NUREG-0996

Safety Evaluation Report

related to the renewal of the operating license for the University of Oklahoma Research Reactor

Docket No. 50-112

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

September 1983

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ABSTRACT

This Safety Evaluation Report for the application filed by the University of Oklahoma for a renewal of Operating License R-53 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 and operated by the University of Oklahoma and is located on the campus in Norman, Cleveland County, Oklahoma. The staff concludes that the Aerojet General Nucleonics (AGN) reactor facility can continue to be operated by University of Oklahoma without endangering the health and safety of the public.

TABLE OF CONTENTS

		Page
ABSTRACT		iii
1	INTRODUCTION	1-1
	1.1 Summary and Conclusions of Principal Safety Considerations	1-2
	1.2 Reactor Description	1-3
	1.3 Reactor Location	1-3
	1.4 Shared Facilities and Equipment	1-3
	1.5 Comparison With Similar Facilities	1-3
	1.6 Nuclear Waste Policy Act of 1982	1-4
2	SITE CHARACTERISTICS	2-1
	2.1 Geography	2-1
	2.2 Demography	2-1
	2.3 Nearby Industrial, Transportation, and Military Facilities	2-1
	2.4 Meteorology	2-1
	2.5 Geology	2-2
	2.6 Hydrology	2-2
	2.7 Seismology	2-3
	2.8 Conclusion	2-3
3	DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS	3-1
	3.1 Wind Damage	3-1
	3.2 Water Damage	3-1
	3.3 Mechanical System and Components	3-1
	3.4 Conclusion	3-1
4	REACTOR	4-1
	4.1 Facility Description	4-1
	4.2 Reactor Core	4-2
	4.2.1 Core Assembly	4-2
	4.2.2 Fuel Elements	4-2
	4.2.3 Safety and Control Rods	4-3
	4.2.4 Conclusion	4-3
	4.3 Reactor Tank Shielding	4-3
	4.4 Reactor Operation	4-4

TABLE OF CONTENTS (Continued)

			Page
	4.5	Dynamic Design Evaluation	4-4
		4.5.1 Reactivity Limit	4-4
		4.5.2 Fuel Temperature	4-5
		4.5.3 Conclusion	4-5
	4.6	Operational Practices	4-5
	4.7	Conclusions	4-6
5		REACTOR COOLING SYSTEM	5-1
	5.1	Reactor Core Cooling System	5-1
	5.2	Coolant Purification System	5-1
	5.3	Conclusion	5-1
6		ENGINEERED SAFETY FEATURES	6-1
7		CONTROL AND INSTRUMENTATION SYSTEMS	7-1
	7.1	Systems Summary	7-1
	7.2	Control System	7-1
		7.2.1 Nuclear Control System	7-1
		7.2.2 Process Control System	7-2
	7.3	Instrumentation System	7-2
		7.3.1 Nuclear Instrumentation	7-2
		7.3.2 Process Instrumentation	7-2
	7.4	Conclusions	7-3
8		ELECTRIC POWER SYSTEM	8-1
	8.1	Main Power	8-1
	8.2	Emergency Power	8-1
	8.3	Conclusions	8-1
9		AUXILIARY SYSTEMS	9-1
	9.1	Fuel Handling and Storage	9-1
	9.2	Fire Protection System	9-1
	9.3	Ventilation System	9-1
	9.4	Conclusions	9-1
10		EXPERIMENTAL PROGRAMS AND FACILITIES	10-1
	10.1	Experimental Programs	10-1
	10.2	Experimental Facilities	10-1

TABLE OF CONTENTS (Continued)

Page

10.2.1 Fixed-Location Facilities 10-1
10.2.2 Variable-Location Facilities 10-2

10.3 Conclusion 10-2

11 RADIOACTIVE WASTE MANAGEMENT 11-1

11.1 Airborne Wastes 11-1
11.2 Solid Wastes 11-1
11.3 Liquid Wastes 11-1
11.4 Conclusion 11-1

12 RADIATION PROTECTION PROGRAM 12-1

12.1 ALARA Commitment 12-1
12.2 Health Physics Programs 12-1
12.3 Radiation Sources 12-2
12.4 Routine Monitoring 12-2
12.5 Personnel Radiation Monitoring 12-3
12.6 Potential Dose Assessments 12-3
12.7 Conclusion 12-3

13 CONDUCT OF OPERATIONS 13-1

13.1 Overall Organization 13-1
13.2 Training 13-1
13.3 Emergency Planning 13-1
13.4 Operational Review and Audit 13-1
13.5 Physical Security Plan 13-1
13.6 Conclusion 13-2

14 ACCIDENT ANALYSIS 14-1

14.1 Summary 14-1
14.2 Tornadoes 14-1
14.3 Rapid Insertion of Reactivity (Nuclear Excursion) . . . 14-1

14.3.1 Maximum Power Excursion Scenarios 14-
14.3.2 Conclusion 14-2

14.4 Loss-of-Coolant/Shielding Water 14-2
14.5 Effects of Fuel Aging 14-4
14.6 Maximum Hypothetical Accident 14-5

14.6.1 Fuel Loading Accident 14-5
14.6.2 Conclusion 14-6

15 TECHNICAL SPECIFICATIONS 15-1

TABLE OF CONTENTS (Continued)

		Page
16	FINANCIAL QUALIFICATIONS	16-1
17	OTHER LICENSE CONSIDERATIONS	17-1
	17.1 Prior Reactor Utilization	17-1
	17.2 Multiple or Sequential Failures of Safety Components	17-2
18	CONCLUSIONS	18-1
19	REFERENCES	19-1

LIST OF FIGURES

1.1	Reactor Assembly	1-5
1.2	University of Oklahoma Main Campus	1-6
4.1	First-Floor Plan of Nuclear Engineering Laboratory Building	4-7
4.2	Reactor Area Floor Plan	4-8
4.3	Experimental Facilities In Core	4-9
4.4	Control and Safety Fuel-Element Assembly	4-10
13.1	Administrative Organization	13-3

LIST OF TABLES

7.1	Scram-Producing Reactor Safety Channels	7-4
12.1	Recent Exposure History of Reactor Personnel	12-4
14.1	Radiation Exposures Following Loss of Tank Water for Two Reactor Operation Times at 100 W	14-6

1 INTRODUCTION

The University of Oklahoma (UO) (licensee) submitted a timely application to the U.S. Nuclear Regulatory Commission (NRC) (staff) to increase thermal power level from 15 W to 100 W and for renewal of the Class 104 Operating License (R-53) for a period of 20 years for its Aerojet General Nucleonics (AGN) 211P (hereinafter called UO AGN or facility) research reactor by letter dated October 6, 1978, with supporting documentation and supplements thereafter. The University of Oklahoma has been operating the AGN research reactor since it was initially licensed December 29, 1958. The university currently is permitted to operate the reactor within the conditions authorized in past amendments in accordance with Title 10 of the Code of Federal Regulations, Paragraph 2.109 (10 CFR 2.109), until NRC action on the renewal request is completed.

The staff technical safety review, with respect to issuing a renewal operating license to the UO, has been based on the information contained in the renewal application and supporting supplements, plus responses to requests for additional information. The renewal application as supplemented includes the Physical Security Plan, Technical Specifications, Environmental Impact Analysis, Safety Analysis Report, Reactor Operator Requalification Program, and Emergency Plan. This material, with the exception of the Physical Security Plan, is available for review at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

The renewal application contains the information regarding the original design of the facility and includes information about modifications to the facility made since initial licensing. The Physical Security Plan is protected from public disclosure under 10 CFR 2.790(d) (1) and 10 CFR 9.5(a)(4).

The purpose of this Safety Evaluation Report (SER) is to summarize the results of the safety review of the UO AGN research reactor and to delineate the scope of the technical details considered in evaluating the radiological safety aspects of continued operation. The SER will serve as the basis for renewal of the license for operation of the UO reactor at steady-state thermal power levels up to 100 W.

The facility was reviewed against the requirements of 10 CFR 20, 30, 50, 51, 55, 70, and 73; applicable regulatory guides (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 dose values with related standards in 10 CFR 20, the standards for protection against radiation during normal operation, both for employees and the public.

This Safety Evaluation Report was prepared by Harold Bernard, Project Manager, Division of Licensing, Office of Nuclear Reactor Regulation, Nuclear Regulatory Commission. Major contributors to the technical review include the project manager and A. Chu of the NRC staff and C. Thomas and G. Linder of Los Alamos National Laboratory (LANL) under contract to the NRC.

1.1 Summary and Conclusions of Principal Safety Considerations

The staff 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 Office of Inspection and Enforcement, and onsite observations.

The principal matters reviewed for the UO AGN research reactor and the conclusions reached were the following:

(1) The design, testing, and performance of the reactor structures, systems, and components important to safety during normal operation are inherently safe, and safe operation can reasonably be expected to continue at the higher power level of 100 W thermal.

(2) The expected consequences of a broad spectrum of postulated credible accidents have been considered, emphasizing those likely to cause release of fission products from fuel element cladding failure. The staff performed conservative analyses of serious credible accidents and determined that the calculated potential radiation doses outside of the reactor room are a small fraction of the levels in 10 CFR 20 for doses in unrestricted areas.

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

(4) The systems provided for 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 is reasonably achievable (ALARA).

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

(6) The financial data and information provided by the licensee are such that the staff has determined that the licensee has sufficient revenues to cover operating costs and to ensure protection of the public from radiation exposures when operations are terminated.

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

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

(9) The licensee has submitted an Emergency Plan dated October 27, 1982, using guidance that was current at the time of license renewal application in accordance with the requirements of 10 CFR 50.54.

1.2 Reactor Description

The UO AGN research reactor is a small pool-type reactor that has been in operation at 15W thermal since 1959. The fuel elements are composed of a homogeneous mixture of 20% enriched uranium dioxide particles and polyethylene moderator. The elements are coated with epoxy to retain fission products. The critical mass of the reactor is about 785 gms for the standard configuration and 1,154 gms for the mode using the flux trap. The core assembly, containing the fuel elements and 32 graphite reflector elements, is suspended in an 800-gal water-filled cylindrical aluminum tank. The 8-ft-high tank is mostly below floor level and is shielded with concrete blocks. Beam tubes emerge from the concrete shield into an experimental pit area below the floor as shown in Figure 1.1. A fuel storage pit on the opposite side of the reactor pool allows for storage of spare and spent fuel elements.

The reactor can be used in two configurations: one a standard 3 by 4 close-packed configuration and the other one using a flux trap and consisting of 15 to 20 fuel elements.

The reactor has operated safely at 15 W power level for 23 years, without incident, using the standard design of 10 fuel elements. In 1982, the UO received an amendment to the license permitting a core configuration using a flux trap and up to 20 fuel elements.

1.3 Reactor Location

The reactor is located in the Nuclear Engineering Laboratory (NEL) Building, which is on the Norman Campus of the University of Oklahoma. The NEL Building is constructed of brick and reinforced concrete. The building is bordered by a street on the west and parking areas on the north, east, and south. The location of the NEL Building on the campus is shown in Figure 1.2.

The State Mental Hospital is located about 1.6 mi northeast of the reactor site east of the Norman City limit. The hospital has approximately 1,000 acres of land and 111 buildings for about 955 employees and about 3,200 patients.

The Cerebral Palsy Institute is located approximately 2 mi northeast of the reactor site.

The U.S. Naval Air Technical Training Center is about 1 mi south of the reactor location. About 3,600 persons are employed or stationed at the Center.

1.4 Shared Facilities and Equipment

A graphite pile and a subcritical assembly that are used for reactor physics and reactor engineering courses also are contained in the reactor room. The remainder of the building is used for laboratories, classrooms, and offices. A second floor room, directly over the reactor pool, is controlled by the reactor staff and is used for storage.

1.5 Comparison With Similar Facilities

The UO AGN research reactor was designed and built by Aerojet General Nucleonics as a standard teaching system. The UO AGN (pool-type) reactor is now the only remaining one of its kind in operation in the U.S. The reactor fuel from the AGN-211 reactor at the University of West Virginia, Morgantown, that had been operated at a power level of 75 W for a number of years, was recently shipped to the UO facility. An AGN-211 reactor at the University of Basel (Switzerland) has a nominal power of 2 kW and limited operation to 1 kW with the polyethelene-coated fuel particles. In order to operate at power levels above 1 kW, the University of Basel reactor replaced the AGN polyethelene fuel with metal-clad UO_2 elements.

