APPLICATION FOR RENEWAL OF CLASS 104

FACILITY LICENSE R-110 FOR OPERATION

OF THE AGN-201 REACTOR

BY:

IDAHO STATE UNIVERSITY

POCATELLO, IDAHO

November 23, 1995

1. COMPLIANCE OF FACILITY WITH APPLICABLE STANDARDS AND CRITERIA

Idaho State University has operated the AGN-201M reactor, serial #103, under license #R-110, issued by the USNRC, since 1967. The records indicate that operation has been in accordance with applicable USNRC rules and regulations, with the exception of a few minor violations found during USNRC inspections over the past twenty-eight years.

In 1967 the Atomic Energy Commission approved Idaho State University's license application for an AGN-201 reactor with an initial licensed power of 100 mWt. In 1975 the USNRC approved the relocation of the reactor from the Physical Science Building to the College of Engineering Lillibridge Engineering Laboratory on the ISU campus. Later, the USNRC approved a license amendment to increase the licensed power to 5 Wt.

2. EMERGENCY AND SECURITY PLANNING

Emergency and Physical Security procedures have been prepared and used for the AGN-201 reactor facility since its licensing in 1967. Both the Emergency Plan and the Physical Security Plan are audited, for implementation purposes, by the Reactor Safety Committee on a biennial basis. Revision 5 of the Emergency Plan is the currently approved edition and Revision 3 of the Physical Security Plan is the currently approved edition. Copies of both plans are in Appendix A.

The ISU AGN-201 reactor facility is also submitting Revision 6 of the Emergency Plan and Revision 4 of the Physical Security Plan to the USNRC for approval during this license renew review.

3. TECHNICAL SPECIFICATIONS

Amendment 4 of the Technical Specifications was prepared by ISU and approved by the USNRC, April 22, 1988. It is located in Appendix B. A revision to the Technical Specifications will be submitted for USNRC approval at a later time, in early 1996.

4. OPERATOR LICENSING AND REQUALIFICATION PROGRAM

Attached in Appendix C is the Operator Requalification program submitted under Docket # 50-284, License # R-110 and approved by the USNRC on August 17, 1995. The requalification program is rigorously followed and required records are kept of the requalification activity.

5. FINANCIAL CONSIDERATIONS

The AGN-201 reactor is owned and operated by Idaho State University. The University provides a budget adequate to employ a full-time Reactor Supervisor and to provide for the replacement of defective components. Other expenses incurred in the operation of the reactor are normally covered by the College's operating budget.

Information on the financial qualifications of the University to operate and decommission the AGN-201 reactor facility is located in Appendix D.

6. ENVIRONMENTAL CONSIDERATIONS

No environmental effects should result from use of this reactor. Since the AGN- 201 reactor has a dry core of uranium-impregnated polyethylene, sealed in an aluminum tank, there is no significant release of radioactivity to the environment. There is effectively little to no solid waste associated with the reactor operation. What solid waste that is generated is disposed of in accordance with the University Radiation Safety Policy Manual. Health physics support for the AGN-201 reactor is provided by the University's Technical Safety Office and by the licensed reactor operators of the facility. A copy of the Environmental Report is included in Appendix E.

7. SAFETY ANALYSIS REPORT

The Hazards Summary Report for the AGN-201 reactor prepared for the original license application in 1967 has been recently updated and re-formatted as a Safety Analysis Report. This new Safety Analysis Report is attached as Appendix F.

APPLICATION FOR RENEWAL OF CLASS 104 FACILITY LICENSE R-110 FOR OPERATION OF THE AGN-201 REACTOR

APPENDIX A

EMERGENCY PLAN AND PHYSICAL SECURITY PLAN

BY:
IDAHO STATE UNIVERSITY
POCATELLO, IDAHO

NOVEMBER 23, 1995

EMERGENCY PLAN FOR

THE NUCLEAR FACILITIES AT

LILLIBRIDGE ENGINEERING LABORATORY

AT IDAHO STATE UNIVERSITY

APPROVED: 11/22/1995

REVISION 6

EMERGENCY PLAN
CONTENTS

1.0 INTRODUCTION

This emergency plan shall be used as a plan of action to follow in the event of a nuclear incident in the Nuclear Operations Area of Lillibridge Engineering Laboratory located at Idaho State University, Pocatello, Idaho.

The Nuclear Operations Area contains an AGN-201 nuclear reactor manufactured by Aerojet General Nucleonics in 1956 and a Subcritical Assembly. The reactor and Subcritical Assembly are owned by Idaho State University and operated under USNRC License Nos.-110 and SNM-1373, respectively. The maximum power at which the reactor is licensed to operate is 5 W. The fuel for both facilities consists of uranium enriched to 19.88% U-235.

The AGN-201 reactor system consists of two basic units, the reactor and the control console. The reactor unit includes the core, consisting of uranium dioxide dispersed in polyethylene, a graphite reflector, and the lead and water shielding. Fuel loaded control and safety rods are installed vertically from the bottom of the reactor unit. These rods pass by the nuclear instrumentation which measures the power level. Rod movement is achieved by the use of control rod drive mechanisms which provide safe and efficient operation of the reactor. The weight of the reactor unit, with the water shield, is 20,000 pounds; the weight of the reactor control console is 800 pounds.

The AGN-201 reactor is located in Room 20 in the basement of the Lillibridge Engineering Laboratory building at Idaho State University. Refer to Appendix A for the floor plans of the laboratory.

The Subcritical Assembly is located in Room 23.

2.0 DEFINITIONS

Emergency Planning Zone - Rooms 20 and 23 on the first level of the Lillibridge Engineering Laboratory building.

Operations area - The area inside Rooms 20 and 23.

Operations boundary - The walls, ceilings, and doors of Rooms 20 and 23.

Nuclear facility - Consists of an AGN-201 nuclear reactor and the subcritical assembly.

Emergency Response team - Consists of health physicists, NRC- licensed reactor operators, senior reactor operators, and Radiation Safety Officer, Reactor Administrator, Reactor Supervisor.

Radiation accident - Any release of radioactivity which may injure or contaminate a person.

Nuclear incident - Any unusual circumstances or occurrence that could lead to or cause damage to the reactor and/or subcritical assembly nuclear fuel or nuclear fuel cladding.

Nuclear emergency - Any emergency which combines a radiation accident with any nuclear incident.

Emergency Support Center - Is located in the northeast corner of the Machine Shop near the large bay door. It serves as a coordination area for all emergency response agencies for addressing the emergency.

3.0 ORGANIZATION AND RESPONSIBILITIES

The Idaho State University personnel who will respond to a nuclear incident are the Reactor Administrator, Reactor Supervisor, Radiation Safety Officer, Technical Safety Office, licensed reactor operators and Campus Security. They will be advised by the College of Engineering faculty. There are no other local support organizations directly committed to respond to a nuclear incident other than the normal response provided by the local fire and police departments.

In order for the emergency plan to function as intended, it is essential that all coordinating personnel at Idaho State University be aware of their areas of responsibility and assure that their facilities and equipment are available and operational. The following is a list of University personnel and their areas of responsibility:

1. The **Reactor Administrator and/or Reactor Supervisor** are responsible for:
   a. Operations at Idaho State University nuclear facilities should a nuclear incident occur.
   b. Notification of the State Police and Idaho Department of Health and Welfare in the event of a radiation accident.
   c. Requests for medical assistance or notification of a local hospital to prepare for patient care.
   d. Safety regulations and practice with the nuclear facilities.
   e. Internal operations and assignments.
   f. Routine checking of safety equipment and safety within the facility and assuring that employees are knowledgeable of equipment operation.
   g. Requests for additional fire fighting assistance, and advising the fire Marshall concerning the hazards of the nuclear facilities.
   h. Evacuation plans and assembly areas.
   I. Maintaining up-to-date notification roster of appropriate personnel and agencies.
   j. Personnel accountability procedures in the Emergency Planning Zone.
   k. The training of University personnel who are responsible to act under this emergency plan.
   l. In the absence of both the Reactor Administrator and Reactor Supervisor, the Radiation Safety Officer shall assume their duties.

2. The **Radiation Safety Officer** is responsible for:
   a. Health physics assistance at Idaho State University.
   b. Authorizing volunteer emergency workers to incur radiation exposure in excess of normal occupational limits.
   c. Manning check points or control points for surveying personnel and equipment.

d. Health physics at the emergency site and scheduling of personnel and working times in radiation areas.
e. Monitoring teams and environmental sampling at Idaho State University, analysis of samples, and maintenance of records.
f. Decontamination procedures and control.
g. Health physics for contaminated personnel until they are attended by the proper medical personnel.
h. Insuring that all necessary health physics information is communicated to the appropriate agencies.
I. Personnel monitoring, personnel radiation records, and for scheduling of personnel for the operations team.
j. In the absence of the Radiation Safety Officer, the Reactor Administrator or the Reactor Supervisor will assume these duties.

3. **Idaho State University Campus Security** will be responsible for:
a. Establishing area control and manning of c! points.
b. Traffic control and traffic counting.
c. Assistance in communications and information dissemination.
d. Assisting State Police in the event of a radiation accident.

Refer to Appendix B for a list of organizations who have indicated they will provide assistance upon request as part of their normal duties or services.

4.0 EMERGENCY PROCEDURES

Table 1 shows the Emergency Classification Scheme for potential emergency situations which may occur at the College of Engineering. However, in view of the operating history, fuel inventories, and physical size of the AGN-201 Reactor and the Sub-critical Assembly, an emergency is not expected to extend beyond the emergency classification designation of "Notification of Unusual Events" as defined in U.S. NRC NUREG-0849.

Emergency Class	Action Level	Purpose
Notification of Unusual Events	•Civil Disturbance •Bomb Threat •Theft of Special Nuclear Material •Fire or Explosion •Nuclear Emergency	(1) Ensure that the first step in any response later found to be necessary has been carried out, (2) bring the operating staff to a state of readiness, and (3) provide systematic handling of unusual events, information and decision-making.

Table 1. Emergency Classification Scheme

The Emergency Planning Zone (EPZ), consists of Rooms 20 and 23, in the basement of the Lillibridge Engineering Laboratory (See Figure 1 in Appendix A). This emergency plan shall apply to the EPZ. There are no postulated accidents for the AGN-201 Reactor or Subcritical Assembly which would result in the exposure of 1 rem whole body or 5 rem thyroid beyond the operations boundary.

In the event of an incident which requires evacuation of the Lillibridge Engineering Laboratory, i.e., fire or explosion anywhere within the Laboratory, all personnel within the EPZ shall proceed to the southeast corner of the machine shop area near the double doors for accountability and radioactive contamination checking. If radioactive contamination is suspected, the potentially contaminated personnel will be separated from other personnel.

The emergency exposure guidelines are the same as the radiation dose standards for individuals in restricted areas as specified in 10 CFR 20.1201. These guidelines are sufficient when the size and postulated radiation accidents are considered for the nuclear facilities at Idaho State University.

The College of Engineering will maintain emergency procedures for dealing with various emergencies including:

- nuclear emergencies
- bomb threats
- fires or explosions
- theft or attempted theft of Special Nuclear Material
- and civil disturbance.

The response procedures describe the type of response to be accomplished and the duties and responsibilities of the security organization and the nuclear facility management involved in the response. An up-to-date notification roster will be maintained in the Reactor Supervisor's office, in the College of Engineering administrative office, the evacuation area in the Machine Shop and the Campus Security office. The notification roster indicates the names and telephone numbers of those who will be notified immediately of any emergency and also the names and telephones numbers of those who may be called upon to assist. Refer to Appendix B for the notification roster. (Note: Updating this roster does not constitute a revision to the Emergency Plan.)

A number of radiation monitoring devices are maintained at the Technical Safety Office. In addition, radiation and contamination monitoring devices for emergency use only, are stored in the file cabinet in the Machine Shop. The Radiation Safety Officer will determine which devices are to be used. A Geiger counter will be the standard device for monitoring. All personnel entering a radiation area or a suspected radiation area shall have some method of determining the radiation field and personal dose. No person shall enter a suspected radiation area unless under the direction of the Radiation Safety Officer. The exception shall be for the police and the fire departments. If it is necessary that the police or firemen enter a radiation area without personal radiation monitoring devices, the Radiation Safety Officer shall be informed immediately and will then survey the affected person for contamination and arrange for a whole body assay if necessary.

The L.E.L. Building fire alarm system is continuously monitored by the ADT Security Company. In the event a fire alarm pull station is activated, an auto dialer sends a signal to ADT, who in turn calls the ISU Campus Security Office to inform them of an alarm signal at the L.E.L. Building.

4.1 Nuclear Emergency

A nuclear emergency shall be any emergency which combines a radiation accident with any other nuclear incident. Evacuation of the Lillibridge Engineering Laboratory building may or may not be required for a nuclear emergency. If the emergency is strictly a radiation accident and **NOT** combined with fire or explosion, building evacuation will be ordered if the radiation levels are above 10 mR/hr outside the operations boundary or if there are airborne radioactive materials.

In the improbable event that the Nuclear Reactor Laboratory must be evacuated, personnel should leave the Laboratory via either of the accessible two exits. These are: (1) through the double doors to the vestibule and up the stairs and (2) an emergency escape hatch located in the roof. Two sets of stairs and an elevator lead from the first level (basement) to the second level (ground level) floor from which three exit points are available; one to the south and two on either side of the display foyer. A ladder leads to an escape hatch in the ceiling of the reactor laboratory which can only be opened from the inside.

The emergency exit sequence shall be as follows:

(1) Personnel in adjacent laboratory spaces shall be warned by operators to initiate evacuation;

(2) The first person to reach the Emergency Ventilation Cut-Out Switch (located on the south wall, across from the Reactor Supervisor's office, shown in Figure 1 in Appendix A, will trip all ventilation , except exhaust hoods, off line in the building preventing any further air exchange;

(3) If time permits, radiological monitoring equipment will be taken from the reactor laboratory. If this is not possible other monitoring equipment has been stored at the Physical Science Building and evacuation area in the machine shop for emergencies;

(4) The building fire alarm will be sounded to evacuate the entire building. The locations of the local fire alarms are at the bottom of the staircase on the south side (or on the way to the staircase on the north side) of the building and on the east wall of the reactor laboratory;

(5) The last person leaving the reactor laboratory area will shut all doors. All persons leaving the EPZ area shall proceed to the southeast corner of the machine shop area near the double doors for accountability and checking for radioactive contamination.

(6) Operating Staff personnel shall proceed to the Fire Department Emergency Command Center (if established) and provide information and assistance to the On Scene commander;

(7) The University Administration will be notified as soon as is practicable. The Reactor Administrator and the Reactor Supervisor will be notified

immediately. They will, in turn, determine which State and Federal agencies shall be notified. The Radiation Safety Officer will be responsible for thorough radiation monitoring. The Nuclear Reactor Laboratory will be reentered as radiological levels permit and then only by authorization of the Reactor Supervisor or his designated representative. The reactor system will be checked for damage. Refer to Appendix C for the Emergency Evacuation Plan.

4.2 Bomb Threat

1. Shut down the reactor.
2. The person receiving the threat should obtain as much information as possible. Ask the following questions:
    a. Where is the bomb?
    b. What kind of bomb is it?
    c. What time will it go off?
    d. Why are you doing this?
3. Notify:
    a. Idaho State University Campus Security
    b. Pocatello Police Department
4. The campus security officer, upon being notified of the threat, will proceed immediately to notify the following offices:
    a. Chief of Campus Security
    b. Pocatello Police Department
    c. University Administration
    d. Reactor Administrator
    e. Reactor Supervisor
5. The campus security officer will record the name and location of the person receiving the threat.
6. The removal or transfer of any radioactive material will be the responsibility of the Reactor Administrator and/or the Reactor Supervisor.
7. College of Engineering staff will assist with any subsequent searches of the Lillibridge Engineering Laboratory building.
8. The Operating Staff will notify the U.S. Nuclear Regulatory Commission, Region IV.

4.3 Fire or Explosion

1. Scram the reactor by pushing the power off button and check that the rods have scrammed by any of the following methods:
   a. Rods engaged lights out.
   b. Decreasing current trace on channel 2 and 3 strip charts.
   c. Period meter pegged low.
2. As soon as the scram is verified, evacuate the building.
3. Notify the Pocatello Police Department by the quickest available means, i.e., radio, fire alarm, telephone. (911)
4. The Pocatello Fire Department will:
   a. Proceed to the area.
   b. Notify Pocatello Police Department and Idaho State University Campus Security for traffic control.
5. ISU Campus Security will notify the campus maintenance department. The maintenance department will provide a person to secure or activate building systems and alarms as necessary.
6. Notify the Reactor Administrator and/or Reactor Supervisor who will in turn notify:
   a. Idaho State University Radiation Safety Officer.
   b. U.S. Nuclear Regulatory Commission Region IV.

4.4 Theft or Attempted Theft of Special Nuclear Material

1. If an indication of a theft or an attempted theft exists in or around Rooms 20 or 23 of the Lillibridge Engineering Laboratory, immediately notify the Campus Security who will in turn notify:
   a. Chief of Campus Security
   b. Pocatello Police Department
   c. Reactor Administrator
   d. Reactor Supervisor
2. The Reactor Administrator and/or the Reactor Supervisor will proceed immediately to the Lab and inspect and inventory all Special Nuclear Material. If a theft or attempted theft of Special Nuclear Material has occurred, the U.S. Nuclear Regulatory Commission, Region IV will be notified immediately.

4.5 Civil Disturbance

1. Notify the Chief of Campus Security.
2. Campus Security will post guards in the basement by the Sub-critical Assembly lab and Reactor Laboratory.
3. Notify the Pocatello Police Department for riot and incident control.

5.0 EMERGENCY FACILITIES AND EQUIPMENT

The Emergency Support Center is the northeast corner of the Machine Shop near the large overhead door, Appendix A, Figure 2. Emergency control directions will be given from this area.

An ambulance shall be dispatched for any person who may be injured. If that person is also contaminated with radioactive materials, the receiving hospital shall be informed the injured person is potentially contaminated.

If a person is not injured but contaminated with radioactive materials, decontamination procedures will begin under the direction of the ISU Radiation Safety Officer.

The only emergency communications system in addition to the normal telephones are the hand-held radios which are used by campus security. Emergency communications will have to be by word of mouth if the telephone system or the Campus Security radios are inoperable.

6.0 MAINTAINING EMERGENCY PREPAREDNESS

The Reactor Administrator and the Reactor Supervisor are responsible to ensure the proper execution of the Emergency Plan.

The training of University personnel who are responsible to act under this emergency plan is the responsibility of the Reactor Administrator and the Reactor Supervisor with the assistance of the Technical Safety Office in the area of radiological control. Off-site support organizations are responsible for their own training.

The Idaho State University Reactor Administrator and the Reactor Supervisor will provide a training program at least once a year to train other University personnel who may be called upon to assist in the improbable event of a nuclear incident.

University personnel who would be involved in a nuclear incident will be tested by annual drills. This will be accomplished by the unannounced initiation of an emergency drill by the Reactor Administrator or by the Reactor Supervisor with written permission from the Reactor Administrator. Outside agencies will be contacted in advance and informed of the drill. University personnel will carry through with this action as though it were an actual emergency. Records of these drills will be entered into the facility operating records by the Reactor Supervisor or a licensed Senior Reactor Operator or Reactor Operator.

It is the responsibility of the Reactor Supervisor and/or Reactor Administrator to ensure that Letters of Agreement with off-site support organizations, who respond to nuclear emergencies, be updated on a biennial basis, but not to exceed 30 months. These letters shall be kept in the Reactor Supervisor's office.

The Emergency Plan shall be audited under the cognizance of the Reactor Safety Committee at least once every two years. They shall evaluate the effectiveness of the plan and note the results of the evaluation in their minutes. They shall also approve any changes which may be made to the plan.

Emergency equipment used for fire fighting, radiation detection and air sampling shall normally be checked for proper operation annually, but in no case shall the check be greater than 16 months. Batteries in portable equipment shall be checked prior to each use and annually unless previous experience dictates a more frequent check is required. A complete stock of replacement batteries shall be available for all battery powered emergency equipment. Emergency equipment will be inventoried annually.

APPENDIX A - FIGURES

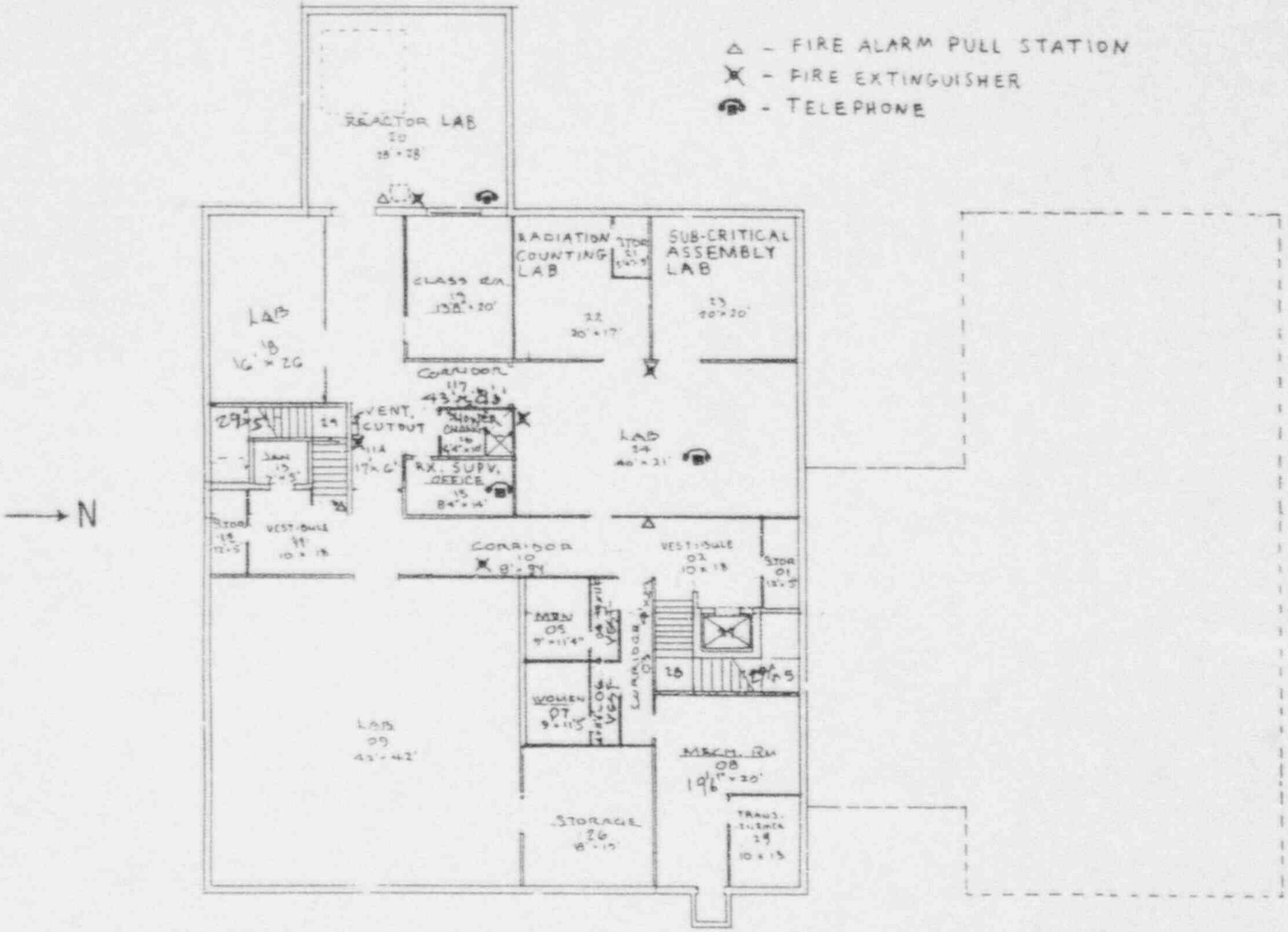


Figure 1 - First (basement) LEL floor plan

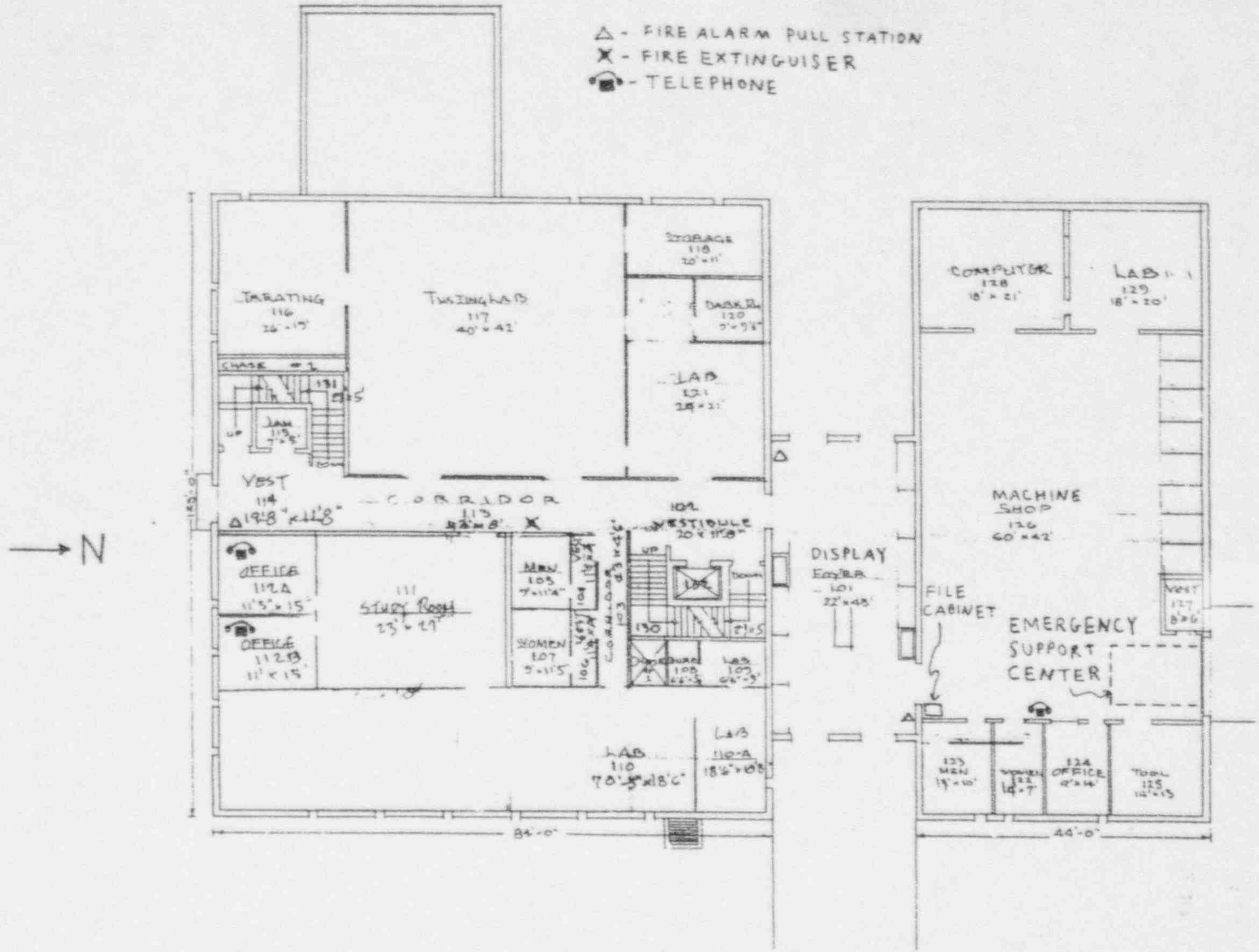


Figure 2 - Second (ground) level of LEL floor plan

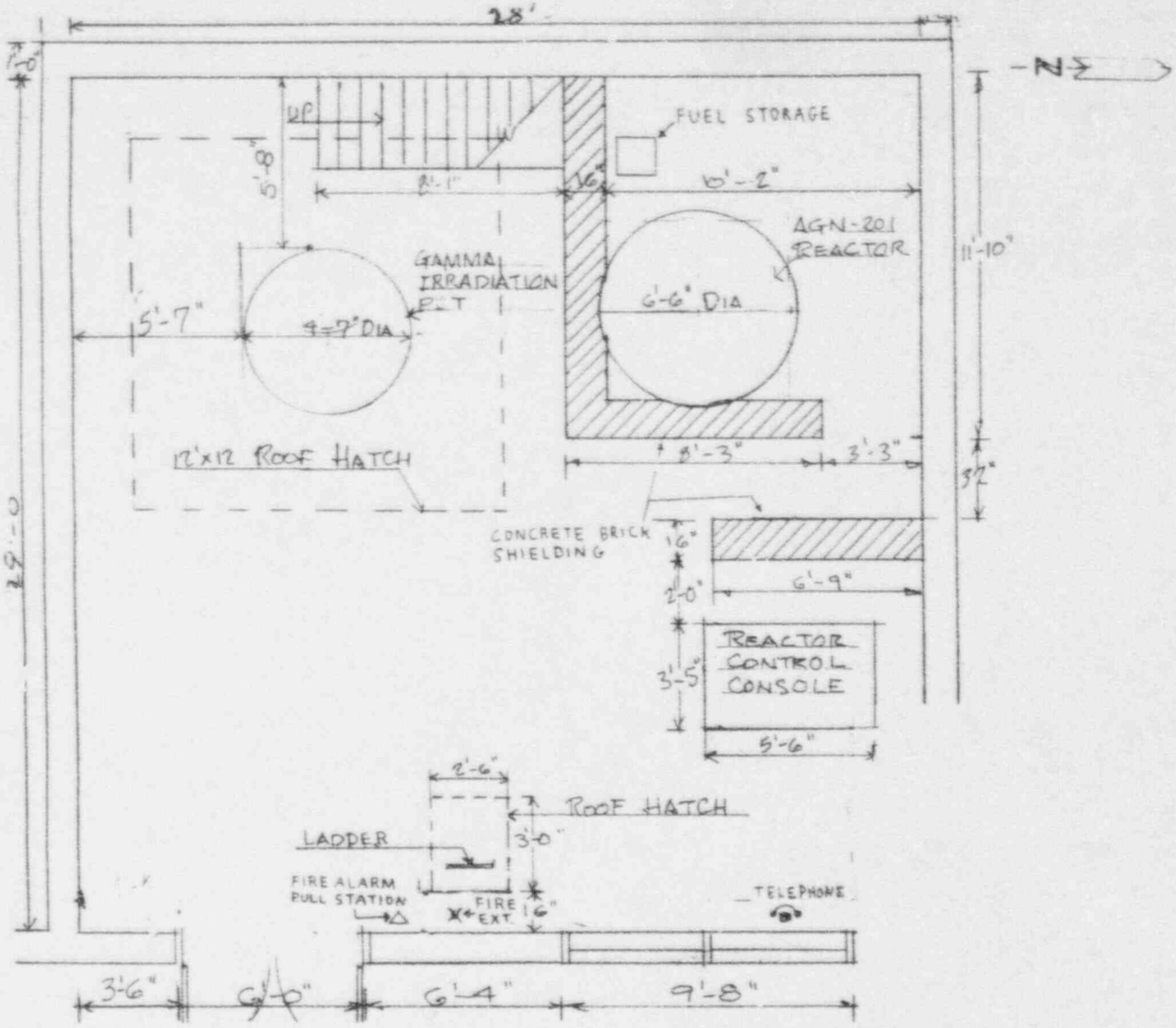


Figure 3 - Reactor laboratory floor plan (basement)

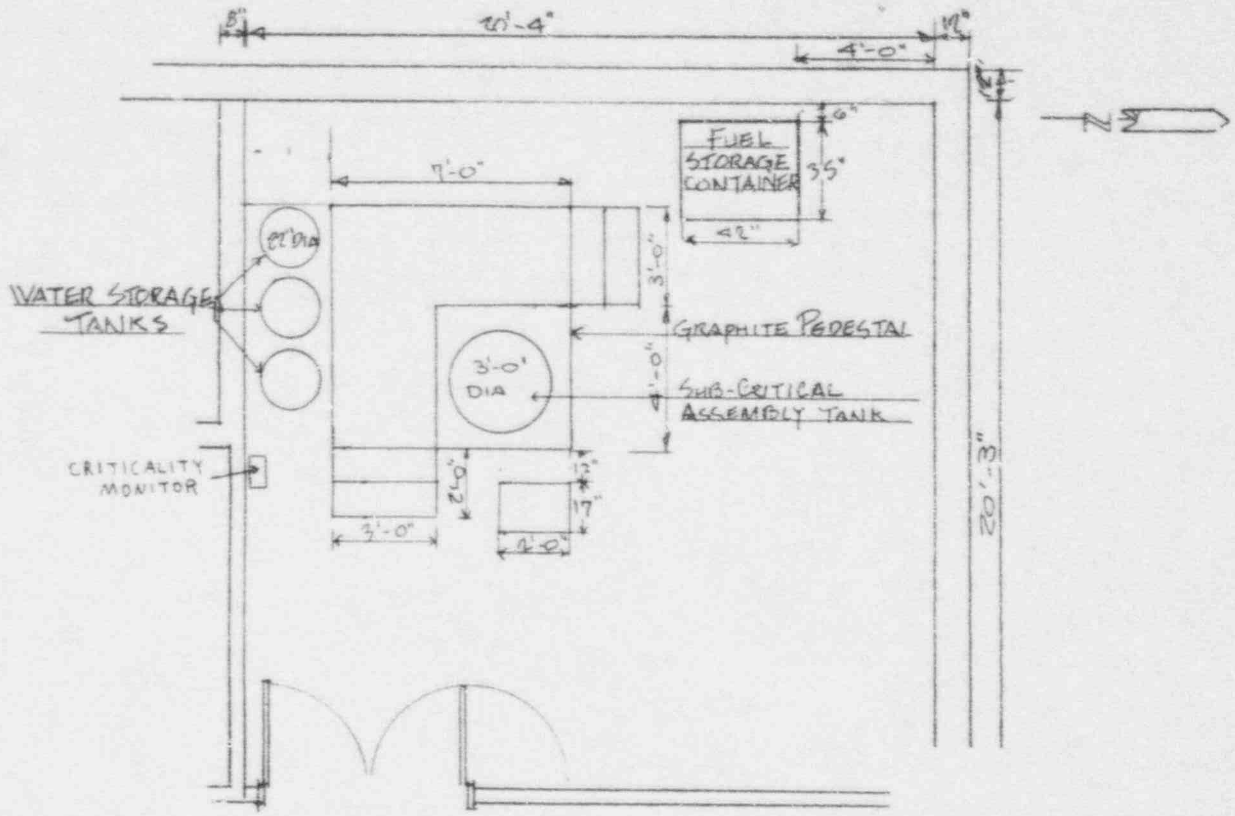


Figure 4- Sub-critical laboratory floor plan (basement)

APPENDIX B
NOTIFICATION ROSTER

REACTOR ADMINISTRATOR	A. STEPHENS	HOME: 208-524-0905 WORK: 208-526-4907
REACTOR SUPERVISOR	D. CLOVIS	HOME: 208-234-0581 WORK: 208-236-3637
REACTOR OPERATORS:	J. BENNION	HOME: 208-233-3239 WORK: 208-236-3351
	B. BOSTON	HOME: 208-232-2191 WORK: 208-236-2311
	K. BUNDE	HOME: 208-234-0581 WORK: 208-533-7473
	J. MCWHIRTER	HOME: 208-523-6404 WORK: 208-236-4021
RADIATION SAFETY OFFICER	T. GESELL	HOME: 208-237-1076 WORK: 208-236-3669
ISU CAMPUS SECURITY		208-236-2515
ISU ADMINISTRATION		208-236-3440

OFF-SITE SUPPORT ORGANIZATIONS

* POCATELLO POLICE DEPARTMENT	911
* POCATELLO FIRE DEPARTMENT	911
* IDAHO STATE POLICE	208-232-1426
* BANNOCK REGIONAL MEDICAL CENTER	208-239-1000
NUCLEAR REGULATORY COMMISSION	817-860-8100 AFTER 3:15PM(MST)301-816-5100

* Indicates ISU has Letters of Agreement with these agencies.

UPDATED______

APPENDIX C.

EMERGENCY EVACUATION PLAN

1. The licensed reactor operator is cognizant of the detailed emergency plan. HE/SHE WILL BE IN CHARGE OF EVACUATION, if present when the emergency is discovered.
2. Use the normal room exit and building exits if possible. The escape hatch located in the roof is to be used only if normal exits are blocked by fire or radiation. Make sure exits from the Reactor Laboratory are closed after all persons are out.
3. The radiological monitoring instrument on the reactor console and the reactor log book will be brought from the laboratory room by the reactor operator.
4. If the radiation levels are above 10 mR/hr outside the operations area of the Nuclear Reactor Laboratory (Rm 20) or Subcritical Assembly Laboratory (Rm 23), **OR** if there are airborne radioactive materials, the reactor operator will order building evacuation.
5. The reactor operator shall initiate building evacuation by tripping one of the building fire alarms located at the bottom of the staircase on the south side (or on the way to the staircase on the north side) of the building, or in the Nuclear Reactor Laboratory.
6. The first person to reach the Emergency Ventilation Cutout Switch (located on the south wall, across from the Reactor Supervisor's office) will turn all ventilation fans off.
7. The reactor operator shall notify the Reactor Supervisor and/or the Reactor Administrator immediately.
8. The Reactor Supervisor and/or the Reactor Administrator shall be in charge of all building re-entry.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

PHYSICAL SECURITY PLAN FOR

LILLIBRIDGE ENGINEERING LAB

IDAHO STATE UNIVERSITY

REVISION 4

APPROVED: 11/22/1995

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

PHYSICAL SECURITY PLAN

CONTENTS

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

1.0 PURPOSE

This Security Plan describes the physical protection system and security organization which will provide protection against sabotage and detect the theft of Special Nuclear Material at the Idaho State University Nuclear Reactor Laboratory and Subcritical Assembly facilities. It demonstrates compliance with 10 CFR 50.34(c), 10 CFR 73.40 and 10 CFR 73.67.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

2.0 GENERAL PERFORMANCE OBJECTIVES

The "General Performance Objectives" of the physical protection system and security organization described in this plan are:

- To minimize risk of acts of sabotage.

- To minimize the possibilities of unauthorized removal of special nuclear material consistent with the potential con sequences of such actions.

- To facilitate the location and recovery of missing Special Nuclear Material.

To achieve these objectives, the physical protection system shall provide early detection of unauthorized access or activities by an external adversary within the controlled access areas containing Special Nuclear Material. This will assure proper placement and transfer of custody of Special Nuclear Material.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

3.0 IDENTIFICATION OF SPECIAL NUCLEAR MATERIAL

FORM	USE	ENRICHMENT	AMOUNT	
CONTAINED NON EXEMPT MATERIAL			URANIUM	U-235
UO_2 Powder Fuel in Polyethylene	AGN-201 Reactor	19.88%	3371 gm	670 gm
	Spare Fuel	19.88%	27 gm	6 gm
U-Al alloy	Subcritical	19.84%	7615 gm	511 gm

EXEMPT MATERIAL		CONTAINED Pu
Pu-Be	(2)Sealed Sources	Total 20 gm (18 gm Pu-239)

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

4.0 IMPLEMENTATION

This security plan shall be fully implemented upon approval by the Reactor Administrator, Reactor Safety Committee, Director of Security for Idaho State University, and the U.S. Nuclear Regulatory Commission.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

5.0 CONTROLLED ACCESS AREAS

5.1 Use

All Special Nuclear Material may be used only within a Controlled Access Area (CAA).

Rooms 20 and 23 of the Lillibridge Engineering Laboratory are CAAs (See Appendix C - Figure 1). Room 20 is the Reactor Lab shown in Appendix C - Figure 3. It is surrounded on four sides with reinforced concrete walls. Three walls are surrounded by earth to ten feet above the floor level. The fourth wall has a double door and an observation window. Access is through this door. The Observation Room is locked except when in use. The roof of the reactor lab is a four inch concrete slab with openings as shown in Appendix C - Figure 2. The escape hatch can only be opened from the inside. The roof hatch has an exterior padlock. Both the escape hatch and roof hatch are made of plate steel. Room 23, shown in Appendix C - Figure 4, has two exterior walls with earth backfill. The other walls are double drywall construction. A double door provides access. No windows are located in the room, except for the observation windows located in the doors. The false ceiling in the room is hung from a reinforced concrete slab. Special Nuclear Materials stored in this room are kept in a padlocked steel Fuel Storage Container shown in the figure.

All access doors to the CAAs are double locked. The keys are held only by the Reactor Administrator, Reactor Supervisor, and Licensed Reactor Operators/Senior Reactor Operators, and those that the Reactor Administrator deems necessary.

5.2 Storage

All Special Nuclear Material is located only within a Controlled Access Area. All Special Nuclear Material at the facility is stored in designated areas within the CAAs. The designated areas are shown in Appendix C - Figures 3 and 4.

The Fuel Storage Container in Rm. 23 is a standard commercial security cabinet approved by the General Services Administration as Class 6 or equivalent. The Fuel Storage Container is locked or attended by an authorized individual. Five high strength padlocks are used on this security cabinet.

The storage container in Rm. 20 is an air-tight metal container that uses one high-strength padlock.

Access doors are Schlage Type 6 locks. The doors are dead-bolt locked when the rooms are not in use.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

6.0 DETECTION OF THEFT OR ATTEMPTED THEFT

The detection of theft or attempted theft of Special Nuclear Material is facilitated by administrative controls, locks and random patrols by Idaho State University Security personnel.

When the CAAs are not occupied during the normal working day, they are kept double locked with the area monitored by the presence of authorized personnel in the CAAs

During non-working hours, holidays, and weekends, the areas are monitored by the Idaho State University Security Public Safety Officers. Randomly, approximately every four hours, the Public Safety Officers physically check the status of CAAs and document their check by keying in on a bar code reading system station.

The Public Safety Officers are trained in accordance with the Reserve Officer Training Program sponsored by the Police Officer Standards and Training (POST), which is administered by the ISU School of Applied Technology. This includes arrest and self-defense techniques.

All ISU Public Safety Officers are trained annually in the emergency procedures contained in Appendix B.

In the event of an actual or attempted theft of SNM, the ISU Public Safety personnel will be notified via telephone (an operator will man the phone 24 hours per day). ISU Public Safety will in turn notify the Pocatello Police Department.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

7.0 SECURITY RESPONSE

The University facility security organization is made up of the following components:

University Security Organization

1) Director of Security
2) University Security Officer
3) Security Supervisor
4) Public Safety Officers (six)

Operating Organization (See Appendix A)

1) Reactor Administrator
2) Reactor Supervisor
3) Reactor Operators

Radiological Controls Organization

1) Radiation Safety Officer
2) Radiation Safety Officer Staff

The Reactor Administrator or his designated representative has overall responsibility for the initiation and implementation of the security program at the ISU facility. The principal local law enforcement agency is the Pocatello Police Department. Secondary law enforcement agencies are the Bannock County Sheriff and the Idaho State Police. The University Security Department maintains a liaison with the Pocatello Police Department which has committed to provide a response force when requested. The response time is normally less than ten minutes.

The University Security Department maintains at least one watchman per shift capable of providing surveillance of the CAAs. The University Security Department will (1) maintain surveillance of the CAAs, (2) maintain liaison with the local law enforcement agency, (3) notify the local law enforcement agency of any unauthorized penetrations or activities in the CAAs requiring their attention, and (4) notify the facility Operating Organization of any unauthorized penetrations or activities in the CAAs.

At least once each twenty-four months, The Reactor Safety Committee will conduct a security program audit.

8.0 RESPONSE PROCEDURES

The facility will maintain response procedures for dealing with threats or actual thefts of Special Nuclear Material. Response procedures for the following security incidents are maintained at the facility by the Reactor Supervisor and the University Security Department:

1) Bomb Threat
2) Fire or explosion
3) Theft of Special Nuclear Material
4) Civil disturbance

The response procedures describe the type of response to be accomplished, the duties and responsibilities of the security organization and the Operating Organization involved in the response, law enforcement assistance capabilities, and law enforcement response agreements.

The Nuclear Regulatory Commission will be notified in the event of theft or attempted theft of Special Nuclear Material. (See Appendix B.C.2)

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

APPENDIX A

OPERATING ORGANIZATION

POSITION NUMBER	NAME	TELEPHONE
Reactor Administrator	A. Stephens	Work: 208-526-4907 Home: 208-524-0905
Reactor Supervisor	R. Clovis	Work: 208-236-3637 Home: 208-234-0581
Reactor Operators	J. Bennion	Home: 208-233-3239 Work: 208-236-3351
	B. Boston	Home: 208-232-2191 Work: 208-236-2311
	K. Bunde	Home: 208-234-0581 Work: 208-533-7473
	J. McWhirter	Home: 208-523-6404 Work: 208-236-4021

Note: Appendix A will only be forwarded to the applicable ISU personnel when changes to the phone list are made.

RADIOLOGICAL CONTROLS ORGANIZATION

Radiation Safety Officer	T. Gesell	Work: 208-236-3669 Home: 208-237-1076
Radiation Safety Officer Staff		Work: 208-236-2311 Pager: 208-236-9101

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

APPENDIX B - EMERGENCY PROCEDURES

A. Bomb Threats to Lillibridge Engineering Laboratory

1. The person receiving the threat should obtain as much information as possible. Ask the following questions:

   a. Where is the bomb located?
   b. What type of bomb is it?
   c. What time will it go of?
   d. Why are you doing this?

2. Notify:

   a. Idaho State University Campus Security, 236-2515
   b. Pocatello Police Department, 911

3. The security officer, upon being notified of the threat, will immediately notify the following offices:

a.	Chief of Campus Security	Work: 208-236-2515 Home: 208-237-5429
b.	Pocatello Police Department	911
c.	Bannock County Sheriff	208-236-0623
d.	I.S.U. Administration	208-236-3440
e.	Reactor Administrator A. Stephens	Work: 208-526-4907 Home: 208-524-0905
f.	Reactor Supervisor R. Clovis	Work: 208-236-3637 Home: 208-234-0581
g.	<u>Reactor Operators:</u> J. BENNION	Home: 208-233-3239 Work: 208-236-3351

	B. BOSTON	Home: 208-232-2191 Work: 208-236-2311
	K. BUNDE	Home: 208-234-0581 Work: 208-533-7473
	J. MCWHIRTER	Home: 208-523-6404 Work: 208-236-4021
h.	Radiation Safety Officer Staff Office:	208-236-2311 Pager:208-236-9101
I.	USNRC Region IV After 3:15 pm MST	817-860-8100 301-816-5100

Note: Appendix B will only be forwarded to the applicable ISU personnel when changes to the phone list are made.

4. The security officer will record the name and location of the person receiving the threat.

5. The removal or transfer of any radioactive material will be the responsibility of the Reactor Administrator and/or the Reactor Supervisor.

6. College of Engineering/Radiation Safety Officer Staff will assist with any subsequent searches of the Lillibridge Engineering Laboratory.

B. Fire or Explosion at the Lillibridge Engineering Laboratory

1. Notify the Pocatello Fire Department by the quickest available means, i.e., radio, fire alarm, telephone 911.

2. The Fire Department will:

   a. Proceed to the area, but be sure that the fire fighters are wearing protective clothing and breathing devices.

   b. Notify Pocatello Police Department, Bannock County Sheriff, and/or Idaho State University Security to post guards around the area, and to keep out any unauthorized vehicles and persons.

   c. Notify the Reactor Administrator and/or Reactor Supervisor who will, in turn notify:

      (1) U.S. Nuclear Regulatory Commission Region IV
      (2) Idaho State University Radiation Safety Officer Staff

3. The Reactor Supervisor and Radiation Safety Officer will assist by monitoring the area for radioactive contamination.

CONTAINS 10 CFR 2.790 (d)
INFORMATION WITHHELD FROM
PUBLIC DISCLOSURE

C. Theft of Special Nuclear Material from Lillibridge

1. If an indication of a theft or an attempted theft exists in or around Rooms 20 and 23 of the Lillibridge Engineering Laboratory, immediately notify the I.S.U. Security who will, in turn, notify:

   a. Chief of Campus Security
   b. Pocatello Police Department
   c. Reactor Administrator
   d. Reactor Supervisor
   e. Radiation Safety Officer Staff

2. The Reactor Administrator and/or the Reactor Supervisor will proceed immediately to the Nuclear Reactor Laboratory and inspect and inventory all Special Nuclear Material. If a theft or attempted theft of Special Nuclear Material has occurred, the U.S. Nuclear Regulatory Commission Region IV will be notified immediately at 817-860-8100 (or 301-816-5100 after 3:15 P.M. MST).

