UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENVIRONMENTAL ASSESSMENT

FOR THE

TRAINING AND RESEARCH REACTOR OF THE

UNIVERSITY OF MARYLAND

LICENSE NO. R-70

DOCKET NO. 50-166

Description of Proposed Action

This Environmental Assessment is written in connection with the proposed renewal for 20 years of the operating license of the Maryland University Training Reactor (MUTR) of the University of Maryland at College Park, Maryland, in response to a timely application from the licensee dated May 23, 1980, as supplemented. The proposed action would authorize continued operation of the reactor in the manner that it has been operated since facility license No. R-70 was issued in 1974. Currently there are no plans to change any of the structures or operating characteristics associated with the reactor during the renewal period requested by the licensee.

Need for the Proposed Action

The operating license for the facility was due to expire in June 1980. The proposed action is required to authorize continued operation so that the facility can continue to be used in the licensee's mission of education and research.

Alternatives to the Proposed Action

The only reasonable alternative to the proposed action that was considered was not renewing the operating license. This alternative would have led to cessation of operations, with a resulting change in status and a likely small impact on the environment.

Environmental Impact of Continued Operation

The MUTR operates in an existing shielded water tank inside an existing multiple purpose building, so this licensing would lead to no change in the physical environment.

Based on the review of the specific facility operating characteristics that are considered for potential impact on the environment, as set forth in the staff's Safety Evaluation Report (SER)[1] for this action, it is concluded that renewal of this operating license will have an insignificant environmental impact. Although judged insignificant, operating features with the greatest potential environmental impact are summarized below.

Argon-41, a product from neutron irradiation of air during operation, is the principal airborne radioactive effluent from the MUTR during routine operations. Conservative calculations by the staff, based on the total amount of Ar-41 released from the reactor during a year, predict a maximum potential annual whole body dose of less than 1 millirem in unrestricted areas. Radiation exposure rates measured outside of the reactor facility building are consistent with this computation.

The staff has considered hypothetical credible accidents at MUTR and has concluded that there is reasonable assurance that such accidents will not release a significant quantity of fission products from the fuel cladding and, therefore, will not cause significant radiological hazard to the environment or the public.

This conclusion is based on the following:

a) the excess reactivity available under the Technical Specifications is insufficient to support a reactor transient generating enough energy to cause overheating of the fuel or loss of integrity of the cladding,

b) at steady-state power levels of 250 kilowatts, the inventory of fission products in the fuel cannot generate sufficient radioactive decay heat to cause fuel damage even in the hypothetical event of instantaneous total loss of coolant, and

c) the hypothetical loss of integrity of the cladding of the maximum irradiated fuel rod will not lead to radiation exposures in the unrestricted environment that exceed guideline values of 10 CFR 20.

In addition to the analyses in the SER summarized above, the environmental impact associated with operation of research reactors has been generically evaluated by the staff and is discussed in the attached generic evaluation. This evaluation concludes that there will be no significant environmental impact associated with the operation of research reactors licensed to operate at power levels up to and including 2 MWt and that an Environmental Impact Statement is not required for the issuance of construction permits or operating licenses for such facilities. We have determined that this generic evaluation is applicable to operation of the MUTR and that there are no special or unique features that would preclude reliance on the generic evaluation.

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[1] NUREG-1043, "Safety Evaluation Report Related to the Renewal of the Operating License for the Training and Research Reactor at the University of Maryland."

Agencies and Persons Consulted

The staff has obtained the technical assistance of the Los Alamos National Laboratory in performing the safety evaluation of continued operation of the MUTR facility.

Conclusion and Basis for Final No Significant Impact Finding

Based on the foregoing considerations, the staff has concluded that there will be no significant environmental impact attributable to this proposed license renewal. Having reached this conclusion, the staff has further concluded that no Environmental Impact Statement for the proposed action need be prepared and that a Final No Significant Impact Finding is appropriate.

Dated: August 7, 1984

ENVIRONMENTAL CONSIDERATIONS REGARDING THE LICENSING OF RESEARCH REACTORS AND CRITICAL FACILITIES

Introduction

This discussion deals with research reactors and critical facilities which are designed to operate at low power levels, 2 MWt and lower, and are used primarily for basic research in neutron physics, neutron radiography, isotope production, experiments associated with nuclear engineering, training and as a part of the nuclear physics curriculum. Operation of such facilities will generally not exceed a 5 day week, 8 hour day or about 2000 hours per year. Such reactors are located adjacent to technical service support facilities with convenient access for students and faculty.

Sited most frequently on the campus of large universities, the reactors are usually housed in already existing structures, appropriately modified, or placed in new buildings that are designed and constructed to blend in with existing facilities.

Facility

There are no exterior conduits, pipelines, electrical or mechanical structures or transmission lines attached to or adjacent to the facility other than utility service facilities which are similar to those required in other campus facilities, specifically laboratories. Heat dissipation is generally accomplished by use of a cooling tower located on the roof of the building. These cooling towers are on the order of 10' x 10' x 10' and are comparable to cooling towers associated with the air-conditioning system of large office buildings.

Make up for this cooling system is readily available and usually obtained from the local water supply. Radioactive gaseous effluents are limited to Ar 41 and the release of radioactive liquid effluents can be carefully monitored and controlled. These liquid wastes are collected in storage tanks to allow for decay and monitoring prior to dilution and release to the sanitary sewer systems. Solid radioactive wastes are packaged and shipped off-site for storage at NRC approved sites. The transportation of such waste is done in accordance with existing NRC-DOT regulations in approved shipping containers.

Chemical and sanitary waste systems are similar to those existing at other university laboratories and buildings.

Environmental Effects of Site Preparation and Facility Construction

Construction of such facilities invariably occurs in areas that have already been disturbed by other university building construction and in some cases solely within an already existing building. Therefore, construction would not be expected to have any significant affect on the terrain, vegetation, wildlife or nearby waters or aquatic life. The societal, economic and esthetic impacts of construction would be no greater than that associated with the construction of a large office building or similar university facility.

Environmental Effects of Facility Operation

Release of thermal effluents from a reactor of less than 2 MWt will not have a significant effect on the environment. This small amount of waste heat is generally rejected to the atmosphere by means of small cooling towers. Extensive drift and/or fog will not occur at this low power level.

Release of routine gaseous effluent can be limited to Ar 41 which is generated by neutron activation of air. This will be kept as low as practicable by minimum air ventilation of the tubes. Yearly doses to unrestricted areas will be at or below established limits. Routine releases of radioactive liquid effluents can be carefully monitored and controlled in a manner that will ensure compliance with current standards. Solid radioactive wastes will be shipped to an authorized disposal site in approved containers. These wastes should not amount to more than a few shipping containers a year.

Based on experience with other research reactors, specifically TRIGA reactors, operating in the 1 to 2 MWt range, the annual release of gaseous and liquid effluents to unrestricted areas should be less than 30 curies and 0.01 curies respectively.

No release of potentially harmful chemical substances will occur during normal operation. Small amounts of chemicals and/or high-solid content water may be released from the facility through the sanitary sewer during periodic blowdown of cooling tower or from laboratory experiments.

Other potential effects of the facility, such as esthetics, noise, societal or impact on local flora and fauna are expected to be too small to measure.

Environmental Effects of Accidents

Accidents ranging from the failure of experiments up to the largest core damage and fission product release considered possible result in doses of only a small fraction of 10 CFR Part 100 guidelines and are considered negligible with respect to the environment.

Unavoidable Effects of Facility Construction and Operation

The unavoidable effects of construction and operation involves the materials used in construction that cannot be recovered and the fissionable material used in the reactor. No adverse impact on the environment is expected from either of these unavoidable effects.

Alternatives to Construction and Operation of the Facility

To accomplish the objectives associated with research reactors, there are no suitable alternatives. Some of these objectives are training of students in the operation of reactors, production of radioisotopes, and use of neutron and gamma ray beams to conduct experiments.

Long-Term Effects of Facility Construction and Operation

The long-term effects of research facilities are considered to be beneficial as a result of the contribution to scientific knowledge and training.

Because of the relatively low amount of capital resources involved and the small impact on the environment very little irreversible and irretrievable commitment is associated with such facilities.

Costs and Benefits of Facility and Alternatives

The costs are on the order of several millions of dollars with very little environmental impact. The benefits include, but are not limited to some combination of the following: conduct of activation analyses, conduct of neutron radiography, training of operating personnel and education of students. Some of these activities could be conducted using particle accelerators or radioactive sources which would be more costly and less efficient. There is no reasonable alternative to a nuclear research reactor for conducting this spectrum of activities.

Conclusion

The staff concludes that there will be no significant environmental impact associated with the licensing of research reactors or critical facilities designed to operate at power levels of 2 MWt or lower and that no environmental impact statements are required to be written for the issuance of construction permits or operating licenses for such facilities.

NUREG-1043

Safety Evaluation Report

related to the renewal of the operating license for the training and research reactor at the University of Maryland

Docket No. 50-166

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1984

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GPO Printed copy price: $4.50

NUREG-1043

Safety Evaluation Report
related to the renewal of the operating license for the training and research reactor at the University of Maryland

Docket No. 50-166

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

March 1984

ABSTRACT

This Safety Evaluation Report for the application filed by the University of Maryland (UMD) for a renewal of operating license R-70 to continue to operate a training and research reactor facility has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned and operated by the University of Maryland and is located at a site in College Park, Prince Georges County, Maryland. The staff concludes that this training reactor facility can continue to be operated by UMD without endangering the health and safety of the public.

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1 INTRODUCTION

The University of Maryland (UMD, licensee) submitted a timely application to the U.S. Nuclear Regulatory Commission (NRC/staff) for renewal of the Class 104 Operating License (R-70), for its TRIGA*-type training reactor. The application, with supporting documentation, was by letter dated May 23, 1980, signed by the president of the university and properly notarized. The application requested renewal of the license for a period of 20 years. The licensee is permitted to operate the reactor within the conditions authorized in past license amendments in accordance with Title 10 of the Code of Federal Regulations, Paragraph 2.109 (10 CFR 2.109), until NRC action on the renewal request is completed.

The renewal application references information regarding the original design of the reactor facility and contains information about modifications to the facility made since initial licensing.

The application also includes a revised Safety Analysis Report, information for an environmental impact appraisal, financial information, an Operator Requalification Program, revised Technical Specifications, and a revised Physical Security Plan which is protected from public disclosure under 10 CFR 2.790(d)(1) and 10 CFR 9.5(a)(4).

The staff's technical review with respect to issuing a renewal operating license to UMD has been based on visits to the facility and on the information contained in the renewal application and supporting documents plus responses to requests for additional information. This material is available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555. This Safety Evaluation Report (SER) was prepared by R. E. Carter, Project Manager, Division of Licensing, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. Major contributors to the technical review include the project manager and J. E. Hyder, S. Pillay, and J. L. Sapir of the Los Alamos National Laboratory under contract to NRC.

The purpose of this SER is to summarize the results of the safety review of the Maryland University Training Reactor (MUTR) and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for renewal of the license for operation of the MUTR facility at steady-state power levels up to and including 250 kW. The facility was reviewed against Federal regulations (10 CFR 20, 30, 50, 51, 55, 70, and 73), applicable regulatory guides (principally Division 2, Research and Test Reactors) and appropriate accepted industry standards (American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series). Because there are no specific accident-related regulations for research reactors, the staff has at times compared calculated hypothetical radiation dose values with related standards in 10 CFR 20, "Standards for Protection Against Radiation," both for employees and the public.

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*Training, Research, Isotopes General Atomic.

The MUTR was initially licensed for operation at 10 kWt in October 1960 as an open pool-type reactor. From 1960 until 1969 the MUTR operated with fuel of the Materials Testing Reactor (MTR) type. In 1969 the licensee initiated a long-range program of upgrading the reactor facility, and after major changes in fuel type, control rod systems, and control instrumentation, the MUTR was relicensed in June 1974 to operate with TRIGA-type uranium-zirconium-hydride fuel at steady-state power levels up to 250 kW. The MUTR has been operated under that license since 1974.

1.1 Summary and Conclusions of Principal Safety Considerations

The staff's evaluation considered the information submitted by the licensee, past operating history recorded in annual reports submitted to the Commission by the licensee, and reports by the Commission's Office of Inspection and Enforcement. In addition, as part of its licensing review of several TRIGA reactors, the staff obtained laboratory studies and analyses of several accidents postulated for the TRIGA reactor. The resolution of principal issues reviewed for the MUTR reactor were

(1) The design, testing, and performance of the reactor structure and systems and components important to safety during normal operation are inherently safe, and safe operation can reasonably be expected to continue.

(2) The expected consequences of a broad spectrum of postulated credible accidents have been considered, emphasizing those that could lead to a loss of integrity of fuel-element cladding. The staff performed conservative analyses of the most serious credible accidents and determined that the calculated potential radiation doses outside the reactor room would not exceed 10 CFR 20 doses for unrestricted areas.

(3) The licensee's management organization, conduct of training and research activities, and security measures are adequate to ensure safe operation of the facility and protection of special nuclear material.

(4) The systems provided for the control of radiological effluents can be operated to ensure that releases of radioactive wastes from the facility are within the limits of the Commission's regulations and are as low as reasonably achievable (ALARA).

(5) The licensee's Technical Specifications, which provide operating limits controlling operation of the facility, are such that there is a high degree of assurance that the facility will be operated safely and reliably.

(6) The financial data provided by the licensee are such that the staff has determined that the licensee has sufficient revenues to cover operating costs and eventually to decommission the reactor facility.

(7) The licensee's program for providing for the physical protection of the facility and its special nuclear material complies with the requirements of 10 CFR 73.

(8) The licensee's procedures for training reactor operators and the plan for operator requalification are acceptable. These procedures give reasonable assurance that the reactor facility will be operated with competence.

(9) The licensee has submitted an Emergency Plan in compliance with the existing applicable regulations. This item is discussed further in Section 13.3.

1.2 Reactor Description

The MUTR is a heterogeneous open pool-type TRIGA reactor. The core is cooled by natural convection of light water, moderated by zirconium hydride (ZrH_x) and water, and reflected by water and graphite. The core is located near the bottom of an aluminum tank that is embedded in an above ground poured concrete shielding structure. The reactor core currently contains 93 standard stainless-steel-clad uranium-zirconium hydride ($U-ZrH_x$) fuel rods. The rods are assembled in units of four to a cluster that are spaced in a rectangular array by a 13-cm-thick aluminum grid plate.

The reactor is designed and licensed to operate at a steady-state thermal power level of 250 kW, using uranium fuel enriched to less than 20% in the U-235 isotope.

1.3 Reactor Location

The reactor is located on the ground floor in the Chemical and Nuclear Engineering Building on the main campus of the University of Maryland, College Park, Prince Georges County, Maryland.

1.4 Shared Facilities and Equipment and Special Location Features

The reactor facility shares its utilities--electricity, water, natural gas, nonradioactive sewage, and the like--with the remainder of the Chemical and Nuclear Engineering Building. The reactor room has its own heating, cooling, and ventilation units, and a primary coolant system that transfers heat in a heat exchanger system to a single-pass city water secondary loop.

1.5 Comparison With Similar Facilities

The reactor fuel rods are similar to those in most of the 58 TRIGA-type reactors in operation throughout the world, 27 of which are in the United States, and 24 of these are licensed by the NRC. The instruments and controls are typical of the TRIGA reactors and similar in principle to most of the nonpower reactors licensed by the NRC.

1.6 Nuclear Waste Policy Act of 1982

Section 302(b)(1)(B) of the Nuclear Waste Policy Act of 1982 provides that the NRC may require, as a precondition to issuing or renewing an operating license for a research or test reactor, that the applicant shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. DOE (R. L. Morgan) has informed the NRC (H. Denton) by letter dated May 3, 1983, that it has determined that universities and other government agencies operating nonpower reactors have entered into contracts with DOE that provide that DOE retain title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing.

Because the University of Maryland has entered into such a contract with DOE the applicable requirements of the Waste Policy Act of 1982 have been satisfied.

2 SITE CHARACTERISTICS

2.1 Geography

The site for the MUTR is on the northeastern quadrant of the main campus located at College Park, Prince Georges County, Maryland. The reactor is about 14 km northeast of the center of Washington, D.C., and 6 km from the nearest point of the District of Columbia line.

The general terrain surrounding the reactor site is characterized by low, gently rolling hills. The reactor building is on the relatively flat portion of the campus near lower elevations. To the east the ground slopes slightly downward toward a shallow stream, Paint Branch, about 366 m away.

The relative location of the MUTR within the campus is at the center of the concentric circles shown on the map of the University of Maryland College Park Campus, Figure 2.1. Figure 2.2 shows towns and residential areas adjacent to the reactor site. The nearest off-campus public residence is approximately 370 m from the building housing the reactor.

2.2 Demography

The College Park Campus of the University of Maryland has a peak daytime population (students, faculty, and other persons) of approximately 20,000. The distribution of people within the campus during working hours is summarized in Table 2.1. This is based on maximum occupancy expected for each university building during the academic year except for special occasions such as athletic events. The average peak daytime population tabulated in Table 2.1 with use of Figure 2.1 permits a detailed analysis of the campus population distribution in all areas and directions immediately around the reactor building. Concentric circles have been imprinted on Figure 2.1 to assist in the analysis of the population distribution. The peak daytime population within approximately 457 m of the reactor building is 12,000. Figure 2.2 shows that the university is located generally within a high population-density suburban area of the greater metropolitan area north of Washington, D.C.

2.3 Nearby Industrial, Transportation, and Military Facilities

2.3.1 Transportation Routes

The Washington National Airport is approximately 16 km from the university campus, and only a small minority of the larger commercial flights have air-routes over the campus. The university airport is approximately 1 km from the reactor building, but only small, private planes make use of this facility. The nearest railroad carrying interstate freight also is at a distance of approximately 1 km, as is the nearest multilane highway carrying large amounts of freight traffic. The Capital Beltway, a major automotive traffic artery, is approximately 3.8 km to the north of the reactor building.

