August 5, 2002

Mr. Gary N. Nugent
Chief Executive Officer
Department of Veterans Affairs
Nebraska - Western Iowa Health Care System
4101 Woolworth Avenue
Omaha, NE 68105

SUBJECT: DEPARTMENT OF VETERANS AFFAIRS ALAN J. BLOTCKY REACTOR FACILITY - AMENDMENT NO. 11 RE: RENEWAL OF FACILITY OPERATING LICENSE NO. R-57 (TAC NO. MB8345)

Dear Mr. Nugent:

The U.S. Nuclear Regulatory Commission (NRC) has issued Amendment No. 11 for Facility Operating License No. R-57 for the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System, Alan J. Blotcky Reactor Facility in response to the application for renewal dated May 10, 1993, as supplemented on March 1, 1995, December 17, 1997, March 12, April 5, July 29, November 24 and December 2, 1999, January 4, September 25, October 2 and October 24, 2000, and August 8 and October 16, 2001. This amendment renews the operating license for twenty years from its date of issuance.

In accordance with our practice, we have restated the license in its entirety, incorporating all the changes and amendments made since the issuance of the original license.

The Department of Veterans Affairs has recently decided to decommission the Alan J. Blotcky Reactor Facility and has informed the NRC of the decision. Because the updated technical specifications issued with this license renewal add flexibility to the performance of surveillance requirements that will be useful during the time period prior to approval of a decommissioning plan, the NRC has decided to issue this license renewal.

Enclosed with the amended license is a copy of the notice of renewal that is being sent to the Office of the Federal Register for publication, and the Safety Evaluation Report associated with the renewal. The Environmental Assessment was sent to you under separate cover. If you have any questions, please contact me at 301-415-1127.

Sincerely,

/RA/

Alexander Adams, Jr., Senior Project Manager
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-131

Enclosures:
1. Amendment No. 11
2. Notice of Renewal
3. Safety Evaluation Report

cc w/ enclosures:

Please see next page

cc:

Mayor
City of Omaha
Omaha, NE 68102

Mr. John P. Claassen
Reactor Manager/Supervisor
Omaha Veterans Administration
Medical Center
4101 Woolworth Avenue
Omaha, NE 68105

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

Julia Schmitt, Program Manager
Department of Health and Human Services
P.O. Box 95007
Lincoln, NE 68509-5007

M. Brenda Hebert (12C1)
Department of Veterans Affairs
810 Vermont Avenue, N.W.
Washington, DC 20420

Enclosed with the amended license is a copy of the notice of renewal that is being sent to the Office of the Federal Register for publication, and the Safety Evaluation Report associated with the renewal. The Environmental Assessment was sent to you under separate cover. If you have any questions, please contact me at 301-415-1127.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

RENEWAL OF FACILITY OPERATING LICENSE

DOCKET NO. 50-131

DEPARTMENT OF VETERANS AFFAIRS

NEBRASKA - WESTERN IOWA HEALTH CARE SYSTEM

ALAN J. BLOTCKY REACTOR FACILITY

Amendment No. 11
License No. R-57

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment for renewal of Facility Operating License No. R-57 filed by the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System (the licensee) dated May 10, 1993, as supplemented on March 1, 1995, December 17, 1997, March 12, April 5, July 29, November 24 and December 2, 1999, January 4, September 25, October 2 and October 24, 2000, and August 8 and October 16, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

    B. Construction of the Alan J. Blotcky Reactor Facility (the facility) was completed in substantial conformity with Construction Permit No. CPRR-36 dated June 24, 1959, the provisions of the Act, and the rules and regulations of the Commission;

    C. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    D. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's rules and regulations;

    E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the rules and regulations of the Commission;

    F. The licensee is a Federal Agency which will use the facility for the conduct of educational training and academic research purposes, and need not furnish proof of financial protection as would otherwise be required by subsection 170a of the Act;

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. The issuance of this license is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession and use of the byproduct and special nuclear materials as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30 and 70, including Sections 30.33, 70.23, and 70.31.

2. Facility License No. R-57 is hereby amended in its entirety to read as follows:

A. The license applies to the TRIGA nuclear research reactor located at the Alan J. Blotcky Reactor Facility owned by the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System. The facility is located at the Nebraska - Western Iowa Health Care System, Omaha Division (formerly known as the VA Medical Center Omaha) in Omaha, Douglas County, Nebraska, and is described in the licensee's amendment application for renewal of the license dated May 10, 1993, as supplemented.

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System:

(1) Pursuant to Section 104c of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location in Omaha, Douglas County, Nebraska, in accordance with the procedures and limitations described in the application and set forth in this license;

(2) Pursuant to the Act and 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material," to receive, possess, and use in connection with operation of the facility:

a. Up to 3.3 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel;

b. Up to 20 grams of contained uranium-235 of any enrichment in the form of fission chambers; and

c. To possess, use, but not separate, such special nuclear material as may be produced by the operation of the facility.

(3) Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material," to receive, possess and use in connection with operation of the facility:

a. Up to 8 curies of polonium-beryllium in the form of sealed sources;

b. Up to 4 curies of americium-beryllium in the form of sealed sources;

c. Up to 1.5 curies of cesium-137 in the form of sealed sources;

d. Up to 10 millicurie each of iodine-129, barium-133, lead-210, cobalt-60, and technetium-99 in the form of sealed sources; and

e. To possess, use, but not separate, except for byproduct material produced in non-fueled experiments, such byproduct material as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in Parts 20, 30, 50, 51, 55, 70, and 73 of 10 CFR Chapter I, to all applicable provisions of the Act, and to the rules, regulations and orders of the Commission now, or hereafter in effect, and to the additional conditions specified below:

(1) <u>Maximum Power Level</u>

The licensee is authorized to operate the reactor at steady-state power levels not in excess of 20 kilowatts (thermal).

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 11, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

D. This license is effective as of the date of issuance and shall expire twenty years from its date of issuance.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Patrick M. Madden, Section Chief
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosure: Appendix A Technical Specifications

Date of Issuance: August 5, 2002

ENCLOSURE TO LICENSE AMENDMENT NO. 11

FACILITY OPERATING LICENSE NO. R-57

DOCKET NO. 50-131

Replace the following pages of Appendix A, "Technical Specifications," with the enclosed pages.

Remove	Insert
Cover Page thru 31	Cover Page thru 34

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

NOTICE OF RENEWAL OF FACILITY OPERATING LICENSE NO. R-57

DEPARTMENT OF VETERANS AFFAIRS

NEBRASKA - WESTERN IOWA HEALTH CARE SYSTEM

DOCKET NO. 50-131

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 11 to Facility Operating License No. R-57 for the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System (the licensee), which renews the license for operation of the Alan J. Blotcky Reactor Facility located at the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System, Omaha Division (formerly known as the VA Medical Center Omaha) in Omaha, Nebraska.

The facility is a non-power reactor that has been operating at a power level not in excess of 20 kilowatts (thermal). The renewed Facility Operating License No. R-57 will expire twenty years from its date of issuance.

The amended license complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I. Those findings are set forth in the license amendment. Opportunity for hearing was afforded in the notice of the proposed issuance of this renewal in the FEDERAL REGISTER on January 26, 1995, at 60 FR 5228. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

Continued operation of the reactor will not require alteration of buildings or structures, will not lead to significant changes in effluents released from the facility to the environment, will not increase the probability or consequences of accidents, and will not involve any unresolved issues concerning alternative uses of available resources. Based on the foregoing and on the Environmental Assessment, the Commission concludes that renewal of the license will not result in any significant environmental impacts.

The Commission has prepared a "Safety Evaluation Report Related to the Renewal of the Operating License for the Research Reactor at the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System, Omaha Division" for the renewal of Facility Operating License No. R-57 and has, based on that evaluation, concluded that the facility can continue to be operated by the licensee without endangering the health and safety of the public.

The Commission also prepared an Environmental Assessment which was published in the FEDERAL REGISTER on November 27, 2001, (66 FR 59267) for the renewal of Facility Operating License No. R-57 and has concluded that this action will not have a significant effect on the quality of the human environment.

For further details with respect to this action, see: (1) the application for amendment dated May 10, 1993, as supplemented on March 1, 1995, December 17, 1997, March 12, April 5, July 29, November 24 and December 2, 1999, January 4, September 25, October 2 and October 24, 2000, and August 8 and October 16, 2001, (2) Amendment No. 11 to Facility Operating License No. R-57; (3) the related Safety Evaluation Report and (4) the Environmental Assessment dated November 20, 2001. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Documents related to this license renewal dated on or after November 24, 1999, may be accessed through the NRC's Public Electronic Reading Room on the internet at http://www.nrc.gov. If you do not have access to ADAMS or it there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room (PDR) Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 5TH day of August 2002.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

/RA/

Patrick M. Madden, Section Chief
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Safety Evaluation Report Related to the Renewal of the Operating License for the Alan J. Blotcky Reactor Facility at the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System, Omaha Division

August 2002

Office of Nuclear Reactor Regulation

ABSTRACT

This safety evaluation report (SER) summarizes the findings of a safety review conducted by the staff of the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Reactor Regulation (NRR). The staff conducted this review in response to a timely application filed by the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System (the licensee or VA) for a 20-year renewal of Facility Operating License R-57 to continue to operate the Alan J. Blotcky Reactor Facility (AJBRF or the facility). The facility is located in the basement of the Nebraska - Western Iowa Health Care System, Omaha Division (formerly known as the VA Medical Center Omaha) in Omaha, Nebraska. In its safety review, the staff considered information submitted by the licensee (including past operating history recorded in the licensee's annual reports to the NRC), as well as inspection reports prepared by NRC personnel and first-hand observations. On the basis of this review, the staff concludes that the VA can continue to operate the facility, in accordance with its application, without endangering the health and safety of the public.

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1 INTRODUCTION

1.1 Overview

By letter (and supporting documentation) dated May 10, 1993, as supplemented on March 1, 1995, December 17, 1997, March 12, April 5, July 29, November 24 and December 2, 1999, January 4, September 25, October 2 and October 24, 2000, and August 8 and October 16, 2001, the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System (VA or the licensee) submitted to the U.S. Nuclear Regulatory Commission (NRC or the Commission) a timely application for a 20-year renewal of the Class 104c Facility Operating License No. R-57 (NRC Docket No. 50-131). Such a renewal would authorize continued operation of the TRIGA-type research reactor (the reactor) located at the Alan J. Blotcky Reactor Facility (AJBRF or the facility) at the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System, Omaha Division (formerly known as the VA Medical Center Omaha) in Omaha, Nebraska. Until the staff completes action on the renewal request, the licensee is permitted to operate the reactor under the conditions authorized in past amendments in accordance with Title 10, Section 2.109 of the *U.S. Code of Federal Regulations* (10 CFR 2.109).

The staff's review, with respect to renewing the AJBRF operating license, was conducted on the basis of information contained in the renewal application, as well as supporting supplements and licensee responses to staff requests for additional information. Specifically, the renewal application included financial statements, the safety analysis report, an environmental report, the Operator Requalification Program, and Technical Specifications (TSs). The licensee also requested that the staff consider as part of the application the Emergency Plan and Physical Security Plan previously filed with the NRC. The licensee has since updated the Emergency Plan in response to a request for additional information issued by the staff as part of the license renewal process, and as part of the licensee's routine maintenance of the Emergency Plan under 10 CFR 50.54(q). The licensee subsequently requested that the requirement to maintain a security plan be removed from the license. This request was considered by the staff as part of the license renewal review. As part of the review, the staff also reviewed annual reports of facility operation submitted by the licensee and inspection reports prepared by NRC personnel. Several site visits were conducted at the facility to observe facility conditions.

The Department of Veterans Affairs has recently decided to decommission the Alan J. Blotcky Reactor Facility and has informed the NRC of the decision. Because the updated technical specifications issued with this license renewal add flexibility to the performance of surveillance requirements that will be useful during the time period prior to approval of a decommissioning plan, the NRC has decided to issue this license renewal.

With the exception of the Physical Security Plan, this material may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. Documents related to this license renewal dated on or after November 24, 1999, may be accessed through the NRC's Public Electronic Reading Room on the Internet at http://www.nrc.gov. If you do not have access to ADAMS, if there are problems in accessing the documents located in ADAMS, or if you want access to documents before November 24, 1999, contact the NRC Public Document Room Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov. The Physical Security Plan is protected from public disclosure under 10 CFR 2.790.

In conducting its safety review, the staff evaluated the facility against the requirements of 10 CFR Parts 20, 30, 50, 51, 55, 70, and 73; applicable regulatory guides; and relevant accepted industry standards, such as the American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series. The staff also referred to the guidance contained in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors." Because there are no specific accident-related regulations for research reactors, the staff compared calculated dose values for accidents with related standards in 10 CFR Part 20 (the standards for protecting employees and the public against radiation). Amendments to 10 CFR Part 20 (20.1001 through 20.2402 and Appendices) became effective January 1, 1994. Among other things, these amendments changed the dose limits for occupationally exposed persons and members of the public, as well as the concentrations of radioactive material that are allowed in effluents released from licensed facilities. The licensee must follow the requirements of 10 CFR Part 20, as amended, for all aspects of operation regarding the AJBRF. However, in conducting the accident evaluation, the staff used the dose limits in 10 CFR Part 20 that have been historically applied to accidents at this reactor (10 CFR 20.1 through 20.602 and Appendices referred to as the "old" Part 20). See NUREG-1537, Chapter 13 for additional discussion of accident dose limits.

The purpose of this safety evaluation report (SER) is to summarize the findings of the staff's safety review of the facility and to delineate the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for renewing the license for operation of the reactor at thermal power levels up to and including 20 kW.

This SER contains 18 chapters which discuss the following topics:

- Chapter 1 contains a summary and conclusions regarding the principal safety considerations of the staff review, the history and a general description of the facility, information on shared facilities and equipment, comparison with similar facilities, and how the licensee complies with the Nuclear Waste Policy Act of 1982.

- Chapter 2 describes the site and applicable site characteristics, including geography, demography, meteorology, hydrology, geology, seismology, and interaction with nearby installations and facilities.

- Chapter 3 describes the design bases of facility structures, systems, and components and the responses to environmental factors on the reactor site.

- Chapter 4 describes the design bases and the functional characteristics of the reactor core and its components. In this chapter, the safety considerations and features of the reactor are discussed.

- Chapter 5 discusses the design bases and describes the function of the reactor coolant and associated systems, including the primary and secondary coolant systems, and coolant makeup and purification systems.

- Chapter 6 lists the design bases and describes the function of the facility engineered safety features (ESFs) that may be used to mitigate consequences of postulated accidents at the facility.

- Chapter 7 lists the design bases and describes the function of the instrumentation and control (I&C) systems and subsystems at the facility, placing emphasis on safety-related systems and safe reactor shutdown.

- Chapter 8 lists the design bases and describes the functions of the electrical power systems at the facility.

- Chapter 9 lists the design bases and describes the functions of auxiliary systems, such as fuel handling and storage, fire protection warning and communication systems, and research facilities.

- Chapter 10 lists the design bases and describes the functions of the experimental facilities. Non-power reactors are designed with irradiation capabilities for research, education, and technological development. This chapter discusses the characteristics of experiment and irradiation facilities and the basis of the proposed experimental programs.

- Chapter 11 lists the design bases and describes the functions of the radiation protection and the radioactive waste management programs at the facility. The description of the radiation protection program includes health physics staffing and procedures, monitoring programs for personnel exposures and effluent releases, and assessment and control of radiation doses, both to workers and the public. The facility program to maintain radiation exposures and releases as low as reasonably achievable (ALARA) is described in this chapter. The program for radioactive waste management is described including the control and disposal of radiological waste from both reactor operations and experimental programs. The impact on the facility staff and members of the public from radioactive effluent releases from the facility are described.

- Chapter 12 lists the bases and describes the functions of plans and procedures for the conduct of facility operations. These include discussions of the management structure, personnel training and evaluation, provisions for safety review and auditing of operations by the Radiation Safety Committee, and other required functions, such as reporting, and security and emergency planning.

- Chapter 13 lists the bases, scenarios, and analyses of accidents at the reactor facility, and describes the maximum hypothetical accident, which is a fission product release from one fuel element whose cladding fails in air. The radiological consequences from analyzed accidents to the facility staff and members of the public are discussed.

- Chapter 14 discusses the TSs, which state the operating limits and conditions and other requirements for the facility to acceptably ensure protection of the health and safety of the public.

- Chapter 15 concerns financial qualifications of the licensee for continuing operations and decommissioning.

- Chapter 16 discusses prior reactor utilization focusing on aging of the fuel and safety systems.

- Chapter 17 contains the major conclusions of the staff's review of the licensee's renewal application.

- Chapter 18 contains references used for the staff review.

This SER was prepared by Mr. Alexander Adams Jr., Senior Project Manager, from the NRC's Office of Nuclear Reactor Regulation (NRR), Division of Regulatory Improvement Programs, Operating Reactor Improvements Program, Research and Test Reactors Section. Other major contributors to the technical review included R.E. Carter, of the Idaho National Engineering and Environmental Laboratory (INEEL), under contract to the NRC.

1.2 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 of safety inspections by the NRC staff. 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 AJBRF reactor were:

- The design 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.

- 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 licensee 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 Part 20 doses for unrestricted areas.

- 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.

- The systems provided for the control of radiological effluents can be operated to ensure that releases of radioactive materials from the facility are within the limits of the Commission's regulations and are ALARA.

- The licensee's TSs, which provide limits controlling operation of the facility, are such that there is a high degree of assurance that the facility can be operated safely and reliably. There has been no significant degradation of equipment, and the TSs will continue to ensure that there will be no significant degradation of equipment.

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

- 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 Part 73.

- 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.

- The licensee maintains an Emergency Plan in compliance with the existing applicable regulations which provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

On the basis of these findings, the staff concludes that the VA can continue to operate the AJBRF, in accordance with its application, without endangering the health and safety of the public.

1.3 History

On June 24, 1959, the U.S. Atomic Energy Commission issued to the VA a Construction Permit (CPRR-36). This permit authorized the VA to construct a General Atomics TRIGA-type research reactor at the Veterans Administration Hospital in Omaha, Nebraska. On June 26, 1959, Facility Operating License No. R-57 was issued by the Atomic Energy Commission authorizing VA to operate the TRIGA reactor at steady-state power levels up to 10 kW(t). The reactor first reached criticality on June 30, 1959. Amendment No. 2 to the license issued in September 1963 increased the steady-state thermal power level of the reactor to 18 kW(t) and Amendment No. 9 issued in April 1991 increased the power level to 20 kW(t). The license has been renewed twice prior to this renewal with the last renewal issued in August 1983 with a term of 10 years. During the 1983 renewal, the facility description, organization and safety evaluation were updated.

Since the 1983 license renewal, two license amendments have been issued. Amendment No. 9 issued in April 1991, increased the reactor licensed power level from 18 kW(t) to 20 kW(t) and approved the installation of a microprocessor-based neutron monitoring instrumentation system. Amendment No. 10, issued in May 2000, decreased the frequency of reactor fuel inspections based upon a long history of acceptable fuel performance. These changes to the TSs are reflected in this license renewal.

1.4 Reactor Description

The AJBRF is located at the Nebraska - Western Iowa Health Care System, Omaha Division, City of Omaha, Douglas County, Nebraska. The reactor is housed in the basement of the southwest wing of the hospital building (see Figure 1.1). The hospital building is located approximately 330 ft (100 meters) from the nearest dwelling in the unrestricted area.