D. Civil disturbance in the vicinity of Lillibridge

1. Notify the Chief of Campus Security

2. I.S.U. Security will post guards in the basement of the Laboratory.

3. Notify the Pocatello Police Department and/or the Bannock County Sheriff for riot and incident control.

APPENDIX C - FIGURES

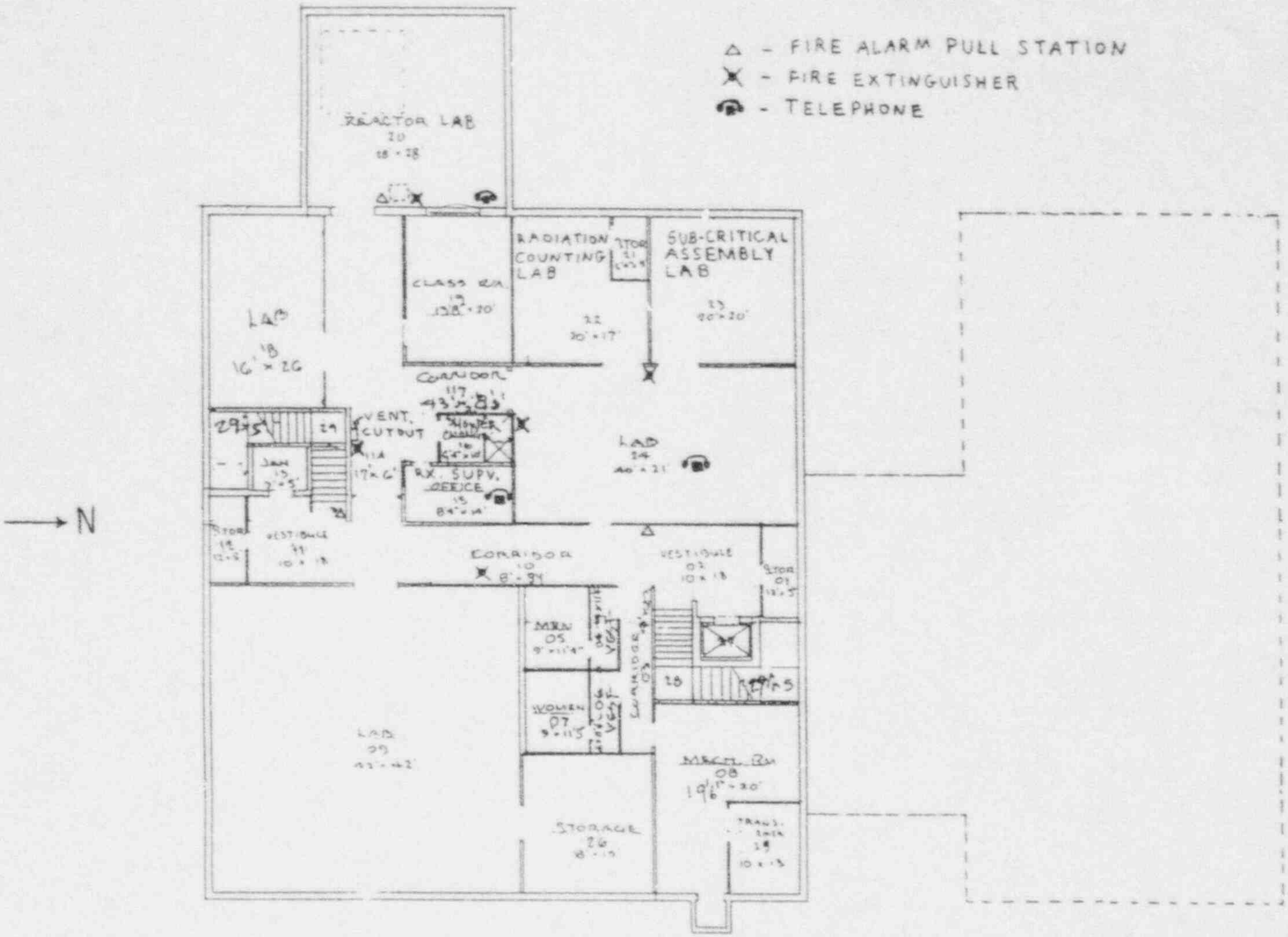


Figure 1 - First (basement) LEL floor plan

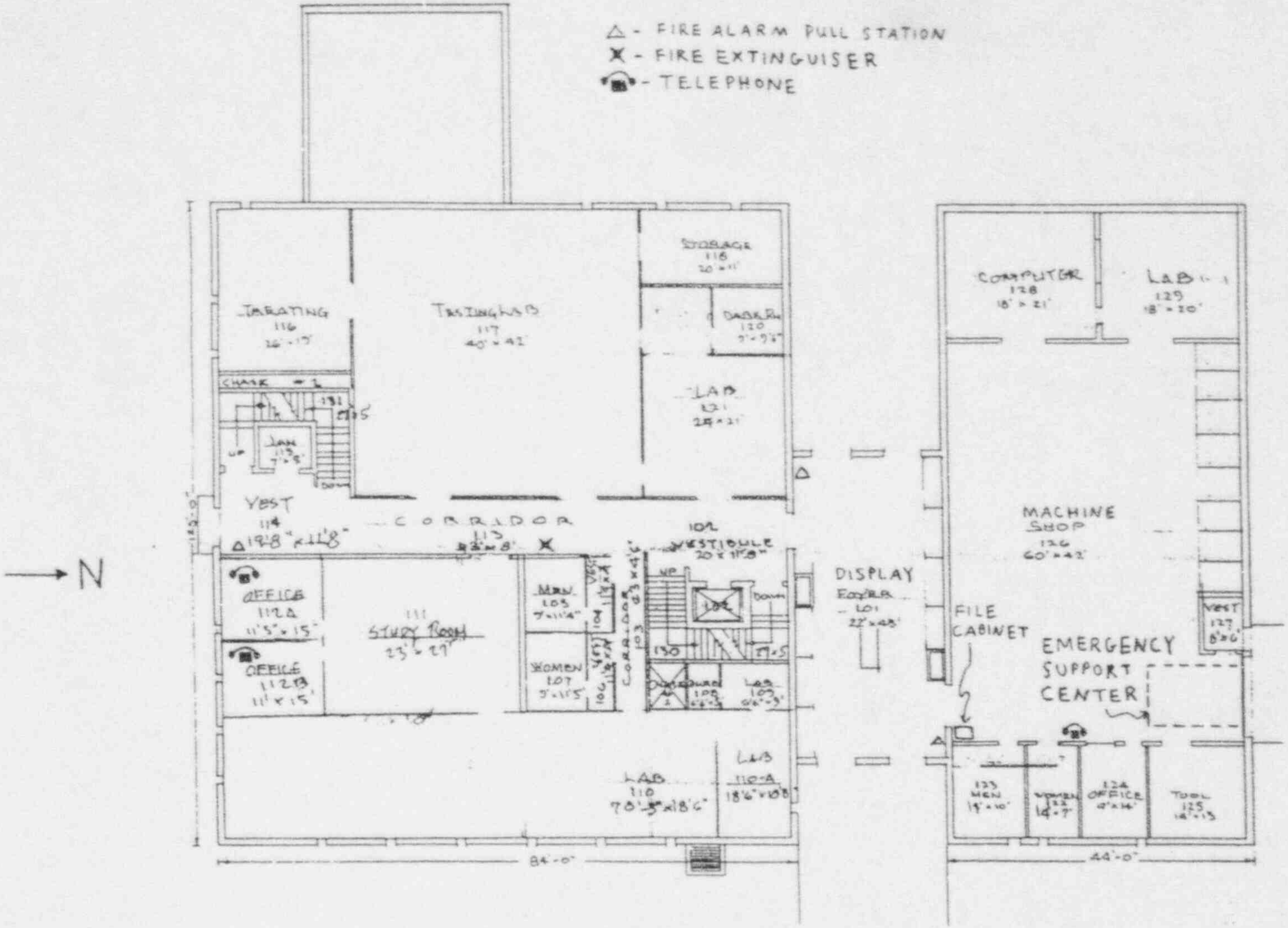


Figure 2 - Second (ground) level of LEL floor plan

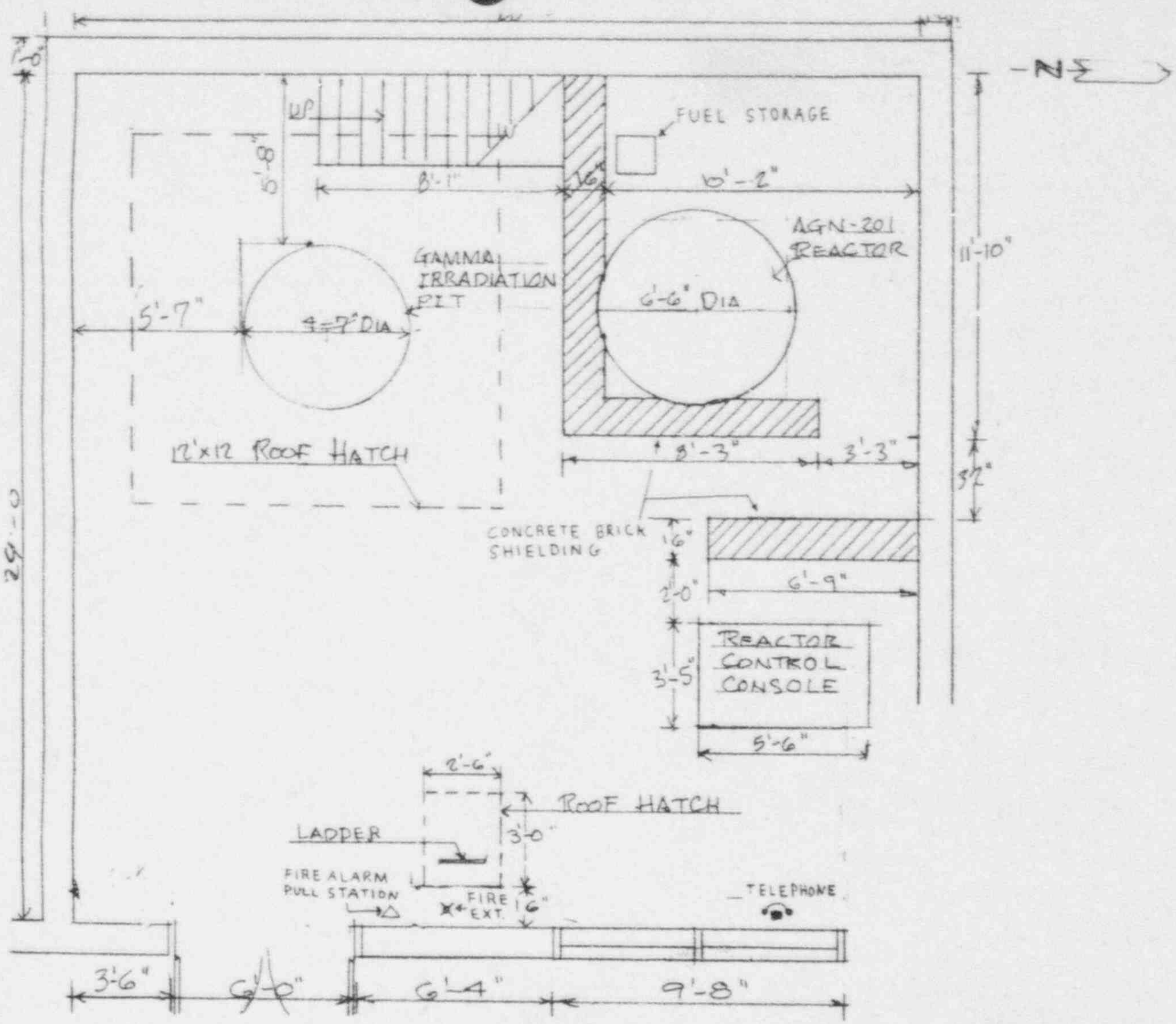


Figure 3 - Reactor laboratory floor plan (basement)

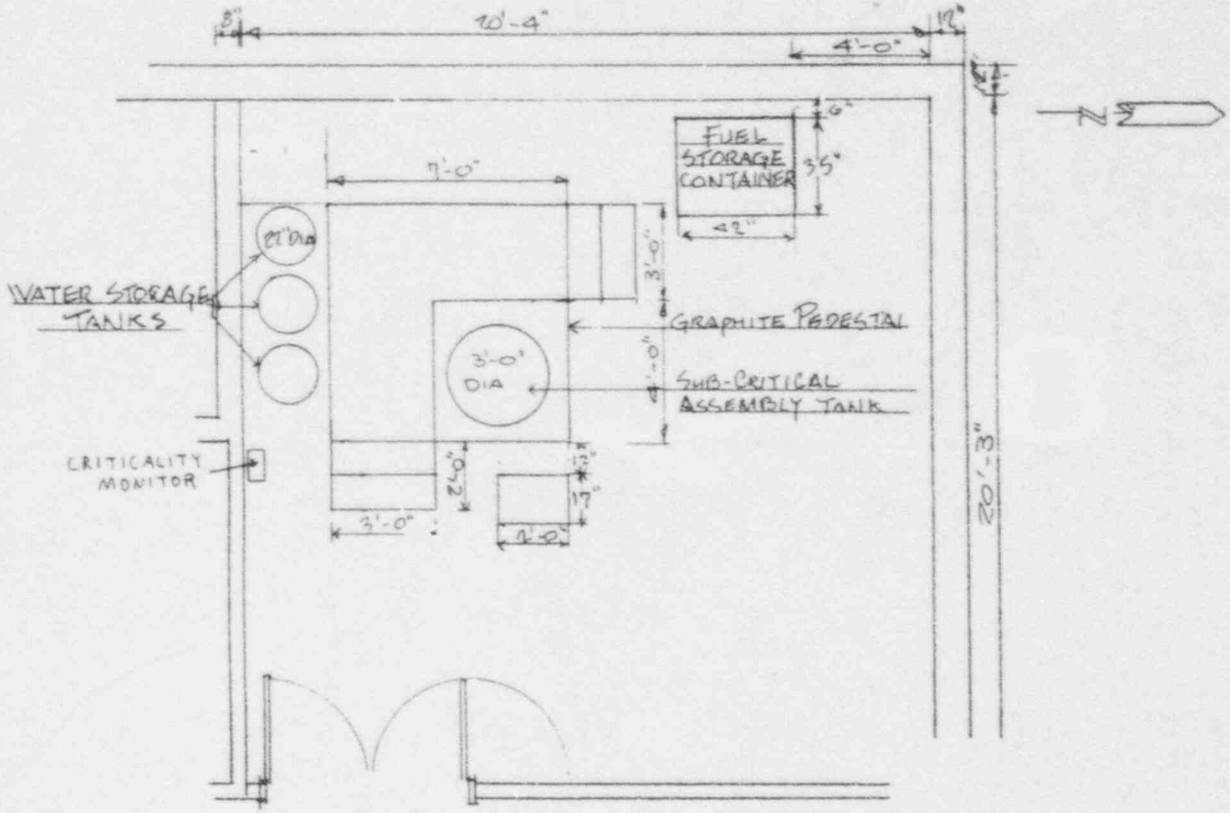


Figure 4 - Sub-critical assembly laboratory floor plan (basement)

APPLICATION FOR RENEWAL OF CLASS 104 FACILITY LICENSE R-110 FOR OPERATION OF THE AGN-201 REACTOR

APPENDIX B

TECHNICAL SPECIFICATIONS FOR AGN-201 M REACTOR, SERIAL NO. 103

BY:
IDAHO STATE UNIVERSITY
POCATELLO, IDAHO

NOVEMBER 23, 1995

APPENDIX A

TO FACILITY OPERATING

LICENSE NO. 4-110

TECHNICAL SPECIFICATIONS

FOR

IDAHO STATE UNIVERSITY AGN-201 M REACTOR (SERIAL #103)

DOCKET NO. 50-284

Amendment No. 4
Date: April 22, 1988

1.0 DEFINITIONS

The terms Safety Limit (SL), Limiting Safety System Setting (LSSS), and Limiting Conditions for Operation (LCO) are as defined in 50.36 of 10 CFR Part 50.

1.1 Channel Calibration - A channel calibration is an adjustment of the channel such that its output responds, within acceptable range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including equipment, actuation, alarm, or trip.

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

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

1.4 Experiment -

a. An experiment is any of the following:

(1) An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;

(2) An evaluation or test of a reactor system operational, surveillance, or maintenance technique; or

(3) The material content of any of the foregoing, including structural components, encapsulation or confining boundaries, and contained fluids or solids.

b. Secured Experiment - Any experiment, or component of an experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means. The restraint shall exert sufficient force on the experiment to overcome the expected effects of hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment or which might arise as a result of credible malfunctions.

c. Unsecured Experiment - Any experiment, or component of an experiment is deemed to be unsecured whenever it is not secured as defined in 1.4.b above. Moving parts of experiments are deemed to be unsecured when they are in motion.

d. Movable Experiment - A movable experiment is one which may be inserted, removed or manipulated while the reactor is critical.

e. Removal Experiment - A removable experiment is any experiment, experimental facility, or component of an experiment, other than a permanently attached appurtenance to the reactor system., which can reasonably be anticipated to be moved one or more times during the life of the reactor.

1.5 Experimental Facilities - Experimental facilities are those portions of the reactor assembly that are used for the introduction of experiments into or adjacent to the reactor core region or allow beams of radiation to exit from the reactor shielding. Experimental facilities shall include the thermal column, glory hole, and access ports.

1.6 Explosive Material - Explosive material is any solid or liquid which is categorized as a Severe, Dangerous, or Very Dangerous Explosion Hazard in "Dangerous Properties of Industrial Materials" by N. I. Sax, Third Ed., (1968), or is given an Identification of Reactivity (Stability) index of 2, 3, or 4 by the National Fire Protection Association in its publication 704-M, 1966, "Identification System for Fire Hazards of Materials," also enumerated in the "Handbook for Laboratory Safety," 2nd Ed. (1971) published by the Chemical Rubber Company.

1.7 Measuring Channel - A measuring channel is the combination of sensor, lines, amplifiers, and output devices which are connected for the purpose of measuring or responding to the value of a process variable.

1.8 Operable - Operable means a component or system is capable of performing its intended function in its normal manner.

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

1.10 Potential Reactivity Worth - The potential reactivity worth of an experiment is the maximum absolute value of the reactivity change that would occur as a result of intended or anticipated changes or credible malfunctions that alter experiment position or configuration.

Evaluations of potential reactivity worth of experiments also shall include effects of possible trajectories of the experiment in motion relative to the reactor, its orientation along each trajectory, and circumstances which can cause internal changes such as creating or filling of void spaces or motion of mechanical components. For removable experiments, the potential reactivity worth is equal to or greater than the static reactivity worth.

1.11 Reactor Component - A reactor component is any apparatus, device, or material that is a normal part of the reactor assembly.

1.12 Reactor Operation - Reactor operation is any condition wherein the reactor is not shutdown.

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

1.14 Reactor Shutdown - The reactor shall be considered shutdown whenever:

a. Either: 1. all safety and control rods are fully withdrawn from the core, or

2. the core fuse melts resulting in separation of the core,

and:

b. The reactor console key switch is in the "off" position and the key is removed from the console and under the control of a licensed operator.

1.15 Safety Channel - A safety channel is a measuring channel in the reactor safety system.

1.16 Static Reactivity Worth - The static reactivity worth of an experiment is the value of the reactivity change which is measurable by calibrated control or regulating rod comparison methods between two defined terminal positions or configurations of the experiment. For removable experiments, the terminal positions are fully removed from the reactor and fully inserted or installed in the normal functioning or intended position.

2.0 SAFETY LIMITS AND LIMITED SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the maximum steady state power level and maximum core temperature during steady state or transient operation.

Objective

To assure that the integrity of the fuel material is maintained and all fission products are retained in the core matrix.

Specification

a. The reactor power level shall not exceed 100 watts.

b. The maximum core temperature shall not exceed 200°C during either steady state or transient operation.

Basis

The polyethylene core material does not melt below 200°C and is expected to maintain its integrity and retain essentially all of the fission products at temperatures below 200°C. The Hazards Summary Report dated February 1962 submitted on Docket F-15 by Aerojet-General Nucleonics (AGN) calculated a steady state core average temperature rise of 0.44C/watt. Therefore, a steady state power level of 100 watts would result in an average core temperature rise of 44°C. The corresponding maximum core temperature would be below 200°C thus assuring integrity of the core and retention of fission products.

2.2 Limiting Safety System Settings

Applicability

This specification applies to the parts of the reactor safety system which will limit maximum power and core temperature.

Objective

To assure that automatic protective action is initiated to prevent a safety limit from being exceeded.

Specification

a. The safety channels shall initiate a reactor scram at the following limiting safety system settings:

Channel	Condition	LSSS
Nuclear Safety #2	High Power	$\leq$ 10 watts
Nuclear Safety #3	High Power	$\leq$ 10 watts

b. The core thermal fuse shall melt when heated to a temperature of about 120°C resulting in core separation and reactivity loss greater than 5% $\Delta k/k$.

Basis

Based on instrumentation response times and scram tests, the AGN Hazards Report concluded that reactor periods in excess of 30-50 milliseconds would be adequately arrested by the scram system. Since the maximum available excess reactivity in the reactor is less than one dollar the reactor cannot become prompt critical and the corresponding shortest possible period is greater than 200 milliseconds. The high power LSSS of 10 watts in conjunction with automatic safety systems and/or manual scram capabilities will assure that the safety limits will not be exceeded during steady state or as a result of the most severe credible transient.

In the event of failure of the reactor to scram, the self-limiting characteristic due to the high negative temperature coefficient, and the melting of the thermal fuse at a temperature below 120°C will assure safe shutdown without exceeding a core temperature of 200°C.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 Reactivity Limits

Applicability

This specification applies to the reactivity condition of the reactor and the reactivity worths of control rods and experiments.

Objective

To assure that the reactor can be shut down at all times and that the safety limits will not be exceeded.

Specification

a. The available excess reactivity with all control and safety rods fully inserted and including the potential reactivity worth of all experiments shall not exceed 0.65% $\Delta k/k$ referenced to 20°C.

b. The shutdown margin with the most reactive safety or control rod fully inserted shall be at least 1% $\Delta k/k$.

c. The reactivity worth of the control and safety rods shall ensure sub-criticality on the withdrawal of the coarse control rod or any one safety rod.

Basis

The limitations on total core excess reactivity assure reactor periods of sufficient length so that the reactor protection system and/or operator action will be able to shut the reactor down without exceeding any safety limits. The shutdown margin and control and safety rod reactivity limitations assure that the reactor can be brought and maintained subcritical if the highest reactivity rod fails to scram and remains in its most reactive position.

3.2 Control and Safety Systems

Applicability

These specifications apply to the reactor control and safety systems.

Objective

To specify lowest acceptable level of performance, instrument set points, and the minimum number of operable components for the reactor control and safety systems.

TABLE 3.1

SAFETY CHANNEL	SET POINT	FUNCTION
Nuclear Safety #1 Low Power	5% Full Scale	Scram at source levels < 5% of full scale
Nuclear Safety #2 High Power	10 Watt	Scram at power > 10 watt
Low Power	3.0×10^{-13} amps	Scram at source levels < 3×10^{-13} amps
Reactor Period	5 sec	Scram at periods < 5 sec
Nuclear Safety #3 (Linear Power)		
High Power	10 Watt	Scram at power > 10 watt
Low Power	5% full scale	Scram at source levels < 5% of full scale
Manual Scram	-----	Scram at operator option
Radiation Monitor	-----	Alarm at or below level set to meet requirements of 10 CFR Part 20

Specification

a. The total scram withdrawal time of the safety rods and coarse control rod shall be less than 200 milliseconds.

b. The average reactivity addition rate for each control or safety rod shall not exceed 0.065% $\Delta k/k$ per second.

c. The safety rods and coarse control rod shall be interlocked such that:

   1. Reactor startup cannot commence unless both safety rods and coarse control rod are fully withdrawn from the core.

   2. Only one safety rod can be inserted at a time.

   3. The coarse control rod cannot be inserted unless both safety rods are fully inserted.

d. All reactor safety system instrumentation shall be operable in accordance with Table 3.1 with the exception that Safety Channels 1 or 3 may be bypassed whenever the reactor control or safety rods are not in their fully withdrawn position.

e. The shield water level interlock shall be set to prevent startup and scram the reactor if the shield water level falls 10 inches below the highest point on the reactor shield tank manhole opening.

f. The shield water temperature interlock shall be set to prevent reactor startup and scram the reactor if the shield water temperature falls below 15°C.

g. The seismic displacement interlock sensor shall be installed in such a manner to prevent reactor startup and scram the reactor during a seismic displacement.

h. A loss of electric power shall cause the reactor to scram.

Basis

The specifications on scram withdrawal time in conjunction with the safety system instrumentation and set points assure safe reactor shutdown during the most severe foreseeable transients. Interlocks on control and safety rods assure an orderly approach to criticality and an adequate shutdown capability. The limitations on reactivity addition rates allow only relatively slow increases of reactivity so that ample time will be available for manual or automatic scram during any operating conditions.

The neutron detector channels (nuclear safety channels 1 through 3) assure that reactor power levels are adequately monitored during reactor startup and operation. Requirements on minimum neutron levels will prevent reactor startup unless channels are operable and responding, and will cause a scram in the event of instrumentation failure. The power levels initiate redundant automatic protective action at power level scrams low enough to assure safe shutdown without exceeding any safety limits. The period scram conservatively limits the rate of rise of reactor power to periods which are manually controllable and will automatically scram the reactor in the event of unexpected large reactivity additions.

The AGN-201's negative temperature coefficient of reactivity causes a reactivity increase with decreasing core temperature. The shield water temperature interlock will prevent reactor operation at temperatures below 15°C thereby limiting potential reactivity additions associated with temperature decreases.

Water in the shield tank is an important component of the reactor shield and operation without the water may produce excessive radiation levels. The shield tank water level interlock will prevent reactor operation without adequate water levels in the shield tank.

The reactor is designed to withstand 0.6g accelerations and 6 cm displacements. A seismic instrument causes a reactor scram whenever the instrument receives a horizontal acceleration that causes a horizontal displacement of 1/16 inch or greater. The seismic displacement interlock assures that the reactor will be scrammed and brought to a subcritical configuration during any seismic disturbance that may cause damage to the reactor or its components.

The manual scram allows the operator to manually shut down the reactor if an unsafe or otherwise abnormal condition occurs that does not otherwise scram the reactor. A loss of electrical power de-energizes the safety and coarse control rod holding magnets causing a reactor scram and thus assuring safe and immediate shutdown in case of a power outage.

A radiation monitor must always be available to operating personnel to provide an indication of any abnormally high radiation levels so that appropriate action can be taken to shut the reactor down and assess the hazards to personnel.

3.3 Limitations on Experiments

Applicability

This specification applies to experiments installed in the reactor and its experimental facilities.

Objective

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

Specification

a. Experiments containing materials corrosive to reactor components or which contain liquid or gaseous, fissionable materials shall be doubly encapsulated.

b. Explosive materials shall not be inserted into experimental facilities of the reactor or stored within the confines of the reactor facility.

c. The radioactive material content, including fission products of any experiment shall be limited so that the 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 doses stated in 10 CFR Part 20 for persons occupying (1) unrestricted areas continuously for two hours starting at time of release or (2) restricted areas during the length of time required to evacuate the restricted area.

d. The radioactive material content, including fission products of any doubly encapsulated experiment shall be limited so that the complete release of all gaseous, particulate, or volatile components of the experiment shall not result in exposures in excess of 0.5 Rem whole body or 1.5 Rem thyroid to persons occupying an unrestricted area continuously for a period of two hours starting at the time of release, or exposure in excess of 5 Rem whole body or 30 Rem thyroid to persons occupying a restricted area during the length of time required to evacuate the restricted area.

Basis

These specifications are intended to reduce the likelihood of damage to reactor components and/or radioactivity releases resulting from an experiment failure and to protect operating personnel and the public from excessive radiation doses in the event of an experiment failure.

3.4 Radiation Monitoring, Control, and Shielding

Applicability

This specification applies to radiation monitoring, control, and reactor shielding required during reactor operation.

Objective

To protect facility personnel and the public from radiation exposure.

Specification

a. An operable portable and installed radiation survey instrument capable of detecting gamma radiation shall be immediately available to reactor operating personnel whenever the reactor is not shutdown.

b. The reactor room shall be considered a restricted area whenever the reactor is not shutdown.

c. The following shielding requirements shall be fulfilled during reactor operation:

   1. The reactor shield tank shall be filled with water to a height within 10 inches of the highest point on the manhole opening.

   2. The thermal column shall be filled with water or graphite except during a critical experiment (core loading) or during measurement of reactivity worth of thermal column water or graphite, or when the neutron radiography collimator is being used.

   3. The movable shield doors above the thermal column shall be maintained in a closed position whenever the reactor is operated at a power greater than 0.5 watts.

Basis

Radiation surveys performed under the supervision of a qualified health physicist have shown that the total gamma, thermal neutron, and fast neutron radiation dose rate in the reactor room, at the closest approach to the reactor outside the designated high radiation areas is less than 25 mrem/hr at reactor power levels less than 5.0 watt.

The facility shielding in conjunction with designated restricted radiation areas is designed to limit radiation doses to facility personnel and to the public to a level below 10 CFR 20 limits under operating conditions, and to a level below Criterion 19, Appendix A, 10 CFR 50 recommendations under accident conditions.

4.0 SURVEILLANCE REQUIREMENTS

Actions specified in this section are not required to be performed if during the specified surveillance period the reactor has not been brought critical or is maintained in a shutdown condition extending beyond the specified surveillance period. However, the surveillance requirements must be fulfilled prior to subsequent startup of the reactor.

4.1 Reactivity Limits

Applicability

The specification applies to the surveillance requirements for reactivity limits.

Objective

To assure that reactivity limits for Specification 3.1 are not exceeded.

Specification

a. Safety and control rod reactivity worths shall be measured annually, but at intervals not to exceed 16 months.

b. Total excess reactivity and shutdown margin shall be determined annually, but at intervals not to exceed 16 months.

c. The reactivity worthy of an experiment shall be estimated or measured, as appropriate, before or during the first startup subsequent to the experiment's insertion.

Basis

The control and safety rods are inspected and their reactivity worths measured annually to assure that no degradation or unexpected changes have occurred which could adversely affect reactor shutdown margin or total excess reactivity. The shutdown margin and total excess reactivity are determined to assure that the reactor can always be safely shutdown with one rod not functioning and that the maximum possible reactivity insertion will not result in reactor periods shorter than those than can be adequately terminated by either operator or automatic action. Based on experience with AGN reactors, significant changes in reactivity or rod worth are not expected within a 16-month period.

4.2 Control and Safety System

Applicability

This specification applies to the surveillance requirements of the reactor control and safety system.

Objective

To assure that the reactor control and safety systems are operable as required by Specification 3.2.

Specification

a. Safety and control rod scram times and average reactivity insertion rates shall be measured annually, but at intervals not to exceed 16 months.

b. Safety and control rods and drive shall be inspected for deterioration at intervals not to exceed 2 years.

c. A channel test of the following safety channels shall be performed prior to the first reactor startup of the day or prior to each operation extending more than one day.

   Nuclear Safety #1, #2, and #3
   Manual scram

d. A channel test of the seismic displacement interlock shall be performed semiannually.

e. A channel check of the following safety channels shall be performed daily whenever the reactor is in operation:

   Nuclear Safety #1, #2, and #3

f. Prior to each day's operation or prior to each operation extending more than one day, safety rods #1 and #2 shall be inserted and scrammed to verify operability.

g. The period, count rate, and power level measuring channels shall be calibrated and set points verified annually, but at intervals not to exceed 16 months.

h. The shield tank water level interlock, shield water temperature interlock, and seismic displacement safety channel shall be calibrated by perturbing the sensing element to the appropriate set point. These calibrations shall be performed annually, but at intervals not to exceed 16 months.

i. The radiation monitoring instrumentation shall be calibrated annually, but at intervals not to exceed 16 months.

Basis

The channel tests and checks required daily or before each startup will assure that the safety channels and scram functions are operable. Based on operating experience with reactors of this type, the annual scram measurements, channel calibrations, set point verifications, and inspections are of sufficient frequency to assure, with a high degree of confidence, that the safety system settings will be within acceptable drift tolerance for operation.

4.3 Reactor Structure

Applicability

This specification applies to surveillance requirements for reactor components other than control and safety rods.

Objective

To assure integrity of the reactor structures.

Specification

a. The shield tank shall be visually inspected every two years. If apparent excessive corrosion or other damage is observed, corrective measures shall be taken prior to subsequent reactor operation.

b. Visual inspection for water leakage from the shield tank shall be performed annually. Leakage shall be corrected prior to subsequent reactor operation.

Basis

Based on experience with reactors of this type, the frequency of inspection and leak test requirements of the shield tank will assure capability for radiation protection during reactor operation.

4.4 Radiation Monitoring and Control

Applicability

This specification applies to the surveillance requirements of the radiation monitoring and control systems.

Objective

To assure that the radiation monitoring and control systems are operable and that all radiation areas within the reactor facility are identified and controlled as required by Specification 3.4.

Specification

a. All portable and installed radiation survey instruments assigned to the reactor facility shall be calibrated under the supervision of the Radiation Safety Officer annually, but at intervals not to exceed 16 months.

b. Prior to each day's reactor operation or prior to each reactor operation extending more than one day, the reactor room high radiation alarm shall be verified to be operable.

c. A radiation survey of the reactor room and reactor control room shall be performed under the supervision of the Radiation Safety Officer annually, but at intervals not to exceed 16 months, to determine the location of radiation and high radiation areas corresponding to reactor operating power levels.

<u>Basis</u>

The periodic calibration of radiation monitoring equipment and the surveillance of the reactor room high radiation area alarm will assure that the radiation monitoring and control systems are operable during reactor operation.

The periodic radiation surveys will verify the location of radiation and high radiation areas and will assist reactor facility personnel in properly labeling and controlling each location in accordance with 10 CFR 20.

5.0 DESIGN FEATURES

5.1 Reactor

a. The reactor core, including control and safety rods, contains approximately 670 grams of U^{235} in the form of 20% enriched UO_2 dispersed in approximately 11 kilograms of polyethylene. The lower section of the core is supported by an aluminum rod hanging from a fuse link. The fuse melts at temperatures below 120°C causing the lower core section to fall away from the super section reducing reactivity by at least 5% $\Delta k/k$. Sufficient clearance between core and reflector is provided to insure free fall of the bottom half of the core during the most severe transient.

b. The core is surrounded by a 20 cm thick high density (1.75 gm/cm3) graphite reflector followed by a 10 cm thick lead gamma shield. The core and part of the graphite reflector are sealed in a fluid-tight aluminum core tank designed to contain any fission gases that might leak from the core.

c. The core, reflector, and lead shielding are enclosed in and supported by a fluid-tight steel reactor tank. An upper of "thermal column tank" may serve as a shield tank when filled with water or a thermal column when filled with graphite.

d. The 6-1/2 foot diameter, fluid-tight shield tank is filled with water constituting a 55 cm thick fast neutron shield. The fast neutron shield is formed by filling the tank with approximately 1000 gallons of water. The complete reactor shield shall limit doses to operating personnel in restricted and unrestricted areas to levels less than permitted by 10 CFR 20 under operating conditions.

e. Shielding is provided by a concrete wall constructed of 4" X 8" X 16" concrete blocks and 4" X 8" X 12" barytes concrete blocks for 5 watt operation. The blocks are held to close dimensional tolerance in manufacture and stacked in such a manner that voids in the completed wall are at a minimum. Near the beam ports and glory hole, high density blocks are used between 40 inches and 112 inches above the base. The use of these blocks further reduces radiation level in these areas. Overhead shielding is provided by 8 inch thick barytes blocks (minimum density 3.7 gcc).

As detailed in the amendment for 5 watt operation for Aerojet-General Nucleonics, dated 11 February 1957, and on file with the Commission in Docket 50-32, an 18 inch additional concrete shield wall was sufficient to maintain sub-tolerance radiation levels external to the shield when operating at 5 watts. Subsequent analysis by Aerojet-General Nucleonics indicated that 16 inches of ordinary concrete shielding was sufficient. Twelve inches of barytes concrete is more effective than 18 inches of ordinary concrete.

The radiation levels associated with 5 watt operation (peak thermal flux of 2.5×10^8 n/cm^2 -sec) have been calculated by Aerojet-General Nucleonics. The table below gives Aerojet-General results for gamma dose for an 18" concrete shield:

Energy (MeV)	u	ux	B	e^{-ux}	Dose (mrem/hr)
3.0	0.081	3.64	1.9	0.027	1.5
2.2	0.105	4.73	2.2	0.0088	1.0
				Total =	2.5 mrem/hr (18" concrete)

Changing the e^{-ux} for a 16" concrete wall and using the same buildup factors yields the following table:

Energy (MeV)	u	ux	B	e^{-ux}	Dose (mrem/hr)
3.0	0.081	3.30	1.9	0.037	2.1
2.2	0.105	4.28	2.2	0.014	1.2
				Total =	3.3 mrem/hr (16" concrete)

The National Naval Medical Center reported neutron dose rates were less than 0.2 mrem/hr for 18" shield.

Although calculations and operating surveys of similar facilities show that at 5 watts power there will be no area on the reactor floor outside the concrete shield where the total radiation exceeds tolerance levels; nevertheless, the reactor floor is a control area with restricted access.

f. Two safety rods and one control rod (identical in size) contain up to 20 grams of U^{235} each in the same form as the core material. These rods are lifted into the core by electromagnets, driven by reversible DC motors through lead screw assemblies. De-energizing the magnets causes a spring-driven, gravity-assisted scram. The fourth rod or fine control rod (approximately one-half the diameter of the other rods) is driven directly by a lead screw. This rod may contain fueled or unfueled polyethylene.

5.2 Fuel Storage

Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in locked rooms in the College of Engineering laboratories. The storage array shall be such that k_{eff} is no greater than 0.8 for all conditions of moderation and reflection.

5.3 Reactor Room

a. The reactor room houses the reactor assembly and accessories required for its operation and maintenance.

b. The reactor room is a separate room in the Lillibridge Engineering Laboratory, constructed with adequate shielding and other radiation protective features to limit doses in restricted and unrestricted areas to levels no greater than permitted by 10 CFR 20, under normal operating conditions, and to a level below Criterion 19, Appendix A, 10 CFR 50 recommendations under accident conditions.

c. Access doors to the reactor room are self-locking.

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

The administrative organization for control of the reactor facility and its operation shall be as set forth in Figure 1 of these specifications. The authorities and responsibilities set forth below are designed to comply with the intent and requirements for administrative controls of the reactor facility as set forth by the Nuclear Regulatory Commission.

6.1.1 University Officer

The University Officer is an administrative officer responsible for the University and in whose name the application for licensing is made.

6.1.2 Dean, College of Engineering

The Dean of the College of Engineering is the administrative officer responsible for the operation of the College of Engineering.

6.1.3 Reactor Administrator

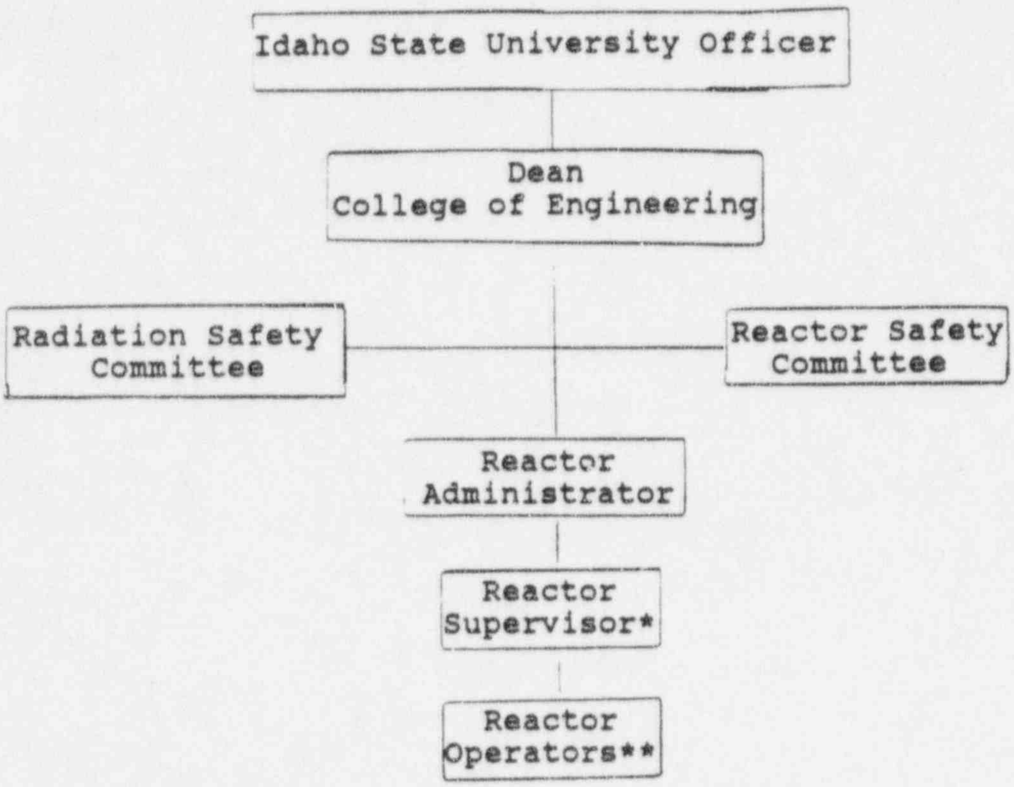
The Reactor Administrator is the administrative officer responsible for the operation of the AGN-201M Reactor Facility. In this capacity he shall have final authority and ultimate responsibility for the operation, maintenance, and safety of the reactor facility within the limitations set forth in the facility license. He shall be responsible for appointing personnel to all positions reporting to him as described in Section 6.1 of the Technical Specifications. He shall seek the advice and approval of the Radiation Safety Committee and/or the Reactor Safety Committee in all matters concerning unresolved safety questions, new experiments and new procedures, and facility modifications which might affect safety. He shall be an _ex officio_ member of the Reactor Safety Committee.

6.1.4 Reactor Supervisor

The Reactor Supervisor shall be responsible for the preparation, promulgation, and enforcement of administrative controls including all rules, regulations, instructions, and operating procedures to ensure that the reactor facility is operated in a safe, competent, and authorized manner at all times. He shall direct the activities of operators and technicians in the daily operation and maintenance of the reactor; schedule reactor operations and maintenance; be responsible for the preparation, authentication, and storage of all prescribed logs and operating records; authorize all experiments, procedures, and changes thereto which have received the approval of the Reactor Safety Committee and/or the Radiation Safety Committee and the Reactor Administrator; and be responsible for the preparation of experimental procedures involving use of the reactor.

Figure 1

Administrative Organization of the Idaho State University AGN-201M Reactor Facility NRC License R-110



* Requires NRC Senior Operators License

** Requires NRC Operators License except where exempt per 10 CFR 55 paragraph 55.9

Persons holding positions on the Administrative Organization shall meet or exceed the qualification requirements of ANS1/ANS -15.4-1977 (N380), "Selection and Training of Personnel for Research Reactors."

6.1.5 Reactor Operators

Reactor Operators shall be responsible for the manipulation of the reactor controls, monitoring of instrumentation, operation of reactor related equipment, and maintenance of complete and current records during operation of the facility. Reactor Operators who are exempt from holding an NRC license per 10 CFR 55 paragraph 55.9 shall only operate the reactor under the direct and immediate supervision of a licensed Reactor Operator.

6.1.6 Reactor Safety Committee

The Reactor Safety Committee shall be responsible for, but not limited to, reviewing and approving safety standards associated with the use of the reactor facility; reviewing and approving all proposed experiments and procedures and changes thereto; reviewing and approving all modifications to the reactor facility which might affect its safe operation; determining whether proposed experiments, procedures, or modifications involve unreviewed safety questions, as defined in 10 CFR 50 paragraph 50.59(c), and are in accordance with these Technical Specifications; conducting periodic audits of procedures, reactor operations and maintenance, equipment performance, and records; review all reportable occurrences and violations of these Technical Specifications, evaluating the causes of such events and the corrective action taken and recommending measures to prevent reoccurrence; reporting all their findings and recommendations concerning the reactor facility to the Reactor Administrator.

6.1.7 Radiation Safety Committee

The Radiation Safety Committee shall advise the University administration and the Radiation Safety Officer on all matters concerning radiological safety at University facilities.

6.1.8 Radiation Safety Officer

The Radiation Safety Officer shall review and approve all procedures and experiments involving radiological safety. He shall enforce all federal, state, and university rules, regulations, and procedures relating to radiological safety. He shall perform routine radiation surveys of the reactor facility and report his findings to the Reactor Administrator. He shall provide personnel dosimetry and keep records of personnel radiation exposure. He shall advise the Reactor Administrator on all matters concerning radiological safety at the reactor facility. The Radiation Safety Officer shall be an *ex officio* member of the Reactor Safety Committee.

6.1.9 Operating Staff

a. The minimum operating staff during any time in which the reactor is not shutdown consist of:

   1. One licensed Reactor Operator in the reactor control room.

   2. One other person in the reactor room or reactor control room certified by the Reactor Supervisor as qualified to activate manual scram and initiate emergency procedures.

   3. One licensed Senior Reactor Operator readily available on call. This requirement can be satisfied by having a licensed Senior Reactor Operator perform the duties stated in paragraph 1 or 2 above or by designating a licensed Senior Reactor Operator who can be readily contacted by telephone and who can arrive at the reactor facility within 30 minutes.

b. A licensed Senior Reactor Operator shall supervise all reactor maintenance or modification which could affect the activity of the reactor.

6.2 Staff Qualifications

The Reactor Administrator, the Reactor Supervisor, licensed Reactor Operators, and technicians performing reactor maintenance shall meet the minimum qualifications set forth in ANS 15.4, "Standards for Selection and Training of Personnel for Research Reactors." Reactor Safety Committee members shall have a minimum of five (5) years experience in their profession or a baccalaureate degree and two (2) years of professional experience. Generally, these committee members will be made up of University faculty, but outside experience may be sought in areas where additional experience is considered necessary by the Reactor Administrator.

6.3 Training

The Reactor Administrator shall be responsible for directing training as set forth ANS 15.4, "Standards for Selection and Training of Personnel for Research Reactors." All licensed reactor operators shall participate in requalification training as set forth in 10 CFR 55.

6.4 Reactor Safety Committee

6.4.1 Meetings and Quorum

The Reactor Safety Committee shall meet as often as deemed necessary by the Reactor Safety Committee Chariman but shall meet at least once each calendar year. A quorum for the conduct of official business shall be the chairman, or his designated alternate, and two (2) other regular members. At no time shall the operating organization comprise a voting majority of the members at any Reactor Safety Committee meeting.

6.4.2 Reviews

The Reactor Safety Committee shall review:

a. Safety evaluations for changes to procedures, equipment or systems, and tests or experiments, conducted without Nuclear Regulatory Commission approval under the provision of 10 CFR 50 paragraph 50.59, to verify that such actions do not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems that change the original intent or use, and are non-conservative, or those that involve an unreviewed safety question as defined in 10 CFR 50 paragraph 50.59.

c. Proposed tests or experiments which are significantly different from previously approved tests or experiments, or those that involve an unreviewed safety question as defined in Section 50.59, 10 CFR 50 paragraph 50.59.

d. Proposed changes in Technical Specifications or licenses.

e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of facility equipment that affect nuclear safety.

g. Reportable occurrences.

h. Audit reports.

6.4.3 Audits

Audits of facility activities shall be performed under the cognizance of the Reactor Safety Committee but in no case by the personnel responsible for the item audited. These audits shall examine the operating records and encompass but shall not be limited to the following:

a. The conformance of facility operation to the Technical Specifications and applicable license conditions, at least once per 12 months.

b. The performance, training, and qualifications of the entire facility staff, at least once per 24 months.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety, at least once per calendar year.

d. The Facility Emergency Plan and implementing procedures at least once per 24 months.

e. The Facility Security Plan and implementing procedures, at least once per 24 months.

6.4.4 <u>Authority</u>

The Reactor Safety Committee shall report to the University officer and shall advise the Reactor Administrator on those areas of responsibility outlined in Section 6.1.6 of these Technical Specifications.

6.4.5 <u>Minutes of the Reactor Safety Committee</u>

The Chairman of the Reactor Safety Committee shall direct the preparation, maintenance, and distribution of minutes of its activities. These minutes shall include a summary of all meetings, actions taken, audits, and reviews.

6.5 <u>Approvals</u>

The procedure for obtaining approval for any change, modification or procedure which requires approval of the Reactor Safety Committee shall be as follows:

a. The Reactor Supervisor shall prepare the proposal for review and approval by the Reactor Administrator.

b. The Reactor Administrator shall submit the proposal to the Chairman of the Reactor Safety Committee.

c. The Chairman of the Reactor Safety Committee shall submit the proposal to the Reactor Safety Committee members for review and comment.

d. The Reactor Safety Committee can approve the proposal by majority vote.