There are no nearby major military installations nor any large industrial complexes that give rise to heavy vehicular traffic on the highways within several hundred meters of the reactor building.

2.3.2 Nearby Facilities

The campus is located in an area containing no major industries. There are, however, nearby shopping areas, parks, schools, and urban homes and apartments.

Because there are no industrial or military facilities in the near vicinity of the reactor site that could directly or indirectly cause accidental damage to the reactor, the staff concludes that such accidents need not be hypothesized.

2.4 Meteorology

Washington, D.C., and Prince Georges County, Maryland, lie near the western edge of the Atlantic coastal plain. The proximity of the ocean has a marked influence on the weather conditions.

The warmest weather occurs, on the average, during the middle of July with maximum average daily temperature of 87.4°F. The record high temperature of 106°F occurred on July 20, 1930. The coldest weather usually occurs in the late January and early February, when the maximum average temperature is 43.8°F and the minimum average temperature is 28.4°F.

The normal annual precipitation is about 40 in. Because of a uniform moisture supply throughout the year, no well-pronounced wet or dry seasons are evident. Thunderstorms during the summer months often bring sudden and heavy rain showers. Most thunderstorms are not accompanied by high winds, although on June 9, 1929, a thunderstorm with wind gusts up to 100 mph was recorded. The reactor site is somewhat protected from high winds by the higher grounds to the west. Two hailstorms have been recorded with the resultant damage of $100,000 or more; one in April 1938 and the other in May 1953. Tornadoes occur rarely, but three of them with resulting damage of $100,000 or more occurred; two in April 1932 and one in November 1927. In April 1973, a tornado struck in the vicinity of suburban Fairfax, Virginia, causing an estimated $15,000,000 damage.

Tropical disturbances occasionally influence the local weather with high winds and heavy rainfalls, but extensive damage from this cause has been rare. Three major hurricanes have been recorded. On October 15, 1954, Hurricane Hazel caused a peak gust of wind at 98 mph, but only 1.73 in. of rainfall was recorded. On August 12 and 13, 1955, Hurricane Connie produced 6.60 in. of rainfall. Flooding from the rains of Hurricane Agnes in 1973 caused an estimated $300,000,000 damage in Virginia, Maryland, Delaware, and the District of Columbia, but no significant damage at the site of the reactor.

Snow accumulations of more than 10 in. are relatively rare. The greatest recorded snowfall from a single storm was 28 in. accumulating in 2 days in January 1922, but snowfalls of this magnitude are extremely rare.

Because the reactor site is close to Washington, D.C., information on weather is based on climatological data for that city. Table 2.2 summarizes the weather characteristics in the vicinity of the reactor site.

On the basis of the meteorological data presented in the licensee's SAR, the staff concludes that the meteorological conditions at the reactor site do not pose a significant risk of damage to the reactor nor render the site unacceptable for the facility.

2.5 Geology

The site is underlain by about 61 m of soils belonging to the Potomac Group of Lower Cretaceous age, principally the Patuxent Formation overlain by a thin layer of Arundel clay. Precambrian igneous and metamorphic basement rocks underlie the Patuxent Formation. The basement is exposed at the surface at the Fall Zone, a few km west of the site, and its surface slopes to the east. The overlying Coastal Plain strata dip toward the east and thicken in that direction. Many faults have been mapped in the Piedmont west of the Fall Zone where the basement outcrops or is exposed in excavations. It is likely that similar faults are present in the basement beneath the site. Most of these faults are several hundred million years old and do not penetrate the overlying Coastal Plain sediments such as the Patuxent Formation and Arundel clay at the site. However, several faults have been mapped within a radius of 80 km of the site that do offset Coastal Plain strata (Darton, 1950; Dryden, 1932; Jacobeen, 1972; and Mixon and Newell, 1970). These faults were evaluated by the NRC during licensing activities for several nuclear power reactor sites such as Douglas Point, Summit, Hope Creek and North Anna and found to be noncapable within the meaning of Appendix A to 10 CFR 100.

The Patuxent Formation is a predominantly white, yellow, gray, and brown sand interbedded with sandy clay. It is this layer that contains appreciable amounts of water, yielding several hundred gallons a minute to drilled wells. Arundel clay is a reddish brown material that is not an important water-bearing formation. Its porosity is high but its water permeability is quite low. Locally the sediments have been indurated by calcium carbonate and iron oxide, which results in an unusually low porosity.

2.6 Hydrology

The principal streams within Prince Georges County flow in a southerly direction and are tributary to either the Potomac River or the Chesapeake Bay. Figure 2.3 shows the location of the principal streams.

From the reactor site surface, runoff can be expected to follow the gradual slope to the east toward Paint Branch about 366 m from the reactor building. A 48-in. storm-sewer system at the university leads to the east and terminates at Paint Branch in about the same location.

Paint Branch flows to the southeast and joins the Northeast Branch of the Anacostia River about 2 mi from the reactor site. The Anacostia River flows into the Potomac River at Washington, D.C. In general, all streams within several miles of the university flow to the south and eventually join the Potomac River.

The sanitary sewer drainage system carries waste water and sewage from the reactor building. It consists of an 8-in. sewer line, which joins a 15-in. line about 213 m east of the reactor building. The 15-in. sanitary sewer continues eastward for another 152 m where it joins the 36-in. trunk sewer belonging to the Washington Suburban Sanitary Commission. The Washington Suburban Sanitary Commission's lines feed into the District of Columbia's sewer lines and subsequently lead to the Blue Plains Treatment Plant along the Potomac River in Washington, D.C.

The reactor site, the University of Maryland, and all surrounding towns are supplied with water from the facilities of the Washington Suburban Sanitary Commission. This water is obtained from the Patuxent River upstream of Laurel, Maryland, about 16 km northeast of the reactor site and is treated at the Patuxent Filtration Plant in Laurel. No drainage from the vicinity of the reactor site leads into this water supply system.

On the basis of the above information, the staff concludes that the hydrological conditions at the reactor facility do not render the site unacceptable for the training reactor location.

2.7 Seismology

The earthquake risk in the District of Columbia and Maryland is characterized as a seismic risk zone where only minor earthquake damage may be expected. Historically there have been 12 earthquakes within about 80 km of College Park, Maryland. The largest of these had a maximum Modified-Mercalli intensity (MMI) of V. The earliest recorded earthquake in Maryland occurred in Annapolis in 1758. Maryland was in the felt zone for the great earthquake series of 1811-1812, which was centered in Missouri. The most severe earthquake recorded in Virginia history (Giles County, 1897) shook most of Maryland. An earthquake centered near Luray, Virginia, in 1918 was reported felt in College Park. A single felt report was received from West Hyattsville, Maryland, in 1969 associated with a minor earthquake near Elgood, West Virginia. The Charleston, South Carolina, earthquake of 1886 was reported to have had an MMI of IV in the College Park area.

The staff concludes that the history of infrequent earthquake activity and no damaging earthquakes in the vicinity of the site in recorded history supports the conclusion that the risk of seismic-induced hazards to the MUTR is not significant.

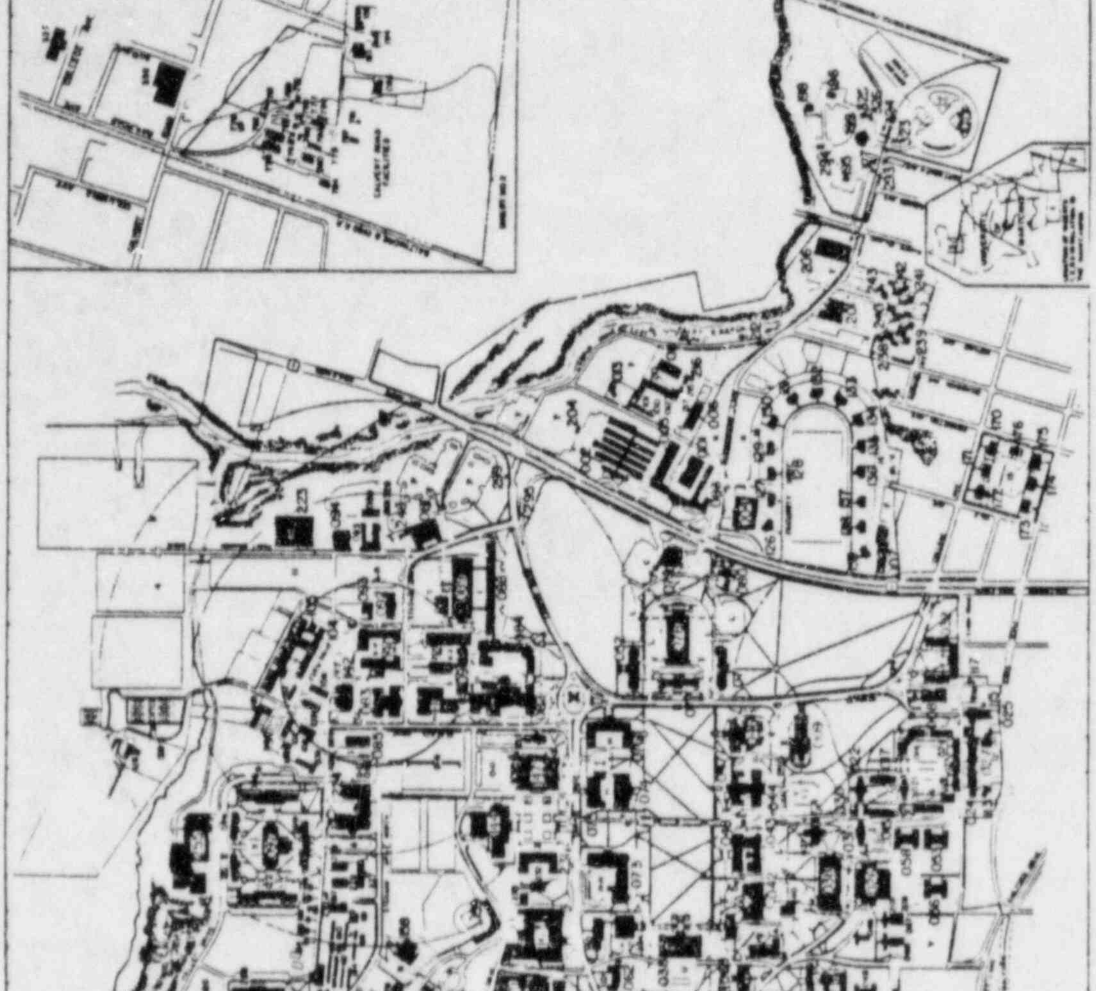

Figure 2.1 Map of the University of Maryland College Park Campus
Source: University of Maryland SAR, 1980

Figure 2.2 Towns and residential areas adjacent to the site
Source: University of Maryland SAR, 1980

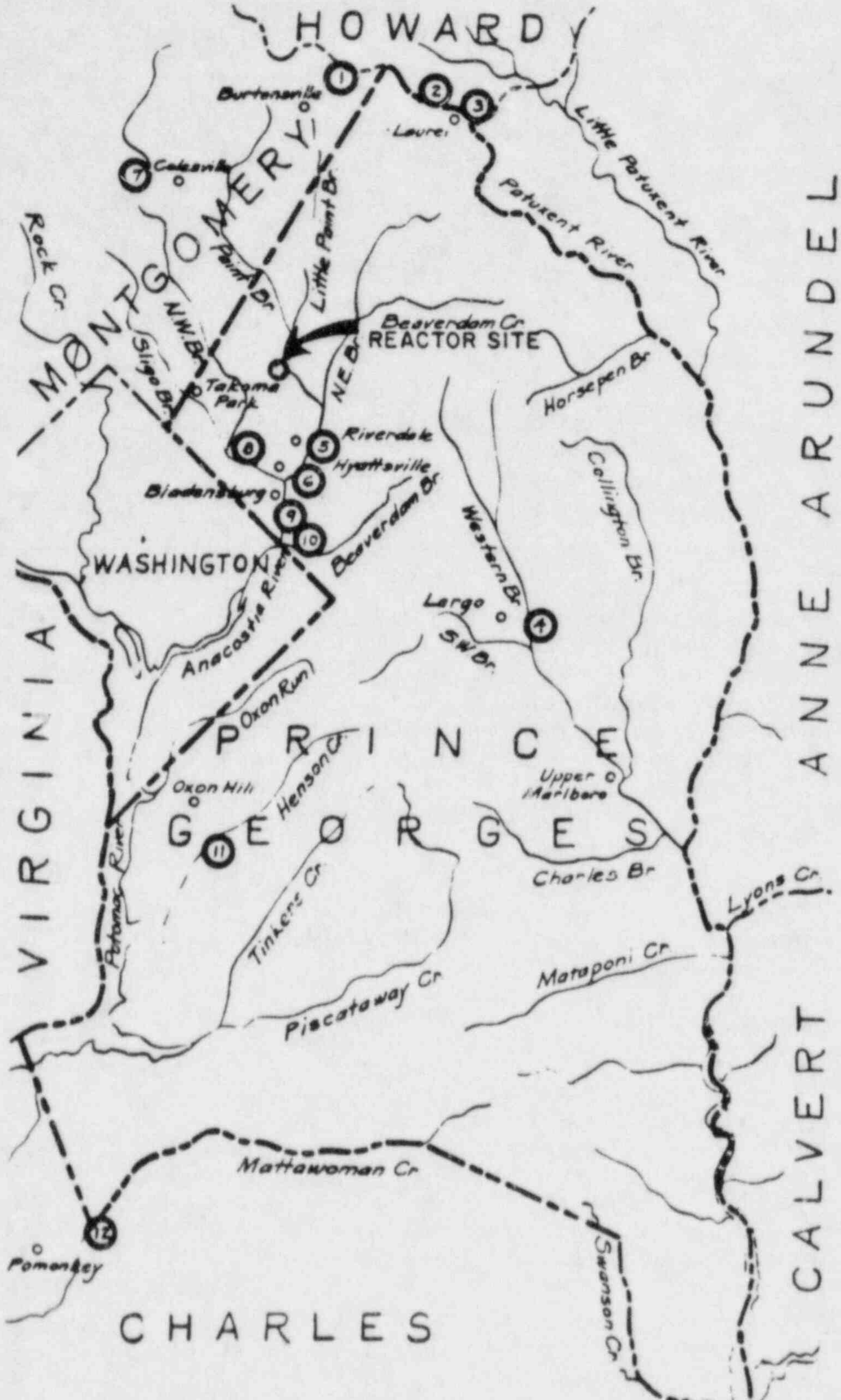

Figure 2.3 Location of principal streams
Source: University of Maryland SAR, 1980

Table 2.1 Estimated population during working hours

Distance from reactor, meters	Total
0-152	2480
152-305	5740
305-457	4278

Table 2.2 Climatological profile, Washington, D.C., and environs

Parameters	Values
Precipitation (in.)	
Average yearly rainfall	40.0
Average yearly snowfall	1.94
Average wettest month - August	4.75
Average dry month - January	2.59
Greatest rainfall in 24 hours (August 1928)	7.31
Greatest snowfall in 24 hours (January 1922)	25.0
Greatest unmelted snowfall (February 5-13, 1899)	34.2
Relative Humidity (%)	
Lowest monthly average humidity	58
Highest monthly average humidity - August, September	72
Lowest recorded value of humidity - March 11, 1929	6
Yearly average humidity	64
Temperature (°F)	
Yearly mean daily maximum	66.3
Yearly mean daily minimum	48.3
Highest monthly average maximum - July	87.4
Highest monthly average minimum - July	69.4
Lowest monthly average maximum - January	43.8
Lowest monthly average minimum - January	28.4
Record highest - July 20, 1930	106
Record lowest - February 11, 1899	-15

Table 2.2 (Continued)

Parameters	Values
Weather Conditions (average number of days/year)	
Clear	102
Cloudy	158
Partly cloudy	105
Rain (0.01 in. or more)	112
Snow (0.01 in. or more - melted)	5
Thunderstorms	29
Maximum temperature - 90°F or more	37
Minimum temperature - 32°F or below	83
100% relative humidity frequence	10.4
Heavy fog visibility 0.40 km or less	13

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

The licensee's Final Safety Analysis Report (May 1980) provides information on the design and functions of the reactor building and the reactor systems and auxiliary systems.

3.1 Wind Damage

The Washington metropolitan area experiences very few extreme wind conditions such as tornadoes or inland hurricanes. Further, the reactor building is constructed of a steel frame and poured concrete walls and floor, with a brick facing on the walls, and the aluminum reactor tank is embedded in a poured concrete biological shield that is faced with steel plates in the lower regions. On the basis of the above information, the staff concludes that wind or storm damage to the MUTR reactor facility is very unlikely.

3.2 Water Damage

The reactor building is situated near the bottom of a gently sloping terrain, but well above the flood plain. Therefore, the staff concludes that there is reasonable assurance that significant damage to the reactor because of flooding is not likely enough to render the site unsuitable as the location of the reactor.

3.3 Seismic-Induced Reactor Damage

The information on past seismic activity and the likelihood of future earthquakes in the Washington, D.C., area indicate that the MUTR is located in a region of low probability of severe seismic activity. In the event of an earthquake causing catastrophic damage to the reactor building and/or the reactor pool, water might be released. However, Section 14 of this SER shows that loss of coolant in the MUTR does not lead to core damage, and mechanical damage to fuel cladding would release only a small fraction of the fission product inventory. On the basis of these considerations, the staff concludes that the risk of radiological hazard resulting from seismic damage to the reactor facility is not significant.

3.4 Mechanical Systems and Components

The mechanical systems important to safety are the neutron-absorbing control rods suspended from the reactor superstructure. The motors, gear boxes, electromagnets, switches, and wiring are all above the level of the water and readily accessible for visual inspection, testing, and maintenance. A preventive maintenance program has been in effect for many years at the MUTR to ensure that operability of the reactor systems is in conformance with the performance requirements of the Technical Specifications.

