UNIVERSITY OF CALIFORNIA-IRVINE
NUCLEAR REACTOR FACILITY
LICENSE NO. R-116
DOCKET NO. 50-326

LICENSE RENEWAL APPLICATION

SAFETY ANALYSIS REPORT,
AND
TECHNICAL SPECIFICATIONS

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

University of California, IRVINE
Nuclear Reactor Facility

License R-116
Docket 50-326

SAFETY ANALYSIS REPORT
REVISION 1
OCTOBER 1999

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1. INTRODUCTION AND GENERAL CONSIDERATIONS

1.1 Introduction

This SAFETY ANALYSIS REPORT (SAR) supports an application to the United States Nuclear Regulatory Commission (NRC) by the University of California at Irvine (UCI) for renewal of a Class 104c license, number R-116, of a TRIGA® Mark I pulsing reactor. The reactor is owned by the Regents of the University of California and operated by the Department of Chemistry at UCI. The reactor was designed and built by Gulf General Atomic, Inc., was installed in 1969 in [redacted] of Rowland Hall on the main UCI campus and is known as the UCI Nuclear Reactor Facility (UCINRF). This SAR is an extensive revision and reformatting of that submitted in 1968 prior to any operation. For example, where possible, projections for values of parameters have been replaced by actual measurements. For this reason, line revisions have not been identified. Subsequent revisions will be specifically marked.

The reactor was designed to be safely operated in steady-state mode at power levels up to at least 1 megawatt and to be pulsed repetitively to yield a burst having a prompt energy release of about 16 Mw-sec, a peak power of about 1,200,000 kW and a pulse width at half maximum of about 11 milliseconds. The power levels attainable depend on fuel load, control rod configurations, and operational parameter limitations. The present license request is for 250 kilowatt operation. The reactor configuration, selected for safe installation in a multi-purpose building on an urban campus, has a fixed core installed in [redacted] pool tank with no beam ports.

The safety of the TRIGA® reactor lies in the large prompt negative temperature coefficient that is an inherent characteristic of the uranium-zirconium hydride fuel-moderator material. Thus, even when large, sudden, reactivity insertions are made and the reactor power rises on short period, the excess reactivity is compensated automatically because the fuel temperature rises simultaneously so that the system returns quickly to a normal power level before any heat is transferred to the cooling water.

The inherent prompt shutdown mechanism of TRIGA® reactors was demonstrated extensively during more than 10,000 transient tests conducted at prototype TRIGA® reactors in Gulf General Atomic's laboratory at San Diego. These tests involved step reactivity insertions up to 3.5% $\delta k/k$. This demonstrated safety has permitted the location of TRIGA® reactors in urban areas in buildings without pressure-type containment. Currently, many other TRIGA® fueled reactors are in operation and several are equipped for pulsing operation. The inherent safe performance of the fuel and core design has been demonstrated time and again, including the 30 years of successful operation at this facility.

UCINRF uses only one type of fuel element: "standard" TRIGA® stainless steel clad fuel with less than 20% enriched uranium. Each element contains approximatel[redacted] of total uranium (less tha[redacted] of U-235). The fuel is a solid homogeneous mixture of uranium (8.5% by weight) and zirconium hydride (H/Zr ratio of 1.7). Safety is related to the large prompt negative temperature coefficient of about -1.3×10^{-4} $\delta k/k$ - ^0C. This type of fuel has received extensive testing without incident in TRIGA® reactors around the world at considerably higher power and pulsing levels than those used or contemplated at UCINRF.

The reactor facility includes a separate stand-alone facility for gamma irradiation, radioassay equipment, and laboratories for preparing samples and handling irradiated samples. UCINRF is used primarily by the Department of Chemistry in its programs of research and teaching in the field of radioanalytical chemistry. Other users from both on and off campus departments and organizations mostly carry out analytical determinations using variations on the sensitive neutron activation analysis technique.

1.2 Summary and Conclusions on Principal Safety Considerations

The UCINRF TRIGA® Mark I reactor design makes safe and adequate provisions for normal operating considerations such as heat disposal during full power operation and radiation dose rates, including nitrogen-16 and argon-41 generation. It does this by providing adequate depth of pool water, appropriate room ventilation rates, pool stirring, and minimal areas of air exposed to high neutron flux. Laboratory facilities, and a fume hood, together with fixed radiation monitoring stations are provided for irradiated sample handling from several experimental facilities that are included. The reactor room and the laboratories form a confinement volume, with normal reduced pressure operation to assist in control of airborne radioactivity. 24-hour surveillance of this area is provided by both area and particulate radiation monitors in order to alert staff to any mishap in the facility.

Technical specification limitations are imposed on core excess fuel and samples inserted into core to limit the possibility of inadvertent reactivity introduction to well below the capacity of the core fuel to recover without failure. Fuel temperature is continuously monitored during operation and a scram level is set well below agreed fuel temperature safety limits to preclude fuel damage.

In addition, possible accidents considered in the body of this report include the maximum hypothetical accident (MHA) of loss of pool water with simultaneous fuel element cladding failure, release of radioactivity from experiment failure, and loss of pool water.

The Maximum Hypothetical Accident for UCINRF is considered to be the failure of a fuel element cladding, in air. Estimates indicate that, even with conservative assumptions, potential radiation exposures are well within acceptable limits for on-site and off-site personnel. Warning systems and emergency procedures are in place to mitigate such effects.

Experiment failure possibilities are addressed by using operation limitations on experiments as well as adequate shielding and ventilation. An automatic ventilation shut-down is provided to reduce the potential for release of airborne radioactivity beyond the boundaries of the facility. Emergency procedures call for building evacuation should limits on static radiation levels be exceeded.

Loss of Pool Water can result only from rupture of the tank holding the cooling and shielding water, a circumstance considered highly unlikely. If it were to occur, high radiation levels would eventually result in the building, but only at least five hours after sounding of a low water level alarm. Calculations indicate that the fuel element cladding will remain intact and thus no fission products will be released so that no injuries to operating personnel or danger to the surrounding population would result.

Inadvertent Reactivity Insertions are limited by limiting the reactor loading to a maximum of 2.1% δk/k ($3.0) excess reactivity above a cold, critical and compact condition without xenon. The TRIGA® reactor system is unusual in that rapid insertion of this amount of reactivity in a very short time is routine for many existing TRIGA® reactors using this fuel with no safety consequences.

1.3 Facility History of Operations.

1.3.1 Licensing History

Special Nuclear Material License SNM-1143 issued August 5th 1969. (Docket 70-1202)

Construction Permit CPRR-107 issued May 5th, 1969; Notice filed 4/15/69. (Docket 50-326)

Amendment 1 to Construction Permit: 9/10/69 to permit fuel follower control rod and instrumented element installation.

Operating License R-116 (Docket 50-326) issued November 24th 1969; Notice filed 10/22/69.

Operating License Amendments Issued:

1. 1/7/74 increase U-235 possession limit fro[redacted] t[redacted].
2. 6/2/82 revisions to Security Plan.
3. 5/25/94 revision of surveillance requirements in Technical Specifications.
4. 1/22/99 revision of surveillance requirements in Technical Specifications.

Tech Spec Changes Approved (recent changes have been classified as license amendments (see above):

1. 6/30/70 AEC initiated - added surveillance of transient rod drives.
2. 12/17/70 pool water level surveillance when measurement channel out of service.

1.3.2 Operating History

The reactor was first taken critical on November 25th 1969. Since that date until June 30th 1999, the following have been achieved:

		Annual Average (approximate)
Daily startups and shutdowns	5139	171
Hours critical	7480:19	250
Megawatt hours	1,307.4	44 (translates to 176 full power equivalent hours)
Pulses	978	none since 1993

1.4 Facility Modifications.

Very few physical modifications have been made to this facility since initial installation in 1968. The following is a brief list of physical changes.

1. Water filled central thimble replaced with central dry tube. This necessitated adding an additional fuel element to restore reactivity excess, and flux peaking in the core center was reduced.. This was for greater experimental convenience in sample irradiation.
2. Two additional pneumatic transfer systems were added, one cadmium lined in an E-ring fuel element position, one in an F-ring fuel element position.
3. A separate pool water purification pump, with appropriate piping modifications, was added to run 24 hours a day, so that the coolant pump could be reserved for reactor operations only.
4. A new coolant water flow valve and controller was installed to update control (manual to electronic) and replace an aging valve.
5. New radiation monitoring equipment was installed to directly replace earlier models no longer supported.. This practice is ongoing with further updates about to be installed
6. Acquisition of additional previously used fuel and a new fuel element, a fuel follower control rod, and instrumented elements as backups to the original complement.
7. Recently (1998/99), some basic refurbishment was carried out in which facility flooring (containing asbestos) was removed and replaced with vinyl sheet flooring, walls were repainted, wall deterioration was repaired, and laboratory cabinetry damaged by water spills was repaired.

2. SITE CHARACTERISTICS

2. 1 Location (Figs. 2-1, 2-2, and 2-3)

The Irvine Campus of the University of California is located in Orange County, some 40 miles south-east of the center of Los Angeles and is about 4 miles from the Pacific Coast. Situated on slopes at the northern extremity of the San Joaquin Hills, the campus overlooks the Santa Ana basin to the north and the inland end of Newport Bay to the west; higher land of the Hills extends south and east to just over 1000 feet above sea level The campus has an average elevation of about 100 feet.

The 1500 acres of land owned by the Regents of the University is within the boundaries of the extensive Irvine Ranch, controlled by the Irvine Company. By agreement between the Irvine Company and the Regents, a joint Master Development Plan has provided for the development of the surrounding area in close conjunction with the University. A Research Park is the latest co-development, under construction in 1999. Since its incorporation, in 1971, the city of Irvine encompassess most of the immediately surrounding community.

Much of the anticipated development of UCI and the surrounding community from rural into urban/suburban, projected in 1968, has been accomplished. A new growth cycle for the campus and the surrounding business community is underway and will continue for the next 15-20 years.

2.2 Population

2.2.1 Community Population

Orange County is a densely populated urban area with 2.6 million residents in 32 cities and a further 0.2 million in unincorporated areas (1999 California Dept of Finance). UCI is situated about in the middle (N, S, and E directions) and the borders in those directions are about 30 km (19 miles). Thus 2.8 milion residents are within a 30 km zone. A 10 km zone encompasses most of the cities of Newport Beach (to the south west), Costa Mesa to the north west, and Irvine (the remainder). A small part of Santa Ana would be included. The populations of these cities are listed in Table 2.1 together with their growth over the last 10 years. It is difficult to project furture growth in an area mostly, but not quite, "built out", but a conservative estimate would be to anticipate growth over the next 20 years to equal that over the last 10. The inclusion of the whole of Santa Ana most of which lies outside the 10 km zone is very conservative. These projections are made in the Table 2.1.

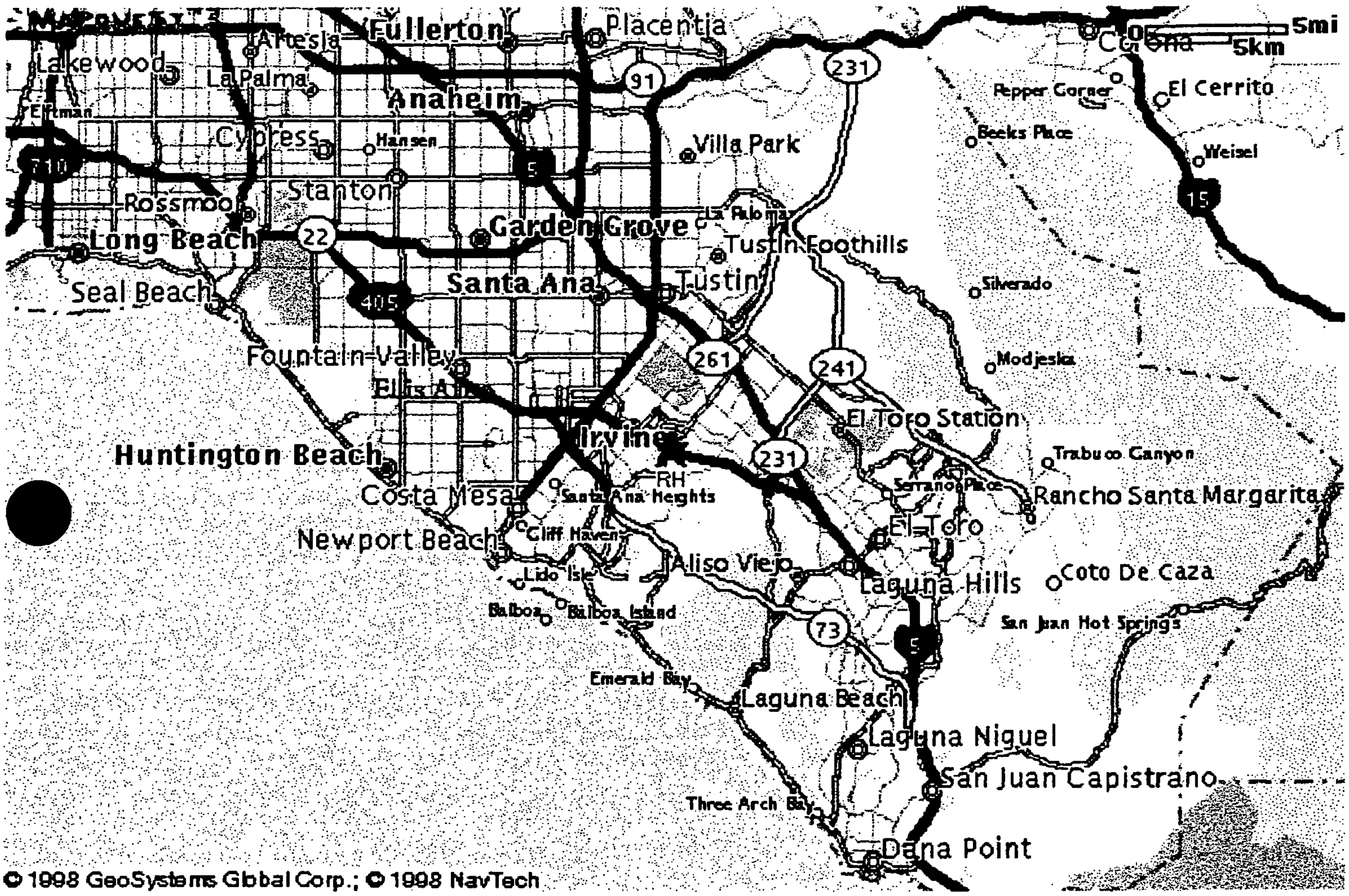


Fig 2.1 Location of Orange County and City of Irvine

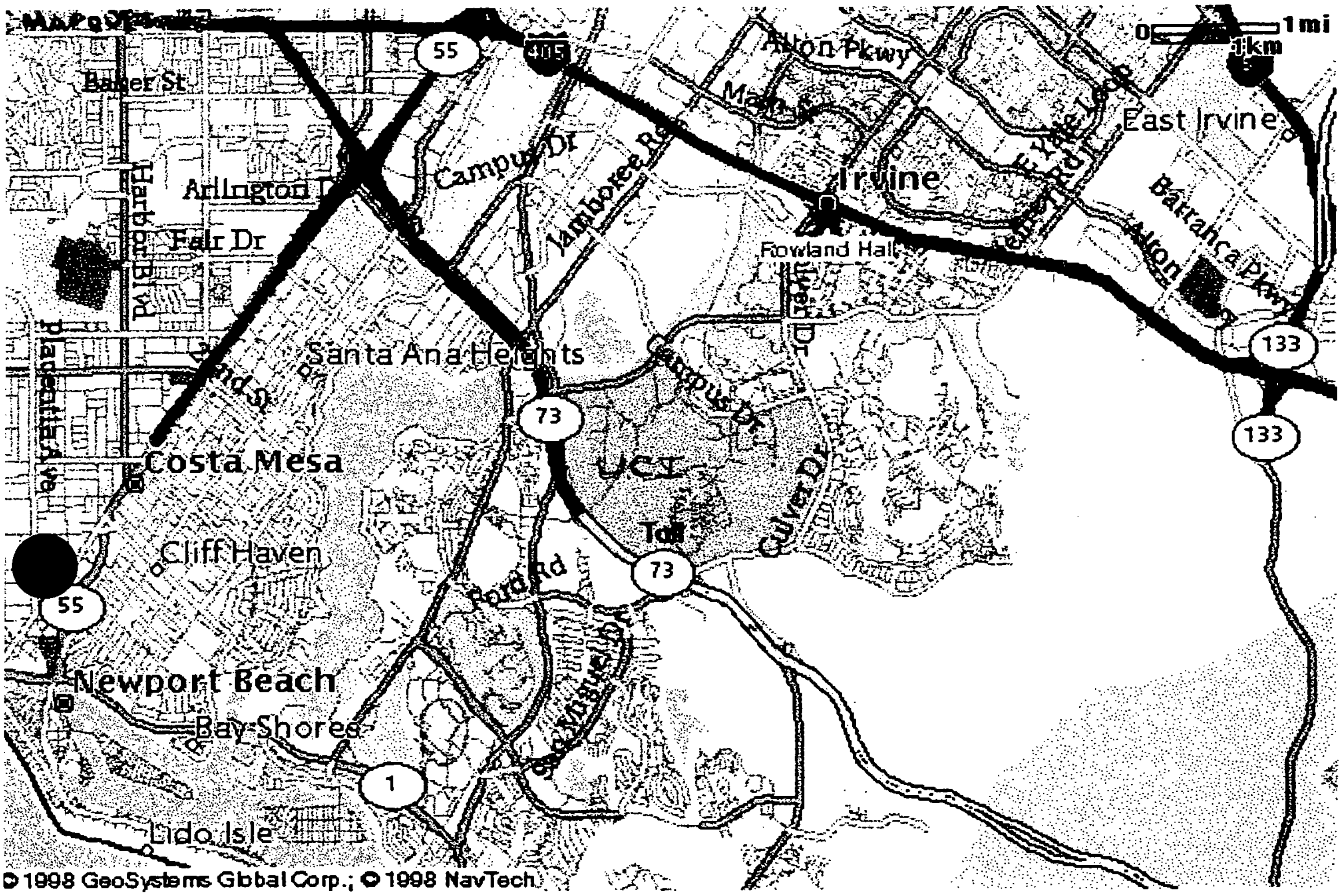


Fig 2.2 Location of UCI Campus

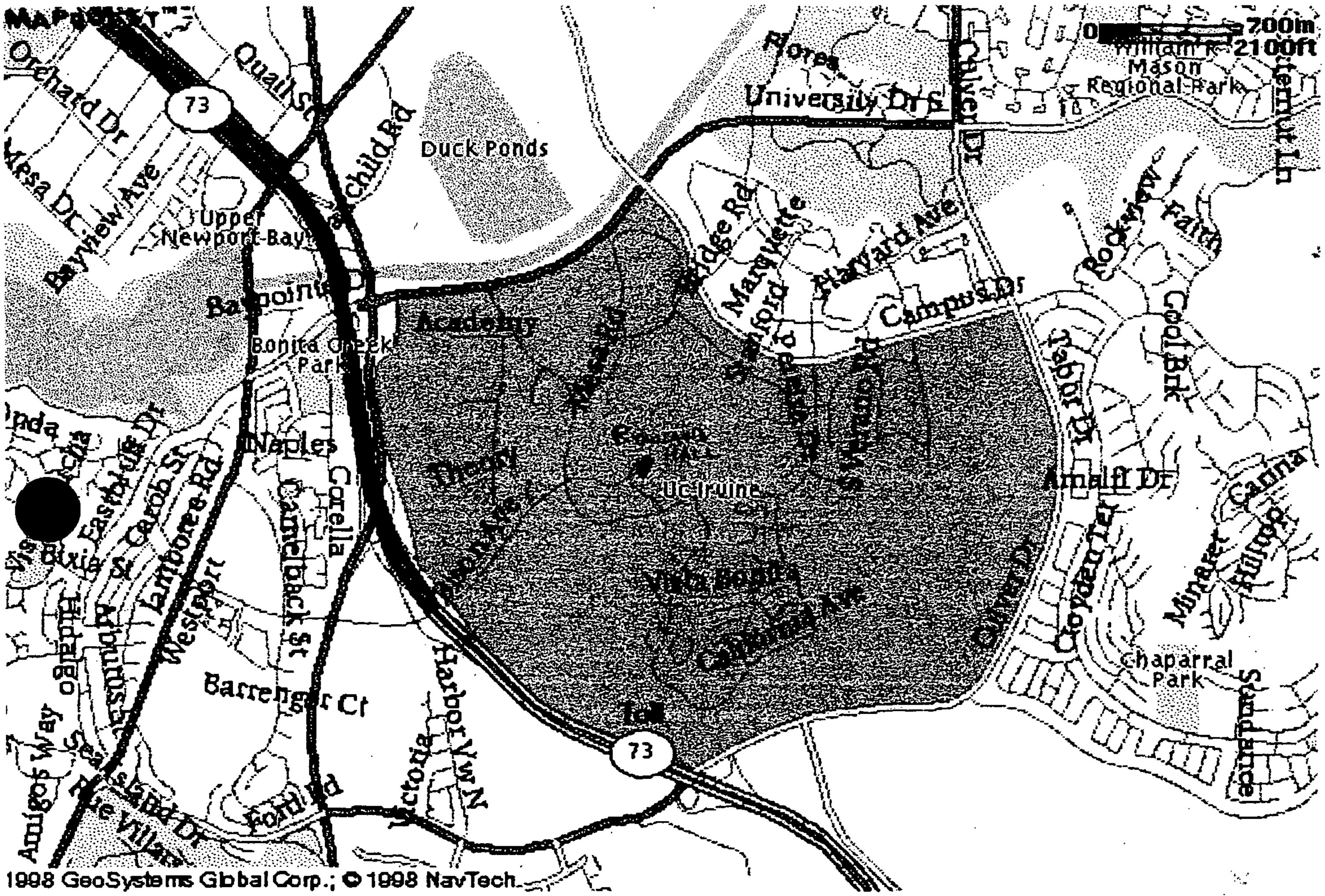


Fig 2-3 Location of Rowland Hall (Bison Rd)

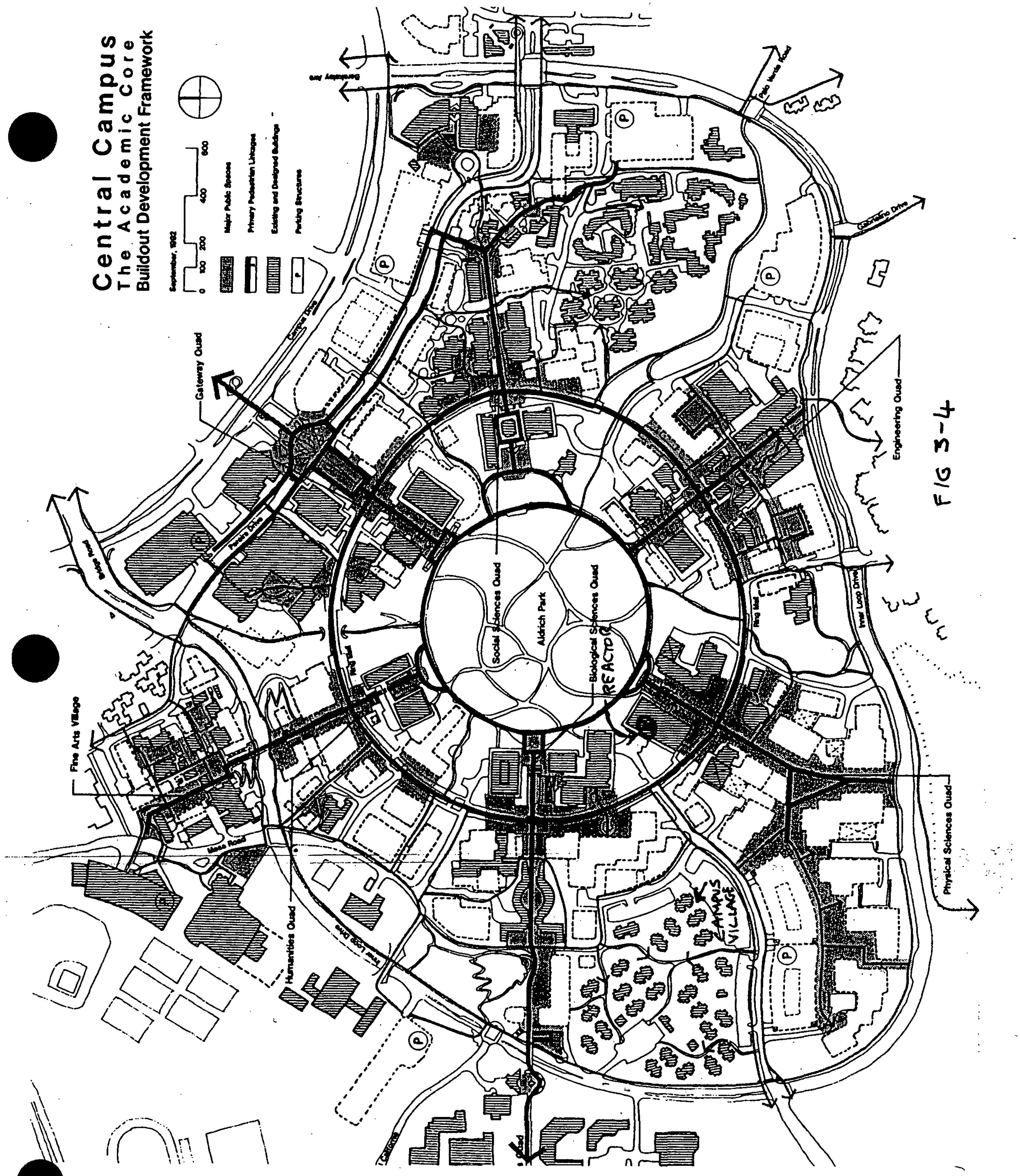

Central Campus
The Academic Core
Buildout Development Framework
September, 1992
0 100 200 400 600
Major Public Spaces
Primary Pedestrian Linkages
Existing and Designed Buildings
Parking Structures
Gateway Quad
Campus Drive
Bridge Road
Fine Arts Village
Mesa Road
Humanities Quad
Inner Loop Drive
Ring Mall
Social Sciences Quad
Aldrich Park
Biological Sciences Quad
REACTOR
Engineering Quad
Physical Sciences Quad
CAMPUS VILLAGE
Berkeley Ave
Palo Verde Road
Gabrielino Drive

FIG 3-4

Table 2.1 Estimated Populations of Cities Within 10 km of UCI

City	Population, 1999	Growth, % in 10 years	Projection, 2020
Newport Beach	74,000	11	82,000
Costa Mesa	105,600	10	116,000
Irvine	136,600	24	170,000
Santa Ana	315,000	7	337,000
TOTAL	631,200		705,000

Upcoming census data will permit revision of these figures.

2.2.2 Campus Population

Residential

At present (1998 data), approximately 5,600 students reside within the University boundary. In addition, 684 faculty and staff live on University Hills. Projections have been made for the year 2005 which anticipates housing 10,400 students and 1100 faculty on the main campus.

The closest complex to the reactor is Campus Village with a mean residential distance about 1000 feet to the west and the closest dwelling unit about 500 feet to the west (see Fig. 2-4). Campus village houses 800 undergraduates in 200 apartments.

Daytime Occupation

UCI Campus has approximately 2500 faculty and staff on the main campus, with a student population of about 18,000. By 2005 these numbers are expected to rise to 4600 and 25,000 respectively. The maximum campus expected population would be the sum of these figures: currently 20,500 rising to 30,000 by 2005. It is unclear what goal is served by trying to estimate this information further. There will be a large number of indiviuals most of the time within a 1 km radius of the reactor facility.

2.3 Proximity of Industrial and Military Installations

TABLE 2-2

DISTANCES AND DIRECTIONS OF NEARBY INSTALLATIONS

Reservoirs	Elevation (ft)	Distance (miles)	Direction
Bonita	120	1.0	S
San Joaquin	470	1.8	S

Big Canyon	290	2.4	SSW
Sand Canyon	163	2.6	E
Airports			
Orange County Airport		2.3	NW

OC Airport (SNA) had 3.5 million emplanments in 1995. This is expected to rise to 5.9 million by 2005.

The fate of El Toro Marine Base, a distance of 7 miles away, currently under conversion to civilian use is unclear. There is a plan to creat a major intenational airport on that site. This is receiving a lot of political opposition. At the moment flight operations by the military have ceased both at El Toro and at the closer Santa Ana Marine Corps Air Facility (now closed), so there is less chance than before of an air incident near UCI. None of the proposed landing or take-off flight paths from SNA or El Toro fly over UCI. There are no plans to use the santa Ana facility for flight operations at this time.

Industry

A large number of small and medium sized corporations have facilities or headquarters in the surrounding area. A joint development by UCI and the Irvine company is occurring on the edge of campus, less than 1 km from the facility. The Food and Drug Administration is constructing a major laboratory research facility adjacent to the campus. Such development will continue to occur. Most of this business is "light industry" and does not pose a special hazard to facillity operations, nor the facility to them. Much of the activity is very similar to that occurring in UCI research facialities of rowland Hall and adjacent buildings.

2.3 SITE HAZARD ANALYSIS

A hazard tool recently available for assessment is provided by PROJECT IMPACT an ESRI/FEMA project to provide hazard assessment for a number of potential climatic and geological hazards. This is located at http://www.esri.com/hazards/ which can be searched to produce a hazard area map for hazards due to:

Flood; Hail Storms; Hurricane; Wind Storms and Earthquakes (Recent and Historic)

Sesimic Hazard Analysis is also available for this site at

http://www.consrv.ca.gov/dmg/shezp/maps/m_tus5.htm

Copies of these are included at the end of this section. They do not identify any special hazard from any of these for the site at Rowland Hall, or the immediate area.

2.3 CLIMATOLOGY

Extremes of climate are rare in Southern California Coastal regions. Historical data show that frost is infrequent (average 1-2 nights per year); snow, sleet and hail are almost unknown; rainfall is concentrated in winter storms and the yearly total varies considerably between low ranges.

The most important climatic feature from the safety aspect are the frequent occurrence of temperature inversions. These phenomena have been the subject of a special report by the U.S. Weather Bureau[1] and much of the following discussion is based on its findings. A strong diurnal pattern pertains for much of the year with moderate winds form the south west (on shore) during the day and from the east-north east at night (off shore). Relatively calm periods occur twice a day during each transition, whne low temperature inversions can form. This pattern is interspersed with strong "Santa Ana" winds from the north-east, mostly in the winter season. Clearly, the effect of dispersion from an incident at UCI would be highly dependent on the time of year and time of day at which it happened. the polciy therefore is to use highly conservative atmospheric conditions for general assessment purposes, and attempt to perform detailed predictions if and when an incident occurs.

2.4 GEOLOGY

2.4.1 General

The Campus is situated in the south-east corner of the Los Angeles Basin in the Peninsular Ranges Province[2] In this region, the basement rocks are overlain with Cenozoic strata in northwest to west-northwest trending fault zones. Unconformities are frequent in late miocene and early pliocene because of considerable tectonic unrest during deposition.

2.4.2 Surface Features

Exposure details reflect the considerable unconformity noted above (Fig. 2-6[3]). The northwest section of the University property is overlain with varied marine terrace deposits; to the south and east the exposure is of the Topanga formation of Middle Miocene Age. The central portion is distinguished as the Paularino Member consisting of a sequence of interbedded sandstone and breccia.[2]

2.4.3. Reactor Site--Test Borings

Information on the reactor location is available from test borings made for Rowland Hall building,[4] the reactor tank site[5] and for utility tunnel extensions.[6] The borings confirm predictions made from the surface map. At the reactor tank site as shown in Fig. 2-7, quaternary sand extends to a depth of only five feet and overlies the Paularino formation. The report details

"the sand is firm and cemented; the underlying shale andsandstone are very firm to hard

[1] U.S. Weather Bureau, Technical Paper No. 54. 'Meterological Summaries Pertinent to Atmospheric Transport and Dispersion Over Southern California', Department of Commerce, Washington, D.C. 1965.

[2] Geology of the Los Angeles Basin--an Introduction, Geological Survey Professional Paper 420-A, R F Yerkes, T.H. McCulloh, J.E. Schoellhamer and J.G. Vedder

[3] Oil and Gas Investigations--Geoiogical Map of San Joaquin Hills--San Juan Capistrano Area, Orange County J G. Vedder, et. al..

[4] Leroy Crandall and Associates~ Los Angeles Job 64562

[5] Idem, Job No. 64562-B

[6] Geolabs, Inc., Van Nuys and Santa Ana. W.O. i718-AP-67

and highly cemented."

The boring extended some 10 feet below the level of the base of the reactor tank No ground water was encountered in this or any other drilling at the site.

2.5 HYDROLOGY

Most of the year is dry at this site. Drainage in the rain season is to the north west by natural canyons which are being left on the site.Rowland Hall is located on a ridge and this, together with the porous nature of the subsurface rocks means that neither groundwater nor flooding is likely to be any problem for the reactor foundation. Hazard maps show this location as not subject to 100 year flood potential. Recent news suggests that the surrpnding areas previously identified as subject to 100 year flooding will shortly be reclassifed as unaffected owing to improvements in drainage systems in the County.

