NUREG-1283

Safety Evaluation Report
related to the renewal of the operating license for the research reactor at Purdue University

Docket No. 50-182

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1988

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555

2. The Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082

3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances*.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Information Support Services, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG-1283

Safety Evaluation Report
related to the renewal of the operating license for the research reactor at Purdue University

Docket No. 50-182

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1988

ABSTRACT

This Safety Evaluation Report for the application filed by Purdue University for a renewal of Operating License R-87 to continue to operate a research reactor has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is owned by Purdue University and is located on the campus in West Lafayette, Indiana. On the basis of its technical review, the staff concludes that the reactor facility can continue to be operated by the university without endangering the health and safety of the public or the environment.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

FIGURES

TABLES

1 INTRODUCTION

Purdue University (Purdue/licensee) submitted a timely application to the U.S. Nuclear Regulatory Commission (NRC/staff) for renewal of the Class 104 Operating License (R-87) for its open-pool-type research and training reactor. The application, with supporting documentation, was transmitted by letter dated June 30, 1986, as supplemented, requesting renewal of the license for a period of 20 years. The licensee is permitted to operate the reactor within the conditions authorized in the existing license, as amended, in accordance with Title 10 of the Code of Federal Regulations, Paragraph 2.109 (10 CFR 2.109), until NRC action on the renewal request is completed.

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

The application also includes a revised Final Safety Analysis Report (FSAR), information for an environmental impact assessment, financial information, an Operator Requalification Program, and revised Technical Specifications. Supplemental information included revisions to the Purdue University Physical Security Plan, which is withheld from public disclosure in accordance with 10 CFR 2.790.

The staff's technical review with respect to issuing a renewal operating license to Purdue was based on visits to the facility and on the information contained in the renewal application and supporting documents, plus responses to requests for additional information. This material is available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555. This Safety Evaluation Report (SER) was prepared by R. E. Carter and A. Adams, Jr., Project Managers, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. Major contributors to the technical review include the Project Managers and C. H. Cooper and W. R. Carpenter of the Idaho National Engineering Laboratory under contract to NRC.

The purpose of this SER is to summarize the results of the safety review of the Purdue University reactor (PUR) and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. This SER will serve as the basis for renewal of the license for operation of the PUR facility at power levels up to and including 1000 W thermal (Wt). The facility was reviewed against Federal regulations (10 CFR 20, 30, 50, 51, 55, 70, and 73), applicable regulatory guides (principally Division 2, Research and Test Reactors), and appropriate accepted industry standards [American National Standards Institute/American Nuclear Society (ANSI/ANS) 15 series]. Because there are no specific accident-related regulations for research reactors, the staff has at times compared calculated hypothetical radiation dose values with related standards in 10 CFR 20, "Standards for Protection Against Radiation," for employees and the public.

The PUR was initially licensed for operation at 1.0 kWt in August 1962 as an open-pool-type reactor, with fuel of the Materials Testing Reactor (MTR) type. Only minor modifications have been made to the reactor since the initial licensing.

1.1 Summary and Conclusions of Principal Safety Considerations

The staff's evaluation considered the information submitted by the licensee, past operating history recorded in annual reports submitted to the Commission by the licensee, reports by the Commission's Region III, and onsite observations. In addition, as part of its licensing review, the staff obtained laboratory studies and analyses of credible accidents postulated for plate-type reactors. The principal safety matters reviewed for the PUR and the conclusions reached follow:

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

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

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

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

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

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

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

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

(9) The licensee's Emergency Plan provides reasonable assurance that the licensee is prepared to assess and respond to emergency events.

1.2 Reactor Description

The PUR is a heterogeneous, swimming-pool-type nonpower reactor. The core is cooled by natural convection of light water, moderated by water, and reflected by water and graphite. The reactor core is located near the bottom of a water-filled tank surrounded and supported by a concrete shielding structure. The reactor core rests on supports on the bottom of the tank, and the control mechanisms and detectors are suspended from a support plate at the top of the tank.

The reactor core is composed of approximately 16 fuel elements positioned in holes in an aluminum grid plate. The grid plate contains a rectangular matrix of holes to allow the changing of fuel element locations and the insertion of graphite reflector elements to displace reflector water. Each fuel element consists of several thin metal plates assembled into a unit about 7 cm by 7 cm with an active fuel length of ~0.60 m. Fuel elements of this general configuration were first designed for and used in the Materials Testing Reactor (MTR) and thus are referred to as MTR-type fuel elements. Three of the fuel elements were fabricated with the four middle plates missing, providing space for the positioning and movement of the reactor control rods.

Reactivity of the reactor core is changed by the operator moving the control rods that are suspended from fail-safe electromagnets. The ionization chambers used for sensing neutron and gamma-ray fluxes are suspended near the core. The control console, from which the operator can observe the reactor pool and the top structures, is located adjacent to the reactor. The control console consists of typical read-out and control instrumentation.

1.3 Reactor Location

The PUR is housed in a small room designed and dedicated for that purpose in the Duncan Annex of the electrical engineering building on the east side of the campus of Purdue University in the city of West Lafayette. The nearest larger city is Lafayette, which is located about 2 km from the site.

1.4 Shared Facilities and Equipment and Special Location Features

The electrical engineering building is in close proximity to other buildings on the campus and obtains utility services such as water, electricity, and sanitary sewage from the main campus systems. There are no special features associated with the facility location.

1.5 Comparison With Similar Facilities

The fuel used in the PUR is based on the MTR design and is very similar to the fuel used in approximately 50 other research reactors operating in the United States and at least 25 reactors operating in foreign countries. The control and instrumentation systems, although different in detail, are based on the same operating principles as those used for these other 75 research or test reactors.

1.6 Nuclear Waste Policy Act of 1982

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

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

2 SITE CHARACTERISTICS

2.1 Geography

The PUR is located near the eastern edge of the Purdue University campus, in West Lafayette, Tippecanoe County, Indiana. There are few large centers of population in Indiana, and the nearest large city is Indianapolis, approximately 100 km to the southeast of the site.

The general terrain of and around the campus and the reactor site itself are located in a relatively flat area.

The location of the PUR within the campus is shown in Figure 2.1. The nearest off-campus residential area is approximately 50 m from the reactor building. Figure 2.2 shows the location of Lafayette with respect to other major cities in Indiana, and to Chicago, Illinois.

2.2 Demography

The daytime population on the campus near the reactor site is normally less than 40,000 people, including students and university staff. Because the campus is adjacent to a residential area, the permanent population within 2 km during working hours is normally about 30,000 people. Most of the population of the Lafayette area resides within 8 km of the reactor site.

2.3 Nearby Industrial, Transportation, and Military Facilities

There is no large industry, heavily traveled transportation route, or military installation in or near Purdue, nor is there a heavily traveled airport within several kilometers.

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

2.4 Meteorology

The general climate of Tippecanoe County is continental with hot summers and cold winters. The seasons are strongly marked, and the weather is frequently changeable. The average annual temperature is about 10°C. The mean temperature in January, the coldest month, is about -4°C, and in July, the warmest month, it is 23°C. Prevailing winds are from the west or southwest during the winter and from the south during the summer. The average annual precipitation is about 0.9 m; July is the wettest month, and February is the driest. This region of the United States is subjected to tornado activity, primarily during the late spring and early summer. In Tippecanoe County as a whole, 25 tornados have been recorded during the past 35 years, with no significant damage occurring on the Purdue campus.

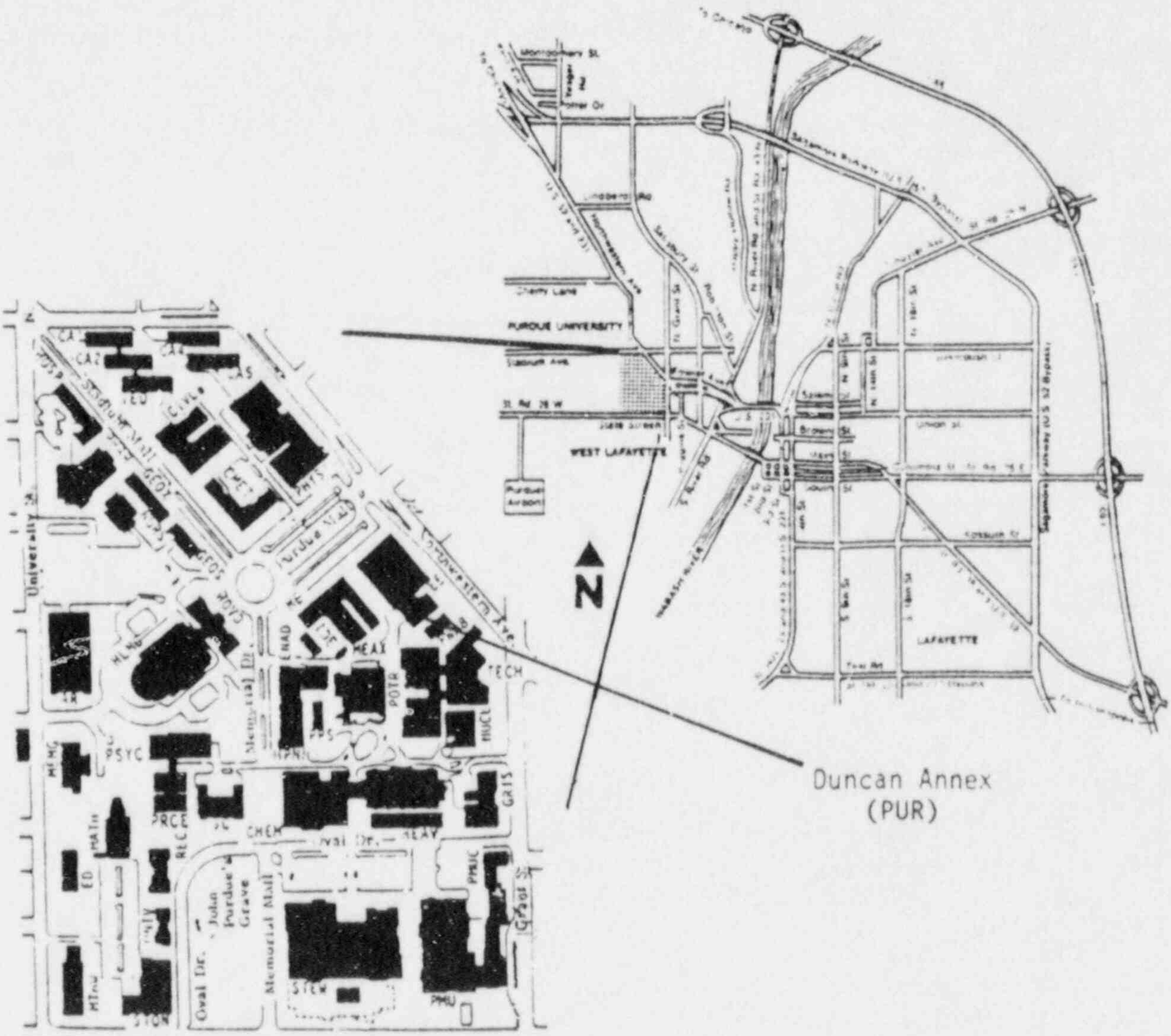

Figure 2.1 Purdue University main campus and the
Lafayette-West Lafayette vicinity

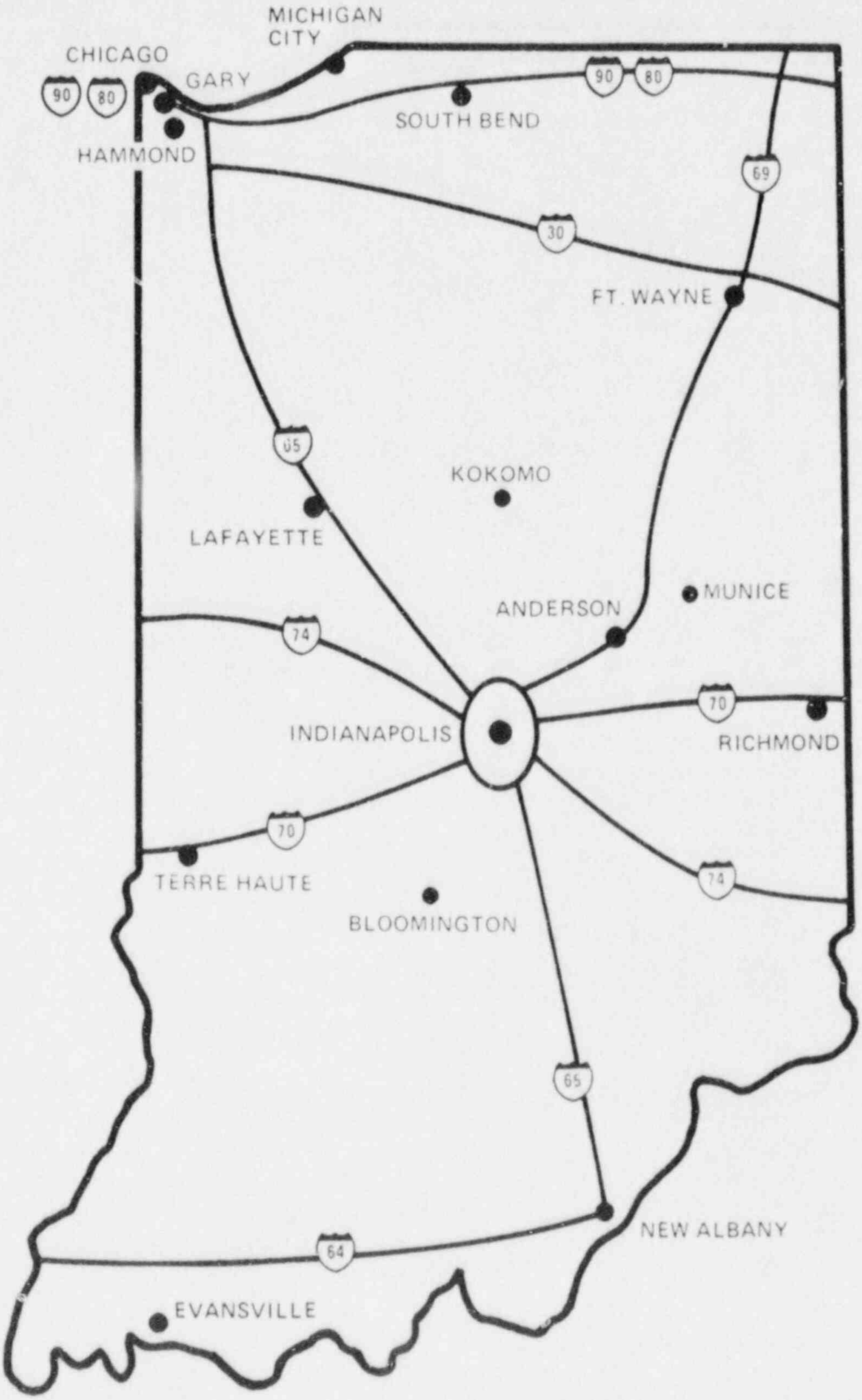

Figure 2.2 State of Indiana

2.5 Geology

The county lies within the Tipton Till Plain of Indiana and is a section of the Till Plains subprovince of the U.S. Central Lowlands physiographic province. Most of the soils in this area are derived from the glacially deposited material. Extensive upland areas are covered with a thin mantle of loose deposits. A few areas are covered with soils of alluvial, colluvial, or organic origin. Glacial drift covers the bedrock to a depth ranging from a few feet to more than 300 feet. The underlying bedrock, consisting of flint, shale, sandstone, and limestone of the Mississippian period, is exposed as rock terraces in the Wabash Valley and on the upland in the western part of the county. Purdue University is located above an extensive glacial deposit of sand and gravel.

