the pool 38 inches below the pool surface, thus limiting water loss to that level reduction upon failure. The discharge pipe from this system is discussed in the paragraph immediately above.

The 4 beam ports penetrate the pool wall at mid-core level. The beam ports are operated with a watertight flange on the outside end which will prevent leakage should the in-pool portion of the beam port fail. The in-pool portion of the tubes are aluminum pipes with welded end closure and bolted flanged connection to the beam port shutter assembly. The four beam ports have a common drain system, but the discharge valve for the drain system is maintained closed during operation. The beam ports also have vent connections which connect to the Beam Port and Thermal Column Ventilation system (see Chapter 9, Section 9.1). The vent connections are equipped with valves (normally kept open for ventilation flow), but which may be closed if a beam port leak occurs. Finally, each beam port vent has a check valve which allows air flow only out of the vent, thus preventing pressure differences between the beam ports from causing circulation between beam ports.

The thermal column case also penetrates the pool wall. It is of welded aluminum construction and has no valves or flanges which could be opened to drain the pool. Thus, the experimental facilities are the only penetrations which could, upon failure, cause loss of pool water sufficient to expose the core.

A fuel storage pit, [redacted] for storage of fuel elements awaiting return to a DOE facility and for storage of fuel in a shielded location, should maintenance of the in-pool portions of the reactor be necessary. Further information on irradiated fuel storage can be found in section 9.2 of this report.

The pool water is kept within the following limits:

Temperature (at Core cooling water inlet)	<130°F
Resistivity	>2 x 10^5 0hm - cm
Radioactivity	<10CFR Part 20 Appendix B Table 3 values for radioisotopes with >24 hour half-life

The reactor core is cooled by natural convection of pool water through the core. The 130°F temperature limit is imposed by demineralizer resin tolerance and by humidity control considerations.

The resistivity limit is set to reduce corrosion effects, extending the expected lifetime of the fuel elements and controlling water radioactivity. Routine checks of resistivity are made to determine the necessity of regenerating the demineralizer.

The radioactivity of the pool water is continuously monitored by an area monitor station located near the demineralizer. Should the pool water reach the activity limit above, the reading on this area monitor will increase during periods when the reactor is not operating. In addition, water samples are routinely analyzed for activity by other methods which give a more exact identification of quantity and type of activity present.

No problem has been experienced in maintaining pool water radioactivity below the indicated limits in nearly 40 years of operation.

4.4 Biological Shield

The reactor is shielded by concrete and water (See **Figure 4-16**). At normal pool level the core is covered by 20 feet of water. The shield at core level consists of about 3 feet of water [redacted] Denser concrete is used in the thermal column door and beam port plugs. Calculations and measurements of radiation levels for 1000 kW operation are (excepting N^{16} activity) discussed below:

Surface of shield, excepting beam port and thermal column openings - less than 1.5 mrem/hr.
Pool surface (leakage radiation) (No N^{16}) less than 15 mrem/hr.

"Hot spots" - measurements have shown that higher radiation levels exist around the beam ports and thermal column. Measurements of the maximum radiation levels at these "hot spots" at 1000 kW are about 10 mrem/hr around the beam ports and 40 mrem/hr at the hottest spot around the thermal column door. The dose one foot away from the hot spots is about 5 mrem/hr.

Since the third-floor classrooms and offices are above the level of the pool curb, an analysis of the effect of water loss on persons in these areas with partial loss of water from the pool was performed and reported to NRC to support changes in the Emergency Plan[1]. It was assumed that pool water level dropped to 15 feet below the pool curb one hour after the reactor was shut down from full power. Even with this large loss in water level, the third-floor level of the Mechanical Engineering Building is still shielded [redacted] Thus, only scattered radiation from the Reactor Laboratory roof would give a significant dose rate to these areas. The dose rate 15 feet below the normal pool level was measured with an underwater detector, and would result in a 24-hour dose at the pool curb of 219 mrem, while the dose at the roof surface in the same time period would be 120 mrem. Scattered dose from the roof structure would result in a dose of no more than 0.1 mrem in the third floor of the Mechanical Engineering Building in the 24 hour period.

Pool Surface Radiation Levels - N^{16} Activity

The radiation level due to N^{16} activity at the pool surface directly above the core when operating at 1000 kW would range from 80 to 220 mrem/hr if no N-16 control system were in operation (variability is due to changes in surface flow patterns). The N-16 diffuser system is normally in operation, however, reducing the dose rates at the pool surface to 2-4 mrem/hour at the pool surface. These radiation levels are low enough that no hazard will exist to personnel outside the Reactor Laboratory or in normally occupied levels within the Reactor Laboratory. Radiation levels on the walkway surrounding the pool are around 20 mrem/hr while the reactor is operating at 1000 kW without the diffuser operating and <0.5 mrem/hour with the diffuser operating.

All of the Reactor Laboratory outside of the console area is posted as a radiation area and a radioactive materials area. A cable and switch arrangement is positioned on the north stairway to the pool surface so that an alarm will be sounded should entry to that area be made while the reactor is operating, thus assuring that personnel will not enter the area without knowledge of the reactor operator.

The south stairway, leading from the console area to the pool surface does not have a cable and switch arrangement as does the north stairway. Access to these stairs is gained only through the console area and is well monitored. No difficulty has been experienced in maintaining radiation does to individuals well below those doses permitted in 10 CFR 20.

Heating Effects in Shield and Thermal Column

Heating effects caused by absorption of gamma radiation and fast neutrons are within allowable limits. For all calculations, it was assumed that the pool water was at the 130° temperature limit, and the reactor was operated continuously at 1.5 MW.

The heating in the concrete shield is approximately 20% of the maximum suggested by Rockwell[2]. Analysis of the heating rate in the lead shield for the thermal column indicates that the maximum temperature of the lead will be less than 217°F. Calculation of the graphite temperature in the thermal column indicates a maximum of 244°F.

4.5 Nuclear Design

4.5.1 Normal Operating Conditions
4.5.2 Reactor Core Physics Parameters

Note: These two sections are combined to enable concise inclusion of the measured core parameters of the several cores which have been operated under the license.

Core Arrangements

The use of the reactor as a training and research tool requires flexibility of core arrangement. Permitted arrangements are subject, however, to the following criteria:

a. A mixed core must contain at least 9 FLIP fuel bundles (clusters)

b. FLIP fuel must be located in a central contiguous region
c. The core must be a close packed array except for single fuel element positions or fuel bundle positions on the core periphery
d. Calculations indicate that operation of a specific core will be within technical specification limits on power generation per element and fuel temperature.

When the Safety Analysis Report for converting UWNR from flat-plate to TRIGA fuel was written, expected performance was based on computations and on the behavior of a "prototype" TRIGA Mark III reactor, the Torrey Pines Reactor at General Atomics. The prototype reactor used individual TRIGA fuel elements in a right circular cylindrical array typical of TRIGA reactors; UWNR uses four-element bundles in a rectangular arrangement in the grid box provided for the original flat-plate fuel. The uranium loading in the prototype was 8 wt% uranium, while UWNR has a uranium loading of 8.5 wt%. Both the prototype and initial UWNR TRIGA cores had stainless steel clad, and both used 20% enriched uranium. The heat transfer characteristics were quite similar, although the diameter of the clad for UWNR was slightly smaller to fit to the grid box array spacing. UWNR also differed by having shrouds dividing the core box into three regions. These shrouds guide control blades, but also introduce water gaps within the core lattice.

The prototype was operated for many years at steady-state power levels up to 1500 kW and thousands of pulses up to 6000 MW. In this report, although the prototype performance characteristics are indicated and sometimes compared to UWNR, most of the information is based on the measured performance of the cores which are currently operable under the present license and technical specifications.

The current core is an all-FLIP (stainless steel clad) configuration consisting of 23 FLIP fuel bundles. Cores with 9 and 15 FLIP fuel bundles also have been operated for significant times, and have been thoroughly tested for conformance to technical specifications and the predictions and descriptions in this and the previous Safety Analysis Report. It is planned that the all-FLIP core will continue to be the operating core until funding (and analysis) for refueling with LEU is completed, at which time this SAR will be amended. It may become necessary to revert to the 15- or 9-bundle FLIP cores before refueling, however, in order to maintain less than a formula quantity of HEU fuel at non-self protecting levels during the return of the HEU to DOE.

Standard TRIGA fuel cores[3]

This core, shown in **Figure 4-17** , and a succeeding core enlarged to 30 fuel bundles because of fuel burnup was operated from November 1967 until March 1974 when it was shut down. Measured core parameters for the initial 25 bundle version of the core are presented below . Several variations in the peripheral reflector configuration were included in this series of cores, including up to 18 reflector elements around the periphery, and some cores with special voided-center or Bismuth-center reflectors.

Core Designation	A25-R10
Excess reactivity	4.82 % ρ
Shutdown Margin	5.17 % ρ
Transient Rod worth	2.10 % ρ
FP reactivity defect	2.95 % ρ
Peak pulse power	1930 MW
Prompt neutron lifetime	42E-6 sec

Figure 4-17 Standard TRIGA Fuel Core

The measured fuel temperatures (**Figure 4-18**) in the UWNR standard TRIGA core almost matched those in the prototype, although this could have differed significantly, depending upon instrumented element placement in the two cores. The UWNR instrumented element was located near the core center in grid position D4NW as indicated in **Figure 4-17** .

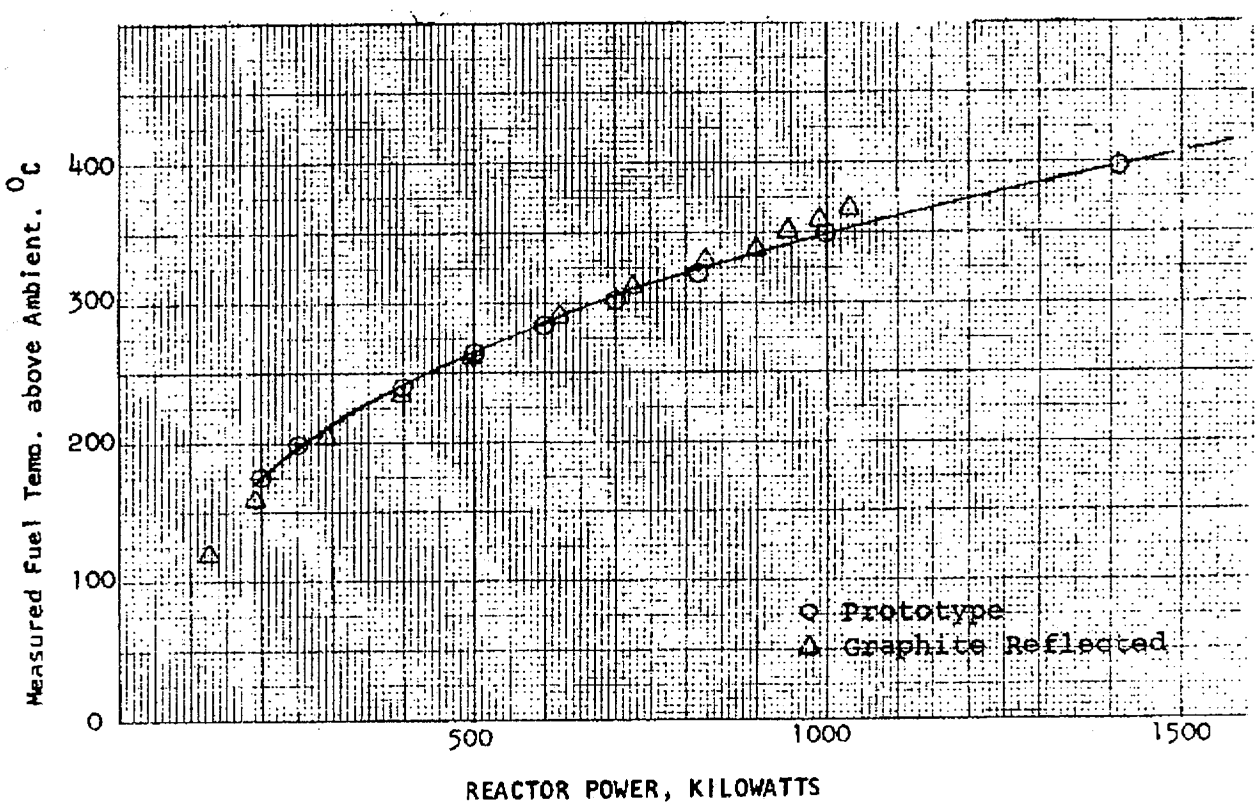


Figure 4-18 Fuel Temperature- Standard Fuel

When the reactivity loss from power operation for UWNR with the original TRIGA core is compared to the prototype (**Figure 4-19**) it is apparent that the power defect for UWNR is significantly larger than for the prototype. In both cases, the core had been pulsed a significant number of times before the temperature measurements were made, so the difference is not from clad stretching in pulsing. The large loss was considered to be a function of the different core geometry and reflection making the leakage change more drastically with temperature in UWNR.

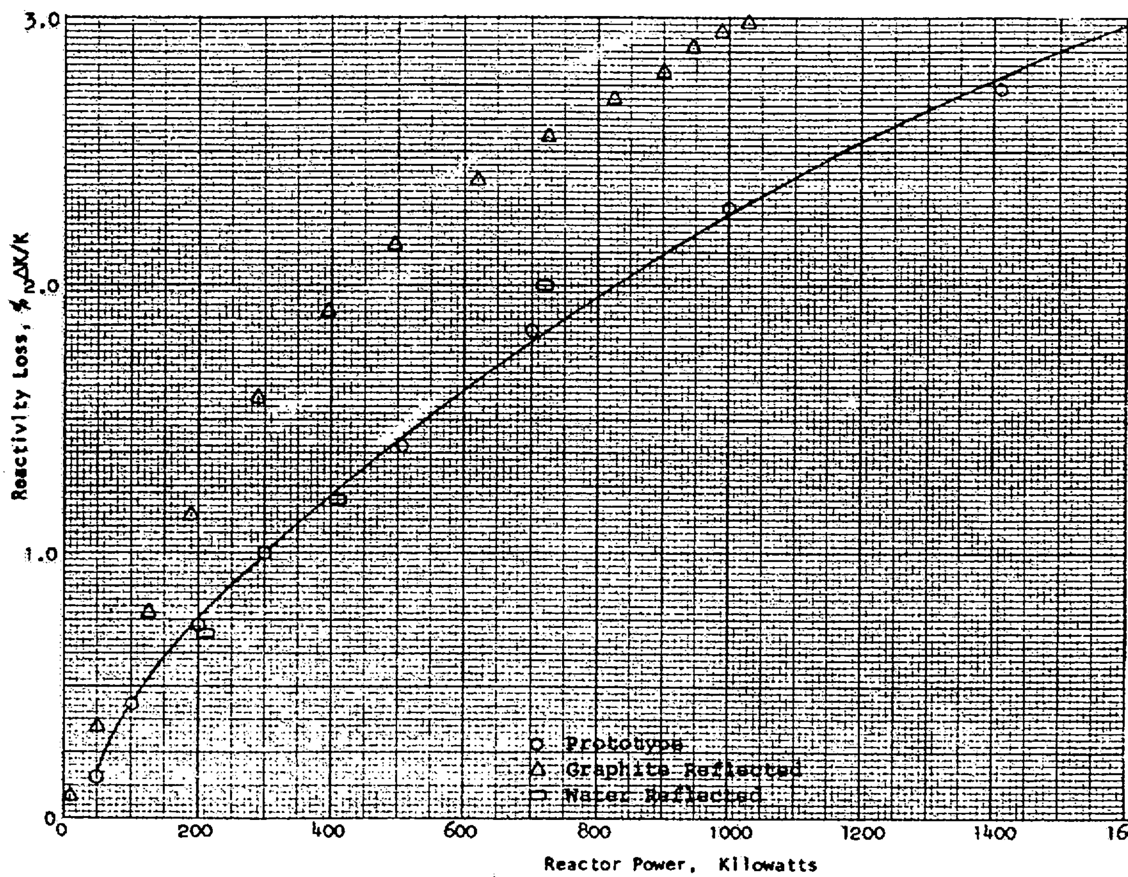


Figure 4-19 Power Defect vs. Power- Standard TRIGA Core

When pulsing behavior of the two cores was compared, other differences were expected and found. The pulsing behavior differed from the typical TRIGA core primarily due to the water gaps in the control blade shrouds and the graphite reflector, both of which increased neutron lifetime, resulting in longer periods for the same pulsed reactivity insertion, and thus broader pulses. Later graphs, **Figure 4-35** through **Figure 4-37** , compare the pulsing behavior of this and all of the other pulsing cores with that of the prototype TRIGA reactor. Note that the pulsed reactivity addition limit for the standard TRIGA fuel was 2.1 % ρ , instead of the 1.4 % ρ for cores containing FLIP fuel. Pulsing behavior differences will be discussed later in this report.

Reactivity Effects In Standard Fuel Cores

The reactivity effect of fuel bundles in different lattice positions have been estimated from measured and calculated values. The position codes in Table 4-1 are those shown in **Figure 4-15** . The reactivity value given is the worth of adding or removing a fuel bundle while the remainder of a 25-bundle core is already present. The worth of the bundles when added in an approach to critical would be radically different, since cores are loaded as compact cores during the loading sequence; that is, the fuel loading plan is planned to assure that the next fuel bundle loaded will have a smaller reactivity effect than the bundles previously loaded.

Table 4-1 Fuel Bundle Worths in UWNR Cores- all in % ρ

Position	Water Reflected	Graphite Reflected
A	2.60	4.0
B	1.95	3.46
C	1.22	2.18
C'	1.16	1.15
D	0.77	-
E	1.76	2.76
F	1.85	1.55

The reactivity effects of reflector variations also have been measured and /or estimated and are indicated in Tables 4-2 and 4-3. First, the worth of both a graphite reflector element and a voided reflector element are indicated relative to a water reflector, with the position codes being those shown in ? .

Table 4-2 Reflector Element Reactivity Worths

Position	Replace water with graphite % ρ	Replace water with air % ρ
1	1.228	-0.239
2	0.180	-0.198
3	0.058	-0.064
4	0.076	-0.083
5	0.115	-0.126
6	0.157	-0.172
7	0.029	-------

Next, the effect of other changes that affect reflection are indicated. Most of these were measured in a core of standard TRIGA fuel.

Table 4-3 Reactivity Effect of Reflector Region Changes

Condition	Result-%ρ
Flooding all 4 beam ports	+0.0005
Flooding pneumatic tube	+0.002
Pneumatic tube samples water filled Cadmium filled	-0.0003
Dropping fuel bundle on top of core	+0.5
Adding fuel bundle on side of core	+0.77

Cores containing FLIP fuel

The longer operating lifetime for FLIP fuel was the major reason for selecting this fuel type for refueling the University of Wisconsin Nuclear Reactor. The higher enrichment of FLIP fuel coupled with erbium poisoning provides the longer operating lifetime, but it also causes changes in operating characteristics relative to standard fuel. The prototype FLIP core was also the Torrey Pines TRIGA Mark III fueled with FLIP fuel. The most marked changes from use of FLIP fuel are a reduction of prompt neutron cycle time to about 10E-6 seconds at beginning of core life (20E-6 at end of core life) and a temperature coefficient that is strongly temperature dependent. (Figure 3-16, page 3 of reference)[4]. These data are for the prototype reactor; values in UWNR were expected to and do differ because of the water gap in the control blade shrouds and the graphite reflector, making the neutron lifetime considerably longer in UWNR FLIP cores than it was in the standard TRIGA core.

In addition, the harder spectrum in a FLIP core leads to power peaking in regions near water gaps. This leads, in a compact core, to a peaking factor within a FLIP element of 1.43. If a large water-filled flux trap is located adjacent to an element, the peaking factor in the element can increase to 2.65 peak/average within the cell.

Thermal and hydraulic parameters of FLIP fuel remain the same as standard fuel.

FLIP fuel elements are not mixed with standard elements in the same fuel bundle at Wisconsin. Thus, the smallest increment of FLIP fuel addition possible will be three FLIP elements (in a bundle containing the transient rod guide tube). Placing such a bundle in the center of a 5 x 5 array of standard TRIGA fuel leads to the highest value of power peaking possible, with resultant power generation of 31.2 kW in each element. Although no operation with this core is anticipated or desired, other TRIGA reactors have operated with power generation rates at least as high as 32kW per element.

Addition of less than five FLIP fuel bundles (24 FLIP elements) was not considered useful for a full power operating core, since it would not provide sufficient additional reactivity to compensate for burnup in the standard elements.

Calculations were performed for cores with 1, 2, 5, 9, 15 and 25 FLIP -bundles in central contiguous regions of the core. All calculations were for a 5 x 5 array of fuel bundles with the transient rod guide tube in the fuel bundle at grid-position D5. Calculations were performed with a two-dimensional diffusion theory code (Exterminator 2) . Standard seven group cross sections obtained from Gulf-General Atomic were used in the calculations. The accuracy of the calculations was checked by analysis of cores with known values of K_{eff} and power density. Results on calculation of mixed cores and FLIP cores were found to be consistent with similar calculations performed elsewhere [5]. Subsequent computations using a 3-dimensional diffusion theory code (DIF-3D) in support of use of LEU fuel agreed well with the results for Exterminator-2 for the all-FLIP core.

FLIP fueled cores can experience significant power peaking which must be considered in permissible fuel arrangements and setting of limiting safety system settings. The power produced in individual fuel elements was predicted from the computations done for safety analyses. The following table shows both the power density in individual fuel elements and the worth of a fuel bundle loaded in a particular location (if applicable to the condition) for several different core arrangements analyzed. See **Figure 4-15** for position descriptions.

Table 4-4 Maximum Power Density and Reactivity Worth of Fuel Bundles- Cores Containing FLIP FUEL

Core arrangement (keyed to fuel bundle locations in **Figure 4-15**) NOTE: D5 is a 3-element bundle with transient rod guide tube	Power in Maximum Element- kW	Reactivity Effect of Removing Fuel Bundle in Indicated Position-%ρ
5 FLIP +20 Standard fuel bundles (FLIP in Positions A & B)	21.4	----
Replace FLIP in E5 with H_2O	28.3	2.83
9 FLIP +16 Standard fuel bundles (FLIP in Positions A, B, and E)	18.1	----
Replace FLIP in D5 with H_2O	20.0	0.93
Replace FLIP in C5 or E5 with H_2O	25.9	1.69
Replace FLIP in D4 or D6 with H_2O	23.2	0.98
Replace FLIP in E4, C4, C6, or E6 with H_2O	22.3	1.49
15 FLIP +10 Standard fuel bundles (FLIP in Positions A, B, C, E, &F)	17.2	----
Replace FLIP in D5 with H_2O	19.0	0.87
Replace FLIP in C5 or E5 with H_2O	24.6	1.65
Full FLIP- 25 FLIP fuel bundles (FLIP in all positions except D)	15.5	----
Replace FLIP in D5 with H_2O	17.2	0.79
Replace FLIP in C6 or D6 with H_2O	20.0	0.51
Replace FLIP in C5 or E5 with H_2O	22.1	1.42

Reference to the table above (Table 4-4) shows that power generation in any individual element is well below 23 KW in all compact FLIP fuel arrangements. Further, the presence of a 3-inch square water gap in the FLIP fuel region will result in power generation rates below 23 KW/element in most of the cores.

The initial operational mixed core contained nine FLIP fuel bundles (35 elements), and the calculations indicate that flux traps could not be permitted for full power operation in this arrangement for locations C5, E5, D4, and D6 if the maximum kW/element is to be kept below

23 kW. The combination of fuel bundle proximity to control blade shrouds and the transient rod guide tube causes the greatest power peaking in any of these cores.

It is also apparent from comparing Tables 4-1 and 4-4 that the reactivity worth of an individual FLIP bundle is lower than that of a standard fuel bundle, even in mixed cores.

The Technical Specifications under which the facility has operated since conversion to FLIP fuel required at least nine (9) FLIP fuel bundles (35 elements) in a central contiguous location with no water gaps larger than a single element except on the core periphery. As a result, the maximum power density in any fuel element at 1,000 kW was limited to 18.1 kW for any of the cores considered. This is approximately 11% higher than the maximum in an all standard fuel core

Three different FLIP-containing cores have been operated at UWNR. Characteristics of each core, measured during the startup and acceptance testing of each core, are shown in the following sections. During core test programs, one quadrant of each core containing FLIP fuel was mapped for temperature by moving an instrumented fuel element into each unique core position. Interpretation of the fuel measurements was complicated by instrumented element failures so that measurements were made with different instrumented elements in different cores as explained below. The standard fuel temperature measurements were made using two different instrumented elements, since the original standard instrumented element failed before the 15-bundle FLIP core was tested. There was, however, only one standard instrumented element in the core at a time, so no comparisons between the indication of the elements in the same core position are available. Two instrumented FLIP elements were available, and both were used in the temperature mapping. The individual FLIP instrumented elements were both placed in at least one common position to enable comparison between the indication of the different elements in the same core position. However, because one instrumented element had all thermocouples fail, three different instrumented FLIP elements have been used in the tested cores, and widely varying temperatures (as much as 130° C) were measured when these different elements were placed in the same core position. This makes interpretation of the predicted and measured fuel temperatures more difficult, but the conclusion reached during the test programs was that the results were reasonably consistent with the predicted values, considering that the non-instrumented fuel assemblies probably have as large a range of heat transfer characteristics as the instrumented elements do. Data tables from these core test programs include fuel temperatures as predicted and measured.

First mixed core- 9 FLIP bundles and 16 standard bundles [6]

This initial mixed core, **Table 4, Figure 4-20** and **Figure 4-21** , was operated from March 1974 through December 1977.

Some parameters of this core were:

Core Designation	F25-R10
Excess reactivity	4.05 % ρ
Shutdown margin	3.60 % ρ
Transient Rod worth	1.37 % ρ
FP reactivity defect	1.92 % ρ
Peak pulse power	805 MW
Prompt neutron lifetime	29.7E-6 sec

[redacted]

Figure 4-20 9-Bundle FLIP Core

Core Position		KW In Element	Predicted Temp. ^{o}C	Measured Temperatures (^{o}C) Bdle 41 FLIP	Bdle 42 FLIP	Bdle 22 STD
D 5	NE	18.1	372	Can't Measure		
	NW	16.1	360	Can't Measure		
	SW	18.1	372	Can't Measure		
E 5	NE	17.4	363	402	365	
	NW	16.1	350	395	356	
	SW	16.9	360	402	365	
	SE	17.0	361	397	361	
E 4	NE	15.4	342	382	343	
	NW	15.4	342	383	346	
	SW	16.1	350	384	348	
	SE	17.0	361	380	343	
D 4	NE	16.1	350	397	359	
	NW	16.1	350	399	353	
	SW	17.4	363	400	363	
	SE	16.2	350	398	342	
F 5	NE	9.3	272			290
	SE	6.7	239			238
F 4	NE	8.0	257			260
	NW	8.8	266			280
	SW	6.3	238			233
	SE	5.7	223			216
F 3	NE	5.8	230			200
	NW	6.8	242			222
	SW	4.9	210			185
	SE	4.6	204			175
E 3	NE	7.4	250			240
	NW	6.6	237			269
	SW	6.9	242			275
	SE	7.4	250			246
D 3	SW	6.9	242			280
	SE	8.3	261			260

Figure 4-21 Power/element and Temperature -9 FLIP Bundle Core

Fuel temperatures in the 9-Bundle FLIP core are shown in **Figure 4-22** . Instrumented elements were located in grid position F5NE for bundle 22 (standard fuel), grid position C4SW for bundle 41, and grid position E5NE for bundle 42. The instrumented element in bundle 42 was located immediately adjacent to the transient rod guide tube. Although the power density was much higher in the FLIP fuel in this partial FLIP core, the fuel temperatures were reasonable.

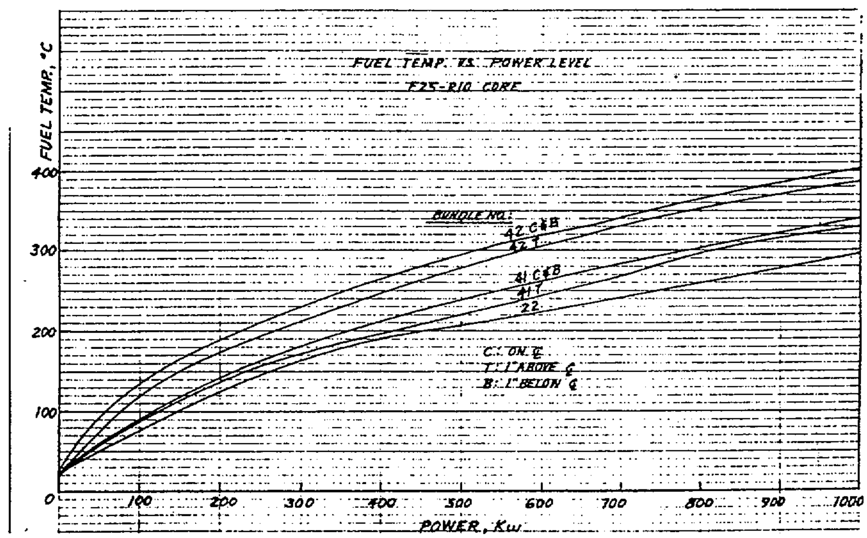


Figure 4-22 Fuel Temperatures vs. Power- 9 Bundle FLIP

Power Defect vs. Power - 9 Bundle FLIP

The power defect was strongly affected by the FLIP fuel. See **Figure 4-23** . Since the temperature coefficient in FLIP fuel becomes more negative as temperature rises, the total power defect during normal full power operation is smaller, while accident response remains essentially the same as in standard TRIGA fuel.

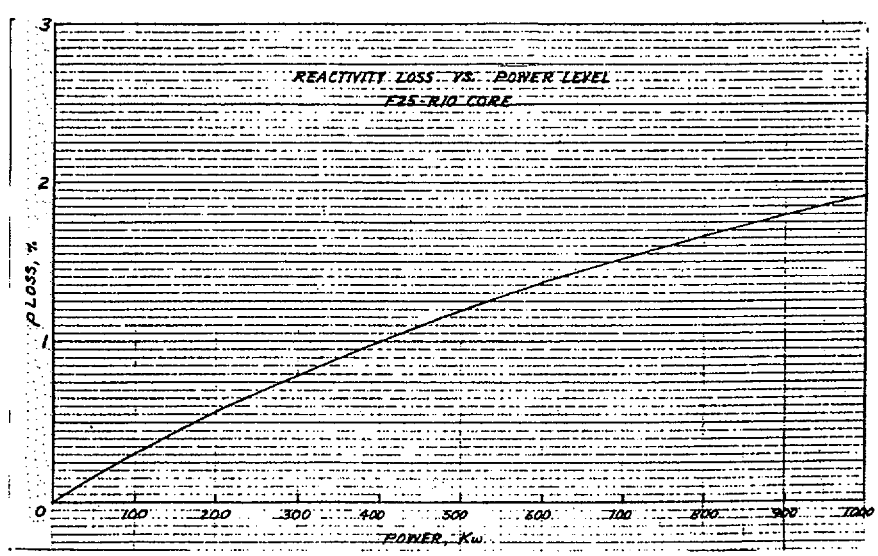


Figure 4-23 Power Defect vs. Power - 9 Bundle FLIP

Second mixed core - 15 FLIP fuel bundles and 10 standard bundles [7]

This core, shown in **Figure 4-24** and **Figure 4-25** , was operated from January 1978 until June 1979, although some initial high power operation with a core intermediate between this and the all FLIP core was done to make the initial shipment self-protecting before the final batch of fuel was received.
Some parameters for this core were:

Core Designation	G25-R10
Excess reactivity	3.87 % ρ
Shutdown Margin	3.90 % ρ
Transient Rod worth	1.38 % ρ
FP reactivity defect	1.75 % ρ
Peak pulse power	930 MW
Prompt neutron lifetime	24E-6 sec

[redacted]

Figure 4-24 15 Bundle FLIP core

Core Position		KW in Element	Predicted Temp. (26 or 41)°C	Measured Temperatures (°C) Bdle 41 Center TC	Bdle 42 Center TC	Bdle 26 Center TC
D5	NE	17.2	363		C A N ' T M E A S U R E	
	NW	16.0	349		C A N ' T M E A S U R E	
	SW	17.2	363		C A N ' T M E A S U R E	
E5	NE	16.5	355		435	
	NW	15.3	342		---	
	SW	15.8	347		361	
	SE	15.8	347		361	
E4	NE	11.9	304		340	
	NW	14.6	334		355	
	SW	15.0	338		345	
	SE	13.3	318		322	
D4	NE	12.5	309			
	NW	15.2	340			
	SW	16.4	354	269	400	
	SE	12.6	310		359	
E3	NE	9.7	277	237	415	
	NW	9.9	280	290		
	SW	11.2	294	247		
	SE	10.6	288	294		
D3	NE	11.0	293			
	NW	10.4	285			
	SW	10.4	285	301		
	SE	11.0	293	250		
F5	NE	8.8	267			271
	NW	8.8	267			
	SW	6.4	237			
	SE	6.4	237			237
F4	NE	7.5	252			246
	NW	8.4	262			263
	SW	6.1	231			220
	SE	5.5	222			213
F3	NE	5.4	220			203
	NW	6.3	236			220
	SW	4.8	208			177
	SE	4.5	200			163

Figure 4-25 Power/element and Temperature - 15 FLIP Bundle Core

With the 15-bundle core, FLIP fuel characteristics become more pronounced . The standard fuel instrumented element in standard fuel bundle 25 was located in grid position F5NE, and the measured temperatures are shown in **Figure 4-26** .

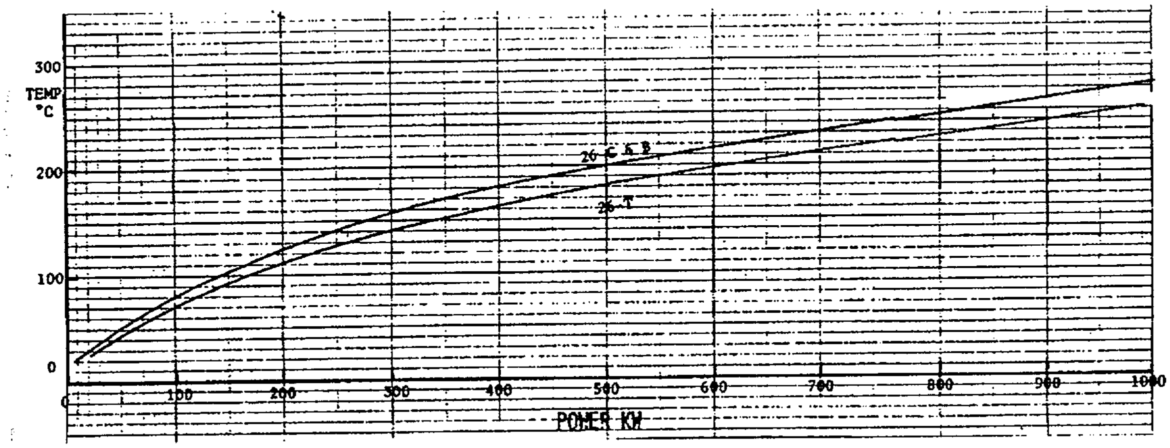


Figure 4-26 Standard Fuel Temperature vs. Power -15 Bundle FLIP

Figure 4-27 shows the temperatures for the instrumented element in fuel bundle F41, located in grid position E3NE. The unusual behavior of the bottom thermocouple (41-B in the figure) was due to the beginnings of failure of the thermocouple due to internal shorting.

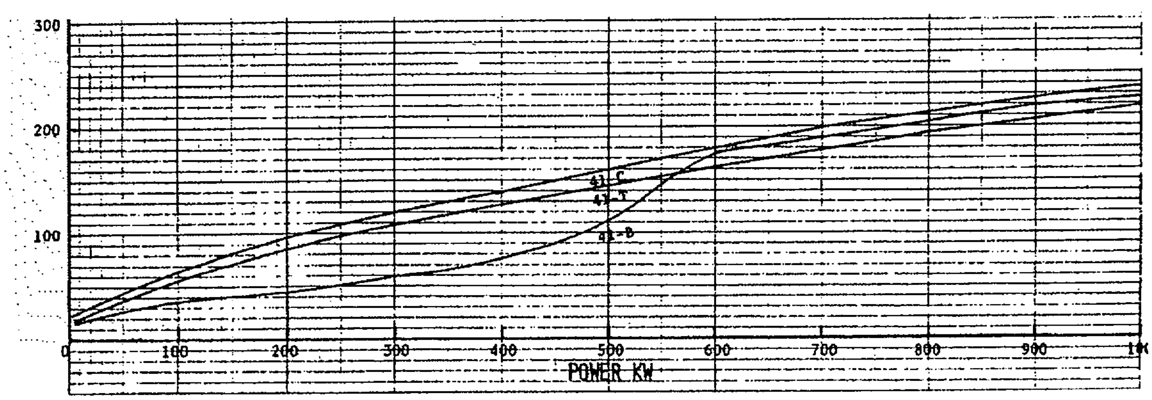


Figure 4-27 Bundle 41 Fuel Temperature vs. Power -15 Bundle FLIP

Figure 4-27 shows the temperature for the fuel bundle F42 instrumented element vs. reactor power.

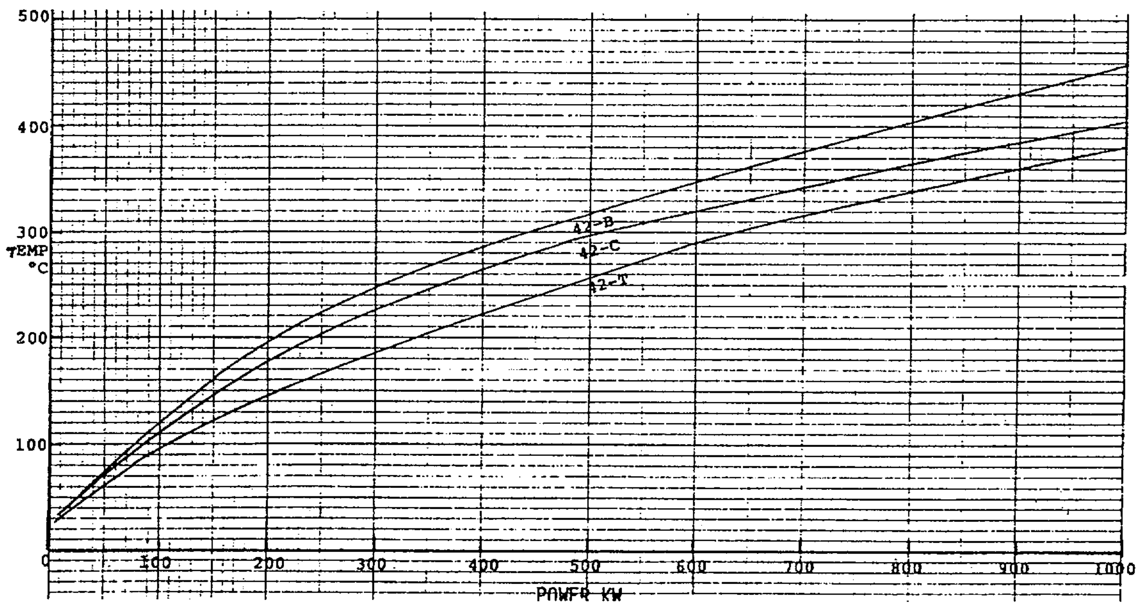


Figure 4-28 Bundle 42 Fuel Temperature vs. Power - 15 Bundle FLIP

The power defect for this core is shown in **Figure 4-28** . Again, the effect of the FLIP fuel results in a still lower power defect with the higher amount of power generation in the FLIP portion of this core. Pulsing behavior also showed a further reduction in the prompt neutron lifetime and thus faster periods for the same reactivity input in a pulse.