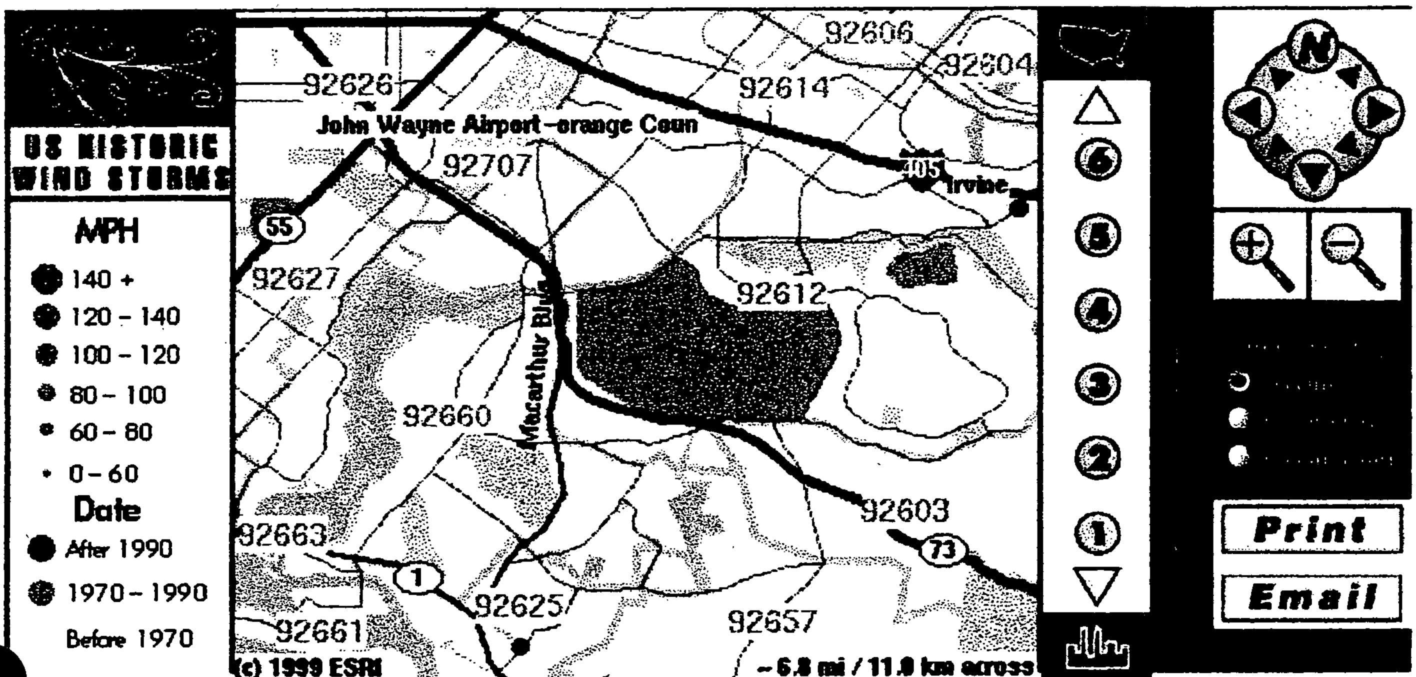
2.6 SEISMOLOGY

The Los Angeles Basin Area of Southern California has a history of considerable seismic activity and many fault zones exist. Traditionally the UCI region was considered relatively stable. Official hazard maps drawn recently (see maps at the end of this section) reflect this and show low hazard. However, new information is being reviewed (Prof. Lisa Grant, private communication) which may question this conclusion. Unfortunately it will not be released/published in time for this submission. For this reason, we plan to submit a COMPLETELY REVISED SEISMIC HAZARD ANALYSIS FOR THIS REACTOR SITE DURING THE NEXT FEW MONTHS. For this reason the remainder of this section is presently deliberately blank.

SITE CHARACTERISTICS ADDENDUM

Hazard Impact Maps (FEMA/ESRI)

PROJECT IMPACT
ESRI and FEMA
PROJECT IMPACT
Hazard Information and Awareness



Map Notes: The historic wind storm data displayed on this site was collected by the National Weather Service. The maps indicate the relative intensity of the historic wind storms as measured by miles per hour (MPH). For more information on the online hazard maps, please refer to the Frequently Asked Questions page.

Current Map View:

Historic Wind Storms

Change View

Make New Map

Return to Home Page

PROJECT IMPACT

ESRI and FEMA

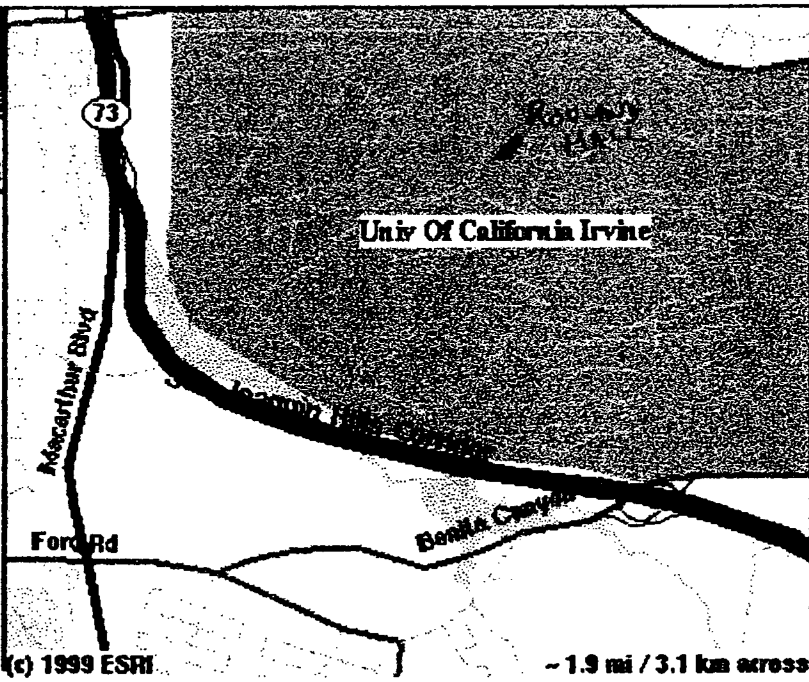
PROJECT IMPACT

Hazard Information and Awareness

US FLOOD HAZARD AREAS

Flood Data

- 100 - Yr Flood
- 500 - Yr Flood
- Water Bodies
- No Data



Print

Email

Map Notes: The **FEMA Digital Q3 Flood Data** displayed on this Web site is developed by scanning the existing Flood Insurance Rate Map (FIRM) hardcopy and capturing a thematic overlay of flood risks. Digital Q3 Flood Data files contain only certain features from the FIRM hardcopy in effect at the time of scanning and do not replace the existing FIRM hardcopy maps. The Q3 Flood Data is being displayed here with basemap data from the GDT Dynamap/2000 data set. The Q3 Flood Data is currently available for approximately **1,200 counties** across the United States.

Current Map View:

Flood Hazard Areas

Change View

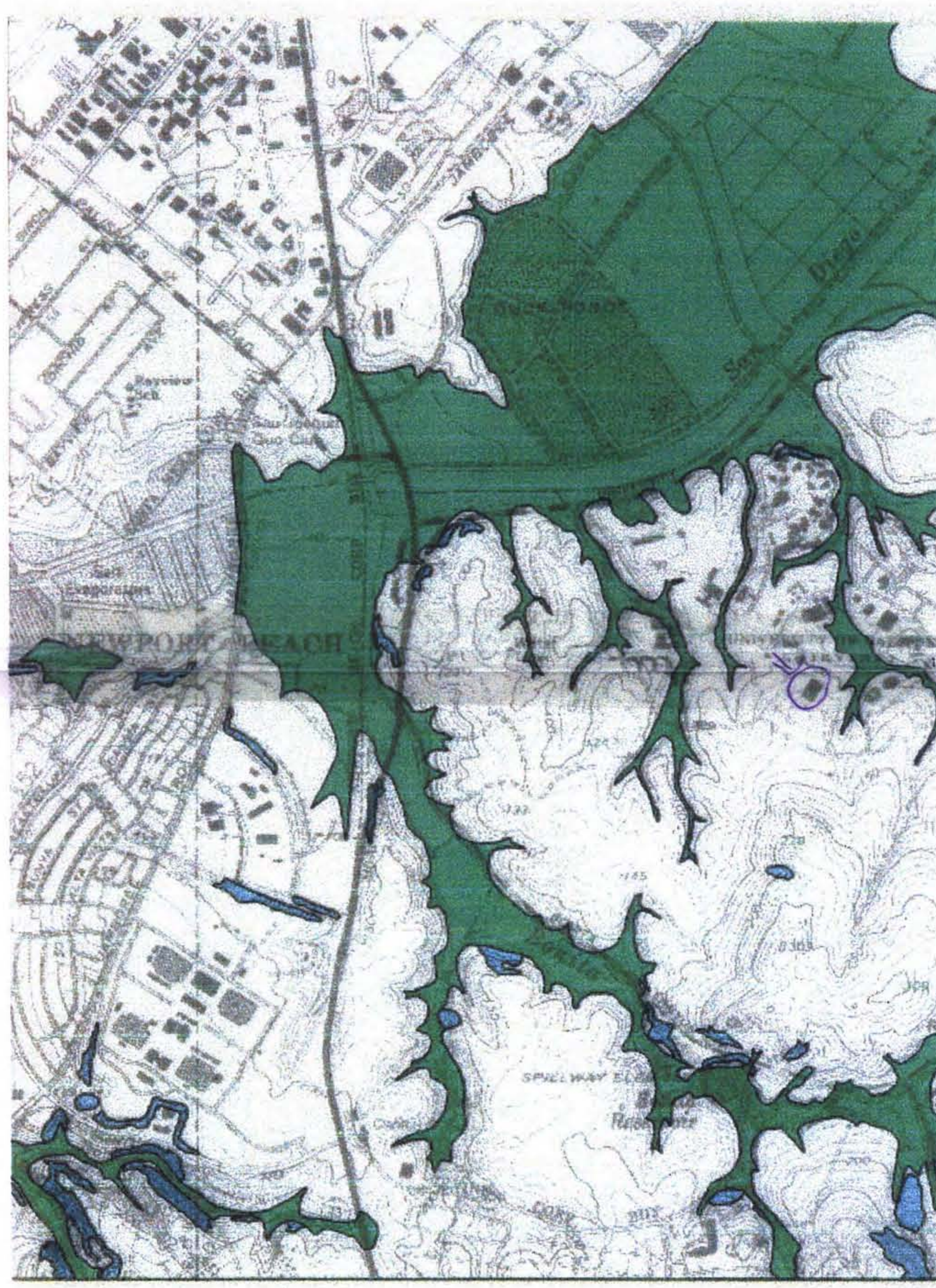
The maps displayed on this site should be considered an **advisory tool** for general hazard awareness, education, and flood plain management. The flood hazard maps displayed on this site are **not the legal document** to be used when making a single site flood hazard determination. For more information on these maps, please refer to the **Frequently Asked Questions** page.

Make New Map

Return to Home Page

monitor make it unlikely that the image you see below is original scale. NOTE: For the full legend, please see the main SHZ Map page for the Tustin Quadrangle.

Wrong map? If your site of interest is on an adjacent map, simply click on the adjacent arrow and that map (if available) will load.



ase map prepared by U.S. Geological Survey, 1965, photorevised 1981.
Graphic layout of original zone map by J.E. Arthur; web conversion and layout by Ted Smith, Barbara Wanish, and Beatriz Crowl.

REFERENCES USED TO COMPILE THIS MAP
Tustin Quadrangle

Seismic hazard evaluation of the Tustin 7.5-minute quadrangle, Los Angeles County, California: California Department of Conservation, Division of Mines and Geology Open-File Report 97-20.

APERTURE CARD

Also Available on Aperture Card

9910290290-01

MAP EXPLANATION
Zones of Required Investigation:

Liquefaction

Areas where historic occurrence of liquefaction, or local geological, geotechnical and groundwater conditions indicate a potential for permanent ground displacements such that mitigation as defined in Public Resources Code Section 2693(c) would be required.

Earthquake-Induced Landslides

Areas where previous occurrence of landslide movement, or local topographic, geological, geotechnical and subsurface water conditions indicate a potential for permanent ground displacements such that mitigation as defined in Public Resources Code Section 2693(c) would be required.

PURPOSE OF MAP

This map will assist cities and counties in fulfilling their responsibilities for protecting the public safety from the

3. DESIGN OF ROWLAND HALL AND REACTOR FACILITY

3.1 GENERAL

In this section, the general features of facility construction are described, with special focus on the features designed to accommodate the presence of the reactor.

The UCI TRIGA® reactor core is located in a tank [redacted] of Rowland Hall (RH)which was built in 1968-69, and renamed in 1998. The location of the building at UCI has been identified in Section 2.

The building is a general purposes building with research and teaching laboratories, lecture halls, classrooms, offices and workshops. Currently, occupancy is shared among Chemistry, Earth Systems Science, Mathematics and Physics Departments. [redacted]

In addition to the lower service level, RH has five floors and a penthouse (fan rooms) level. Floor plans are included in Appendix A together with a cross section drawing in Appendix B.

The building was designed with a structural steel frame with reinforced concrete floors acting as diaphragms in distributing loads to vertically resisting elements. Space frame calculations were performed by Bole and Wilson, Licensed Structural Engineers, Long Beach, to ensure compliance with Uniform Building Code Requirements (1964) for Zone 3 for resistance to seismic forces. Recently these criteria have been re-examined in the light of recent experiences with steel frame buildings, following guidance in FEMA 267 and 267A, developed following the 1994 Northridge earthquake some 70 miles distant. As a result, thRowland Hall has been declared POOR whereas prior ratings would have been GOOD or FAIR in terms of potential for life safety. A draft plan for seismic upgrade is provide in Appendix A. No specifics relate to the reactor design, but to the general lieklihood of general building damage in the Penthouse and upper floors. Funding for the plan has not yet been allocated, but is being pursued.

3.2 FIRST FLOOR RH

The area immediately above the reactor tank [redacted] is research laboratories and corridor space (Appendix A). Long residence by any person in such areas is extremely unlikely; however, in order to allow for possible future operation of the reactor at 1,500 kw, [redacted] At 250 kw the dose rate is less than 0.05 mr/hr and therefore, there is no danger from gamma radiation to persons in these first floor areas.

3.3 SERVICE FLOOR

The service floor (Appendix A) houses a machine workshop, mechanical rooms and specialized laboratories [redacted]
The other laboratories are used primarily for heavy equipment, and vibration sensitive instruments. One area is designed as a high pressure cell with heavy steel lining to the interior walls and an easily 'blown' wall to the exterior.

3.4 REACTOR FACILITY AREA

[redacted]

Floor covering in the area is sheet vinyl to minimize cracks which could trap a radioactive spill.

3.4.1 Reactor Room-[redacted]

The reactor is situated near the bottom of a water-filled pit located as shown in Fig. 3-1. The pit is lined with aluminum, 1/4 inch thick on the sides and 1/2 inch thick on the bottom. The liner and the concrete pit is constructed to the design and specifications used by Gulf General Atomic for their Mark III TRIGA installation at San Diego (see Section 4.1). The design is considered to be more than adequate to endure earthquake induced ground motions. An extensive leak-testing procedure coupled with weld radiography at 20% ot the welds was performed to ensure integrity of the aluminum liner. The leak tests were repeated with the liner in place on its foundation to ensure against damage during shipping and handling. The tank was double wrapped with hot tarred felt to provide a water barrier to the surrounding concrete.

A strong aluminum railing surrounds the edge of the pool as only the reactor end is completely covered by gratings.

[redacted]) below a 10, 000 pound capacity monorail hoist. The rail runs from the large access door to the loading dock, over the reactor pit. Operation is locally controlled from a pendant box. The hoist can handle a modest sized fuel cask if used fuel needs to be removed from the pool. It was used to add "previously irradiated" fuel to the pool in 1974. So far no fuel has ever been removed from the pool at this facility.

[redacted]

The facility houses a Cs^{137} source self-contained gamma irradiator. This is of the self-contained, fully shielded type and is certified for operation in an uncontrolled area. No extra precautions are necessary because this is placed [redacted].

3.4.2 Control Room-[redacted]

The control room contains the reactor control and has good visual access to the reactor. Area monitor and security monitor instrumentation is located adjacent to the reactor control console. Portable monitoring equipment is kept here--the only access to the reactor area is through the control room except under special circumstances when the doors to the loading dock is opened, in which case personnel control is exercised. These doors are not opened while the reactor is operating. The door between the reactor room and the control room has rubber seals and is provided with a closer so that it will normally seal the confinement area.

3.4.3 "Wet" Laboratory-[redacted]

A stainless steel fume hood with exhaust rate of about 500 cfm is installed in the laboratory. The vent, in common with all other ventilation from the building, exhausts vertically at the penthouse level above the roof. The send/receive station for the reactor pneumatic sample transfer system is located as shown (Fig. 3-2). The exhaust from this is led into the main reactor room exhaust system. It is not intended that moderate or high levels of radioactivity shall be processed without special provision for shielding and waste disposal. The reactor administration will require that due provision be made before authorization of production of such quantities of activity.

3.4.4 "Dry" Laboratory Room-[redacted]

No special facilities are provided in this room. It is used for instrumental experimental set-ups and preparation of samples prior to irradiation.

3.5 AIR CONFINEMENT

As mentioned above, the reactor room, laboratory and counting room is designed to act as a confinement area for contaminated air. All penetrations of walls, ceilings and floors

by pipes and ducts are specially sealed. The large doors to the loading dock and the door to the control room have weather seals. The ventilation system is designed to maintain a negative air pressure of at least 0.15 in of water in this area both with respect to the outside and to the rest of the building.

3.6 VENTILATION SYSTEM

3.6.1 Normal Operation

The ventilation system for this area is shown in Fig. 3-3. A schematic of the control system is given in Fig. 3-5. In normal operation, fresh air is drawn into the building at the service floor level and split to form a hot and cold plenum which is distributed around the building. A local thermostat controls mixing dampers at the entrance to the area. Temperatures and static pressures in the main supply ducts are maintained by automatic controls. The complete system is under the control of the Central Plant which is manned 24 hours a day.

In the reactor area system, in addition to the mixing control dampers, an additional static pressure control and several emergency isolation dampers are provided. The first is designed to regulate the supply air to the reactor room so that the negative pressure requirement is met; the second operates in emergency status as described below.

An exhaust system operates on a 24 hour basis from the reactor room. Air from the laboratory and counting room is also exhausted by this route via louvres in the doors to these rooms. The exhaust rate is approximately 4300 cfm vertically at the penthouse level (Fig. 3-6). No regulating damper is included in this system, but a shut-off damper (see below) is incorporated.

3.6.2 Emergency operation

In the unlikely event of an accidental release of particulate or gaseous radioactivity into the reactor room air this should be exhausted slowly to the atmosphere to allow dilution and filtered to remove particulates. Thus an emergency purge exhaust is installed, actuated by a signal from a continuous air particulate monitor (described below(3.7.1)).

An independent fan can exhaust air from the reactor room at a rate of about 250 cfm through a Magnamedia Berryllium-collector style filter when actuated by a continuous air monitor signal (vide infra). At the same time, isolation dampers (Fig. 3-5) close the air supply to the reactor room, laboratory and counting room and the regular reactor room exhaust system and the exhaust fans for the latter and the fume hood are switched off. The pneumatic transfer system blower is also switched off, and audio and visual alarms indicate this condition is actuated.

In the emergency condition, air inlet is only the natural leakage rate of the area which should be less than 250 cfm exhaust capacity so that negative pressure is maintained.

3.7 RADIATION MONITORING SYSTEM

3.7.1 Continuous Air Monitor--Operation of Emergency System

Experience with an incident at the University of Illinois TRIGA reactor and more recently at reed College, indicate that in the case of leakage of fission products from a fuel element, the release is rapidly detected by a continuous air monitor operating alongside the reactor. This procedure was thus selected for UCINRF. A continuous air monitor is located adjacent to the reactor bridge and draws air through a tube from beneath the reactor bridge over the water in the pool. The unit provides audio-visual alert and alarm levels. This unit is designed to be operative at all times and a periodic testing procedure is a part of the written operating instructions for the reactor. The unit also responds to overall radiation (gamma) levels in the facility, and so provides additional back-up information to the facility monitoring equipment, sending a high level alarm signal through the security system. A high alarm actuates the emergency exhaust mode (see Fig 3-5 and section 3.6.2 above). This unit is supplied with electric power through circuits associated with the RH Emergency Power system (diesel generator) to provide continuous service even in emergencies.

3.7.2 Radiation Area Monitor System

A multi-station area monitoring system is installed with control and readout adjacent to the reactor control console. Each station operates independently for failure back-up and has "failure", "alert" and "alarm" level settings and indicators. As well as audio and visual alarms, high level alarms are also sent to the central security monitoring station. The current system provides four general and two specific location sensors:

General:
1) Ceiling immediately over reactor core
2) Ceiling over fuel storage pit area
3) East wall of reactor room below window to corridor
4) South wall of laboratory [redacted]

Specific:
5) Rotating Specimen Rack and Central Thimble unload ports
6) Pneumatic Transfer System Unload Terminus

The object of these locations is to maintain a check at the most likely positions of release of radiation to uncontrolled areas. Local warning and readout is provided in addition to central control at locations 3) and 4). This unit is supplied through circuits associated with the RH Emergency Power system (diesel generator) to provide continuous service even in emergencies.

4. REACTOR DESCRIPTION

4.1 General Design Features.

The reactor core and reflector assembly is located below approximately 20 feet of water in an open pool (Fig. 4-1). The pool consists of an aluminum tank [redacted] wide by [redacted] long and [redacted] deep (Fig. 4-2), supported by a reinforced concrete shield. At the core end, [redacted] of concrete surround the tank and [redacted] in of concrete are below the tank to reduce soil activation (Fig. 4-3).

UCINRF reactor utilizes stainless steel clad fuel-moderator elements, developed by Gulf General Atomic, consisting of a homogeneous mixture of uranium enriched to 20% in U^{235}, with zirconium hydride, approximately $Zr_1H_{1.7}$. The unique feature of these elements is the prompt negative temperature coefficient of reactivity, which gives the TRIGA® reactor its built-in safety by automatically returning the reactor power to its normal level in the event of a power excursion.

The reactor core (Fig. 4-1) consists of a cylindrical lattice of the fuel elements, control rods and a polonium-beryllium neutron source. Water occupies about one-third of the core volume. The core is presently limited to be loaded to give an excess reactivity of 2.1% $\delta k/k$. The fuel elements have graphite end sections serving as top and bottom reflectors. Neutron reflection in the radial direction is provided by approximately 1 foot of graphite enclosed in an aluminum housing. In addition, initially, in the outer regions of the core there are some graphite dummy elements.

All the core components are contained between top and bottom aluminum grid plates. The elements, control rods and dummy elements are located in the 126 holes arranged in six concentric rings around a central hole in the top plate.

Versatile and accurate control of the power level of the reactor is provided by four control rods. Two poison rods with fueled followers are adjusted by electrically driven rack and pinion mechanisms. The third poison rod has an air filled follower and is connected with an adjustable travel to a pneumatic-electromechanical drive system. The fourth rod is pneumatically driven but no travel adjustment is possible. This rod has a double length poison section so that initial acceleration on fast removal is achieved with no change in reactivity. Thus the actual reactivity change occurs at a higher rate than with rod 3. For high level pulsing when both rods are used, a delay circuit is set so that both rods reach their upper limits at the same time.

The UCINRF reactor is well equipped for specimen irradiation. A rotary 'lazy susan' device rests in a well in the top of the graphite reflector, providing watertight irradiation space which may be loaded from the bridge over the reactor tank at any time. 40 positions are provided for samples so that they can be irradiated under uniform neutron flux conditions using this facility. Two high speed (less than 1 second transfer time) pneumatic transfer systems permits research with extremely short-lived radioisotopes, one has a cadmium liner to reduce thermal neutron exposure of samples. These systems can deliver samples to various detectors set up in the reactor room. One moderate speed (4 second) pneumatic transfer system is used for regular work with short-lived radioisotopes. Sample loading and retrieval is in the facility laboratory [redacted] adjacent to the reactor room. The in-core termini for all three systems are located in various fuel element positions in core.

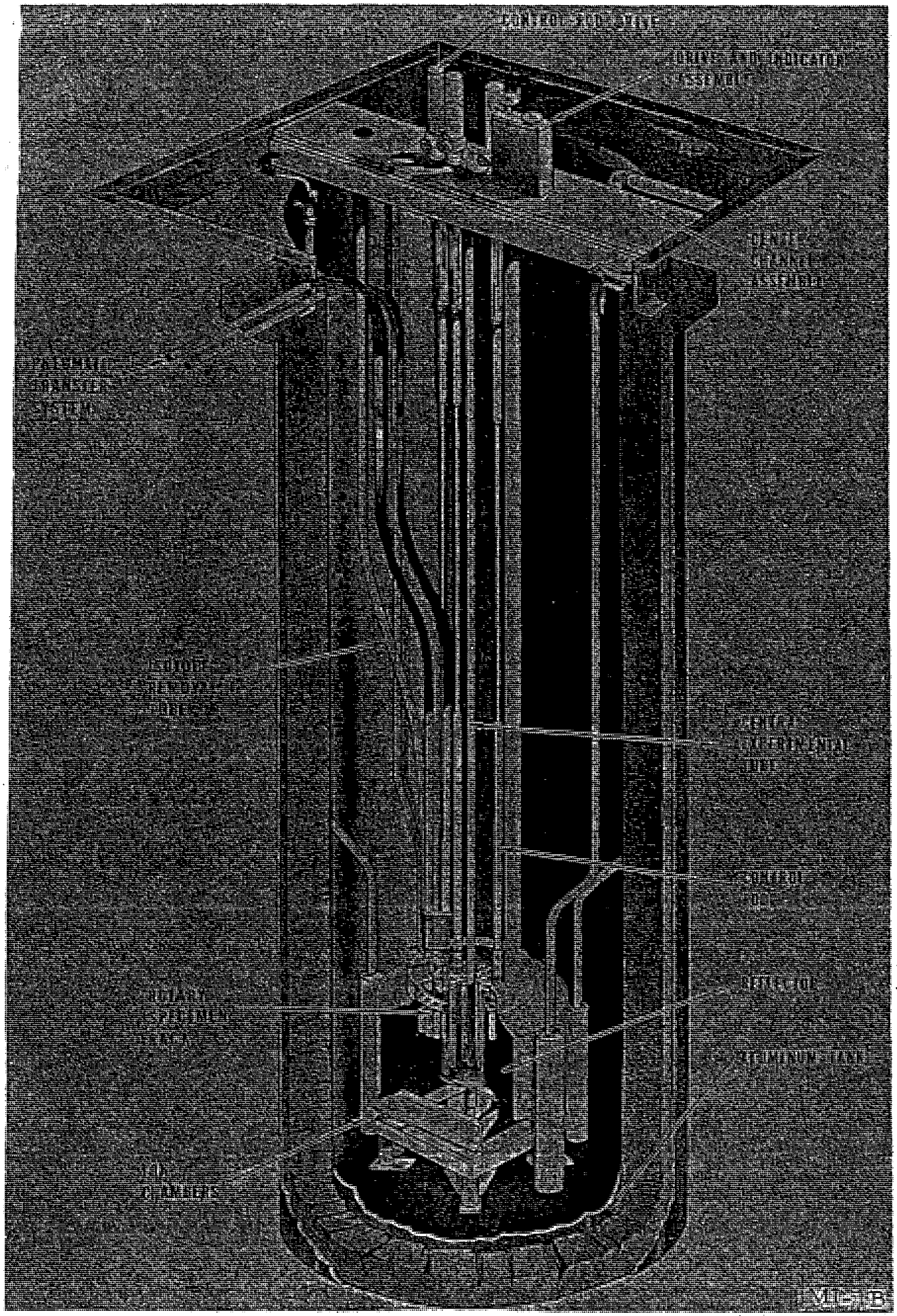


FIG. 4-1 Cross Section of TRIGA Mark 1 Reactor

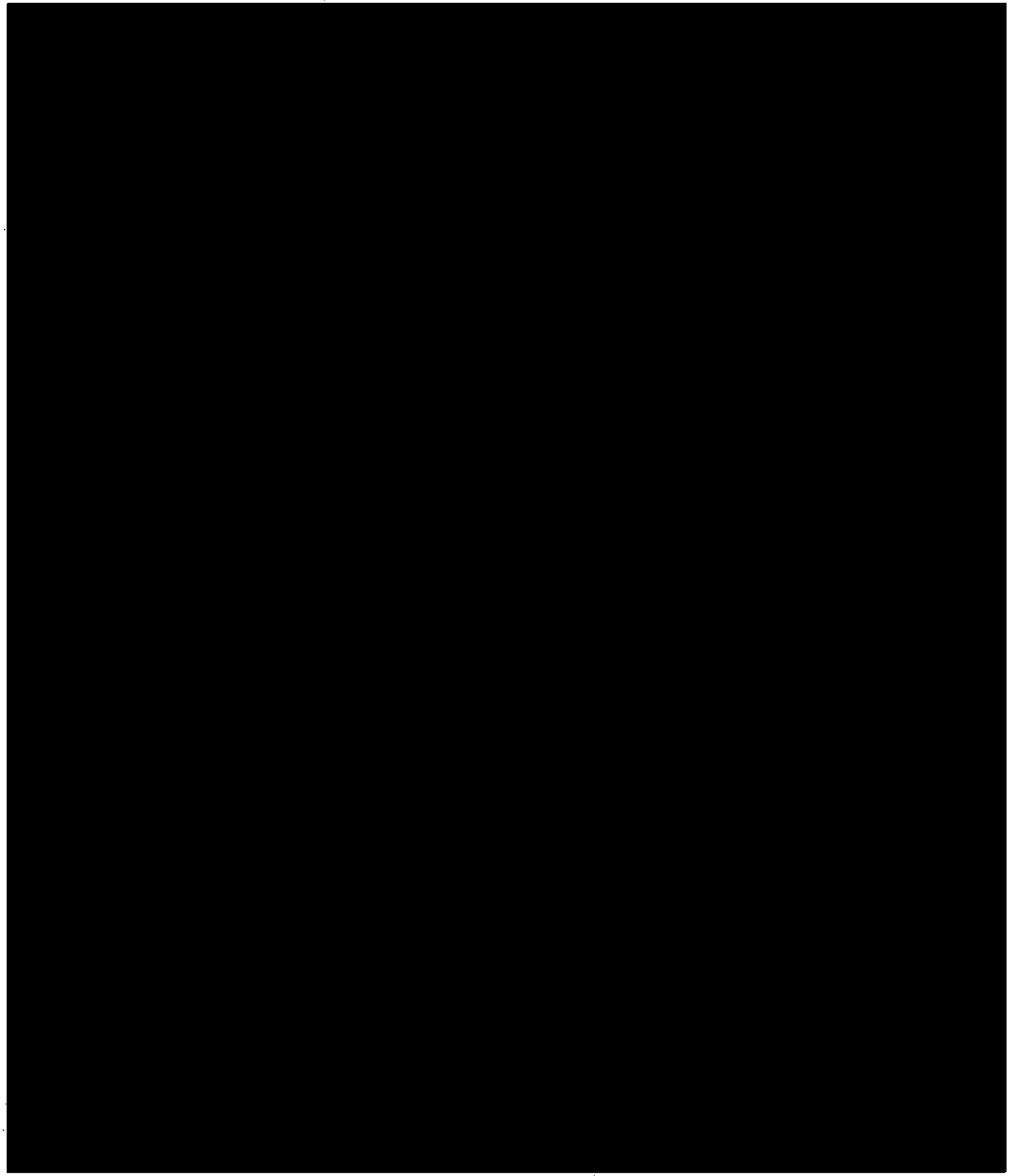


FIG. 4-2 Reactor Tank - Aluminum Liner

[redacted]

FIG. 4-3 Concrete Reactor Tank

A maximum flux position is provided in a dry tube called the Central Thimble at the center of the core. Experimental positions can also be created at other locations in the core by removal of certain portions of the upper grid plate following rearrangement of the fuel elements.

Instrumentation is provided to monitor, indicate and record the neutron flux and power level and their rates of change. Bulk-water temperature, fuel element temperature and water cooling system inlet and outlet temperatures are also displayed at the control console. Two modes of operational control are possible:

Mode 1. steady state, with manual or servo control.

Mode 2. transient (pulse) operation.

In addition to the reactor control instrumentation, an interlock system prevents reactor operation unless prescribed safety conditions have been met.

The reactor core is designed to be cooled by natural convection of demineralized water in the pool. Water from the pool is pumped through a filter and demineralizing unit which removes dissolved and particulate impurities to reduce the radioactivity accumulation and the corrosion rate of components in the tank. Cooling of the demineralized water is effected by heat exchange with the University's chilled water system.

4.2 Fuel-Moderator Elements

The active section of each fuel-moderator element (Fig 4-4) is [redacted] long and [redacted] in diameter. The fuel is a solid homogeneous mixture containing 8.5% by weight of uranium enriched to 20% in U^{235} and zirconium hydride to approximately $ZrH_{1.7}$. To facilitate hydriding, a 0.18 in. diameter hole is drilled through the center of the active section; a zirconium rod is inserted in this hole after hydriding is complete. Cylindrical graphite slugs, abou[redacted] are placed above and below the fuel to act as reflectors and the assembly is heliarc welded into a [redacted] thick stainless steel can. With the stainless steel end fixtures the element is approximately [redacted] long and weighs [redacted] lbs. The average U^{235} content is abou[redacted] grams. The top end fitting is grooved and specially shaped to lock into a fuel handling tool and is designed to allow passage of cooling water around it through the top grid plate in which it locates. The bottom end fitting supports the entire weight of the element on the lower grid plate. Serial numbers are inscribed on the top end fixture of every element.

One instrumented element has three thermocouples embedded in the fuel. As shown in Fig 4-5, the sensing tips of the fuel element thermocouples are located approximately halfway between the zirconium rod and the cladding and at the horizontal center of the fuel section and 1 inch above and below the center. The leadout wires pass through a seal contained in a [redacted] stainless steel tube welded to the upper end fixture. This tube projects about [redacted] above the upper end of the element and is extended by lengths of [redacted] tubing connected by 'Swagelok' unions to provide a watertight conduit carrying the leadout wires above the water surface in the pool. In all other respects, the instrumented element is identical to the standard element.

[redacted]

FIG. 4-4 Stainless-steel-clad Fuel-moderator Element

[redacted]

FIG. 4-5 Instrumented Stainless-steel-clad Fuel-moderator Element

The reactor core is loaded to produce a cold, clean excess reactivity of less than 2.1% k/k($3.0). Initially, 68 elements were required including one instrumented element and two fueled control rod followers to achieve an excess of S2.84. 22 graphite-filled dummy elements were also installed. The core as of June 30th 1999 has 82 elements (including an instrumented element and 2 fuel follower control rods), 33 graphite reflector elements, and 6 remaining water filled fuel element positions for a cold, clean excess of $2.86.

4.3 Control Rods And Drives

4.3.1 Control Rod Description

Two motor driven rods—one shim and one regulating-control reactor power during steady state operation. These rods (Fig. 4-6) are encased in stainless steel cans 43-in. long and 1-3/8-in. diameter and pass through 1-1/2-in. diameter guide holes in the top and bottom grid plates. An aluminum 'safety' plate is attached to the reflector beneath the lower grid plate (Fig. 4-6) to prevent the possibility of a rod accidentally dropping out of the core. The upper section of each rod is graphite the next 15 in. is the poison (graphite impregnated with boron carbide), the follower section consists of 15 in. of standard fuel and the bottom section is 6-1/2-in. of graphite.

The third rod (Fig 4-7) is the adjustable transient rod. This rod is approximately 37 in. long with a 15 in. section of borated graphite poison and a 21 in. long air space encased in aluminum. The rod slides in a thin walled aluminum tube that passes through the top and bottom grid plate and is screwed into and supported by the aluminum 'safety' plate beneath the grid.

The fourth control rod is the fast transient rod and it acts as a safety rod in steady state operation. This rod has a double length borated graphite poison section encased in aluminum. The rod is guided by a thin-walled guide tube which passes through the grid plates and screws into the safety plate (Fig. 4-8).

The control rods have a stroke of approximately 14-in, except for the fast transient rod which travels approximately 30 inches.

4.3.2 Rack and Pinion Drives for Standard Rods

Each drive (Fig. 4-9) consists of a motor driving a rack and pinion gear. The control rod is attached to an extension rod carrying a piston and a magnet armature. The armature engages to the rack mechanism through an electromagnet. In the event of a power failure or scram signal, the magnet is be de-energized and the control rod will fall into the core. The piston moves inside a barrel and provides dashpot action when close to the lower end of travel. The barrel is well ventilated at its upper end so that the piston and rod can move freely. The drive motor for the shim rod can insert or withdraw at a rate of approximately 19-in. per minute. Electrical dynamic and static braking on this motor is used to provide fast stops and to limit coasting or overtravel. The regulating rod drive motor is a variable-speed servomotor with a tachometer generator for rate feedback; it will insert and withdraw the rod at a maximum rate of about 24 in. per minute. Position indication for either rod is given by a helipot connected to the pinion which feeds back to voltage meters in the console.