6.6 <u>Procedures</u>

There shall be written procedures that cover the following activities:

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

b. Fuel movement and changes to the core and experiments that could affect reactivity.

c. Conduct of irradiations and experiments that could affect the safety of the reactor.

d. Preventive or corrective maintenance which could affect the safety of the reactor.

e. Surveillance, testing, and calibration of instruments, components and systems as specified in Section 4.0 of these Technical Specifications.

f. Implementation of the Security Plan and Emergency Plan.

The above listed procedures shall be approved by the Reactor Administrator and the Reactor Safety Committee. Temporary procedures which do not change the intent of previously approved procedures and which do not involve any unreviewed safety question may be employed on approval by the Reactor Supervisor.

6.7 Experiments

a. Prior to initiating any new reactor experiment an experimental procedure shall be prepared by the Reactor Supervisor and reviewed and approved by the Reactor Administrator and the Reactor Safety Committee.

b. Approved experiments shall only be performed under the cognizance of the Reactor Supervisor.

6.8 Safety Limit Violation

The following actions shall be taken in the event a Safety Limit is violated:

a. The reactor will be shutdown immediately and reactor operation will not be resumed without authorization by the Nuclear Regulatory Commission (NRC).

b. The Safety Limit violation shall be reported to the appropriate NRC Regional Office of Inspection and Enforcement, the Director of the NRC, and the Reactor Safety Committee not later than the next working day.

c. A Safety Limit Violation Report shall be prepared for review by the Reactor Safety Committee. This report shall describe the applicable circumstances preceding the violation, the effects of the violation upon facility components, systems or structures, and corrective action to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the NRC, and Reactor Safety Committee within 14 days of the violation.

6.9 Reporting Requirements

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the appropriate NRC Regional Office.

6.9.1 Annual Operating Report

Routine operating reports covering the operation of the unit during the previous calendar year should be submitted prior to June 30 of each year.

The annual operating reports made by licensees shall provide a comprehensive summary of the operating experience having safety significance that was gained during the year, even though some repetition of previously reported information may be involved. References in the annual operating report to previously submitted reports shall be clear.

Each annual operating report shall include:

(1) A brief narrative summary of:

a. Changes in facility design, performance characteristics, and operating procedures relating to reactor ssafety that occured during the reporting period.

b. Results of major surveillance tests and inspections.

(2) A monthly tabulations showing the hours the reactor is operating.

(3) List of the unscheduled shutdowns, including the reasons therefore and corrective action taken, if any.

(4) Discussion of the major safety related corrective maintenance performed during the period, including the effects, if any, on the safe operation of the reactor and the reasons for the corrective maintenance required.

(5) A brief description of :

a. Each change to the facility to the extent that it changes a description of the facility in the application for license and amendments thereto.

b. Changes to the procedures as described in Facility Technical Specifications.

c. Any new or untried experiments or tests performed during the reporting period.

(6) A summary of the safety evaluation made for each change, test, or experiment not submitted for NRC approval pursuant to 10 CFR 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety question existed and that no change to the Technical Specifications was required.

(7) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as determined at or prior to the point of such release or discharge.

a. Liquid waste-

Total estimated quantity of radioactivity released (in curies) and total volume (in liters) of effluent water (including diluent) released.

b. Airborne waste-

Total estimated quantity of radioactivity released (in curies) determined by an approved sampling and counting method.

c. Solid waste-

(i) Total amount of solid waste packaged (in cubic meters)

(ii) Total activity in solid waste (in curies)

(iii) The dates of shipments and disposition if shipped off site)

(8) A description of the results of any environmental radiological surveys performed outside the facility.

(9) Radiation Exposure - A summary of radiation exposures greater than 100 mrem (50 mrem for persons under 18 years of age) received during the reporting period by facility personnel or visitors.

6.9.2 Reportable Occurrences

Reportable occurrences, including causes, probable consequences, corrective actions and measures to prevent recurrence, shall be reported to the NRC.

a. Prompt Notification with Written Followup-

The types of events listed shall be reported as expeditiously as possible by telephone and telegraph to the Director of the appropriate NRC Regional Office, or his designated representative no later than the first working day following the event, with a written followup report within two weeks. Information provided shall contain narrative material to provide complete explanation of the circumstances surrounding the event.

(1) Failure of the reactor protection system subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reached the setpoint specified as the limiting safety system setting in the technical specifications.

(2) Operation of the reactor when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the technical specifications.

(3) Abnormal degradation discovered in a fission product barrier.

(4) Reactivity balance anomalies involving:

a. disagreement between expected and actual critical positions of approximately 0.3% Δ k/k;

b. exceeding excess reactivity limits;

c. shutdown margin less conservative than specified in technical specifications.

(5) Failure or malfunction of one (or more) component(s) which prevents, or could prevent, by itself, the fulfillment of the functional requirements of systems used to cope with accidents analyzed in Safety Analysis Report.

(6) Personnel error or procedural inadequacy which prevents, or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in Safety Analysis Report.

(7) Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the Safety Analysis Report or in the basis for the Technical Specifications that have permitted reactor operation in a manner less conservative than assumed in the analyses.

(8) Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the Safety Analyses Report or Technical Specification basis, or discovery during plant life of conditions not specifically considered in the Safety Analysis Report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

6.10 <u>Record Retention</u>

6.10.1 Records to be retained for a period of at least five years:

a. Operating logs or data which shall identify:

(1) Completion of pre-startup checkout, startup, power changes, and shutdown of the reactor.

(2) Installation or removal of fuel elements, control rods or experiments that could affect core reactivity.

(3) Installation or removal of jumpers, special tags or notices, or other temporary changes to reactor safety circuitry.

(4) Rod worth measurements and other reactivity measurements.

b. Principal maintenance operations.

c. Reportable occurrences.

d. Surveillance activities required by technical specifications.

e. Facility radiation and contamination surveys.

f. Experiments performed with the reactor.

This requirements may be satisfied by the normal operations log book plus:

1. Records of radioactive material transferred from the facility as required by license.

2. Records required by the Reactor Safety Committee for the performance of new or special experiments.

g. Changes to operating procedures.

6.10.2 Records to be retained for the life of the facility:

a. Gaseous and liquid radioactive effluents released to the environs.

b. Appropriate offsite environmental monitoring surveys.

c. Fuel inventories and fuel transfers.

d. Radiation exposures for all personnel.

e. Updated as-built drawings of the facility.

f. Records of transient or operational cycles for those components designed for a limited number of transients or cycles.

g. Records of training and qualification for members of the facility staff.

h. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

i. Records of meetings of the Reactor Safety Committee.

APPLICATION FOR RENEWAL OF CLASS 104
FACILITY LICENSE R-110 FOR OPERATION
OF THE AGN-201 REACTOR

APPENDIX C

REACTOR OPERATOR REQUALIFICATION PROGRAM

BY:
IDAHO STATE UNIVERSITY
POCATELLO,IDAHO

NOVEMBER 23, 1995

REACTOR OPERATOR REQUALIFICATION PROGRAM

FOR THE IDAHO STATE UNIVERSITY

REACTOR

Idaho State University
College of Engineering
P.O. Box 8060
Pocatello, Idaho 83209

Rev. 2, 08/17/95

REACTOR OPERATOR
REQUALIFICATION PLAN
CONTENTS

1.0 PURPOSE

This document sets forth the requirements for the Reactor Operator and Senior Reactor Operator Requalification Program for Idaho State University's (ISU) AGN-201 Nuclear Reactor (Docket #50-284) in accordance with Title 10 of the Federal Code of Regulations Part 55.59 (10CFR55.59). The purpose of the requalification training program is to ensure that all operations personnel maintain proficiency at a level equal to or greater than that required for initial licensing.

2.0 SCHEDULE

A complete requalification training program shall be offered biennially. The program consists of lectures, on the job training, and written, oral, and console evaluations. The classroom retraining includes eight different lectures to be offered at least once during the biennium. The evaluations shall be conducted annually. Each operator shall be required to perform licensed functions for at least four hours during each quarter. The performance of licensed functions entails:

1) Performance of corrective maintenance.
2) Performance of preventive maintenance or surveillance.
3) Radiological work under the reactor license.
4) Making preparations to the facility to perform an experiment with the reactor.
5) Securing from an experiment with the reactor.
6) Reactor console run time.
7) Administering reactor console exams to senior reactor and reactor operators.

Each operator licensee shall complete the program biennially. The licensee shall enter the requalification program on the date the Nuclear Regulatory Commission issues a new license. The licensee shall continue in the requalification program until either the expiration date of the current license or the date at which the current license is terminated.

3.0. LECTURE

The program shall include coverage of the following eight topics which shall be offered at least once during the requalification training period:

	Topic	Reference
A.	Nuclear Reactor Theory	Standard Nuclear Engineering Text
B.	Radiation Control and Safety	10CFR Parts 20 and 30, ISU Radiation Safety Manual

3.0 LECTURE (Continued)

	Topic	Reference
C.	Governing Regulations	10CFR Parts 19, 50, 55, and 70
D.	Reactor Design	Reactor Facility Study Material
E.	Reactor Control and Safety Systems	Reactor Facility Study Material
F.	Reactor Operating Characteristics	Reactor Facility Study Material
G.	All Reactor Facility Procedures, Policies, and Rules	Operating Procedures, Emergency, Plans, Physical Security Plan, Maintenance and Surveillance Procedures, etc.
H.	Technical Specifications and License Conditions	Technical Specifications and License Conditions

Each lecture shall include a brief review of the last Reactor Safety Committee Meeting Minutes with an emphasis on approved changes to the reactor facility procedures. All of the maintenance and surveillance procedure entries since the last lecture shall also be reviewed. The frequent updates shall ensure that the operators are current on all reactor facility activities. Any operator may be assigned to present a lecture.

4.0 ON THE JOB TRAINING

Each operator shall perform licensed functions for at least four hours per quarter to satisfy 10CFR55.53(e). Each operator shall demonstrate familiarity with the following activities at least once during the biennial period;

1. prestart checks,
2. startup, and
3. termination.

This training shall be evaluated by any licensed operator. As a minimum, to demonstrate proficiency at manipulating the reactor facility controls, each operator will perform at least one complete Operating Procedure #1 (O.P. #1) Startup and Shutdown per quarter, provided that reactor operation is possible. Senior reactor operators may not take credit for their required O.P. #1 Startup and Shutdown per quarter by directing another operator in reactor facility manipulations. Operators may not take credit for their one required O.P. #1 Startup and Shutdown per quarter by directing an operator for the purposes of reinstatement.

5.0 EVALUATIONS

The ability of the operator to perform licensed functions shall be determined through evaluations which shall be conducted annually. These evaluations shall include written and console examinations. These examinations may be administered in any order, at any time during the year, and on different dates.

5.1 Written Examinations

The written examination shall be administered as a closed book exam in a controlled area. The operators shall reference only retained knowledge and shall have only paper, pencils, erasers, and calculators to complete the exam. The content of the examinations shall satisfy the requirements of 10CFR55.41 and may include requirements of 10CFR55.43. The Reactor Supervisor (RS) and the Reactor Administrator (RA) shall be responsible to prepare, administer, and grade the written examination. Therefore, the RS and the RA are exempt from the written examination.

5.2 Console Examinations

Each operator shall demonstrate familiarity with the following operator activities during the console examination:

1. prestart check,
2. startup,
3. operation at power, and
4. termination.

The console examinations are required only during those years in which reactor operation is possible. Console examinations may be evaluated by any **senior** reactor operator. Every licensee shall participate in the console examination.

5.3 Grading

The criteria for grading the assignment of pass/fail are established as follows:

5.3.1 Written Examination

The licensee shall be assigned a rating of either SATISFACTORY or UNSATISFACTORY. In order to obtain a rating of SATISFACTORY, the licensee shall attain a minimum score of 70% in each section of the examination. If the licensee fails to attain a rating of SATISFACTORY, the licensee shall be removed from his/her licensed duties and enrolled in an accelerated training program in the deficient area.

5.3.2 Console Examination

The licensee shall be assigned a rating of either SATISFACTORY or UNSATISFACTORY. In order to attain a rating of SATISFACTORY, the licensee should demonstrate an understanding of the operation of all apparatus and mechanisms. This is evaluated through the ease and smoothness the operator performs the prestart checks, startup, power operation, and termination. If the licensee fails to attain a rating of SATISFACTORY, the licensee shall be removed from his/her licensed duties and enrolled in an accelerated training program in the deficient area.

6.0 OPERATOR REINSTATEMENT

An operator may be removed from active status by failing to actively perform the functions of an operator during any calendar quarter or by failing to attain a satisfactory grade on an evaluation exam. The calendar quarters are as follows: January through March, April through June, July through September, and October through December. 10CFR55.53(f) outlines the requirements for operator reinstatement.

If an operator has not actively performed the functions of an operator during a calendar quarter, he/she shall satisfactorily demonstrate his/her competence before resuming his/her licensed functions. This is accomplished by performing at least six hours of licensed functions, including at least one O.P #1 Startup and Shutdown, under the direction of a licensed operator. Upon completion of this activity, the operator shall be certified for operation by the Reactor Supervisor.

If an operator has failed to attain a satisfactory grade on any evaluation, he/she shall demonstrate his/her competence before resuming his/her duties. This is accomplished through participation in additional training in the area of deficiency. Upon completion of the training, the operator shall be certified for operation by the Reactor Supervisor after successfully completing another evaluation in the area of deficiency.

7.0. RECORDS

Operator Requalification tracking shall be maintained through a number of logs and forms. Lecture attendance shall be maintained on the Requalification/Training Lecture Forms. Each operator will record their performance of licensed functions upon completion on the Individual Operator Licensed Function Tracking Form. The annual written examination key shall be kept as part of the Operator Requalification records.

A record shall be maintained for each licensee and shall contain a current copy of the licensee's reactor operator license, copies of all written examinations administered to the licensee during the requalification period, the medical examination form from the licensee's last medical exam, and the licensee's Requalification Program Progress Checklist.

The checklist shall contain the record of attended lectures, on the job training, written and console examination evaluations, a record of operator reinstatement, medical examination completion date, and medical examination due date. Additional forms may be kept in the licensee's record to provide supporting documentation and may include license applications and renewal.

IDAHO STATE UNIVERSITY
NUCLEAR ENGINEERING LABORATORY
REQUALIFICATION PROGRAM PROGRESS CHECKLIST

Operator ______________________ License No. ____________________

License Effective Date: __/__/__ License Expiration Date: __/__/__

Training period (2 years): Beginning: __/__/__ Ending: __/__/__

Lecture Program	Date	Instructor
1. Nuclear Reactor Theory	__/__/__	____________
2. Radiation Control and Safety	__/__/__	____________
3. Governing Regulations	__/__/__	____________
4. Reactor Design	__/__/__	____________
5. Reactor Control and Safety Systems	__/__/__	____________
6. Reactor Operating Characteristics	__/__/__	____________
7. All Reactor Facility Procedures, Plans, Policies, and Rules	__/__/__	____________
8. Technical Specifications	__/__/__	____________
and License Conditions	__/__/__	____________

On the Job Training — LICENSED FUNCTIONS

	Date	Hours	Date	Hours	Date	Hours	Total
Qtr. 1	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
2	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
3	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
4	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
5	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
6	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
7	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____
8	__/__/__	____	__/__/__	____	__/__/__	____	____
	__/__/__	____	__/__/__	____	__/__/__	____	____

On the Job Training (cont.)

Prestart Check completed: __/__/__ Examiner: ________________
Startup completed: __/__/__ Examiner: ________________
Termination completed: __/__/__ Examiner: ________________

Evaluations (satisfactory/unsatisfactory)

Written Year One Date Administered __/__/__ Evaluation ________________
Year Two Date Administered __/__/__ Evaluation ________________

Console Year One	Date	Examiner	Evaluation
1. Prestart Check	__/__/__		
2. Reactor Startup	__/__/__		
3. Power Operation	__/__/__		
4. Termination	__/__/__		

Console Year Two	Date	Examiner	Evaluation
1. Prestart Check	__/__/__		
2. Reactor Startup	__/__/__		
3. Power Operation	__/__/__		
4. Termination	__/__/__		

Operator Reinstatement

Failure to complete calendar quarter licensed functions requires certification for operation by the Reactor Supervisor. This is accomplished by serving 6 hours of supervised licensed functions and performing one O.P. #1 Startup and Shutdown.

Quarter____________ Date____________ Rx. Supervisor________________

Quarter____________ Date____________ Rx Supervisor________________

Quarter____________ Date____________ Rx. Supervisor________________

An unsatisfactory grade in the evaluations require certification for operation by the Reactor Supervisor. This is accomplished through additional training.

Topic____________ Date____________ Rx. Supervisor________________

Topic____________ Date____________ Rx. Supervisor________________

Topic____________ Date____________ Rx. Supervisor________________

Medical Examination Completion Date: __/__/__
Medical Examination Due Date: __/__/__

REQUALIFICATION/TRAINING LECTURE FORM

Topic________________ Instructor____________ Date____________

Names of lecture attendees

Maintenance and Surveillance Log entries cove‑ed in training session:

Log	Entry Date	Log	Entry Date

Updated Procedures covered in training session:

Procedure	Approval Date	Procedure	Approval Date

INDIVIDUAL OPERATOR LICENSED FUNCTION TRACKING FORM

Operator Name: ____________________

Date	Duration of Functions (hrs)	Licensed Functions Performed

APPLICATION FOR RENEWAL OF CLASS 104 FACILITY LICENSE R-110 FOR OPERATION OF THE AGN-201 REACTOR

APPENDIX D

FINANCIAL QUALIFICATIONS

BY:
IDAHO STATE UNIVERSITY
POCATELLO, IDAHO

NOVEMBER 23, 1995

FINANCIAL QUALIFICATIONS

FINANCIAL QUALIFICATIONS
CONTENTS

1.0 FINANCIAL QUALIFICATIONS

This section of the application for renewal of the ISU AGN-201M reactor license includes information showing that the University has more than sufficient funds necessary to cover the estimated costs to safely operate the reactor facility and subsequently decommission it.

2.0 ESTIMATED COSTS OF OPERATIONS

The estimated operating costs for the first five (5) years of operation after renewal of the license are based on actual facility operating costs during the past three years. We do not expect any major increases in activities above the level that has existed here for the past several years, and thus do not expect any major increases in University operating expenses beyond those due to inflation, salary increase trends, etc.

If any major new activities were to involve the reactor, funding for them would have to come from the external organization causing these activities. However, a five (5) watt reactor has few substantial uses involving research or commercial utilization, as shown by the fact that it has been used almost exclusively since it was first licensed as only an educational and training facility.

It is to be noted that the reactor is "elderly" and that excessive repair costs might be expected in the coming years. However, the University has a second complete reactor console in storage, available to be used as a spare or for spare parts. Also, the College of Engineering has ample shop capacity and expertise to repair or build new mechanical equipment for the reactor, and the reactor is not part of any highly-used, tight-schedule type of operation, so repair costs can be controlled without major increases. Further, the income from the primary use of the reactor, to support one laboratory class each semester, is not sufficient to justify major upgrades of the reactor. On this basis, the estimated costs are shown below.

	1996	1997	1998	1999	2000
Estimated Ann. Oper. Costs, thousands	$60	63	66	69	73

These estimates are probably on the high side, considering recent enrollment trends, but are based on the actual costs of operation shown in Table 1.

Table 1
ISU Reactor Facility Expenses

	'93-'94	'94-'95	'95-'96
Salaries	35,200	33,100	40,600
Fringe Benefits	7,500	7,900	8,600
Major Equipment	21,245*	15,200	
Reactor Sharing Program	6,000	4,000	7,000
Misc. Materials & Supplies	450	500	600
Total	70,395	60,700	56,800

*$17,500 provided by USDOE for air monitoring equipment

3.0 SOURCES OF FUNDING

As shown in Table 1, the principal expenses are the salaries and fringe benefits, with major equipment and reactor sharing program expenses making up the remainder. The salaries constitute a line item in the College of Engineering budget, so the University is the source of this funding. The University is also the source of the occasional major equipment purchases obtained to keep the laboratory class updated to ABET Engineering accreditation standards for the laboratory. The USDOE did provide nearly all the funding for the purchase of air monitoring equipment in 1993. The USDOE is also the source of the annual Reactor Sharing Grant program funds. Thus commercial activities do not provide for any of the operating expenses, and in fact yield less than $500 per year, at most. If the USDOE funds were not available, the operation of the facility, supporting two laboratory classes, could continue, but if the University funding were to cease, the facility would have to be shutdown.

The latest financial statement (1994) for the University, included in the Appendix, shows the results of recent growth in student enrollment from about 7,000 students in 1985 to over 12,000 in 1995. Although a state University, a substantial portion of the budget comes from non-State sources. This diversity of income helps to assure a financially strong organization.

4.0 DECOMMISSIONING COSTS

When Idaho State University decides to decommission the AGN-201 reactor, the College of Engineering will prepare a Decommissioning Plan and submit it for USNRC review and approval. That Plan, with changes for "lessons learned", will follow closely the steps and procedures followed by the University of Utah in its recent decommissioning of its AGN-201 reactor. ISU will, like U of U, use in-house labor to accomplish the decommissioning work; will make arrangements for the transportation and USDOE Oak Ridge acceptance of the AGN-201 fuel; will survey and segregate activated and uncleanable-contaminated reactor components and materials from those which are non-activated and those which have been decontaminated; will dispose of activated and contaminated items following USNRC and University regulations and will return useful, non-activated nor contaminated components and materials to College equipment and inventory supplies.

We estimate the costs of decommissioning the reactor in this way to be $50,000 in 1996 dollars. This is based on actual costs incurred by U of U in decommissioning its AGN-201 reactor. If the decommissioning were to occur at the end of the twenty year license period, we estimate the costs at that time to be $100,000. However, the accuracy of that number is largely dependent on the regulatory requirements at that time, which may be significantly different. In any event, Vice President Lawson's signature on the cover letter of this application for renewal of the reactor license, confirms the University's understanding of the financial commitment and its intent to provide the necessary funding for the decommissioning of the reactor at the time this is undertaken.

APPENDIX: 1994 ISU Financial Report

Deloitte & Touche LLP

Suite 1800
101 South Capitol Boulevard
Boise, Idaho 83702

Telephone: (208) 342-9361

INDEPENDENT AUDITORS' REPORT

To the State Board of Education:

We have audited the accompanying balance sheet of the Idaho State University as of June 30, 1994, and the related statements of changes in fund balances and current funds revenues, expenditures and other changes for the year then ended. These financial statements are the responsibility of the University's management. Our responsibility is to express an opinion on these financial statements based on our audit.

We conducted our audit in accordance with generally accepted auditing standards. Those standards require that we plan and perform the audit to obtain reasonable assurance about whether the financial statements are free of material misstatement. An audit includes examining, on a test basis, evidence supporting the amounts and disclosures in the financial statements. An audit also includes assessing the accounting principles used and significant estimates made by management, as well as evaluating the overall financial statement presentation. We believe that our audit provides a reasonable basis for our opinion.

In our opinion, such financial statements present fairly, in all material respects, the financial position of Idaho State University at June 30, 1994, and the changes in its fund balances and its current funds revenues, expenditures and other changes for the year then ended in conformity with generally accepted accounting principles.

Deloitte & Touche LLP

October 14, 1994

IDAHO STATE UNIVERSITY

BALANCE SHEET
JUNE 30, 1994 WITH COMPARATIVE TOTALS FOR 1993

	Current Funds			
ASSETS	**Unrestricted**	**Auxiliary Enterprises**	**Restricted**	**Student Loan Funds**
Cash and cash equivalents	$2,622,339	$ 24,453	$(1,462,255)	$ 249,359
Student loans receivables, less allowance for doubtful loans of $416,156				2,237,730
Accounts receivable, less allowance for doubtful accounts of $296,161	612,690	625,506	806,531	68,670
Unbilled charges			2,109,817	
Due from other funds	90,198	2,631	36,833	
Due from State of Idaho	2,068,057	68,823		
Accrued interest receivable	57,896		254	
Investments	2,552,829			
Investments held in trust				
Inventories	257,270	775,979		
Property, plant and equipment				
	$8,261,279	$1,497,392	$ 1,491,180	$2,555,759

LIABILITIES AND FUND BALANCES

	Unrestricted	Auxiliary Enterprises	Restricted	Student Loan Funds
LIABILITIES:				
Accounts payable and accrued liabilities	$5,863,445	$ 822,177	$ 682,526	$ 621
Accrued interest payable				
Due to other funds	55,180	9,882	81,048	
Deposits		92,159		
Deferred revenue	904,897			
Amounts held in custody for others				
Notes and bonds payable				
	6,823,522	924,218	763,574	621
FUND BALANCES:				
Unrestricted:				
General	1,437,757	573,174		
Unexpended plant				
Renewal and replacement				
Retirement of indebtedness				
Restricted:				
General			727,606	126,589
Unexpended plant				
Renewal and replacement				
Retirement of indebtedness				
U.S. government grants refundable				2,428,549
Net investment in plant				
	1,437,757	573,174	727,606	2,555,138
	$8,261,279	$1,497,392	$ 1,491,180	$2,555,759

See notes to financial statements.

Combined Plant Funds	Agency Funds	Totals	
		1994	1993
$ 5,814,426	$(12,599)	$ 7,235,723	$ 7,229,082
		2,237,730	2,181,397
	82,794	2,196,191	2,048,257
	11,665	2,121,482	1,846,916
15,579	938	146,179	156,665
		2,136,880	3,010,128
222,842		280,992	262,702
9,043,864		11,596,693	12,827,616
1,370,071		1,370,071	1,414,568
		1,033,249	930,074
111,759,251		111,759,251	104,197,766
$128,226,033	$ 82,798	$142,114,441	$136,105,171
$ 221,681		$ 7,590,450	$ 8,177,324
194,338		194,338	183,338
	$ 69	146,179	156,665
		92,159	87,294
		904,897	861,429
	82,729	82,729	87,869
13,572,242		13,572,242	13,893,256
13,988,261	82,798	22,582,994	23,447,175
		2,010,931	3,356,351
2,389,429		2,389,429	1,604,976
642,748		642,748	604,634
658,344		658,344	669,548
		854,195	656,003
11,211,501		11,211,501	11,866,162
71,813		71,813	71,813
1,061,349		1,061,349	1,129,077
		2,428,549	2,365,050
98,202,588		98,202,588	90,334,382
114,237,772		119,531,447	112,657,996
$128,226,033	$ 82,798	$142,114,441	$136,105,171

IDAHO STATE UNIVERSITY

STATEMENT OF CHANGES IN FUND BALANCES
YEAR ENDED JUNE 30, 1994 WITH COMPARATIVE TOTALS FOR 1993

	Current Funds			Student Loan Fund
	Unrestricted	Auxiliary Enterprises	Restricted	
REVENUES AND OTHER ADDITIONS:				
Unrestricted current funds revenues	$64,491,848	$16,387,880		
Student fees			$ 430,837	
Private gifts, grants and contracts			7,012,878	$ 75
Government grants and contracts			13,442,116	39,093
Investment income			2,908	
Interest on loans receivable				59,973
Expended for plant facilities, including $4,880,752 charged to current funds' expenditures				
Retirement of indebtedness				
Public works projects completed				
Proceeds from issuance of bonds and notes payable				
Other additions			145,099	46,817
Total revenues and other additions	64,491,848	16,387,880	21,033,838	145,958
EXPENDITURES AND OTHER DEDUCTIONS:				
Educational and general expenditures	64,721,224		20,256,769	
Auxiliary enterprises expenditures		16,642,120		
Indirect costs recovered			1,055,426	
Loan cancellations, write-offs and provisions for doubtful accounts				50,988
Expended for plant facilities, including $73,591 of non-capitalized repairs and maintenance				
Retirement of indebtedness				
Interest on indebtedness				
Disposal of plant facilities				
Additions to indebtedness				
Other deductions				39,231
Total expenditures and other deductions	64,721,224	16,642,120	21,312,195	90,219
TRANSFERS AMONG FUNDS - ADDITIONS (DEDUCTIONS):				
Mandatory:				
Principal and interest				
Loans and matching grants	(335,722)		328,823	6,899
Net transfers (voluntary)	(466,282)	(59,800)	148,587	
Total transfers	(802,004)	(59,800)	477,410	6,899
NET INCREASE (DECREASE) FOR THE YEAR	(1,031,380)	(314,040)	199,053	62,638
FUND BALANCES, BEGINNING OF YEAR	2,469,137	887,214	528,553	2,492,500
FUND BALANCES, END OF YEAR	$ 1,437,757	$ 573,174	$ 727,606	$2,555,138

See notes to financial statements.

Plant Funds				Totals	
Unexpended Plant	Renewal and Replacement	Retirement of Indebtedness	Net Investment in Plant	1994	1993
				$ 80,879,728	$ 75,054,209
$ 1,554,519				1,985,356	1,563,150
72,000			$ 1,321,502	8,406,455	5,423,432
				13,481,209	13,993,834
456,816	45,990	115,259		620,973	427,722
				59,973	68,005
			6,322,571	6,322,571	4,602,602
			306,721	306,721	2,324,052
			1,062,399	1,062,399	163,603
					13,000,000
		62,038		253,954	358,963
2,083,335	45,990	177,297	9,013,193	113,379,339	116,979,572
				84,977,993	76,519,430
				16,642,120	16,149,574
				1,055,426	938,427
				50,988	58,895
1,515,410				1,515,410	634,533
		306,721		306,721	2,324,052
		772,794		772,794	600,240
			1,144,987	1,144,987	1,376,085
					13,000,000
		218		39,449	156,718
1,515,410		1,079,733	1,144,987	106,505,888	111,757,954
(923,298)		923,298			
485,165	(7,876)	(99,794)			
(438,133)	(7,876)	823,504			
129,792	38,114	(78,932)	7,868,206	6,873,451	5,221,618
13,471,138	676,447	1,798,625	90,334,382	112,657,996	107,436,378
$13,600,930	$714,561	$1,719,693	$98,202,588	$119,531,447	$112,657,996

IDAHO STATE UNIVERSITY

STATEMENT OF CURRENT FUNDS REVENUES, EXPENDITURES AND OTHER CHANGES
YEAR ENDED JUNE 30, 1994 WITH COMPARATIVE TOTALS FOR 1993

	Unrestricted	Auxiliary Enterprises
REVENUES:		
Appropriated general education revenues:		
State general account - general education	$37,149,654	$ 1,093,446
Endowment income	1,857,713	
Student fees and miscellaneous receipts	8,761,951	
Idaho dental education program	438,000	
Museum of Natural History	405,400	
Family practice	265,400	
Vocational education	5,923,572	
Federal grants and contracts	12,170	
State grants and contracts	120,340	
Private gifts, grants and contracts	468,022	365,978
Other student fees	4,613,763	4,268,768
Sales and services of educational departments	1,307,499	
Sales and services of auxiliary enterprises		10,659,688
Indirect costs recovered	1,055,426	
Other sources	2,112,938	
Total revenues	64,491,848	16,387,880
EXPENDITURES AND MANDATORY TRANSFERS:		
Educational and general:		
Instruction	37,141,029	
Research	1,111,536	
Public service	535,946	
Academic support	5,009,482	
Libraries	2,843,863	
Student services	3,178,219	
Institutional support	6,786,847	
Operations and maintenance of plant	5,914,366	
Scholarships and fellowships	2,199,936	
Educational and general expenditures	64,721,224	
Mandatory transfers:		
Loans and matching grants	335,722	
Principal and interest		
Total educational and general	65,056,946	
Auxiliary enterprise		
Expenditures		16,642,120
Total expenditures and mandatory transfers	65,056,946	16,642,120
OTHER TRANSFERS AND ADDITIONS (DEDUCTIONS):		
Excess (deficiency) of restricted receipts over transfers to revenue		
Net transfers, voluntary	(466,282)	(59,800)
Total other transfers and additions (deductions)	(466,282)	(59,800)
NET INCREASE (DECREASE) IN FUND BALANCES	$(1,031,380)	$ (314,040)

See notes to financial statements.

	Restricted	Totals 1994	Totals 1993
		$ 38,243,100	$36,280,112
		1,857,713	1,783,088
		8,761,951	7,874,503
		438,000	408,800
		405,400	0
		265,400	0
		5,923,572	5,588,374
	$ 9,691,017	9,703,187	10,217,560
	2,642,152	2,762,492	2,914,362
	7,012,878	7,846,878	5,677,640
	430,837	9,313,368	8,059,971
	5,963	1,313,462	1,425,729
		10,659,688	10,215,385
		1,055,426	938,427
	145,099	2,258,037	2,083,579
	19,927,946	100,807,674	93,467,530
	2,613,547	39,754,576	34,393,455
	4,791,642	5,903,178	4,117,047
	2,845,869	3,381,815	3,596,036
	203,766	5,213,248	5,054,899
	116,681	2,960,544	2,816,618
	804,386	3,982,605	3,929,712
	38,146	6,824,993	6,005,498
	434,158	6,348,524	6,042,746
	8,408,574	10,608,510	10,563,419
	20,256,769	84,977,993	76,519,430
	(328,823)	6,899	
			221,887
	19,927,946	84,984,892	76,741,317
		16,642,120	16,149,574
	19,927,946	101,627,012	92,890,891
	50,466	50,466	(15,963)
	148,587	(377,495)	(1,087,329)
	199,053	(327,029)	(1,103,292)
	$ 199,053	$ (1,146,367)	$ (526,653)

APPLICATION FOR RENEWAL OF CLASS 104
FACILITY LICENSE R-110 FOR OPERATION
OF THE AGN-201 REACTOR

APPENDIX E

ENVIRONMENTAL REPORT

BY:
IDAHO STATE UNIVERSITY
POCATELLO, IDAHO

NOVEMBER 23, 1995

ENVIRONMENTAL REPORT

CONTENTS

1.0 INTRODUCTION

The Idaho State University (ISU) AGN-201M reactor facility has been operated by the College of Engineering and its precedent entities since the college was granted facility operating license No. R-110 by the U.S. Atomic Energy Commission in 1967. The reactor is used as a primary instructional and research tool in meeting the educational goals of the nuclear science and engineering program. Additionally, the reactor is used by students and faculty from other departments and colleges at ISU, including the departments of Physics and Chemistry and the College of Pharmacy. The reactor supports basic nuclear engineering laboratory experiments, limited isotope production, and the training of student reactor operators. The reactor facility is operated in conformance with all applicable federal, state, local, and University rules and regulations.

Located on the ISU campus, the AGN-201 reactor laboratory consists of a one-story extension of the Lillibridge Engineering Laboratory (LEL) building. The reactor laboratory comprises a room 7.3-m high (24 ft) whose floor plate measures approximately 8.5m x 8.5 m (28 ft x 28ft). The floor of the laboratory is approximately 3 m (10 ft) below ground level. Three of the laboratory walls are of monolithic reinforced-concrete construction with a brick veneer and are the exterior walls of the building extension. The interior wall adjoining the main LEL building is constructed of concrete block and mortar. The roof of the reactor laboratory is a 10-cm-thick (4 in) concrete slab supported by corrugated steel plates and I-beams. Areas surrounding the exterior walls of the laboratory are bordered by trees and lawn; the nearest sidewalks are about 5 m from the north and west walls.

The AGN-201M reactor is a self-contained, thermal reactor with a compact core of 20-percent-enriched uranium oxide homogeneously dispersed in a matrix of low-density polyethylene. The core is contained within an hermetic aluminum can that isolates the reactor fuel and prevents the escape of fission-product gases to the environment. Because of the low thermal power level of the reactor no external means are required for cooling the reactor core. Heat transfer of fission energy occurs through conduction from the fuel through the graphite reflector and lead shield surrounding the core to the approximately 3790 liters (1000 gallons) of water contained within the shield tank. In addition to serving as a fast neutron shield, this water acts as a massive thermal sink. Hence, there are no thermal or radioactive liquid effluents from the facility.

Originally, the reactor operating license allowed operation at power levels up to 100 mW. In 1977, the operating license was amended to allow operation up to a maximum power of 5W after high-density, concrete shielding with a minimum 30 cm (1 ft) thickness was installed around the reactor to reduce the radiation fields generated at increased power levels. Radiation levels at the surface of the shielding during operation at full power are generally below 5 mrem hr^{-1}.

A comprehensive radiation protection program is administered by the Radiation Safety Division of the ISU Technical Safety Office (TSO). The TSO, whose Director is also the

University Radiation Safety Officer, provides administrative and technical support for users of radioactive materials at ISU. All faculty, staff and students with regular duties in the reactor laboratory have been issued individual radiation dosimeters which measure the equivalent dose from beta, gamma, and neutron radiation. Visitors to the laboratory are provided with self-reading pocket dosimeters for monitoring exposure to radiation. The cumulative exposure to ionizing radiation in and around the reactor laboratory is measured by thermoluminescent dosimeters located at fixed environmental monitoring stations within the laboratory. Personnel and environmental dosimetry are analyzed on a quarterly basis.

2.0 RECENT REACTOR OPERATING HISTORY

A summary of reactor operations from the past seven years is presented in Table 1. Annual average values for the number of hours of operation and energy release are given in Table 2. On average, the reactor is operated 116.4 hr each year at a mean power level of 0.81 W. Most reactor operations occur during the fall and spring semesters of the academic calendar year (September through May) in conjunction with the teaching of undergraduate and graduate nuclear engineering laboratory classes.

Table 1. Reactor operating hours and annual energy release for the period 1988-94.

	1988	1989	1990	1991	1992	1993	1994
January	17	13.7	7.8	2.6	0	6.1	6.5
February	27.5	14.9	20.6	6.8	24	8.5	8.8
March	22.4	11.9	34.6	5.9	12.4	8.8	9.1
April	24.8	11.4	23.4	8.6	4.9	16.5	4.9
May	5.1	5.3	10.7	2.9	5	3	1.3
June	2.6	0.5	14.5	3.1	7.1	0	6.5
July	3.2	0.8	4.5	0	0	3	2.5
August	4.2	5	2.1	4.3	7.6	8	6
September	7.7	11.8	10.2	16.7	14.3	31.7	37.8
October	15.8	12.9	12.2	18	44.7	23.2	26
November	20.8	11.8	9.8	10.9	20.8	12.3	26.5
December	3.1	1.6	4.3	17.3	5.9	13.2	6.6
Total Operation (hr)	154.2	101.6	154.7	97.1	146.7	134.3	142.5
Total Energy Release (W-hr)	156	110	157	64	96	103	72

Table 2. Average annual values for AGN-201 reactor operations.

Annual Operation	116.4 Hours
Energy Generation	94.8 watt-hours
Mean Power Level	0.81 watts

3.0 GASEOUS EFFLUENT: ARGON-41 PRODUCTION

Argon-41, a short-lived (1.82 hr half-life) radioactive noble gas, is generated near the reactor core by neutron capture in argon-40, a minor constituent of air. Argon is present in dry air to the extent of 0.93 percent by volume or 1.3 percent by mass. Air may be present in reactor experimental facilities such as the glory hole and/or the four access ports. Air is also in solution in the shield tank water. Argon-41 produced from the activation of dissolved air in the tank water is negligible because: (1) the concentration of argon in the tank water is small, (2) the thermal neutron flux in the shield water is several orders of magnitude lower than in the reactor core, and (3) the access cover retards diffusion of the gas from the shield tank into the reactor room.

An upper limit on ^{41}Ar concentrations during reactor operation at full-licensed power has been established by considering the production of ^{41}Ar assuming that the saturation activity has been reached and that the air within the reactor has been irradiated for a "long" time (at least seven half-lives, or about 13 hours in order to approach the saturation activity. Reactor operations for this length of time are not currently performed nor are they envisioned in the future. The second assumption is reasonable since the ventilation system operates continuously with a volumetric flow rate to the reactor room of about 7.83E+5cm^3 s^{-1} (1660 cfm). Credit for radioactive decay is not taken to increase conservativism.

At the maximum operating power of 5W, the average thermal neutron flux in the glory hole is approximately 1.75E+8 n cm^{-2} s^{-1}. The resulting saturation activity of ^{41}Ar is equal to the rate of ^{41}Ar production within the glory hole which is 10,600 atoms s^{-1}. In terms of activity, this production rate is 30.2 pCi s^{-1}. Consequently, the maximum concentration of ^{41}Ar within the reactor laboratory is 3.85E-11 μCi cm^{-3}. In restricted areas such as the reactor room, the Derived Air Concentration (DAC) establishes the maximum permissible concentration of airborne radioactive materials to which occupational workers may be exposed. The DAC for ^{41}Ar as promulgated by the U.S. Nuclear Regulatory Commission in 10 CFR 20 Appendix B is 3E-6 μCi cm^{-3}. Thus, the maximum ^{41}Ar concentration in the reactor laboratory is 0.0013 percent of established limits.

Air flow from the reactor laboratory room is mixed with return air from the rest of the LEL building. Approximately 80 percent of the total air flow is recycled to the building while the remainder is discharged from a roof vent located some 9m above ground level. Air flow leaving the reactor room is diluted by a factor of about 27 before being discharged to the environment. Return air is diluted further by mixing with fresh makeup air before being recirculated to the building. The calculated maximum effluent concentration of ^{41}Ar at the penthouse discharge vent is 1.5E-12 μCi cm^{-3}. The calculated ^{41}Ar concentration in the recycled air to unrestricted areas in the LEL building is approximately 1.0E-12 μCi cm^{-3}.

These concentrations are insignificant in comparison with the maximum permissible concentration in unrestricted areas established by the U.S. Nuclear Regulatory Commission. The maximum permissible effluent concentration for ^{41}Ar provided by 10 CFR 20 Appendix B is 1E-8 μCi cm^{-3}. This value is the maximum concentration of ^{41}Ar allowed in any unrestricted areas. The maximum concentration of ^{41}Ar in the unrestricted areas of the LEL building from the ventilation system recycle is 0.010 percent of the maximum permissible level. The ^{41}Ar concentration discharged to the atmosphere from the exhaust stack is about 0. 015 percent of the maximum permissible level. This estimated value is much greater than the actual concentration at ground level since it does not take into account further dilution from the vent.

Based on the average annual reactor operation data given in Table 2, the total activity of ^{41}Ar generated in one year is 2.1 μCi. Neglecting radioactive decay , the concentration of ^{41}Ar released to the environment, averaged over a one-year period is 8.5E-14 μCi cm^{-3}. This value is 0.00085 percent of the permissible level. The results of these calculations clearly show that any hazard associated with the production of ^{41}Ar by operation of the AGN-201 reactor at Idaho State University is minuscule.

4.0 OTHER EFFLUENTS

Except for induced radioactivity in deliberately irradiated experiments, no liquid or solid radioactive effluents are generated by the AGN-201 reactor. Because of the low neutron flux of the reactor, neutron activation of reactor components is insignificant. Experience gained by the licensees of other AGN-201 reactor facilities during the decommissioning of this type of reactor has demonstrated that activation products in reactor structures and components were generally below core components that had been in direct contact with the uranium-impregnated polyethylene fuel disks. Contaminated surfaces were limited to the interior surface of the core, in- core graphite reflector elements, and the interior surfaces of the aluminum clads of the safety and control rods.

Byproduct material, consisting primarily of short-lived isotopes produced by the irradiation of experiments, is transferred as necessary to the TSO under the Idaho

State University Board Scope license for final disposition in accordance with all current federal, state and local regulations. Records of all the radionuclides produced by the reactor are maintained in a Radioisotope Production and Disposition Log. The thermal output from the reactor operating at full-licensed power is inconsequential and has less environmental impact than the thermal load produced by the fluorescent lamps that illuminate the reactor laboratory.

5.0 ENVIRONMENTAL MONITORING

Radiation levels in the unrestricted areas surrounding the reactor during operation at full-licensed power are measured and documented on an annual basis. The maximum equivalent dose rate at the interior surface of the exterior walls of the reactor room is 0.5 mrem hr^{-1}. No detectable neutron radiation has been observed outside of the LEL building. All radiation dose rates in unrestricted areas of the LEL are below threshold levels. Accumulated equivalent doses within the restricted areas of the reactor Laboratory are measured by LiF thermoluminescent dosimeters (TLDs) at various fixed environmental stations. These TDLs are processed quarterly and provide an upper limit on the cummulative environmental dose from reactor operations. Measured doses from the last 2 years are summarized in Table 3. The TDLs are capable of detecting a minimum dose of 10 mrem. Doses listed as <10 mrem were below this threshold. From Table 3, a maximum annual environmental doseof about 50 mrem is produced as a consequence of AGN-201 reactor operations. This dose is a fraction of the annual background radiation dose from cosmic rays and naturally occuring radioactive materials found in the soil and building materials.

Water samples from the AGN shield tank are routinely analyzed for radioactive contamination. The samples are counted using a liquid scintillation detector system which measures energetic emissions in two energy regions spanning 0-420 keV and 420-1000 keV. Results from the latest surveillance, performed 2/21/95 showed there was no contamination present in the water samples above normal levels for the local culinary water supply.

Air particulate samples taken around the glory hole and thermal column of the AGN-201 reactor have also been analyzed. The very small activities measured in the samples, an average of 4 dpm in the 0-420 keV energy region and 6 dpm in the 420-1000 keV region, are comparable to the activities of air samples obtained at other areas of the campus and are most likely due to the decay of radon progeny.

Contamination surveys of the reactor laboratory are performed semiannually or as needed by trained personnel from the Technical Safety Office. Surveys include wipe samples from the reactor tank, thermal column, glory hole, and concrete shield. Results of these surveys show that removable contamination is less than 20 dpm over an area of 100 cm^2

Table 3. Summary of environmental doses 1993-1995.

Location	Quarterly Total Effective Dose Equivalent (mrem)								
	1995	1994				1993			
	1st Quarter	4th Quarter	3rd Quarter	2nd Quarter	1st Quarter	4th Quarter	3rd Quarter	2nd Quarter	1st Quarter
Interior (East) Wall	<10	<10	30	<10	<10	**	**	**	**
Reactor Console	<10	40	10	<10	<10	10	10	20	10
Exterior (West) Wall	<10	<10	<10	<10	<10	<10	<10	<10	<10

**This Environmental Monitoring Station was established at the end of 1993; therefore, no data exist prior to 1993.

All routine radiation surveillance and monitoring indicate that the AGN reactor core is intact. No radioactive fission products, gases or otherwise, have been detected outside of the reactor core. It is possible, however, that radioactive contamination may result from the activation of contaminants on irradiated experiments, or from the breakage of irradiated experiments or other encapsulated radioactive materials in use at the laboratory. In such cases, approved procedures will be followed to clean up any such contamination as soon as it occurs. Under no circumstance is radioactive contamination allowed to remain in place indefinitely.

6.0 CONCLUSION

A comprehensive radiation protection program is in place which oversees routine, periodic radiation and contamination surveillance and environmental monitoring of the reactor facility to ensure that the integrity of the reactor core is preserved and that all radiation exposures are kept as low as reasonably achievable. The small amount of radioactive materials generated by the reactor in support of the engineering curriculum and research programs are produced, used, controlled, and ultimately disposed of in accordance with all pertinent federal, state, local, and University regulations. The benefits gained from the operation of the ISU AGN-201 reactor in terms of contributions to scientific knowledge and training of future engineers and scientists in the areas of nuclear and radiological science far outweigh any minor, negative environmental effects.

Continued operation of the reactor will not require any alteration of the existing building or structures, will not lead to changes in effluents released from the facility to the environment, and will not increase the probability or consequences of accidents. Operation of the ISU AGN-201 reactor has no significant impact on the environment.

APPLICATION FOR RENEWAL OF CLASS 104 FACILITY LICENSE R-110 FOR OPERATION OF THE AGN-201 REACTOR

APPENDIX F

SAFETY ANALYSIS REPORT

BY:
IDAHO STATE UNIVERSITY
POCATELLO, IDAHO

NOVEMBER 23, 1995

SAFETY ANALYSIS REPORT

CONTENTS

CONTENTS (CONT'D)

CONTENTS (CONT'D)

SAFETY ANALYSIS REPORT

LIST OF FIGURES

SAFETY ANALYSIS REPORT

LIST OF TABLES

1.0 INTRODUCTION

This report supports an application to the United States Nuclear Regulatory Commission (NRC) by Idaho State University (ISU) at Pocatello, Idaho, for the renewal of the class 104, Facility License R-110 for the University's AGN-201 nuclear reactor. Because of the extensive operating experience acquired by ISU and others with the AGN-201 reactor, the system has a well established and corroborated data base, so no research and development activities were required to evaluate the system. It has been possible to use accepted, standard safety analysis techniques with regard to evaluating the characteristics of this facility. The reactor is administered and operated by Idaho State University for education, research, and training of students. However, it is deemed that the facility benefits all interested parties within the regional area.