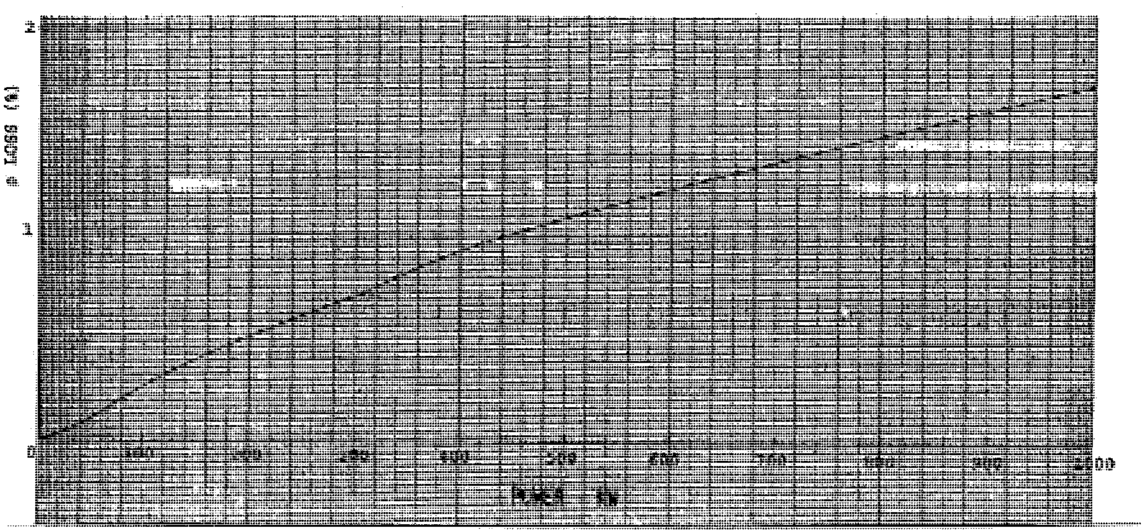


Figure 4-29 Power Defect vs. Power - 15 Bundle FLIP

All-FLIP core [8]

The reactor has been operated with cores containing only FLIP fuel since July 1979. Two variants of this core have been commonly used, differing only in the number of graphite reflectors used and location of irradiation facilities. The arrangement shown in **Figure 4-29** with the characteristics indicated below and in **Figure 4-31** is the most-used variant.

Some measured parameters for this core were:

Core Designation	I23-R10
Excess reactivity	
4.23 % ρ	
Shutdown Margin	3.80 % ρ
Transient Rod worth	1.395% ρ
FP reactivity defect	1.53 % ρ
Peak pulse power	950 MW
Prompt neutron lifetime	23E-6 sec

[redacted]

Figure 4-30 All-FLIP Core

Core Position		KW in Element	Predicted Temp. (#42)°C	TEMPERATURES AT FULL POWER					
				#41			#42		
				Bot	Ctr	Top	Bot	Ctr	Top
D5	NE	16.0	490	Can't Measure					
	NW	14.9	470	Can't Measure					
	SW	16.0	490	Can't Measure					
E5	NE	15.5	480	365	382	353			
	NW	14.4	460						
	SW	14.6	465						
	SE	14.6	465	340	362	338			
E4	NE	11.9	415	300	315	295			
	NW	13.6	442	334	350	328			
	SW	13.7	450	330	350	326			
	SE	12.1	420	305	325	305			
D4	SW	15.2	475	390	405	380	492	432	407
	SE	15.1	470				426	380	343
E3	NE	14.2	455		277	259	375	337	300
	NW	12.2	420	275	290	270			
	SW	9.9	385	275	285	265			
	SE	10.6	395		241	224			
F5	NE	11.6	415				450	400	350
	SE	7.9	345				341	301	268
F4	NE	9.8	380				410	362	325
	NW	11.0	405				442	395	352
	SW	7.5	340				340	300	270
	SE	6.8	325				305	270	240
F3	NE	8.7	360				318	282	241
	NW	8.1	345				362	320	280
	SW	5.9	300				220	245	280
	SE	6.7	325				242	220	198

Figure 4-31 Power/element and Temperature - All FLIP Core

Fuel temperatures in the all-FLIP core are shown in **Figure 4-31** and **Figure 4-32** . The bottom thermocouple in fuel bundle 41had developed a short from one side of the couple to ground, and thus reads well below the actual temperature in this graph. The instrumented element in bundle F41 was in grid position E3NE, while that for bundle F42 was in grid position D4SW next to the transient rod guide tube.

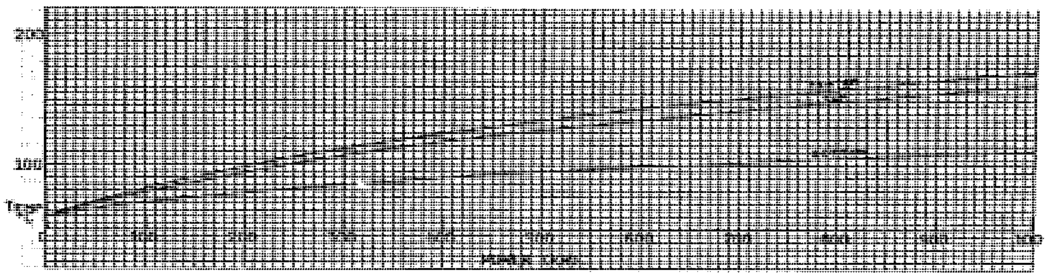


Figure 4-32 Bundle 41 Fuel Temperature vs. Power - All FLIP

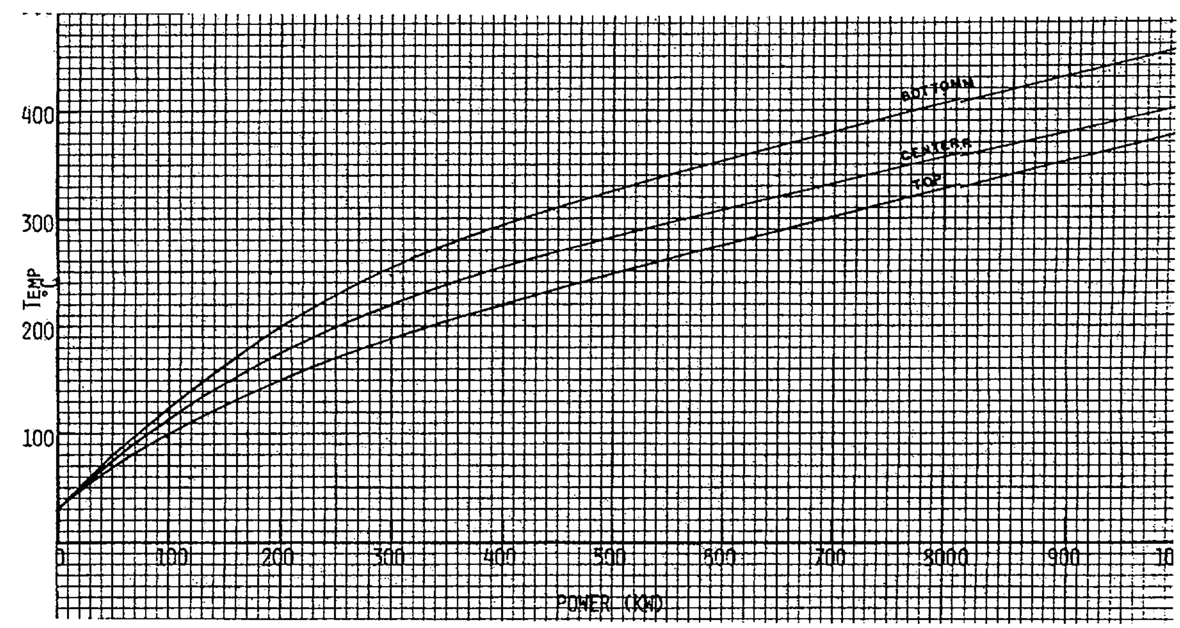


Figure 4-33 Bundle 42 Fuel Temperature vs. Power - All FLIP

Figure 4-33 shows the power defect versus power for this core. In this all-FLIP core the power defect for licensed full power decreased slightly from the 15-bundle FLIP core value. At the end of 1999, after more than 477 MWd operation, the power defect at licensed full power remains at 1.51 % ΔK/K.

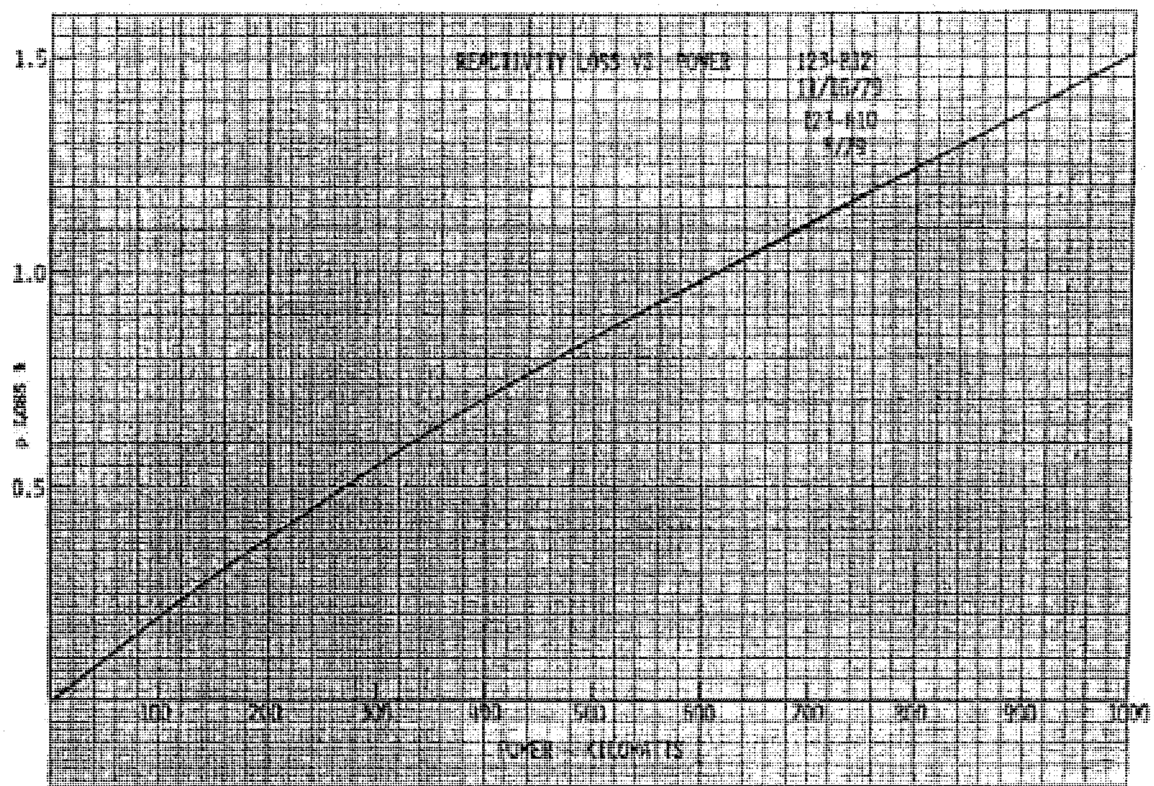


Figure 4-34 Power Defect vs. Power - All FLIP

Isothermal Temperature Coefficient

The coolant water temperature in the prototype was varied over wide ranges (20° to 60°C) to measure the resulting reactivity change. The measurements were made at power levels of less than 10 watts. The coefficient is slightly positive with a net gain in available reactivity of 0.077% over the range indicated. The average coefficient,0.0019%/°C, is small enough that it is essentially negligible for normal operating conditions.

The effect of the water gap left in the shrouds when the control blades are withdrawn was expected to increase the temperature coefficient by about 20% in the UWNR, giving a temperature coefficient estimated at 0.0024%/°C. This value was small enough to be considered negligible for normal operating conditions. Values measured during startup testing were 0 and 0.0042, but with vary large uncertainty in the values because of other possible reactivity variations that might occur (bubbles, variation of rest position of control blades and the transient rod, and other extremely small variations). Considering these other variations it is not possible to see any change in reactivity in the UWNR cores as the bulk water temperature changes for either the standard, mixed, or all-FLIP cores.

Pulse Parameters

Measurements were made of the various parameters relating to pulsing operation of the prototype and of the UWNR cores. The most important of these are given below for step insertions of reactivity up to 2.1% ΔK/K for the standard-fueled cores and 1.4% ΔK/K for the cores that contain FLIP fuel. The data for the prototype TRIGA Mark III core, the UWNR standard TRIGA core (both water and graphite reflected), and the mixed cores are indicated in the figures referenced below.

Period and Pulse Width

During pulsing operation the reactor is placed in a super-prompt-critical condition in which the asymptotic period is related to the prompt reactivity insertion divided into the prompt neutron cycle time. The pulse width is inversely related to the prompt reactivity insertion. Behavior of the different cores and the prototype is indicated in **Figure 4-35** with points of inverse period and FWHM shown on the same graph. The plots show the results of plotting the reciprocal of the measured period versus the prompt reactivity insertion. Since the period data were obtained from an oscillographic recording of the reactor power versus time at a portion of the pulse before fuel temperature limiting effects have begun, the accuracy of the measurements is not so good as for other parameters, and some scatter in the data are expected. As can be seen, the minimum period in standard fuel obtained for reactivity insertions of 2.1% ΔK/K is about 2.6 msec while that of the FLIP core is 3.4 msec for a 1.4% ΔK/K insertion.

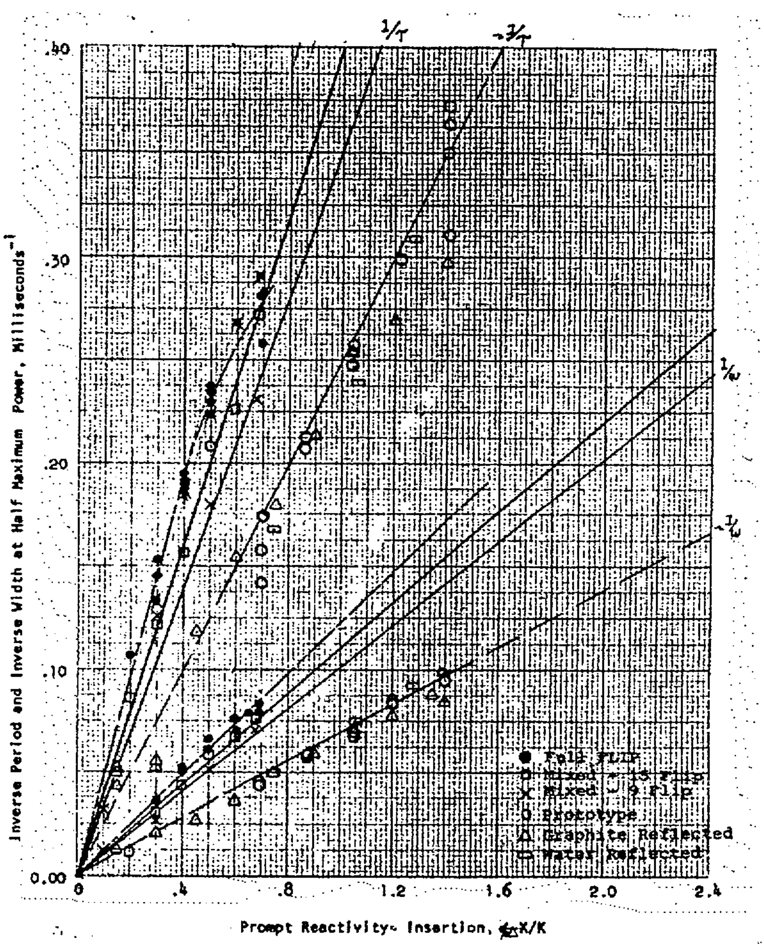


Figure 4-35 Inverse Period and Inverse Width at Half-Max vs. Prompt Reactivity Insertion

As FLIP fuel replaces standard fuel in the mixed cores, the decrease in prompt neutron cycle time results in a different straight line plot for each core. Further, it becomes apparent on the all-FLIP core that the faster cycle time, even for the smaller prompt reactivity insertion allowed for the mixed cores, causes the data for larger reactivity insertions to depart from a straight line. This is because the transient rod has not completed its travel before the reactor reaches a substantial power level, thus resulting in a longer period as negative reactivity is inserted by the fuel temperature coefficient.

Pulse Width

The width of the power pulse is most conveniently described as the time interval between half-power points. Also shown in **Figure 4-35** are plots of the reciprocal of the measured width versus prompt reactivity insertion. Each of the cores has a different straight line plot, again due to the increasingly short prompt neutron cycle time as the amount of FLIP fuel increases.[2]

When the pulse width (Full Width at Half Maximum power) of the various cores is compared to the prompt period as shown in **Figure 4-36**, all of the cores conform fairly well to the same straight line because the difference in prompt neutron cycle time is not a factor in the relationship between these two variables as it is in the reactivity-period and reactivity-FWHM relationships.

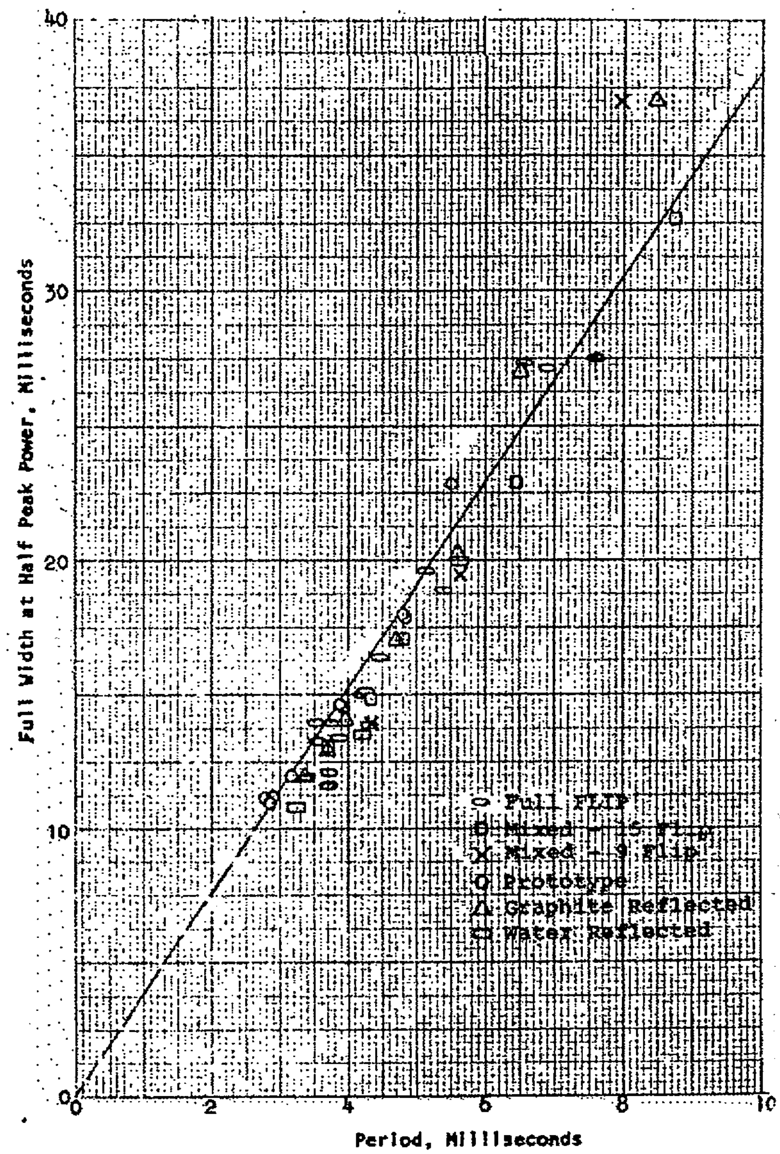


Figure 4-36 FWHM vs Period

Peak Power

Peak power in a pulse is proportional to the square of the prompt reactivity insertion, while energy generated in a pulse is directly related to the prompt reactivity insertion. **Figure 4-37** shows the interrelationship between maximum transient power and pulse width. Peak power is plotted against the square of inverse FWHM in order to get a straight line plot. The standard and FLIP fueled cores show differences due to the shorter prompt neutron cycle time.

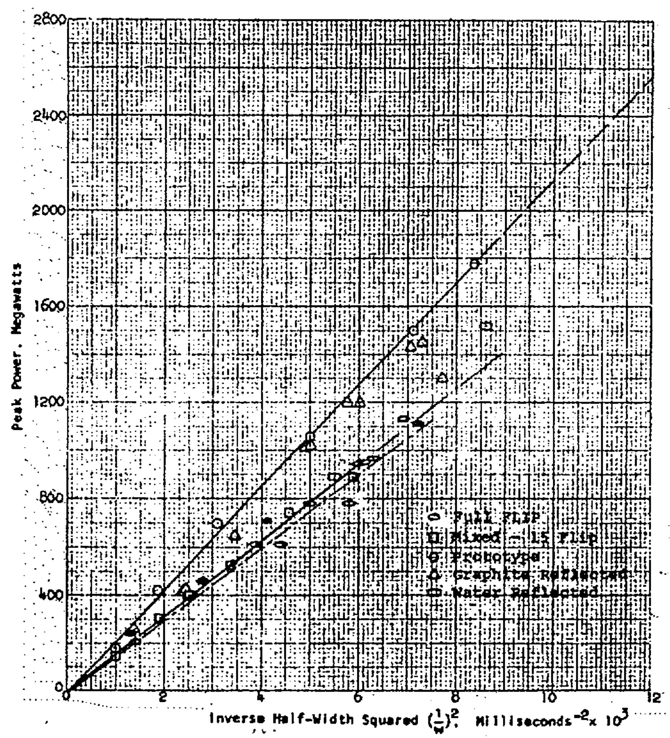


Figure 4-37 Peak Power vs Inverse Half-width Squared

Figure 4-38 shows the relationship between initial period and peak power and seems to be fit fairly well by a single straight line. Note that the prompt reactivity insertion is limited to 1.4 % ΔK/K for the FLIP cores.

For a given core configuration, the peak power, integral power in the prompt burst, and width of the pulse are determined by the reactivity insertion made. It can be seen from the plots that the peak power is controllable over a rather wide range since this parameter is very nearly proportional to $(\Delta K/K - 0.7\%)^2$. Pulse width and integral powers, on the other hand, are approximately linear functions of reactivity insertions above prompt critical so that their range is more limited.

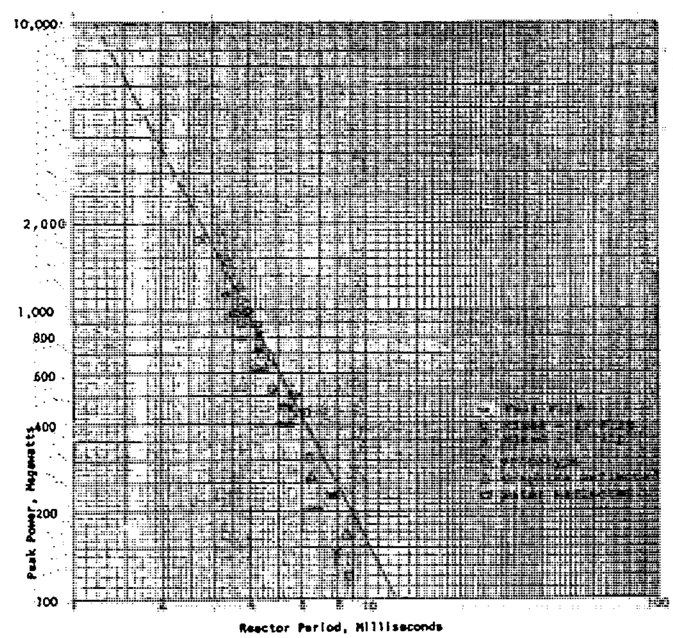


Figure 4-38 Peak Power vs. Reactor Period

4.5.3 Operating Limits

For previous operation of the reactor at 1 MW, the reactivity has been less than 4.9 % ΔK/K above clean cold critical. The reactivity is allocated approximately as indicated below:

Power Coefficient	1.75% ΔK/K
Xenon Poisoning	1.75% ΔK/K
Control & Flux Balancing	1.40% ΔK/K

No specific amount of negative reactivity available from control element action is specified, since the requirement on minimum shutdown margin assures safe shutdown from any operating condition.

The minimum shutdown margin with the most reactive control element and any un-scrammable control elements full out will not be less than 0.2% ΔK/K. Shutdown margin is verified by calibrated control element positions and by rod-drop measurements.

The limitations on cores containing FLIP fuel will maintain power density to levels capable of natural convection cooling during power operation up to 1.5 MW power. This limitation will also assure that power density in any fuel element will be below that at which loss of reactor coolant will result in fuel damage.

In addition, the maximum reactivity for an experiment is limited to 1.4 % ΔK/K All in-pool experiments will be constrained at least as well as the fuel bundles. In-core experiments are designed so they are constrained by the grid or grid box structure, although part of their support may be from other pool structure.

Should an experiment having the maximum reactivity worth allowed for all experiments (1.4% ΔK/K) fail, the resulting step change in reactivity worth would be less than that deliberately inserted during pulsing operation.

Should the beam ports and pneumatic tube flood while the reactor is operating at full power, a step reactivity addition of 0.07% ΔK/K would result. This reactivity change is so small that it would not cause any disruption of normal operation.

If a gross departure from procedure were to be made and a fuel element bundle were added to the outside of the core while operating at full power, the maximum reactivity that would result would be about 0.7% ΔK//K. This is a reactivity smaller than that routinely inserted during pulsing operation.

Despite the built-in safeguards and inherent safety of the reactor and its fuel, great attention is paid to proper supervision of operation and adherence to procedures approved by competent authority. It is the policy of the University of Wisconsin that standard operating procedures are carefully prepared and reviewed, strictly followed, and kept current. Likewise, competent supervision assures that operation is kept within the limits set by licenses, technical specifications, existing procedures, and general good practice.

4.6 Thermal-Hydraulic Design

General Atomic has done extensive thermal-hydraulic computations over the years, including one study specifically directed at the use of four-element bundles of TRIGA fuel as a replacement fuel for reactors which were originally fueled with flat-plate fuel elements and operated at power levels up to 2,000 kW with natural convection cooling [9]. The Puerto Rico Nuclear Center TRIGA-FLIP reactor used 4-element fuel bundles and operated at power levels much higher than the 1,000 kW steady-state power level of UWNR with natural convection cooling [10]. The thermal and hydraulic design for operation of the Torrey Pines Thermionic Reactor, section 3.3 of the reference, describes the core parameters for a very similar core [11]. The conclusions of these references were that four element bundles of TRIGA fuel could be used for power levels up to 2000 kW with natural convection cooling. The University of Wisconsin Nuclear Reactor has been operated with natural convection cooling at steady-state power levels up to 1,000 kW for many years with no cooling problems and no fuel damage. Many other TRIGA reactors have been operated at power levels up to 1.5 MW with natural convection cooling and no cooling problems.

4.7 References

1. Letter submitted under Docket 50-156 from R. J. Cashwell, Reactor Director, to Theodore S. Michaels, Senior Project Manager, USNRC, August 7, 1990

2. Reactor Shielding Design Manual, Theodore Rockwell III, Editor, McGraw Hill Book Company, 1956, page 178

3. Memo No. 4, Report on Refueling the University of Wisconsin Nuclear Reactor, R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, March 1968

4. GA-9064, Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, Section 3.2 and Figure 3-16, General Atomic, Jan. 5, 1970

5. Same as 3

6. Core Test Program UWNR Mixed TRIGA-FLIP Core (9 FLIP Bundles),R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, July 1974

7. Core Test Program UWNR Mixed TRIGA-FLIP Core (15 FLIP Bundles),R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, February 1978

8. Core Test Program All FLIP Core) ,R. J. Cashwell, Nuclear Engineering Department, University of Wisconsin, January 1980

9. Steady State Thermal Analysis for the Proposed Use of TRIGA Fuel Elements in MTR Reactors, GA-5708, General Atomic, 1965

10. Safeguards Summary Report for the TRIGA-FLIP Reactor at the Puerto Rico Nuclear Center, PRNC 123, Puerto Rico Nuclear Center, November 11, 1969.

11. Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor, GA-9064, Gulf General Atomic, January 5, 1970

5 REACTOR COOLANT SYSTEMS

5.1 Summary Description

The pool water is cooled by the system shown schematically in **Figure 5-1.** The design basis of the reactor coolant system is to dissipate 1.0 MW with primary temperatures approximately 80 °F and prevent the inadvertent loss of pool water. The system, however, performs no safety function.

The system consists of three loops; the closed-loop primary coolant system, the closed-loop intermediate coolant system and the closed-loop campus chilled water system. Heat from the primary coolant system is transferred to the intermediate coolant system through the primary heat exchanger. Heat from the intermediate coolant system is then rejected to the campus chilled water system through the intermediate heat exchanger. The system is designed to maintain a pressure gradient towards the pool in order to prevent the inadvertent loss of pool water.

5.2 Primary Coolant System

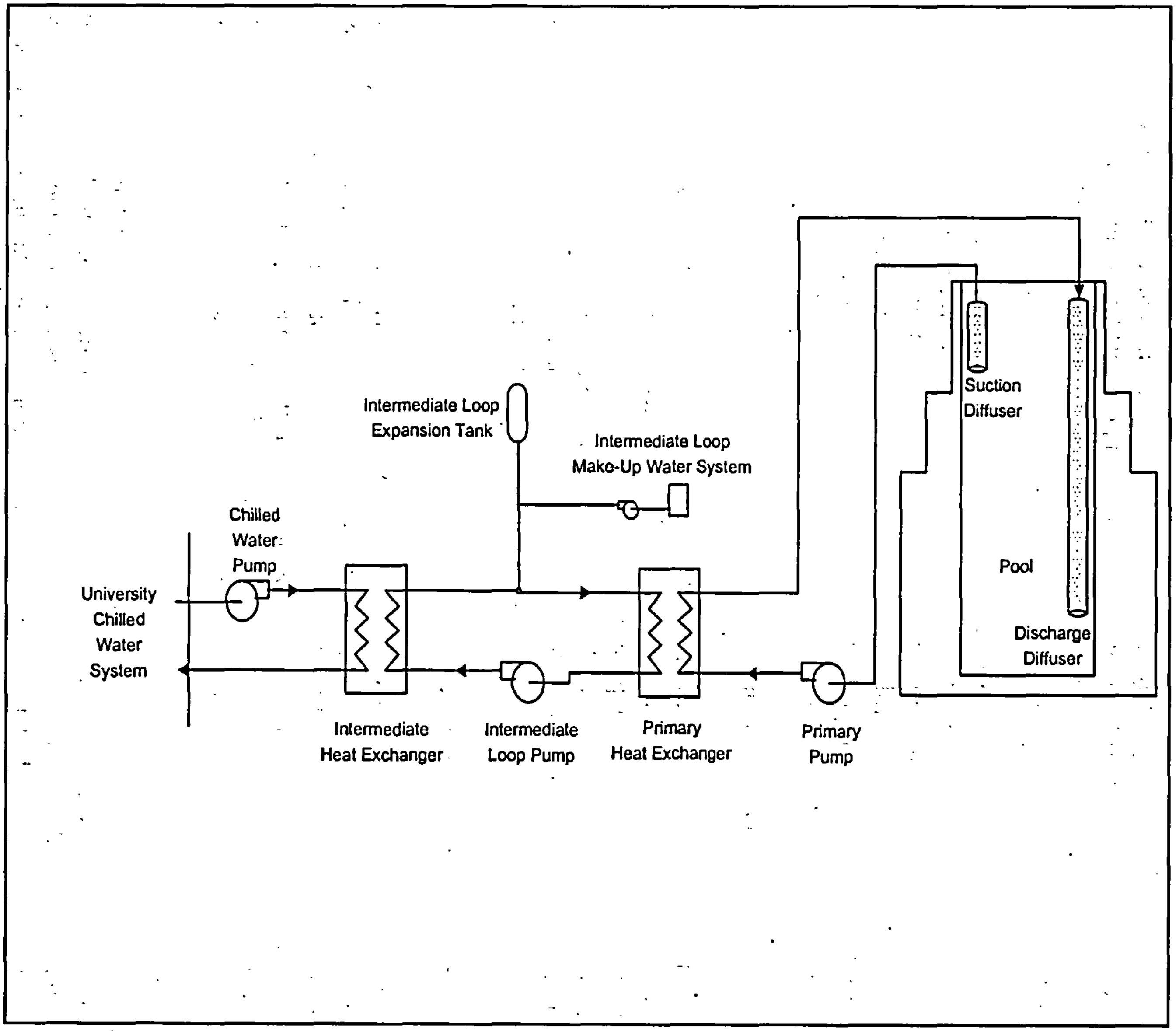
The primary coolant system is composed of a pump, isolation valves and various devices used to extract flow rate, temperatures and pressures. Stainless steel components and piping are used in the system in order to maintain primary water quality more easily. The primary system continuously circulates pool water through the primary heat exchanger. The intake and outlet diffusers include siphon breaker holes to preclude draining more than 1 foot of water even in the case of a pipe rupture. This will maintain at least 19 feet of water above the active core.

5.3 Intermediate Coolant System

The intermediate coolant system consists of a pump, isolation valves and various devices used to extract temperatures and pressures. Stainless steel components and piping are used in the system to maintain reactor grade quality water in the intermediate coolant system. Circulation in this system is maintained by the intermediate pump, discharging through the intermediate heat exchanger, where it will reject heat to the campus chilled water system. The cold water will then circulate through the primary heat exchanger to cool the primary water and return to the pump suction.

The intermediate coolant system is equipped with a pressurized expansion tank and a make-up water system. The expansion tank will accommodate volumetric changes in the intermediate system process fluid and maintain the intermediate system pressure above the primary coolant system pressure under both static and operational conditions. By maintaining the intermediate system pressure higher than the primary system, should a leak occur, it would result in intermediate water entering the primary system thereby protecting against the inadvertent loss of pool water. A pressure sensor provides indication at the control console and an interlock prevents starting the primary pump unless the intermediate loop pump is running.

Figure 5-1 Cooling System Schematic



Should leakage occur, it could be detected in three ways. First, the intermediate loop will be unable to maintain pressure and a low pressure annunciator will alarm at the reactor control console. Second, the pool level float switch will be actuated by high pool water level should as much as 150 gallons of intermediate water enter the pool system. Finally, if the integrity of the intermediate water heat exchanger is also compromised, an influx of degraded quality intermediate water will increase conductivity in the pool water.

5.4 Campus Chilled Water System

The campus chilled water system consists of carbon steel piping, pump, isolation valves and various devices used to extract temperatures and pressures. Circulation in this system is maintained by the chilled water pump, taking a suction on the main campus chilled water system. The pump discharges through a filter and into the intermediate heat exchanger, where it cools the intermediate loop water and returns to the main campus chilled water system.

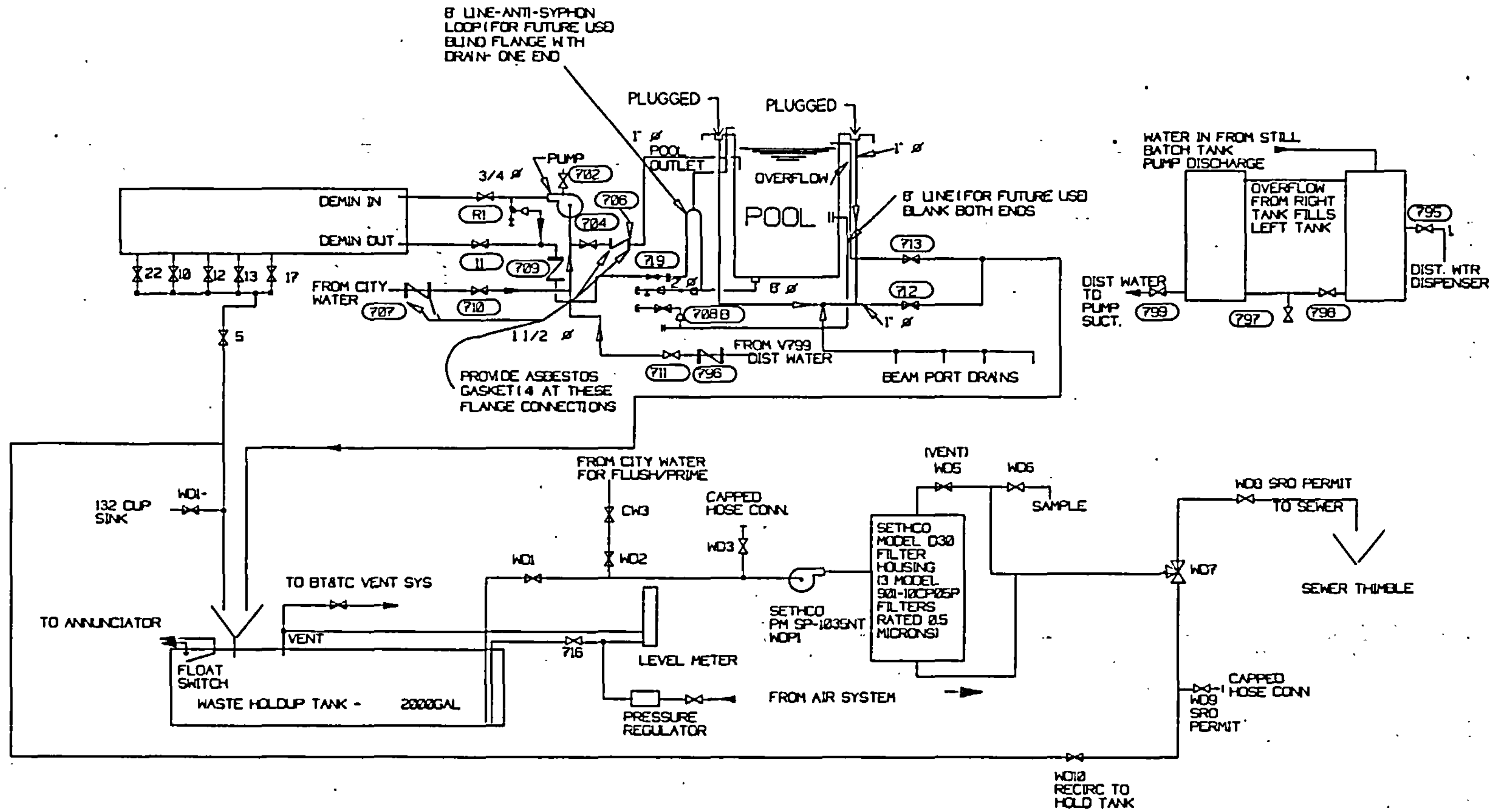
The campus chilled water loop is maintained at a higher pressure than the intermediate system. This pressure gradient will insure that in the extremely unlikely event of leaks in both the primary coolant system and intermediate coolant system heat exchangers and loss of intermediate system pressure that inadvertent loss of pool water will be physically impossible.