The recent history of operation of the MUTR indicates few malfunctions of electro-mechanical systems, no persistent malfunction of any one component, and most malfunctions were one of a kind (i.e., few repeats). (See Inspection Reports and licensee reports to the Commission.) On the basis of the above information the staff concludes that there has not been significant deterioration of equipment with time or with operation and there is reasonable assurance that continued operation of the MUTR facility will not increase the risk to the public.

3.5 Conclusions

The MUTR was designed and built to withstand all credible and probable wind and water events associated with the site. A seismic event has a small likelihood of occuring and the consequences of such an event would not pose a significant radiological hazard to the public (see Section 14). There is no evidence of significant deterioration of systems or components. Therefore, the staff has concluded that the construction of the facility is acceptable and that continued operation, as proposed, will not cause significant radiological risk to the public.

4 REACTOR

The Maryland University Training Reactor (MUTR) is a heterogeneous pool-type research reactor incorporating TRIGA-type solid uranium-zirconium hydride ($U-ZrH_x$) fuel/moderator elements. The reactor core is immersed in an open tank of light water that acts as moderator, coolant, and partial shield. Heat generated from fission is removed from the fuel by natural convection of the water coolant.

Reactor control is achieved by insertion and withdrawal of neutron-absorbing control rods. The reactor does not have an installed pulsing capability. The MUTR was first licensed to operate in October 1960 at a power level of 10 kW using MTR-type fuel elements. In 1974 the reactor was converted to operate with the current $U-ZrH_x$ fuel at an authorized maximum steady-state power level of 250 kW. The reactor has been used in a broad range of teaching, research, and service programs in basic and applied areas of science and engineering.

4.1 Reactor Core

The reactor core consists of a lattice of 93 cylindrical $U-ZrH_x$ fuel-moderator rods assembled into four-rod clusters (elements) and three control rods. The fuel elements are held in a rectangular array by cluster adapters inserted into a bottom grid plate. The grid plate provides a 9-by-5 array of holes that will accommodate either fuel clusters or graphite reflector assemblies. The active (or fueled) region of the reactor core forms an approximate rectangular parallelepiped ~46 cm long, 30 cm wide, and 38 cm high and contains 3.4 kg of ^{235}U. Water coolant/moderator occupies approximately 30% of the core volume.

The radial neutron reflector is composed partially of graphite in the form of reflector assemblies and the thermal column assembly and partially of the pool water in which the core is immersed. Top and bottom axial neutron reflection is provided by graphite slugs incorporated into the individual fuel rods. A schematic of the core configuration is shown in Figure 4.1.

4.1.1 Fuel Elements

The MUTR uses TRIGA-type stainless-steel-clad cylindrical fuel rods in which the enriched uranium is homogeneously mixed with a zirconium hydride neutron moderator. The fuel consists of cylindrical rods of $U-ZrH_x$ containing 8.5 W% uranium enriched to slightly less than 20%. The nominal weight of ^{235}U in each unused fuel rod is 37 g. The hydrogen-to-zirconium atom ratio of the fuel-moderator material is approximately 1.7:1. To facilitate hydriding, a 0.46-cm-diameter hole was drilled through the center of the active fuel section; a zirconium rod was inserted into this hole after hydriding was completed.

The fuel section of each rod is 0.38 m long and 3.5 cm in diameter. Graphite end plugs that are 8.6 cm long and located above and below the fuel region serve as axial neutron reflectors. The fueled section and graphite end plugs are contained in a 0.051-cm-thick, type 304, stainless-steel-walled can, which is welded to top and bottom stainless-steel end fittings. Each rod is about 0.9 m long and weighs about 3.4 kg. These rods are nearly identical to standard TRIGA fuel rods, except that they are made slightly smaller--3.58 cm outer diameter versus 3.73 cm outer diameter--to maintain the proper metal-to-water ratio in the core in the four-rod cluster geometry.

One fuel position contains a special instrumented fuel rod into which three thermocouples were fitted during fabrication. The sensing tips of the fuel-element thermocouples are located near the vertical centerline, at the axial center of the fuel section and 2.5 cm above and below the center, respectively. In other respects, the instrumented fuel rod is identical to the normal fuel rod. The thermocouples monitor the fuel rod temperatures and provide a scram signal in the event of high fuel temperature.

A four-rod fuel cluster was developed to allow conversion of MTR-type reactors to TRIGA-fueled reactors. The cluster consists of an aluminum bottom adapter, four stainless-steel-clad fuel-moderator rods, and an aluminum top handle. The four fuel rods are threaded into the bottom adapter, which is designed to fit into and rest on the existing MTR-type bottom grid plate. The top handle of the cluster serves as a lifting fixture and a spacer for the upper ends of the fuel rods. A stainless-steel locking plate fastens the top handle to the fuel elements. A drawing of a fuel cluster and a detail of an individual fuel rod is shown in Figure 4.2.

Graphite reflector assemblies may be used to fill grid positions not occupied by fuel-moderator elements, control rods, or other core components. These assemblies are of the same general dimensions as the four-rod fuel clusters but are filled with graphite contained in aluminum cans. Several grid plate positions are not used and remain vacant during operation.

4.1.2 Control Rods

Power levels in the MUTR are regulated by three control rods--two shim rods and one regulating rod. The neutron poison material in all three rods is powdered boron carbide contained in a sealed aluminum tube.

Overall, the control rods are 43 cm long and 3.18 cm in diameter with a poison section length of 38 cm. The control rod tube is threaded into a connecting rod that extends from the core to the drive mechanism mounted on the reactor bridge support structure.

To accommodate a control rod, one fuel rod in a cluster is replaced by a control rod guide tube that has the same outside diameter and threaded bottom fitting as a fuel rod. The upper aluminum handle and locking plate on a normal fuel cluster was modified to accommodate the control rod guide tube.

The control rod locations in the current core configuration are shown in Figure 4.1.

4.2 Reactor Tank

The reactor tank is an all-welded cylindrical aluminum vessel made of 0.95-cm plate on the sides and 1.25-cm aluminum plate on the bottom. The tank is 2.1 m in diameter and 6.48 m deep, with a capacity of 6,000 gal.

The reactor core is positioned near the bottom of the tank under approximately 5.3 m of light demineralized water, which serves as a radiation shield, neutron moderator, and reactor coolant. The natural thermal convection of the water transfers the heat generated in the core to the pool water.

When necessary, the pool water may be pumped through an external heat exchanger system that ultimately disposes of the heat to city water that is released to the sanitary sewer system. There are five flanged nozzles welded on to the reactor tank: four are for two radial experiment beam tubes and the two ends of a through tube, and the fifth is for the thermal column.

The reactor tank incorporates a fuel storage rack, as described in Section 9 of this report.

4.3 Support Structure

The core is supported vertically by a lower aluminum grid plate 0.127 m thick, 0.74 m long, and 0.47 m wide. The plate itself is bolted to four corner grid-plate stands, which are, in turn, bolted to pads welded to the bottom of the reactor tank. The grid plate is positioned 0.33 m from the reactor tank floor.

The grid plate contains a 9-by-5 pattern of holes 5.7 cm in diameter. The bottom fuel cluster adapter is designed to fit snugly into these holes and rest on the grid plate, thereby providing support and alignment for the fuel element clusters.

The top of the cluster adapter contains four tapped holes into which the fuel rods are threaded. The bottom end fittings on the fuel rods incorporate a flange at the base of the threads, allowing the fuel to seat firmly on the adapter and be rigidly supported in cantilever fashion. The top handle of the cluster serves as a lifting fixture and provides vertical alignment for the upper ends of the fuel rods.

A bridge support structure mounted on the reactor tank provides support for in-core detectors, control rod drives, startup neutron source mechanism, water diffuser pipes, pool water instrumentation, and a pneumatic sample irradiation tube.

4.4 Reactor Instrumentation

The operation of the MUTR is monitored by instrumentation channels that measure fuel element temperature, neutron flux, radiation exposure rate, and pool water level. Thermocouples in an instrumented fuel rod provide continuous information on fuel material temperature during steady-state operation. This signal is displayed on the control console and is used to initiate a reactor scram if preset temperature limits are exceeded.

Three neutron-sensitive channels (one fission chamber, one compensated ion chamber, and one uncompensated ion chamber) indicate reactor power over the range from neutron source level of ~1 mW to full power. The power is displayed at the reactor console, and the signals from the fission chamber and the uncompensated ion chamber are used to initiate a reactor scram if preset power levels are exceeded. The reactor period also is measured and displayed at the reactor control console and used to initiate a reactor scram.

Area radiation monitors provide an indication in the control room of gamma radiation levels at selected locations within the reactor building. In case preset radiation limits are exceeded, these monitors will initiate audio and visual alarms both locally and in the control room, scram the reactor, and turn off operating exhaust fans. Further details concerning the reactor instrumentation system are discussed in Section 7 of this report.

4.5 Biological Shield

The biological shield of the MUTR consists of ordinary concrete, steel plates, and water. The pool tank and shield structure provide a minimum of 0.6 m of water and 1.98 m of concrete on all sides of the core, except the thermal column side. This thickness extends from floor level to 3.35 m above the floor or 2.44 m above the core centerline. For the next 0.6 m above this level, the concrete shield is 1.5 m thick and then is reduced to 0.9 m thick to the top of the reactor tank. Vertical shielding is provided by 5.3 m of pool water above the core.

Special shielding plugs are provided for the experiment facilities. Shielding through the thermal column includes 1.5 m of graphite, 0.3 cm of boral, 2.5 cm of steel, a layer of lead bricks (10 cm), and 1.0 m of concrete. Inner and outer shield plugs are provided for the two beam tubes and each end of the through tube. These plugs, along with additional materials, include 0.3 cm of boral, 0.64 cm of aluminum, 1.47 m of concrete, 10 cm of lead, and 5.1 cm of steel.

4.6 Dynamic Design Evaluation

Operation of the reactor is accomplished by the use and manipulation of the control rods in response to observed changes in measured reactor parameters such as neutron flux (reactor power) and fuel temperature provided by the instrumentation channels. In addition, interlocks in the control rod circuits prevent inadvertent reactivity additions, and a scram system initiates rapid, automatic shutdown if safety settings are exceeded.

Further stability and safety during both steady-state and transient conditions are incorporated into TRIGA-type reactors by virtue of the large, prompt, negative temperature coefficient of reactivity inherent in the U-ZrH_x fuel-moderator material. The negative temperature coefficient is primarily a result of the neutron spectrum-shifting properties of ZrH_x at elevated temperatures, which increase the leakage of slow neutrons from the fuel-bearing material into the water moderator where they are absorbed preferentially. The reactivity decrease is a prompt effect because the fuel and ZrH_x are mixed intimately, and thus the ZrH_x temperature rises essentially simultaneously with the reactor power. An additional contribution to the prompt, negative temperature coefficient of reactivity is the Doppler broadening of ^{238}U resonances at elevated temperatures that increases nonfissioning neutron capture in these resonances (GA-0471, 1958; Simnad, 1976; GA-4314, 1980).

The large, prompt, negative temperature coefficient inherent in ZrH_x fuel rapidly and automatically acts to compensate for insertions of excess reactivity, terminating potential excursions without depending on electronic or mechanical protective systems or operator protective action. Therefore, it serves as a backup safety feature mitigating the effects of accidental reactivity insertions. On the basis of this temperature coefficient feed-back mechanism, TRIGA-type reactors with fuel exactly like the MUTR have been licensed and operated routinely in a pulsing mode, with step insertions of reactivity up to 3.5% Δk/k (GA-0471, 1958; Simnad, 1976; GA-4314, 1980).

4.6.1 Excess Reactivity and Shutdown Margin

The proposed Technical Specifications for the MUTR limit the maximum excess reactivity to 2.5% Δk/k (~3.60$) in the cold, xenon-free condition. The Technical Specifications also require a minimum shutdown margin of 0.35% Δk/k (0.50$) with the highest worth control rod fully withdrawn.

The total reactivity worth of all nonsecured experiments is limited to 1.00$ and a single secured experiment to 1.00$. When multiple experiments are being conducted, the individual worth of any individual experiment cannot exceed 1.00$.

The vendor of TRIGA-type fuel has conducted numerous experiments to evaluate the fuel performance employing rapid insertions of reactivity up to 3.5% Δk/k (5.00$). The experiments revealed no apparent fuel damage. These experiments encompass all credible reactivity insertions possible with the authorized reactivity conditions at the MUTR. Accidents are discussed in Section 14.

The excess reactivity of the current MUTR core is 1.21% Δk/k (1.73$). The control rod worths are 2.80$ and 3.00$ for the shim rods and 2.42$ for the regulating rod, yielding a total rod worth of 8.22$. Under these conditions, the shutdown margin with the highest-worth rod fully withdrawn is 3.49$ (1.73$ - 2.80$ - 2.42$). Therefore, the current configuration meets the excess reactivity and shutdown requirements. With all control rods fully inserted, the current core is subcritical by approximately 6.49$.

4.6.2 Normal Operating Conditions

The MUTR Technical Specifications impose a safety limit of 1,000°C for the maximum fuel temperature under any conditions of operation. The safety limit for high-hydride ($ZrH_{1.7}$) stainless-steel-clad fuel elements is based on preventing excessive stress buildup in the cladding because of hydrogen pressure from the thermal disassociation of ZrH_x. Based on theoretical and experimental evidence (Simnad, 1976; GA-4314, 1980), the limit of 1,000°C represents a conservative value to provide confidence that the fuel elements will maintain their integrity and that no cladding damage will occur. A further provision in the Technical Specifications limits reactor power level to provide assurance that the safety limit will not be exceeded. At the maximum licensed steady-state power level of 250 kW, the maximum fuel temperature is ~300°C. Limiting safety systems settings are established in the Technical Specifications to shut the reactor down if the steady-state power level were to exceed 300 kW or the measured fuel temperature were to exceed 400°C. Based on radial and local power distributions in the reactor core, these requirements ensure that the safety limit of 1,000°C will not be exceeded or approached in any fuel rod in the core.

4.6.3 Assessment

The staff concludes that the inherent large, prompt, negative temperature coefficient of reactivity of U-ZrH_x fuel moderator provides a basis for safe operation of the MUTR in the steady-state mode. Furthermore, the Technical Specifications provide that the core excess reactivity and experiment reactivity worths be limited so that the reactor can always be brought to a subcritical condition even if the highest worth control rod were fully withdrawn. The current core configuration meets all of these requirements, and any other core configuration made by the licensee also must be governed by the Technical Specifications.

The safety limit at the MUTR is based on theoretical and experimental investigations and is consistent with that used at other similar reactors. Adherence to this safety limit should provide assurance that fuel element integrity will be maintained. Operating data at the maximum licensed steady-state power show that the maximum fuel element temperatures remain well below the prescribed safety limit. Research reactors with TRIGA-type fuel similar to the MUTR have demonstrated safe and reliable operation at steady-state power levels up to 1.5 MW and pulse reactivity insertions up to 5.00$ (Simnad, 1976; GA-4314, 1980). On the basis of the above considerations, the staff concludes that there is reasonable assurance that the MUTR can be operated safely at 250 kW steady-state power level with the reactivity conditions as limited by the Technical Specifications.

4.7 Functional Design of Reactivity Control System

The power level in the MUTR is controlled by the operation of three control rods (two shim and one regulating rod) all of which contain boron as the neutron poison. The location of the three control rods is shown in Figure 4.1. Rod movement is accomplished using rack-and-pinion electromechanical drives. Each control rod drive system is energized from the control console through its own independent electrical cables and circuits, which tends to minimize the probability of multiple malfunctions of the drives. On receipt of a scram signal, all three control rods will fall by gravity into the core, thereby shutting down the reactor.

4.7.1 Control Rods

The control rod drive assemblies are mounted on a bridge assembly over the pool and each consists of a nonsynchronous, two-phase electric motor coupled to a rack-and-pinion drive system. A draw tube connected to the rack supports an electromagnet that, in turn, engages an iron armature attached to the upper end of a long connecting rod. The control rod absorber is attached to the lower end of the connecting rod. During normal operation, the electromagnet is energized and in contact with the armature, and the motorized system will insert or withdraw the control rod at a constant rate of approximately 48 cm/min. If power to the electromagnet is interrupted, the connecting rod is released and the control rod falls by gravity into the core, rapidly decreasing reactivity of the reactor (scramming).

Limit switches mounted on the drive assembly actuate circuits that indicate on the control console the up (fully withdrawn) and down (fully inserted) position of the magnet, the down position of the rod, and whether the magnet current is on and the magnet is in contact with the rod. In addition, a potentiometer connected to the pinion gear generates position indications that are displayed on the control console.

4.7.2 Scram-Logic Circuitry and Interlocks

The scram-logic circuitry and interlocks ensure that several reactor core and operational conditions must be satisfied for reactor operations to occur or continue. The scram-logic circuitry incorporates a set of open-on-failure logic relay switches in series. Any scram signal or component failure in the scram logic will result in a loss of voltage to the electromagnets on the control rods, causing the rods to scram and shut down the reactor.

The Technical Specifications for the MUTR require the operability of one fuel element temperature scram, two reactor power level scrams, and at least two area radiation level scrams. Also required are scrams in the event of a loss of electrical power or loss of high voltage supply to the ionization chambers. A manual scram is required to allow the operator to shut down the reactor if an unsafe or abnormal condition occurs. In addition to the scrams required by the Technical Specifications, a fast period scram also is available.

The control rods must insert 90% of their full reactivity worth in less than 1 sec following a scram signal. Appropriate surveillance checks, tests, and calibrations are required by the Technical Specifications to verify continued operability and satisfactory performance of the scram functions.

Several safety interlocks are incorporated into the control rod circuitry to prevent inadvertent reactivity insertions. Control rod withdrawal is prevented unless an adequate neutron source signal is available to allow controlled startup of the reactor. Another interlock prevents the withdrawal of more than one control rod at a time. Finally, control rods cannot be withdrawn if plugs from the beam or through tubes have been removed unless a special bypass key is used. Use of this key is controlled by the reactor operator under approved operating procedures.