The TRIGA reactor is an open tank-type heterogeneous, light-water-cooled reactor. The core is moderated by zirconium-hydride and water and reflected by water and graphite. It is located near the bottom of a steel tank in a cylindrical pit below ground level. The concrete-lined tank rests on a concrete slab. The reactor core currently consists of 57 uranium-zirconium-hydride ($U\text{-}ZrH_x$) fuel elements where 56 are aluminum-clad and one is stainless-steel clad. The elements are spaced in grid plates so that about 33 percent of the core volume is occupied by water. Shielding above the reactor core is provided by 16 ft (4.9 m) of water, and the core is cooled by natural convection of the water in the tank.

The reactor is designed and licensed to operate at steady-state thermal power levels up to 20 kW with a maximum available excess reactivity of 1.00$. It attained criticality with 54 fuel elements containing about 1.9 kg (4.2 lb) of uranium-235. The uranium in the fuel is enriched to less than 20 percent uranium-235.

1.5 Shared Facilities and Equipment and Special Location Features

The AJBRF is on the basement floor of the hospital building, where it is used primarily in research and radioisotope production related to the diagnosis and treatment of disease. Utilities such as municipal water and nonradioactive sewage, natural gas, and electricity are provided for joint use in the entire building. The AJBRF has a separate ventilation system exhausting through filters to the outside environment and a chiller system dedicated to removing heat from the reactor water. Research and preparation laboratories are part of the AJBRF, and the chemical hoods in these laboratory rooms are separately exhausted on the hospital building roof.

1.6 Comparison with Similar Facilities

The reactor core geometry is similar to that of most of the approximately 50 TRIGA reactors in operation throughout the world, about half of which are in the United States. However, the cladding for the majority of the AJBRF reactor fuel is aluminum, which was used only in the first few TRIGA reactors. The instruments and controls are similar to those on other research reactors licensed by the NRC.

1.7 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 for the disposal of high-level radioactive waste and spent nuclear fuel. The Department of Energy (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 non-power reactors have entered into contracts with the Department of Energy that provide that the Department of Energy retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing.

Because the Department of Veterans Affairs is an agency of the Federal government, it is in conformance with the Waste Policy Act of 1982.

Figure 1.1
Reactor Facility

2 SITE CHARACTERISTICS

2.1 Reactor Site

The hospital building is built on a knoll, at an elevation of 1215 ft (370 m) above mean sea level (MSL) in a commercial area within the city limits of Omaha, Douglas County, Nebraska. Omaha, which is on the eastern border between Nebraska and Iowa on the Missouri River, is at an elevation of 980 to 1280 ft (300 to 390 m) above MSL. Thus, the hospital is on some of the highest ground within the city. To the north is a large county hospital, to the south a commercial district, to the west a residential area, and to the east a golf course. The medical center grounds are sufficiently large so that the nearest offsite dwelling is more than 330 ft (100 m) away.

2.2 Demography

The metropolitan statistical area of Omaha includes suburbs in both Nebraska and Iowa with a 1990 population of 618,000, which has shown an 8 percent increase since 1980 (2000 census data is not yet available for the metropolitan area). Population within the city of Omaha itself has increased from about 336,000 in 1990 to 390,000 in 2000. The change in area population since the 1930s has shown a definite trend toward the northwest, west, and especially toward the southwest.

2.3 Nearby Industrial, Transportation, and Military Facilities

Omaha has no major heavy industry, but there are railroad yards 2 mi (3.2 km) to the southeast and 4.5 mi (7 km) to the east. There are railroad lines 2 mi (3.2 km) to the east and south. Offutt Air Force Base, is some 8 mi (13 km) to the southeast. The Omaha airport is more than 6 mi (10 km) from the facility. A low altitude airway [3000 to 17,000 ft (900 m to 5000 m) MSL] passes near the vicinity of the site. The nearest interstate highways (I-80 to the south or I-480 to the east) are more than 1 mi (1.6 km) away from the facility.

The staff concludes that because of the lack of significant industrial facilities and the distance of the highway, airport and rail lines from the facility, no significant risk is posed by industry, transportation, or military facilities to the continued safe operation of the facility.

2.4 Climatology and Meteorology

The climatology of the licensee’s site is described in the following sections. This includes information on precipitation, winds, and temperature. The sources of meteorological data to be used in case of an emergency is also discussed.

2.4.1 Climatology

Omaha is situated on the west bank of the Missouri River; the river level at Omaha is normally 965 ft (293 m) above MSL. The rolling hills in and around Omaha rise to 1300 ft (395 m) above MSL.

The climate is typical continental, with relatively warm summers and cold, dry winters. It is situated midway between two distinctive climatic zones—the humid east and the dry west. Fluctuations between these two zones produce periods of weather conditions that are characteristic of either zone or combinations of both. Omaha is also affected by most storms that cross the country. This causes frequent and rapid changes in weather, especially during the winter.

Most of the precipitation falls during sudden showers or thunderstorms from April to September. Of the total precipitation, about 75 percent falls during this 6-month period, predominantly as evening or night showers and thunderstorms. Although winters are relatively cold, precipitation is light, with only 10 percent of the total annual precipitation falling during the winter. Sunshine is fairly abundant, ranging from around 50 percent of the possible in the winter to 75 percent of the possible in the summer.

2.4.2 Temperature and Wind Variability

The prevailing winds at the Omaha airport are SSE during most of the year, shifting to the NNW during the winter quarter. The mean wind speed at the Omaha airport is about 10 mph (16 kph). The maximum wind speed recorded at the Omaha airport was 109 mph (175 kph).

Temperatures range from below 0°F (-18°C) in the winter to above 100°F (38°C) in the summer. The mean date of the last killing freeze in spring is April 14, and the mean date of the first killing freeze in autumn is October 20. The longest freeze-free period on record is 219 days in 1924, and the shortest period, 152 days in 1885. The average length of the freeze-free period is 188 days.

2.4.3 High Winds

High winds result from thunderstorms and intense low-pressure systems that traverse the region.
For the facility site, tornadoes of any wind intensity have an average probability of being observed of 1.3×10^{-3} per year. For the time period 1950 to 1996, 1759 tornadoes were observed in the state of Nebraska. Tornadoes have been recorded in the general area of the site. From 1950 to 1995, there have been 12 tornados in Douglas County. On May 6, 1975, a tornado that hit Omaha resulted in three deaths and up to $500 million in property damages.

The area around the facility is the hospital tornado shelter. The reactor is at the bottom of a pool placed in the floor of the basement of a multistory hospital building. The reactor is surrounded by poured concrete walls with no windows and with 2 to 4 in (7 to 11 cm) of concrete overhead. For these reasons, the staff concludes that tornado damage to the reactor itself is very unlikely.

2.4.4 Sources of Meteorological Data for Emergencies

Local meteorological measurements for use in evaluating accidental gaseous releases from the hospital building are not available; however, regional meteorological data can be obtained from the National Weather Service at the Omaha airport (Eppley Airfield). The meteorological data available from this source would enable the licensee to predict the dispersion in the unlikely event of an accident-related gaseous release to the environment.

2.4.5 Conclusions

The meteorological characteristics of the AJBRF site and vicinity are quite variable, in terms of both temperature extremes and wind direction and speed. While tornadoes are not uncommon in Nebraska, the staff concludes, on the basis of the above discussion, that the strike probability is acceptably low for any given location (such as the hospital building). The procedure established by the licensee for collecting meteorological information to be used during a facility emergency is acceptable to the staff. Therefore, the staff concludes that there are no unique meteorological conditions that could produce or cause a significant risk to the continued safe operation of the AJBRF.

2.5 Geology

The Omaha area lies within the Dissected Till Plains of the Central Lowland Physiographic Province of the United States. The topography is gently rolling, and the ground surface at the hospital building lies at an elevation about 1200 ft (370 m) above MSL. This elevation represents some of the highest ground within the city limits of Omaha, being about 275 ft. (80 m) above the level of the Missouri River.

The surface soils in the Omaha area are primarily loess and glacial drift deposits. Two stages of glaciation, the Nebraskan and the Kansan, left thick deposits of till overlying bedrock. It is believed that much of the glacial till has been eroded in the vicinity of the hospital building and that not more than 100 ft (30 m) remains. The till consists mainly of lean and gravelly clays with a few lenses of sand gravel. The exact depth to bedrock directly below the AJBRF site is not known but is estimated to vary between MSL elevation of 1000 and 1050 ft (300 and 320 m), on the basis of the nearest top bedrock information.

The loess at the site are of the Peorian and Loveland Formations of the late Pleistocene Epoch. The soil classification of the Peorian indicates that the material consists predominantly of clayey silts and lean clay. The soil of the Loveland formation varies from clayey silt to fat clay with minor amounts of sand and clayey sand in the basal part of the formation. At the hospital building site, the Peorian is from 30 to 45 ft (10 to 15 m) thick and the Loveland is more than 60 ft (20 m) thick. This would mean that the total thickness of the overburden is approximately 100 ft (30 m).

Bedrock in this area is limestone and shale of the Pennsylvania period. The surface of the bedrock is very irregular because of an extensive period of erosion that followed the uplift of the area in early Pennsylvania time and continued to the Pleistocene Epoch. This uplift brought the granite basement to within 600 ft (180 m) of the surface in certain areas, forming a ridge known as the Nemaha Ridge or Arch. A major structure, the Humboldt Fault, which has a throw of more than 900 ft (275 m), is associated with the Nemaha Arch. The Humboldt structure zone is assumed to continue through a point in the southeastern city limits of Omaha. The Humboldt Fault has not been considered to be a capable fault within the meaning of Appendix A, 10 CFR Part 100, based on investigations for the Cooper, Fort Calhoun, and Wolf Creek Nuclear sites.

2.6 Hydrology

Because no piezometers were installed or observation wells drilled at the site, there is no definite information as to the exact depth of the water table. However, on the basis of logs of borings drilled in 1946, the zone of saturation appears to be below 65 ft (20 m), although there is some indication of perched water levels in the soil strata as high as 15 ft (4.6 m). Furthermore, because the hospital building is sited on a knoll, there is reasonable assurance that neither surface nor groundwaters make the location unsuitable for the facility.

Groundwater will flow to the southwest from the hospital building site. The groundwater will travel downward through relatively impermeable loess until it reaches impermeable glacial till. Groundwater will travel to the Big Papillion Creek which runs in a southeasterly direction approximately 2 mi (3 km) west and 4 mi (6 km) south-west of the site. The groundwater will follow the creek to the Missouri River.

The nearest wells are located about 1.6 mi (2.5 km) west on the Aksarben grounds and 4.2 mi (7 km) southwest (84^{th} and L streets) of the site. The Aksarben well is not in the direction of groundwater flow and the other well is beyond the Big Papillion Creek groundwater flow path. The licensee calculates that it would take at least 100 years for water to travel from the hospital building site to the Big Papillion Creek. This would allow a substantial time for any radionuclides that did not bind to the soil to decay before reaching the public water supply.

Because of the location of the facility site on the highest ground elevation within Omaha, the staff concludes that surface water features will not impact the site.

2.7 Seismology

The AJBRF site is located in Seismic Risk Zone 1 of the United States (Algermissen, 1969), which is defined as “Minor damage, distant earthquakes may cause damage to structures with fundamental periods greater than 1.0 seconds, corresponding to intensities V and VI of the Modified Mercalli scale.” Intensity VI on the Modified Mercalli intensity scale is described as “Felt by all; many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight.”

The largest earthquakes within 200 mi (320 km) of the AJBRF site have had maximum Modified Mercalli intensities of VII. The closest Modified Mercalli intensity VII earthquake was about 60 mi (100 km) from the site and it is estimated that the intensity at the site from this event was about Modified Mercalli intensity V. There have been no reports or physical evidence of earthquakes at the site and no indication of significant building damage due to earthquakes in the area of the site.

This tectonically stable region is characterized by relatively low intensity as well as relatively low frequency of earthquakes. Therefore, the staff concludes that the history of no significant earthquake damage in the site region supports the conclusion that seismic-induced risk to the AJBRF is not significant. Furthermore, if the hospital building were damaged, the radioactive fuel would be safely contained within the pool below ground level. Seismic induced damage to the facility is discussed in Section 3.3 of this SER and accidents that could be caused by a seismic event, such as loss of coolant are discussed in Section 13.

2.8 Conclusions

On the basis of the above considerations regarding both natural and man-made hazards, the staff concludes that there is no significant risk associated with the site that would make it unacceptable for continued operation of the facility.

Figure 2.1
AJBRF Vicinity

3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 Facility Description

The facility is located in Room B526 of the basement of the Nebraska - Western Iowa Health Care System, Omaha Division Medical Center. The room is considered a restricted area with locked doors and entrance controlled by reactor laboratory personnel [required by TS 5.1(1)]. The minimum free volume in the reactor area is 25,000 ft^3 (708 m^3) [required by TS 5.1(4)]. The main hospital building is 11 stories high and was constructed in 1951. The building is constructed of brick and reinforced concrete construction, including the ceilings and floors. Walls surrounding the facility are of brick, cinder block and reinforced concrete construction. Normal access to the facility is through door SW2 (see Figure 1.2). There is also an access door to a stairwell at the back of the facility.

The reactor is located near the bottom of a cylindrical pool 20 ft (6.1 m) below the floor of the facility room. The only access to the reactor pool is from the top. The reactor control console is located near the reactor pool in the same room. The facility also contains several rooms for sample preparation and a room for isotope storage. Three additional rooms, containing a laboratory (SW2B), walk-in refrigerator (SW2D), and an electron microscope (SW2A) are within the site boundary because entrance to the facility is required to reach these rooms.

The facility ventilation system is designed such that during operation there is a slight negative pressure in the facility as compared with outside pressure. However, the room is not designed to be a confinement. The ventilation system is discussed in Chapter 6 of this SER.

3.2 Wind Damage

Meteorological data indicate a low frequency of tornadoes and effects of high winds. The facility is in the basement of the 11-story hospital building, surrounded by poured concrete walls with no windows and with 3 to 4 in (7 to 10 cm) of concrete overhead. Therefore, the NRC staff concludes that the design to mitigate significant wind damage to the facility is acceptable.

3.3 Water Damage

The hospital building is situated on a knoll about 1215 ft (370 m) above sea level, which is much higher than most of the ground within the Omaha city limits and approximately 275 ft. (80 m) above the level of the Missouri River. Therefore, the staff concludes that there is reasonable assurance that potential damage to the reactor by flood or groundwater is small.

3.4 Seismic-Induced Reactor Damage

Analyses of newspaper accounts since 1867 indicate that the site is in a tectonically stable region characterized by low level as well as low frequency of earthquakes. The site is located in Seismic Risk Zone 1 of the United States. There is a risk of slight damage, principally to poorly built or designed structures. Because of the location of the facility in a low seismic risk zone and the features of the hospital building described in Section 3.1, the staff concludes that the risk of seismic damage to the facility is small.

3.5 Mechanical Systems and Components

The mechanical systems of importance to safety are the neutron–absorbing control rods suspended from the reactor superstructure. The control rods are attached to their drive mechanisms by electromagnets. When power to the electromagnets is interrupted, the control rods fall into the reactor core by gravity. The motors, gear boxes, electromagnets, switches, and wiring are above the level of the water in the reactor pool and are readily accessible for testing and maintenance. A preventive maintenance program has been in operation for many years at the facility to conform and comply with the performance requirements of the TSs.

The effectiveness of this preventive maintenance program is attested to by the small number and types of malfunctions of equipment over the years of operation. These malfunctions have generally been one of a kind (that is, no repeats) and/or of components that were fail safe or self annunciating (see Inspection Reports and reports of Reportable Occurrences from the licensee, Docket No. 50–131). Therefore, the staff concludes that there appears to be no significant uncompensated deterioration of equipment with time or with operation. Thus, there is reasonable assurance that continued operation for the requested period of renewal will not increase the risk to the public.

3.6 Conclusions

On the basis of the above considerations, the staff concludes that the AJBRF was designed and built to withstand all credible and probable wind, water and seismic damage contingencies associated with the site. The design and performance of the safety systems have been verified through more than 40 years of operation. Accordingly, the staff concludes that the reactor systems and components are adequate to provide reasonable assurance that continued operation will not cause significant radiological risk to the health and safety of the public.

4.0 REACTOR

The AJBRF TRIGA reactor (Figure 4.1) is a research reactor designed, fabricated, and installed by General Atomics. The reactor first achieved criticality on June 30, 1959. It is a below-grade, open-tank type, light-water moderated, cooled, and shielded reactor that currently is authorized to operate in the steady-state mode at thermal power levels up to 20 kW with an excess reactivity limitation of 1.00$. Beta-effective for a TRIGA reactor core very similar to this one has been shown to be 0.79% $\Delta k/k$. Unlike most TRIGA reactors, this reactor has not been licensed to pulse.

The reactor is used as a source of ionizing and neutron radiation for research in biology and medicine, including nuclear medicine, clinical chemistry, radiobiology, and biomedical applications of neutron activation techniques.

4.1 Reactor Core

The reactor core (Figure 4.2) forms a right circular cylinder consisting of a compact array of 56 aluminum-clad and one stainless steel-clad uranium-zirconium hydride (U-ZrH_x) fuel moderator elements, graphite dummy elements, three boron carbide control rods, control rod guides, a startup neutron source, and irradiation facilities. The fuel elements are spaced so that about 33 percent of the core volume is occupied by water, yielding a fuel-to-hydrogen ratio resulting in a critical mass near the minimum value for 20 percent enriched uranium fuel. The elements are held with their long axes vertical in concentric rings by an upper and a lower grid plate. The reactor currently requires 57 fuel elements to achieve criticality and to provide the authorized excess reactivity (1.00$) necessary to meet operating requirements. The balance of the 91 fuel element positions in the grid plate is occupied by experimental facilities or neutron-reflector elements, in which the U-ZrH_X fuel is replaced by graphite (required by TS 5.3.1). To ensure that the reactor will be subcritical during fuel movement, TS 3.1.3(1) requires that fuel elements are not to be inserted or removed from the core unless the reactor is subcritical by more than the worth of the most reactive fuel element.

To assure that there have been no changes in a core configuration, TS 4.1.3 requires a visual observation of the reactor core be made before each initial daily startup of the reactor. The observation is recorded in the daily startup checklist. The observation will assure that the fuel elements, control rods, detectors and experimental facilities are in place as specified in the safety analysis report and that the core is free of any extraneous material.

The core is immersed in purified water and surrounded by a cylindrical graphite reflector that is completely encased in a welded aluminum container. The flooding of the reflector container in the event of a leak would decrease reactivity, because of the additional neutron absorption in hydrogen.

4.1.1 Fuel Elements

The reactor currently uses 56 TRIGA (Figure 4.3) aluminum-clad cylindrical fuel elements and one stainless steel-clad element.

The fuel elements are visually examined remotely for physical damage at least once every five years with at least 20 percent of the fuel elements examined each year. The examination includes inspection for swelling, cracks, corrosion and pitting to detect cladding deterioration from erosion, corrosion or other damage. If an annual examination identifies damaged fuel, the entire core is inspected. Fuel elements are removed from the core if a clad defect exists as indicated by the release of fission products or visual observation. Any indication of the release of fission products by the facility monitoring instruments will be considered a clad defect and damaged fuel will be assumed to be in the core. The reactor will not be operated with damaged fuel except to detect and identify damaged fuel for removal (required by TSs 3.1.5 and 4.1.4). To ensure the reactor is maintained in a subcritical condition, all control rods are fully inserted into the core during the fuel element inspection [required by TS 3.1.3(2)]. For those rare instances where fuel element cladding fails, it is normally detected when fission products are released into the primary coolant and the facility air during reactor operation. Fuel elements with damaged cladding do not normally release fission products when the reactor is shut down. The damaged fuel is located by rotating elements out of and into the core and operating the reactor. This rotation will narrow the number of possible elements with damage until the damaged element is found. Damaged elements can also be found by sampling water near each element during operation to detect fission products. Fuel cladding failure or a release of fission products during operation of the AJBRF has never occurred. Regular examination of the AJBRF fuel elements has shown no indication of any surface deterioration, swelling, or bending. Because of the low power level of the reactor, requirements to monitor primary water chemistry and radioactivity in the primary water, and the excellent operating history of the facility, the fuel inspection surveillance requirement may be postponed if the reactor is shut down.