The report contains information on the Pocatello, Idaho location and site characteristics such as meteorology, geology, seismology and demography, etc. It also includes a description of the Lillibridge Engineering Laboratory building where the reactor is housed on the ISU campus and a general description of the reactor and listing of its characteristics. The report concludes with a detailed safety analysis of the reactor.

The reactor is operated at steady state power levels up to 5 W_{th}. The total operational fuel loading provides a maximum of 0.65% excess reactivity above a delayed critical condition. The safety of the AGN-201 reactor design lies in its negative temperature coefficient, thermal safety fuse and its nuclear instrumentation safety systems. Thus, even if a sudden reactivity insertion were made, the reactor power rise would be attenuated by the negative temperature coefficient and terminated by the melting of the thermal safety fuse which otherwise holds the upper and lower halves of the core together.

The principal applicant involved in the continued operation of the ISU AGN-201 reactor is the Vice President for Academic Affairs of Idaho State University. Responsibility for the safety and general operation of the reactor has been delegated to the College of Engineering.

2.0 LOCATION AND SITE CHARACTERISTICS

Information in this section includes maps, building drawings and data on local population characteristics; meteorology, geology, hydrology and seismology of southeastern Idaho.

2.1 Location and Demography

Figure 2.1-1 is a map of southeast Idaho showing county and state demarcations, major rivers, lakes and reservoirs and the principal cities in the region. Pocatello in Bannock County and Idaho Falls in Bonneville County are the largest cities. The U.S. Department of Energy's Idaho National Engineering Laboratory is about 50 miles northwest of Pocatello. Hill Air Force Base is about 120 miles south of Pocatello in the vicinity of Salt Lake City, Utah. Idaho State University is located in Pocatello, as shown in Figure 2.1-2. The major employers of this city of approximately 50,000 people are two phosphate rock processing companies, the Union Pacific Railroad and the University. The major north-south highway, Interstate 15, passes to the east of the city and the campus. An intersection of I-15 with Interstate 84 which heads west toward the cities of Twin Falls and Boise is located just north of the city. The Pocatello Municipal Airport is about 7 miles northwest of the campus on Interstate 84. The Union Pacific Railroad east-west main line tracks go through the center of the city, about 2 miles west of the campus. The city lies in the Portneuf River valley at a general elevation of 4470 ft bordered on the southwest and northeast by hills of the Bannock mountain range which rises about 3,000-4,000 ft above the valley floor. Bannock County population information, including distribution by race, is given in Table 2.1-1 and Pocatello city government financial information if given in Table 2.1-2. Figure 2.1-3 is an ISU campus map showing the location of the Lillibridge Engineering Laboratory as well as the other buildings, etc. on the campus. The University presently has somewhat more than 12,000 students and 1200 faculty, the College of Engineering being the smallest of the colleges in the University. Table 2.1-3 includes demographics of the estimated 10,430 people who reside within a one mile radius of the Lillibridge Engineering Laboratory. The Laboratory building is about 300 ft south of Carter Street. The nearest residential area extends northward from that street. The Physical Science building is located between the Laboratory and the Carter Street residences.

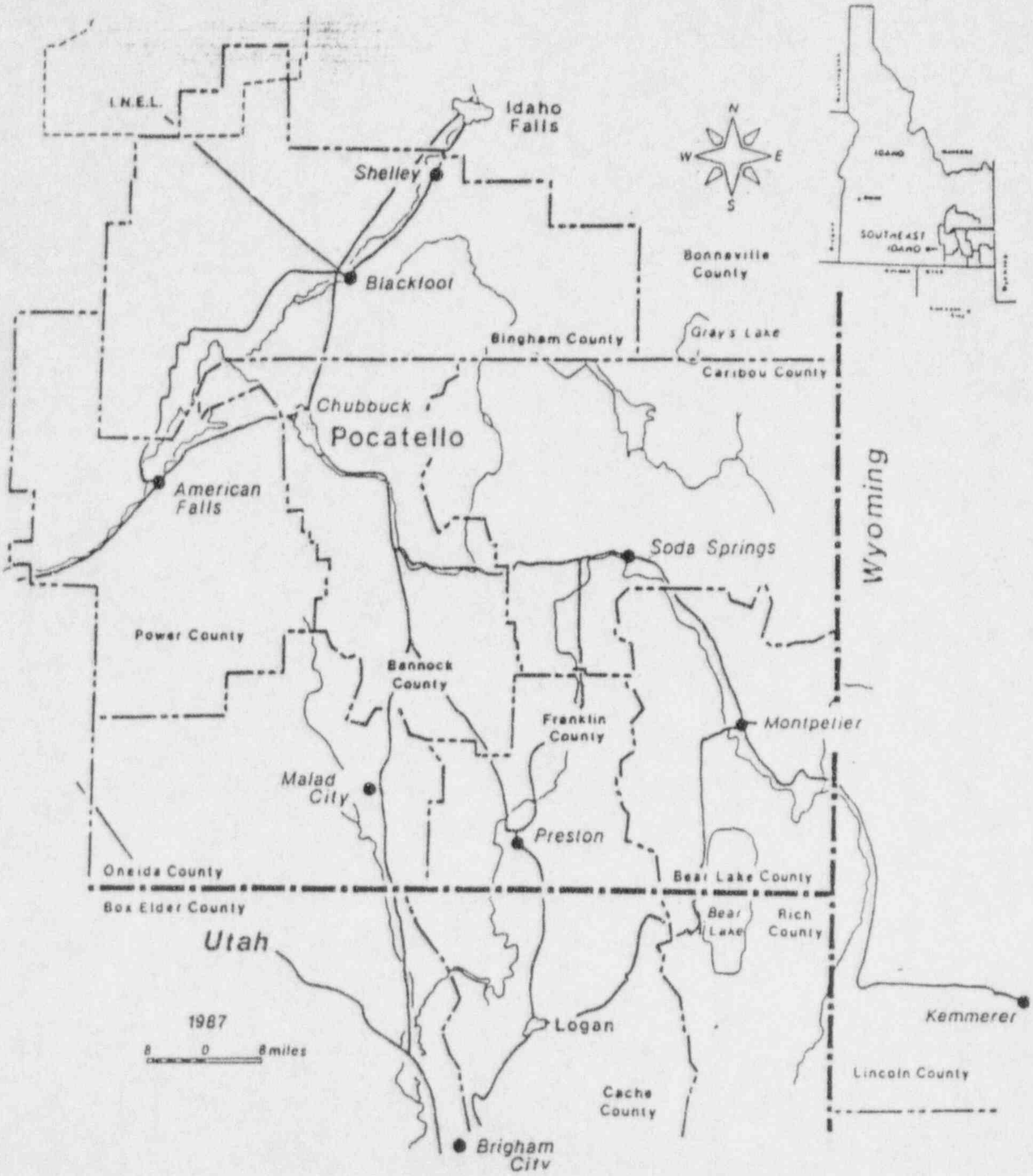


Figure 2.2-1 Map of Southeast Idaho.

Figure 2.1-2 Map of the City of Pocatello, Idaho

9511280385-01

Table 2.1-1 Area and Population

Metro-politan area code[1]	State and county code[2]	County	Land area,[3] 1990 (Sq. mi.)	Population: 1992 Total persons	Population: 1992 Rank[4]	Population: 1992 Per square mile[3]	Population: 1990	Population: 1980	Net change, 1980–1992: Number	Net change, 1980–1992: Percent	Population characteristics, 1990 — Race: White	Race: Black	Race: American Indian, Eskimo, or Aleut	Race: Asian or Pacific Islander
			1	2	3	4	5	6	7	8	9	10	11	12
	16 000	IDAHO	82 751	1 065 885	X	13	1 006 749	944 127	121 758	12.9	950 451	3 370	13 780	9 365
1080	16 001	Ada	1 055	223 335	229	212	205 775	173 125	50 210	29.0	198 888	958	1 382	2 887
...	16 003	Adams	1 365	3 536	2 950	3	3 254	3 347	189	5.6	3 203	2	41	1
[illegible]	16 005	Bannock	1 113	68 561	862	62	66 026	65 421	3 140	4.8	61 742	431	1 678	712
...	16 007	Bear Lake	971	6 258	2 731	6	6 084	6 931	-673	-9.7	5 999	-	25	5
...	16 009	Benewah	776	8 107	2 562	10	7 937	8 292	-185	-2.2	7 278	6	602	28
...	16 011	Bingham	2 095	39 613	1 032	19	37 583	36 489	3 124	8.6	32 439	39	2 615	273
...	16 013	Blaine	2 645	14 883	2 015	6	13 552	9 841	5 042	51.2	13 241	10	53	104
...	16 015	Boise	1 903	4 035	2 913	2	3 509	2 999	1 036	34.5	3 431	2	35	14
...	16 017	Bonner	1 738	28 967	1 347	17	26 622	24 163	4 804	19.9	26 210	37	220	71
...	16 019	Bonneville	1 869	77 395	599	41	72 207	65 980	11 415	17.3	69 246	297	391	687
...	16 021	Boundary	1 269	8 639	2 502	7	8 332	7 289	1 350	18.5	7 950	3	150	26
...	16 023	Butte	2 233	2 940	3 002	1	2 918	3 342	-402	-12.0	2 829	-	22	5
...	16 025	Camas	1 075	755	3 125	1	727	818	-63	-7.7	712	2	8	3
1080	16 027	Canyon	590	96 260	496	163	90 076	83 756	12 504	14.9	80 445	175	687	987
...	16 029	Caribou	1 786	7 115	2 645	4	6 963	8 695	-1 580	-18.2	6 824	7	22	13
...	16 031	Cassia	2 567	20 159	1 690	8	19 532	19 427	732	3.8	17 580	3	170	96
...	16 033	Clark	1 765	798	3 123	Z	762	798	-	-	688	-	5	-
...	16 035	Clearwater	2 462	8 666	2 499	4	8 505	10 390	-1 724	-16.6	8 262	10	180	21
...	16 037	Custer	4 926	4 049	2 911	1	4 133	3 385	664	19.6	4 044	2	33	19
...	16 039	Elmore	3 078	20 570	1 667	7	21 205	21 565	-995	-4.6	18 898	777	171	453
...	16 041	Franklin	666	9 471	2 436	14	9 232	8 895	576	6.5	9 052	5	38	12
...	16 043	Fremont	1 867	11 240	2 284	6	10 937	10 813	427	3.9	10 273	9	68	37
...	16 045	Gem	563	12 556	2 187	22	11 844	11 972	584	4.9	11 322	13	139	53
...	16 047	Gooding	731	12 030	2 225	16	11 633	11 874	156	1.3	10 886	7	43	31
...	16 049	Idaho	8 485	14 191	2 063	2	13 783	14 769	-578	-3.9	13 378	3	346	34
...	16 051	Jefferson	1 095	17 486	1 833	16	16 543	15 304	2 182	14.3	15 627	7	122	40
...	16 053	Jerome	600	15 684	1 953	26	15 138	14 840	844	5.7	14 304	9	115	54
...	16 055	Kootenai	1 245	77 450	598	62	69 795	59 770	17 680	29.6	68 461	94	675	326
...	16 057	Latah	1 077	31 786	1 246	30	30 617	28 749	3 019	10.5	29 388	174	206	709
...	16 059	Lemhi	4 564	7 080	2 648	2	6 899	7 460	-380	-5.1	6 773	2	49	23
...	16 061	Lewis	479	3 628	2 943	8	3 516	4 118	-490	-11.9	3 322	4	169	18
...	16 063	Lincoln	1 206	3 425	2 960	3	3 308	3 436	-11	-.3	3 231	3	22	12
...	16 065	Madison	472	23 953	1 509	51	23 674	19 480	4 473	23.0	22 741	43	108	296
...	16 067	Minidoka	760	20 167	1 688	27	19 361	19 718	449	2.3	16 540	43	201	100
...	16 069	Nez Perce	849	34 938	1 151	41	33 754	33 220	1 718	5.2	31 681	48	1 692	211
...	16 071	Oneida	1 200	3 469	2 957	3	3 492	3 258	211	6.5	3 431	4	19	8
...	16 073	Owyhee	7 678	8 545	2 514	1	8 392	8 272	273	3.3	6 935	22	276	78
...	16 075	Payette	408	17 477	1 834	43	16 434	15 825	1 652	10.4	15 210	14	189	158
...	16 077	Power	1 406	7 520	2 617	5	7 086	6 844	676	9.9	6 157	7	203	40
...	16 079	Shoshone	2 634	13 644	2 102	5	13 931	19 226	-5 582	-29.0	13 620	16	182	40
...	16 081	Teton	450	3 864	2 925	9	3 439	2 897	967	33.4	3 360	2	13	1
...	16 083	Twin Falls	1 925	56 000	793	29	53 580	52 927	3 073	5.8	51 202	65	309	524
...	16 085	Valley	3 678	6 935	2 666	2	6 109	5 604	1 331	23.8	5 988	8	60	27
...	16 087	Washington	1 456	8 723	2 497	6	8 550	8 803	-80	-.9	7 860	7	46	130

Table 2.1-2 City Government Finances and Climate

City	City government finances, 1990-1991 – Con. General expenditure: Total ($1,000)	Per capita[1] (Dollars)	By selected function, percent for – Education	Public welfare	Health and hospitals	Police protection	Fire protection	Highways	Sewerage and solid waste management	Climate[2] Average daily temperature (Degrees Fahrenheit) – Mean: January	Mean: July	Limits: January[3]	Limits: July[4]	Annual precipitation (Inches)	Heating degree days[5]	Cooling degree days[6]
	179	180	181	182	183	184	185	186	187	188	189	190	191	192	193	194
FLORIDA – Con.																
Lakeland city	68 650	973	-	-	-	15.8	7.0	11.4	16.8	61.1	82.5	50.4	92.2	47.54	588	3 546
Lake Worth city	29 071	1 018	-	-	-	22.1	13.7	3.8	24.7	65.1	82.2	55.7	89.9	60.75	323	3 891
Largo city	43 145	657	-	-	-	14.4	10.6	3.0	38.1	59.9	82.1	50.0	90.2	43.92	726	3 396
Lauderdale Lakes city	8 315	304	-	-	7.6	20.3	13.5	4.1	-	67.2	82.6	57.9	90.1	60.64	205	4 124
Lauderhill city	14 928	300	-	-	4.3	25.1	23.9	5.8	-	67.2	82.6	57.9	90.1	60.64	205	4 124
Margate city	15 965	371	-	-	Z	28.8	10.0	7.2	-	66.2	82.5	56.8	91.2	59.15	262	4 038
Melbourne city	45 774	767	-	Z	Z	16.1	9.8	7.7	18.8	60.9	81.1	50.7	90.0	45.49	644	3 193
Miami city	312 189	871	Z	.3	Z	25.3	14.2	2.5	13.3	67.2	82.6	59.2	89.0	55.91	200	4 198
Miami Beach city	147 783	1 595	-	Z	3.6	18.6	6.4	1.8	7.5	68.1	82.6	62.3	86.9	45.35	139	4 157
Miramar city	15 968	393	-	-	-	27.9	19.2	.6	12.4	66.3	82.4	56.8	89.8	63.01	273	4 012
North Lauderdale city	NA	NA	NA	NA	NA	NA	NA	NA	NA	66.2	82.5	56.8	91.2	59.15	262	4 038
North Miami city	24 135	483	-	-	-	22.9	-	4.8	10.5	66.3	82.4	56.8	89.8	63.01	273	4 012
North Miami Beach city	23 606	673	-	-	-	26.9	-	3.0	22.1	68.1	82.6	62.3	86.9	45.35	139	4 157
Oakland Park city	17 322	658	-	-	3.2	29.4	11.2	3.4	20.6	67.2	82.6	57.9	90.1	60.64	205	4 124
Ocala city	39 858	948	-	-	-	16.9	10.3	17.7	9.6	57.5	81.5	45.1	92.2	51.59	930	3 046
Orlando city	612 895	3 721	-	-	-	5.4	3.0	2.7	9.6	59.7	82.3	48.6	91.5	48.11	686	3 361
Ormond Beach city	22 548	759	-	-	-	13.3	7.1	14.5	9.7	57.5	81.2	46.9	89.8	47.89	909	2 919
Palm Bay city	26 304	420	-	-	-	20.6	15.4	20.9	9.7	60.9	81.1	50.7	90.0	45.49	644	3 193
Panama City city	21 229	618	-	-	.7	16.3	14.4	13.1	8.6	50.7	80.8	39.1	90.0	65.06	1 661	2 409
Pembroke Pines city	32 345	494	-	-	3.0	21.5	15.8	8.0	.8	66.3	82.4	56.8	89.8	63.01	273	4 012
Pensacola city	75 580	1 299	-	-	.2	10.8	9.0	6.2	4.6	50.6	82.1	41.4	89.9	62.25	1 617	2 636
Pinellas Park city	22 593	520	-	-	.1	19.7	12.3	18.8	15.3	60.7	83.1	52.9	89.8	48.62	603	3 626
Plantation city	26 410	396	-	-	-	38.8	3.0	3.8	1.1	67.2	82.6	57.9	90.1	60.64	205	4 124
Pompano Beach city	67 327	930	-	-	4.7	25.5	8.9	2.2	2.7	66.2	82.5	56.8	91.2	59.15	262	4 038
Port Orange city	20 209	572	-	-	-	12.5	9.7	8.6	10.9	57.5	81.2	46.9	89.8	47.89	909	2 919
Port St. Lucie city	12 473	223	-	-	1.7	34.7	-	19.2	-	62.5	81.4	51.6	90.4	50.06	490	3 441
Riviera Beach city	32 165	1 164	-	-	1.4	22.0	10.2	3.5	13.8	65.1	82.2	55.7	89.9	60.75	323	3 891
St. Petersburg city	266 142	1 124	-	-	1.5	13.9	5.4	9.5	9.5	60.7	83.1	52.9	89.8	48.62	603	3 626
Sanford city	18 263	564	-	-	.1	22.8	11.7	13.9	6.0	58.2	81.5	46.8	91.6	48.81	831	3 004
Sarasota city	78 699	1 544	-	-	-	14.7	10.0	9.2	25.8	60.2	81.4	49.0	90.8	53.71	678	3 186
Sunrise city	44 473	690	-	-	-	15.9	13.4	5.8	13.0	67.2	82.6	57.9	90.1	60.64	205	4 124
Tallahassee city	118 534	950	-	.2	.2	15.4	8.2	13.6	21.6	50.5	81.3	38.1	91.3	65.71	1 705	2 518
Tamarac city	16 573	414	-	-	.5	28.6	11.4	3.8	11.2	66.2	82.5	56.8	91.2	59.15	262	4 038
Tampa city	343 731	1 226	-	-	1.1	14.6	5.9	8.8	29.8	59.9	82.1	50.0	90.2	43.92	726	3 396
Titusville city	18 949	481	-	-	.1	20.8	12.1	15.0	12.0	58.6	81.3	47.3	91.4	54.07	803	3 057
West Palm Beach city	82 516	1 220	-	-	2.3	20.9	9.1	4.8	10.3	65.1	82.2	55.7	89.9	60.75	323	3 891
GEORGIA	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Albany city	50 394	645	-	-	-	12.5	9.2	4.1	16.0	46.5	80.8	33.8	91.9	51.48	2 205	2 206
Athens city	23 629	517	-	.3	-	19.4	-	8.9	32.8	41.8	79.6	32.0	89.6	49.74	2 893	1 709
Atlanta city	572 897	1 454	2.9	.1	-	13.1	7.0	4.7	14.0	41.0	78.8	31.5	88.0	50.77	2 991	1 667
Augusta city	47 992	1 075	-	-	-	11.5	7.3	11.1	10.2	43.9	80.8	32.0	91.7	44.86	2 565	1 948
Columbus city (remainder)	107 068	599	-	5.3	2.6	17.1	9.1	5.2	16.4	45.7	81.9	35.2	91.8	51.00	2 261	2 284
East Point city	22 567	656	-	-	-	25.5	14.5	8.0	15.3	41.0	78.8	31.5	88.0	50.77	2 991	1 667
La Grange city	71 480	2 793	8.9	.2	63.5	6.0	3.5	3.1	6.8	43.4	79.0	31.9	90.1	54.52	2 667	1 696
Macon city	59 067	554	-	-	-	20.8	19.1	7.0	4.8	45.4	81.2	34.2	91.9	44.63	2 334	2 125
Marietta city	31 866	722	-	-	2.1	18.2	17.3	4.8	16.6	41.0	78.8	31.5	88.0	50.77	2 991	1 667
Rome city	24 622	812	Z	-	.4	12.4	20.0	9.6	20.1	38.4	76.9	27.0	87.3	55.33	3 467	1 337
Roswell city	34 343	717	-	-	-	17.4	3.5	3.1	13.9	41.0	78.8	31.5	88.0	50.77	2 991	1 667
Savannah city	116 375	846	-	.7	-	14.6	7.4	7.1	15.4	46.9	81.8	38.1	91.1	49.22	1 847	2 365
Smyrna city	31 069	1 003	-	-	-	13.9	11.6	6.6	17.8	41.0	78.8	31.5	88.0	50.77	2 991	1 667
Valdosta city	18 617	466	-	-	.7	19.0	14.8	6.2	26.1	49.0	80.7	36.7	92.1	52.24	1 844	2 350
Warner Robins city	18 026	412	-	-	-	21.0	14.0	7.6	28.5	45.4	81.2	34.2	91.9	44.63	2 334	2 125
HAWAII	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Hilo CDP	X	X	X	X	X	X	X	X	X							
Honolulu CDP	[7]820 933	[7]982	[7]-	[7]-	[7]1.2	[7]12.5	[7]5.0	[7]5.5	[7]22.4	71.7	75.8	63.6	83.0	129.19	-	3 284
Kailua CDP	X	X	X	X	X	X	X	X	X	71.4	78.9	62.4	87.7	21.53	-	3 845
Kaneohe CDP	X	X	X	X	X	X	X	X	X	71.3	77.0	65.5	82.2	78.91	-	3 482
Mililani Town CDP	X	X	X	X	X	X	X	X	X	71.3	77.0	65.5	82.2	79.91	-	3 482
Pearl City CDP	X	X	X	X	X	X	X	X	X	71.4	78.9	62.4	87.7	21.53	-	3 845
Waimalu CDP	X	X	X	X	X	X	X	X	X	71.4	78.9	62.4	87.7	21.53	-	3 845
Waipahu CDP	X	X	X	X	X	X	X	X	X	71.4	78.9	62.4	87.7	21.53	-	3 845
IDAHO	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Boise City city	72 163	574	-	-	.4	13.7	14.4	.7	20.4	29.0	74.0	21.6	90.2	12.11	5 861	754
Idaho Falls city	26 757	609	-	-	3.2	16.3	16.9	6.5	23.7	18.2	66.6	10.0	86.0	10.88	8 060	305
Lewiston city	15 509	552	-	-	-	12.5	15.1	8.7	41.7	33.6	74.1	27.6	89.0	12.43	5 270	814
Nampa city	12 338	435	-	-	-	16.2	11.4	10.0	23.2	29.0	74.0	21.6	90.2	12.11	5 861	754
Pocatello city	20 460	444	-	-	4.6	15.0	12.4	15.7	16.8	23.3	70.6	14.4	88.1	12.14	7 180	421
Twin Falls city	[8]10 810	[8]392	[8]-	[8]-	[8]-	[8]17.3	[8]11.7	[8]15.5	[8]24.0	26.9	68.8	18.6	85.0	10.40	6 769	379

[1]Based on resident population enumerated as of April 1, 1990. [2]Represents normal values based on 30-year period, 1961-1990; see text for more information. [3]Average daily minimum. [4]Average daily maximum. [5]One heating degree day is accumulated for each whole degree that the mean daily temperature is below 65 degrees Fahrenheit. [6]One cooling degree day is accumulated for each whole degree that the mean daily temperature is above 65 degrees Fahrenheit. [7]Data are for Honolulu City/County. [8]Data are for 1986-1987.

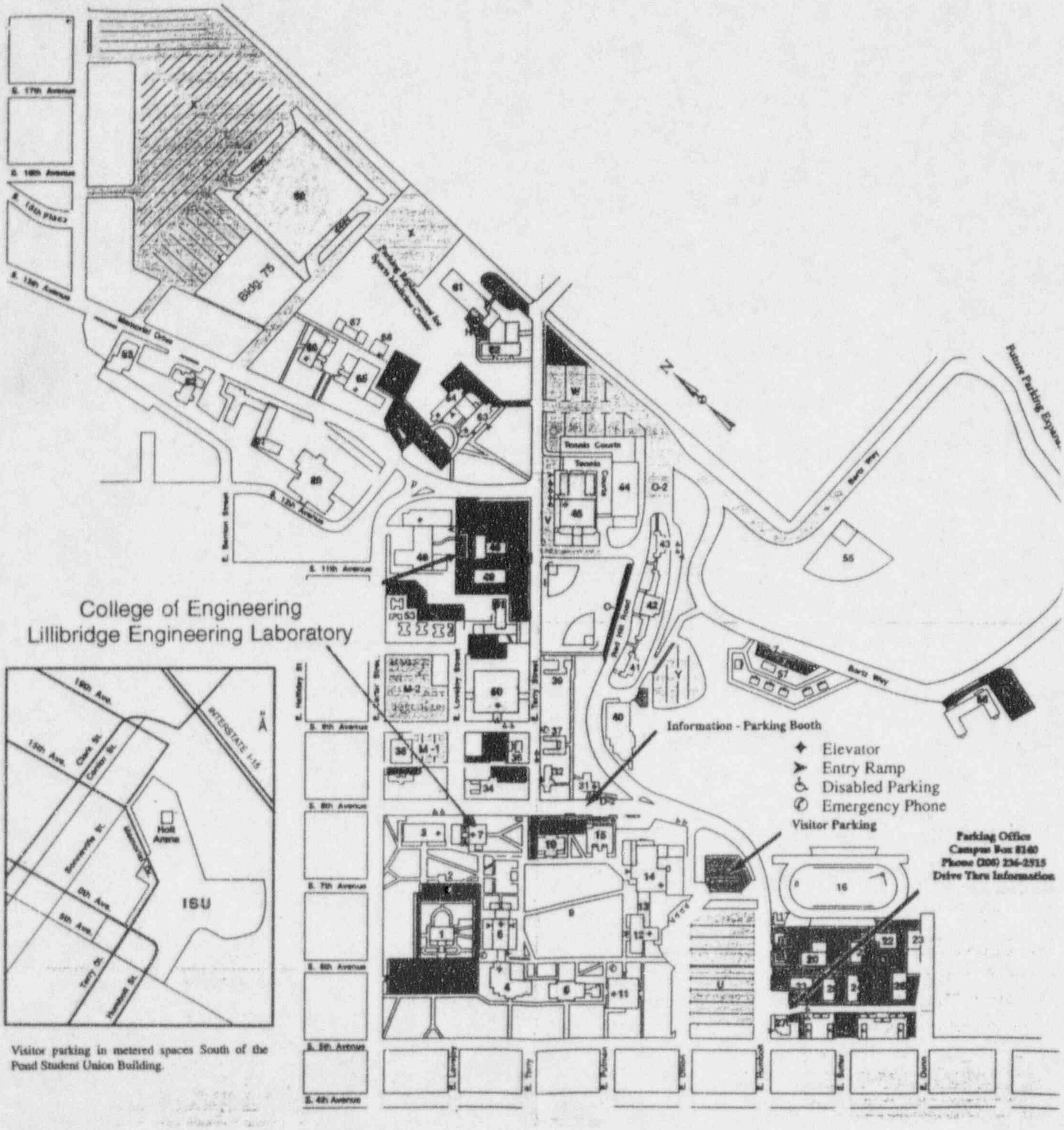


Figure 2.1-3 Campus of Idaho State University.

Table 2.1-3 Census Estimates

PCensus-USA | Desktop Demographics

Center for Business Research and Services -- I.S.U.
1994 Equifax/NDS Census Estimates
Prepared for the I.S.U. College of Engineering - 2/1/95

CENSUS UPDATE SUMMARY	College of Eng. (1 mile radius)	
1994 Estimated population (% base)	10,430	
1994 Population by age:		
0 to 4 years	678	7%
5 to 17 years	1,437	14%
18 to 24 years	2,163	21%
25 to 34 years	1,661	16%
35 to 44 years	1,479	14%
45 to 54 years	839	8%
55 to 64 years	643	6%
65 to 74 years	773	7%
75 years and over	757	7%
1994 Average age	35.4	
1994 Median age	31.4	
1994 Population by race		
White	9,697	93%
Black	182	2%
Asian	551	5%
Hispanic	534	5%
1994 Estimated households (% base)	4,748	
1994 Households by income		
$ 0 to $ 15,000	1,802	38%
$ 15,000 to $ 25,000	893	19%
$ 25,000 to $ 35,000	709	15%
$ 35,000 to $ 50,000	540	11%
$ 50,000 to $ 75,000	509	11%
$ 75,000 to $100,000	166	3%
$100,000 to $150,000	83	2%
$150,000 and over	46	1%
Average Household Income ($)	29,566	
Per Capita Income ($)	14,496	
1999 Projected population	10,792	
1999 Projected households	5,014	

2.2 Meteorology

2.2.1 Introduction

The climate of Pocatello is semiarid and may be described as a middle latitude steppe climate, where temperatures are relatively high in summer but fall to freezing in the winter and rainfall is sparse and characterized by great variability. Pocatello lies in a valley two to five miles wide (Figure 2.2-1), with mountains on either side rising three thousand to four thousand feet above the valley floor. About three miles north of the city center, the valley broadens and merges into the gently rolling topography of the of the Snake River Plain. The official Weather Bureau station is located on the southern margin of the Snake River Plain approximately seven miles northwest of town. Because Pocatello is situated in the Portneuf Valley between two spurs of the Bannock Range, weather data obtained at the Weather Bureau may not reflect the actual weather conditions in the city itself.

Records from a second-order weather station (one at which only temperature and rainfall data were recorded) which existed for about 8 years in the northeastern section of the city indicated that temperatures in Pocatello are two to three degrees higher than those at the official weather station except in winter when they may be more than 10 degrees higher. Except in summer, storms bring rain or snow to the whole area, but the location of Pocatello near to the mountains causes some variation in the amount and distribution of precipitation received at the Weather Bureau station. In summer, precipitation is usually produced by thunderstorms which are extremely localized.

2.2.2 Temperature

The average annual temperature of Pocatello is 47.2 °F and monthly temperatures average from 24°F in January to 71.4°F in July. Temperatures in July may reach 100°F or more for short periods, while January temperatures of -30°F have been recorded. The daily range of temperature is also high, reaching 30°F or more in July. This high diurnal range in summer is due to the fact that while daytime temperatures may reach 90°F or more, excessive re-radiation at night under cloudless skies causes the temperature to drop.

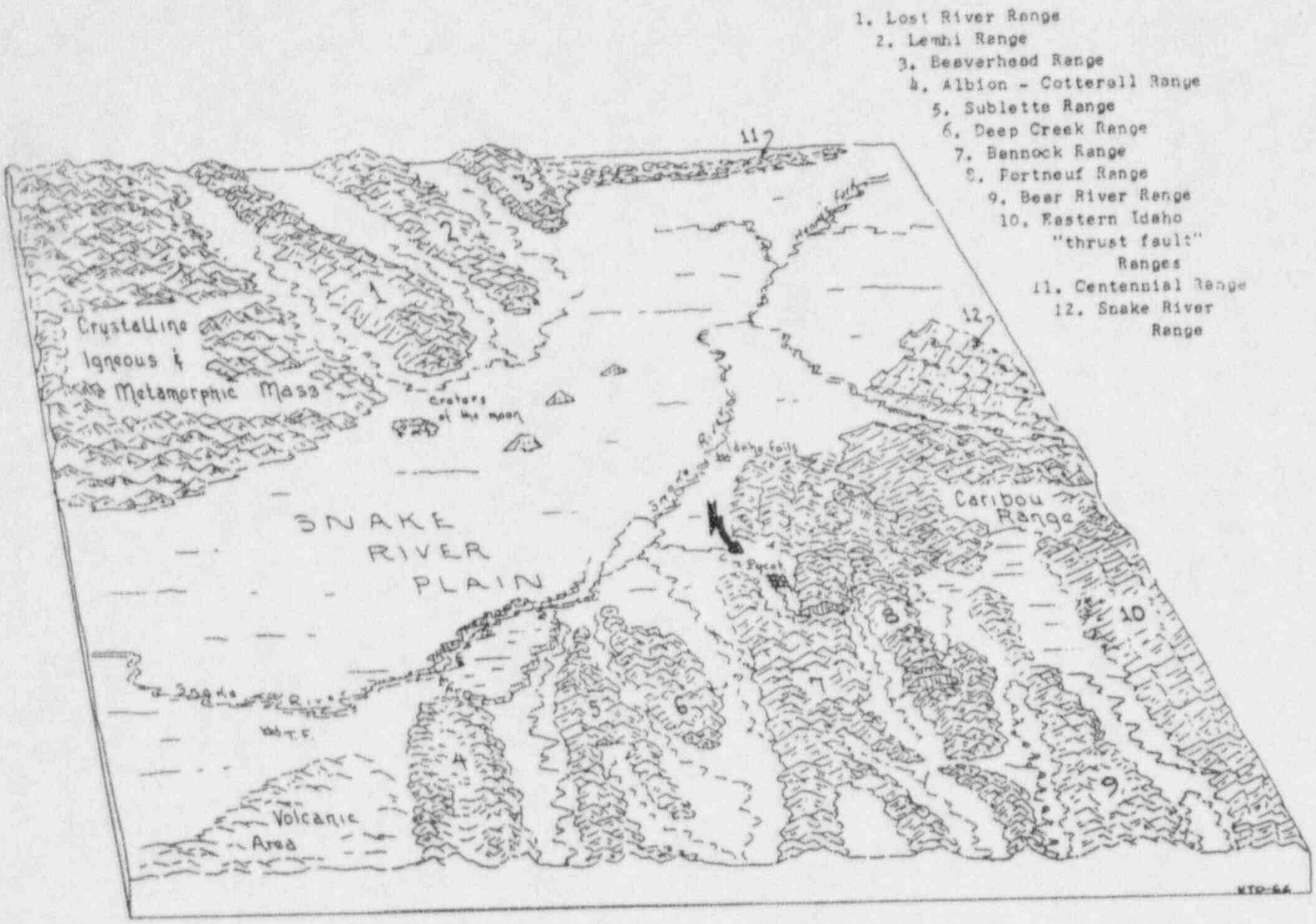


Figure 2.2-1 Physiographic location.

In winter, moist maritime air masses from the Pacific bring periods of mild weather when winds blow from the southwest, but otherwise temperatures may stay below freezing for several days and the daily minima approach or fall below zero. Frost depth to two or three feet is common during the winter season.

In the spring, temperatures gradually rise but freezing temperatures at night are general through most of April. The average first occurrence of 32 degrees Fahrenheit in the fall is September 20 and the average last occurrence in the spring is May 20. The first cold wave may appear during late November but usually not until late December.

2.2.3 Precipitation

Average annual precipitation for Pocatello as recorded at the U.S. Weather Bureau is 12.32 inches but the precipitation varies greatly in both amount and distribution. In the thirty year period between 1964 and 1993, seven years had less than 10 inches of precipitation, the minimum for any year being 5.34 inches in 1966. Six out of thirty years had precipitation over 14 inches, the maximum for one year being 20.33 inches in 1983.

During the winter, precipitation falling as snow sometimes accumulates to a depth of a foot or more, but snow depth on the valley floor (elevation about 4470 feet) reaches only 5 or 6 inches at most, and the snow usually melts after a few weeks, during a thaw.

The mountains surrounding Pocatello receive more moisture than the town itself and are covered with snow at higher elevations from November to May. Dry land wheat is raised on the hillsides near Pocatello where slopes are not too steep to prevent cultivation but agriculture can be carried on in the valley only by irrigation.

Precipitation is distributed unevenly through the year with 82% of the annual precipitation falling during the period of October through June and only 18% during the months of July, August and September. The fact that precipitation minima occur during the season of high temperatures when evaporation rates are also high, results in dry summers. During the summer, precipitation usually falls as local showers accompanied by light-to-moderate thunderstorms, and occasionally by hail. Damage by cloudbursts is rare in Pocatello because contour furrowing done by the Civilian Conservation Corps in the 1930's on the hill slopes above the city has prevented excessive runoff. Cloudy and unsettled weather prevails throughout the winter and spring with measurable amounts of precipitation on about one-third of the days.

2.2.4 Wind

Pocatello lies in the belt of westerlies, consequently, the prevailing wind direction is from the southwest. Average wind speeds are 9 to 10 miles per hour but on rare occasions, during heavy winds, gusts up to 68 miles per hour have been recorded. Winds of 20 to 30 miles per hour may blow continuously for several days in the spring.

Windstorms associated with cyclonic systems and cold fronts do some damage to trees each year, often causing temporary disruption of power and communication facilities; only minor damage is done to structures. Storms of this type may occur from October to June, while during the remaining three months of the year, high winds are almost invariably associated with thunderstorms.

No official wind-recording instruments are located in Pocatello or in the valley adjacent to it. Movement of smoke from the stacks of two chemical plants located on the northern outskirts of the city indicates that, at times, air may be moving up-valley on the western side of the valley and down-valley during the early morning hours along the eastern side. Winds also appear to blow down-valley during the early morning hours along the eastern side of the valley. These mountain and valley winds are light; velocities are probably not more than 1-3 miles per hour.

2.2.5 Other Climatic Factors

Relative Humidity - Relative humidity is higher in winter and spring and during these seasons is near 70 to 80 per cent. During the summer months, relative humidity is never greater than 50 per cent during the day, not exceeding 65 per cent at night.

Fog - The Weather Bureau records an average of 10 days of heavy fog per year for Pocatello, and nearly half of these (four) come in January. The valley and mountain winds tend to prevent the formation of fog in the valley and so occurrences of fog in the city itself averages considerably less than 10 days per year.

Sunshine - Sunny skies prevail over Pocatello during the summer when roughly 50 per cent of the days in July, August, and September are cloudless, and possible sunshine rises to 80 per cent. Cloudiness increases in winter and spring. December and January are the cloudiest months, when the sky is more than eight-tenths covered about two-thirds of the time.

2.2.6 Adverse weather effects on the ISU AGN-201 Reactor

Flood

In the unlikely event of a flood, no special precautions are necessary other than those normally taken in the event of a flood at an industrial site. The reactor will be secured and not operated at this time. The radiological hazard problems are not severe as the reactor is built to withstand a minor flood (one foot of water). In the event of a major flood where the reactor might be overturned or carried away, there is again no serious problem since the self-contained reactor has been designed to withstand such an emergency.

Storm

It is highly unlikely that a storm could damage the AGN-201 reactor; however, in the event of a severe storm, the reactor will be shut down and secured. It should also be noted that there is no recorded history of tornadoes in Pocatello, Idaho.

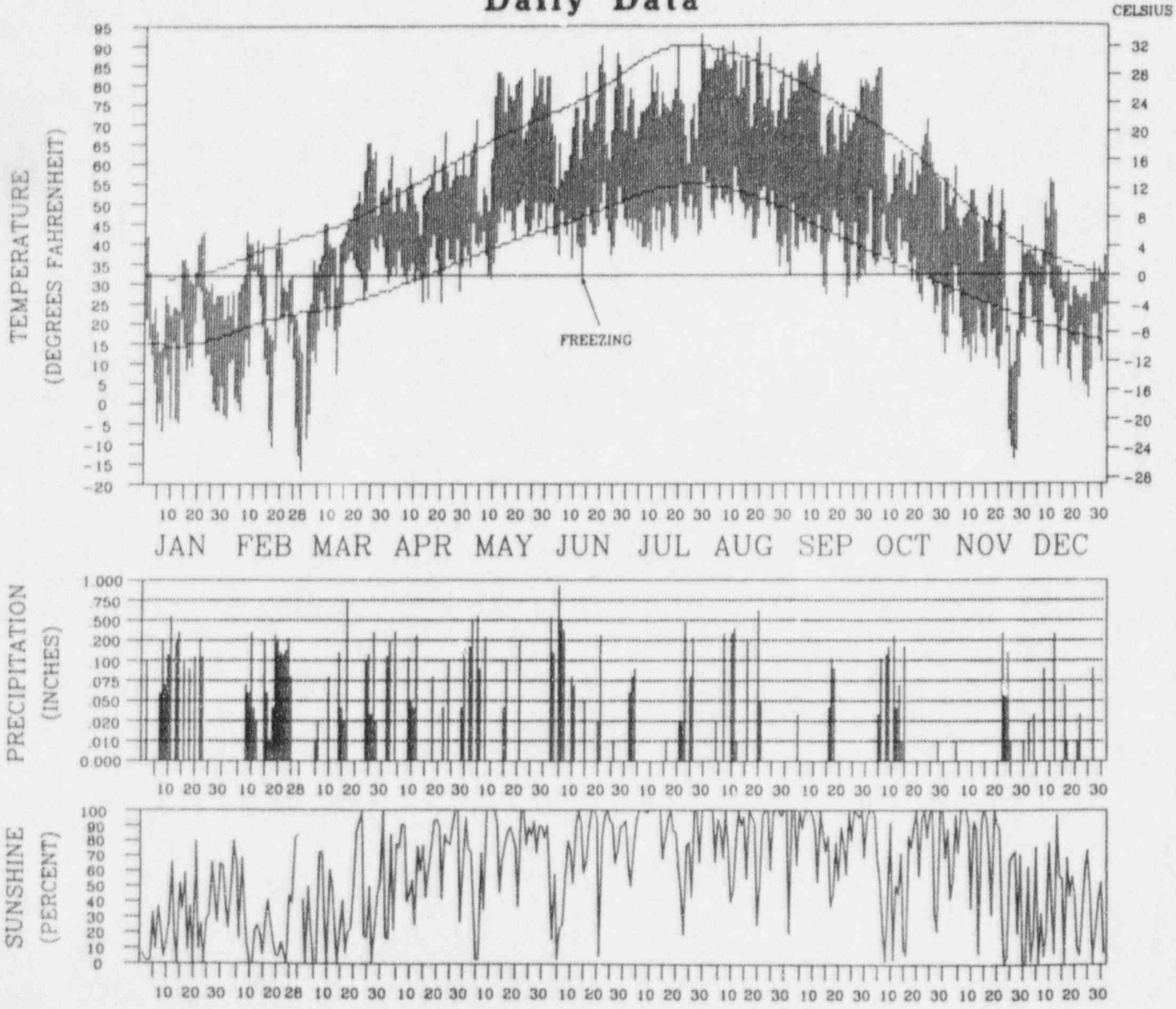
Local climatological and meteorological data from the Pocatello Weather Bureau station is given in Tables 2.2-1 through 2.2-6.