Limit switches mounted on each drive assembly stop the rod drive motor at the top and bottom of travel and provide switching for console indicator lights.

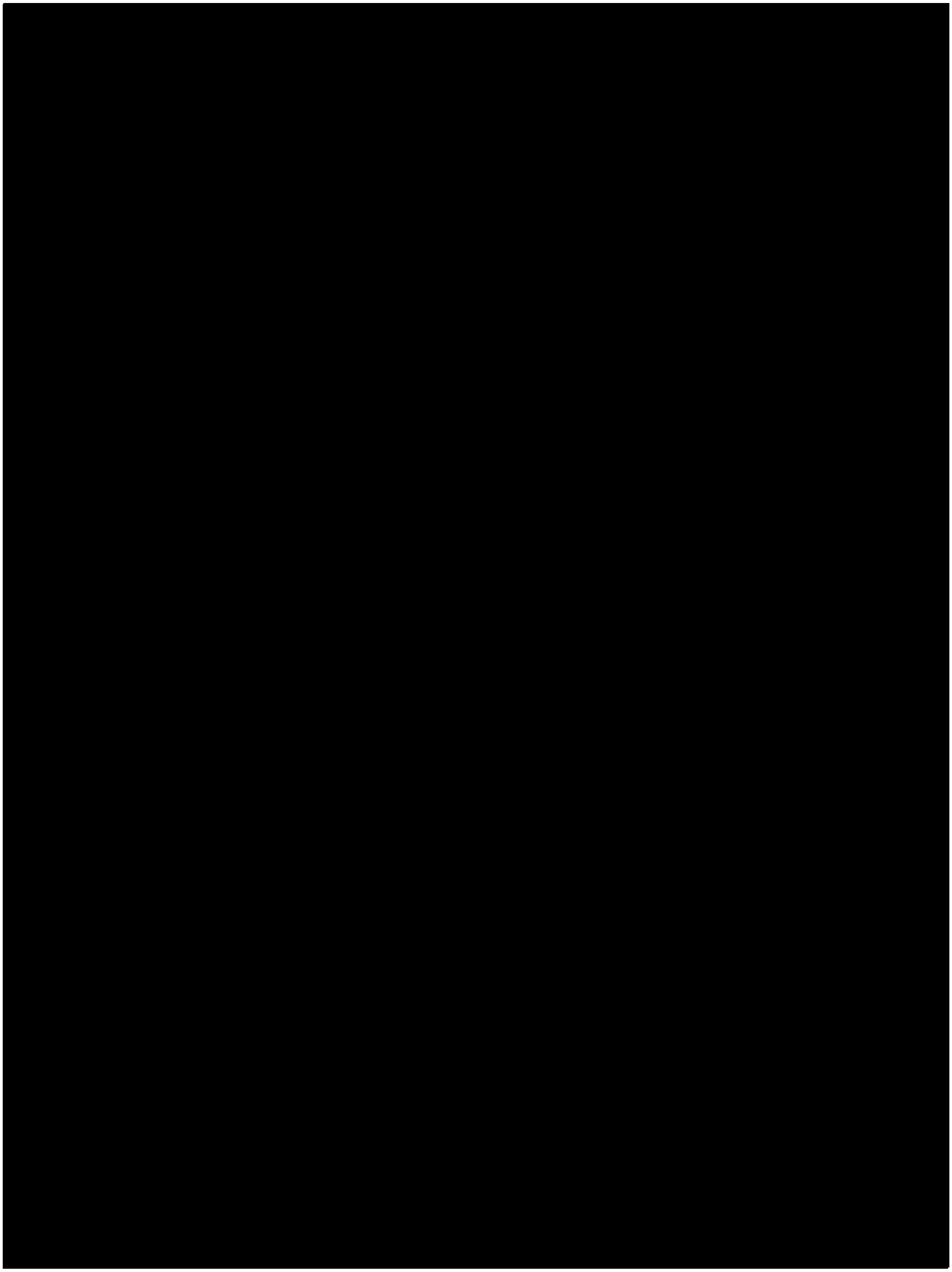


FIG. 4-6 Fueled-follower Control Rod in Core

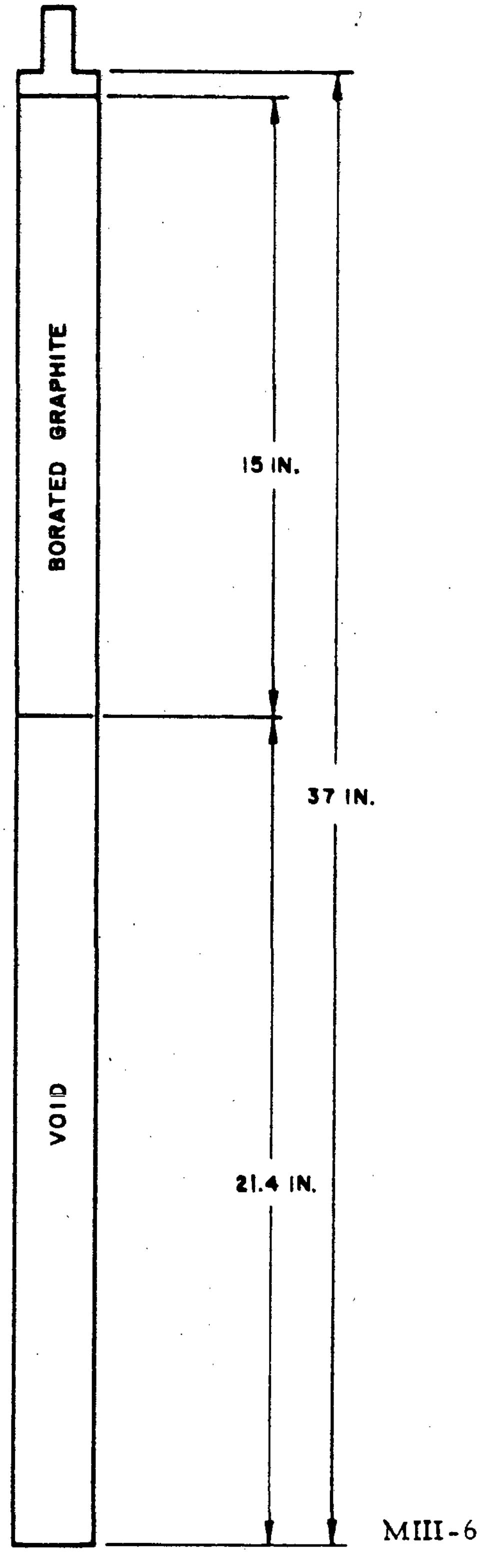


FIG. 4-7 Transient Control Rod

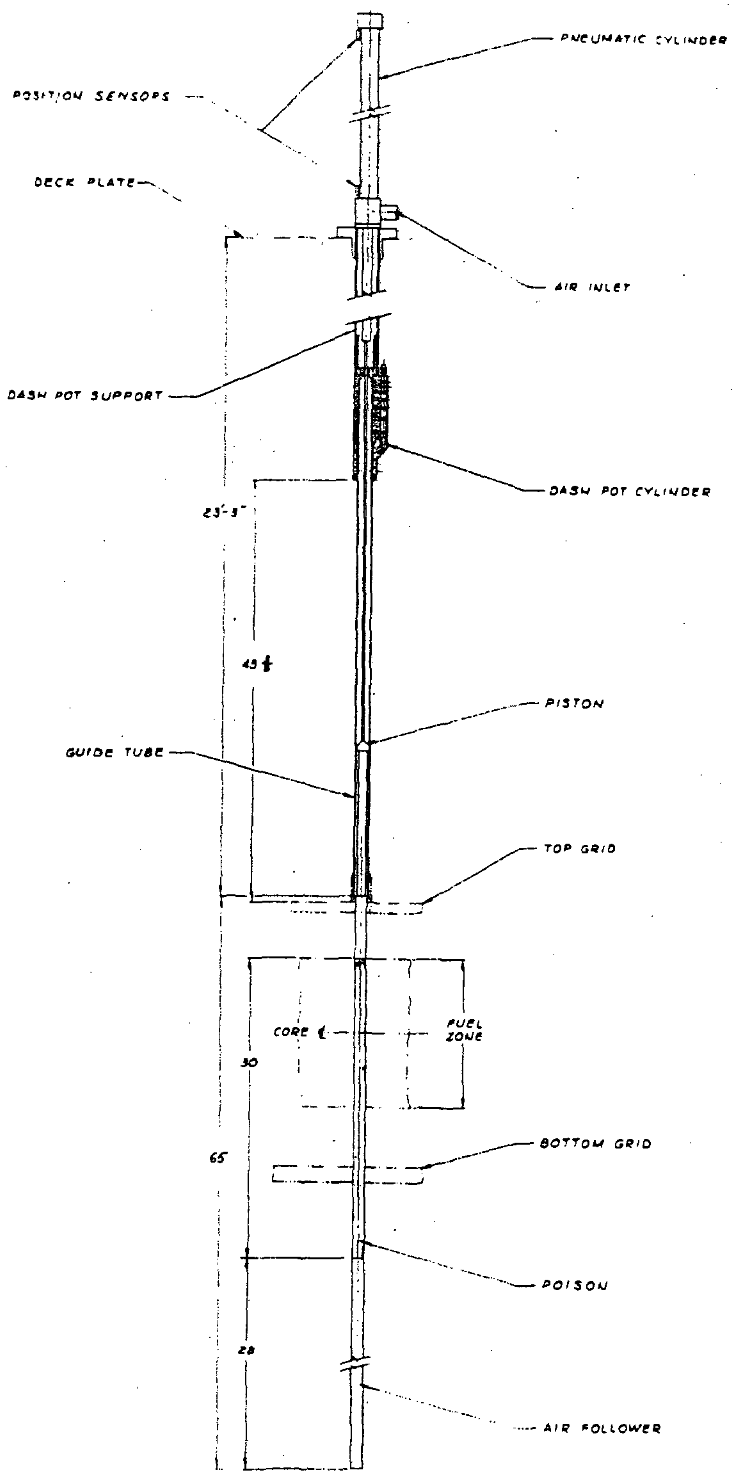


FIG. 4-8 Fast Transient Rod Drive

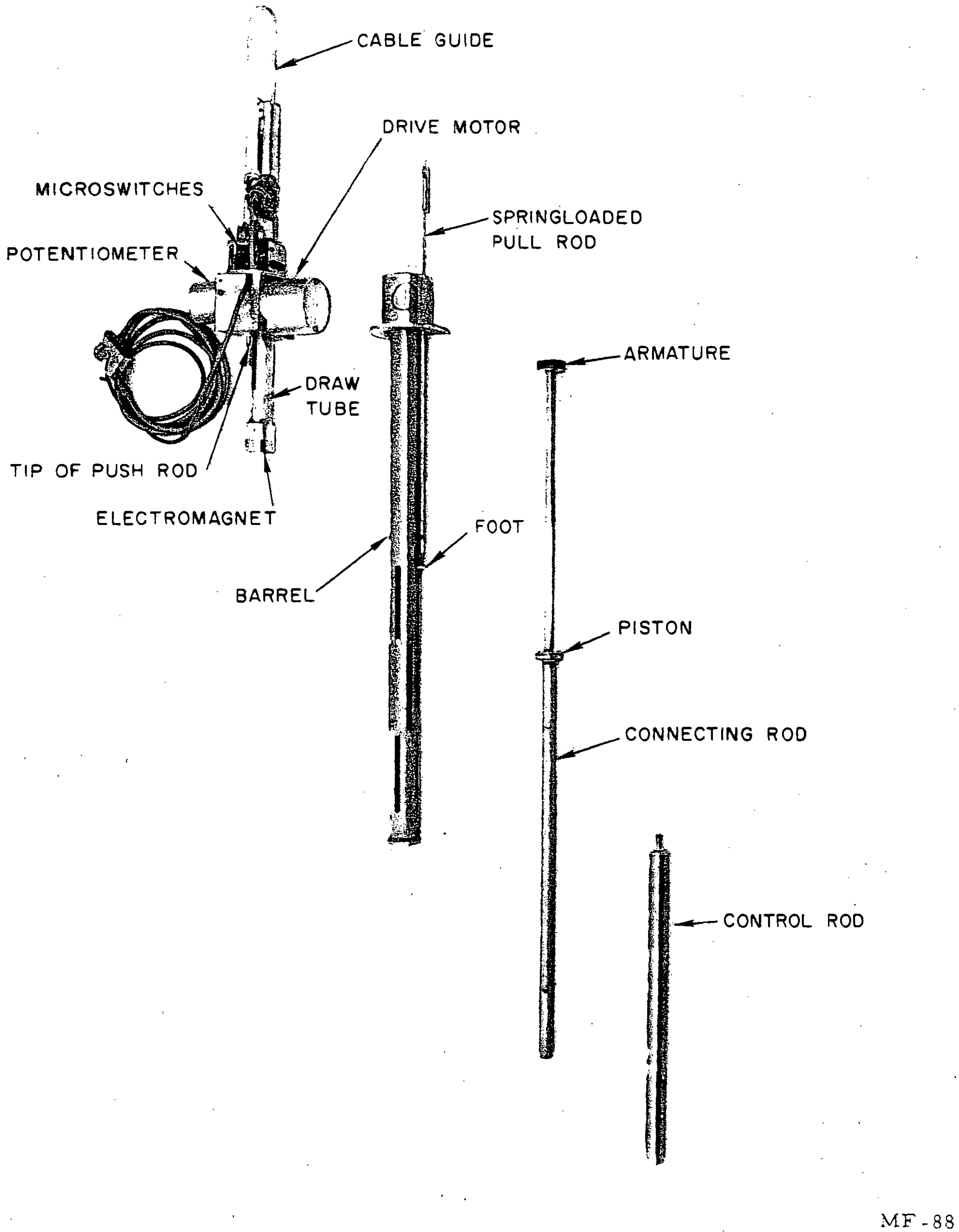


FIG. 4-9 Rack and Pinion Drive

4.3.3 Adjustable Transient Rod Drive

The pneumatic electromechanical drive system (Figs. 4-10 and 4-11) permits ejection of a predetermined amount of the transient control rod from the core.

The pneumatic portion is basically a single acting pneumatic cylinder with the piston attached by a connecting rod to the control rod. Compressed air is admitted at the lower end of the cylinder to drive the piston upwards. The air being compressed above the piston is forced out through vents at the upper end of the cylinder. At the end of its stroke the piston strikes the anvil of a shock absorber. The piston is thus decelerated at a controlled rate at the end of its stroke. This action minimizes rod vibration after transit.

An accumulator tank mounted under the bridge stores the compressed air for operation. A three way solenoid valve controls the air supplied to the cylinder. De-energizing the solenoid interrupts the air and relieves the pressure in the cylinder so that the rod will reinsert under gravity except when air is supplied to the cylinder.

To control the overall travel and hence the reactivity inserted for the pulse, the cylinder may be raised or lowered. This is done by an electric drive independently of the piston or control rod. The motor drives a ball nut assembly through a worm gear. The balls engage in threads on the outside of the air cylinder which can thus be raised or lowered to limit the upper position of travel of the transient rod. A series of limit switches, as on the standard rods, provides indication of the status of all parts of the system.

4.3.4 Fast Transient Rod Drive

The fast transient rod is driven by a pneumatic cylinder, and a hydraulic dashpot provides deceleration at the end of the stroke.

A guide tube, which is supported by the 'safety' plate, passes through the two grid plates and fits into the end of the dashpot assembly; the guide tube is not rigidly attached to the dashpot. Since the dashpot is attached firmly to the bridge, the deceleration forces are transmitted to the bridge rather than to the core structure. The kinetic energy of this system is higher than that of the adjustable transient rod, but the deceleration forces are lower since a longer (6 inch) deceleration stroke is provided. The length of the power stroke is 30 inches. The cylinder is provided with low friction seals so that the rod will drop freely back after the transient. A magnetic pick-up is attached to the outside of the cylinder to indicate the end of the reactivity insertion portion of the stroke. The signal is used during set-up to coordinate the reactivity insertions of the two transient rods.

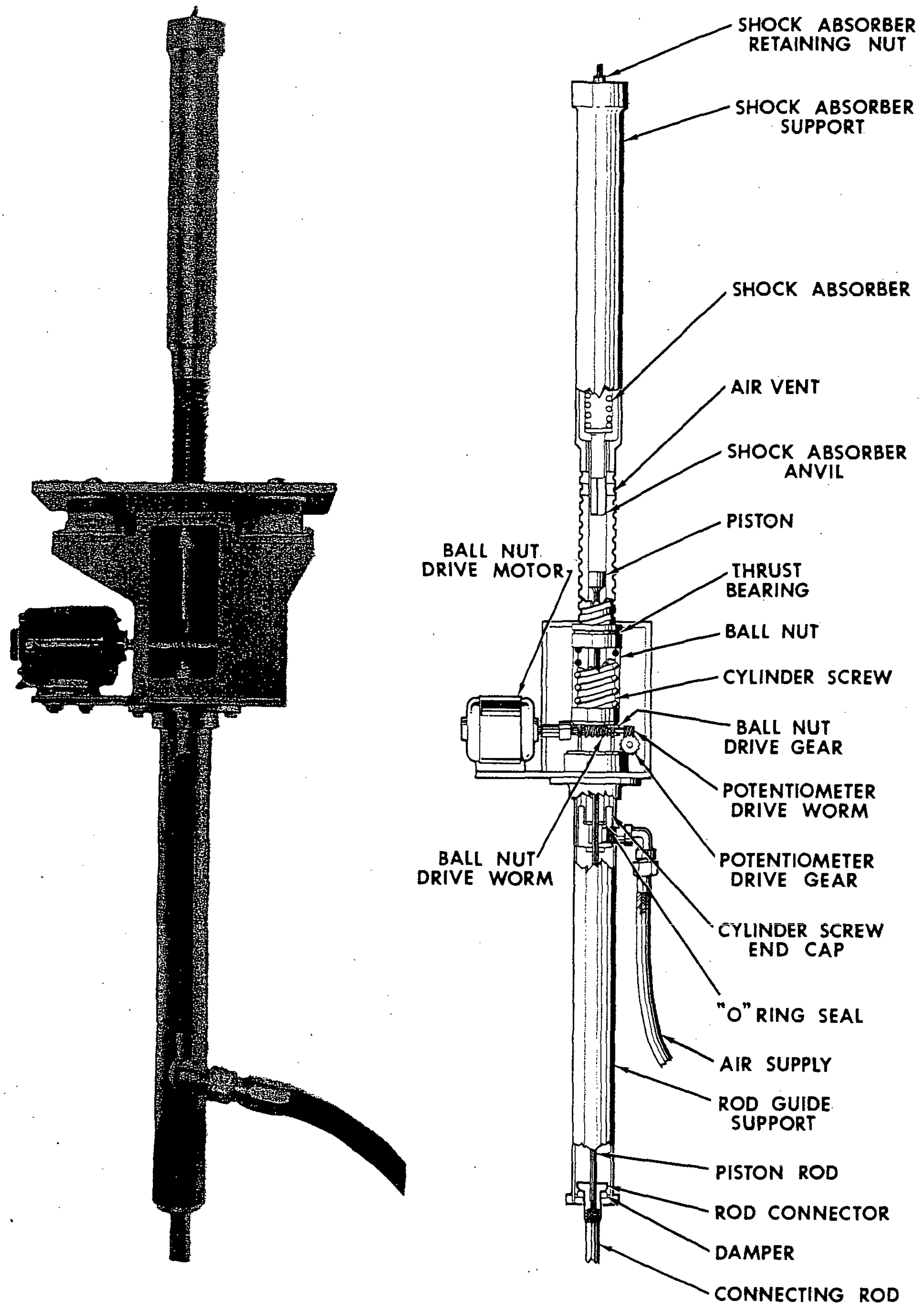


FIG. 4-10 Pneumatic-electromechanical Transient Rod Drive

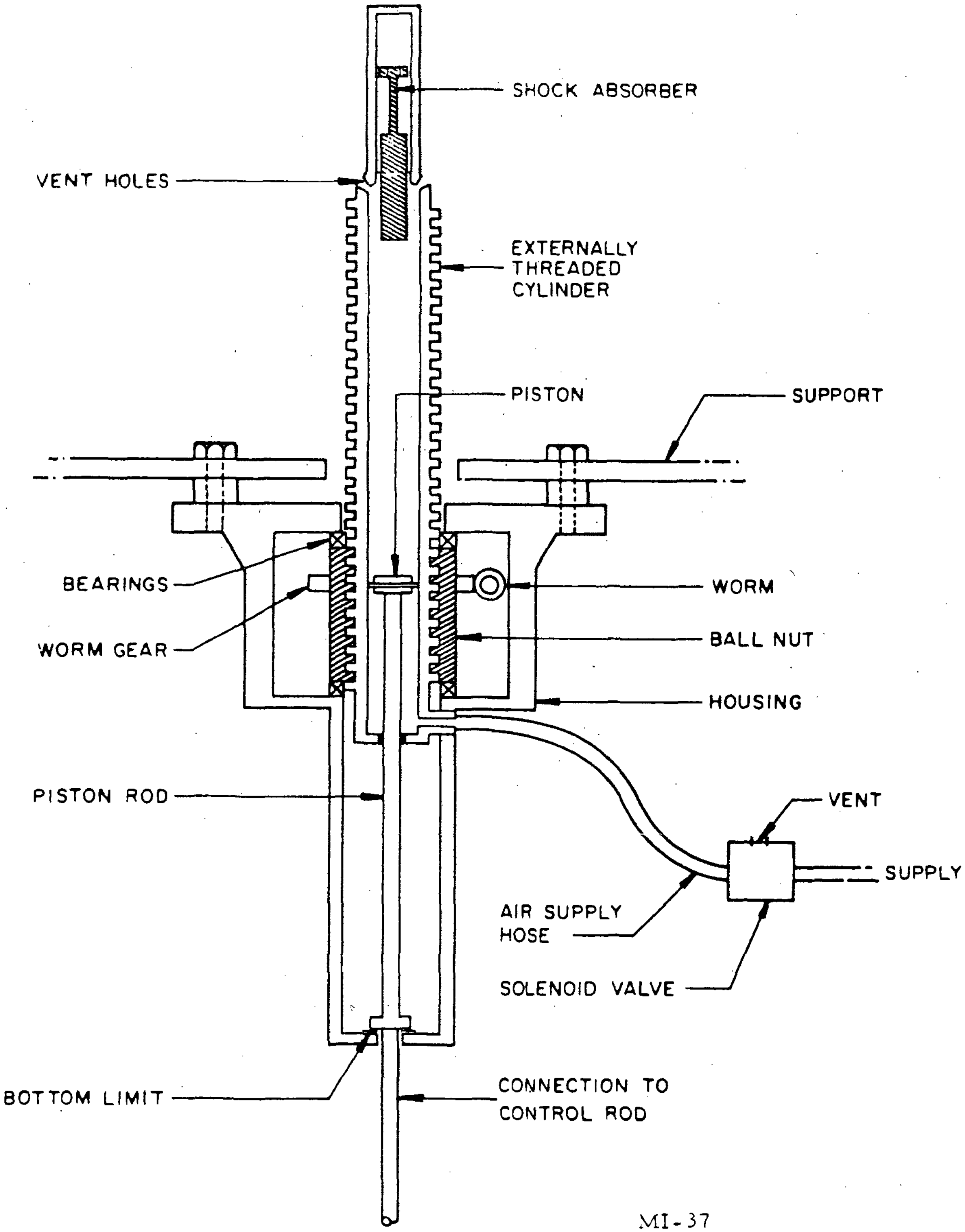


FIG. 4-11 Schematic Drawing of Transient Rod Drive

4.4 Basic Reactor Components

The reactor core and reflector assembly (Fig. 4-12) form a cylinder approximately 43 in. in diameter and 23 in. high. The core consists of a lattice of fuel-moderator elements, graphite dummy elements and control rods surrounded by a graphite reflector.

4.4.1 Reflector

The reflector is a ring shaped block of graphite in a leak tight welded aluminum can. It is 11-3/16 in. thick radially with an inside diameter of 20-7/8 in., and is 22-1/2 in. high. A well is provided in the top for the rotary specimen rack. The latter is a self contained unit and does not penetrate the sealed reflector block in any way. The reflector assembly rests on an aluminum platform about 2 ft from the bottom of the tank.

4.4.2 Grid Plates and Safety Plate

The top grid plate is a 5/8 in. thick aluminum plate (reduced to 3/8 in. thick in the central region) that provides accurate lateral positioning for the core components. The plate is supported by a ring welded to the top inside surface of the reflector container and is anodized to resist wear and corrosion. One hundred and twenty-six (126) holes, 1.505 in. in diameter, are drilled through the top grid plate in six circular bands about a center hole to locate the lattice components (Fig. 4-13). The 1.505 in. hole in the center accommodates the central thimble. Small holes at various positions permit the insertion of foils for flux mapping in the core.

A hexagonal section can be removed from the center of the upper grid plate for insertion of specimens up to 4.4 in. in diameter into the region of highest flux; this requires prior relocation of the six fuel elements from the B ring to the outer portion of the core and removal of the central thimble.

Two generally triangular-shaped sections are cut out of the upper grid plate, each encompassing two E and one D ring holes. When fuel elements are placed in these locations a special fixture provides lateral support. With the fixture and fuel removed, specimens up to 2.4 in. diameter may be inserted. Two 3/8 in. diameter holes between the F and G rings of the grid plate locate and provide support for the neutron source holder at alternate positions.

The differential area between the triangular-shaped spacer blocks at the top of the fuel elements and the round holes in the top grid plate permits passage of cooling water through the plate. The bottom grid plate is 3/4 in. thick aluminum which supports the entire weight of the core and provides accurate spacing of the fuel-moderator elements. Six pads welded to a ring which is in turn welded to the reflector container support the bottom grid plate

Holes in the bottom grid plate are aligned with fuel element holes in the top plate. They are countersunk to receive the lower end fixture of the fuel-moderator elements or the adaptor end of any experimental device. A central hole, 1.505 in. diameter, in the lower grid serves as a clearance hole for the central thimble. Nine additional holes of this size are aligned with upper grid plate holes to provide passage for the fuel-follower control rods. In those positions not used for control rods, adapters are fitted so that a standard element may occupy the position. The adapter can be removed with a special handing tool if necessary.

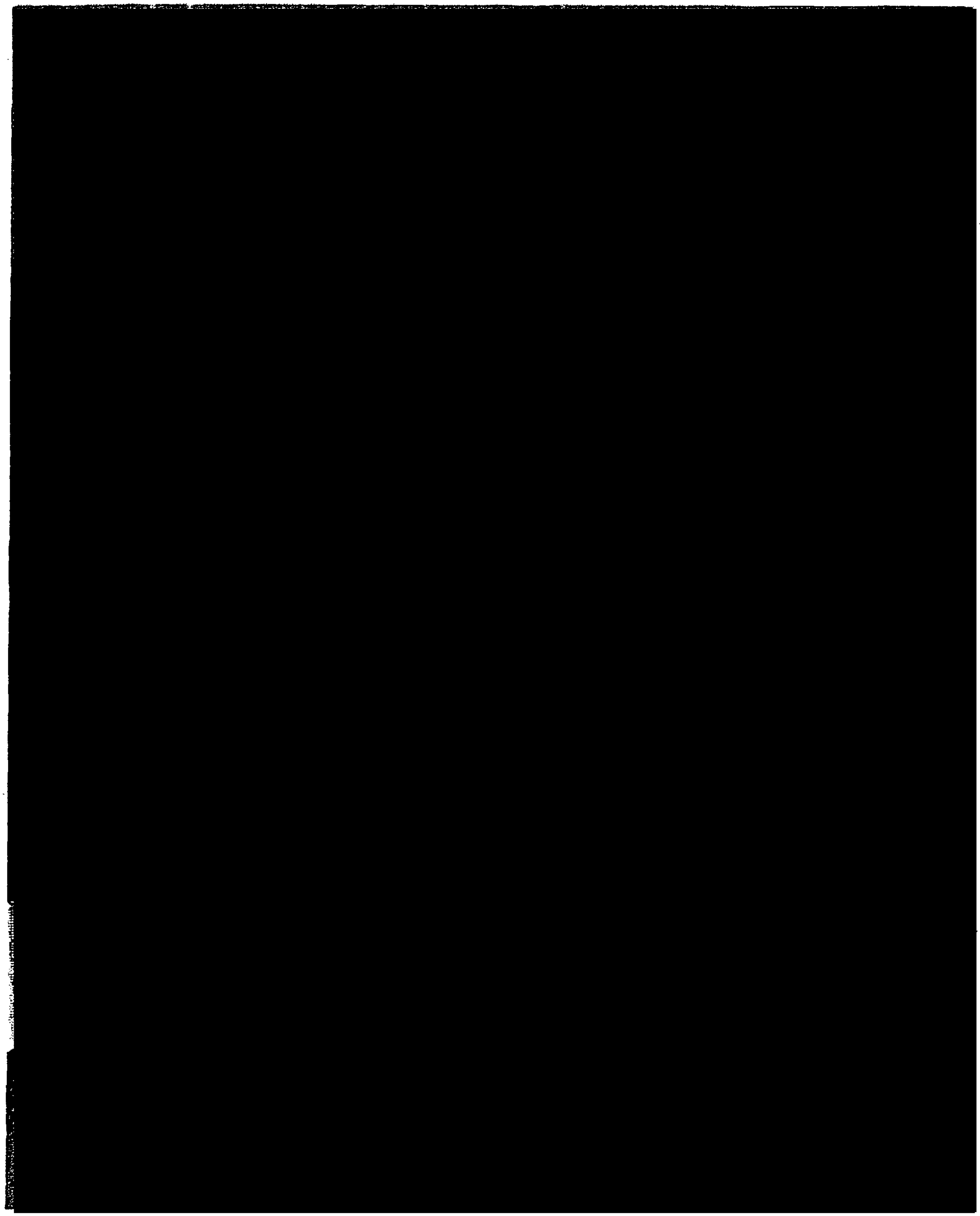


FIG. 4-12 Reactor Core Assembly

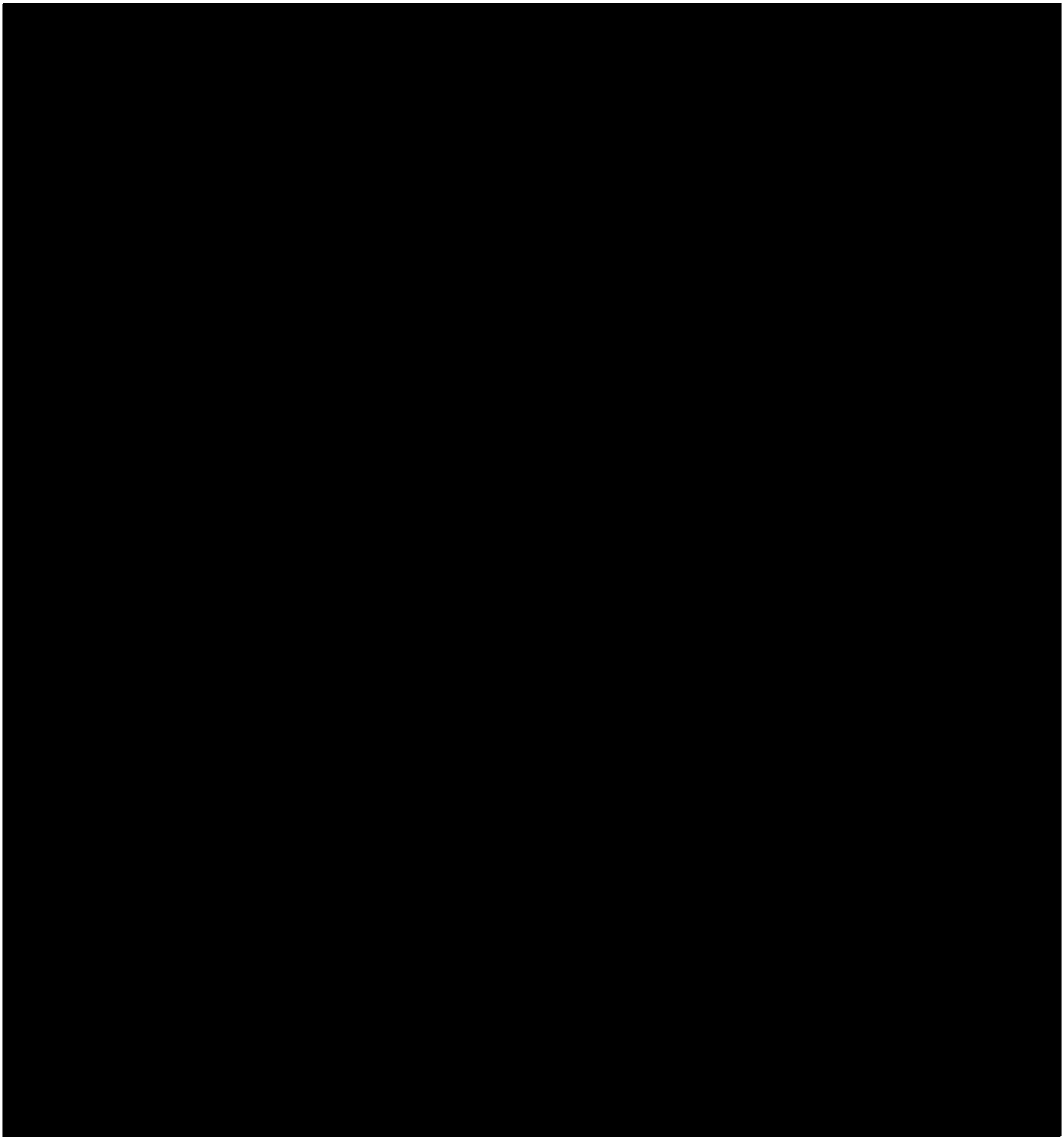


FIG. 4-13 Grid Array

The safety plate is a 1/2 in. thick aluminum plate welded to the extension of the inner reflector liner about 16 in. below the bottom grid plate. This plate precludes the possibility of a control rod dropping out of the core.

4.4.3 Reactor Bridge

A steel bridge spans the reactor pool. It is formed by two heavy channel beams with several 5/8 in. thick steel plates bolted end to end between them. The bridge supports the rod drives and the loading and drive mechanism for the rotary specimen rack; it is designed to support the weight of a 3,000 lb fuel cask.

The remaining portion of the reactor pool at the core end is covered with clear plastic sheet supported by aluminum grates. The grates can support the weight of people and apparatus and can be raised to give access to the pool.

4.4.4 Neutron Source and Holder

A [redacted] curie americium-beryllium neutron source is installed for start-up. The doubly encapsulated source in a cylindrical aluminum holder will fit into either of two diametrically opposed holes in the top grid plate located between the F and G rings. A flange at the upper end of the holder rests on the top grid plate and provides support and proper positioning of the source.