5.5 Primary Coolant Cleanup System and
5.6 Primary Coolant Makeup Water System

The pool make-up and clean-up system is shown schematically in **Figure 5-2** . Water is circulated from the pool surface, through the pump, through the demineralizer, and then into the pool under the core box and coolant header. The pump maintains about 10 gallons/minute flow through the demineralizer. The demineralizer is a mixed-bed type with provisions for regeneration of resins or discharge of spent resin and loading with new resin. A water softener supplies softened water for regeneration of the demineralizer.

Normally, make-up water is supplied by the still. The still delivers water to a storage tank from which it is pumped (by the pool recirculating pump) into the pool to maintain pool water level. Although distilled water is normally used for makeup, alternate flow paths allow softened or city water to be fed through the demineralizer into the pool. In either case, impurities in make-up water are reduced to less than 1 ppm before going into the pool. Wastes from regeneration of the demineralizer are discussed in the next section.

Flow from the demineralizer to the pool is through valve 11, check valve 709 which prevents back flow, and valve 719 into the 8 inch pipe loop and into the bottom of the grid box. The 8 inch line is equipped with a siphon breaker at the top of the pool so that rupture of the line at the demineralizer outlet or of the 8 inch line outside the shield cannot drain the pool to a level that will uncover the core. A second 8 inch line is flanged off on both ends. The 8 inch lines were originally installed to allow a forced-convection cooling mode, but the lines are used only as indicated above.



PIPING SCHEMATIC -MAKEUP AND WASTE

Figure 5-2 Pool Makeup, Cleanup, and Waste Disposal

A two inch line whose rupture could have caused loss of pool water has been permanently plugged inside the concrete shield and is presently sealed off outside the shield. A pool drain line and valve have been eliminated. There are no valves in the system that, if opened, can drain the pool.

Should valve number 5 (shown in **Figure 5-2**) be left open upon placing the system in its normal operating condition, as much as 400 gallons of pool water could be pumped to the holdup tank. No further loss of water would then occur, since check valve 709 will prevent reverse flow from the 8 inch pipe loop to the demineralizer and the siphon breaker at the top of the loop will prevent additional water loss.

All operations involving the make-up and clean-up systems are performed by written checklist-type procedures designed to prevent draining of the pool.

5.7 Nitrogen-16 Control System

Nitrogen-16 suppression is accomplished by a jet-type diffuser system. This system pumps about 80 gallons of water per minute from near the pool surface through a single nozzle having a 0.75 inch wide, 6.5 inch long opening. The nozzle is located 5.5 feet above and 0.5 feet east of the core, with the diffusing stream directed downward at a 45 degree angle toward the west end of the pool.

The pump for the N-16 suppression system is located on the outside east face of the reactor's concrete pool shield structure, about 8 feet below the pool surface. The system is constructed with siphon breaker holes which preclude draining more than one foot of water from the pool in the event of a pipe rupture.

5.8 Auxiliary Systems Using Primary Coolant

There are none.

5.9 References

There are none.

6 ENGINEERED SAFETY FEATURES

6.1 Summary Description

Engineered safety features are not required for this reactor due to low operating power and good fission product retention in the fuel. A confinement with a controlled ventilation system is provided, however, to reduce the consequence of fission product release from fuel or experiment malfunctions to even lower levels

6.2 Detailed Descriptions

6.2.1 Confinement

The Reactor Laboratory is a 43 by 70 foot room of conventional construction within the Mechanical Engineering Building ,with a ceiling height of approximately 36 feet in most of the room. The portion of the ceiling above the console area is at a height of 22 feet. **Figure 6-1** through **Figure 6-5** show the outlines of the room and location of major reactor components.

The floor of the room is concrete laid on the ground. The walls are concrete and brick. [redacted]

The console area is located in the southwest corner of the Reactor Laboratory. It is separated on the north and east sides from the laboratory proper by wire reinforced glass provided to reduce noise originating from the cooling system and other pumps and equipment. Two doors, one on the east and one on the north of the console area open into the remainder of the Reactor Laboratory. The console area is served by an air handling unit which circulates air from the Reactor Laboratory through the console area for ventilation and air conditioning and returns the air to the Reactor Laboratory. The console is therefore within the confinement system of the Reactor Laboratory

The Reactor Laboratory has three windows which face the parking lot and stadium. There are five single doors; two opening on the west wall into the Nuclear Engineering Laboratory at ground level, one opening on the west wall into the basement level of the Nuclear Engineering Laboratory, one opening into the parking area, and one opening into the Heat Power Laboratory. One double door opens into the Nuclear Engineering Laboratory to the east of the Reactor Laboratory. All doors have glass panes which are covered with expanded steel gratings. Interior doors are fire doors and are not weather-stripped. No special seals are provided for lines that penetrate the walls. All doors and windows are normally closed and locked for security and air flow control considerations.

[redacted]

Figure 6-1 Reactor Laboratory Basement Floor Plan

The Nuclear Engineering Laboratory to the east of the Reactor Lab (Room 5 of the Mechanical Engineering Building) is used as a research laboratory and shop, but it also contains a subcritical assembly, a small hot cell, and a storage room which contains low-level radioactive waste.

The basement level of the Nuclear Engineering Laboratory to the west contains small rooms having concrete walls [redacted]. The small rooms house the pneumatic tube equipment and dispatch station, a sample preparation room, air activity monitor equipment, and both general storage and radioactive storage areas. An instrumentation shop is located on the east end, while the west end of the room is a counting laboratory with HPGe, NaI, and alpha-beta-gamma counters.

[redacted]

Figure 6-2 Reactor Laboratory Facing South

The ground-floor level of the Nuclear Reactor Laboratory on the west houses graduate student research and office areas and an office area for the Reactor Laboratory.

Plans exist for building a room inside the Reactor Lab which will include the upper level along the north wall. It would be used either for offices for the Reactor Laboratory or for an electronics shop.

[redacted]

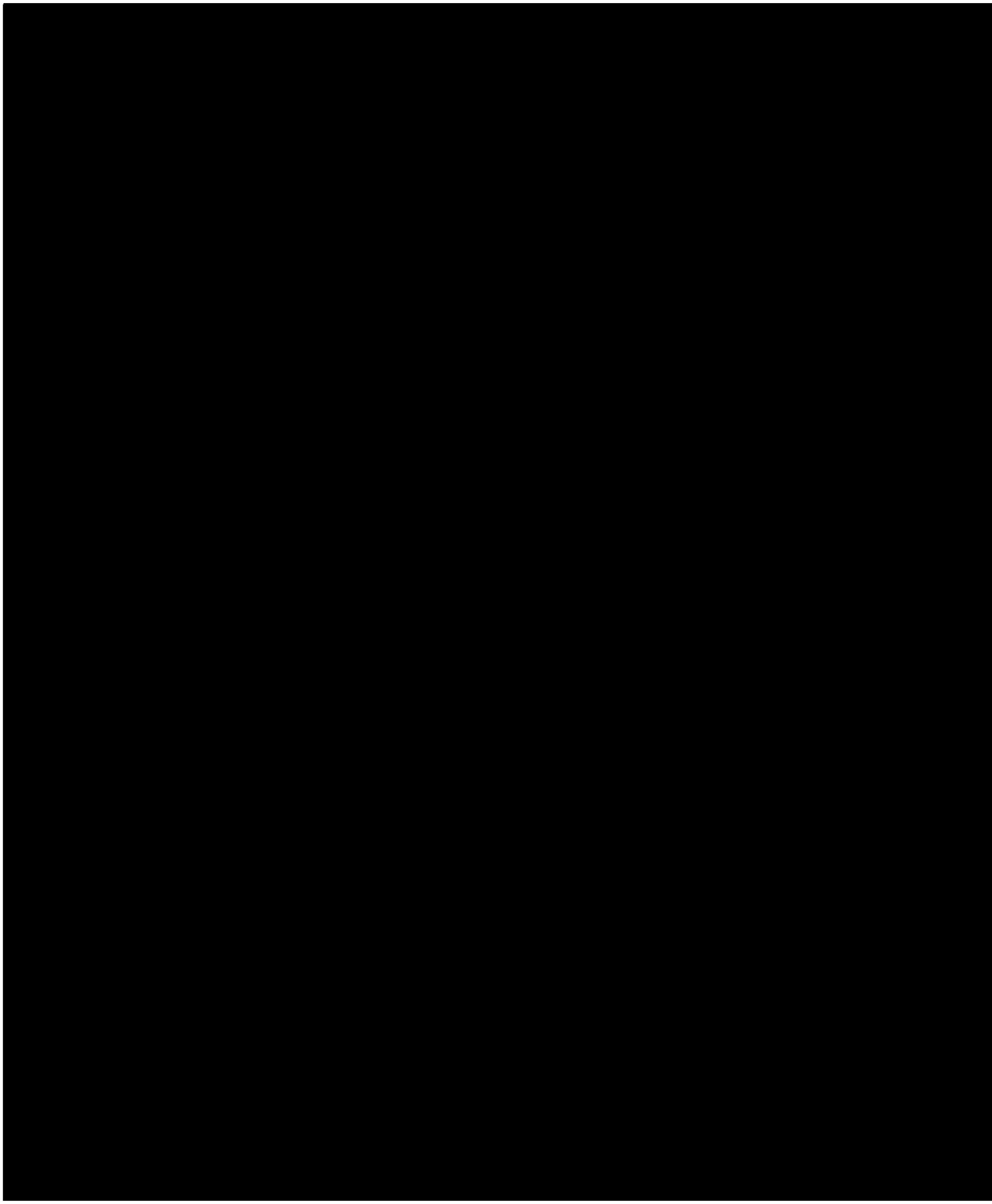


Figure 6-3 Reactor Laboratory View Facing North

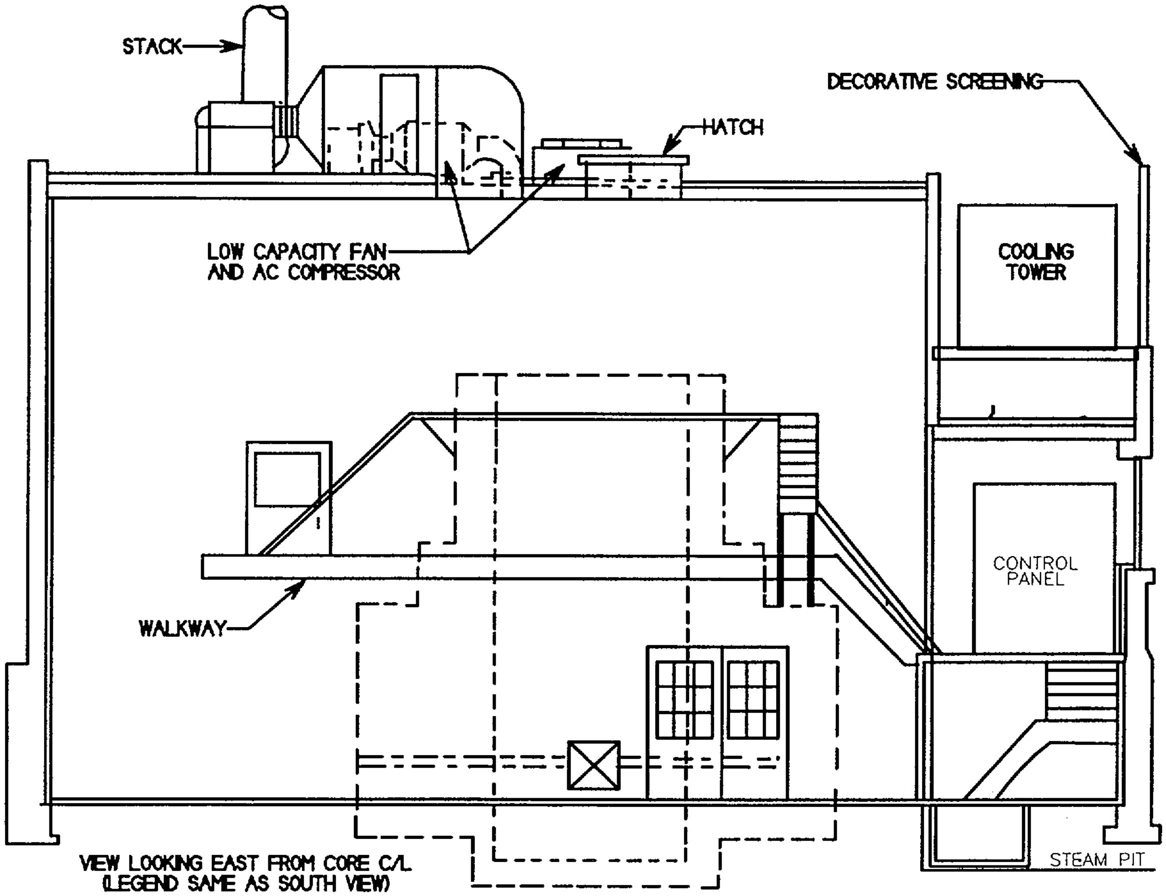


Figure 6-4 Reactor Laboratory View Facing East

Figure 6-5 Reactor Laboratory View Facing West

6.2.3 Containment

No containment is needed or provided.

6.2.3 Emergency Core Cooling System

No emergency core cooling system is required due to the low operating power.

6.3 References

There are no references for this chapter.

7 INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The reactor operates in three standard modes:

Mode 1 Manual or automatic operation at power levels up to 1,000 KW.

Mode 2 Square-wave operation (reactivity insertions to reach a desired steady state power level essentially instantaneously) at power levels between 100 and 1,000 KW.

Mode 3 Pulsed operation produced by rapid transient rod withdrawal that results in a step insertion of reactivity up to the reactivity limit established in the Technical Specifications.

A selector switch is provided to select manual, automatic, square-wave, or pulsing modes of operation.

Operation is from a console displaying all pertinent reactor operation conditions. Instrumentation is entirely analog, except for a digitally based strip-chart recorder and a digital pulse recorder used to display the power trace during pulsing operation.

7.2 Design of Instrumentation and Control Systems

Four ranges of radiation-based power instrumentation, with significant overlap, are provided to cover the operating range from source level to the maximum permitted pulse power. Fuel temperature is also measured and used by the reactor protection system. In addition, other process variables are measured, but not used in the reactor protection system. **Figure 7-1** shows the instrumentation and control system for the UWNR.

7.2.1 Design Criteria

The instrumentation and control system provides the following functions:

- Provides the operator with information on the status of the reactor
- Provides the means for insertion and withdrawal of control elements
- Provides for automatic control of reactor power level
- Provides the means for detecting over-power or fuel over-temperature and automatically scram the control elements to terminate the condition
- Provides auxiliary trip functions based on possible loss of operability of the channels providing the overpower protection
- Provides a record of operation and radioactivity discharged from the stack

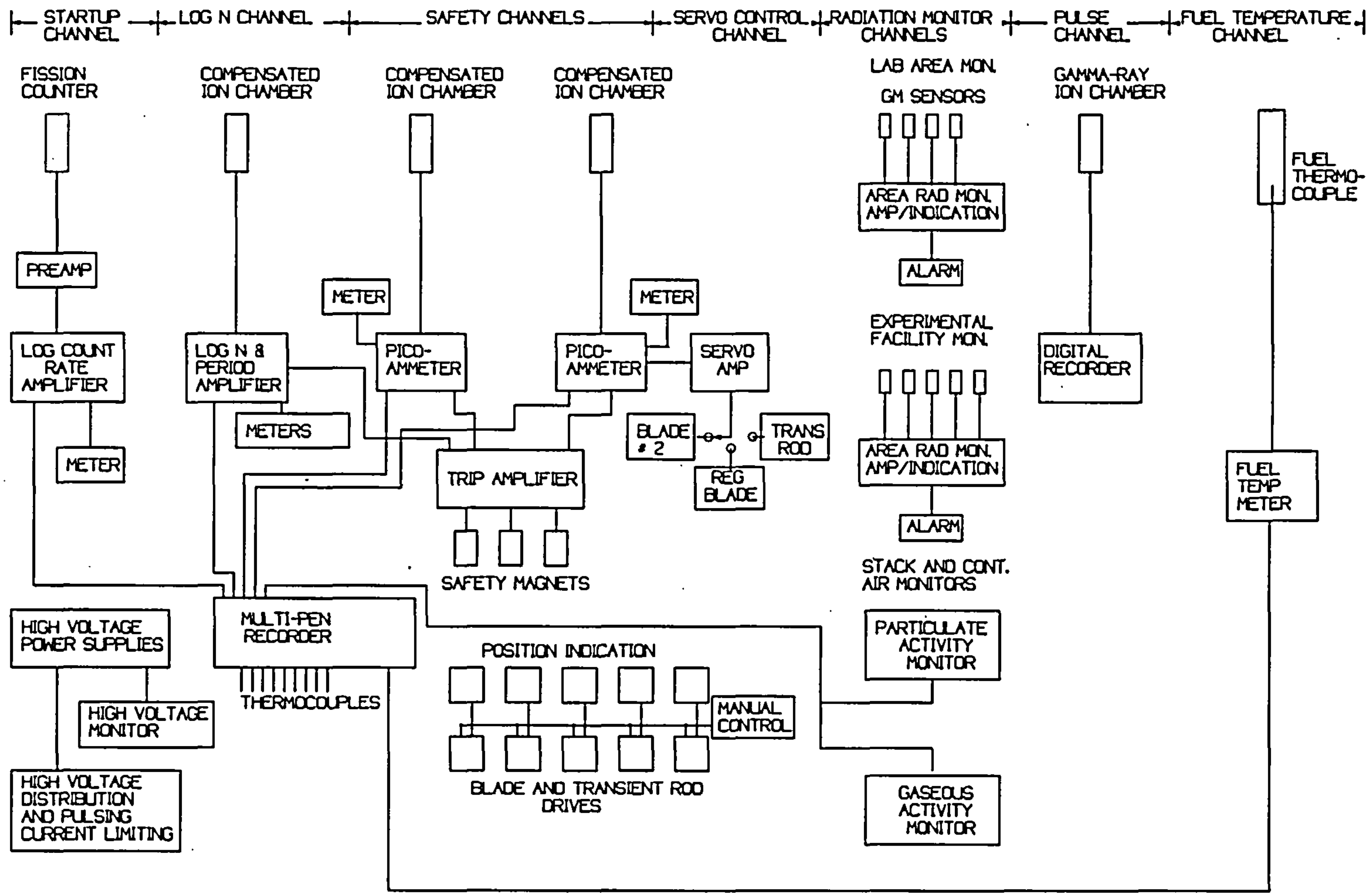


Figure 7-1 Instrumentation and Control Block Diagram

7.2.2 Design-Basis Requirements

The primary design basis for TRIGA reactor safety is the safety limit on fuel temperature. A trip on high fuel temperature is set to assure that the fuel temperature will not be exceeded. Since fuel temperature measurement includes time lag due to thermocouple response time, a reactor trip based on reactor power level as measured by a neutron sensing system is provided.

7.2.3 System Description

Start-up Channel

As shown in **Figure 7-1**, the sensing element for this channel is a fission counter with a drive which can be positioned by the console operator. The counter has a range from 2 nv to 10^6 nv. Since the counter is moveable, its effective range is thus from about 2 micro-watts to 2 MW. The pulses from the startup counter are amplified and converted to a logarithmic count-rate displayed on a meter and recorded. The amplified pulses may also be sent to a scaler that is used for subcritical measurements. The amplifier includes a bistable which allows control element withdrawal only if the count rate is greater than 2 counts/second. Another bistable provides protection to the fission counter by preventing insertion of the fission counter drive when the count rate is too high. The start-up channel, in the full in position, overlaps the low end of the safety channel range instruments.

Log N - Period Channel

This channel monitors the power level of the reactor over the range from 0.1 watt to full power. The Log N - period amplifier detects the signal from a compensated ionization chamber and amplifies the signal to provide a 7-decade logarithmic display proportional to power level. The amplifier also extracts period (startup rate) information. The Log N signal is recorded and operates a bistable used in pulse and square wave modes to prevent firing the transient rod when above 1 kW. The period signal is recorded, displayed on a meter on the console, and fed to the automatic control channel when the mode switch is in AUTO mode.

Pulse Power Channel

Current from a gamma ionization chamber (or an uncompensated neutron ionization chamber) is fed to a digital data acquisition channel. In "PULSE" mode, a signal concurrent with firing the transient rod causes data to be recorded and displayed on a monitor. Information on peak power and integrated power in the pulse is automatically computed and displayed. Peak fuel temperature after the pulse is recorded by the main console strip-chart recorder.

Safety Channels

Two safety channels monitor reactor power level from about 0.1 watt to full power. The signal from each channel originates in a compensated ionization chamber. The chamber signal is fed into a solid state picoammeter. The picoammeter includes normally open relay contacts which open on an overpower condition to cause a reactor scram. Should either or both safety channel scram signals be present, the reactor shuts down. The power level scram trip point is set to 1.25 times the maximum operating level. The safety channels provide an additional interlock signal to pulse and square wave logic to prevent firing the transient rod unless the picoammeters are on the full-power range.

Temperature Measurements

Fuel element internal temperature is indicated at the console. It causes an alarm and scram at the limiting safety system setting.

The temperature of the bulk pool water is measured at the core inlet by a thermocouple. This temperature is indicated on a recorder and causes an alarm and a scram on high temperature.

Primary cooling, intermediate cooling, and campus chilled water systems inlet and outlet temperatures, and demineralizer inlet temperature are indicated on the system temperature recorder. An alarm on this recorder indicates excessive temperature at any of these points.

7.2.4 System Performance Analysis

The instrumentation and control systems have been in routine operation for the almost 40 years of operating history. All of the measuring instruments have been replaced over the years with instruments incorporating advances in electronics but meeting or exceeding the original design criteria. The switch to entirely solid-state electronics has resulted in marked improvement in stability and reliability.

Limiting safety system settings, limiting conditions for operation, surveillance requirements, and action statements concerning the control and instrumentation systems are detailed in the proposed Technical Specifications in Chapter 14.

7.2.5 Conclusion

Operation during the term of the license has shown the instrumentation and control system to be capable of performing all intended functions with excellent stability and reliability. The system is expected to continue to perform the intended functions.

The safety limits of fuel temperature and reactor power are adequately protected by the safety channels and the fuel temperature channels, each of which will cause a reactor shutdown if the limiting safety system settings are exceeded. Additional components which cause trips on loss of conditions necessary for continued power operation (loss of high voltage to detectors, loss of water from the pool, LogN channel not selected to operate mode, pool water temperature above the temperature used in calculation of event consequences, loss of electrical power) provide assurance that the primary protection equipment will operate as planned.

7.3 Reactor Control System

Mode Switch

The mode switch and associated logic circuits provide the following capabilities and operating restrictions for the different positions. The switch is a cam operated switch which is rotated through consecutive positions. From " MANUAL", rotating clockwise places the switch in "AUTO" mode; rotating counter-clockwise places the switch first in "SQUARE WAVE" mode, then in "PULSE" mode. Returning from "PULSE" to "MANUAL" requires going through the "SQUARE WAVE" position. **Table 7-1** details the conditions and restrictions invoked in the different mode switch positions.

Manual Operation

For manual operation the control elements are slowly withdrawn to obtain the desired power level. At this level the reactor may continue to be operated manually or it may be switched to automatic control. The automatic control channel maintains power level by servo control of the regulating blade, transient rod, or #2 control blade. **Figure 7-1** shows a block diagram of the control system.

Square Wave Operation

This mode is provided for those applications which require that the power level be brought rapidly to some high level, held there for a period of time, and then reduced rapidly producing a square wave of power.

In the square wave mode the reactor is brought to a level of less than 1000 watts in the manual mode. The mode switch is then changed to the square wave position. Changing of the mode switch to the "PULSE" position removes an interlock that prevents application of air to the transient rod unless the transient rod is in the full "IN" position. A preadjusted step reactivity change is then made to bring the reactor to preset power levels between 100 and 1000 kW. The reactivity step change is made with the transient rod. The automatic control system inserts additional reactivity required to maintain the preset power level as the fuel heats up. The operator must manually augment the reactivity inserted by the servo because the transient rod does not have sufficient worth to overcome the power defect at high power levels. The linear power level scram is maintained at 1.25 P max. and an interlock prevents initiation of this mode if the range switch is not on the full power range setting.

Pulsing Operation

The reactor is brought to a power level of less than 1000 watts in steady state mode. The mode switch is then changed to pulsing mode. When the switch is in pulsing mode the normal neutron channels are disconnected and a high level pulsing chamber is connected to read out the peak power of the pulse on a fast digital recorder provided for that purpose. Changing of the mode switch to the "PULSE" position removes an interlock that prevents application of air to the transient rod unless the transient rod is in the full "IN" position. Fuel temperature is recorded during pulsing operation. The pulse channels are also indicated on **Figure 7-1** .

Table 7-1 Mode Switch Functions

Mode	Conditions/restrictions
Manual ("MANUAL")	Transient rod may be fired only if scram is reset and transient rod drive is at the "IN" position.
Automatic ("AUTO")	Same as Manual, except automatic control system can control power, subject to period limit
Square-Wave ("SQUARE WAVE")	Transient rod can be fired from other than full in position only if scram is reset, both safety channels are on top range, and power level does not exceed 1 kW as indicated by the LogN channel. The period channel in the LogN amplifier is defeated (restored after a short time delay when the switch is returned to "MANUAL" position). Automatic control system can control power if actual power is within ±5% of scheduled power.
Pulse ("PULSE")	Period channel remains defeated . Prohibits control blade withdrawal. If the scram is reset, both safety channels are on top range, and power level does not exceed 1 kW as indicated by the LogN channel; (a) Transient rod can be fired from other than full in position, (b) High voltage is removed from the fission counter, safety channel CICs , and the LogN CIC . (c) Signals from all CICs are directed to ground rather than to instrument input. (d) The transient rod drops automatically 15 seconds or less after it is fired (e) The pulse power level channel is sent a signal causing it to record the pulse power trace. When returning to the "SQUARE/WAVE" position, high voltage is restored to the detectors immediately and the signals to the neutron measuring instruments are restored after a short time delay (to prevent damage to instrument inputs from the transients resulting from high voltage restoration).

Control Element Operation

There are five control elements; three shim-safety blades, a transient control rod, and a regulating blade. The shim-safety blades and the transient control rod have scram capability. The following conditions must be met before any control element drive can be withdrawn (raised), either manually or by the automatic control system:

1. No scram conditions present and scram relays reset;
2. Count-rate on startup channel greater than 2 counts per second;
3. Fission Counter not in motion;
4. Console key switch set to "ON" position;

There are no interlocks or permissives which restrict insertion (lowering) of control element drives. Insertion is accomplished by placing the individual momentary-contact control switches in the "IN" position, by a maintained-contact "RUNDOWN" switch which inserts all control element drives to the "IN" limit. The three shim/safety blade drives also automatically run to the "IN" limit when a SCRAM has occurred.

Safety Blade Control

The three safety blades are manually controlled by individual pistol-grip switches with LOWER, OFF, and RAISE positions, with spring return to OFF. One safety blade may be selected to be controlled by the automatic level control system. The position of each safety blade is indicated by separate digital read-outs, and the indicator lights on the console show when each drive is at its "IN" or "OUT" limit and when the blade magnets are engaged with the armatures. Position indication is accurate to ±0.02 inches.

The safety blades will scram from any position when stationary or during withdrawal and insertion. In the event of a scram the safety blade drives automatically run in to their "IN" limits.

Regulating Blade Control

The regulating blade has identical position indication and "IN" and "OUT" limit indication. It is manually controlled by a separate pistol-grip switch and may be controlled by the automatic level control system.

Transient Rod Control

Manual movement of the transient rod drive is controlled by the console-mounted, switch/light, push buttons which not only control movement, but also indicate in and out limits. Position indication is accurate to 0.02 inches. The transient rod drive may be selected to be controlled by the automatic level control system.

Air pressure is used to fire the transient rod to the selected position in pulse and square-wave modes and to engage and hold the transient rod at the drive position in manual and automatic modes. Additional lighted push-button switches are installed to support these functions.. Illumination of the "READY/FIRE T ROD" indicator/switch indicates that the permissives for firing are met and the rod has not been fired. Illumination of the "ENG'D/AIR" indicator/switch indicates movement of the transient rod from the full-in rest position as a result of air having been applied. (Applying air while the transient rod drive is in the full in position causes the piston in the drive to move upward by compressing the spring inside the shock absorber). Pressing the "ENG'D/AIR" switch removes the air from the drive, causing the transient rod to drop to the full-in rest position.

Since the transient rod control is capable of introducing step changes in reactivity up to the Technical Specification limit, logic circuits are provided to assure the transient control rod is fired only under the appropriate conditions.

Automatic Level Control System

The servo amplifier (level controller) controls reactor power level in automatic and square-wave modes. The servo amplifier output drives either the regulating blade, transient rod, or a safety blade as selected by a servo element selector switch on the console. Only one element at a time may be selected; selecting another replaces the previously selected element. The servo amplifier responds to a power level signal from one of the safety channel picoammeters and controls speed and direction of the servo element through a servo motor.

In automatic mode, the automatic level control channel uses period information from the Log N - period channel to limit control element withdrawal to maintain a period longer than a pre-selected level. In square-wave mode a servo error circuit is employed. This circuit allows servo operation only when the servo error is less than 5%. Servo error (the difference between scheduled power in % and the picoammeter indication in % of full scale) is indicated at the console in both "square-wave" and "automatic" modes.

Additional indicators on the reactor control cubicle are provided to indicate "AUTO ON" and scheduled power.

7.4 Reactor Protection System

Scram Circuits

The scram circuit, which initiates shutdown by dropping the shim-safety blades and the transient rod, is shown in **Figure 7-2** . Scram is accomplished by de-energizing the scram relays under one of the following conditions:

1. Manual scram;
2. Fuel temperature above LSSS;
3. Power level greater than 1.25 P max;
4. High voltage failure in control console;
5. Loss of control power;
6. Log N - period amplifier calibrate switch not in input position;
7. Coolant temperature at core coolant entrance above 130°F;
8. Pool water level low.

In addition to these scram functions, the transient control rod logic includes a timer which causes the transient rod to drop to the full in position within 15 seconds after the transient rod has been fired in pulse mode only.

The key-operated console MASTER switch, designated 4S2 on the figures, is a cam-operated switch with a large number of contacts which are selectively operated in the three switch positions. (OFF, ON, and TEST). Six different contact sets must be closed (and are closed only in the "ON" position of the switch) in order to reset the scram relays and apply power to the trip amplifier which supplies DC voltage to the shim-safety blade magnets.

A normally-open contact of Relay 6K1A in the transient rod logic circuit opens when the relay is de-energized, resulting in the drop of the transient rod.

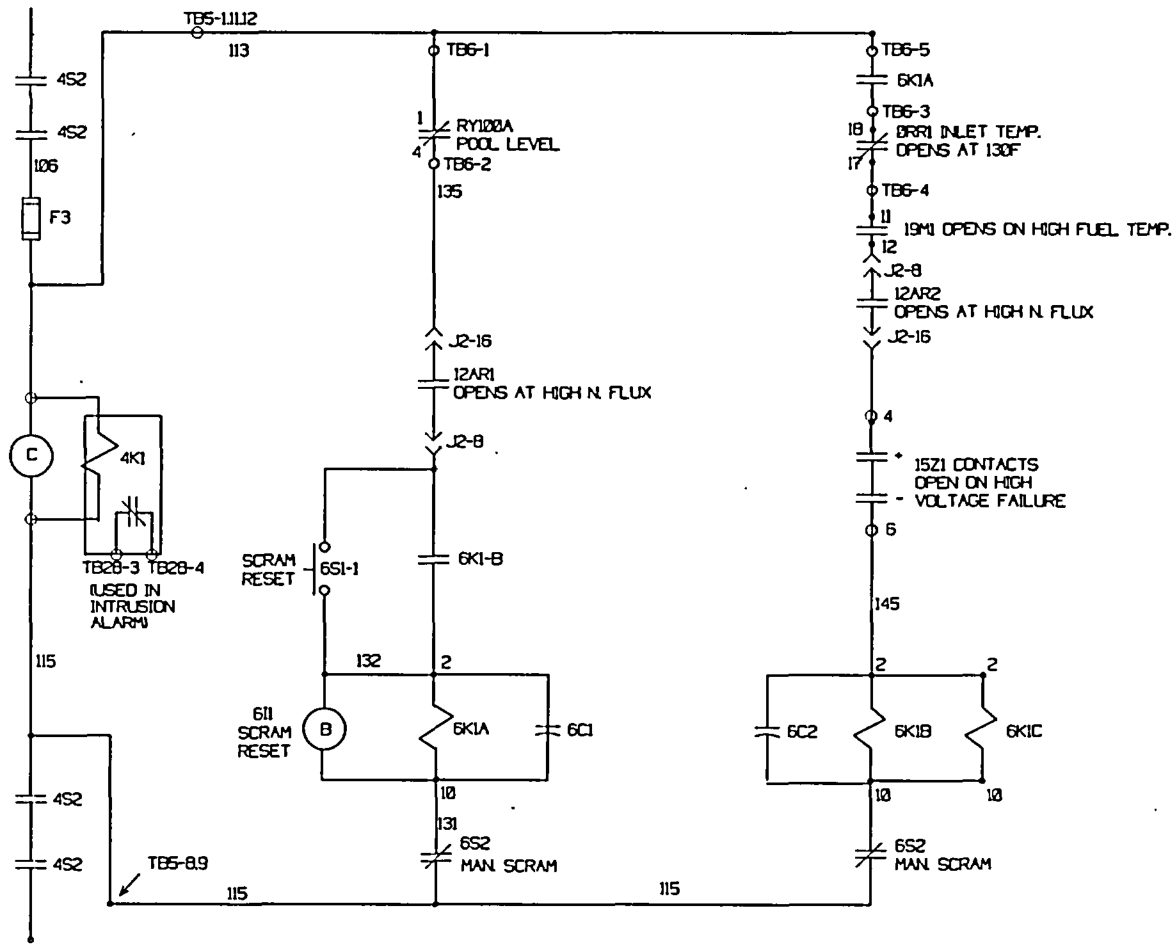


Figure 7-2 Scram Logic Elementary Diagram

Normally-open contacts of both 6K1A and 6K1B are in series with the alternating current power supply to the trip amplifier as shown in **Figure 7-3** . Magnet power is turned off if either or both of the scram relays de-energizes.

7.5 Engineered Safety Features Actuation Systems

There are no engineered safety features actuation systems.

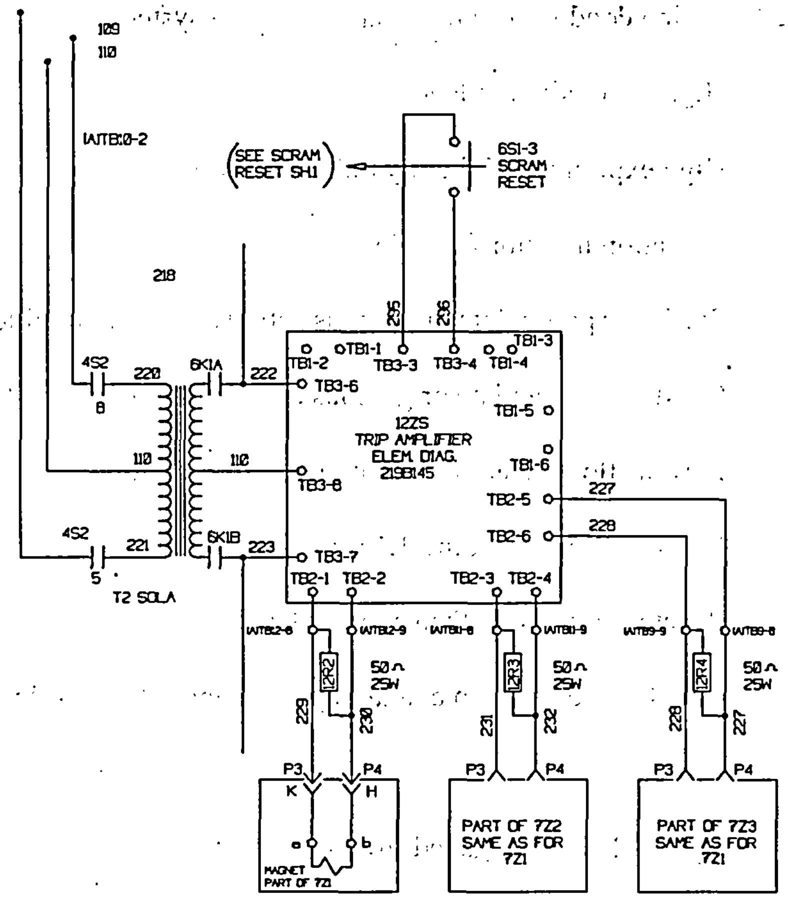


Figure 7-3 Magnet Power Supply

7.6 Control Console and Display Instruments

Alarm and Indicator System

When an abnormal condition develops, an audible signal sounds and a lighted annunciator begins to flash rapidly. The operator may press the acknowledge button to silence the audible signal, at which time the audible signal stops and the lighted annunciator goes to steady illumination. When the condition is corrected, the lighted annunciator goes to slow flash, and the light is extinguished when the operator presses the Reset button.