The fuel in aluminum-clad fuel elements is 1.41 in (3.58 cm) in diameter by 14 in (35.6 cm) long and is a solid homogenous mixture of an U-ZrH_X alloy containing 8 weight-percent uranium enriched to less than 20 percent in uranium-235 [the TS 5.3.2(1) limit which applies to both aluminum and stainless steel-clad fuel is a maximum of 9 weight-percent uranium enriched to less than 20 percent uranium-235]. The nominal initial weight of uranium-235 in each fuel element is 38 g. The hydrogen-to-zirconium ratio in the aluminum clad elements is approximately 1.0 [required by TS 5.3.2(2)]. A thin wafer at each end of the active fuel contains samarium oxide as a burnable poison. TS 5.3.2(4) which applied to both aluminum and stainless steel-clad fuel elements states that any burnable poison shall be any integral part of the as-manufactured fuel element. A burnable poison is defined as a material fixed in place in the core for the specific purpose of compensating for fuel burnup and/or other long-term reactivity adjustments. The fuel is jacketed with a watertight 0.030 in (0.076 cm) thick aluminum tube [required by TS 5.3.2(3)]. Four-inch (10.2-cm) sections of graphite are inserted in the tube above and below the fuel to serve as top and bottom neutron reflectors for the core. Aluminum end fixtures are attached to both ends of the tube. The overall length of each fuel element is 28.8 in (0.72 m).

The stainless steel-clad fuel elements are also a homogenous mixture of U-ZrH_X alloy containing approximately 8.5 weight-percent uranium enriched to less than 20 percent in uranium-235. The hydrogen-to-zirconium ratio is approximately 1.65 [TS 5.3.2(2) limit is a nominal 1.7]. The fuel in each fuel element is 1.43 in (3.63 cm) in diameter by 15 in (38.1 cm) long. Aluminum-samarium wafers are located at each end of the active fuel as a burnable poison. Later versions of the fuel may contain erbium as the burnable poison. The fuel is jacketed with a 0.20 in (0.05 cm) [required by TS 5.3.2(3)] thick stainless steel watertight tube. Graphite reflector plugs 3.45 in (8.8 cm) long are located above and below the fuel and serve as neutron reflectors. Stainless steel end fixtures are attached to both ends of the tube. The overall length of the fuel element is the same as that of the aluminum-clad fuel element.

Fuel elements are removed from operation if the burnup of uranium-235 in the fuel matrix exceeds 50 percent of the initial concentration [required by TS 3.1.5(2)]. To date, no fuel elements at the AJBRF have reached this limit.

The staff has reviewed the use of cores containing fuel elements with both types of cladding and with mixtures of cladding types. The staff finds that mixing aluminum and stainless steel-clad elements will not result in significant change in the reactor performance and concludes that there is reasonable assurance that the reactor is capable of safe operation, as limited by its TSs, with a core containing either or both types of fuel element cladding. Core conditions are controlled by the most limiting fuel type which is aluminum-clad fuel. The staff notes that there is extensive operating experience with both types of fuel elements under conditions (power level and pulsing) that are more severe than those experienced under the operating conditions authorized for the reactor. Furthermore, the fuel has been located in the highly purified pool water, so cladding degradation by corrosion is expected to be negligible.

4.1.2 Reflector

The core lattice is surrounded by a ring of graphite 12 in (30.5 cm) thick and 22 in (56 cm) high with an inside diameter of 17 in (43 cm) and an outside diameter of 42 in (107 cm). The graphite reflector assembly is encased in a welded aluminum can to prevent the penetration of water. The reflector assembly rests on the reflector platform and provides the support for the two grid plates.   In addition, core top and bottom axial reflection is provided by the graphite plugs incorporated into both ends of the individual fuel elements. Graphite-reflector elements, which are fuel element cans filled with graphite instead of fuel, are placed in core lattice positions not occupied by other core components.

4.1.3 Control Rods

The power levels in the TRIGA reactor are regulated by three (safety, shim and regulating) aluminum-clad boron-carbide (neutron absorbing material) control rods which have scram capability. TS 5.3.3 allows control rods to contain borated graphite, B_4C powder, or boron and its compounds as a poison contained in a suitable cladding material such as aluminum or stainless steel to ensure mechanical stability during movement and to isolate the poison from the pool water environment.   The control rods operate in perforated aluminum guide tubes. The guide tubes are supported by the bottom grid plate, and the upper grid plate provides lateral support. Each control rod has an extension tube that connects to a drive mechanism through an armature and electromagnet system.

One control rod, designated as the safety rod, is routinely withdrawn to its limit from the core during normal operation. A second control rod, the shim rod, is used for coarse reactivity changes. The safety and shim rods each have a reactivity worth of approximately 2.25$. The third control rod is used as a regulating rod for fine control of reactor power and has a worth of approximately 0.85$. The regulating rod is used in conjunction with a servo-amplifier system to provide automatic control of the reactor at steady power and to bring the reactor up to power on preset periods of 30 or 60 seconds.

The control rods are visually inspected for deterioration on average annually. During this inspection, the rod drive and scram mechanisms are also inspected (required by TS 4.2.6). For control rod removal, the reactor must be more than five elements short of critical with all rods fully withdrawn [required by TS 3.1.3(3)]. Periodic visual examination of control rods performed by the licensee showed pitting, which required replacement of four control rods (one in 1964, two in 1966 and one in 1973). General Atomics concluded on the basis of analysis of a rod replaced in 1966 that the pits were probably a result of iron particles embedded in the aluminum cladding of the rod during manufacture. The manufacturing and inspection process was modified to decrease the likelihood of such inclusions. Periodic inspection since the last replacement of a control rod (November 1973) has revealed no evidence of pitting.

4.1.4 Neutron Source

The reactor utilizes a doubly encapsulated americium-beryllium neutron source to ensure that there are neutrons and observable indication for safe reactor startup. The source is of a standard acceptable design, typical of those used in other licensed non-power reactors. The shape of the source holder is similar to a TRIGA fuel element, so it could be placed in any fuel element location.

4.1.5 Conclusions

The staff has reviewed the information regarding the reactor fuel, core arrangement, reflector, control rods, and the neutron source and found that the design performance capability and performance history of the components are adequate to provide reasonable assurance of continuing operability to provide safe operation and shutdown of the reactor during the proposed license renewal period.

4.2 Reactor Tank

The reactor core is near the bottom of a below-grade cylindrical pit (Figure 4.1) 20 ft (6.1 m) below ground level located in the basement of the 11-story hospital building. The pit contains a 6 ft 10 in (2.1 m) inside diameter steel tank with a ¼ in (0.64 cm) thick wall. The tank rests on a 1 ft (0.28 m) thick concrete slab. A 10 in (0.25 m) thick poured concrete wall surrounds the outside of the tank. The concrete slab and wall provide a protective barrier between the tank and the surrounding soil. The inside of the tank is covered on the sides by a layer of “gunite” 2 in (5 cm) thick and on the bottom by a layer of poured concrete 4 in (10 cm) thick. The entire inner surface is coated with two applications of a waterproof epoxy resin coating. Since 1959, visual inspection of the tank by the licensee shows no evidence of deterioration of the tank, and there has been no indication of unexplained loss of water. The steel tank and external concrete will inhibit flow of any pool water that might leak through the epoxy and gunite liner.

The reactor tank contains approximately 4,000 gal (15,000 l) of water with a normal shielding depth of 16 ft (4.9 m) above the top grid plate. TS 3.3(4) requires a minimum of 15 ft (4.6 m) of water covering the core. TS 3.1.4 requires a float alarm switch to be operable to provide a visual and audible alarm at the hospital switchboard (which has a person present continuously) and a visual alarm on the reactor console if the pool water level falls to less than 12 ft (3.6 m) above the top of the core. TS 4.3.3 requires that this float switch be channel tested monthly to ensure that the system is operable.

The natural thermal convection of this water can disperse the heat generated in the core by the normal operation of the reactor. The pool water is pumped through a chiller unit (refrigerator) that ultimately disposes of the heat to the outside atmosphere. The pool water inlet pipe in the cooling system is 13 ft (4 m) above the top of the core, limiting the amount of coolant that would be lost in the event of a coolant piping rupture.

If the external cooling system were to fail with the reactor operating at 20 kW, the rate of rise of the temperature of the water in the reactor tank will be less than 2° F (1° C) per hour providing adequate time for corrective action to be taken. In the event of the loss of all coolant, the natural convection of air through the core will maintain its temperature below the cladding failure level, and all the fission products generated during reactor operation will be retained within the individual elements (see Chapter 13).

Three storage pits are located in the reactor room floor adjacent to the reactor tank. The pits are vertical 10 in (0.25 m) diameter steel pipes 10 ft (3 m) long and are lined with an organic coating. The pits may be filled with water and used for temporary storage of irradiated specimens or irradiated fuel elements. The storage pits have the capacity to safely store the entire reactor core. The maximum number of fuel elements that could be forced into a single layer in a storage pit is about 37, yielding a flooded reactivity, $k_{eff,}$ of less than 0.50. The licensee has performed calculations that show that the k_{eff} for the normal storage of up to 25 fuel elements per pit is less than 0.8 which meets the requirement in TS 5.3.4 of k_{eff} no greater than 0.9 for all cases of moderation and reflection using light water.

4.3 Support Structures

A fixed bridge spans the reactor tank at floor level in the reactor room (Figure 4.1). The control rod drives, specimen-removal drive mechanism, rotary specimen-rack drive mechanism, central irradiation thimble, pneumatic transfer tube system, and control sensors are located on and may be suspended from the bridge.

The reactor core components are supported by top and bottom grid plates. The bottom grid plate supports the weight of the fuel elements, the pneumatic transfer tube, the central thimble, and the control rod guide tubes (Figure 4.2). The top grid plate provides lateral position control only. The bottom grid plate is attached to the underside of the neutron reflector container. The reflector is mounted on structural supports that rest on and are attached to the reactor tank bottom. A recess is provided within the reflector for the rotary specimen rack.

4.4 Shielding

Because the reactor tank is entirely imbedded in earth and concrete below floor level, the only area of personnel access is from the top, in the reactor room. The usual 16 ft (4.9 m) of water above the core provides more than adequate attenuation of the neutrons and gamma rays between the core and the working area at the top of the pool. Evaluation of neutron leakage into soil surrounding the outer tank wall indicates that activation and transport of likely materials in the soil would lead to negligible exposures in the unrestricted area, and are acceptable.

4.5 Reactor Instrumentation

The operating condition of the reactor is monitored by two neutron detector channels. One consists of a boron lined uncompensated ion chamber whose analog signals indicate reactor power level, and can initiate a reactor scram if the level exceeds the setpoint. The other consists of a low noise fission chamber whose digitized signals indicate both linear and logarithmic values of reactor power level, show reactor period, and can initiate a reactor scram on power level. This system, called the NM-1000, was authorized by license amendment in 1991. The nuclear control and process control instrumentation are discussed in Chapter 7.

4.6 Dynamic Design Evaluation

The safe operation of a TRIGA reactor during normal operations is accomplished by the control rods and is monitored accurately by the core power level (neutron) detectors. A backup safety feature of a TRIGA reactor is the reactor core's inherent large, prompt, negative temperature coefficient of reactivity, resulting from an intrinsic molecular characteristic of the U-ZrH_X matrix at elevated temperatures. The negative temperature coefficient results principally from the neutron hardening properties of the fuel matrix at elevated temperatures, which increases the leakage of neutrons from the fuel-bearing material into the water moderator material, where they are absorbed preferentially. This reactivity decrease is a prompt effect because the fuel and ZrH_x are mixed homogeneously, thus the ZrH_x temperature rises essentially simultaneously with fuel temperature which is directly related to reactor power. An additional contribution to the prompt, negative temperature coefficient is the Doppler broadening of uranium-238 resonances at high temperatures, which increases nonproductive neutron capture in these resonances.

Because of the large, prompt, negative temperature coefficient, a step insertion of excess reactivity resulting in an increasing fuel temperature will be compensated for by the fuel matrix rapidly and automatically. This can terminate the resulting power excursion without any dependence on the electronic or mechanical reactor safety systems or the actions of the reactor operator. Also, changes of reactivity resulting in a change in fuel temperature during steady-state operation can be rapidly compensated for by the fuel matrix, thus limiting the reactor steady-state power level (GA-E-117-833, 1980; Simnad et. al., 1976). Similarly, this inherent characteristic of the U-ZrH_X fuel has been the basis for designing TRIGA reactors with a pulsing capability as a normal licensed mode of operation. However, the AJBRF is not authorized for or provided with the transient rod and instrument systems to implement pulse mode operation. TRIGA reactors have been routinely pulsed with fuel similar to the AJBRF with reactivity insertions of more than 2.00$. Potential accidents are discussed in Chapter 13.

4.6.1 Excess Reactivity, Experiment Worth, and Shutdown Margin

The maximum power excursion transient that could occur would be one resulting from the inadvertent rapid insertion of the total available excess reactivity. TS 3.1.1 limits excess reactivity to 1.00$ under xenon free, cold (20 °C or 68 °F) critical conditions. TS 4.1.1 requires that the excess reactivity be determined on average annually, and after changes in the core, in-core experiments, or control rods for which the predicted change in reactivity exceeds the absolute value of the required shutdown margin. A reactivity transient accident based on this limitation is analyzed in Chapter 13, showing that no fuel failure or other reactor damage would result, so the primary fission product barrier (fuel cladding) would remain intact.

TS 3.7.1(1) limits the reactivity worth of individual secured experiments to 1.00$ and moveable experiments to 0.85$. A secured experiment is defined in the TSs as any experiment that is held in a stationary position relative to the reactor by mechanical means. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces than can arise as a result of credible malfunctions. In addition, TS 3.7.1(3) requires the reactor to be shut down during the changing or moving of any secured experiment. A moveable experiment is defined in the TSs as an experiment where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating. To help ensure that the reactivity of higher worth experiments is known, TS 3.7.1(4) requires the actual experiment worth to be measured and recorded at the time of initial insertion of any experiment whose estimated worth is greater than 0.40$. The sum of the absolute worths of all experiments in the reactor and in the associated experimental facilities is limited to 1.00$. Therefore, reactivity changes resulting from experiment malfunctions are enveloped by the analyses of 1.00$ excess reactivity addition.

The shutdown margin, as defined and required by TS 3.1.2, is greater than 0.51$ with (1) all experiments in their most reactive state; (2) the reactor in the reference core condition (cold, critical, xenon-free condition), and (3) the highest worth control rod fully withdrawn. Under these conditions, the negative reactivity would be at least 1.00$, fully satisfying the TS requirement. TS 4.1.2 requires that the shutdown margin be determined on average annually, and after changes in the core, in-core experiments, or control rods.

The worth of the control rods must be accurately known to determine compliance with the reactivity limits for the reactor. TS 4.2.1 requires the integral worth of all control rods to be determined on an average annually and after changes of the core or control rods.

4.6.2 Other Reactor Physics Parameters

The staff has reviewed the coefficients of reactor moderator void and temperature to verify that they are negative. TRIGA reactors are designed so that the core provides a slightly under-moderated neutron spectrum at ambient temperature and with all spaces in the core filled with either fuel, control rods, or neutron reflector elements. TS 5.3.1 requires the reactor core to consist of a compact array with all core positions filled by fuel, control rods, experimental facilities or graphite-reflector elements. The core of the AJBRF TRIGA is designed so that displacing moderator liquid or increasing fuel temperature causes a loss of reactivity, promoting stability of reactor operation.

4.6.3 Normal Operating Conditions

The authorized maximum thermal steady-state power level of the reactor is 20 kW. TS 2.2 imposes a limiting safety system setting of 20 kW as measured by the calibrated power channels to prevent the maximum fuel temperature from reaching the safety limit given in TS 2.1 of 500 °C (932 °F).

Calculations performed by General Atomics and confirmed by experiments indicate that no cladding damage occurs at peak fuel temperatures as high as approximately 530 °C (986 °F) for low-hydride-type (U-$ZrH_{1.0}$), aluminum-clad elements (Simnad 1980), and 1175 °C (2150 °F) for high-hydride-type (U-$ZrH_{1.65}$), stainless-steel-clad elements (Coffer et al., 1966; Simnad, 1980; Simnad et al., 1976). The licensee has proposed limits based on the more limiting aluminum clad fuel. Cladding damage in the high-hydride-type, stainless-steel fuel is caused by a pressure buildup in the element as a result of the evolution of hydrogen produced by dehydriding of the fuel with increasing temperature. The pressure internal to the fuel element reaches the point where the cladding fails. Cladding damage in the low-hydride-type, aluminum-clad fuel is caused by a phase change in the fuel matrix that occurs at about 530 °C (986°F). The phase change causes the fuel to swell which causes the cladding to fail. The 500 °C (932 °F) safety limit for the reactor is determined by the cladding damage threshold temperature of the low-hydride-type, aluminum-clad fuel element.

The limiting safety system setting of 20 kW ensures that a considerable margin of safety exists. TRIGA reactors using aluminum-clad fuel elements with a hydrogen-to-zirconium ratio of 1.0 have demonstrated safe and reliable routine operations at power levels up to 250 kW (Simnad et al., 1976). At 250 kW, the maximum temperature rise of fuel elements in the core B-ring was about 180 °C (356°F). Fuel in the B-ring is normally the hottest in the core. The data indicates that the temperature of the hottest fuel in the reactor core will remain significantly below the 530 °C (986 °F) limiting temperature established for aluminum cladding failure in low-hydride (U-$ZrH_{1.0}$) TRIGA fuel elements (Simnad, 1980). NRC has licensed TRIGA reactors with stainless-steel clad high-hydride fuel elements (U-$ZrH_{1.6 \text{ to } 1.7}$) at power levels over 2000 kW, a factor of 100 greater than the AJBRF reactor limiting safety system setting.

4.6.4 Conclusions

The staff concludes that the inherent large, prompt, negative temperature coefficient of the U-ZrH_X fuel-moderator, and the negative core temperature and moderator void coefficients, provide a basis for safe operation of the AJBRF reactor in the steady-state mode. Furthermore, on the basis of the above information, the staff concludes that: (1) the limitation on total excess reactivity of 1.00$; (2) a limitation on total absolute experiment reactivity worth of 1.00$; (3) a limitation of 1.00$ on individual secured and 0.85$ on movable experiments; and (4) operation in compliance with TS minimum shutdown margin requirements provides assurance that operation of the AJBRF reactor to support the experimental program will pose no threat to the heath and safety of the public. In addition, the staff concludes that the negative shutdown margin of 0.51$ under defined conditions is sufficient to ensure that the reactor can be safely shut down under all credible operational conditions.

The safety limit and limiting safety system setting for the reactor are based on theoretical and experimental investigations and are consistent with those approved by NRC and used at other TRIGA-type reactors. Operating data at the maximum authorized reactor power level of other TRIGA reactors provide confidence that the maximum fuel element temperatures will be maintained far below the prescribed safety limits.