4.5 NUCLEAR DESIGN

4.5.1 General Characteristics

The various characteristics and parameters associated with operating the University of California at Irvine TRIGA® have been extrapolated from parameters from other TRIGA® reactors, and from calculations that have been correlated with experience with these same reactors. Specifications for the fuel in the reactor are given in Table 4-1.

Table 4-1

TRIGA® STAINLESS-STEEL CLAD FUEL-MODERATOR SPECIFICATIONS

Over-all length, in.............	[redacted]
Outside diameter, in............	[redacted]
Fuel outside diameter, in........	[redacted]
Fuel length, in..................	[redacted]
Fuel composition.................	U-ZrH
Weight of U-235, g...............	[redacted]
Uranium content, wt-%............	~8.5
Uranium enrichment, %............	< 20
Hydrogen-to-zirconium atom ratio.	1.65 to 1.70
Cladding material...............	304 stainless steel
Cladding thickness, in...........	0.020

4.5.2 Reactor Nuclear Parameters

The following parametric values were estimated for the UCI TRIGA from calculations correlated with measurements made on other reactors of the same generic type. Even though these were carried out prior to the first installation, they have been entirely satisfactory in application to this facility. Thus, although more recent methodologies may have been employed at other facilities, it is felt that the values given in this section are reasonable and justified by experience in facility utilization. Table 4-2 summarizes the reactor core parameters.

Figure 4-14 shows the estimated reactivity loss as a function of operating power for a 70-element core. Figure 4-15 indicates the corresponding maximum fuel element temperature.

Figure 4-16 shows the neutron flux as a function of radius at the axial centerline. These fluxes were calculated using the GAZE code developed by General Atomic. This is a one-dimensional diffusion-theory code which will allow full scattering matrices (up or down scatter over the entire group structure) and will allow up to 96 energy groups.

Two other Gulf General-Atomic-developed codes were used to generate the cross-sections for the reactor problem. GAM-I generates fast-neutron cross-sections by solution of the P_1 equations from which a space-independent flux is derived for a given composition; the space-independent flux is used to average fine group cross-sections. Thermal-group cross-sections were generated in essentially the same manner using the GATHER-1 code. However, scattering kernels were used to describe properly the interactions of the neutrons with the chemically bound moderator atoms. Cross-sections above 1 eV were calculated by GAM-1 and below 1 eV by GATHER-1.

Homogenization of the core, to account for the fact that the thermal neutrons see a heterogeneous array of fuel and water, was accomplished by the use of cell calculations which used the transport theory GAPLSN code. This was done in order to calculate group-dependent disadvantage factors for each region of the cell. New GATHER-1 problems were then run using these disadvantage factors to produce cross-sections for the homogenized cell which would give the same reaction rates as the heterogeneous cell.

Table 4-2

CORE NUCLEAR PARAMETERS USED AND DETERMINED

Cold, clean, critical loading	~60[a] elements
	[redacted]
Operational loading	~70[a] elements
	[redacted]
l-Prompt neutron lifetime	43 μsec
β-Delayed neutron fraction	0.7%
α-Prompt temperature coefficient	13.4×10^{-5} δk/k / °C

T_f-Average fuel temperature	23°C	200°C
T_w-Average water temperature	23°C	23°C
η	2.073	2.0645
p	0.9157	0.6991
p	0.9517	0.9481
ε	1.054	1.055
k_∞ (=ηfp)	1.4683	1.4435
τ	22.23 cm^2	22.22 cm^2
L^2	1.9118 cm^2	2.1110 cm^2
B^2	0.018785 cm^{-2}	0.017235 cm^{-2}
k_{eff}	1.00	1.00

Φ/P (0<E<1 eV) - (operational loading) 1.2×10^7 n/cm^{2}sec-watt

P_{max}/P_{avg} ~2.0

[a]Including two fueled control rod followers and one instrumented element.

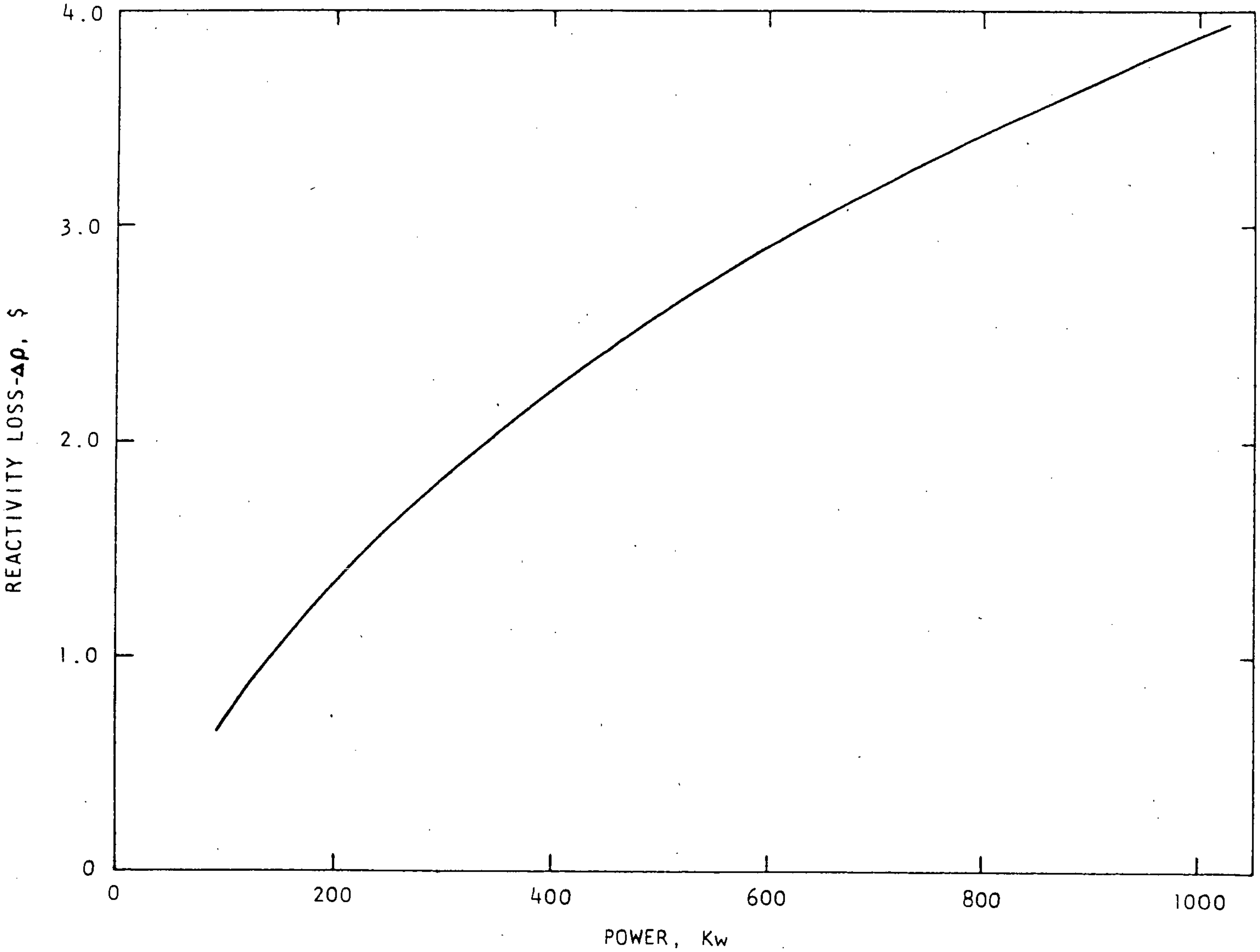


FIG. 4-14 Estimated Reactivity Loss vs. Power

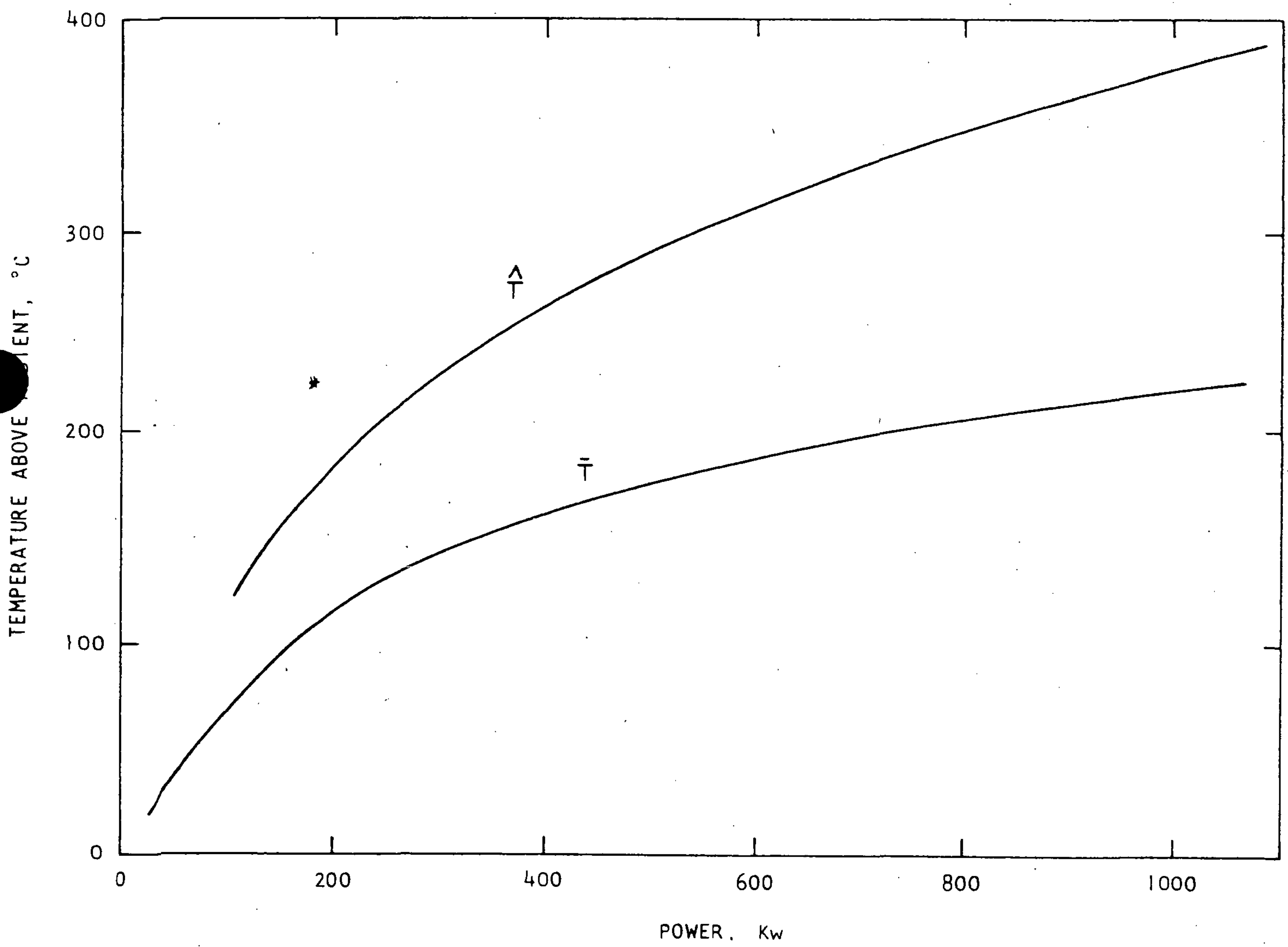


FIG. 4-15 Estimated Maximum B-Ring and Average Fuel Temperature vs Power.

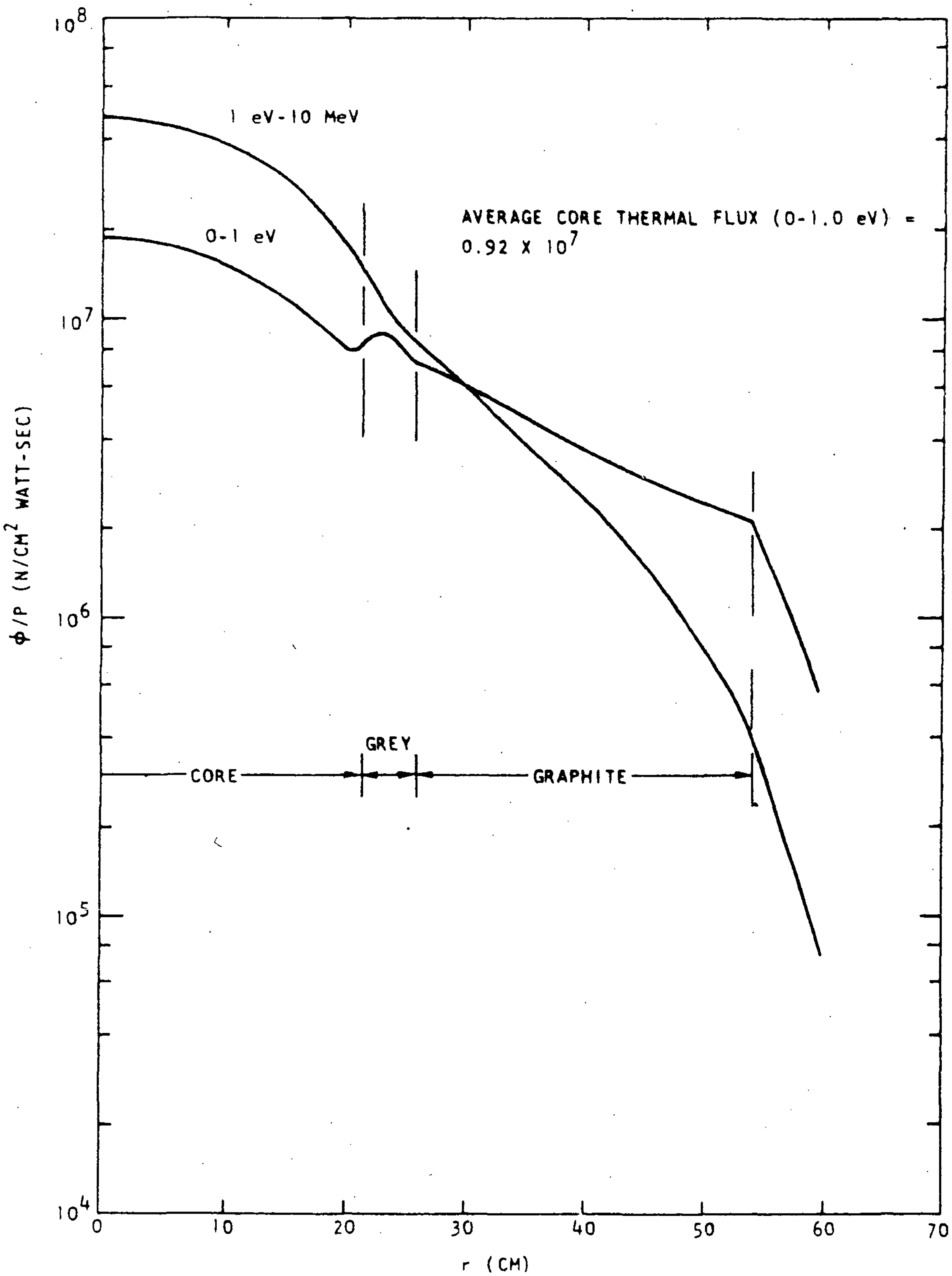


FIG. 4-16 Axial Mid-Plane Flux per Watt vs Radius

4.5.2 Reactivity Effects-Experimental Facilities

The reactivity changes associated with the insertion or removal of experiments is controlled through technical specification limits so that inadvertent changes whilst operating are limited to $1.0. Table 4-3 is presented to provide a guide to the order of magnitude of the effects associated with introducing experiments.

Table 4-3

APPROXIMATE REACTIVITY EFFECTS

ASSOCIATED WITH EXPERIMENTAL FACILITIES

Facility	Change	Worth (% δk/k)
Central grid hole	fuel element versus water	+0 9
Central thimble	void versus water	– 0.6
Triangular cutout	three fuel elements versus water	+0.8
Triangular cutout	void versus water	–0.23
Pneumatic transfer tube (G-ring)	void versus water	–0.10
Rotary specimen rack	void versus water	–0.1 to –0.3

4.5.4 Fuel Element Worths

The worth of fuel elements is dependent on their position in the core as well as other parameters which have minor effects, such as pool temperature. Measurements of fuel element worth in various TRIGA reactors have been used to extrapolate the estimated worth as given in Table 4-4.

TABLE 4-4

ESTIMATED FUEL ELEMENT WORTH

COMPARED WITH WATER FOR EACH CORE RING

Grid Position	Worth (% δ k/k)
B	0.81
C	0.67
D	0.50
E	0.33
F	0.22
G	0.15

4.5.5 Control Rod Worths (as of 2/10/99)

ROD	Core Grid Position	Worth ($)	Worth (%δk/k)
REG	C10	2.79	1.95
SHIM	C4	3.70	2.59
ATR	D1	1.82	1.27
FTR	F13	0.70	0.49

4.5.6 Reactor Power Pulsing

The reactor loading will be a maximum of 2.1% δk/k ($3.00) excess reactivity above a cold, critical condition without xenon. A compact condition is a core loaded with all of the innermost lattice positions containing fuel or control rods excepting the center grid position.

Operating experience with TRIGA reactors indicates that a 2.1% δk/k step addition of reactivity from low power presents no hazard. TRIGA reactors have been pulsed safely thousands of times with reactivity additions up to 3.5% δk/k. A typical plot of power pulse and the fuel temperature for a 2.1% δk/k addition is shown in Fig. 7-4. The resulting power pulse attains a peak power of about 1200 Mw with a reactor period of 3.1 msec, a total energy release during the pulse of about 16 Mw-sec, and a pulse width of about 11 msec. Fuel temperatures are less than 450°C. These values, which were derived by use of the Fuchs-Nordheim model have been confirmed by actual operating experience. This model yields the coupled set of differential equations:

$$\ell \dot{P} = (\rho - \alpha T)P \quad \text{and} \qquad (1)$$

$$C\dot{T} = P - P_0 \quad \text{with} \qquad (2)$$

$$C = C_0 + \gamma T \qquad (3)$$

where
- P = Power level (P_0 = initial power), watts
- ρ = Reactivity above prompt critical
- α = Magnitude of the negative temperature coefficient, $^{\circ}C^{-1}$
- T = Temperature °C (average over fuel) above the equilibrium temperature at P_0
- ℓ = Prompt neutron lifetime, sec
- C = Heat capacity of the fuel in the core, w-sec/°C
- C_0 = Heat capacity at the equilibrium temperature corresponding to P_0
- γ = Rate of change of heat capacity with temperature, w-sec/$^{\circ}C^2$

The above lumped parameter system neglects heat transfer and delayed neutron effects and averages space and neutron energy variations so that all coefficients are assumed constant.

Combining equations (1) and (2) yields

$$\frac{dP}{dT} = \frac{(\rho - \alpha T)(C_0 + \gamma T)P}{\ell(P - P_0)} \tag{4}$$

Integrating, using the condition that T = 0 when $P = P_0$ yields

$$\ell\left[(P - P_0) - P_0 \ell n \frac{P}{P_0}\right] = T\left[\rho C_0 + (\gamma\rho - \alpha C_0)T/2 - \frac{\alpha\gamma T^2}{3}\right]. \tag{5}$$

Maximum (or minimum) temperatures occur when

$$\frac{dT}{dt} = 0 = P - P_0 \qquad \text{or when} \qquad P = P_0$$

Then,

$$T\left[\rho C_0 + (\gamma\rho - \alpha C_0)\frac{T}{2} - \frac{\alpha\gamma T^2}{3}\right] = 0. \tag{6}$$

The roots of this equation are

$$T = 0 : \frac{-3}{8}(\sigma - 1)\left\{1 \pm \left[1 + \frac{16}{3}\frac{\sigma}{(\sigma - 1)^2}\right]^{1/2} T_f\right\}, \tag{7}$$

where

$$\sigma = \frac{\alpha C_0}{\gamma\rho}, \qquad \text{and} \qquad T_f = 2\,\frac{\rho}{\alpha}$$

and the positive sign is taken when $\sigma < 1$. Values of the parameters are

$\alpha = 1.34 \times 10^{-4}\ {}^{o}C^{-1}$ and

$C_0 = \left[0.718 \times 10^5 + 0.825 \times 10^2 (T_{SS} - 25)\right]$ watt-sec/°C

(conservatively assumed to be only 66 element core)

$\gamma = 0.825 \times 10^2\, watt - \sec/{}^{o}\, C^2$

T_{SS} = fuel temperature at P_0.

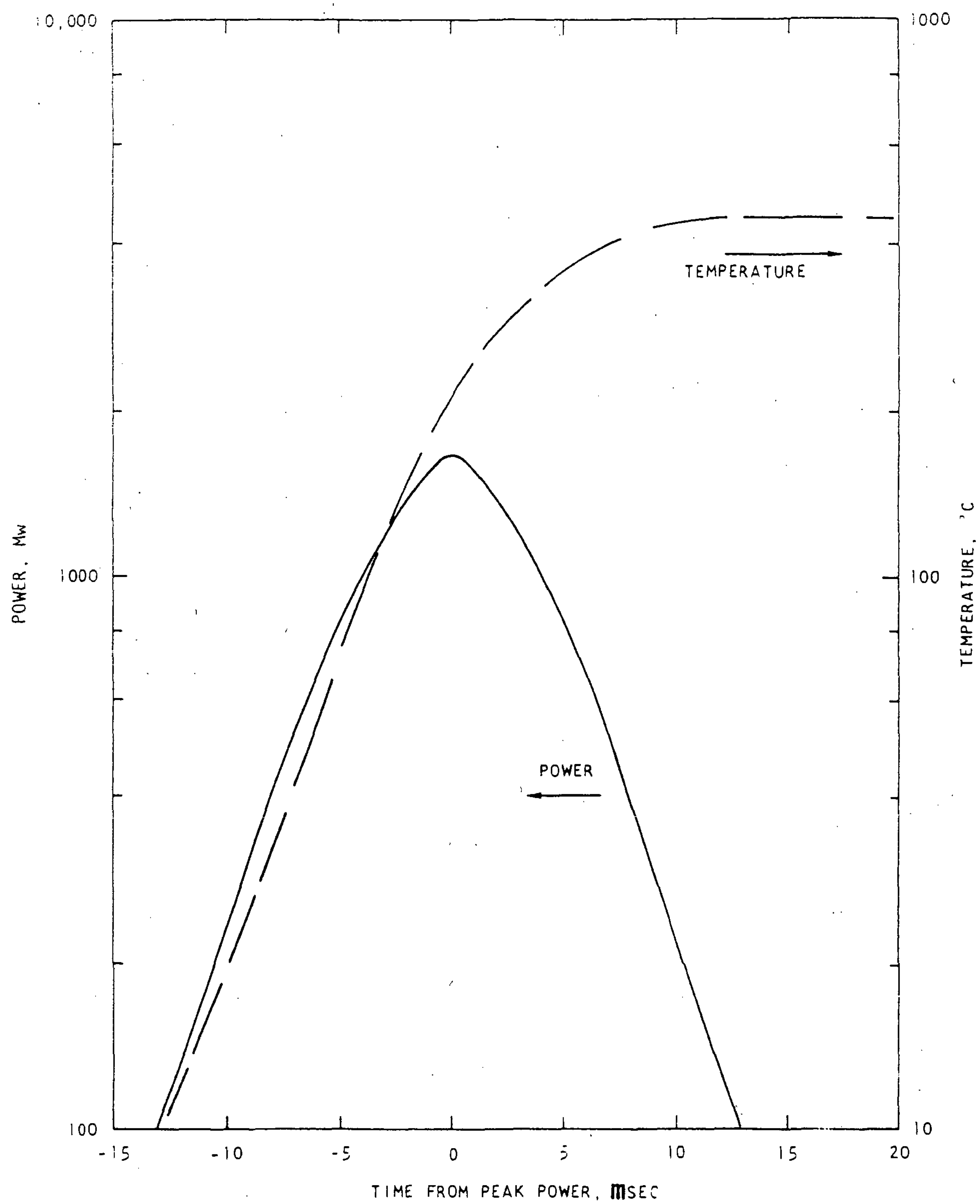


FIG 4-17 Calculated Peak Power and Fuel Temperature Rise vs. Time for $3.0 Pulse

5. REACTOR COOLANT SYSTEM

5.1 INTRODUCTION

TRIGA® reactors up to 1 megawatt power level and beyond are designed, and have operated successfully over many reactor years, with convection cooling. Sufficient water and flow channels exist within the core structure becaiuse of the grid plate and fuel element design to permit this. In the case of standard low density fuel such as in UCINRF, there is no concern about cladding temperatures overheating or approaching DNB limits, so no discussion of such is warrented at this facility. Cooling of the primary water is provided in order to enable operations over long periods without placing undue stress on the pool liner or other reactor components. Thus the reactor may be safely operated with no cooling available. Cooling is actuated on an as needed basis at this facility.

Administrative controls are used to limit the pool temperature to below 30°C and in practice operation has never continued above 25°C. Maintaining close limits shoudl extend the life of the tank and other components and gives additional assurance that reactor parameters, such as measued neutron fluxes, do not vary as a result of expansion of detector support structures, etc.

To minimize the effects of corrosion and to keep water radioctivity contamination as low as possible, a water purification system is provided. This system operates on a 24 hours a day basis. Contaminants are removed onto removable filters and/or mixed bed ion exchange resin, which are disposed through low level waste disposal procedures on an as needed basis.

5.2 WATER COOLING SYSTEM

A diagram of the water cooling circulation is shown in Fig. 5-1. Brief specifications for the components are given below.

Pump	Centrifugal, 7.5 hp, close coupled, self priming, stainless steel body and impeller and mechanical shaft seal.
Heat exchanger (Fig. 5-2)	All parts that are in contact with the demineralised water are Type 304 stainless steel. The exchanger is designed for a capacity of 880,000 Btu/hr. The shell is supplied at > 50°F by the University chilled water system flowing in black iron piping and heavily chemically treated to reduce corrosion. The connection and control system is shown in Fig. 5-3.
Controller	Watlow, Series 96. Input from platinum resistance sensor (RTD) stainless steel clad in pool. The control is normally set to maintain pool temperature at 19.0 ± 1.0 °C. Gate valve controls on outflow. (Fig 5-3)

4.3 WATER PURIFICATION SYSTEM

Water clarity and purity is maintained by constant circulation through a filter and ion exchange resin. A ppol skiimer helps to remove surface contamination. Purity is assessed by conductivity measurements taken during daily start-up exercises. New resin is installed when the conductivity consistently approaches 2 micromhos/cm. Make up water is added manually, when needed, from the Rowland Hall deionized water system. A conductivity probe is provided so make-up water can be checked before a substantial amount is added to the pool. Components are described briefly below.

Pump	Centrifugal, self priming type, 1.5 hp, close coupled, plastic impeller and housing, ceramic mechanical shaft seal.
Filters	Three replaceable fiber cartridges 25 micron rating.
Pressure gauges	Before and after filter gauges measure the drop across the filter as an aid in determining the extent of filter clogging. Pressure gauges are also located at the entry to each demineralizer bed to provide indication of possible clogging.
Demineralizer	2 3 cubic foot tanks, for a total of 6 cf, in a parallel flow system. Each contains mixed cation and anion resin (initially in H and OH forms) for ion removal and stabilization of pH.
Conductivity meter.	Measures the conductivity up and downstream of the demineralizer as a test of its efficiency. Switched sensor in make-up water delivery line. Temperature compensation adjustment.
Flow meter	Range 0 to 30 gallons per minute. Normally 15-20 gpm.
Water surface skimmer	Collects foreign particles that float on the surface Water at the surface flows over the top on the floating portion of the skimmer. Clenaed manually when clogged.

6. ENGINEERED SAFETY FEATURES

6.1 Introduction

The following considers additional details of the features of the reactor and building included in the design to ensure that the reactor may be operated under specified conditions with no hazard to the health and safety of the operator, other occupants of the Physical Science complex, or to the general public.

The reactor is to be located in a large building with an estimated daytime occupancy of some 1,000 persons. Other buildings, and student residential areas are nearby. Thus an attempt is made to eliminate as many of potential nuclear hazards as possible. The choice of the TRIGA® system with its inherent prompt shutdown mechanism is the most important factor in this respect.

Factors considered are features relating to reactivity limitations, experimental limitations, and normal operating parameters relating to radiation levels and releases.

6.2 Reactivity Considerations

Reactivities of fuel elements in various positions are given in Table 6-1. The maximum insertion in the case of an improper procedure such as the insertion of a fuel element into an empty position in the B-ring with the core in a critical condition would be $1.16. This would result in a step insertion of only $0.16 prompt critical, so a pulse of no hazard would occur.

TABLE 6-1

REACTIVITY OF FUEL ELEMENTS IN VARIOUS POSITIONS IN CORE

Grid Position (Ring)	Worth ($)
B	1.16
C	0.96
D	0.71
E	0.47
F	0.31
G	0.21

The reactivity worths of the control rods as of 2/10/99 are given in Table 6-2.

TABLE 6-2

ESTIMATED CONTROL ROD NET WORTH

Position	Rod	Worth ($)
C 4	Regulating rod with fueled follower	$2.79
C ring	Shim rod with fueled follower	$3.70
D ring	Adjustable transient rod - void follower	$1.82
F ring	Fast transient rod - void follower	$0.70
		$9.01

The total worth gives a shutdown margin of over $6 over the normal excess loading of $3.0. With the maximum worth rod stuck out the reactor remains shut down by $5.3.

With a core loading of $3.0, the maximum reactivity that can be inserted in step manner is $3.0 above cold, or only $2.0 prompt reactivity. This has been shown to be entirely safe for the TRIGA® core. The combination of core loading limits, reactor control rod drives and control systems with interlocks is designed to prevent any accidental insertion even of such magnitude.

6.2 Production and Handling of Radioisotopes in Experiments.

Neutron activation in the reactor can produce large quantities of radioisotopes which may have intense radiation. Subsequent handling must be supervised by individuals experienced in the detection and evaluation of radiological hazards. The reactor staff will be so trained and will supervise all handling within the reactor area. Outside this area, isotopes come under the control of the Radiation Safety Committee and State of California license authority.

It may be calculated that at 250 kW the reactor could produce about 2,000 curies of Cobalt-60 in the rotary specimen rack. However, the activity would be distributed between the 40 sample positions, with two sample containers in each position. Thus the maximum that could be withdrawn at one instant is about 25 curies. Such a source is intense but may be handled by competent personnel.

A further possible source of intense radiation is an irradiated fuel element. The equilibrium radioactivity associated with a single element irradiated at 250 kW is estimated to be about 20,000 curies at shut-down. Such an element may be safely stored under water and the UCI pool is provide with storage racks for such storage. In the unlikely event that prompt removal is necessary, a fuel element cask may be utilized, and a hoist has been installed for this eventuality.

The risk of release or radioactivity by breakage of an irradiated sample can be reduced by careful design of handling procedures and attention to details of irradiation, such as the integrity of sample containers and the effect of radiation heating on a sample. Such criteria are examined closely during experiment approval. The use of non-porous flooring in the facility is to aid in preventing long-term contamination. Supplies for emergencies are reviewed in Chapter 11.

6.3 Features to Address Operational Radiation Levels (for Accidents see Chapter 13)

6.3.1 Direct Radiation Exposure

The depth of pool water, and a thickened over tank ceiling are design features aimed at reducing dose rates to facility personnel and the surroundings. The dose rates measured directly over the core at various locations with the reactor operating at 250 kilowatts are given in Table 6-3. Because of the short time duration, integrated exposures during a full power pulse (up to 1000 megawatts peak) are equivalent to approximately 1.0 min. of steady state 250 kilowatt operation, and are thus negligible.

TABLE 6-3

MEASURED DOSE RATES DURING FULL POWER (250 KILOWATT) OPERATION

AT LOCATIONS DIRECTLY ABOVE CORE CENTER

Location	Approximate Dose Rate (millirad/hour)
Background in Building [redacted]	0.02
Pool Surface	2.0
Ceiling	0.15 to 0.4*
Floor of Room Overhead	0.04 to 0.1*

* Varies as Argon-41 level in room approaches equilibrium over several hours. Lower value is early in short operation.

These values are in good agreement with those predicted (in the 1968 SAR) prior to reactor operation.

We predict a maximum possible annual individual exposure as follows:

Mean dose level = 0.07 mr/hr – 0.02 (background) = 0.05 mr/hr = 0.05 mrem/hr (for radiation filtered through 17 inches of concrete).

Run time per year = 176 hours full power equivalent (Section 1.3.2)

Thus an individual could obtain an annual exposure of 0.05 x 176 = 8.8 millirem (0.09 mSv).

This estimate is conservative because to receive this exposure the individual would have to:

a) lie on the floor over core center

b) time occupancy to coincide with all reactor operation at full power

The ceiling thickness is thus more than adequate in meeting 20.1301 requirements for operation up to 10 times this value, either in power level or hours, or a combination thereof, as planned.

For operational exposure in the reactor room, an individual accumulates an exposure from working at the pool surface during all hours of reactor operation of:

176 hr x 2.0 millirem/hr = 252 millirem/year.

This exposure represents a measured composite of direct radiation and radiation from gas cloud immersion (see below) as measured with a single monitor device. Since the isotopes involved (argon and nitrogen) do not have specific organ interactions, this may be taken to represent a reasonable value for TEDE. This is therefore predicted to be approximately 1/20th of the exposure limit established by 10 CFR 20.1201 for adults. Thus a combination of operating hours or power levels up to 20 times this value will meet requirements, even for the highly unlikely individual present on the reactor bridge during all hours of operation.

Very slight neutron leakage has been detected from this operating reactor above the rotary specimen rack drive shaft equivalent to 0.5 ± 0.2 mrem/hr at 250 kW steady state power. No measurable neutron exposure can be found elsewhere in the facility. The only exposure would be to a person's extremity because of the physical location. No person's extremity would be in this area with the reactor operating for more than a few minutes per year. This potential exposure has not been included in the above estimate.

6.3.2 Production And Release Of Radioactive Gases

The UCINRF has two distinct sources for the production of radioactive gases: (1) air dissolved in the pool; (2) irradiation of air present in experimental facilities. In 1968, estimates were made of the contributions to airborne release from the facility of these two sources. Since operation commenced actual measurements have been made to ascertain the true levels. Provisions have been made at this facility to minimize personnel exposures from radioactive gases.

6.3.2.1 Nitrogen-16

O-16 captures fast neutrons to form N-16 by way of the (n,p) reaction. O-16 is present in air and in the water itself, so the pool water, as well as air spaces in the reactor are potential sources of N-16. However, because the in-core cooling water is the only area significantly exposed to neutrons with above thermal energies, this is the main source. N-16 has only a 7 second half life, so while it can contribute to radiation exposure of individuals in the facility, essentially none reaches stack level, some 90 feet away. UCINRF has a "diffuser" consisting of a return water nozzle from the reactor cooling water system directed across the top of the reactor core. When the cooling water is circulated, this stirs the tank and markedly increases the rise time of nitrogen gas in the pool, in relation to its half-life. In consequence there is a substantial reduction in the radiation level observed in the room when the system is on. In implementation of ALARA, this system is actuated except for low power runs, or operation at full power of only a few minutes.