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 licensee shall have entered into an agreement with the Department of Energy (DOE) for the disposal of high-level radioactive waste and spent nuclear fuel. DOE (R.L. Morgan) has informed the NRC (H. Denton) by letter dated May 3, 1983, that it has determined that universities and other government agencies operating nonpower reactors have entered into contracts with DOE that provide that DOE retain title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing.

Because the University of Oklahoma has a contract with the Department of Energy, it is in conformance with the Waste Policy Act of 1982.

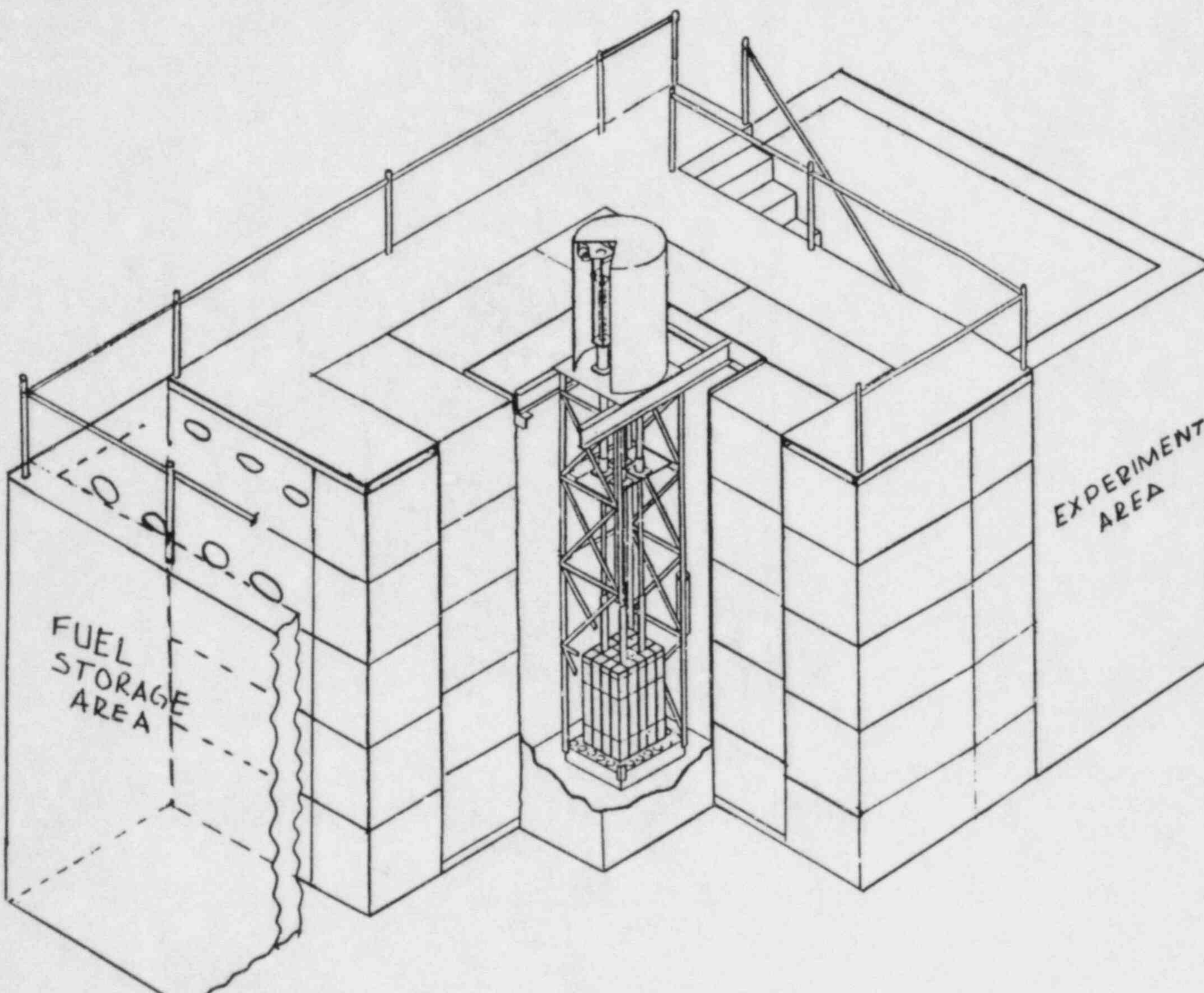

Figure 1.1 Reactor assembly

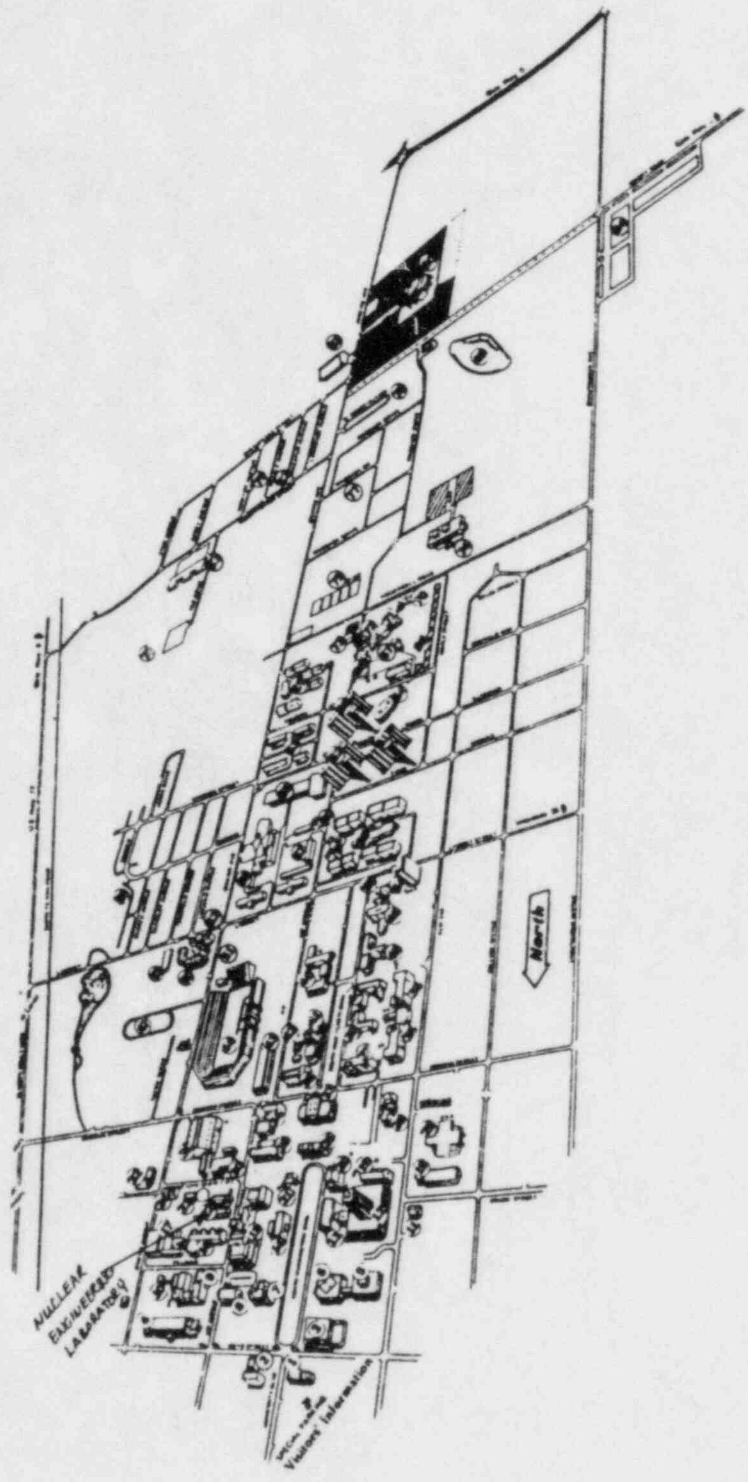

Figure 1.2 University of Oklahoma main campus

2 SITE CHARACTERISTICS

2.1 Geography

The University of Oklahoma is located in the city of Norman, Cleveland County, Oklahoma. Norman is approximately 22 mi from Oklahoma City, the state's largest city. The reactor site is on the university's main campus, inside the city limits, and approximately 3/4 mi from the Main Street business district. The surrounding land is used primarily for residential purposes. Norman is situated on a plateau overlooking the valley of the South Canadian River. This river is essentially dry except during extended periods of rainfall in its drainage area. The river bed is located approximately 3 mi south of the city limit.

2.2 Demography

The population of Norman is approximately 72,000; the UO campus population consists of about 22,000 students and 6,500 staff, including part-time and temporary employees (part of whom are students). The remainder of Cleveland County has a population of approximately 7,000 to 10,000.

2.3 Nearby Industrial, Transportation, and Military Facilities

There is no heavy industry nearby. The Oklahoma City plant of the Oklahoma Gas and Electric Company is located approximately 22 mi north of the reactor site. The UO operates its own electric power plant, which is located about 500 ft east of the reactor site.

The major transportation route in the area is State Highway 9, which passes to the east of the city limit. U.S. Highways 40 and 35 are also in the vicinity of Norman.

The U.S. Naval Air Technical Training Center is approximately 1 mi south of the reactor site.

In view of the safe operating history of the past 23 years and the nature and location of nearby industrial, transportation, and military facilities, the staff concludes that these facilities pose no significant risk to the safe operation of the UO reactor.

2.4 Meteorology

Oklahoma, because of its geographical location, is in a transition zone between subtropical climates to the south and the colder continental climates to the north. Occasionally during the winter, Arctic storms move into the State, producing weather disturbances and low temperatures. Moist air masses from the Gulf of Mexico move inland in a north-northeast direction. The State also is a part of the transition area between the more humid eastern United States and the more arid areas of the southwest. The United States Weather Bureau takes complete surface observations at Will Rogers Airport, 18 mi northwest of Norman. Personnel at Tinker Air Force Base also make weather observations.

The Norman, Oklahoma, area is subject to tornadoes. However, there is a tornado-alert system that informs the region of the possibility of such an event. Procedures require that the reactor be shut down and not operated during such an alert.

The average annual precipitation--nearly all as rainfall--at Norman (1894-1956) is 33.22 in. The maximum annual precipitation recorded to date is 55.83 in. in 1923. The minimum of 18.03 in. precipitation was recorded in 1954. Months of average maximum precipitation are May and June, while the average minimum precipitation occurs in the months of January and February.

Temperatures in the State vary from day to day. July and August are the hottest months with temperatures averaging about 82°F. January is the coldest month with an average temperature of 38°F. December, January, and February will include temperatures below freezing. Temperatures below 0°F usually occur in the State sometime during each winter.

2.5 Geology

The bedrock throughout the area is Hennessey Shale of Permian Age. The shale is red, compact, noncalcareous and impermeable. It reaches to a depth of 500 ft at the reactor site.

The soil is Bethany silt loam. Surface and internal drainage are very slow. The soil has a high-water-holding capacity and absorbs most of the precipitation.

Surface soil to a depth of about 15 in. is a dark grayish brown, granular, slightly acid silt loam. This grades into the upper subsoil, 4 to 8 in. thick, which is neither tight nor hard and is readily penerated by moisture. The upper subsoil grades into a lower subsoil of brown, firm, blocky clay that continues to depths of 40 to 50 in. with little change. Next in profile is a brown, heavy, noncalcareous clay, mottled with yellow and reddish brown. It grades at depths of 6 to 8 ft into alkaline, calcareous, reddish silty clay or silty shale.

Structurally the site lies near the southern end of the Nemaha Uplift. The McClain County fault zone, which consists of a number of fault-bounded blocks that are about 25 mi by 37 mi long, underlies the site (NUREG/CR-3117). There is no geologic evidence that indicates the fault zone is younger than Paleozoic (at least 240 million years). On the basis of the available evidence, there are no capable faults as described in Appendix A, 10 CFR 100, in the site vicinity.

2.6 Hydrology

The reactor site and immediate surrounding area are essentially flat. Minor slopes are produced by street construction. Drainage from the reactor site is into a storm sewer system, eventually reaching the South Canadian River bed some 3 mi south of the reactor location.

The soil is relatively impervious and permits little ground water movement. There are no shallow wells in the area using water from the soil zones. Because of the impervious nature of the shallow zones there is no danger of contamination of the Garber Sandstone Reservoir, the source of the local potable water supply.

The reactor pool always contains water, and as the fuel storage pit arrangement has been designed against critically considering flooding; flooding resulting from either hydrological anomalies or leakage from the core, therefore, will not produce any radiological accidents or consequences.