On the basis of the above considerations, the staff concludes that there is reasonable assurance that the AJBRF reactor can be operated safely at 20 kW, as limited by TS requirements.

4.7 Functional Design of the Reactivity Control System

The power level in the reactor is regulated by the use of three standard control rods, which contain boron as the neutron-absorbing material. The control rods are driven vertically in or out of the core by electro-mechanical drive mechanisms. Each control rod drive system is energized from the control console through independent electrical cables and circuits, which tends to limit the probability of simultaneous malfunctions of the drives. All of the three control rods are designed to be released through the safety circuitry to fall by gravity into the core on the receipt of a scram signal or interruption of electrical power. An electrical interlock prevents raising more than one control rod at a time (required by TS 3.2.4).

4.7.1 Control Rod Drives

The control rod drive mechanisms (Figure 4.4) are located on the bridge at the top of the reactor pool structure (at floor level of the reactor room) and consist of a motor and reduction gear that drive a rack-and-pinion system. Potentiometers provide rod position information at the control console for the shim, safety and regulating rods. The control rod extension tube and dash pot are connected through an electromagnet and armature. In the event of electrical power failure or a scram signal, all of the electromagnets are deenergized and the control rods are released to fall into the core by gravity. The drive motors are non-synchronous, single phase, and reversible.

Electrical dynamic and static braking on the drive motors are used for fast stops. Switches on the drive assembly limit the drive motions and indicate at the console the up and down positions of the magnet, the down position of the rod, and armature-magnet contact. The control rod drive mechanisms have a stroke of approximately 15 in (0.38 m). The maximum rod withdrawal rate is 12 in (30 cm) per minute, with a maximum reactivity insertion rate of about 0.05$ per second. However, TS 3.2.2(1) limits the maximum reactivity insertion rate of standard control rods to less than 0.10$ per second.

4.7.2 Scram-logic Circuitry

The scram (protective action) circuitry ensures that essential reactor core and operational conditions are satisfied for reactor operation to occur or continue. The minimum conditions that must be satisfied are specified in the TSs. The scram logic circuitry is a fail-safe system such that any scram signal to or component failure in the scram logic system will result in the loss of control rod magnet power, releasing the control rods and causing a reactor shutdown. Input signals to the scram circuitry are supplied from several process variables and power level sensors that operate independently of each other to ensure redundancy. (The details of individual sensors are presented in Chapter 7.)

The following scrams are required by TS 3.2.3:

- Linear power greater than 100 percent of licensed power.
- Percent power greater than 100 percent of licensed power.
- Loss of high voltage to the ion chamber power supply.
- Loss of high voltage to the fission counter power supply.
- Console scram button.
- Magnet current key switch.
- Watchdog timer.

The time between activation of the scram logic system and the full insertion of each control rod is limited to less than two seconds by TS 3.2.1 to ensure adequate safety for the reactor and fuel elements for the range of anticipated operations at the facility. TS 4.2.3 requires the scram times of all control rods to be measured on an average annually or whenever work is done on the rods or rod drive systems.

4.7.3 Conclusions

The reactor is equipped with safety and control systems typical of most non-power reactors. The system contains an acceptable number of control rods and sufficient independent, redundant scrams. The staff concludes that there is sufficient redundancy of control rods so that the reactor can be brought to a safe shutdown even if the most reactive control rod fails to insert upon receiving a scram signal. More than one nuclear instrumentation channel monitors the neutron density (power level), providing redundancy to mitigate consequences of single malfunctions.

In addition to the active electro-mechanical safety system for normal and abnormal operation, the large, prompt, negative temperature coefficient of reactivity inherent in the $U\text{-}ZrH_x$ fuel moderator discussed in Section 4.6 provides a backup safety feature. The physical behavior of this fuel limits the steady-state power level and terminates inadvertent transients that produce large increases in temperature.

Based on the above discussion, the staff concludes that the reactivity control systems of the AJBRF reactor are designed and can function adequately to ensure safe operation and safe shutdown of the reactor under all credible normal and off-normal operating conditions.

4.8 Operational Practices

The licensee has implemented a thorough preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is not operated at power without the appropriate safety-related components operable. The reactor is operated by NRC-licensed personnel in accordance with explicit operating procedures, which include specific responses to any reactor control signal. All proposed experiments involving use of the reactor are reviewed by the Reactor Safeguards Committee before the installation of the experiment in the reactor or its experimental facilities for potential effects on the reactivity of the core or damage to it as well as for possible radiological effects on the health and safety of employees and the general public.

4.9 Conclusions

On the basis of the information presented above, the staff concludes that the reactor is designed and built according to standard industrial practices. It consists of components representing hundreds of reactor years of operation and includes redundant safety-related systems.

The staff review of the facility has included studying its specific design and installation, its controls and safety instrumentation, its specific preoperational and operating procedures, and its operational limitations as identified in the current and proposed TSs and all other pertinent documents associated with the license renewal. The design features of the reactor are similar to those typical of the research reactors of the TRIGA type operating in many countries of the world, 19 of which are licensed by NRC. On the basis of this review of the AJBRF reactor and experience with these other facilities, the staff concludes that there is reasonable assurance that the AJBRF reactor is capable of safe operation, as limited by the TSs, for the period of the requested license renewal.

Figure 4.1
Reactor and Pit

Figure 4.2
Core and Reflector Assembly

Figure 4.3
TRIGA Fuel Element

Figure 4.4
Control Rod Drive Mechanism

5.0 REACTOR COOLING SYSTEM

The reactor cooling system is shown schematically in Figure 5.1. The reactor core is located in a tank of demineralized water and cooled by natural thermal convection. The reactor heat is removed from the cooling water by a five-ton refrigeration unit and discharged to the atmosphere by an air-cooled condenser outside the building. The system contains about 4000 gal (15,000 l) of coolant. The system can be inspected to detect deterioration and components can be readily replaced if needed.

Cooling water flows from the reactor tank to a water monitor chamber where temperature, conductivity, and gamma radioactivity are monitored. The circulating pump takes water from the monitor chamber and discharges it through a refrigerated heat exchanger, filter, demineralizer, flow meter, and back into the reactor tank. There is piping bypassing the filter and demineralizer so that coolant can circulate when one or both of these units are being serviced. There is a skimmer at the pool surface. When the skimmer is operating, part of the coolant loop flow is through the skimmer, cleaning the pool surface of debris or contamination. TS 3.3 requires that the conductivity of the pool water not exceed 5μmhos/cm when averaged over one month and the pH of the coolant be maintained in the range of 5.0 to 7.5. TS 4.3.2 requires that the conductivity be measured at least weekly and the pH at least monthly. These limits have been shown to limit corrosion in aluminum and in stainless-steel systems.

Reactor coolant lost by evaporation from the pool surface is replaced by manually pouring demineralized water directly into the reactor pool. This consists of about 15 gal (60 l) per year. There is no direct piping connection between potable water supplies and reactor primary coolant and thus no way to introduce primary water into the potable water supply.

The cooling water loop inlet takes suction from the reactor pool at a point about 13 ft (4 m) above the top of the reactor core. A piping rupture would leave the reactor core adequately cooled and shielded. The maximum amount of coolant loss would be about 130 gal (500 l). This water would be contained within the reactor room or concrete enclosure housing the cooling system components. The radioactive material concentration in the coolant is normally well below the limits contained in 10 CFR Part 20 for release to the environment. TS 3.3(3) requires that the radioactivity in the coolant not exceed 0.1 μCi/ml. Approaching this limit would indicate a potential failure of fuel cladding and would be investigated by the licensee. TS 4.3.1 requires that the reactor coolant be sampled for gross activity monthly and for isotope identification quarterly. This frequency is sufficient to detect cladding failure and to establish long term trends.

The instrumentation and controls associated with the reactor cooling system are described in Chapter 7.

The staff concludes that the reactor cooling system is adequate to remove sufficient heat from the fuel to prevent overheating under all normal and abnormal operating conditions. On the basis of the staff's review of the system design and operating experience since 1959, there is reasonable assurance that the system can continue to function adequately for the duration of the proposed license renewal.

Figure 5.1
Reactor Cooling System

6.0 ENGINEERED SAFETY FEATURES

The only feature designed to mitigate the consequences of a nuclear accident at the reactor facility is the reactor room ventilation system (see Chapter 13). However, specific credit for mitigation is not included in the accident analyses.

The reactor room ventilation system provides heated or cooled 100 percent outside air to the AJBRF at the rate of 1520 cfm (2600 m^3/hr) through six ceiling outlet ducts. The exhaust effluent of 2970 cfm (5000 m^3/hr) exits the reactor room into the outside air by means of an exhaust fan installed in the outside wall of the building to the water treatment pit and to the outside at ground level [required by TS 5.4 (1)]. TS 3.5 (1) requires the exhaust fan to be operating during reactor operation and to have a flow rate of at least 2970 cfm (5000 m^3/hr).

In addition, two laboratory fume hoods exhaust a total of approximately 919 cfm (1560 m^3/hr) by means of fans installed on the roof of the hospital building. Although not required by the TSs to be in operation during routine reactor operation, the licensee normally keeps these hoods running. One hood is required to be in operation by the TSs when the pneumatic tube is in use. TS 3.5 (2) requires the pneumatic tube system to exhaust into one of the fume hoods with a nominal flow rate of 250 cfm (425 m^3/hr) during pneumatic tube operation. The fume hood exhausting the pneumatic tube system is required to be exhausted to the roof [required by TS 5.4 (1)] of the Medical Center. The fume hood exhaust system for the pneumatic tube has a flow switch with an audible alarm that will indicate if the exhaust fan stops [required by TS 5.4 (2)] which is tested prior to each day's use of the pneumatic system for proper operation and after repair or maintenance to the system [required by TS 4.5 (1)]. This helps to ensure that argon-41 produced in the pneumatic tube system will be properly vented to the environment. The pneumatic tube system is one of the largest potential sources of argon-41 during reactor operation. Exhausting this source of argon to the roof of the Medical Center allows for a large degree of dispersion and dilution in the environment.

TS 4.4 (2) requires that a daily check of the ventilation system operability be performed prior to reactor operation. Because of the intake and exhaust flow rates discussed above, the reactor room is kept at a slight negative pressure during reactor operation with respect to the rest of the hospital [required by TS 3.4 (1)]. To help maintain this negative pressure, TS 3.4 (2) requires that the doors to the facility be closed except for normal entry during reactor operation.
TS 4.4(1) requires a daily check prior to reactor operation to ensure that all doors to the reactor facility are closed. The reactor area exhaust fan is normally operated continuously and has a starter switch mounted on the reactor console so that it can be started or stopped manually. The fan is equipped with a gravity-operated damper on the exhaust side, so that the exhaust damper will close when the fan is switched off or its power is interrupted. In addition, when the fan is stopped, a duct pressure control closes an absolute damper in the air supply duct and simultaneously causes an alarm to be initiated on the hospital central control system [which in accordance with TS 4.5 (2) is tested for proper operation monthly], which is monitored continually. Thus, in case of an emergency, a single switch on the reactor console can stop air from entering or leaving the reactor laboratory. If the exhaust fan stops, the hospital ventilation engineers are immediately notified by the Medical Center computer system.

The fume hoods and exhaust fans share in the removal of airborne radioactive materials, including argon-41, and fission products if a maximum hypothetical accident were to occur. Therefore, their use helps limit the radiation doses to reactor room occupants, and to lower the concentration of such radiation sources in the unrestricted area because of the two exhaust locations.

The NRC staff reviewed the design, maintenance, operation and TS requirements of the reactor room ventilation system. The staff concludes that the reactor ventilation system equipment and procedures are adequate to provide controlled release of airborne radioactive effluents during normal operations and in the event of abnormal or accident conditions, and that the reactor staff, researchers, and the public will be adequately protected from airborne radioactive hazards related to reactor operations. On the basis of the staff's review of the operational experience of the facility and TS requirements for operability and testing of the system, the staff concludes that degradation of components will be detected and components replaced as needed, therefore, there is reasonable assurance that the systems discussed in this section of the SER can continue to operate safely, as limited by the TSs for the proposed license renewal.

7.0 INSTRUMENTATION AND CONTROL SYSTEMS

The instrumentation and control systems for the AJBRF TRIGA reactor are similar to those used in other research reactors in the United States. The nuclear fission process is controlled by using three neutron-absorbing control rods. The control and instrument systems are interlocked to provide automatic and manual protective (scram) capability in case of a reactor malfunction and to provide the means for operating the various components in a manner consistent with design objectives. In 1991, a license amendment authorized the replacement of part of the original analog instrumentation and protective circuitry with a microprocessor-based system (for an evaluation of the hardware and software associated with the microprocessor-based system see Amendment No. 9 to Facility Operating License No. R-57 issued on April 12, 1991). A schematic of the instrumentation and protective systems is shown in Figure 7.1. The minimum required reactor scram channels and interlocks are shown in Table 7.1 and the required minimum measuring channels are shown in Table 7.2. TS 4.2.4 requires channel tests and checks, as applicable, of all scram channels and interlocks required by TS 3.2.3, 3.2.4, and 3.2.5 before each reactor startup at the beginning of each operating day after a secured shutdown and after maintenance. The fission counter power supply, and the watch dog timer are tested monthly.

7.1 Control Console

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

- rod control switches for raising and lowering the control rods
- position indicators to show the position of the control rods to within 0.2 percent
- annunciator lights to indicate the up or down position of each rod and rod-magnet contact
- linear and log-N power recorders
- reactor period, power level, pool temperature, and log count-rate meters
- radiation monitor alarm lights
- microprocessor control and display panels
- additional pilot lights to indicate power on, cooling system on, and startup source strength, and
- other annunciator lights to indicate the source of a scram signal.

Automatic scram (TS 3.2.3) is initiated by:

- an excessive reactor power level as indicated by either a wide range fission chamber or an uncompensated ion chamber channel. TS 4.2.5 requires the calibration of the power measuring channels by the calorimetric method annually or after any modification or repair of the measuring system or change in the fuel configuration

- a signal from the watchdog timer
- a neutron detector power supply (ion chamber or fission counter) failure, or
- an electrical power interruption.

Manual scram can be initiated by the operator by means of either the console scram button or the magnet current key switch. The magnet current key switch breaks only the rod magnet circuit so that the rest of the console may be operated without rod withdrawal if the switch is off. Following a scram or the dropping of a rod and after the rod reaches the full-in position, the drive mechanism automatically follows the rod down to reestablish contact.

7.2 Instrumentation System

The instrumentation system used in the reactor is composed of both nuclear control and process instrumentation circuits. The instrumentation system provides annunciation and/or indication at the control console. In addition, an automatic scram function is provided through the scram-logic units discussed below. Additional features of the instrumentation system include alarms, interlocks and rod-drive inhibits.

7.2.1 Nuclear Instrumentation

The nuclear instrumentation of a research reactor provides information necessary for the operator to evaluate the operating status and perform appropriate manipulation of nuclear controls, and initiate protective action to prevent reactor operation beyond acceptable limits. In accordance with a license amendment in 1991, the licensee replaced some of its initial analog instrumentation by a system based on digital computer techniques, the NM-1000. The reactor is authorized to operate with two neutron detecting channels, which achieves minimum redundancy and diversity (Figure 7.1).

The analog channel derives its signals from an uncompensated boron-coated ion chamber, and is used principally as a monitoring safety channel near the licensed power level. The second channel derives its neutron signal from a low noise, uranium-coated fission counter. The signal from the fission counter amplifier is conditioned in the microprocessor whose output provides continuous reactor power measurements from shutdown source level to above maximum licensed power level, a range of about ten decades. At thermal power levels up to about one kilowatt, the microprocessor counts individual fission pulses and converts them to a counting rate that is linear with power level. From about one kilowatt to above licensed power level, the microprocessor cannot respond to individual pulses because the rate is too high, and instead uses a Campbelling process. This analyzes the statistical fluctuations in the ion current, producing a root-mean-square signal proportional to the reactor thermal power. These two segments of the power signal are matched to produce a continuous output signal over the entire power range. This signal is the source for a wide range logarithm of the power, a rate of change of power (reactor period), and a multi-range linear power monitor. The linear channel and the reactor period signals are coupled into the reactor servo system for automatic power increase and steady power control. The automatic regulating channel consists of a servo-amplifier that controls the regulating rod and thus keeps the reactor power level constant. The servo-amplifier is activated by an error signal that is governed by the setting of the power demand-control in relation to the actual reactor power level. Because period information also is employed, the servo-amplifier may be used to automatically bring the reactor up to a power level, within the limits of the worth of the regulating rod, on a preset period of either 30 or 60 seconds. Automatic changes in power level on these periods are possible. The servo-amplifier will allow quick recovery to bring the power level back to within about 1 percent of the original value, even when a step change in reactivity of up to several tenths of one percent of $\Delta k/k$ is made.

The neutron-sensing chambers are hermetically sealed in aluminum cans and are mounted on the outside of the reflector so that their positions may be vertically adjustable in order to change sensitivity.

7.2.2 Process Instrumentation

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

- The water radioactivity monitor comprises a gamma radiation detector and a countrate meter circuit that gives both audible and visible alarms if the gamma activity in the pool water reaches a preset value.

- The water conductivity monitor consists of a conductivity probe and Wheatstone bridge circuit. Regular measurements of the conductivity are made to ensure that neutron activation of pool water impurities will be small and that chemical corrosion of fuel cladding is limited. Experience has shown that the buildup of radioactive isotopes in the coolant is negligible if the average conductivity during operation does not exceed five micromhos per centimeter (See Section 5).

- The water temperature monitor consists of a resistance bulb thermometer that senses the bulk pool temperature. Temperature indication is provided on the control console. This system is required to be operational whenever the reactor is in operation (required by TS 3.2.5). The reactor is shut down if the temperature exceeds 35 °C (95 °F) [required by TS 3.3 (5)]. Previous experience and calculations for TRIGA reactors operating at power levels up to 250 kW indicate that with the bulk pool temperature and therefore, the inlet coolant temperature, limited to 35 °C (95 °F), the maximum temperature of reactor fuel under steady-state operating conditions will be well below the phase change temperature of 530 °C (986 °F) for the $ZrH_{1.0}$ alloy.

- The water level monitor consists of a float switch and associated circuitry. This provides both an audible and visual alarm at the hospital switchboard which is continuously monitored and a visual alarm on the reactor console if the water level is less than 12 ft (3.6 m) above the top of the core (TS 3.1.4). TS 3.3 (4) requires a minimum depth of 15 ft (4.5 m) of water above the core for reactor operation. Not only does this requirement ensure sufficient vertical shielding, but also ensures that natural convection cooling would not be inhibited.

- The area radiation monitor is a calibrated, non-jamming gamma ray monitor with an audible alarm located in the reactor laboratory. The monitor is positioned a short distance from the reactor isotope removal tube. The normal alarm set point is 2 mR per hour (required by TS 3.6.1).

- The continuous airborne radiation monitor is a calibrated, recording, continuous air monitor located in the reactor room near the top of the reactor [the sample point is within 5 meters (16 ft) of the pool at the pool access level]. The monitor can detect both gaseous and particulate radioactivities. The monitor location is consistent with the expected area of maximum airborne activity under both normal and abnormal conditions. The alarm set point (2000 pCi/ml) (required by TS 3.6.1) is based on detecting 70 percent of the occupational derived air concentration (DAC) values for particulate activity for isotopes in the ranges 84-105 and 129-149. The monitor also contains a charcoal filter to provide the capability of monitoring for airborne radioiodines.