6.3.2.2 Argon-41

In the UCINRF only the pneumatic tube system, the central thimble, and the rotary specimen rack contain air. There are no permanent beam tubes or exposure rooms. ^{41}Ar is produced by neutron capture by ^{40}Ar in the air and is the most significant hazard during normal operation. As a noble gas with a short half-life, argon-41 contributes to internal and external exposure from cloud submersion, but not to any committed dose.

Experience with this and other TRIGA reactors has shown that no detectable release of ^{41}Ar occurs from the rotary specimen rack or the central thimble during full power operation because there is no significant "flushing" of these regions. A small amount is released from the pool water.

The pneumatic tube (PT) systems discharge directly into the reactor room exhaust duct. The concentration in the diluted discharge of the main PT system has been measured to be 6×10^{-8} microcuries/mL with the reactor operating at 250 kilowatts. The flow rate of this discharge is 1.2×10^{8} mL per minute. Measurements were also made of ^{41}Ar over the pool water after the reactor had been operated for several hours. Based on dilution estimates this is found to add less than 1×10^{-8} microcurie per mL. Thus the estimated maximum release rate at the exhaust stack of the facility during pneumatic transfer system operation is 7×10^{-8} microcuries per mL, and with reactor operation without the PT in use is $< 1 \times 10^{-8}$ microcuries/mL.

For comparison, the 10CFR 20 Appendix B values for ^{41}Ar are:

Table 1, Col. 3: DAC (inhalation)	**= 3×10^{-6} microcuries/mL**
Table 2, Col. 1: Air	**= 1×10^{-8} microcuries/mL**

In terms of occupational exposure, the individual in the reactor room is in a concentration at the DAC during prolonged reactor operation. Since neither occupancy, nor operation, is continuous, TEDE from this source is acceptable.

6.3.2.2.1 Argon-41 Dispersion in the Atmosphere

It was calculated above that the maximum release of ^{41}Ar at the stack is at a concentration of 7×10^{-8} μc/mL.

To estimate the hazard to the community of such a constant source of gaseous activity is to assume a Gaussian form for the plume dispersion[1], certain values for the diffusion parameters and a mean wind velocity corresponding to a certain meteorological condition. This method tends to give a conservative account since it is improbable that such parameters will remain constant over long periods of time. Any variation will tend to reduce the dose received at any particular location away from the source.

In order to obviate some uncertainty, an attempt is made to establish a lower limit of exposure by placing a $CaSO_4$/Dy environmental dosimeter pack in the effluent stack to represent the exposure that an individual at that location would receive. The pack is exchanged and read quarterly. Within the limits of error, the readings over many years have been consistently within the expected background variation (approximately 10 mrem computed annually). This meets the requirement of 20.1101 (d) without invoking dilution from mixing beyond the stack.

As an alternative at UCINRF, computations are made each year based on the conservative

[1] F. A. Gifford, Nuclear Safety, V. 2, No. 2, Dec. 1960, p. 56, and Nuclear Safety, Vol. 2, No, 4, July 1961, p. 47.

assumption that all reactor runs are at full power. The number of minutes of PT operation is recorded, and the full annual production of ^{41}Ar computed from the measurements listed above.

For example, in 1998, 85 hours of equivalent full power operation, and 441 minutes of PT operation at full power were assumed. The resulting calculated stack release is 9.3 x 10^3 microcuries in a volume of 6.3 x 10^{13} mL to yield an average concentration of 1.5 x 10^{-10} microcuries/mL Constant submersion in a hemispherical cloud of that concentration would yield an individual exposure of 0.7 millirem. This is considerably below the desired limit of 10 millirem (10CFR 20.1101(d) and would not be measurable by means of the dosimeters used.

These estimates assume submersion at the point of release. Application of a Gaussian Plume model with stable wind conditions, and point of release at 86 feet (26.2 m) above ground, gives values that are well below the 10 CFR 20, Appendix B, Table 2, column 1 value of 1 x 10^{-8} microcuries/mL. This value is attained only during operation of the pneumatic transfer system, typically for less than 500 minutes per year. Regular operation gives less than 1/7th of the value.

6.3.2.2.2 Concentration of ^{41}Ar at the Air Intakes of Rowland Hall.

The safety of persons inside the building must also be considered. The possibility of contamination of the air taken into the main building by the reactor effluent is considered here.

The important factors are:

(1) One air intake is at the south-west corner of the first floor level of the building, some 70 feet below the exhaust level and up-wind to the prevailing south-southwest daytime winds. The other is down wind at the northeast corner, but 86 feet below the Penthouse level.

(2) In addition to the reactor room exhaust, a large number of fans exhaust on to the Penthouse level. On the west side, over 48,000 cfm is exhausted on a 24 hr basis. The east side is similar. Thus a volume of air about 100,000 cfm will be exhausted at most times. When compared with the reactor exhaust of 4,324 cfm this air provides dilution by over a factor of ten - to well below the 1 x 10^{-8} DAC value as air leaves the roof area.

(3) The roof exhausts in Rowland Hall are predominantly fume hood vents from chemistry research and teaching laboratories, so efforts are taken to avoid air recycling to meet OSHA standards. Because of the intense odor of certain chemicals, failure of the system is readily detectable! Special vertical constrictions have been added to all roof exhausts (including those from the reactor) to create greater vertical velocity on calm air days. All hoods and vents are operating 24 hours a day. As a result all switches that could be used to turn off fan motors by individual laboratory users have been disabled, and alarms added to many hoods to alert personnel if flow is reduced below optimum values. All the air systems in Rowland Hall are tested on a regular (at least annual) basis to assure proper function.

These changes and policies help to assure that air from the reactor ventilation is diluted and kept away from this and other local buildings.

7. REACTOR INSTRUMENTATION AND CONTROL SYSTEMS

7.1 CONTROL CONSOLE AND NEUTRON DETECTORS

All the functions essential to the operation of the reactor are controlled by the operator from a desk-type control console in the control room (Fig 7-1). The instrumentation contained in the console is connected to the control rod drives, the facility interlock system, and various detectors positioned around the reactor core. Three commercial analog neutronics modules (with digital readouts) manufactured by Gamma-metrics, Inc., are incorporated:

(1) Wide Range Linear Monitor (fission chamber signal input)

(2) Wide Range Monitor (CIC signal input)

(3) Power Range Monitor (UCIC signal input)

These channels cover the power ranges indicated in Fig. 7-2.

A cabinet adjacent to the console contains control and readout for the remote area monitor and other alarm indicator lights.

7.1.1 Control Console

The meters, switches, and recorder used to operate the reactor are mounted in the console as follows:

1. A console power (POWER ON) switch;
2. A three-position OPERATE key switch;
3. Five sets of three control-rod adjustment switches (UP and DOWN) and indicator lights (ON and CONT) for the shim(SHIM), regulating (REG), and adjustable transient (ATR) control rods (two unused sets are provided);
4. One TRANSIENT ROD FIRE switch;
5. Three LCD ROD POSITION indicators, one each for REG, SHIM, and ATR postions;
6. A MODE SELECTOR switch;
7. A WIDE RANGE LINEAR (WRL) Monitor with associated meter, range indicator, MANUAL/AUTO/250 kW mode switch, and REMOTE/LOCAL switch. In the adjacent rack are the test/calibrate controls for this channel. When in REMOTE, the readouts are on the control console face.
8. A WIDE RANGE (WR) Monitor with digital LEVEL PERCENT and PERIOD SECONDS indicators, bar indicators for these and trip settings, test and calibrate controls.
9. A POWER RANGE Monitor with digital LEVEL PERCENT, PEAK POWER (MW) and ENERGY (MW-sec) indicators, bar indicators for above and trip settings, and test and calibrate controls.
10. A dual-pen strip-chart recorder with one pen recording LOG power (WR output), the other recording LINEAR power (WRL output).

11. A %-DEMAND control for servo control setting;
12. A manual SCRAM bar;
13. A FUEL TEMP. and inlet/outlet WATER TEMP meter and selector switch;
14. A digital POOL WATER TEMPERATURE meter indicating in degrees Celsius;
15. Annunciators for SCRAMS (7) , SOURCE (interlock), COOLING, PURIFICATION, and LAZY SUSAN.
16. ARM switches and indicators (ARM and UP) for the ATR and FTR (fast transient rod) air solenoid drives.
17. MAGNET current supplies and indicator modules (not yet activated/installed)

Additional details of functions of these control and indicating devices are as follows:

The console POWER ON switch controls primary ac power to all circuits except the ±25 vdc power supply, clock, and ion chamber power supply. The power supply and clock, which are left on even though the console is not in operation, are controlled by the circuit breaker on the rear center door.

Current to the magnet power supply and recorder chart drive motor is controlled through the three-position OPERATE key switch.

The control-rod adjustment switches, UP, DOWN, and combination CONT/ON (contact/on) switches are provided for the shim and regulating rods. UP and DOWN switches are also provided for the adjustable transient rod cylinder. In addition to rod-adjustment switches, a FIRE switch and ARM switches are also provided for the two transient rods.

The positions of the regulating, shim, and adjustable transient rod drive cylinder relative to their lower limits are registered on separate digital indicators mounted on the control panel.

A three-position mode selector switch is located in the lower left corner of the central control panel. The switch is used to select any one of the three modes of operation for the reactor: STEADY-STATE (manual or AUTOMATIC); and pulsing operation (PULSE).

The dual-pen strip-chart recorder is centrally located in the upper center portion of the console. One pen provides a linear indication of power and the other provides a log power trace.

The %-DEMAND control is used in conjunction with the squarewave (optional) and steady-state automatic control servo.

A SCRAM bar provides a means of manually scramming the reactor.

7.2 CONTROL AND SAFETY INSTRUMENTATION

Indications of control-rod positions, power levels, fuel temperatures, and rate of power changes are necessary in the operation of the reactor.

A simplified block diagram illustrating the basic circuitry of the reactor's control and safety instrumentation is shown in Fig. 7-3. The individual control and safety channels are described briefly in the following paragraphs.

7.2.1 Control-rod Drive Switches and Circuits

The control-rod circuit consists of the switches and indicating devices used in operatinq the two standard (rack-and-pinion) control-rod drives. The six illuminated pushbutton switches in the circuit are arranged in the center of the control panel in two vertical rows, one row for each control rod. Each row of switches contains a white DOWN switch, a red UP switch, and a blue and yellow double-pushbutton CONT/ON (contact/on) switch.

Illumination of the various switches signifies the following conditions:

DOWN switch: The DOWN light indicates that the control-rod and drive are at their lower limits.

UP switch: The UP light indicates that the controlrod drive is at its upper limit.

CONT/ON switch: The CONT side of the double pushbutton switch indicates con~act between the control-rod assembly armature and the control-rod drive electromagnet The ON side ~f the switch indicates that the electromagnet power is on The magnet power circuit is energized by a key switch located on the left side of the control panel.

When the double CONT/ON pushbuttons are depressed, magnet current is interrupted and the ON lights are extinguished. If a control rod is above the down limit, the rod falls back into the core and the CONT light is extinguished until the automatic magnet-drive-down control lowers the magnet and contact is again made with the rod. Releasing the button closes the magnet circuit and magnet current is restored.

7.2.2. Adjustable Transient Rod Drive Switches and Circuits.

The adjustable transient rod may be used as a steady-state control rod as well as a transient rod. The pneumatic cylinder can be driven up or down by depressing the appropriate switch on the console. The associated position indicator reads out the position of the pneumatic cylinder. The UP and DOWN pushbuttons become illuminated when the cylinder reaches the respective limits of travel.

To fire the adjustable transient rod the FIRE button must first be armed by depressing the adjustable transient rod ARM switch to the right of the recorder. This completes the circuit between the FIRE button and the solenoid-operated air valve. One-half of this ARM button will become illuminated when the FIRE button is armed; the other half is illuminated whenever the control rod is not in the full down position. (After the FIRE button has been armed and pushed.) The control rod may be reinserted into the core by pushing the ARM button a second time.

For steady-state operation, the pneumatic cylinder is driven to its full down position and air pressure is applied under the piston. Thus, when the pneumatic cylinder is driven up, the control rod is also raised. The system is interlocked so that in the steady-state mode, air cannot be applied until the cylinder is in the full down position.

For pulsing and square-wave operation the cylinder is adjusted to the position required for a particular reactivity insertion and the control rod is ejected by applying air pressure under the piston. In the PULSE mode the solenoid-operated air valve will automatically return to the vent position, permitting the control rod to drop back into the core by gravity. The timing of the automatic return is adjustable up to 10 sec.

7.2.3 Fast Transient Rod Switches and Circuitry

The fast transient rod may be operated as a safety rod as well as a transient rod. In either case, the drive must be armed in the same fashion as the adjustable transient drive only using the fast transient rod ARM switch. To use the fast transient rod as a safety rod during steady-state or square wave operation, the adjustable transient rod and all steady-state control rods must first be in the full down position. Then the fast transient rod can be withdrawn by pushing the ARM button and then the FIRE button. The fast transient rod may be reinserted (without effecting the position of the other rods) by pushing its ARM button a second time.

For pulsing when both transient rods are to be used, the rods are individually armed by depressing their respective ARM buttons. They are then both fired together upon depressing the FIRE button.

The reactivity insertion of the adjustable transient rod takes place in about 100 msec, whereas the reactivity insertion of the fast transient rod takes place in about 50 msec. To prevent pulse clipping, the end of the reactivity insertions of both rods should occur at the same time. Coordination of the two transient rods is provided by means of a delay circuit. This delay circuit is adjustable between 20 msec and 100 msec.

A magnetic sensor is provided on each transient rod drive at the end of the reactivity insertion. The signal generated by these sensors, when the drives are fired, can be displayed on a scope during set-up operations. The delay circuit can then be adjusted so that the displayed signals occur at the same time interval after depressing the FIRE button.

7.2.4 Wide Range Monitor (WR) and Period Channel

The WR channel receives its input from a fission counter through a special low noise cable into a low-noise preamplifier. The channel converts the signal to logarithmic information over a range of 10^{-8} to 200 percent of full power. An adjustable trip level is provided to assure sufficient neutrons are presne for start-up. This is set at 1×10^{-7} percent, corresponding to about 10 nv. The trip level can be read on a bargraph. A digital (E-format) readout is provided as well as an LCD bargraph indicator over the full range. An output is directed to the console recorder (red pen) to provide a continuous record of Log Power.

The WR channel also provides a digital period readout (-30 seconds to + 3 seconds) together with a bargraph. An adjustable period trip level is also provided and displayed by bargraph (currently set at 3.2 seconds). Test and calibrate modes are provided for each output.

A NON-OPER trip is initiated during calibration or if detector High Voltage is disabled. Detector Highh Voltage may be read at a front panel test point.

7.2.5 Wide Range Linear Power Channel

The Linear Power channel consists of a compensated ion chamber coupled to a Wide Range Linear (WRL) Monitor installed (because of physical limitations) in a rack adjacent to the main reactor console. In order to provide clear indications to the reactor operator, the power level displays from this unit are duplicated in the left facing control console panel, together with the MODE selection switches. This panel is activated by use of a REMOTE/LOCAL switch, normally left in REMOTE. Test and calibration controls are provided on the main (rack mounted) unit front panel as are readouts of the high voltage and compensation voltage applied to the detector.

During steady-state operation, the WRL channel provides linear power level indications from below source level to full power. The linear power circuit consists of a neutron-sensitive compensated ion chamber, amplifiers, a digital range indicator and meter readout. In AUTO mode the unit switches ranges automatically up at about 80% of range and down at about 20% of range (except in the top range). A MANUAL mode can be selected in which the operator selects the range. A 250kW mode can be selected if it is desired to use the channel as a percent of full power channel. A lighted mode indicator is provided on the console. A servo output is provided for automatic reactor flux control, but is not presently in use/connected.

An adjustable trip is provided on all ranges to initiate reactor scram if the linear power is allowed to rise beyond a percentage (currently set at 105%) of scale reading. A NON-OPER trip is actuated if the high voltage fails, or the unit is being calibrated. An additional adjustable bistable trip is provided that can be adjusted over the entire range. This is used for a "PULSE" interlock which prevents activating the pulse mode circuitry for the reactor if the measured linear power level is too high. This is currently set to trip at about 750 watts.

An output signal from the WRL is used to drive the blue pen of the recorder to provide continuous record of (linear) power level. No range information is transmitted to the recorder.

7.2.6 Power Range Channel

The Power Range Channel operates consist of an uncompensated ion chamber and a Power Range Monitor. The current from the chamber can be measured from 1.56×10^{-6} amps to 1×10^{-3} amps and provides indication from 0 to 125% power on a digital indicator and a bargraph. An adjustable trip is currently set at 109% and is also indicated on a bargraph. In the pulse mode. the circuit can accumulate nv over time, determine the maxium (peak power) level (with four operator-selected ranges: 2000, 1000, 500 and 200 MW), and store a value for accumulated (nvt) MW-seconds at 2 secs and/or 10 seconds into the pulse with a 50 MW-sec upper range. Associated calibration and testing circuits are included.

A NONOPER trip actuates if high voltage failure occurs. The detector high voltage can be measured at a front panel test point.

7.2.7 Fuel Temperature Thermocouples and Meters

A fuel thermocouple is connected to the WATER/FUEL TEMP. meter mounted on the front of the control panel. This meter indicates fuel temperature during all modes of reactor operation; a scram circuit is associated with this circuit to scram the reactor on high fuel temperature. Test and calibrate circuits (which place simulated thermocouple signals onto the meter circuit) are provided.

7.2.8 Water Temperature Monitoring Channels

Reactor pool water temperature is monitored in the reactor tank by means of a stainless steel clad thermistor probe connected to a standard commercial temperature monitor circuit with a digital readout in degrees Celsius. This value is constantly displayed. The probe measures the water temperature at about 2 feet below the water surface.

An additional readout is provided on the cooling water circuit controller in the reactor room. This is also a thermistor probe measureing pool temperature a few inches below the water surface but at the opposite side of the pool.

Water temperatures can also be determined by the console operator in the water-cooling circuit by using a spring loaded selector switch which enables readout on the fuel temperature meter. Tip-sensitive nickel wire detectors are located in the inlet and outlet piping from the reactor tank and connected to a bridge circuit.

7.3 INTEGRATION OF CONTROL AND SAFETY CIRCUITRY

During reactor operation, the functions of the individual control and safety channels are intimately related. In this section, each allowed mode of reactor operation is briefly described, and operator interaction and safety concerns addressed. Block diagrams illustrating the integration of the reactor control and safety channels with regard to specific operating modes are also provided.

7.3.1. Steady-State Operation - Manual and Automatic

Manual and automatic steady-state reactor control is used for reactor operations from source levels to power levels up to 250 kw (see Fig. 7-4). This mode is used for manual reactor startup, change of power level, and steady-state operation. Three power level channels described above provide indication at all levels. Two recorder traces are available indicating log and linear power. The linear power range is indicated by a lighted digital display on the Wide Range Linear Monitor console display panel. Trips are enabled on all three channels to scram the reactor if excessive power is indicated on any one. Period indication is available which can alert the operator to unanticipated rapid changes in flux.

Automatic control of the reactor is permissible at power levels from 1 w to 250 kw. Automatic control involves the use of flux controller servo signal used to control the position of the regulating rod.

Fuel temperature is displayed on the WATER/FUEL TEMP. meter and a scram set point is maintained well below any fuel safety limit for operation.

7.3.2 Transient (PULSE) Operation

This mode is used to produce short duration pulses of high peak power. During transient operation, the high voltage is lowered to the fission chamber and the compensated ion chamber. The uncompensated ion chamber operayting with the Power Range Monitor is used. An interlock is provided that prevents firing of the transient rods if the reactor power is above 1 kw.

A block diagram illustrating the integration of the control and safety circuitry for transient operation is shown in Fig. 7-7.

Fuel temperature continues to be monitored in this mode and a fuel temperature scram is obtained if the fuel exceeds a preset (steady state) temperature on the meter.

7.4 SAFETY DEVICES (SUMMARY)

7.4.1 Scrams

1. Wide Range Monitor channel, fission chamber, 0-100% power.
2. Wide Range Linear Monitor, compensated ion chamber, 0-100% of any range.
3. Power Range Monitor, uncompensated ion chmaber, 0-100% power.
4. Manual SCRAM bar.
5. Detector high voltage supply failure, provided on each channel.
6. Console power failure.
7. Seismic switch, set approx MM III motion.

7.4.2 Interlocks

1. To assure minimum source strength before control rods can be withdrawn.
2. To prevent withdrawal of two control rods simultaneously.
3. To ensure that pulsing cannot occur with reactor power greater than 1 kw.
4. To prevent application of air to the fast transient rod in the steady state mode unless all other rods are fully inserted.
5. To prevent application of air to the adjustable transient rod in the steady state mode unless the cylinder is in the 'down' position.
6. To prevent movement of any control rod except transient rods in pulsing mode.

7.4.3 Failure Mode Analysis

An analysis of possible modes of failure of the reactor scram circuits and the control rod drive circuits is presented in Appendix IV.

8. ELECTRICAL POWER

UCINRF is supplied with routine electric power through a UCI system that has at least two major feeds to campus to avoid complete power loss. The reactor is an extremely small component of power use in Rowland Hall and the campus. The provisions for dealing with loss of this power in the facility are described in Section 13. 8. However, the design of this type of TRIGA® reactor does not require electric power to be maintained in a safe shutdown condition.

UCI has a strong interest beyond the reactor in maintainance of stable power to the science buildings. Much expensive equipment and samples rely on close and constant temperature control. For this reason maintenance of electric power and/or rapid restoration of power following any outage is a very high priority independently of any concerns regarding the nuclear reactor. The University of California system is such a large organization that is able to negotiate reliable and economical power supply in the California electricity market.

9. AUXILIARY EQUIPMENT

9.1 Underwater lights

Illumination of the reactor core is provided, when needed, by sealed beam lights encased in water-tight housings supported from the aluminum channel at the top of the tank.

9.2 Fuel element storage racks

[redacted]

9.3 Fuel Handling Tool

[redacted]

9.4 Fuel-Element Inspection Tool

The fuel-element inspection tool is used to accurately inspect a fuel element or control rod with fuel follower for longitudinal growth and for bowing in excess of 0.062 in.

The upper support plate of the tool is mounted with two 1/2-in. bolts on the aluminum channel at the top of the reactor tank and extends downward 12 ft into the tank, permitting the inspection of an irradiated fuel element and providing approximately 9 ft of shielding water over the element. The tool can be installed at different locations around the pool to facilitate measurement of all core elements one at a time. All parts of the tool in contact with water are either aluminum or stainless steel. The aluminum support-tube structure has a hole at the bottom end and another at the top to allow water to fill the interior of the pipe.

The bowing of a fuel element is detected by a carefully machined cylinder (a go/no-go gauge) attached to the bottom of the tool. If a fuel element will slide completely into the cylinder, its bow, if any, is less than 0.062 in. If the element passes through its cylinder, it will come to rest on a lever arm.

The length of the fuel element is measured by pushing downward (approximately 10 lb of force is required) on the indexing rod until the indexing plug shoulders on the indexing plate This places the upper surface of the fuel-element triangular spacer at an indexed position common to all fuel elements measured. Pushing the fuel element downward to this position forces the lower lever downward an amount that varies with the length of the fuel element being measured. This lever actuates a push rod extended to the top of the unit and its displacement is measured by a dial indicator.

A "standard" dummy element was furnished with the inspection tool; it is a solid piece of aluminum with the same dimensions and the same top and bottom end fixtures as those on a regular fuel element. This standard must be inserted to calibrate the tool. The dial indicator is zeroed with the standard in the device. Every fuel element in the core can then be measured and its length compared with that of the standard. Careful records are kept of each fuel element, grid location, and length, compared with the standard. Instrumented (thermocouple) fuel elements can also be inspected with this tool. By using a different indexing plate, a longer go/no-go gauge, and a "standard" dummy fuel follower standard element, the fuel follower control rods can also be inspected.

Fuel elements that do not pass the elongation and bowing tests will not be used for further operations. The Technical Specifications establish agreed limits (bowing in excess of 0.062 in. and longitudinal growth in excess of 0.10 in.) based on past engineering studies and prudent factors beyond which elements must be removed from the core.

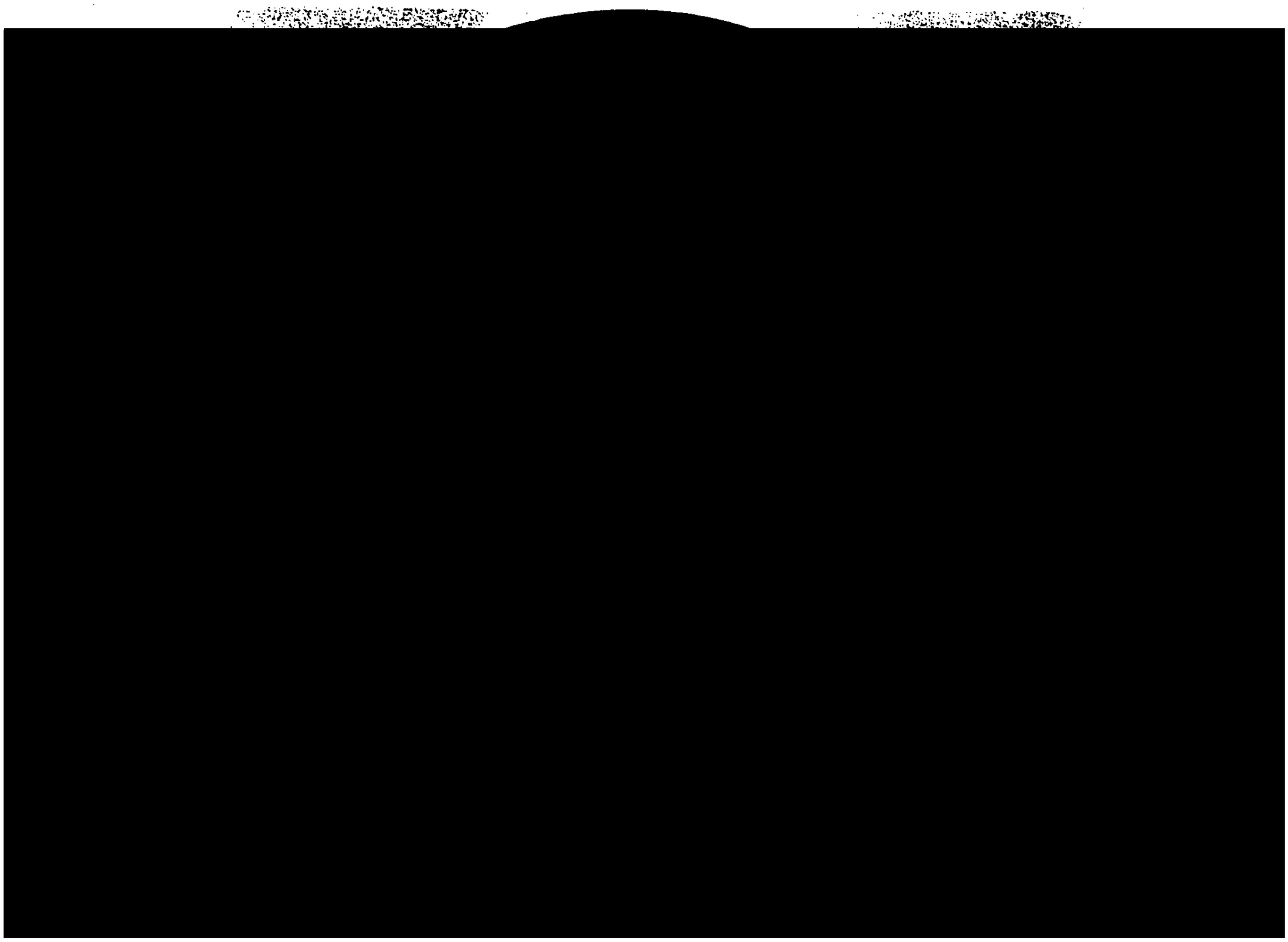


FIG. 9-1 Fuel Element Handling Tool

10. IRRADIATION FACILITIES

10.1 In-Pool and In-Core Facilities

Specimens may be exposed by placing them in the reactor pool adjacent to the core in watertight containers. Higher fluxes may be obtained for small specimens by using positions within the core created by adjustment of fuel element positions.

10.2 Rotary Rack Facility (Lazy Susan)

The rack (Figs. 10-1 and 10-2) supports 40 aluminum tubes which serve as receptacles for 1.12 in. diameter specimen tubes. The rack can be rotated manually or electrically, the drive on the bridge being transmitted through a shaft to a stainless steel sprocket and chain drive contained within the watertight aluminum housing. The assembly rests in a well in the graphite reflector. The drive shaft contains plastic to prevent radiation streaming up to the bridge.

Specimen containers are inserted and removed through a 1.33 in. I.D. tube extending from the rack housing to the bridge. This tube is offset with large-radii bends to avoid direct line radiation streaming from the core (see Fig. 4-l).

The drive assembly on the bridge includes a 40 position indicator dial, a crank for manual rotation, and a motor and slip clutch for electrical rotation. The locking handle must be lifted to rotate the rack. At the desired rack position, the locking handle is lowered locking the specimen rack in place. After each position is loaded in this way, the locking handle is latched up and the rack rotated electrically to provide uniform exposure of specimens.

Specimens are removed by means of a weighted retrieval unit that catches in the top of specially designed sample containers and is raised and lowered on a string or wire.

Useable space in the specimen container is about 2 cubic inches, in a cylinder 3.81 in. long by 0.8l in diameter. Two such containers may be inserted in each rotary rack position.

10.3 Pneumatic Transfer Systems

A moderate speed pneumatically operated transfer system (Fig. 10-3) extends from a receiver-sender station in the laboratory adjacent to the reactor room into a terminus in the core. A blower exhausts air from the system through a filter into the reactor room exhaust duct and the vacuum is used to draw samples into or out of the in-core terminus. The terminus is located in one of the holes in the grid and is supported by the bottom grid plate. The specimen capsule will come to rest vertically at about the mid plane of the core. A timer and control is located at the send-receive station. The air flow is controlled by a solenoid valve; after crossing the filter it is fed into the main reactor room exhaust duct. A permanent area radiation monitor is located at the send-receive station.