The land surface of Tippecanoe County is flat to rolling, except where the major streams have cut deeply into the surface. The entire county lies within the drainage basin of the Wabash River and its tributaries. The land slopes generally southwestward with the streams flowing westward. Two main tributaries, the Tippecanoe River and Wild Cat Creek, enter the Wabash River upstream from the campus. Minor tributaries include Little Pine Creek, Indian Creek, Burnetts Creek, Mott's Creek, Sugar Creek, Buck Creek, Wea Creek, and Flint Creek.

2.6 Hydrology

Most of Tippecanoe County is covered by glacial drift. The drift ranges in thickness from a thin veneer to about 435 feet and was deposited on a bedrock surface that was eroded by a preglacial drainage system. Much of the surface drift consists of glacial till. Water-laid cross-bedded sand and gravel are associated with the till. The subsurface glacial deposits also include much till with interbedded sand and gravel. Locally, clay deposits are as much as 106 feet thick. Within the drift, five sheetlike water-bearing units are differentiated in parts of the county. Ground water within these units occurs under artesian and water-table conditions. Locally these may occur within the same unit.

This area was repeatedly glaciated during the Pleistocene epoch. Before glacial times, a giant drainageway, now known as the Teays River, flowed from the Appalachian Mountains across Ohio and passed northwestward through the present site of Lafayette-West Lafayette. Illinoian ice dammed the preglacial Teays River channel and ponded the relatively small glacial Lake Lafayette. An outlet channel, developed to drain this preglacial lake, was subsequently perpetuated as the present Wabash River drainage line southwestward from the Lafayette-West Lafayette area.

The elevation of the Purdue University campus is approximately 706 feet, and the level of the Wabash River is approximately 510 feet. With this difference of over 100 feet, the flow of both surface water and ground water is in a generally easterly and southerly direction toward the Wabash River, which flows around two sides of the campus.

Any leakage of contaminated water from the PUR represents no potential hazard to either the West Lafayette or Purdue University water supply, since these flows are away from the well fields of both. The Wabash River represents a natural barrier between the reactor and the Lafayette well fields, so no potential hazard exists there.

2.7 Seismology

Information on seismic activity in the Central United States (NUREG/CR-1577, "An Approach to Seismic Zonation for Siting Nuclear Electric Power Generating Facilities in the Eastern United States," May 30, 1981) shows that the PUR is located in that portion of Indiana that lies in a zone for low seismic activity, within which might result only minor damage to structures caused by distant earthquakes.

The three most significant seismic-source zones that are closest to West Lafayette are

(1) the New Madrid area of southeastern Missouri

(2) the Wabash Valley Fault system of southwestern Indiana and southeastern Illinois

(3) the Anna, Ohio area

Reasonable estimates of the maximum magnitude events that could occur in those areas give values of 7.4, 6.6, and 6.3 (body wave motion) for the seismic zones, respectively. Based on the distance from these zones (400, 200, and 200 km, respectively) and attenuation curves, estimates for peak horizontal acceleration at West Lafayette for maximum magnitude events that are likely to occur at these three seismic zones show that these events would cause insignificant damage to Purdue buildings.

The staff concludes that the history of earthquake activity with no damaging historic earthquakes near West Lafayette supports the conclusion that the risk of seismically induced hazards to the PUR is not significant.

2.8 Conclusion

The staff has reviewed and evaluated the PUR site for both natural and manmade hazards and concludes that there are no significant risks associated with the site that make it unacceptable for the continued operation of the reactor.

3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

The licensee's Safety Analysis Report provides information on the design, construction, and functions of the reactor building, reactor systems, and auxiliary systems.

3.1 Reactor Building

The Duncan Annex of the electrical engineering building is constructed of brick, concrete block, and reinforced concrete and was originally designed as a large high-voltage laboratory. It was subsequently subdivided into offices, classrooms, and laboratories. The reactor is located in the southwest corner on the ground floor in a high bay area of the building. Figure 3.1 shows the floor plan of the nuclear engineering laboratories, including the reactor room.

The outside air supply and exhaust both pass through high-efficiency particulate air filters. The reactor room is maintained at negative air pressure (minimum 0.05 inch of water). All doors to the reactor room have foam rubber seals. Steam heat is used to heat the room, and a room air conditioner circulates and cools the reactor room air.

The only floor drain to the sewers is sealed except for a vent opening. This vent is raised about 2 feet above the floor and has a filtered inverted opening. Condensate from the air conditioner is released to this drain through an opening 12.0 feet above the floor.

3.2 Wind and Water Damage

The Purdue University campus area experiences few extreme wind conditions such as tornados or inland hurricanes. Furthermore, the reactor building is constructed from concrete blocks and the reactor pool is formed of steel-reinforced poured concrete. The reactor site is well above the flood plain; therefore, wind or water damage to the PUR facility is very unlikely.

3.3 Seismically Induced Reactor Damage

The information on past seismic activity and the likelihood of future earthquakes in the area of the Purdue University campus indicates that the PUR is in a region where there is low probability of severe seismic activity. If an earthquake should cause catastrophic damage to the reactor building and/or the reactor pool, water might be released. However, Section 14 of this SER shows that loss of coolant in the PUR would not lead to core damage, and mechanical damage to fuel cladding would release only a small fraction of the very low inventory of fission products.

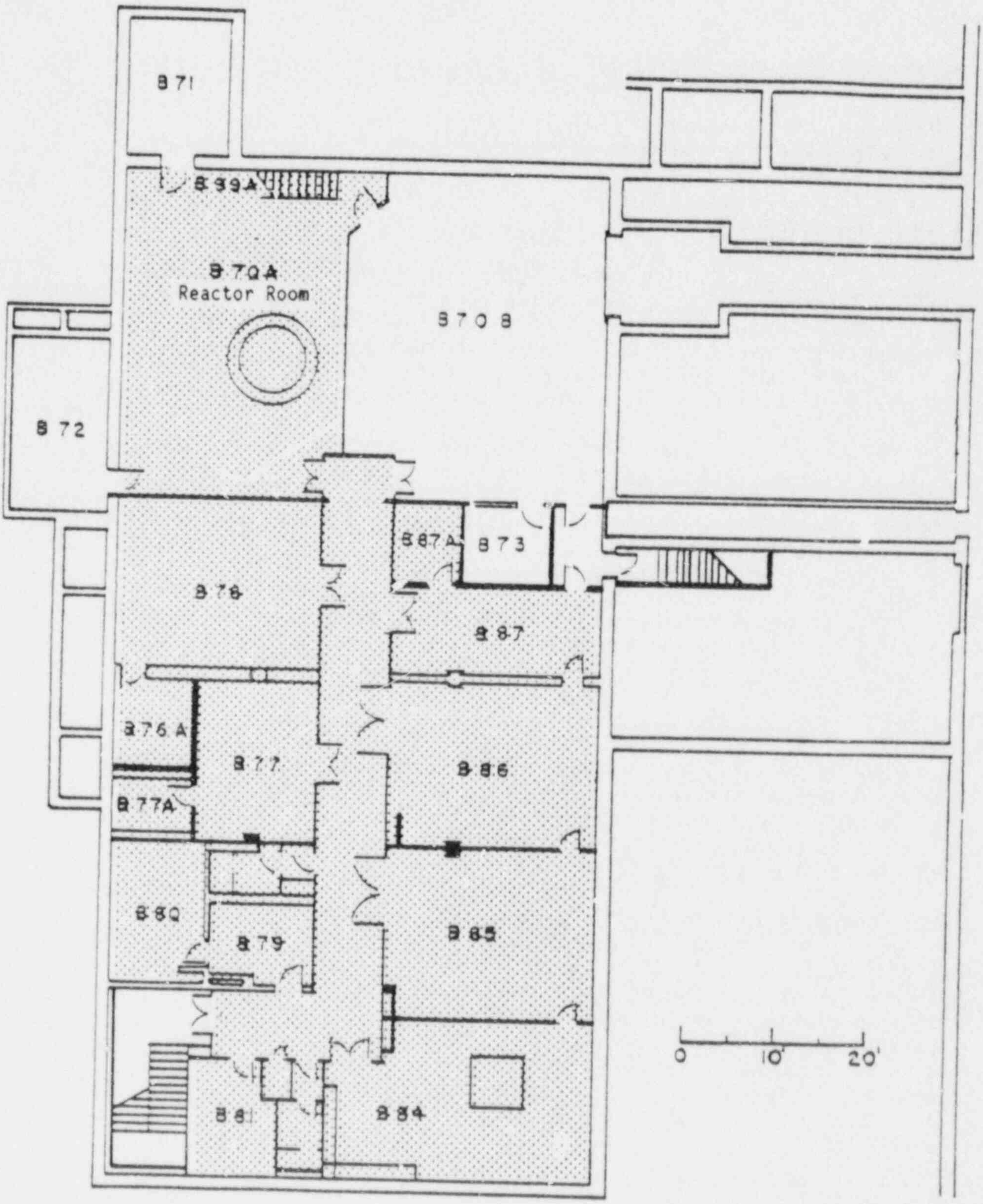

Figure 3.1 Floor plan of nuclear engineering laboratories, including reactor room

3.4 Mechanical Systems and Components

The mechanical systems of importance to safety are the neutron-absorbing control rods suspended from the reactor superstructure. The motors, gear boxes, electromagnets, switches, and wiring are above the pool-water level and readily accessible for testing and maintenance. The staff has addressed the effects of aging on the continued performance of these components in Section 17 of this SER.

3.5 Conclusion

On the basis of the above considerations, the staff concludes that the PUR facility was designed and built to adequately withstand all credible and likely wind, water, and seismic damage associated with the site. These considerations indicate that natural events would lead to small reactor-related consequences to the environment. Furthermore, the design and performance of the safety systems have been verified by more than 25 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 REACTOR

The PUR was built by Lockheed Nuclear Products and initially attained criticality in August 1962. This reactor uses MTR-type 93% enriched uranium-235 (U-235) aluminum-clad fuel plates that are assembled into fuel assemblies and placed into a graphite-reflected region to form the reactor core. The reactor core is immersed in an open tank of light water that serves as the neutron moderator, coolant, and shield. The reactor operates at a maximum power level of 1 kW. The reactor power is regulated by inserting or withdrawing neutron-absorbing control rods.

The reactor is used as a neutron source for activation analysis studies, academic research, and the limited production of radioactive isotopes. It also is used as a training facility for the nuclear engineering educational program. The PUR is operated for an average of about 13 kWh/yr. The principal design parameters for the current core configuration are listed in Table 4.1.

The PUR facility layout in the Duncan Annex of the electrical engineering building is shown in Figure 4.1.

4.1 Reactor Core

The core of the reactor is 30.48 cm square and 60.96 cm high. It consists of 13 fuel assemblies and 3 control rod assemblies. Each fuel assembly consists of up to 10 aluminum enriched-uranium alloy plates. Each control rod assembly consists of up to six plates and two aluminum guard plates with space for control rods. Adjustments to ensure that maximum excess reactivity is not exceeded are effected by substituting dummy fuel plates for uranium plates. The reactor is cooled and moderated by a pool of light water. The 4 x 4 array of fuel assemblies is reflected on all sides with graphite-reflector elements and on the top and bottom with water. The 20 reflector elements are composed of graphite waterproofed with epoxy resin and are contained in standard fuel assembly cans. One row of six graphite-reflector elements is designed to hold samples for isotope production (see Figure 4.2).

4.1.1 Fuel Assemblies

The MTR-type fuel plates are 93% enriched U-235 metal alloyed with 1100 aluminum alloy and clad with 0.05-cm 1100 aluminum alloy, with a total thickness of 0.15 cm. These flat MTR-type fuel plates are then inserted in aluminum canisters. Up to 10 fuel plates (7 x 64 x 0.15 cm overall dimensions) are contained in each of 13 standard fuel assemblies. Up to six plates plus two guard plates of 6061 aluminum alloy are contained in each of three control assemblies. The number of fuel plates in the fuel assemblies can be adjusted to provide for a maximum excess reactivity of 0.6% $\Delta k/k$.