TS 3.6.1 requires the radiation monitoring channels to be in operation whenever the reactor is operating. However, for periods of time required for maintenance not exceeding one month, the TSs permit the replacement of the monitoring device with portable instruments that perform essentially the same function as the replaced monitor. Additional portable radiation monitoring devices that are not part of the required area and airborne monitoring systems are available.

TS 4.6(1) requires an annual calibration of the radiation monitors and calibration after maintenance, according to the manufacturer’s recommendations. TS 4.6(2) requires a channel test of the radiation monitors daily prior to reactor operation. If the reactor is not operated for a period of time, the channel test must be performed at least monthly.

7.3 Control System

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

7.3.1 Nuclear Control System

The nuclear control system consists of the safety, shim, and regulating rods and their associated drive mechanisms. A discussion of the control mechanism is presented in Chapter 4. The logic of the control instrumentation includes the following:

- The drive mechanisms consist of two-phase motors and reduction gears driving racks and pinions.

- The control rods are magnetically coupled to the drive shafts and can be released for gravity insertion.

- The control rods can only be withdrawn singly. Gang operation is prevented by an interlock [required by TSs 3.2.2 (2) and 3.2.4].

- A rod interlock prohibits the withdrawal of control rods when the reactor period is less then three seconds. The purpose of the interlock is to prevent operating the reactor on a short period that could cause power to quickly increase and exceed the limiting safety system setting (20 kW reactor power).

- The speed of rod withdrawal is limited to about 12 in (30 cm) per minute to ensure a conservatively safe rate of reactivity insertion. The reactive insertion of control rods is limited to less than 0.10$ per second by TS 3.2.2 (1). The withdrawal time of the safety rod is measured daily prior to the first start up of the reactor. Withdrawal times of the shim and regulating rods are measured on an average annually. Insertion speeds of all control rods are measured annually (required by TS 4.2.2). The regulating rod may be used for automatic servo control of reactor power at a constant preset level or for bringing the reactor to power on a preset period.

The short period rod withdrawal interlock replaces a short period scram that occurred if the reactor period was less than seven seconds. This short period interlock is approved for use at other similar TRIGA reactors such as the reactor at Reed College. It is shown in Section 13 of this SER that rapid insertion of the total available excess reactivity would not cause the maximum fuel temperature to reach the safety limit. Although this reactor does not have the capability to pulse, TRIGA reactors similar to the AJBRF TRIGA are pulsed routinely on very short periods without damage to the fuel cladding. The reactor control system has existing software and hardware for implementation of this interlock because it was approved by NRC for use on a reactor operated by the designer and reactor vendor, General Atomics. The verification and validation of the necessary software have already been performed and have been evaluated and approved by NRC. Based on the discussion above, the replacement of the period scram with a period interlock is acceptable to the staff.

The licensee requested that the requirement for a rod interlock that prohibits withdrawal of the shim and regulating rods unless the safety rod is fully withdrawn and a rod interlock that the safety rod cannot be withdrawn unless the shim and regulating rods are fully inserted be eliminated from the TSs. This requirement dates to an early generation of TRIGA control console where the location of the safety rod was not displayed to the operator. The operator only had indication of the safety rod being fully inserted or fully withdrawn. The purpose of the interlocks was to allow movement of the shim and regulating rods only when the location of the safety rod was known. The licensee replaced this early control console with a newer version that displays the location of the safety rod. Therefore the staff concludes that the two interlocks discussed above are no longer required and can be removed from the TSs.

7.3.2 Process Control Systems

The process control systems consist of the circuitry and devices required to energize and deenergize the coolant pump and reactor room ventilation systems.

7.4 Conclusions

The NRC staff reviewed the design, maintenance, operation and TS requirements of the instrumentation and control system. The staff concludes that the control, instrumentation, and protective systems at the AJBRF are acceptably designed and maintained. The quality of workmanship evident in the installation is acceptable. All electrical power and instrumentation wiring is protected from physical damage by conduit and/or cable trays. Redundancy in the important area of reactor power measurements is ensured by overlapping ranges of the log-N and linear power channels. The control and protective system is designed so that the reactor is automatically shut down if electrical power is lost, and if any one of several protective trips is activated.

The NM-1000 system is similar to systems installed and in use with several other licensed TRIGA reactors, all of which operate at maximum thermal power levels at least a factor of ten above the AJBRF. The use of the NM-1000 and one other linear safety channel provide minimum acceptable redundancy and diversity in the protective nuclear instrumentation.

On the basis of an analysis of the control, instrumentation, and reactor protective systems, the staff concludes that the systems comply with the requirements and the performance objectives of the TSs and they are acceptable to ensure safe operation and shutdown of the facility. On the basis of the staff's review of the operational experience of the facility and TS requirements for operability and testing of the system, the staff concludes that degradation of components will be detected and components replaced as needed, therefore, there is reasonable assurance that the instrumentation and control systems discussed in this section of the SER can continue to operate safely, as limited by the TSs for the proposed duration of the license.

Table 7.1
Minimum Reactor Scrams and Interlocks

Safety Channel	Function	Setpoint
Percent power	Scram	100% of licensed power
Linear power level	Scram	100% of licensed power
Magnet current key switch	Scram	Manual
Console scram button	Scram	Manual
Ion chamber power	Scram	loss of high voltage
Fission counter power supply	Scram	loss of high voltage
Watchdog timer	Scram	Key software tasks take longer than 1.5 secs
Reactor Period	Prevents rod withdrawal	Interlock if period < 3 seconds
Neutron count rate (startup)	Prevents rod withdrawal	Interlock if count rate < 2 counts per second
Simultaneous manual withdrawal of two rods	Prevents rod withdrawal	
Pool level	Warning	Alarm when water level is less than 12 ft (3.6 m) above top of the core
Bulk pool temperature	Meter indication	Reactor shutdown (manual) if temperature $\geq$35 °C (95 °F)

Table 7.2
Required Minimum Measuring Channels

Channel	Number Operable	Function
Startup (NM-1000, fission chamber)	1	Monitor subcritical multiplication for startup
Power level (NM-1000, fission chamber)	1	Input for safety power level scram and to digital display unit and recorder
Log N (NM-1000, fission chamber)	1	Wide range power level and display on digital unit and on recorder
Period (NM-1000, fission chamber)	1	Input for period display on digital unit and period interlock
Percent power (ion chamber)	1	Input for power level scram and display on analog meter
Pool water temperature	1	Display on analog meter*

* For purposes of maintenance the in-line thermistor may be replaced by a thermistor placed in the reactor tank and read on a separate meter.

Figure 7.1
Block Diagram of Instrumentation

Figure 7.2
Functional Diagram of the NM-1000

Figure 7.3
Schematic Representation of Conditions Leading to a Scram on the TRIGA Mark I Reactor

8.0 ELECTRICAL POWER SYSTEM

The electrical power requirements of the AJBRF are supplied by three circuits from the hospital electrical distribution system.

The AJBRF has no emergency electrical power system except for two battery-powered lanterns that activate when the building power fails. In the event of loss of electrical power, the control rods are released to fall into the core by gravity, causing shutdown of the reactor. When this 20 kW reactor is made subcritical, the decay heat in the fuel is readily dissipated in the ambient coolant without a significant temperature increase. No fixed radiation monitors are supplied with alternative or emergency electrical power.

On the basis of its review, the NRC staff concludes that the electrical power system is acceptable and that an emergency power system is not necessary to ensure and maintain safe shutdown of the reactor.

9.0 AUXILIARY SYSTEMS

The auxiliary systems discussed in this section include the fuel handling and storage systems, fire protection provisions, other research facilities and warning and communication systems. The ventilation system is discussed in Section 6, "Engineered Safety Features" and radioactive waste storage is discussed in Section 11, "Radiation Protection Program and Radioactive Waste Management."

9.1 Fuel Handling and Storage

Handling of fuel elements is done in the pool water by using long-handled tools. If a fuel element is to be removed from the pool, it would first be placed in a shielded cask.

If a fuel element were to become damaged, it would be removed from the pool and placed in one of three pits in the reactor room floor. These pits may be filled with water for additional shielding, but analyses have shown that water would not be required to maintain acceptable removal of decay heat from the fuel. These pits can also be used for storage of undamaged irradiated fuel, or other radioactive components. The storage pits are discussed further in Section 4.2, “Reactor Tank,” of this SER.

9.2 Fire Protection System

The reactor room has two fire alarm boxes. A smoke detector is located in the corridor 111 ft (34 m) north of the reactor room door. Three fire extinguishers are available (two carbon dioxide and one dry chemical) to facility personnel for fighting small fires. The reactor room is equipped with a sprinkler system that is dry until heat sensors activate to fill the system and trigger an audible alarm when the temperature reaches 135 °C (275 °F). The sprinkler heads open at 165 °C (329 °F). Any fires that cannot be controlled by operating personnel will be dealt with by the Omaha, Nebraska, municipal fire department. The fire department can be called by telephone 911, by two-way radios that the Medical Center Police have, or by direct alarm box. Response time is less than 10 minutes. Personnel from the fire department are briefed on special hazards, including radiological hazards, that might be encountered in fighting fires at the facility. The facility Emergency Plan and Emergency Implementing Procedures contain the facility staff response to fires. Training is discussed in the approved Emergency Plan. Combustible materials are controlled to low levels in the reactor room.

9.3 Research Facilities

The reactor is an integral part of the biomedical research laboratory located at the hospital. Byproduct materials produced by reactor experimental irradiations are handled and processed in reactor laboratory spaces where the same radiation protection principles are employed, the same protective measures are implemented, and the same controls of radioactive materials are achieved as applicable to reactor operations. Radioactive materials used in laboratories outside of the AJBRF are under the Veterans Administration byproduct materials license.

9.4 Warning and Communication Systems

In the event of a fire or radiological emergency, an alarm is sounded by means of a switch on the reactor console. The alarm can be activated for the reactor facility or for the basement and first floors of the hospital building.

The Medical Center has an audio page system consisting of speakers strategically placed throughout the Hospital and Research Buildings. These are controlled by the switchboard operator. There is also a medical center page system where key personnel including physicians, administrative and engineering personnel, and the Radiation Safety Officer carry pocket pagers that can be operated by any telephone in the Medical Center. Telephones are located in the reactor facility and throughout the Medical Center. The Medical Center Police have two-way radios.

9.5 Conclusions

The staff has reached the following conclusions on the facility auxiliary systems:

- The fuel handling and storage system designs are adequate to ensure that reactor fuel can be moved, serviced, and stored without danger to operating personnel or the public because of fuel radioactivity or a possible accidental criticality event.

- The fire protection provisions are consistent with similar provisions at NRC-licensed non-power reactor facilities and are acceptable to detect and respond to fire events.

- Use of byproduct material in research facilities associated with the reactor is well controlled and will be conducted safely.

- The warning and communication systems are adequate to ensure that sufficient warning can be given of abnormal events and that appropriate communications can be conducted.

On the basis of the above findings, the staff concludes that the reactor facility auxiliary systems can provide the necessary service to the reactor facility for the requested license renewal period.

10.0 EXPERIMENTAL PROGRAMS

The reactor serves as a source of ionizing and neutron radiation for research, education, and radionuclide production. In addition to in-pool irradiation capabilities, the experimental facilities include a pneumatic transfer system, a rotary specimen rack, and a central thimble. The TSs limit the effect on reactivity of all experiments, and provide means for technical and safety review.

10.1 Experimental Facilities

The reactor experimental facilities include:

- Irradiation of samples in the reactor pool
- Pneumatic transfer system
- Rotary specimen rack, and
- Central Thimble

10.1.1 Pool Irradiations

The open pool of the reactor permits bulk irradiations in the water outside the cylindrical graphite reflector. The decision to perform experiments in the reactor pool (as opposed to using the pneumatic transfer system or the specimen rack) is dictated by specimen size and the type and radiation source strengths required. The actual placement of experiments or samples in the core region may also be impacted by their effect on excess reactivity.

10.1.2 Pneumatic Transfer System

A 1.25 in (3 cm) outside diameter pneumatic transfer tube is provided for the rapid transport of samples to and from the region of the reactor core. The sample holders can be inserted or removed while the reactor is in operation through a constant exhaust system that is vented through a filter to one of the fume hoods. The system has automatic timing controls. This facility is used principally for the production of isotopes with short half lives. The specimens are inserted into and removed from the pneumatic system in a fume hood in the reactor facility, with shielding as necessary. The fume hood is near the reactor console and under direct control of the reactor operator. Any airborne radioactive material in the pneumatic system air is controlled and exhausted from the reactor area by the fume hood blower. The fume hood ventilation system is discussed in Section 6 of this SER.

10.1.3 Rotary Specimen Rack

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

The rotary specimen rack is enclosed completely in a welded aluminum container. The aluminum ring is located at approximately the level of the top grid plate, with the specimen cups extending from the ring down to about 4 in (10 cm) below the top of the active lattice. In the radial direction, the centers of the cups are about 4 in (10 cm) outside of the inner edge of the graphite reflector assembly. The container enclosing the rotary specimen rack has been designed to ensure that it will remain watertight. Furthermore, the system is designed so that flooding this container will decrease the reactivity of the reactor because of the increased neutron absorption in the hydrogen.

10.1.4 Central Thimble

A central thimble is provided to permit irradiations or experiments in the region of maximum neutron flux density. It consists of a vertical 1.5 in (3.4 cm) inside diameter aluminum tube leading from the top of the reactor pool through the center of the reactor core. The bottom of the tube is capped, but holes drilled in the wall of the tube directly above the upper grid plate ensure that the portion within the fueled region will be filled with water during reactor operation. Samples in water tight containers can be lowered down the tube for irradiation.

The shield water can be removed from the portion of the central thimble above the upper grid plate using air pressure to force the water out of the tube through the holes in the tube wall. This provides a highly collimated beam of neutron and gamma radiation for experiments. The maximum radiation dose on the next floor directly above the reactor is 1.5 mR per hour with the thimble, which has 2 in (5 cm) of lead shielding above it, operating as a beam tube. This radiation level is within the limits given in 10 CFR 20.1301 for individual members of the public.

Lead bricks are stacked around the central thimble before the shield water is removed, and the highest radiation dose in the reactor room when the thimble is used as a beam tube is less than 2 mR per hour. The central thimble has been used only once for a beam to determine radiation dose levels and has demonstrated that with additional shielding [2 to 4 in (5 to 10 cm) of lead] above the thimble, it could be used with no significant hazard to the hospital staff or to the public.

10.2 Experimental Limits

A number of requirements are placed on the conduct of experiments in the TSs to help ensure that experiments are carried out safely. These limitations are in the areas of reactivity limits, experiment materials, and experiment failure and malfunction.

Two types of experiments are defined in the TSs, movable and secured experiments. Moveable experiments are defined as those where it is intended that all or part of the experiment may be moved in or near the core or into and out of the reactor while the reactor is operating. The reactivity of moveable experiments is closely controlled because they represent a routine change in reactivity during reactor operation. Secured experiments are those held in a stationary position relative to the reactor by mechanical means. The restraining forces of the experiment must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions. Because they are designed not to move and thus introduce reactivity into the reactor during operation, this type of experiment normally has higher reactivity limits than moveable experiments. TS 3.7.1 (3) requires the reactor to be shutdown when changing or moving a secured experiment.

TS 3.7.1 (1) limits the reactivity of each moveable experiment to 0.85$ and each secured experiment to 1.00$. Experiments can have positive or negative reactivity upon insertion into the reactor. It is possible that two experiments inserted into the reactor with opposite reactivity effects could have a net affect of zero on the reactor. However, upon removal an experiment has an opposite reactivity affect. Therefore, the total absolute value of the experiments in a reactor is also controlled with TS 3.7.1 (2) limiting the absolute worth of experiments in the reactor to 1.00$. If the estimated reactivity worth of an experiment is greater than 0.40$, TS 3.7.1 (4) requires that the actual experiment worth be measured and recorded at the time of initial insertion into the reactor. This helps to ensure that experiments will not be placed into the reactor that are in violation of reactivity limits.

Experiment materials are controlled to limit radioactive material that could be released if experiment failure occurs or to prevent damage to the reactor if experiment failure occurs. TS 3.7.2 (1) requires that experiments containing liquid, gas or potentially corrosive materials will be doubly encapsulated. This is to provide two barriers between the experiment and the pool water decreasing the probability of experiment failure. TS 3.7.2 (2) prohibits the irradiation of compounds highly reactive with water, potentially explosive materials and liquid fissionable materials in the reactor because of these material's potential to damage the reactor and release radioactive materials into the reactor facility. Fueled experiments (containing uranium) shall not be irradiated in the reactor except for the activation of uranium foils [required by TS 3.7.2(7)]. These foils are usually used for calibration purposes. This TS controls the amount of radioactive material produced in the reactor. TS 3.7.2 (6) states that no experiment should be performed unless the material content, with the exception of trace constituents, is known. This prevents the creation of unknown amounts and types of radioactive material.

The amount of radioactive material in experiments is controlled by TS 3.7.2 (4). The radioactive material is limited so that the complete release of all gaseous, particulate, or volatile components from the encapsulation will not result in doses in excess of the annual limits stated in 10 CFR Part 20. TS 3.7.3 (1) contains assumptions that must be applied to experiment failure analyses to ensure that calculations are sufficiently conservative. There is an experiment design requirement in TS 3.7.3 (2) that experiments be designed such that they will not contribute to the failure of other experiments, core components, or principle physical barriers or to the uncontrolled release of radioactivity. However, if a failure occurs and releases material which could damage the reactor fuel or structure by corrosion or other means, TS 3.7.2 (5) requires that removal and physical inspection of potentially damaged components be performed to determine the consequences and need for corrective action. The results of the inspection and any corrective action taken are reviewed by the Reactor Director and determined to be satisfactory before reactor operation is resumed.

10.3 Experimental Review

A Reactor Safeguards Committee (RSC) that reports to the Chief Executive Officer of the Health Care System provides an independent review of changes in reactor operating procedures, proposed changes in TSs or other license conditions, and all experimentation affecting reactor operation. This committee is composed of individuals collectively having a broad spectrum of expertise in radiation or reactor-related technology or both (see Section 12 of this SER).

The limitations on experiments are specified in Section 3.7 of the TSs as discussed above. TS 6.5 requires experiments to be carried out in accordance with written procedures properly reviewed and approved. All new experiments or class of experiments are reviewed by the RSC and approved in writing by the RSC and the Reactor Director/Supervisor prior to initiation. Substantive changes to previously approved experiments are made only after review by the RSC and approval in writing by the Reactor Director/Supervisor. Minor changes that do not significantly alter the experiment may be approved by the Reactor Director/Supervisor or a designated shift Senior Reactor Operator.

In addition to ensuring safe reactor use, the review and approval processes provide for personnel specifically trained in radiological safety and reactor operations to consider and recommend alternative operational conditions (such as different core positions, power levels, or irradiation times) that might decrease personnel exposure and/or the potential release of radioactive materials to the environment; in accordance with ALARA principles.

10.4 Conclusions

The staff concludes that the design of the experimental facilities, combined with the detailed review and administrative procedures applied to all research activities, give reasonable assurance that experiments (1) are unlikely to fail, (2) are unlikely to release significant radioactivity to the environment, (3) are unlikely to cause damage to the reactor systems or its fuel, and (4) are not likely to prevent safe shutdown of the reactor. Therefore, on the basis of its review of the facility and operating experience since 1959, the staff considers that reasonable provisions have been made so that the experimental programs and use of the experimental facilities do not pose a significant risk of damage to the reactor or of uncontrolled release of radioactivity to the environment, or of unacceptable radiation exposure of staff or the public. The staff also concludes that there is acceptable assurance that the reactor can continue to operate safely with its experimental program, as limited by its TSs for the purposed duration of the license.