Fast pneumatically operated transfer systems are similarly constructed and installed, except the maximum sample capsule size is much smaller. These are operated by compressed nitrogen gas

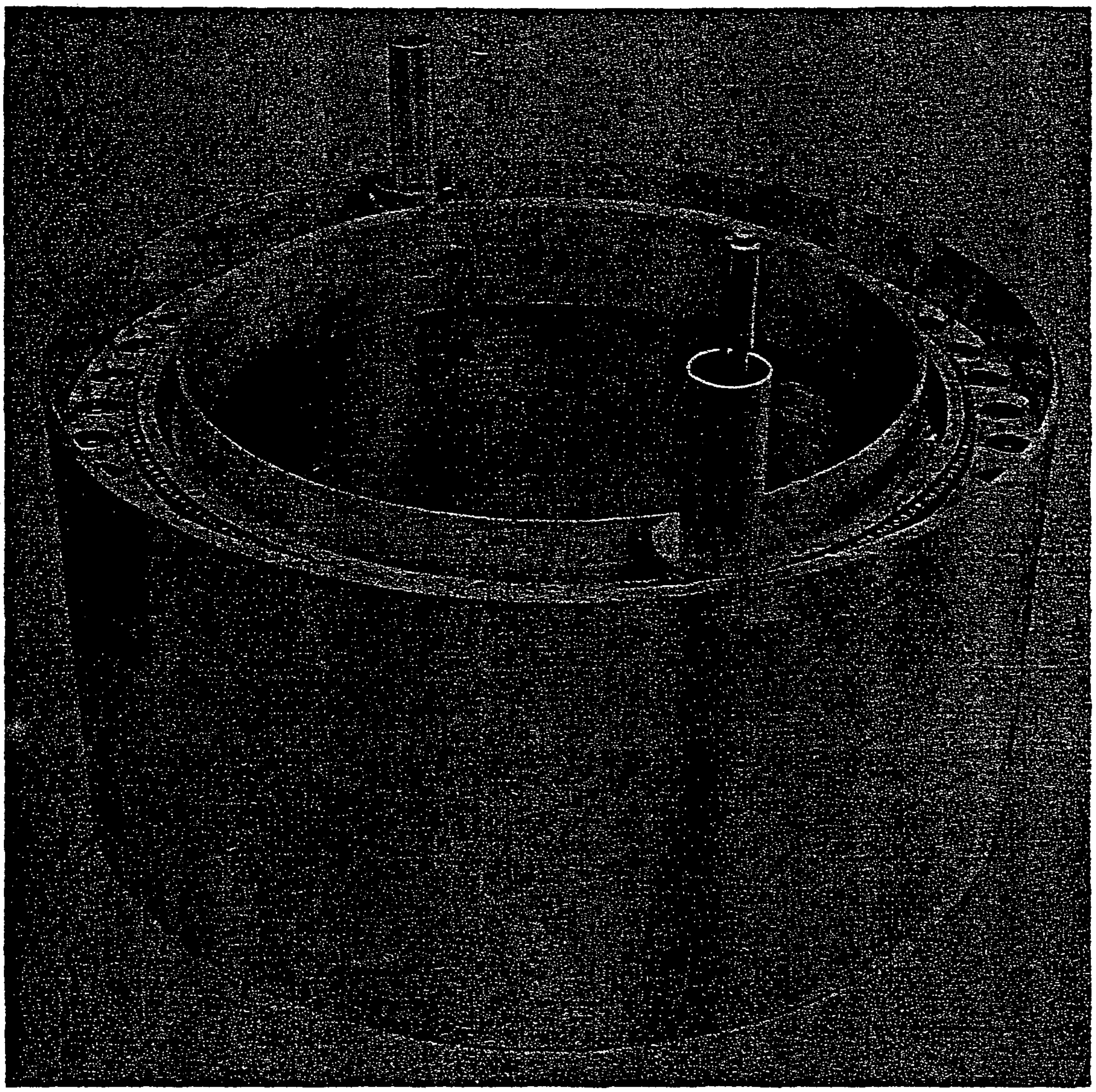


FIG. 10-1 Rotary Specimen Rack (Lazy Susan) Assembly

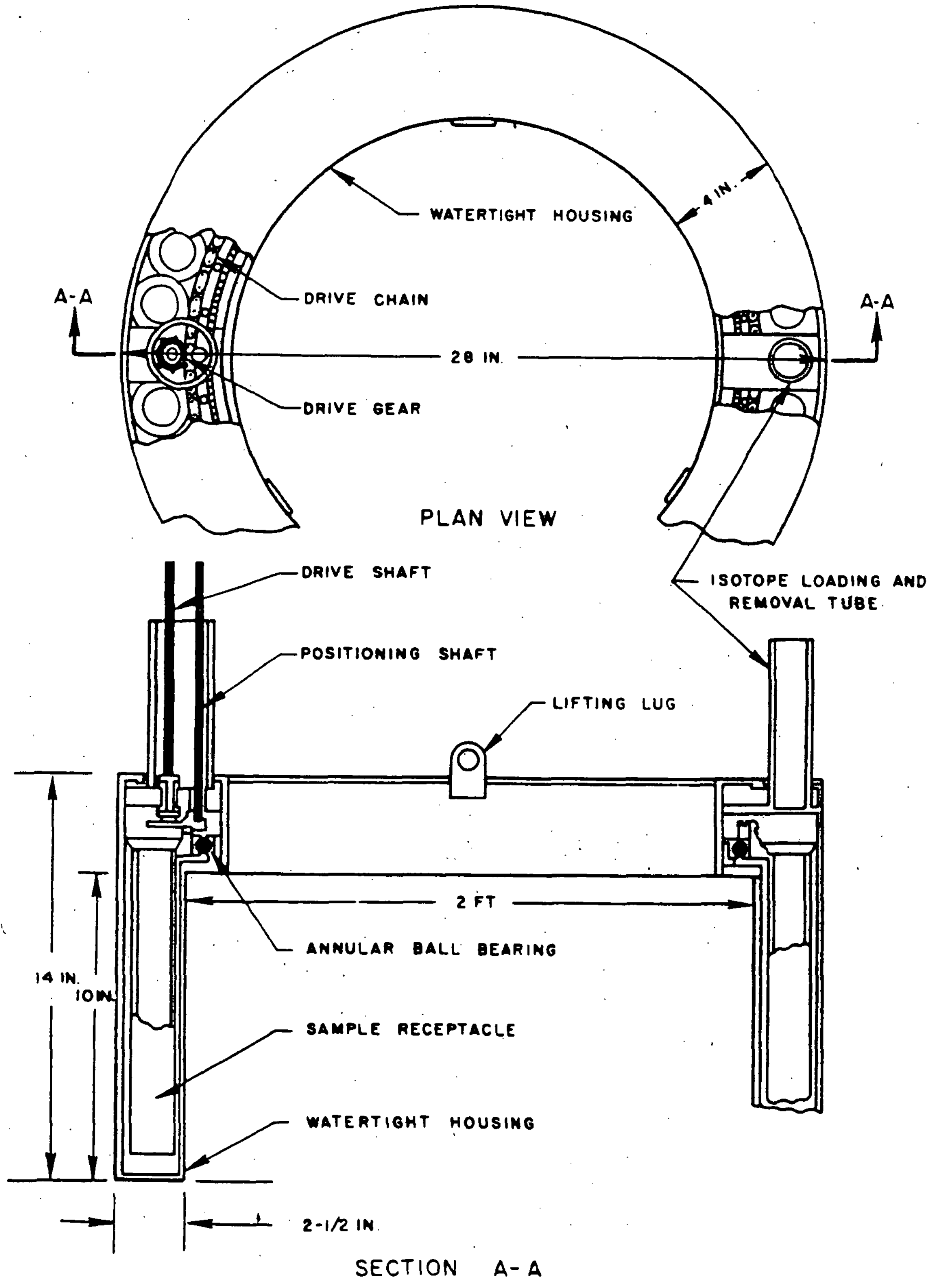


FIG. 10-2 Schematic of Rotary Specimen Rack Assembly

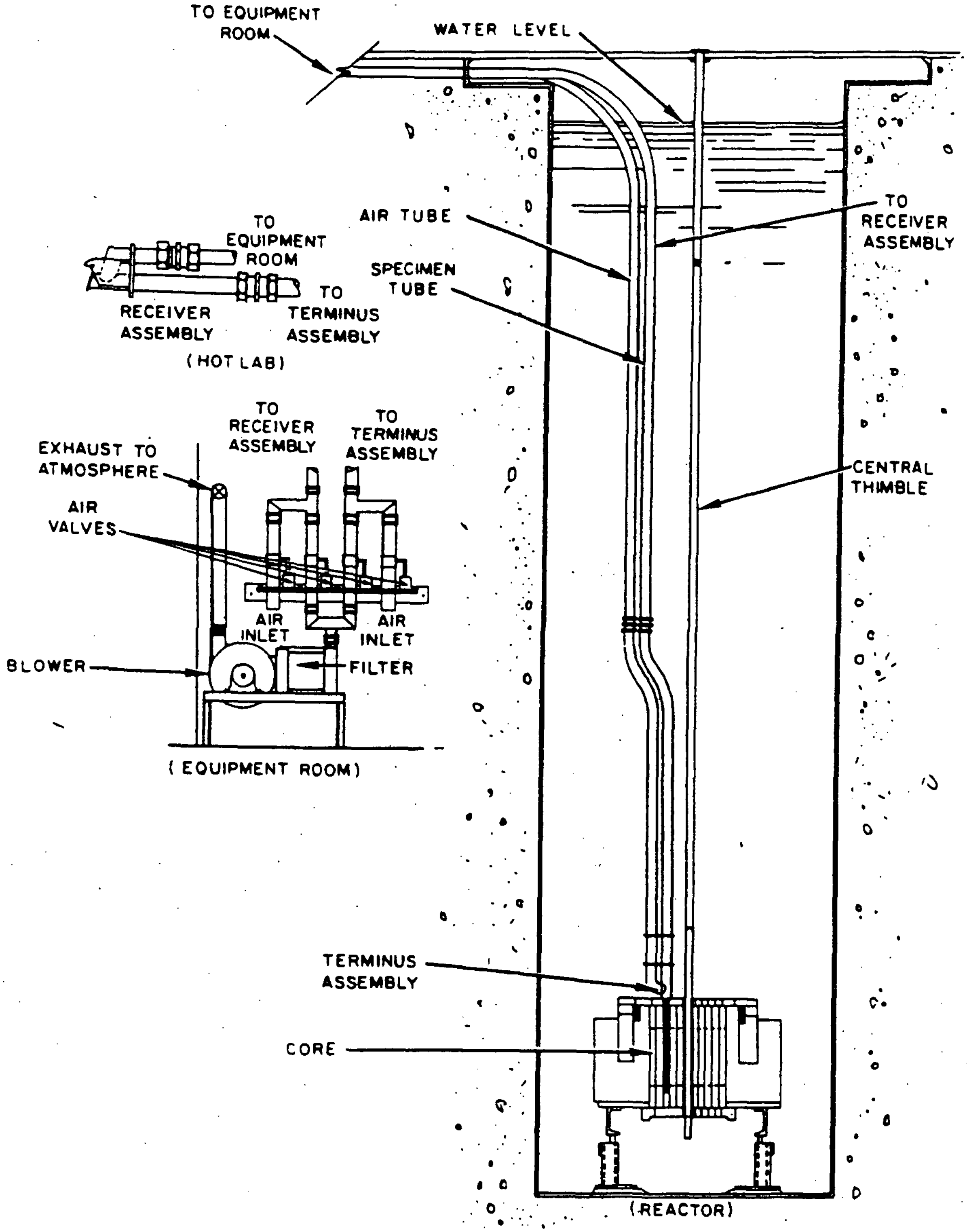


FIG. 10-3 Pneumatic Transfer System

using solenoid control valves and a send-receive unit is located in the reactor room, so they may be attached directly to detector systems. One of the transfer systems has thin cadmium sheet surrounding the in-core terminus, but separated from the pool water by aluminum tubing. All systems are fixed so they cannot be inadvertently removed from core whilst the reactor is in operation.

10.4 Central Thimble

The central thimble, located at the center of the core provides space for the irradiation of small samples at the point of maximum flux. The thimble is an aluminum tube 1-1/2 in. outside diameter, with a wall thickness of 0.083 in. It extends from the bridge where it is bolted to one side, with a curved tube to enter through the central grid plate holes, terminating with a rod at the lower end which rests on the safety plate. The tube is constructed so samples rest at approximately core center-line. The curves in the tube prevent radiation streaming to the room.

A padlock is maintained on the entry funnel cap to this device so that samples cannot be placed in this location without operator permission.

11. RADIATION PROTECTION AND WASTE MANAGEMENT PROGRAM

UCINRF relies heavily on the UCI Radiation Safety Program, approved by the State of California and set forth in the UCI Radiation Safety Manual. This program incorporates by explicit reference, applicable portions of Title 10, Code of Federal Regulations; California Health and Safety Code, Division 20; and California Administrative Code, Title 17. Ultimate responsibility for this program rests with the Chancellor of UCI. The Reactor Supervisor is the responsible Principal Investigator for the Radiation Use Authorization (RUA) issued by the UCI Radiation Safety Program to permit work in the facility.

The program within the facility has a number of components:

- fixed area and particulate monitors to warn personnel of unusual circumstances
- portable monitors for use by experimenters and others handling samples
- personal dosimetry for experimenters (including UCI students in classes), staff and visitors
- regular surveys to uncover unusual radiation levels
- wipe tests on a regular schedule to warn of removable radioactive contamination
- environmental monitoring
- program surveillance by EH&S staff using both measurements and record inspection
- waste management
- maintenance of emergency supplies
- Standard Operating Procedures for operation, maintenance and calibration activities
- Standard Operating Procedures establishing action levels and response to unusual findings and occurrences

The records of monitoring activities for both normal and unusual are retained at either the reactor facility, or EH&S, or both.

11.1 Fixed Monitoring

These instruments have been described fully in Section 3.7.

11.2 Portable Monitors

At least one portable monitor measuring beta-gamma radiation at low levels (G-M type) and one measuring higher levels (air filled ion chamber type) are kept in working order in the control room except when in immediate use in the facility use. These instruments are calibrated at least annually by EH&S, who also provide back-ups when these instruments need repair.

11.3 Personnel Dosimeters

Personnel monitoring for experimenters (including students in laboratory courses using the facility) and staff is under the direction of EH&S who provide a commercial beta-gamma TLD badge service, exchanged quarterly. These are supplemented in the reactor area by pocket dosimeters which are also used for occasional visitors. Tour groups are admitted of up to 30 individuals measured by at least 3 randomly distributed dosimeters. No exposure above zero level has ever been recorded for such a group at this facility.

11.4 Regular Area Surveys

Regular Area Surveys for Beta-Gamma levels are conducted in locations and frequencies depending on workload and likelihood of creation of areas with significant dose rates above background. The locations and frequencies are detailed in facility Standard Operating Procedures.

Occasional (approximately annually) surveys of the facility are conducted for neutron and overall gamma dose rates to assure that the facility profile is as originally measured shortly after start-up.

11.5 Wipe Tests for Removable Contamination

100 cm^2 area wipes are made in over 15 key locations in the facility on a regular basis. Additional wipe surveys are made independently by EH&S on a quarterly basis. Beta sensitive assay is used in both cases.

11.6 Environmental Dosimetry

$CaSO_4$/Dy chips in special commercial environmental packs are placed in 10 locations at UCI and changed quarterly. Results from reading of these by the commercial supplier are presented in each facility annual report. The locations have been chosen to survey the immediate confines of the facility, and to survey more remote locations on campus to establish baselines for comparison in the event of a release incident.

11.7 Oversight of UCINRF Radiation Protection Program by EH&S.

As mentioned in the introduction, EH&S has several (at least three full time) staff with radiological safety experience. This staff and other part-time staff provide constant oversight/surveillance of the program. This includes examination of all commercially read dosimeters, review of facility records, radioisotope inventory control, and direct supervision of unusual shipments. Overall performance is reviewed annually as part of the required renewal program for the RUA for the Reactor Supervisor which covers all work with radioisotopes and radiation in the facility.

11.8 Waste Management Program.

11.8.1 Low level Radioactive Waste

All low level radioactive waste created as a result of facility operations (such as filters and resin) and experiments (gloves, discarded samples, contaminated apparatus) is transferred from the facility to the EH&S radioisotope waste handling program. While waste remains within the facility it is kept in specially marked containers, available for both liquid and dry waste, that are specified for use by the campus Radiation Safety Program. Records of transfers are maintained by EH&S and the facility. At the present time, all waste is retained for decay at UCI under the supervision of the **State of California** license to the campus.

11.8.2 High Level Radioactive Waste (spent fuel)

No spent fuel has so far been transferred from UCINRF. The fuel is the property of the US Department of Energy and an agreement exists that when no longer needed, such fuel will be transferred to the Idaho National Engineering and Environmental Laboratory (INEEL) for eventual disposal. Planning for receipt of this fuel is complete and described in an extensive document dated July 18th 1997 reference EDH-14-97 (Appendix C). This plan identifies the year 2029 as final use year, with 2032 for shipment.

11.9 Emergency Supplies

Emergency supplies, mostly relating to radiological incidents are maintained in at least three separate locations. These supplies are checked at least annually for inventory control and for possible deterioration of components. Supplies include such items as first aid supplies; rolls of: absorbent paper, paper towels, masking tape; coveralls and booties; mops; spill control materials (water absorbing); glass fiber and charcoal filters for air samplers; flashlights and batteries, radiation monitors.

Supplies are maintained in:

- the reactor control room for immediate use
- a closet across the hallway from the reactor control room, in Rowland Hall, but accessible without entering the facility (see Fig 3-2)
- an emergency response vehicle (kept at the EH&S waste site)

In addition, assay equipment, and additional supplies are maintained at the EH&S laboratories and waste handling facility (soon to be moved to a new campus location).

11.10 Standard Operating Procedures (SOP)

Standard Operating Procedures are maintained for all operation, maintenance and surveillance procedures utilized at the facility. Changes in these procedures can be made by the Reactor Supervisor with the approval of the ROC. Additional requirements are imposed by the campus RUA. Changes that relate to aspects of the UCI Radiation Safety Program must also be approved by the campus RSO and the RSC.

12. CONDUCT OF OPERATIONS

This section describes the basic organization and requirements to operate the UCINRF as a basis for Section 6 of the Technical Specifications. (See Section 14). Details are found in the Standard Operating Procedures at the facility. The organization is designed to promote efficient handling and performance of experiments and yet to establish adequate safeguards for the health and safety of the community. Each unit in the network is discussed briefly. Also discussed are criteria for the examination of proposed experiments and for ensuring safe operation of equipment. The suggestions are not meant to be limiting and are subject to reconsideration by the appropriate administration in the light of further operational experience.

12.1 Organization

The reactor is operated within the School of Physical Sciences at the University of California, Irvine (UCI). All UCI units including the Dean of Physical Sciences and the Office of Environmental health and Safety report ultimately to the UCI Executive Vice Chancellor, who is delegated by the Regents of the University and the Chancellor of UCI to act as the authorized license manager for the US Nuclear Regulatory Commission on their behalf.

The current administrative organization is shown in Fig. 12-1. Each component is described in more detail below.

FIG 12.1 ADMINISTRATIVE ORGANIZATION OF UCINRF

School of Physical Sciences

Reactor Operations Committee

UCI Office of Environmental Health and Safety

Reactor Administrator

UCI Radiation Safety Officer

Reactor Supervisor

Reactor Operations

12.1.1 Reactor Administrator

The Reactor Administrator has the ultimate responsibility for the facility meeting USNRC codes and guides. In addition he will select a responsible and competent person as Reactor Supervisor. He has delegated implementation of operations to the Reactor Supervisor.

12.1.2 Reactor Supervisor

The Reactor Supervisor is the operational administrator of the facility--an permanent faculty or staff member of the University holding a Senior Operators License. He is directly responsible to the Reactor Administrator for the running of the facility; and thus enforces administrative rules and operating procedures, is responsible for all planning and scheduling and reviews and classify all proposed new experiments and modifications to the reactor and submits them to the Reactor Operations Committee (ROC) for approval. He consults directly with both the ROC and the Radiation Safety Committee (RSC) concerning all safety aspects of reactor operation. Within the reactor facility he will enforce compliance with the requirements of the NRC and the RSC and be responsible for monitoring operations. All other reactor operators and trainees are under his or her control.

The Reactor Supervisor is responsible for the operation, maintenance and scheduling of all equipment within the facility including the gamma-irradiation source.

A clear succession system is established in the Standard Operating Procedures and the Emergency Plan when a need for action arises in the absence of the Reactor Supervisor.

12.1.3 Reactor Operations Committee (ROC)

This Committee is established by the University to advise the Reactor Administrator and Reactor Supervisor regarding reactor operation. Members are persons having experience in reactor or radiation physics, medical physics or chemistry. The Reactor Supervisor and the campus Radiation Safety Officer are members ex-officio.

The Reactor Supervisor reports to this Committee on any problems concerned with the reactor. In particular, the approval of this Committee is required for start-up of the reactor following major modification or repair. He must be able to satisfy the Committee at all times that the reactor is in good operating order. The Committee may forbid operation if it does not believe that everything is in order for safe operation.

All Class II experiments (vide infra) and facility modifications must be approved by this Committee.

12.1.4 Radiation Safety Committee (RSC)

The radiation Safety Committee is a university administrative committee serving by appointment to the UCI Executive Vice Chancellor, that is responsible for the administration of the Radiation Safety Program at UCI. It has legal responsibility to the State of California for the safe use of radiation and radioisotopes at UCI. It meets with the campus Radiation Safety Officer on a regular basis to perform its review function. All operations involving isotopes inside and outside the reactor facility are subject to its scrutiny. The Reactor Supervisor is a permanent ex officio member of this committee. This committee is chaired by a UCI Academic Senate faculty member whose research work involves expertise in the use of radioisotopes. The regulations of this committee are enforced within the reactor area by the Reactor Supervisor under the surveillance of the staff of the Radiation Safety Office.

12.1.5 UCI Radiation Safety Officer

An experienced Health Physicist is on the staff of the Office of Environmental Health and Safety. He is responsible for the radiological health and safety of the community. Working through the Reactor Supervisor he will ensure the safety of operations in the reactor area and is responsible for the transfer of isotopes outside the Chemistry Department. He is continually available for consultation with the Reactor Supervisor and in the event of any emergency. He is a member, ex-officio, of both the RSC and the ROC. He has additional staff members trained in radiological safety who form a trained emergency response team and can be additional consultants regarding safe operations with radioisotopes.

12.1.6 Reactor Operators

All reactor operators are licensed by the USNRC to operate this reactor. Most operators at this facility have become qualified as Senior Operators. Any Senior Operator may temporarily assume the duties of Reactor Supervisor. In addition to their reactor specific training and certification, all operators are trained in radiation safety and authorized as part of the UCI radiological safety program.

12.1.7 Reactor Assistant Staff

Assistants are used at the facility to assist with routine surveillance and/or experimental procedures. They are trained and authorized in the use of radiation and radioisotopes as part of the UCI Radiation Safety Program.

12.2 Administrative Controls And Procedures

12.2.1 Introduction

This section describes the aspects of procedures pertinent to the safe and secure operation of the facility and the performance of experiments. The Reactor Supervisor is responsible for ensuring compliance with the established controls.

12.2.2 Access to the Reactor Area

Normal access to the reactor and its associated laboratories is through the control room. Possession of both a door key and a key card are needed to enter the facility. Further details of this entry control process are in the facility Security Plan. Authorization for unescorted access is controlled by the Reactor Supervisor who retains and issues all keys and key cards to the facility. Persons entering the reactor room are required to carry personal dosimeters or to be part of a non-working (visitor) group assigned dosimeters. The double doors to the loading dock are normally closed and locked. Emergency exit may be made through these doors or through the control room. Persons who are authorized to access at any time are required to have a knowledge of safety and emergency procedures within the reactor area. Emergency instructions require a member of the reactor staff or the E. H. and S. Officer shall be contacted prior to entry by Police or Firemen except in cases of extreme life-threatening emergency, when they may break in.

12.2.3 Reactor Operation

An Operator-in-Charge is designated for each reactor operation to establish responsibly for assuring that the reactor and the facility equipment are operated in accordance with Technical Specifications and Standard Operating Procedures. He or she is also responsible for assuring that all other personnel in the facility follow practices in accord with UCI radiation safety program and other safety requirements. The Operator in Charge shall specifically determine that:

(a) he or she is in a position to operate the control of the reactor at all times the reactor is critical.

(b) the correct start-up and/or shut-down procedures are followed, Technical Specification conditions for operation are met, and that log entries are made for all operations.

(c) any experiment is correctly authorized and that any requirements have been complied with.

(d) the experimenter's procedure conforms to RSC recommended practice and the experimenter is working within a valid Radiation Use Authorization (RUA) covering work in them facility.

(e) all samples removed from the reactor are monitored, their activity levels recorded, and any necessary temporary access barriers or shielding are used to minimize personnel exposure.

(f) the experimenter and the Reactor Supervisor are informed in case of any unusual or unexpected incident, apparent equipment or instrument failure or malfunction.

(g) radiation safety assistance is summoned if unexpectedly high radiation levels are experienced.

The Operator will normally satisfy most of (c) and (d) by ensuring that the proposed experiment and schedule has been correctly approved. All members reactor operators are expected to be experienced in basic radiation safety procedures, so that adequate safety is assured.

12.2.4 Routine Monitoring and Test Procedures

A program is established for regular surveillance testing of all safety equipment, procedures and certain reactor components. Frequencies are established for key items in the Technical Specifications (Section 14). A reactor start-up checklist assures that these have been accomplished and that items especially related to safe operation are operational prior to the reactor becoming critical.

Any malfunction of safety related systems shall be sufficient cause for suspension of reactor operation until the fault is corrected.

The results of all surveillance tests are recorded in log book kept in the control room.

12.2.5 Authorization of Experiments

All experiments proposed for the reactor are designated Class I or Class II by the Reactor Supervisor. In addition, procedures must be approved by the RSC as part of a UCI Radiation Use Authorization(RUA).

Class I experiments include all repeats of successful experiments and those that are but minor modifications to a previous experiment. The Reactor Supervisor and any Senior Reactor Operator has the authority to approve and schedule these experiments. Most irradiations (*vide infra*) are Class I experiments.

Class II comprises all other experiments which must be submitted to the Reactor Operations Committee for review and approval.. Experiment requests will normally be submitted to the ROC by the Reactor Supervisor after evaluation by him and at least one other ROC member. The experimenter is responsible for obtaining or modifying an RUA, if not already authorized.

Certain criteria are used in experiment evaluation. These criteria shall include, but not necessarily be limited to the following:

a. An experiment shall mean any addition of apparatus, devices or structure to the core or experimental facilities which is not a normal part of these facilities.

b. An irradiation shall mean the placement of a movable sample, in a container, in an existing experimental facility designed for such purpose.

c. The sum of the absolute reactivity worths of all experiments in the reactor and samples in the irradiation facilities shall not exceed 2.1% δk/k ($3.0). This figure includes the total potential reactivity insertion which might arise from experiment malfunction, accidental flooding or voiding, accidental insertion or removal of samples or experiment devices or structures.

c. A conceivable failure of the experiment shall not lead to direct failure of any core element or control rod, or any other experiment.

d. A fueled experiment shall be limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 1.5 curies and the maximum Strontium-90 inventory does not exceed 5 millicuries.

e. Where the possibility exists that the failure of an experiment (other than a fueled experiment) could release radioactive gases or aerosols to the reactor room, the quantity of material shall be limited such that the airborne radioactivity concentration will not exceed the limits set by Title 10, CFR 20, Appendix B, Table 1 for ALI or DAC assuming 100% of the gases or aerosols escape.

f. In consideration of the requirements of par. e. if either a hold-up tank, closing automatically at high radiation level, or a filter installation designed to retain >0.3 micron particles with an efficiency of >99%, is provided, then the assumption may be made that only 10% of the gases or aerosols in the first instance, or of the aerosols only in the second, can escape.

g. For materials with boiling point above 55°C and where vapors formed by boiling the material could only escape through a column of water above the core, the assumption may be made that only 10% of such vapors can escape.

h. Known explosive materials such as gunpowder, dynamite, TNT, or nitroglycerine shall not be irradiated in quantities greater than 25 milligrams. In addition the pressure produced in the experiment container by detonation of the explosives shall be calculated or experimentally demonstrated to be less than the design pressure of the container.

i. In the event of any failure or release of material occurring which might damage the reactor in any way, the reactor shall be physically inspected to determine the consequences and need for corrective action before proceeding with any further experiments.

12.3 Operator Training

Operator training at this facility has been under the direct supervision of Senior Reactor operators, and carried out in accordance with appropriate regulations in 10 CFR Part 55. Only when trainees have demonstrated adequate background knowledge following training in radiological safety practices will they be permitted to commence operational training at the control console. Trainees will be expected to participate in facility surveillance and maintenance over a period of time in order to gain further familiarity with the facility components and operating procedures. Trainees will be expected to pass written examinations covering those areas identified in 10CFR55.41 and 55.43 and prepare for Senior status.

12.4 Licensed Operator Requalification Program

The objective of the requalification program is to assure continued safe operations at the facility by maintaining adequate familiarity with facility equipment and normal operating and emergency procedures. Emphasis shall be placed on those areas key to routine and emergency operations.

The program should take into account the special nature of operations at this facility and the size and skill and experience of the operating staff. Each cycle will be over a two year period.

Operators are expected to maintain familiarity with the facility by performing frequent control manipulations during each six month period that the reactor is operational, including startups and shutdowns, and operation at full licensed power. Observation of these activities by a Senior Reactor Operator designated as the examiner may constitute an operational examination to meet requalification provided documentation of successful performance is made.

The nature of staffing and educational opportunities at this facility (no nuclear engineering program exists, nor is required by regulations) makes a formal lecture program impossible. Licensed operators will review and discuss appropriate categories of information at least once a year in order to meet this requirement. Such reviews will especially focus on facility changes and surveillance and maintenance. Review of emergency and safety practices will occur through participation in annual emergency drills. A written examination will be taken every two years in the required subject areas as long as at least two licensed SROs are at the facility. One SRO will be designated as the examiner, and others the examinees in any given cycle.

13. ACCIDENT ANALYSES

13.1 INTRODUCTION

NUREG-1537 calls for nine categories of typical accidents to be addressed. Those that have potential for UCINRF are discussed in some detail, others are briefly reviewed. General issues regarding the safety of TRIGA® reactors have been the subject of several reports over the last 30 years. The most recent and complete summary is presented in the SAR (revision 2) for the McClellan Nuclear Radiation Center (MNRC) where accident considerations for a 2 megawatt TRIGA® with mixed fuel loads are detailed.

The specific scenarios considered are:

13.2 Maximum Hypothetical Accident (MHA) - rupture of a single fuel element cladding in air.

13.3 Insertion of excess reactivity.

13.4 Loss of coolant.

13.5 Loss of coolant flow.

13.6 Mishandling or malfunction of fuel.

13.7 Experiment malfunction.

13.8 Loss of normal electricl power.

13.9 External events.

13.10 Mishandling or malfunction of equipment.

13.2 Maximum Hypothetical Accident - Rupture of Single Fuel Element in Air.

The MHA for TRIGA® reactors has been assumed to be the release oif fission products from an element whose cladding has been stripped, in air. The problem with such a hyppothesis is that assumptions maust be made about the immediate prior history of the element in order to establish an appropriate temperature to use. If the element ruptured as a direct result of reactor operations, the most likely scenario would be during a pulse, when elements arew subjected to the maxiumum stress. However, this could only happen under water, since the reactor cannot reach operational reactivity, let alone pulse insertion, if the core contains substantial air. Thus one conclusion is that the most probable hypothesis is that the cladding becomes damaged during fuel handling out of core, when its temperature would be the result of residual decay heating only.

We first examine the conclusion that residual heating and fission product gas pressure build-up by itself is unlikely to result in cladding rupture.

13.2.1 Integrity of Fuel Element Cladding - Likelihood of Rupture

The calculation below shows that rupture of fuel element cladding is not likely following loss of cooling water from the tank. The calculation assumes that the reactor has been operating at 250 kw for a very long time prior to loss of the water which is considered to be lost instantaneously following reactor shut-down. The maximum temperature reached by the fuel, and consequently the stainless-steel cladding, is less than 160°C. This temperature is such that the pressure exerted by trapped air and fission product gases is less than 30 psi. This pressure produces a stress of 985 psi, whereas the yield stress for the stainless-steel cladding is about 23,000 psi at 160°C. Therefore, the fission products will be retained in the fuel elements. The same conclusion has been reached at TRIGA® facilities operating at much higher pwoer levels.

It should also be noted that although the only mechanism considered herein for heat removal after the water loss is through natural convection of air through the core, some heat will be removed by conduction to the grid plates, and some will be removed by radiation.

Method of Calculation

Use was made of a two-dimensional, transient heat transport computer code, entitled RAT, developed at Gulf General Atomic, for calculating the maximum temperature in the core after a water loss.

It was assumed that at the time at which the water was lost, the temperature distribution in the fuel element considered in this calculation was equal to the average temperature in the hottest (B ring) fuel element during steady-state operation of the reactor at 250 kw.

It was also conservatively assumed that the reactor had been operating for a very long time at 250 kw, and with only 62 elements in the core. The rate of energy release in the B ring element was determined from consideration of the energy deposition of fission product gammas and betas only. The energy release from delayed neutrons is relatively small (about 1700 watt-sec total in the B ring element) and has an average decay constant of about 0.08 sec^{-1}.

The after-shutdown power density (in Btu/hr-ft^3) in the B ring fuel element is given by:

$$\frac{q}{v} = 0.1p\frac{P}{V_f}\cos\left[0.78\frac{\pi}{L}\left(x-\frac{L}{2}\right)\right]\left\{\left[t+t_o+10\right]^{-0.2} - 0.87\left[t+t_o+2x10^7\right]^{-0.2} - 0.053\right\} \quad (1)$$

where

p = Peak-to-average power density in the core = 2.0

P = Operating reactor power = 8,525 x 10 Btu/hr (250 kw)

V_f = Volume of the fuel in the core = [redacted] ft^3

L = Length of the fuel = [redacted]

x = Distance measured from the bottom of the fuel element, ft

t = Time after the core is exposed to the air, sec

t_o = Time from shutdown to the time the core is exposed, sec

Equation (1) is a modification of the Untermeyer-Weill formula that matches the work of Stehn and Clancy[1] to about 5 x 10^4sec after shutdown. It is also conservatively assumed that all the energy produced by fission product decay in the element is deposited in the element.

While the decay gammas and betas are raising the fuel element temperature, the flow of air between the fuel elements will be removing heat, tending to lower the fuel temperature. The air velocity through the channel can be determined by setting the frictional pressure loss equal to the buoyancy. Entrance and exit losses will be about 2 to 5% of the friction losses and have been ignored.

$$\delta P \text{ (buoyancy)} = \delta P \text{ (friction)} \qquad (2)$$

The term on the left is given by

$$\delta P \text{ (buoyancy)} = (\rho_o - \rho_1)\frac{L}{2} \qquad (3)$$

where L is the length of the channel and ρ_o and ρ_1 are the entrance and exit air densities, respectively.

Since the frictional pressure drop calculations for laminar flow in non-circular channels are incorrect when expressed in terms of the hydraulic radius, the pressure drop for the TRIGA® reactor must be predicted by other means. The method selected was to convert the free-flow area into an annulus around the fuel element. With an annular space of inner diameter D_1 and outer diameter D_2, the frictional pressure drop becomes[2]

$$\delta P \text{ (Friction)} = \frac{32\mu vL}{g\left[D_2^2 + D_1^2 - \frac{D_2^2 - D_1^2}{\ln(D_2/D_1)}\right]} \qquad (4)$$

where
- μ = Viscosity of air, lb/hr-ft
- v = Velocity of the air, ft/hr
- L = Length of the fuel element, ft
- D_1 = Fuel element diameter, ft
- D_2 = D_1 + 2b, ft.

The term b in D_2 should be the effective separation distance between the B ring fuel element and those in the C ring. The use of the minimum separation distances as b would yield too large a pressure drop and is considered too restrictive. The use of the average separation distance, based on the free flow between the B ring element and the C ring, would yield too low a pressure drop

[1] J.R. Stehn and E. F. Clancy, "Fission Product Radioactivity and Heat Generation," Paper No. 1071, Proc. Second United Nations Con. on the Peaceful Uses of Atomic Energy, Geneva, 1958.

[2] W. H. Adams, "Heat Transmission", McGraw Hill, New York, N.Y. 1954, p. 149.

since the pressure drop is not a linear function of the separation distance. As an approximation, b is taken as the mean of the two values, that is

$$b = \frac{1}{2}(0.01130 + .04930) = 0.0303 \text{ ft.} \tag{5}$$

For the channel between the B ring and the C ring, the equation for the pressure balance (2) becomes

$$(\rho_o - \rho_1) = 6.88 \times 10^{-5}\,\overline{\mu v}, \tag{6}$$

where $\overline{\mu}$ is the average viscosity of the air (lb/hr-ft) in the channel and is a function of the entrance and exit temperatures, $\overline{v}$ is the average air velocity (ft/hr) in the channel, and the entrance and exit densities are ρ_o and ρ_1(lb/ft^3).

The mass flow rate of air in the channel is

$$w = \overline{v\rho}A_C \tag{7}$$

where $\overline{\rho}$ is the average density of air in the channel, and A_c is cthe flow area associated with the channel. Combining equations (6) and (7), one obtains the mass flow rate

$$w = 405.(\rho_o - \rho_1)\,\overline{\rho}\ /\ \overline{\mu}. \tag{8}$$

Assuming that the average properties in equation (8) are the average of the properties at the entrance and exit, equation (8) becomes

$$w = 405.\frac{(\rho_0^2 - \rho_1^2)}{(\mu_0 + \mu_1)}. \tag{9}$$

Over the range of temperatures of interest the properties of air have been approximated by linear equations. Thus,

$$\rho = \frac{1}{2.5x10^{-2}T} lb/ft^3$$

and (10)

$$\mu = (0.0113 + 0.6017x10^{-4}T) lb/hr - ft$$

where T is the temperature in °Rankine.

Using these expressions in equation (9) one obtains

$$w = \frac{3.24x105(T_1^2 + T_0^2)}{(T_1^2 + T_0^2)[0.01135 + 0.30085x10^{-4}(T_1 + T_0)]} lb/hr. \tag{11}$$

The determination of the amount of heat removed by this air flowing past the element rests on the evaluation of a heat transfer coefficient. For a free-standing cylinder cooled by natural circulation, the heat transfer coefficient is given, conservatively, by

$$h = 0.531\frac{k_f}{L}[Gr\,\mathrm{Pr}]^{0.25} \qquad (12)$$

where k_f is the thermal conductivity of air film at temperature T_f (Btu/hr-ft-°F), Gr is the Grashof number, and Pr is the Prandtl number. Because the fuel element is surrounded by adjacent elements, the flow will probably not be laminar, even at low Reynolds numbers, and the heat transfer correlation should be better than that assumed, perhaps by as much as a factor of 2.

Again, over the temperature range of interest, one can write for the thermal conductivity and specific heat of the air

$$k_a = (0.0009 + 0.26 \times 10^{-4}T)\ \text{Btu/hr-ft-°F and } C_{pa} = 0.240\ \text{Btu/lb-°F} \qquad (13)$$

with temperature, T, in °R. Using these values and those in equation (10) in equation (12) one obtains equation (14):

$$h = 50.16\left[\{0.0009 + 0.13x10^{-4}(T_w + T_a)\}^3 \left\{\frac{[1.25x10 - 2(Tw + Ta)]^{-2}[0.5(Tw + Ta)]}{[0.01335 + 0.30085x10^{-4}(Tw + Ta)]}\right\}^{-1}(Tw + Ta)\right]^{0.25}$$

where T_w is the wall temperature, T_a is the bulk air temperature,

and T_f is the average of the two, all in °R.