Table 4.1 PUR principal design parameters

Parameter	Value
Maximum power level	1 kW
Geometry of core	0.3 x 0.3 x 0.6 m
Moderator-coolant	Light water
Maximum excess reactivity	0.6% $\Delta k/k$
Prompt neutron lifetime	77.2×10^{-6} s
Fuel assemblies	
Number (total)	16
Standard-type	13
Control assembly-type	3
Number of plates per standard assembly	10
Number of plates per control assembly	6
Plate dimensions	7.0 x 64 x 0.15 cm
Active fuel length	59.4 cm
U-235 per plate	16.5 g
Water gap	0.53 cm
Cladding	0.051-cm aluminum
Enrichment	93%
Reflector	
Material on sides	Graphite
Number of graphite assemblies	20
Control rods and drives	
Number of regulating rods	1
Number of shim safety rods	2
Total number of control rods	3
Measured worth of control rods	
Regulating rod	0.26% $\Delta k/k$
Shim safety rod no. 1	5.0% $\Delta k/k$
Shim safety rod no. 2	2.4% $\Delta k/k$
Rod speed out	
Regulating rod	45.0 cm/min
Shim safety rods	11.2 cm/min
Scram time for complete insertion	1 s
Material	
Regulating rod	Hollow stainless steel
Shim-safety rods	Solid borated stainless steel
Size	
Regulating rod	1.3 x 5.7 x 64.8 cm
Shim-safety rods	1.3 x 5.7 x 64.8 cm
Maximum rate of reactivity change	
Regulating rod	0.006% $\Delta k/k/s$
Shim-safety rod no. 1	0.031% $\Delta k/k/s$
Shim-safety rod no. 2	0.013% $\Delta k/k/s$

Table 4.1 (Continued)

Parameter	Value
Average rate of reactivity change	
Regulating rod	0.0031% Δk/k/s
Shim-safety rod no. 1	0.015% Δk/k/s
Shim-safety rod no. 2	0.007% Δk/k/s
Reactivity effects	
Temperature coefficient	
Calculated	-2.1×10^{-2}% Δk/k per °C
Measured	-3.4×10^{-2}% Δk/k per °C
Void coefficient (measured)	-2.6×10^{-2}% Δk/% void
Process water	
Resistivity	>330,000 ohm-cm
pH	5.5 ± 1
Flow rate	1.89 L/s

4.1.2 Control Rods

Three control rods are used to control and regulate the power levels in the PUR: one regulating rod and two shim-safety rods. Each of the three rods operates within a hollow guide tube. The neutron absorber in the regulating rod is stainless steel, and the neutron absorber in the shim-safety rods is borated stainless steel. Each control rod is 64.7 cm long and has a vertical travel of ~61 cm. The cross-sectional dimensions are 1.3 x 5.7 cm for all the rods. The maximum rate of withdrawal for the control rods corresponds to 0.031% Δk/k/s and 0.013% Δk/k/s for the two shim-safety rods and 0.006% Δk/k/s for the regulating rod.

The neutron-absorbing sections of the shim-safety rods are supported by electromagnets that release the rods in a scram. The scram time for complete insertion of these shim-safety rods is 1 second. The regulating rod is mechanically connected to its drive and does not scram.

4.1.3 Neutron Source

The PUR utilizes a 5-Ci plutonium-beryllium neutron startup source. The source is located in a special reflector element source holder adjacent to and just outside the graphite reflector (see Figure 4.2). The source can be withdrawn from its in-core position manually by means of an attached steel cable that is connected to the top of the source holder cap. An indicator light coupled to the startup meter at the control console shows whether the source is in or out of the core.

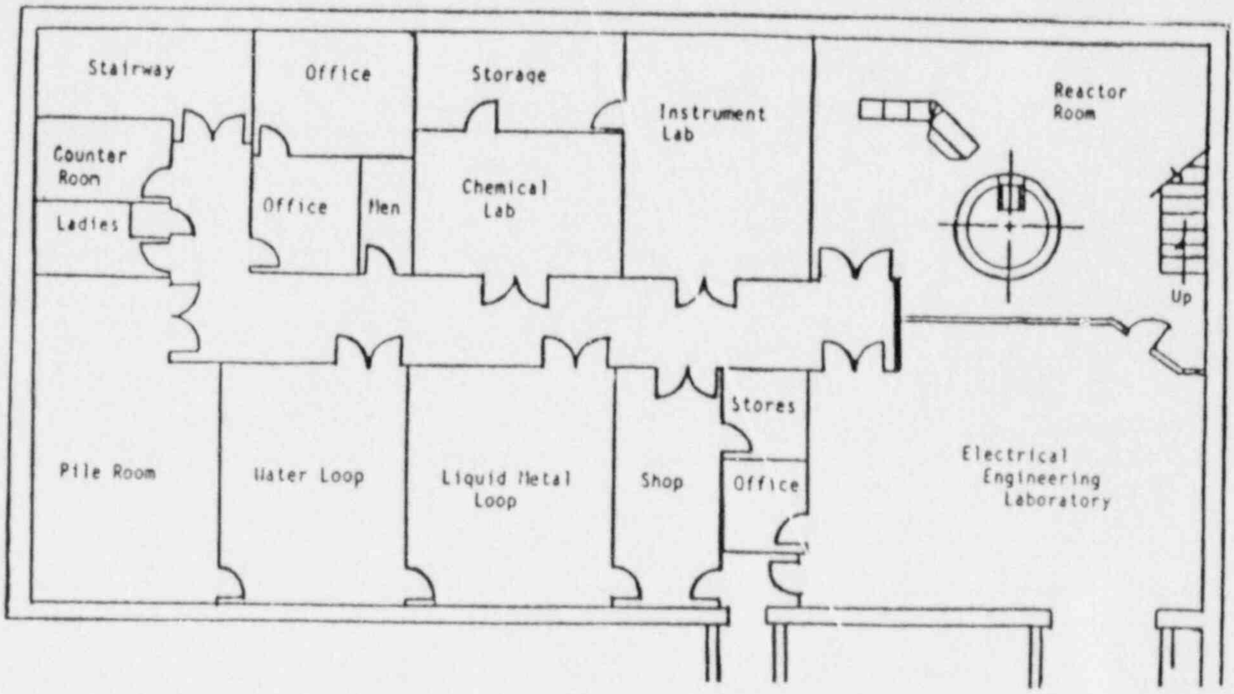

Figure 4.1 Facility layout

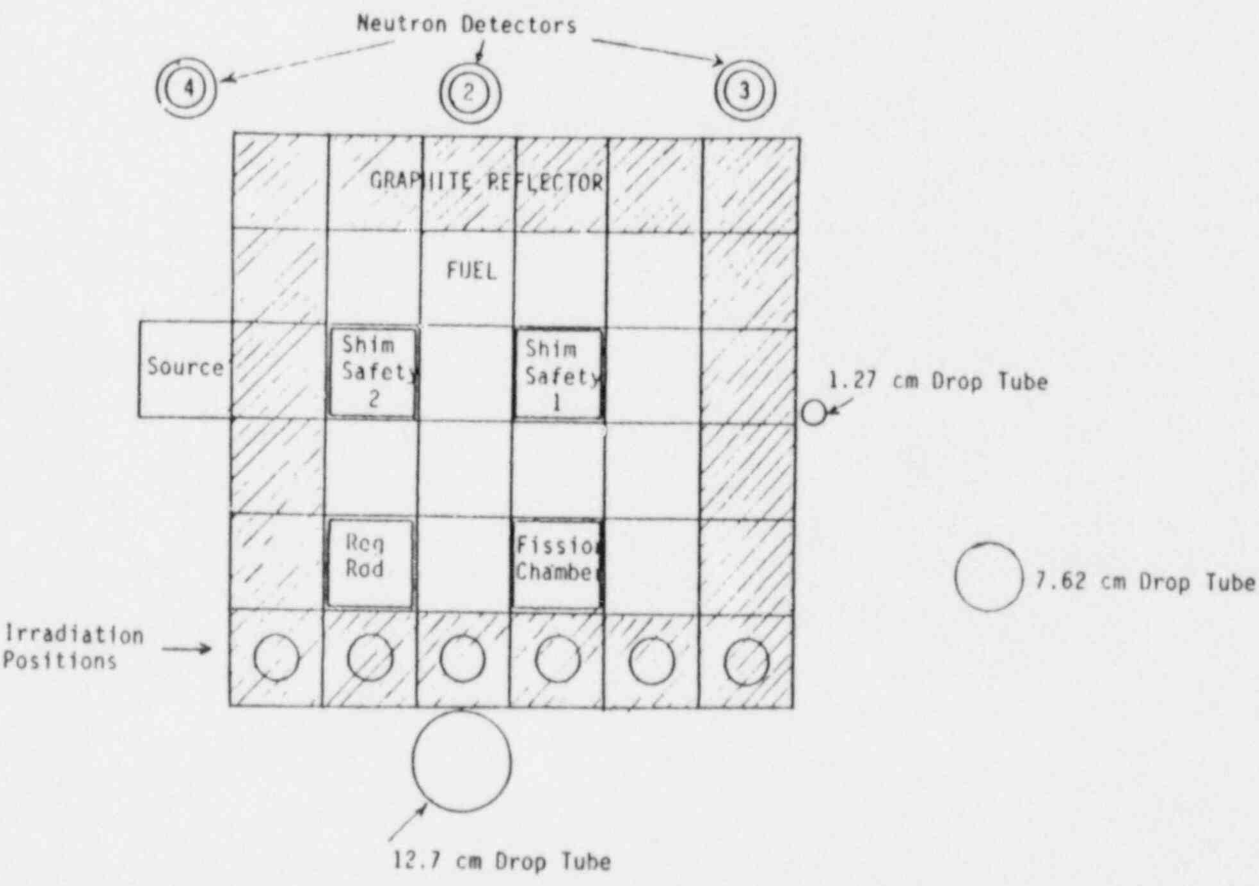

Figure 4.2 Core configuration

4.2 Reactor Pool and Biological Shield

The reactor core is located within two coaxial tanks that form the reactor pool. The outer tank rests on a concrete pad 4.6 m below floor level. The reactor pool is built below floor level except for the 1-m wall that serves as a biological shield for the operators and experimenters. The pool is contained in a cylindrical tank 5.3 m deep and 2.4 m in diameter. The core is located to one side to provide additional space for experiments. Opposite from the reactor core, two fuel storage racks are mounted on the tank floor. These fuel storage racks are fabricated of aluminum and contain a boral sheet in their centers as a neutron-absorption material.

The supports for the drive mechanisms for the control rods, the fission chamber and the source, and the neutron detectors are fastened to the support plate at the top of the tank. A traversing mechanism was mounted on the top of the reactor pool wall after the reactor was built. A lightweight, portable aluminum bridge can be placed across the pool for maintenance and fuel-handling operations.

Shielding over the core is provided by 4 m of water, which reduces the radiation level at the top of the pool to less than 1 mrem/h when the core is operating at 1 kW. The concrete pad, reactor tank, and distance reduce the maximum radiation level at the control console area to less than 0.1 mrem/h at 1 kW.

4.3 Grid Plates and Core Support Structure

A 7 x 11 position grid plate supports the 16 fuel assemblies and 20 reflector and isotope-production elements. The approximate active core dimensions are 30.5 x 30.5 x 61 cm. The core structure is centered approximately 76 cm from the center of the reactor tank and 9 cm from the bottom of the tank.

4.4 Reactor Instrumentation

The nuclear operation of the PUR is monitored by four neutron sensitive channels (two of which are always on range) that indicate thermal power level over the entire operating range of the reactor. These channels initiate scram signals if preset neutron flux levels are reached. The bulk reactor coolant temperature is measured manually with a thermometer placed in the pool water. The instrumentation and control systems are discussed in detail in Section 7.

4.5 Dynamic Design Evaluation

The PUR is operated by manipulating control rods in response to changes in the neutron flux (power) measured by the instrument channels. There are interlocks to prevent inadvertent reactivity additions and a scram system to initiate a rapid shutdown (reactor scram) if a preset power limit has been reached. Additionally, the measured temperature coefficient is negative over the operating temperature range. In the unlikely event of inadvertent high-power operation leading to high temperatures, this negative temperature coefficient of reactivity will tend to limit the reactor power.

4.5.1 Excess Reactivity and Shutdown Margin

Excess reactivity is defined as that value of reactivity that would occur if all control rods were completely removed from the reactor core. Reactivity is measured for a given core loading starting from a just-critical cold, clean core. A designated core loading may include irradiation facilities, such as the isotope-production elements, or other facilities of such nature that they become a portion of the core when installed.

Excess reactivity must be built into the reactor core in order to compensate for a number of reactivity losses. Also, a sufficient reactivity must be available to allow for an adequate reactor period for the PUR. This reactivity value has been determined to be not more than 0.6% $\Delta k/k$ and is the maximum allowed under any operating condition by the Technical Specifications.

The Technical Specifications require that the control rods provide a shutdown margin greater than 1.0% with the highest-worth control rod fully withdrawn and with the highest-worth experiment (0.4% $\Delta k/k$ for each secured experiment or 0.3% $\Delta k/k$ for each movable or unsecured experiment) in its most reactive state under any conditions of operation. This is to provide assurance that the reactor can be shut down safely even if one control rod did not insert.

The current core configuration has an excess reactivity of 0.48% $\Delta k/k$. The individual control rod worths are shown in Table 4.1. The total rod worth is 7.66% $\Delta k/k$. The shutdown margin for the current core configuration with the highest-worth rod fully withdrawn is 2.18% $\Delta k/k$.

Therefore, the current core configuration meets both the shutdown and the excess reactivity requirements. With all rods fully inserted, the core is subcritical by 7.18% $\Delta k/k$.

4.5.2 Conclusion

The Technical Specifications require that all three control rods be operable and the reactor can be brought to a subcritical condition even if the highest-worth control rod is totally removed from the core. These requirements ensure an adequate shutdown margin and provide sufficient redundancy in the unlikely event of a control assembly malfunction. Limiting the total excess reactivity of the core plus installed experiments to less than 0.6% $\Delta k/k$ allows for adequate reactor control under normal circumstances and prevents a prompt power excursion under any postulated abnormal circumstances.

On the basis of the above considerations, the staff concludes that excess reactivity will be limited sufficiently and adequate redundant shutdown capability is provided to ensure safe, controlled operation of the PUR. In addition, the 1.0% $\Delta k/k$ shutdown margin with the highest-worth rod fully withdrawn and the highest-worth experiment in its most reactive position provides reasonable assurance that the reactor can be shut down adequately under all postulated operating conditions.