4.7.3 Assessment

The MUTR is equipped with safety and control systems typical of most nonpower reactors. The control rods, rod drives, scram circuitry, and interlocks have performed reliably in the MUTR since they were installed in the early 1970's, and similar equipment has performed reliably in many other TRIGA reactors for at least 20 years.

The control system provides for an orderly approach to criticality and for safe reactor shutdown during both normal and emergency conditions. There is sufficient redundancy of control rods to ensure safe shutdown even if the most reactive rod fails to insert upon receiving a scram signal. The reactivity worth and speed of travel of the control rods are adequate to provide complete control of the reactor system during operation from a shutdown condition to full power. Interlocks are provided to preclude inadvertent rod movement that might lead to hazardous conditions. Redundant neutron sensors and scram circuits, incorporated to shut the reactor down automatically, mitigate the consequences of single malfunctions. Area radiation monitors cause a reactor scram in case of abnormally high radiation levels, and a scram bar allows the operator to initiate a manual scram whenever such action is needed.

In addition to the electromechanical safety controls for normal and abnormal operation, the large, prompt, negative temperature coefficient of reactivity inherent in the U-ZrH_x fuel moderator (discussed in Section 4.6) provides a unique backup safety feature. The reactor shutdown mechanism of this fuel terminates reactor transients that produce large increases in temperature and will limit the steady-state power level. Because this inherent shutdown mechanism acts to limit the magnitude of a possible transient accident, it would mitigate the consequences of such accidents and can be considered to be a fail-safe safety feature.

On the basis of the above considerations, the staff concludes that the reactivity control systems of the MUTR are designed and function acceptably to ensure safe operation and safe shutdown of the reactor under all operating conditions.

4.8 Operational Procedures

The University of Maryland has implemented administrative controls that require review, audit, and written procedures for all safety-related activities. A Reactor Safety Committee reviews all aspects of the TRIGA reactor operation to ensure that the reactor facility is operated and used within the terms of the facility license and consistent with safety of the public as well as of the operating personnel. The responsibilities of this committee include review and approval of operating procedures, experiments, and proposed changes to the facility or its Technical Specifications. The committee audits reactor operations periodically, as prescribed in the Technical Specifications.

Written procedures (reviewed and approved by the Reactor Safety Committee) have been established and are followed for safety-related activities, including reactor startup, operation, and shutdown; preventive or corrective maintenance; and periodic inspection, testing, and calibration of reactor equipment and instrumentation.

The facility has implemented a preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is not operated at power without all of the safety-related components required by the Technical Specifications fully operable.

The reactor is operated by trained NRC-licensed personnel in accordance with the above-mentioned procedures. All proposed new experiments involving the use of the MUTR are evaluated by the Reactor Director and reviewed by the Reactor Safety Committee for potential impact on the reactivity of or damage to the reactor, as well as possible consequences to the health and safety of employees and the general public.

4.9 Conclusion

The staff review of the MUTR facility has included studying its specific design and installation, its control and safety systems, and its operating procedures. As noted earlier, these features are similar to those typical of the research reactors of the TRIGA type operating in many countries of the world, more than 20 of which are licensed by the NRC. There are currently 11 TRIGA reactors operating at 1 MW or greater with no safety-related problems. On the basis of the review of the MUTR and experience with these other facilities, the staff concludes that there is reasonable assurance that the MUTR, as limited by its Technical Specifications, is capable of continued safe operation.

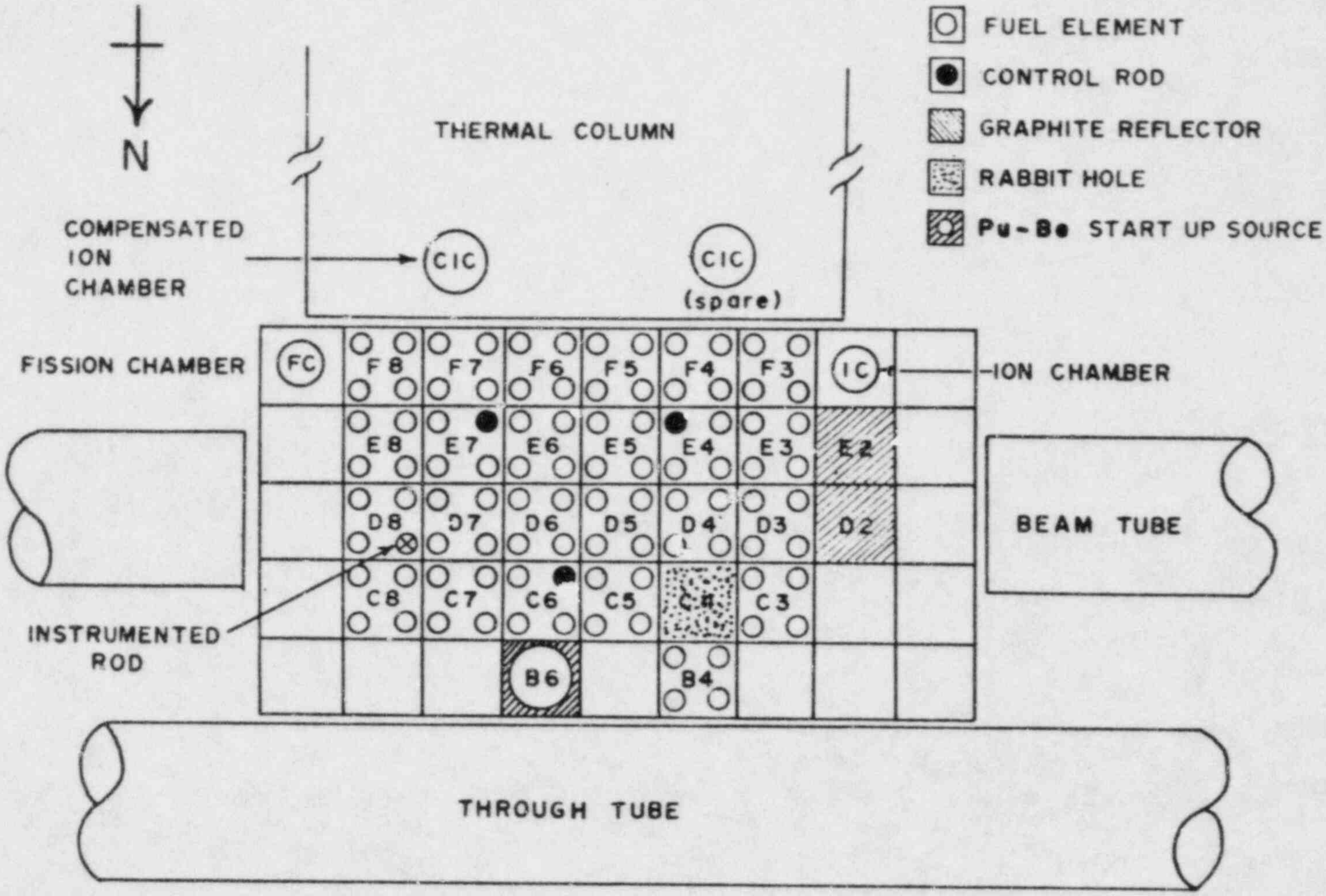

Figure 4.1 MUTR core configuration

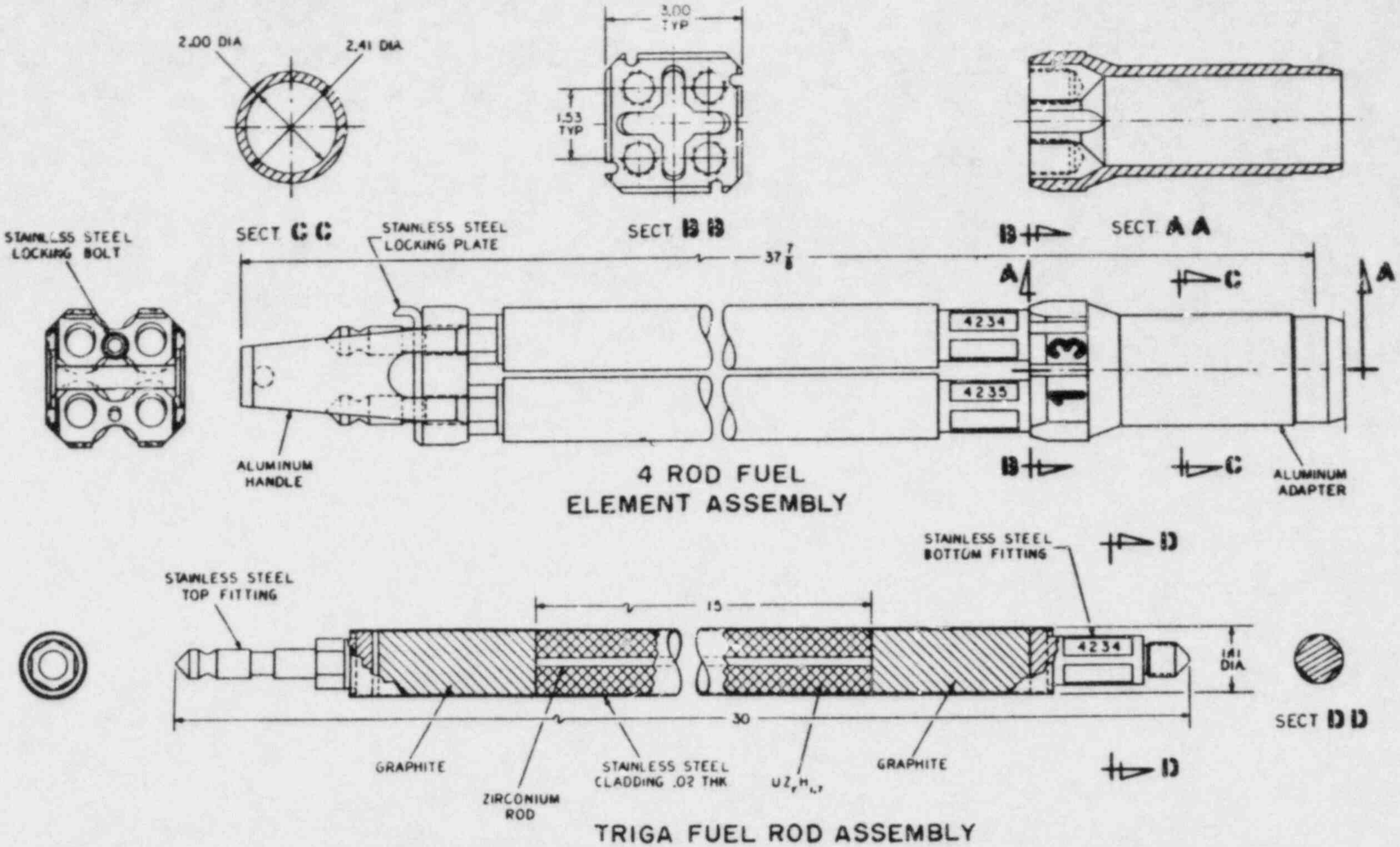

Figure 4.2 TRIGA fuel rod assembly

5 REACTOR COOLANT AND ASSOCIATED SYSTEMS

The reactor coolant at the MUTR is deionized light water in which the reactor core is immersed in a deep cylindrical tank. The heat generated within the fuel during reactor operation is transferred to the pool water by natural convection. The coolant water from the pool may be withdrawn at a rate of 120 gal/min using a pump and circulated through a closed primary loop where it is cooled and purified. Two heat exchangers (shown in Figure 5.1) are connected in series with the option to include or bypass the second heat exchanger as desired.

5.1 Cooling System

The primary cooling system consists of the reactor pool, a particulate filter, the primary coolant pump, two heat exchangers, and a demineralizer. Also included in the system is a syphon break in the inlet line to the pool, which prevents draining of the reactor pool in case of an external pipe rupture. The system contains instruments necessary for the reactor operator to monitor the flow, waterborne radioactivity level, water temperature, pressure and conductivity at various points in the system. The secondary water system for the heat exchangers is an open loop originating from the city water system, passing through the shell side of the heat exchangers, and discharging into the city sanitary sewage system. The primary side of the cooling system is maintained at a lower pressure than that of the secondary loop to prevent potential leakage of primary water to the environment.

5.2 Nitrogen-16 Diffuser

Some of the ^{16}O present in the pool water is activated to ^{16}N by an ^{16}O (n,p) reaction as the water passes through the core of the reactor during operation. Because ^{16}N is a high energy gamma-emitter, it has the potential to increase the gamma field on the bridge above the pool water. To reduce this gamma field, a diffuser system is installed to force water over the surface of the core and thereby increase the transport time for ^{16}N to rise to the surface, allowing additional decay time for this short-lived (T 1/2 = 7.1 sec) radionuclide. This diffuser system is normally used only when the reactor is operated at power levels above 50 kW, in order to help limit occupational radiation exposures.

5.3 Primary Coolant Purification System

The coolant purification system is part of the primary closed loop consisting of the reactor pool, pump, particulate filter, heat exchanger, and a mixed-bed demineralizer loop. The demineralizer loop is tapped between the reactor pool inlet and the outlet lines as shown in Figure 5.1. The demineralizer is a rechargeable type containing 34 l of mixed-bed resin. Ionized species of water-soluble materials are removed by the demineralizer during the passage of water through the unit. Only ~10% of the water in the coolant loop is diverted through the demineralizer loop. Conductivity probes located at the inlet and outlet of the demineralizer units determine the effectiveness of the water purification system.

5.4 Primary Coolant Makeup System

The pool water makeup sytem consists of a flow regulator and a microfilter device for the city water, which then is passed through a separate demineralizer before it enters the primary coolant system. This makeup system is normally valved off from the primary coolant heat exchanger and purification system.

5.5 Conclusion

The staff concludes that the cooling system of MUTR is of an adequate size, design, condition, and maintenance level to ensure adequate cooling of the reactor under operating conditions authorized in the MUTR operating license.

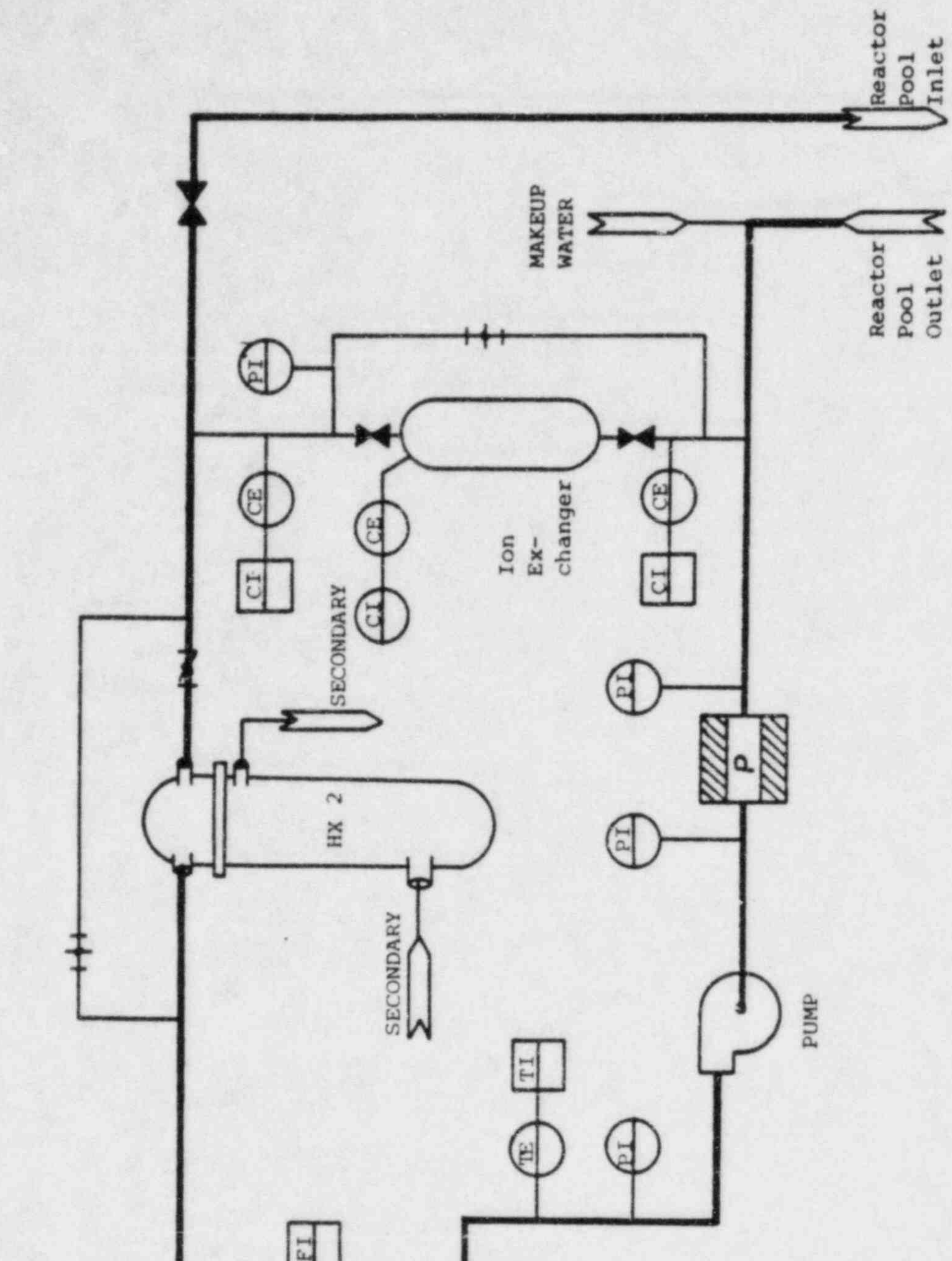

Figure 5.1 Primary coolant system

6 ENGINEERED SAFETY FEATURES

The only engineered safety system associated with the MUTR facility that is not directly associated with reactor control is the reactor room ventilation system. Under normal conditions, the significant airborne radioactive materials formed as a result of reactor operations are ^{41}Ar and ^{16}N. No fission products are permitted to escape beyond the fuel cladding. This is discussed in more detail in Section 11.