Table 2.2-1 Local and Climatological Data

ISSN 0198-1803

1993 LOCAL CLIMATOLOGICAL DATA
ANNUAL SUMMARY WITH COMPARATIVE DATA
POCATELLO,
IDAHO

Daily Data



TEMPERATURE DEPICTS NORMAL MAXIMUM, NORMAL MINIMUM AND ACTUAL DAILY HIGH AND LOW VALUES (FAHRENHEIT)
PRECIPITATION IS MEASURED IN INCHES, SCALE IS NON-LINEAR
SUNSHINE IS PERCENT OF THE POSSIBLE SUNSHINE

I CERTIFY THAT THIS IS AN OFFICIAL PUBLICATION OF THE NATIONAL OCEANIC AND ATMOSPHERIC ADMINISTRATION, AND IS COMPILED FROM RECORDS ON FILE AT THE NATIONAL CLIMATIC DATA CENTER, ASHEVILLE, NORTH CAROLINA, 28801

noaa NATIONAL OCEANIC AND ATMOSPHERIC ADMINISTRATION — NATIONAL ENVIRONMENTAL SATELLITE, DATA AND INFORMATION SERVICE — NATIONAL CLIMATIC DATA CENTER ASHEVILLE NORTH CAROLINA

Kenneth D Hadeen
DIRECTOR
NATIONAL CLIMATIC DATA CENTER

Table 2.2-2 Meteorological Data for 1993

METEOROLOGICAL DATA FOR 1993

POCATELLO, IDAHO

LATITUDE: 42°55' N LONGITUDE: 112°36' W ELEVATION: FT. GRND 4454 BARO 4464 TIME ZONE: MOUNTAIN WBAN: 24156

	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEP	OCT	NOV	DEC	YEAR
TEMPERATURE °F:													
Averages													
-Daily Maximum	28.2	29.7	45.7	55.1	71.8	71.3	76.3	81.9	76.2	62.1	41.3	35.1	56.2
-Daily Minimum	8.0	11.3	26.4	33.7	43.3	43.8	45.9	47.0	38.8	32.7	13.8	17.1	30.2
-Monthly	18.1	20.5	36.1	44.4	57.6	57.6	61.1	64.5	57.5	47.4	27.6	26.1	43.2
-Monthly Dewpt.	13.0	14.7	28.1	30.1	39.9	42.0	41.8	43.8	32.5	32.8	14.1	20.0	29.4
Extremes													
-Highest	43	44	65	68	84	90	93	92	88	84	59	56	93
-Date	22	19	24	21	25	20	28	19	10	5	3	10	JUL 28
-Lowest	-7	-17	-9	25	31	32	38	32	26	12	-14	1	-17
-Date	7	28	1	20	9	13	13	31	22	30	26	25	FEB 28
DEGREE DAYS BASE 65 °F:													
Heating	1445	1240	890	612	226	235	136	67	219	539	1115	1200	7924
Cooling	0	0	0	0	3	16	24	57	2	1	0	0	103
% OF POSSIBLE SUNSHINE	29	30	38	66	75	70	83	84	81	67	66	36	63
AVG. SKY COVER (tenths)													
Sunrise - Sunset	7.9	7.8	8.4	7.6	6.3	5.2	3.5	3.4	4.4	6.5	6.2	8.8	6.3
Midnight - Midnight	8.1	7.9	8.1	7.1	6.1	5.0	3.3	3.4	3.9	5.8	5.4	8.3	6.0
NUMBER OF DAYS:													
Sunrise to Sunset													
-Clear	3	3	1	5	6	10	18	19	14	6	8	0	93
-Partly Cloudy	7	7	7	6	11	12	7	6	8	9	8	6	94
-Cloudy	21	18	23	19	14	8	6	6	8	16	14	25	178
Precipitation .01 inches or more	13	16	12	12	8	11	9	8	4	10	6	9	118
Snow, Ice pellets, hail 1.0 inches or more	11	9	0	2	0	0	0	0	0	0	2	3	27
Thunderstorms	0	0	2	2	4	7	4	7	3	2	0	0	31
Heavy Fog, visibility 1/4 mile or less	10	9	5	0	0	0	0	0	0	1	3	5	33
Temperature °F													
-Maximum													
90° and above	0	0	0	0	0	1	2	3	0	0	0	0	6
32° and below	22	18	3	0	0	0	0	0	0	0	6	13	62
-Minimum													
32° and below	31	25	18	12	2	1	0	1	7	15	29	30	171
0° and below	11	7	2	0	0	0	0	0	0	0	4	0	24
AVG. STATION PRESS. (mb)	862.9	862.5	865.2	861.8	861.2	861.5	862.5	864.2	864.7	866.2	866.2	866.9	863.9
RELATIVE HUMIDITY (%)													
Hour 05	82	83	87	78	79	80	75	78	68	82	73	85	79
Hour 11 (Local Time)	75	76	69	55	47	52	45	40	32	53	54	76	56
Hour 17	76	74	64	44	35	45	34	29	25	44	50	76	50
Hour 23	83	83	82	66	63	64	62	64	49	70	65	83	70
PRECIPITATION (inches):													
Water Equivalent													
-Total	2.30	2.04	1.68	1.54	1.93	2.95	1.07	1.90	0.27	1.01	0.54	0.65	17.88
-Greatest (24 hrs)	0.69	0.41	0.79	0.33	0.59	1.18	0.47	0.67	0.15	0.31	0.31	0.30	1.18
-Date	10-11	19-20	16-17	4	5- 6	5- 6	23	20-21	16-17	11-12	22-23	11-12	JUN 5- 6
Snow, Ice pellets, hail													
-Total	29.6	21.3	1.5	5.5	0.2	T	0.0	T	0.0	0.0	6.2	6.3	70.6
-Greatest (24 hrs)	8.1	4.1	0.9	3.0	0.2	T	0.0	T	0.0	0.0	3.0	2.5	8.1
-Date	10-11	21-22	10	12	8	11		10			22-23	12	JAN 10-11
WIND:													
Resultant													
-Direction (!!)	205	256	255	241	233	248	245	230	240	248	220	213	236
-Speed (mph)	2.6	1.6	3.0	7.7	4.1	5.3	6.3	2.8	3.5	3.2	5.5	4.0	4.0
Average Speed (mph)	8.8	8.3	8.4	11.6	9.4	10.0	9.9	8.8	8.4	8.8	9.3	8.7	9.2
Fastest Mile													
-Direction (!!)	27	29	15	28	30	28	15	15	23	26	25	24	30
-Speed (mph)	40	46	32	32	46	32	28	31	28	24	30	30	46
-Date	22	20	26	1	3	27	29	11	12	24	3	4	MAY 3
Peak Gust													
-Direction (!!)	W	W	S	W	W	W	SE	SE	SW	W	SW	W	W
-Speed (mph)	63	56	40	48	56	49	38	45	38	38	44	45	63
-Date	22	20	23	1	3	27	29	11	12	5	18	4	JAN 22

Table 2.2-3 Normals, Means and Extremes

NORMALS, MEANS, AND EXTREMES

POCATELLO, IDAHO

ATITUDE: 42°55'N LONGITUDE: 112°36'W ELEVATION: FT. GRND 4454 BARO 4464 TIME ZONE: MOUNTAIN WBAN: 24156

	(a)	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEP	OCT	NOV	DEC	YEAR
TEMPERATURE °F:														
Normals														
-Daily Maximum		32.2	38.4	46.7	57.5	67.5	78.0	88.1	86.3	75.1	62.5	45.2	33.7	59.3
-Daily Minimum		14.4	19.8	25.9	32.3	39.6	47.3	53.0	50.9	42.8	33.5	26.0	15.8	33.4
-Monthly		23.3	29.1	36.3	44.9	53.6	62.7	70.6	68.7	59.0	48.0	35.6	24.8	46.4
Extremes														
-Record Highest	44	57	65	75	86	93	103	102	104	98	91	71	59	104
-Year		1974	1992	1986	1992	1954	1988	1991	1990	1976	1992	1975	1980	AUG 1990
-Record Lowest	44	-30	-33	-12	15	20	30	34	30	19	10	-14	-29	-33
-Year		1962	1985	1985	1970	1972	1989	1981	1992	1985	1971	1993	1990	FEB 1985
NORMAL DEGREE DAYS:														
Heating (base 65°F)		1293	1005	890	603	353	125	9	29	218	527	882	1246	7180
Cooling (base 65°F)		0	0	0	0	0	56	183	144	38	0	0	0	421
% OF POSSIBLE SUNSHINE	44	40	53	61	66	68	75	83	81	79	71	47	40	64
MEAN SKY COVER (tenths) Sunrise - Sunset	44	8.0	7.5	7.1	6.7	6.2	4.9	3.5	3.8	4.1	5.1	7.2	7.9	6.0
MEAN NUMBER OF DAYS:														
Sunrise to Sunset														
-Clear	44	2.7	4.0	5.0	6.3	7.5	11.7	17.4	15.4	15.2	12.0	5.1	3.4	105.5
-Partly Cloudy	44	6.7	6.4	8.2	7.9	9.9	9.6	9.0	10.6	8.1	8.4	7.2	6.8	98.8
-Cloudy	44	21.7	17.9	17.8	15.8	13.6	8.7	4.6	5.0	6.8	10.6	17.7	20.8	160.9
Precipitation .01 inches or more	44	12.0	10.3	10.2	8.3	9.2	6.9	4.2	4.5	4.7	5.3	9.2	10.9	95.8
Snow, Ice pellets, hail 1.0 inches or more	44	3.3	2.1	2.1	1.3	0.2	0.0	0.0	0.0	0.*	0.5	1.9	3.0	14.4
Thunderstorms	43	0.*	0.1	0.4	1.0	3.4	4.7	5.6	5.1	2.7	0.5	0.1	0.1	23.8
Heavy Fog Visibility 1/4 mile or less	43	4.6	3.3	1.6	0.4	0.3	0.*	0.*	0.0	0.1	0.4	1.9	4.3	16.9
Temperature °F														
-Maximum														
90° and above	30	0.0	0.0	0.0	0.0	0.0	4.2	14.4	12.3	1.9	0.*	0.0	0.0	32.8
32° and below	30	15.0	7.5	2.0	0.*	0.0	0.0	0.0	0.0	0.0	0.2	4.0	12.7	41.4
-Minimum														
32° and below	30	28.1	25.1	24.5	15.6	4.2	0.3	0.0	0.1	3.3	14.7	23.4	28.5	167.9
0° and below	30	5.6	2.2	0.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.9	3.9	12.8
AVG. STATION PRESS.(mb)	20	865.6	864.4	860.9	861.4	861.1	862.3	863.7	863.9	864.3	865.3	864.7	865.5	863.6
RELATIVE HUMIDITY (%)														
Hour 05	30	79	80	77	70	70	71	65	62	65	70	77	79	72
Hour 11 (Local Time)	30	75	70	60	48	44	42	35	34	38	47	66	75	53
Hour 17	30	71	62	51	38	34	32	24	23	28	37	60	70	44
Hour 23	27	78	76	69	58	56	54	47	44	50	58	73	78	62
PRECIPITATION (inches):														
Water Equivalent														
-Normal		1.04	0.92	1.26	1.20	1.35	1.02	0.65	0.67	0.85	0.91	1.16	1.11	12.14
-Maximum Monthly	44	3.24	2.63	2.95	3.30	3.29	3.30	2.28	3.98	3.43	2.56	2.84	3.39	3.98
-Year		1980	1986	1983	1963	1980	1967	1984	1968	1982	1956	1983	1983	AUG 1968
-Minimum Monthly	44	0.21	0.12	0.10	0.06	0.19	0.02	T	T	0.00	0.00	0.01	0.07	0.00
-Year		1992	1970	1965	1977	1969	1974	1988	1958	1987	1988	1976	1989	OCT 1988
-Maximum in 24 hrs	44	0.97	0.67	1.27	1.25	1.67	1.18	0.98	1.16	1.13	1.82	0.85	0.94	1.82
-Year		1970	1983	1990	1976	1970	1993	1965	1972	1982	1976	1969	1983	OCT 1976
Snow, Ice pellets, hail														
-Maximum Monthly	44	29.6	21.3	16.6	15.5	5.5	0.2	0.0	T	2.0	12.6	27.5	33.7	33.7
-Year		1993	1993	1985	1976	1983	1981		1993	1965	1971	1985	1983	DEC 1983
-Maximum in 24 hrs	44	10.1	6.4	8.4	10.0	5.2	0.2	0.0	T	2.0	8.0	7.8	10.8	10.8
-Year		1950	1984	1985	1976	1983	1981		1993	1965	1980	1992	1988	DEC 1988
WIND:														
Mean Speed (mph)	41	10.6	10.6	11.2	11.7	10.6	10.1	9.1	8.9	9.1	9.3	10.3	9.9	10.1
Prevailing Direction through 1963		SW	SW	SW	SW	SW	SW	SW	SW	SW	SW	SW	SW	SW
Fastest Mile														
-Direction (!!!)	44	SE	W	W	S	W	W	W	SW	W	SW	W	NW	W
-Speed (MPH)	44	61	57	72	61	61	50	57	54	57	54	67	57	72
-Year		1952	1963	1955	1956	1953	1955	1968	1966	1961	1966	1955	1981	MAR 1955
Peak Gust														
-Direction (!!!)	10	W	S	SW	SW	SW	SW	W	SW	NW	S	S	S	W
-Speed (mph)	10	68	60	64	55	61	55	66	68	51	51	58	54	68
-Date		1990	1985	1987	1988	1991	1986	1986	1985	1989	1991	1988	1987	JAN 1990

Table 2.2-4 Precipitation & Average Temperature

PRECIPITATION (inches) POCATELLO, IDAHO

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEP	OCT	NOV	DEC	ANNUAL
1964	0.76	0.34	0.78	1.46	1.41	2.08	0.30	0.24	T	0.50	0.85	2.95	11.67
1965	0.71	0.38	0.10	1.65	0.84	1.12	1.19	0.78	0.85	0.02	1.21	1.20	10.05
1966	0.27	0.51	0.42	0.31	0.93	0.21	0.10	0.02	0.82	0.40	0.63	0.72	5.34
1967	1.08	0.18	1.21	1.81	1.14	3.30	0.22	0.13	0.14	0.80	0.51	0.91	11.43
1968	0.75	1.11	1.29	0.39	1.70	1.84	0.10	3.98	0.64	0.58	0.92	0.78	14.08
1969	1.80	0.94	0.40	0.34	0.19	2.02	0.14	0.54	0.14	0.56	0.87	1.10	9.04
1970	1.98	0.12	1.11	1.28	2.03	1.18	1.25	0.11	0.80	0.68	1.60	1.13	13.27
1971	1.47	0.62	1.90	2.33	0.50	1.26	0.32	0.58	2.06	2.29	1.81	1.69	16.83
1972	1.45	0.92	0.61	1.36	0.54	1.29	0.56	1.36	1.14	1.39	0.44	1.89	12.95
1973	1.04	1.00	1.45	0.70	0.44	0.87	1.84	0.13	2.29	1.19	1.87	0.84	13.66
1974	1.55	0.66	1.66	1.40	1.28	0.02	0.14	0.15	0.07	1.99	0.77	0.93	10.62
1975	0.65	1.51	1.71	1.51	1.64	0.73	1.61	0.02	0.03	2.54	0.95	0.59	13.49
1976	0.45	1.42	1.08	2.82	0.51	0.98	0.77	0.74	0.45	1.83	0.01	0.20	11.26
1977	0.70	0.37	0.79	0.06	2.09	0.77	0.58	0.32	0.99	0.18	0.94	1.08	8.87
1978	1.17	1.07	1.16	1.24	1.99	0.22	0.03	0.41	0.87	0.06	1.39	0.74	10.35
1979	1.09	1.17	0.79	0.63	0.33	0.96	0.59	0.70	0.23	0.79	1.12	0.40	8.80
1980	3.24	1.12	1.52	0.55	3.29	0.86	0.72	1.25	1.34	1.88	1.13	0.37	17.27
1981	0.72	0.47	1.69	1.27	3.24	0.68	0.29	0.32	0.08	1.52	1.67	2.22	14.17
1982	1.40	0.91	1.98	1.02	1.06	0.47	1.27	2.02	3.43	0.99	1.17	2.00	17.72
1983	0.48	1.12	2.95	0.72	1.99	1.19	1.13	1.67	1.64	1.21	2.84	3.39	20.33
1984	0.49	2.06	0.63	2.13	0.69	1.33	2.28	0.71	0.50	0.89	0.96	0.53	13.20
1985	0.71	0.96	1.55	0.29	1.27	1.04	0.48	0.05	1.42	0.50	2.37	1.17	11.81
1986	1.07	2.63	0.87	2.56	1.90	0.33	0.12	0.14	1.33	0.39	0.89	0.20	12.43
1987	1.03	0.71	0.84	0.47	2.02	0.67	1.93	0.66	0.00	0.25	0.44	1.21	10.23
1988	1.06	0.21	0.83	0.97	0.78	0.37	T	0.20	0.04	0.00	2.27	1.19	7.92
1989	0.59	1.36	2.91	0.33	0.76	0.62	0.11	0.43	0.48	0.84	1.03	0.07	9.53
1990	0.53	0.24	2.28	1.43	1.37	0.75	0.14	0.70	0.33	0.31	0.92	1.15	10.15
1991	0.71	0.28	2.25	2.56	3.02	0.59	0.12	0.85	0.66	0.74	1.65	0.22	13.65
1992	0.21	0.65	0.47	0.72	0.25	1.34	0.71	0.37	0.08	0.98	1.37	1.82	8.97
1993	2.30	2.04	1.68	1.54	1.93	2.95	1.07	1.90	0.27	1.01	0.54	0.65	17.88
Record Mean	1.18	1.01	1.23	1.27	1.36	1.05	0.65	0.70	0.81	0.98	1.03	1.06	12.32

See Reference Notes on Page 6B
Page 4A

AVERAGE TEMPERATURE (deg. F) POCATELLO, IDAHO

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEP	OCT	NOV	DEC	ANNUAL
1964	19.2	16.9	27.8	42.7	53.6	60.0	72.6	67.5	57.3	49.3	35.0	29.1	44.2
1965	29.8	30.8	32.5	46.8	50.5	60.6	68.6	65.8	51.8	50.6	40.6	25.6	46.1
1966	25.7	26.5	37.1	44.2	56.9	62.2	72.2	67.6	61.7	44.6	37.7	24.0	46.7
1967	29.8	33.1	37.5	40.7	52.9	59.5	71.5	71.0	61.7	47.3	35.4	19.7	46.7
1968	21.2	32.6	39.5	41.7	51.7	61.1	71.6	63.0	56.5	46.0	33.4	24.8	45.3
1969	29.1	25.7	29.2	44.9	58.1	59.9	69.7	70.5	62.1	42.7	35.5	28.5	46.3
1970	29.9	37.2	36.1	39.0	53.6	63.1	71.0	72.1	53.7	43.0	37.6	25.0	46.8
1971	27.2	27.8	32.6	42.6	52.7	60.2	68.9	71.1	52.7	42.8	32.7	22.6	44.5
1972	22.9	29.0	41.0	43.0	53.8	62.3	68.6	69.5	55.6	47.8	36.0	18.3	45.6
1973	19.8	27.0	35.9	43.4	55.7	63.4	70.6	69.8	57.3	48.7	36.7	29.6	46.5
1974	22.5	29.8	38.5	45.5	53.3	66.6	71.4	66.8	59.4	48.6	37.1	24.7	47.0
1975	22.4	28.3	35.5	39.9	50.8	61.5	73.8	66.6	60.0	48.1	33.4	32.5	46.1
1976	26.9	29.4	32.6	44.3	57.7	61.6	72.0	65.9	61.1	46.8	37.7	28.4	47.0
1977	17.1	27.5	34.9	49.9	51.7	68.9	70.8	68.9	59.7	50.4	37.1	32.9	47.5
1978	31.1	33.6	42.9	45.9	51.0	61.8	70.1	66.9	58.1	49.5	31.8	20.0	46.9
1979	10.7	27.9	36.1	44.0	54.4	61.7	71.4	69.2	64.7	51.1	28.4	29.3	45.7
1980	24.7	34.5	36.2	48.0	51.4	59.2	68.6	65.2	58.9	46.2	34.7	31.7	46.6
1981	28.5	29.5	39.0	47.5	51.6	61.8	69.6	72.0	61.4	45.4	38.1	30.8	48.0
1982	20.2	23.5	37.7	41.3	51.8	61.8	67.9	70.8	58.3	45.2	31.9	25.3	44.7
1983	31.5	34.0	40.3	42.3	51.5	60.7	67.0	71.1	60.2	49.3	36.3	19.9	47.0
1984	17.7	20.3	34.2	42.8	54.5	59.0	70.1	70.3	58.3	43.8	34.8	19.6	43.8
1985	12.9	18.5	27.5	49.0	56.9	64.1	73.0	66.2	54.2	45.5	26.7	11.4	42.2
1986	23.4	35.5	44.2	45.0	52.8	67.4	67.1	70.5	54.1	48.0	36.0	24.5	47.4
1987	19.2	33.6	39.3	51.5	57.7	65.2	67.7	67.2	60.7	50.4	36.3	26.0	47.9
1988	21.3	32.8	37.6	48.6	54.2	69.6	73.3	68.7	58.0	55.3	35.3	21.8	48.0
1989	19.4	18.9	38.3	49.2	53.4	62.1	73.1	66.8	59.6	46.8	37.0	26.7	46.0
1990	30.0	28.3	40.0	50.1	51.8	62.5	70.9	69.0	65.7	47.8	37.7	14.8	47.4
1991	19.4	36.7	38.5	44.4	51.1	61.1	70.9	72.4	60.0	47.6	32.8	26.4	46.8
1992	26.0	38.0	44.1	51.3	60.2	65.9	67.4	69.6	59.3	51.4	28.8	22.1	48.5
1993	18.1	20.5	36.1	44.4	57.6	57.6	61.1	64.5	57.5	47.4	27.6	26.1	43.2
Record Mean	24.0	29.3	37.0	46.0	54.4	62.8	71.4	69.6	59.7	48.8	36.1	26.6	47.2
Max	32.5	38.0	46.8	57.8	67.5	77.3	87.7	85.7	75.0	62.3	46.0	35.0	59.3
Min	15.5	20.6	27.1	34.1	41.3	48.3	55.1	53.4	44.3	35.3	26.2	18.3	35.0

Table 2.2-5 Heating Degree Days & Cooling Degree Days

HEATING DEGREE DAYS Base 65 deg. F POCATELLO, IDAHO

SEASON	JULY	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	TOTAL
1964-65	0	68	223	478	894	1105	1082	952	999	543	443	143	6930
1965-66	16	60	393	440	725	1216	1211	1072	861	618	253	140	7005
1966-67	2	26	133	623	812	1263	1083	888	846	720	372	178	6946
1967-68	0	3	128	545	883	1398	1352	931	783	689	405	154	7271
1968-69	9	143	260	581	942	1240	1105	1094	1104	594	223	167	7462
1969-70	24	17	105	685	878	1124	1080	772	888	771	348	141	6833
1970-71	4	2	333	675	817	1236	1166	1034	997	666	376	168	7474
1971-72	19	13	374	684	964	1309	1300	1039	735	654	346	101	7538
1972-73	30	9	281	527	860	1445	1394	1057	895	643	290	135	7566
1973-74	12	18	237	500	843	1089	1313	979	815	580	359	78	6823
1974-75	11	42	173	500	829	1246	1312	1021	908	745	435	122	7344
1975-76	11	40	168	521	943	1001	1172	1026	999	615	219	140	6855
1976-77	1	34	143	561	811	1127	1478	1043	926	445	406	6	6981
1977-78	1	38	189	447	831	990	1045	875	678	567	431	119	6211
1978-79	10	59	241	473	990	1387	1678	1034	887	623	320	155	7857
1979-80	1	10	66	422	1090	1100	1245	876	888	507	415	184	6804
1980-81	5	57	182	576	899	1026	1125	988	801	516	407	127	6709
1981-82	16	9	130	603	800	1051	1382	1161	838	703	406	139	7238
1982-83	38	0	216	607	987	1227	1029	863	760	675	415	141	6958
1983-84	61	3	163	478	854	1392	1461	1291	949	658	333	205	7848
1984-85	5	7	221	648	897	1400	1612	1298	1157	472	246	85	8048
1985-86	0	44	321	597	1143	1657	1281	818	636	592	389	35	7513
1986-87	28	5	326	519	864	1247	1410	875	792	398	231	64	6759
1987-88	46	31	144	445	854	1203	1348	926	843	485	340	38	6703
1988-89	1	3	229	295	888	1330	1408	1292	820	469	355	113	7203
1989-90	0	47	159	556	831	1183	1079	1022	766	438	399	139	6619
1990-91	3	20	53	526	813	1555	1406	786	813	610	425	122	7132
1991-92	3	1	176	533	960	1188	1201	776	639	407	154	97	6135
1992-93	23	63	175	417	1078	1323	1445	1240	890	612	226	235	7727
1993-94	136	67	219	539	1115	1200							

See Reference Notes on Page 68.

COOLING DEGREE DAYS Base 65 deg. F POCATELLO, IDAHO

YEAR	JAN	FEB	MAR	APR	MAY	JUNE	JULY	AUG	SEP	OCT	NOV	DEC	TOTAL
1969	0	0	0	0	15	23	177	195	25	0	0	0	435
1970	0	0	0	0	0	91	197	230	4	0	0	0	522
1971	0	0	0	0	0	29	146	212	9	0	0	0	396
1972	0	0	0	0	7	26	150	152	6	0	0	0	341
1973	0	0	0	0	7	94	194	174	14	0	0	0	483
1974	0	0	0	0	1	130	215	102	12	0	0	0	460
1975	0	0	0	0	2	23	290	98	23	4	0	0	440
1976	0	0	0	0	1	46	227	70	28	0	0	0	372
1977	0	0	0	0	1	129	190	164	35	0	0	0	519
1978	0	0	0	0	0	28	177	123	41	0	0	0	369
1979	0	0	0	0	0	64	207	145	63	0	0	0	479
1980	0	0	0	3	0	17	125	70	6	0	0	0	221
1981	0	0	0	0	0	38	168	232	31	0	0	0	469
1982	0	0	0	0	2	48	133	187	16	0	0	0	386
1983	0	0	0	0	4	19	131	199	25	0	0	0	378
1984	0	0	0	0	15	33	168	178	25	0	0	0	419
1985	0	0	0	0	5	65	257	90	3	0	0	0	420
1986	0	0	0	0	17	110	100	185	4	0	0	0	416
1987	0	0	0	0	8	74	134	104	23	0	0	0	343
1988	0	0	0	0	8	181	261	123	25	0	0	0	598
1989	0	0	0	1	4	34	255	109	10	0	0	0	413
1990	0	0	0	0	0	68	194	152	79	0	0	0	493
1991	0	0	0	0	0	12	192	237	32	0	0	0	473
1992	0	0	0	0	15	73	103	212	11	2	0	0	416
1993	0	0	0	0	3	16	24	57	2	1	0	0	103

Table 2.2-6 Snowfall

SNOWFALL (inches)

POCATELLO, IDAHO

SEASON	JULY	AUG	SEP	OCT	NOV	DEC	JAN	FEB	MAR	APR	MAY	JUNE	TOTAL
1964-65	0.0	0.0	0.0	0.0	3.7	6.9	4.7	3.7	0.9	1.2	0.4	0.0	21.5
1965-66	0.0	0.0	2.0	0.0	2.0	12.1	4.8	8.7	1.5	1.3	T	0.0	32.4
1966-67	0.0	0.0	0.0	1.1	2.5	5.5	3.3	3.5	5.7	13.9	0.3	0.0	35.8
1967-68	0.0	0.0	0.0	T	4.4	13.9	11.6	4.7	5.9	1.1	T	0.0	41.6
1968-69	0.0	0.0	T	0.0	5.1	17.9	7.7	15.2	5.7	T	0.0	0.0	51.6
1969-70	0.0	0.0	0.0	T	T	7.1	1.9	0.5	9.5	11.5	T	0.0	30.5
1970-71	0.0	0.0	T	0.1	1.4	9.9	13.1	5.9	5.2	0.3	T	T	35.9
1971-72	0.0	0.0	T	12.6	9.6	17.6	10.6	6.9	1.0	0.9	0.0	0.0	59.2
1972-73	0.0	0.0	0.0	0.5	2.7	17.4	12.9	7.0	8.7	4.1	0.0	0.0	53.3
1973-74	0.0	0.0	0.0	2.0	8.7	5.9	8.8	5.9	6.5	12.0	T	T	49.8
1974-75	0.0	0.0	T	2.3	3.7	7.7	7.4	8.8	8.3	11.4	4.9	T	54.5
1975-76	0.0	0.0	0.0	7.6	8.6	3.1	4.5	13.6	7.0	15.5	0.0	T	59.9
1976-77	0.0	0.0	0.0	0.0	0.2	1.5	10.3	0.6	7.6	0.6	1.1	0.0	21.9
1977-78	0.0	0.0	0.0	0.3	2.6	4.2	5.8	7.8	3.6	0.4	0.9	0.0	25.6
1978-79	0.0	0.0	T	0.0	11.5	7.6	14.5	9.7	3.1	3.7	0.3	T	50.4
1979-80	0.0	0.0	0.0	0.8	8.9	2.5	12.7	1.9	7.2	1.5	T	0.0	35.5
1980-81	0.0	0.0	0.0	8.0	2.7	2.4	5.9	2.9	6.3	0.3	1.0	0.2	29.7
1981-82	0.0	0.0	0.0	1.9	2.3	17.9	22.3	5.8	13.6	2.4	0.2	0.0	66.4
1982-83	0.0	0.0	T	0.5	7.3	18.3	6.2	4.9	10.5	5.3	5.5	0.0	58.5
1983-84	0.0	0.0	1.0	0.0	10.7	33.7	6.5	20.1	4.3	9.1	0.2	0.0	85.6
1984-85	0.0	0.0	T	3.7	4.3	5.7	9.3	11.9	16.6	1.3	0.0	0.0	52.8
1985-86	0.0	0.0	T	1.4	27.5	11.5	5.7	6.0	3.7	11.5	1.8	0.0	69.1
1986-87	0.0	0.0	T	0.0	8.6	1.7	13.1	4.5	2.7	2.1	0.0	0.0	32.7
1987-88	0.0	0.0	0.0	0.0	1.6	8.4	8.9	1.1	3.3	1.5	2.6	0.0	27.4
1988-89	0.0	0.0	0.0	0.0	7.7	17.0	6.2	12.0	5.9	0.3	0.5	0.0	49.6
1989-90	0.0	0.0	0.0	8.2	2.5	1.2	3.5	3.1	14.7	T	1.6	0.0	34.8
1990-91	0.0	T	0.0	0.1	3.8	15.9	8.0	1.5	8.3	9.8	1.8	0.0	49.2
1991-92	0.0	0.0	0.0	2.5	9.1	1.9	3.1	0.9	0.6	0.4	0.0	0.0	18.5
1992-93	0.0	0.0	0.0	0.1	15.2	19.9	29.6	21.3	1.5	5.5	0.2	T	93.3
1993-94	0.0	T	0.0	0.0	6.2	6.3							
Record Mean	0.0	T	0.1	1.9	5.3	8.9	10.0	6.2	5.9	4.4	0.6	T	43.2

2.3 Geology and Hydrology

2.3.1 General physiographic setting

Pocatello is located on the boundary between the northeastern corner of the Great Basin Section of the Basin and Range Physiographic Province and the southeastern edge of the Snake River Plain Section of the Columbia River Plateau Province (as shown in Figure 2.2-1). The area thus has stratigraphic, structural, petrologic and geomorphic characteristics of both areas. The Basin and Range Province, particularly in the Great Basin Section, is characterized by alternating basins and mountain ranges; the basins commonly are partially graded up onto the mountain sides by Bogota (filled) and pediment (cut) surfaces. The Snake River Plain is an arcuate, flat-surfaces basalt plateau, dissected by the Snake River and to a lesser extent by some of its tributaries on the western end. Both physiographic areas are relatively young geologically. Seismic and volcanic activity during "Recent" time and the freshness of tectonic forms and surficial extrusive rocks show that the area is still in the process of evolving.

2.3.2 Local geology and physiography

The city of Pocatello is in the valley bottom of the Portneuf River, a tributary to the Snake River. The Portneuf drains about 1200 square miles of the Bannock and Portneuf mountain ranges east and southeast of Pocatello. The city is bordered on the southwest and northeast sides by hills generally less than 4000 feet high, with the general terrain becoming more mountainous to the southeast. The river valley is flat floored and at the townsite widens abruptly to the northwest out onto the Snake River Plain.

The bedrock floor of the valley is buried under some 200 feet of alluvium; whether the original shape and depth of the valley was primarily structural (down-faulted) or erosional is unknown. A much larger river (Bear River) probably occupied the valley prior to the diversion of that river to the south 30,000 years or so ago, possibly accounting for the size of the valley relative to that of the current river. Regardless of its original configuration, the valley has been altered by several episodes of cutting and filling since its inception. Broad benches slope 300-400 feet per mile toward the valley from each side; these are remnants of a Pleistocene valley fill. Much of this sequence has been removed down the axis of the valley, and the surfaces are currently being dissected by small tributaries to the Portneuf. A basalt flow about the same age as the diversion of the Bear River (about 30,000 years) floors part of the valley upstream from the main part of the city.

Idaho State University buildings are located on terraces 40-60 feet above the valley bottom on the northeast side of the valley. They are built on a combination of the valley-fill alluvium and loess (wind-blown silt deposits), both of which are common in the area. The southeastern corner of the campus is bordered by a bedrock hill composed of Cambrian (Brigham) Quartzite; the block has been elevated along a high-angle fault which may extend to the northwest under part of the campus or which may terminate at the end of the quartzite block. If the fault does extend under the alluvium beneath the campus, the fault is apparently entirely pre-alluvium in age, as no displacement in the alluvium along strike with the fault has been noted.

2.3.3 Subsurface water of Pocatello

Gravel lenses and beds near the center of the Portneuf valley supply water for Pocatello. The non-artesian production is from about 30 to 40 feet in depth. The water is pumped to storage tanks located in the hills surrounding the city. The water table fluctuates about 5 to 10 feet annually, with those wells near the river showing greater fluctuation. No wells have been drilled for water in the immediate University area, however, porosity and permeability are probably relatively low toward the east side of the valley. Most of the water source for the producing aquifers is associated with influent from the river which is located toward the southwest side of the valley about 0.8 miles from the University. A well, recently drilled, a mile east of the University at an elevation of about 4700 feet produced only several hundred gallons per minute from 270 feet depth, and was abandoned for insufficient yield. The closest producing well to the University site is about 0.6 miles south, at an elevation of about 4450 feet; this well produces water from about 4410 elevation, 90 feet below the level of the campus. Considerations of the height of the water table, the elevation of the campus, and the character of the subsurface suggest that no ground water problems will be encountered in the University area.

2.3.4 Surface waters

During the early spring of 1962 and 1963, flooding occurred in the low areas of the valley along the river to elevations of about 4460 feet. This flooding was due to unseasonably warm weather and runoff of meltwater into the river over a frozen subsurface in the mountains southeast of Pocatello. A U.S. Corps of Engineers project to straighten and line the banks of the river with concrete has eliminated flooding; even without this correction there would be little chance of flooding in the future to anywhere near the height of the campus. Local drainage is generally away from the Lillibridge Engineering Laboratory so that no surface drainage problems associated with surface runoff are likely.

2.4 Seismology

This report was prepared for the College of Engineering by James E. Zollweg, Northwest Geophysics, August 16, 1995.

2.4.1 Introduction

Pocatello lies along the boundary of two major geologic provinces. The Eastern Snake River Plain (ESRP) is the generally flat plain north of the city. The highland area around the city lies within the Basin and Range province. The Basin and Range province is a region of generally moderate to high seismicity and there are numerous late Cenozoic faults within 80 miles. Seismic safety has been an important consideration in eastern Idaho because of the proximity of the Idaho National Engineering Laboratory to active late Quaternary faults of the Northern Basin Range province, which lies north of the ESRP. Careful analysis of potential maximum earthquakes is required, and this analysis will rely heavily on current expert opinion regarding seismic safety issues.

2.4.2 Methodology

A three-step methodology was used in the assessment of seismic risk at Pocatello. The first of these is consideration of the distribution of historical seismicity. The second is consideration of the proximity of potentially active surface faults. The third element, utilized if the first two steps do not indicate a larger magnitude event controls seismic risk, is consideration of a "floating" earthquake that can occur essentially anywhere within the Pocatello region.

2.4.3 Historical seismicity catalogs

A search was made of available on-line earthquake catalogs to assess the frequency of magnitude 4 or greater events in the Pocatello area. The search area chosen was 41 to 44.5^0 north latitude and 110 to 115^0 west longitude. Seismic monitoring of eastern Idaho is splintered between several organizations and their catalogs are not routinely integrated. Most of the catalogs are not published in any usual sense; they are maintained by their individual organizations as part of routine seismic monitoring efforts. The level of completeness, accuracy, and ending date differ between the catalogs. Those catalogs consulted were developed by Woodward-Clyde Federal Services (see Wong et al., 1992 for a map), the University of Utah, the Idaho National Engineering Laboratory, the Montana Bureau of Mines and Geology, Boise State University, and the

U. S. Geological Survey. The Woodward-Clyde catalog was derived from the other catalogs and other sources of historic data, but only runs through late 1989. It was the primary data source for events through its completion date. The composite catalog is believed to be essentially complete through mid-1994.

Figure 2.4-1 shows the resulting epicenter map, and Table 2.4-1 gives epicentral data for the events within 130 km (approximately 80 miles) of 42.868^0 north, 112.435^0 west (the approximate location of the Pocatello business district and Idaho State University). Smaller earthquakes have occurred closer to Pocatello, but the distribution of magnitude 4+ events may be a better index to active source areas. No magnitude 4 earthquake has occurred closer than about 60 km (about 35 mi) to Pocatello. The ESRP, which lies to the north of Pocatello, is nearly aseismic at the magnitude 4 level and is not considered to be a likely source of events capable of causing damage in Pocatello. High activity occurs to the south, east, and northeast at greater distances than 35 mi. Many of these earthquakes have been felt in Pocatello, although damaging levels of intensity are not believed to have been achieved historically. The closest magnitude 6+ earthquake was the 1975 Pocatello Valley earthquake (local magnitude 6.0), about 56 mi from Pocatello, and the closest magnitude 7+ earthquake was the 1983 Borah Peak earthquake (surface wave magnitude 7.3), about 105 mi from Pocatello.

It is concluded that the historical earthquake catalog shows Pocatello is in an area of relatively low seismicity, with higher seismicity areas occurring at least 35 mi from the city. This would suggest that Pocatello is at little risk from a nearby source of large earthquakes, but the historic record is short. Geologic evidence discussed in the next section is felt to be of greater importance than the historical seismicity catalog.

2.4.4 Faults

Faults in the Basin and Range province usually have long recurrence times between large events, typically 1,000 to 100,000 years on the same fault segment. Therefore, the historical catalog is not the only indication of the potential for damaging earthquakes at Pocatello. Figures 2.4-2 and 2.4-3 (taken from Hilt, 1993) show that while mapped late Cenozoic faults occur within a few miles of Pocatello, the nearest fault with probable late Quaternary movement is located about 100 mi southeast of Pocatello (it is the Bear Lake fault). Late Quaternary movement is known on faults on the north side of the ESRP at about the same distance from Pocatello. Faults with late Cenozoic movement but no evidence for late Quaternary movement are generally considered inactive for seismic hazard

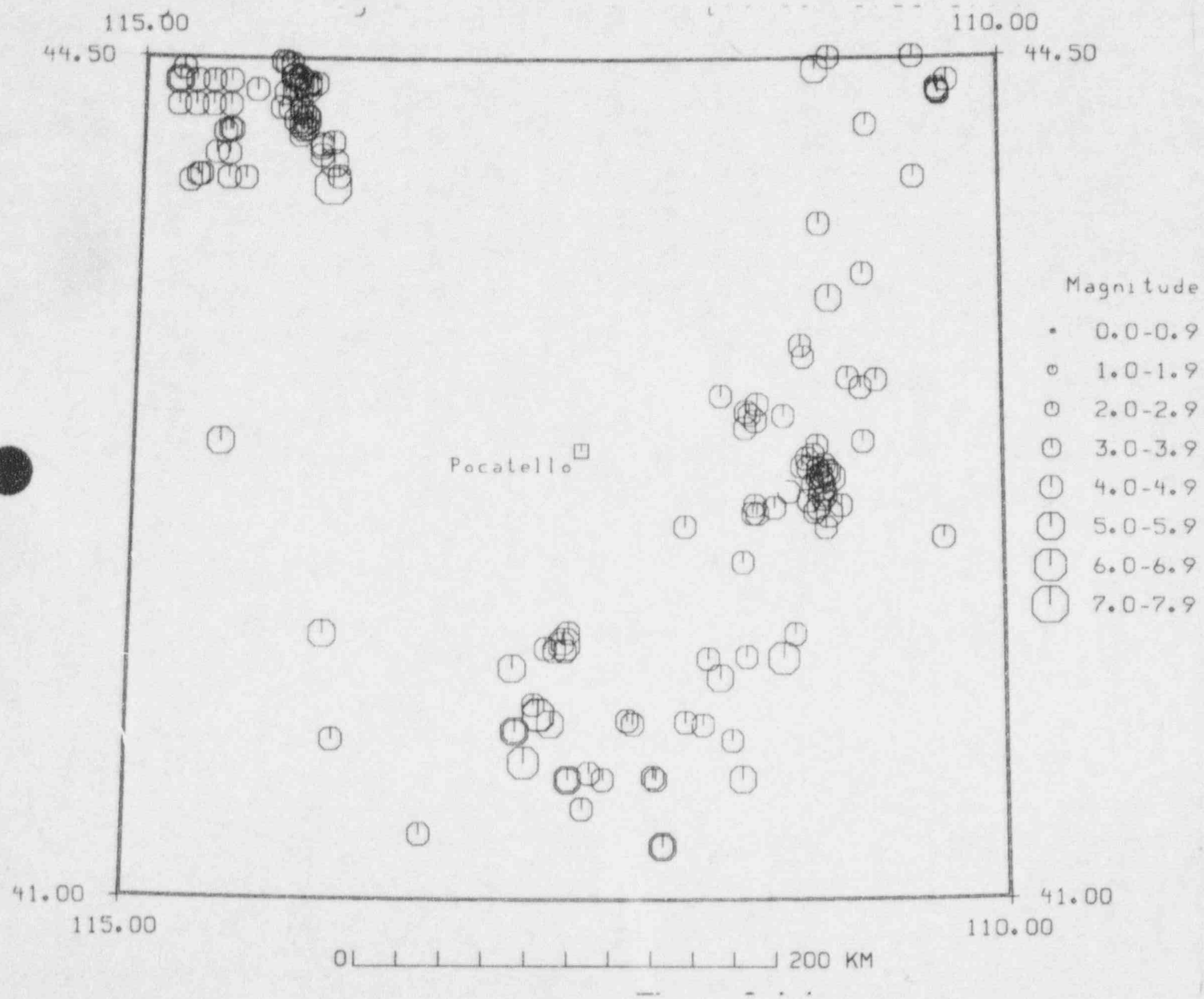


Figure 2.4-1 Epicenter map.

Table 2.4-1 Earthquakes within 130 km of Pocatello, ID (1909-1994)

DATE-(UTC)-TIME yy mm/dd hh:mm:ss	LAT(N) deg.	LON(W) deg.	MAG	COMMENTS
1909 10/06 02:50:00	41.75	112.65	6.0	77.0 mi S of Pocatello
1909 11/17 06:30:00	41.73	112.15	4.3	78.9 mi S of Pocatello
1915 07/30 18:50:00	41.73	112.15	4.3	78.9 mi S of Pocatello
1930 06/12 09:15:00	42.60	111.00	5.8	75.4 mi ESE of Pocatello
1934 03/12 15:05:48	41.76	112.66	6.6	76.8 mi S of Pocatello
1934 04/14 21:26:32	41.71	112.60	5.4	79.0 mi S of Pocatello
1934 05/06 08:09:42	41.95	112.81	5.5	65.7 mi SSW of Pocatello
1946 05/06 02:30:00	41.71	112.11	4.3	80.1 mi SSE of Pocatello
1960 08/07 16:27:16	42.40	111.50	4.9	57.6 mi SE of Pocatello
1962 08/30 13:35:24	41.91	111.61	5.7	77.3 mi SSE of Pocatello
1962 09/05 03:00:00	42.00	111.70	4.3	70.8 mi SSE of Pocatello
1969 09/19 09:31:45	43.05	111.41	4.4	53.2 mi ENE of Pocatello
1969 09/19 13:33:15	42.98	111.41	4.9	51.7 mi E of Pocatello
1969 09/19 19:57:18	43.00	111.26	4.4	59.9 mi E of Pocatello
1969 09/19 23:58:06	42.95	111.48	4.1	48.4 mi E of Pocatello
1973 04/14 06:45:46	42.03	112.61	4.4	57.9 mi S of Pocatello
1975 03/27 04:48:51	42.05	112.53	4.2	55.7 mi S of Pocatello
1975 03/28 02:31:05	42.05	112.51	6.0	55.8 mi S of Pocatello
1975 03/29 13:01:19	42.01	112.51	4.7	57.8 mi S of Pocatello
1975 03/30 06:56:28	42.01	112.56	4.1	58.1 mi S of Pocatello
1976 11/05 02:48:55	41.80	112.68	4.0	74.4 mi S of Pocatello
1978 10/24 20:30:59	42.55	111.83	4.1	37.6 mi SE of Pocatello
1978 11/30 06:53:40	42.10	112.50	4.7	52.6 mi S of Pocatello
1981 12/09 08:15:04	42.63	111.41	4.1	53.5 mi ESE of Pocatello
1982 05/30 11:06:42	42.68	111.23	4.0	62.2 mi E of Pocatello
1982 10/14 04:10:23	42.60	111.41	4.7	54.9 mi ESE of Pocatello
1982 10/14 11:09:29	42.60	111.43	4.1	53.7 mi ESE of Pocatello
1983 02/08 10:54:54	43.30	111.16	4.2	70.7 mi ENE of Pocatello
1985 07/02 03:03:56	43.25	111.15	4.0	70.2 mi ENE of Pocatello
1987 03/18 00:00:42	42.61	111.31	4.1	59.2 mi ESE of Pocatello
1988 11/13 11:53:24	42.61	110.91	4.4	78.4 mi ESE of Pocatello
1988 11/19 20:00:53	42.00	111.46	4.3	77.0 mi SE of Pocatello
1992 11/10 10:46:18	43.08	111.61	4.8	43.6 mi ENE of Pocatello
1992 11/10 10:54:50	43.00	111.45	4.9	50.3 mi ENE of Pocatello
1992 11/11 12:08:07	43.01	111.48	4.0	49.4 mi ENE of Pocatello
1994 02/02 11:04:25	42.75	111.06	4.0	69.9 mi E of Pocatello
1994 02/03 07:14:51	42.75	111.03	4.5	71.0 mi E of Pocatello
1994 02/03 09:05:03	42.75	110.96	5.8	74.3 mi E of Pocatello
1994 02/03 09:47:36	42.71	111.03	4.0	71.7 mi E of Pocatello
1994 02/03 09:58:40	42.75	111.03	4 2	70.9 mi E of Pocatello

Table 2.4-1 Continued.

DATE-(UTC)-TIME yy mm/dd hh:mm:ss	LAT(N) deg.	LON(W) deg.	MAG	COMMENTS
1994 02/03 10:25:51	42.78	111.11	4.0	66.9 mi E of Pocatello
1994 02/03 11:19:07	42.76	111.00	4.7	72.6 mi E of Pocatello
1994 02/03 11:46:50	42.78	111.15	4.0	64.9 mi E of Pocatello
1994 02/03 12:04:57	42.71	111.08	4.4	68.7 mi E of Pocatello
1994 02/04 02:42:12	42.70	111.03	5.2	71.8 mi E of Pocatello
1994 02/04 03:10:08	42.83	111.08	4.0	68.1 mi E of Pocatello
1994 02/04 21:49:12	42.61	111.05	4.0	71.7 mi ESE of Pocatello
1994 02/07 06:35:47	42.65	111.03	4.8	72.7 mi E of Pocatello
1994 02/07 12:15:45	42.66	111.01	4.5	73.3 mi E of Pocatello
1994 02/10 00:56:11	42.88	111.06	4.3	69.3 mi E of Pocatello
1994 02/11 04:24:29	42.81	111.11	4.0	66.3 mi E of Pocatello
1994 02/11 14:59:50	42.76	111.00	5.3	72.8 mi E of Pocatello
1994 02/14 16:55:34	42.80	111.01	4.0	71.5 mi E of Pocatello
1994 03/03 07:13:17	42.78	111.05	4.1	69.8 mi E of Pocatello
1994 04/07 16:16:44	42.53	111.01	4.8	75.5 mi ESE of Pocatello
1994 04/08 07:26:21	42.60	111.08	4.1	71.1 mi ESE of Pocatello
1994 04/10 20:04:09	42.65	111.11	4.6	68.4 mi ESE of Pocatello

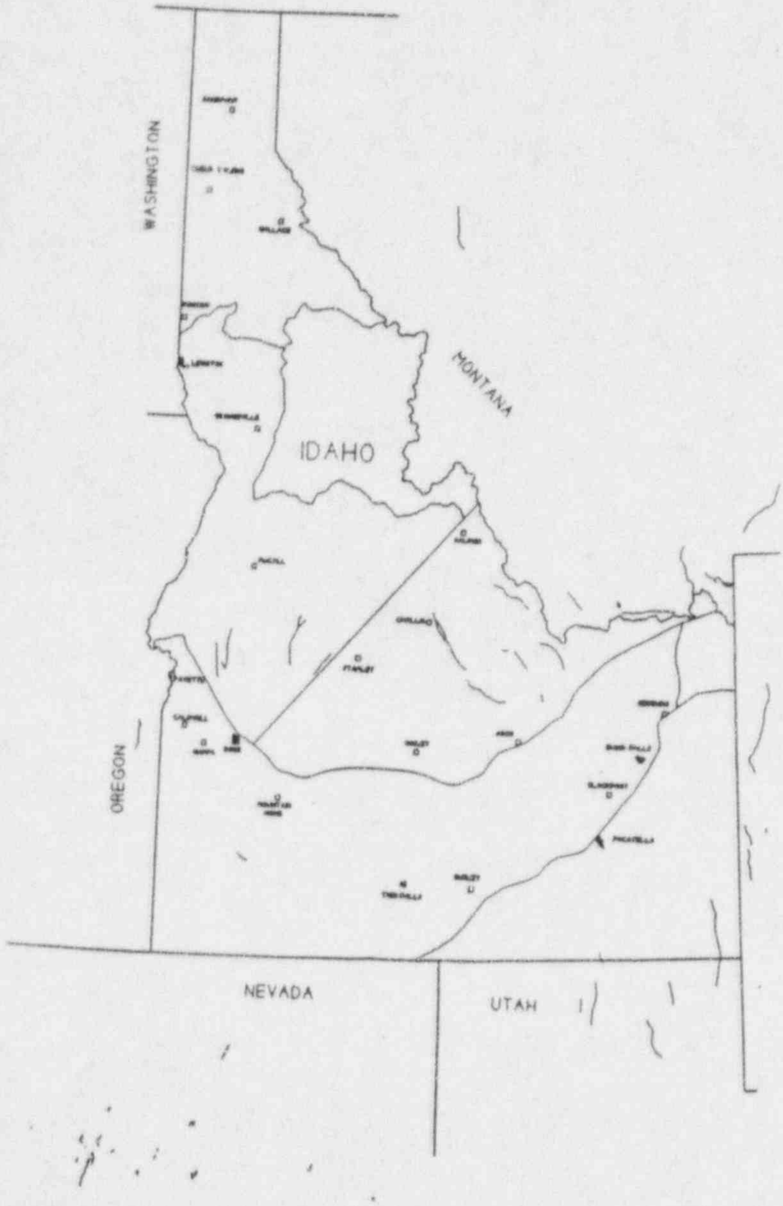


Figure 2.4-2 Faults with late Quaternary motion in Idaho

Figure 2.4-3 Late Cenozoic faulting in Idaho

evaluation purposes. Most late Quaternary faults have expectable maximum magnitudes between about surface wave magnitude 6.8 and 7.6.

Anders et al. (1989) believe that the location of active segments of faults around the ESRP is a function of distance from the ESRP axis, and propose a physical model to explain an increase in strain rates with distance from the ESRP. The Anders et al. model would suggest that faults in the immediate vicinity of Pocatello have very low strain rates, and consequently long recurrence times (in excess of 10,000-100,000 years) are probably to be expected. If the Anders et al. model is correct, there is little risk from a large earthquake occurring in the immediate vicinity of Pocatello. Major earthquakes (magnitude 6.8+) are likely to occur no closer to Pocatello than the regions of observed high seismicity, and the Anders et al. model suggests that the more active faults would be located even farther from the city. It is therefore concluded that the known late Quaternary faults are not greater sources of seismic risk to Pocatello than possible blind faults existing near the city, as will be discussed in the next section.

2.4.5 Floating earthquakes

Blind faults (those not recognized at the earth's surface) exist in southeast Idaho, as is proven by the 1994 local magnitude 5.8 Draney Peak earthquake and the 1975 local magnitude 6.0 Pocatello Valley earthquake. These earthquakes did not occur on known faults. To account for such events, seismologists have used the concept of a "floating earthquake" which is customarily chosen as being 1/2 magnitude unit larger than the largest historical event that is not associated with a known fault. The largest such events are the Draney Peak and the Pocatello Valley earthquakes. Thus, a reasonable estimate of the magnitude of the floating earthquake is 6.5. The floating earthquake is customarily placed at a distance of 25 km from the site in question. Since Pocatello lies at the edge of the Basin and Range province in which the Draney Peak and Pocatello Valley earthquakes occur, the possibility of blind faults near Pocatello cannot be ruled out. Therefore, a magnitude 6.5 earthquake at a distance of 25 km is chosen to be the event controlling seismic hazard at Pocatello. The choice of this event is a state-of-the-art assessment and may change in the future as a better understanding of southeastern Idaho seismotectonics develops.

<u>Design acceleration for floating earthquake</u>. The horizontal peak ground acceleration for a floating earthquake of magnitude 6.5 at a distance of 25 km from Pocatello can be calculated from relationships presented by Joyner and Boore (1981). Because of the lack of significant historical

seismicity within 35 mi of Pocatello, it is felt that use of the Joyner and Boore (1981) 50th percentile formula is sufficiently conservative; a higher percentile formula would be justified if significant seismicity were located near the city. The Joyner and Boore (1981) relationship is:

$$\log A = -1.02 + 0.249\ Mw - \log r - 0.00255\ r$$

where A is horizontal peak ground acceleration in g, Mw is earthquake moment magnitude (roughly equivalent to local magnitude at magnitude 6.5), and r is a distance term defined as:

$$r = (d^2 + 7.3^2)^{0.5}$$

where d is the closest approach of the seismogenic structure in km. For the floating earthquake, d is 25 km. It should be noted that the Joyner and Boore (1981) curve was developed mainly from California area data and application to southeast Idaho can be expected to be only approximate.