4.6 Functional Design of Reactivity Control Systems

The power level of the PUR is controlled by three control rods: two shim-safety rods and one regulating rod. The two shim-safety rods can be scrammed, and solid borated stainless steel is used as the neutron absorber. The rods are connected to their drives by electromagnets. If electrical power to these electromagnets is interrupted for any reason, the armatures of the two shim-safety rods are released and they fall by gravity into the core. For the single regulating rod, hollow stainless steel is used as the neutron absorber. This rod is mechanically attached to its drive and does not fall into the core during scram. The core locations of the three control rods are shown in Figure 4.2.

Each rod-drive system is energized from the control console through its own independent circuits. A manual scram at the control console is possible for the two shim-safety control rods, or they can be scrammed automatically by the safety circuits. Although the regulating rod cannot be scrammed, certain manual or automatic trips can cause the regulating rod to be driven downward into the core.

4.6.1 Control Rod-Drive Assemblies

The tubular control rod-drive assemblies are mounted in a vertical position over the core. Motion is imparted to a control rod by a positive upward force on the extension rod, which is coupled to an electromagnet (except for the regulating rod), and to a screw mechanism, which is rotated through a fixed nut by the drive motor at the upper end of the tube. All rod drives are supplied with instant reversing induction motors. Downward motion results from the positive down-drive of the screw mechanism by the drive motor.

Limit switches are provided on the drive mechanism for the rod-drive units. The following switches are provided: jam, up, 2/3 up, and down. In addition, the safety rods are provided with bottom and magnet-engage switches. One coarse and one fine position indicator located on the console indicate the position of each rod. These indicators are located on the console and provide readings to 0.01 cm.

4.6.2 Control Rod Circuitry and Interlocks

Three identical control channels are used for the two shim-safety rods and the regulating rod. A pushbutton switch selects an individual rod to be controlled. All control rods can be inserted simultaneously into the core by a gang-lower switch when shutdown is desired. This switch cannot cause the control rods to be gang raised under any circumstance. Control-console indicators for each rod include upper limit and lower limit. Additionally, engage (magnet coupled) and shim range (rod not bottomed) indications are provided for the shim-safety rods.

Rod positions are indicated on a coarse vertical scale and on a selectable digital readout device having a resolution of 0.01 cm. The true value of the rod position is known to ±0.03 cm, which is equivalent to a maximum reactivity uncertainty of less than 0.01% $\Delta k/k$.

Two types of automatic action are incorporated into the reactor safety system to correct abnormal or unplanned conditions: trip and rod insert. In a trip, the shim-safety rods are dropped by removing the current from the magnets. A rod insert (set back) will cause all three rods to drive downward into the core. Both actions are of the latching type, and manual reset of the safety system is required to return to the normal conditions. A complete description of the scram system with setpoints is contained in Section 7.

4.6.3 Conclusion

The PUR is equipped with safety and control systems, control rods, rod drives, scram-logic circuitry, and interlocks that have performed reliably and satisfactorily in the PUR for 25 years.

The control systems allow for an orderly approach to criticality and for safe shutdown of the reactor during normal and abnormal conditions. There is sufficient redundancy of control rods to ensure safe reactor shutdown, even if the most reactive rod fails to insert on receiving a scram signal. Interlocks prevent inadvertent rod withdrawal and, thus, inadvertent positive reactivity changes. A manual scram button allows the operator to initiate a scram independently for any conditions deemed to require a prompt shutdown.

On the basis of the above discussion, the staff concludes that the reactivity control systems of the PUR are designed adequately and will function to provide a reasonable assurance of safety.

4.7 Operational Procedures

The PUR operates under Technical Specifications that direct the operation, audit, and surveillance of the reactor and provide procedural reviews for all safety-related activities. Written procedures have been established for safety-related and operational activities that include reactor startup, operation, and shutdown; maintenance; and calibration of equipment and instrumentation. In addition, the reactor is operated by trained NRC-licensed personnel in accordance with the above-mentioned procedures and Technical Specifications.

4.8 Conclusion

The staff review of the PUR facility has included studying its specific design and installation, control and safety systems, and operational limitations, as identified in the Technical Specifications. The staff concludes that the PUR was designed and built according to good industrial practices. The staff further concludes that there is sufficient shutdown margin to ensure that the PUR can be adequately shut down under all anticipated normal and abnormal operating conditions.

The design features of the PUR are similar to those of many pool-type research reactors operating in many countries of the world. On the basis of its review of the PUR and its experience with similar facilities, the staff concludes that this reactor is capable of safe operation, as limited by its Technical Specifications.

5 REACTOR COOLANT AND ASSOCIATED SYSTEMS

5.1 Primary Cooling System

The energy produced in the core is dissipated to the pool water as heat by the natural convection of the approximate 22,700 L of demineralized water in the reactor pool. The pool water is maintained at the ambient temperature of the environment by heat conduction to the ground and air and by some evaporation of water from the pool surface. To raise the temperature of the pool water 10C°, the reactor would have to operate continuously for more than 200 hours at full power (1 kW), assuming no heat loss. Thus, pool heatup poses no constraint on the anticipated operating schedule of the PUR (~13 kWh/yr).

5.2 Process Water System

The process water system is assembled in one unit and contains a pump, filter, demineralizer, valves, flow meters, and a heat exchanger (see Figure 5.1). The heat-removal capacity of the heat exchanger is 10.5 kW. It was designed to maintain the reactor pool temperature at 75°F during continuous operation at 10 kW. The demineralizer contains a removable cartridge that is monitored continuously for radioactivity buildup. This system limits, by the use of filters and ion-exchange resin, the aluminum corrosion rate, corrosion product buildup, and neutron activation of impurities in the coolant.

5.3 Primary Coolant Makeup Water System

Makeup water for the pool is taken batchwise from the Purdue University water line and is passed through the demineralizer enroute to the pool. A vacuum breaker excludes any possibility of siphoning pool water into the supply line. The pool makeup water system, in addition to the demineralizer, also includes a normally closed manual shutoff and throttle valve and a check valve.

5.4 Primary Coolant Chiller System

Although the chiller is not needed for present operations, it remains available if required. Calculations indicate that the temperature rise rate while operating the PUR at a power level of 1 kW would be 4.65×10^{-2} °C/h, based on a mass of water equal to 1.85×10^{4} kg. This takes no credit for heat loss to the surrounding sand and gravel or loss by evaporation. Experimentally, no temperature increase has been observed using the pool thermometer following 8 hours of operation at 1 kW. The chiller is designed with three loops to prevent the spread of radioactive contamination from the primary loop to the heat dump. The pool water passes through the primary loop, and a freon refrigerant is in the secondary loop. Campus water in the third loop is used to remove the heat and is discharged into the campus sewer system. Radioactive contamination cannot pass through the three-loop system, unless at least two pipe failures were to occur simultaneously with significant abnormal radioactivity in the primary coolant, and the chiller system was in operation.

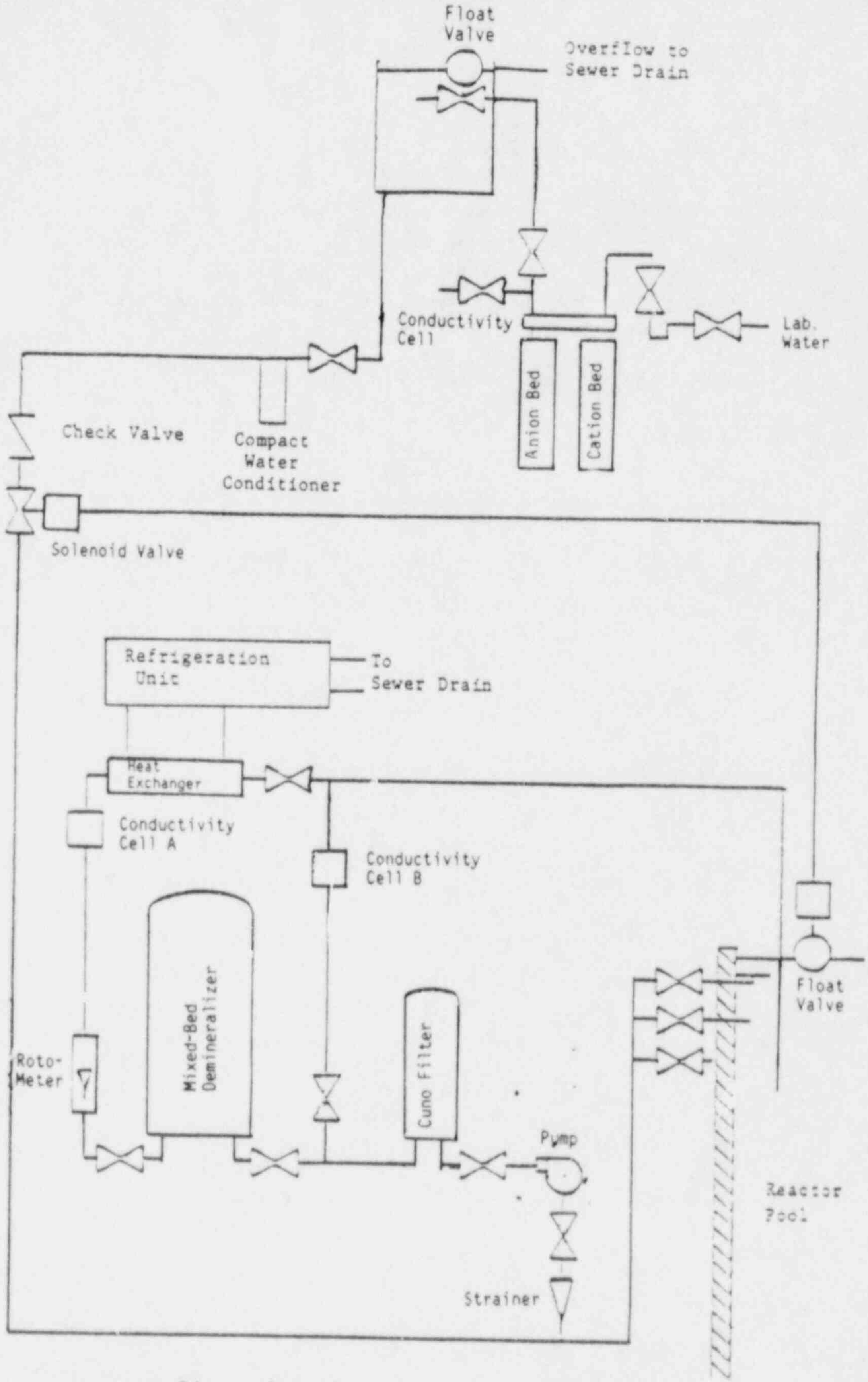

Figure 5.1 Reactor water process system

5.5 Conclusion

The staff concludes that the reactor coolant system at the PUR facility is of proper size, design, and condition and is maintained properly to ensure adequate cooling of the reactor at the power level specified in the PUR operating license. Also, the process water system can limit both corrosion and radioactivity problems associated with coolant contamination.

6 ENGINEERED SAFETY FEATURES

Engineered safety features are those features or systems that mitigate the potential consequences of accidents. The only systems that could be considered as engineered safety features associated with the PUR facility are the ventilation system and the drain system. These systems are designed to limit the uncontrolled release of airborne and liquid radioactive materials under normal operating conditions as well as accident conditions.

6.1 Ventilation System

The outside air supply and room exhaust are passed through high efficiency particulate air (HEPA) filters (see Figure 6.1). The reactor room is maintained at negative air pressure (minimum 0.13 cm of water). All doors to the reactor room have foam rubber seals. Steam heat is used to heat the room, and a room air conditioner circulates and cools the reactor room air. Under emergency conditions, the exhaust system and the air conditioner will be shut off and the sealed room will prevent the rapid spread of contamination.

6.2 Drain System

The only floor drain to the sewers is sealed except for a vent opening. This vent is raised about 1.2 m above the floor and has a filtered inverted opening. Condensate from the air conditioner is released to this drain through an opening 3.66 m above the floor. During an emergency, the valve on the drain from the condensate holdup tank is shut off with the same switch that shuts off the exhaust system. The condensate is held until it is tested for radiological contamination before it is released to the sewer.

6.3 Conclusion

On the basis of the evaluation of potential accidents at the PUR facility that are discussed in Section 14 of this SER, the staff concludes that no significant amounts of airborne radioactivity or liquid waste would be released into or out of the reactor room. Therefore, the available ventilation system is considered acceptable.