2.7 Seismology

Norman, Oklahoma, is in Cleveland County, which is a relatively active seismic area. The largest historical earthquake within 200 mi of Norman is the April 9, 1952, magnitude 5.5 maximum Modified Mercalli (MM) intensity VII event which occurred about 25 miles from the site. This earthquake resulted in an MM intensity of V in Norman (Murphy and Cloud, 1954). MM intensity V is described as "Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbances of trees, poles and other tall objects sometimes noticed. Pendulum clocks may stop." Even if an earthquake of this size (MM VII) were to occur near the site, damage would be negligible in buildings of good design and construction and slight to moderate in well-built ordinary structures.

Accordingly, the staff concludes that on the basis of the historic seismicity coupled with the inherent safety of the reactor, there is no significant hazard to the reactor, operating personnel, or the contiguous public.

2.8 Conclusion

On the basis of the above information, the staff 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

3.1 Wind Damage

The University of Oklahoma Norman Campus is an area subject to tornadoes. However, Norman has a city-wide tornado alert system (sirens) which, according to the licensee, operates very effectively. Procedures require that the reactor not be operated if a tornado alert is in effect.

3.2 Water Damage

The natural drainage and storm sewers at the reactor site appear to be effective as there has been no flooding in the past history of the University of Oklahoma. Because of that past history and the design and configuration of the reactor, the staff concludes that water damage to the reactor by flood or ground water is unlikely.

3.3 Mechanical System and Components

The reactor unit consists of a core assembly in a water-filled tank suspended by means of an aluminum frame attached to a steel I-beam bridge mounted at the top of the tank. The mechanical system of importance to safety is the control rod actuator assembly, which is mounted on the bridge with the vertical absorber control rods mounted beneath.

3.4 Conclusion

The UO facility was designed and built to withstand all credible and probable wind and water damage contingencies associated with the site. As stated in Section 2.7, a seismic event has a small likelihood of occuring and the consequences of such an event would be minimal and, therefore, need not be evaluated explicitly.

4 REACTOR

The University of Oklahoma (UO) reactor is an AGN research reactor that had been manufactured by Aerojet General Nucleonics. The AGN-211 reactor series are swimming-pool-type research reactors using uranium enriched to 20% in the isotope ^{235}U as fuel mixed in a polyethylene matrix. The UO AGN research reactor is equipped with a 7/8-in.-diameter horizontal glory hole, a 4-in.-diameter horizontal beam port, and a 4-in.-diameter access port. The reactor currently is licensed to operate at a steady-state power level of 15 W; the licensee, however, has requested permission in the license renewal application to operate at 100 W and has performed analyses in support of this request. The staff has reviewed the license renewal application and supporting documentation of various potential accidents to the reactor during steady-state operation at 100 W.

The UO reactor is used for undergraduate and graduate-level training in nuclear engineering, as well as for research in nuclear technology.

The discussion in the following sections is based on information obtained from reports and documents submitted by the licensee and Aerojet General Nucleonics and discussions with UO personnel.

4.1 Facility Description

The reactor is located on the first floor in the Nuclear Engineering Laboratory Building (Figure 4.1). As shown in Figure 1.1, the reactor and its concrete-block shielding are installed in a below-ground pit that is approximately 30.5 ft long and 10.25 ft wide. An experimental area and a fuel storage pit are located on opposite sides of the reactor in the pit (Figure 4.2). Two 4-in. beam tubes emerge from the concrete shield into the experimental area (Figure 4.3). The floor of the experimental area is approximately 10 ft below the reactor room floor. The fuel storage pit is used for storage of spare fuel elements.

Access to the reactor area (Rooms 103 through 107A and B, Figure 4.1) is limited, and keys are assigned only to members of the reactor staff. All persons entering the reactor area during reactor operation are required to wear personnel dosimeters.

Outside the reactor area and immediately adjacent to it are graduate student offices and nuclear engineering laboratories as shown in Figure 4.1. These facilities occupy the entire north half of the first floor. The south half of the first floor is occupied by Computer and Personnel Department personnel.

The second floor of the building is occupied by the UO Personnel Department staff. However, the room directly over the reactor pool is controlled by the reactor staff and is used for storage. Operating procedures require that the room be locked before startup and during reactor operation. This is a precautionary measure because the measured gamma-ray dose rate in the center of the room with the reactor operating at 15 W is about 0.09 mR/h with no detectable neutron dose. The projected gamma-ray dose rate at the proposed 100-W power level is 0.6 mR/h. The staff notes that the licensee plans to use 1 in. of steel plate or equivalent gamma shielding on top of the reactor whenever the reactor will operate at 50 W or more, to reduce the gamma-ray dose rate in the second floor storage room below this projected 0.6 mR/h level.

4.2 Reactor Core

The reactor core can be arranged in two configurations; (1) a standard arrangement that contains 11 fuel elements plus a 1/3 element (785 gms) and (2) a flux-trap arrangement that contains up to 20 fuel elements. A fine and a coarse control rod, 2 safety rods, and a variable number of graphite reflector elements complete the core. The core assembly is suspended in a water-filled (800-gal) tank by means of an aluminum frame attached to a steel I-beam bridge mounted on the top of the tank. The control rod actuator assemblies are mounted on the bridge. The reactor core is in a 3/8-in.-thick, 40-in. by 60-in. by 8-ft-high aluminum tank shielded by a number of 20-in. by 20-in. by 40-in. concrete blocks. A platform and railing are secured to the top of the reactor.

4.2.1 Core Assembly

The core assembly consists of 2-7/8-in. by 3-1/6-in. by 29-in. fuel elements. The base of each element has an aluminum guide that fits into the aluminum grid plate. The grid plate is essentially identical to the one used in the Oak Ridge Bulk Shielding Facility and is supported from the I-beam bridge by an aluminum frame. The bridge also acts as a support for the ion chambers.

The fuel elements are removable from above by grasping the element handle with the proper removal tool. The control rods fit into the core assembly in such a manner that a control rod cannot be removed without removing a fuel element.

The licensee's Technical Specifications limit an operating reactor core to a maximum of 20 fuel elements in any core configuration. Typical core configurations are (1) a standard 12-fuel-element parallelepiped with a graphite or water reflector, or (2) a flux-trap configuration with up to 20 fuel elements with a graphite or water reflector.

The lattice is arranged so that each fuel element in the core assembly is surrounded by 1/8-in. of water. Operation at a power level of 1 W results in a temperature rise at the center of the element of approximately 0.04C°; hence, at 100 W a fuel temperature equilibrium of approximately 4C° over ambient is expected. (This temperature increase will reduce the excess reactivity by about 0.1% $\Delta k/k$.)

4.2.2 Fuel Elements

Each fuel element is 2-7/8 in. by 3-1/16 in. by 29 in. and has a bottom nose-piece and a top handle. The center 10 in. of each element is composed of the homogenous mixture of UO_2 in polyethylene (56 mg $^{235}U/cm^3$) in the form of 20-m-diameter particles. The density of the UO_2 is 318 mg/cm^3.

A fuel element, as shown in Figure 4.4 is constructed as follows:

(1) bottom aluminium mounting section
(2) 5 in. of graphite reflector

(3) 10 in. of fuel
(4) 5 in. of graphite reflector
(5) 2 in. of lead shadow sheild
(6) the top aluminum removal fitting

The licensee has obtained 12 additional fuel elements from the University of West Virginia, as discussed in Section 1.5.

4.2.3 Safety and Control Rods

The reactor has a fine adjustment rod and a coarse control rod plus two safety rods. The safety rods are made of boral, the coarse control rod is made of aluminium-clad cadmium, and the fine adjustment control rod is made of stainless steel. The control rods are in the form of 2-3/4-in.-wide blades that fit into slots in the edge of four specially constructed fuel elements (Figures 4.4). As noted previously, the slot is designed so that it is necessary to remove a fuel element before the control rod can be removed.

During operation, the actuator rods of the control rods are held to the rod drives by electromagnets. A scram signal deenergizes the holding magnets, allowing the safety rods and coarse control rod to be driven into the core under the combined influence of gravity and the scram springs. The total rod injection time is less than 500 msec. The rods are decelerated by a dashpot in the last 2 cm of travel. The fine control rod is rigidly mounted to its carriage and is driven into the core by the automatic return system that drives the magnet/lead screw mechanisms to their full "in" position after a scram. All rod operation is manual (no automatic control is provided). Operational interlocks and the scram circuit logic are described in Section 7.

The reactivity worth of the coarse control rod and each of the safety rods depends on the core configuration and varies from approximately 1 to 1.5% $\Delta k/k$. The fine control rod is worth approximately 0.45% $\Delta k/k$ in any core configuration. The maximum speed of travel of the rods is 0.3 cm/sec, yielding a maximum reactivity change of about 2.8×10^{-4} $\Delta k/k$/sec.

The UO has a complete duplicate of the control and safety rod system that was obtained from the University of West Virginia in 1980. This spare system gives the licensee the capability of replacing the system currently in use in total or as components fail.

4.2.4 Conclusion

The staff has reviewed the information regarding the reactor fuel and the safety and control rod systems and concludes that the design and performance capability of the components and the onsite availability of replacement components is adequate to ensure the safe operation of the reactor at power levels up to 100 W for the proposed license renewal period.

4.3 Reactor Tank Shielding

Shielding in the radial direction is provided by the 40-in. by 60-in. water tank and by 40 in. of ordinary concrete. The concrete shielding is provided by solid blocks that are 20 in. by 20 in. by 40 in. and are placed in such a manner that the cracks overlap and the integrity of the shield is not broken. Shielding in the vertical direction is provided by about 5 in. of graphite and 2 in. of lead attached to the top of each fuel element (Figure 4.4) and by about 5 ft of water above the top of the reactor core. For operation at power levels of 50 W or more, 1 in. of steel plate or equivalent gamma shielding will be placed across the top of the pool. This will ensure that the radiation level at the control console will be no more than 0.1 mR/h with the reactor operating at 100 W.

The staff concludes that the inherent shielding of the UO AGN research reactor and the additional shielding planned for operations at power levels of 50 W or more is adequate to protect the environment and the health and safety of the general public and the reactor staff.

4.4 Reactor Operation

The operation of the reactor core is monitored by two separate detector channels. These channels use gamma compensated ion chambers to monitor the neutron flux density and provide trip signals to the safety circuits. The instrumentation is described in more detail in Section 7.

4.5 Dynamic Design Evaluation

The safe operation of an AGN research reactor is accomplished by the control and safety rods and is monitored by the core power-level (neutron density) detectors. The reactor's inherent negative temperature coefficient provides backup reactivity control and serves to self-limit any potential excursion.

4.5.1 Reactivity Limit

The licensee's Technical Specifications limits the total excess reactivity, including the static positive reactivity worth of any experiment, to not more than 0.65% $\Delta k/k$ for any pool temperature. This is well below the value necesary to go prompt critical. The total excess reactivity limitation applies to both the standard and flux-trap core configurations discussed in Section 4.2.1. Reactor fuel loading constraints are used to control the total core excess reactivity. Accidents resulting from the sudden removal of cadmium from the glory hole or insertion of a fuel element into a flux-trap are analyzed in Section 14.

The minimum shutdown margin as required by the licensee's Technical Specifications is 0.5% $\Delta k/k$ with the safety or control rod of highest worth fully removed. The temperature coefficient of reactivity, obtained by heating the pool water to equilibrium and then taking measurements while cooling, was determined to be negative 2×10^{-4} $\Delta k/k$ per Centigrade degree in the range of 23 to 28°C. This large, negative temperature coefficient is a reflection of the large role played by the polyethelene when it heats up.

The licensee states that the limitations on total core excess reactivity ensure reactor periods of sufficient length so that the reactor protection system will be able to shut the reactor down without exceeding any safety limits. The licensee also states that the shutdown margin and control and safety rod reactivity limitation ensure that the reactor can be brought and maintained subcritical if the highest worth rod fails to scram and remains in its most reactive position. The staff concurs with the licensee's findings.

4.5.2 Fuel Temperature

Aerojet General Nucleonics (AGN-211 Preliminary Design Report, undated and Reactor Hazards Evaluation Report, 1957) calculated the temperature rise at the centerline of the fuel element with the reactor operating at 1 W. The elements were treated as 3-in. slabs. The calculated temperature rise above the surface temperature was 0.07F° (0.04C°). The extrapolated temperature rise at the centerline of a fuel element at 100 W reactor operation is 7F° (3.9C°). Assuming a surface temperature of 68°F (20°C) the centerline fuel temperature would be about 75°F (23.9°C). Thus, the estimated centerline fuel temperature for operation at 100 W is well below the Technical Specifications safety limit of 392°F (200°C).