11.0 RADIATION PROTECTION PROGRAM AND RADIOACTIVE WASTE MANAGEMENT

11.1 Radiation Protection Program

As required by 10 CFR 20.1101(a), the licensee has a structured radiation protection program with a health physics staff equipped with radiation detection equipment to determine, control, and document occupational radiation exposures at the reactor facility. TS 6.3 (1) requires that the radiation safety program complies with the requirements of 10 CFR Part 20. This TS also states that additional guidance for the radiation safety program may be found in ANSI/ANS 15.11-1993, “Radiation Protection at Research Reactor Facilities.” This standard is generally supported by the NRC staff and is in use at many non-power reactors. The facility monitors effluents to ensure that the releases are in compliance with applicable regulations.

11.1.1 ALARA Commitment

The licensee has formally established the policy that all reactor-related operations are to be planned and conducted in a manner to maintain all radiation exposures ALARA. This policy is implemented by specific guidelines and procedures. TS 6.3 (4) states that management is committed to practice an effective ALARA program that is aimed at making every reasonable effort to maintain radiation exposure as far below the limits specified in 10 CFR Part 20 as practicable. The TS also states that the ALARA program should apply to facility staff, facility users, the general public and the environment. All proposed experiments and procedures at the reactor are reviewed for ways to reduce the potential exposures of personnel. Any unanticipated or unusual reactor-related exposures are investigated by the radiation safety officer, the reactor operations staff, and the RSC to develop methods to prevent recurrences.

11.1.2 Health Physics Program

Important aspects of the facility health physics program are discussed in this section of the SER.

11.1.2.1 Health Physics Staffing

The normal, full-time health physics staff at the hospital consists of one professional and one technician with additional support as needed. The onsite staff has sufficient training and experience to direct the radiation protection program for the research reactor. The Radiation Safety Officer has been given the responsibility, the authority, and adequate lines of communication to provide an effective radiation safety program. TS 6.3 (2) assigns the Radiation Safety Officer or his designate the responsibility for implementing the radiation protection program at the reactor facility using the regulations and ANSI/ANS 15.11-1993. TS 6.3 (3) has the Radiation Safety Officer reporting to Level 1 management (Chief Executive Officer) through the Chief of Staff.

11.1.2.2 Procedures

Detailed written procedures have been prepared that address the health physics activities required for the reactor facility and associated research programs. TS 6.4 (5) requires procedures for personnel radiation protection, consistent with applicable regulations or guidelines. The TS requires the procedures to include management commitments and programs to maintain exposures and releases ALARA in accordance with the guidelines of ANSI/ANS 15.11-1993. TS 6.4 (8) requires procedures for use of byproduct material and shipment of byproduct material. Copies of these procedures are readily available to all personnel.

11.1.2.3 Experimental Support

The health physics staff participates in experiment planning by reviewing all proposed procedures for ways to control personnel exposures and limit the generation of radioactive waste. Approved procedures specify the type and degree of health physics involvement in each activity. As examples, operating procedures require that changes in experimental setups include a survey by health physics personnel using portable instrumentation, and all items removed from the reactor room must be surveyed.

11.1.2.4 Non-Routine Tasks

One-of-a-kind, short-term tasks (such as non-routine maintenance activities) are occasionally performed in potential radiation or contamination areas, but only after detailed staff review. The work is then performed with health physics coverage.

11.1.2.5 Training

All reactor facility 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 currently are given an examination on health physics practices and procedures during each requalification cycle. The level of any retraining given is determined by the examination results.

11.1.3 Radiation Sources

The major radiation sources that are of concern to the reactor radiation protection program are discussed below.

11.1.3.1 Reactor

Sources of radiation directly related to reactor operations include radiation from the reactor core, ion exchange and filter equipment in the demineralizer system, airborne radioactive materials (primarily argon-41), and several sealed sources needed for reactor startup and radiation detector calibrations.

The reactor U-ZrH_x alloy fuel is contained in aluminum or stainless steel cladding. Radiation exposures from the reactor core are reduced to acceptable levels by water and concrete shielding. The ion exchange resins and the filters are changed routinely while only low levels of radioactive materials have accumulated, thereby limiting personnel exposure.

11.1.3.2 Extraneous Sources

Sources of radiation that may be considered as incidental to normal reactor operation, but are associated with reactor use, include radioactive isotopes produced 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 standard protective measures of time, distance, and shielding.

11.1.4 Radiation Monitoring

Aspects of the licensee’s radiation protection program concerning the routine monitoring of radiation are discussed in this section of the SER.

11.1.4.1 Instrumentation

The reactor facility has a variety of detecting and measuring instruments available for monitoring potentially hazardous ionizing radiation. The instrument calibration procedures and techniques ensure that any credible type of radiation and any significant radiation intensities will be detected promptly and measured correctly.

11.1.4.2 Fixed Position Monitors

The reactor facility uses several fixed-position radiation monitors placed at strategic locations in the reactor room. Area radiation monitors (the monitor at pool level is required by TSs) provide audible and visible alarms if radiation levels exceed setpoints. The licensee has two continuous air monitors (one monitor is required by the TSs) which are located in the reactor room. A water radioactivity monitor would detect a gross fuel cladding failure. (See Section 7.4.2 for additional information on the fixed position monitors.)

11.1.4.3 Effluent Monitoring

The monitoring of airborne and liquid effluents released from the facility into the environment is discussed in this section of the SER.

11.1.4.3.1 Airborne Effluents

Monitored radioactive airborne effluents from the facility consist principally of activated argon-41. The airborne radioactivity is also monitored to provide prompt indication of any abnormal concentrations being discharged to the environment, from fuel-cladding failure, for example. This is accomplished by withdrawing a representative room air sample from a point near the top of the reactor through a continuous air monitor. This monitor also is provided with a charcoal filter for monitoring the presence of iodine radioactivity (an airborne fission product). The output of the monitor is indicated on a meter having adjustable alarm setpoints, and a continuous record also is provided.

11.1.4.3.2 Liquid Effluents

The reactor generates no radioactive liquid effluents, so no monitor for such materials is required. Radioactive liquid waste generated in the research program is stored and may be released into the sanitary sewer in accordance with regulatory requirements.

11.1.4.4 Environmental Monitoring

Radioactive argon-41 gas is the only potentially radioactive material released to the environment as a result of the routine operation of the reactor. The routine gaseous effluent measurements consist of those recorded by the continuous air monitor, and the monthly exposure data obtained from film badges located within the reactor room, at the exhaust stack output, and at the water treatment pit output. The pit exhaust represents the airborne exhaust to the environment because most reactor room air is discharged at that location. The net integrated exposure in the pit for a typical operating year was about 20 mrem.

11.1.5 Occupational Radiation Exposures

The personnel monitoring program for radiation exposure at the facility and the results of personnel monitoring are described below.

11.1.5.1 Personnel Monitoring Program

Reactor facility personnel exposures are measured by the use of film badges assigned to individuals who might be exposed to radiation. In addition, self-reading pocket ion chambers or electronic dosimeters are used. Instrument dose rate and time measurements are used to ensure that administrative occupational exposure limits are not exceeded. These limits are in conformance with the limits specified in 10 CFR Part 20.

11.1.5.2 Personnel Exposures

The reactor facility personnel annual exposure history has shown very low doses. For the years 1996 to 2000, no monitored person received a dose greater than 100 mrem per year.

11.2 Radioactive Waste Management

Radioactive waste from research reactors normally is in the form of a gas, liquid, or solid. Presently, only gaseous radioactive waste resulting from facility operations is discharged to the environment from the AJBRF. There has not been any liquid or solid waste released from the facility in recent years. Liquid waste may be released to the sanitary sewer system or solidified and packaged as solid waste and transferred to an approved disposal or processing site in accordance with applicable regulations. Solid waste may be packaged and transferred to an approved disposal or processing site in accordance with applicable regulations. TS 3.6.2 states that normal releases of radioactive effluents from reactor operation shall not exceed 10 CFR Part 20 limits.

11.2.1 Airborne Waste

The potential radioactive airborne waste includes neutron-activated gaseous argon-41 and nitrogen-16, and dust particulates in the dry experimental facilities. No fission products escape from the fuel cladding during normal operations. The amount of nitrogen-16 or activated dust particulates that escapes into the facility air during full power operation is very small. Argon-41 is the primary radioactive effluent released into the environment by the facility.

The radioactive airborne argon-41 is produced principally by the neutron irradiation of the argon found in air dissolved in the pool water and of the air in the pneumatic transfer system and the rotary specimen rack. Nitrogen-16 is produced by the $O^{16}(n,p)N^{16}$ reaction by fast neutrons as the coolant passes through the reactor core. Nitrogen-16 decays with a 7-second half life and experiences substantial decay during the time it takes water to go from the reactor core to the pool surface.

The staff has reviewed the licensee's assumptions and computations on the production rate of argon-41 and nitrogen-16 and finds them to be acceptable. The licensee assumed that the reactor operates for 2000 hours/year with 8 hour operating days. Actual operation has averaged 344 hours per year over the last 10 years. For the pneumatic tube, it was assumed that 3000 samples were irradiated per year which is greater than historical usage. It was assumed that 100 samples per year were inserted into the rotary sample rack and that the rack was sealed during operation. Because the average temperature of the pool water in this reactor does not increase much during operation, the rate of exchange of dissolved gases between the water and the room is not large. Hence, most of the dissolved argon-41 will beta decay within the water. Furthermore, because most of the argon-41 produced in the air in the rotary specimen rack will not exchange rapidly with room air, it will decay in situ. The licensee's calculations result in 260 mCi (9620 MBq) of argon-41 released from the reactor pool, 134 mCi (4958 MBq) of argon 41 from the pneumatic tube and 110 mCi (4070 MBq) of argon-41 from the rotary sample rack for a total release from reactor facility of 504 mCi (18,648 MBq) per year. Argon-41 from the reactor pool and rotary sample rack are released into the reactor room air and are released into the environment by the facility ventilation system to the reactor pit. The pneumatic tube argon-41 is released into the hood exhaust and is then released to the environment on the roof of the main hospital building. If conditions dictate, the exhaust fan can be turned off and automatic dampers will close off the air inlet to and the air exhaust from the reactor room.

The licensee calculated the amount of production of nitrogen-16 that will occur in the reactor core. The nitrogen-16 is carried to the pool surface by the thermal plume of the natural convection cooling of the reactor core. The licensee assumed that the nitrogen-16 rises directly to the pool surface although coolant returning to the pool through the coolant return pipe will disrupt the upward flow of the water containing the nitrogen-16. The low power level of the reactor results in a low production rate of nitrogen-16 and a weak thermal plume that takes 230 seconds to travel the 16 ft (4.9 m) from the top of the core to the pool surface. Given the half life of nitrogen-16 of 7 seconds, the nitrogen-16 decays through over 32 half lives during its ascent to the pool surface. The nitrogen-16 that escapes the pool into the reactor room air is not detectable.

11.2.2 Liquid Waste

Some activities associated with the normal research operations that are conducted within the reactor facility are capable of generating liquid radioactive waste. However, over the last 10 years, there has been no liquid radioactive waste released resulting from operation of the reactor.

All potentially radioactive liquid waste is stored to allow radioactive decay and then released to the sanitary sewer system as non-radioactive waste after it has been determined that it complies with the limits governing the release of soluble byproducts. If a release of radioactive liquid to the sanitary sewer were necessary, dilution of the radioactive waste with the sewer liquid with which it combines would be made to ensure that all releases are in compliance with applicable regulations in 10 CFR Part 20, Subpart K, “Waste Disposal,” for release of liquid radioactive materials.

11.2.3 Solid Waste

Low-level solid waste can be generated as a result of reactor operations. It consists primarily of ion exchange resins, filters, potentially contaminated paper and gloves, and occasional small, activated components. Over the last 10 years, these wastes have been allowed to decay in storage and were disposed of as non-radioactive waste. If necessary, radioactive solid waste would be packaged in accordance with applicable NRC (10 CFR Part 71) and Department of Transportation (DOT) (49 CFR) regulations and transferred from the facility in accordance with applicable regulations.

11.2.4 Potential Dose Assessments

Natural background radiation levels in the Omaha area result in an exposure of about 80 mrem per year to each individual residing there. At least an additional 10 percent (approximately 8 mrem per year) will be received by those living in a brick or masonry structure. Radon can add as much as 200 mrem per year to a person’s background dose. Any medical diagnosis X-ray examination will add to this natural background radiation, increasing the total accumulative annual exposure.

As noted above, argon-41 and nitrogen-16 are the two principal airborne radionuclides formed during routine operation of the reactor. Nitrogen-16 decays with a 7-second half life, so no measurable quantities escape or are released from the reactor building, leaving argon-41 as the principal, and usually the only, airborne radionuclide that could pose a routine radiological risk in both the restricted and unrestricted areas.

The staff expects licensees to conduct a detailed examination regarding the formation, release, and exposure parameters of argon-41. The purposes of this examination are to assess the potential doses with acceptable accuracy, and to demonstrate that the methods used to analyze radiologic effects in both the restricted and unrestricted environments are sufficiently understood and available for the licensee to assess doses resulting from possible inadvertent releases of airborne radioactive materials.

The licensee has estimated (using applicable methods) the formation of both of these nuclides, in various reactor operations.

11.2.4.1 Unrestricted Area

Conservative calculations by the licensee based on the maximum possible amount of radioactivity routinely released by reactor operations predict the exposure to the maximum exposed individual in the unrestricted area and at the nearest residence of less than one mrem per year based on best estimates of annual argon-41 production rates of 504 mCi (18,648 MBq) per year. The calculations for the unrestricted environment are based on semi-infinite cloud assumptions, so are overestimated by at least an order of magnitude. The staff considers this production and release rate to be a reasonable estimate on the basis of its knowledge of the facility and the reactor operating schedule. However, even if the facility were to operate continuously and experimental facility use would double, exposures would remain below one mrem per year. The staff independently calculated predicted doses that confirmed the licensee's results.

11.2.4.2 Restricted Area

The licensee provided information concerning the sources of argon-41 and nitrogen-16 in the restricted area during normal operation of the reactor. The licensee determined the maximum equilibrium argon-41 concentration in the reactor room would be about 9×10^{-9} μCi/ml for argon-41 released from the reactor pool and 1.3×10^{-8} μCi/ml for release from the rotary sample rack for a total concentration of 2.2×10^{-8} μCi/ml, which is less than the derived air concentration (DAC) given in 10 CFR Part 20, Appendix B, Table 1, Column 3 for argon-41 of 3×10^{-6} μCi/ml. The licensee calculated by acceptable finite room methods the annual potential exposures to the maximum exposed worker (assuming 2000 hours of reactor operation per year), of approximately 0.5 mrem. The staff has reviewed the licensee's assumptions and computations on the production rate of argon-41 and agrees with the licensee's methods and results. This exposure is due to argon-41 as no measurable nitrogen-16 leaves the reactor pool.

The licensee calculated the dose rate at the top of the pool from nitrogen-16 in the water. The calculated dose rate was 2×10^{-7} mrem/hour, which is not significant.

Personnel exposure to the radiation from chemically inert argon-41 is limited (1) by dilution of this gas in the reactor room, (2) by prompt removal of this gas from the reactor room and experimental areas by the ventilation system, and (3) by its discharge to the atmosphere, where it is diluted and diffused further before reaching occupied areas offsite.

11.3 Conclusions

The staff concludes that radiation protection receives appropriate support from the licensee's administration. Among other guidance, the staff's review considered the guidance of ANSI/ANS 15.11, 1993, "Radiation Protection at Research Reactor Facilities." On the basis of this review, the staff reached the following conclusions:

- The radiation protection program is acceptably staffed and equipped.

- The reactor health physics staff has adequate authority and lines of communication.

- The radiation protection procedures are integrated into facility operations and research plans.

- Surveys verify that operations and procedures achieve ALARA principles.

- The effluent monitoring programs and calculational procedures are adequate to promptly identify significant uncontrolled releases of radioactivity and to predict maximum exposures to individuals in the unrestricted area. These maximum levels are predicted by acceptable methods and are not more than one mrem per year, a very small fraction of applicable regulations and guidelines specified in 10 CFR Part 20.

- The reactor radiation protection program is acceptably implemented because there have been no instances of reactor-related exposures of personnel above applicable regulations and no unidentified or uncontrolled significant releases of radioactivity to the environment during the past years of reactor operation.

- There is reasonable assurance that personnel and procedures will continue for the duration of the license renewal to protect the health and safety of the public, the facility staff, and the environment from significant radiation exposures related to normal reactor operations.

- Waste management activities at the reactor facility have been conducted and can be expected to continue to be conducted in a manner consistent with both 10 CFR Part 20 and ALARA principles.

- The licensee's systems and procedures limit the production of argon-41 and nitrogen-16, and control potential exposures of facility staff. Conservative computations (by both the licensee and the staff) of the quantities of these gases released beyond the limits of the reactor facility give reasonable assurance that potential doses to the public as a result of argon-41 would not be significant, even if there were a major increase in the operating schedule and sample irradiations at the reactor.

12.0 CONDUCT OF OPERATIONS

The conduct of operations involves the administrative aspects of facility operation, the facility emergency plan and facility security. The administrative aspects of facility operations are the facility organization, training, operational review and audits, procedures, required actions, and records and reports.

12.1 Overall Organization

Responsibility for the safe operation of the AJBRF is vested within the chain of command shown in Figure 12.1. Sections 6.1.1 and 6.1.2 of the TSs provide details of the management requirements of the reactor. The Reactor Director/Supervisor is delegated responsibility for overall facility operation. The Reactor Director/Supervisor is responsible to the Chief Executive Officer and the Associate Chief of Staff for Research for safe operation and maintenance of the reactor and its associated equipment. Individuals at the various management levels, in addition to responsibility for the policies and operation of the reactor facility, are responsible for safeguarding the public and facility personnel from undue radiation exposure and for adhering to all requirements of the operating license and TSs. The Reactor Director/Supervisor delegates the succession to this responsibility during his absence.

TS 6.1.3 (1) contains the minimum staffing requirements when the reactor is not secure. A licensed reactor operator must be present in the reactor room. A licensed senior reactor operator may substitute for the reactor operator and serve as both reactor operator and senior reactor operator. The reactor does not have a control room. The control console is located near the reactor pool. A second person must be present at the reactor room able to carry out prescribed written instructions. This would typically be initiation of the emergency plan procedures if the reactor operator would become incapacitated. This second person may be unexpectedly absent for as long as two hours to accommodate a personal emergency provided immediate action is taken to obtain a replacement. A designated Senior Reactor Operator must be present in the reactor room or readily available on call. Readily available on call means that the Senior Reactor Operator has been specifically designated and the designation is known to the operator on duty, the Senior Reactor Operator keeps the operator on duty informed of how to be contacted, and the Senior Reactor Operator is capable of reaching the facility within a reasonable time [30 minutes or within a 15-mi (24-km) radius] under normal conditions.

TS 6.1.3(2) requires a list of facility personnel (management, radiation safety and other operations personnel) by name and phone number be readily available for use by the reactor operator. Certain events require the presence at the facility of a Senior Reactor Operator [TS 6.1.3(3)]. These events are initial startup and approach to power, all fuel or control-rod relocations within the reactor core region, and recovery from unplanned or unscheduled shutdowns or significant power reductions.