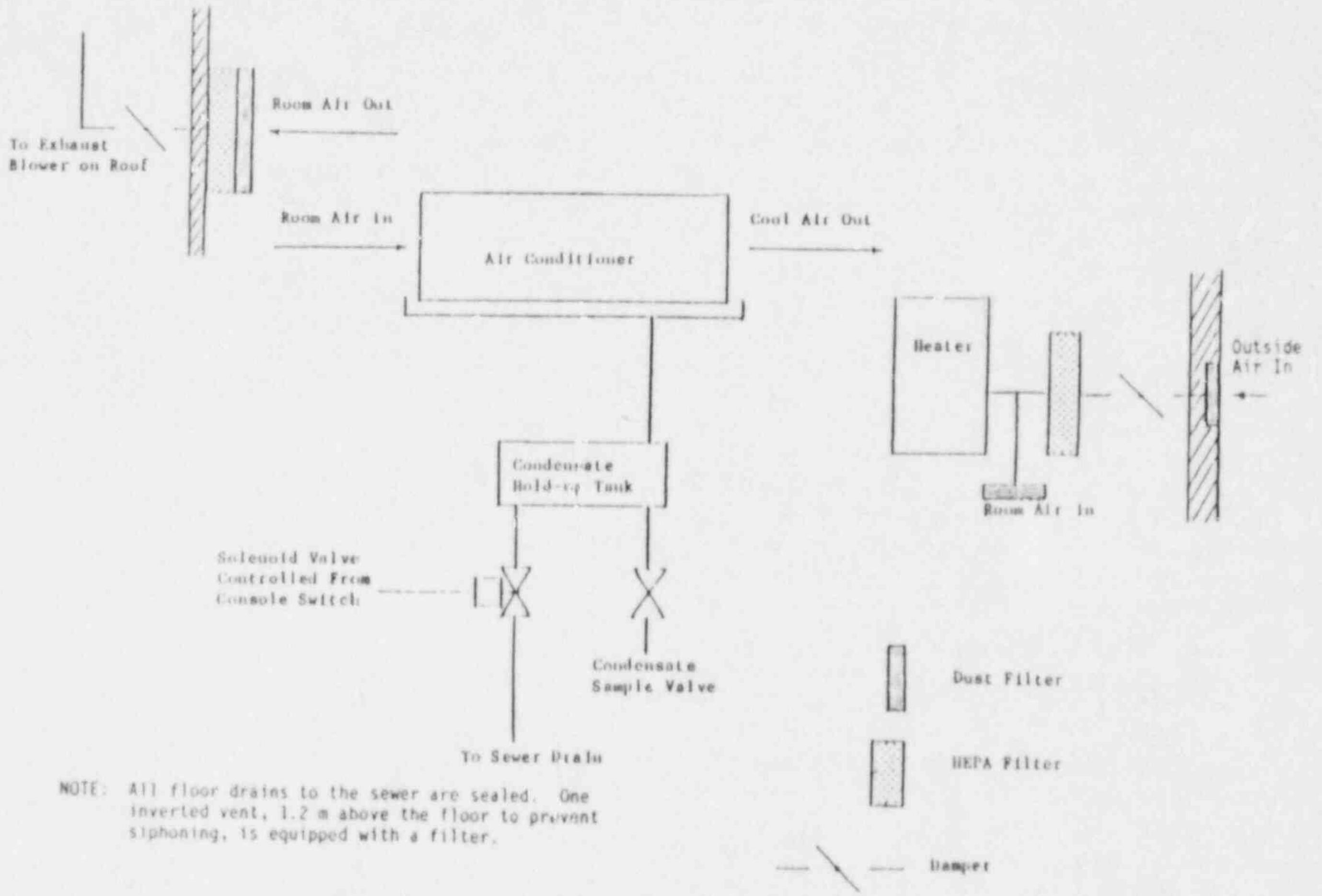

Figure 6.1 Reactor room ventilation and cooling system

7 CONTROL AND INSTRUMENTATION SYSTEMS

The major components of the PUR control and instrumentation systems, including rod controls, annunciators, pen recorders, and meters, are located in the control console. The control console is designed to provide maximum visibility of the instruments and accessibility to the controls and indicators. All indicators and controls necessary for startup and shutdown operations are located in one group in front of the operator.

7.1 Reactor Control System

The reactor control system at the PUR facility, consisting of both nuclear and process instrumentation, provides reactor control during normal operations and ensures safe shutdown in the event of abnormal operation (see Figure 7.1). Interlocks are provided between the instrumentation system and the scram system to provide positive control of the reactor and essentially eliminate the chances of accident initiation.

7.1.1 Control Rod Drives

Three identical and independent control channels are used for the shim-safety and regulating rod-drive systems. A pushbutton switch selects an individual rod to be controlled. All control rods can be inserted simultaneously into the core by a gang lower switch when shutdown is desired. This switch cannot cause the control rods to be gang raised under any circumstance. A complete description of the PUR control rod-drive system is given in Section 4.

7.1.2 Servo Control System

A servo control system provides automatic control once the reactor has reached the desired power level. The servo control system senses deviations from an adjustable setpoint on the channel no. 3 linear power recorder and adjusts the position of the regulating rod to maintain the reactor at a constant power level. Servo permit circuitry actuates the console alarm buzzer if the reactor power deviates by more than 5% from the setpoint, indicating a malfunction of the system. A deviation meter is located on the console.

7.1.3 Neutron Source Drive

A motorized neutron source drive is provided to raise the source through a travel of approximately 1.8 m to the "full out" position. The two-position system is operated by raise-lower switches at the console with limit switches to indicate the source upper-limit or lower-limit positions.

7.1.4 Fission Chamber Dr[illegible]

Controls for [illegible] motor [illegible] channel no. 1 fission detector are also located on the con[illegible] position indicator and a selectable fine position indicator [illegible] system is selected and coupled to the drive s[illegible]tch in the same [illegible] control rod drives. Indicator lights

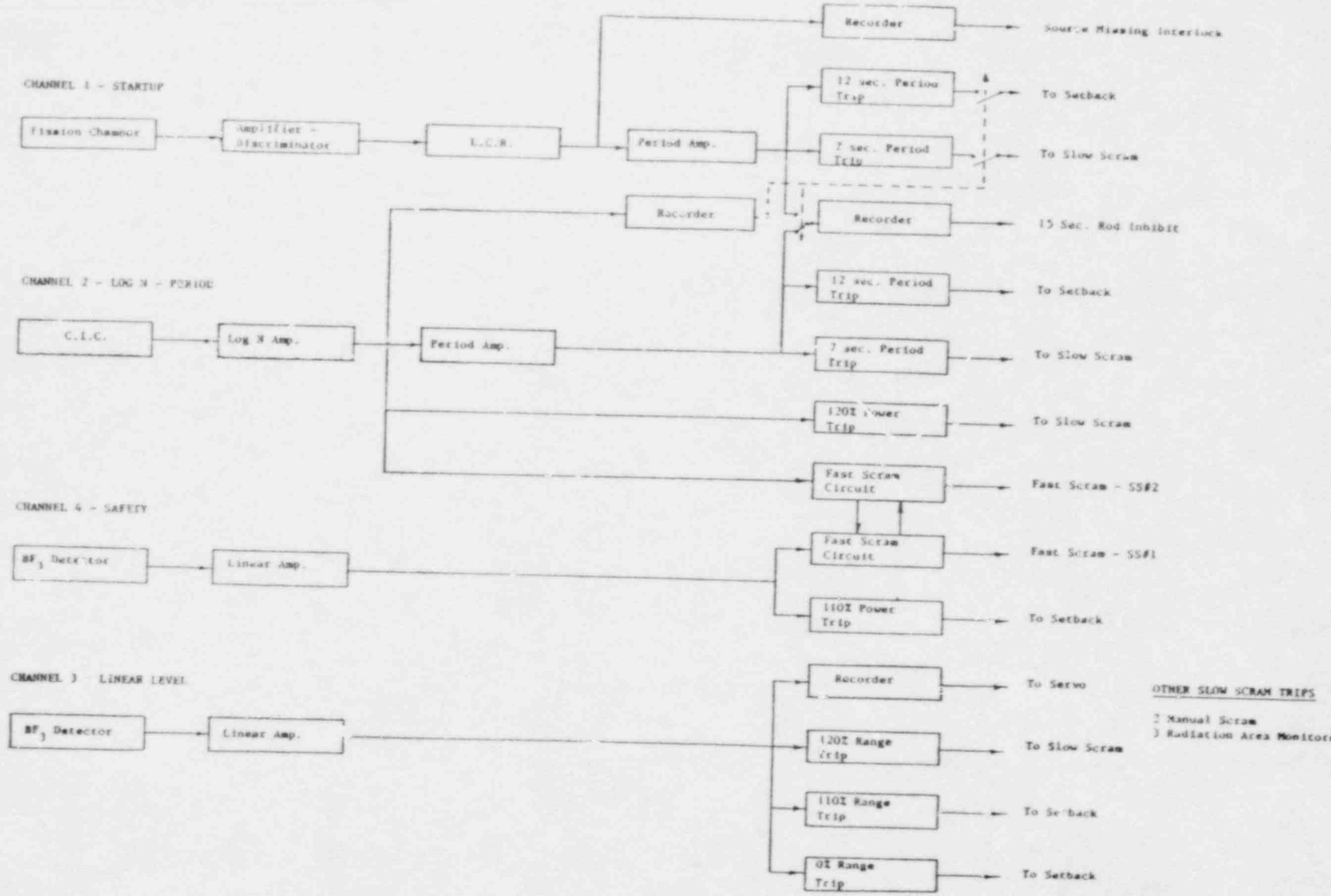

Figure 7.1 Reactor control system

note the upper- and lower-limit positions; however, the detector may be placed at any position within its range.

7.1.5 Annunciator and Alarm Systems

When a system trip occurs, or when other abnormal system conditions are sensed, an alarm (buzzer) sounds and an illuminated indicator is lighted on the control console indicating the source of the trouble. An annunciator acknowledge button may be used to reset the buzzer.

7.2 Reactor Instrumentation

The function of the reactor instrumentation is to provide adequate information for the operator and to generate signals to control the reactor or initiate trips. The nuclear instrumentation consists of a fission chamber, a compensated ion chamber, and two uncompensated ion chambers. All neutron detectors are arranged near the reactor core to ensure high sensitivity and to facilitate repair, maintenance, and repositioning. The detectors are in watertight aluminum tubes. The fission chamber is provided with a motor-driven positioning mechanism and position-indication system; the other detectors are manually adjustable.

7.2.1 Channel No. 1 - Startup Channel

The startup channel is used to monitor the neutron flux. It consists of a movable fission chamber, preamplifier, pulse amplifier, scaler for accurate counting, log count rate and period amplifier, and count rate recorder, and shares a period recorder with channel no. 2. The range of this equipment is from 1 to 10^4 counts/s (about 1.5×10^{-5} to 1.5×10^{-1} W) with periods from -30 to +3 s.

7.2.2 Channel No. 2 - Log N and Period Channel

The log N channel indicates the reactor power level over the range from 0.0001 to 300% power level (10^{-3} W to 3 kW). The detector for this channel is a compensated ionization chamber followed by a log N amplifier plus period instrumentation with outputs to the log N recorder and to the period recorder shared with channel no. 1.

7.2.3 Channel No. 3 - Linear Power

The linear level channel is capable of measuring neutron flux in a reactor operating range from about 10^{-5} W (shutdown) to >100 kW. The sensing element is a BF_3 ionization chamber coupled to a micro-microammeter.

7.2.4 Channel No. 4 - Safety Channel

The safety channel utilizes a BF_3 ionization chamber and feeds directly into the safety amplifiers. The sensitive range of this instrument is from a few percent to at least 150% of licensed power, linearly. Its output is indicated on the instrument chassis (instrument panel). The purpose of this channel is solely to provide reactor trip.

7.2.5 Temperature and Water Monitor Channels

Water temperature in the PUR pool is measured by a thermometer suspended by a string in the pool. Water level is measured by a scale immersed in the pool. Water conductivity is displayed on the console by two meters registering the output of two conductivity cells that measure the pool water before and after it passes through the demineralizer.

7.2.6 Radiation Monitoring Instruments

The radiation monitoring system consists of three fixed-position remote area monitors (RAMs) and one continuous air monitor (CAM). All of the RAM alarms initiate a reactor scram. The CAM detects airborne particulates and alarms. A continuous sample is drawn from the reactor room through the CAM filter, which is checked semimonthly for gross beta-gamma activity.

7.3 Scram System and Interlocks

Three types of action are incorporated into the control system to correct for abnormal reactor conditions:

- fast scram - initiated by short reactor period on channel no. 2 or high flux on channel no. 4 - interrupts the current to the control rod magnets electronically

- slow scram - initiated by short reactor period on channel no. 1 or no. 2 or by high flux on channel no. 2 or no. 3 - causes the rods to drop by removing power to the magnet power supply by means of a relay

- rod insert (setback) - initiated by short reactor period on channel no. 1 or no. 2 or by high flux on channel no. 3 or no. 4 - causes all three rods to be driven downward into the core

All three actions are of the latching type and require manual reset before return to the normal operating condition.

In addition to scrams initiated by high flux and reactor period, reactor scram or rod insert at PUR is initiated by any of the following signals:

- manual pushbutton
- compensated ion chamber (CIC) power supply failures
- safety amplifier trouble
- area high radiation

7.4 Conclusion

The control and instrumentation systems at the PUR are designed to provide reliability and flexibility. There is adequate redundancy and diversity in the nuclear flux (power) monitoring circuits. In particular, nuclear power measurements are overlapped in the ranges of the startup, log-N, linear power, and percent power level (safety) channels. In view of the simple nature of the open pool, the temperature and water monitor instrumentation is considered adequate. On the basis of the above information, the staff concludes that the control and instrumentation systems at the PUR facility comply with the requirements and performance objectives of the Technical Specifications and applicable regulations and are acceptable to ensure safe operation and shutdown of the reactor.

8 ELECTRIC POWER

8.1 Electrical Power System

The electrical power for building lighting and reactor instrumentation is single-phase, 60 Hz, 120/240 V, which is furnished through a transformer and several control panels located throughout the building.

8.2 Emergency Power

The reactor will scram in the case of an electrical power interruption because the control rods are supported by electromagnets. Because the decay heat generated in the core following a scram is not enough to cause fuel damage (see Section 14), emergency power is not required to maintain the reactor in a safe shutdown condition. Power for the facility intrusion detectors is supplied by a 12-V battery that is checked monthly and replaced biannually. If an electrical outage should occur, this battery would supply the necessary power for these instruments for at least 24 hours. Battery-powered emergency lighting is also available to facilitate personnel movement during a power outage. If a power outage should occur, no radiation monitors would be operating except for the portable, hand-held battery-powered type.

8.3 Conclusion

The staff concludes that the design of the electrical power system, coupled with the fact that the reactor will scram in the event of a power failure, is acceptable for continued operation of the PUR.

9 AUXILIARY SYSTEMS

The auxiliary systems considered are the ventilation system, the fire protection system, the fuel storage system, the heating and air conditioning system, and the crane system.

9.1 Ventilation System

The ventilation system is considered to be an engineered safety feature and is discussed in Section 6 of this report.

9.2 Fire Protection System

The PUR and the building where the reactor is located are intrinsically fireproof, and in the event of a fire no special precautions are required. The fire protection system for the reactor facility is typical of those at most university low-power research reactors. There are two portable fire extinguishers in the reactor room located at either end of the room. If a fire should occur, the reactor would be shut down and the supervisor, or alternate, would be notified. Normal fire procedures for the building are in place and are expected to preclude accumulation of flammable materials. A fire station is located on campus (~1/2 mile from the reactor) and is available on short notice to assist in case of a fire.