The licensee has noted that the AGN reactor at the University of West Virginia was operated routinely at a 75-W power level for a number of years with no apparent effect in the fuel and that the University of Basel (Switzerland) operated an AGN at 1,000 W for short periods of time with standard polyethylene fuel without any detrimental effects on the fuel.

On the basis of the above information, the licensee states that the reactor can be operated at 100 W with reasonable assurance that there will be no loss of fuel integrity and no significant change in fission product retention. The staff concurs with the licensee's findings.

4.5.3 Conclusion

The staff concludes that the inherent, large, negative temperature coefficient of the UO AGN research reactor provides a basis for the safe operation of the reactor for the period of the license renewal.

Furthermore, the staff concludes that (1) the limitation on total excess reactivity including the static positive reactivity of all experiments and samples with all safety and control rods fully withdrawn of 0.66% $\Delta k/k$ at any pool temperature, (2) the limitation on the reactivity worth of the control and safety rods so that withdrawal of the coarse control or either safety rod will not result in criticality, and (3) operation in compliance with Technical Specification minimum shutdown margin requirements provides assurance that the UO experimental and training programs will pose no threat to the health and safety of the public. In additon, the staff concludes that the minimum shutdown margin of 0.5% $\Delta k/k$ with the most reactive safety or control rod fully withdrawn is sufficient to ensure that the reactor can be adequately shut down under all likely conditions. The staff also concludes that the fuel temperatures with the reactor operating at 100 W are low enough to ensure that there will be no loss of fuel integrity resulting from the proposed increase in power level.

4.6 Operational Practices

The UO has implemented a preventive maintenance program that is supplemented by a detailed preoperational checklist to ensure that the reactor is not operated at power without the appropriate reactor safety components operable as described in Section 7 and delineated in Table 7.1. The reactor is operated by or under the immediate supervision of NRC-licensed personnel in accordance with explicit operating procedures that include specific responses to any reactor control signal. All proposed experiments, procedures and procedure changes, and proposed modifications to the reactor and its associated components are reviewed by the Reactor Safety Committee for potential effects on the reactivity of the core, or damage to it, as well as for possible effects on the health and safety of employees and the general public and must be approved by the committee before implementation.

4.7 Conclusions

On the basis of the information presented above, review of operating records, and a site visit, the staff concludes that the UO AGN research reactor is designed and built according to good industrial practices. It consists of standardized components representing many reactor-years of operation and includes redundancy of safety-related systems.

The staff review of the UO AGN reactor facility has included studying its specific design and installation, its controls and safety instrumentation, its specific preoperational and operating procedures, its operational limitations as identified in the original and revised Technical Specifications, and all other pertinent documents associated with the license review. Review of reactor-related exposures of personnel above indicate exposures that are a small percentage of 10 CFR 20 regulations and no releases of radioactivity to the environment. Furthermore, the staff considers that there is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public and the reactor staff and students during reactor operations at power levels up to 100 W for the requested renewal period.