For the purpose of this analysis the heat transfer along the graphite end reflectors was neglected, and the fuel element was considered to be a [redacted] cylinder of $U\text{-}ZrH_{1.7}$ with thermal conductivity and specific heat given by:

$$k_f = (10.7 - 6.42 \times 10^{-4}T)\ \text{Btu/hr-ft-°F and } C_{pf} = (26.3 + 0.0245\ T)\ \text{Btu/ft}^3\text{-°F} \quad (15)$$

where T is the local temperature in °R. The temperature drop in the clad was ignored because, at the time of peak temperature, it has been calculated to be insignificant (1 - 5°C). For the computer program, the fuel element was divided into five radial and five axial regions, and the temperature in each region was computed as a function of time after complete water loss.

In Figure 13-1 the maximum fuel temperature calculated is plotted as a function of time after the loss of cooling water. The loss of water in the core was assumed to take place within seconds after shutdown, and the initial fuel temperature is 113°C.

To determine the pressure exerted on the cladding by released hydrogen, fission products, and air trapped in the fuel can, the conservative assumption will be made that the entire system is at peak fuel temperature, i.e., about 152°C.

The total number of fission product nuclei released to the gap between the fuel and clad was determined from Blomeke and Todd[3] and the results of the experiment described in Section 13.7.

[3] J. O. Blomeke and Mary F. Todd, "U-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time", ORNL- 2127.

The total quantity of Br, I, Kr, and Xe released to the gap in the B ring fuel element after continuous operation at 250 kw for about one year will be

$$N_i = 0.028 \text{ x } 6.866 \text{ x } 10^{20} = 1.933 \text{ x } 10^{19} \text{ atoms.} \qquad (16)$$

The number of gram-atoms in the gap is

$$n_{fp} = \frac{1.933 \text{x} 10^{19}}{6.02 \text{x} 10^{23}} = 3.216 \text{ x } 10^{-5} \text{ gram-atoms.} \qquad (17)$$

The partial pressure exerted by the fission products gases is

$$P_{fp} = n_{fp} RT \qquad (18)$$

where, initially, the volume, V, is taken as the 1/8-inch space between the end of the fuel and the reflector end piece. This is quite conservative because the graphite reflector pieces have a porosity of 20% and the fission product gases can expand into the graphite. The initial volume, then, is

$$V = \pi r^2 h = \pi (1.80)^2 \, 0.317 \text{ cm}^3 = 3.23 \text{ cm}^3; \qquad (19)$$

where the radius of the fuel material, r, is about [redacted] and the space width, h, is [redacted] cm).

Thus the initial pressure exerted by all the fission product gases is

$$P_{fp} = \frac{3.216x10^{-5}}{3.23} RT = 0.997 \text{ x } 10\text{-}5 \text{ RT} \qquad (20)$$

The partial pressure of the air in the fuel element is

$$P_{air} = \; = 4.46 \text{ x } 10^{-5} RT, \qquad (21)$$

and the total pressure exerted by the air and fission products is

$$P_r = \left(1 + \frac{P_{fp}}{P_{air}}\right) P_{air} = \left(1 + \frac{0.997 \text{x} 10^{-5}}{4.46 \text{s} 10^{-5}}\right) P_{air} = 1.223 P_{air} \qquad (22)$$

Also $$P_{air} = 14.7 \frac{T}{273} \frac{V_o}{V} \qquad (23)$$

where P_{air} is in ib/in.^{2}, and T is in °K.

The equilibrium hydrogen pressure over the U-$ZrH_{1.7}$ fuel material is simply a function of the fuel temperature. At a temperature of 152°C, the hydrogen pressure is negligible (see Figure 13-2).

The total gas pressure at the maximum fuel temperature of 152°C is

$$P = PH + Pr = 0 + 1.223 \, (14.7) \left(\frac{152 + 273}{273}\right) \frac{3.23}{V} = \frac{90.0}{V} \text{ psi.} \qquad (24)$$

Ignoring the negligible increase in volume caused by the expansion of the clad the total pressure then, is

$$P = \frac{90.0}{V} = \frac{90.0}{V_o} = \frac{90.0}{3.23} = 27.8 \text{ psi.} \qquad (25)$$

The tangential stress in the fuel element cladding when subjected to an internal pressure, P is

$$S = Pr/t, \qquad (26)$$

where r is the radius of the fuel element can [redacted] and t is the wall thickness [redacted] or

S = [redacted] P. (27)

The stress in the cladding is

$$S = 35.4(27.8) = 985 \text{ psi.} \qquad (28)$$

From the United States Steel Company handbook, "Steels for Elevated Temperatures", stress for type 304L stainless steel at 152°C is >23,000 psi.

Consequently it is concluded that, subsequent to the complete loss of cooling water, the release of hydrogen from the fuel and the expansion of air and fission product gases in the space between fuel and graphite end pieces will not normally result in the rupture of the fuel element cladding.

13.2.2 Consequence of Cladding Failure of Single Fuel Element.

The effect of a fuel element rupture resulting in a release of fission products from a B - ring fuel element was evaluated in the 1968 SAR. More recently the issue of fission product release fraction for TRIGA® fuel[4] was reviewd. The value proposed remains the same, so that the earlier analysis reamins valid. The potential release is the product of the Fission Product Inventory and the Release Fraction.

13.2.2.1 Fission Product Inventory

The inventory of fission product gases, halogens and noble gases in the B-ring fuel element in which the average power density is 1.6 times greater than the core average, was determined by assuming infinite operation at 250 kw with 70 elements in the core. Using the data developed by Blomeke and Todd[5] results in the compilation of the volatile fission products given in Table 13-5.

Two groups of gaseous fission products have been recognized. The first comprises bromine and iodine isotopes that will dissolve if water is present, and be airborne if it is not The second group comprises the insoluble volatiles: krypton and xenon isotopes. They are the major source of radioactivity in the room (and outside) if the pool water is present.

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4 USNRC, Safety Evaluation Report on High Uranium Content, Low Enriched Uranium-Zirconium Hydride Fuels for TRIGA® Reactor, NUREG-1282, August 1987

5 J. 0. Blomeke and Mary F. Todd, "U-235 Fission Product Production as a Function of the Thermal Neutron Flux, Irradiation Time, and Decay Time", ORNL-2127, Oak Ridge National Laboratory, August 1957-November 1958.

TABLE 13-5

GASEOUS FISSION PRODUCTS IN ONE B-RING FUEL ELEMENT 250 kw SATURATED

ACTIVITY–70 ELEMENT CORE

	Nuclide	Decay Constant (hr^{-1})	Inventory (curies)
Group I			
	Br 83	3.02×10^{-1}	24.59
	Br 84	1.31×10^{0}	56.47
	Br 84m	6.95×10^{0}	1.163
	Br 85	1.39×10^{1}	76.47
	Br 87	4.49×10^{1}	137.5
	I 129	4.60×10^{-12}	51.24
	I 131	3.58×10^{-3}	148.6
	I 132	3.07×10^{-1}	225.3
	I 133	3.34×10^{-2}	332.3
	I 134	7.93×10^{-1}	389.1
	I 135	1.04×10^{-1}	302.6
	I 136	2.90×10^{1}	158.9
		Total Iodines	1608.
		Total Group I	1904.
Group II			
	Kr 83m	3.66×10^{-1}	24.59
	Kr 85m	1.59×10^{-1}	76.47
	Kr 85	7.67×10^{-6}	15.37
	Kr 87	5.35×10^{-1}	138.9
	Kr 88	2.50×10^{-1}	190.2
	Kr 89	1.31×10^{1}	236.2
	Kr 90	7.55×10^{1}	265.6
	Kr 91	2.54×10^{2}	158.1
	Xe 131m	2.41×10^{-3}	1.486
	Xe 133m	1.26×10^{-2}	7.998
	Xe 133	5.50×10^{-3}	332.3
	Xe 135m	2.67×10^{0}	90.79
	Xe 135	7.60×10^{-2}	245.3
	Xe 137	1.07×10^{1}	30.26
	Xe 138	2.45×10^{0}	281.0
	Xe 139	6.08×10^{1}	290.7
	Xe 140	1.56×10^{2}	305.2
		Total Group II	2690.

13.2.2.2 Fission Product Release Fraction

The recommended formula for release fraction quoted in NURGE 1282 is:

$$\textit{release fraction} = 1.5 \times 10^{-5} + 3600 \times \exp(-13400/T) \text{ with T in Kelvin.}$$

Although the central UCI TRIGA elements may reach peak temperatures somewhat greater than 300°C in a pulse, the release of fission products from the fuel should still be mostly from recoil as this temperature is sustained for a very short time during each pulse. Below 570 K (300°C) the second term describing non-recoil release does not add significantly. Thus we may assume a relase fraction of 1.5×10^{-5} in evaluating the MHA for UCINRF.

Multiplying this release fraction by the inventory of gaseous fission products produced in the fuel, as given in Table 13-5, gives the total activity that would be released should the integrity of a fuel element cladding be compromised. These values are shown in Table 13-6

	Total Curies
Noble Gases	4.0×10^{-2}
Iodine	2.4×10^{-2}
Halogens	2.9×10^{-2}

13.2.2.3 Pool Water Activity

We review briefly the course of events if the rupture did occur in water, in which case the soluble (Group I) halogen products will remain in the water, totalling 2.9×10^{-2} curies. Since the volume of water in the reactor pool is 8.7×10^{7} cm^3, the activity concentration is $3.3 \times 10^{-4} \mu c/cm^3$. In 24 hours the activity would decrease to $5.8 \times 10^{-5} \mu c/cm^3$. The activity remains moderately high because of the small decay constants for I-129, I-131, and I-133. The demineralizer can be used to safely concentrate these for subsequent disposal. 10 millicuries of activity on the resin is a readily handled quantity by trained personnel.

13.2.2.4 Predicted Exposures to Personnel Inside the Reactor Room

The maximum exposure to a person in the reactor room will occur if the fission products from a ruptured element are distributed instantaneously within the room and no air change occurs.

The total release of noble gases is 4.0×10^{-2} curies and the volume of the reactor room is 4.8×10^{8} cm^3 so that the concentration is 0.84×10^{-4} microcuries/cm^3, or 3.1 dis/sec-cm^3. If the room is assumed to be equivalent to a hemisphere of radius R = 606 cm, the dose rate at the center is given by

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

Sv	=	source strength, dis/sec-cm^3
K	=	conversion factor for flux--dose rate
		4.2×10^4 γ/cm^2-sec per mr/min for photons
		of 0.7 Mev (average energy of f.p. decay)
μ'	=	attenuation coefficient for air
	=	3.5×10^{-5} cm^{-1} for 0.7 Mev γ

Thus the maximum estimated deep dose rate from nobel gases alone (water remaining in the pool) is 2.2 mr/min. If the halogens were to be released as well, this would be increased to about 3.7 mr/min. Thus an individual could remain in the room for about 25 min before receiving an exposure of 100 mr. The controlling factor, however will be a longer term dose to the thyroid from halogen activity in the second case. In either event the dose rates are such as to sound off the radiation alarms in the area so that the room can be cleared rapidly.

In addition it is planned that the emergency purge exhaust would be operating in such an accident, venting air at the rate of 250 cfm. This decreases the dose rate to a person in the reactor room.

The dose to the thyroid can be calculated at any time (t) from[6]

$$D_t = \frac{(5.92\times10^2)A_\tau f_a \bar{E}\left[1 - e^{-\lambda_e t}\right]}{m\lambda_e} \text{ (rads)}$$

where,

A_τ	=	inhaled iodine, in curies
fa	=	fraction which is deposited in critical organ
$\bar{E}$	=	effective energy absorbed by thyroid per disintegration (Mev)
λ_e	=	effective decay constant, comprising both radioactive decay and biological elimination
m	=	mass of the thyroid in grams

in a long time, $t = \infty$, and using T_e ($= 0.693/\lambda_e$), the equation becomes:

$$D_\infty = \frac{(8.54\times10^2)A_\tau f_a \bar{E} T_e}{m} \text{ (rads)}$$

6 J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 1962.

$$= \text{constant x} \sum_i A_{\tau i} \bar{E}_i T_{ei}$$

where the parameters are for the ith isotope.

The inhaled iodine is given by

$$A_{\tau i} = R_\tau \bullet \frac{A_i}{v}$$

where $\frac{A_i}{v}$ is the activity concentration, and R the breathing rate for time τ. If τ is made equal to 1 sec, then

$$D_\infty = \frac{(8.54 \times 10^2) f_a R}{m} \sum_i \frac{A_i E_i T_{ei}}{v} \text{ (rads/sec)}$$

For the standard man[7]

$$f_a = 0.23$$
$$m = 20 \text{ grams}$$
$$R = 10 m^3/8 \text{ hr}$$
$$= 3.47 \times 10^{-4} m^3 sec^{-1}$$

and the value of the constant is thus 3.41×10^{-3}.

The data necessary for the summation are in the citred reference[8], and the necessary activity concentrations are calculated from Table 13-5 using 1.5×10^{-5} for the release fraction. The summation for I-131 through I-136 yields a value for D_∞ = 0.45 rads/sec.

Assuming that 50 rems to the thyroid is a limiting annual exposure[9] a person will have approximately 2 minutes to evacuate from the reactor room. In fact the time will be longer than this since the room exhaust will be in operation and will reduce the dose rate.

13.2.5 Predicted Exposure Outside the Building

On detection of release of radioactive fission products into the reactor room, the continuous air monitor will close the normal exhaust and start the emergency purge. The room air will then be exhausted through a filter at 250 cfm (0.118 m^3/sec) which will not remove the noble gases and may not remove the iodines released in the event of pool water loss.

At the penthouse level, on the west side of the building where the emergency purge exhausts, there are also 44 other exhaust fans, excluding those from the reactor area. All of these, with an exhaust rate totalling over 50,000 cfm are designed to operate 24 hrs a day. Assuming that a

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7 "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)," Health Physics 3, June 1960.

8 J. J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, 1962.

9 10CFR 20.1201(1)(i)

fraction may, for any reason, be inoperative, the total volume of air discharged onto the westside of the penthouse level will not be less than 40,000 cfm.

From the above, it is reasonable to assume a conservative dilution of the exhaust effluent by a factor of 160. Using this, and assuming no climatic dilution outside the building, the maximum dose to an individual's thyroid may be calculated.

The calculation assumes:

(1) Complete mixing in the reactor room at all times

(2) The person is immersed in the effluent from the building in which the concentration of radioactivity X_t at any time t is equal to $C_t/70$ where C_t is the concentration in the exhaust and in the room.

With (1) C_t is given by a simple first order rate law: $C_t = C_0 \exp\left[-\frac{a}{V}t\right]$

where	a	=	exhaust rate in m^3/sec
	V	=	room volume in m^3
	C_o	=	initial concentration, at t = 0

the inhaled activity $A_{dt} = R.\ X_t.\ dt$ as above and thus the total amount inhaled up to time t is

$$A_t = \frac{RC_o}{160}\int_o^t \exp\left[-\frac{a}{V}t\right]dt$$

$$= \frac{RC_oV}{160a}\left[1-\exp(-\frac{a}{V}t)\right]$$

If the exposure is for an infinite time,

$$A_\infty = Rc_0V/a.160$$

and, comparing with the previous calculation, the total dose

$$\text{int } D_\infty = \frac{D_\infty(rads/\sec)\bullet V(m^3)}{160a(m^3/\sec)}$$

$$= 11 \text{ rads}$$

If the person were to remain in the effluent cloud for only one hour, the corresponding figure is about 7 rads. Thus we may conclude that for a location on the roof of the building, the maximum possible dose to the thyroid from a fission product release will be considerable. Since the building of the roof is not open to the public and is approximately 100 feet above ground level, the general public would be unable to approach closely to the initial source cloud.

Gulf General Atomic has developed computer programs which take allowance both for decay and for further meteorological dilution.[10] Calculations using these programs for a similar situation to that presented here[11] indicate that these factors will reduce the maximum dose to a person a short distance away from the building by several orders of magnitude. Thus for a slow leak rate from the room (a = 1.69 x 10^{-2} m^3/sec) and very stable atmospheric conditions (Pasqual F), they found predicted exposures of:

	Whole Body γ (millirem)	Thyroid (millirem)
Outside Max dose (12 hours) (General Public)	2.9 x 10^{-3}	2.6
Inside Max dose (1 hour) Occupational Personnel	6.0 x 10^{-3}	7,000

Similar calculations using more sophisticated programming for MNRC[12] resulted in exposure predictions within very similar ranges when coorrected for their greater source term by roughly a factor of 8.

The high variability of local climatic conditions actually prevalent when an accident occurs make it relatively ineffective to try to predict accident exsposures with any greater degree of certainty. It seems very likely that the present predictions will be conservative, and that accidents of this hypothetically very serious nature at this facility will not result in unacceptably hazardous exposures either to personnel in the facility, nor to the general public. In order to further guard against the likelihood of public exposures, operational regulations at UCINRF preclude the presence of any inessential or untrained personnel within the facility whenever fuel handling operations are in progress. Planned events involving handling of irradiated fuel elements in air would be subjected to even greater restrictions and personnel training, and exclusion of the public from areas adjacent to the facility. This was done during the transfer of irradiated fuel elements into the facility in 1974. The protocol for that procedure is on file for further use and inspection.

13.3 Insertion of Excess Reactivity

By deliberate violation of the operating licence and several interlocks and scrams an accidental insertion of reactivity may be made whilst the reactor is at a steady state power greater than 1 kw. While it is inconceivable that a full pulse insertion can be made from full steady state power, this scenario will be examined to establish an extreme upper limit to the potential for fuel cladding stress at this facility.

The measured B-ring fuel temperature at 250 kw is 240°C

The Fuchs-Nordheim Model can be used to estimate the fuel temperature from a pulse:. the

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10 GA-6511, "GAD∅SE and GA∅SET--Programs to Calculate Environmental Consequences of Radioactivity Release". GAMD-5939, "PHICHK and PHIBAR programs to Calculate Atmospheric Dispersion".

11 Addendum to GA-6499, "Hazards Report for the Oregon State University 250 kw TRIGA Mark II reactor--dated June 23, 1965", April 20, 1966.

12 SAR, McClellan Nuclear Radiation Facillity, 1999, Chapter 13.

parameters and equations needed have been given in Section 4.

Then, $C_o = [0.718 \times 10^5 + 0.825 \times 10^2(70)]$ w-sec-/°C $= 0.776 \times 10^5$ w-sec-/°C

The insertion of \$ 3.0 gives a value for

$$\rho = (\$\,3.0 - \$\,1.0) \times (0.7 \times 10^{-2}) = 1.4 \times 10^{-2}$$

$$\text{whence, } \sigma = \frac{1.34\text{x}10^{-4} \cdot 0.776\text{x}10^{5}}{0.825\text{x}10^{2} \cdot 1.4\text{x}10^{-2}} = 9.00$$

$$\text{and } \overline{T_f} = \frac{2 \bullet 1.4\text{x}10^{-2}}{1.34\text{x}10^{-4}} = 208.9°\text{C}$$

$$\text{Thus, } \overline{T} = \frac{-3}{8}(9.00-1)\left\langle 1 - \left[1 + \frac{16}{3}\frac{9.00}{(9.00-1)^2}\right]^{1/2}\right\rangle 208.9$$

$= 202.4$ °C temperature rise for the average fuel element.

To determine the maximum temperature in the hottest fuel element, the average energy release is determined and then multiplied by the peak-to-average power ratio to obtain the maximum energy release in the center element. Then one returns to the energy--temperature equation to determine the maximum temperature.

Let $\overline{E}$ = the energy necessary to raise the average core temperature from $\overline{T}_{SS}$ to $\overline{T}_f$, then

$$\overline{E} = \int_0^{\overline{T}_f - \overline{T}_{ss}} C\,dT = \int_0^{\overline{T}_f - \overline{T}_{ss}} \left[C_{\overline{T}_{ss}} + gT\right] dT =$$

$$\left[C_{\overline{T}_{ss}} T + \frac{g}{2}T^2\right]_0^{\overline{T}_f - \overline{T}_{ss}} = C_{\overline{T}_{ss}}(\overline{T}_f - \overline{T}_{ss}) + \frac{g}{2}(\overline{T}_f - \overline{T}_{ss})^2 \text{(I)}$$

$$= 0.776x10^5(202.4) + \frac{0.825x10^2}{2}(202.4)^2$$

$$= 1.74 \times 10^7 \text{ w-sec}$$

The peak-to-average power ratio for the core during a pulse of this magnitude is very nearly 2.0 so the energy release corresponding to the peak power is equal to 3.48×10^7 w-sec.

Rewriting the above equation for the energy (I) gives:

$$C_{\hat{T}_{ss}}(\hat{T}_f - \hat{T}_{ss}) + \frac{g}{2}(\hat{T}_f - \hat{T}_{ss})^2 - \hat{E} = 0$$

$$\text{Thus,}\quad \hat{T}_f - \hat{T}_{ss} = \frac{-C_{\hat{T}_{ss}}}{g} + \left[\frac{C^2_{\hat{T}_{ss}}}{g^2} + \frac{2\hat{E}}{g}\right]^{1/2}$$

where $\hat{C}_{T_{SS}} = 0.718 \times 10^5 + 0.825 \times 10^2\,(240)$

$= 0.916 \times 10^5$ w-sec/°C

Then $$\hat{T}_f - \hat{T}_{SS} = \frac{-0.916x10^5}{0.825x10^2} + \left[\left(\frac{0.916x10^5}{0.825x10^2}\right)^2 + \frac{6.96x10^7}{0.825x10^2}\right]^{1/2}$$

= 330°C temperature rise for the hottest element.

and assuming simple additivity: $\hat{T}_f = 330 + 240 = 570$ °C.

Experimentally, a similar result is obtained if we estimate the peak temperature for a \$3.0 pulse from measurements on the most recent largest pulses (extrapolated from \$2.55 pulses giving 240± 5 °C) of about 330-350°C. Conservatively, we thus estimate maximum fuel temperature reached from this hypothetical accident of 600°C. Coincidentally this is chosen as the SCRAM setting for the fuel temperature meauring channel.

Discussion in the MNRCSAR (p 4-59) proposes a conservative safety limit of 1100 °C for stainless steel clad $ZrH_{1.7}$ fuel if the cladding temperature is below 500°C at which stress due to the hydrogen pressure will be well below the ultimate cladding strength. The maximum temperature estimated for safe operation is 900°C at high cladding temperatures (>500°C). At 602°C, the stress will have several factors of safety, even if the cladding temperature rose to the same temperature. It is also pointed out (op.cit.) that reported attempts to measure hydrogen pressures in pulses gave values three times less than calculated values.

Thus for circumstances in which a \$3.0 pulse is added to UCINRF core operating at 250kW power, there is still a large margin of safety below the safety limit for fuel temperature in the hottest element. The average element will experience stress conditions at least a factor of 2 lower.

The examination of this "ultimate" accident is designed to place an upper boundary limit on any reactivity incident. Any realistic scenario would be well below these estimates.

8.4 Loss Of Reactor Pool Water (Coolant)

8.4.1 Mechansims and Rate of Loss of Coolant.

The cooling system and purification system pool outlets to their respective pumps are both situated at less than three (3) feet below the normal water level. Thus "pumping out" water from the pool in the event of a pipe breakage could only lower the water level by less than 3 feet. No significent hazard results from such lowering. All components of the water piping on both systems are above pool level, so no syphoning of water from the pool can occur.

Rapid and complete loss of the water in the reactor pool is considered to be highly improbable: in fact it is likely that such an event could only arise from an earthquake shock so intense that the reactor hazard would only be one of many serious hazards to persons in Rowland Hall. However, provision has been made for warning of the loss of pool water as an isolated event and calculations have been made of the fuel temperatures and radiation levels to be anticipated.

A float switch alarm, operating on a 24 hour basis is provided in the tank to alarm at the UCI Police Dispatch Desk if the water should reach about 6 feet below its normal level. This also alrms in the reactor control room. As for any other "trouble" alarm report, the Police Department notifies reactor staff immediately. Operating Procedures exist to delineate subsequent actions. A radiation area monitor station on the ceiling dirctly over the pool will also report an alarm should the water fall to a potentially dangerous level.

The time it would take for the pool to drain may be conservatively estimated by considering a cataclysm that resulted in the complete opening to the soil of the bottom of the tank. Under these conditions the flow of water through the soil can be described by Darcy's law,[1] where i is the pressure gradient

$$\frac{dx}{dt} = Ki$$

in meters of water head per meter, and K is a constant depending on the nature of the soil, having the units m/year/mH_2O)/m. The time to drain the tank is then t = 1/K, years/meter(H_2O). The permeability K has a maximum value for pure sand of about 3×10^3.[1] Thus to drain the tank to the bottom of the core (22 ft) or 6.7 meters the time is 0.0022 years or 19.3 hours. This allows ample time for emergency procedures to be enacted.

Any effect on reactor fuel has been considered in section 13.2 above.

8.4.2 Radiation Levels After Loss of Pool Water

Dose rates have been calculated for two locations: 1) at the top of the pool, 2) at the floor level of the classroom over the reactor. The reactor was assumed to have been operating for a long time at 250 kw prior to loss of water.

The dose rates calculated for instantaneous loss of water at shutdown are given in Table 8-2, and are clearly hazardous. However, as shown above, instantaneous loss of all the water above the core is not conceivable and it is more realistic to estimate the dose rates with partial water loss. These values are given in Table 8-3.

Thus more than 5 hours is provided by the alarm before the dose rate approaches the "permissible" 0.24 mr/hr level in the area above the reactor. Such time is ample to clear the area, investigate the damage and initiate remedial action. In addition a rough estimate of the radiation dose at the side of the pool from scattered radiation indicates that an individual who did not expose himself or herself to the direct radiation from the core could work there for at least 12 hours, 1 day after the accident, without receiving a dose in excess of that permitted by 10 CFR 20.101 for a calendar quarter. This time would be sufficient to view the interior of the pool with a mirror and to continue emergency repairs.

TABLE 13-2

CALCULATED RADIATION DOSE RATES AFTER

TOTAL LOSS OF REACTOR POOL WATER

Time after shutdown	Direct Radiation Levels, R/hr	
	(1)	(2)
	Top of Pool	Floor of Classroom
0.1 hr	1.1×10^3	2.5
1.0 hr	5.9×10^2	1.4
10 hr	2.8×10^2	0.67
1 day	2.0×10^2	0.48
1 week	9.1×10^1	0.22
1 month	16.5	0.04

TABLE 13-3

CALCULATED RADIATION DOSE RATES WITH

LOSS OF POOL WATER AT 0.4 METERS PER HOUR

Time after shutdown	Water Depth over core (meters)	Direct Radiation at Floor of Classroom
		(mr/hr)
0.1 hr	6.1	$< 10^{-10}$
1.0 hr	5.6	$< 10^{-10}$
5.0 hr	4.1 alarms*	1.3×10^{-10}
10.0 hr	2.1	2.8×10^{-2}
15.2 hr	0	ca 550

* Activated by float switch

Method of Calculation

The following assumptions were made:

1). The core had been operating at 250 kw for a very long time (1000 hr) prior to shutdown.

2). The reactor is shutdown as soon as the leak develops.

3). The core can be approximated to a point source with effective strength determined by the photons escaping from an isotropic volume-equivalent spherical source.

4). The photon energy is only from fission decay and can be averaged to 1.0 MeV.

5). Attenuation by core components other than fuel elements is negligible.

Assumptions 1) and 5) are conservative, and 4) is optimistic. The net effect is conservative.

Effective Source Strength

The fission product decay photon source strength is given by Perkins and King[13] and the value of $\Gamma_{(total)}$ was assumed to be 1.0 Mev. The source strengths obtained are given in Table 13-4.

Each fuel element has a volume of 400 cm^3, thus a 67 element core has a volume of 2.68×10^4 cm^3 which is equivalent to a solid sphere of 18.4 cm radius. The fraction of photons which escape from the core is given[14] as a function of $\mu_c r_c$, and is 0.197, where μ_c is the attenuation coefficient of the core material for 1 Mev photons (= 0.207 / cm [15]) and r_c is the radius of the sphere: 18.4 cm. The effective source strength is the product of the source strength and the fraction escaped.

TABLE 13-4

SOURCE STRENGTH FOR SPHERICAL

REACTOR CORE OPERATED FOR 1000 HOURS AT 250 KW

Time after shutdown	Source Strength (MeV/sec-250 kw)	Effective Source Strength (MeV/sec-250 kw)
0.1 hr	1.62×10^{16}	3.19×10^{15}
0.5 hr	1.13×10^{16}	2.22×10^{15}
1.0 hr	0.9×10^{16}	1.77×10^{15}
5.0 hr	5.5×10^{15}	1.08×10^{15}
10.0 hr	4.25×10^{15}	0.84×10^{15}
24.0 hr	3.0×10^{15}	5.9×10^{14}
1week	1.37×10^{15}	2.7×10^{14}
1month	2.5×10^{14}	4.9×10^{14}

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[13] J. F. Perkins and R. W. King, Nuel. Sci. Engin., 3, 726 (1958).

[14] K. K. Aglintsev, Applied Dosimetry, (Eng. Ed.), Ili, London (1965).

[15] GA-5400, p 8-53

Dose Rates-Direct Radiation

The dose rate is computed on the basis that the core is a point source, 21 feet below the top of the pool:

$$D = \frac{S_{eff}}{K.\,4\pi R^2} Be^{-\mu x}$$

where D = the dose rate in rads/hr (= rem/hr for 1 Mev γ)

S_{eff} = effective source strength (col. 3., Table 13-4)

R = distance between core and location considered.

K = conversion factor for Mev/cm^2 to rad/hr (= 5.77×10^5 for 1 Mev photons)

B = build-up factor for water[16], and concrete[17]

μ = linear attenuation coefficient, cm^{-1} (ref. 1) for 1 Mev photons

$\mu_{concrete}$ = 0.149 cm μ_{water} = 0.067 cm

x = thickness of material, cm. ([redacted] cm for concrete ceiling of reactor room). Attenuation by air was neglected.

13.5 Loss of Coolant Flow

UCINRF reactor operates with convection cooling so forced cooling is unnecessary. Shutdown and repair would be intiated following any event that could resrict water flow through the core. Procedural limitations restrict operation when cooling is unavailable to keep the pool temperature below 25°C.

13.6 Mishandling or Malfunction of Fuel

Fuel that is mishandled (e.g dropped) must be inspected and measured before being used in core. At other TRIGA® facilities fuel elements have been found to develop "pin hole" leaks while being used at high power levels. Standard Operating Procedures are written to provide a protocol for locating such an element and removing it from core. The goal ios to minimize radioactivity releases whilst providing a logical way to locate the offending fuel. Routine operations would be suspended following any mishandling or malfunction of fuel. The MHA (Section 13.2) examines thw rost case scenario of fuel malfunction or mishandling.

13.7 Experiment Malfunction

16 H. 0. Wyckoff, "Radiation Attenuation Data", in Radiation Hygiene Handbook, H. Blatz, ed. McGraw Hill, New York, 1959. Chapter 8.

17 Reactor Handbook, Vol. III, Part B, "Shielding", (1962).

All experiments are reviewed for assessment of hazards of malfucntion prior to approval. Specific types of experiments are limited by Technical Specifications include those involving uranium fuel, potentially explosive materials, and those which may release radioactive gases or aerosols. Limitations are imposed to reduce the potential for radiation exposure to personnel or damage to the reactor core.

13.8 Loss of Normal Electric Power

The UCINRF reactor control console is designed so that a reactor scram of all control rods occurs upon loss of regular electric power. Reactor shutdown is thus assured.

To provide information to personnel during such eventuality, the facility radiation monitors (RAM and CAM systems) and alarms and the security monitoring and alarm system are wired to a building emergency diesel generator designed to pick up emergency lighting and power mode automatically on failure of regular electric power. Emergency lights are located in the control room and in the main reactor room. This generator is located in Rowland Hall such that it is likely to be able to operate even in the aftermatch of a major seismic event. This unit is under surveillance and maintenance, including routine testing, by the University Facilities Management group, and is not considered to be a part of the reactor facility.

13.9 External Events

The potential for serious external impact of climate or other irregularities is low at this facility. Section 2 has reviewed this. The facility construction [redacted] a large structure with [redacted] further protects the facility from such influences. [redacted] The probabilty that flying debris or falling aircraft could cuase significant damage is very low. The adjacent loading dock has extensive drains that will obviate flooding from an outside source.

The threat of damage from deliberate human activities (other than aircraft) is considered in the Security Plan. In operation for the past 30 years there have so far been no direct threats. Threats to place explosives in nearby lecture halls have been issued but have been found to be false. Procedures call for facility staff to be alerted during such incidents, and for all buildings to be evacuated.

The potential for seismic damage is considered elsewhere (section 2).

[NOTE this needs to be updated in 1999-2000 in the light of new assessment of seismic hazrd to the campus area to be released in late 1999.]

13.10 Mishandling or Malfunction of Equipment

Standard Operating Procedures require personnel awareness and training for use of facility equipment. As part of a large university School of Physical Sciences, the facility is subject to general rules and regulations regarding safety of all operations. Highly skilled and trained Facility Maintenance staff are available and on call 24 hours a day. The School employs a professional, resident, Facilities Manager to supervise maintenance and repair of existing buildings, remodelling, and new construction. The School also employs a fulltime resident Safety Manager to provide surveillance and assistance in general and chemical safety. All operations are subject to surveillance by the staff of the Environmental Health and Safety Office which employs several trained radiation safety specialists in addition to personnel qualified in all areas of safety and environmental protection. These are always readily available as consultants for planning or emergencies, and to provide special assistance, for example when unusual operations such as transfer of fuel, shipment of isotopes, or disposal of radioactive materials are conducted.

Since students are frequently involved in carrying out experiments within the facility, they are always carefully instructed in operation of equipment. Any operatuons that could affect the reactor core or fuel are always carefully supervised by trained reactor operators.

Specific Operating Procedures are written in anticipation of common malfunctions, such as the fracture of a sample transfer capsule.

Procedures also call for facility operations to be suspended following any perceived malfunction of safety related equipment or in the case where routine measurements indicate that such equipment may not be functioning properly.

14. TECHNICAL SPECIFICATIONS

TECHNICAL SPECIFICATIONS

FOR THE

UNIVERSITY OF CALIFORNIA, IRVINE
TRIGA MARK I NUCLEAR REACTOR

REVISED:

NOVEMBER 1998

TABLE OF CONTENTS

1.0 DEFINITIONS

The following frequently used terms are defined to aid in the uniform interpretation of these specifications.

1.1 Reactor Shutdown - The reactor is in a shutdown condition when sufficient control rods are inserted so as to assure that it is subcritical by at least $1.00 of reactivity.