9.3 Fuel Storage System

The only fuel at the PUR facility is the existing core. The fuel is handled outside the core configuration only for experimental or inspection purposes. During some experiments, fuel elements are stored in one of the two fuel racks located inside the vessel on the bottom. Poison (boral steel plates) and geometry are used in these racks to ensure subcriticality. An annual fuel inspection involves removal of only one element outside the vessel at a time.

9.4 Heating and Air Conditioning System

The PUR facility heating and air conditioning system is integral to the ventilation system discussed in Section 6 of this report. The air-operated dampers in this system are supplied pressurized air from the university system. The actuators are designed to close the dampers if a loss of air should occur.

9.5 Crane System

The PUR facility has a 2-ton crane that runs along one axis over the center of the reactor tank and is normally positioned at the end of the track. This crane is used only occasionally under the supervision of the reactor supervisor for installing special experiments, and is maintained in good working order.

9.6 Conclusion

The staff concludes that the auxiliary systems at the PUR facility are designed, operated, and maintained adequately and are capable of performing their intended functions of helping to ensure the safe operation of the facility.

10 EXPERIMENTAL PROGRAMS

The PUR is used in support of educational programs in physical, biological, and pharmaceutical sciences. The reactor also is a source of ionizing radiation and neutrons used for various research programs.

10.1 Experimental Facilities

10.1.1 Reflector Tubes

Six positions in the graphite-reflector grid along one side of the core are utilized for special isotope-production elements (see Figure 4.2). These elements are identical to the graphite-reflector elements except for central access holes to accommodate samples up to 3.8 cm in diameter and up to 61 cm long. Currently, the sample capsules loaded into these locations are 7.6 cm long with an outside diameter of 2.5 cm and are fabricated of aluminum. Experiments located in these tubes are considered secured in that they are not moved during reactor operation.

10.1.2 Drop Tubes

Three drop tubes are currently utilized at the PUR. Their positions are shown in Figure 4.2. All of these tubes extend from the level of the active core to a height above the pool water level that is sufficient to prevent inadvertent flooding. Reactivity addition, should accidental flooding occur, is discussed in Section 14. The inside diameters of the tubes are 1.3, 7.6, and 12.7 cm. The 7.6-cm tube is fabricated of polyvinyl chloride; the other two are fabricated of stainless steel. Experiments contained in these tubes are properly secured to a suitable tether and then lowered into the tube. Because they may be moved during reactor operation, they are classed as nonsecured experiments.

10.2 Experiment Review

Specific procedures to allow placement, operation, and removal of all experiments proposed for the PUR are reviewed and approved by the Committee on Reactor Operation to establish if they fall within an envelope of previously accepted reactivity addition and radiological consequences.

10.3 Conclusion

The staff concludes that the design of the experimental facilities, combined with the review and administrative procedures applied to all research activities, is adequate to ensure that experiments are not likely to fail, are unlikely to release significant radioactivity to the environment, and are unlikely to damage the reactor systems or the fuel. Operating experience provides further assurance that the experimental program at the PUR facility will be conducted safely in the future. Therefore, the staff concludes that reasonable provisions have been made so the experimental programs and facilities do not pose a significant risk to the reactor core or risk of radiation exposure to the public.

11 RADIOACTIVE WASTE MANAGEMENT

The PUR produces essentially no radioactive waste during normal operation because of the low-power level and limited operating schedule.

11.1 ALARA Commitment

The university's commitment to the ALARA (as low as is reasonably achievable) principle was established by the Radiological Control Committee in 1951 and was recently (November 19, 1986) restated by the President of Purdue University. Purdue University is committed to a policy of making every reasonable effort to keep radiation exposures as far below the specified regulatory limits as is reasonably achievable. Thus, the underlying philosophy of the radiological control operations of the university will be to maintain radiation exposures as low as is reasonably achievable, which is in keeping with the recommendations of the National Council on Radiation Protection and Measurements, the National Academy of Sciences-National Research Council, and other independent scientific organizations. The principle of ALARA is also codified as part of the NRC regulations in 10 CFR 20.1(c), which state that licensees should "make every reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as far below the limits specified in the part as practicable."

11.2 Waste Generation and Handling Procedures

11.2.1 Solid Waste

The disposal of high-level radioactive waste in the form of spent fuel is not anticipated for the term of the license. Low-level radioactive solid waste generated at the facility consists of potentially contaminated paper and gloves and solid samples produced for experiments and is usually less than 1 ft^3/yr. This waste is collected in specially marked containers and is disposed of under the university's By-Product License 13-02812-04. The water process system uses demineralizer resins to remove impurities from the primary coolant and collect any radioactive ions in the water. These resins are continuously monitored and are periodically replaced. To date, no radioactive materials have been detected before shipment.

11.2.2 Liquid Waste

No radioactive liquid wastes except those that might be produced from student or faculty experiments are generated as a result of normal reactor operations. The pool water is analyzed periodically for radioactivity. Any detected short-lived activity is allowed to decay and would not constitute disposable liquid radioactive waste. Any wastes generated by research activities are disposed of under the university's By-Product License.

11.2.3 Airborne Waste

The airborne waste that could be present at the PUR facility would be composed of argon-41, tritium, nitrogen-16, and activated dust particles.

Argon-41 is produced by thermal neutron activation of argon-40 in the air dissolved in the pool water. No detectable traces of argon-41 from air dissolved in the water have been observed or are expected at the PUR facility because of its low operating power levels.

The most likely source of tritium (H-3) is the pool water. From monthly water samples, the level of tritium has been found to be very low (much less than the most restrictive maximum permissible concentration).

The principal potential source of nitrogen-16 is from the fast neutron interaction with oxygen in the pool water. The nitrogen must then diffuse to the surface of the pool before it is released to the atmosphere. In normal operation, currents that might be established in the reactor pool would be very small and with the short half-life (7.14 seconds), most of the nitrogen-16 would decay before it reached the surface. No nitrogen-16 has been detected in the reactor room.

The reactor room air is sampled continuously for particulates by a CAM. Filters are changed and analyzed semimonthly for gross alpha and beta activity using a windowless flow proportional counter.

11.3 Conclusion

The staff has reviewed the operational history of the PUR and concludes that no significant wastes are generated as a result of its normal operation. However, should any significant waste ever be generated, acceptable provisions for the radioactive waste management activities at the facility have been adopted and are expected to be continued to ensure compliance with 10 CFR 20 and the ALARA principle.

12 RADIATION PROTECTION PROGRAM

Purdue University has a structured radiation safety program. Policies for the program are determined by the Radiological Control Committee established by the President of the university (See Figure 12.1). The program is administered by the Radiological Control Officer and staff. The staff is equipped with radiation detection instrumentation to determine, control, and document occupational radiation exposures at the reactor facility and all laboratories using radioisotopes at the university under By-Product License 13-02812-04. Routine surveys of the reactor room include analysis of the reactor pool water and reactor room air.

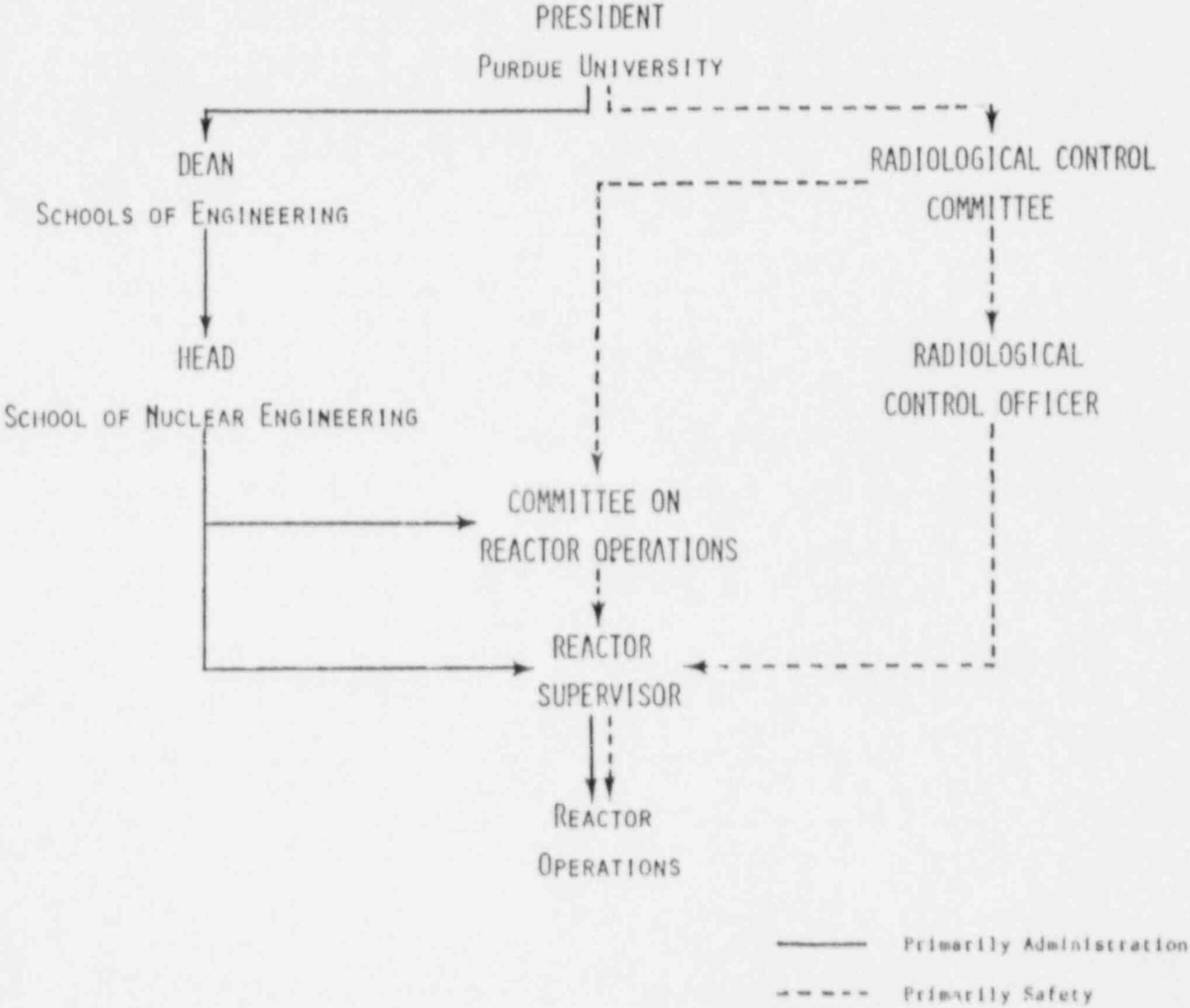

Figure 12.1 Organizational structure for PUR operations

12.1 ALARA Commitment

The university is committed to the ALARA principle, and the Office of Radiological and Chemical Control makes every effort to keep doses as low as is reasonably achievable (ALARA). All unanticipated or unusual exposures are investigated by the Radiological Control Committee and the operations staff to develop methods to prevent recurrences.

12.2 Health Physics Program

At present, the university has a full-time health physics staff consisting of a Radiation Safety Officer, Assistant Radiation Safety Officer, two health physicists, environmental waste technician, and appropriate secretarial support. The health physics staff performs all routine surveys and is available for consultation on all matters concerning radiation safety.

There are no documented procedures to ensure consultation on matters concerning radiation safety as opposed to ensuring availability of the health physics staff. However, specific reactor operating procedures require that radiological control personnel be present when specified operations are performed. Because the Radiological Control Officer is an official member of the Committee on Reactor Operations, this ensures the relationship is more than casual (see Figure 12.1).

12.2.1 Procedures

Written procedures have been prepared that address routine health physics monitoring at the PUR facility. These procedures identify the interactions between the operations and health physics personnel and the administrative limits to control exposure. Copies of these procedures are available to the operations and research staffs and administrative personnel.

12.2.2 Instrumentation

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

12.2.3 Training

All reactor-related personnel are required to attend a radiation safety training session before they begin work at the reactor. Additional training to illustrate the ALARA principle and methods to minimize exposures is provided to those personnel working directly with radioactive materials. Retraining for reactor operators in radiation safety is also provided periodically.

12.3 Radiation Sources

12.3.1 Reactor

The reactor core is the primary source of radiation directly related to reactor operations. Radiation exposure rates from the reactor core are reduced to acceptable levels by the water in the pool and concrete shielding.

12.3.2 Extraneous Sources

Sources of radiation associated with reactor use include radioactive isotopes produced for research, activated components of experiments, and activated samples. Personnel exposure from these sources is strictly controlled by fully developed procedures that follow normal health physics principles.

12.4 Routine Monitoring

12.4.1 Fixed-Position Monitors

Three fixed-position remote radiation area monitors (RAMs) with adjustable alarm setpoints and one continuous air monitor (CAM) are located in the reactor room. The CAM air filters are changed and analyzed semimonthly.

The RAMs are set to alarm at 7.5 mR/h. This setpoint was calculated using the 10 CFR 20 limit of 3 R per quarter maximum whole-body dose and assuming 10 weeks per quarter and 40 hours per work week. Instruments are calibrated semiannually by radiological control personnel. The RAMs are calibrated for exposure rate, and the CAM is calibrated for detection efficiency.

12.4.2 Wipe Tests

Wipe tests of exposed surfaces of the reactor room are made monthly. Water samples are taken and counted monthly. All samples and material removed from the reactor are checked for levels of activity, and wipe tests are made for loose contamination.

12.5 Occupational Radiation Exposures

12.5.1 Personnel Monitoring Program

Film badges and thermoluminescent dosimeter (TLD) finger rings are assigned to all approved reactor personnel. In addition, self-reading pocket dosimeters and dose rate instruments are used to administratively keep occupational exposures below the regulatory limits in 10 CFR 20. Students and visitors are provided with self-reading pocket dosimeters.