The resulting horizontal peak ground acceleration is 0.13 g. Such an acceleration indicates that minor structural damage in poorly-built facilities is about the most that can be expected in the Pocatello area. Since such an event has not occurred historically, this seems to be a conservative estimate suitable for most uses.

References

Anders, M. H., J. W. Geissman, L. A. Piety, and J. T. Sullivan (1989), Parabolic distribution of circumeastern Snake River Plain seismicity and latest Quaternary faulting: Migratory pattern and association of the Yellowstone hotspot, Journal of Geophysical Research 94, 1589-1621.

Hilt, A. P. (1993), Seismic Siting Considerations in Idaho, M. S. Thesis, University of Idaho, Moscow, 136 p.

Joyner, W. B., and D. M. Boore (1981), Peak horizontal acceleration and velocity from strong-motion records including records from the 1979 Imperial Valley, California earth quake, Bulletin of the Seismological Society of America 71, 2011-2038.

Wong, I., K. Coppersmith, W. Silva, R. Youngs, T. Sawyer, M. Hemphill-Haley, C. Stark, P. Knuepfer, R. Castro, F. Makdisi, D. Wells, S. Chiou, R. Bruhn, and W. Daning (1992), Earthquake Ground Motion Evaluations for the Proposed New Production Reactor at the Idaho National Engineering Laboratory, Volume I: Deterministic Evaluation, Informal Report EGG-GEO-10304, Woodward-Clyde Consultants, Geomatrix Consultants, and Pacific Engineering and Analysis, Oakland, CA.

3.0 LILLIBRIDGE ENGINEERING LABORATORY

This section contains information about the Laboratory building and the Nuclear Operations Area in its basement.

3.1 General Description

As shown in Figure 2.1-3, the Laboratory is located on the west side of South 8th Street in Pocatello, just south of the Carter Street intersection at the northeast corner of the "lower" campus. The major axis of the rectangular building is parallel to the street and in approximate north-south direction. The building has three (3) floors: a basement (first) level, a ground (second) level and an upper (third) level. The ground level floor is at 4500 ft. elevation above sea level. The building was designed by C. A. Sundberg and Associates, constructed by Taysom Construction and first occupied September 14, 1971. The building conforms with the zone 3 seismic design requirements of the Uniform Building Code, with walls made up of concrete block with interior and exterior brick facing. Floors are of precast concrete slab design. The building has approximately 28,500 sq. ft. divided as follows:

Laboratories - 11,900 sq. ft.
Offices - 2,900
Classrooms - 1,600
Workshop & misc. - 12,100

The northern-most portion of the building is the single story Mechanical Shop and experimental fluids research area. It has a thick floor to support heavy machining equipment, and no basement. Two areas in the Shop are to be used in case of reactor emergencies. These areas and their usage are discussed in the Emergency Plan. The Reactor Laboratory extends westward from the western side of the main building. The upper portion of this Laboratory extends above ground the height of the first story of the main building. The floor of the Laboratory is the building basement level.

The building is supplied with 110, 240 and 480 VAC power and is heated with low pressure steam from the campus system. The building has its own recirculating ventilation and cooling system which is interconnected with the Reactor Laboratory ventilation system as described below. Figure 3.1-1 shows the layout of the top floor. The space is devoted to the College of Engineering Office, Dean's Office, some College faculty offices, the Computer laboratory, large conference room and some staff and graduate student offices. The ground floor level shown in Figure 3.1-2 is comprised of the Thermal Fluids Laboratory, Materials and Measurements Laboratory, Structures and Geotechnics Laboratory, some graduate student offices and a College Library/study room. The Mechanical Shop is also on this level. As shown in Figure 3.1-3, the basement houses the Electrical Laboratory, Environmental Laboratory, a

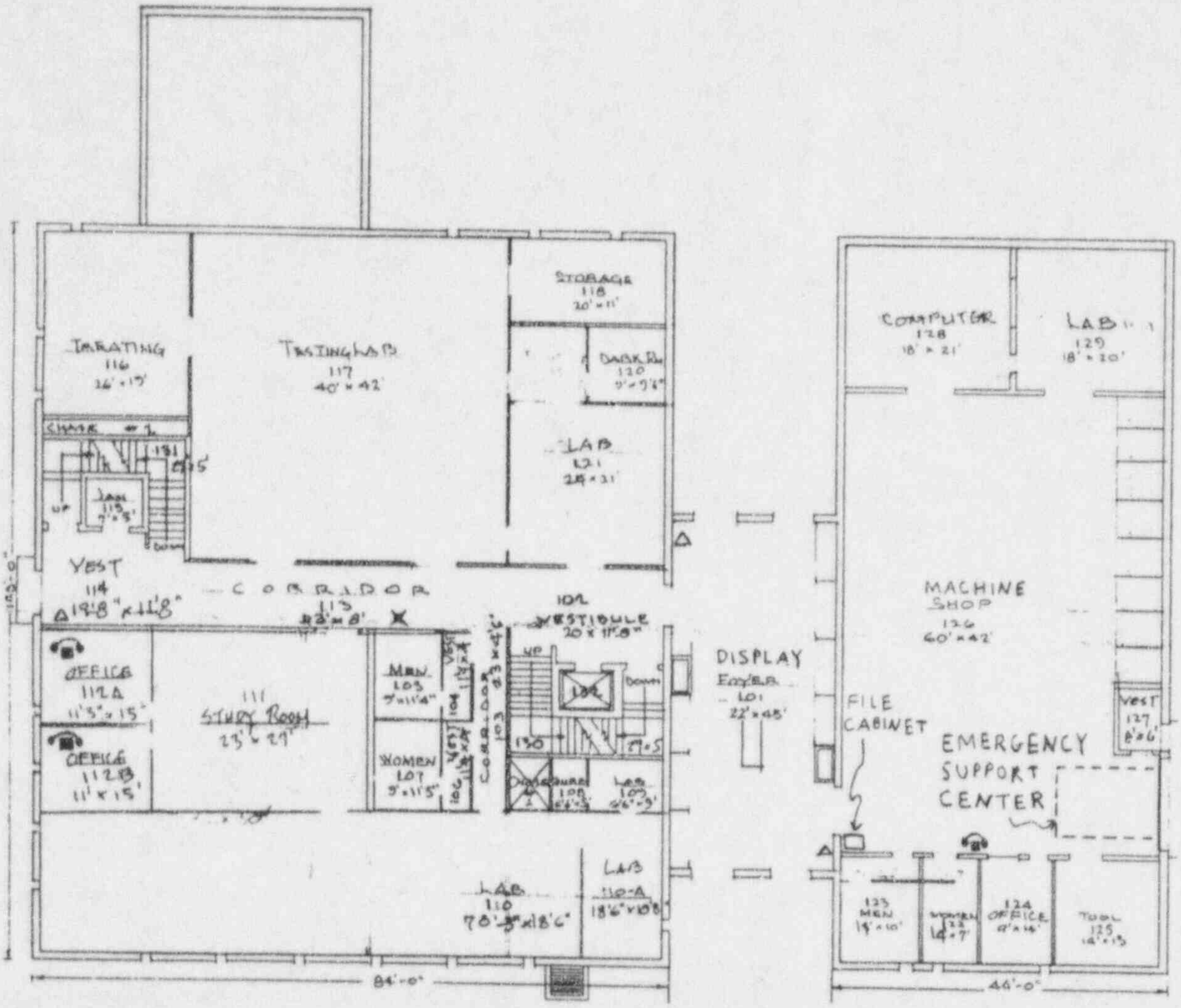


Figure 3.1-2 LEL Second-Level (ground) Floor Plan.

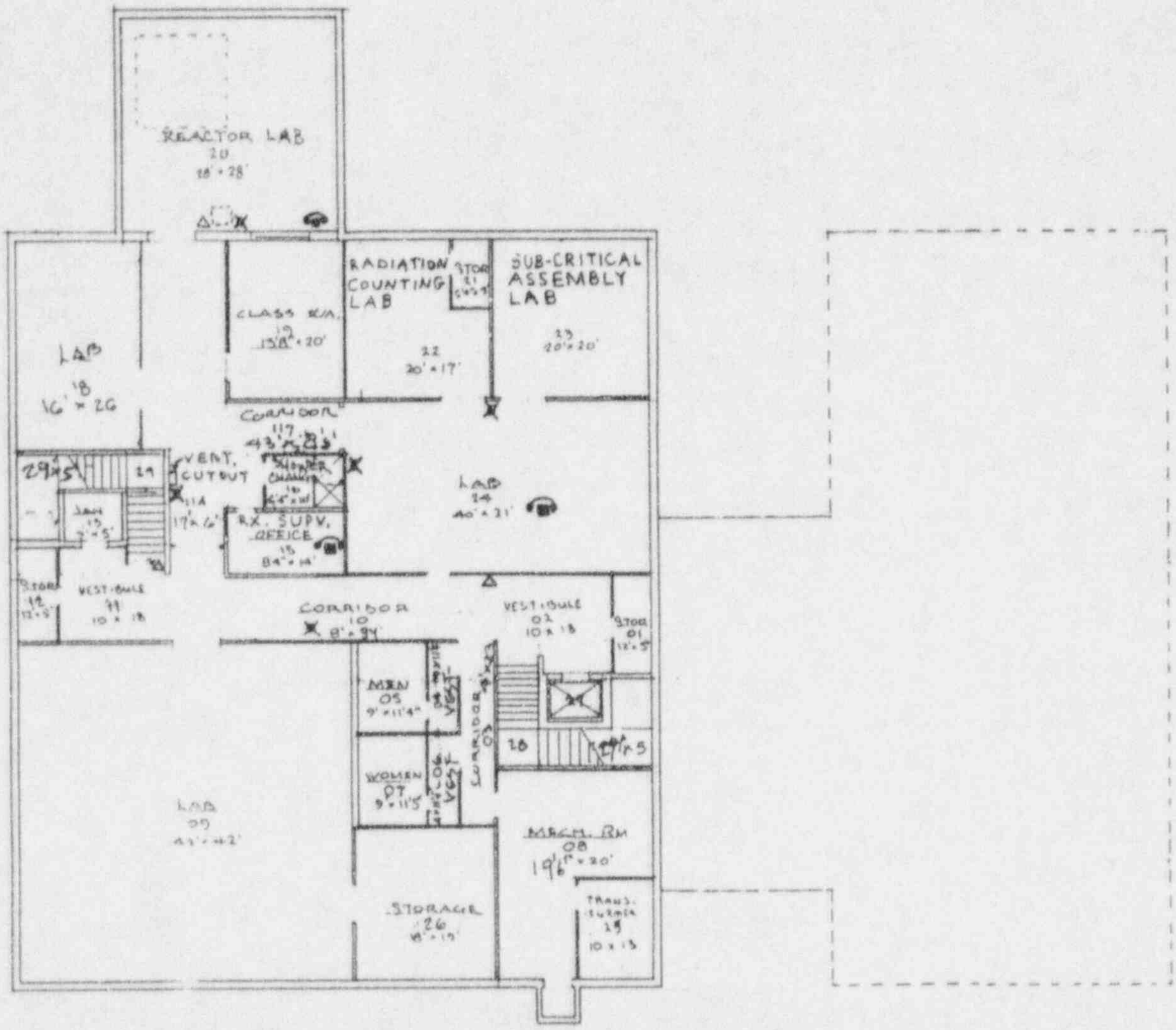


Figure 3.1-3 LEL First-Level (basement) Floor Plan.

graduate student office area, mechanical equipment room and the Nuclear Operations Area.

The building ventilation and cooling system recirculates about 80% of the flow, discharges 20% from a roof vent and takes in about 20% in fresh (outside) makeup air before being distributed throughout the building. The exhaust from the Reactor Laboratory mixes with air from the rest of the building on its way to the roof vent. The building air and Reactor Laboratory exhaust mix in the ratio 27:1, respectively. In the event of an emergency, the Lillibridge ventilation system fans can be turnded off to reduce the emission and recirculation of any airborne radioactive contamination.

3.2 Nuclear Operations Area

This area comprises the main portion of the basement level of the building and includes the Reactor Laboratory, Subcritical Assembly Laboratory, Counting Laboratory, Reactor Supervisor's Office and the Reactor Observation/Conference Room.

3.2.1 Reactor laboratory

This 28 ft. square, 24 ft. tall room houses the AGN-201 reactor, as shown in Figure 3.2-1. The Labortory is identified as Room 20. The 6.5 ft. diameter reactor tank is located in the northwest corner of the room behind an interior L-shaped shield wall. Since the Laboratory is at the basement level, earth shielding exists on the outside of the west and north (and south) walls in this northwest corner, for a height of 10 ft. up from that baement floor. Access to the Reactor Laboratory is through double doors on the east wall which is the west wall of Lillibridge. Emergency egress is provided via a ladder to a personnel roof hatch. Another roof hatch, 12 ft. square, provides a means of moving large equipment into and out of the Reactor Laboratory. The equipment hatch doors are locked externally with a padlock. The personnel hatch is locked on the inside. An overhead crane with trolley provides large equipment lift coverage over the entire Laboratory floor area. A covered 4½ ft. diameter cylindrical steel-lined pit extends 16 ft. below the Reactor Laboratory floor just south of the reactor tank. Access to the top of the reactor tank is via stairs located along the west wall of the laboratory. The reactor control console is located to the east of the reactor tank in the northeast corner of the room. The console can readily be seen from the Observation/Conference Room adjacent to the east wall of the Reactor Laboratory, through full height windows in that wall.

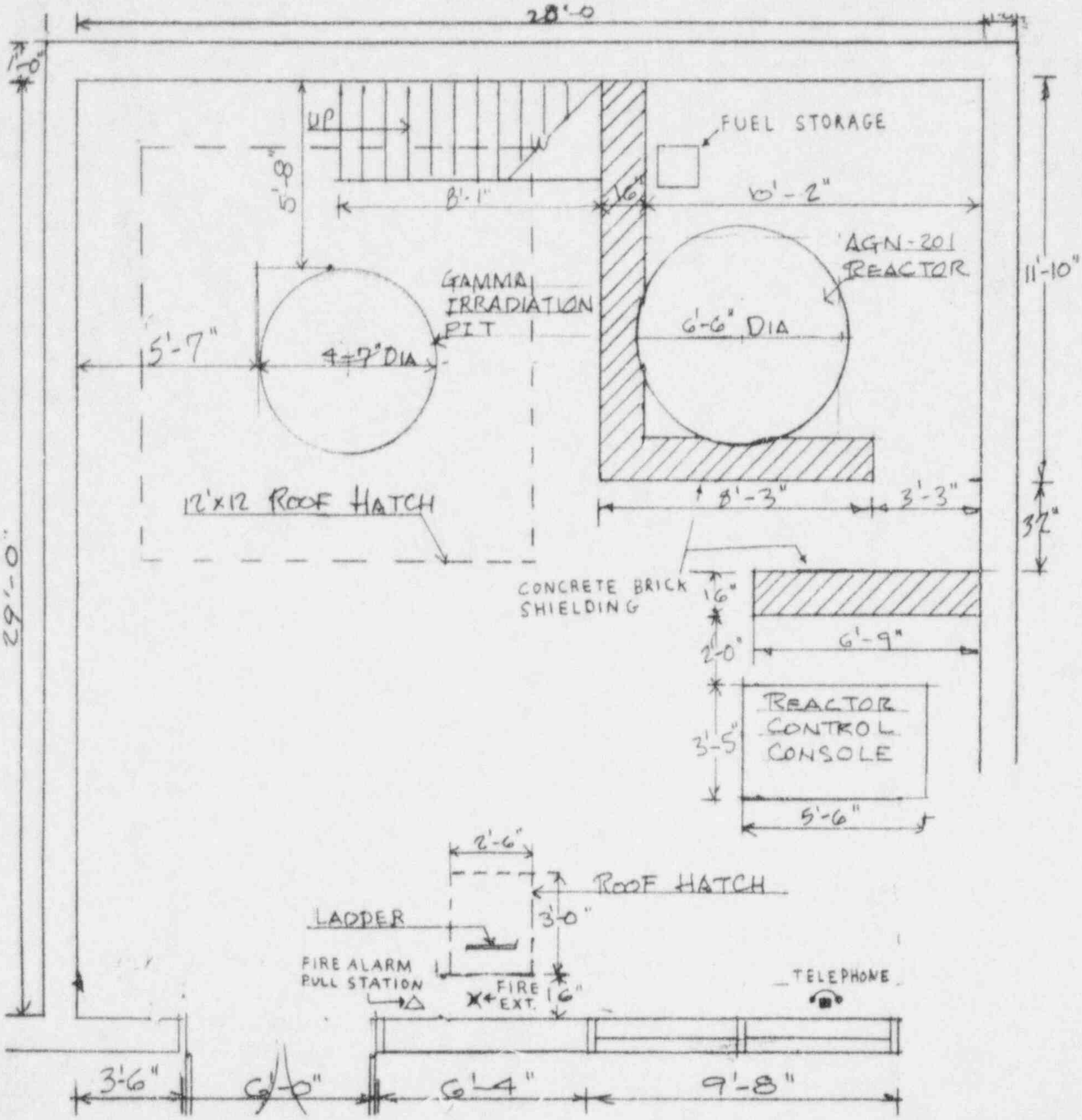


Figure 3.2-1 Reactor Laboratory Floor Plan (basement).

The Reactor Laboratory is supplied with 110 and 240 VAC power and fluorescent lighting but no plumbing. A telephone is located at the control console; a fire extinguisher and the fire alarm are located on the east wall adjacent to the double doors. Shielding walls, and radiation fields with the reactor at full power of 5W, are discussed in Section 5.3. An earthquake detector is located on the support skirt of the reactor tank. various radiation monitoring devices are located within the Laboratory.

3.2.2 Counting laboratory

The Counting Laboratory, Room 22, is located north of the Observation/Conference room, between it and the Subcritical Assembly Laboratory. The Counting Laboratory has a small fume hood and associated compresed air, water, etc. plumbing. Also located in this Laboratory is a locked closet for storage of small, sealed radiation sources. The primary equipment in the Laboratory is a liquid nitrogen cooled germanium detector based computerized spectroscopy system. The Laboratory also has the outboard terminus of a rabbit system connected to the AGN-201 reactor, and an associated counting system.

3.2.3 Subcritical laboratory

This Laboratory, Room 23, contains a subcritical assembly used in the undergraduate and graduate laboratory courses. As shown in Figure 3.2-2, the 19.9% enriched fuel, when not in use, is stored in the locked Fuel Storage Container. The water tank used to hold the fuel for experiments is mounted on a large set of high purity graphite blocks, the whole assembly constituting a sigma pile. The Laboratory is equipped with both neutron and gamma radiation monitors with associated alarms. The subcritical assembly water tank is connected by piping and a pump to three smaller plastic water storage tanks located at an elevation lower than the assembly water tank. No other plumbing exists in the room. Neither beryllium, beryllium oxide, nor heavy water is are permitted in this Laboratory without Reactor Supervisor permissions..

3.2.4 Reactor observation room, supervisor's office

As noted above, the Observation/Conference room, is adjacent to the Reactor Laboratory and has full height windows in their common wall. This is identified as Room 19. It is locked when not in use, since a radiation field of 6 mr/hr is present in the room when the reactor is operating at a 5W power level. Access to this room, when the reactor is operating, is restricted to authorized personnel. In all other parts of the Nulcear Operations Area, radiation levels are less than 0.1 mr/hr when

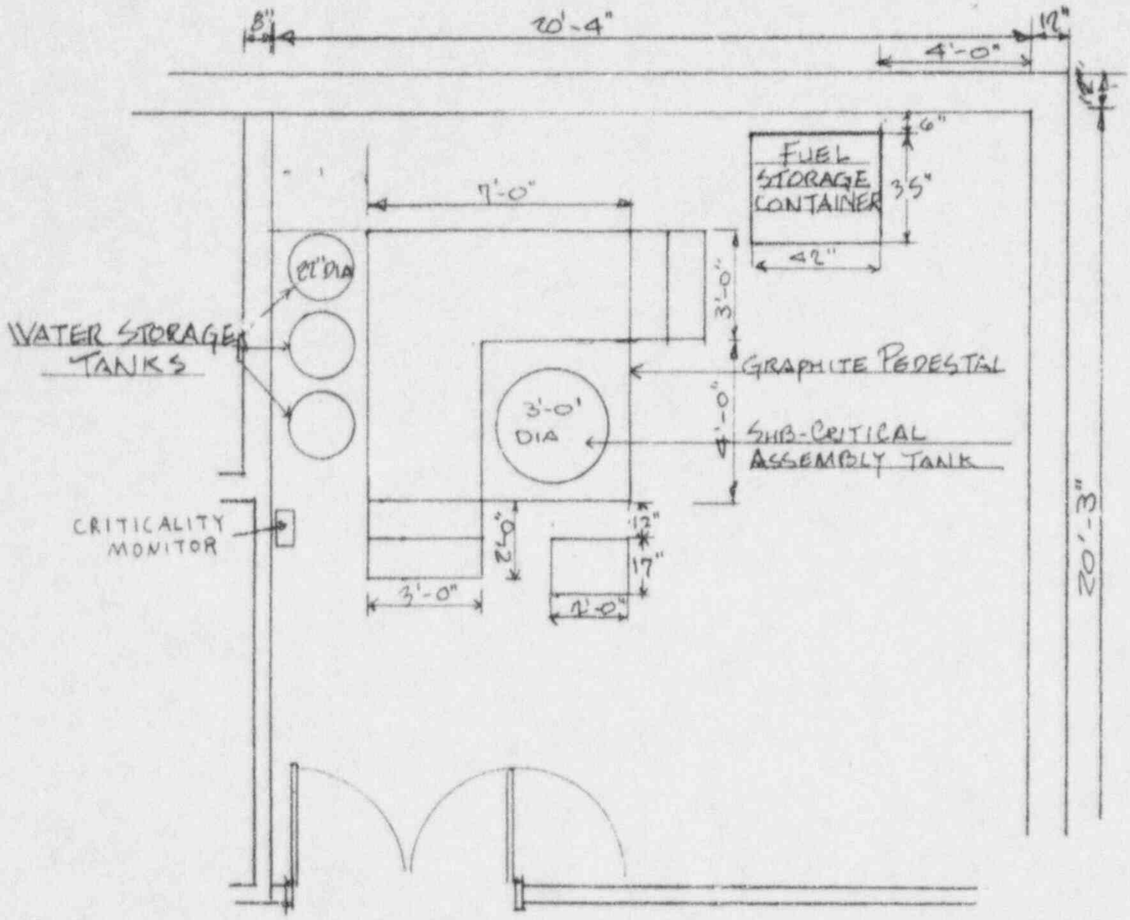


Figure 3.2-2 Subcritical Assembly Laboratory Floor Plan (basement).

the reactor is operating at 5W.

As shown in Figure 3.2-1, the Reactor Supervisor's office, Room 15, is located east of the Observation/Conference room and adjacent to a room containing an industrial shower and a storage area for film badges, dosimeters and other radiation monitoring equipment. The ventilation cutout switch, which turns off the Lillibridge ventilation system fans is located on the wall south of the Supervisor's office.

4.0 AGN-201 Reactor

This section contains a description and detailed technical information about the AGN-201 reactor.

4.1 Introduction

The AGN-201 reactor operating at Idaho State University is essentially identical to all the AGN-201 reactors which have operated around the world since 1957. Therefore, the reactor description given in the Safety Analysis Report for the original Aerojet General AGN-201 reactor (Hazard Report and Preliminary Design Report by Aerojet General Nucleonics, Docket F-40) is applicable to this reactor as are the Hazard and Safety Analysis Reports for other AGN-201 reactors on file with the NRC.

The characteristics and operating parameters of this reactor have been calculated and experimentally determined from data obtained from this and other AGN-201 reactors.

The critical mass of the reactor is approximately 665 grams of ^{235}U. Since the loading density is 61 mg/cm^3, the estimated core volume is 12.6 liters. Allowing for control and safety rods, etc., the right circular cylinder is 25.8 cm (10.15") in diameter by 23.75 cm (9.34") high. In order to facilitate critical assembly, the core is fabricated in four 3.9 cm (1.56") high sections and five thinner sections. The thinner sections allow variations of the amount of ^{235}U to be inserted in smaller increments in the range from 550 grams to 700 grams total core weight. Each of the four bottom fuel discs has four holes to permit penetration by the safety and control rods; two holes accommodate the safety rods and the other two holes accommodate the control rods. The glory hole, which is 2.38-cm (15/16") inside diameter, is located at the intersection of the two upper thick discs. The core discs are formed by pressing under high pressure a homogeneous powdered mixture of polyethylene and UO_2 (61 mg $^{235}U/cm^3$) in the form of 20 micron diameter particles. The total density of uranium is then 305 mg/cm^3 yielding a weight ratio of 1 gm of UO_2 to 3.16 gm polyethylene (assuming 20%-enriched ^{235}U).

The core tank has been designed to contain any radiolytic and fission-product gases that might diffuse out of the fuel discs. Sixty-five mil commercial (6061 T6) aluminum is used throughout as the structural material. The core tank may be considered to be made of an upper and lower section, separated by a thin aluminum plate in the same plane as the glory hole that bisects the main core cylinder. This baffle serves to separate the core into two halves, and is part of the fuel system. Detachable top and bottom cover plates as well as control- and safety-rod thimbles form an integral part of the core tank.

The lower section of the core tank contains approximately one-half of the core material as well as part of the graphite reflector. These core sections and the inner piece of graphite reflector are supported by an aluminum rod projecting downward from the thermal safety fuse link. Ample space at the bottom of the cylinder, coupled with a tapered graphite-to-graphite joint, is provided to insure free-fall of the bottom half of the core plus reflector sections, should the fuse melt from an accidental nuclear excursion. The upper section of the core tank is filled with five fuel discs and part of the reflector. A space for core expansion and gas accumulation is provided in the top section of the core tank.

The reflector consists of 20 cm of high density (1.75 gm/cm^3) graphite on all sides of the core. Appropriate holes are provided for the glory hole, the four safety and control rods, and the four access ports. The top cover of the core tank is removable, as is the top plug of graphite, to permit access to the core. The complete core tank may be removed, also. Ten centimeters of lead completely surround the core and the reflector. The graphite reflector weighs about 320 kg (700 lb), and the lead shield weighs about 3400 kg (7,500 lb).

The lead shielding, reflector, and core are enclosed in and supported by a thick steel tank (47.5 cm radius). The outer tank acts as a secondary container for the core tank assembly and is completely fluid-tight. A removable tank is provided over the top of the core to permit access to the core tank. This removable tank may be filled with water for shielding or with graphite when a thermal column is desired. The outer steel tank is completely surrounded on all sides by a water tank for shielding against fast neutrons. The control rods and safety rods enter through the bottom of the tank.

4.2 AGN-201 Characteristics

The AGN-201 is a low-power reactor that has the distinct advantage of small size, which enables it to be readily located in existing buildings. The reactor is employed for use in education and research. Because of the low operating power of the reactor, no external cooling system is required. The main design objectives were to make the reactor as safe and foolproof as possible, to use a minimum critical mass, and to provide a high analytical sensitivity.

The AGN-201 reactor system consists of two basic units, the reactor and the control console (Figure 4.2-1). The reactor unit includes the uranium-polyethylene core, graphite reflector, lead and water shielding, control rod drive mechanisms, and neutron detectors. Fuel-loaded control and safety rods are inserted vertically from the bottom of the reactor unit, passing by the instruments which measure the power level and the control mechanisms which provide for the safe and efficient operation of the reactor. The mass (weight) of the reactor unit, with the water shield in place, is 9100 kg (20,000 lb); the mass (weight) of

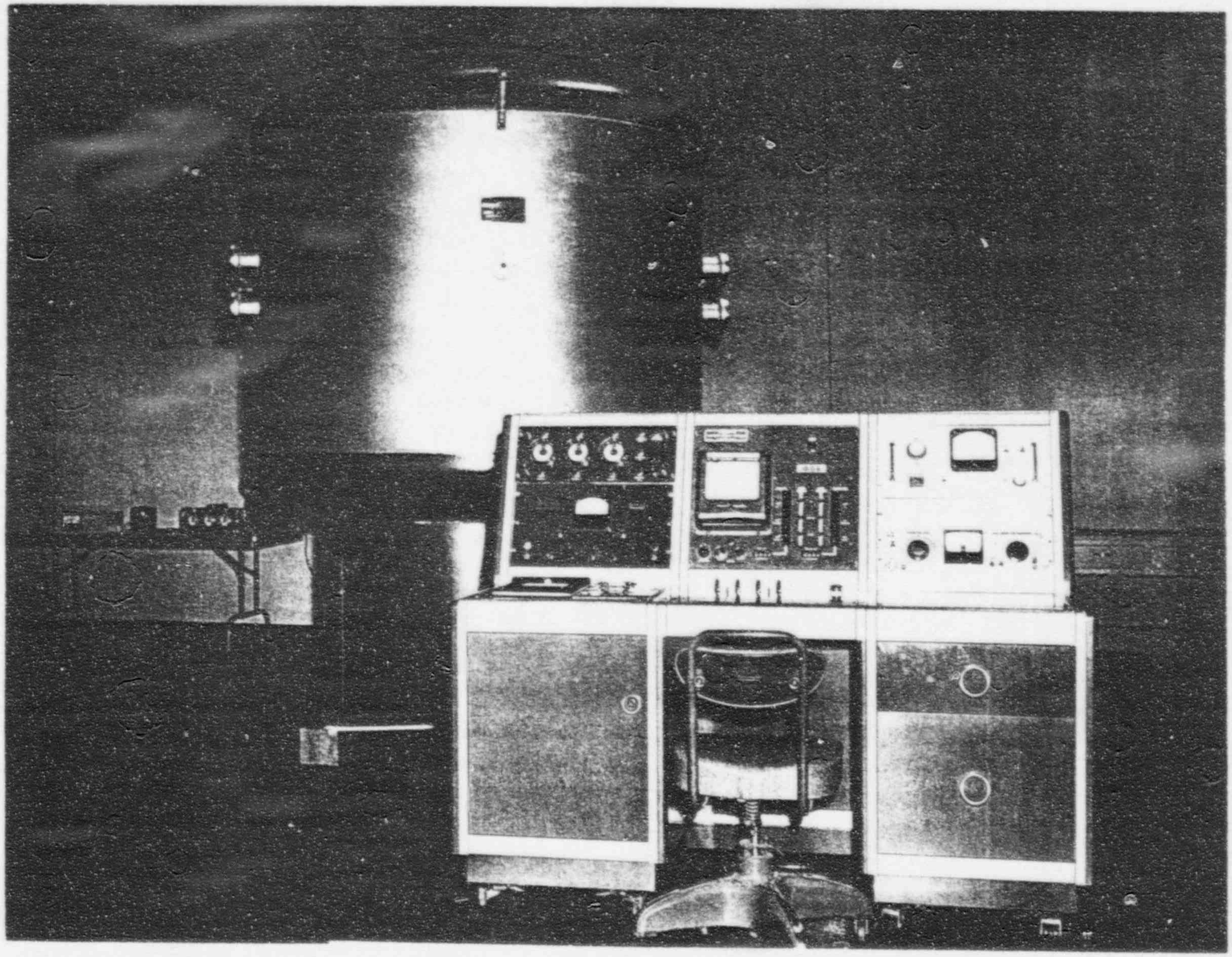


Figure 4.2-1 AGN-201 nuclear reactor and prototype control console.

the console unit is 360 kg (800 lb). Power requirements for operation of the control console and associated instrumentation are 2 kW of 110 VAC, single phase, 60-cycle electrical power.

The AGN-201 is a homogeneous, thermal reactor as shown in Figure 4.2-2. The core is made up of a series of nine circular discs, 25 cm in diameter and of varying thickness, which consist of UO_2 embedded in radiation-stabilized polyethylene. The core has a critical mass of approximately 665g ^{235}U. The cylindrical core is surrounded by a graphite reflector, 20-cm thick, and is shielded by 10-cm lead and 55-cm water. As shown in Figure 4.2-3. the core and part of the reflector are contained in an aluminum tank which has re-entrant thimbles in its base, into which the four control and safety rods are inserted. Each rod contains fuel material sealed in an aluminum capsule, thereby increasing reactivity as the rods are inserted into the core. Both safety rods and one control rod are held in position by electromagnets for scram purposes. Each of the four rods is driven by a lead screw.

The core is divided into half-sections with a plane of separation occurring at the level of the one-inch diameter glory hole that passes completely through the core. The lower section is held in place by a thermal fuse link which is designed to soften at 100° C, and when it functions as designed, permits the bottom core section to drop 5 cm (2") to the bottom of the core tank. This separation of the core results in a reduction in reactivity of from 5 to 10%. The excess reactivity is limited to a maximum of 0.24% $\Delta k/k$ or a minimum period of about 15 sec by the amount of fuel available.

The two safety rods and the coarse control rod are each worth approximately 1.25% $\Delta k/k$ ($1.69). The fine control rod, which can be loaded with either fuel material or polyethylene, is currently loaded with fuel and worth about 0.31% $\Delta k/k$ ($0.42). The neutron flux is monitored by three independent detectors: two BF_3 ionization chambers and a BF_3 proportional counter, all of which are located in the water tank just outside the lead shield. The detectors are connected respectively to a logarithmic micromicroammeter, a linear micromicroammeter, and pulse amplifier and count rate meter located in the reactor console. Each indicator is connected through a sensitrol relay to a scram circuit. Additional safety interlocks provide for reactor shutdown if the level of the shield water drops, if the reactor temperature falls below 15° C, or if an earthquake occurs. Sequential interlocks are also present to ensure that the proper operational method is followed. Detailed characteristics of the reactor are given in Table 4.2-1.

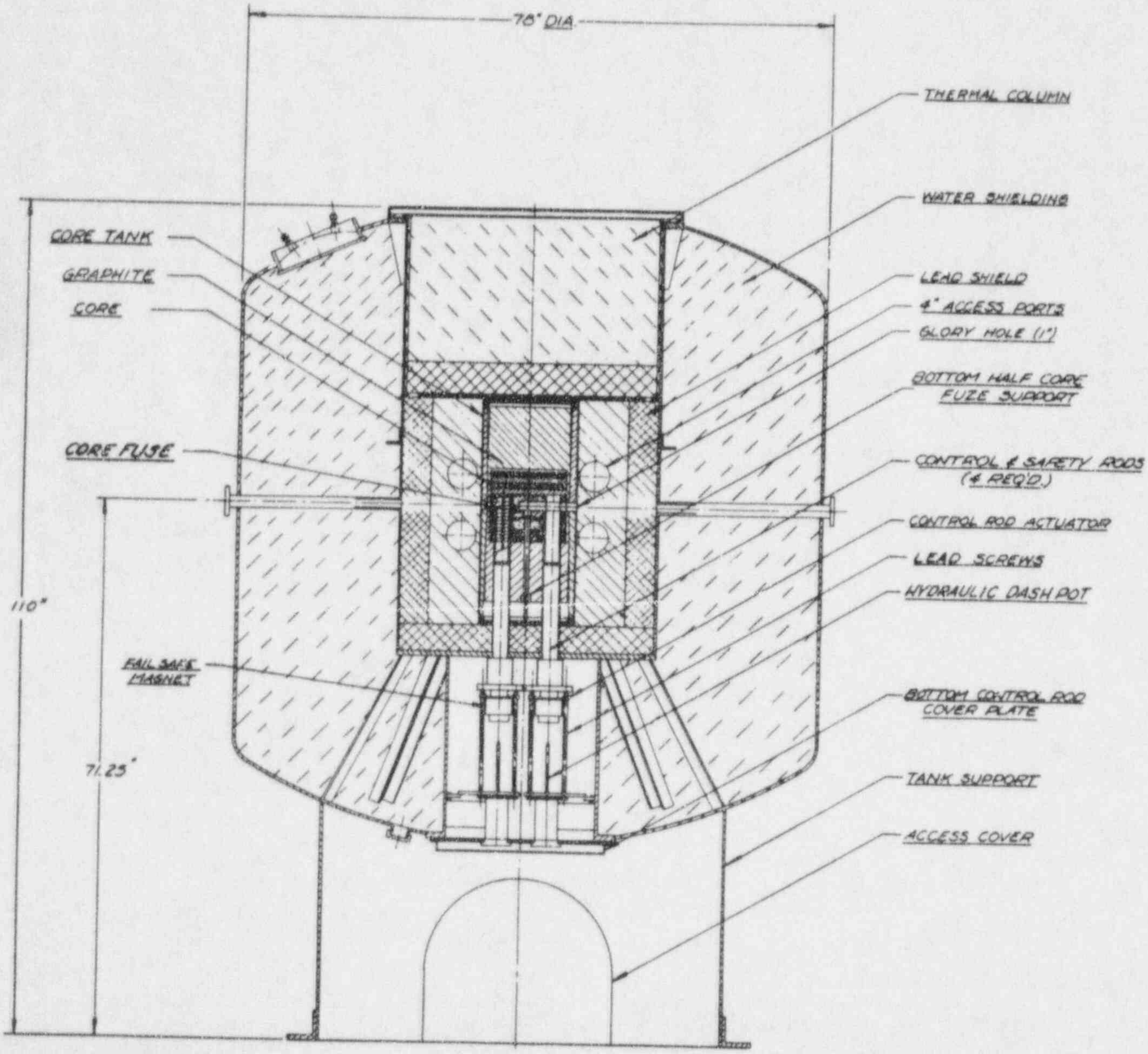


Figure 4.2-2 AGN-201 reactor unit.

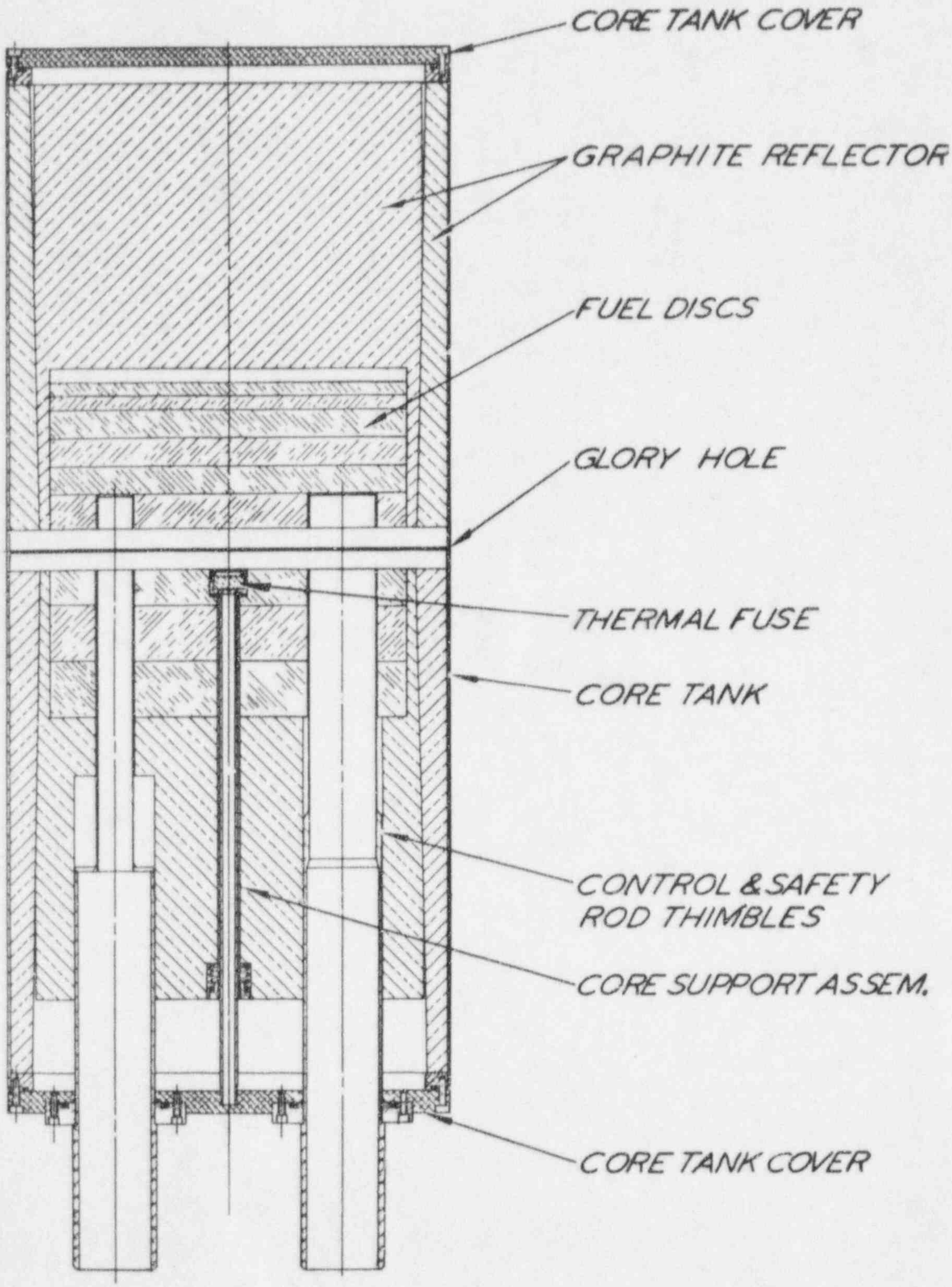


Figure 4.2-3 AGN-201 core tank and contents.

Table 4.2-1
Reactor Characteristics

a. General	
Type	Homogeneous thermal reactor
Principle Uses	Education and Training
Maximum Operating Power	5 watts
Core	UO_2 (20%-enriched in ^{235}U) particles homogeneously distributed in solid polyethylene moderator, cylindrical geometry
Reflector	Graphite
Shield	Lead and water
Control	Two safety and two control rods, all fuel loaded
b. Fuel	
Fuel Material	20%-enriched UO_2
Fuel Disc	Right circular cylinder fabricated from a compressed mixture of fuel material and polyethylene powder Specifications: UO_2 powder ^{235}U enrichment - 19.5 $\pm$ 0.5% Particle size - 15 $\pm$ 10 μm Polyethylene powder: Particle size - 100 μm Purity - commercial grade
Approximate Core Fuel Disc Sizes and ^{235}U Content	9 in number (25.6-cm diameter) Four discs 4-cm high (96 g ^{235}U ea) Three discs 2-cm high (58 g ^{235}U ea) Two discs 1-cm high (29 g ^{235}U ea)

Table 4.2-1 (cont'd)

Approximate Control and Safety Rod Fuel Disc Sizes and Total ^{235}U Content for All Safety and Control Rods	12 discs 4.7-cm diameter & 4-cm high (3.6 g ^{235}U ea) 4 discs 2.3-cm diameter & 4-cm high (0.9 g ^{235}U ea)
Total Fuel Loading	672.93 g
^{235}U Density	0.061 g $^{235}U/cm^3$
Normal Lifetime of Fuel Discs	Indefinite
Core Fuse	Small, right circular cylinder, 2.2-cm diameter & 0.9-cm high, 0.40 g ^{235}U. Fabricated from mixture of fuel material and polystyrene powder.

c. <u>Reactor</u>

Core-containing Vessel	Gas-tight, 65-mil (6061T6) aluminum cylindrical tank, 32.2-cm diameter and 76-cm high. Vessel has a 2.54-cm glory hole on the horizontal center line and appropriate control rod thimbles. Tank contains core fuel discs and graphite reflector plugs.
Core description	9 fuel discs, 25.6-cm diameter, 23.8-cm high, separated in half at glory-hole mid-plane by a thin aluminum baffle. Lower half of core contains appropriate holes for control and safety rod thimbles and is part of the safety fuse system.
Reflector 20	Heavy-density (1.75 g/cm^3) graphite, approximately cm on all sides of core. Top and bottom reflector in core tank, side reflector surrounds core tank and contains four 10-cm through-holes running tangentially to core tank.
Gamma Shield	10 cm of lead completely surrounding reflector.
Reactor Tank	80-mm-thick (5/16") steel tank, 95-cm diameter, and 148-cm high; contains core tank, reflector, and lead shield, and appropriate holes for control and safety rods, glory hole, and access ports. Gas-tight vessel when all seals are made. Upper portion of tank contains a removable thermal column tank.

Table 4.2-1 (cont'd)

Fast Neutron Shield	55-cm of water surrounds the reactor tank except at its bottom face.
Water Tank	Steel tank serving as the main structural tank, 198-cm (6.5 ft) diameter and contains 3800 l (1,000 gal) of water. Tank is supported at its bottom by the reactor skirt.
Reactor Dimensions	198-cm (6.5 ft) diameter and 290-cm (9.5 ft) high.
Reactor Weight	6800 kg (15,000 lb) (less shield water).
Reactor Control	Two safety rods, 14.4-g ^{235}U each. One coarse control rod, 14.4-g ^{235}U. One fine control rod, 3.6-g ^{235}U. All rods, with the exception of the fine control rod, which is mechanically coupled, are magnetically coupled to carriage which is driven in a vertical direction on a lead screw by a reversible DC motor. Total travel distance is about 25 cm.
Experimental Facilities	2.22-cm (7/8") glory hole passing through the center of core at core median plane. Four, 10-cm-diameter (4") access ports passing through the graphite reflector tangentially to the core. Thermal column tank above the core.
Neutron Source	Ra(Be) source utilizing (α,n) reaction; 10-mg $RaCl_2$ mixed homogeneously with Be powder. Neutron yield is approximately 10^5 n/sec.

d. <u>Nuclear Data</u>

(1) Fuel Loading

(a) Approximate Critical Mass 665 g ^{235}U

(b) Excess Reactivity at 20°C
with Glory Hole Empty 0.18% $\Delta k/k$ (\$0.24)

Table 4.2-1 (cont'd)

(2) Neutron Flux	
Average Thermal Flux	1.5×10^8 n/cm^2-s at 5 W
Peak Thermal Flux	2.5×10^8 n/cm^2-s at 5 W
(3) Reactivity Worth of Reactor Components	
(a) Safety and Coarse Control Rods	1.25% $\Delta k/k$ ($1.69) (each)
(b) Fine Control Rod	
Fuel-loaded	0.310% $\Delta k/k$ ($0.42)
Polyethylene-loaded	0.155% $\Delta k/k$ ($0.21)
(c) Standard Fuel filling Glory Hole	
At Core Surface	0.042% $\Delta k/k$ gm^{-1} ($0.06)
At Core Center	0.100% $\Delta k/k$ gm^{-1} ($0.14)
(d) Polyethylene in Glory Hole (completely filled)	0.29% $\Delta k/k$ ($0.39)
(e) Access Port Plugs	
1 Wood Plug	0.002% $\Delta k/k$ ($0.003)
1 Section Pb	0.015% $\Delta k/k$ ($0.02)
1 Section Graphite	0.194% $\Delta k/k$ ($0.26)
Total Worth of Plugs in one Access Port	0.422% $\Delta k/k$ ($0.57)
(f) Temperature Coefficient of Reactivity (Approximate)	-0.025% $\Delta k/k$ °C^{-1}
(g) Reactor Sensitivity at Core Center, Measured with 1/*v* Absorber	-0.14% $\Delta k/k$ cm^{-2}

(4) Pertinent Figures

(a) Control Rod Calibration Given in Figure 4.2-4

(b) Inhour Equation Given in Figure 4.2-5

(c) Flux Plot Given in Figure 4.2-6

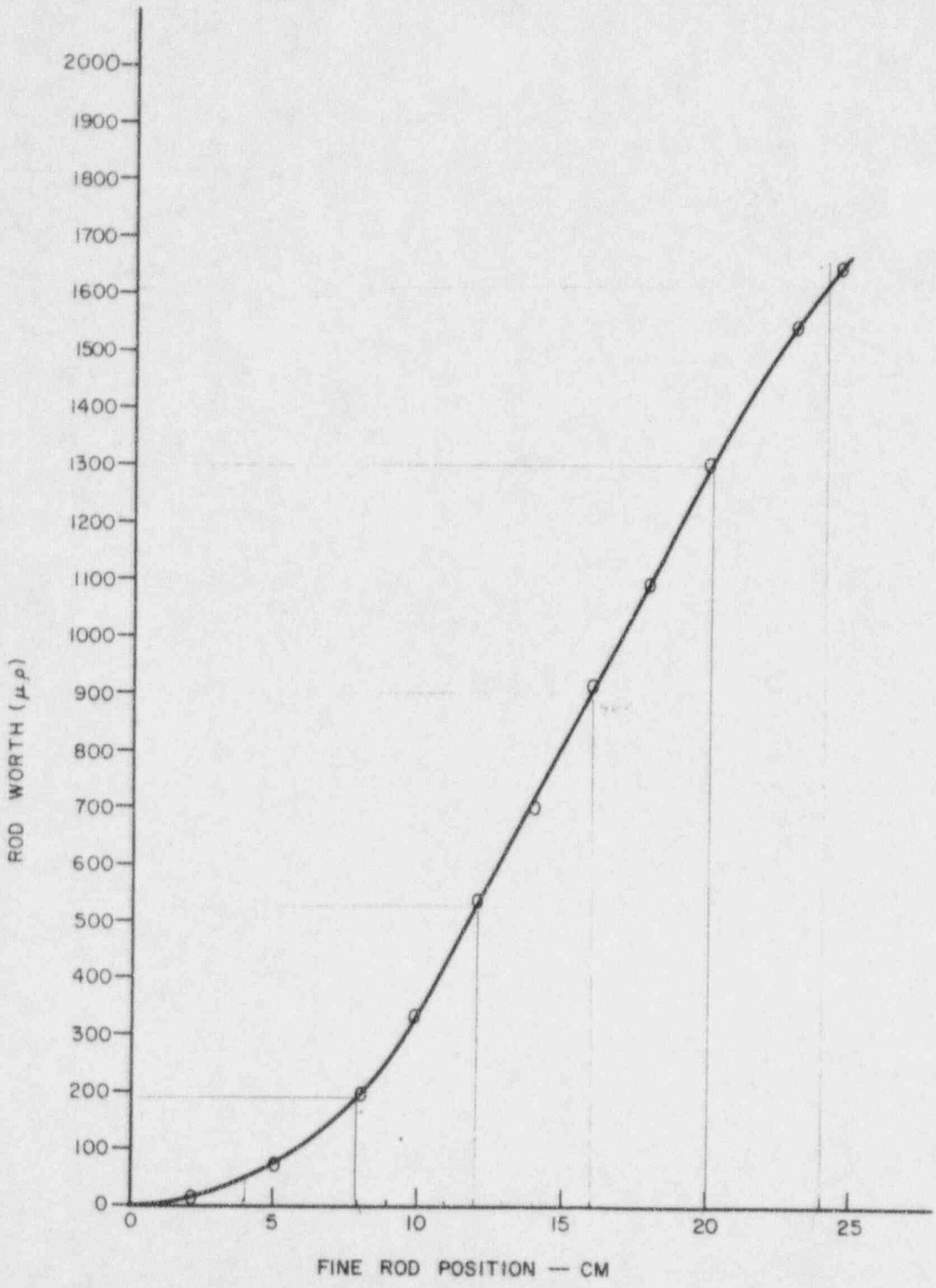


Figure 4.2-4 Fine control rod calibration curve.