12.2 Training

Most of the training of reactor operators is done by the in-house personnel. TS 6.1.4 requires training and requalification of personnel to be in compliance with 10 CFR Part 55. Additional guidance for selection, training and requalification of operators used by the licensee is found in

ANSI/ANS 15.4 - 1988, “Selection and Training of Personnel for Research Reactors.” As part of the license renewal the staff reviewed the “Omaha Veterans Administration Medical Center TRIGA Reactor Requalification Plan.” The plan discusses the schedule of training, lectures and written examinations, on the job training, oral and operating examinations, document review requirements, overall evaluation of operators, absence from licensed activities, exemptions to the program, recordkeeping, and administration of the program. The staff concludes that it meets the applicable regulations of 10 CFR Part 55, and follows the guidelines of ANSI/ANS 15.4.

12.3 Operational Review and Audits

The Reactor Safeguards Committee (RSC) provides independent review and audit of facility activities reporting to Level 1 management (See figure 12.1). The requirements for the Committee are contained in TS 6.2. The members collectively represent a broad spectrum of expertise in the appropriate reactor technology. RSC members may be from within or outside the operating organization and are appointed to the Committee by Level 1 management. The RSC has a minimum of four members with the Associate Chief of Staff for Research as the Chairman, the Radiation Safety Officer as an ex-officio member and the Reactor Director/Supervisor as a member. Persons on the RSC will have at least five years of professional work experience in their discipline or specific field represented on the Committee. Qualified and approved alternate members may serve on a temporary basis. However, no more than two alternative members may participate on a voting basis at any one time.

The operations of the RSC are in accordance with an established charter. The Committee meets at least once per calendar year and more often as needed to effectively monitor facility activities. A quorum is not less than one-half the membership where the operating staff does not constitute a majority of those present. The use of subgroups is discussed in the RSC charter. Meeting minutes are reviewed, approved and disseminated within a month following RSC meetings.

The following items are reviewed by the RSC:

- Determinations that proposed changes in the facility and to procedures as described in the safety analysis report, and tests or experiments not described in the safety analysis report do not meet the criteria of 10 CFR 50.59(c)(2). The criteria determine when a license amendment would be needed for a change, test or experiment.

- New procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

- New experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

- Proposed changes in the TSs and license.

- Violations of TSs, license or internal procedures or instructions having safety significance.

- Operating abnormalities having safety significance.

- Reportable occurrences.

- Audit reports.

The audit function of the RSC includes selective but comprehensive examination of operating records, logs and other documents. Audits can include discussions with cognizant personnel and observation of operations. The following items are audited:

- Facility operations for compliance to the TSs and applicable license conditions are audited annually.

- The requalification program for the operating staff is audited at least every other calendar year.

- Results of action taken to correct deficiencies that may occur in reactor facility equipment, systems, structures, or methods of operation that affect reactor safety are audited at least once per calendar year.

- The reactor facility emergency plan and implementing procedures are audited at least once every other calendar year.

Deficiencies that are uncovered by an audit that affect reactor safety are immediately reported to Level 1 management. Written reports of audit findings are submitted to Level 1 management and the RSC within three months after completion of the audit.

12.4 Procedures

The licensee has developed a comprehensive set of written operating procedures for all aspects of facility operation as required by TS 6.4. These procedures address (1) startup, operation and shutdown of the reactor; (2) fuel loading, unloading, and movement within the reactor; (3) maintenance of major components of systems that could have an effect on reactor safety; (4) surveillance checks, calibrations, and inspections required by the TSs or those that may have an effect on reactor safety; (5) personnel radiation protection, consistent with applicable regulations or guidelines and that include commitment and programs to maintain exposures and releases ALARA; (6) administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity; (7) implementation of the emergency plan and security procedures; (8) use and shipment of byproduct material; and (9) any additional plans that may be deemed necessary for operation of the facility.

Substantive changes to procedures require documented review by the RSC and approval by the Reactor Director/Supervisor. Minor modifications to procedures that do not change their original intent (and may be made under 10 CFR 50.59) may be made by the Reactor Director/Supervisor, but need to be approved by the RSC within 14 days.

12.5 Required Actions

Certain events require specific licensee actions in accordance with TS 6.6. If the reactor fuel element safety limit is exceeded, the reactor is shut down and the violation is reported to the Reactor Director/Supervisor and the Nuclear Regulatory Commission by telephone to the Operations Center no later than the following working day. Reactor operation will not resume until authorized by the Nuclear Regulatory Commission. A written report is submitted to the Nuclear Regulatory Commission within 30 days of the violation. The report describes the applicable circumstances leading to the violation including, when known, the cause and contributing factors, the effects of the violation on reactor facility components, systems or structures and on the health and safety of the public, and corrective action taken to prevent recurrence. The report is reviewed by the RSC and any follow-up report is submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation.

The licensee is also required by the TSs to take specific action if there is a release of radioactivity from the site above allowed limits or if a reportable event occurs. Reportable events are any of the following:

- Operation with actual safety system settings for required systems less conservative than the limiting safety system settings specified in the TSs.

- Operation in violation of limiting conditions for operation established in the TSs unless prompt remedial action is taken.

- A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdown.

- An unanticipated or uncontrolled change in reactivity greater than 1.00$ except for reactor scrams resulting from a known cause.

- Abnormal and significant degradation in reactor fuel or cladding, or both, or coolant boundary (excluding minor leaks) which could result in exceeding prescribed radiation exposure limits of personnel or environment, or both.

- An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations.

If any of these events occur, reactor conditions are returned to normal or the reactor is shut down. If the reactor is shut down to correct the situation, the reactor will not be restarted without authorization of the Reactor Director/Supervisor. The occurrence is reported to the Reactor Director/Supervisor and to the Nuclear Regulatory Commission. Notification to the Nuclear Regulatory Commission is by telephone to the Operations Center no later than the following working day with a written report to the Document Control Desk within 30 days. The occurrence is also reviewed by the RSC at their next meeting.

12.6 Reports and Records

The TSs require the licensee to make routine and special reports to the Nuclear Regulatory Commission. Some of the special reports (event reports) were discussed above. The other special report requirement in TS 6.7.2 (2) is for the licensee to report in writing to the Document Control Desk within 30 days permanent changes in the facility organization involving Level 1 and 2 personnel and significant changes in the Safety Analysis Report.

The routine report submitted by the licensee is an annual report for the previous calendar year submitted by March 31 to the Document Control Desk. The report contains the following information:

- A narrative summary of reactor operating experience including the energy produced by the reactor.

- The unscheduled reactor shutdowns including, where applicable, corrective action taken to preclude recurrence.

- A tabulation of major preventative and corrective maintenance operations having safety significance.

- A brief description, as required by 10 CFR 50.59, of any changes, tests and experiments, including a summary of the evaluation of each.

- A summary of the nature and amount of radioactive effluents released or discharged to environs beyond the effective control of the licensee as determined at or before the point of release or discharge. The summary includes to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution is less than 25 percent of the concentration allowed, a statement to that effect is sufficient.

- A summarized result of environmental surveys, if any, performed outside of the facility.

- A summary of exposures received by facility personnel and visitors if greater than 25 percent of that allowed.

In addition to the requirements of the regulations, TS 6.8 contains requirements for records retention. Records to be retained for the life of the facility include:

- Gaseous and liquid radioactive effluents released to the environs.

- Radiation exposure for all personnel monitored.

- Drawings of the reactor facility.

Records that are retained for a period of at least five years (or for the life of the component if less than five years) include:

- Normal reactor facility operation (not including supporting documents which are retained for one year or one inspection cycle, whichever is longer).
- Principal maintenance operations.
- Reportable occurrences.
- Surveillance activities required by the TSs.
- Reactor facility radiation and contamination surveys where required by applicable regulations.
- Experiments performed with the reactor.
- Fuel inventories, receipts and shipments.
- Approved changes in operating procedures.
- Records of meeting and audit reports of the RSC.

In addition, records of retraining and requalification of licensed operators are maintained at all times the individual is employed or until the individual's license is renewed.

12.7 Physical Security

The licensee has requested that the license condition, 2.C.(3) requiring that a physical security plan be maintained be deleted as part of this license renewal. The regulations in 10 CFR 73.67 (c)(1) require facilities to maintain a physical security plan when they possess special nuclear material of moderate strategic significance or 10 kg or more of special nuclear material of low strategic significance. The licensee's requested possession limits are less than these amounts. Nonetheless, because the reactor license authorizes possession of special nuclear material of low strategic significance, the licensee must maintain security in accordance with the provisions of 10 CFR 73.67(f), "Fixed Site Requirements for Special Nuclear Material of Low Strategic Significance."

The licensee is authorized to possess up to 3.3 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 in the form of reactor fuel; up to 20 grams of contained uranium-235 of any enrichment in the form of fission chambers; and such special nuclear material as may be produced by the operation of the facility.

The definition of Special Nuclear Material of Low Strategic Significance in 10 CFR 73.2 is less than an amount of special nuclear material of moderate strategic significance. Special Nuclear Material of Moderate Strategic Significance is defined as (1) more than 1000 grams of uranium-235 contained in uranium enriched to 20 percent or more of the uranium-235 isotope, or more than 500 grams of uranium-233 or plutonium, or in a combined quantity of more than 1000 grams when computed by the equation, grams = (grams contained U-235) + 2 (grams U-233 + grams plutonium) or (2) 10 kilograms or more of uranium-235 contained in uranium enriched to 10 percent or more but less than 20 percent of the uranium-235 isotope.

The licensee's possession limits fall within the definition of Special Nuclear Material of Low Strategic Significance. The 3.3 kilograms of contained uranium-235 enriched to less than 20 percent in the isotope uranium-235 is less than the 10-kilogram limit. The reactor produces a small amount of plutonium during operation. Given the license limit of 20 grams of contained uranium-235 of any enrichment (assumed to be over 20 percent enriched in the uranium-235 isotope), operation of the reactor can produce 490 grams of plutonium before the 1000 gram limit computed by the equation above would be violated. However, some of the plutonium produced by the reactor is also consumed as fuel. The licensee has produced less than one gram of plutonium in 42 years of reactor operation so it is unlikely that 490 grams of plutonium will be produced over the 20-year term of the license renewal. However, the licensee and the Nuclear Regulatory Commission inspection program will monitor special nuclear material levels to ensure that the licensee's material continues to meet the definition of Special Nuclear Material of Low Strategic Significance.

The regulations in 10 CFR 73.67(f) require licensees to (1) store or use material only within a controlled access area, (2) monitor with an intrusion alarm or other device or procedures the controlled access areas to detect unauthorized penetrations or activities, (3) assure that a watchman or offsite response force will respond to all unauthorized penetrations or activities, and (4) establish and maintain response procedures for dealing with threats of thefts or thefts of this material. The licensee shall retain a copy of the current response procedures as a record for three years after the close of period for which the licensee possesses the special nuclear material under each license for which the procedures were established. Copies of superseded material must be retained for three years after each change. The licensee is aware of these requirements. The Nuclear Regulatory Commission inspection program will verify that these requirements are met.

12.8 Emergency Planning

Regulations in 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 to 10 CFR Part 50. As part of the original application for license renewal, the licensee referred the staff to the existing emergency plan. Consequently, the licensee submitted an updated emergency plan dated November 24, 1999. The plan was reviewed against NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors." The licensee maintains an acceptable Emergency Plan that complies with the requirements of Appendix E to 10 CFR Part 50, following the guidance of Regulatory Guide 2.6, and ANSI/ANS 15.16, "Standard for Emergency Planning for Research Reactors."

12.9 Conclusions

On the basis of the above discussions, the staff concludes that the licensee has sufficient experience, management structure, and procedures to provide reasonable assurance that the reactor will continue to be managed in a way that will cause no significant risk to the health and safety of the public. The staff has reviewed the licensee's proposed organization, training, operational review and audits, procedures, required actions, and records and reports against the guidance given in ANSI/ANS-15.1-1990, "American National Standard for the Development of

Technical Specifications for Research Reactors," which is supported by the NRC staff for the conduct of operations. The licensee's proposed conduct of operations in these areas is consistent with the guidance of the standard and is therefore, acceptable to the staff.

The staff concludes that the removal from the license of the requirement that the licensee maintain a physical security plan is in accordance with the regulations and that the licensee will maintain physical security in accordance with the requirements of 10 CFR 73.67(f).

The staff concludes that the licensee's emergency plan meets the requirements of the regulations in 10 CFR Part 50, Appendix E and is therefore acceptable.

Figure 12.1
Facility Organization

13.0 ACCIDENT ANALYSIS

To help establish safety limits, limiting safety system settings and limiting conditions for operation of the reactor, the licensee analyzed potential reactor transients and other hypothetical accidents. The licensee's analysis has included the potential effects of natural hazards as well as potential accidents involving the operation of the reactor. Specifically, the licensee analyzed the potential effects of such events on the reactor fuel and the health and safety of the public. The staff then evaluated the licensee's analytical assumptions, methods, and results, and added some considerations of its own, as discussed below. In addition, the NRC staff has obtained independent analyses of accidents with TRIGA-fueled reactors (NUREG/CR-2387), and has compared those results with accidents analyzed by the licensee.

None of the credible accidents postulated would lead to the failure of the cladding of any fuel pins or the uncontrolled release of fission products. However, the licensee postulated an enveloping event involving the rupture of the cladding of an irradiated fuel pin in air which is the standard enveloping event for TRIGA design research reactors. This event would lead to the maximum potential radiation hazard to facility personnel and members of the public. The licensee makes no assumptions as to the cause of the failure. The licensee evaluated only the potential consequences of this event, not the likelihood or mechanisms of the event's occurrence. This worst-case scenario for this accident has been designated as the maximum hypothetical accident (MHA) and for a TRIGA reactor, for purposes of classification, is referred to as the "fuel-handling accident." The licensee and the staff have evaluated other possible accident sequences that originate in the intact reactor core; none pose a significant risk of cladding failure or release of fission products. If this cladding were ruptured, noble gases and halogen fission products could escape. An MHA is defined as a postulated accident with potential consequences greater than those from any event that can be mechanistically analyzed. Thus, the staff assumes that the accident occurs but does not attempt to describe or evaluate deterministically the mechanical details of the initiation of the accident or the probability of its occurrence. Only the consequences are considered.

The following potential accidents were considered for evaluation and analysis:

- MHA (fuel handling accident)
- natural phenomena
- rapid insertion of reactivity (a nuclear excursion)
- loss of coolant
- misplaced experiments
- mechanical rearrangement of fuel

13.1 MHA (Fuel Handling Accident)

This potential accident covers various incidents to one or more fuel elements in which the fuel cladding might be breached or ruptured. The worst case scenario assumes that the cladding of the most irradiated fuel element fails in the reactor room air after a long run at full licensed power so that the inventories of all radionuclides of significance in the scenario are at their maximum values. The licensee did not try to develop a detailed mechanistic scenario, but assumed that the cladding of one fuel element fails and that all of the fission products accumulated in the gap are released abruptly. Thereupon, these nuclides would diffuse in the ambient air (reactor room) or water (reactor pool), depending on the location at the time of clad failure.

Several series of experiments at General Atomics have obtained data on the species and fractions of fission products released from U-ZrH_X under various conditions (Simnad et. Al., 1976; Foushee and Peters, 1971; Baldwin, Foushee, and Greenwood, 1980). The noble gases were the principal species found to be released, and, when the fuel specimen was irradiated at temperatures below about 350 °C (662 °F), the fraction of the total inventory that was released could be summarized as a constant equal to 1.5×10^{-5}, independent of operating temperature or operating history. Given the temperature of irradiation, this release fraction could be reasonably applied to TRIGA reactors operating up to about 800 kW steady state power which includes the licensee's reactor at 20 kW steady state power. Because the noble gases do not condense or combine chemically, it is assumed 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 (356 °F). Therefore, some of the radioiodines will be trapped by materials with which they come in contact, such as water, and reactor or building structures. In fact, evidence indicates that most of these iodines will either not become or not remain airborne under many accident scenarios applicable to non-power reactors. However, to be certain that the fuel cladding failure scenarios discussed below led to upper limit dose estimates for all events, the licensee assumed that 100% of the iodines in the gap also become airborne. This assumption will lead to computed thyroid doses that could be a couple of orders of magnitude higher than actual doses.

The staff has reviewed various acceptable methods for computing the dose within and beyond the confines of the reactor facility in case of a fission product release and has independently calculated and confirmed the licensee's dose predictions for the accident scenarios. Three scenarios were considered by the staff and the licensee as discussed in the following sections. In appropriate cases, doses to the most exposed worker and at the location of the nearest permanent residence and the most exposed member of the public were analyzed.

13.1.1 Scenario for Failure of Fuel Element in Air

The MHA is based on a single fuel element cladding failure in air in the reactor room. The analysis is based on the following general assumptions:

- Failure occurs immediately after an extended reactor operation. The power history is 40 years at 1.5 kW to represent average actual operation history of the facility followed by 20 years at 20 kW to represent the maximum possible operation during the renewal period.

- All noble gases and halogen radionuclides in the gap, including the iodines, are released into the room air (release fraction from fuel to the gap is 1.5×10^{-5}).

- The radionuclides released are uniformly instantaneously distributed throughout the reactor facility volume of 7.1×10^{8} cm^{3}.

- A core array of 57 fuel elements with the failed element leading the average core power per element by a factor of 2 (0.70 kW).

- No decrease in source strength resulting from radioactive decay occurs during the exposure period.

- Inhalation dose conversion factors are taken from Federal Guidance Report No. 11.

For the analysis of the most exposed worker the following additional assumptions are made in addition to the general assumptions given above:

- The most exposed worker is exposed to the initial concentration of radionuclides for one hour before exiting the reactor room. This is equivalent to the reactor laboratory and radio chemical hood exhaust fans being shut down, and the damper in the air supply system closed, so that the airborne radioactivity is confined to the reactor room.

- Doses are delivered by inhalation of iodines and by immersion in a finite-sized room containing the noble gases.

For the analysis of the most exposed member of the public and the nearest permanent residence the following additional assumptions are made in addition to the general assumptions given above.

- The ventilation system is in operation. The air exhaust fan does not shut down and the air supply system damper does not close. However, it is assumed that all of the reactor room air is exhausted at the exhaust fan. The event is assumed to only involve a ground level release.

- The most exposed member of the public is 102 m (335 ft) and the nearest permanent residence is 158 m (520 ft) from the release point.

- The exposure from the release lasts an hour.

- Horizontal and vertical diffusion coefficients were estimated from the curves in RG 1.145. Doses were calculated using methods in RG 1.109.

The calculated dose to the most exposed worker is given in Table 13-1. Exposures were on the order of about 1 mrem deep dose equivalent, 720 mrem committed dose equivalent to the thyroid, and 23 mrem committed effective dose equivalent. The calculated dose to the most exposed member of the public and at the nearest permanent residence is given in Table 13-2. The dose was on the order of one mrem.

The location of the most exposed member of the public was the nearest point off the hospital site. Because the licensee can quickly control persons on their site, calculations were not performed for persons on site. However, for the license renewal issued in 1983, a fuel clad failure in air was also analyzed with different assumptions (the most significant being reactor power at 18 kW and an exposure time of five minutes which is the time for the reactor ventilation to vent the reactor room). Doses were calculated for a person 33 ft (10 m) from the ground-level release point. The doses were calculated as 1.6 mrad beta dose, 9.6 mrad gamma dose, and 0.45 rem thyroid dose commitment. These doses are within the limits for radiation dose to members of the public given in the “old” Part 20 which has historically applied to this research reactor (10 CFR 20.1 through 20.602 and Appendixes) (See Section 1.1 of this SER). See Section 14 of NUREG-0998, “Safety Evaluation Report Related to the Renewal of the Operating License for the Research Reactor at the Omaha Veterans Administration Medical Center,” dated July 1983 for additional details concerning this calculation.