The following conditions will actuate the alarm system:

1. High area radiation level;

2. High experimental facility radiation level;

3. Radiation monitor failed low;

4. Stack Air particulate or gaseous activity above normal level;

5. CAM Air particulate or gaseous activity above normal level;

6. Low air flow in stack or continuous air monitors;

7. Neutron flux exceeding 1.15 times the normal value;

8. Reactor period less than a preset level;

9. Count rate on startup counter approaching saturation level;

10. Any scram;

11. Safety blade disengaged from magnet;

12. Failure of high voltage power supply;

13. Water level in pool two or more inches above or below normal (also gives an alarm at Police and Security Headquarters);

14. Fuel element temperature high;

15. Core inlet temperature above preset level;

16. Cooling system temperatures above preset level;

17. Thermal Column door open;

18. Chain switch across stair actuated or entry to High Radiation Area;

20. Hold tank full;

21. Pneumatic tube blower on;

22. Intermediate coolant system low pressure.

To provide operating information for the reactor operator, the following indicator lights are provided:

1. Scram reset;
2. Safety blade magnet engaged;
3. Power on;
4. Control elements in (distinct light for each);
5. Control elements out (distinct light for each);
6. Automatic control on;
7. "Rabbit" in reactor.

Status of the ventilation system is indicated by fan-running lights for the room exhaust, beam port and thermal column exhaust, and emergency exhaust fans.

Operating controls and indicators at the control console for the cooling system include switches to operate primary, intermediate, chilled water, and diffuser pumps (the hydraulic irradiation facility pump starts when the diffuser pump starts) along with indicators which light when the pumps have discharge pressure. In addition, a pressure switch on the intermediate coolant system provides indication that the system is pressurized.

7.7 Radiation Monitoring Systems

Radiation Monitors

The radiation monitors are arranged into three systems; the primary area monitors, experimental facility area monitors, and air activity monitors.

The primary area monitors are located as follows:

1. Demineralizer area;
2. On the reactor bridge about one foot above the water surface;
3. Beside the thermal column door;
4. In the control console area.

All Area Radiation monitor units have ranges from 0.1 to 10000 mr/hr.

Unit 1 supplies information on radiation level from the demineralizer. It is set to alarm at a radiation level just above that expected in a normal run. Unit 2, located just above the pool water level, alarms at a radiation level just above that reached during full power operation. Unit 3 is located beside the thermal column. It too is set to alarm just above normal operating level. This unit will give an alarm if the thermal column door is left open when the reactor is operated at any substantial power. Unit 4 indicates the dose rate in the console area. The 4 units indicated above are connected to the Reactor Laboratory evacuation alarm. An alarm from one of these units will sound the evacuation alarm if it is not corrected by the operator within 30 seconds (See procedure UWNR 150).

The Experimental Facility Area Radiation Monitor is an area radiation monitor system installed to preclude the possibility of unknowingly generating high radiation levels by operating the reactor at high power levels with the beam ports open, or by return of an intensely radioactive pneumatic tube sample. The sensors for this system are installed on the walls of the Reactor Laboratory in direct line with the beam ports and at the pneumatic tube send-receive station. The system gives visual and audible alarms at the console if the radiation level exceeds a preset value. The pneumatic tube monitor also provides local alarm and indication. The monitors are normally set to alarm at a radiation level equivalent to a dose rate of 50-100 mrem/hr at the beam port flange (10 mrem/hr at the detector location). The pneumatic tube monitor also is normally set to 10 mrem/hr, but the pneumatic tube operating procedure states that it may be set to a higher level if calculated sample activity is expected to result in a higher reading.

The stack air monitor measures both particulate and gaseous activity of the air discharged from the stack. Particulate activity is collected on a filter tape and counted with a thin end-window GM tube and count-rate meter. Gaseous activity is measured with a large Kanne ionization chamber feeding a picoammeter/integrator. The system, therefore, operates by detecting β activity. Both particulate and gaseous activity levels are recorded, and provide annunciation should preset levels be exceeded. In addition, gaseous activity levels are integrated to provide a record of total gaseous activity discharged from the stack.

The sensitivity of the particulate activity monitor allows detection of concentrations of about 1.0E-10 μC/ml of a material with a single β particle emitted per disintegration. The efficiency is higher if more than one β particle is emitted per disintegration. The sensitivity of the gaseous activity monitor is such that a concentration of about 1.0E-6 μCi/ml of Ar^{41} at the stack discharge can be detected by the instrument. The efficiency varies with the number of β particles emitted by the isotope being detected. The primary activity expected to be present in the stack discharge is Ar^{41} activity.

An identical instrument, provided as a backup for the Stack Air Monitor, is operated as a Continuous Air Monitor. It samples the atmosphere immediately above the surface of the reactor pool, although it can be made to sample other locations when desired. The backup air activity monitor can be connected to the stack monitor flow path, should the stack monitor fail.

7.8 References

There are no references for this chapter.

8 ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

There are no electrical power supplies that are critical for maintaining the facility in safe shutdown, even for extended periods of time.

The electrical power distribution for the Reactor Laboratory is quite complex, as the facility was built into an existing building and initial electrical power sources were appended to those for the building, using the most economical power routing and distribution path at the time. Additional capability was installed each time substantial amounts of new equipment was installed. For this reason the 120 VAC equipment is powered from diverse services; no single breaker or distribution panel provides isolation of all of the 120 VAC power to the laboratory. All services except for the 480 VAC 3 phase service originates in transformers and distribution panels in the north wing of the Mechanical Engineering Building (including a transformer bank located underground but outside the walls of the building). This equipment is located in rooms 50 and 52 of the Mechanical Engineering Building. Reactor laboratory personnel do not have access to these rooms and do no control manipulations on these services.

480 VAC 3 Phase Electrical Power

A single breaker [redacted] in [redacted] Engineering Research Building feeds all of the Mechanical Engineering 480 VAC services associated with the Reactor Laboratory. A panel in [redacted] Mechanical Engineering serves the Mechanical Engineering services, with breaker 7 in that panel providing the feed to the Reactor Laboratory which connects into the 480 VAC panel located immediately behind (east of) the Reactor Console.

Equipment served by this panel includes (1) the N-16 Diffuser and Whale Pumps (same breaker but separate local disconnect switches); (2) The Primary Coolant System Pump; (3) The Secondary Cooling System Heat Exchanger Pump; (4) The Secondary Cooling System Tower Pump; (5) the West Cooling Tower Fan; and (6) the East Cooling Tower Fan. Each of these pumps or fans can be remotely started from a graphic panel on the Reactor Console.

240 VAC 3 Phase Electrical Power

This service enters the Reactor Laboratory through the east wall and supplies the loads with disconnects and controllers located on the large plywood panel just to the north of the center of the west wall of the laboratory. These loads include the main laboratory air conditioning system (including the three convectors), the Beam Port and Thermal Column Ventilation fan, the Room Exhaust fan, the Emergency Exhaust fan, the air motor that opens the louvers for the Emergency Exhaust, and the overhead crane. This service also supplies power to the 3-phase outlet on the east end of the pool curb.

240/120 VAC 1 Phase Electrical Power

This service enters the Reactor Laboratory through the east wall and supplies Electrical Panel #2, where it is split to provide most of the 120 VAC loads inside the Reactor Laboratory. Breaker 1 supplies the Reactor Console, Breaker 19 supplies the Waste Storage (Hold) Tank pump, and the remaining breakers supply the lighting and 120 VAC outlets in the Reactor Laboratory. Some of the 120 VAC outlets on the west side of the Reactor Laboratory are supplied from a breaker panel on the east wall of room 5 Mechanical Engineering.

All 240 VAC 1 Phase loads on this service are supplied from Electrical Panel #1, and include the demineralizer pump, the motor for the thermal column door, and the air conditioner unit for the control room.

Some reactor equipment is located in the 40 area of the Mechanical Engineering Building basement west of the Reactor Laboratory. This area has four electrical services, including a 230 VAC 3 phase service and three 120 VAC single phase services. The pneumatic tube blower as well as some shop equipment is powered with the 240 VAC 3 phase service. The pneumatic tube control system and the stack and continuous air monitors are powered from one of the three 120 VAC services.

8.2 Emergency Electrical Power Systems

There are no emergency electrical power systems.

9 AUXILIARY SYSTEMS

9.1 Heating, Ventilation, and Air Conditioning Systems

Heating and cooling systems are designed so that they do not cause interchange of the atmosphere within the Reactor Laboratory with surrounding areas.

The ventilation system is designed to prevent the spread of airborne particulate radioactive material into occupied areas outside the Reactor Laboratory. It removes particulates with high-efficiency filtration and assures that all releases of either gaseous and particulate activity are monitored and discharged at an elevated release point. Accidents which might result in discharge of radioactive material from the stack are discussed elsewhere in this report, and remarks may be found there indicating the concentrations which might be expected. In addition, a portion of the ventilation system vents the beam ports, thermal column, and liquid waste holdup tank to assure that air flow is from the Reactor Laboratory into these facilities.

Heating and Air Conditioning

The Reactor Laboratory is heated by the exposed steam pipes running along the east wall of the room and by steam heated convectors located in the laboratory. The convectors circulate the air in the room, and do not cause exchange of air with other areas.

Two air conditioning systems are provided. The first, of about 5-ton capacity, is controlled by a thermostat located in the control console and provides cooling air and humidity control for the instrumentation. Part of the cooled air from this unit is exhausted into the enclosed control room area of the Reactor Laboratory. The system brings in no outside air, nor does it discharge any air from the laboratory. The second air conditioning unit, of about 20-ton capacity, cools and alleviates humidity problems in the remainder of the Reactor Laboratory. It consists of an air-cooled condenser and compressor located on the roof of the building and evaporator-convectors located in the Reactor Laboratory. This unit does not cause air flow into or out of the Reactor Laboratory.

Condensate water from both air conditioning systems goes to the building sewers. Samples may be obtained for assay.

Ventilation

The ventilation system for the Reactor Laboratory consists of three subsystems; Room Exhaust, Emergency Exhaust, and Beam Port and Thermal Column Ventilation. All three sub-systems exhaust into the Reactor Stack which discharges above the roof of the east wing of the Mechanical Engineering Building at a point 17 meters above ground level. The stack air activity monitor continuously monitors the stack to maintain a record of radioactivity in the air being discharged. The ventilation system is shown schematically in **Figure 9-1** , while location of major components is indicated in **Figures 6-3** through **6-5** of Chapter 6. All fans and filters are mounted on the roof, and the filtration takes place on the suction side of the fan.

Room Exhaust System

The roof-mounted room exhaust fan has a capacity of about 960 cfm at 1.5 inches suction pressure. It is normally "ON" to assure that any air flow is from adjacent areas into the Reactor Laboratory. The fan takes its suction through a roof penetration on the east side of the laboratory. It discharges through a filter package consisting of a pre-filter and a high-efficiency filter rated at 99.97% efficiency for 0.3 micron particles. Filter pressure drop is indicated within the reactor laboratory. Minimum flow rate through the filters with the room closed is approximately 600 cfm.

Beam Port and Thermal Column Ventilation System

The Beam Port and Thermal Column Ventilation system is designed to sweep out the Ar^{41} activity present in an experimental facility when the facility is opened. During ordinary operation the experimental facilities are closed and there is an essentially zero rate of discharge. When a beam port flange or the thermal column door is opened there is a slug of activity discharged. The average concentration discharged will, therefore, be extremely low due to dilution by the other blowers and the fact that no activity is discharged most of the time. Section 11.1.1.1 of this report discusses the levels of activity discharged.

The Beam Port and Thermal Column Ventilation blower has a rated capacity of 960 cfm at 1.5 inch suction pressure and is sized to maintain an air velocity of about 40 feet per minute into all beam ports and the thermal column, should all be opened simultaneously. Normal flow rate with the system sealed is about 360 cfm. The thermal column shielding door is weather-stripped to maintain a nearly airtight seal. An air-operated flapper valve at the end of the duct connected to the thermal column and beam port vents is normally open. This maintains a slight negative pressure within the thermal column when the door is closed, but prevents the full static suction of the fan from forcing air in through the thermal column door seals. When the thermal column door is opened, however, the flapper valve closes and full system suction is impressed on the thermal column to cause the desired in-flow of air. In addition, a ball check valve within the thermal column vent prevents mixing of the air in the thermal column with the ventilation flow except when the door is not sealed. The facility ALARA program identified this feature as one which resulted in a significant reduction in activity discharged through the stack without increasing the Ar-41 dose within the laboratory. Also incorporated into the system for ALARA considerations are ball-check valves in the vent line from each beam port. Because the pressure drop in the BP&TC Ventilation System duct causes lower pressures in beam ports closer to the blower, the common drain connection for the four beam ports allows air flow through the drain connections from higher to lower pressure beam ports. The check valve prevents flow into the beam port vents, thus preventing circulation through the drain system and substantially reducing the amount of Ar-41 discharged to the atmosphere.

The effluent from the system is filtered and discharged into the Reactor Laboratory stack. Filter pressure drop is indicated in the laboratory. The filter and blower are at roof level to prevent leakage of water should a beam port rupture and fill with water.

The vent of the waste holdup tank is also connected to the Beam Port and Thermal Column Ventilation system in order to assure any gaseous effluent from the holdup tank is discharged through a monitored release path. In addition, the stack air activity monitor discharges the sample stream extracted from the stack into this ventilation system.

Emergency Exhaust Fan

The larger fan (the emergency exhaust fan) is for use in those cases in which it is considered desirable to completely change the air in the Laboratory to prevent spread of contamination to adjacent occupied areas. The fan takes a suction on the Reactor Laboratory through a set of air-operated louvers mounted in the roof just north of the reactor pool. (The louvers are necessary to prevent condensation in the filter bank and ducts during cold weather). The fan has a free air capacity of about 9600 cfm and delivers greater than 5000 cfm through the high-efficiency filters with the room closed. This filter bank does not have pre-filters.

This fan may also be operated with a door in the duct between the filter bank and the fan manually opened to force large quantities of unfiltered outside air through the stack to dilute air exhausted from the stack to permissible concentration of radioactive materials. Use of the emergency exhaust fan is governed by the Emergency Plan, which states that the decision to operate the fan should be reached by common consent of the Emergency Coordinator and the campus Health Physics organization.

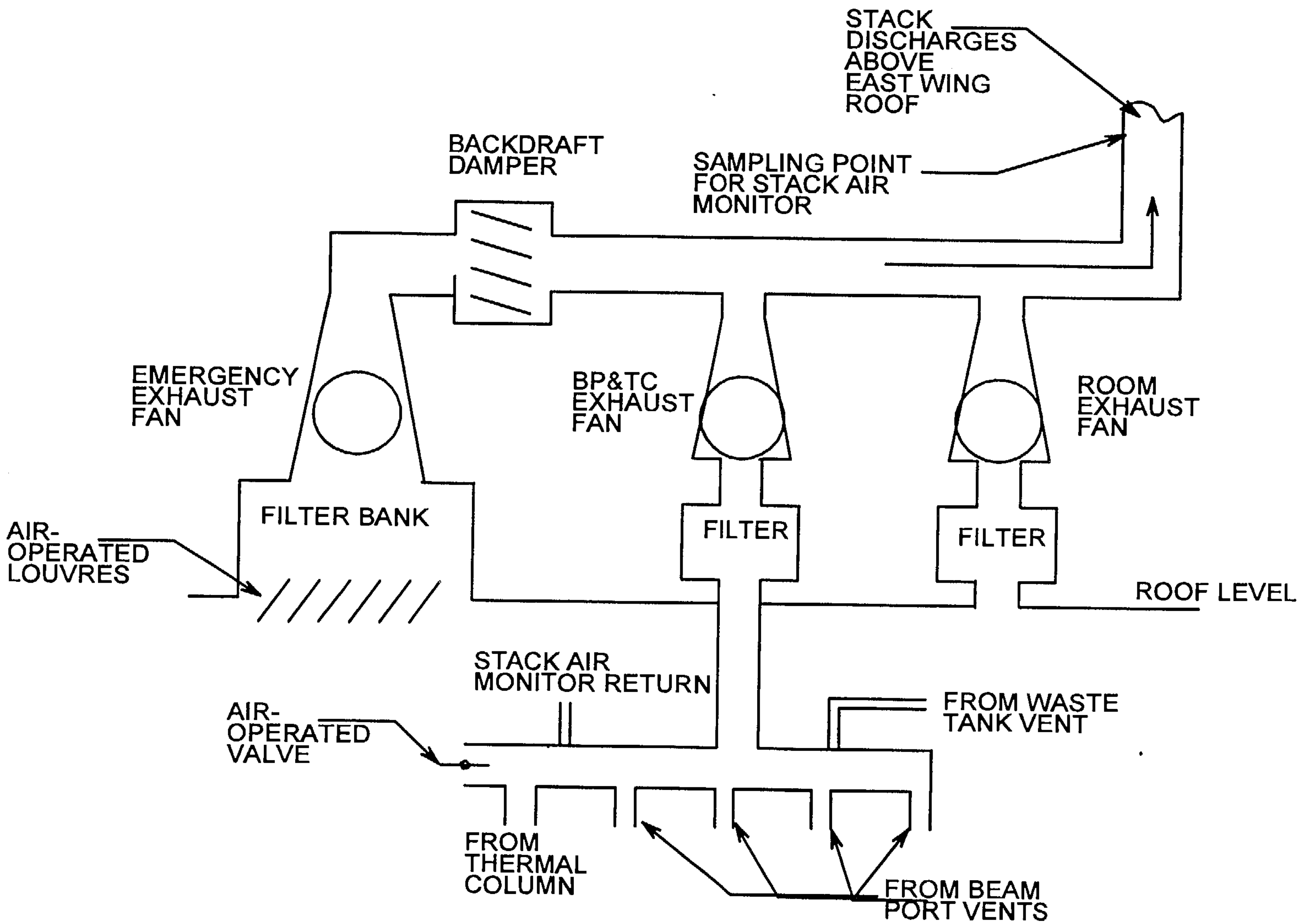


Figure 9-1 Ventilation System Schematic

[redacted]

Figure 9-2 Fuel Storage Positions

9.2 Handling and Storage of Reactor Fuel

9.2.1 Fuel Handling

[redacted]

Individual fuel elements (excepting instrumented elements which are handled using the conduit protecting the thermocouple leads) are seldom handled in the facility once initial assembly into bundles takes place. [redacted] This tool is used primarily in connection with the maintenance and measurement tool discussed in Section 9.2.3.

9.2.2 Fuel Storage

New (unirradiated) reactor fuel can be handled manually and is stored in a steel safe welded to the pool catwalk handrails. Those needing further details on new fuel storage should consult the facility security plan.

Sufficient storage room is provided for all on-site irradiated fuel. The fuel storage locations are indicated in **Figure 9-2** . All fuel storage facilities are designed to allow sufficient convective water flow to remove decay heat.

A fuel storage pit [redacted] with an aluminum-clad lead shielded cover contains three aluminum fuel storage baskets constructed with aluminum clad cadmium poison plates. These baskets hold up to 60 fuel bundles for storage. The multiplication constant (K_{eff}) for the three storage baskets with full complement of fuel elements is less than 0.8.

Three additional aluminum fuel racks, each holding up to 9 fuel bundles, are attached to the pool wall at about the level of the grid box. Although no neutron absorbers are built into these racks, their geometry keeps the multiplication constant for these racks, when fully loaded, to less than 0.8.

9.2.3 Fuel Bundle Maintenance and Measurements

Fuel bundle disassembly, assembly, and fuel element bow and elongation measurements are conducted underwater using a special tool (shown in **Figure 9-3** with a dummy element installed) designed for both these purposes. This tool is used under about 18 feet of water, where [redacted]

Required measurements of fuel element bow and elongation also are made with the maintenance and measurement tool. Because of the excessive time and added handling required to disassemble the bundle and measure each element in a separate measuring tool, the tool was designed to make the measurements without disassembly. **Figure 9-4** shows the three sensors employed (a portion of the housing is removed in this view.) Each uses a differential transformer as a transducer to give a remote electrical output proportional to displacement of the sensors.

The X and Y sensors employ spring-loaded aluminum wheels attached to the differential transformer cores. When the bundle is lowered into the tool the wheels are forced back and they then ride on the fuel element clad surface. These sensors are adjusted to give a zero signal for a standard fuel element dummy.

The length sensor differential transform is actuated by one lobe of a cam. A second lobe of this cam is rotated into contact with the top edge of the fuel element cladding by a leaf spring attached to the operating rod. The cam pivots out into the measuring position only when the operating rod is fully withdrawn. The length sensor is also adjusted to zero output for the standard fuel element dummy.

A readout device is positioned on the pool curb or reactor bridge and connected to the underwater portion of the tool. Differential transformer core position is indicated by a meter for length measurements, and a recorder output is provided to the horizontal axis of an X-Y recorder for the bow measurements.. Polarity is set so an increase in element length or a bow away from the center-line of the bundle gives a positive meter indication or recorder readout.

The dummy fuel bundle has one dummy element exactly 0.100 inches longer than the other three elements. This element also has a section in which the radius has been reduced by 0.060 inch. By using this element the attenuation and zero controls on the readout box may be adjusted to give a calibrated readout of bow and length.

After calibration of the tool, measurement can be made on standard fuel elements. The standard setup used gives a 1 cm horizontal displacement for 0.060 inch transverse bend (bow) and a meter reading of length in thousandths of an inch deviation from the dummy element reference length. A trace is drawn for both the X and Y sensors while the length measurement meter reading is manually recorded on the form. A complete set of measurements for all four elements in a bundle can be completed in about twenty minutes.

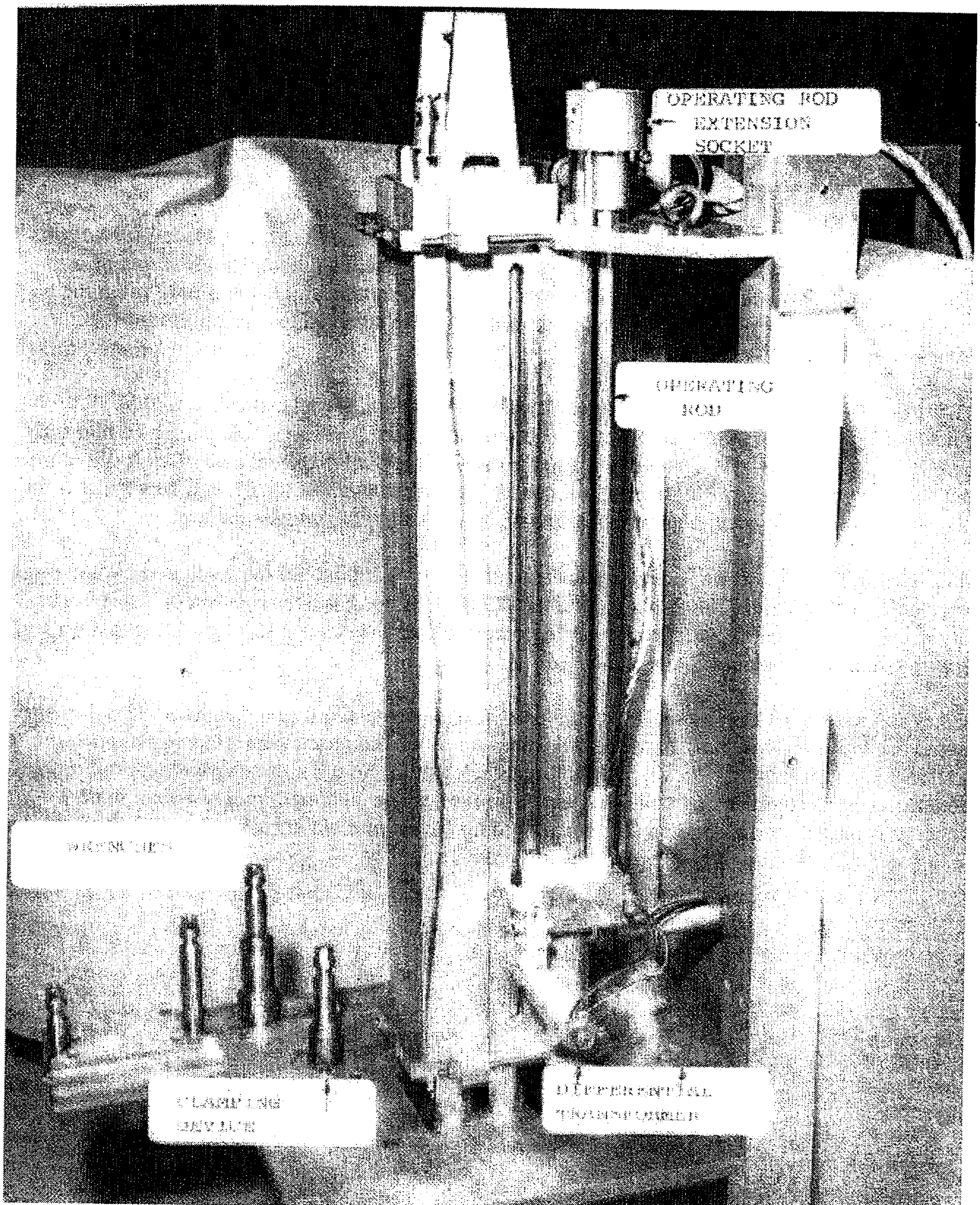


Figure 9-3 Fuel Maintenance and Measuring Tool

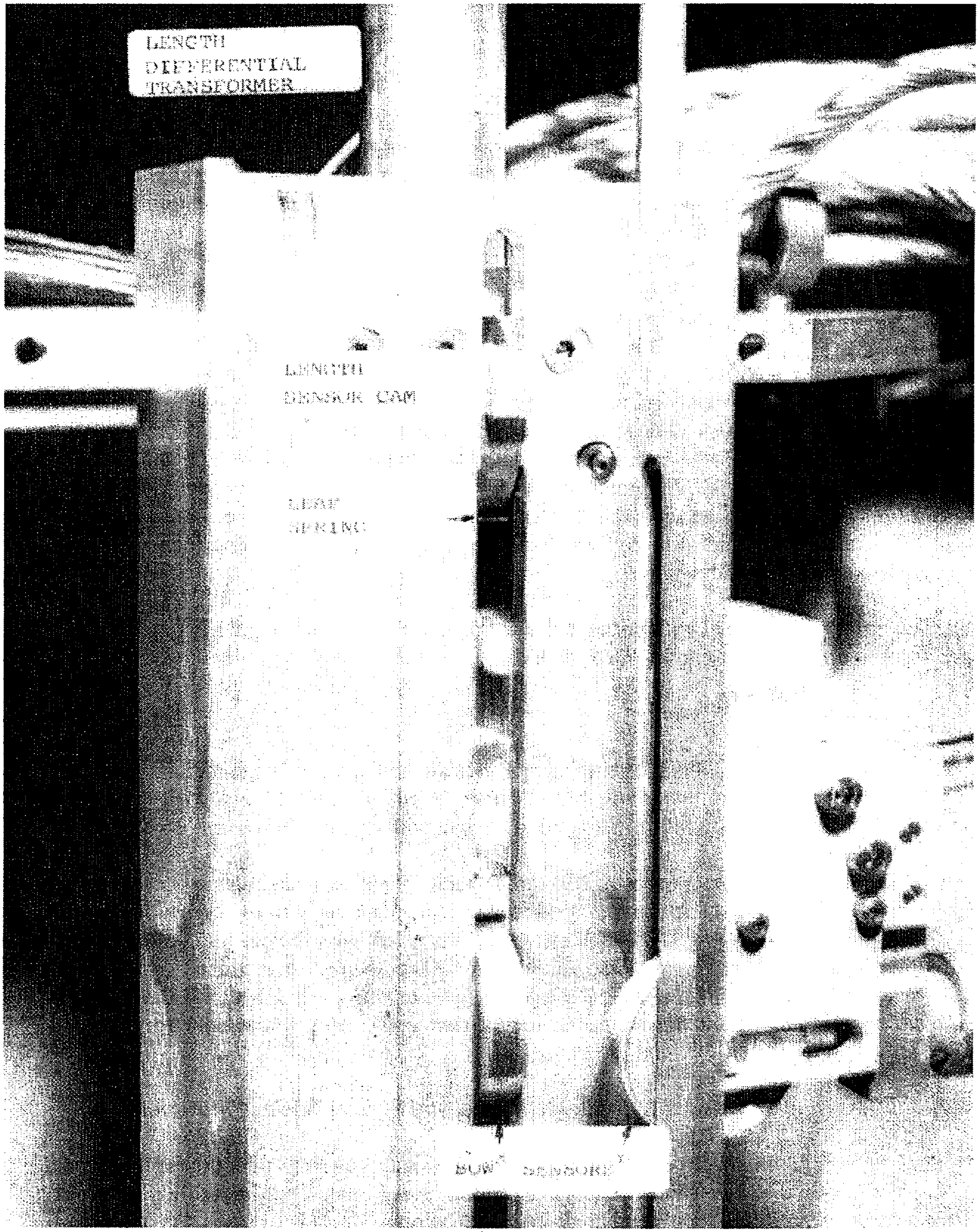


Figure 9-4 Bow and Elongation Sensors

Since the X and Y readouts are 90° apart, the maximum possible bow will be the square root of the sum of the squares of the bows indicated by direct measurement. As long as neither measured bow exceeds 0.088 inch, no calculations or other measurements are necessary. If either bow measurement exceeds 0.088 inch, then the square root of the sum of the squares of the measured bows must be calculated to determine whether or not this resultant is less than 1/8 inch. If the calculated number is less than 1/8 inch, the element is within technical specifications. Should the calculated bow exceed 1/8 inch, the crowsfoot wrench may be used to rotate the element being measure to that the reading of one bow sensor is maximized and the true bow may be determined directly to see whether it exceeds technical specification limits.

9.3 Fire Protection Systems and Programs

The fire detection and protection systems in the laboratory meet the local and state requirements. Interior doors from the Reactor Laboratory to the remaining parts of the building are also fire doors that meet local codes. All walls between the reactor laboratory and the remainder of the building are masonry. Both heat and smoke detectors within the laboratory are connected to the building fire alarm system. The fire alarm system alarms locally (for evacuation notification) [redacted]

The laboratory is equipped with portable fire extinguishers which are regularly inspected and serviced by the University Safety Department.

9.4 Communication Systems

The Reactor Laboratory and supporting laboratories are equipped with commercial telephones. Two lines are available at the reactor control center. A cellular phone is also kept in the control room except when being used for maintaining contact with the control room.

An intercom system is installed in the Reactor Laboratory and surrounding supporting offices and laboratories. This system provides two-way communications with each other station and an all-call capability for paging. A second intercom, controlled from the reactor control center, provides an additional communication link to the pool top, the pneumatic tube operation station, the Reactor Director's office, and the Reactor Supervisors office. It also allows the reactor operator to communicate with persons outside the principal doors into the laboratory.

9.5 Possession and Use of Byproduct, Source, and Special Nuclear Material

All activities using radioactive and special nuclear materials covered under the reactor license take place within the Mechanical Engineering Building and a small portion of the Engineering Research Building in the rooms and areas indicated below:

- Reactor Laboratory (Room 30/130);
- Room 132 (ground floor west of Reactor Laboratory);
- Rooms 140A-J (ground floor west of Reactor Laboratory);
- Room 5 , including the individually enclosed and lettered rooms 5E and 5F within room 5 (basement level east of Reactor Laboratory);
- Rooms 40-47 (basement level west of Reactor Laboratory);

the portion of the basement hallway of the Engineering Research Building (including the ramp) between Room 5 Mechanical Engineering Building and the ramp up to ground level (this ramp is the heavy object access route to the Reactor Laboratory over which spent fuel shipments must travel).

Radioactive and special nuclear material use outside these areas is conducted under license 48-09843-18, the University of Wisconsin Broad Scope License.

9.6 Cover Gas Control in closed Primary Coolant Systems

There is no cover gas control in the primary coolant system.

9.7 Other Auxiliary Systems

There are no other auxiliary systems required for safe reactor operation.

9.8 References

There are no references.

10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 Summary Description

Facilities are provided to permit use of radiation from the reactor in experimental work without endangering personnel. Facilities provided with this reactor include four beam ports, a thermal column, and pneumatic and hydraulic irradiation transfer systems. All systems are designed to control radiation exposure to personnel using the facility as well as members of the public. Consideration is given to controlling the Ar-41 effluent from the experimental facilities in order to meet both the limits on releases to the environment and exposure of personnel within the laboratory.

10.2 Experimental Facilities

Thermal Column

The thermal column, **Figure 10-1** , is a graphite-filled, horizontal penetration through the biological shield which provides neutrons in the thermal energy range (about 0.025 eV) for irradiation experiments. The column, which is about 8 feet long, is filled with about 6 feet of graphite. A small experimental air chamber between the face of the graphite and the thermal column door has conduits for service connections (air, water, electricity) to the biological shield face. Detectors for the safety channels and the LogN channel are located within the thermal column. The location of the thermal column is indicated in **Figure 10-2** .

Personnel in the building are protected against gamma radiation from the column by a dense concrete door which closes the column at the biological shield. The door moves on tracks set into the concrete floor perpendicular to the shield face.

A ventilation system maintains a low pressure within the thermal column so that air flow is into the column when the door is open. The door is gasketed so that air flow is very small when the door is closed. When the door is opened, however, an air velocity of about 40 feet per minute into the column prevents the Ar-41 activity from diffusing into the Reactor Laboratory. Section 9.1 contains further information on the ventilation system for the thermal column and beam ports.

An annunciator is activated whenever the thermal column door is not fully closed. In addition, an area radiation monitor beside the thermal column door will give an alarm should the reactor be operated at a substantial power with the door open.

[redacted]

Figure 10-1 Thermal Column

Beam Ports

Four 6-inch beam ports penetrate the shield and provide fluxes of both fast and thermal neutrons for experimental use. **Figure 10-2** indicates the positions of the beam ports with respect to the grid box and shield while **Figure 10-3** shows the construction of the beam ports .

The ports are air-filled tubes, welded shut at the core ends and provided with water-tight covers on the outer ends. The portions of the ports within the pool are made of aluminum, while the portions within the shield are steel.

A shutter assembly, made of lead encased in aluminum, is opened for irradiations by a cable lifting device that extends to the pool curb. When closed, the shutter shields against gamma rays from the shut-down core, allowing experiments to be loaded and unloaded without excessive radiation exposure to personnel. A drain line is attached to the bottom of the shutter housing, while a vent line attaches to the top of the shutter housing. All beam port drains combine before exiting the concrete shield, where a stop valve is provided..

When beams are not being extracted shielding plugs are installed in the outer end of each port, filling almost all of the volume within. These plugs, made of dense concrete in aluminum casings, have spiral conduits for passage of instrument leads. These plugs completely stop the beam of radiation and minimize the production of Ar-41 in the beam ports.

[redacted]

Figure 10-2 Location of Beam Ports & Thermal Column

Sealed aluminum cans are installed in the in-pool portion of the beam ports unless the particular beam port experiment requires installation of a collimator or filter in that location. These cans contain the Ar-41 produced and further minimize the release of Ar-41 activity. Additional control of Ar-41 activity released is accomplished by bellows seals on the lifting cables and maintaining the valve on the common drain header closed.

Since extremely high radiation levels could exist should the reactor be operated at substantial power levels with the shielding plugs removed, a beam port monitoring system is provided. The system consists of radiation detectors mounted on the walls in line with each beam port and a read-out device at the console which gives an audible and visual alarm should a preset radiation level be exceeded. The system is set to alarm at a radiation level equivalent to a dose rate of about 60 mrem/hour at the beam port openings.

The thermal column-beam port ventilation system (Section 9.1) exhausts the beam ports through the vent pipes shown in **Figure 10-3** . Vent pipes are connect to the ventilation system through a check valve which prevents back-flow into the vent and an isolation valve which may be closed should the beam port fill with water. With the beam port open, a linear flow velocity of about 40 feet per minute is maintained into the port opening, preventing diffusion of the airborne activity into the laboratory. With the beam port closed Ar-41 is almost entirely contained within the beam port.

[redacted]

Figure 10-3 Beam Port

Pneumatic Tubes

Pneumatic tubes are used to irradiate samples for short times and when the sample must be processed immediately after irradiation, as in neutron activation analysis of short-lived radioisotopes. The currently installed pneumatic tube system conveys samples from a basement room to an irradiation position beside the core (**Figure 10-4**). The "rabbits" used in the system will convey samples up to 1-1/4 inches diameter and 5-1/2 inches long, although the gross weight of a sample is kept below 12 ounces. Although the polyethylene rabbits used in the system can withstand longer irradiations, this facility is usually used for shorter irradiations of small objects. The system operates as a closed loop with CO_2 cover gas controlling generation and discharge of Ar-41 activity.

The start and stop switches for the pneumatic tube blower, indications of "Blower On" and "Rabbit in Reactor", and an "Emergency Return" switch (which allows the reactor operator to eject rabbits if desired) are on the reactor control console. In addition, an annunciator is actuated when the blower is started. All indications and controls except for the blower start capability are duplicated at the pneumatic tube control center in a basement laboratory immediately west of the Reactor Laboratory. Automatic timing of irradiations is done at this control center, and rabbits are inserted, dispatched, and removed at this location. An area monitor indicates radiation level and gives a visible and audible alarm should the radiation level exceed a preset level at each station. The preset level is selected according to the computed activity of the sample being irradiated.

A "Receive Only" rabbit station is sometimes installed to deliver samples to other laboratory areas. Both this and the "Send/Receive" station are installed in fume hoods with high efficiency filters to control any releases from sample failures. Sample activity is limited to a level which, should the sample rupture upon discharge from the system, will result in keeping the concentration exhausted below 10 CFR Part 20 limits for unrestricted areas when averaged over a period of one month.

The reactivity effect from a sample is restricted to less than 0.2% ρ. . Tests run with water and cadmium samples indicate that sample reactivity effects will normally be less than 0.01% ρ. Static reactivity measurements will be run for samples of fissionable material or particularly strong absorbers such as some of the rare earths.

Since the pneumatic tube penetrates the shield below water level, a leak in the tubing could drain the pool. Gate valves located in the tubing just outside the shield automatically close when the blower is turned off. Should these valves fail to close, water will not be lost from the system from a break inside the pool unless another break in the tubing occurs outside the pool. Further, to drain more than 8 feet of water from the pool a siphon action would have to be set up. A siphon action is prevented by a solenoid valve controlled siphon breaker at the highest point in the system. The solenoid valves close when the blower motor is energized. When the blower motor is not energized, the solenoid valves open and check valves will then allow air to enter the system if a siphon action starts. Normally these check valves prevent loss of cover gas from the system.

The system is operated using a written check-list type procedure to assure that the built-in safe-guards remain effective.

Grid Box Irradiation Facilities

[redacted]

Figure 10-4 Pneumatic Tube System Layout

Irradiation of larger samples and most irradiations of more than twenty minutes duration are performed in irradiation facilities on the core periphery inside the grid box.

Radiation baskets (**Figure 10-5**) are 3-inch square aluminum containers which fit into the grid plate and may contain one or more samples. The bottom end boxes are similar to those of reflector elements, thus positioning the devices in fixed positions relative to the core. These devices may contain internal shelves or other positioning devices to position samples in fixed positions.