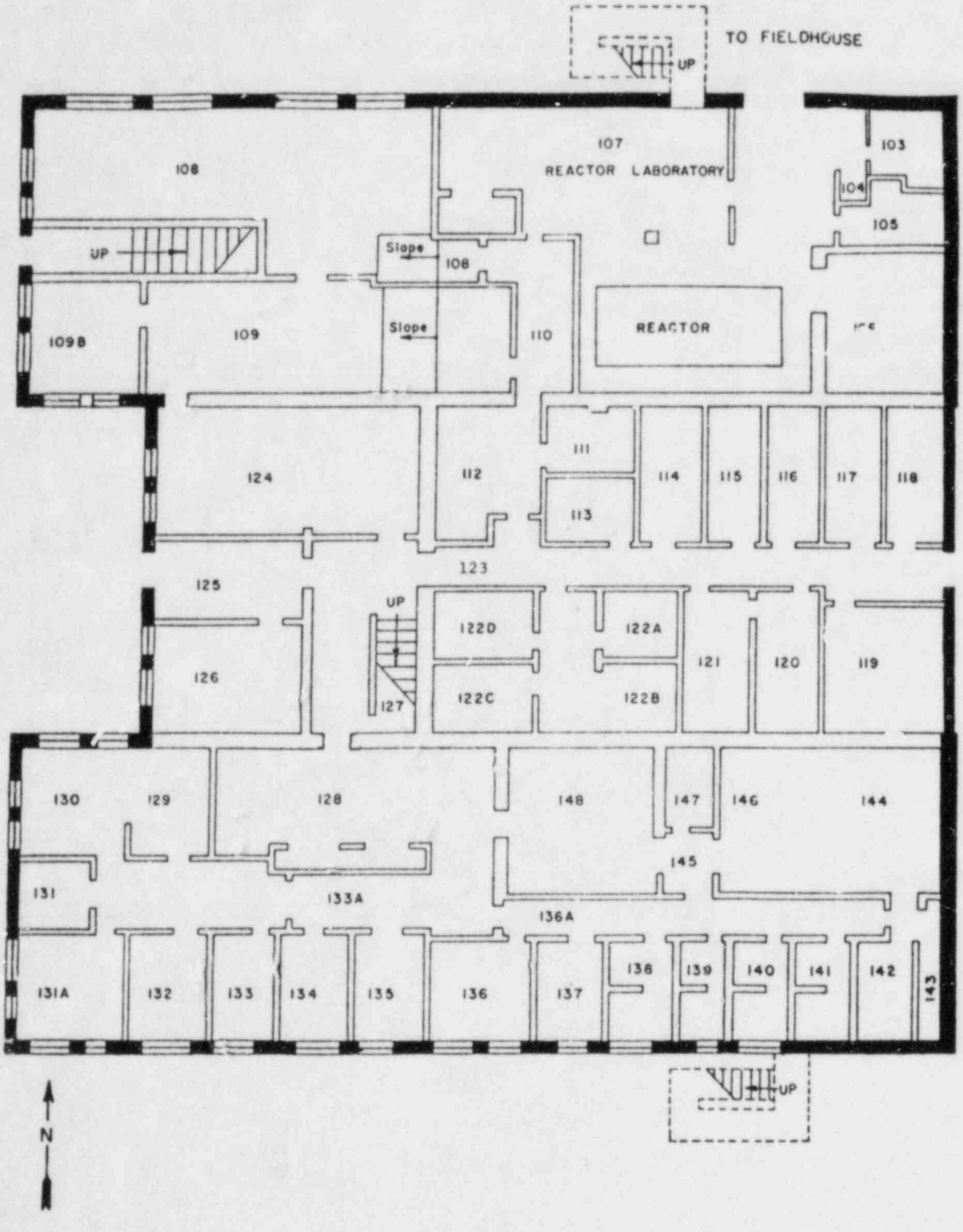

Figure 4.1 First-floor plan of Nuclear Engineering Laboratory Building

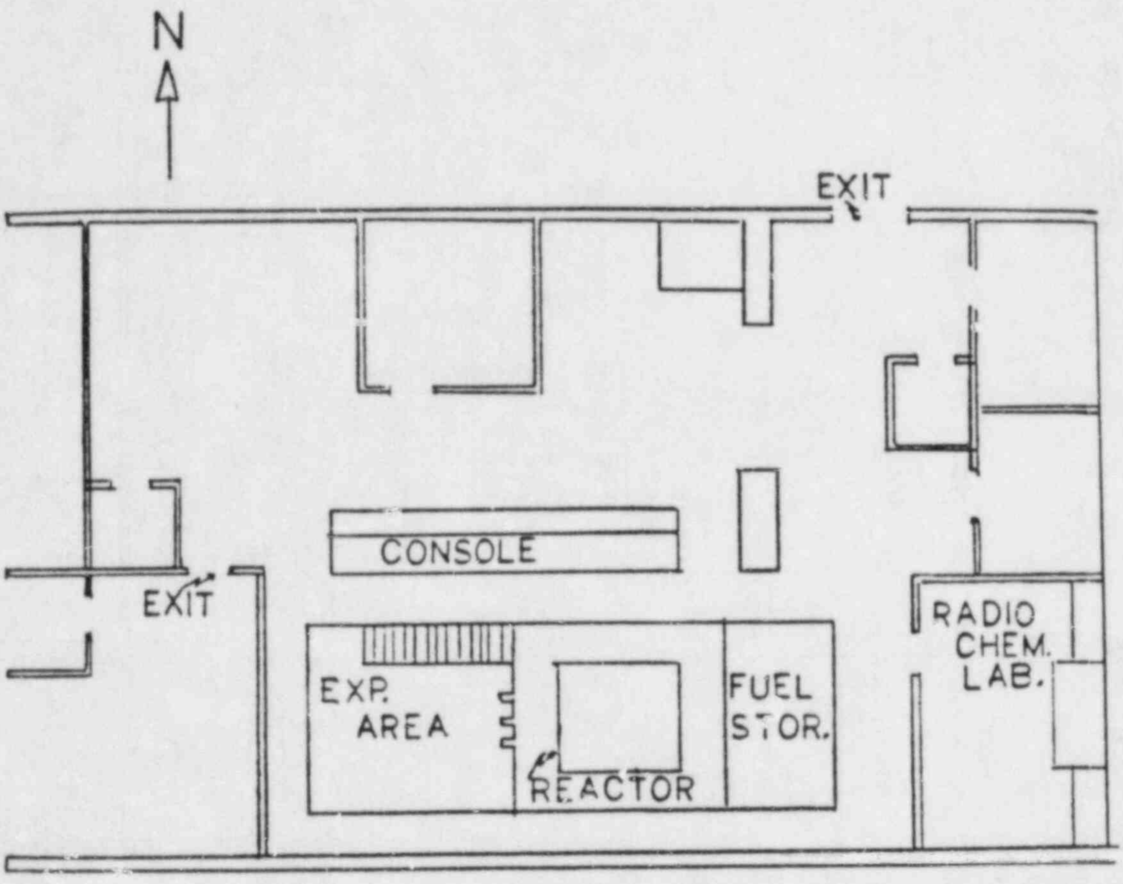

Figure 4.2 Reactor area floor plan

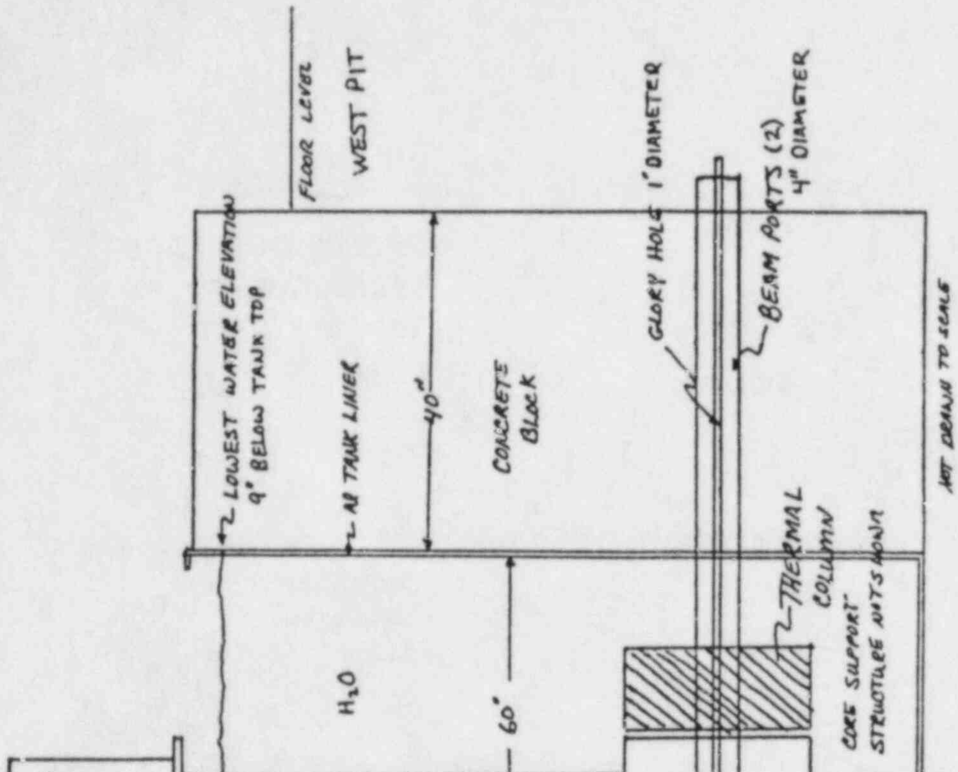

Figure 4.3 Experimental facilities in core

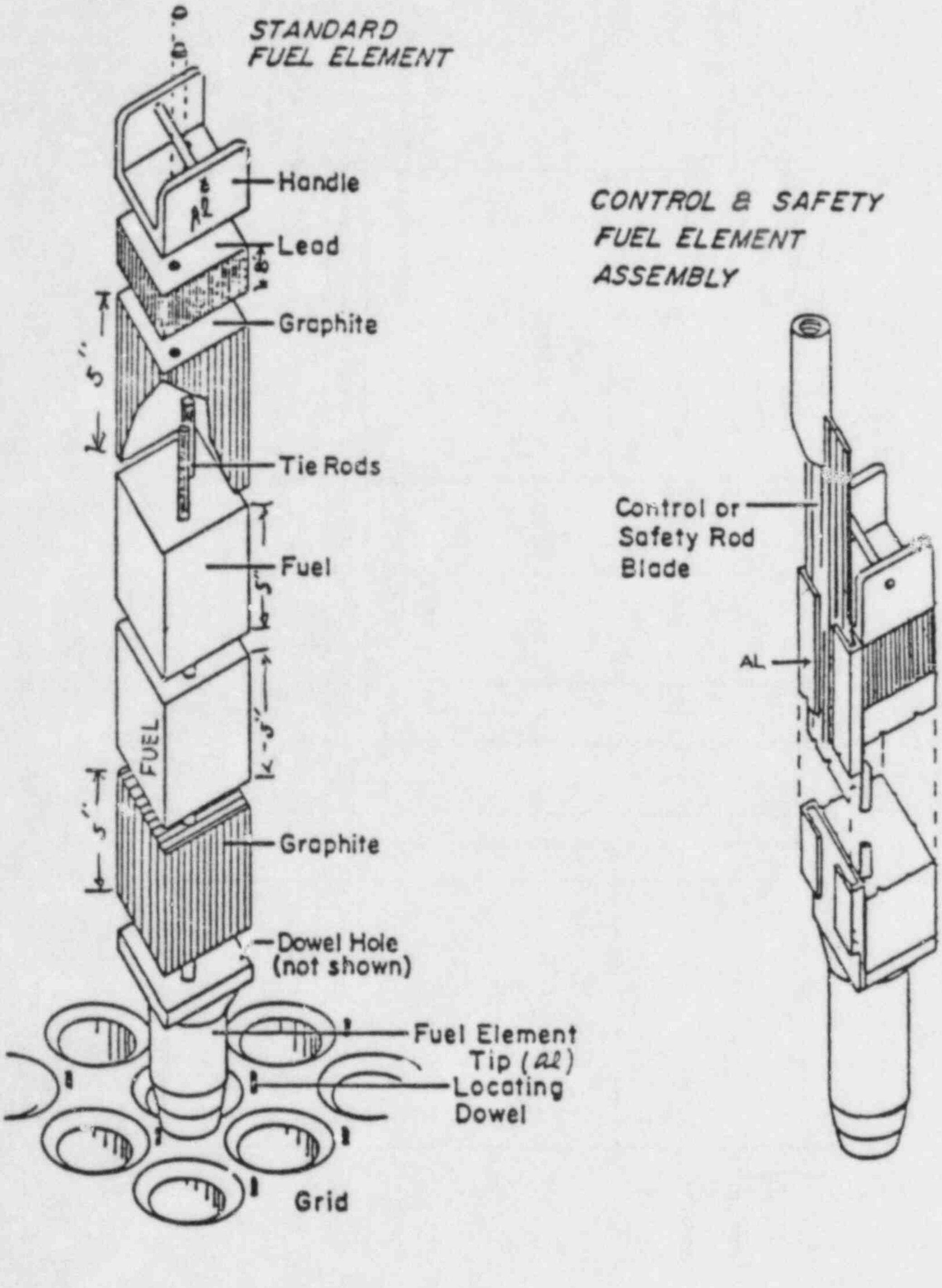

Figure 4.4 Control and safety fuel-element assembly

5 REACTOR COOLING SYSTEM

5.1 Reactor Core Cooling System

The reactor core is submerged in demineralized water in an 800-gal aluminum tank and is cooled by natural convection. When the reactor is operated under steady-state conditions at 100 W, the bulk pool temperature rises a negligible amount before achieving thermal equilibrium. As discussed in Section 14, various postulated accident scenarios would produce temperature rises of only 1-4C°.

5.2 Coolant Purification System

About 5 gal per minute of water are pumped from the pool through a demineralizer filter and back into the pool. A radioactivity detector and a conductivity meter also are installed in this loop. The outputs from these instruments are displayed at the control console.

5.3 Conclusion

The staff concludes that the 800-gal reactor water pool is adequate to remove heat from and to prevent overheating of the fuel under all normal and off-normal operating conditions.

6 ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) are systems provided to mitigate the radiological consequences of design-basis accidents. No ESF systems are provided at the UO facility.

The OU reactor is a 100-W AGN-211 reactor. Therefore, the fission product inventory is very low. In addition, the analyses of accidents in Section 14, including the maximum hypothetical accident, indicate that the resultant radiological consequences would be small fractions of the limits specified in 10 CFR 20.

Therefore, the staff concludes that the operation of the UO AGN reactor without any ESF systems does not pose a radiological hazard to the public or to the environment in the event of an accident.

7 CONTROL AND INSTRUMENTATION SYSTEMS

7.1 Systems Summary

The control and instrumentation systems for the OU AGN research reactor are similar to those generally used in other research reactors of a similar size in the United States. Control of the nuclear fission process is achieved by using coarse and fine control rods and two safety rods. The control and instrumentation systems are interlocked to provide automatic and manual scram capability in case of reactor malfunction and to provide the means for operating the various components of the reactor in a manner consistent with design objectives. The scram-producing reactor safety channels, functions, and set points are shown in Table 7.1. The licensee's Technical Specifications require that the reactor safety channels shall be operable in accordance with Table 7.1, whenever the reactor control or safety rods are not in their fully inserted position.

7.2 Control System

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

7.2.1 Nuclear Control System

A detailed description of the nuclear control system that consists of the safety rods, the fine and coarse control rods, and their associated drive mechanisms is presented in Section 4.

The safety rods and the coarse control rod are interlocked so that (1) only one safety rod can be withdrawn at a time, (2) the coarse control rod cannot be withdrawn unless both safety rods are fully withdrawn, and (3) the safety rods cannot be withdrawn (cocked) unless the control rods (both fine and coarse) are fully inserted.

The rods are controlled by manual holddown (spring-return) switches located on the control console. The positions of the control rods are indicated to the nearest 0.01 cm on the console. The location of any of the three magnets at either its upper or lower position is indicated by an appropriate light on the control console. Contact between the magnets and their associated actuator rods also is indicated by control console lights.

The reactivity worth of each of the safety rods and the coarse control rod depend on the core configuration and varies from about 1.0 to 1.5% $\Delta k/k$. The worth of the fine control rod is approximately 0.45% $\Delta k/k$ for all core configurations. The licensee's Technical Specifications require that the scram insertion times of the safety and coarse control rods shall not exceed 500 msec.

7.2.2 Process Control System

The process control system consists of the circuitry and devices required to energize and deenergize the water purification system pump and the ventilation system. Failure of the water purification system pump for any reason results in a scram.

7.3 Instrumentation System

The instrumentation system is composed of both nuclear and process instrumentation circuits.

7.3.1 Nuclear Instrumentation

The instrumentation discussed below provides the operator with the necessary information to properly manipulate the nuclear controls.

The log power and period channel (Nuclear Safety Channel 1) comprises a compensated ion chamber, a power supply, a logarithmic micro-microammeter, a recorder, a period signal and scram, and a two-contact meter relay. This channel covers the power range from source level to full power and will produce a scram if the power level is less than 10^{-7} W or greater than 150 W. This channel also will cause a scram if the reactor period is less than 5 sec.

The linear power channel (Nuclear Safety Channel 2) comprises a compensated ion chamber, a power supply, a linear micro-microammeter with range switch, a recorder, and a meter. This channel indicates power level from source level to full power by means of the range switch and will produce a scram when the power level becomes appreciably greater than full-scale on any range. However, the scram level is nominally set for 120 W (120% of full power).

Both compensated ion chambers are located in watertight cans submerged in the shield/reflector water pool above and to the side of the core. The appropriate controls and the meters and recorders are located on the control console. A manual (operator-controlled) scram also is located on the control console.

7.3.2 Process Instrumentation

The instrumentation discussed below senses and monitors parameters associated with the pool water and radiation levels.

The pool water radioactivity monitor is composed of a gamma-radiation monitor located in the center of a cylindrical tank filled with pool water. A pump circulates the pool water through the water-radioactivity monitor, to the filter-demineralizer system, and back to the reactor tank. The water intake to the monitor/cleanup system is located at the bottom of the reactor tank near the bottom edge of the graphite reflector. This monitor provides visual indication on the control console.

The pool water conductivity monitor consists of a conductivity probe and Wheatstone bridge circuit. Measurements are made before and immediately after reactor operation to ensure that the pool water impurity activities will be negligible. The licensee's Technical Specifications require that the pool water conductivity be maintained at less than 15 μmho/cm.

The pool water level monitor consists of a float switch and the associated circuit. This monitor provides an audio and visual alarm at the control console and initiates a scram if the pool water level drops to 9 in. below the reactor tank top.

The pool water temperature monitor consists of a resistance bulb thermometer that senses the bulk pool temperature. Temperature indication is provided on the control console. A scram is initiated if the bulk pool temperature falls below 5°C.

The air monitor consists of a gamma-radiation monitor and the associated electronic recording and air sampling system. This system takes air directly from the surface of the pool water into a sensitive air-ionization chamber. At a power level of 15 W the system normally does not record the presence of any gaseous fission products. Conservative calculations by the staff indicate the sensitivity may be sufficient to detect the presence of such fission products at a power level of 100 W. The monitor will be calibrated to permit quantification of any measured activity.

The pool water radioactivity monitor is the only radiation monitor required by the licensee's Technical Specifications to be operational when the reactor is operating. This channel, which provides visual indication on the control console, ensures that abnormal levels of radioactivity will result in operator response. However, because the reactor has not been operated previously at 100 W, the licensee plans a thorough radiation monitoring program as part of the step-wise power increase plan described in the Technical Specifications. The radiation monitoring program will include, but not be limited to, constant monitoring of the air over the reactor pool, using the air monitor described above, and radiation surveys around the reactor, the control console, in adjacent rooms, and those above the reactor laboratory. The results of the radiation monitoring program will provide a basis for evaluating both the continuous monitoring of the pool water radioactivity and the continuous monitoring of potential airborne radioactivity. Both the licensee and the staff believe that continuous monitoring of the pool water radioactivity is adequate. However, the licensee's operational procedures require the air monitor and appropriate radiation monitors to be operational whenever the reactor is operated at power levels above 15 W.

7.4 Conclusions

From review of drawings, reports, and a site visit, the control and instrumentation system at the UO AGN research reactor facility is well designed and maintained. All power and instrumentation wiring is protected from physical damage by conduit and/or cable trays. The specifications of the individual components are in excess of minimal requirements for the overall system. Redundancy in the crucial areas of power measurements is ensured by overlapping ranges of the log power and linear power channels. The control system is designed so that the reactor shuts down automatically if electric power is lost. In addition, the radioactivity monitoring systems and procedures for reactor room, pool water, and personnel are adequate for the proposed reactor operating conditions.

On the basis of the above analysis of the control and instrumentation systems, the staff concludes that both systems satisfy all existing regulations and that they are adequate to ensure the safe operation of the facility.

Table 7.1 Scram-producing reactor safety channels

Device (Safety Channel)	Function	Limiting Safety System Settings
Log power channel (Nuclear Safety Channel 1)	High-power limit Low-power limit Short period limit	150 W 10-70W 5 s
Linear power channel (Nuclear Safety Channel 2)	High power	120 W
Reactor tank water level	Low pool water level	9 in. below reactor tank top
Pool water temperature	Low pool water temperature	5°C or less
Manual scram	Normal shutdown	Scram on operator decision

8 ELECTRICAL POWER SYSTEM

The electrical power system at the OU reactor facility is a standard and well-accepted electrical supply system designed and constructed to specifications similar to those at other research reactor facilities.

8.1 Main Power

Electrical power for building lighting and equipment is supplied by a single 20-amp circuit from the Nuclear Engineering Laboratory distribution system. The main power control panel is located in the electrical utility room. Fluorescent lighting is used throughout the building.