13.1.2 Failure of Fuel–Cladding Within Pool Water

The licensee also analyzed a fuel-cladding failure with the fuel rod in the reactor core under water. It was assumed that almost all of the noble gases escape from the pool water, but that none of the iodine isotopes do. The other assumptions are effectively the same as for the previous sections. The results are given in Table 13.3.

13.1.3 Conclusions

In accordance with the discussions and analysis above, the staff concludes that if one fuel element from the reactor were to release all the noble gaseous and iodine fission products accumulated in the fuel cladding gap, radiation doses to both occupational personnel and to the public in unrestricted areas would be far below the limits of 10 CFR Part 20. This conclusion is valid even for the very unlikely accident scenario selected, namely, that the clad failure occurs immediately after an extended full power operation, that all of the gap radioactivity, including all iodines, is immediately dispersed uniformly within the reactor room, and the maximum exposed individual in the unrestricted area remains 102 m (335 ft) from the release point and is exposed for one hour, which is a greater period of exposure time than would happen in reality. The staff further notes that the assumptions and methods used in the calculation (e.g., the licensee assumed a semi-infinite cloud of released noble gas) are very conservative. The results at the location of the nearest private residence are also based on one hour of exposure, but the released cloud would pass that point well within that time frame. In addition, radiation doses to persons on the hospital grounds 33 ft (10 m) from the ground-level release point would be within the limits of the “old” 10 CFR Part 20 historically used for accident acceptance criteria at this reactor.

The staff compared the licensee’s results with its estimates based on more realistic finite sized plumes of airborne radioactivity, which confirms that the licensee’s results are conservative. The staff also concludes that the licensee has the capability to evaluate airborne releases of radioactive materials.

13.2 External Events

The licensee has considered the potential effects of natural phenomena such as earthquakes and tornadoes on the reactor and concluded that the hazards to the reactor are not radiologically significant. The area is seismically stable, characterized by earthquakes of low intensity as well as low frequency. Examination of seismic events since 1867 indicates no significant damage. Tornadoes are more frequent than earthquakes; however, the fact that the reactor is in the basement of the hospital building and is surrounded by poured concrete walls with no windows and with 3 to 4 in (7 to 10 cm) of concrete overhead makes tornado damage very unlikely. The staff agrees with the licensee's conclusion that the hazards from these natural phenomena are not significant.

In the case of failure of electric power, the control rod electromagnets will be deenergized and the rods will fall into the core, shutting the reactor down. The staff concludes that loss of electric power will lead to a safe shutdown of the reactor with no impact on the facility staff or the public.

As discussed in Section 2, a low altitude airway passes near the vicinity of the hospital. However, the probability of an aircraft striking the hospital is very low. Because of the location of the reactor facility in the basement of the hospital and the reactor core near the bottom of an in-ground tank, the possibility of damage is low. The staff concludes that the hazards from aircraft are not significant.

13.3 Rapid Insertion of Reactivity (Nuclear Excursion)

The maximum power excursion (transient) that could occur would be one resulting from the inadvertent rapid insertion of the total available excess reactivity. The TRIGA fuel loading is limited by TS 3.1.1 to 1.00$ excess reactivity above clean-cold critical. The staff believes that this is a reasonable limitation based on operational experience. Because TS 3.7.1 limits the reactivity worth of a single experiment to 1.00$, it is conceivable that a step reactivity insertion of 1.00$ can be obtained. Although neither the licensee nor the staff has been able to postulate a credible mechanism that would result in a step insertion that is rapid enough to cause a transient based on prompt neutrons alone, it has been assumed for purposes of the analysis that such an event does occur. The staff notes that the reactor is neither authorized to nor equipped for pulse mode operation.

A failure of the recorder could lead to an inadvertent withdrawal of the regulating rod. The typical worth of the regulating rod is 0.50$, which is less than the reactivity worth of experiments. Therefore, rapid addition of this reactivity will be bounded by the analysis of the rapid addition of experiment reactivity. A slower addition of the reactivity would result in power increasing on a short period. The reactor safety system would initiate a power level scram at a power level of 20 kW. Power overshoot (which occurs during the time needed for the safety system to drop the control rods into the core) would be minimal. Similar TRIGA reactors are licensed to power levels of 300 kW.

General Atomics demonstrated by experimentation with the prototype Torrey Pines TRIGA reactor (GA-0531, 1958; GA-0722, 1959) that the insertion of 2.00$ excess reactivity caused no damage to the reactor nor any significant radiation exposure to individuals near the reactor or in the surrounding area. This reactivity insertion yielded a reactor period of 10 msec and a peak power of approximately 250 MW. The prompt, negative temperature coefficient of the reactor fuel terminated the transient and limited the total energy release from the transient to a level that caused no fuel damage. Within 30 seconds after initiation of the transient, the reactor power level had returned to a quasi-equilibrium level of 200 kW. The maximum fuel temperature was about 360 °C (680 °F), well below the temperature (550 °C or 1022 °F) at which fuel with a hydrogen-to-zirconium ratio of 1.0 undergoes a phase transition (GA-4314, 1980). Extensive further experience with TRIGA reactors has confirmed the inherent safety of the reactors under transient mode operation. Thus, if all of the excess reactivity (1.00$) authorized for the reactor were inserted rapidly, the resultant transient would not approach those that have been demonstrated as safe for routine transient mode operation of other TRIGA-type reactors with similar fuel.

On the basis of the above considerations presented by the licensee and extensive experience with other TRIGA reactors, the staff concludes that there is no credible nuclear excursion possible with the reactor 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 a reactor transient.

13.4 Loss of Coolant

Because there are a number of floors in the hospital building immediately above the reactor that are normally occupied, the loss of the coolant which acts as a radiation shield in the vertical direction could result in potential radiation exposure to occupants of these areas. In addition, such a loss of coolant would result in increases in temperatures of the fuel and cladding. The licensee's analysis indicates that the loss of coolant accident can occur by only two mechanisms; the tank may be pumped dry; or a tank failure may allow the water to drain into the soil.

The tank outlet water line extends only 3 ft (1 m) below the normal water level. Therefore, even if the water system is operated inappropriately – if, for example, it is operated when the pump discharge line has been disconnected for repairs, the tank cannot be accidentally pumped dry. In the event that it is necessary to drain the tank, the fuel will first be removed in shielded casks. The recirculating pump does not have sufficient suction head to drain the tank, so another more powerful pump would be required, with a temporary suction line inlet below the core.

Tank failure could possibly be caused by a severe earthquake or major settling of the building foundation. As noted in Section 13.2, the hazard from earthquakes is not significant. At the time of construction of the reactor facility, there was no evidence of foundation failure during the previous eight years of the building's existence. Subsequent examination of the reactor tank has shown no evidence of deterioration. As described in Section 4.2, the reactor tank has five barriers that prevent coolant leakage from the tank. Two of these barriers are waterproof - the epoxy resin coating and the welded steel tank. The other three barriers (gunite, reinforced concrete, and the surrounding soil) would present a very high resistance to water leakage. The core drilling made at the reactor location before construction shows the soil to be clay silt and glacial clay, both of which are almost impervious to water flow (Abbot, 1956).

Even though the likelihood of a loss of shielding water is considered to be very small, the licensee performed detailed calculations to evaluate the radiological hazard associated with this accident. The ORIGEN code was used to model the gamma ray source strength, and the MCNP code was used to model gamma ray transport from the core to various locations in the reactor room and hospital building. If the reactor had been operating for a long period of time (1000 hours) at 20 kW before instantaneously losing all of the shielding water, the integrated dose that an individual could receive in the first floor area immediately above the reactor would be about 12 Rem in the first hour. Prompt evacuation of the area would reduce this potential dose significantly. The radiation dose at the tank top from the unshielded core would be about ten times this, but would be collimated by the tank walls, so that workers in the reactor room could refill the tank with water, without being overexposed. Radiation levels in the reactor room from scattered radiation from the collimated beam would be about 1.5 Rem/hr one hour after the loss of coolant.

It is important to note that because of the design of the reactor tank, instantaneous loss of coolant cannot occur. To ensure that personnel would be alerted to a loss of coolant accident, a float switch is installed in the reactor tank to actuate an audible alarm located at the control console and the hospital switchboard if the water level falls to within 12 ft (3.6 m) of the top of the core. There is reasonable assurance that the switchboard operator would be able to initiate corrective action before the core would be completely uncovered.

Because the water is required for adequate neutron moderation, its removal would terminate any significant neutron chain reaction. However, the residual radioactivity would continue to deposit heat energy within the fuel. In the loss of coolant accident scenario, it is assumed that sufficient water is lost to uncover the core and that subsequent heat removal from the fuel is provided only by air convection. From the data of Table IV, TID-14844 (AEC, 1962), for example, it is estimated that the initial rate of fission product heating in an average fuel element would be less than 20 W, if loss of all water and a reactor shutdown were simultaneous and instantaneous. On the basis of this power level, it is further estimated that air circulation would prevent the temperature of any fuel element from rising more than 100 °C (212 °F). This temperature, is well below the safety limit (500 °C or 932 °F) for the reactor fuel.

On the basis of the above considerations, the licensee concluded and the staff agrees that the possibility of loss of coolant is very unlikely and that consequences would be unlikely to cause damage to the reactor or result in unacceptable radiation exposure to staff or occupants of the hospital. The staff also concludes that a rapid loss of coolant from the reactor tank following extended operation at 20 kW would not result in the melting of the fuel or cladding or loss of cladding integrity from other related causes.

13.5 Misplaced Experiments

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. In that case, the sample might become overheated or develop pressures that could cause failure of the experiment container. As discussed in Section 10, all new experiments are reviewed before insertion, and all experiments in the region of the core are isolated from the fuel cladding by at least one barrier - such as the pneumatic transfer tube, the central thimble, or the reflector can. TS limits control experiment content and radioactive material content.

The staff concludes that the experimental facilities, procedures for experiment review and TS limits at the reactor facility 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 exceed the limits in 10 CFR Part 20 (“new” Part 20 because experiments are considered part of current operation).

13.6 Mechanical Rearrangement of the Fuel

This type of potential accident would involve the failure of some reactor system or could involve an externally originated event which disperses the fuel and, in so doing, breaches the cladding of one or more fuel elements. The staff has not developed scenarios for accidents such as these. Thus, there is no basis for deciding if any arbitrary scenario is credible. Based on the licensee's calculations of the results of the failure of one fuel element under water, even if the entire core was damaged under water, the doses to the licensee's staff and public would be within "old" 10 CFR Part 20 limits.

The configuration of fuel elements in a TRIGA core is optimum for maximum reactivity. Therefore, any arbitrary accidental rearrangement would be likely to decrease available reactivity. The staff concludes that no accidental mechanical rearrangement that is credible would lead to an event with more severe consequences that those accidents considered in Sections 13.1 or 13.3.

13.7 Conclusions

The staff has reviewed the licensee's analyses of potential accidents at the reactor facility. The staff concludes that the licensee has postulated and analyzed sufficient accident initiating events and scenarios to demonstrate that the reactor is designed acceptably to avoid inadvertent reactor damage that could prevent safe shutdown, so there is reasonable assurance that no credible accident would cause significant or undue radiological risk to the facility staff, the environment, or the public.

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Table 13.1

Doses to the Most Exposed Worker
due to the MHA, failure of fuel clad in air (in mrem)

DDE[a] (Noble Gases)	CDE[b] Thyroid	CEDE[c] (Iodine)
0.06	720	23

a. Deep Dose Equivalent, Immersion In Finite Room
b. Committed Dose Equivalent
c. Committed Effective Dose Equivalent

Table 13.2

Doses at the location of the Most Exposed Member of the Public (MMP)
and the Nearest Permanent Residence (NPR) (in mrem), due to MHA

	DDE[a] (Noble Gases)	CDE[b] (Iodine)
MMP	0.0035	0.63
NPR	0.0015	0.27

a. Deep Dose Equivalent, Immersion in Semi-infinite Cloud
b. Committed Dose Equivalent

Table 13.3

Doses to Most Exposed Worker (MEW), Most Exposed Member of the Public (MMP),
and at the location of the Nearest Permanent Residence (NPR) (in mrem),
due to failure of fuel clad in water

	DDE (Noble Gases)
MEW	0.06 a
MMP	0.0021 b
NPR	0.00087 b

a. Deep Dose Equivalent, Immersion In Finite Room
b. Deep Dose Equivalent, Immersion in Semi-infinite Cloud

14.0 TECHNICAL SPECIFICATIONS

The licensee's TSs evaluated in this licensing action define certain features, characteristics, and conditions governing the continued operation of this facility. These TSs are explicitly included in the renewal license as Appendix A. Formats and contents acceptable to the NRC have been used in the development of these TSs, and the staff has reviewed them using ANSI/ANS-15.1-1990 and NUREG-1537 as guides. In addition to the written application, on November 28, 2001, the NRC Project Manager and AJBRF Reactor Director/Supervisor discussed seven minor changes to the TSs to correct errors in format and grammar. These changes did not affect the meaning of any of the TSs involved. On May 9, 2002, the NRC Project Manager and AJBRF Reactor Director/Supervisor discussed and agreed to a change to TS 6.8.1. The records retention period for reportable occurrence is changed from five years to lifetime of the facility to be consistent with 10 CFR 50.36(C)(2). The corrections are reflected in the TSs issued with the renewal license amendment.

On the basis of its review, the staff finds that the TSs provide acceptable assurance that the assumptions and conditions of the licensee's safety analysis will be met. Facility operation within the limits of the TSs will not result in offsite radiation exposures in excess of 10 CFR Part 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 accident events.

15.0 FINANCIAL QUALIFICATIONS

The Nebraska - Western Iowa Health Care System is part of the Department of Veterans Affairs which is under the Executive Branch of the Federal Government. It operates on an annual allocation of funds governed by the Bureau of the Budget.

The TRIGA reactor is operated by the Nebraska - Western Iowa Health Care System. The Chief Executive Officer has committed to supporting the reactor program. Therefore, 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).

16.0 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 accident event could cause some exposure. The maximum hypothetical accident would not lead to a dose to the most exposed individual, either occupational or member of the public, greater than applicable guidelines or regulations ("old" 10 CFR Part 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 the staff must also consider whether operation will cause significant degradation in safety features. Furthermore, because loss of integrity of fuel cladding is the MHA, the staff has considered mechanisms which could increase the likelihood of failure. Possible mechanisms are: (1) radiation degradation of fuel element 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.

16.1 Fuel Element Aging

The aluminum-clad low-hydride TRIGA fuel in the core has been in use since 1959 and has been subjected to low burnup levels of U-235. 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. Because the reactor operates at a maximum power level of 20 kW, the temperature of the fuel does not exceed 100 $^\circ$C (212 $^\circ$F) during normal operation. At this temperature, the pressure of the air and/or hydrogen within the cladding does not increase significantly.

Fuel aging should be considered normal with use of the reactor and is expected to occur gradually. There is some evidence that the U-ZrH_x fuel tends to fragment with use, probably as a result of the stresses caused by high temperature gradients and high rate of heating during pulsing (GA-4314, 1980) (ref. 8). Some of the possible consequences of fragmentation are a decease in thermal conductivity across cracks, leading to higher central fuel temperatures during steady-state operation (temperature distributions during pulsing would not be affected significantly by changes in conductivity because a pulse is completed before significant heat redistribution by conduction occurs), and more fission products would be released into the cracks in the fuel.

With regard to the first item above, hot cell examinations of thermally stressed hydride fuel bodies have shown relatively widely spaced radial cracks that would cause minimal interference with radial heat flow (GA-4314, 1980). However, after pulsing, TRIGA reactors have exhibited small increases in both steady state fuel temperatures and power reactivity coefficients. At power levels of 500 kW, temperatures have increased by approximately 20 $^\circ$C (36 $^\circ$F) and power reactivity coefficients by approximately 20% (GA-5400, 1965). General Atomics has attributed these changes to an increased gap between the fuel material and cladding caused by rapid fuel expansion during pulse heating, which reduces the heat transfer. Experience has shown that the observed changes occur mostly during the first several pulses and have essentially saturated after 100 pulses. Because the reactor is not operated in the pulse mode, changes in steady state fuel temperature and power reactivity coefficients are not an issue for continued operating of the reactor with its current fuel inventory.

Two mechanisms for fission product release from TRIGA fuel have been proposed (GA-4314, 1980). The first mechanism is fission fragment recoil into gaps within the fuel cladding. This effect predominates up to about 400 °C (752 °F) and is independent of fuel temperature. The licensee's operating fuel temperatures have never exceeded 400 °C (752 °F); thus, this will be the main effect. General Atomics 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 the existing gap at some other location. Such a narrowing allows more fission fragments to traverse the gap and become embedded in the fuel or cladding material on the other side. In a closed system in which the density of the fuel is constant, the average gap size and therefore the fission product release rate remains constant, independent of the degree to which fuel material is broken up.

Above approximately 400 °C (752 °F), the controlling mechanism for fission product release is diffusion through the ZrH, and the amount accumulated in the gap is dependent on fuel temperature and fuel surface-to-volume ratios. In the licensee's fuel this mechanism is not significant because of the low fuel temperature and low utilization of the reactor.

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, corrosion of the aluminum cladding is expected to be negligible.

The fuel is handled as infrequently as possible, consistent with periodic surveillance. Any indications of possible damage or degradation are investigated. 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 tube or the central thimble.

The staff concludes that the likely processes of aging of the U-ZrH_x fuel moderator under low power, steady-state, non pulsing operation would not cause significant changes in the operating temperature of the fuel or affect the accumulation of gaseous fission products within the cladding. The staff also concludes that there is reasonable assurance that fuel aging will not significantly increase the likelihood of fuel cladding failure, or the quantity of gaseous fission products available for release in the event of loss of cladding integrity for standard TRIGA fuel operated under the conditions of the AJBRF.

16.2 Aging of Safety Components

The reactor contains redundant safety-related measuring channels and control rods. Failure of all but one control rod would not prevent a reactor shutdown to a safe condition. In addition, safety system failures normally lead to a reactor scram. Important parameters (such as reactor power) have redundant safety systems. If a failure occurs that would prevent a scram upon receipt of a valid scram signal in one channel, the redundant channel would scram the reactor. The staff review has revealed no mechanism by which failure or malfunction of one of these safety-related components could lead to a non-safe failure of a second component.

The licensee 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 electro-mechanical instrumentation. There is no indication of significant degradation of the instrumentation, and the staff further concludes that the preventive maintenance program outlined in the licensee’s procedures would lead to adequate identification and replacement before significant degradation occurred. As an example, the control console system was replaced by a new, state-of-the-art system during the 1990s. 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 licensee 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 a component malfunction.

16.3 Conclusions

In addition to the considerations discussed above, the staff reviewed licensee event reports and NRC inspection reports. On the basis of this review and the above considerations, the staff concludes that there has been no significant degradation of equipment due to aging and that facility management will continue to maintain and operate the reactor so that there is no significant increase in the radiological risk to facility employees or the public.

17.0 CONCLUSIONS

Based on its evaluation of the application as set forth above, the staff has reached the following conclusions:

- The application for renewal of Operating License No. R-57 for its research reactor filed by the Department of Veterans Affairs, Nebraska - Western Iowa Health Care System dated May 10, 1993, 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.

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

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

- 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.

- 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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