Two types of hydraulic transfer systems have been used for most irradiations within the core box. The smaller hydraulic irradiation device consists of an irradiation terminal which fits into a vacant grid position, a loading device and control valve assembly just beneath the pool surface, and connecting aluminum and polyethlyene tubing. This system is powered by bypass flow in the N-16 diffuser system. The sample carrier is a 20 ml liquid scintillation vial. This system is not presently in use, having been replaced with a system which can handle larger sample sizes.

Figure 10-6 shows one of the larger hydraulic tubes (called a whale tube at the facility). In this facility sample movement is powered by a separate pump located beneath the north side of the reactor bridge. The bottom ends of these tubes also fit into the grid plate, and the top of the tube is fastened to the bridge structure to provide further support and prevent inadvertent movement. The motor for the pump is electrically paralleled to the diffuser pump and thus runs when the diffuser pump is in operation. The pump takes its suction just below the pool surface and directs its flow to a jet pump near the bottom end of each tube, causing sufficient induced flow down the tube to move samples to the irradiation position and hold them in place. Samples which float return to the top of the tube where they are retained until removal by operating personnel. Non-floating samples can be removed with a retriever tool, or they may be installed with a retrieving string or wire attached. Flow direction and "sample in" indicators and controls are located at the pool top and control console.

Rupture of piping connected to the hydraulic irradiation facilities will not result in loss of pool water due to their locations within and immediately above the pool. Reactivity effects of samples are much smaller than those associated with installation and removal of conventional irradiation baskets with samples in them. The remarks regarding reactivity effects for samples in the pneumatic tube apply to these facilities.

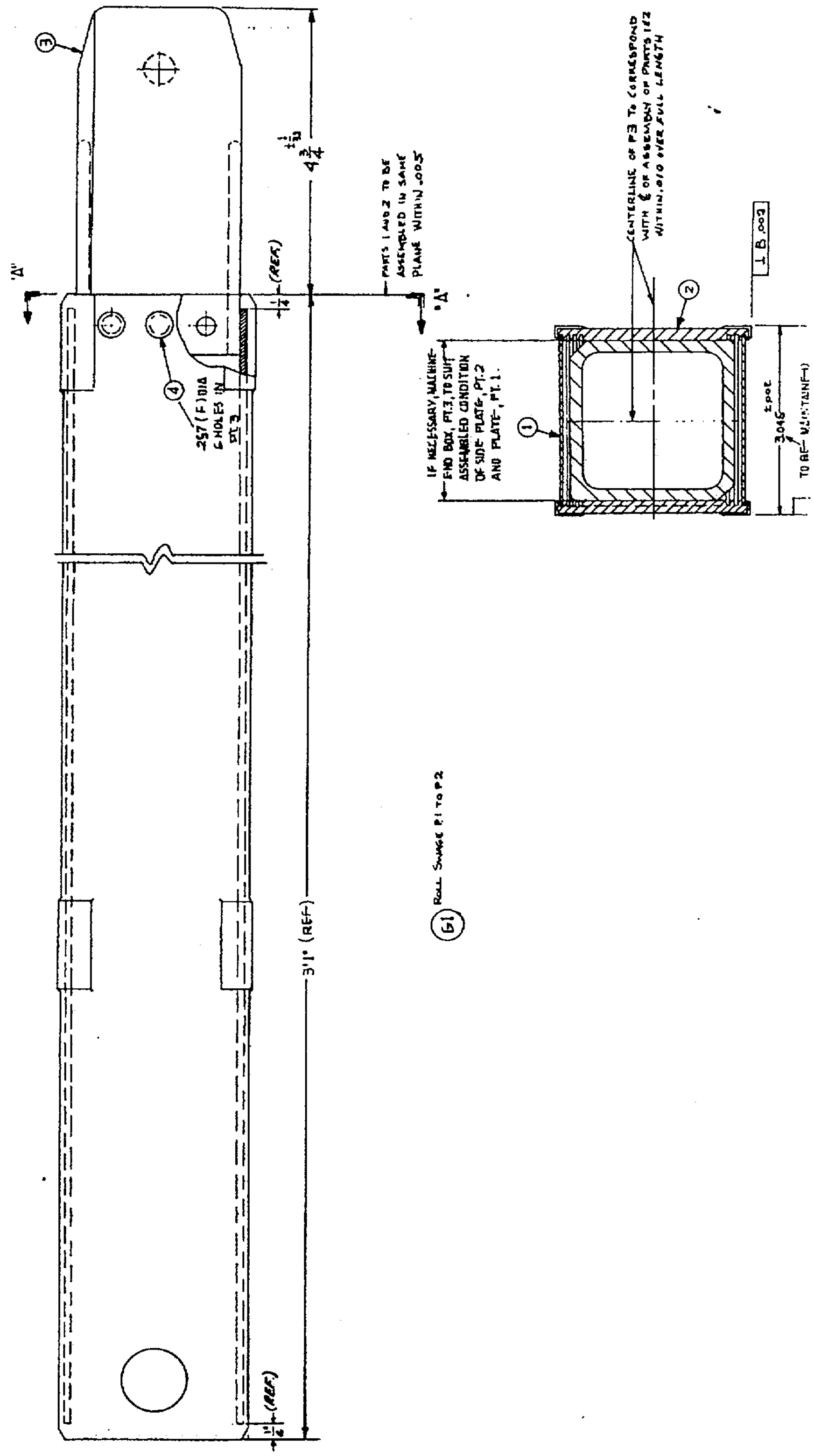


Figure 10-5 Radiation Basket

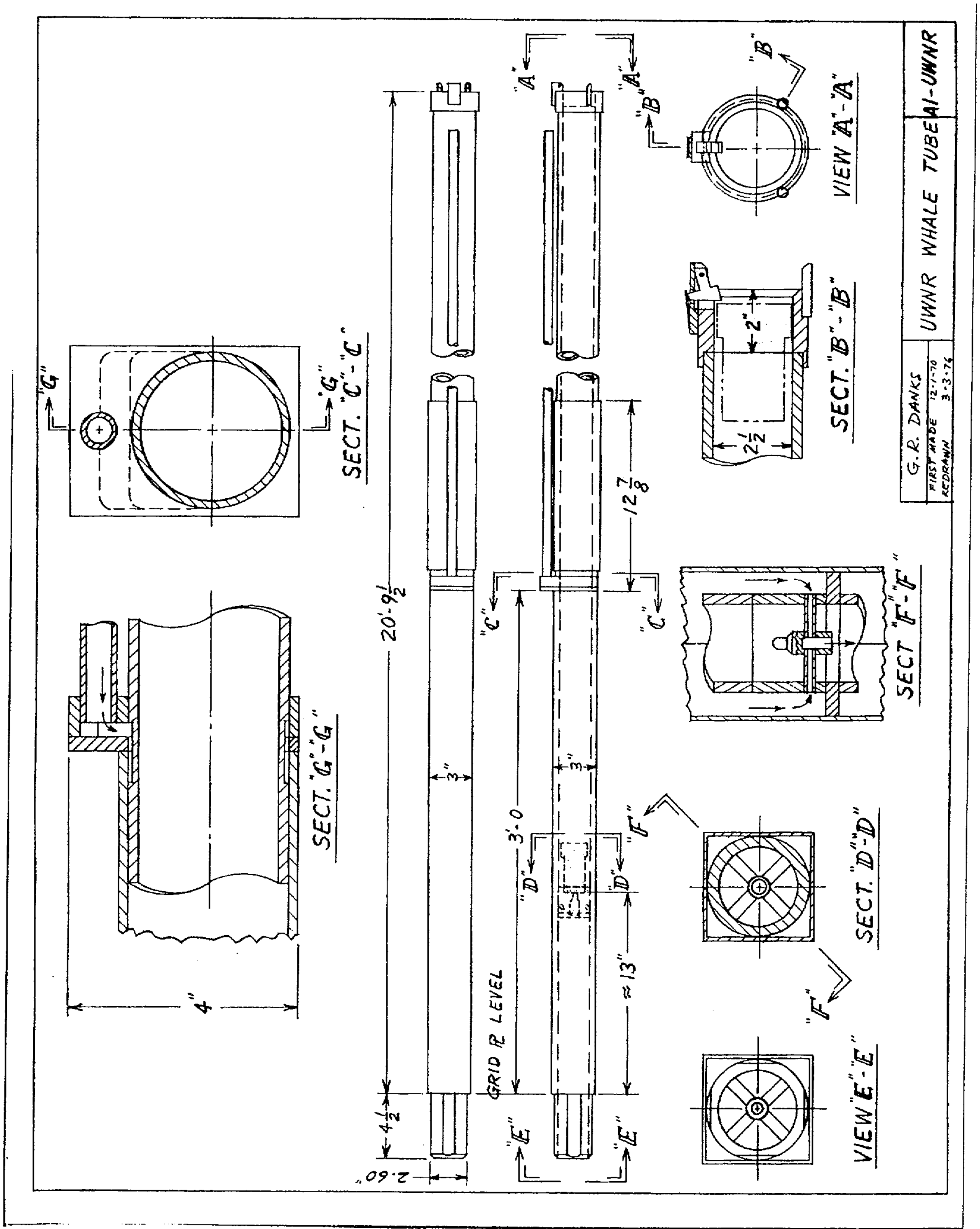


Figure 10-6 Whale Tube

10.3 Experiment Review

A body of operating procedures is in place to assure that experiments are conducted in a manner that will ensure the protection of the public. Experiment review meets the requirements of Regulatory Guide 2.2[1] and standard ANSI N401-1974/ANS-15.6 [2] as modified by Regulatory Guide 2.4 [3].

UWNR 002, Experiment Standing Operating Instructions, defines several classes of experiments that are routinely conducted and states the limitations and precautions to be observed as well as the methodology to be used. Control Element Calibrations, reactivity coefficient measurements, in-core neutron flux distribution measurements and sample irradiation/isotope production experiments are specifically defined in these instructions.

Since sample irradiation and isotope production are major experimental activities at UWNR, several addition standard operating procedures and limitations are in effect for these activities. Limits on potential airborne radioactivity produced in the event of sample breakage are included in the procedures to assure releases will not exceed those considered in the safety analysis report or those permitted under technical specifications. Irradiation of fueled experiments is controlled so that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 Curies and the maximum Sr-90 inventory is not greater than 5 millicuries. UWNR 002 allows SRO approval of irradiations meeting the requirements of UWNR 131 up to the limits for routine approval of gas, dust, highly volatile material, and fissionable material stated on UWNR 130, Request For Isotope Production. Approval for amounts in excess of the UWNR 130 limits (but still less than the UWNR 131 limits) require approval by a subcommittee of the Reactor Safety Committee. Approval for limits above those stated in UWNR 130 but below 10CFR Part 20 limits if released require approval by the Reactor Safety Committee. Other written procedures in the UWNR 130 series, including sample packaging requirements for the different irradiation facilities and approvals, are in effect for operation of all experimental facilities. Irradiation of material that is to be transferred to the campus broad radioisotope license requires both written and telephone approvals to assure that the recipient of the material is permitted possession and use of the material under that license.

For other experiments the senior reactor operator (SRO) responsible for operation when the experiment is performed classifies the experiment as routine (previously approved and performed), modified routine (determined not to be significantly different from previously performed experiment), special (not previously approved, but within constraints of technical specifications), or special requiring NRC approval (involving technical specification changes or unreviewed safety questions) .

Routine experiments may be approved by the SRO without further evaluation. For other experiments, the SRO evaluates the experiment in terms of its effect on reactor operation and the possibility and consequences of experiment failure, including consideration of chemical reactions, physical integrity, design life, proper cooling, reactivity effects, and interaction with core components. If the SRO classifies the experiment as modified routine he may approve operation of the experiment if he determines and specifies in a written record that the hazards associated with the modified routine experiment are not significantly greater or different from those involved with the corresponding routine experiment.

If the experiment is determined to be a special experiment, an experiment review questionnaire (UWNR 030) which includes description of the experiment including materials inserted, thermodynamics, reactivity effects, radioactivity, shielding, instrumentation related to control panel instruments, administration, and procedures, as well as a safety analysis must be completed and reviewed. Special experiments must be reviewed and approved by the Reactor Director and the Reactor Safety Committee. Favorable evaluation of an experiment shall conclude that failure of the experiment will not lead directly to damage of reactor fuel or interference with movement of a control element. Special requiring NRC approval experiments require local approvals as well as approval by NRC.

10.4 References

1. Regulatory Guide 2.2, Development of Technical Specifications for Experiments in Research Reactors, US Nuclear Regulatory Commission, November 1973

2. American National Standard ANSI N401-1974/ANS 15-6, Review of Experiments for Research Reactors, American Nuclear Society, November 19, 1974

3. Regulatory Guide 2.4, Review of Experiments for Research Reactors, U. S. Nuclear Regulatory Commission, May 1977

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

This chapter deals with the program and procedures for dealing with radioactive materials, radiation, and radioactive waste management. Since the Nuclear Reactor Laboratory is a part of the University of Wisconsin-Madison the campus radiation safety regulations govern activity under the reactor license. Information on these campus regulations was previously submitted as part of the application for license 48-09843-18 under docket 030-03465/030-17753 with Reference No. 0780-00052/070-00134. The Radiation Safety Regulations[1] are also available by internet access at (http://wiscinfo.doit.wisc.edu/safety/Radiation/regs.pdf). This information is incorporated by reference as part of this Safety Analysis Report.

The intent of the campus radiation safety program is to maintain radiation exposure to experimenters, students, and the general public as low as reasonably achievable as well as below regulatory limits while using radiation and radioactivity for teaching and research purposes. The implementation of the campus program within the activities of the Nuclear Reactor Laboratory has the same intent.

11.1 Radiation Protection

The radiation protection program at the reactor facility, while conforming to the campus program, has some specific aspects that apply only to the reactor facility. For instance, the design of the experimental facilities, the reactor pool, and the reactor shield includes protective measures and devices which limit radiation exposures and release of radioactive material to the environment. Information on these aspects of the radiation control program is included in the sections of this report that describe that equipment. General requirements, such as dosimeter use and records, certification of training, survey frequency, leak testing of sources, and overall ALARA program are discussed in the campus documentation. The remaining portions of this chapter will deal with the issues specific to the reactor.

11.1.1 Radiation Sources

11.1.1.1 Airborne Radiation Sources

Releases from abnormal reactor operations
The fuel retains the fission products, with releases to the environment only if the fuel clad is breached. This possibility is one of the accidents considered in Chapter 13 of this report in the analysis of the maximum hypothetical accident This event would result in maximum dose to personnel within the Reactor Laboratory and the maximum dose released to the environment. The maximum occupational dose calculated is 10 millirem whole body, 1 rad to the lung, and 18.9 rads to the thyroid, while the maximum dose to persons in unrestricted areas will be less than 0.153 rem whole body and 1.019 rad to the thyroid.

Releases from normal reactor operations

Argon-41 is the only activity released in significant quantities during normal operations. Calculations and measurements have been performed to determine production and release rates of the various activities that might be discharged due to normal operation. The calculation method used for Ar-41 release is shown in Appendix A, Section B,

Due to the operation of the beam port and thermal column ventilating system and the laboratory exhaust fan, the airborne activity levels in the laboratory are low. Some Ar-41 is produced in the dissolved air in the pool water as it passes through the reactor core and is released as the water is warmed while passing through the core. Some of the resulting activity eventually reaches the pool surface where it is released to the laboratory atmosphere. The concentration of Ar-41 in the air immediately above the pool surface during full-power operation reaches about one-third of the DAC for occupational exposure; as this air diffuses throughout the laboratory, the activity in the laboratory as a whole is at least a factor of 6 below the DAC. Therefore, further discussion will be concerned with the activity released to the atmosphere.

The maximum release rate of Ar-41 would occur with the reactor operating continuously at 1 MW and all four beam ports and the thermal column open. Such operation is not reasonable, but it does establish an upper limit to the activity that might be discharged. This maximum release rate is 13.3 μCi/sec, giving an Ar-41 concentration at the stack outlet of 2.4×10^{-5} μCi/ml. The EPA COMPLY program[2] indicates that the maximally exposed receptor would receive a dose of 0.6 mrem/year if all activity generated were discharged continuously.

The maximum concentration to which the public would be exposed (using Gifford's model as discussed in Appendix A) in this case would be about 3×10^{-8} μCi/ml.

As previously indicated, the above maximum value is for a situation not likely to occur during operation. The usual procedure is to have the experimental facilities in a no-flow condition if possible. Under no-flow conditions the beam port and thermal column ventilation system keeps the pressure in the experimental facilities lower than room pressure, and the activity produced in the facilities remains there and decays. The ALARA measures taken on the experimental facilities limits the typical release rate to about 10% of the production rate. Historically, in the year in which the maximum recorded Ar-41 release to the environment occurred (1993) the COMPLY program indicated a resulting dose of 3.4E-3 mrem/year.

One theoretically important consideration in the analysis of a reactor location is the effect on surrounding unrestricted areas of a spillage of radioactive materials. A release of radioactive material might occur, for example, if a highly-volatile liquid were irradiated in the reactor for the production of isotopes. If, while it was being transferred from the reactor to a cask, it were dropped and its container broken, the atmosphere within the Reactor Laboratory could become conceivably contaminated; further, this atmosphere could conceivably be released to the surroundings in such a fashion as to present an exposure in unrestricted areas. This problem may be of importance when the material being irradiated is highly volatile, or is a solid in finely-powdered form. For a typical solid or liquid spill, no special problems exist other than the direct radiation from the sample and cleaning up contamination. Since the level of radiation will be known for each sample, adequate equipment for handling the sample will be available when the material is discharged from the reactor. Equipment adequate for cleanup of spills will be kept available so that spills can be dealt with immediately, lessening the possibility of spreading contamination to adjacent areas. The remainder of this section will deal with gases, highly volatile liquids, or powdered samples which might cause air-borne activity in the event of a spill. This problem is handled at Wisconsin by a combination of administrative and operational procedures. For routine operations,, a concerted effort will be made to keep the concentration of contaminants in the atmosphere released from the Reactor Laboratory well below the limits as stated in Table 2, Appendix B, 10 CFR Part 20, "Standards for Protection Against Radiation". Among the procedures which will be followed to achieve this goal will be the double-encapsulating of materials to be exposed in the reactor in aluminum containers (for long exposure) or sealed polyethylene containers for exposures of less than 4×10^{17} thermal neutrons/sq. cm. with accompanying gamma ray and fast neutron fluxes. Only members of the reactor staff (or selected and trained individuals working under their supervision) will be permitted to handle these capsules within the Reactor Laboratory and the capsules will normally be opened only at appropriate locations outside the laboratory. Further, a log book will be maintained of all material exposures. However, because accidents can occur, and the amount of radioactivity which will be generated in any one sample of material will be limited. Specifically, this amount of radioactivity will be limited such that, should a container be broken and its contents disperse in the air within the Reactor Laboratory, the concentrations discharged through the stack when averaged over one year will be within the maximum concentrations of 10 CFR Part 20 Appendix B Table 2. Since the large fan has a capacity of 9,800 cfm through its filters, the weekly flow of dilution is 1.4×10^{14} ml. Normal approvals will be given for concentrations considerably smaller than these, however, and samples of such size as to approach these limits must have special approvals. These approvals will consider all other activity discharged, and will insure that the total stack discharge lies within permissible limits should the sample rupture.

Pneumatic tube stations are located outside the Reactor Laboratory and thus are not subject to the laboratory ventilation system. Each station is installed within a fume hood having a face velocity of > 100 lfpm to protect the system operator in case of sample breakage. The blowers for the fume hoods have free air capacities of 1200 ft^3/minute. The air discharged from each hood is passed through high efficiency filters and then exhausted to the atmosphere.

Although special packaging requirements are enforced to prevent breakage of pneumatic tube samples, such breakage may occur. Therefore, a more restrictive limit is placed on the activity which may be produced in the pneumatic tube. Volatile sample size is limited to that level which, diluted by the flow through the hood during a one-week period, (3.4×10^{11} ml) will result in average concentrations less than 10 CFR Part 20 Appendix B Table 2 limits. The blowers operate automatically whenever the pneumatic tube system is used. As with the other samples, the maximum activities generated must have special approvals, and only quantities considerably smaller are routinely approved.

Finally, no such sample breakage has ever occurred during previous operations involving thousands of sample irradiations, and such breakage is considered quite unlikely to occur.

11.1.1.2 Liquid Radioactive Sources

The only activity produced in liquid form in amounts sufficient to be a personnel exposure hazard is Nitrogen-16, which is produced in the reactor coolant as it passes through the reactor core when operating at power levels above 100kW. N-16 is controlled by use of the diffuser system (discussed in Section 5.6), which reduces the dose rate at the pool surface to 2 to 3 mrem/hour during full power operation. If the diffuser system fails during full power operation the dose rate at the pool surface is less than 100 mrem/hour.

Small quantities of liquid radioactive waste are generated by regeneration of the demineralizer and from liquids irradiated as part of sample irradiation. The radiation level from such liquids is extremely low and does not produce radiation exposure hazards. Disposal of this material is addressed in section 11.2.3. Releases are made to the sewer system within 10CFR Part 20 Appendix B Table 3 limits. Annual liquid releases have ranged from 0 to 10,000 gallons, with 3000 gallons being typical.

11.1.1.3 Solid Radioactive Sources

The major source of radiation and radioactivity is the fission product generation in the reactor fuel. Typical four-element fuel bundles will generate fields of 100 to more than 1000 R/hour in air at 3 feet if removed from the reactor pool. (The reactor is operated on a schedule that assures that less than 5 Kg of FLIP fuel material will drop below the 100 R/hour dose rate limit even if regular operation is suspended for a period of several weeks, in order to meet the requirements of the facility security plan). As long as the fuel is contained within the pool filled with water this source of radiation dose presents no personnel hazard. Loss of pool water is considered in Chapter 13, with the conclusion that the dose rates from pool water loss after long periods of operation could result in high radiation levels at the pool top (1200 R/hour one day after shutdown), but not so high that persons could not perform corrective actions to restore enough pool level to reduce the dose rate to tolerable levels (dose rate at the pool top level when shielded by the pool curb would be about 240 mrem/hour at the same decay time). Further, the pool is designed to preclude loss of pool water, and operation would not take place if there were any difficulty in maintaining pool level.

Other possibilities of significant radiation exposure from solid radioactive material are the standard 20% enriched TRIGA core, samples irradiated for isotope production, reactor components which have spent a long time near the core, and the reactor startup source. All of these are small sources compared to fuel fission product activity in the operating core. Dose rates from the old fuel are several orders of magnitude lower than those from the operating core. Sample handling equipment and procedures and use of aluminum for almost all structure near the core reduce exposure rates from samples and activated materials to levels which generate no significant personnel hazard during operation or maintenance of the reactor. For example, the shim-safety blades, reflector elements, and transient control rod have maximum radiation levels of a few R/hour at contact after a week of reactor shutdown. Activity produced during irradiations is calculated before the irradiations are performed and equipment and procedures are in place to deal with the activity after the irradiation is completed.

11.1.2 Radiation Protection Program

11.1.3 ALARA Program

The University Radiation Safety Regulations[1] are written to incorporate ALARA principles and practices. The Nuclear Reactor Laboratory policies and procedures reflect the commitment to ALARA principles. An annual ALARA review is conducted jointly by campus Safety Department health physics staff and the Reactor Laboratory staff with a report of the results of the review being submitted to the Reactor Director and the Reactor Safety Committee.

11.1.4 Radiation Monitoring and Surveying

The campus regulations[1] specify requirements on monitoring and surveying. Procedures for reactor operation reflect these requirements. Installed radiation and air activity monitors are described in Section 7.7 of this report. Area radiation surveys are conducted each month, including checks for contamination and particulate air activity. Sample irradiation procedures and forms require checks of radiation level each time a sample is removed from an irradiation facility. Experiment reviews and approvals required radiation surveys for new experiments and modifications of experiments.

11.1.5 Radiation Exposure Control and Dosimetry

The campus regulations[1] specify requirements on radiation control and dosimetry, and the Safety Department administers the dosimetry program. TLD dosimeters are used for operating personnel and students using the laboratory on a regular basis, and electronic dosimeters are used and records are maintained for tour groups and visitors.

Experiment approval requires that no High Radiation Areas are created external to the experiment shielding. Some experiments have shield cavities large enough for personnel entry, however, and higher radiation levels can exist inside the shield. Should an experiment design be approved with a High Radiation Area or Very High Radiation Level within the experiment shield, any entry points to the area will be provided with an alarm that sounds at the reactor console and at the entry point if a person attempts to open the entry point barrier.

Radiation doses received by visitors and tour groups are so low that they cannot be measured; tour groups are not allowed in any area with dose rate exceeding 0.5 mrem/hour. No student dosimeter has ever received a measurable exposure from reactor operation. Occupational exposures of operations and maintenance personnel have historically been very low, seldom exceeding 0.5 Rem TEDE in a year and usually below 100 mrem/year.

11.1.6 Contamination Control

The campus regulations[1] specify requirements on Contamination Control. As noted in section 11.1.4, monthly contamination surveys are conducted. Laboratory policy is that no detectable removal contamination is allowed; any contamination discovered is immediately decontaminated.

For routine cleaning, the laboratory has cleaning equipment which is dedicated to use in the laboratory area, and custodial personnel use this equipment in order to prevent the possibility of spreading unidentified contamination. Floor sweepings are surveyed for radioactivity before disposal.

11.1.7 Environmental Monitoring

Environmental TLD monitors are used and evaluated on a quarterly basis. The dosimeters are distributed around the engineering campus so that they surround the Reactor Laboratory. At the present time more than 25 points are monitored. Effluent concentrations are measured at the point of release.

11.2 Radioactive Waste Management

The campus regulations[1] specify requirements for dealing with radioactive waste on campus. The Reactor Laboratory follows the campus regulations.

11.2.1 Radioactive Waste Management Program

This is a campus program.

11.2.2 Radioactive Waste Control

This is a campus-wide program. Liquid waste from beam port drains, pool overflow, laboratory floor drains, and demineralizer regeneration is stored in a 2000-gallon holdup tank, and other liquid radioactive wastes generated in the laboratory are collected in local containers. Filled local containers may be dumped into the holdup tank.

11.2.3 Release of Radioactive Waste

Solid radioactive waste is transferred to the Safety Department for disposal.

Liquid wastes can be transferred to the Safety Department, but most are placed into the holdup tank. The Reactor Laboratory occasionally discharges liquid waste from the holdup tank to the sewer system. All discharges are filtered so that no particulate activity above 0.4 micron size is discharged. Sampling, analysis, and release of the holdup tank contents are governed by a written procedure that assures releases are within 10CFR Part 20 Appendix B Table 3 limits and that the pH is within local limits for discharge to the sewer.

11.3 Bibliography

1. Radiation Safety Regulations, University of Wisconsin-Madison, Revision 2, January 1997
This document is available in PDF format at URL
http://wiscinfo.doit.wisc.edu/safety/Radiation/regs.pdf

2. U. S. Environmental Protection Agency, COMPLY Program Rev. 2, October 1989

12 CONDUCT OF OPERATIONS

12.1 Organization

12.1.1 Structure

Figure 12-1 is a chart indicating operating organization. Position responsibilities and authority are summarized in the following sections.

12.1.2 Responsibility

University Radiation Safety Committee

To exercise its prerogatives (as a campus-wide committee appointed by the Chancellor of the University of Wisconsin-Madison Campus to review all activities on campus which involve the use of radiation) in reviewing all activities related to the Reactor Laboratory.

To advise the Reactor Director of all studies and/or actions taken with regard to the Reactor Laboratory.

To overrule the Reactor Director where necessary in carrying out its function.

To supply health physics services to the University.

University Radiation Safety/Health Physics (Part of University Safety Department)

Note: The group performing radiation safety related activities within the Safety Department are, on the UW-Madison campus, called both "Health Physics" and "Radiation Safety" and is listed in the campus directory under both names. When either term is used in this document it refers to the same organization.

To assist the University Radiation Safety Committee by conducting inspections, making recommendations, maintaining records, and establishing procedures for emergency operations, waste disposal, etc.

To provide similar inspections and service functions to the Reactor Safety Committee.

Reactor Director

To approve all policy decisions and all basic regulations, basic instructions, and basic procedures governing the use and operation of the reactor and related facilities.

To designate the Reactor Supervisor and other Senior operators.

To take cognizance of all recommendations and actions by the University Radiation Safety Committee (which relate to the reactor facility) and the Reactor Safety Committee.

To appoint qualified members to the Reactor Safety Committee as necessary.

Reactor Safety Committee

Review and approval of new experiments utilizing the reactor facilities;

Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;

Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specifications;

Review of abnormal performance of plant equipment and operating anomalies having safety significance; and

Review of unusual or reportable occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50.

Review of audit reports.

Review of violations of technical specifications, license, or procedures and orders having safety significance.

Reactor Supervisor

To initiate and enforce policies, administrative rules, regulations, and operating procedures relating to the Reactor Laboratory, subject to the appropriate approvals of the Reactor Safety Committee, the University Radiation Safety Committee, and the Reactor Director.

To ensure that all activities within the Reactor Laboratory are in accordance with prior approvals from the appropriate committees or from the Reactor Director.

The Reactor Supervisor shall have authority to authorize experiments and/or procedures which have been approved by the Reactor Safety Committee. He will prepare specific detailed procedures based on the general procedures approved by the Committee.

To see that all proper records are kept.

To maintain a Senior Operator's License.

To appoint Reactor Operators.

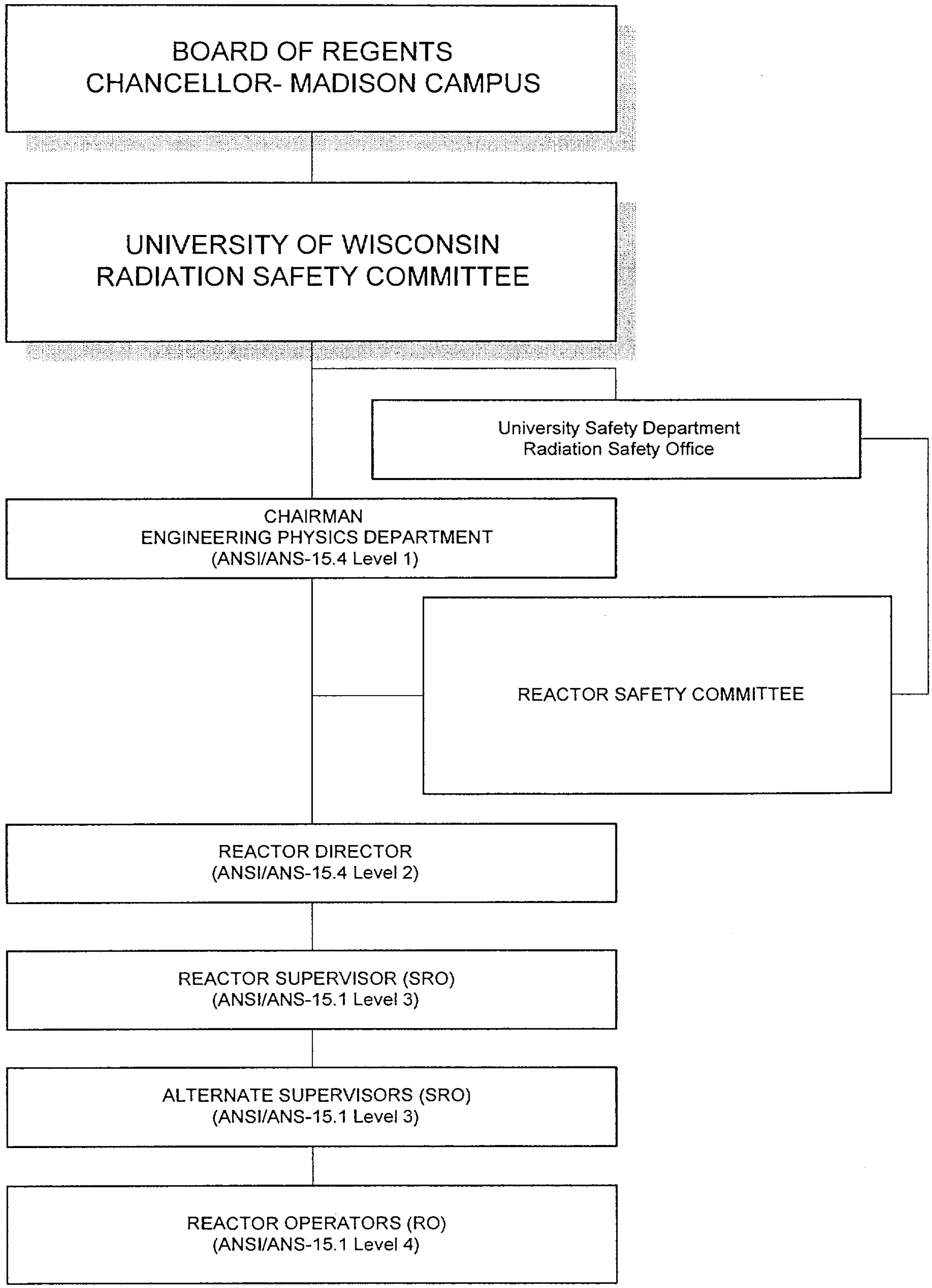


Figure 12-1 Organization Chart

The Reactor Supervisor or another Senior operator shall be in charge of the Reactor Laboratory at all times (although not necessarily physically present). The individual in charge, if physically present, shall be responsible for prompt execution of emergency procedures. The Reactor Supervisor or another Senior operator will be present at the facility during fuel manipulation, reactor start-up and approach to power, and recovery from unscheduled scrams and shut-downs. He shall be available on call at other times during reactor operation.

To be responsible for safety in the Reactor Laboratory, including responsibility for health physics matters.

To advise and prepare information for the committees concerned with the Reactor Laboratory, and to present such information to the committees

Senior Operators (alternate Supervisors)

To accept responsibility for safe and efficient operation of the Reactor Laboratory when designated by the Reactor Supervisor.

To maintain a Senior Operator's License.

Reactor operators

To hold a Reactor operator's License.

To conform to all rules and regulations for operation of the reactor.

A reactor operator will be present at the control console at all times when the reactor is in operation.

To monitor laboratory activities from a health-physics standpoint.

12.1.3 Staffing

The minimum staffing when the reactor is not secured shall be:

1. A licensed reactor operator in the control room (if senior operator licensed, may also be the person required in (c))
b. A second designated person present at the facility complex able to carry out prescribed written instructions.
c. A designated senior reactor operator shall be readily available at the facility or on call.

A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator.

A licensed senior reactor operator shall be present at the facility for:

a. Initial startup and approach to power.
b. All fuel handling or control-element manual manipulations.
c. Relocation of any in-core experiment with a reactivity worth greater than 0.7% ΔK/K.
d. Recovery from unplanned or unscheduled shutdown or significant power reduction.

12.1.4 Selection and Training of Personnel

The selection, training of operations personnel meets or exceeds the requirements of ANSI/ANS-15.4-1988 Sections 4-6[1]. The operator training program includes sufficient radiation safety training to meet the requirements of 10 CFR Part 19 and the campus Radiation Safety Regulations. The operator training program is offered as an elective four credit-hour course with a formal training manual, homework, and practical exercises (OJT) included. It includes the equivalent of 0.5 weeks of reactor fundamentals, 1.25 weeks of systems coverage, 0.5 weeks of systems observation, and 0.8 weeks of control room operations administered over the period of one semester. The course is completely described on the internet at <http://www.engr.wisc.edu/ep/neep/courses/neep234.html>.

The operator proficiency maintenance program (re-qualification program) fully meets the requirements of 10CFR Part 55 and is formalized as a facility procedure, UWNR 004[2], which received NRC approval upon initial implementation and is reviewed annually by the facility operating organization along with other facility procedures. The program includes written, oral, and performance testing as well as emergency procedure drills and classes on changes in experiments, facility equipment, and procedures.

12.1.5 Radiation Safety

Radiation safety aspects of facility operation are routinely performed by members of the reactor operating staff, including routine radiation and contamination surveys and sampling of water and air samples. The campus radiation safety organization (see chapter 11) established to oversee all activities involving ionizing radiation on campus is part of the university Safety Department, and thus is an independent organization which reports to the central campus administration. The radiation safety organization has the authority to interdict or terminate radiation safety related activities conducted under the reactor license.

12.2 Review and Audit Activities

12.2.1 Composition and Qualifications

The Reactor Safety Committee is appointed by the Reactor Director. Minimum committee size is six members, one of whom is a health physicist from the University of Wisconsin Safety Department Radiation Safety/Health Physics office. Other members are faculty and staff of the university selected based on expertise to assure that the following disciplines are represented:

1. Reactor Physics - Nuclear Engineering;
2. Mechanical Engineering - Heat transfer and fluid mechanics;
3. Metallurgy/Materials
4. Instruments and Control Systems;
5. Chemistry and Radio-chemistry;
6. Radiation Safety.

Reactor operations staff is not precluded from membership on the committee as long as such members do not reach a majority of a quorum for voting. The health physics personnel who perform the monthly audits and inspections are invited to the meetings, but are not necessarily members of the committee.

12.2.2 Charter and Rules

The Reactor Safety Committee operates with a written charter which specifies the manner in which business is conducted. The charter includes rules on meeting frequency (at least annually), voting rules, agenda, quorums, use of subcommittees, minutes, and methods and content of submissions to the committee. Provisions for use of telephone polls or subcommittees for approval of items not requiring a formal meeting are also a part of the charter.

12.2.3 Review Function

The reactor director or his designee reviews all written operating procedures at least annually. Results of this review, along with suggested procedure revisions, are submitted to the Reactor Safety Committee for approval, or re-affirmation if no changes are deemed necessary..

The review responsibilities of the Reactor Safety Committee shall include, but are not limited to, the following:

1. Review and approval of new experiments utilizing the reactor facilities;

2. Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;

3. Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specifications;

4. Review of abnormal performance of plant equipment and operating anomalies having safety significance; and

5. Review of unusual or reportable occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50.

6. Review of audit reports.

7. Review of violations of technical specifications, license, or procedures and orders having safety significance.

12.2.4 Audit Function

A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office represents the University Radiation Safety Committee and conducts an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Health Physics Office are also used by the Reactor Safety Committee, with the scope of the audit extended to cover license, technical specification, and procedure adherence.

12.3 Procedures

Written operating procedures are used to assure the safety of operation of the reactor. Procedure use does not preclude the use of independent judgement and action should the situation require such. Operating procedures are in effect for the following items:

(1) Testing and calibration of reactor operating instrumentation and controls, control rod drives, area radiation monitors, and air particulate monitors;
(2) Reactor startup, operation, and shutdown;
(3) Emergency and abnormal conditions, including provisions for evacuation, reentry, recovery, and medical support;
(4) Fuel element and experiment loading or unloading;
(5) Control rod removal or replacement;
(6). Routine maintenance of the control rod drives and reactor safety and interlock systems or other routine maintenance that could have an effect on reactor safety;
(7) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes; and
(8) Civil disturbances on or near the facility site.