8.2 Emergency Power

No emergency power is provided for the UO reactor operation. Because the UO reactor will scram in case of a power interruption and the decay heat generated in the core after scram is minimal, no emergency power is supplied except battery-operated emergency lighting for personnel movement during an emergency.

8.3 Conclusions

The above factors, and the fact that the reactor will scram in the event of power failure and that ambient pool cooling can remove decay heat, leads the staff to conclude that the electrical power system is adequate for safe operation of the UO reactor.

9 AUXILIARY SYSTEMS

9.1 Fuel Handling and Storage

Handling of fuel elements is done in the pool water by using long-handled tools. An overhead crane is deemed not required.

All fuel elements not in the reactor are stored in the fuel storage pit immediately adjacent to the reactor. Each fuel element is stored below floor level in a steel cylinder; calculations indicated that the array will remain in a subcritical configuration if flooded and in an infinite water reflector. Some fuel elements are stored in steel shipping containers in the fuel storage pit when the entire core is unleaded.

9.2 Fire Protection System

The fire protection system consists of wall-mounted fire extinguishers.

9.3 Ventilation System

The ventilation system is composed of an air handling unit located in the reactor room. This unit is equipped with cooling and heating capability. The unit takes fresh air makeup from the outside and recirculates air throughout the reactor room. Exhaust from the reactor room is through vents located in the toilets (Rooms 103 and 105) and the adjacent radio chemical laboratory (Room 108; see Figure 4.1). These rooms vent to the outside. The controls to shut off the ventilation system fan are located in the control room.

9.4 Conclusions

The staff concludes that the fuel-handling and storage facilities are appropriate for the reactor size and use. The staff also concludes that the fire protection system is adequate for the facility.

10 EXPERIMENTAL PROGRAMS AND FACILITIES

10.1 Experimental Programs

In addition to being an integral part of the nuclear engineering undergraduate and graduate programs, the UO AGN research reactor supports the various experimental programs of the staff and students. Most of the experimental work involves activation of various materials and subsequent analyses. These irradiated materials may be foils or small samples to evaluate reactor parameters or material composition (neutron activation analysis) or to use as tracers in various studies.

All proposed new experiments must be reviewed by the university's Radiation Safety Officer (RSO) and the Reactor Safety Committee. The RSO may be a member of the Reactor Safety Committee. The reviews by the RSO and the Reactor Safety Committee are carried out

(1) to ensure that accidents causing changes in composition and geometry of the experiments will not cause positive changes or ramps in reactivity that might place the reactor on unsafe periods

(2) to provide assurance of mechanical integrity, chemical compatibility, and adequate protection against any other potential hazard

(3) to provide assurance that in the event of an accident, the postulated complete release of all gaseous, particulate, or volatile components from the experiment will not result in doses in excess of 10% of the equivalent annual dose limits of 10 CFR 20 for persons in restricted areas during the time required to evacuate the restricted areas

Irradiation of explosive materials is not permitted in the program.

10.2 Experimental Facilities

The experimental exposure facilities, described below, consist of fixed-location facilities accessible from the experimental area (Figure 4.3) and the variable-location facilities accessible from the reactor bridge.

10.2.1 Fixed-Location Facilities

The fixed-location facilities consist of a horizontal glory hole, a horizontal access port, and a horizontal beam port.

The horizontal glory hole has an inside diameter of 7/8 in. and goes through the reactor core. Samples may be placed in the glory hole at varying positions in the core and reflector.

The 4-in.-diameter horizontal access port extends through the graphite reflector to one side of the reactor core. Samples may be placed within the port or a collimated beam of radiation may be brought out for irradiations external to the reactor biological shield.

The 4-in.-diameter horizontal beam port extends to the face of the graphite reflector. As with the access port, samples may be irradiated within the beam port or a beam may be extracted for use external to the reactor biological shield. The neutron flux in the beam port is more highly thermalized than that in the access port.

All of the facilities described above are constructed of aluminium and are watertight.

10.2.2 Variable-Location Facilities

The reactor is authorized to operate with either a standard 12-element or a 20-element flux-trap core configuration (see Section 4.2.1). The flux-trap configuration provides for operation with up to 20 fuel elements and a flux trap consisting of a one- or two-element void. These variable fuel and reactor configurations make it possible to test different types of reflector elements or to leave water-filled voids in the core where large samples may be irradiated.

10.3 Conclusion

The staff concludes that the design of the experimental facilities, together with the limitations for experiments delineated in the Technical Specifications, ensure proper and safe experimental programs.

11 RADIOACTIVE WASTE MANAGEMENT

Because of the low power level and limited operating schedule of the UO AGN research reactor, there has been a negligible generation of radioactive waste. It is anticipated that increasing the operating power level from 15 to 100 W will not increase appreciably the amount of radioactive waste generated.

11.1 Airborne Wastes

Although analysis of pool water has demonstrated the presence of short-lived gaseous fission products (^{16}N, ^{41}Ar, ^{87}Kr, ^{88}Kr, ^{137}Xe, and ^{138}Xe), the activity level is so low that an air monitor, drawing air directly from the surface of the pool water, does not normally record the presence of gaseous activity during operation at 15 W. However, at 100-W operation, the water activity is expected to increase by a factor of about 6.7 because of reactor operation at and the airborne radioactivity may increase to detectable but insignificant levels. The airborne wastes resulting from operation at 100 W are expected to be negligible and will pose no hazard within or outside of the facility. The staff analysis of release of gaseous activity under accident conditions is presented in Section 14.

11.2 Solid Wastes

Potential solid wastes generated as a result of reactor operation include ion-exchange resins and filters, contaminated paper and gloves, and occasional small activated components.

Some of the reactor-based research results in the generation of solid low-level radioactive waste in the form of contaminated paper, gloves, glassware, and irradiated materials. This solid waste generation typically has contaminated submillicurie quantities of radionuclides per year.

Solid waste either is allowed to decay to levels consistent with disposal as nonradioactive waste or is packaged and shipped to an NRC-approved disposal site by the University Radiation Safety staff in accordance with applicable NRC and Department of Transportation regulations.

11.3 Liquid Wastes

Normal reactor operations or reactor-based research produce small volumes of radioactive liquid wastes. Any such wastes either are disposed of by release to the sanitary sewer system in accordance with applicable NRC regulations or are solidified and disposed of as solid radioactive waste by the University Radiation Safety staff as noted above.

11.4 Conclusion

The staff has reviewed the operational history of the reactor and notes that for operation at 15 W there is normally no detectable airborne radioactivity.

The staff's analysis discussed in Section 14 indicates that airborne radioactivity resulting from operation at 100 W will be insignificant. Therefore, the staff concludes that because the doses in the reactor laboratory in other restricted areas and in unrestricted areas will be negligible if not nonexistent.

The staff concludes that the waste management activities of the UO AGN research reactor facility have been and are expected to continue to be conducted in a manner consistent with 10 CFR 20 and ALARA principles.

12 RADIATION PROTECTION PROGRAM

The OU has developed a structured radiation safety program with an adequate radiation safety staff and appropriate detection equipment (1) to determine, control, and document occupational radiation exposures at its reactor facility and (2) to ensure compliance with applicable regulations concerning releases of radioactive materials to restricted and unrestricted areas.

12.1 ALARA Commitment

In support of the university's administrative objectives, the Radiation Safety and Reactor Safety staffs and the Reactor Safety Committee have established the policy that all university activities and operations are to be conducted in a manner to keep all radiation exposures as low as is reasonably achievable (ALARA). All proposed experiments and procedures at the AGN research reactor are reviewed to minimize potential exposures of personnel. Any unanticipated or unusual reactor-related exposures are investigated by both the Radiation Safety and Reactor Safety staffs to develop methods to prevent recurrences.

12.2 Health Physics Programs

The full time Radiation Safety staff at the UO consists of one professional and one technican and is independent of the UO AGN research reactor line of responsibility. In addition, the licensed reactor operators are qualified in health physics and perform radiation protection duties as required.

The Radiation Safety staff provides radiation safety support to the entire university complex, including many radioisotope laboratories. However, the staff believes that the UO AGN research reactor staff is adequately trained in health physics to provide radiation protection support consistent with the training/research efforts within the facility.

Detailed written procedures have been prepared that address the Radiation Safety staff's various activities and the support that it is expected to provide to the routine operations of the Reactor Safety staff. These procedures identify the interactions between the Radiation Safety staff and the Reactor Safety staff and experimenters. The procedures also specify administrative limits and action points, as well as appropriate responses and corrective action to be taken, if these limits or action points are reached or exceeded. Copies of these procedures are readily available to the Reactor Safety staff and experimenters and to the Radiation Safety and administrative personnel.

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

All reactor operators are trained in health physics and certified by the Radiation Safety Officer (RSO). The training program is designed to identify the particular hazards of each specific type of work to be undertaken and methods to negate their consequences. Retraining in radiation safety is provided as part of the operator requalification program. All reactor operators are given an examination annually on radiation safety practices and procedures. All of the above-discussed health physics training is provided by the Radiation Safety staff.

12.3 Radiation Sources

Sources of potential radiation directly related to reactor operations include radiation from the reactor core, the ion-exchange column and filter in the water cleanup system, the startup source, irradiated fuel, and, possibly at 100-W operation, radioactive gases.

The reactor fuel consists of fine UO_2 particles in a polyethylene block, which in turn is coated with epoxy. Operation at 15 or 100 W results in a small fission product inventory that decays to a low level in a few days. A fuel element can be handled without shielding 24 hours after shutdown from a 15-W run. Increasing the power level to 100 W will result in higher residual activity levels that may necessitate shielding, remote techniques, or longer decay times for fuel handling. Radiation exposures from the operating reactor core are reduced to acceptable levels by concrete shielding, the water-filled tank, and, for operation at power levels above 50 W, by supplemental steel shielding on top of the reactor tank.

The ion-exchange resins and filters are changed as required to minimize personnel exposure.

Sources of radiation that may be considered as incidental to the normal reactor operation but are associated with reactor use, include radioactive isotopes produced for research, activated components of experiments, and activated samples or specimens. Personnel exposure to radiation from such intentionally produced radioactive materials as well as from the required manipulation of activated experimental components is controlled by carefully developed and reviewed operating procedures that use the normal protective measures of time, distance, and shielding.

12.4 Routine Monitoring

The AGN research reactor facility uses a pool-water radioactivity monitor, an air monitor sampling air drawn from the surface of the reactor pool water, and several portable survey meters. The pool-water radioactivity monitor is required by the Technical Specifications to be operational whenever the reactor is operational. The air monitor is required by the licensee's operational procedures for operations at power levels in excess of 15 W. The radiation survey meters are operating whenever the reactor is in use.

The Radiation Safety staff participates in experiment planning by reviewing all proposed procedures for methods of minimizing personnel exposures and limiting the generation of radioactive waste. Approved procedures specify the type and degree of health physics involvement in each activity. As an example, standard operating procedures require that changes in experimental setups include a radiation survey by health-physics-qualified personnel, and all items removed from reactor room must be surveyed and appropriately tagged by health physics-qualified personnel.

12.5 Personnel Radiation Monitoring

The UO monitoring program for personnel using the AGN research reactor is described in the university's radiation safety procedures. To summarize the program, exposures are measured by the use of TLD or pocket dosimeters assigned to individuals who might be exposed to radiation. All persons entering the reactor facility are considered to be such individuals.

The UO AGN research reactor personnel exposure history of the last 5 years is given in Table 12.1.

12.6 Potential Dose Assessments

Natural background radiation levels in the Norman Campus area result in an exposure of about 110 mrems per year to each individual residing there (A. W. Klement et al., 1972). At least an additional 7% (approximately 8 mrems per year) will be received by those living in a brick or masonry structure. Any medical diagnostic X-ray examination will add to this natural background radiation, increasing the total of accumulated annual exposure.

Conservative calculations by the staff indicate that operation of the UO AGN research reactor at power levels up to 100 W will result in insignificant exposures from airborne radioactivity to individuals either inside or outside the facility. As shown in Table 12.1 radiation exposures from operation of the reactor at 15 W to individuals in both the restricted and unrestricted areas have been demonstrated to be insignificant. Similarly, projected exposures for operations at power levels up to 100 W will be well below the 10 CFR 20 limits. The additional steel shielding across the top of the pool level for operation of the reactor at power levels above 50 W will reduce the gamma radiation levels at the control console to 0.1 mr per hour and is consistent with the licensee's commitment to the ALARA principle.