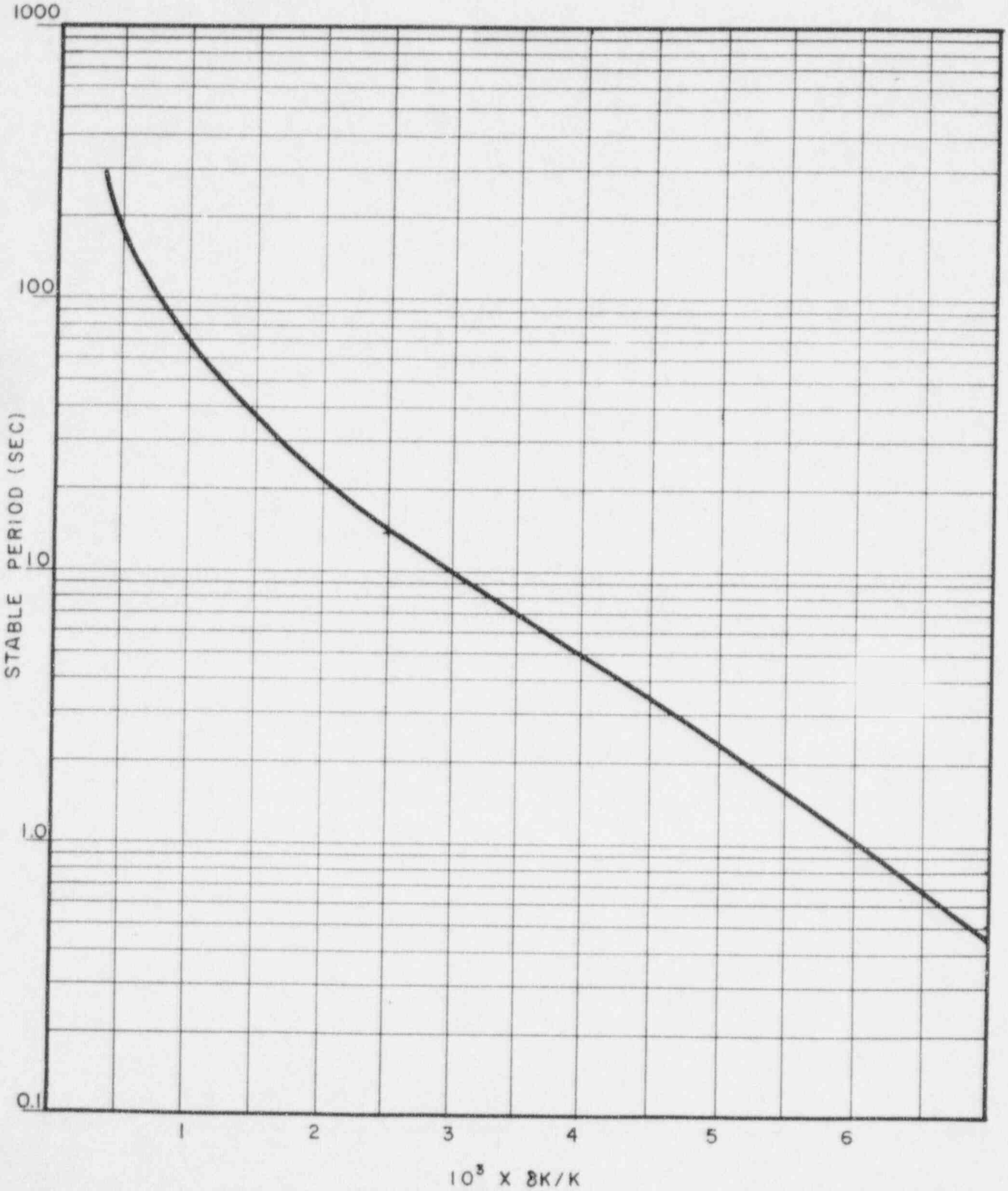


Figure 4.2-5 AGN-201 Inhour equation.

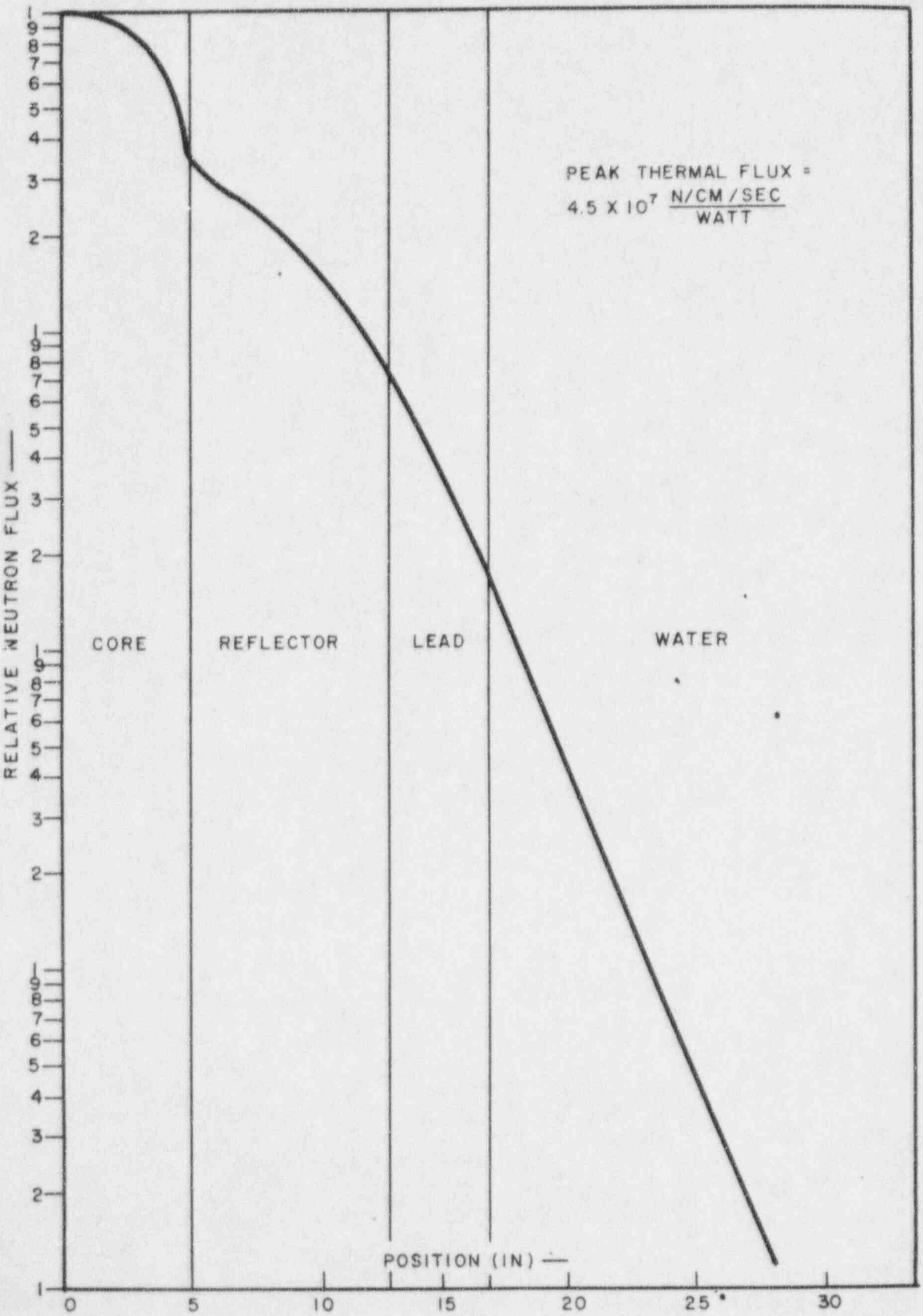


Figure 4.2-6 Horizontal thermal neutral flux through glory hole at 100 mW.

4.3 Control

4.3.1 Control rods

The AGN-201 reactor has two safety and two control rods. Three of these, the two safety rods and the coarse rod, are identical in design although their functions are different. Each contains about 15 grams of ^{235}U and operates in a manner such that reactivity is increased as the rod is inserted. The amount of reactivity each rod controls is nearly proportional to the amount of fuel material contained in the rod. With the same uranium concentration in the rods as in the core, and a total of 14.4 grams of ^{235}U in each rod, 1.25% of the reactivity is contained in each of the three large diameter rods. The fine control rod is smaller in diameter and is loaded with fuel material for a reactivity worth of 0.31%. Its function is to permit fine adjustment of the power level of the operating reactor.

Figure 4.3-1 shows the design of an active rod which fits into a 5-cm (2-in) diameter hole in the core. The active length of the rod is 15 cm of UO_2 embedded in radiation-stabilized polyethylene identical in composition to the reactor core. This active fuel material is enclosed in two aluminum containers, the outermost providing the fluid seal from the core tank, and the innermost aluminum container sealing the active fuel material in the rod. By this design, a double fluid-tight seal is maintained for the core as well as for the control and safety rods.

For small adjustments of the reactivity of the reactor, the control rods offer a convenient method of adding or removing fuel. As such, they are designed for use during start-up as well as for small adjustments in available reactivity under normal operating conditions.

The maximum ^{235}U in the rods (at 61 mg/cm^3) is 14.4 gm (266 cm^3) which control 1.25% reactivity. For safety reasons, the safety rods contain more than 5 gm of ^{235}U: the reactor will remain subcritical if any of the large-diameter rods is fully withdrawn from the core.

The safety rods are in the safe or subcritical positions when hey are in their outermost, fully-withdrawn position. The total distance of travel is about 25 cm and in the fully-withdrawn position the active fuel in the rod is just inside the lead shield and partially within the graphite reflector. The rods are inserted sequentially by a drive mechanism. The release mechanism is constructed in such a manner that if a scram signal is received during insertion, both rods are instantaneously ejected to their outmost positions.

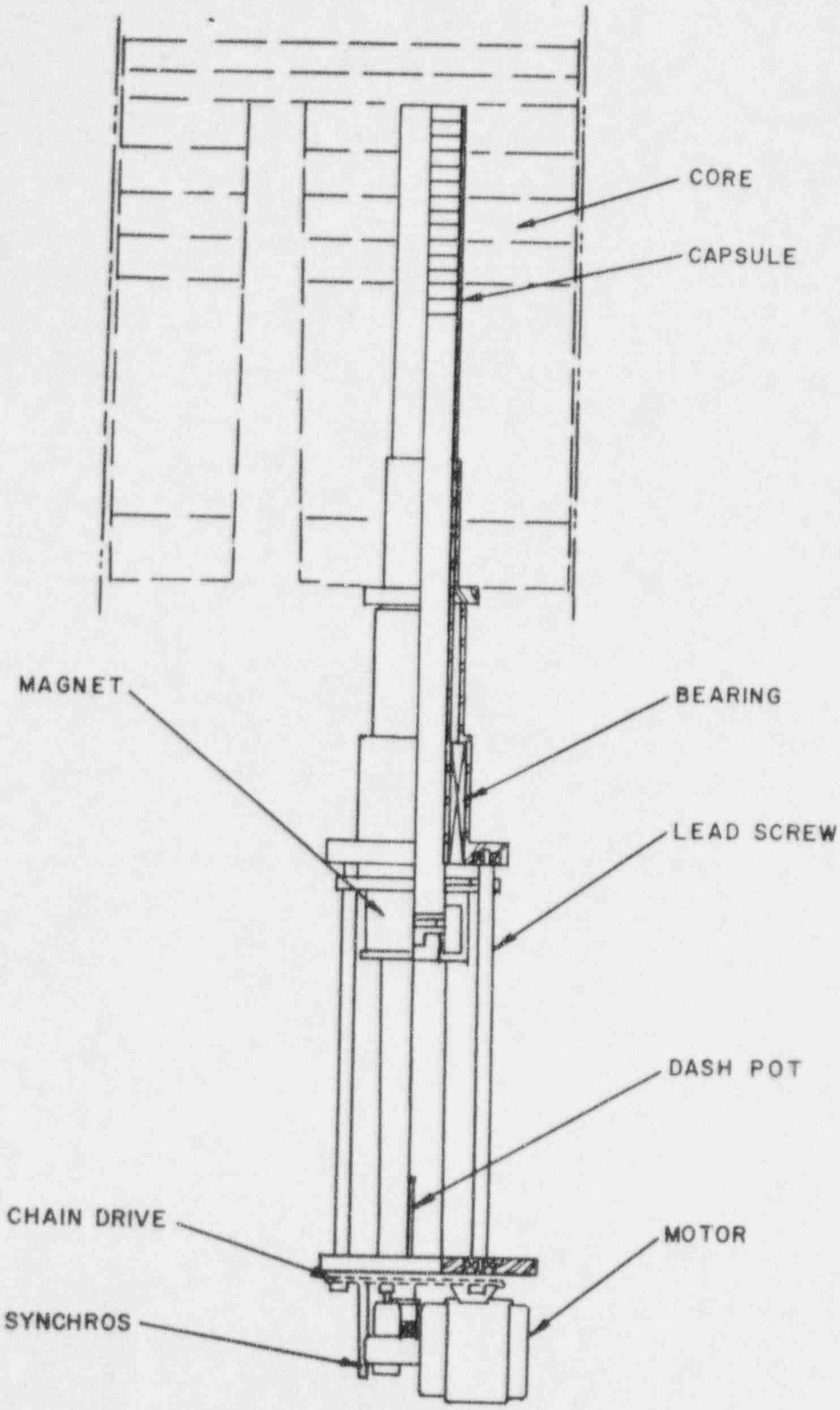


Figure 4.3-1 AGN-201 control rod and drive mechanism.

The safety system is a "fail safe" design in that the scram signal opens the holding electromagnets allowing the rods (except for the fine control rod) to be accelerated downward and out of the core by both gravity and spring loading. The spring constant is such that the rods are initially accelerated with a force of 5 g, requiring a total ejection time of approximately 120 milliseconds. The reactivity change of both safety rods is minus 0.7% ($0.95) during the first 50 milliseconds. Each scammable rod is decelerated by an hydraulic (oil-filled) or pneumatic dashpot during the last 10 cm of travel.

Both the coarse and fine control rods are driven by reversible DC motors through lead screw assemblies which are controlled by switches at the control console. The maximum speed of travel of the rods is 1.0 cm/sec, yielding a maximum reactivity change of $2 \times 10^{-4}\%\ \Delta k/k\ sec^{-1}$ for the coarse rod. The positions of both control rods are indicated on the control console. In the event of a scram, the coarse rod and safety rods are automatically and instantaneously ejected to the safe position. At startup, interlocks prevent coarse rod movement unless the safety rods are fully inserted or "cocked". The safety rods cannot be cocked until the control rods have reached their safe or starting positions. The fine control rod has too little reactivity to be of practical value in the event of a scram. Consequently, on receipt of a scram signal it is driven out of the core by its reversible DC motor at the rate of about 0.5 cm/sec.

4.3.2 Instrumentation System

The instrumentation for the AGN-201 reactor comprises three neutron-sensitive monitors: one BF_3 proportional counter and two BF_3 ionization chambers. These neutron detectors are located in the shielding water tank just outside the reactor tank (as shown in Fig. 4.3-2). They are placed in water-tight aluminum cans and can be removed from the reactor through the access cover at the top of the water tank.

The control system is very similar to that used for other low power reactors (1) and is shown schematically in Figure 4.3-3. The ion chambers are connected through micro-microammeter amplifiers to a relay and recorder system. Similarly, the BF_3 counter is connected to a counting ratemeter and, in turn, to the relay system. The relays that form the scram bus may be set to scram the reactor at any predetermined instrument reading. Scrams may also be initiated by the earthquake detector, the water shielding level detector, water temperature detector, and the manual scram button.

The recorder may be switched to monitor any of the three neutron detection devices. The #1, BF_3 counter may be connected to a scaler for

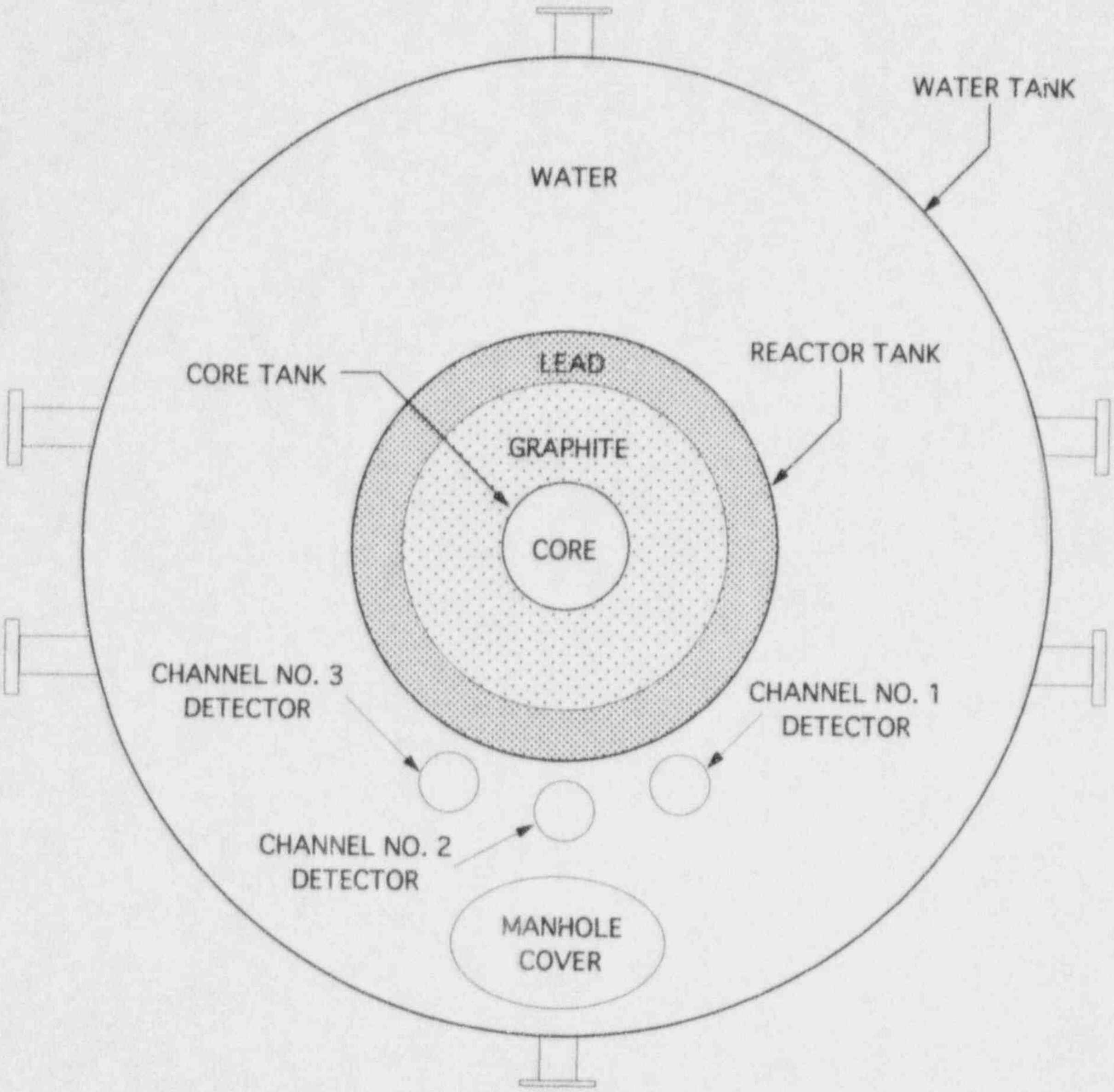


Figure 4.3-2 Cross section of reactor showing locations of neutron detectors.

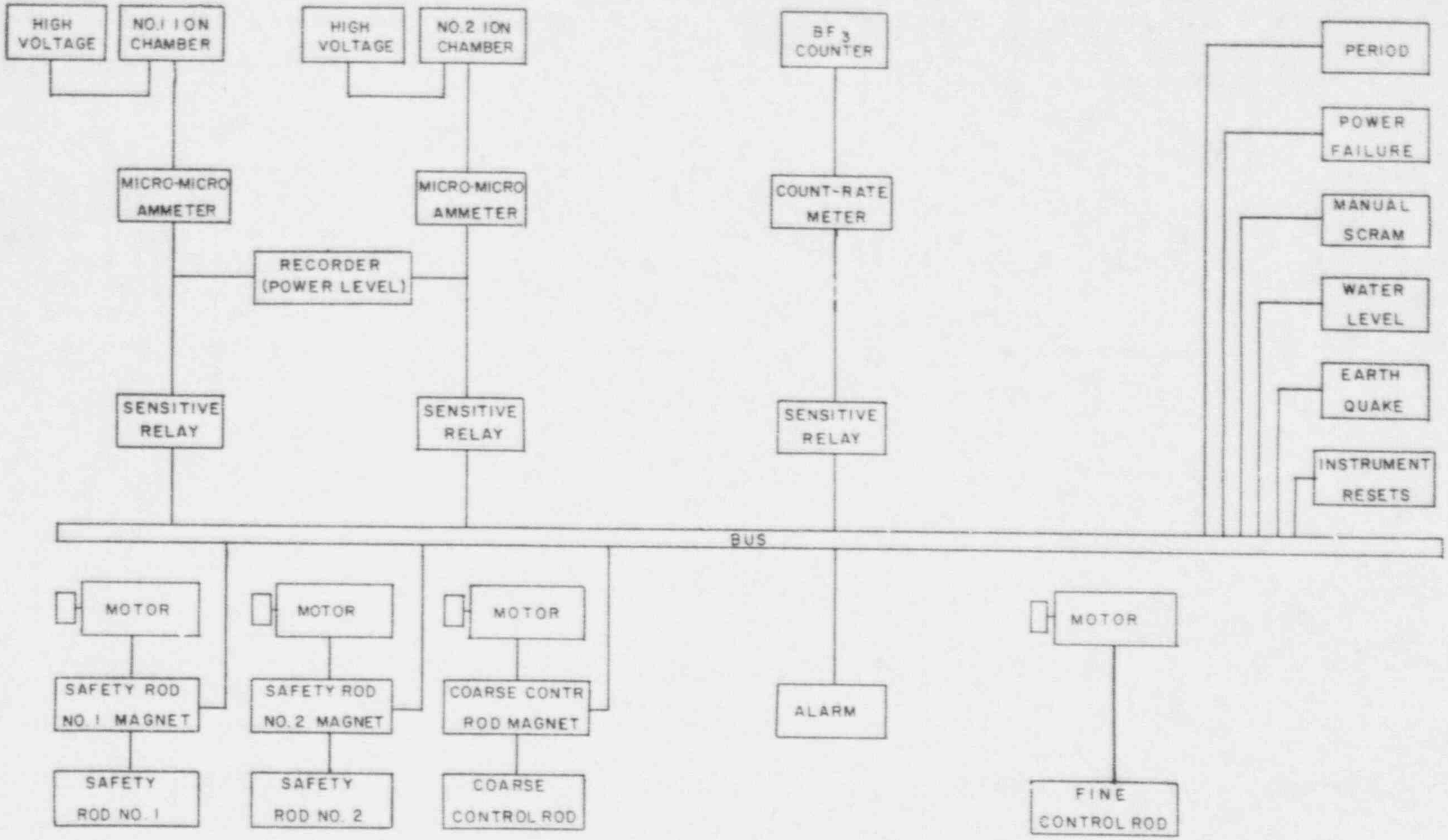


Figure 4.3-3 AGN-201 control system.

a more accurate determination of the counting rate than can be read on the counting rate meter. The time constant of the ion chamber circuit may be varied from 0.5 to 40 seconds.

In the event of an abnormal condition, any one of the three detectors or other safety devices actuates the scram system causing automatic and rapid ejection of both the safety rods and the control rods, and operation of the scram warning light. The AGN-201 system is designed to fail safe. In the event of an external power failure, the reactor will scram. A criticality monitor will continue to operate in a power failure to provide the reactor operator with continuous radiation readings. The control console for the AGN is shown in Figure 4.3-4.

Channel No. 1. The components of the nuclear safety Channel No. 1 neutron monitoring system are shown in Figure 4.3-5. The primary purpose of Channel No. 1 is to provide low-level protection so that the reactor cannot be started up without a neutron source. A secondary purpose, although not required by the Technical Specifications, is to provide an additional high-level trip mechanism.

The detector for Channel No. 1 is a $^{10}BF_3$-filled proportional counter. This detector is mounted in a buoyant plastic tube which is held down in the water shield by a solenoid arm during reactor startup. At power levels greater than 0.1 watt, the plastic tube is allowed to float upward to an upper, fixed position by activating the solenoid switch. This control provides a two-position setting for the detector, allowing Channel No. 1 to be operable over the entire power range of the reactor. The sensitivity of the detector has been adjusted by the installation of a partial cadmium jacket so that in the lower position it provides a few counts per second with the source in the proper position and about 9,000 counts per second when the reactor is at 5 watts.

A preamplifier is located in the cable tray on top of the reactor for signal conditioning of the pulses from the detector. Power for the preamplifier is provided through the signal cable which extends from the preamplifier to the linear amplifier in the control console. Therefore, no additional power cord is required for the preamplifier.

The high voltage power supply required for operation of the detector is mounted in the Channel No. 1 NIM-bin in the console. A coaxial cable delivers the high voltage to the preamplifier.

The signal from the preamplifier enters the calibration check chassis before it reaches the linear amplifier. This chassis provides a means of testing channel operation prior to startup. The test switch on this chassis

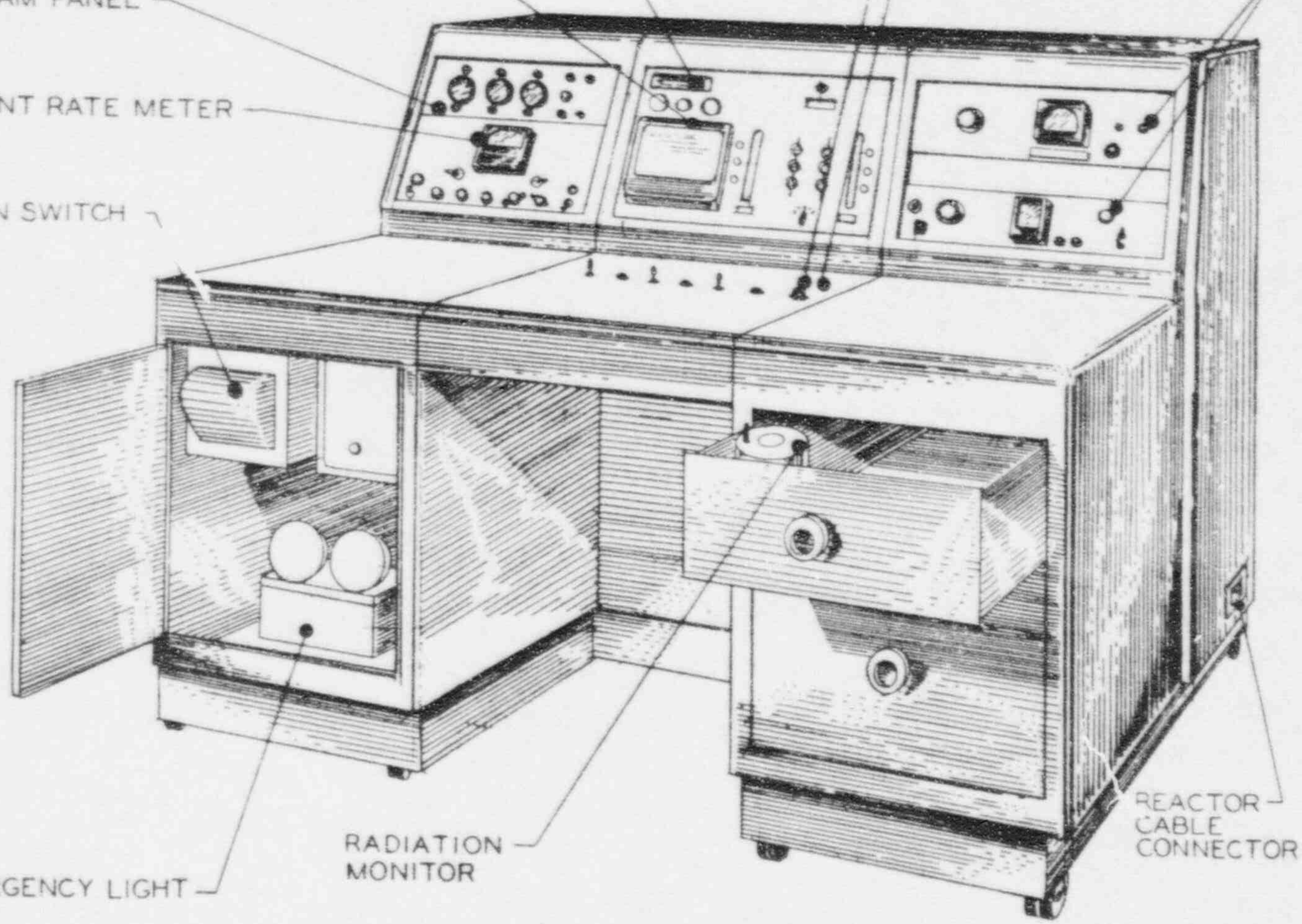


Figure 4.3-4 Original equipment AGN-201 control console.

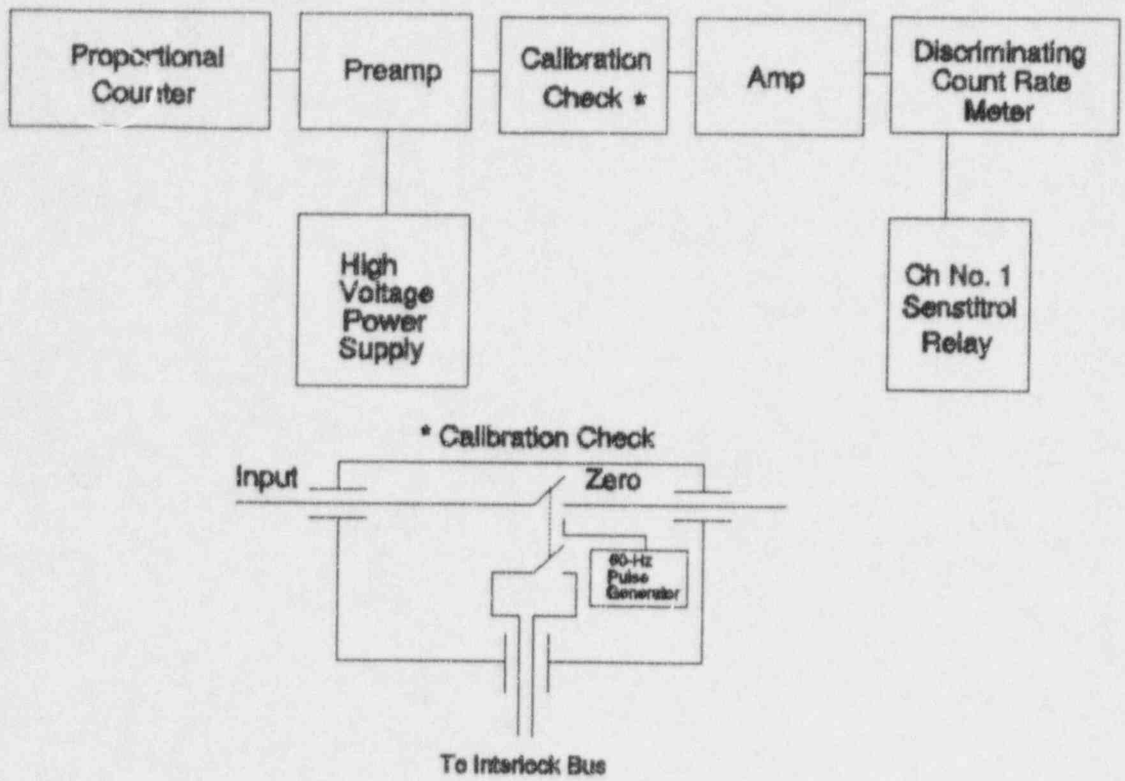


Figure 4.3-5 Block diagram for nuclear safety channel No. 1.

provides "Zero", "Calibrate" (60-Hz signal), and "Operate" positions. Calibration of the instrument meter is made with an internal 60-Hz circuit. If the switch is in any position other than "Operate", the scram interlock bus is opened. This chassis also contains a push button that activates the solenoid to raise the proportional counter to a less sensitive position.

The linear amplifier, which is also located in the NIM bin, provides the pulse amplication necessary to drive the count rate meter.

The count rate meter, also located in the NIM bin, has an internal discriminator that is set at two volts. The range of the meter varies from 10 to 10,000 counts per second with a selector switch providing three scales per decade in a 1-2-5 sequence. The output of the count rate meter drives the coil of the Channel No. 1 sensitrol relay on the Safety Chassis. This sensitrol provides both high and low trips. A scaler may be connected to the count rate meter for a more accurate measurement of the neutron count rate than can be read from the count rate meter.

Channel 2. The components of Channel No. 2 are shown in Figure 4.3-6. The primary purpose of Channel No. 2 is to provide both high- and low-level protection on a non-switchable neutron monitoring channel. A secondary purpose is to provide backup power level indication.

The Channel No. 2 detector is a $^{10}BF_3$-filled ionization chamber. This detector is mounted in a water-proof housing in the water shield tank. Two coaxial cables, one for the detector signal and one for high voltage, extend from the detector through the cable tray to the control console. The high voltage power supply for Channel No. 2 is mounted on the rear of the console.

Channel No. 2 uses a Keithley Model 420 Log n Amplifier and Period Meter for detector signal processing. The output from one of the BF_3 ionization chambers is fed to a logarithmic micromicroammeter located in the reactor console. This ammeter is a vacuum-tube electrometer covering the range from 10^{-13} to 10^{-6} amperes without range switching. The logarithmic scale is obtained by negative feedback from output to input with a diode whose emission is limited by negative plate-to-cathode voltages. The ammeter provides an internal calibration circuit consisting of a mercury battery and appropriate resistors. A calibration switch is provided with positions for "Operate", "10^{-7}", "10^{-11}", and "Amplifier Balance". If the calibration switch is in any position other than the "Operate" position, the scram interlock bus opens. Channel calibration is performed by adjusting potentiometers located on the instrument front panel to a corresponding calibration signal read from the channel display meter. The instrument has a period range from infinity to 3 seconds.

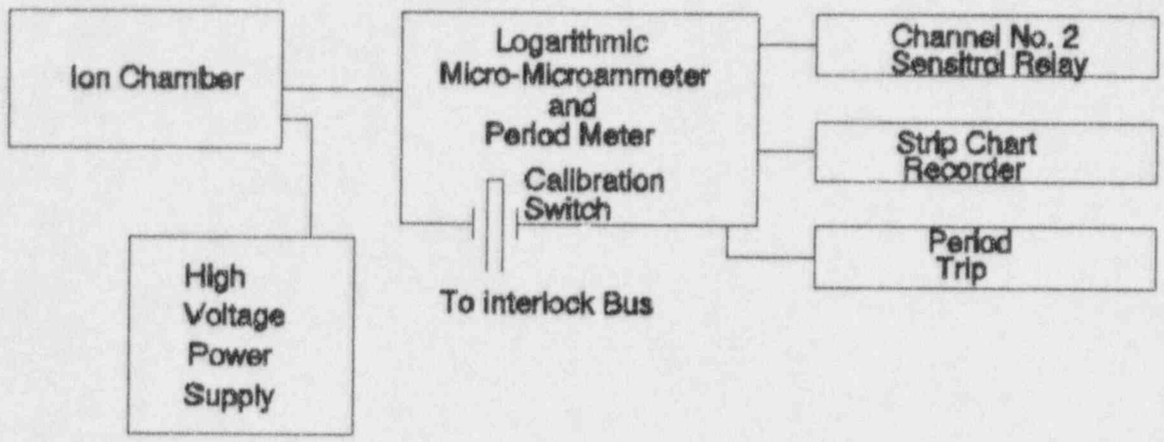


Figure 4.3-6 Block diagram for nuclear safety channel No. 2.

The output signal from the Channel No. 2 ammeter is fed into three devices. The signal is connected to the Channel No. 2 sensitrol relay on the Safety Chassis which provides high- and low-level trip protection. The signal is also provided to a period meter and trip circuit which actuates at a preset period greater than or equal to 5 seconds. In addition, the signal drives a strip chart recorder located in the console.

Channel No. 3. The components of Channel No. 3 are shown in Figure 4.3-7. The primary purpose of Channel No. 3 is to provide both high- and low-level protection on a switchable neutron monitoring channel. A secondary purpose is to provide power level indication.

The Channel No. 3 detector is a $^{10}BF_3$-filled ionization chamber. The detector is mounted in a water-proof housing in the water shield tank. Two coaxial cables, one for the detector signal and the other for the high voltage, extend from the detector through the cable tray to the control console.

The high voltage power supply for Channel No. 3 is mounted on the rear of the console. The output signal of the BF_3 ionization chamber is fed to a linear micromicroammeter located in the reactor console. This ammeter, a Keithly Model 410, has a current range from 3×10^{-13} to 10^{-3} ampere. The maximum current used during reactor operation is 3×10^{-7} Amp. A mechanical stop has been inserted in the range selector switch to prevent higher ranges from being selected. The instrument is a vacuum tube electrometer designed for measuring small currents. A "Zero Check" push button is provided so that the instrument zero may be examined and adjusted prior to startup.

Two outputs from the amplifier are provided. One is connected to the Channel No.3 sensitrol relay in the Safety Chassis which provides both high- and low-level trip protection on any range. The output is also fed to a strip chart recorder located on the console.

Recorders. Two strip chart recorders are located in the center panel of the reactor console. One recorder is connected to and registers the output signal from the logarthmic amplifier of Channel No. 2. The other recorder is connected and registers the output signal from the linear amplifier of Channel No. 3. The main recorder controls consist of an attenuation or damping control, a zero-adjust control, and a calibration control for adjusting the span of the instrument. The recorders also have variable chart speeds.

Safety Chassis. The Safety Chassis provides control of the current to the electromagnets which support the two safety and coarse control rods in

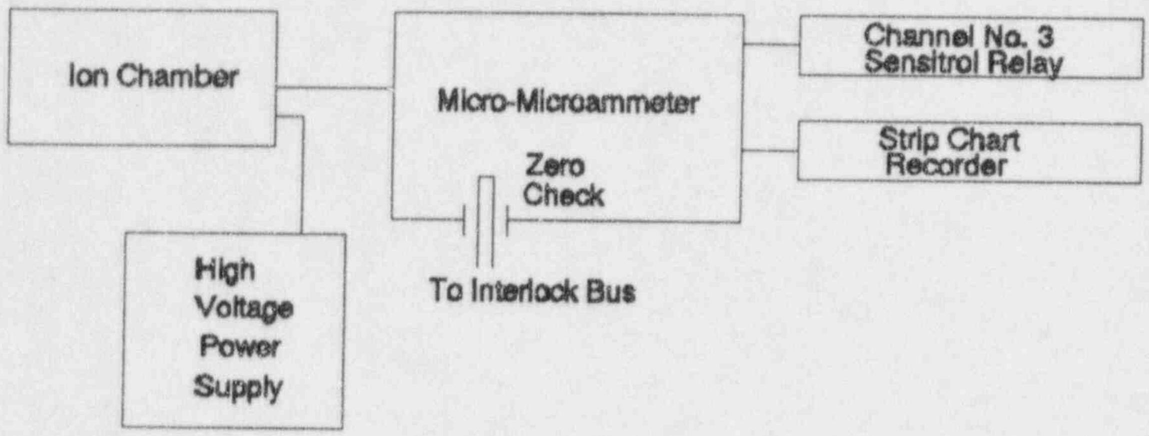


Figure 4.3-7 Block diagram for nuclear safety channel No. 3.

the reactor. The Safety Chassis is located in the upper left corner of the control console. A simplified circuit diagram of the Safety Chassis is shown in Figure 4.3-8.

The electromagnet current is controlled by a 6L6 electron vacuum tube that is operated as an electronic switch. The three electromagnets of the safety and coarse control rods are in series with the plate circuit of the 6L6 tube. When the 6L6 is in the conducting mode of operation, plate current flows through all three electromagnets. When the 6L6 tube is not conducting, no current flows to the electromagnets.

Conduction through the 6L6 is controlled by the grid voltage. In the normal conducting state, the grid voltage is established by the cathode resistor in the circuit. This resistor may be varied to adjust the magnitude of the magnet current. The adjustment is internal and does not need to be reset unless a new 6L6 tube is installed.

In the nonconducting or scram state, the grid voltage is made negative by about 50 volts by applying voltage from the -90 volt supply through appropriate resistors. The tube thus cuts off current to the electromagnets and causes them to de-energize. The scram state is actuated by any of the switches shown on the circuit diagram or by opening any of the interlocks in the scram interlock bus.

When the scram state is initiated, a magnet-current reversal relay discharges a large capacitor through the electromagnets in the opposite direction from that in which current to the electromagnets normally flows.

This reverse current coupled with the residual magnetism present in the ferrous rod support plate provides an immediate repulsive force that breaks the rod support plate free from the electromagnet thereby ensuring prompt rod release and ejection from the core.

The overcurrent meter relay is mounted on the rear of the console rather that in the Safety Chassis. It provides protection from the possibility of a grid to cathode short in the 6L6 vacuum tube. Should such a condition occur, the tube current would increase and could not be shut off by making the grid negative. The overcurrent relay, which is designed to sense this increase in current, would disconnect the 260-volt plate supply from the tube which would thereby terminate current flow.

The reset relay ensures that once plate current is stopped it cannot be restarted until the reset button is pushed, even though the cause of the scram condition may have been cleared.

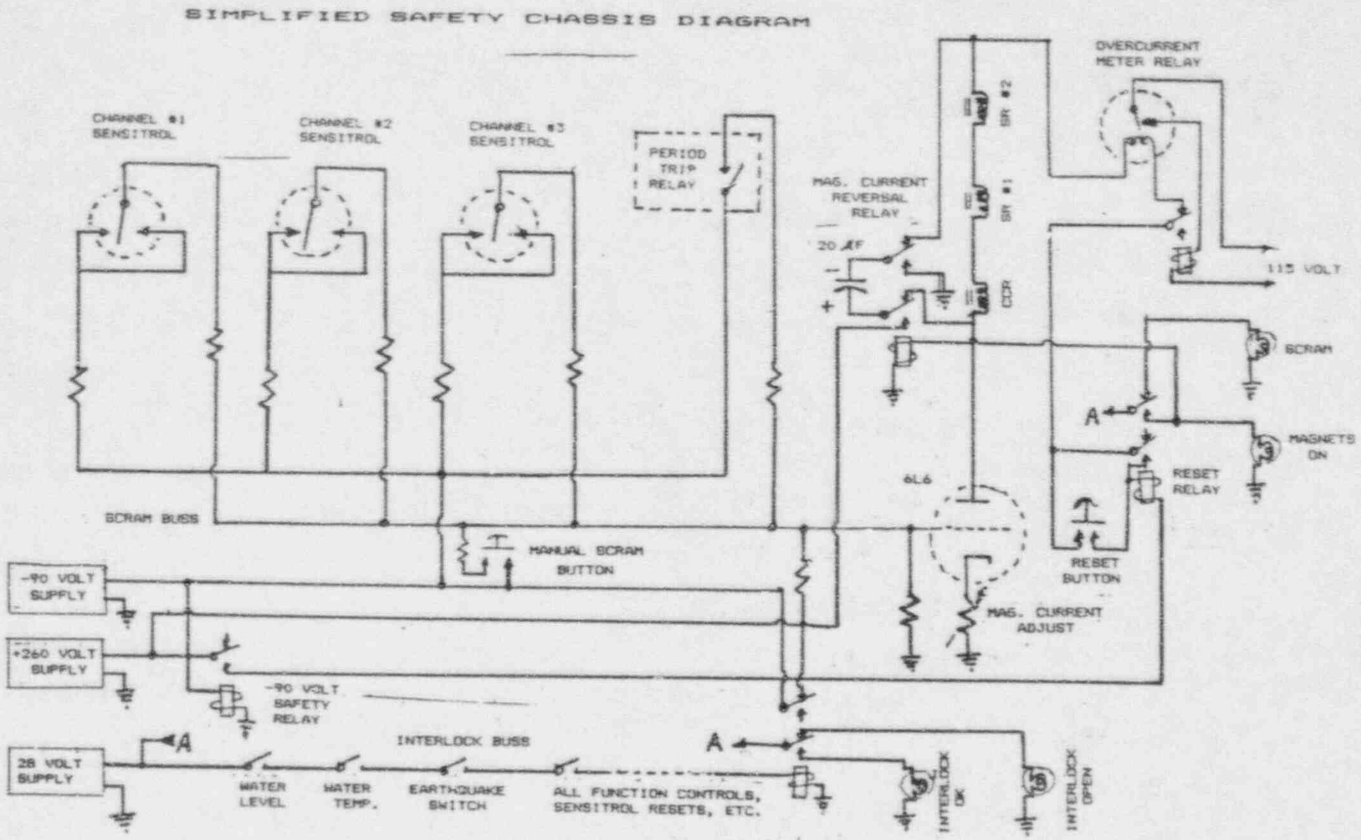


Figure 4.3-8 Simplified circuit diagram of Safety Chassis.

4.3.3 Reactor scram system

The main scram system, which is put to use when a high- or low-level trip in any of the sensitrol relays occurs, utilizes a negative voltage supply which cuts off the current passing through an electron tube. This method of scramming the reactor is also used when a period trip occurs or the interlock circuit opens. Figure 4.3-9 below is a simplified diagram of the Reactor Scram System.

When negative voltage is applied to the control grid of the 6L6 vacuum tube, the plate current which is used in part to energize the safety and coarse rod magnets is terminated causing the safety and coarse control rods drop out of the core to their down, fully-withdrawn positions. Scram relay K-4 also opens, lighting the scram warning light, and turning off the "Magnets On" light. As seen in the diagram, if the negative voltage supply fails, relay K-7 opens and the B+ to the 6L6 electron tube is removed, holding the reactor in a scram condition. The two other methods for scramming the reactor, manual scram and stop button which removes the AC power from the Safety Chassis and motor panel, function so as to remove the B+ voltage used by the 6L6 electron tube. The manual scram button, which is normally closed, is installed in the B+ line. When the button is pushed downward, this line is physically broken, cutting off the scram tube and scramming the reactor. When the AC power is removed from the scram chassis by pushing the stop button, the B+ power supply is turned off and the reactor is scrammed.