Substantive changes to the above procedures may be made only with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made by the Senior Operator in control or his designated alternate. All such temporary changes are documented and subsequently reviewed by the Reactor Safety Committee.

12.4 Required Actions

In the event a safety limit is exceeded:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.
(2) An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.7 of the technical specifications, and
(3) A report shall be prepared which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee (RSC) for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

A reportable occurrence is defined as any of the following that occur during reactor operation:

(1) Operation with any safety system setting less conservative than specified in the technical specifications;
(2) Operation in violation of a Limiting Condition for Operation listed in the Technical Specifications;
(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function;
d. Any unanticipated or uncontrolled change in reactivity greater than 0.7% $\Delta K/K$, excluding reactor trips from a known cause;
(5) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits; and
(6) Abnormal and significant degradation in reactor fuel or cladding which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

In the event of an reportable occurrence as defined in the Technical Specifications, the following actions shall be taken:

(1) The reactor shall be shut down.
(2) The Reactor Director or his designated alternate shall be notified and corrective action taken with respect to the operations involved,
(3) The Director or his designated alternate shall notify the Chairman of the Reactor Safety Committee,

(4) A report shall be made to the Reactor Safety Committee which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and

(5) A report shall be made to the NRC.

12.5 Reports

Reports will be made to NRC in accordance with the following:

(1) An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted (in writing to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555) within six months following the end of each calendar year, providing the following information:

a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

c. The number of emergency shutdowns and inadvertent scrams, including reasons therefor;

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required:

e. A brief description, including a summary of the safety evaluations of changes in the facility or in the procedures and of tests and experiments carried pursuant to Section 50.59 of 10 CFR Part 50.

f. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

Liquid effluents (summarized on a monthly basis)

(1) Liquid radioactivity discharged during the reporting period .tabulated as follows:

(a) Total estimated radioactivity released (in curies).

(b) The isotopic composition if greater than 1 x 10^{-7} microcuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

(d) Average concentration at point of release (in microcuries/cc) during the reporting period and the fraction of the applicable limit in 10CFR20.

(2) Total volume (in gallons) of effluent water (including diluent) during periods of release.

Gaseous Waste (summarized on a monthly basis)

(1) Radioactivity discharged during the reporting period (in curies) for:

(a) Gases.

(b) Particulates with half lives greater than eight days.

The estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis and the fraction of the applicable 10CFR 20 limits for these values..

Solid Waste

(1) The total amount of solid waste packaged (in cubic feet).

(2) The total activity involved (in curies).

(3) The dates of shipment and disposition (if shipped off site).

g. A summary of radiation exposures received by facility personnel and visitors, including dates and time of significant exposures and a summary of the results of radiation and contamination surveys performed within the facility; and

h. A description of any environmental surveys performed outside the facility.

(2) A report within 60 days after completion of startup testing of the reactor (in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555 with a copy to the NRC compliance inspector assigned to the facility) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level describing the

measured values of the operating conditions or characteristics of the reactor under the new conditions including:

a. An evaluation of facility performance to date in comparison with design predictions and specifications, and

b. A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

(3) A report of any of the following not later than the following day by telephone or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report describing the circumstances of the event and sent within 14 days to U.S. Nuclear Regulatory commission, Attn: Document Control Desk, Washington, D.C. 20555, with a copy to the NRC inspector assigned to the facility:

(a) Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;
(b) Any violation of a safety limit; and
(c) Any reportable occurrences.

(4) A written report within 30 days in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555, of:

a. Permanent changes in facility organization at Reactor Director or Department Chair level.
b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report.

12.6 Records

The following records are retained for a period of at least five years or for the life of the component involved if less than five years.

(1) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),
(2) Principal maintenance activities,
(3) Reportable occurrences,
(4) Surveillance activities required by the Technical Specifications,
(5) Reactor facility radiation and contamination surveys where required by applicable regulations,
(6) Experiments performed with the reactor,

(7) Fuel inventories, receipts, and shipments,
(8) Approval of changes in operating procedures,
(9) Records of meeting and audit reports of the review and audit group.

Operator qualification and re-qualification records will be retained for at least one cycle of the re-qualification program.

The following records will be retained for the lifetime of the reactor facility. (Note: Retention of annual reports which contain the information in items (1) and (2) are considered as suitable records for those items.
(1) Gaseous and liquid radioactive effluents released to the environs,
(2) Offsite environmental monitoring surveys required by technical specifications,
(3) Radiation exposures for all personnel monitored,
(4) Updated, corrected, and as-built drawings of the facility.

12.7 Emergency Planning

The Emergency Plan for the University of Wisconsin Nuclear Reactor was prepared to meet the requirements of ANSI/ANS 15.16-1978 [3] as amplified by Nureg-0849 [4]. This plan was submitted to NRC for review in May 21,1980, with subsequent revisions in October 25, 1982, and May 17, 1984. By letter dated July 25, 1984 NRC indicated that the plan met the requirements referenced above. The plan was again modified and submitted to NRC on May 16, 1990, with supporting information submitted on August 12, 1990. NRC notification that the revision was acceptable was received in a letter dated April 26, 1991. The plan was again modified (Revision 4) to reflect the changes in section number and nomenclature of 10CFR Part 20 and submitted to NRC on February 17, 1994 and April 22, 1994. This version is the current version in use at the facility.

The Emergency Plan indicates response capabilities for emergency conditions arising in connection with operation of the reactor. It includes identification of various precursor conditions (loss of electrical power, fires ,reactor pool leaks, riots, etc) and the consequences for various independent or simultaneous precursor. The plan includes the event classification system. The dose to which people could be exposed under various conditions is indicated, as are the actions that can be taken to minimize the consequences of the emergency. Detailed emergency implementing procedures have been developed and are referenced in the plan.

Primary responsibility for emergency planning and response is given to the Reactor Director. Delegation of responsibility and authority in the absence of the Reactor Director is specified. The Emergency Plan and implementing procedures are reviewed annually to assure that any required changes are incorporated into the plan.

12.8 Security Planning

The facility physical security plan, UWNR 003, was initially submitted to NRC on October 18, 1988. The security plan was revised and submitted again on June 17,1991 and was found to meet the applicable requirements, including the format of RG 5.59 [5]. The plan will require revision as a result of this Safety Analysis Report, since some figures from the previous Safety Analysis Report are included by reference. These changes will be made during the usual annual reviews of the plan once this SAR becomes the document referenced in the reactor operating license.

The security plan indicates the measures provided to protect special nuclear material, including details of the protective equipment and police agencies, and is thus withheld from public disclosure. The Reactor Director is responsible for administering the security program and assuring that it is updated as required.

12.9 Quality Assurance

Since no construction permit is sought in the application for renewal of the license for the University of Wisconsin Nuclear Reactor, no description of a quality assurance program for the design and construction of the structures, systems, and components of the facility is included. This section describes the Quality Assurance program that is in place to govern safe operation and modification of the facility. This program meets the applicable requirements of Regulatory Guide 2.5 [6] and ANSI/ANS-15.8-1995[7]

The Reactor Director has responsibility for the quality assurance activities, and thus has the authority to identify problems, to initiate corrective actions, and to insure that corrective actions are performed. He exercises QA oversight by assuring that operating and maintenance procedures include specific requirements to assure that modification, maintenance, and calibration of safety-related systems are performed in a manner that maintains the quality and reliability of equipment. Further, experiment reviews use written requirements to assure that installation and operation of the experiment does not degrade the performance of safety equipment. Modification of safety-related equipment is planned and reviewed using formal written checklist-type procedures that assure that equipment continues to meet the original specifications. Most of the reactor equipment in use in the facility does not have formal QA documentation because it was built before the QA requirements were in effect. This equipment is covered under the provisions of section 4 of ANSI/ANS-15.8. Several instruments are replacements for the vacuum-tube electronics originally provided by the reactor manufacturer, General Electric Company. This replacement equipment was designed, built, and tested to meet the original specifications stated in the equipment manuals provided with the General Electric equipment. After-maintenance checks, alignment, and calibration of the replacement equipment still assures the equipment meets the original equipment specifications.

Procedures include schedules of equipment maintenance and calibration, and provide records that such functions have been completed. Calibration procedures include requirements that critical equipment and instruments used in the calibrations are themselves currently calibrated (when appropriate).

12.10 Operator Training and Requalification

Operator training and Requalification programs are briefly described in section 12.1.4. The Requalification plan at UWNR is published as a standard procedure (UWNR 004, "University of Wisconsin Nuclear Reactor Operator Proficiency Maintenance Program") which was submitted to NRC on October 24, 1973 and revised on February 7, 1974. By letter dated March 29, 1974 we were notified by NRC that the program meets the requirements of Section 50.54(i-1) of 10CFR Part 50 and Appendix A of 10 CFR Part 55. Since the program is a numbered procedure it is reviewed by management on an annual basis.

12.11 Startup Plan

The facility has been in routine operation for many years, so a startup plan is not included in this Safety Analysis Report for license renewal.

12.12 Environmental Reports

On January 23, 1974 the AEC staff concluded in a memorandum addressed to D. Skovholt and signed by D. R. Miller, "that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 Mwt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities."

Since this Safety Analysis Report is written in support of extending the license expiration date for an additional 20 years, no changes in land and water use are contemplated. Emissions of radioactive materials or other effluents will not change as a result of extending the license term.

12.13 References

1. Standard ANSI/ANS-15.4-1988, Selection and Training of Personnel For Research Reactors, American Nuclear Society, June 9, 1988 ANSI Approval

2. UWNR 004, Operator Proficiency Maintenance Program

3. ANSI/ANS 15.16-1978, "Emergency; Planning for Research Reactors", ANS, LaGrange Park, Illinois, 1978

4. NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors", USNRC, October 1983

5. Regulatory Guide 5.59, Revision 1, "Standard Format and Content for A Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance, US Nuclear Regulatory Commission, February 1983

6. Regulatory Guide 2.5, Revision 0-R, :Quality Assurance Program Requirements for Research Reactors, October 1977

7. ANSI-15.8-1995, "quality assurance program requirements for research reactors", ANS, La Grange Park, Illinois, 1995

13 ACCIDENT ANALYSIS

NUREG -1537[1] divides accident analysis into initiating and consequences sections. In this report the two sections are combined; that is, the analyses of the consequences of accidents are grouped with the accident-initiating events and scenarios in order to reduce duplication which would otherwise occur. In addition, a better appreciation of the likelihood and consequences of accidents is afforded. The sections are numbered to correspond to the numbering system of NUREG-1537.

13.1 Accident-Initiating Events and Scenarios
13.2 Accident analysis and Determination of Consequences

NUREG/CR-2387 [2] reports an independent study of accidents in TRIGA-type reactors and concludes "The only potential for offsite exposure appears to be from a fuel-handling accident that, based on highly conservative assumptions, would result in dose equivalents of ≤1 mrem to the total body from noble gases and ≤1.2 rem to the thyroid from radioiodines." Notwithstanding that conclusion, the following sections reiterate the analysis done for a license amendment allowing the University of Wisconsin Nuclear Reactor to operate with FLIP or mixed Standard-FLIP cores, using values specific to the UWNR reactor location and characteristics.

13.1/2.1 Maximum Hypothetical Accident

The maximum hypothetical accident for UWNR is postulated as damage to a fuel element resulting in failure of the fuel cladding. It is postulated that this damage occurs after a very long time of operation at 125% of full power (the power level limiting safety system setting) and that it occurs in the fuel element with the highest power density possible in permitted UWNR mixed core fuel loadings. In a compact 9-Bundle FLIP core the highest power density is 18.1 kW at 1000 kW; the corresponding number for the 15-bundle FLIP core is 17.2 kW, while the value for the currently-used all FLIP core is only 15.2 kW. Continuous operation at the power level scram setpoint is highly unlikely, but for this <u>hypothetical</u> computation the power level in the maximally exposed fuel element of the 9-bundle core is assumed to be 23 kW (1.25 times 18.1 kW rounded to the next highest number).

The likelihood of a major fuel element cladding failure is considered small. The elements must meet rigid quality control standards; pool water quality is carefully controlled; and much care is taken in handling fuel. Such clad failures are, however, possible and the remainder of this section is concerned with the consequences of such a failure.

The release of radioactivity by corrosion and leaching by the pool water has been measured at Gulf General Atomic. About 100 micrograms of U-ZrH per square centimeter of exposed fuel surface per day is released for shutdown conditions. This release is easily controlled by isolating the leaking element in a container provided for that purpose. The gaseous and highly volatile fission products that have collected in the space between fuel and cladding would be the activity contributing to personnel hazards.

Fission Product Inventory in Fuel Element

The quantity of these volatile and gaseous fission products was determined by the use of Perkins and King[3] data. Column B of Table 13.1 indicates the fission product activities in the fuel element exposed to the maximum power density.

Fission Product Release Fraction

The release of fission products from U-ZRH fuel elements has been extensively studied by Gulf General Atomic and others. The results of this work indicate that the release of fission product gases into the gap between fuel and cladding is given by the following relationship:

$$FR=1.5E\text{-}5 + 3.6E3\exp(-1.34E4/T)$$

where T is the maximum fuel temperature (°K) in the element during normal operation.

The maximum fuel temperature in a fuel element operated in the steady-state mode at 23 KW will be less than 440 °C. Calculations of release fraction however, are based on 600 °C in order to assure a conservative result.

The release fraction corresponding to 600 °C is 7.9 E-4. Applying this fraction to the total inventory of the fuel element as given in column B of Table 13.1 gives the released activity as shown in column C of the table.

For the purpose of further calculations, it is assumed that all gaseous fission products are released to the room air whether the pool is filled with water or not. For soluble volatiles, calculations assume all activity is absorbed in pool water for calculations of pool water activity (column D). For calculations of air activity, the assumption is made that 10% of the volatiles escape with the pool filled with water (columns E and F) and 100% escape with the pool empty.

Activity in Pool Water

If 100% of the soluble fission products are absorbed in the pool water, the resulting activity level will be 0.075 μCi/ml. Within 24 hours the level would be reduced by radioactive decay to about 0.012 μCi/ml. After 24 hours the activity decay rate would be chiefly determined by the I131 half life (8.05 days). The demineralizer will remove most of this activity, giving a radiation dose rate of about 88 mrem/hr at one meter after the activity is deposited in the resins. The resins can be dumped to an underground storage pit or the underground liquid waste holdup tank where the activity will decay without hazard to personnel.

Fission Product Release to Air within the Reactor Laboratory

Calculations were performed to determine (a) the dose rate due to gamma emitters uniformly dispersed throughout the volume of the reactor lab, (b) the dose to the lungs from beta emitters for an individual remaining in the laboratory for ten minutes; and (c) the dose to the thyroid of an individual remaining in the room ten minutes. For the latter calculations, it is assumed that 10% of the iodine radioisotopes escape from the pool water. In addition to these calculations, a computation of the number of DAC hours indicates that a person present in the room for 10 minutes after the release would receive less than the annual limit on intake for occupational exposure..

(a) Whole body exposure due to gamma emitters

The amount of insoluble volatiles released to the room would be 5.89 Ci. If this activity is distributed uniformly in the laboratory volume, the resulting concentration would be 2.95E-3 $\mu Ci/cm^3$. The resulting maximum dose rate is calculated to be 60 mrem/hr. An individual remaining in the laboratory for 10 minutes after a release would receive a whole body dose of 10 mrem.

(b) Dose to the lungs

The lung is the critical organ when considering the effects of inhaling the insoluble volatiles from a ruptured fuel element. The beta emitting nuclides become more important than those emitting gamma rays since all the decay energy is absorbed in lung tissue.

The calculation outlined in the appendix indicates the lung exposure for an individual remaining in the laboratory for 10 minutes after a clad rupture to be 1.0 rad.

(c) Thyroid dose

The thyroid dose to a person in the reactor room was calculated assuming that he remained in the laboratory for 10 minutes after the fission product release. If the pool water is not lost and 10% of the halogens released escape into the atmosphere, the concentrations of the various iodine isotopes would be as presented in Table 1. In a ten minute period the lungs would be exposed to the iodine isotope activities shown in Table 2. As before, it was assumed that the "standard man" breathes 1.25 M^3/active-hour and his lungs hold 3 liters of air. A conservative calculation results in a dose to the thyroid of 18.9 rads.

Although all doses were calculated based on an individual remaining in the laboratory for ten minutes, emergency procedures require immediate evacuation after scramming the reactor, and re-entry to the area is made using self-contained breathing apparatus. Actual doses in the event of the accident would be a factor of 10 less than calculated, considering reasonable evacuation times.

TABLE 13.1 FISSION PRODUCT RELEASE FROM CLAD RUPTURE

A ISOTOPE	B SATURATED INVENTORY (Ci)	C RELEASED INVENTORY (Ci)	D AMOUNT IN WATER (Ci)	E AMOUNT IN AIR (Ci)	F LABORATORY CONCENTRATION (μCi/ml)	G Part 20 TABLE 1 DAC (μCi/ml)	H CONCENTRATION DISCHARGED (μCi/ml)	I Part 20 TABLE 2 MAX.EFFLUENT CONCENTRATION (μCi/ml)	J RATIO COL. H/I
Br 82	30	0.024	0.024	0.002	1.2E-06	2E-06	1.30E-10	6E-09	0.0217
83	105	0.083	0.083	0.008	4.2E-06	3E-05	4.40E-10	9E-08	0.0049
84	194	0.153	0.154	0.015	7.7E-06	2E-05	8.20E-10	8E-08	0.0103
'85	253	0.200	0.200	0.020	1.0E-05	1E-07	1.07E-09	1E-09	1.0700
'87	600	0.473	0.475	0.047	2.4E-05	1E-07	2.53E-09	1E-09	2.5300
TOTAL Br		0.933		0.093					
I '130m	200	0.158	0.158	0.016	7.9E-06	1E-07	8.40E-10	1E-09	0.8400
131	563	0.446	0.446	0.045	2.2E-05	2E-08	2.38E-09	2E-10	11.9000
132	855	0.677	0.677	0.068	3.4E-05	3E-06	3.61E-09	2E-08	0.1805
133	1282	1.015	1.015	0.102	5.1E-05	1E-07	5.41E-09	1E-09	5.4100
134	1554	1.230	1.230	0.123	6.2E-05	2E-05	6.56E-09	6E-08	0.1093
135	1185	0.938	0.938	0.094	4.7E-05	7E-07	5.00E-09	6E-09	0.8333
'136	602	0.477	0.477	0.048	2.4E-05	1E-07	2.54E-09	1E-09	2.5400
TOTAL I		4.941		0.494					
Kr 83m	105	0.084		0.084	4.2E-05	1E-02	4.40E-10	5E-05	0.0000
85m	253	0.200		0.200	1.0E-04	2E-05	1.07E-08	1E-07	0.1070
85	51	0.040		0.040	2.0E-05	1E-04	2.13E-09	7E-07	0.0030
87	486	0.386		0.385	1.9E-04	5E-06	2.05E-08	2E-08	1.0250
88	699	0.556		0.555	2.8E-04	2E-06	2.95E-08	9E-09	3.2778
'89	855	0.669		0.669	3.4E-04	1E-07	3.61E-08	1E-09	36.1000
TOTAL Kr		1.935		1.935					
Xe 131m	5	0.004		0.004	2.0E-06	4E-04	2.10E-10	2E-06	0.0001
133m	31	0.025		0.025	1.6E-05	1E-04	1.33E-09	6E-07	0.0022
133	1282	1.015		1.015	5.1E-04	1E-04	5.41E-08	5E-07	0.1082
135m	350	0.277		0.277	1.4E-04	9E-06	1.48E-08	4E-08	0.3700
135	1243	0.984		0.984	4.9E-04	1E-05	5.25E-08	7E-08	0.7500
'137	1185	0.938		0.938	4.7E-04	1E-07	5.00E-08	1E-09	50.0000
138	894	0.707		0.707	3.5E-04	4E-06	3.77E-08	2E-08	1.8850
TOTAL Xe		3.950		3.950					

' indicates generic <2 hr half-life values used

Continuation of Table 13.1

SELECTED RELEASE TOTALS	
Halogen Gamma Emitters	5.2 Ci
Halogen Beta Emitters	5.8 Ci
Total Halogens	5.87 Ci
Insoluble Gamma Emitters	3.52 Ci
Insoluble Beta Emitters	5.50 Ci
Total Insoluble Volatiles	5.89 Ci

Release of Fission Products to Unrestricted Areas

Columns H, I, and J of Table 13.1 are concerned with the exposure of personnel outside the restricted area. Calculations were performed as indicated in the appendix. The maximum concentrations which might be expected in unrestricted areas were calculated under the assumption that venting took place in the time required for the ventilation system to make one complete change in the laboratory (4238 seconds). Wind velocity was assumed to be the lowest average for any month.

The total dose to personnel in the unrestricted area is independent of whether the large exhaust fan or the normal ventilating fan is used; the concentration would be considerably higher if the larger fan were used, but the period of exposure would be proportionally shorter. It is also emphasized that the total exposure figure is a maximum to be expected at any point other than within the areas evacuated in the event of an accidental release.

The total of the ratios of instantaneous individual concentrations to 10CFR Part 20 Appendix B Table 2 maximum air concentrations for discharge is calculated to be 119.1, where the maximum air concentration values are for unrestricted areas, 168 hours per week. When averaged over a year's time, the resulting average concentration is 0.016 of the maximum indicated by 10 CFR Part 20 for non-occupational exposure in unrestricted areas. Even with the effluent discharge from normal operation (see Section 11.1.1) the total concentration to which personnel might be exposed is below the 10CFR Part 20 limits.

A more conservative calculation which assumes zero stack height (see Appendix A) was performed. This analysis is applicable to a situation in which the laboratory ventilation system fails and the release takes place through building leaks. For purposes of comparison, it was again assumed that the release occurred in the time required for the ventilation system to make an air change in the laboratory. The effect of this analysis is to multiply the values in columns H and I by a factor of 10.17, giving a resulting average concentration (yearly average) of 0.224 times 10CFR Part 20 Appendix B Table 2 limits.

Finally, an additional calculation was performed assuming 100% release of Br and I and the more conservative calculation (zero stack height) of atmospheric dilution. The resulting summation of ratio of concentrations to release limit in this case would be 3450. Averaged over a year's time, the resulting concentration (yearly average) is 0.655 times the 10CFR Part 20 Appendix B Table 2 limits, still within the permissible release concentration when averaged over a period of one year.

As indicated in the table, releases are in all cases less than the 10 CFR Part 20 Appendix B Table 2 limits when averaged over one year. As a backup check to assure that these calculations were conservative, cases equivalent to MHA were entered into the EPA COMPLY program[4] at level 4. Six of the radioisotopes in Table 13.1 are not in the COMPLY listing of radioisotopes (BR-85, Br-87, I-130m, I-136, Kr-89, and Xe-137) and are thus not included in the COMPLY results. Further, COMPLY does not permit zero height releases from a building, so the stack height was input as 1 meter. For the release with the ventilation system operable and the pool filled, COMPLY indicated an annual dose of 1.7E-2 mrem, with 1.4E-2 mrem due to Iodine. For the release with pool water lost and the ventilation system inoperable (assumed stack height of 1 meter) the COMPLY program indicated an annual dose of 0.1 mrem and 0.1 mrem from Iodine.

Table 13.2 Maximum Exposures In Unrestricted Areas from Maximum Hypothetical Accident

Assumed Failures	Total Body Dose	Thyroid Dose	Fraction of Part 20 Annual Limits
Fuel clad leak with normal operation of ventilation system; pool filled	0.006 rem	0.010 rad	0.016
Fuel clad leak with failure of ventilation system; pool filled	0.084 rem	0.102 rad	0.224
Fuel clad leak with failure of ventilation system and concurrent loss of pool water	0.153 rem	1.019 rad	0.655

13.1/2.2 Insertion of Excess Reactivity

The worst case result of insertion of excess reactivity would be insertion of the maximum allowed experiment reactivity worth or ejection of the transient rod (1.4% ΔK/K) while the reactor is operating at maximum steady-state power.

Calculations[5] performed by Gulf General Atomic indicate that a peak temperature of 1150 °C in FLIP fuel will not produce a stress in the fuel clad in excess of the ultimate yield strength. Further, TRIGA fuel with a H/Zr ratio of at least 1.65 has been pulsed to temperatures of about 1150 °C without any damage to the clad[6]. In a mixed FLIP-Standard TRIGA core the peak temperatures in FLIP fuel are much higher than in standard fuel due to the peaking of the power distribution near water gaps. For this reason the subsequent analysis in this section is concerned with internal temperatures in FLIP fuel elements.

A worst case core arrangement is considered, in which a FLIP element is located adjacent to a 3-inch square water gap. The power density in the FLIP element is at the maximum permissible value based on consideration of the loss of coolant accident (23 KW when the core is operating at 1 MW). The core is operating at the power level scram point of 1.25 MW, and the transient control rod is fired to initiate a pulse.

Pulses of 2.1% ΔK/K fired in standard TRIGA at this facility have had energy releases of less than 20 MW seconds. FLIP and mixed cores have been operated with maximum reactivity insertions for pulses reduced to 1.4%ΔK/K because of the shorter prompt neutron lifetime in FLIP fueled cores. Typical 1.4% reactivity pulses in an all-FLIP core have energy releases of only about 14 MW seconds, while mixed cores have slightly lower releases. Computations will be done for 20 MW second release.

The limitation of experiment reactivity to 1.4% ΔK/K will insure that reactivity insertions from experiment removal or failure will insure that such an accident will result in consequences no worse than those considered here.

Firing the transient rod while at full power is prevented by interlocks and administrative requirements. Removal of an experiment while operating at full power would not result in a reactivity insertion rate as large as that resulting from firing the transient rod, and the most likely result of experiment removal under the conditions assumed would be a reactor scram from power level, and fuel temperature trips. Further, experiments having worths approaching 1.4% Δ K/K are fastened to prevent inadvertent removal, and administrative restrictions do not allow such manipulations while the reactor is in operation. The predicted conditions establish an upper limit for a reactivity accident.

Fuel Temperatures from Operation at the Scram Point

Calculations for the SAR[7] of the Puerto Rico Reactor resulted in the information presented in the lower curve in **Figure 13-1** . This curve shows the fuel temperature distribution at the axial centerline in a FLIP fuel element operating at conditions of slightly higher power density than that assumed here. The Puerto Rico case is an element operating at a power density in the maximum element of 1.4 times the average of 22.3 KW/element. The axial peaking factor is 1.3. Calculations done for UWNR considered the case of an element operating at 23 KW times the ratio of 1.25 of scram setting/licensed power level, with the same axial peaking factor of 1.3. Using these numbers, the fuel centerline and average temperatures will be lower in the UWNR core, but the temperature at the outer surface of the fuel would be approximately the same in both cases.

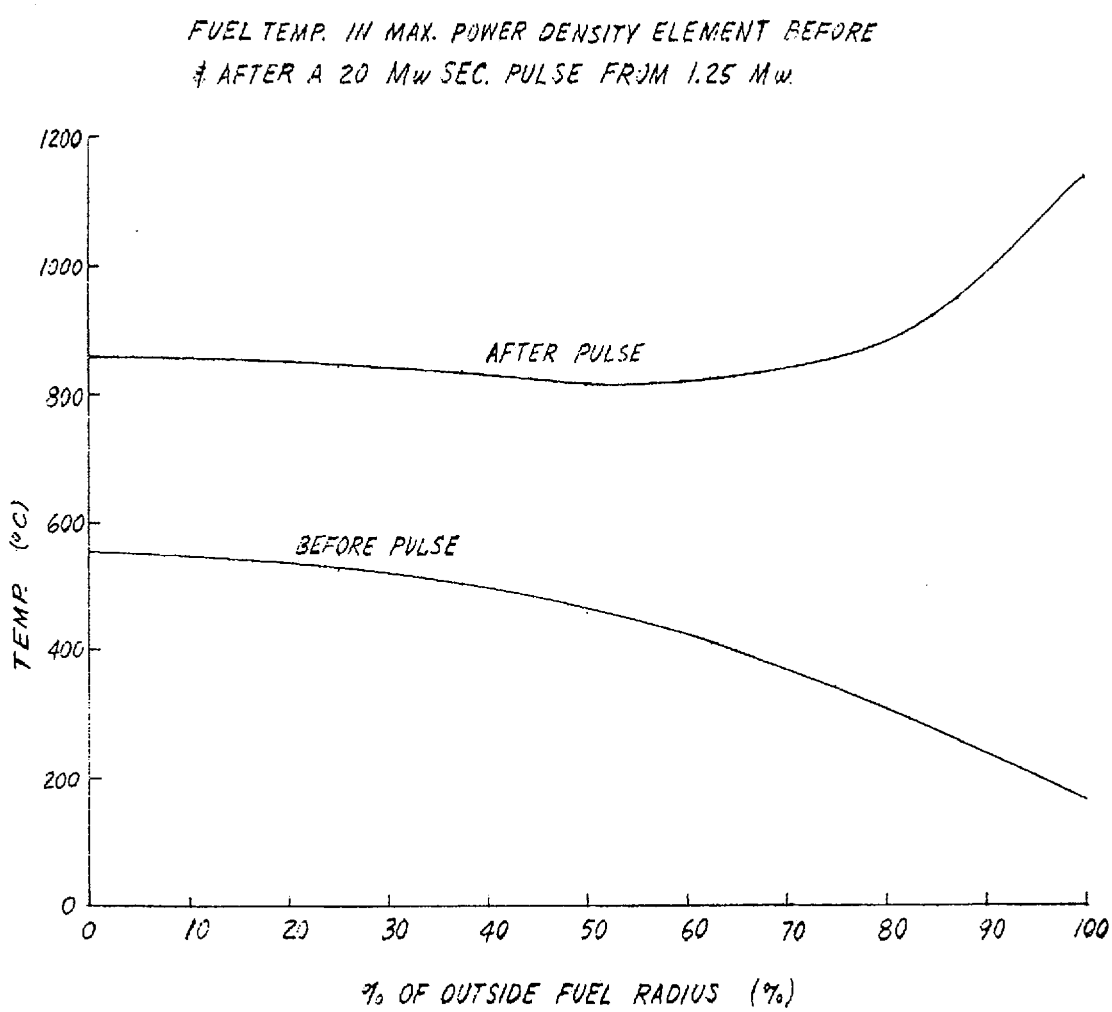


Figure 13-1 Fuel Temperature Distribution in a Fuel Element

Temperature after Pulse

Firing a pulse while at the scram point would cause the reactor to scram from power level and fuel temperature scrams. The entire pulse energy release is used, however, in the following analysis.

The temperature distribution in the fuel element immediately after a 20 MW second pulse is plotted as the top curve in **Figure 13-1** . The peaking factor within a FLIP element adjacent to a 3-inch square water-gap is 2.49, and an axial peaking factor of 1.3 is used as in the steady state conditions. The energy deposited in the element under consideration is calculated using the same peaking factor (power in maximum element/ power in average element in core) which resulted in the 23 KW steady state level.

The maximum adiabatic temperature reached in the element will occur at the outer surface of the fuel element adjacent to the water-gap. This maximum temperature would be 1133 C, slightly below the safety limit of 1150 C.

Although such an event is considered highly unlikely, it would not cause fuel damage or release of fission products from the reactor.

After these computations were completed, NRC requested that the accident be re-evaluated for the permitted cores under the technical specifications that were proposed for UWNR. The major changes were limiting the minimum FLIP content to be 9 fuel bundles (35 elements, since the transient rod is located within the 9 central fuel bundles in the UWNR core).

The re-analysis was based on a lower power in maximally exposed fuel element (18.1 kW instead of 22.3 kW) and a limitation of the reactivity insertion from 2.1% ΔK/K to 1.4%V. First, the temperature of the fuel in the maximum element will be lower at the beginning of the pulse by about 70°C. Second, the use of a compact array of nine (9) FLIP bundles reduces the possible peaking factor within a FLIP element from the 2.49 value used in the original calculation to a value of 2.03 for a FLIP element beside the transient rod guide tube (this is the position with highest power density in the core.) Finally, reduction of allowable pulsed reactivity insertion from 2.1% ΔK/K to 1.4% ΔK/K will substantially reduce the energy generation in a pulse, while the limitation of experiment worth to 1.4% ΔK/K will provide similar safeguards for experiment failure or removal. Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of 1.4% ΔK/K resulted in a maximum fuel temperature rise of approximately 400 °C [8], and measurements at Wisconsin confirmed that prediction.

Consideration of all these differences shows a peak fuel temperature of about 450 °C lower than that indicated above. It is therefore concluded that fuel damage would occur in neither case, but with a much larger safety margin in the more restrictive case considered here.

13.1/2.3 Loss of Coolant

Although there is little likelihood of complete loss of water from the reactor pool, an analysis is made to demonstrate that such loss will not damage reactor fuel.

Possible Means of Water Loss

The pool is contained within the thick reinforced concrete reactor shield which will maintain its integrity under the most severe earthquake that would be expected in this area.

A sheared and open beam port could drain the water level to mid-core height in about 400 seconds, but water would still be in contact with the fuel and would prevent excessive temperatures.

The 8-inch stainless steel pipes built into the pool walls for possible future use in a forced convection cooling system are flange sealed on the outer ends. In addition, one of these pipes has a loop and a siphon breaker extending well above the core so that a rupture cannot lower pool level below the core. The other pipe is flange sealed inside the pool and penetrates the shield wall well above the core. Rupture of either of these lines will not uncover the core.

Rupture of the piping in the demineralizer could cause only slight water loss due to location of the outlet lines from the pool and a check valve at the demineralizer outlet.

Radiation Levels Due to Unshielded Core

Calculations of radiation levels at various points in the Reactor Laboratory were made assuming operations at 1000 kW for an infinite time. Doses from direct and scattered radiation were considered, with the scattered dose calculated for the case of a thick concrete ceiling nine (9) feet above the pool. Results of the calculations are given in Table 13-3.

Table 13.3 Calculated Radiation Dose Rates After Pool Water Is Lost

Time After Shutdown	Dose at Floor Level R/hr	Dose at Console R/hr	Direct Radiation at Pool Curb R/hr	Pool Top Scattered Radiation R/hr
10 seconds	1.6	2.0	1.0E4	2.6
1 day	0.17	0.24	1.2E3	0.30
1 week	0.08	0.12	5.4E2	0.14
1 month	0.03	0.04	1.4E2	0.04

These levels are not too high to allow emergency repairs to be made. Facility emergency procedures cover the situation of pool water loss.

Fuel Temperature After Loss of Pool Water

Calculations performed at Texas A & M University have treated the loss of coolant accident in detail, based on reactor shutdown 15 minutes before the core is uncovered. At Wisconsin, the pool level scram would cause automatic shutdown much sooner, as the A & M calculation is based on pool drainage by rupture of a 10-inch line. Other pertinent parameters of the two facilities are identical. The calculations employed the Gulf computer code TAC for calculation of system temperatures.

The results of these calculations (Pages 25-31 of Texas A & m University Nuclear Science Center Amendment II to the Safety Analysis Report, November 1, 1972 submitted under Docket for License R-83) indicate that for a maximum power density of less than 21 kW/element for standard fuel and 23 kW/element for FLIP fuel, loss of coolant water would not result in fuel clad failure and release of fission products.

13.1/2.4 Loss of Coolant Flow

Not applicable; natural convection cooling

13.1/2.5 Mishandling or Malfunction of Fuel

Reference 1 states that this condition produces the maximum consequence to the public. This accident is therefore included as the maximum hypothetical accident (Section 13.1/2.1), when combined with failure of the ventilation system and loss of pool water. The effect of a fuel clad failure with normal pool level and with the ventilation system operating normally has no significant effect on the public.

13.1/2.6 Experiment Malfunction

Experiment reactivity worth and composition are controlled and limited so that experiment failure will not insert a step change in reactivity greater than 1.4% $\Delta K/K$ (fixed experiments; movable experiments are limited to 0.7% $\Delta K/K$. Procedure for experiment review includes consideration of chemical and explosive hazards to the reactor. Any experiment containing fissionable material is limited so that production of gaseous and volatile fission products results in releases lower than that considered in section. Therefore, experiment malfunction will not result in consequences more severe than those listed in other parts of this chapter.

13.1/2.7 Loss of Normal Electrical Power

13.1/2.7 Loss of Normal Electrical Power

Loss of normal electrical power will cause the reactor to shut down. It will not result in any release of radioactive material or increase the dose to the population. Emergency core cooling engineered safety systems are not required. The maximum hypothetical accident analysis does include loss of the ventilation system in the analysis, thus effectively including loss of electrical power.

13.1/2.8 External Events

Since the safety of a TRIGA reactor is so strongly a function of the fuel composition and characteristics, none of the usual external event initiators will cause any effect on the public. As stated in chapter 2 of this report, floods and hurricanes are an insignificant threat to the safety of the reactor (Section 2.4), with the 100 year flood causing nearby Lake Mendota to expand by only 30 feet, not threatening the laboratory. Further, should the water level within the room rise above ground level it would not affect the safety of the reactor. Tornados do occur in the Midwest, but damage to the concrete shield which protects the reactor core is incredible. The seismicity of the area is extremely low, with the estimated 50 year peak ground acceleration to a seismic event is less than 0.01 g (Section 2.5.4). Since no engineered structures other than the reactor shield are required to provide protection to the reactor, such an acceleration will have no effect on the reactor. Likewise, though aircraft collisions with the building are not impossible, they are unlikely due to the location of flight paths. Further, such impacts will not breach the concrete shield at core level. Impacting the outer walls of the building will not result in radiation being released.

[redacted]

13.1/2.9 Mishandling or Malfunction of Equipment

An analysis was made of the possibility that loss of water from the reactor or from the radioactive liquid waste storage tanks could affect the city water supply, and a negative result was obtained as indicated below. Madison obtains its drinking water supply from several wells drilled into the Cambrian sandstone described above. The location of these wells is shown on **Figure 2.10**, and they supply the University as well as the city. All of these wells are cased from ground level into the sandstone so as to keep out water from the glacial deposit. The closest well to the reactor site is about 2,000 feet southeast.

The Reactor Laboratory floor drain empties into the hold tank. Should the entire contents of the pool be let out into the room, however, some water could escape into the sewer system through a drain thimble into which waste water is pumped from the hold tank or a similar thimble into which blowdown from the cooling tower is directed. There are four methods by which water may leave the reactor room:

(1) by flow pumped from the radioactive waste storage tanks through the elevated drain thimble provided for emptying the tanks;

(2) directly into the drain thimble should the pool be completely ruptured, thus reaching a level high enough to overflow into the thimbles or escape from the laboratory and enter floor drains in surrounding areas;

(3) by loss through the floor and into the ground; and

(4) by rupture of the liquid radioactive waste storage tank directly into the soil under the laboratory floor.