1.2 Reactor Secured - The reactor is secured when all. the following conditions are satisfied:

a. The reactor is shutdown;

b. Power to the control rod magnets and actuating solenoids is off, and the key removed;

c. No work is in progress involving fuel or in-core experiments or maintenance of the core structure, control rods, or control rod drive mechanisms.

1.3 Reactor Operation - The reactor is in operation when it is not secured.

1.4 Standard Control Rod - A standard control rod is one having rack and pinion, electric motor drive, and scram capability.

1.5 Transient Control Rod

a. Adjustable Transient Rod - an adjustable transient rod is one having both pneumatic and electro-mechanical drives and with scram capability.

b. Fast Transient Rod - A fast transient rod is one that is pneumatically operated and has scram capability.

1.6 Operable - A system or device is operable when it is capable of performing its intended functions in a normal manner.

1.7 Cold Critical - The reactor is in the cold critical condition when it is critical with the fuel and bulk water temperatures the same (≈20°C).

1.8 Steady-State Mode - The reactor is in the steady-state mode when the reactor mode selection switch is in the steady-state or automatic position. In this mode, reactor power is held constant or is changed on periods greater than three seconds.

1.9 Pulse Mode - The reactor is in the pulse mode when the reactor mode selection switch is in the pulse position. In this mode, reactor power is increased on periods less than one second by motion of the transient control rod(s).

1.10 Experiment - An experiment is:

a. Any apparatus, device or material placed in the reactor core region, in an experimental facility, or in-line with a beam of radiation emanating from the reactor;

b. Any operation designed to measure reactor characteristics.

1.11 Untried Experiment - An untried experiment is any experiment not previously performed in this reactor.

1.12 Experimental Facilities - Experimental facilities are the pneumatic transfer systems, central thimble, rotary specimen rack, and the in-core facilities (including single element positions, three-element positions, and the seven element position).

1.13 Abnormal Occurrence - An abnormal occurrence is any of the following:

a. Any actual safety system setting less conservative than specified in the Limiting Safety System Settings section of the Technical Specifications;

b. Operation in violation of a limiting condition for operation;

c. An engineered safety system component failure which could render the system incapable of performing its intended function;

d. Release of fission products from a fuel element;

e. An uncontrolled or unanticipated change in reactivity;

f. An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

1.14 Standard Thermocouple Fuel Element - A standard thermocouple fuel element is a standard fuel element containing three sheathed thermocouples imbedded near the axial and radial center of the fuel element.

1.15 Measured Value - The measured value of a process variable is the value of the variable as it appears on the output of a channel.

1.16 Measuring Channel - A measuring channel is the combination of sensor, lines, amplifiers and output device which are connected for the purpose of measuring the value of a process variable.

1.17 Reactor Safety System - The reactor safety system is that combination of channels and associated circuitry which forms the automatic protective system for the reactor or provides information which requires manual protective action to be initiated.

1.18 Operating - Operating means a component or system is performing its intended function in its normal manner.

1.19 Channel Check - A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with other independent channels or methods measuring the same variable.

1.20 Channel Test - A channel test is the introduction of a signal into the channel to verify that it is operable.

1.21 Channel Calibration - A channel calibration is an adjustment of the channel such that its output responds, with acceptable range and accuracy, to known values of the parameter which the channel measures.

1.22 Reference Core - A reference core is a core with a configuration similar to the core configuration existing at the initial start-up of the reactor.

1.23 Ring - A ring is one of the six concentric bands of fuel elements surrounding the central opening of the core. The rings are designated by the letters B through G, with the letter B used to designate the innermost band.

1.24 Three Element Positions - Two generally triangular-shaped sections cut out of the upper grid plate, one encompassing ring holes D5, E6 and E7 and the other D14, E18 and E19. When fuel elements are placed in these locations a special fixture provides lateral support. With the fixture and fuel removed, an experiment up to 2.4 in. in diameter may be inserted.

1.25 Seven Element Position - A hexagonal section which can be removed from the upper grid plate for insertion of specimens up to 4.4 in. in diameter after relocation of the six B-ring elements and removal of the central thimble.

1.26 Closed Packed Array - A closed packed array is a fuel loading pattern in which the fuel elements are arranged in the core by filling the inner rings first.

1.27 Surveillance Activities - Activities required at pre-defined intervals to assure performance of reactor and safety related components. During prolonged periods when the reactor remains shutdown, Technical Specification Surveillance Requirements 4.1 (fuel element dimensions), 4.2 (control rod integrity), and 4.3 (fuel temperature safety limit) may be deferred. However, they must be completed prior to reactor start-up except for 4.2 (a), 4.3 (d), and 4.3 (f) which require reactor operation in order to be accomplished and must be completed within 30 days of restarting reactor operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limit - Fuel Element Temperature

Applicability

This specification applies to the fuel element temperature.

Objective

The objective is to define the maximum fuel element temperature that can be permitted with confidence that no fuel element cladding damage will result.

Specification

The temperature in a stainless steel clad, high hydride fuel element shall not exceed 1000°C under any conditions of operation.

Bases

The safety limitations of the TRIGA fuel are described in the Safety Analysis Report (SAR) for the UC Irvine TRIGA, Section 8. The important process variable for a TRIGA reactor is the fuel element temperature. This parameter is well suited as a single specification since it can be measured. A loss in the integrity of the fuel element cladding could arise from an excessive build-up of pressure between the fuel moderator and the cladding. The pressure is caused by the presence of fission product gases and the dissociation of the hydrogen and zirconium in the fuel moderator. The magnitude of this pressure is determined by the fuel moderator temperature.

The safety limit for the stainless steel clad, high hydride ($ZrH_{1.7}$) fuel element is based on data (SAR pages 8.38 through 8.40 and University of Illinois SAR pages III-56 through III-59) which indicate that the stress in the cladding (due to the hydrogen pressure from the dissociation of the zirconium hydride) will remain below the yield stress provided the temperature of the fuel does not exceed 1000°C.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the trip setting for the fuel element temperature channel.

Objective

The objective is to prevent the safety limit from being exceeded.

Specification

For a core composed entirely of stainless steel clad, high hydride fuel elements, limiting safety system settings apply according to the location of the standard thermocouple fuel element which shall be located in the B-or C-ring as indicated in the following table:

Location	Limiting Safety System Setting
B-ring	800°C
C-ring	755°C

Bases

Stainless steel clad, high hydride fuel element: The limiting safety system settings that are indicated represent values of the temperature, which if exceeded, shall cause the reactor safety system to initiate a reactor scram. Since the fuel element temperature is measured by a fuel element designed for this purpose, the limiting settings are given for different locations in the fuel array. Under these conditions, it is assumed that the core is loaded so that the maximum fuel temperature is produced in the B-ring. If the fuel element temperature is measured in the C-ring, the respective temperature is the limiting safety system setting.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity

Applicability

These specifications apply to the reactivity condition of the reactor, and the reactivity worths of control rods and experiments, and apply for all modes of reactor 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.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. The shutdown margin referred to the cold, xenon-free condition, with the highest worth rod fully withdrawn, is greater than $0.50;

b. The total reactivity worth of the two transient control rods is less than $3.00;

c. Any experiment with a reactivity worth greater than $1.00 is securely fastened so as to prevent unplanned removal from or insertion into the reactor;

d. The excess reactivity is less than $3.00;

e. The reactivity worth of an individual experiment is not more than $3.00;

f. The total reactivity worth of all experiments is limited so that the shutdown margin referred to the cold xenon-free condition with all rods in is at least $0.50;

g. The total of the absolute values of the reactivity worth of all experiments in the reactor is less than $3.00;

h. The drop time of a standard control rod from the fully withdrawn position to 90 percent of full reactivity insertion is less than one second; and

i. The neutron power level indication on the startup channel is greater than 1×10^{-7}% of full power..

Bases

The shutdown margin required by specification 3.1a is necessary so that the reactor can be shut down from any operating condition and remain shutdown after cooldown and xenon decay even if one control rod (including a transient control rod) should stick in the fully withdrawn position.

Specification 3.1b is based on Section 8.5 of the SAR. The power level at which a pulse could be initiated in an accident may be as high as 100°C At 100 kw, the peak temperature of the fuel will be 115°C. The calculations indicate that a $3.00 pulse will result in a peak temperature of only 502°C, well below the safety limit.

Specification 3.1c is based on the same calculations. By restricting each experiment to $1.00, an additional margin is provided to allow for considerable uncertainty in experiment worth.

Specification 3.1c through 3.1g are intended to provide additional margins between those values of reactivity changes encountered during the course of operations involving experiments and those values of reactivity which, if exceeded, might cause a safety limit to be exceeded.

Specification 3.1 h is intended to assure prompt shutdown of the reactor in the event a scram signal is received.

Specification 3.1 i is intended to assure that sufficient neutrons are available in the core to provide a signal at the output of the startup channel during approaches to criticality.

3.2 Pulse Operation

Applicability

These specifications apply to operation of the reactor in the pulse mode.

Objective

The objective is to prevent the fuel temperature safety limit from being exceeded during pulse mode operation.

Specifications

The reactor shall not be operated in the pulse mode unless, in addition to the requirements of Section 3.1, the following conditions exist:

a. The transient rods are set such that their reactivity worth upon withdrawal is less than $3.00; and

b. The steady-state power level of the reactor is not greater than 1 kilowatt.

Bases

Specification 3.2a is based on Figure 7-4 of the SAR which shows that the temperature rise expected for a pulse insertion of $3.00 is less than 500°C.

Specification 3.2b is intended to prevent inadvertent pulsing from a high steady-state power level such that the final peak temperature might approach the safety limit.

3.3 Reactor Instrumentation

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 the measuring channels described in the following table are operable and the information is displayed in the control room:

Measuring Channel	Minimum Number Operable	Operating Mode in which Required
Fuel Element Temperature	1	All Modes
Reactor Power Level	2	Steady-State
Reactor Power Level (high range)	1	Pulse Mode
Startup Power Level	1	During Reactor Startup
Area Radiation Monitors	2	All Modes
Continuous Air Radiation Monitor	1	All Modes

Bases

The fuel temperature displayed at the control console gives continuous information on the process variable which has a specified safety limit.

The neutron detectors assure that measurements of the reactor power level are adequately covered at both low and high power ranges.

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.

3.4 Reactor Safety System

Applicability

This specification applies to the reactor safety system channels.

Objective

The objective is to require the minimum number of reactor safety system channels that must be operable in order to assure that the fuel temperature safety limit is not exceeded.

Specification

The reactor shall not be operated unless the safety system channels described in the following table are operable.

Measuring Channel	Minimum Number Operable	Function	Operating Mode in which Required
Fuel Element Temperature	1	Scram	All Modes
Reactor Power level	1	Scram	Steady-State Mode
Reactor Power Level	1	Prevent transient rods firing when power is >1 kilowatt	Pulse Mode
Manual Button	1	Scram	All Modes
Seismic Switch	1	Scram	All Modes
Startup Power level	1	Prevent control rod withdrawal when power level indication is less than 1×10^{-7} %.	Reactor Startup
Standard Control Rod Position	1	Prevent application of air to fast transient rod when all other rods are not fully inserted	Steady-State Mode
Adjustable Transient Cylinder Position	1	Prevent application of air to adjustable transient rod unless cylinder is fully down	Steady-State Mode

Bases

The interlocks which prevent the firing of the transient rods in the steady-state mode or if the power level is greater than 1 kilowatt prevent inadvertent pulses. The interlock to prevent startup of the reactor with less than 1×10^{-7}% power indicated on the startup channel assure that sufficient neutrons are available to assure proper operation of the startup channel.

The fuel temperature scram provides the protection to assure that if a condition results in which the limiting safety system setting is exceeded, an immediate shutdown will occur to keep the fuel temperature below the safety limit. The power level scram is provided as added protection against abnormally high fuel temperature and to assure that reactor operation stays within the licensed limits. The manual scram allows the operator to shut down the system if an unsafe or abnormal condition occurs. The seismic switch will shut down the reactor if major earth movement (M.M. VI or above) occurs in case the operator is prevented from operating the manual scram at the time.

3.5 Release of Argon 41

Applicability

This specification applies to the release of radioactive argon 41 from the facility exhaust system.

Objective

The objective is to assure that exposures to the public resulting from the release of argon 41 generated by reactor operation, will not exceed the limits of 10 CFR Part 20 for unrestricted areas.

Specification

Releases of argon 41 from the reactor room exhaust shall not be made in concentrations greater than 4×10^{-8} μc/ml averaged over a year.

Basis

It is shown in Section 8.4.4 of the SAR, pages 8-18 through 8-23 that the release of argon 41 will be diluted by a factor of at least 40 in reaching a potential exposure site even in the poorest dispersion conditions. At a concentration level of 1×10^{-9} μCi/ml, for constant immersion, the maximum conceivableannual exposure will be 5 mrem to an individual and is well within acceptable limits.

3.6 Ventilation System

Applicability

This specification applies to the operation of the reactor facility 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 facility and building ventilation system is in operation and the emergency exhaust shutdown_system has been verified to be operable within the preceding 30 days. An exception may be made for periods of time not to exceed two days to permit repairs to the system. During such periods of repair:

a. The reactor shall not be operated in the pulse mode; and

b. The reactor shall not be operated with experiments in place whose failure could result in the release of radioactive gases or aerosols.

Basis

It is shown in Section 8.7.5 of the SAR that operation of the emergency exhaust shutdown system reduces off-site doses to below 10 CFR Part 20 limits in the event of a TRIGA fuel element failure, and in 8.4.4 and 8.4.5 that operation of the normal system adequately dilutes the argon 41 released even under unusual experimental operations. The specifications governing operation of the reactor while the ventilation system is undergoing repair preclude the likelihood of fuel element failure during such times. It is shown in Section 8.6 that, if the reactor were to be operating at full steady-state power, fuel element failure will not occur even if all the reactor tank water were to be lost immediately.

3.7 Pool Water Level

Applicability

This specification applies to the pool water level.

Objective

The objectives are to assure that an adequate level of water is maintained above the core and that prompt corrective action will be initiated in the unlikely event that pool-water leaks from the tank.

Specification

The pool water level shall normally be maintained approximately 19 feet above the reactor top grid plate. A pool water level measuring channel shall sound an alarm at the UCI Police Dispatch Desk if the water level in the reactor tank drops to 13 feet or less above the top grid plate. The measuring channel shall be operable except during periods of maintenance on the channel. If the measuring channel is inoperable, the level of the pool water shall be verified to be normal by visual observation at least every ten (10) hours. Whenever the duration of inoperability exceeds five (5) consecutive days, the reactor shall not be operated until repairs are completed and normal operation of the water level measuring channel has been verified. If either the alarm actuates or visual observation indicates that water level is not normal, prompt corrective action shall be taken.

Basis

Section 8.6 of the SAR discusses the results of loss of pool water from the Irvine TRIGA reactor. Section 8.6.2 shows that fuel cladding rupture is unlikely even following operation at full licensed power. Calculations in Section 8.6.3 indicate that ten hours after a leak develops in the pool or five hours after the water level (13 ft) alarm sounds, the radiation levels in the room above the reactor facility would be 0.028 mr/hr with the reactor shutdown. Both instrument and visual monitoring at the intervals specified will provide adequate time for corrective action. Written procedures, approved in accordance with Specification 6.3, shall define emergency actions to be taken.

3.8 Limitations on Experiments

Applicability

This specification applies to experiments placed in the reactor and its experimental facility.

Objective

The objective is to prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

Specifications

The reactor shall not be operated unless the following conditions exist:

a. Fueled experiments are limited such that the total inventory of iodine isotopes 131 through 135 in the experiment is not greater than 0.3 curies and the Strontium 90 inventory is not greater than 1 microcurie;

b. The quantity of known explosive materials to be irradiated is less than 25 milligrams and the pressure produced in the experiment container upon accidental detonation of the explosive has been experimentally determined to be less than the design pressure of the container; and

c. Experiments containing materials corrosive to reactor components, compounds highly reactive with water, potentially explosive materials or liquid fissionable materials are doubly encapsulated.

Basis

It is shown in the SAR p. 8.53, that a release of 0.024 curies of iodine activity will result in a maximum dose to the thyroid of a person in an unrestricted area of less than 1/20 of the permissible dose. The limit on iodine inventory is set at 10 times this value. The limit for Strontium 90 is that which corresponds to the iodine yield of 0.3 curies for a given number of fission events and would be no hazard. Specifications 3.8b and 3.8c reduce the likelihood of damage to reactor components resulting from experiment failure.

4.0 SURVEILLANCE REQUIREMENTS

4.1 Fuel

Applicability

This specification applies to the surveillance requirement for the fuel elements.

Objective

The objective is to assure that the dimensions of the fuel elements remain within acceptable limits.

Specifications

a. The standard fuel elements shall be measured for length and bend at intervals separated by not more than 500 pulses of magnitude greater than $1.00 of reactivity, but the intervals shall not exceed 36 months. Fuel follower control rods shall be measured for bend at the same time interval.

b. A fuel element indicating an elongation greater than 1/10 of an inch over its original length or a lateral bending greater than 1/16 of an inch over its original bending shall be considered to be damaged and shall not be used in the core for further operation.

   A fuel follower control rod shall be considered to be damaged and shall not be used for further operation if it indicates a lateral bending greater than 1/16 of an inch over the fuel containing portion of the rod.

c. Fuel elements in the B- and C-ring shall be measured for possible distortion in the event that there is indication that fuel temperatures greater than the limiting safety system setting on temperature may have been exceeded.

Bases

The most severe stresses induced in the fuel elements result from pulse operation of the reactor, during which differential expansion between the fuel and the cladding occurs and the pressure of the gases within the elements increases sharply. The above limits on the allowable distortion of a fuel element have been shown to correspond to strains that are considerably lower than the strain expected to cause rupture of a fuel element and have been successfully applied at other TRIGA installations. The surveillance interval is selected based on the past history of more frequent , uneventful, inspections for over 20 years at this facility and experience at other TRIGA facilities with similar power levels, fuel type, and operational modes. It is also designed to reduce the possibilities of mechanical failures as a result of handling elements, and to minimize potential radiation exposures to personnel.

4.2 Control Rods

Applicability

This specification applies to the surveillance requirements for the control rods.

Objective

The objective is to assure the integrity of the control rods.

Specifications

a. The reactivity worth of each control rod shall be determined annually, but at intervals not to exceed eighteen months.

b. Control rod drop times shall be determined annually, but at intervals not to exceed eighteen months.

c. The control rods shall be visually inspected for deterioration at intervals not to exceed three years.

d. On each day that pulse mode operation of the reactor is planned, a functional performance check of the transient (pulse) rod system shall be performed.

   Annually, at intervals not to exceed eighteen months, the transient (pulse) rod drive cylinder and the associated air supply system shall be inspected, cleaned, and lubricated as necessary.

Bases

The reactivity worth of the control rods is measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worths of experiments inserted in the core. The visual inspection of the control rods and measurement of their drop times are made to determine whether the control rods are capable of performing properly. The surveillance intervals are selected based on the past history of more frequent , uneventful, inspections for over 20 years at this facility and experience at other TRIGA facilities with similar power levels, fuel type, and operational modes. They are also designed to reduce the possibilities of mechanical failures as a result of handling control rods, and to minimize radiation exposures to personnel.

4.3 Reactor Safety System

Applicability

This specification applies to the surveillance requirements for the measuring channels of the reactor safety system.

Objective

The objective is to assure that the safety system will remain operable and will prevent the fuel temperature safety limit from being exceeded.

Specifications

a. A channel test of each of the reactor safety system channels shall be performed prior to each day's operation or prior to each operation extending more than one day.

b. A channel check of the fuel element temperature measuring channel shall be performed daily whenever the reactor is in operation or when pulse operation is planned.

c. A channel check of the power level measuring channels shall be performed daily whenever the reactor is in operation.

d. A channel calibration by the calorimetric method shall be made of the power level monitoring channels annually, but at intervals not to exceed eighteen months.

e. A calibration of the temperature measuring channels shall be performed annually, but at intervals not to exceed eighteen months. This calibration shall consist of introducing electric potentials in place of the thermocouple input to the channels.

f. A verification of the original calibration of the temperature measuring channels shall be performed annually, but at intervals not to exceed eighteen months. This verification shall consist of. comparing the measured temperature in a reference core at a known power level with the temperature measured in the reference core during the initial startup of the reactor.

Basis

The daily tests and channel checks will assure that the safety channels are operable. The annual calibrations and verifications will permit any long-term drift of the channels to be corrected. The history of operations at this facility over the last 20 years has shown that annual checks will allow correction for the very small amounts of drift observed.

4.4 Pool Water Level Channel

Applicability

This specification applies to the pool water level channel required by Section 3.7 of these specifications.

Objective

The objective is to assure that the channel is operable.

Specifications

The pool water level measuring channel shall be verified to be operable at intervals not to exceed two months.

Basis

This verification will assure that a continued warning system for a loss-of-coolant accident is maintained,

4.5 Radiation Monitoring Equipment

Applicability

This specification applies to the radiation monitoring equipment required by Section 3.3 of these specifications.

Objective

The objective is to assure that the radiation monitoring equipment is operating and to verify the appropriate alarm settings.

Specification

The alarm -set points for the radiation monitoring instrumentation shall be verified daily during periods when the reactor is in operation.

Basis

Surveillance of the equipment will assure that sufficient protection against radiation is available.

4.6 Maintenance

Applicability

This specification applies to the surveillance requirements following maintenance of control or safety system.

Objective

The objective is to assure that a system is operable before being used after maintenance has been performed.

Specification

Following maintenance or modification of a control or safety system or component, it shall be verified that the system is operable prior to its return to service.

Basis

This specification assures that work on the system or component has been properly carried out and that the system or component has been properly reinstalled or reconnected before reliance for safety is placed on it.

5.0 DESIGN FEATURES

5.1 Reactor Fuel

Applicability

This specification applies to the fuel elements used in the reactor core.

Objective

The objective is to assure that the fuel elements are of such a design and fabricated in such a manner as to permit their use with a high degree of reliability with respect to their mechanical integrity.

Specifications

a. Standard Fuel Element: The standard fuel element shall contain uranium-zlrconium hydride, clad in 0.020 inch of 304 stainless steel. It shall contain a maximum of 9.0 weight percent uranium which has a maximum enrichment of 20 percent. There shall be 1.55 to 1.80 hydrogen atoms to 1.0 zirconium atom.

b. Loading: The elements shall be placed in a closely packed array except for experimental facilities or for single positions occupied by control rods and a neutron start-up source.

Basis

These types of fuel elements have a long history of successful use in TRIGA reactors.

5.2 Reactor Building

Applicability

This specification applies to the building which houses the reactor facility.

Objective

The objective is to assure that provisions are made to restrict the amount of release of radioactivity from the reactor facility.

Specifications

a. The reactor shall be housed in a closed room designed to restrict leakage when in operation, when the facility is unmanned, or when spent fuel is being handled exterior to a cast.

b. The minimum free volume of the reactor room shall be 1,000 cubic feet.

c. The building shall be equipped with a ventilation system capable of exhausting air or other gases from the reactor room at a minimum of 70 feet above ground level.

Basis

In order that the movement of air can be controlled, the reactor area contains no windows that can be opened. The room air is exhausted through an independent exhaust and discharged at roof level with other exhausts to provide dilution.

5.3 Fuel Storage

Applicability

This specification applies to the storage of reactor fuel at times when it is not in the reactor core.

Objective

The objective is to assure that fuel which is being stored will not become supercritical and will not reach unsafe temperatures.

Specifications

a. All fuel elements shall be stored in a geometrical array where the keff is less than 0.8 for all conditions of moderation.

b. Irradiated fuel elements and fuel 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 800°C.

Basis

New fuel is stored in their shipping containers. Hot fuel is stored in pits described in the submittal dated June 5, 1969. These pits are designed to hold 19 elements, an amount which cannot form a critical array. Very hot fuel is stored in racks in the main tank where cooling water is provided.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

a. The reactor facility shall be an integral part of the School of Physical Sciences of the University of California, Irvine. The reactor shall be related to the University structure as shown in Chart I.

b. The reactor facility shall be under the direction of the Reactor Administrator who shall be a tenure member of the UCIfaculty and supervised by a Reactor Supervisor who shall be a qualified licensed senior operator for the facility. The Reactor Supervisor shall be responsible for assuring that all operations are conducted in a safe manner and within the limits prescribed by the facility license and the provisions of the Reactor Operations Committee.

c. There shall be a Radiation Safety Officer responsible for the safety of operations from the standpoint of radiation protection. The Radiation Safety Officer shall report to the Office of Environmental Health and Safety which is an organization independent of the reactor operations organization as shown in Chart I.

CHART I

School of Physical Sciences
|
Reactor Operations Committee
|
Reactor Administrator
|
Reactor Supervisor

Office of Environmental Health and Safety
|
Radiation Safety Officer

Radiation Safety Officer ‹- - - - - - - -› Reactor Supervisor

Reactor Operations

6.2 Review

a. There shall be a Reactor Operations Committee which shall review reactor operations to assure that the facility is operated in a manner consistent with public safety and within the terms of the facility license.

b. The responsibilities of the Committee 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, and Technical Specifications;
3. Determination of whether a proposed change, test, or experiment would constitute an unreviewed safety question or a change in the Technical Specifications;
4. Review of the operation and operational records of the facility;
5. Review of abnormal performance of plant equipment and operating anomalies;
6. Review of unusual or abnormal occurrences and incidents which are reportable under 10 CFR 20 and 10 CFR 50; and
7. Approval of individuals for the supervision and operation of the reactor.

c. The Committee shall be composed of at least five members, one of whom shall be a health physicist designated by the Office of Environmental Health and Safety of the University. The Committee shall be proficient in all areas of reactor operation and reactor safety. The membership of the Committee shall include at least one member who is not associated with the School of Physical Sciences..

d. The Committee shall have a written statement defining such matters as the authority of the Committee, the subjects within its purview, and other such administrative provisions as are required for effective functioning of the Committee. Minutes of all meetings of the Committee shall be kept.

e. A quorum of the Committee shall consist of not less than a majority of the full Committee and shall include the chairman or his designee.

f. The Committee shall meet at least semi-annually, at intervals not to exceed nine months.

6.3 Operating Procedures

Written procedures, reviewed and approved by the Reactor Operations Committee, shall be in effect and followed for the following items. The procedures shall be adequate to assure the safety of the reactor but should not preclude the use of independent judgment and action should the situation require such.

a. Startup, operation, and shutdown of the reactor.

b. Installation or removal of fuel elements, control rods, experiments, and experimental facilities.

c. Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary coolant system leaks, and abnormal reactivity changes.

d. Emergency conditions involving potential or actual release of radioactivity, including provisions for evacuation, re-entry, recovery, and medical support.

e. Maintenance procedures which could have an effect on reactor safety.

f. Periodic surveillance of reactor instrumentation and safety systems, area monitors and continuous air monitors.

Substantive changes to the above procedures shall be made only with the approval of the Reactor Operations Committee. Temporary changes to the procedures that do not change their original intent may be made by the Reactor Supervisor. All such temporary changes to procedures shall be documented and subsequently reviewed by the Reactor Operations Committee.

6.4 Action to be Taken in the Event a Safety Limit is Exceeded

In the event a safety limit is exceeded, or thought to have been exceeded:

a. The reactor shall be shut down and reactor operation shall not be resumed until authorized by the NRC.

b. An immediate report of the occurrence shall be made to the Chairman of the Reactor Operations Committee, and reports shall be made to the NRC in accordance with Section 6.7 of these specifications.

c. A report shall be made 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 reoccurrence. This report shall be submitted to the Reactor Operations Committee for review, and a suitable similar report submitted to the NRC when authorization to resume operation of the reactor is sought.

6.5 Action to be Taken in the Event of an Abnormal Occurrence

In the event of an abnormal occurrence, as defined in Section 1.13 of the specifications, the following action shall be taken:

a. The Reactor Supervisor shall be notified and corrective action taken prior to resumption of the operation involved.

b. A report shall be made 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 reoccurrence. This report shall be submitted to the Reactor Operations Committee for review.

c. Where appropriate, a report shall be submitted to the NRC in accordance with Section 6.7 of these specifications.

6.6 Plant Operating Records

a. In addition to the requirements of applicable regulations, and in no way substituting therefor, records and logs shall be prepared and retained for a period of at least 5 years of the following items, as a minimum:

1. Normal plant operation;

2. Principal maintenance activities;

3. Abnormal occurrences;

4. Equipment and component surveillance activities;

5. Gaseous and liquid radioactive effluents released to the environs;

6. Off-site environmental monitoring surveys;

7. Fuel inventories and transfers;

8. Facility radiation and contamination surveys;

9. Radiation exposures for all personnel;

10. Experiments performed with the reactor.

b. Updated, corrected, and as-built drawings of the facility shall be retained for the facility life.

6.7 Reporting Requirements

In addition to the requirements of applicable regulations, and in no way substituting therefore, reports shall be made to the NRC as follows:

a. An immediate report (by telephone and telegraph to the NRC Headquarters Office) of:

1. Any accidental off-site release of radioactivity above permissible limits, whether or not the release resulted in property damage, personal injury or exposure; and

2. Any violation of a safety limit.

b. A report within 24 hours (by telephone or telegraph to the NRC Headquarters Office) of:

1. Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring

during operation of the reactor;

2. Incidents or conditions relating to operation of the facility which prevented or could have prevented the performance of engineered safety features as described in these specifications; and

3. Any abnormal occurrences as defined in Section 1.13 of these specifications.

c. A report within 10 days (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) of:

1. Any significant variation of measured values from a corresponding predicted or previously measured value of safety-connected operating characteristics occurring during operation of the reactor;

2. Incidents of conditions relating to operation of the facility which prevented or could have prevented the performance of engineered safety features as described in these specifications; and

3. Any abnormal occurrences as defined in Section 1.13 of these specifications.

d. A report within 30 days (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) of:

1. Any substantial variance from performance specifications contained in these specifications or in the Safety Analysis Report;

2. Any significant change in the transient or accident analyses as described in the Safety Analysis Report;

3. Any changes in facility organization; and

4. Any observed inadequacies in the implementation of administrative or procedural controls.

e. A report within 60 days after criticality of the reactor (in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) upon receipt of a new facility license or an amendment to the license authorizing an increase in reactor power level or the installation of a new core, describing the measured values of the operating conditions or characteristics of the reactor under the new conditions, including:

1. Total control rod reactivity worth;

2. Reactivity worth of the single control rod of highest reactivity worth;

3. Total and individual reactivity worths of any experiments inserted in the reactor; and

4. Minimum shutdown margin both at room and operating temperatures.

f. A routine report in writing to the Document Control Desk, USNRC, Washington, D. C. 20555) within 60 days after completion of the first six months of facility operation and at the end of each 12-month period thereafter, providing the following information:

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

2. A tabulation showing the energy generated by the reactor (in megawatt hours), the amount of pulse operation, the number of hours the reactor was critical;

3. The number of emergency shutdowns and inadvertent scrams, including the reasons therefore;

4. Discussion of the major maintenance operations performed during the period,

including the effect, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required;

5. A summary of each change to the facility or procedures, tests, and experiments carried out under the conditions of Section 50.59 of 10 CFR 50;

6. 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;

7. A description of any environmental surveys performed outside the facility; and

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

6.8 Review of Experiments

a. All proposed experiments utilizing the reactor shall be evaluated in writing by the experimenter and the Reactor Supervisor (and the Radiation Safety Officer when appropriate) to assure compliance with the provisions of the utilization license, theTechnical Specifications, and 10 CFR 20. If, in the judgment of the Reactor Supervisor, the experiment meets with the above provisions and is not an "untried experiment" the experiment may be scheduled. Otherwise it will be submitted to another member of the Reactor Operations Committee for written evaluation and thence to the Reactor Operations Committee for final approval as indicated in Section 6.2 above. When pertinent, the evaluation shall include:

1. The reactivity worth of the experiment;

2. The integrity of the experiment, including the effects of changes in temperature, pressure, or chemical composition;

3. Any physical or chemical interaction that could occur with the reactor components; and

4. Any radiation hazard that may result from the activation of materials or from external beams.

b. Prior to the performing of an experiment not previously performed in the reactor, it shall be reviewed and approved in writing by the Reactor Operations Committee. Their review shall consider the following information:

1. The purpose of the experiment;

2. A procedure for the performance of the experiment; and

3. The written evaluations made as in Paragraph a. above.

c. A request for radioisotopes or the irradiation of materials shall be handled in the same manner as many other experiment except that a series of irradiations can be approved as one experiment. The expiration date for such approvals shall be one year or the expiration date of the applicant's appropriate radioactive materials license. For each irradiation, the applicant shall submit an "Irradiation Request" to the Reactor Supervisor. Thisd request shall contain information on the target material including the amount, chemical form, and packaging. For the purposes of Paragraph a. above, routine irradiations, which do not contain nuclear fuel or known explosive materials and which do not constitute a significant threat to the integrity of the reactor or to the safety of individuals, may be classified as "tried experiments".

d. In evaluating experiments, the following assumptions shall be used for the purpose of determining whether failure of the experiment would cause the appropriate limits of 10 CFR 20 to be exceeded:

1. If the possibility exists that airborne concentrations of radioactive gases or aerosols may be released within the facility, 100 percent of the gases or aerosols will escape;
2. If the effluent exhausts through a filter installation designed for greater than 99 percent efficeincy for 0.3 micron particles, at least 10% of gases or aerosols will escape; and
3. For a material whose boiling point is above 55°C and where vapors formed by boiling this material could escape only through a column of water above the core, at least 10% of these vapors will escape.

SECTION 15

FINANCIAL CAPABILITY

INFORMATION SUBMITTED WITH APPLICATION LETTER, 10/21/99

FedEx United States

Registration
Shipping
Tracking
Dropoff Locator
Rate Finder

Service Guide
eBusiness Tools
About FedEx
Index|Search
Contact Us

Home

Results

800992629871

Email these tracking results to one or more recipients

Airbill Number : 800992629871

- Delivered To : Ship'g/Receiv'g
- Delivery Location : WASHINGTON DC
- Delivery Date : 10/22
- Delivery Time : 11:47
- Signed For By : R.GRACE
- Status Exception : Rcpt not in/Business closed
- Scan Activity :
  - Delivered GAITHERSBURG MD 10/22 11:47
  - Delivery Exception GAITHERSBURG MD 10/22 10:25
  - Placed on Van GAITHERSBURG MD 10/22 07:56
  - Arrived at FedEx Destination Location GAITHERSBURG MD 10/22 07:30
  - Left FedEx Ramp DULLES VA 10/22 06:48
  - Left FedEx Sort Facility MEMPHIS TN 10/22 02:07
  - Left FedEx Sort Facility MEMPHIS TN 10/22 02:01
  - Left FedEx Origin Location IRVINE CA 10/21 18:06
  - Picked up IRVINE CA 10/21 17:04

E-mail Your Package Tracking Results

Enter up to three E-mail addresses, add your message and submit.

1.
2.
3.

Enter message to be sent with this email.

Send Email

If you have any questions about your shipment, please send us e-mail or contact Customer Service.