6.1 Ventilation System

The reactor building has a ventilation system that is independent of the rest of the engineering building. There are two roof-mounted ventilating exhaust fans and two motor-operated intake louvers. The fans and louvers can be electrically controlled from the control room and three other locations on the ground floor of the reactor building. The ventilation system has a capacity for exhausting 170 m^3/min of air to the roof vents, which are 7.3 m above ground level. Under normal conditions, two motor-operated intake louvers mounted on the ground floor (west wall) of the reactor building provide the air supply into the reactor room. There are also air conditioners on the exterior west wall of the building that act only as air inlets. Air from the west balcony laboratories exhausts into the main reactor bay area through two motorized louvers and one air conditioner mounted in the sample preparation laboratory.

Any reactor scram automatically initiates a fail-safe spring-loaded mechanism that secures the ventilation system. It also is possible to secure the ventilation system by emergency manual shutdown from four locations in the building without causing a reactor scram.

The ventilation system and the automatic shutdown controls are required by Technical Specifications to be tested before each day's reactor operation and before the startup of an operation extending for more than a day.

6.2 Conclusion

The equipment and procedures for the reactor building ventilation system are designed to control the release of airborne radioactive effluents during normal operation and to limit the release of airborne radioactivity in the event of abnormal or accident conditions. On the bases of the above information, and discussions in Sections 11, 12, and 14 of this SER, the staff concludes that the ventilation system, as it currently exists, is acceptable and provides assurance that the public will not be exposed to significant quantities of airborne radioactive effluents related to MUTR operations.

7 CONTROLS AND INSTRUMENTATION

The controls and instrumentation for the MUTR are similar to those used in many other research reactors in the United States. The nuclear fission process is regulated by three neutron-absorbing control rods. The control and instrument systems are interlocked to provide for automatic and manual scram in case of abnormal reactor operation and provide the means for operating the various components in a manner consistent with design objectives. A block diagram of the reactor control circuits is shown in Figure 7.1.

7.1 Control Console

The reactor control console contains the controls and indicating and recording instrumentation required for operation of the reactor. The control panel contains (1) rod control switches for raising and lowering the control rods; (2) rod-position indicators to show the positions of the two shim-safety rods and the regulating rod; (3) annunciator lights to indicate the up or down position of each rod and contact between the rod armatures and lifting magnets; (4) linear and logarithmic neutron (log-n) power recorders; (5) reactor period, power level, pool temperature, and log-count rate meters; (6) radiation monitor alarm lights; and (7) additional pilot lights to indicate electrical power on, cooling system pump on, and neutron startup source strength. Other annunciator lights on the console indicate the source of any scram signal.

7.2 Control System

The control system is composed of both nuclear and process control equipment and is designed for redundant operation so that failure or malfunction of individual components essential to the safe operation of the reactor do not preclude safe operation.

7.2.1 Nuclear Control System

The nuclear control system consists of two shim-safety rods and one regulating rod and their associated drive mechanisms. Additional discussion is presented in Section 4.

(1) The drive mechanism consists of a motor and a reduction gear driving a rack-and-pinion gear system.

(2) The control rod extensions are magnetically coupled to the drive shafts and can be scrammed by removal of electrical power to the magnets.

(3) The control rods only can be withdrawn one at a time because of an interlock circuit.

(4) The speed of rod withdrawal is constant and is limited to about 48 cm/min to ensure a safe rate of reactivity insertion. The regulating rod may be used for automatic servo-control of steady-state power at a preset level or for bringing the reactor to power on a preset period.

Automatic scram is initiated by (1) a reactor power level exceeding the safety setting as indicated and set by either of the two power level channels, (2) a reactor period less than a preset value, (3) any high voltage power supply failure, (4) an electrical power failure, or (5) fuel temperature exceeding the safety setting. Manual scram can be initiated by the operator by means of the console scram button or the magnet current key switch. Following a scram and after each rod reaches the full-in position, the drive mechanism automatically follows the rod down to re-establish contact between magnet and armature.

7.3 Instrumentation System

The instrumentation system is composed of both nuclear and process instrumentation circuits. The annunciations and/or indications provided by the electronics system on the control console are noted in Section 7.1 and shown in Figure 7.1, which lists the minimum safety channels for operation prescribed by the Technical Specifications.

7.3.1 Nuclear Instrumentation

This instrumentation provides the operator with the necessary information for adequate manipulation of the nuclear controls. See Figure 7.2 for the range relationship of the in-core detectors.

(1) Wide Range Log-Power Channel - The wide range, log-power channel responds to 10 decades of neutron flux even in a gamma ray background of about 10^5 R/hour. It operates from a single fission pulse chamber and the full 10-decade range is read out on a single meter and on one channel of a dual pen recorder. Using the signal of individual pulses from the fission chamber, the channel (Campbell technique) combines a pulse log count rate system for the lower 5 decades with a log ionization current system for the upper 5 decades to produce a single output signal for the total range of 10 decades. The channel has six calibrate positions for checking its overall alignment and linearity. A minimum neutron source count rate interlock prevents control rod withdrawal unless the measured neutron counting rate exceeds a predetermined value. Also, a reactor period meter obtains a signal from the log power amplifier, which provides an indication of the time rate of change of neutron density calibrated from -30 to +3 sec. The Technical Specifications do not require a scram provision on the reactor period system (see Sections 4.6 and 14.1 for discussion of transients).

(2) Safety (percent power) Channel 1 - The same fission chamber identified above is used for Safety Channel 1. The ionization current signal from the detector is fed to a linear amplifier that converts the chamber current to a proportional voltage output signal that feeds a bistable trip and a percent power meter. Test and calibration signals are built into each channel and are controllable from the front panel. The circuit boards and switches are interlocked to indicate removal of a board or incorrect position of a switch.

(3) Safety (percent power) Channel 2 - The neutron detector for this channel is an uncompensated ion chamber. Other than the detector, this channel is identical to Safety Channel 1. The different type of nuclear detector was chosen to ensure redundancy of core neutron flux information.

(4) Multirange Linear Power Channel - During the steady-state mode of operation, the linear power channel provides power level indications from just above neutron source level to full power. This channel consists of a neutron-sensitive compensated ion chamber, a multirange linear picoammeter, a meter and one channel of the strip-chart recorder, and a 15-position range switch. The range switch is used to select a particular power scale for the recorder display. The linear amplifier's output voltage signal, which is proportional to the input ionization chamber current (a linear function of reactor power), is fed to the reactor power regulation system used for the automatic mode of operation.

(5) Fuel Temperature Monitor - Fuel heating is proportional to reactor power and fuel temperature is related to fuel heating. The fuel temperature monitor is a meter-operated optical relay that derives its signal directly from a thermocouple imbedded in an instrumented fuel element. A reactor trip point is set on the front face of a meter and de-energizes the fuel temperature scram relay. Zero and calibration signals are built into the channel and are controllable from the front panel.

7.3.2 Process Instrumentation

This instrumentation principally monitors parameters associated with both the quality and quantity of the pool water and with radiation exposure rates at several locations around the reactor facility. The readouts of these parameters are all provided at the control console. The coolant water parameters that are measured include pool water depth, primary water conductivity, and primary water flow rate. The radiation levels are measured on the bridge above the reactor, and in the vicinity of the pool water purification system.

7.4 Conclusion

The control and instrumentation system at the MUTR is adequately designed and maintained, as evidenced by a small number of malfunctions during operation. All electrical power and instrumentation wiring is protected from physical damage by conduit and/or cable trays. The specifications of the individual components are in excess of minimal requirements for the overall system. Redundancy in the important area of reactor power measurements is ensured by overlapping ranges of the redundant log-n and linear power channels. The control system is designed and operates so that the reactor is automatically shut down if electrical power is lost.

The staff concludes that the controls and instrumentation at the MUTR satisfy all existing regulations and are acceptable to ensure safe operation and shutdown of the facility.

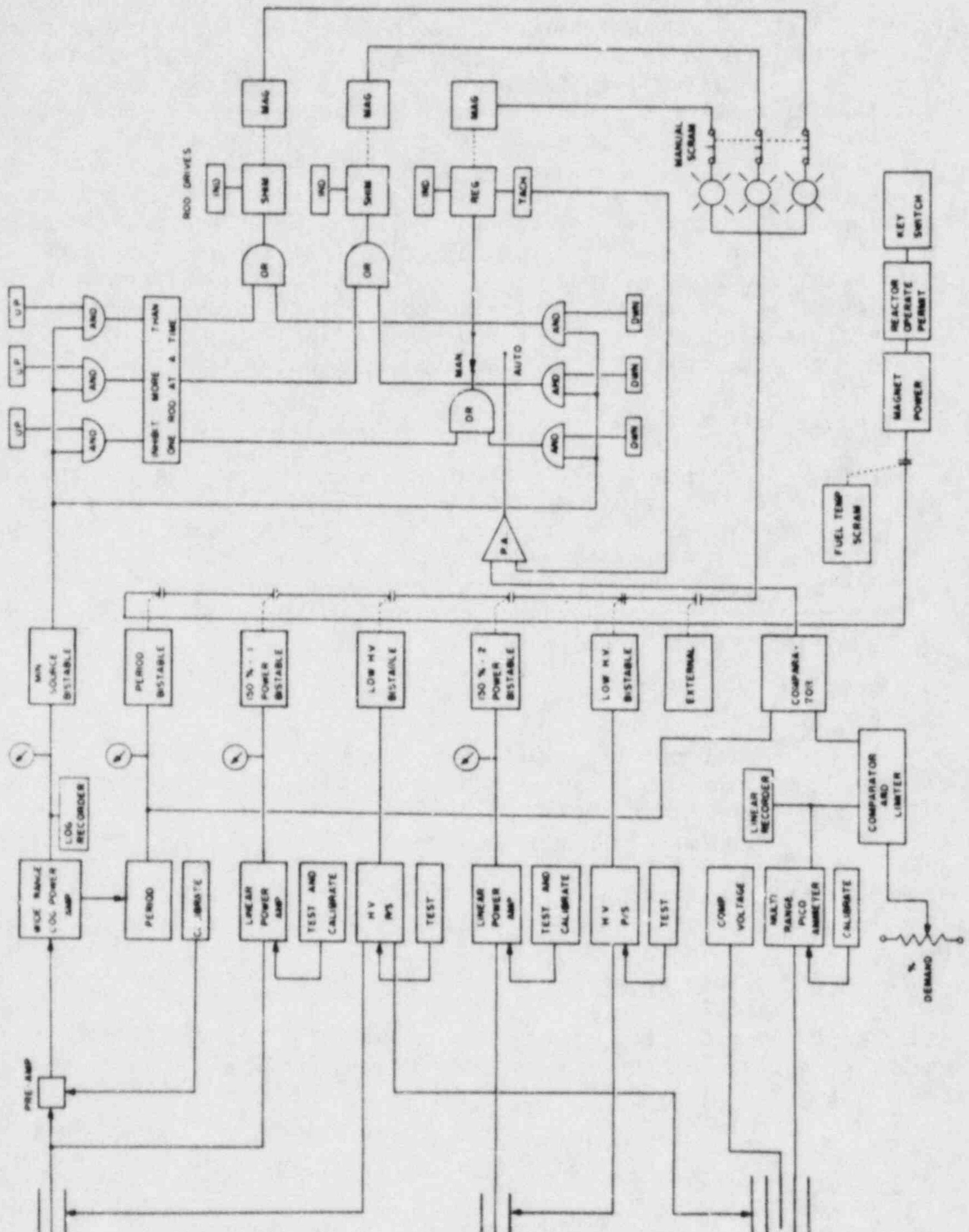

Figure 7.1 Block diagram of instrumentation

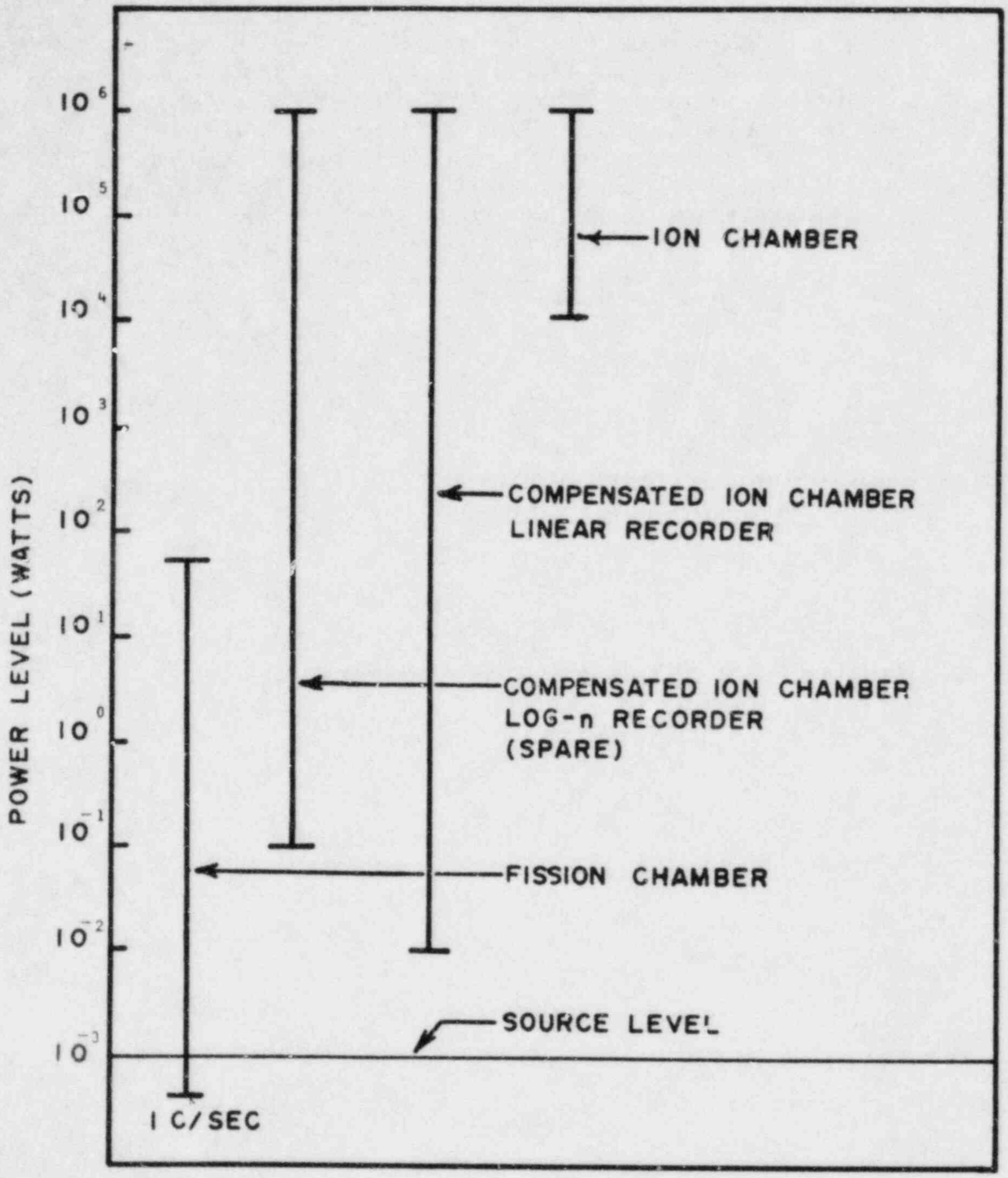

Figure 7.2 Operating range of in-core nuclear detectors

Table 7.1 Minimum reactor safety channels

Safety channel	Functions
Minimum source count rate	Prevent withdrawal of control rods
Safety (percent power) #1	Scram on high power
Safety (percent power) #2	Scram on high power
Fuel temperature	Scram on high temperature

8 ELECTRICAL POWER SYSTEM

Electrical power for building lighting and equipment power is 120/208-V three-phase, four-wire, 60 Hz. The total estimated power requirement for the facility is 300 kVA. The main power control panel is located in the electrical utility room, with subpanels located as required in other areas. Because the MUTR will scram in case of a power interruption and the decay heat generated in the core after scram is not significant (see Section 14), no emergency power is supplied.

The electrical power system at the MUTR is a standard and well-accepted electrical supply system designed and constructed to specifications similar to those at other research reactor facilities. The system consists of standard commercial components. These factors, when considered with the fact that the MUTR will scram in the event of power failure, lead the staff to conclude that the electrical power system is acceptable for continued operation of the MUTR.

9 AUXILIARY SYSTEMS

9.1 Ventilation System

The ventilation system is an engineered safety feature and, therefore, discussed in Section 6 of this report.

9.2 Fire Protection System

Fire protection for the reactor facility is an integral part of the campus-wide fire protection system. The University's Environmental Safety Group is primarily responsible for the campus fire protection, including the reactor facility. This group maintains the fire protection equipment within the reactor facility through a semiannual inspection. There are three fire hydrants within 100 m of the reactor building. The fire alarms at the reactor facility provide remote monitoring at the campus police headquarters. The general fire protection service to the campus is provided by the Prince Georges County Fire Department, which has a fire station at the College Park campus. The University's Environmental Safety Group and the personnel of the Prince Georges County Fire Department are given a tour of the reactor facility at least annually.

9.3 Heating and Air Conditioning

The heating system for the reactor facility consists of baseboard water heaters. The hot water to the system is provided by the campus-wide central heating facility.

The MUTR facility has an independent air conditioning system that is an air chiller using cold water for cooling. The air inside the reactor bay is cooled and recirculated within the reactor bay.

9.4 Fuel Handling and Storage

Fuel that is not in current use in the reactor core may be stored in a fuel storage rack attached around the inside of the reactor tank opposite the thermal column. There are two rows of aluminum cans in this fuel rack, the first row has 10 storage cans and the second row, which is in front of the first, has 3 cans. These fuel racks can hold up to 13 fuel clusters in a noncritical array under 13 ft of light water. Criticality estimates of K_{eff} <0.4 have been made of these storage racks, fully loaded and fully submerged, using the TWOTRAN (GA-8747) computing code.