Analysis for cases (1) and (2)

In so far as the first two discharge paths are concerned, the flow through the drain thimble or floor drains empties into a sanitary sewer main. From there it would travel through mains via a pumping station to the main sewage plant, located south of and outside the corporate limits of the city. From there, the sewage travels through mains an additional five miles to the south before it empties into an open ditch. On the way, any water from the reactor would become considerably diluted since the minimum flow-rate into the ditch is 7,000 gpm whereas the probable maximum rate of entry into the floor drain would not be more than 10 to 100 gpm. These facts, coupled with the fact that stringent administrative precautions will be taken to ensure that water contaminated beyond established tolerance levels is not released to the drain, tend to preclude that the city water supply could be adversely affected by this method.

Analysis for case (3)

The possibility that the city water supply could be affected via the second method (2) is also negligible. The base of the reactor is about 8 feet below ground level, and water cannot be dissipated via surface run-off. Since the walls of the building surrounding the reactor are made of concrete up to the ground level, significant water loss through the floor could result only if the concrete was breached. In fact, it would appear that the only mechanism by which contaminated water could enter the soil would be the result of an earthquake sufficiently severe to rupture both the reactor tank and shield, as well as the floor of the building, at a time when the reactor pool water was radioactive beyond tolerance levels. Such a set of coincidental occurrences is considered extremely remote. Further, even if it did occur, there is no assurance that the water supply would be adversely affected. For example, the nearest city well is about 2,000 feet from the reactor site, and it has been estimated by a ground water specialist that water would flow through the sandstone from the reactor to the well at not more than 0.1 foot per day. Thus, as long as 55 years might be required for the reactor water to reach the well.

Analysis for case (4)

Should the radioactive waste storage tanks rupture, a similar analysis to that in case (3) indicates no adverse effect on the well. Furthermore, the quantities of water likely to be lost are small and activities are expected to be low enough that no hazard exists.

13.3 Summary and Conclusions

None of the accidents considered here will result in consequences to the public health and safety. Even the maximum hypothetical accident does not result in releases of radioactivity in excess of 10CFR Part 20 limits when averaged over a year.

13.4 References:

1. NUREG-1537 Part 1, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, USNRC, February 1996

2. Credible Accident Analyses for TRIGA and TRIGA-fueled Reactors, NUREG/CR-2387, PNL-4028, April 1982

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

4. COMPLY V1.5d, U.S. Environmental Protection Agency, Office of Radiation and Indoor air, Washington DC 20460, October 1989

5. "Safety Analysis Report for the Torrey Pines TRIGA Mark III Reactor", GA-9064, Gulf General Atomic, Jan. 5, 1970.

6. "Annular Core Pulse Reactor", General Dynamics, General Atomic Division Report GACD 6977, Supplement 2, 9/30/66

7. Safeguards Summary Report for the TRIGA-FLIP Reactor at Puerto Rico Nuclear Center, Report PRNC 123, Revision C, November 11, 1969.

8. Docket 50-120, Change No. 11 to the Technical Specifications Facility License R-83, Texas A & M University, Section 3.2 Basis

14 TECHNICAL SPECIFICATIONS

1 INTRODUCTION

1.1 SCOPE

This section of the SAR for license renewal of the University of Wisconsin Nuclear Reactor constitutes the proposed Technical Specifications for that facility as required by 10 CFR 50.36. This document includes "Bases" to support the selection and significance of the specifications. The bases are included for information purposes only, and are not part of the Technical Specifications in that they do not constitute requirements or limitations which the licensee must meet in order to meet the specifications. Dimensions, measurements, and other numerical values given in these specifications may differ slightly from actual values due to construction and manufacturing tolerances or normal degree of accuracy or of instrument readings.

These specifications are re-formatted from the technical specifications in force in 1999. Changes reflect only changes required by name changes or to include information not in the original technical specifications. In addition, certain additions required by NUREG-1537 are included. All substantive changes are denoted by redlining. These technical specifications continue to include use of TRIGA-FLIP and the original LEU TRIGA fuels, either separately or in mixed cores.

1.2 FORMAT

Content and section numbering is in accordance with section 1.2.2 of ANSI/ANS 15.1.

1.3 DEFINITIONS

The terms used herein are explicitly defined to ensure uniform interpretation of the Technical Specifications.

1.3.1 Reactor Operating Conditions

COLD CRITICAL:
The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures both below 125°F.

PULSE MODE (PU)
Pulse mode operation shall mean any operation of the reactor with the mode selector switch in the pulse position.

REACTOR SECURED:
The reactor is secured when:

(1) Either there is insufficient moderator available in the reactor to attain criticality or there is insufficient fissile material present in the reactor to attain criticality upon optimum available conditions of moderation and reflection, or

(2). The following conditions exist:
  a. The reactor is shut down,
  b. The console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator or stored in a locked storage area, and
  c. No work is in progress involving in-core fuel handling or refueling operations, maintenance of the reactor or its control mechanisms, or insertion or withdrawal of in-core experiments with a reactivity worth exceeding 0.7% $\Delta K/K$.

REACTOR SHUTDOWN:
The reactor is shut down when the reactor is subcritical by least 0.7% $\Delta k/k$ of reactivity.

REACTOR OPERATION:
Reactor operation is any condition wherein the reactor is not secured.

REPORTABLE OCCURRENCE:
A reportable occurrence is any of the following that occur during reactor operation:

(1) Operation with any safety system setting less conservative than specified in the technical specifications;
(2) Operation in violation of a Limiting Condition for Operation listed in Section 3;
(3) Operation with a required reactor or experiment safety system component in an inoperative or failed condition which could render the system incapable of performing its intended safety function;
(4) Any unanticipated or uncontrolled change in reactivity greater than 0.7% $\Delta K/K$, excluding reactor trips from a known cause;
(5) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of a condition which could result in operation of the reactor outside the specified safety limits; and
(6) Abnormal and significant degradation in reactor fuel or cladding which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

SHUTDOWN MARGIN:
Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems, starting from any permissible operating condition (assuming the most reactive scrammable control element and any non-scrammable control elements remain full out), and the reactor will remain subcritical without further operator action.

SQUARE WAVE MODE (SW)
Square wave mode operation shall mean any operation of the reactor with the mode selector switch in the square wave position .

STEADY STATE MODE (SS)
Steady state mode operation shall mean operation of the reactor with the mode selector switch in the manual or automatic positions.

1.3.2 Reactor Experiments and Irradiation

EXPERIMENT:
Experiment shall mean
(1) any apparatus, device or material which is not a normal part of the reactor core or experimental facility, or
(2) any activity external to the biological shield using a beam of radiation emanating from the reactor core ,or
(3) any operation designed to measure reactor parameters or characteristics, or any activity external to the biological shield using a beam of radiation emanating from the reactor core:

Classification of experiments shall be:
(1) Routine experiments. Routine experiments are those which have previously been performed at the facility.

(2) Modified routine experiments. Modified routine experiments are those which have not been performed previously but are similar to the routine experiments in that the hazards are neither greater nor significantly different than those for the corresponding routine experiments.

(3) Special experiments. Special experiments are those which are not routine or modified experiments.

EXPERIMENTAL FACILITIES:
Experimental facilities shall mean beam ports, including extension tubes with shields, thermal columns with shields, vertical tubes, through tubes, in-core irradiation baskets, irradiation cell, pneumatic transfer systems and any other in-pool irradiation facilities.

IRRADIATION:
Irradiation shall mean the insertion of any device or material that is not a normal part of the core or experimental facilities into an experimental facility so that the device or material is exposed to a significant amount of the radiation available in that irradiation facility.

SECURED EXPERIMENT:
A secured experiment shall mean any experiment that is held firmly in place by a mechanical device or by gravity, that is not readily removable from the reactor, and that requires one of the following actions to permit removal:

(1) removal of mechanical fasteners
(2) use of underwater handling tools
(3) moving of shield blocks or beam port containers.

1.3.3 Reactor Components

CORE LATTICE POSITION:
A core lattice position is that region in the core (approximately 3" by 3") over a grid hole. It may be occupied by a fuel bundle, an experiment or experimental facility, or a reflector element.

FUEL BUNDLE:
A fuel bundle is a cluster of three or four fuel elements secured in a square array by a top handle and a bottom grid plate adaptor.

FUEL ELEMENT:
A fuel element is a single TRIGA fuel rod of either standard or FLIP type.

FLIP CORE:
A FLIP core is an arrangement of TRIGA-FLIP fuel in the reactor grid plate.

FLIP FUEL:
FLIP fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of about 70% and erbium as burnable poison.

INSTRUMENTED ELEMENT:
An instrumented element is a special fuel element in which thermocouples are embedded for the purpose of measuring fuel temperatures during reactor operation.

MIXED CORE:
A mixed core is an arrangement of standard TRIGA fuel elements and FLIP fuel elements with at least 35 TRIGA-FLIP fuel elements located in a central region of the core.

OPERATIONAL CORE:
An operational core may be a standard core, mixed core, or FLIP core for which the core parameters of shutdown margin, fuel temperature, power calibration, and maximum allowable reactivity insertion have been determined to satisfy the requirements of the Technical Specifications.

REGULATING BLADE:

The regulating blade is a low worth control blade that need not have scram capability. Its position may be varied manually or by the servo-controller.

SHIM-SAFETY BLADE:
A shim-safety blade is a control blade having an electric motor drive and scram capabilities. Its position may be varied manually or by the servo-controller.

STANDARD TRIGA FUEL:
Standard TRIGA fuel is TRIGA fuel that contains a nominal 8.5 weight percent of uranium with a ^{235}U enrichment of less than 20% and no burnable poison.

STANDARD CORE:
A standard core is an arrangement of standard TRIGA fuel in the reactor grid plate.

TRANSIENT ROD:
The transient rod is a control rod with scram capabilities that can be rapidly ejected from the reactor core to produce a pulse. Its position may be varied manually or by the servo-controller. It may have a voided or solid aluminum follower.

1.3.4 Reactor Instrumentation:

CHANNEL CALIBRATION:
A channel calibration consists of comparing a measured value from the measuring channel with a corresponding known value of the parameter so that the measuring channel output can be adjusted to respond with acceptable accuracy to known values of the measured variable.

CHANNEL CHECK:
A channel check is a qualitative verification of acceptable performance by observation of channel behavior.

CHANNEL TEST:
A channel test is the introduction of a signal into the channel to verify that it is operable.

EXPERIMENT SAFETY SYSTEMS:
Experiment safety systems are those systems, including their associated input circuits, which are designed to initiate a scram for the primary purpose of protecting an experiment or to provide information which requires manual protective action to be initiated.

LIMITING SAFETY SYSTEM SETTINGS:
Limiting safety system settings are settings for automatic protective devices related to those variables having significant safety functions.

MEASURED VALUE:
The measured value is the magnitude of that variable as it appears on the output of a measuring channel.

MEASURING CHANNEL:
A measuring channel is the combination of sensor, interconnecting cables or lines, amplifiers, and output device which are connected for the purpose of measuring the value of a variable.

OPERABLE:
A system, device, or component shall be considered operable when it is capable of performing its intended functions in a normal manner.

REACTOR SAFETY SYSTEMS:
Reactor safety systems are those systems, including their associated input circuits, which are designed to initiate a reactor scram for the primary purpose of protecting the reactor or to provide information which requires manual protective action to be initiated.

SAFETY CHANNEL:
A safety channel is a measuring channel in the reactor safety system.

SAFETY LIMITS:
Safety limits are limits on important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity.

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability
This specification applies to fuel element temperature and steady-state reactor power level.

Objective
The objective is to define the maximum fuel element temperature and reactor power level that can be permitted with confidence that no fuel element cladding failure will result.

Specifications

(1) The temperature in a TRIGA-FLIP fuel element shall not exceed 1150°C under any conditions of operation.

(2) The temperature of a standard TRIGA fuel element shall not exceed 1000°C under any conditions of operation.

(3) The reactor steady-state power level shall not exceed 1500 kW under any conditions of operation.

Bases

A loss of integrity of the fuel element cladding could arise from a buildup of excessive pressure between the fuel moderator and the cladding if the fuel temperature exceeds the safety limit. The pressure is caused by air, fission product gases, and hydrogen from dissociation of the fuel moderator. The magnitude of this pressure is determined by the fuel moderator temperature and the ratio of hydrogen to zirconium in the alloy.

The safety limit for the TRIGA-FLIP fuel element is based on data which indicate that the stress in the cladding due to hydrogen pressure from the dissociation of zirconium hydride will remain below the ultimate stress provided the temperature does not exceed 1150°C and the fuel cladding is water cooled [1].

The safety limit for the standard TRIGA fuel is based on data including the large amount of experimental evidence obtained during high performance reactor tests of this fuel. These data indicate that the stress in the cladding (due to hydrogen pressure from the dissociation of zirconium hydride) will remain below the ultimate stress provided that the temperature of the fuel does not exceed 1000°C and the fuel cladding is water cooled [2].

It has been shown by experience that operation of TRIGA reactors at a power level of 1500 kW will not result in damage to the fuel. Several reactors of this type have operated successfully for several years at power levels up to 1500kW. It has been shown by analysis and by measurements on other TRIGA reactors that a power level of 1500 kW corresponds to a peak fuel temperature of approximately 600°C. Thus a Safety Limit on power level of 1500 kW provides an ample margin of safety for operation.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the scram setting which prevents the safety limit from being reached.

Objective

The objective is to prevent the safety limits from being reached.

Specifications

(1) The limiting safety system setting for fuel temperature shall be 400°C (750°F) as measured in an instrumented fuel element. For a mixed core, the instrumented element shall be located in the region of the core containing FLIP type elements.

(2) The limiting safety system setting for reactor power level shall be 1.25 MW.

Bases:

The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated preventing the safety limit from being exceeded. A setting of 400°C provides a safety margin of 750°C for FLIP type fuel elements and a margin of 600°C for standard TRIGA fuel elements. A part of the safety margin is used to account for the difference between the true and measured temperatures resulting from the actual location of the thermocouple. If the thermocouple element is located in the hottest position in the core, the difference between the true and measured temperatures will be only a few degrees since the thermocouple junction is at the mid-plane of the element and close to the anticipated hot spot. If the thermocouple element is located in a region of lower temperature, such as on the periphery of the core, the measured temperature will differ by a greater amount from that actually occurring at the core hot spot. Calculations and measurements made at the facility with all permitted mixed core arrangements indicate that, for this case, the true temperature at the hottest location in the core will differ from the measured temperature by no more than a factor of two. Thus, when the temperature in the thermocouple elements reaches the trip setting of 400°C, the true temperature at the hottest location would be no greater than 800°C providing a margin to the safety limit of at least 200°C for standard fuel elements and 350°C for FLIP type elements. These margins are ample to account for the remaining uncertainty in the accuracy of the fuel temperature measurement channel and any overshoot in reactor power resulting from a reactor transient during steady state mode operation. For a mixed core (i.e., one containing both standard and FLIP type elements), the requirement that the instrumented element be located in the FLIP region of the core provides an even greater margin of safety since the peak to average power ratio within that region will be smaller than over an entire core composed of elements of the same type.

Calculations and measurements for this and similar TRIGA reactors indicate at 1.25 MW, the peak fuel temperature in the most limiting core loading permitted under section 3 of these specifications (9 FLIP bundles) will be less than 600°C so that the limiting power level setting provides an ample safety margin to accommodate errors in power level measurement and anticipated operational transients.

In the pulse mode of operation, the same limiting safety system settings will apply. However, the power level channels do not provide protection in pulse mode, and the temperature channel will have no effect on limiting the peak powers generated because of its relatively long time constant (seconds) as compared with the width of the pulse (milliseconds). The limit on transient rod worth in another specification limits the generated power so that fuel temperatures reached in a transient are smaller than those from full power operation. This transient rod worth limit is less than the reactivity required for steady-state full power. If the transient rod fails to automatically drop after the pulse, fuel temperature reached due to energy generation in the tail of a pulse will be less than that at full power. Only in the case of operation outside the permitted parameters of core composition and reactivity limitations would fuel temperature safety system actuation be needed to provide protection in any operating mode.

3 LIMITING CONDITIONS FOR OPERATION

3.1 Reactor Core Parameters

Applicability
These specifications apply to the reactivity condition of the reactor and the reactivity worths of control rods. They apply for all modes of operation.

Objective
The objective is to assure that the reactor can be shut down at all times and to assure that the fuel temperature safety limit will not be exceeded.

3.1.1 Excess Reactivity

Specifications:
The excess reactivity shall not exceed 5.6% Δk/k.

Bases
As shown in chapter 4 of the SAR, this amount of excess reactivity will provide the capability to operate the reactor at full power with experiments in place. The primary limitation providing reactivity safety , however, is the shutdown margin requirement discussed in the next specification.

3.1.2 Shutdown Margin

Specifications
The reactor shall not be operated unless the shutdown margin provided by control rods shall be greater than 0.2% Δk/k with:

(1) the highest worth non-secured experiment in its most reactive state,

(2) the highest worth control rod and the regulating blade (if not scrammable) fully withdrawn, and
(3) the reactor in the cold condition without xenon.

Bases
The value of the shutdown margin assures that the reactor can be shut down from any operating condition even if the highest worth control rod should remain in the fully withdrawn position. If the regulating blade is not scrammable, its worth is not used in determining the shutdown reactivity.

3.1.3 Pulse Limits

Specifications
(1) The reactivity to be inserted for pulse operation shall be determined and mechanically limited such that the reactivity insertion will not exceed 1.4% $\Delta k/k$.
(2) Pulses shall not be initiated at power levels exceeding 1 kilowatt .

Basis
Measurements performed on the Puerto Rico Nuclear Center TRIGA-FLIP reactor indicated that a pulse insertion of reactivity of 1.4% Δ k/k resulted in a maximum temperature rise of approximately 400°C. With an ambient water temperature of approximately 100°C, the maximum fuel temperature would be approximately 500°C resulting in a safety margin of 500°C for standard fuel and 650°C for FLIP type fuel. Tests done on the mixed and all-FLIP cores [3,4,5,6] indicate that the fuel temperatures measured and calculated for the core arrangements allowed by these specifications do not exceed 683°C in the worst case allowed.

The temperature rise from pulse initiation is in addition to the temperature in the fuel at the time the pulse is initiated. Limiting the initial power level to 1 kW assures that excessive temperatures will not be reached.

These margins allow amply for uncertainties due to the accuracy of measurement or location of the instrumented fuel element or due to the extrapolation of data from the PRNC reactor.

3.1.4 Core Configurations

Applicability
This specification applies to the configuration of fuel and in-core experiments.

Objective
The objective is to assure that provisions are made to restrict the arrangement of fuel elements and experiments so as to provide assurance that excessive power densities will not be produced.

Specifications

(1) The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.

(2) The TRIGA core assembly may be standard, FLIP, or a combination, thereof (mixed core) provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.

(3) The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.

(4) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

(5) Fuel shall not be inserted or removed from the core unless the reactor is subcritical by more than the calculated worth of the most reactive fuel assembly.

(6) Control elements shall not be manually removed from the core unless the core has been shown to be subcritical with all control elements in the full out position.

Bases

(1) Standard TRIGA cores have been in use for years and their characteristics are well documented. The Puerto Rico Nuclear Center and the Gulf Mark III all-FLIP cores have operated and their characteristics are available. Gulf has also performed a series of experiments using standard and Flip fuel in mixed cores and a mixed core has been used successfully in the Texas A&M University TRIGA reactor. In addition, studies performed at Wisconsin for a variety of mixed core arrangements indicate that such cores with mixed loadings would safely satisfy all operational requirements (SAR Chapters 4 and 6).

(2) In mixed cores, it is necessary to arrange FLIP elements in a contiguous, central region of the core to control flux peaking and power generation peak values in individual elements.

(3) Vacant core lattice positions will contain experiments or an experimental facility to prevent accidental fuel additions to the reactor core. They will be permitted only on the periphery of the core to prevent power perturbations in regions of high power density.

(4) The core will be assembled in the reactor grid plate which is located in a pool of light water. Water in combination with graphite reflectors can be used for neutron economy and the enhancement of experimental facility radiation requirements.

(5),(6) Manual manipulation of core components will be allowed only when a single manipulation can not result in inadvertent criticality.

3.1.5 Reactivity Coefficients

Does not apply to TRIGA and TRIGA-FLIP reactors.

3.1.6 Fuel Parameters

Applicability
This specification applies to the dimensional and structural integrity of the fuel elements.

Objective
The objective is to assure that the reactor will not be operated with defective fuel elements installed.

Specifications
The reactor shall not be operated with damaged fuel except for purposes of identifying the damaged fuel. A fuel element shall be considered damaged and must be removed from the core if:

(1) In measuring the transverse bend, its sagitta[7] exceeds 0.125 inch over the length of the cladding;

(2) In measuring the elongation, its length of the cladding exceeds its original length by 0.125 inch; and

(3) A clad defect exists as indicated by detection of release of fission products.

(4) The fuel has not been visually inspected within the previous 15 months.

(5) The burnup of uranium-235 in the UzrH fuel matrix shall not exceed 50 percent of the initial concentration.[8,9]

Basis
The limit of transverse bend has been shown to result in no difficulty in disassembling the core. Analysis of the removal of heat from touching fuel elements shows that there will be no hot spots resulting in damage to the fuel caused by this touching. Experience with TRIGA reactors has shown that fuel element bowing that could result in touching has occurred without deleterious effects. The elongation limit has been specified to assure that the cladding material will not be subjected to stresses that could cause a loss of integrity in the fuel containment and to assure adequate coolant flow through the top grid plate.

3.2 Reactor Control and Safety Systems

3.2.1 Operable Control Rods

Applicability
This specification applies to the number of operable control elements that must exist in order to operate the reactor.

Objective
The objective of this requirement is to insure that the reactor may be shut down from any condition of operation.

Specification
The reactor shall not be operated unless at least three control elements are functioning and scrammable.

Basis: In most cores the limits on shutdown margin actually dictate the number of operable control elements required. Non-pulsing cores do not require presence of a transient control rod if the shutdown margin requirements are met by the control blades.

3.2.2 Reactivity Insertion Rates (Scram time)

Applicability:
This specification applies to the time required for the scrammable control elements to be fully inserted from the instant that a safety channel variable reaches the Safety System Setting.

Objective
The objective is to achieve prompt shutdown of the reactor to prevent fuel damage.

Specification
The scram time measured from the instant a simulated signal reaches the value of the LSSS to the instant that the slowest scrammable control element reaches its fully inserted position shall not exceed 2 seconds.

Basis
This specification assures that the reactor will be promptly shut down when a scram signal is initiated. Experience and analysis have indicated that for the range of transients anticipated for a TRIGA reactor, the specified scram time is adequate to assure the safety of the reactor.

3.2.3 Other Pulsed Operation Limitations

Limitations other than those on core configuration and pulsed reactivity insertion limits are not required on this reactor.

3.2.4 Reactor Safety System

Applicability
This specification applies to the reactor safety system channels.

Objective
The objective is to specify the minimum number of reactor safety channels that must be operable for safe operation.

Specification
The reactor shall not be operated unless the safety channels described in Table 3.2.4 are operable.

TABLE 3.2.4 Reactor Safety System Channels

Safety Channel	Setpoint and Function	Number operable in specified mode		
		SS	SW	PU
(1) Fuel Temperature	Scram if fuel temperature exceeds >400°C in the fuel temperature safety channel. In the event of loss of all available fuel thermocouples and inability to obtain a replacement instrumented fuel element, operation may continue in any operational core if the linear power level scram points are reduced to 110% full power.	1	1	1
(2) Linear Power Level	Scram if power > 125% full power	2	2	-
(3) Manual Scram	Manually initiated scram	1	1	1
(4) Preset Timer	Transient rod scram 15 seconds or less after pulse	-	-	1
(5) Reactor water level	Scram if < 19 feet above top of core	1	1	1
(6) High Voltage Monitor	Scram on loss of high voltage to neutron and gamma ray power level instrument detectors	1	1	1

Bases

(1) and (2)The fuel temperature and power scrams provide protection to ensure that the reactor is shut down before the safety limit on fuel temperature is reached.

The exception is required because FLIP fuel is no longer manufactured. If a core has been tested to meet the definition of an operational core the power level scrams provide adequate protection to assure the LCO of fuel temperature is not exceeded.

(3) The manual scram allows the operator a means of rapid shutdown in the event of unsafe or abnormal conditions.

(4) The preset timer assures reduction of reactor power to a low level after a pulse.

(5) The reactor pool water level scram assures shutdown of the reactor in the event of a serious leak in the primary system or pool.

(6) The high voltage monitor prevents operation of the reactor with other systems inoperable due to failure of the detector high voltage supplies.

3.2.5 Interlocks

Applicability
This section applies to the interlocks which inhibit or prevent control element withdrawal or reactor startup.

Objective
The objective of these interlocks is to prevent operation under unanalyzed or imprudent conditions.

Specifications
The reactor shall not be operated in the indicated modes unless the interlocks in Table 3.2.5 are operable.

Table 3.2.5 Interlocks

		Number operable in specified mode		
Channel	Setpoint and Function	SS	SW	PU
Log Count Rate	Prevent control element withdrawal when neutron count rate < 2 per second	1	1	1
Transient Rod Control	Prevent application of air unless drive is at IN limit.	1	0	0
Log N Power Level	Prevent firing transient rod when power level is above 1 kW and transient rod is not full in.	1	1	1
Pulse Mode Control	Prevents withdrawal of control blades while in pulse mode.	0	0	1

Bases

The Log count rate interlock does not allow control element withdrawal unless the neutron count rate is high enough to assure proper instrument response during reactor startup.

The Transient Rod Control interlock prevents inadvertent addition of excessive amounts or reactivity in steady-state modes.

The Log N interlock prevents firing of the transient rod at power levels above 1.0 kW if the transient rod drive is not in the full down position. This effectively prevents inadvertent pulses which might cause fuel temperature to exceed the safety limit on fuel temperature.

The pulse mode control blade withdrawal interlock prevents reactivity addition in pulse mode other than by firing the transient rod.

3.2.6 Backup Shutdown Mechanisms

Backup shutdown mechanisms are not required for this reactor.

3.2.7 Bypassing Channels

Applicability
This specification applies to the interlocks in Table 3.2.5.

Objective
The objective is to indicate the conditions in which an interlock may be bypassed.

Specification
The Log Count Rate interlock in Table 3.2.5 may be bypassed

1. During fuel loading in order to allow control element withdrawal necessary for the fuel loading procedure or
2. when Log Power Level and Linear Power Level channels are on-scale.

Bases
During early stages of fuel loading the count-rate on the source range channel will be below the interlock setpoint. The bypass allows control element movements necessary for loading fuel with control elements partially withdrawn and for performing inverse multiplication determinations of control element worth and core reactivity status. Once the other power indications are available the startup count rate channel is no longer required, so the interlock no longer serves any purpose.

3.2.8 Control Systems and Instrumentation Required for Operation

Applicability
This specification applies to the information which must be available to the reactor operator during reactor operation.

Objective:

The objective is to require that sufficient information is available to the operator to assure safe operation of the reactor.

Specification

The reactor shall not be operated unless measuring channels listed in Table 3.2.8 are operable.

Table 3.2.8 Instrumentation and Controls Required for Operation

Channel	Function	Number operable in specified mode SS	SW	PU
Fuel Temperature	Input for fuel temperature scram. In the event of loss of all available fuel thermocouples and inability to obtain a replacement operation may continue in any operational core.	1	1	1
Linear Power Level	Input for safety system power level scram	2	2	0
Log Power Level	Wide range power indication, permissive for initiation of Pulse Mode	1	1	0
Startup Log Count Rate	Wide range power indication, permissive for control element withdrawal	1*	1*	0
Pulsing Power Level	Pulse power level indication	0	0	1

* Required during startup only until the Log Power Level and Linear Power Level channels are on-scale

Bases

Fuel temperature indicated at the control console gives continuous information on the process variable which has a specified safety limit. The exception is required because FLIP fuel is no longer manufactured. If a core has been tested to meet the definition of an operational core the power level scrams provide adequate protection to assure the LCO of fuel temperature is not exceeded.

The power level monitors assure that reactor power level is adequately monitored for all modes of operation.

3.3 REACTOR POOL WATER SYSTEMS

Applicability
This specification applies to the pool containing the reactor and to the cooling of the core by the pool water.

Objective
The objective is to assure that coolant water shall be available to provide adequate cooling of the reactor core and adequate radiation shielding and to prevent damage to in-pool components by corrosion.

Specifications

a. The reactor core shall be cooled by natural convective water flow.

b. The pool water inlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve and siphon breaker to prevent inadvertent draining of the pool.

c. Diffuser and other auxiliary systems pumps shall be located no more than 15 feet below the top of the reactor pool.

(4) All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 feet of water from the pool.

(5) A pool level alarm shall indicate loss of coolant if the pool level drops one foot or less below normal level. .

(6) The reactor shall not be operated if the conductivity of the pool water exceeds 5 micromhos/cm (<0.2 MegOhm-cm) when averaged over a period of one week.

(7) The reactor shall not be operated if the radioactivity of pool water exceeds the limits of 10CFR Part 20 Appendix B Table 3 for radioisotopes with half-lives >24 hours.

Bases

(1) This specification is based on thermal and hydraulic calculations which show that the TRIGA-FLIP core can operate in a safe manner at power levels up to 2,700 kW with natural convection flow of the coolant water. A comparison of operation of the TRIGA-FLIP and standard TRIGA Mark III has shown operation to be safe for

the above power level. Thermal and hydraulic characteristics of mixed cores are essentially the same as that for TRIGA-FLIP and standard cores.

(2) The inlet pipe to the demineralizer is positioned so that a siphon action will drain less than 15 feet of water. The outlet pipe from the demineralizer discharges into a pipe entering the bottom of the pool through a check valve prevents leakage from the pool by reverse flow from pipe ruptures or improper operation of the demineralizer valve manifold. In addition, the pipe is has a loop equipped with a siphon breaker which prevents loss of pool water.

(3) In the event of pipe failure and siphoning of pool water, the pool water level will drop no more than 15 feet from the top of the pool.

(4) Other pipes which enter the pool have siphon breakers which prevent pool drainage. Valves are provided for pneumatic tube system lines and primary cooling system pipe. Other piping installed in the pool has blind flanges permanently installed.

(5) Loss of coolant alarm, after one foot of loss, requires corrective action. This alarm is observed in the reactor control room and outside the reactor building.

(6) The conductivity limit assures that materials within the pool will not be degraded and that the radioactivity of the pool water will be minimized.

(7) Analyses in section 12.2.9 of the Safety Analysis Report show that limiting the activity to this level will not result in any person being exposed to concentrations greater than those permitted by 10CFR Part 20.

3.4 Containment or Confinement

Applicability
These specifications apply to the room housing the reactor and the ventilation system controlling that room.

Objective
The objective is to provide restrictions on release of airborne radioactive materials to the environs.

Specifications
(1) The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2,000 cubic meters.

(2) All air or other gas exhausted from the reactor room and associated experimental facilities shall be released to the environment a minimum of 17 meters above ground level.

Bases
Calculations in Chapter 13 of the Safety Analysis Report show that exposure of occupants of the Laboratory can be kept below 10 CFR part 20 limits for occupational exposure under accident conditions if the room volume is 2,000 m^3. Calculations in Chapter 13 of the SAR based on release of radioactive effluent at ground level show that concentrations of radioactive materials are within limits of 10 CFR Part 20 for non-restricted areas during the accidents considered. Further calculations based on release at the stack height show a further reduction by a factor of 10 due to operation of the ventilation system and release of effluent at a height of 17m.

3.5 Ventilation Systems

Applicability.
This specification applies to the operation of the reactor laboratory ventilation system.

Objective
The objective is to assure that the ventilation system is in operation to mitigate the consequences of the possible release of radioactive materials resulting from reactor operation.

Specification.
The reactor shall not be operated unless the laboratory ventilation system is in operation, except for periods of time not to exceed two days, to permit repairs of the system.

Bases

It is shown in the SAR Chapter 11 that Argon-41 release at zero stack height results in concentrations less than the concentrations permitted for non-restricted areas. Further, the calculations indicate that operation of the ventilation system reduces the concentration to which the public would be exposed by a factor of 10 below this limit. Exposures in the event of a fuel element cladding leak are also calculated based on non-operation of the ventilation system. Therefore, operation of the reactor with the ventilation system shut down in order to make repairs assures the degree of control on which the calculations are based.

3.6 Emergency Power

Emergency power systems are not required for this facility.

3.7 Radiation Monitoring Systems and Effluents

3.7.1 Monitoring Systems

Applicability

This specification applies to the radiation monitoring information which must be available to the reactor operator during reactor operation.

Objective
The objective is to assure that sufficient radiation monitoring information is available to the operator to assure safe operation of the reactor.

Specification
The reactor shall not be operated unless the radiation monitoring channels listed in Table 3.7.1 are operable.

Table 3.7.1 Radiation Monitoring Systems

Radiation Monitoring Channels*	Function	Number
Area Radiation Monitor	Monitor radiation levels within the reactor room	3
Exhaust Gas Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
Exhaust Particulate Radiation Monitor	Monitor radiation levels in the exhaust air stack	1
Environmental Radiation Monitors	TLD dosimeters evaluated on a quarterly basis record exposure in area surrounding the stack	4

*For periods of time for maintenance to the radiation monitoring channels, the intent of this specification will be satisfied if they are replaced with portable gamma sensitive instruments having their own alarms or which shall be kept under visual observation.

Basis
The radiation monitors provide information to operating personnel of any impending or existing danger from radiation so that there will be sufficient time to evacuate the facility and take the necessary steps to prevent the spread of radioactivity to the surroundings. The environmental monitors are placed in areas immediately surrounding the reactor laboratory to record actual dose that would have been delivered to a person continually present in the area.

3.7.2 Effluent (Argon-41) Discharge Limit

Applicability
This specification applies to the concentration of Ar-41 which may be discharged from the facility.

Objective
The objective is to insure that the health and safety of the public are not endangered by the discharge of Ar-41.

Specification
The concentration of Ar-41 in the effluent gas from the facility, as diluted by atmospheric air in the lee of the facility as a result of the turbulent wake effect, shall not exceed 1×10^{-8} µCi/ml averaged over one year.

Basis:
Part 20 CFR Appendix B, Table II specifies a limit of 1×10^{-8} µCi/ml for Ar-41. Chapter 11 and Appendix A of the SAR substantiates a release level of 4.1E-9 µCi/ml for a 3.54 meters/second (lowest monthly average) wind speed if all Ar-41 produced were continuously discharged. The dilution factor by which emitted material is diluted is 4.1E-4 µCi/ml per Ci/second discharged.

3.8 Experiments

Applicability
This specification applies to experiments installed in the reactor and its experimental facilities.

Objective
The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

3.8.1 Reactivity Limits

Specifications

The reactor shall not be operated unless the following conditions governing experiments exist:

(1) The reactivity worth of non-secured experiments shall not exceed 0.7% Δ k/k.

(2) The reactivity worth of any single experiment shall not exceed 1.4% Δ k/k.

Bases:

(1) This specification is intended to provide assurance that the worth of a single unfastened experiment will be limited to a value such that the safety limit will not be exceeded if the positive worth of the experiment were to be suddenly inserted (SAR Chapter 13).

(2) The maximum worth of a single experiment is limited so that its removal from the cold critical reactor will.not result in the reactor achieving a power level high enough to exceed the core temperature safety limit. Since experiments of such worth must be fastened in place, its

removal from the reactor operating at full power would result in a relatively slow power increase such that the reactor protective systems would act to prevent high power levels from being attained. SAR accident analysis includes a sudden addition of 1.4% Δ k/k from firing the transient control rod while operating at the power level scram pqint, a more severe transient than that which could result from removal of a fixed experiment with the same reactivity worth.

3.8.2 Materials

Specifications:

(1) Explosive materials, such as gunpowder, TNT, nitroglycerin, or PETN, in quantities greater than 25 milligrams shall not be irradiated in the reactor or experimental facilities. Explosive materials in quantities less than 25 milligrams may be irradiated provided the pressure produced upon detonation of the explosive has been calculated and/or experimentally demonstrated to be less than the design pressure of the container.

(2) Experiment materials, except fuel materials, which could off-gas, sublime, volatilize, or produce aerosols under (1) normal operating conditions of the experiment or reactor, (2) credible accident conditions in the reactor, or (3) possible accident conditions in the experiment shall be limited in activity such that if 100% of the gaseous activity or radioactive aerosols produced escaped to the reactor room or the atmosphere, the airborne concentration of radioactivity averaged over a year would not exceed the limit of Appendix B of 10 CFR Part 20.

(3) In calculations pursuant to (2) above, the following assumptions shall be used:

(1) 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.

(2) If the effluent from an experimental facility exhausts through a filter installation designed for greater than 99% efficiency for 0.3 micron particles, at least 10% of these vapors can escape.

(3) For materials whose boiling point is above 130°F 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 can escape.

(4) An atmospheric dilution factor of 4.1×10^{-4} for gaseous discharges from the facility.

(4) Each fueled experiment shall be controlled such that the total inventory of iodine isotopes 131 through 135 in the experiment is no greater than 1.5 curies.

Bases:

(1) This specification is intended to prevent damage to reactor components resulting from failure of an experiment involving explosive materials.

(2) Specifications (2) and (3) are intended to reduce the likelihood that airborne activities in excess of the limits of Appendix B of 10 CFR Part 20 will be released to the atmosphere outside the facility boundary of the UWNR.

(3) The dilution factor is based on computations reported in Chapter 11 and Appendix A of the Safety Analysis Report.

(4) The 1.5 curie limitation on iodine 131 through 135 assures that in the event of failure of a fueled experiment leading to total release of the iodine, the exposure dose at the exclusion area boundary will be less than that allowed by 10 CFR Part 20 for an unrestricted area.

3.8.3 Experiment Failure and Malfunctions

Specification:
If a capsule fails and releases material which could damage the reactor fuel or structure by corrosion or other means, removal and physical inspection shall be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken shall be reviewed by the Reactor Director or his designated alternate and determined to be satisfactory before operation of the reactor is resumed.

Bases:
Operation of the reactor with a failed capsule is prohibited to prevent damage to the reactor fuel or structure. Failure of a capsule must be investigated to assure no damage has or will occur.

3.9 Facility Specific LCOs

There are no facility specific LCOs at this facility.

4 Surveillance Requirements

In accordance with section 4.0 of Standard ANSI/ANS-15.1, the following terms for average surveillance intervals shall allow , for operational flexibility only, maximum times between surveillance intervals as indicated below unless otherwise specified within the specification.

Five-year interval not to exceed six years.
Biennial interval not to exceed two and one-half years.
Annual interval not to exceed 15 months.
Semiannual interval not to exceed seven and one-half months.
Quarterly interval not to exceed four months.
Monthly interval not to exceed six weeks.
Weekly interval not to exceed ten days

Daily interval must be done within the calendar day.