12.5.2 Personnel Exposures

Exposures to reactor personnel are monitored with film badges and TLD finger rings, which are read quarterly. If doses are >100 mrem, a record is made of how and why the dose was received. Exposures are generally much less than 100 mrem, except during the annual fuel plate inspection, when a finger ring dose of >100 mrem may be experienced by the plate inspector. Because the reactor personnel and fast breeder blanket facility (FBBF) personnel are the same and use one personnel dosimeter system, it is difficult to determine how much of the "facility" dose is a result of the reactor operations. However, this is not considered necessary because the actual dosages are so low. A summary of the latest 5 years of whole-body exposure to reactor personnel is provided in Table 12.1.

Table 12.1 History of personnel radiation exposure at PUR facility

Whole-body exposure (rem)	Number of individuals				
	1981	1982	1983	1984	1985
<0.1	8	9	6	8	7
>0.1	0	0	0	0	0

12.6 Effluent Monitoring

12.6.1 Airborne Effluents

Potential releases of airborne effluents are monitored with a CAM in the reactor room.

The CAM has never detected an unexplainable level of activity from either the accumulation of particulates or immersion exposure (i.e., immersion gas).

Analysis of the CAM filter generally indicates a concentration less than 3×10^{-16} µCi/cc for particulates, which is well below the most restrictive maximum permissible concentration of 6×10^{-13} µCi/cc for an unknown alpha emitter.

Airborne waste is discussed in Section 11.2.3 of this report.

12.6.2 Liquid Effluents

The reactor generates no detectable radioactive liquid waste during normal operation. To prevent any release of potentially contaminated water to the sewer system, samples are collected and analyzed by standard techniques.

Liquid waste is discussed in Section 11.2.2 of this report.

12.7 Environmental Monitoring

Under the environmental monitoring program, the reactor pool water is sampled monthly, reactor room CAM air samples are analyzed semimonthly, one TLD has been placed in the reactor room, and one TLD has been placed in a classroom above the reactor room. Reactor pool water samples are analyzed for gross gamma, gross alpha, and gross beta activity and for H-3. Reactor room air samples are analyzed for gross alpha and gross beta activity. Results indicate nothing beyond natural background has been detected in the reactor room air and reactor pool water samples. Typical exposures have ranged from those that are minimally detectable to 30 mrem.

12.8 Potential Dose Assessments

Natural background radiation levels in the West Lafayette area result in an average exposure of about 100 mrem/yr. The maximum potential dose outside the reactor room is less than 1 mrem/yr based on film badge data for the classroom above the reactor room; therefore, there is no significant contribution to the background radiation in unrestricted areas.

12.9 Conclusion

The staff concludes that the radiation protection program receives appropriate support from the Purdue University administration. The staff further concludes that (1) the program staff is adequate and is equipped properly, (2) the reactor radiation safety-related staff has adequate authority and lines of communication, (3) the procedures are integrated correctly into the research plans, and (4) surveys verify that operations and procedures achieve ALARA principles. Additionally, the staff concludes that the PUR radiation protection program is acceptable because there have been no instances of reactor-related exposures of personnel above applicable guideline values and no significant releases of radioactivity to the environment have been identified. There is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public during routine or offnormal reactor operations.

13 CONDUCT OF OPERATIONS

13.1 Overall Organization

Responsibility for the safe operation of the reactor facility is vested within the chain of command shown in Figure 12.1. The Head of the School of Nuclear Engineering is delegated responsibility, on behalf of the licensee, for overall facility operation.

13.2 Training

Most of the training of reactor operators is done by in-house personnel. The licensee's Operator Requalification Program was revised in February 1988 in conjunction with this license renewal application, and the staff concludes that it meets the applicable regulations [10 CFR 50.54 (i-1) and 10 CFR 55] and is consistent with the guidance of ANS 15.4.

13.3 Operational Review and Audits

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

13.4 Emergency Planning

10 CFR 50.54(q) and (r) require that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E to 10 CFR 50. An applicable Emergency Plan was submitted by the licensee on May 10, 1984, and approved by the NRC on November 7, 1984.

13.5 Physical Security Plan

The PUR facility has established and maintains a program to protect the reactor and its fuel and to ensure its security. The NRC staff has reviewed the revised Physical Security Plan and concludes that the plan meets the requirements of 10 CFR 73.67 for special nuclear material of moderate strategic significance. The PUR facility's inventory of special nuclear material for reactor operation falls within that category.

Both the revised Physical Security Plan, Revision 3, dated May 15, 1987, and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1). The amendment renewing facility Operating License R-87 incorporates this Physical Security Plan as a condition of the license.

13.6 Conclusion

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

14 ACCIDENT ANALYSES

The consequences of potential accidents in the PUR are limited by the low power level (1 kW) at which the reactor is operated. The low power levels and low use of the PUR result in a very small accumulation of fission products during normal operations and a correspondingly low level of decay heat and radioactivity stored in the fuel elements. Therefore, some accidents normally postulated for nonpower reactors, such as loss of tank water and handling of irradiated fuel, do not constitute a major hazard in the PUR. To pose a significant hazard, an accident must generate and release a significant amount of fission products.

The licensee and the staff both evaluated the potential consequences resulting from (1) a fuel-element-handling accident, (2) maximum reactivity insertions, (3) reactivity insertions from experiments, (4) a loss-of-coolant accident, and (5) failure of a fueled experiment.

14.1 Fuel-Element-Handling Accident

Fuel-element maneuvers are always conducted under water in the reactor pool. The fuel elements are removed from the core and moved into the storage space, one at a time, using a hand-held fuel-handling tool. Normally, fuel is not removed from the pool except for an annual inspection of a fuel element. A fuel element weighs about 3.18 kg (7.0 lb) in air and only about 2.0 kg (4.4 lb) in water.

14.1.1 Scenario

Three potential fuel-handling cases are considered. Case A assumes the element is dropped on top of the core; case B assumes the element drops back into the core position as it is being raised; and case C assumes the element falls flat on top of the core.

14.1.2 Technical Assessment

For case A, if a fuel element should fall from the handling tool during its transfer under water, it is not heavy enough to cause any significant damage. The most severe damage likely to occur would be some denting of the end fittings because the fuel element, since it is an elongated object, would tend to fall in water in a rather upright position. For case B, no damage to the element would be expected if the element fell back into its original position because of the small distance (60.1 cm) it would fall. Also, no reactivity concerns are present because the core would have had sufficient shutdown margin present before the fuel element was removed. For case C, if the element were to fall flat on top of the core, less force would be placed on the element than if it fell on the end; therefore, no damage would be expected.

14.2 Maximum Reactivity Insertion

This hypothetical accident begins with the step insertion of the maximum licensed excess reactivity of 0.6% $\Delta\kappa/k$ into the critical reactor operating at the maximum licensed power of 1.0 kW.

14.2.1 Scenario

Two general cases are considered: Case A assumes that the safety control circuitry is operating and is activated by the reactor power exceeding the 120% (1.2-kW) power setpoint. Scram redundancy is provided under these conditions, because the period would be about 1 second and the short-period trip would also initiate a reactor scram. During the scram, it is assumed that the most reactive shim-safety rod is stuck and the second shim-safety rod drops into the reactor providing a shutdown margin of 1.8% $\Delta k/k$. It is assumed that all negative reactivity provided by the shim-safety rod is added as a linear ramp in 1 second, the time specified in the Technical Specifications for the rods to be fully inserted.

Case B considers the worst failure of the safety system where both rods fail to scram and the reactor is controlled only by the negative temperature feedback of the moderator. In this case, the calculated value of the moderator coefficient is used because it is the lesser of the two values (calculated vs. measured) reported in Table 4.1 and thus ensures conservatism.

The results for case A show reactor power would rise to about 2.8 kW within 55 ms following the step insertion of 0.6% $\Delta k/k$, primarily as a result of the prompt jump. The 120% (1.2-kW) trip, however, would initiate scram within about 3 ms of the step insertion. Between this 3-ms scram initiation and the 55-ms peak power, the scram reactivity is insufficient to control the power rise; however, once the prompt jump to 2.8 kW is completed, the scram reactivity becomes controlling and the PUR is quickly shut down. The calculated results for case A are summarized in Table 14.1. As shown, the power is above its initial value for less than 0.5 second and the resultant energy release is about 0.5 kW-s. This results in a negligible temperature rise in the PUR fuel plates.

The results for case B, where no control rods are inserted and the negative moderator temperature coefficient is considered as the only shutdown mechanism, are presented in Table 14.2. In this case, the analysis shows that the reactor power rises over a period of about 3 minutes to a level of about 380 kW. At this time the 0.6% $\Delta k/k$ step reactivity insertion causing the power rise is completely quenched by the negative moderator temperature coefficient. At this time or before, it is reasonable to assume that another scram or operator-initiated scram occurs and the power level is quickly reduced below 1 kW. These results are consistent with a number of the excursion experiments performed at the BORAX and SPERT facilities. Some of the results of the SPERT-1 experiment using the DU-12/25 core are applicable to the analysis of the PUR because the fuel geometry and composition are very similar. Table 14.3 compares the characteristics of the two reactors. A series of self-limiting power excursion tests was carried out in SPERT-1 using five core loadings. The input variable referred to in these experiments was the reactor period, induced by a

Table 14.1 Results of power transient analysis with ramp insertion of control rod (case A)

Parameter	Value
Initial conditions	
Power level (kW)	1
Temperature (°C)	20
Input values	
Reactivity (step) added (% Δk/k)	0.6
Scram time (s)	1.0
Scram reactivity ramp (% Δk/k/s)	-2.40
Calculated results	
Maximum power level (kW)	2.85
Elapsed time to maximum power (ms)	55
Elapsed time while P>1 kW (ms)	275
Total energy released while P>1 kW (kW-s)	0.5

Table 14.2 Results of power transient analysis with no control rods (case B)

Parameter	Value
Initial conditions	
Power level (kW)	1
Pool temperature (°C)	20
Reactor temperature (°C)	20.6
Moderator temperature coefficient (% Δk/k/°C)	-2.1×10^{-2}
Input values	
Reactivity (step) added (% Δk/k)	0.6
Calculated results	
Maximum power level (kW)	380
Elapsed time to maximum power (min)	~3
Total energy released in 3 min (MW-s)	~54

stepwise reactivity insertion. The results of the calculations of the PUR postulated accident are consistent with the observed results of the SPERT-1 experiments using long periods of about 1 to 3 seconds where absolutely no fuel damage was observed. In fact, the SPERT-1 experiments showed that the fuel could withstand transients with periods as short as 14 ms with no apparent damage to the fuel.

Table 14.3 Comparison of important fuel data for PUR and SPERT-1

Item	PUR	SPERT-1
Geometry	Plate	Plate
Length (cm)	61	61
Width (cm)	7.0	7.6
Thickness (cm)	0.15	0.15
Water gap (cm)	0.53	0.45
Fuel		
Material	U-Al	U-Al
Enrichment (%)	93	93
Thickness (mm)	0.51	0.51
Cladding		
Material	Al	Al
Thickness (mm)	0.51	0.51

Because the results in Table 14.2 are consistent with long-period SPERT-1 tests where no fuel failure was observed and because the available excess reactivity is much less than was added to the SPERT-1 during the short-period tests (as low as 14 ms with no fuel failure), it is concluded that even during the very unlikely event of a safety system failure during the maximum credible reactivity accident, the fuel would not melt and no fission products would be released.

14.2.2 Technical Assessment

The staff believes that the reactivity insertion accidents considered in the Purdue SAR are representative of the most severe transients that can credibly occur at the PUR. The staff has reviewed the licensee's accident assumptions and calculations and finds them conservative, reasonable, and acceptable. The staff, therefore, concludes that it is unlikely that a credible nuclear excursion in the PUR would lead to fuel melting or cladding failure and, consequently, such a transient would not pose a significant hazard to the public.

14.3 Flooding of an Irradiation Facility and Failure of a Movable Experiment

A sudden replacement of a voided (i.e., air-filled) space next to the core by water, such as that resulting from the flooding of an experiment tube, would cause a stepwise reactivity insertion, its magnitude depending on the void volume being replaced and its position relative to the core. Experiments have shown that flooding of the 12.7-cm irradiation tube located outside graphite reflector element F6 with water adds 0.3% $\Delta k/k$ to the core.

Another identified mechanism for suddenly adding reactivity to the critical core at the PUR is the failure of a movable experiment. Similarly, the maximum stepwise reactivity addition is limited by the Technical Specifications to 0.3% $\Delta k/k$.

14.3.1 Technical Assessment

It is shown in Section 14.2 that a sudden reactivity insertion of 0.6% $\Delta k/k$ into a critical core of the PUR can be tolerated with a sufficient safety margin. Therefore, 0.3% $\Delta k/k$ would be enveloped by that analysis.

14.4 Loss-of-Coolant Accident

The reactor pool is designed to prevent unintentional drainage. The pool is constructed of a stainless steel liner and set in a second steel tank with the interstitial region filled with sand. The tank rests on a concrete pad about 4.6 m below the floor of the reactor room, which is in the basement of the building. The pool has no drains or coolant pipes that could open or break. Therefore, a sudden loss of coolant is considered to be extremely unlikely. Furthermore, if the pool drained instantaneously, while the reactor was operating, the loss of water (moderator) would shut down the reactor.

14.4.1 Scenario

Any reasonably conceivable leakage of water from the reactor pool is expected to be rather slow. In such a case, the radiation area monitor mounted directly above the core would detect any additional radiation coming from the core as a result of a decreasing pool water level. Because the pool water level is checked during daily routine operations, any significant leakage would be detected before reactor startup for the day.

14.4.2 Technical Assessment

Although extremely unlikely, if the core were to become immediately uncovered following a 1-kW power run for 24 hours, heat transfer would occur by natural convection of ambient air. The decay power of the PUR immediately after shutdown from full power (1 kW) is about 65 W. The decay power rapidly decreases and is about 35 W after 1 minute of decay and 0.87 W in 24 hours. For this case, the amount of heat removed is proportional to the cladding temperature. No significant temperature increase of the fuel would be expected because heat transfer would occur first to the aluminum fuel assembly cans and then by convection to the air. Even assuming adiabatic conditions for 24 hours (that is, no heat transferred from the fuel), the fuel temperature rise would only be 9.5C°, which is not a significant temperature rise for the PUR fuel.