The fuel element clusters are handled one at a time by a fuel handling tool. This tool consists of an 18-ft-long aluminum rod with a lock-closed claw to engage the fuel cluster handling pin at one end and an operating handle at the other. The tool is lowered to the fuel cluster from the reactor bridge. The claw is closed and locked around the handling pin by a mechanism on the handle, and the cluster is lifted from the core and carried under water to the storage rack. There it is lowered into the holder and the tool is disengaged. The same procedure is used for moving fuel from the storage rack to the core when needed. The fuel cluster handling tool also can be used to move reflector elements. Routine movement of irradiated fuel elements is conducted while they are totally submerged in water to limit exposure to operations personnel.

9.5 Fuel Rod Inspection Tool

The fuel rod inspection tool designed by General Atomic Company can be used for the measurement of a fuel rod for longitudinal growth and bowing. The tool is designed to be mounted and used in the reactor tank, permitting the inspection of an irradiated fuel rod at an elevation that provides approximately 2.7 m of shielding water over the element. The straightness of the fuel rod can be checked by inserting the rod into the cylindrical "go/no-go" gauge attached to the bottom of the tool.

If any tested fuel rod changes exceed manufacturer's limits, it is considered to be damaged and removed from service in accordance with the Technical Specifications.

9.6 Fuel Rod Transfer Cask

A fuel rod transfer cask is designed to permit the safe transfer of irradiated fuel rods and other radioactive materials from the reactor tank to the fuel storage pits (Section 9.7). The cask is a steel casing containing a lead cylinder and weighs approximately 2,591 kg. The cask has 0.51 m outside diameter and a 1.14 m length. A 5-cm-diameter cavity extending almost the full length of the cask can hold a single fuel rod. Eyebolts are provided for the attachment of lifting cables. The bottom and top of the cask contain removable lead shield plugs that lock in place. Written procedures cover the use of the crane over the reactor.

9.7 Fuel Storage Pits

In the event that fuel must be removed from the reactor pool, storage pits are provided in the north floor of the reactor building. This facility is approximately 1.2 m wide and 1.8 m long, containing 24 cylindrical holes in the concrete floor. Each of the holes is about 15 cm in diameter, lined with steel, about 1.5 m deep, and fitted with a removable concrete-filled steel plug about 0.6 m long for shielding. The pit is recessed and is covered with a steel door. The licensee has calculated that this pit, flooded and fully loaded with TRIGA fuel has a K_{eff} of <0.8. This storage facility currently contains no fuel, but it has been used in the past for the storage of irradiated MUTR fuel.

9.8 Conclusion

On the basis of the above considerations, the staff concludes that the auxiliary systems at MUTR facility are designed and maintained adequately, and the systems are acceptable for their intended purposes.

10 EXPERIMENTAL PROGRAMS

The MUTR is used for various experimental programs in addition to its principal use in the educational program at the university. The reactor is a source of gamma and neutron radiations for research and radioisotope production. In addition to in-pool irradiation capabilities, experimental facilities include a pneumatic transfer system, a thermal column, two radial beam ports, and a through-tube beam port.

10.1 Experimental Facilities

10.1.1 Pool Irradiations

The open pool of the reactor permits the irradiation of experiments submerged in the vicinity of the core and thermal column. The decision to perform experiments in the reactor pool instead of using the pneumatic transfer system or a beam tube is dictated by the nature and size of the specimen and the desired type and intensity of radiation fields. The reactivity effect of experiments or samples in the core region of the pool is limited by the Technical Specifications.

10.1.2 Pneumatic Transfer System

A pneumatic transfer system allows small sealed samples to be rapidly transferred between the reactor and the sample preparation laboratory located on the west balcony. The irradiation terminus is in position C-4 of the reactor core, and the receiver terminus is in a shielded glove box location on the west balcony. The controls for the pneumatic transfer system are located in the reactor control room, under the reactor operator's control. The driving force for this system is provided by pressurized carbon dioxide gas. The use of CO_2, instead of air, avoids the production of ^{41}Ar and coincidentally decreases the formation of ^{14}C by several orders of magnitude. The exhaust CO_2 from the system is released directly below the exhaust duct of the reactor room.

10.1.3 Beam Ports and Through Tube

There are four beam port penetrations through the reactor tank wall, two of which are used for a through tube. These penetrations normally contain shielding material in the sections within the concrete biological shield. The shield plugs may be removed to provide external beams of radiation through the beam ports and/or to insert samples for higher neutron flux irradiation. The through tube extends from one side of the pool to the other and can be used for moving samples from one side to the other while exposing them to a radiation field or for insertion of stationary experiments.

10.1.4 Thermal Column

The thermal column is a graphite-filled housing extending to the face of the core through a penetration on the side of the reactor tank wall. The graphite assembly consists of 10-cm-square graphite stringers arranged to form a stepped column 1.52 m long. There are four stringers that may be removed in sections of different lengths to form experimental holes of various sizes.

10.2 Experimental Review

Before any new experiment can be conducted using the reactor or the associated experimental facilities, it is reviewed by the Reactor Safety Committee. The membership of the Safety Review Committee is planned to provide a spectrum of expertise to review the experiments and their potential hazards. This review and approval process for experiments allows personnel trained in reactor operations to consider and suggest alternative operational conditions--such as different core positions, power levels, and irradiation times--that will minimize personnel exposure and/or potential release of radioactive materials to the environment.

10.3 Conclusion

On the basis of the above considerations, the staff has determined that the design of the experimental facilities, combined with the detailed review and administrative procedures applied to all research and educational activities is acceptable to ensure that the experiments are (1) unlikely to fail, (2) unlikely to release significant radioactivity to the environment, and (3) unlikely to cause damage to the reactor system or its fuel. Therefore, the staff concludes that reasonable provisions have been made so that the experimental programs and facilities do not pose a significant risk to the facility staff or to the public.

11 RADIOACTIVE WASTE MANAGEMENT

The major radioactive waste generated by routine reactor operations is airborne activated material, principally ^{41}Ar. A limited volume of radioactive solid waste, principally spent ion exchange resins, is generated by reactor operations, and some additional solid waste is produced by the research programs involving the use of reactor facilities. The facility does not regenerate the coolant purification ion exchanger resin beds; thus, very little radioactive liquid waste is generated at this facility.

11.1 ALARA Commitment

The MUTR is operated with the philosophy of minimizing the release of radioactive materials to the environment, in accordance with a policy of maintaining radiation exposures "as low as is reasonably achievable" (ALARA) (see Section 12.1). The university administration, through the Radiation Safety Officer, instructs all operating and research personnel to develop procedures to limit the generation and subsequent release of radioactive materials.

11.2 Waste Generation and Handling Procedures

11.2.1 Solid Waste

The disposal of high-level radioactive waste in the form of spent reactor fuel is not anticipated during the term of this license renewal. Therefore, the radioactive solid waste generated as a result of reactor operations consists primarily of ion exchange resins and filters, potentially contaminated paper and gloves, and occasional small activated components. Some of the reactor-based research also results in the generation of solid low-level radioactive wastes in the form of contaminated paper, gloves, and glassware. This solid waste has contained less than 1 mCi of radionuclides per year, averaged over the past 5 years.

The solid waste is collected in specially marked barrels. They are held temporarily before being packaged and shipped to an approved disposal site in accordance with applicable regulations. The reactor-related low-level radioactive waste (~2 barrels per year) constitutes a small fraction of that accumulated and disposed of annually from all the campus research programs.

11.2.2 Liquid Waste

Normal reactor operations produce no radioactive liquid waste other than the coolant, containing insignificant amounts of tritium and waterborne activation products. The coolant cleanup system is adequate to purify this water on a continuous basis. There is a liquid waste sump and holdup tank to collect all the liquids from the grill work around the base of the reactor shield and two sinks on the west balcony laboratories. The pool overflow drains directly into the holdup tank. When the sump is full, it is sampled and analyzed to determine the quantity of radioactive materials in the liquid waste. If the concentrations of the radioactive material in the tank are less than the levels specified in 10 CFR 20, the contents are discharged to the campus sanitary sewer system. If the concentrations were initially above 10 CFR 20 limits, the contents of the tank would either be drained and stored for radioactive decay or diluted below the 10 CFR 20 levels and discharged to the sewer system. The total volume of ontaminated liquid waste discharged is typically less than 1,000 gal per month.

11.2.3 Airborne Waste

The potential airborne waste during normal operation is composed of ^{41}Ar and neutron-activated dust particulates. These are produced by the irradiation of air and airborne particulates in the thermal column, beam ports, and through tube. The air is swept constantly from the experimental facilities and discharged into the environment through an exhaust stack at the roof of the building. Another gaseous activation product that can be airborne is ^{16}N, produced within the coolant passing through the core of the reactor. To limit the ^{16}N gas that becomes airborne at higher power levels, a jet of water (diffuser) may be sprayed over the surface of the core. This increases the transport time of the short-lived (7.1 sec) ^{16}N from the core to the surface of the pool and allows additional radioactive decay. As a result of this practice, even the highest exposure rate caused by airborne ^{16}N (on the reactor top) does not contribute a significant dose to personnel. No fission products escape from the undamaged fuel cladding during normal operations, and the Technical Specifications prohibit operation of the reactor with damaged fuel.

The licensee has estimated the formation of ^{41}Ar in the pool water during operation of the reactor at 250 kW and finds that less than 0.1 Ci would be evolved from the water into the reactor room air during a typical operational year of ≤ 30 MW·hr. Assuming that this air is exhausted to the external environment, it is further estimated that the maximum potential dose to an individual in the unrestricted areas would be much less than 1 mrem per year, which is well within the limits of 10 CFR 20 (see Section 12.6.1). The staff also has estimated the radiological consequences of the airborne radioactive waste (NUREG-0851) and agrees with the licensee. Both the licensee and the staff also have considered the release of ^{41}Ar for an operating schedule of 400 MW hours per year, and the potential annual dose in the unrestricted environment is still less than 1 mrem per year.

11.3 Conclusion

The staff concludes that the waste management activities at the MUTR facility have been conducted and may be expected to be conducted in the future in a manner consistent with 10 CFR 20 and with the ALARA principles. Among other guidance, the staff review has followed the methods of American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.11, 1977, "Radiological Control at Research Reactor Facilities."

Because ^{41}Ar is the only significant radionuclide released by the reactor to the environment during normal operations, the staff has reviewed the history, current practices, and future expectations of operations. The staff concludes that the maximum potential doses in unrestricted areas as a result of actual releases of ^{41}Ar have never exceeded or even approached the limits specified in 10 CFR 20 when averaged over a year. Furthermore, the staff's computations of the dose beyond the limits of the reactor facilities give reasonable assurance that the maximum potential doses to the public as a result of ^{41}Ar release would not be significant even if there were major changes in the operating schedule of the MUTR.

12 RADIATION PROTECTION PROGRAM

The University of Maryland has a structured radiation safety program with a trained health physics staff equipped with radiation detection instrumentation to determine, control, and document occupational radiation exposures at its reactor facility.

12.1 ALARA Commitment

The Environmental Safety Department for the campus has established the policy that operations are to be conducted in a manner to keep all radiation exposures ALARA. All proposed experiments and procedures at the reactor are reviewed for ways to minimize the potential exposures of personnel. All unanticipated or unusual reactor-related exposures will be investigated by both the Radiation Safety Office and the reactor operations staff to develop methods to prevent recurrences.

12.2 Health Physics Program

12.2.1 Health Physics Staffing

The normal radiation safety staff at the University of Maryland consists of five professional health physicists and several student technicians. This staff provides radiation safety support to the entire university complex, including three accelerators and many radioisotope laboratories. The routine health physics-type activities at the reactor are performed by the reactor operations staff. The formal health physics staff is available for consultation and the University Radiation Safety Officer is a member of the Reactor Safety Committee.

The staff concludes that the staffing of the program to provide radiation safety support for the research efforts within this reactor facility is acceptable.

12.2.2 Procedures

Detailed written procedures have been prepared that address the radiation safety support that is to be provided to the routine operations of the university's research reactor facility. These procedures identify the interactions between the operational and user personnel. They also specify numerous administrative limits and action points as well as appropriate responses and corrective action if these limits or action points are reached or exceeded. Copies of these procedures are readily available to the operational and research staffs and to the administrative and radiation safety personnel for reviewing their respective responsibilities.

12.2.3 Instrumentation

The University of Maryland has acquired a variety of detecting and measuring instruments for monitoring potentially hazardous ionizing radiation. The instrument calibration procedures and techniques provide assurance that any credible type of radiation and any significant intensities will be detected promptly and measured correctly.

12.2.4 Training

All reactor-related personnel are given an indoctrination in radiation safety before they assume their work responsibilities. Additional radiation safety instructions are provided to those who will be working directly with radiation or radioactive materials. The training program is designed to identify the particular hazards of each specific type of work to be undertaken and methods to mitigate their consequences. Retraining in radiation safety is provided as well. As an example, all reactor operators are given an examination on health physics practices and procedures at least every 2 years. The level of subsequent training given is determined by the examination results.

12.3 Radiation Sources

12.3.1 Reactor

Sources of radiation directly related to reactor operations include radiation from the reactor core, ion exchange columns, filters in the water cleanup systems, and radioactive gases (primarily ^{41}Ar).

The fission products are contained in the stainless-steel cladding of the fuel. Radiation exposures from the reactor core are reduced to acceptable levels in the reactor room by water and concrete shielding. The ion exchange resins and filters are changed routinely before high levels of radioactive materials have accumulated, thereby limiting personnel exposure.

Personnel exposure to the radiation from chemically inert ^{41}Ar is limited by dilution and prompt removal of this gas from the reactor area and its discharge to the atmosphere, where it diffuses and is dispersed further before reaching areas that are routinely occupied by the public.

12.3.2 Extraneous Sources

Sources of radiation that may be considered as incidental to the normal reactor operation, but associated with reactor use, include radioactive isotopes produce for research, activated components of experiments, and activated samples or specimens.

Personnel exposure to radiation from intentionally produced radioactive material as well as from the required manipulation of activated experimental components is controlled by rigidly developed and reviewed operating procedures that use the normal protective measures of time, distance, and shielding.

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

The MUTR facility has three fixed-position radiation monitors: one on the bridge above the reactor, a second near the water purification system, and the third near the reactor room air exhaust fan. All monitors have adjustable alarm set points and read out in the control room. All are required by the Technical Specifications to be operable during reactor operation. The purpose of all of these monitors is to indicate to the operator by signal and alarm that an abnormal level of radiation exists. Procedures specify alarm set points, operability checks, and operator responses.

12.4.2 Experimental Support

The health physics staff participates in experiment planning by reviewing all proposed procedures for methods of minimizing personnel exposures and limiting the generation of radioactive waste. Approved procedures specify the type and degree of radiation safety support required by each activity.

12.5 Occupational Radiation Exposures

12.5.1 Personnel Monitoring Program

The University of Maryland personnel monitoring program is described in its Radiation Safety Instructions. Personnel exposures are measured by the use of film badges assigned to individuals who might be exposed to radiation. In addition, self-reading ion chambers are used, and instrument dose rate and time measurements are used to administratively keep occupational exposures below the applicable limits in 10 CFR 20.

Visitors are provided with self-reading ion chambers for monitoring gamma radiation exposures to which they might be subject while in the facility.

12.5.2 Personnel Exposures

The MUTR personnel annual exposure history for the last 5 years is given in Table 12.1. The generally low annual exposures to individuals indicate acceptable control by the facility management and individual responsibility.

12.6 Effluent Monitoring

12.6.1 Airborne Effluents

As discussed in Section 11, airborne effluents to the environment from the reactor facility consist principally of low concentrations of ^{41}Ar. The small amount of ^{41}Ar released into the reactor room is diluted by diffusion into the $1.7 \times 10^3 m^3$ volume of air in the reactor room. The licensee's conservatively calculated maximum concentration in the reactor room after many hours (several ^{41}Ar half-lives) of operation at full power will be less than 10^{-6} mCi/ml. In actual normal operation, this concentration would not be achieved because the calculations assumed that the exhaust fan is not operating, and, therefore, that the ^{41}Ar accumulates up to equilibrium concentration in the confined reactor room air. In practice, reactor room air is discharged at about 170 m^3/min at a point approximately 7 m above ground level, resulting in additional dilution before it reaches unrestricted areas. On the basis of these estimates, the licensee does not measure the normal operational release of airborne effluents, but maintains room monitors that alarm only if abnormal radiation levels occur. When the prior (and expected) operational history of the MUTR is considered, the staff concludes that concentrations in unrestricted areas, when averaged over a year, have never approached the limits of 10 CFR 20, Appendix B, guidelines and that the monitor for airborne effluents is acceptable.

12.6.2 Liquid Effluent

The reactor generates very limited radioactive liquid waste during routine operations (see Section 11). However, leaks in the primary coolant system do have the potential or releases (the possibility of water leaks between primary and secondary systems in the heat exchanger is addressed in Section 5.1) and experimental activities associated with reactor usage also may generate radioactive liquids.

All drains in the reactor bay lead to the sump in the water treatment room. The sump contents can be pumped to a 416-l waste storage tank for decay or cleanup should significant radioactivity be measured at the time of sampling.

Before any releases of potentially contaminated water to the sanitary sewer system, representative samples are collected and analyzed by standard techniques. If the concentrations of radioactive materials in the waste are less than the guideline values of 10 CFR 20.303, the liquids are discharged directly to the sewer.

12.7 Environmental Monitoring

The environmental monitoring program on and around the campus has been discontinued because of the absence of any statistically significant positive measurements over a period of many years.

12.8 Potential Dose Assessments

Natural background radiation levels in the Washington, D.C., area result in an exposure of 80 to 100 mrem per year to each individual residing there. At least an additional 10% (approximately 8 mrem per year) will be received by those living in a brick or masonry structure. Any medical diagnosis or X-ray examination may add to the exposure from natural background radiations, increasing the total accumulative annual individual exposures.