12.7 Conclusion

The staff considers that radiation protection receives appropriate support from the administration. The staff concludes that (1) the program is properly staffed and equipped, (2) the UO staff has adequate authority and lines of communication, and (3) the procedures are integrated correctly into the training and research plans.

The staff concludes that the monitoring activities conducted by UO AGN research reactor personnel are adequate to promptly identify abnormal operating conditions with potential for release of radioacitivity to the facility or the environs within sufficient time to permit appropriate actions to be taken to eliminate or minimize potential exposures to individuals in either restricted or unrestricted areas.

Additionally, the staff concludes that the UO AGN research reactor radiation protection program is acceptable because the staff has found no instances of reactor-related exposures of personnel above applicable regulations and no releases of radioactivity to the environment. Furthermore, the staff considers that there is reasonable assurance that the personnel and procedures will continue to protect the health and safety of the public and the reactor staff and students during reactor operations at power levels up to 100 W for the requested renewal period.

Table 12.1 Recent exposure history for reactor personnel

	Number of Individuals in Each Range				
Whole-body Exposure Range (rem)	1978	1979	1980	1981	1982
No measureable exposure	14	15	13	16	12
Measurable exposure 0.1	1	3	4	1	4
0.1 to 0.25	6	0	0	0	0
0.25 to 0.5	0	0	0	0	0
0.5 to 0.75	0	0	0	1	0
> 0.75	0	0	0	0	0
Number of individuals monitored	21	18	17	18	16

13 CONDUCT OF OPERATIONS

13.1 Overall Organization

Responsibility for the safe operation of the reactor facility lies within the organizational structure shown in Figure 13.1.

13.2 Training

Most of the training of reactor operators is done by inhouse personnel. The licensee's Operator Requalification Program has been reviewed, and the staff concludes that it meets the applicable requirements of 10 CFR 50.34(b).

13.3 Emergency Planning

10 CFR 50.54(q) and (r) require that a licensee authorized to possess and/or operate a research reactor shall follow and maintain in effect an emergency plan that meets the requirements of Appendix E to 10 CFR 50. In 1979 the guidance available to licensees was contained in RG 2.6 (1978 For Comment Issue) and in ANS 15.16 (1978 Draft). In 1980, new regulations were promulgated, and licensees were advised that revised guidance would be forthcoming. Thus, revised ANS 15.16 (November 29, 1981 Draft) and Regulatory Guide 2.6 (March 1982 For Comment) were issued. On May 6, 1982, an amendment to 10 CFR 50.54 was published in the Federal Register (47 FR 19512, May 6, 1982) recommending these guides to licensees and establishing new submittal dates for Emergency Plans from all research reactor licensees. The deadline for submittal from a licensee in the Class of the University of Oklahoma was November 3, 1982. The licensee made a timely transmittal of an Emergency Plan, thereby complying with existing applicable regulations.

13.4 Operational Review and Audit

The Reactor Hazards and Radiological Safety Committee approves suggested procedures for the purchase, production, storage, use, and disposition of all radioisotopes. The committee is responsible to the Reactor Administrators for the review and evaluation of all proposed operations and procedures for the reactor facility as well as all proposed changes in the reactor system or electrical or component design. The Committee advises and assists on any problem relative to the safe operation of the reactor facility.

13.5 Physical Security Plan

The UO has established and maintains a program designed to protect the reactor and its fuel and to ensure its security. The NRC staff has reviewed the plan and concludes that the plan, as amended, meets the requirements of 10 CFR 73.67 for special nuclear material of low strategic significance. The UO licensed authorization for reactor fuel falls within that category. Both the Physical Security Plan and the staff's evaluation are withheld from public disclosure under 10 CFR 2.790(d)(1) and 10 CFR 9.5(a)(4). Amendment 8 to facility license

R-53, dated June 2, 1982, incorporated the Physical Security Plan as a condition of the license.

13.6 Conclusion

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

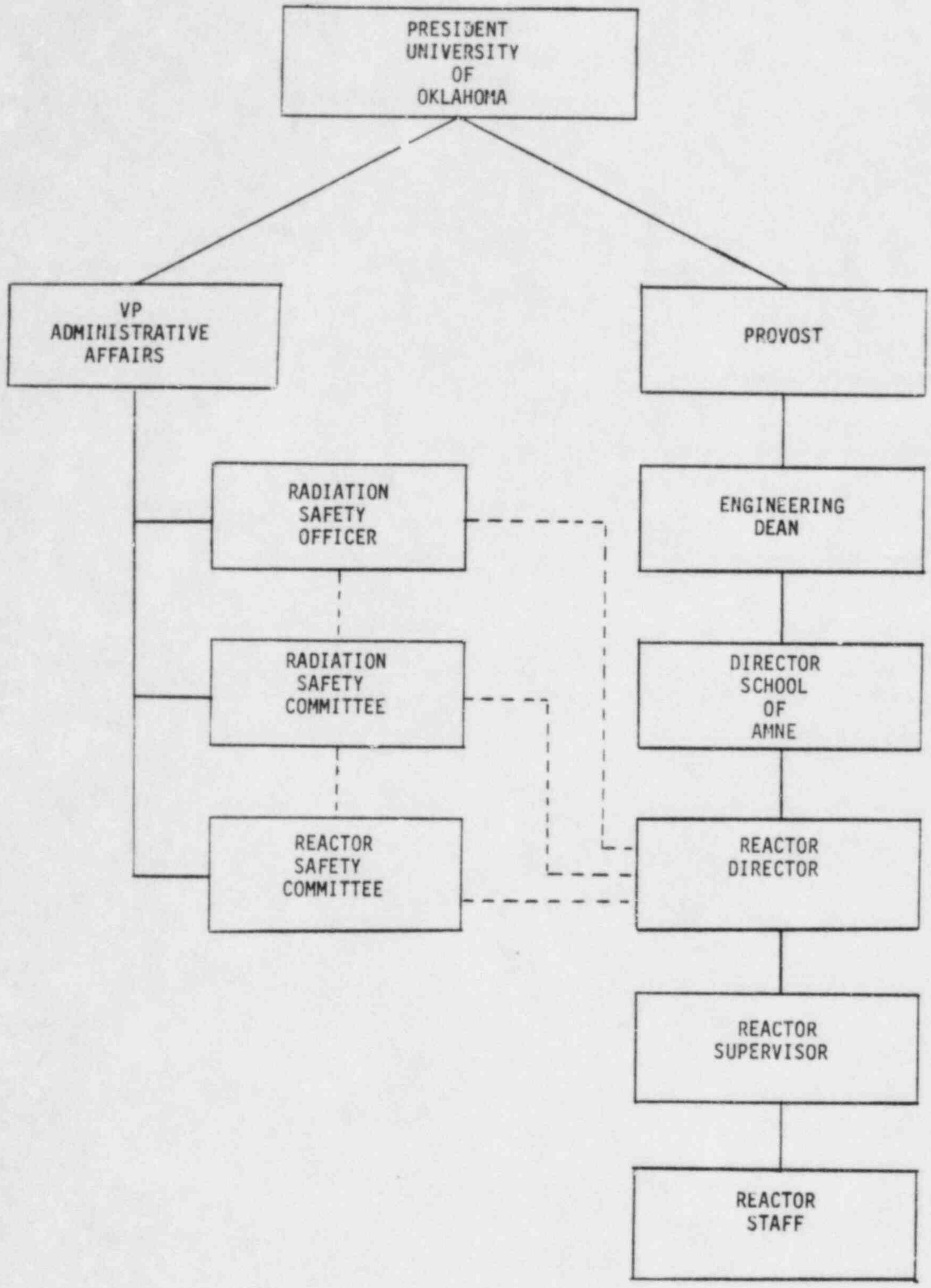

Figure 13.1 Administrative organization

14 ACCIDENT ANALYSIS

14.1 Summary

In evaluating the safety of the operation of the UO AGN research reactor, the licensee and Airojet General Nucleonics (AGN) considered potential accidents to ensure that these events could not result in significant risks to the reactor staff and users or to the public (AGN, October 1957). The potential accidents considered included tornadoes, an inadvertent total excess reactivity insertion during operation of the reactor, a fuel loading incident causing a prompt critical accident, and an accident involving a hypothetical loss-of-pool-water accident.

Of the potential accidents considered, none had consequences greater than the postulated hypothetical (1) insertion of a fuel element in a flux trap or (2) the loss-of-coolant accident following extended reactor operation. The above two accidents have been designated as maximum hypothetical accidents (MHAs) and, for purposes of classification, are referred to as (1) the "fuel-loading accident" and (2) the "loss-of-coolant accident." For an MHA, the staff assumed that the accident occurred but does not try to describe or evaluate all of the mechanical details of the accident or the probability of its occurrence and for which the potential risk to the public health and safety is greater than that from any other credible event.

14.2 Tornadoes

The licensee indicates that the Norman Campus area is subject to tornadoes. There is a city-wide tornado alert system that operates very effectively using sirens. The licensee's procedures prohibit reactor operation during a tornado alert. In addition, the reactor is located in a pit below the floor level of the reactor laboratory. The reactor laboratory, located on the first floor of the Nuclear Engineering Laboratory (NEL) Building, has no windows, brick outer walls, concrete-block inner walls, and 3 to 4 in. of concrete overhead, making tornado damage improbable. The licensee has stated that no tornadoes have actually set down on campus and that any damage to the campus buildings from winds has been superficial. The staff agrees with the licensee and concludes that the prohibition of reactor operation during tornado alerts coupled with the tornado-resistant construction of the facility provides reasonable assurance that the hazard from tornadoes is negligible. There are no other natural phenomena that pose a significant risk.

14.3 Rapid Insertion of Reactivity (Nuclear Excursion)

14.3.1 Maximum Power Excursion Scenarios

The maximum power excursion (transient) that could occur, would be one resulting from the inadvertent rapid insertion of the total available excess reactivity. The UO AGN fuel loading is limited to 0.65% $\Delta k/k$ excess reactivity by the Technical Specifications. As beta effective is 0.0075, this is well below prompt critical for the reactor. Because of these limitations, the licensee states that no insertion of moderator or fuel into the reactor can result in more than a scram on period. For example, rapid removal of a small piece (6 g) of cadmium from the glory hole with a fuel loading total excess reactivity of 0.45% $\Delta k/k$ will result in an instantaneous period of 5 sec causing a period scram (scram setting is 5 sec).

14.3.2 Conclusion

From the above considerations, the staff concludes that instantaneous insertion of all of the available excess reactivity will not result in a prompt critical excursion.

14.4 Loss-of-Coolant Accident

The licensee and the staff have considered possible mechanisms leading to a loss-of-water accident. The only mechanisms deemed possible are (1) pumping the water out; (2) failure of the glory hole, beam tube, or access port; or (3) tank failure.

The tank purification system draws water from below the grid plate. A rupture of the purification system line outside the reactor tank would not drain the tank because an antisyphon hole in the piping within the reactor tank, approximately 9-in. below the top of the reactor tank, prevents draining the tank below the low water alarm/scram setting.

The failure of one of the horizontal experimental facilities also would result in a low-water alarm. The times required to drain the tank through the 7/8-in.-diameter glory hole or the 4-in.-diameter experimental tubes (beam tube or access port) would be about 65 min and 3.2 min, respectively, and the times required to actuate the low-water alarm would be about 8 min and 0.4 min, respectively. As noted above the only potential consequence of the loss-of-water accident would be radiation exposure from the exposed core. Thus, in the event of failure of one of the experimental facilities, isolation of the radiation area(s) until repair of the facility and reflooding of the tank was completed would minimize radiation exposures.

Tank failure could be caused by a severe earthquake, a major settling of the building foundation, or by corrosion of the aluminum tank. The hazard from earthquakes is negligible. There is no evidence of foundation settling. Examination of the reactor tank has shown no evidence of deterioration. Because of the high purity of the tank water and low levels of activated corrosion products, tank corrosion is not expected to be significant.

Loss-of-coolant/shielding water could result in potential radiation exposures to occupants of the reactor laboratory and the area immediately above the reactor. To prevent radiation exposure to individuals in the second-floor area above the reactor under normal or accident conditions, the room is used only as a storeroom, is under the control of the reactor staff, and is locked whenever the reactor is in operation. In addition the metal plate, which is placed over the core during reactor operations above 50 W, will also minimize exposures to both reactor personnel and anyone in the store room on the second floor.