14.5 Maximum Hypothetical Accident

In this section, the failure of an experiment during which fissile material has been irradiated in the reactor is analyzed to assess the hazard associated with this accident. For this analysis, it is assumed that a capsule containing irradiated fissile material breaks and a portion of the fission product inventory becomes airborne. The consequences of the release are analyzed for both the reactor staff and the general public. Because the potential impact of this postulated accident is greater than that for any other accident analyzed, the failure of a fueled experiment is designated as the maximum hypothetical accident at the PUR.

14.5.1 Scenario

In this analysis the consequences of a failed experiment generating 1 W of fission power were studied. The capsule containing the experiment is assumed to break as it is removed from the reactor. The fission products expected to become airborne are the noble gases and elemental iodine. Other fission products and actinides are not volatile at the fueled experiment temperature (which is essentially room temperature). All of the noble gases and 25% of the radioiodine are assumed to be released, which is consistent with Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident," March 1972. No credit for the absorption of iodine in water is taken because of the designation of this event as the maximum hypothetical accident, and because it is postulated to occur in air.

A conservative assumption that the irradiation time was infinite was made in this analysis. Therefore, the fission inventories used in the analysis for some long-lived radionuclides (e.g., krypton-85 or even iodine-131) are overly conservative. Furthermore, it was assumed that the fission products are instantaneously released and uniformly distributed in the 424-m^3 reactor room air volume.

14.5.2 Technical Assessment

The calculated saturation activity for each respective radioiosotope and its concentration in the reactor room after experiment failure is shown in Table 14.4 for a 1-W experiment. This experimental specimen power level corresponds to the amount of fuel that could be allowed by the relevant Technical Specifications. Also shown in this table are the calculated dose rates for the whole body, skin, and thyroid. Under these conditions any one of the RAMs would cause an automatic reactor shutdown and audible and visual alarms in the control room. From past experience, the reactor building can be evacuated within 1.5 minutes. Therefore, it is assumed that the exposure time to the reactor staff is 1.5 minutes, which results in radiation doses of: whole body - 17 mrem, skin - 11 mrem, and committed thyroid - 830 mrem. These doses are <10% of the dose limits as stated in Regulatory Guide 2.2, "Development of Technical Specifications for Experiments in Research Reactors," November 1973, for doubly encapsulated experiments.

This radiation exposure is less than the limits (<10% of the equivalent annual dose stated in 10 CFR 20) established in the Technical Specifications, Section 3.5.f, for a single encapsulated experiment. This experiment corresponds to the irradiation of 1.1 gm of U-235 in the midplane of the isotope irradiation tube located in position F6.

For the radiation calculations outside the reactor building, it was assumed that all fission products released in the reactor building would leak out within 24 hours. Because the reactor room does not have any windows and only a few doors and emergency procedures call for turning off the air exhaust system, the leak rate assumption is considered to be reasonable.

Table 14.4 Dose rates in the reactor room from a failed fuel experiment (power = 1 W)

Isotope	Total (Ci)	Released % (Ci/cm^3)	E-gamma (MeV)	E-beta (MeV)	DCF* (rem/Ci)	Dose rate Gamma** (mrem/h)	Dose rate Beta† (mrem/h)	Dose rate Thyroid (rem/h)
I-131	2.66 E-02	1.57 E-11	3.71 E-01	1.97 E-01	1.00 E+06	3.49 E+00	2.55 E+00	1.96 E+01
I-132	3.75 E-02	2.21 E-11	2.40 E+00	4.48 E-01	6.60 E+03	5.00 E+01	8.16 E+00	1.82 E-01
I-133	5.31 E-02	3.10 E-11	4.77 E-01	4.23 E-01	1.80 E+05	1.41 E+01	1.09 E+01	7.04 E+00
I-134	5.94 E-02	3.30 E-11	1.94 E+00	4.55 E-01	1.10 E+03	6.41 E+01	1.31 E+01	4.82 E+00
I-135	4.69 E-02	2.77 E-11	1.78 E+00	3.08 E-01	4.40 E+04	4.54 E+01	7.02 E+00	1.52 E+00
Kr-33m	5.90 E-03	1.39 E-11	2.50 E-03	1.03 E-02	--	3.41 E-02	1.18 E-01	--
Kr-35m	1.27 E-02	3.00 E-11	1.51 E-01	2.23 E-01	--	4.27 E+00	5.50 E+00	--
Kr-35	2.53 E-03	5.97 E-12	2.11 E-03	2.23 E-01	--	1.19 E-02	1.10 E+00	--
Kr-37	2.00 E-02	4.72 E-11	1.37 E+00	1.05 E+00	--	6.09 E+01	4.08 E+01	--
Kr-38	3.12 E-02	7.36 E-11	1.74 E+00	3.41 E-01	--	1.21 E+02	2.07 E+01	--
Kr-39	3.96 E-02	9.34 E-11	1.60 E+00	1.33 E+00	--	1.41 E+02	1.02 E+02	--
Xe-131m	2.53 E-04	5.97 E-13	2.00 E-02	1.40 E-01	--	1.13 E-02	6.38 E-02	--
Xe-133m	1.35 E-03	3.18 E-12	3.26 E-01	1.55 E-01	--	9.79 F-01	4.07 E-01	--
Xe-133	5.48 E-02	1.29 E-10	3.00 E-02	1.46 E-01	--	3.88 E+00	1.55 E+01	--
Xe-135m	1.77 E-02	4.17 E-11	4.22 E-01	9.74 E-02	--	1.66 E+01	3.35 E+00	--
Xe-135	5.23 E-02	1.23 E-10	2.46 E-01	3.22 E-01	--	2.86 E+01	3.27 E+01	--
Xe-137	5.31 E-02	1.25 E-10	1.50 E-01	1.37 E+00	--	1.77 E+01	1.41 E+02	--
Xe-138	5.57 E-02	1.31 E-10	1.10 E+00	8.00 E-01	--	1.36 E+02	8.66 E+01	--
						7.11 E+02	4.32 E+02	3.3 E+01

*DCF = dose conversion factor.
**Whole body.
†Skin.

Other conservative assumptions in the analysis include

- no radioactive decay (no decrease in source strength)
- no building wake effects and horizontal plume meandering
- release at ground level versus out the 15.2-m exhaust stack
- average wind speed of 1 m/s versus actual average of 3.4 m/s

The calculated dose rates at 100 m (distance assumed as a reasonable distance at which evacuation and control could be accomplished within 2 hours) for an experiment power of 1 W are as shown in Table 14.5. If it is assumed that an individual is located at this point for 2 hours following the fission product release from a postulated experiment failure, then that individual's resulting radiation dose to the whole body would be 0.51 mrem and 20 mrem committed dose equivalent to the thyroid. These doses are less than 10% of the dose limits in Regulatory Guide 2.2 and well within the applicable limits in 10 CFR 20.

14.6 Conclusion

The staff has reviewed the credible potential accidents for the PUR. On the basis of this review, none of the postulated accidents are expected to release significant fission products to the environment or excessive doses to the reactor personnel or to the public. Therefore, the staff concludes that there is reasonable assurance that any accident resulting from continued operation of the PUR would not pose a significant risk to the health and safety of the public.

Table 14.5 Dose rates at 100 m from a failed fuel experiment (power = 1 W)

	Dose rate		
Isotope	Gamma* (mrem/h)	Beta** (mrem/h)	Thyroid (rem/h)
I-131	1.95 E-03	9.06 E-04	6.98 E-03
I-132	1.78 E-02	2.91 E-03	6.49 E-05
I-133	5.01 E-03	3.88 E-03	2.51 E-03
I-134	2.28 E-02	4.67 E-03	1.71 E-05
I-135	1.65 E-02	2.50 E-03	5.41 E-04
Kr-83m	1.21 E-05	4.20 E-05	--
Kr-85m	1.52 E-03	1.96 E-03	--
Kr-85	4.23 E-06	3.90 E-04	--
Kr-87	2.17 E-02	1.45 E-02	--
Kr-88	4.30 E-02	7.36 E-03	--
Kr-89	5.02 E-02	3.64 E-02	--
Xe-131m	4.01 E-06	2.45 E-05	--
Xe-133m	3.48 E-04	1.45 E-04	--
Xe-133	1.30 E-03	5.54 E-03	--
Xe-135m	5.91 E-03	1.19 E-03	--
Xe-135	1.02 E-02	1.17 E-02	--
Xe-137	6.31 E-03	5.03 E-02	--
Xe-138	4.85 E-02	3.08 E-02	--
Total	2.53 E-01	1.73 E-01	1.01 E-02

*Whole body.
**Skin.

15 TECHNICAL SPECIFICATIONS

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

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

16 FINANCIAL QUALIFICATIONS

The PUR is owned and operated by a State educational institution in support of its role in education and research. On the basis of financial information supplied by the licensee in its submittal of June 30, 1986, as supplemented, 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 staff reviewed the licensee's financial status and found it acceptable in accordance with the requirements of 10 CFR 50.33(f).

17 OTHER LICENSE CONSIDERATIONS

Prior Reactor Utilization

In previous sections of this SER, the staff concluded that normal operation of the reactor causes insignificant risk of radiation exposure to the public, and that only an offnormal or accident event could cause some measurable exposure. Even a maximum hypothetical accident would not lead to a dose to the most exposed individual greater than the applicable guideline values of 10 CFR 20.

The staff concluded in its SER for the original Operating License that the reactor was initially designed and constructed to operate safely. During its current review, the staff considered whether prior operation would cause significant degradation in the capability of components and systems to continue to perform their safety functions. Because fuel cladding is the component most responsible for preventing release of fission products to the environment, possible mechanisms that could lead to detrimental changes in integrity were considered. Prominent among the considerations were the following: (1) radiation degradation of cladding integrity, (2) high fuel temperature or temperature cycling leading to changes in the mechanical properties of the cladding, (3) corrosion or erosion of the cladding leading to thinning or other weakening, (4) mechanical damage resulting from handling or experimental use, and (5) degradation of safety components or systems.

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

(1) Nearly identical fuel has been laboratory tested elsewhere and has been exposed under similar irradiation conditions to much higher total radiation doses in operating reactors, such as at the Oak Ridge Research Reactor and the Omega West Reactor (Los Alamos National Laboratory). No significant degradation of cladding has resulted in any of these reactors.

(2) The power density, coolant flow rates, and maximum temperature reached in the PUR fuel are far below similar parameters in some other nonpower reactors using similar fuel. No damage has occurred during normal operations in any of these reactors.

(3) The coolant flow rate at PUR is much lower than that used at several higher powered research reactors using MTR-type fuel. No erosion problems have been observed. At the PUR facility, corrosion is kept to a reasonable minimum by careful control of the conductivity and pH of the primary coolant water.

(4) The fuel is handled as infrequently as possible, consistent with required surveillance. Any indications of possible damage or degradation are investigated immediately, and damaged fuel would be removed from service in accordance with Technical Specifications. All experiments placed near the core are isolated from the fuel cladding by a water gap and at least one barrier or encapsulation.

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

18 CONCLUSIONS

On the basis of its evaluation of the application as set forth in the previous sections, the staff has determined that

(1) The application filed by Purdue University for renewal of Operating License R-87 for its research reactor, dated June 30, 1986, 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 I.

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

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

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

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

19 BIBLIOGRAPHY

American Nuclear Society (ANS), 5.1, "Decay Heat Power in Light Water Reactors," 1978.

---, 15.1, "Standard for the Development of Technical Specifications for Research Reactors," 1982.

---, 15.4, "Selection and Training of Personnel for Research Reactors," 1977.

---, 15.11, "Radiological Control at Research Reactor Facilities," 1977.

---, N401-1974/15.6, "Review of Experiments for Research Reactors," 1974.

Code of Federal Regulations, Title 10, "Energy" (including General Design Criteria), U.S. General Services Administration, Office of Federal Register, National Archives and Records Service, U.S. Government Printing Office, Washington, D.C., January 1987.

Dietrich, J. R., "Experimental Determinations of the Self-Regulation and Safety Operating Water-Moderated Reactors," A/ConF.8/P/481, Argonne National Laboratory, Argonne, IL, June 30, 1955.

Forbes, S. G., et al, "Instability in the SPERT I Reactor, Preliminary Report," IDO-16309, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, October 1956.

Miller, R. N., et al, "Report of the SPERT I Destructive Test Program on an Aluminum, Plate-Type, Water-Moderated Reactor," IDO-16883, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, June 1964.

Nyer, W. E., et al "Experimental Investigations of Reactor Transients," IDO-16285, Idaho Operations Office, Atomic Energy Commission, Idaho Falls, ID, April 1956.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG-1283

2. TITLE AND SUBTITLE
Safety Evaluation Report related to the renewal of the operating license for the research reactor at Purdue University

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
April	1988

5. AUTHOR(S)

6. DATE REPORT ISSUED

MONTH	YEAR
April	1988

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Standardization and Non-Power Reactor Project Directorate
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

11a. TYPE OF REPORT
Safety Evaluation Report

b. PERIOD COVERED (Inclusive dates)

12. SUPPLEMENTARY NOTES

Pertains to Docket No. 50-182

13. ABSTRACT (200 words or less)

This Safety Evaluation Report for the application filed by Purdue University for a renewal of Operating License R-87 to continue to operate a research reactor has been prepared by the Office of Nuclear Reator Regulation of the U. S. Nuclear Regulatory Commission. The facility is owned by Purdue University and is located on the campus in West Lafayette, Indiana. On the basis of its technical review, the staff concludes that the reactor facility can continue to be operated by the University without endangering the health and safety of the public or the environment.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

Non-Power Reactor
Purdue University
License Renewal

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

*U.S.GOVERNMENT PRINTING OFFICE:1988-202-292:80041

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
PERMIT No. G-67

120555078877 1 1AN
US NRC-OARM-ADM
DIV OF PUB SVCS
POLICY & PUB MGT BR-PDR NUREG
W-537
WASHINGTON DC 20555

AT PURDUE UNIVERSITY