Conservative calculations by the staff based on the amount of ^{41}Ar released during normal operations from the reactor facility stack predict a maximum annual exposure of less than 1 mrem in the unrestricted areas.

12.9 Conclusion

The staff considers that radiation protection receives appropriate support from the university administration. The staff concludes that (1) the program is adequately staffed and equipped, (2) the reactor health physics staff has adequate authority and lines of communication, (3) the procedures are integrated correctly into the research plans, and (4) surveys verify that operations and procedures achieve ALARA principles.

The staff concludes that the reactor room monitoring programs conducted by university personnel are acceptable to identify significant releases of radioactivity to predict maximum exposures to individuals in the unrestricted area. These predicted maximum levels (see Section 14) are well within applicable regulations and guidelines of 10 CFR 20.

Additionally, the staff concludes that the University of Maryland radiation protection program is acceptable because the staff has found no instances of reactor-related exposures of personnel above applicable regulations and no unidentified significant releases of radioactivity to the environment. Furthermore, the staff considers that there is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public during future reactor operations.

Table 12.1 Number of individuals in exposure interval

	Number of individuals in each range				
Whole-body exposure range (rem)	1978	1979	1980	1981	1982
Measurable exposure $\leq$ 0.05	9	13	13	22	32
Measurable exposure $\leq$ 0.1	11	14	13	22	35
0.1 to 0.25	0	0	0	0	0
0.25 to 0.5	0	0	0	0	0
0.5 to 0.75	0	0	0	0	0
0.75 to 1	0	0	0	0	0
Number of individuals monitored	11	14	13	22	35

13 CONDUCT OF OPERATIONS

13.1 Overall Organization

Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in Figure 13.1. The Operations Supervisor is delegated responsibility for overall facility operation. He delegates the succession to this reponsibility during his absence.

13.2 Training

Most of the training of reactor operators is done by inhouse personnel. The licensee's Operator Requalification Program has been reviewed, and the staff concludes that it meets the applicable regulations (10 CFR 50.54(i-1)) and Appendix A of 10 CFR 55, and is consistent with the guidance of ANS 15.4.

13.3 Emergency Planning

10 CFR 50.54(q) and (r) require that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E of 10 CFR 50. At the staff's request, as part of the application for license renewal, the licensee submitted a plan following guidance contained in RG 2.6 (1978 For Comment Issue) and in ANS 15.16 (1978 Draft). In 1980, new regulations were promulgated, and licensees were advised that revised guidance would be forthcoming. Thus, revised ANS 15.16 (November 29, 1981, Draft) and RG 2.6 (March 1982 For Comment) were issued. On May 6, 1982, an amendment to 10 CFR 50.54 was published in the Federal Register (47 FR 19512, May 6, 1982) recommending these guides and establishing new submittal dates for Emergency Plans from all research reactor licensees. The deadline for submittal from a licensee in the MUTR class (<2 MW) was November 3, 1982. By letter dated October 27, 1982, the licensee transmitted a revised Emergency Plan in fulfillment of the requirements of the applicable regulations.

13.4 Operational Review and Audits

The Reactor Safety Committee (RSC) provides independent review and audit of facility activities. The Technical Specifications outline the qualifications and provide that alternate members may be appointed by the Chairman. The committee must review and approve plans for modifications to the reactor, new experiments, and proposed changes to the license or to procedures. The committee also is responsible for conducting audits of reactor facility operations and management and for reporting the results thereof to the Chairperson of the Department of Chemical and Nuclear Engineering.

13.5 Physical Security Plan

The MUTR has established and maintains a program to protect the reactor and its fuel and to ensure its security. The NRC staff has reviewed the Physical Security Plan and concludes that the plan, as amended, meets the requirements of 10 CFR 73.67 for special nuclear material of low strategic significance.

MUTR's inventory of special nuclear material for reactor operation falls within that category. Both the Physical Security Plan and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1). Amendment No. 6 to the facility Operating License R-70, dated February 6, 1981, incorporated the Physical Security Plan as a condition of the license.

13.6 Conclusion

On the basis of the above, the staff concludes that the licensee has sufficient experience, management structure, and procedures to provide reasonable assurance that the reactor will be managed in a way that will cause no significant risk to the health and safety of the public.

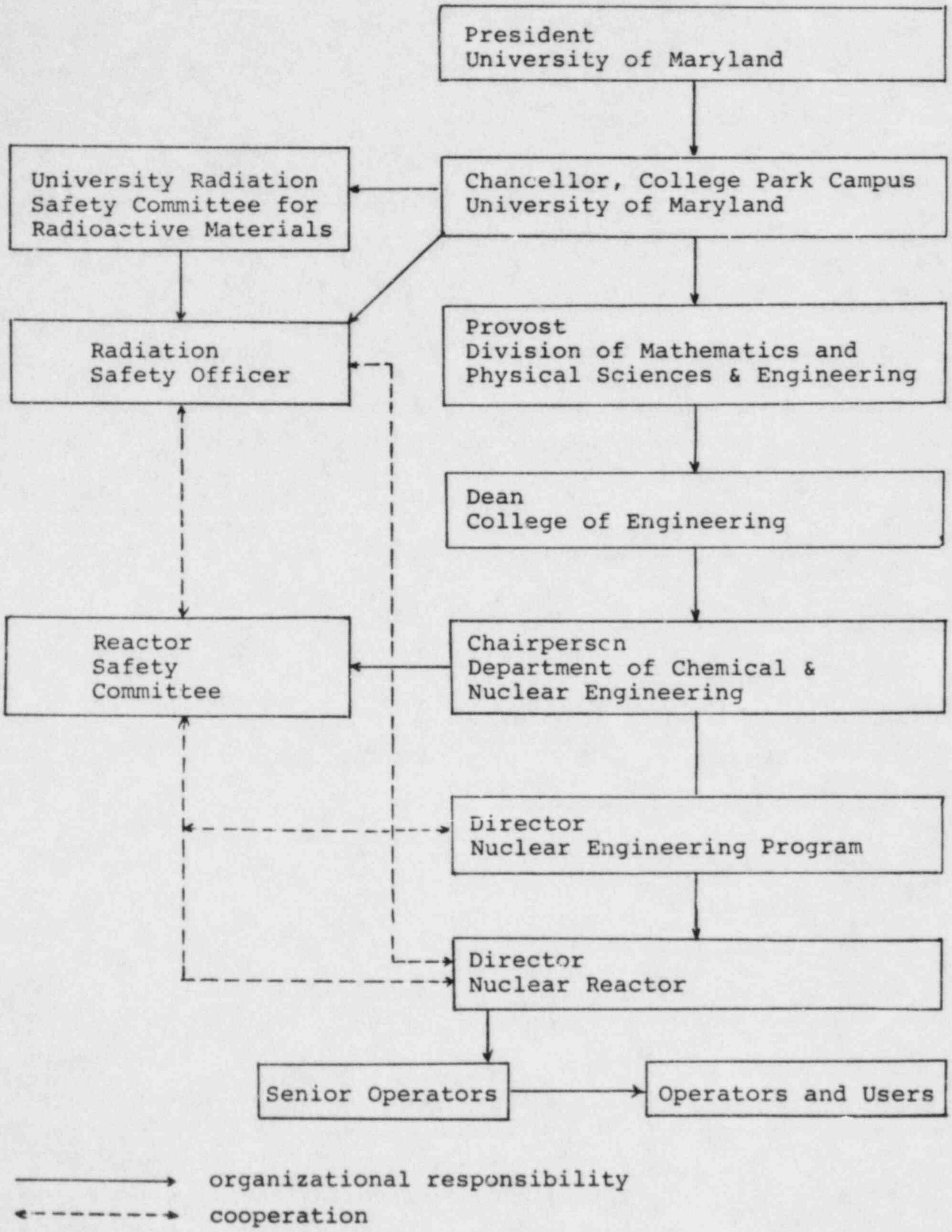

Figure 13.1 Administration chart

14 ACCIDENT ANALYSIS

As part of its evaluation of several pending license renewals for nonpower reactors, the staff asked Battelle Pacific Northwest Laboratory to analyze generic reactor accidents for U-ZrH_x fueled reactors (NUREG/CR-2387) and asked the Los Alamos National Laboratory to evaluate the licensee's submitted documentation and analysis of potential site-specific events. These analyses included the various types of possible accidents and the potential consequences to the public.

The following potential accidents or effects were considered to be sufficiently credible for evaluation and analysis.

(1) rapid insertion of reactivity (nuclear excursion)
(2) loss of coolant
(3) metal-water reactions
(4) misplaced experiments
(5) mechanical rearrangement of the fuel
(6) effects of fuel aging
(7) fuel handling (loss-of-cladding integrity)

Of these potential accidents, only the one leading to the loss-of-cladding integrity of one irradiated fuel cluster in air in the reactor room poses a potential significant impact to the environment outside the MUTR building. For purposes of classification, the staff will call this the "fuel-handling accident." This will be designated as the maximum hypothetical accident (MHA). An MHA is defined as a hypothetically conceived accident for which the risk to the public health and safety is greater than that from any event that can be postulated mechanistically. Thus, the staff assumes that the accident occurs but does not attempt to describe or evaluate all of the mechanical details of the accident or the probability of its occurrence. Only the consequences are considered.

14.1 Rapid Insertion of Reactivity

As discussed in Section 4.6, theoretical calculations have predicted and experimental measurements have confirmed that U-ZrH_x fuel exhibits a strong, prompt, negative temperature coefficient of reactivity. This temperature coefficient not only can terminate a nuclear excursion, but also causes a loss of reactivity as the steady-state temperature of the fuel is raised. These results have been verified at many operating TRIGA reactors. Although it may be possible theoretically to rapidly add sufficient excess reactivity under accident conditions to create an excursion that would not be terminated before fuel damage occurred, the limits imposed by the design and the Technical Specifications of the MUTR give reasonable assurance that such an event will not occur.

14.1.1 Scenario

The most severe reactivity accident postulated for the MUTR is the event in which all of the authorized excess reactivity, 3.60$, is inserted instantaneously into the reactor. The MUTR does not incorporate a fast-moving transient rod, and all three control rods are geared to operate at a maximum withdrawal rate of 48 cm/min. Therefore, the staff has not identified a credible method for inserting all of the excess reactivity instantaneously.

The staff has considered the scenario of the reactor operating at various steady-state power levels between 0 and 250 kW, at which time all of the remaining excess reactivity not compensated by increased temperature of the fuel at that power is inserted rapidly. The analysis neglected reactivity loss as a result of the buildup of ^{135}Xe, a conservative assumption. The staff found that the higher the temperature at which the rapid insertion is initiated, the lower the final temperature of the fuel immediately after the transient. Therefore, the staff has assumed the worst case: initiation of a 3.60$ reactivity insertion with the core at ambient temperature and just critical at effectively zero initial power.

The accident scenarios postulated by the licensee included the dropping of a central fuel cluster worth 4.70$ into position while the reactor was operating at its maximum steady-state power of 250 kW. The licensee has chosen an overly conservative accident scenario that the staff does not consider to be a credible scenario because (1) movement of fuel or rearrangement of the core is administratively prohibited while the reactor is operating and (2) the facility has insufficient spare fuel on hand to fulfill the assumptions.

14.1.2 Assessment

The potential significant consequences of the reactivity insertion accidents, considered by the staff, are melting of the fuel or cladding material and loss of integrity of the cladding as a result of high internal gas pressures and/or phase changes in the fuel matrix. The primary cause of cladding failure at elevated temperatures in stainless-steel-clad rods would be excessive stress buildup in the cladding caused by hydrogen pressure from thermal disassociation of the ZrH_x. Calculations performed by General Atomic Company (GA) and confirmed in many experimental and routine reactor pulses indicate that cladding integrity is maintained at transient peak fuel temperatures as high as 1,175°C (GA-6874, 1967; Simnad, 1976; GA-4314, 1980).

The staff used a Fuchs-Nordheim formulation for a self-limiting reactor excursion, modified to incorporate a temperature-dependent specific heat (Scalettar, 1963; GA-7882, 1967) to calculate the accident scenario. The calculations indicated that there is a maximum fuel material temperature of 685°C in any fuel rod following a 3.60$ reactivity insertion. The total energy released and peak power in the transient were calculated to be 34.5 MW-sec and 3700 MW, respectively. The resultant maximum fuel temperature of 685°C is well below the proposed safety limit for pulsing of 1,150°C (Simnad p 47, 1976).

The staff has also reviewed the literature for large reactivity insertions into reactor cores similar to the MUTR. GA has performed many experiments with reactivity insertions as high as 5.00$ in an 85-element (rod) TRIGA core. GA measured, among other parameters, the temperature of the fuel in the hottest core position, and they examined fuel elements afterward (GA-6874, 1967; Simnad, 1976). There was no indication of excessive stress in the cladding and no indication of either cladding or fuel melting. The measured maximum temperature for the 5.00$ pulse was approximately 750°C, and the estimated peak transient temperature at any localized point in the fuel was 1,175°C. Because the radial temperature distribution in a fuel element immediately following a pulse is similar to the radial power distribution, the peak transient temperature immediately after the pulse is located at the periphery of the hottest fuel element. It will fall rapidly (within seconds) as the heat flows toward the cladding and toward the fuel center. It also was observed that for a 5.00$ pulse the maximum measured transient pressure rise within an instrumented fuel element was far below the predicted equilibrium value at the peak temperature (GA-6874, 1967; GA-9064, 1970; Simnad, 1976). Thus, even the conservative fuel cluster loading accident postulated by the licensee would not lead to loss of integrity of fuel cladding or to fuel melting.

From the above considerations, the staff concludes that there is no credible nuclear excursion in the MUTR, as limited by its Technical Specifications, that could lead to fuel melting or cladding failure resulting from high temperature or high internal gas pressure. Therefore, there is reasonable assurance that fission product radioactivity will not be released from the fuel to the environment as a result of an accidental reactor pulse or excess reactivity insertion.

14.2 Loss of Coolant

A potential accident that would result in increases in the fuel and cladding temperatures is the loss of coolant after the reactor has been operating at steady-state power for some time. Because the water is required for neutron moderation, its removal would terminate any significant neutron chain reaction. However, the residual radioactivity from fission product decay would continue to deposit heat energy in the fuel.

14.2.1 Scenarios

It is assumed that the reactor has been operating at the licensed power of 250 kW long enough to achieve fission product equilibrium (a conservative assumption based on expected usage), and that it is shut down at the initiation of a gross cooling-water leak. It is further assumed that heat is removed by convective water cooling until the entire core becomes uncovered, after which heat removal is provided only by air convection.

Several investigations have evaluated similar scenarios under various assumptions (GA-6596, 1965; Oregon State, 1968; GA-9064, 1970; Texas A & M, 1979). At the power levels existing in the MUTR fuel elements, the peak fuel temperature following an instantaneous complete loss of coolant would be less than 300°C. In any credible scenario, a finite time (minutes) would be required for ~5,000 gal of water to leak out and uncover the core, resulting in somewhat lower ultimate fuel temperatures. Furthermore, the fuel rods would not reach maximum temperatures until at least 1 hour after the loss of coolant. Therefore, some protective action could be taken by the reactor operator who would be present if the reactor had been operating, as assumed.

14.2.2 Assessment

The location of the pool water outlet pipe and siphon breaks installed in the pool water and diffuser inlet pipes would prevent more than a 0.6-m-water drop in case any of these lines should rupture and siphon water from the reactor tank. The water level alarm and room radiation monitors required by the Technical Specifications would alert the operating staff to a low-water condition and allow them to shut the reactor down and initiate corrective action. Even if a coolant loss were to occur following an extended reactor run at the maximum authorized power level of 250 kW, the resultant maximum fuel and/or cladding temperatures would not cause fuel damage or fission product release.

14.3 Metal-Water Reactions

Chemical reactions, especially oxidation, may occur if sufficiently hot metal is brought into contact with water. This has been an area of concern and study in designing reactors since the early 1950s, and there is an extensive body of literature on the subject (Baker and Just, 1962; Baker and Liimatakinen, 1973; Buttrey, 1965; Merten, 1973). From the laboratory tests, it is concluded that the metal (reactor fuel) would have to be heated to very high temperatures (for example, above the melting point) and/or be fragmented into small hot particles and injected into water to support a rapid (explosive) chemical reaction. Either of these conditions implies a prior catastrophic event of some sort, which presumably would have to originate with a nuclear excursion or loss of coolant. In Sections 14.1 and 14.2, fuel rupture or melting from these events were shown not to be credible in a 250-kW U-ZrH_x-fueled reactor as authorized for operation at the University of Maryland.

Additionally, some of the studies (Baker and Liimatakinen, 1973) include metal-air and metal-steam chemical reactions. Violent (explosive) reactions do not appear to be possible in air or steam at atmospheric pressure, even though rapid reactions may occur at sufficiently high temperatures with specially prepared samples and conditions.

In addition to the investigations referenced above, GA has experimentally plunged heated samples of unclad ZrH_x into water to examine possible conditions for initiating and sustaining a metal-water reaction (Lindgren and Simnad, 1979). Up to temperatures of about 1,200°C, there was no apparent chemical reaction of the metal except for the formation of a relatively inert oxide film. Furthermore, most of the hydrogen may have been driven off in the hottest unclad test samples, so the metal surface in contact with the water could have been mostly zirconium rather than the hydride.

On the basis of the above considerations, the staff concludes that there is reasonable assurance that rapid (violent) metal-water, metal-air, or metal-steam reactions will not occur in the MUTR reactor operated at 250 kW and with the excess reactivity authorized in its Technical Specifications.

14.4 Misplaced Experiments

14.4.1 Scenario

This type of potential accident is one in which an experimental sample or device is inadvertently located in an experimental facility where the irradiation conditions could exceed the design specifications of the experiment. In that case, the sample might become overheated or develop pressures that could cause a failure of the experiment container. As discussed in Sections 4, 10, and 13 and as required by the Technical Specifications, all new experiments at the MUTR are reviewed before insertion, and all experiments are separated from the fuel cladding by at least one barrier, such as the pneumatic transfer tube, beam and through tubes, or thermal column.