In addition to potential radiation exposure from a loss-of-water accident, the accident can result in increases in temperature of the fuel. However, because of the low-power level of the reactor (100 W), the decay heat immediately after operation is quite low. The power density of 8.3 W per element for a standard 12-element core and 5 W per element for a 20-element flux-trap core plus the large pool volume assures low fuel temperatures. Because of the low average power density and low fission product inventory, a complete loss of tank water would result in a rise in the element temperatures of approximately 1°C. Heat transfer to the air around the elements would provide adequate cooling. Therefore, there is no concern that fuel will melt.

Even though the possibility of a tank leak is believed to be remote, the staff has performed calculations and the licensee has made experimental measurements to evaluate the radiological hazard associated with this type of accident. As a point of reference for the discussion of these evaluations, the measured dose rate at the surface of the pool water during operation at 15 W is 15 mR per hour and the dose rate to a person standing looking down into the pool is 3 mR per hour. For 100-W operation these dose rates extrapolate to 100 mR per hour and 20 mR per hour, respectively.

The staff has calculated dose rates 5 ft above the top of the reactor core (normal pool water level) for two operating periods (1 hour and 10 hour) and for times after a reactor scram of 1 min, 10 min, 1 day, and 15 days. The total fission product inventories were calculated using the Way-Wigner equation (K. Way and E. P. Wigner, 1948).

$$Ci = 1.4P\left[(t-T_o)^{-0.2} - t^{-0.2}\right]$$

where

Ci = Curies at time t
$t = T_o + t_d$
T_o = operating time in days
t_d = decay time (time after scram) in days
P = power level in watts

The dose rates were calculated using the relation (S. Glasstone and A. Sesonske, 1963),

$$D = \frac{3.7 \times 10^{10} Ci\ E\gamma M_a}{0.02(4\pi R^2)}$$

where

D = gamma dose rate at time t_d in mR/h
Ci = Curies at time t_d
$E\gamma$ = Energy per disintegration = 0.8 Mev
M_a = linear absorption coefficient of air for 0.8 Mev gamma rays = 3.7×10^{-5} cm^{-1}

R = distance from midpoint of core to point 5 ft from top of core = 190.5 cm (75 in.)

For purposes of the dose rate calculation, it has been assumed that each disintegration yields a single photon with an energy of 0.8 Mev and that the core may be treated as a point source.

The results of the calculations are presented in the Table 14.1. These calculations neglect the 2-in. lead shielding on the top of each element.

The licensee has performed an experiment to obtain an estimate of the dose rate from the reactor core. Fifteen days after a run at 15 W the reactor tank was drained until the top of the core was exposed. The dose rate at the top of the core was found to be 0.09 mR/h. The dose rate at 1 day post-shutdown was calculated by the licensee to be about 10 to 50 mR/h. When extrapolated to 100-W operation, these dose rates are 0.6 mR/h and 67 to 333 mR/h, respectively. These correspond to the dose rates of 6 mR/h and 140 mR/h, respectively, calculated by the staff following 1-hour operation at 100 W. The agreement between the licensee's values and those calculated by the staff is not unreasonable.

The radiation from the unshielded core would be highly collimated, so that if an individual did not expose himself directly to the core, he could work in the immediate vicinity of the tank and its biological shield for sufficient time to correct the situation by whatever means was appropriate for the source of the leak. If the leak occurred during a period when the facility was unoccupied, the dose rate would be substantially reduced by the time corrective action was initiated as a result of radioacitve decay, providing more options for correction of the situation.

On the basis of the above considerations, the licensee states that the possibility of loss-of-coolant/shielding water is remote and that the consequences would be unlikely to cause damage to the reactor or result in serious radiation exposure to the staff, students, or other occupants of the NEL Building. The staff concurs with these findings and further concludes that a rapid loss of tank water following extended operation at 100 W would result in a 1C° fuel temperature rise; this precludes fuel melting and fission product release.

14.5 Effects of Fuel Aging

The active portion of the UO AGN fuel consists of fine particles of UO_2 dispersed in polyethylene. Each active fuel section has the appearance of a homogenous, unclad, solid polyethylene block. The licensee notes that the University of West Virginia normally operated the reactor at power levels up to 75 W without any gaseous fission product release problems.

The licensee notes that the fuel has been used for over 20 years with no indication of deterioration. Because the AGN elements are unclad in the usual sense of the term, cladding corrosion or cladding rupture is not a concern with these elements.

On the basis of this information, the staff concludes that there is reasonable assurance that fuel aging will not significantly increase the quantity of gaseous fission products released to the pool water at 100-W operation, above those estimated on the basis of current operational experience. Further, the staff concludes that the licensee's monitoring system, which shifts the air above the pool surface, will provide adequate warning of incipient fuel material deterioration to permit appropriate corrective action so that no significant risk to the health and safety of the staff, students, or public would result from fuel deterioration.

14.6 Maximum Hypothetical Accident

The fuel loading accident, discussed below, has been designated to be one of the two maximum hypothetical accidents (MHAs) considered.

14.6.1 Fuel Loading Accident

The licensee has analyzed an accident resulting from the loading of a fuel element into a flux-trap with an experiment in the glory hole. The base-line conditions for the accident assume that (1) the pool water is at 20°C, (2) the glory hole has an experiment with a positive worth of 0.1%, (3) the total excess reactivity is 0.65% Δk/k, (4) the safety and coarse control rods are worth 1.0% Δk/k, each, (5) the fine control rod is worth 0.45% Δk/k, and (6) all four rods are fully inserted. Under these assumptions the reactor is shut down by -2.8% Δk/k (3 x -0.0% Δk/k -0.45% Δk/k = 0.65% Δk/k).

The scenario assumes that control of the reactor is transferred to a new operator who is instructed to convert the core from a flux-trap core to a standard core. In violation of procedures and though it is not dimensionally possible, it is assumed that the operator loads a fuel element into the empty flux-trap. EXTERMINATOR computer code calculations estimated that an element in a central flux-trap has a reactivity worth of 4.6% Δk/k. The reactor, with all the safety and control rods fully inserted, is then prompt critical with an excess reactivity of 1.8% Δk/k. This reactivity insertion is similar to the 2% Δk/k step insertion analyzed by AGN (October 1957) in the SAR accompanying the original operating license application.

The accident analyzed by AGN assumed that (1) the reactor was operating at 1 W, (2) the energy in the core at the start of the accident was negligible compared with the energy liberated during the accident, (3) no heat was removed from the core during the excursion, and (4) the only means of limiting the excursion was the negative temperature coefficient (no period scram). The results of the analysis indicate that the 2% Δk/k step insertion would result in a period of 5 to 10 msec with the excursion lasting 100 msec. The average core temperature would rise about 72C°, terminating the excursion. The peak power would be 120 MW with a total energy release of about 4.0 megajoules.

Assuming an initial fuel temperature of 20°C, the average fuel temperature would be about 92°C and the maximum fuel temperature would be about 130°C. Because a temperature of at least 200°C is required to melt polyethylene and its softening point is in excess of 150°C, no fission product release would occur; however, as polyethelene softens at about 150°C the fuel could become distorted, and for all practical purposes the polyethylene-fuel portion of the fuel elements might be destroyed. The excursion would result in a dose of 1 to 3 R to a person standing for 1 hour at the edge of the pool and looking down into the reactor. This is a fraction of the annual whole body dose delineated in 10 CFR 20.

At 100-W operation the average fuel temperature would be about 96°C (see Section 4.2.1). As the excursion would result in a total energy release similar to the above mentioned 1-W excursion, the temperature rise would also be about the same as for the excursion from 1-W operation. Calculations indicate that the final temperature would be about 134°C. The consequences would be similar to that described above, i.e., no fission product release but possible fuel distortion.

14.6.2 Conclusion

In accordance with the discussions the above noted conservative assumptions [(2), (3), and (4)] and above analyses, the staff concludes that the hypothetical insertion of a fuel element into a flux trap, which would result in a prompt criticality excursion, would not melt the polyethelene fuel cladding, so there could be no release of fission products and direct radiation would be a small fraction of 10 CFR 20 guidelines. Accordingly, the staff concludes that this postulated maximum hypothetical accident poses no risk to the health and safety of the public or to the reactor staff and students.

Table 14.1 Radiation exposures following loss of tank water for two reactor operation times at 100 W

Post-shutdown times	Operating Times: 1 hour		Operating Times: 15 hours	
	Inventory (Ci)	Dose Rate (R/h)	Inventory (Ci)	Dose Rate (R/h)
1 min	335.0	41.0	446.0	54.0
10 min	122.0	14.8	225.0	27.0
1 day	1.1	0.14	13.0	1.6
15 days	0.045	0.006	0.66	0.081

15 TECHNICAL SPECIFICATIONS

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

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

16 FINANCIAL QUALIFICATIONS

The University of Oklahoma is a state supported institution of higher education. Operation of the reactor is centered in the College of Engineering. The annual cost of operating the reactor is between $20,000 and $25,000 per year. Capital equipment purchases, when required, come from equipment funds within the College of Engineering.

All funds associated with the reactor are derived from appropriations by the State Legislature.

The staff concludes that funds will be made available 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 applicant's financial status was reviewed and found to be acceptable in accordance with the requirements of 10 CFR 50.33(f).

17 OTHER LICENSE CONSIDERATIONS

17.1 Prior Reactor Utilization

Previous sections of this SER conclude that neither normal operation of the reactor nor accident conditions could cause radiation exposures to the public that are more than a small fraction of 10 CFR 20 limits.

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

Because the staff has concluded that the reactor was initially designed and constructed to be inherently safe, the staff also must consider whether operation will cause significant degradation in these features. Possible mechanisms that would lead to detrimental changes in integrity are (1) corrosion, radiation damage, or erosion of the polyethylene cladding leading to blistering or other weakening; (2) mechanical damage as a result of handling or experimental use; and (3) degradation of safety components of systems.

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

(1) Water flow through the core is obtained by natural thermal convection, so the staff concludes that erosion effects as a result of high flow velocity will be negligible. High primary water purity is maintained by continuous passage through the filter and demineralizer system. With conductivity below 15 μmho/cm, corrosion of the aluminum components is expected to be negligible, even over the 20 years of the license renewal period.

    The fuel is periodically examined visually. When required in the past, the epoxy coating on the fuel elements was scraped and recoated. As the effect of the high flux level on the integrity of the cool expoxy coating is not known, the fuel will continue to be visually inspected to determine the condition of the epoxy coating. However, any small quantity of fission products that may escape into the pool water will be removed by the demineralizer system.

(2) The fuel is handled as infrequently as possible, consistent with periodic surveillance. Any indications of possible damage or degradation are investigated immediately. The only experiments that are placed near the core are isolated from the fuel cladding by a water gap and at least one metal barrier. Therefore, the staff concludes that mechanical damage of the core as a result of fuel handling does not constitute a significant risk to the public.

(3) UO 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 concludes that the preventive maintenance program would lead to adequate identification and replacement before significant degradation occurred. Therefore, the staff concludes that there has been no apparent significant degradation of safety equipment and, because there is strong evidence that any future degradation will lead to prompt remedial action at the UO reactor facilty, 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.

For the above reasons, the staff concludes (1) that the risk of radiation exposure to the reactor operators, students, or the public will continue to be well within all applicable regulations and guidelines during the history of the reactor and (2) that there is reasonable assurance that there will be no increase in that risk in any discernible way during this renewal period.

17.2 Multiple or Sequential Failures of Safety Components

Of the many accident scenarios hypothesized for the UO AGN research reactor, none produce consequences more severe than the design-basis accidents reviewed and evaluated in Section 14. Additionally, the reactor contains redundant safety-related measuring channels and control rods. Failure of all but one control rod and all but one safety channel would not prevent reactor shutdown to a safe condition. The staff concludes that there is no mechanism by which failure or malfunction of one of these safety-related components could lead to significant health and safety consequences to the public.

18 CONCLUSIONS

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

(1) The application for renewal of Operating License R-53 for its research reactor filed by the University of Oklahoma, dated October 6, 1978, as amended and supplemented, complies with the requirements of the Atomic Energy Act of 1954, as amended (the Act), and the commission's regulations set forth in 10 CFR Chapter 1.

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

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

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

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

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NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0996
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16. ABSTRACT (200 words or less)

This Safety Evaluation Report for the application filed by the University of Oklahoma for a renewal of operating license number R-53 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 and operated by the University of Oklahoma and is located on the campus in Norman, Cleveland County, Oklahoma. The staff concludes that the AGN-211 reactor facility can continue to be operated by University of Oklahoma without endangering the health and safety of the public.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
Nonpower Reactor University of Oklahoma AGN-201 License Renewal	
17b. IDENTIFIERS/OPEN-ENDED TERMS	

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