Scheduled surveillances, except those specifically required when the reactor is shut down, may be deferred during shutdown periods, but be completed prior to subsequent reactor startup unless operation is required for the performance of the surveillance. Scheduled surveillances which cannot be performed with the reactor operating may be deferred until a planned reactor shutdown. If the reactor is not operational in a particular mode, surveillances required specifically for that mode may be deferred until the reactor becomes operational in that mode.

General

Applicability

This specification applies to the surveillance requirements of any system related to reactor safety.

Objective

The objective is to verify the proper operation of any system related to reactor safety after maintenance or modification of the system.

Specification

Any additions, modifications, or maintenance to the ventilation system, the core and its associated support structure, the pool or its penetrations, the pool coolant system, the rod drive mechanism, or the reactor safety system shall be made and tested in accordance with the specifications to which the systems were originally designed and fabricated or to specifications approved by the Reactor Safety Committee. A system shall not be considered operable until after it is successfully tested.

Bases

This specification relates to changes in reactor systems which could directly affect the safety of the reactor. As long as changes or replacements to these systems continue to meet the original design specifications, then it can be assumed that they meet the presently accepted operating criteria.

4.1 Reactor Core Parameters

Applicability

These specifications apply to the surveillance requirements for measurements, tests, and calibrations of reactor core parameters.

Objective

The objective is to verify the core parameters which are directly related to reactor safety.

Specifications

(1) Excess reactivity
Excess reactivity shall be determined at least annually and after changes in either the core, in-core experiments, or control elements for which the predicted change in reactivity exceeds the absolute value of the specified shutdown margin.

(2) Shutdown margin
The shutdown margin shall be determined at least annually and after changes in either the core, in-core experiments, or control elements.

(3) Pulse limits
The reactor shall be pulsed semiannually to compare fuel temperature measurements (if an operating fuel thermocouple is available) and peak power levels with those of previous pulses of the same reactivity value.

(4) Core configuration
Each planned change in core configuration shall be determined to meet the requirements of Sections 3.1(4) and 5.3 of these specifications before the core is loaded.

(5) Reactivity Coefficients
Power defect and pulsing characteristics shall be measured during startup testing of cores containing different fuel compositions and compared to predictions in the Safety Analysis Report.

(6) Fuel Parameters

a. All fuel elements shall be inspected visually for damage or deterioration annually.

b. Uninstrumented fuel elements which have been resident in the core during the previous year shall be measured for length and sagitta annually. Fuel elements shall not be added to a core unless a measurement of length and sagitta has been completed within the previous fifteen months.

c. Fuel elements in the hottest assumed location, as well as representative elements in each of the rows, shall be measured for possible damage in the event there is indication that the Limiting Safety System Setting may have been exceeded.

Bases

(1),(2) Annual measurements, coupled with measurements made after changes that can affect reactivity values provide adequate assurance that core behavior will be as analyzed. The reactivity values in FLIP fuel change very slowly with fuel burnup.

(3) Semiannual verifications assure no changes in behavior are resulting from fuel characteristic changes.

(4) Checking contemplated core configurations against requirements will prevent inadvertent loading of cores which do not meet power peaking restraints imposed by composition restrictions.

(5) Measurements made during core startup testing are sufficient to assure core behavior will be as analyzed.

(6) Annual inspection of the TRIGA fuel has been shown adequate to assure fuel element integrity through a long history of standard operation.

4.2 Reactor Control and Safety Systems

Applicability
These specifications apply to the surveillance requirements for measurements, tests, and calibrations of the control and safety systems.

Objective
The objective is to verify the performance and operability of those systems and components which are directly related to reactor safety.

Specifications
(1) Reactivity worth of control elements
The reactivity worth of control elements shall be determined upon substantiative changes in core composition or arrangement and annually thereafter.

(2) Control element withdrawal and insertion speeds
Control element drive withdrawal and insertion speeds shall be measured annually and following maintenance to the control element or the control element drive mechanism.

(3) Transient Rod and Associated Mechanism
The transient rod drive cylinder and associated air supply system shall be inspected, cleaned, and lubricated as necessary annually.

(4) Scram times of control and safety elements
The scram time for all scrammable control elements shall be measured annually and following maintenance to the control elements or their drives.

(5) Scram and Power Measuring Channels

a. A channel test of each Reactor Safety System measuring channel in Table 3.2.4 items (1) through (4) and the interlocks in Table 3.2.5 required for the intended modes of operation shall be performed within 24 hours before each day's operation or prior to each operation extending more than one day.

b. A channel test of each item specified in Table 3.2.4 items (5) and (6) shall be performed semi-annually.

(6) Operability Tests
This concern is covered by the General Surveillance criterion at the beginning of this section.

(7) Thermal Power Calibration-Forced Convection
Not applicable to this reactor

(8) Thermal Power Calibration-Natural Convection
A Channel Calibration shall be made of the power level monitoring channels by the calorimetric method upon substantiative changes in core composition or arrangement and annually thereafter.

(9) Control Element Inspection
The control elements shall be visually inspected for deterioration biennially.

Bases

(1) Control element worths change slowly unless the core arrangement is changed, so annual measurement is sufficient to assure safety.

(2) Control element insertion or withdrawal speeds are fixed by the motor design and thus do not change except for extreme binding conditions within the drive.

(3) Transient rod drive and air supply includes filtration and lubrication, so an annual check coupled with pre-startup checks is sufficient to assure operabilty.

(4) Measurement of the scram time on an annual basis is a check not only of the scram system electronics, but also is an indication of the capability of the control rods to perform properly.

(5) The items (1) through (4) in the table are essential safety equipment and thus should be checked frequently, even though no failures have been observed by checkout in nearly 40 years of operation. Frequent testing is unnecessary for item (5), a simple float switch which is very unlikely to fail, and has performed for nearly 40 years without a failure. Testing item (6) , the high voltage monitor scram, results in changing the voltage to the neutron detectors. This introduces step changes into the signal circuits of the measuring channels which can lead to long recovery times and a significant increase in failures of the measuring channels. Further, since the checkout of the linear safety channels is a source check, if high voltage were lost that check would not be possible if the voltage had been lost.

(6) The general requirement for checks of equipment operability after maintenance or modification of systems will reveal any loss of safety functions due to the maintenance or modification.

(8) The power level channel calibration will assure that the reactor will be operated at the proper power levels.

(9) Annual checks in other TRIGA reactors and for nearly 40 years in this reactor have been sufficient to insure no failures due to deterioration.

4.3 Coolant Systems

Applicability

This specification applies to the reactor pool water.

Objective

The objective is to assure the water quality and radioactivity is within the defined limits

Specification

The pool water conductivity and radioactivity shall be measured quarterly.

Bases

Pool water conductivity is continuously monitored, but would be manually monitored on a quarterly basis if the instruments failed.

Radioactivity is indirectly monitored by an area radiation monitor near the demineralizer bed, so gross activity increases would be detected immediately. Experience with TRIGA reactors indicates the earliest detection of fuel clad leaks is usually from airborne activity, rather than pool water activity. The quarterly measurement can identify specific radionuclides.

4.4 Containment or Confinement

No surveillances are required

4.5 Ventilation Systems

Applicability

This specification applies to the building confinement ventilation system.

Objective

The objective is to assure the proper operation of the ventilation system in controlling releases of radioactive material to the uncontrolled environment.

Specification

It shall be verified quarterly and following repair or maintenance that the ventilation system is operable.

Basis

Experience accumulated in over 30 years of operation has demonstrated that the tests of the ventilation system on a quarterly are sufficient to assure the proper operation of the system and control of the release of radioactive material.

4.6 Emergency Electrical Power Systems

Not Applicable. There are no emergency electrical power systems at this facility.

4.7 Radiation Monitoring Systems and Effluents

4.7.1 Radiation Monitoring Systems Applicability

This specification applies to the surveillance requirements for the area radiation monitoring equipment and the stack air monitoring system.

Objective
The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification
The radiation monitoring and stack monitoring systems shall be calibrated annually and shall be verified to be operable by monthly source checks or channel tests .

Basis
Experience has shown that monthly verification of area radiation monitor operability and setpoints in conjunction with the downscale-failure feature of the instrument is adequate to assure operability. Annual calibration is adequate to correct for any variation in the system due to a change of operating characteristics over a long time span. Annual calibrations and monthly source or channel checks of the stack particulate and gaseous monitors, along with the high or low flow alarms associated with the monitor assure operability and accuracy.

4.7.2 Effluents
Applicability
This specification applies to gaseous and liquid discharges from the reactor laboratory.

Objective
The objective is to assure that ALARA and 10CFR Part 20 limits are observed.

Specifications
Liquid radioactive waste discharged to the sewer system shall be sampled for radioactivity to assure levels are below applicable limits before discharge. Results of the measurements shall be recorded and reported in the Annual Report.

The total annual release of gaseous radioactivity to the environment shall be recorded and reported in the Annual Report.

Bases
Liquid waste releases are batch releases, so the liquid can be sampled before release. Air activity discharged is continuously recorded and the integrated release is reported.

4.8 Experiments
No surveillances are required

4.9 Facility-Specific Surveillance
Not applicable. There is no facility-specific surveillance.

5 Design Features

5.1 Site and Facility Description

Specifications

(1) The reactor shall be housed in a closed room designed to restrict leakage. The minimum free volume shall be 2,000 cubic meters.

(2) All air or other gas exhausted from the reactor room and the Beam Port and Thermal Column Ventilation System shall be released to the environment a minimum of 17 meters above ground level.

5.2 Reactor Coolant System

Specifications

(1) The reactor core shall be cooled by natural convective water flow.

(2) The pool water inlet and outlet pipe to the demineralizer shall not extend more than 15 feet into the top of the reactor pool when fuel is in the core. The outlet pipe from the demineralizer shall be equipped with a check valve to prevent inadvertent draining of the pool.

(3) Diffuser and other auxiliary systems pumps shall be located no more than 15 feet below the top of the reactor pool.

(4) All other piping and pneumatic tube systems entering the pool shall have siphon breakers and valves or blind flanges which will prevent draining more than 15 feet of water from the pool.

(5) A pool level alarm shall indicate loss of coolant if the pool level drops approximately one foot below normal level.

5.3 Reactor Core and Fuel

Specifications

(1) TRIGA-FLIP Fuel

The individual unirradiated FLIP fuel elements shall have the following characteristics:

- (1) Uranium content: maximum of 9 Wt-% enriched to nominal 70% Uranium 235.
- (2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.6 H atoms to 1.0 Zr atoms.
- (3) Natural erbium content (homogeneously distributed): nominal 1.5 Wt-%.
- (4) Cladding: 304 stainless steel, nominal 0.020 inch thick.
- (5) Identification: Top pieces of FLIP fuel bundles will have characteristic markings to allow visual identification of FLIP fuel employed in mixed cores.

(2) Standard TRIGA fuel

The individual unirradiated standard TRIGA fuel elements shall have the following characteristics:

(1) Uranium content: maximum of 9.0 Wt-% enriched to a nominal 20% Uranium 235.
(2) Hydrogen-to-zirconium atom ratio (in the ZrH_x): nominal 1.7 H atoms to 1.0 Zr atoms.
(3) Cladding: 304 stainless steel, nominal 0.020 inch thick.

5.2 Reactor Core

Specifications

(1) The core shall be an arrangement of TRIGA uranium-zirconium hydride fuel-moderator bundles positioned in the reactor grid plate.
(2) The Triga core assembly may be standard, FLIP, or a combination thereof (mixed core), provided that any FLIP fuel be comprised of at least thirty-five (35) fuel elements, located in a contiguous, central region.
(3) The reactor shall not be operated with a core lattice position vacant except for positions on the periphery of the core assembly.
(4) The reflector, excluding experiments and experimental facilities, shall be water or a combination of graphite and water.

5.3 Control Elements

Specifications

(1) The safety blades shall be constructed of boral plate and shall have scram capability.

(2) The regulating blade shall be constructed of stainless steel.

(3) The transient rod shall contain borated graphite or boron and its compounds in a solid form as a poison in an aluminum or stainless steel clad. The transient control rod shall have scram capability and may incorporate an aluminum or air follower.

5.4 Fissionable Material Storage

Specifications

(1) All fuel elements shall be stored in a geometrical array where the value of k-effective is less than 0.8 for all conditions of moderation.
(2) Irradiated fuel elements and fueled devices shall be stored in an array which will permit sufficient natural convection cooling by water or air such that the fuel element or fueled device temperature will not exceed design values.

6. Administrative Controls

6.1 Organization

6.1.1 Structure

The reactor facility shall be an integral part of the Engineering Physics Department of the College of Engineering of the University of Wisconsin-Madison. The reactor shall be related to the University structure as shown in SAR Figure 12.1.

The Radiation Safety office performs audit functions for both the Radiation Safety Committee and the Reactor Safety Committee and reports to both committees as well as to the Reactor Director.

6.1.2 Responsibility

The Reactor Director is responsible for all activities at the facility, including licensing, security, emergency preparedness, and maintaining radiation exposures as low as reasonably achievable.

The reactor facility shall be under the direct control of a Reactor Supervisor designated by the Reactor Director. He shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license, procedures, and the requirements of the Radiation Safety Committee and the Reactor Safety Committee.

6.1.3 Staffing

(1) The minimum staffing when the reactor is not secured shall be:

a. A licensed reactor operator in the control room (if senior operator licensed, may also be the person required in (c))
b. A second designated person present at the facility complex able to carry out prescribed written instructions.
c. A designated senior reactor operator shall be readily available at the facility or on call.

(2) A list of reactor facility personnel by name and telephone number shall be readily available in the control room for use by the operator.

(3) A licensed senior reactor operator shall be present at the facility for:

a. Initial startup and approach to power.
b. All fuel handling or control-element relocations.
c. Relocation of any in-core experiment with a reactivity worth greater than 0.7% $\Delta K/K$.
d. Recovery from unplanned or unscheduled shutdown or significant power reduction.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of ANSI/ANS-15.4-1988 Sections 4-6.

6.2 Review and Audit

There shall be a Reactor Safety Committee which shall review and audit reactor operations to assure that the facility is operated in a manner consistent with public safety and within the conditions of the facility license.

6.2.1 Composition and Qualifications

The Committee shall be composed of a least six members, one of whom shall be a Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office. The Committee shall collectively possess expertise in the following disciplines:

1. Reactor Physics - Nuclear Engineering;
2. Mechanical Engineering - Heat transfer and fluid mechanics;
3. Metallurgy
4. Instruments and Control Systems;
5. Chemistry and Radio-chemistry;
6. Radiation Safety.

6.2.2 Charter and Rules

The Committee shall meet at least annually.

The Committee shall formulate written standards regarding the activities of the full committee; minutes, quorum, telephone polls for approvals not requiring a formal meeting, and subcommittees.

6.2.3 Review Function

The responsibilities of the Reactor Safety Committee shall include, but are not limited to, the following:

1. Review and approval of experiments utilizing the reactor facilities;

2. Review and approval of all proposed changes to the facility, procedures, license, and technical specifications;

3. Determination of whether a proposed change, test or experiment would constitute an unreviewed safety question or a change in Technical Specifications;

4. Review of abnormal performance of plant equipment and operating anomalies having safety significance; and

5. Review of unusual or reportable occurrences and incidents which are reportable under 10 CFR Part 20 and 10 CFR Part 50.

6. Review of audit reports.

7. Review of violations of technical specifications, license, or procedures and orders having safety significance.

6.2.4 Audit Function

A Health Physicist from the University of Wisconsin Safety Department Radiation Safety Office shall represent the University Radiation Safety Committee and shall conduct an inspection of the facility at least monthly to assure compliance with the regulations of 10 CFR Part 20. The services and inspection function of the Health Physics Office shall also be available to the Reactor Safety Committee, and will extend the scope of the audit to cover license, technical specification, and procedure adherence.

The committee shall audit operation and operational records of the facility. If the committee chooses to use the staff of the Health Physics organization for the audit function, the reports of audit results will be distributed to the committee and included as an agenda item for committee meetings.

Reactor staff shall perform annual reviews of the requalification program, the security plan, and the emergency plan and its implementing procedures.

6.3 Radiation Safety

The Reactor Laboratory shall meet the requirements of the University Radiation Safety Regulations as submitted for the University Broad License, License Number 48-09843-18 and is subject to the authority of the groups established within that document to insure safety.

The Reactor Director shall have responsibility for maintaining radiation exposures as low as reasonably achievable and for implementation of laboratory procedure for insuring compliance with 10CFR Part 20 regulations

6.4 Procedures

Written operating procedures shall be adequate to assure the safety of operation of the reactor, but shall not preclude the use of independent judgement and action should the situation require such. Operating procedures shall be in effect for the following items:

(1) Testing and calibration of reactor operating instrumentation and controls, control rod drives, area radiation monitors, and air particulate monitors;

(2) Reactor startup, operation, and shutdown;

(3) Emergency and abnormal conditions, including provisions for evacuation, reentry, recovery, and medical support;

(4) Fuel element and experiment loading or unloading;

(5) Control rod removal or replacement;

(6). Routine maintenance of the control rod drives and reactor safety and interlock systems or other routine maintenance that could have an effect on reactor safety;

(7) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms and abnormal reactivity changes; and

(8) Civil disturbances on or near the facility site.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Safety Committee. Temporary changes to the procedures that do not change their original intent may be made by the Senior Operator in Control or his designated alternate. All such temporary changes shall be documented and subsequently reviewed by the Reactor Safety Committee.

6.5 Experiment Review and Approval

(1) Routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval.

(2) Prior to performing any experiment which is not a routine experiment, the proposed experiment shall be evaluated by the senior operator responsible for operation. He shall consider the experiment in terms of its effect on reactor operation and the possibility and consequences of its failure, including where significant, consideration of chemical reactions, physical integrity, design life, proper cooling, interaction with core components, and reactivity effects.

(3) Modified routine experiments may be performed at the discretion of the senior operator responsible for operation without the necessity of further review or approval provided that the evaluation performed in accordance with Section 6.5(2) results in a determination that the hazards associated with the modified routine experiment are

neither greater nor significantly different than those involved with the corresponding routine experiment which shall be referenced.

(4) No special experiment shall be performed until the proposed experiment has been reviewed and approved by the Reactor Safety Committee.

(5) Favorable evaluation of an experiment shall conclude that failure of the experiment will not lead directly to damage of reactor fuel or interference with movement of a control element.

6.6 Required Actions

6.6.1 Action to be Taken in Case of Safety Limit Violation

In the event a safety limit is exceeded:

(1) The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.
(2) An immediate report of the occurrence shall be made to the Chairman, Reactor Safety Committee, and reports shall be made to the NRC in accordance with Section 6.7 of these specifications, and
(3) A report shall be prepared which shall include an analysis of the causes and extent of possible resultant damage, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence. This report shall be submitted to the Reactor Safety Committee (RSC) for review and then submitted to the NRC when authorization is sought to resume operation of the reactor.

6.6.2 Action to be Taken in the Event of an Occurrence of the Type Identified in 6.7.2(1)b., and 6.7.2(1)c.

In the event of an reportable occurrence (1.3.1) the following actions shall be taken:

(1) The reactor shall be shut down.
(2) The Director or his designated alternate shall be notified and corrective action taken with respect to the operations involved,
(3) The Director or his designated alternate shall notify the Chairman of the Reactor Safety Committee,
(4) A report shall be made to the Reactor Safety Committee which shall include an analysis of the cause of the occurrence, efficacy of corrective action, and recommendations for measures to prevent or reduce the probability of recurrence, and
(5) A report shall be made to the NRC in accordance with Section 6.7.2 of these specifications.

6.7 Reports

6.7.1 Operating Reports

(1) An annual report covering the activities of the reactor facility during the previous calendar year shall be submitted (in writing to U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, DC 20555) within six months following the end of each calendar year, providing the following information:

a. A brief narrative summary of (1) operating experience (including experiments performed), (2) changes in facility design, performance characteristics, and operating procedures related to reactor safety and occurring during the reporting period, and (3) results of surveillance tests and inspections;

b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality;

c. The number of emergency shutdowns and inadvertent scrams, including reasons therefor;

d. Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safety of the operation of the reactor and the reasons for any corrective maintenance required;

e. A brief description, including a summary of the safety evaluations of changes in the facility or in the procedures and of tests and experiments carried pursuant to Section 50.59 of 10 CFR Part 50;

f. A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge;

Liquid effluents (summarized on a monthly basis)

(1) Liquid radioactivity discharged during the reporting period tabulated as follows:

(a) Total estimated radioactivity released (in curies).

(b) The isotopic composition if greater than 1×10^{-7} microcuries/cc for fission and activation products.

(c) Total radioactivity (in curies), released by nuclide, during the reporting period based on representative isotopic analysis.

(d) Average concentration at point of release (in microcuries/cc) during the reporting period and the fraction of the applicable limit in 10CFR20.

(2) Total volume (in gallons) of effluent water (including diluent) during periods of release.

Gaseous Waste (summarized on a monthly basis)

(1) Radioactivity discharged during the reporting period (in curies) for:

(a) Gases.

(b) Particulates with half lives greater than eight days.

4. The estimated activity (in curies) discharged during the reporting period, by nuclide, for all gases and particulates based on representative isotopic analysis and the fraction of the applicable 10CFR 20 limits for these values.

Solid Waste

(1) The total amount of solid waste packaged (in cubic feet).

(2) The total activity involved (in curies).

(3) The dates of shipment and disposition (if shipped off site).

g. A summary of radiation exposures received by facility personnel and visitors, including dates and time of significant exposures and a summary of the results of radiation and contamination surveys performed within the facility; and

h. A description of any environmental surveys performed outside the facility.

(2) A report within 60 days after completion of startup testing of the reactor (in writing to the U.S. Nuclear Regulatory Commission, Attn: Document Control Desk, Washington, D.C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level describing the measured values of the operating conditions or characteristics of the reactor under the new conditions including:

a. An evaluation of facility performance to date in comparison with design predictions and specifications, and

b. A reassessment of the safety analysis submitted with the license application in light of measured operating characteristics when such measurements indicate that there may be substantial variance from prior analysis.

6.7.2 Special Reports

(1) There shall be a report of any of the following not later than the following day by telephone or similar conveyance to the NRC Headquarters Operation Center, and followed by a written report describing the circumstances of the event and sent within 14 days to U.S. Nuclear Regulatory commission, Attn: Document Control Desk, Washington, D.C. 20555:

(a) Any accidental release of radioactivity above permissible limits in unrestricted areas whether or not the release resulted in property damage, personal injury, or exposure;
(b) Any violation of a safety limit; and
(c) Any reportable occurrences as defined in Section 1.3.1 of these specifications.

(2) A written report within 30 days in writing to the U.S. Nuclear Regulatory commission, Attn: Document Control Desk, Washington, D.C. 20555 of:

a. Permanent changes in facility organization at Reactor Director or Department Chair level.
b. Any significant change in the transient or accident analysis as described in the Safety Analysis Report;

6.8 Records

6.8.1 Records to be Retained for a Period of at least Five Years or for the Life of the Component Involved if Less than Five Years

(1) Normal reactor facility operation (but not including supporting documents such as checklists, log sheets, etc. which shall be maintained for a period of at least one year),
(2) Principal maintenance activities,
(3) Reportable occurrences,
(4) Surveillance activities required by the Technical Specifications,
(5) Reactor facility radiation and contamination surveys where required by applicable regulations,
(6) Experiments performed with the reactor,
(7) Fuel inventories, receipts, and shipments,
(8) Approved changes in operating procedures,
(9) Records of meeting and audit reports of the review and audit group.

6.8.2 Records to be Retained for at Least One Cycle

Operator qualification and re-qualification records.

6.8.3 Records to be Retained for the Lifetime of the Reactor Facility

Annual reports which contain the information in items (1) and (2) may be used as records for those items.

(1) Gaseous and liquid radioactive effluents released to the environs,
(2) Offsite environmental monitoring surveys required by technical specifications,
(3) Radiation exposures for all personnel monitored,
(4) Updated, corrected, and as-built drawings of the facility.

7 References

1. GA-9064, pages 3-1 to 3-23

2. GA-9064, pages 3-1 to 3-23

3. NE Memo No. 4, Report on Refueling the University of Wisconsin Nuclear Reactor, R. J. Cashwell, March 1968, University of Wisconsin Department of Nuclear Engineering

4. Core Test Program, UWNR Mixed TRIGA-FLIP Core (9 FLIP), R. J. Cashwell, July 1974, University of Wisconsin Department of Nuclear Engineering

5. Core Test Propgram, UWNR Mixed TRIGA-FLIP Core (15 FLIP), R. J. Cashwell, February 1978, University of Wisconsin Department of Nuclear Engineering

6. Core Test Program, All FLIP Core, R. J. Cashwell, January 1980, University of Wisconsin Department of Nuclear Engineering

7. "Sagitta" refers to the bow of the element and means the maximum excursion of the clad surface from a chord connecting the two ends of the clad surface.

8. Simnad and West, 1986

9. NUREG-1282

15 FINANCIAL QUALIFICATIONS

15.1 Financial Ability to Construct a Non-Power Reactor

Not applicable for renewal application.

15.2 Financial Ability to Operate a Non-Power Reactor

The Reactor Laboratory is a part of the Engineering Physics Department of the College of Engineering at the University of Wisconsin-Madison. The teaching mission of the laboratory takes precedence over the research and service missions; for this reason the primary fiscal support for operation of the facility is from the state-funded university budget. Reactor personnel have instructional duties, such as teaching courses, setting up laboratory experiments, and assuring that equipment used in the teaching laboratories is operable. It thus becomes somewhat difficult to allocate funding precisely to just the operation of the reactor. In the following information no attempt is made to separate the instructional component of the budget from the reactor operations part of the budget. For instance, the Reactor Director is teaching two courses during the Spring 2000 semester, but his entire salary is included in the budget below.

Recently the operating budget of the Reactor Laboratory has been considerably higher than what would be predicted from the past history of the facility. This is due in large part to providing replacement of long-term employees who are approaching retirement age with younger workers in time to allow adequate training of the new employees. The total salary expenditures will be reduced significantly after the retirements actually take place, but the estimate below uses the current funding level.

The total operating budget of the laboratory is $388,000. Of this total, state instructional funding covers $299,000, with the remaining expenses split between grants ($48,000) and income generated by reactor services ($41,000).

By expense category, the funds are spent for salary($253,000), fringe benefits ($77,300), supplies and expense ($24,000), and capital equipment ($33,000).

These numbers do not include the infrastructure provided by the University such as electrical power, heating, janitor service, and health physics coverage. Further, the fringe benefits included above are not specifically billed to the department in which the employee works for instructional funding, but come from a campus-wide fund, while the fringe benefits for salaries supported by non-instructional funding are charged to the fund paying the salary.

The instructional funding is appropriated by the state. The administration of the university has been very supportive of the reactor facility and continuation of the Nuclear Engineering curriculum. The fact that the application for renewal of the license is signed by the administration indicates this support.

Much of the capital equipment funding in recent years has come from the DOE program to update the instrumentation and experimental equipment on non-power reactors. In addition, a local utility company has provided matching grants to support instruction in traditional "fission nuclear engineering", and this has contributed to both the supply and expense and capital equipment budget. The combination of funding opportunities has resulted in the reactor and associated laboratories being in excellent condition. The reactor can continue to operate without the outside grant income, but it would not allow for upgrading the equipment as has been done for the last 10 years. However, the present instrumentation and control systems are capable of continuing to operate for another 20 years with no loss of function.

Funds from services provided to persons outside the university have been steady for the last 10 years, with some increases in the last three years. It is expected that this funding will continue for at least the next five years.

15.3 Financial Ability to Decommission the Facility

By letter dated July 19, 1990[1], the University responded to 10CFR Part 50.75 showing the expected cost of decommissioning the reactor. At that time estimated decommissioning cost was $1,200,000 in the year 2000.

Early in 1999 the computations on which the funding plan was made were updated to extend the time of decommissioning to 2020 and to incorporate more recent figures on the cost of disposal of radioactive debris from the decommissioning. The estimate is again based on placing the facility in condition for unrestricted release three years after cessation of operations, now estimated as June 30, 2020. The result of this revised estimate is a decommissioning cost of between $3,000,000 and $8,000,000, with the wide range of cost based primarily upon the uncertainty in disposal cost. Whereas the disposal cost was a minor part of the total decommissioning effort costs for the original computation, it is now by far the largest component of the total cost. Nevertheless, as a state agency, the funding plan remains to obtain the funding when necessary.

15.4 Reference

1. Letter to USNRC Document Control Desk from R. J. Cashwell under Docket 50.156 dated July 19,1990, with attachment signed on behalf of the Board of Regents.

16 OTHER LICENSE CONSIDERATIONS

16.1 Prior Use of Reactor Components

There are no components in use at the University of Wisconsin Nuclear Reactor Laboratory that have had prior use at any other facility or organization. It is conceivable that prior use components could be integrated into Reactor Laboratory systems at some future time. Appropriate analysis and reviews of component replacement will be conducted in accordance to applicable standards, regulations and facility procedures and licensed technical specifications.

16.2 Medical Use of Non-Power Reactors

The University of Wisconsin Nuclear Reactor Laboratory is not engaged nor licensed to conduct any activities for medical use of the facility. Future medical use of the Reactor Laboratory would be conducted pursuant to appropriate license applications and approvals as authorized by the Atomic Energy Act of 1954 as amended.

Appendix A Calculation Methods for Atmospheric Release of Radioactivity

A. Models Used for Calculations in Section 13.1/2.1

For Sutton's diffusion model, the maximum concentration (X_{max}) at any point downwind is given as:

$$(1)\ X_{max} = \frac{2Q}{e\pi\bar{\mu}h^2} \qquad \text{(Reference 1)}$$

where $\bar{\mu}$ is mean wind speed in meters/second
Q is release rate in Ci/second
h is stack height in meters

For the generalized Gaussian Plume Model, the maximum concentration is given by the same equation (Reference 2, equation 8).

For calculations in this report, the following values are used:
$\bar{\mu}$ = lowest monthly average =3.54 meters/second
h = stack height above ground =17.1 meters.
Using these numbers, equation (1) reduces to

$$(2)\ X_{max} = 2.26\text{E-}4\ Q,$$

where X_{max} is in µCi/ml

Reference 2 presents a method applicable to release from buildings with zero stack height to approximate release from leaks in a containment structure. The relation given, as equation 4, is:

$$(3)\ X = \frac{Q}{(\pi\sigma_y\sigma_z + CA)\mu},$$

where X, Q, and μ are defined as above,
C is an empirical constant with a value between 0.5 to 2, and
A is the minimum building cross section.
The σ terms are concerned with atmospheric dispersion, which will be neglected in this analysis, which will result in the equation;

$$(4)\ X = \frac{Q}{CA\mu}$$

Inserting values for the UWNR facility used in the safety analysis for FLIP fuel conversion, and using a value of 1 for C yields:

(5) $$X = \frac{Q}{(1)(30ft)(44ft)(9.29E\text{-}2m^2/ft^2)(3.54m/\text{sec})}, \text{ or}$$

(6) X=2.3E-3 Q with X in units of μCi/ml

a. Whole Body Exposure

The activity concentration of the insoluble volatiles in the reactor room air was determined by dividing released activity by room volume.

$$\frac{A}{V} = \frac{5.89E6\ \mu Ci}{2.00E9\ cm^3} = 2.95E{-3}\ \mu Ci/cm^3$$

Since 3.7E4 dps = 1 μCi, A/V= 109 γ/sec-cm^3

The maximum dose rate is calculated by assuming the room is equivalent to a hemisphere with a radius of 782 cm. In addition, the average gamma energy is 0.7 MeV, the attenuation coefficient for air is 3.5E-5 cm^{-1}, and the flux-to-dose conversion factor is 4.2E4 γ/cm^2/cm^2-mr/min

Using the relationship

$$DR = \frac{30S(1 - \exp(-R\Sigma))}{C\Sigma} \quad \text{where}$$

DR = dose rate in mr/hr
S = Volumetric source strength in γ/sec-cm^2
R = outer radius of hemisphere
Σ = attenuation coefficient for air,

yields a dose rate of 60 mr/hour.

(b) Dose to the Lungs

The dose to the lungs was calculated by first assuming uniform dispersal of the released volatiles in the laboratory volume, giving a concentration of

$$\frac{A}{V} = \frac{5.54E6\ \mu Ci}{2.00E9\ cm^3} = 2.75E{-3}\ \mu Ci/cm^3.$$

Since the "standard man" breathes 1.25 cubic meters of air per active hour, he would breathe approximately 0.21 cubic meters in 10 minutes, the assumed evacuation time. If this number is increased to 0.30 cubic meters to allow for excitement and stress, then his lungs would be exposed to an activity of

$$\frac{A}{V} V = (.75E\text{-}3\ \mu C)\ (3E5\ cm^3) = 825\ \mu Ci.$$

The dose to the lungs is then calculated from the following expression to be 1 Rad.

$$Dose\ (Rad) = \frac{ACR}{m} \sum_{I=1}^{8} \frac{F_i\ E_i}{\lambda_i}(1 - e^{\lambda_i t}), \text{ where}$$

A = Activity exposure (825 μCi)
C = Conversion factors

$$\frac{(3.7E4\ \beta/\sec\text{-}\mu Ci)(1.6E\text{-}6\ erg/MeV)}{100 erg/gm\text{-}rad}$$

R = lung retention factor (0.125 is customary)
m = mass of lungs (1000 grams)
F_i = fraction of total activity
E_i = energy of beta for nuclide i (MeV)
λ_i = radioactive decay constant + biological release constant (6.7E-8 sec^{-1})
t = time of exposure (assumed infinite)

Release rate, Q, for an isotope is the total quantity released to air (column E of Table 13.1, Chapter 13) divided by the assumed release time. The release time used in further calculations is the time for the ventilation system (room air and beam port and thermal column exhaust systems) to make a complete change of air in the Reactor Laboratory.

$$(9) \qquad T_{release} = \frac{2000\ m^3}{0.472\ m^3/\sec} = 4238\ seconds$$

Using the generalized Gaussian Plume Model (equation (2)), and demonstrating with data for Br-83, concentrations released to unrestricted areas (Chapter 13, Table 1 Column H) are calculated as shown below:

$$(10) \qquad X_{Br\text{-}83} = \frac{(0.0083 Ci)(2.26E\text{-}4)}{4238} = 4.43E\text{-}10\ \mu Ci/ml$$

The remaining isotope values are calculated in the same manner.

The activity release was also evaluated through use of equation (6). This calculation would be applicable to release of the activity through the building walls with the ventilation system not operating.

Again using Br-83 as an example, and assuming the same release time as in the previous calculation

$$(11) \quad X_{Br-83} = \frac{(0.0083Ci)(2.3E-3)}{4238} = 4.50E-9 \ \mu Ci/ml$$

This value is a factor of 10.17 greater than that evaluated by the Gaussian Plume Model. All similar values in Chapter 13, Table 1, Columns H and I may be multiplied by this factor for a more conservative case. This calculation was done for the previous Safety Analysis Report using as the building dimensions only the minimum dimensions of the reactor laboratory, a room within the Mechanical Engineering Building. This considerably underestimates the "wake effect" of the actual building. Using values appropriate to the Mechanical Engineering Building reduces this 10-fold increase to approximately 1.

B. Sample Calculations Supporting Section 11.1.1.1

The maximum release rate for Ar-41 activity is 13.3 μCi/second, and the resulting concentration is calculated to be, from equation (2)

$$(7) \ X_{max} = (13.3E\text{-}6)(2.26E\text{-}4) = 3.01E\text{-}9 \ \mu Ci/ml.$$

If calculated using equation (6) (for building wake dilution) using only the building dimension of the Reactor Laboratory room, the resulting value is

$$(8) \ X = (13.3E\text{-}6)(2.3E\text{-}3) = 3.06E\text{-}8 \ \mu Ci/ml.$$

If calculated using equation (6) using the dimensions of the building, rather than just the room dimensions, the resulting value is

$$(9) \ X = (13.3E\text{-}6)(3.1E\text{-}4) = 4.13E\text{-}9 \ \mu Ci/ml$$

It is obvious that all three methods used to calculate the above values cannot both be applicable. Since the reactor is not operated when the ventilation system is not in operation, the value in equation (8) is more realistic, and agrees well with equation (6) when the entire building wake is considered.

References

1 **Meteorology and Atomic Energy**, U. S. Dept of Commerce Weather Bureau, Govt. Printing Office, Washington, DC July 1955

3. F. A. Gifford, Jr, Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model, **Nuclear Safety**, December 1960

4. **Calculation of Distance Factors for Power and Test Reactors**, (TID-14844), USAEC, March 23, 1962

Appendix B Supporting Documents

Task Order No. 2 Under Master Task Agreement No. C96-175937

Page 1

LMITCO FORM
PROC-1811b
07/99

TASK ORDER NO. 2 UNDER MASTER TASK AGREEMENT NO. C96-175937
LOCKHEED MARTIN IDAHO TECHNOLOGIES COMPANY (LMITCO)
2525 Fremont Avenue
P. O. Box 1625, Idaho Falls, ID 83415-3521
OPERATING UNDER U. S. GOVERNMENT CONTRACT NO. DE-AC07-94ID13223

To: University of Wisconsin-Madison
Research Administration
750 University Avenue
Madison, WI 53706-1490

Effective Date: August 26, 1999
Completion Date: November 1, 2001

To: Tom Handland
PI: R. J. Cashwell

This Task Order No. 2 is awarded to:

1. Transfer Reactor Fuel Assistance Subcontract No. C87-101251-002 to the new Master Task Agreement No. C96-175937 as Task Order No. 2.
2. Extend the period of performance to November 1, 2001. This extension is retroactive to November 1, 1998.
3. Confirm the Statement of Work and modification thereto remain unchanged.
4. Assignment: On September 30, 1999, LMITCO's prime contract with DOE will expire. Thus, pursuant to the article in the General Provisions entitled "assignment"" this Task Order is assigned to Bechtel BWXT Idaho, LLC, under its DOE Prime Contract No. DE-AC07-ID13727, effective October 1, 1999.

Procurement Agent: Lynda Keller	Telephone: (208) 526-5597	Cost: $0.00
Ship via: N/A	F.O.B./Trans.: N/A	Cash Terms: Net 0 Days

Billing Address:
Lynda Keller
LMITCO
P. O. Box 1625
Idaho Falls, ID 83415-3521

Signed: Lynda T. Keller 9/14/99
Lockheed Martin Idaho Technologies Company Date

Title: Procurement Agent

Signed: 10/10/99
Subcontractor's Official Date
William J. Vance, Assistant Dean
Title: Research & Sponsored Programs

Return one signed copy of this Task Order to Lynda Keller.

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