14.4.2 Assessment

The staff concludes that the experimental facilities and the procedures for experiment review at the University of Maryland are adequate to provide reasonable assurance that failure of experiments is not likely, and even if failure occurred, breaching of the reactor fuel cladding will not occur. Furthermore, if an experiment should fail and release radioactivity within an experimental facility, there is reasonable assurance that the amount of radioactivity released to the environment would not be more than that from the accident discussed in Section 14.7.

14.5 Mechanical Rearrangement of the Fuel

This type of potential accident would involve the failure of some reactor system, such as the support structure, or could involve an externally originated event that disperses the fuel and in so doing breaches the cladding of one or more fuel elements.

14.5.1 Scenario

The staff has not developed an operational scenario for such accidents that would produce consequences greater than those considered in Section 14.7, which discusses a scenario assuming the failure of the cladding of fuel rods after extended reactor operation and evaluates possible doses resulting from the release of the contained radioactive inventory. This approach should address the spectrum of fuel-cladding failures. (The scenario in which the initiating event causes all of the control rods to somehow simultaneously be ejected from the core initiating a nuclear excursion is encompassed in Section 14.1.)

14.5.2 Assessment

The staff concludes that no mechanical rearrangement that is credible would lead to an accident with more severe consequences than those accidents considered in Sections 14.1 or 14.7.

14.6 Effects of Fuel Aging

The staff has included this process in this section so that all credible effects are addressed. However, as discussed in more detail in Section 17, fuel aging should be considered normal with the continued use of the reactor and is expected to occur gradually. The reactions external to the cladding that might occur also are addressed in Section 17; the possibility of internal reactions is discussed in this section.

14.6.1 Scenario

There is some evidence that U-ZrH_x fuel tends to fragment with use, probably because of the stresses caused by high temperature gradients and the high rate of heating during pulsing (GA-9064, 1970; GA-4314, 1980). On the other hand, the staff is aware of no evidence that steady-state operation causes fragmentation, so the following discussion might not be applicable to the MUTR, which is not authorized for pulse operation. Some of the possible consequences of fragmentation are (1) a decrease in thermal conductivity across cracks, leading to higher central fuel temperatures during steady-state operation, and (2) an increase in the amount of fission products released into the cracks in the fuel.

With regard to the first item above, hot cell examination of thermally stressed hydride fuel bodies has shown relatively widely spaced radial cracks that would cause minimal interference with radial heat flow (GA-9064, 1970; GA-4314, 1980). However, after pulsing, TRIGA reactors have exhibited an increase in both steady-state fuel temperatures and power reactivity coefficients. At power levels of 500 kW, temperatures have increased by ~20C°, and power reactivity coefficients have increased by ~20% (AFRRI, 1960; GA-5400, 1965). GA has attributed these changes to an increased gap between the fuel material and stretched cladding (caused by rapid fuel expansion during pulse heating) that reduces the heat transfer coefficient. Experience has shown that the observed changes occur mostly during the first several large pulses and have essentially saturated after 100 pulses. However, because pulses are not performed, these effects should not occur in the MUTR fuel.

Two mechanisms for fission product release from TRIGA fuel have been proposed by GA (GA-8597, 1968; Foushee and Peters, 1971; GA-4314, 1980; Baldwin et al., 1980). The first mechanism is fission fragment recoil into connected gaps within the fuel cladding. This effect predominates up to about 400°C and is independent of fuel temperature. GA has postulated that in a closed system such as exists in a TRIGA fuel element, fragmentation of the fuel material within the cladding will not cause an increase in the fission product release fraction (GA-8597, 1968). The reason for this is that the total free volume available for fission products remains constant within the confines of the cladding. Under these conditions, the formation of a new gap or widening of an existing gap must cause a corresponding narrowing of an existing gap at some other location. Such a narrowing allows more fission fragments to traverse the gap and become embedded in the fuel or cladding material on the other side. In a closed system, the average gap volume, and, therefore, the fission product release rate, remains approximately constant independent of the extent to which fuel material is broken up.

Above ~400°C the predominant mechanism for fission product release is diffusion, and the fraction released is dependent on fuel temperature history and fuel surface-to-volume ratio. However, release fractions used for safety evaluation are based on a conservative calculation that assumed a fractional release greater than expected in actual operation.

14.6.2 Assessment

The staff concludes that the two likely effects of aging of the U-ZrH_x fuel moderator would not have a significant effect on the operating temperature of the fuel or on the assumed release of gaseous fission products from the cladding. In addition fuel elements are visually inspected periodically at the MUTR for damage or deterioration. Therefore, the staff also concludes that there is reasonable assurance that fuel aging will not significantly increase the likelihood of fuel cladding failure or the consequences calculated for an accidental release in the event of loss of cladding integrity.

14.7 Fuel-Handling Accident

This potential accident, designated as the MHA for the MUTR, includes various incidents to one or more irradiated fuel elements in which the fuel cladding might be breached or ruptured.

14.7.1 Scenario

To be general, the staff let the scenario include the time scale from immediately after a long run at full licensed power to any later time such as, for example, when moving stored irradiated fuel from a rack in the pool into the reactor room. Also to remain general, the staff did not try to develop a detailed scenario, but simply assumed that the cladding of one fuel rod cluster certainly fails and that the volatile fission products accumulated in the free volume between the fuel and the cladding are released abruptly.

As indicated in Section 14.6.1, several series of experiments at GA have given data on the species and fractions of fission products released from $U\text{-}ZrH_x$ under various conditions. The noble gases and halogens were the principal species found to be released. When the fuel specimens were irradiated at temperatures below ~350°C, the fraction released could be summarized as a constant equal to 1.5×10^{-5}, independent of the temperature. At irradiation temperatures greater than ~350°C, the species released remained approximately the same, but the fractions released increased significantly with increasing temperature.

GA has proposed a hypothesis describing the release mechanisms in the two temperature regimes that appears to be valid. It seems reasonable to accept the interpretation of the low-temperature results, which imply that the fraction released for a typical TRIGA fuel rod will be a constant, independent of operating history or details of operating temperatures, and will apply to fuel whose temperature is not raised for long periods of time above approximately 400°C. This means that the 1.5×10^{-5} release fraction can be reasonably applied to TRIGA reactors operating up to at least 800 kW steady-state power, including the MUTR, which is licensed to operate at a maximum steady-state power of 250 kW. Because the noble gases do not condense or combine chemically, it is valid to assume that any released from the cladding will diffuse in the air until their radioactive decay. On the other hand, the iodines are chemically active and are not volatile below about 180°C. Therefore, some of the radioiodines will be trapped by materials with which they come in contact, such as water and structures. In fact, evidence indicates that most of the iodines either will not become or not remain airborne under many accident scenarios that are applicable to nonpower reactors (NUREG-0771). However, to be certain that the fuel-cladding failure scenario discussed below led to upper-limit dose estimates for all events, the staff assumed that 100% of the iodines in the gap become airborne. This assumption will lead to computed doses that may be at least a factor of 100 too high in some scenarios, for example, those in which the pool water is present (NUREG/CR-2387).

The staff analyzed a cladding failure in air of all four rods of a centrally located fuel element cluster, as might occur in a fuel-handling accident, and calculated the resultant doses in the reactor room and the closest unrestricted area outside the reactor building. The calculations assumed the reactor had been operating at 250 kW, the maximum authorized power level, and all fission products had reached their saturated activity (a conservative assumption considering the typical operating history at the MUTR). No radioactive decay was assumed during the time between reactor shutdown and the accident initiation. Scenarios incorporating realistic estimates of these effects (operating history and radioactive decay) could significantly reduce the computed doses. The analysis assumed a fission product release fraction of 1.5 x 10^{-5} of the inventory of both noble gases and halogens; a value appropriate for nonpulsing TRIGA fuel operated at temperatures below 400°C, as is the MUTR. All of the noble gases and halogens in the fuel cladding gap were assumed to be released from the fuel rod and instantaneously to form a uniform distribution in the reactor room air (no plate-out was allowed). The staff calculated the whole-body gamma-ray (immersion) dose and thyroid dose by iodine inhalation to an individual in the reactor room, as well as the integrated dose to an individual in the unrestricted area immediately outside the building. For the within-the-room doses, it was assumed that the ventilation system was shut down at the time of the accident and all the fission products remained in the reactor room. For the outside doses, it was assumed that the ventilation system failed to shut down and operated at its rated capacity, and that the exposed individual in the unrestricted area remained there while all of the contaminated air was exhausted from the building. All dose calculations assumed immersion in a semi-infinite cloud (a very conservative assumption that produces the highest calculated exposures). The resulting calculated doses are presented in Table 14.1.

In their SAR submitted June 1980, the licensee evaluated the effects of a similar accident and obtained resulting doses lower than those calculated by the staff. In his response to staff questions, dated December 19, 1983, the licensee presented an updated calculation of doses which agreed with the staff's results for the same set of assumptions.

14.7.2 Assessment

Because there is no credible way in which the postulated accident could occur without operating personnel being alerted immediately, orderly evacuation of the reactor bay would be accomplished within minutes. Because of the underlying conservative calculative and atmospheric assumptions, the calculated doses to operational personnel and to the most exposed member of the public shown in Table 14.1 are higher than could occur realistically.

On the basis of the discussions and analyses above, the staff concludes that if four fuel rods contained in one fuel rod cluster from the MUTR were to release all noble gases and halogen fission products accumulated in the fuel-cladding gap, radiation doses to both occupational personnel and to the public in unrestricted areas would be within the guidelines of 10 CFR 20. The calculative assumptions correspond to a very conservative scenario. Furthermore, from the results the staff obtained, even if one-half of the fuel rods failed simultaneously, the expected whole-body doses in unrestricted areas outside the reactor building would be less than 25 mrem, and thus would still be well within 10 CFR 20 limits for the public in unrestricted areas.

For calculating potential doses outside the reactor building, the accident scenario assumed that the ventilation system of the building was not closed after the accident. This adds conservatism to the scenario. On the basis of these considerations, the staff concludes that even in the event of a multiple fuel-cladding failure at the reactor, there would be no significant risk to the health and safety of the public.

14.8 Conclusion

The staff has reviewed the credible nuclear excursions and other accidents for the MUTR. On the basis of this review, the postulated accident with the greatest potential effect on the public is the loss of cladding integrity of one irradiated fuel rod cluster in air in the reactor room. The analysis of this accident has shown that even if several fuel clusters failed at once, the expected dose equivalents in unrestricted areas would still be within the guidelines and limits of 10 CFR 20. Therefore, the staff concludes that the design of the facility and the Technical Specifications provide reasonable assurance that the MUTR can be operated with no significant risk to the health and safety of the public.

Table 14.1 Doses resulting from postulated fuel-handling accident

Exposure levels	Whole-body immersion dose	Thyroid dose*
10-min occupational dose in the reactor room	15 mrem	1.4 rem
Maximum individual dose immediately outside building	0.50 mrem	54 mrem

* Total integrated dose equivalent from thyroid uptake of iodines.

15 TECHNICAL SPECIFICATIONS

The licensee's Technical Specifications evaluated in this licensing action define certain features, characteristics, and conditions governing the continued operation of this facility. These Technical Specifications will be explicitly included in the renewal license as Appendix A. Formats and contents acceptable to the NRC have been used in the development of these Technical Specifications, and the staff has reviewed them using the standard ANSI/ANS 15.1-1982 as a guide.

On the basis of its review, the staff concludes that normal reactor operation within the limits of the Technical Specifications will not result in offsite radiation exposures in excess of 10 CFR 20 limits. Furthermore, the limiting conditions for operation, surveillance requirements, and engineered safety features will limit the likelihood of malfunctions and mitigate the consequences to the public of offnormal or accident events.

16 FINANCIAL QUALIFICATIONS

The Maryland University Training Reactor is owned and operated by a state educational institution in support of its role in education and research. On the basis of financial information supplied by the licensee in its May 23, 1980 submittal, the staff concludes that funds will be made available, as necessary, to support continued operations and eventually to shut down the facility and maintain it in a condition that would constitute no risk to the public. The licensee's financial status was reviewed and found to be acceptable in accordance with the requirements of 10 CFR 50.33(f).

17 OTHER LICENSE CONSIDERATIONS

17.1 Prior Reactor Utilization

Previous sections of this SER concluded that normal operation of the reactor causes insignificant risk of radiation exposure to the public and that only an offnormal or accident event could cause some exposure. Even a maximum hypothetical accident (defined as one that is worse than can be mechanistically justified) would not lead to a dose to the most exposed individual greater than applicable guidelines or regulations (10 CFR 20).

In this section, the staff reviews the impact of prior operation of the facility on the risk of radiation exposure to the public. The two parameters involved are the likelihood of an accident and the consequences if an accident occurred.

Because the staff has concluded that the reactor was initially designed and constructed to be inherently safe, with additional engineered safety features, the staff must also consider whether operation will cause significant degradation in these features. Furthermore, because loss of integrity of fuel cladding is the maximum hypothetical accident, the staff has considered mechanisms that could increase the likelihood of failure. Possible mechanisms are (1) radiation degradation of cladding strength, (2) high internal pressure caused by high temperature leading to exceeding the elastic limits of the cladding, (3) corrosion or erosion of the cladding leading to thinning or other weakening, (4) mechanical damage as a result of handling or experimental use, and (5) degradation of safety components or systems.

The staff's conclusions regarding these parameters, in the order in which they were identified above, are

(1) The stainless-steel-clad high-hydride TRIGA fuel in the core has been in use since 1974 and has been subjected to less than 1% burnup of ^{235}U. TRIGA fuel at more extensively used reactors has been in use for many times as much burnup, with no observable degradation of cladding as a result of radiation.

(2) Because the reactor operates at a maximum power level of 250 kW, the temperature of the fuel does not exceed 400°C during normal operation. At this temperature, the pressure of the air and/or free hydrogen within the cladding does not increase significantly.

(3) Water flow through the core is obtained by natural thermal convection, so the staff concludes that erosion effects as a result of high flow velocity will be negligible. High primary water purity is maintained by continuous passage through the filter and demineralizer system. With conductivity below about 5 μmho-cm^{-1}, as limited by the Technical Specifications, corrosion of the stainless-steel cladding is expected to be negligible.

(4) The fuel is handled as infrequently as possible, consistent with periodic surveillance. Any indications of possible damage or degradation are

investigated immediately. The only experiments which are placed near the core are isolated from the fuel cladding by a water gap and at least one metal barrier, such as the pneumatic tubes or the central thimble. Therefore, the staff concludes that loss of integrity of cladding through damage does not constitute a significant risk to the public.

(5) MUTR performs regular preventive and corrective maintenance and replaces components as necessary. Nevertheless, there have been some malfunctions of equipment. However, the staff review indicates that most of these malfunctions have been random one-of-a-kind incidents, typical of even good quality electromechanical instrumentation. There is no indication of significant degradation of the instrumentation, and the staff further concludes that the preventive maintenance program would lead to adequate identification and replacement before significant degradation occurred. Therefore, the staff concludes that there has been no apparent significant degradation of safety equipment and, because there is strong evidence that any future degradation will lead to prompt remedial action by MUTR personnel, there is reasonable assurance that there will be no significant increase in the likelihood of occurrence of a reactor accident as a result of component malfunction.

The second aspect of risk to the public involves the consequences of an accident. On the basis of the discussions in Section 14, the staff concludes (1) that the risk of radiation exposure to the public has been acceptable and well within all applicable regulations and guidelines during the history of the reactor, and (2) that there is reasonable assurance that there will be no increase in that risk in any discernible way during this renewal period. These conclusions are based on operation of MUTR on the maximum schedule and at the maximum power level authorized in the license. However, because the MUTR has not and is not expected to operate on the maximum available schedule, the inventory of radioactive fission products will be far below that postulated in the evaluation of the maximum hypothetical accident both by the applicant and the NRC staff.

17.2 Multiple or Sequential Failures of Safety Components

Of the many accident scenarios hypothesized for the MUTR, none produce consequences more severe than the accidents reviewed and evaluated in Section 14. The only multiple-mode failure of more severe consequences would be failure of the cladding of more than one fuel rod cluster. No credible scenario constructed by the staff has revealed a mechanism by which the failure of integrity of one fuel rod cluster can cause or lead to the failure of additional elements. Therefore, if the cladding of more than one fuel rod cluster should fail, the failures would either be random, or a result of the same primary event. Additionally, the reactor contains redundant safety-related measuring channels and control rods. Failure of all but one control rod and all but one safety channel would not prevent reactor shutdown to a safe condition. The staff review has revealed no mechanism by which failure or malfunction of one of these safety-related components could lead to a nonsafe failure of a second component.

18 CONCLUSIONS

Based on its evaluation of the application as set forth above, the staff has determined that

(1) The application for renewal of Operating License R-70 for its research reactor filed by the University of Maryland dated May 23, 1980, as supplemented, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR, Chapter 1.

(2) The facility will operate in conformity with the application as supplemented, the provisions of the Act, and the rules and regulations of the Commission.

(3) There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public; and (b) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR, Chapter 1.

(4) The licensee is technically and financially qualified to engage in the activities authorized by the license in accordance with the regulations of the Commission set forth in 10 CFR, Chapter 1.

(5) The renewal of this license will not be inimical to the common defense and security or to the health and safety of the public.

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NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER *(Assigned by DDC)* NUREG-1043	
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16. ABSTRACT *(200 words or less)*

This Safety Evaluation Report for the application filed by the University of Maryland, (UMD) for a renewal of operating license R-70 to continue to operate a training and research reactor facility has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is owned and operated by the University of Maryland and is located at a site in College Park, Prince Georges County, Maryland. The staff concludes that this training reactor facility can continue to be operated by UMD without endangering the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
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17b. IDENTIFIERS/OPEN-ENDED TERMS

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MARCH 1984