After the reactor has been scrammed by any one of the above conditions except the stop button, the rod carriages of all four safety and control rods are automatically driven out to their down position. If the AC power is turned off by the stop button, the power to the drive motors is also turned off and the carriages remain in their scram position, although the safety and control rods have ejected from the core by the Safety Chassis. When the AC power is turned back on, the carriages are automatically driven to their lower limits.

All the AC electrical power to the console is supplied through a main power switch which is located inside the reactor console.

4.3.4 Interlock System

Before and during the operation of the reactor, a safety interlock circuit must be satisfied to remove the reactor from a scram condition. The interlock circuit is a simple continuity circuit containing a number of switches and devices which assure that certain safety components of the system are operable. The block diagram shown in Figure 4.3-10

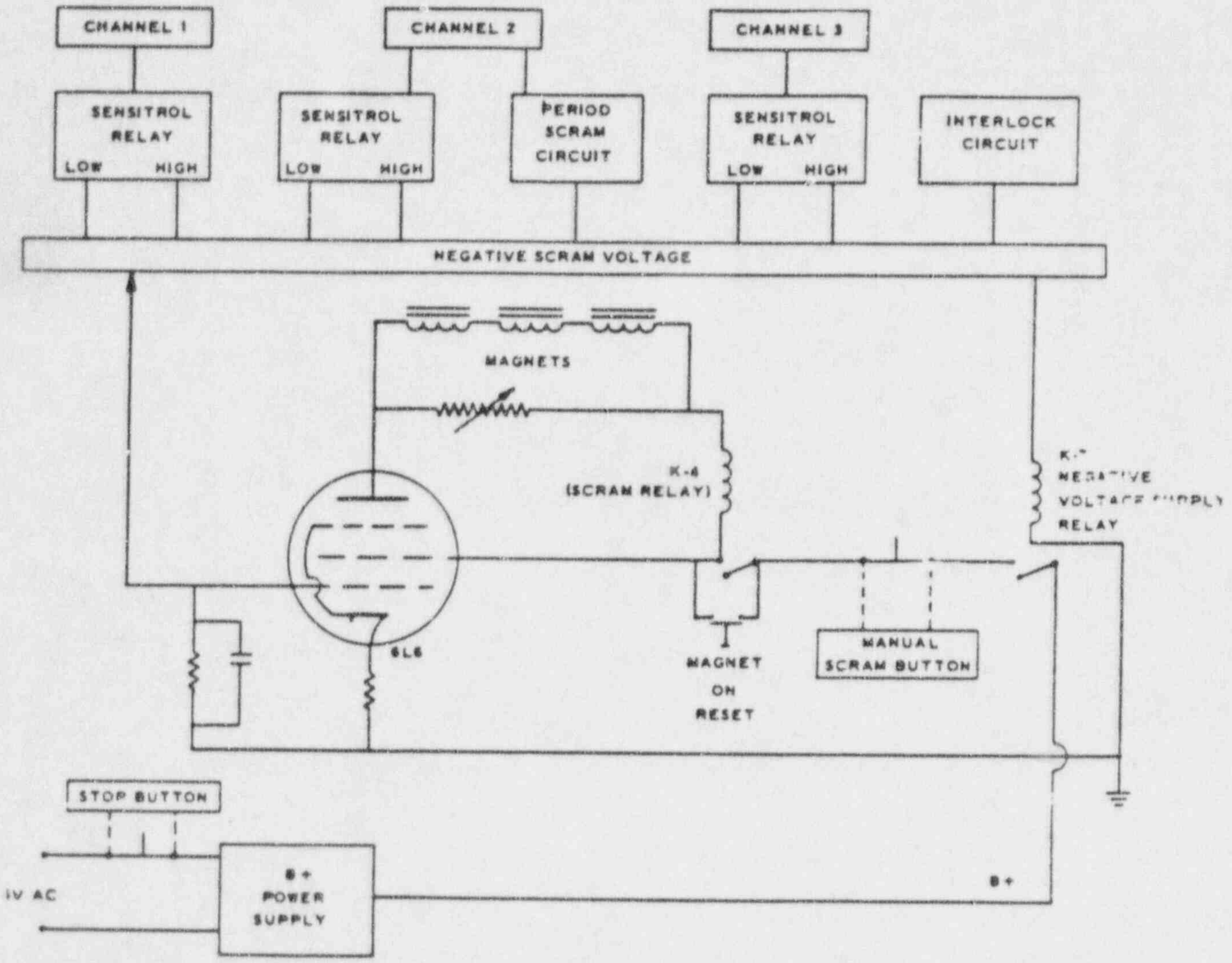


Figure 4.3-9 Simplified diagram of reactor scram system.

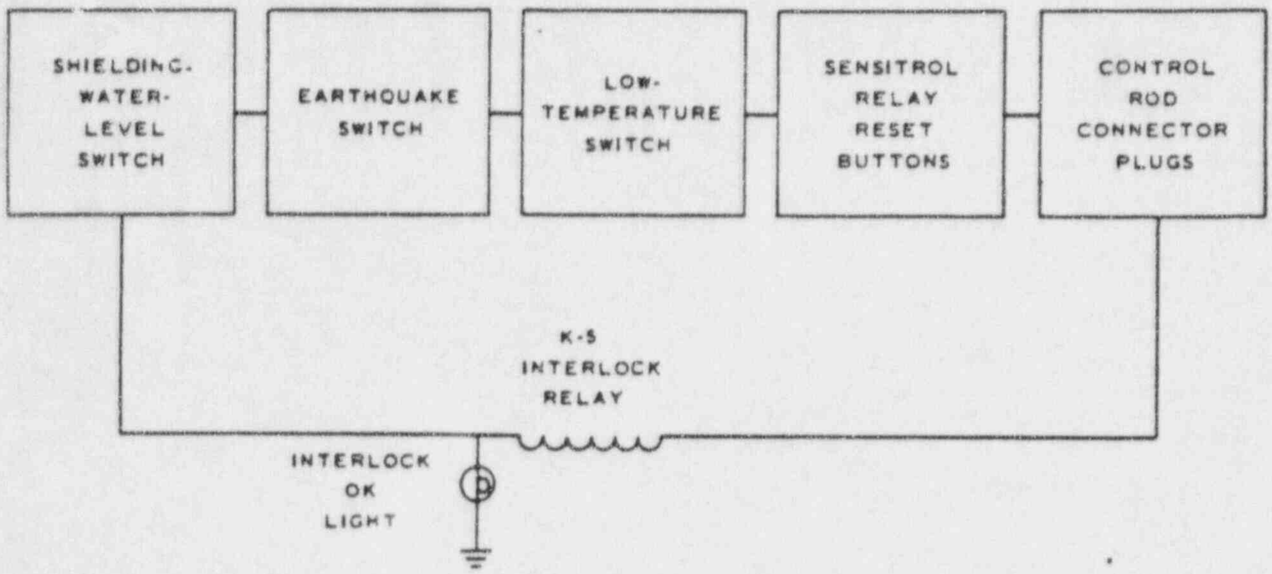


Figure 4.3-10 Block diagram of reactor interlock system.

describes the interlock circuit.

If the interlock circuit opens, relay K-5 opens and a negative voltage is placed on the grid of the 6L6 scram tube, cutting the tube off and hence scramming the reactor.

The shielding water level switch, which consists of a water-tight microswitch and an actuator connected to a float bob, opens the interlock if the shielding water level is less than the minimum allowed.

The earthquake switch consists of a steel ball mounted delicately on two terminal strips to maintain electrical continuity. If the reactor receives a physical shock resulting in a lateral displacement, the ball will move and break the electrical contact being made.

The low-temperature switch, which has been calibrated to open at 15°C, is a simple bimetal thermal switch. As the temperature of the switch reaches this low level, the bimetal strip which makes up one side of the switch bends, breaking the interlock circuit. The bending action takes place because the two metals used in the strip have different linear coefficients of expansion.

To insure that no instrument can be made inoperative during the operation of the reactor by holding its reset button down, all reset buttons are included in the interlock circuit. Pushing one of these buttons physically breaks the continuity interlock circuit and causes a scram condition to exist. In addition, all cable connectors to the control and safety rods have two pins, which have been wired into the interlock circuit. Any connector which is not connected properly will therefore open the interlock circuit. If any of these circuits are interrupted, the "Interlock Open" warning light on the Safety Chassis panel will be energized.

5.0 SAFETY ANALYSIS

Information in this section includes an analysis of the maxumum credible reactivity accident, as well as consideration of radioactive fission product gases, shielding and radiation fluids, etc.

5.1 General

The following section describes the design features of the ISU AGN-201 reactor and building which ensure that the reactor can be operated under the specified conditions with no hazard to the health and safety of the operators, other occupants of the Lillibridge Engineering Laboratory, or to the general public. Also considered are the effects of conceivable accidents due to component malfunction, human error, or force majeure.

The reactor is located in a medium-sized building on the ISU campus with an estimated daytime occupancy of between 100 to 200 people during the academic year. Thus an attempt has been made to eliminate or reduce as many of the normal potential nuclear hazards as possible. Emergency procedures for the AGN facility are given in the facility Emergency Plan.

The primary hazard is the possible over-exposure of personnel to radiation. Such over-exposure may occur in any or all of the following ways: chronic exposure to relatively low radiation levels; acute exposure to high radiation levels from sealed sources; acute exposure to elevated radiation levels as a result of an inadvertent power excursion: and exposure to, and possible inhalation and/or ingestion of, uncontained radioactive fission products. The purpose of this chapter is to define and evaluate these hazards and to discuss the various safety features of the AGN-201 reactor. The hazards set forth here have been documented and evaluated by personnel from this facility and other AGN facilities, extant and decommissioned. An NRC evaluation of the hazards associated with operation of the AGN-201 reactor is given in Docket F-32.

Under normal operating conditions, the tank and concrete shielding are sufficient to protect personnel from undue exposure to radiation. Precautions have been taken through the scram systems and the administrative limit on excess reactivity to insure that a nuclear excursion does not occur. However, if this improbable accident does happen, a thermal safety fuse is utilized to minimize consequences and prevent recurrence of the event. In addition, sufficient shielding is provided to protect nearby personnel from serious exposure. The core is designed to insure that no radioactive gas escapes during any accident. The hazards during and following reactor operations are discussed in detail. These are those that occur from the normal operating conditions at a power level of 5 watts and the accident conditions resulting from a 2-percent step increase in reactivity.

5.2 Reactivity Considerations

Reactivities of core components have been given in Table 4.2-1. The insertion of up to 2-percent reactivity ($2.70) in the case of an improper procedure, such as the insertion of a fueled experiment into the glory hole with the reactor at delayed critical is conceiveable, but highly improbable. Such a step insertion should not damage the core as a result of the ensuing excursion.

The estimated reactivity worth of each of the control rods is given in Table 4.2-1. The total worth of all rods is about 4.0% Δk/k ($5.41) which gives a shutdown margin of 3.35% Δk/k ($4.53) over the maximum allowable excess reactivity loading of 0.65% Δk/k ($0.74). With the maximum worth rod stuck in the core, the shutdown margin is about 2.2% Δk/k ($2.97).

5.3 Radiation and Shielding

5.3.1 Shielding

The standard radiological shielding tank of the AGN-201 is designed to allow continued access to the vicinity of the reactor at power levels up to 0.1 W. Additional shielding for operation at power levels up to 5 W has been provided by a concrete wall constructed of dense 10 cm x 20 cm x 40 cm concrete blocks and 10 cm x 20 cm x 30.5 cm barytes concrete blocks. The blocks are held to close dimensional tolerance and stacked in such a manner that voids in the completed wall are at a minimum. Around the beam ports and glory hole, high density blocks are used between 1.02 m and 2.85 m above the base of the reactor. The use of these blocks further reduces radiation levels near these areas. Overhead shielding is provided by two 30.5-cm-thick, sliding shield doors that may be opened to permit access to the thermal column when the reactor is shutdown or operating at low power.

As detailed in the amendment for 5-watt operation for Aerojet-General Nucleonics, dated 11 February 1957, and on file with the Commission in Docket 50-32, a 45.7-cm (18-in) additional concrete shield wall was sufficient to maintain subtolerance radiation levels (i.e., less than 10 mrem/hr) external to the shield when operating at 5 W. Subsequent analysis by Aerojet-General Nucleonics indicated that 40.6 cm of ordinary concrete shielding was sufficient. The essential features of the ISU AGN-201 shielding are shown in Figures 5.3-1 through 5.3-3

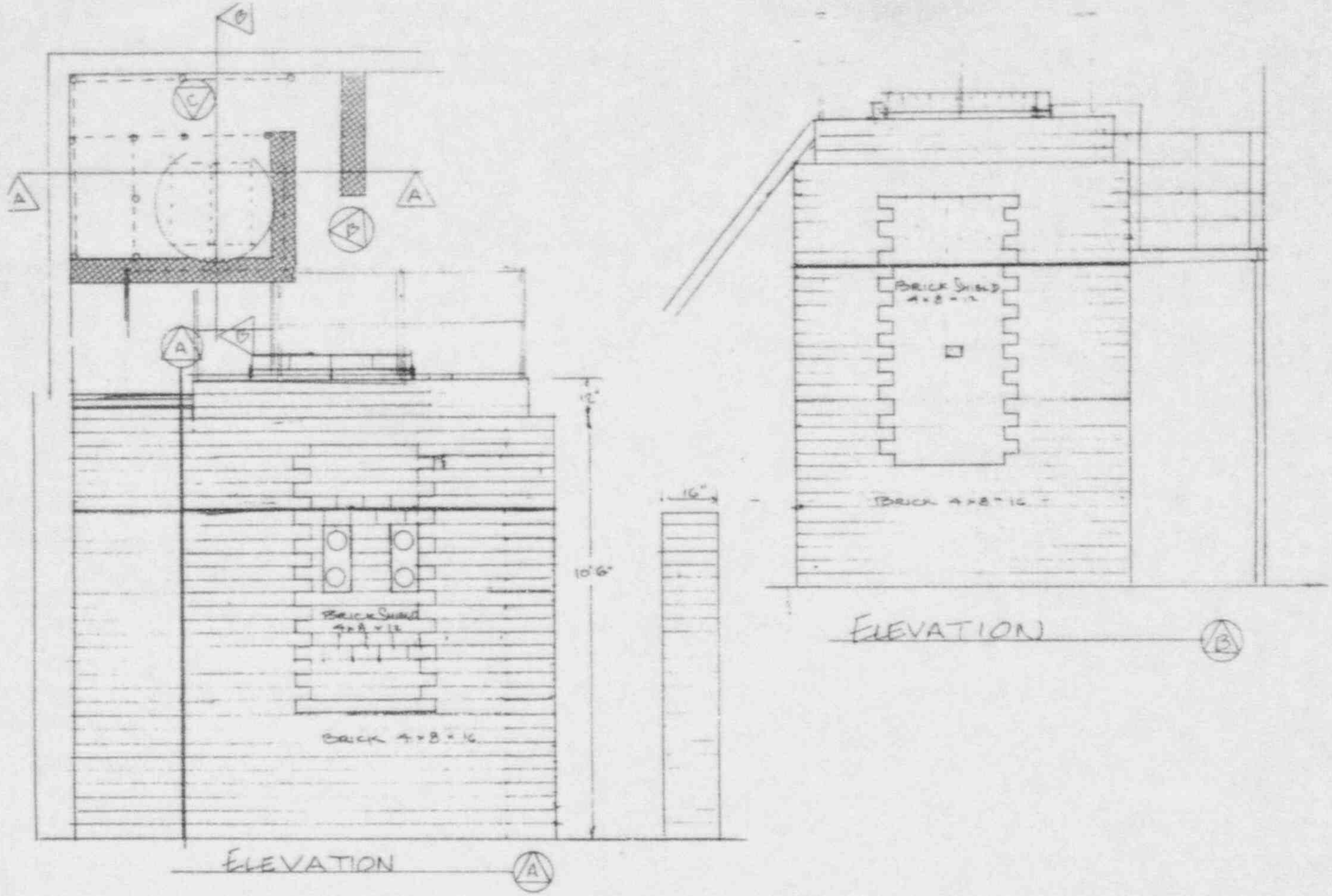


Figure 5.3-1 External concrete block shielding for AGN-201 reactor. Plan view of shielding. Elevation A is view facing north. Elevation B is view facing west.

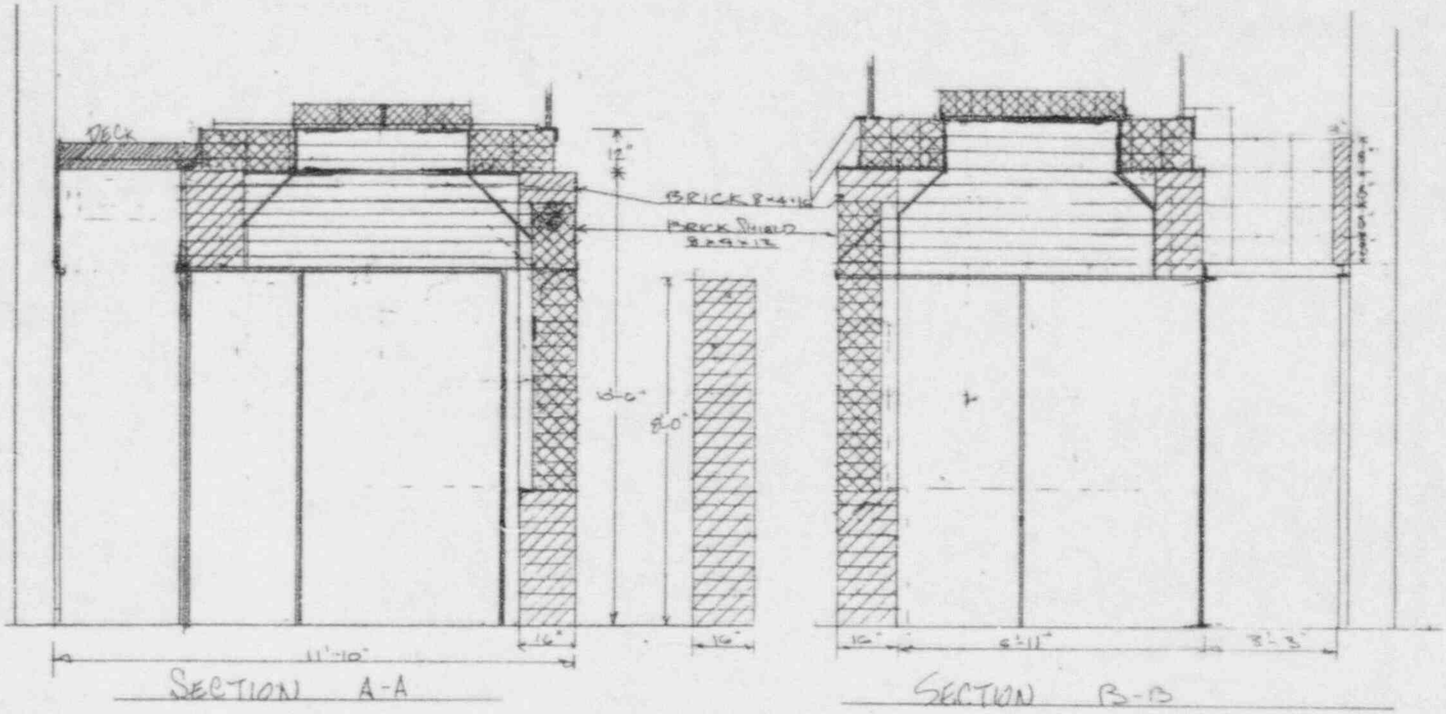


Figure 5.3-2 External concrete block shielding for AGN-201 reactor. Section A-A is a cut-away view facing north. Section B-B is a cut-away view facing west.

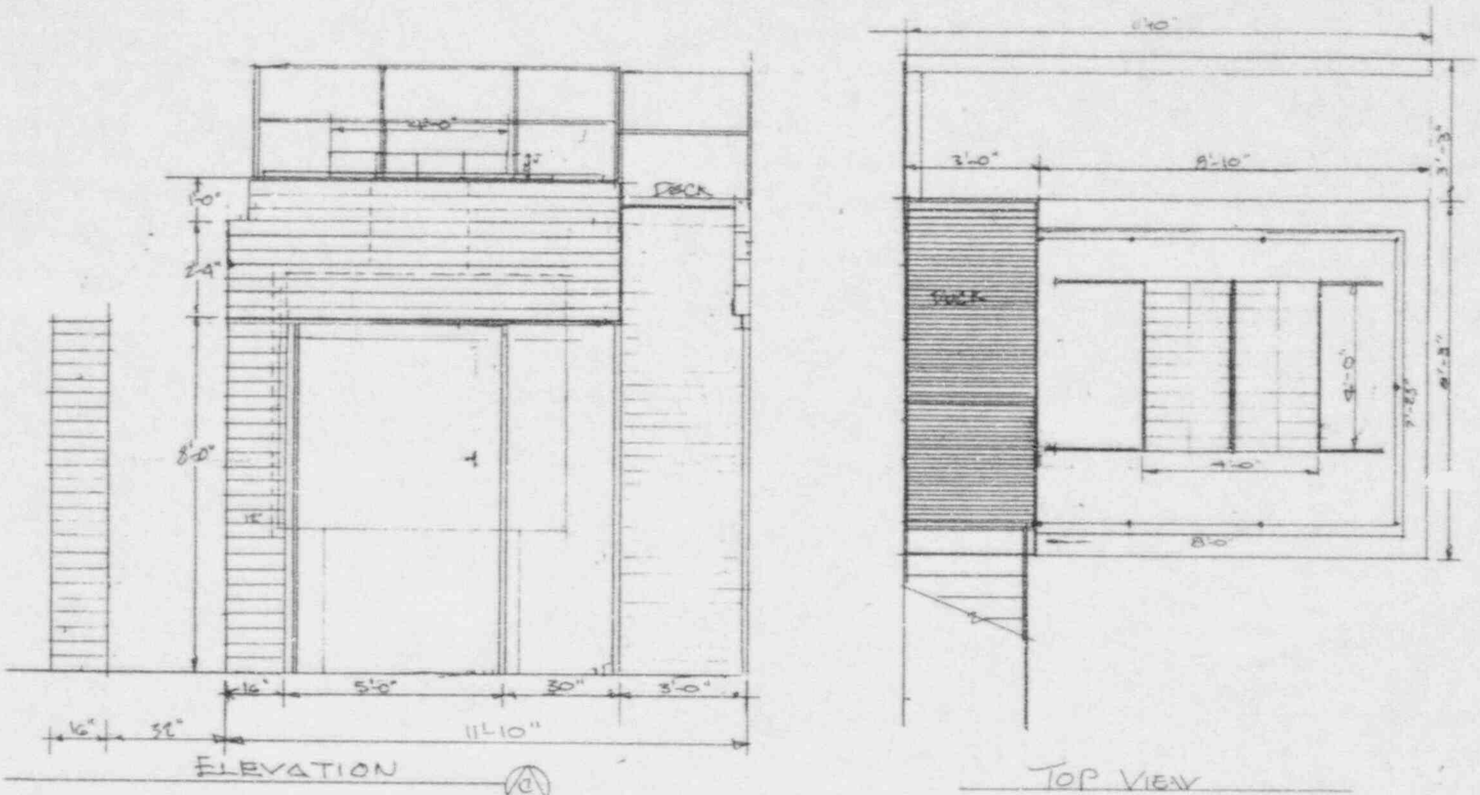


Figure 5.3-3 External concrete block shielding for AGN-201 reactor. Elevation C is facing south. Top view of shielding.

The radiation levels associated with 5-W operation of the ISU AGN-201 (peak thermal flux of 2.5 x 10^8 n/cm^2-s) have been extensively measured. Although operating surveys of the ISU AGN-201 show that at 5-W power there are no areas on the reactor floor outside the concrete shield where the total radiation exceeds tolerance levels (i.e., greater than 10 mrem/hr), nevertheless, access to the reactor floor is restricted.

At the maximum operating power, the shielding from 10 cm of lead and 55 cm of water and 41 cm of concrete is sufficient to reduce the equivalent dose rate to personnel next to the reactor concrete shield to less than 10 mrem/hr (0.1 mSv/hr). A nuclear excursion generating a total energy of 5.8 MJ, although impossible to achieve unintentionally without additional fuel, would result in a total radiation dose of 3.2 rem (32 mSv) to a person standing next to the reactor shield. This exposure is considerably below the minimum medically detectable dose which is generally quoted to be about 20 rem (200 mSv).

5.3.2 Operational Radiation Levels at Full-Power Operation

Complete gamma and neutron dose measurements have been made for the ISU AGN-201 reactor at power levels from essentially zero to the maximum licensed power level of 5 W. Measurements inside the concrete block shielding indicate that the average dose rate from neutrons and gammas at 30 cm from the surface of the AGN tank, while operating at a steady-state power level of 5 W, is about 230 mrem/hr (2.3 mSv/hr). However, access to this area during full-power operation is restricted. Outside the concrete shielding, doses averaged over the shield surface are about 8 mrem/hr (80 μSv/hr) total, while average total dose rates at the AGN console are 0.62 mrem/hr (6.2 μSv/hr) at 5 W. Figures 5.3-4 and 5.3-5 provide typical calculated radiation levels associated with 0.1-W and 5-W operation, respectively.

At maximum power, the equivalent dose rate measured at the entrance to the reactor laboratory is 0.7 mrem/hr (7 μSv/hr). Other areas adjacent to the laboratory have comparable dose rates. The dose rate on the roof of the reactor room immediately above the reactor is 0.15 mrem/hr (1.5 μSv/hr). The area above the reactor is not normally occupied. Outside of the reactor room, the maximum dose rate occurs in the Observation Classroom, LEL Room 19, at a position on the wall separating the Reactor Room and the Observation Room along the line-of-sight from the AGN glory hole. The dose rate at this location is about 7.5 mrem/hr (75 μSv/hr) from a well-collimated beam of radiation emerging from the gloryhole which covers a small area because of limited beam divergence. The Observation Classroom is a restricted access area and is generally

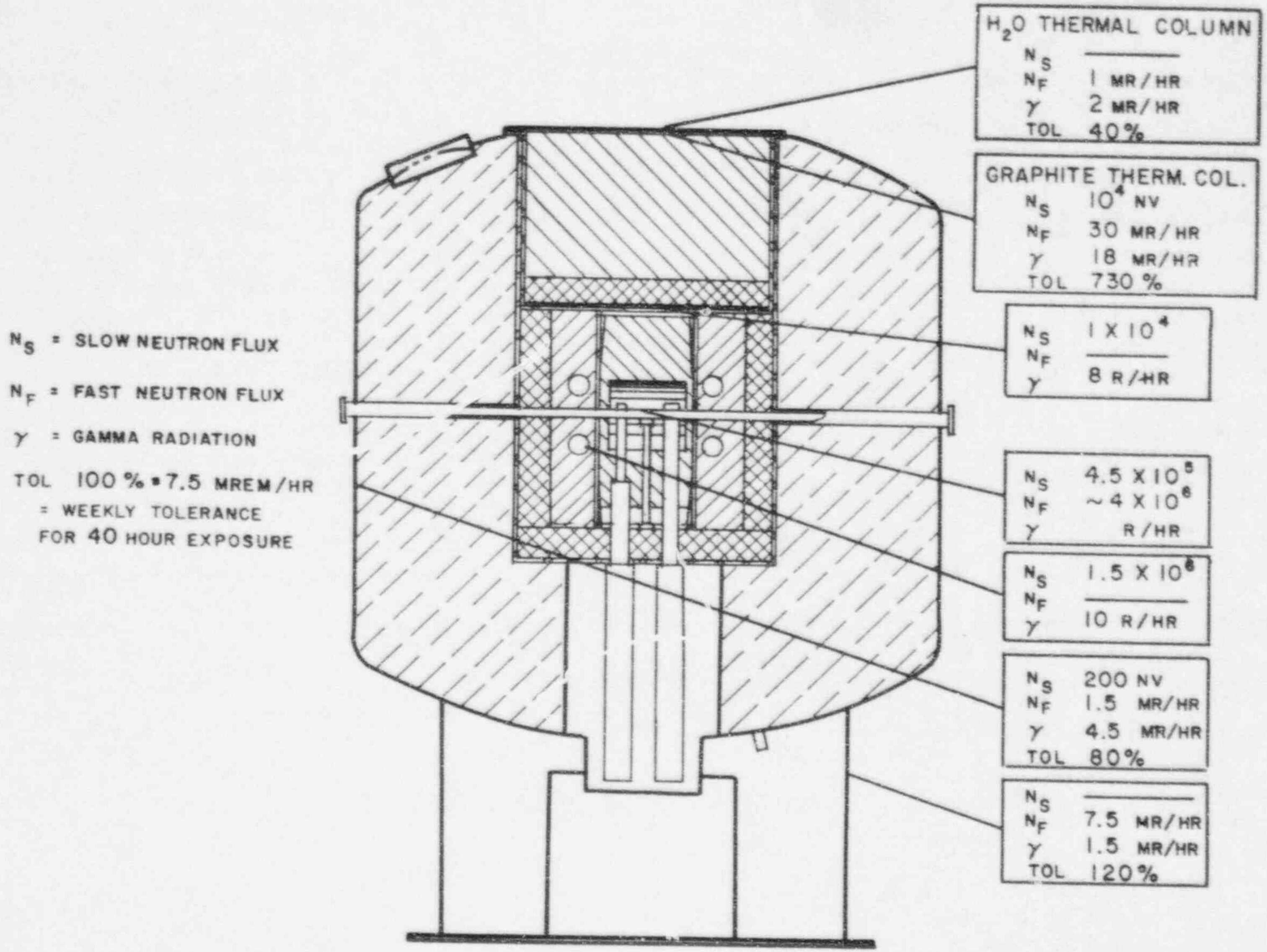


Figure 5.3-4 Radiation levels of the AGN-201 reactor operating at 100 mW.

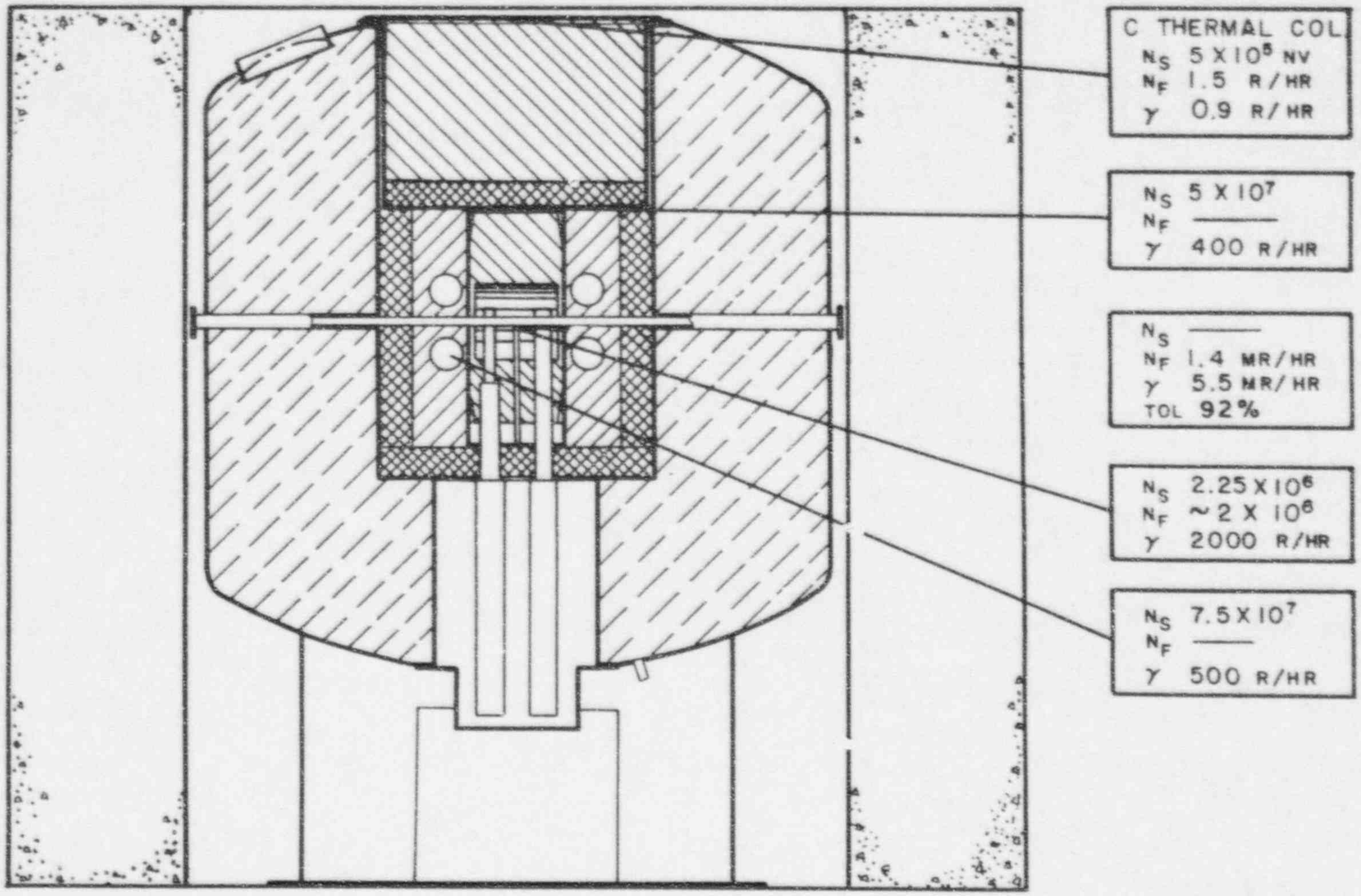


Figure 5.3-5 Radiation levels of the AGN-201 reactor operating at 5 W.

not ccupied by students during reactor operations.

The recommended limit[1] for uncontrolled university areas is 0.1 rem/year (1 mSv/yr) to any student or member of the general population. If a 5-hour week, 30-week reactor operation schedule is assumed, this results in a 0.1 rem/year (1.0 mSv/yr) dose to a receptor at the entrance to the reactor room. The assumption of a 5-hour exposure per week is conservative in that most reactor operations in which students may be exposed to elevated radiation levels are associated with the teaching of an undergraduate or graduate reactor laboratory course and are limited to one three-hour period per week. Therefore, the AGN-201 reactor is adequately shielded for 5-W operation under such an expanded operating schedule.

For persons working in the reactor room, under surveillance, the ISU occupational dose limit[2] is the more restrictive of (1) 1,000 mrem/yr (10 mSv/yr) total effective dose equivalent or (2) 10,000 mrem/yr (100 mSv/yr) for the sum of the deep-dose and committed dose equivalent to any individual organ or tissue, excluding the lens of the eye. A student would have to remain at the concrete shield in the reactor room for about 4 hours per week for 30 weeks each year to receive an equivalent dose of this magnitude from the reactor operating at 5 W.

5.3.3 Radiation damage to fuel matrix

Low-density polyethylene, a polymeric organic material, can sustain radiation damage when exposed to neutron bombardment and the radiation emissions from the decay of fission products. In tests performed by Aerojet-General Nucleonics, more than fifty small samples of core material were exposed in the CP-5 reactor at Argonne National Laboratory in a flux of approximately 10^{12} n/cm^2-s for periods of up to 1 week. An analysis of these samples indicated that radiation damage manifests itself in reduced density and loss of hydrogen from the polyethylene after exposures of approximately 1 week for a fluence of 6 x 10^{17} n/cm^2. By extrapolating these data, on the assumption that the time-integrated flux (fluence or nvt) is responsible for the radiation damage at an average power of 5 W, the core life is approximately 200 years for continuous operation. It is a reasonable assumption that a lower flux for a

---

[1] Radiation Safety Policy Manual, Rev. 2, Radiation Safety Division, Technical Safety Office, Idaho State University, September 1994.

[2] Radiation Safety Policy Manual, Rev. 2, Radiation Safety Division, Technical Safety Office, Idaho State University, September 1994.

correspondingly longer time would result in no more radiation damage than occurred in the high flux tests conducted in the CP-5. Therefore, the core should have an adequate lifetime if exposed to no more than an average continuous power level of 5 W.

5.3.4 Production and handling of radioisotopes

Neutron activation in the AGN-201 reactor can produce only very limited quantities of radioisotopes and any induced radioactivity in reactor structures is negligible. Subsequent handling of radionuclides is supervised by individuals experienced in the detection and evaluation of radiological hazards. The reactor staff has been trained in such procedures and supervises all handling of radioactive materials within the reactor area. Outside this area, the use of radionuclides comes under the control of the ISU Radiation Safety Committee.

The maximum amount of activity which can ve produced by one irradiation is given by the product of the specific activity and the mass of material irradiated. There are three primary limitations on the amount of mass which can be used. The next obvious of these is the amount of space available in the reactor for irradiation. In the AGN-201 reactor, the useful volume of the glory hole is approximately 50 cm^3. This is a sufficient volume fo rmost purposes. For those cases in which this volume is insufficient, the 10-cm-diameter access ports may be used instead of the glory hole.

The second limitation in the total mass of material that may be irradiated is the effect of this material on the criticality of the reactor. Since the material must absorb neutrons to become activated, its insertion into or near the core decreases the reactivity of the system. This effect can be compensated by the use of the control rods. Safety considerstions, however, prohibit operating the reactor at more than 0.24% excess reactivity so that the total amount of material which may be inserted is that which will decrease core reactivity by 0.24% $\Delta k/k$.

The trial limitation on the amount of material arises from loss of the neutron absorption efficiency because of self-absorption whereby the outer portion of the material shields the inner portion fromthe neutrons. This effect is present primarily in strongly absorbing materials, such as indium and gold.

Finally, there is the constraint in the number of excess neutrons available if one were to attenpt to capture all of the neutrons leaking from the reactor core. At 5-W power, this constraint is 10^{11} neutrons per second which corresponds to a total saturation activity of 2.5 Ci. Thus, the AGN-201 reactor operating at a power level of 5W is capable of producing 15mCi sources in the glory hole and up to 150 mCi sources in the access ports. However, it is impossible by any method of capturing the excess neutrons to produce more than 25 Ci cources.[3] This quantity of activity can be handles by facility personnel under the administrative controls which are presently implemented.

The risk of release of radioactivity by breakage of an irradiated sample is reduced by careful design of handling and encapsulating practices and attention to details of irradiation, such as the effect of radiation on the sample. Such criteria are examined closely with respect to experiments. The use of non-porous paintwork in the Reactor Laboratory is an aid in preventing long-term contamination. Emergency decontamination supplies are on hand at all times, as well as contamination survey instruments.

5.4 Production and release of radioactive gases

Radioactive argon-41 and nitrogen-16 are produced by neutron reactions with air and water in the vicinity of the core of the reactor. Air may be contained in experimental facilities (glory hole & access ports) and is in solution in the tank water.

5.4.1 Production of argon-41

Experience with AGN reactors operating at higher power has shown that no significant release of Ar-41 occurs from the glory hole during power operation at 5 W or less. The Naval Post Graduate School found some Ar-41 activity by irradiating a sample of air at atmospheric pressure in a closed tubular container just filling the AGN glory hole to the boundaries of the core. The irradiated air was transferred to a chamber counter with thin-walled glass G-M tube. Decay was followed over approximately one half-life and was consistent with the decay of Ar-41. The measured activities agreed with those estimated from a calculated efficiency of the counter. On the basis of Naval Post Graduate School operating experience, Ar-41 will not be formed in significant concentrations under the skirt at operation at 5 W. Since the resulting peak Ar-41 activity for the air volume in a sealed empty glory hole is only 45 times greater than the MPC value for Ar-41, natural diffusion and mixing of this irradiated air

---

[3] A.T. Biehl et. al., "The AGN-201, a safe low power, portable, nuclear reactor, Nucleonics, Sept.1957.

volume will easily reduce the average air activity in the vicinity of the reactor to less than 1% of MPC values for uncontrolled areas. Also, the reactor area is presently and will continue to be a control area with limited access. Thus, no hazard from Ar-41 is anticipated, as shown below.

The maximum equilibrium concentration of Ar-41 produced can be easily calculated. At 5 W, the average thermal neutron flux is 1.75×10^8 n/cm^2-s. The saturation reaction rate, $\Re$, for Ar-41 production is given by

where

$$\begin{aligned} \Re &= \Sigma_\gamma \phi \\ &= \sigma_\gamma \frac{\zeta m N_A}{A} \phi \end{aligned} \tag{1}$$

σ_γ = microscopic cross section for $^{40}Ar(n,\gamma)^{41}Ar$ [cm^2],
ζ = natural abundance of Ar-40 [dimensionless],
m = the mass of Ar-40 contained within the volume of the glory hole fully contained within the reactor core [grams],
N_A = Avogadro's number [mol^{-1}],
A = atomic mass of Ar-40 [grams mol^{-1}], and
ϕ = average thermal neutron flux [n cm^{-2} s^{-1}].

The mass of Ar-40 is calculated based on the assumption of that the air entrapped within the glory hole is a dry, ideal gas at standard temperature and pressure with argon comprising 1.3% of air by mass. Thus, there are 6.2 mg of Ar-40 contained within the 394 cm^2 of air entrapped within the portion of the glory hole fully contained within the reactor core. The resulting Ar-41 production rate is 10,600 atoms/s, or in terms of activity, the production rate is 30.2 pCi/s.

The equilibrium concentration of Ar-41 in the reactor room at a power level of 5 W is given by

$$C_{41} = \frac{L}{V_R r} \tag{2}$$

where

C_{41}= concentration of Ar-41 [μCi/cm^3],
L = leakage rate of Ar-41 from the core (assumed to be equal to the production rate) [μCi/s],
V_R = the volume of the reactor room [cm^3], and

r = fractional volumetric exchange rate of air in the reactor room equal to the ventilation flow rate, Q [cm^3/s], divided by the reactor room volume.

The dimensions of the reactor room are 8.5 m x 8.5 m x 7.0 (28 ft x 28 ft x 23 ft) which give a volume of 5.11×10^8 cm^3 (8,032 ft^3). During normal operation, the ventilation flow rate is 7.83×10^5 cm^3/s (1,660 cfm). Substitution of these numerical values into Equation (2) yields an equilibrium concentration for Ar-41 of 3.85×10^{-11} $\mu Ci/cm^3$.

From 10 CFR 20 Appendix B, the limiting values for the Derived Air Concentration (DAC) and effluent concentration for Ar-41 are 3×10^{-6} µCi/ml and 1×10^{-8} µCi/ml, respectively. DAC values establish the limiting concentrations of airborne radioactive materials to which occupational workers may be exposed while the effluent concentration provides limiting concentrations for the general public. Thus, the maximum equilibrium concentration of Ar-41 in the reactor room air with the AGN reactor operating at 5 W is 0.0013% of the DAC limit and 0.39% of the effluent concentration limit.

Air flow from the reactor laboratory room is mixed with return air from the rest of the LEL building. Approximately 80 percent of the total air flow is recycled to the building while the remaining 20 percent is discharged from a roof vent located a minimum of 9 m above ground level. The air flow leaving the reactor room is diluted by a factor of about 27 before being discharged to the environs. Return air is diluted further by mixing with fresh makeup air before being recirculated to the building. The calculated maximum effluent concentration of Ar-41 at the penthouse discharge vent is 1.5×10^{-12} $\mu Ci/cm^3$. The Ar-41 concentration in the recycle air to unrestricted areas in the LEL building is approximately 1.0×10^{-12} $\mu Ci/cm^3$.

These concentrations are insignificant compared with the maximum permissible concentration in unrestricted areas which is given by the maximum effluent concentration. The maximum concentration of Ar-41 in unrestricted areas of the LEL building from the ventilation system recycle is 0.010 percent of maximum permissible level. The concentration of Ar-41 in air discharged to the environment from the exhaust vent is about 0.015 percent of the maximum permissible level. This value is much greater than the resulting concentration at ground level since the analysis of the ground-level concentration does not take into account the additional dilution that occurs upon discharge from the vent.

5.4.2 Release of argon-41 from tank water

Argon-40 atoms are present in solution in the tank water and produce Ar-41 on neutron irradiation. However, the Ar-41 produced by the activation of dissolved air in the tank water is negligible for the following reasons: (1) the concentration of argon in the tank water is small, (2) the thermal neutron flux in the shield water is several orders of magnitude lower than in the reactor core, and further, the tank access cover retards diffusion of the gas from the shield tank to the atmosphere. An estimate of the number of Ar-41 atoms released into the room air and the concentration in the discharge can be made and results show that this value is small (about 2×10^{-13} μCi/cm^3 air at 5 W) compared to the Ar-41 production from air entrapped in an open glory hole.

5.4.3 Production and release of nitrogen-16

Simple calculations and experimental measurements show that production and release of nitrogen-16 is insignificant for the AGN-201 reactor operating at power levels up to 5 W.

5.5 Maximum Credible Reactivity Accident

The total excess reactivity of the AGN-201 reactor is approximately 0.25% $\Delta k/k$ (\$0.33). However, for the purposes of the AGN-201 Safety Analysis it has been assumed conservatively that the reactivity can be instantaneously increased to 2% $\Delta k/k$ above delayed critical with the reactor in operation at a power level ≤ 5 W. This situation clearly cannot arise during the normal course of operation of the system, but it could possibly occur if improper extraneous materials are inserted into the reactor, for example, through the glory hole. This instantaneous 2% positive reactivity insertion is taken to be the maximum credible accident for the AGN-201 reactor.

Historically, this scenario has been analyzed using one group theory with one group of delayed neutrons. The analysis was first performed by Aerojet-General Nucleonics and published in the Hazards Summary Report for the AGN-201 Reactor in August 1957. This simplified analysis assumed that: (1) at time zero a 2% step increase in reactivity was inserted with the reactor at 100 mW and (2) the energy in the core at time zero was negligible in comparison with the energy liberated during the ensuing excursion. Further, the analysis assumed that there was no energy transfer from the reactor core during the excursion.

A complete description of the analysis may be found in the Hazards Summary Report and will not be repeated here, but the results of their analysis are summarized below. Numerical solution of the governing equations by the finite-difference method yielded a total energy release of 1.7 MJ and raised the average core temperature about 71° C. The excursion produced a peak power of 54.4 MW at 204 ms after the reactivity insertion. These results were adequate to demonstrate that the core material would not sustain catastrophic damage nor release any fission products because the average core temperature remained well below the melting point of the polyethylene matrix, which is about 200° C.

For the purposes of this document, a more detailed analysis is performed using one neutron energy group with six delayed neutron precursor groups. The same assumptions are made regarding the 2% step increase in reactivity at time zero, negligible energy accumulation at the start of the excursion, and the excursion is modelled as an adiabatic process. With these assumptions, the applicable kinetics equations are then

$$\frac{dn}{dt} = \frac{\rho - \alpha T - \beta}{\ell} n + \sum_{i=1}^{6} \lambda_i C_i \tag{3}$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{\ell} n - \lambda_i C_i \qquad i=1,2,\ldots,6 \tag{4}$$

$$P = \Sigma_f \, n \, v \, \gamma \, V \tag{5}$$

and

$$\frac{dT}{dt} = \frac{P}{\delta V C_p} \tag{6}$$

where

n = space-averaged neutron density in reactor [n/cm^3],
ρ = core reactivity [dimensionless],
α = temperature coefficient of reactivity [°C^{-1}],
ℓ = neutron generation time [s],
β = fraction of delayed neutrons [dimensionless],
β_i = delayed neutron fraction for the ith group of delayed neutron precursors [dimensionless],
λ_i = decay constant for the ith group of delayed neutron precursors [s^{-1}],
C_i = space-averaged density of the ith group of delayed

neutron precursors [cm^{-3}],
P = reactor power [W],
Σ_f = macroscopic fission cross section [cm^{-1}],
v = average thermal neutron speed [cm/s],
γ = recoverable energy per fission [J/fission],
V = volume of reactor [cm^3],
δ = density of the core material [gm/cm^3],
C_p = specific heat at constant pressure [J/gm-°C], and
T = core temperature [°C].

Equations (5-3) and (5-4) represent the usual point-reactor kinetics equations; Eq. (5-5) gives the thermal power output of the reactor; and Eq. (5-6) gives the temperature change in the reactor core. In solving these equations the following numerical values were used:

ρ = 0.02,
α = -1.66×10^{-4} °C^{-1},
ℓ = 7.5×10^{-5} s,
β = 0.0074,
Σ_f = 0.074 cm^{-1},
v = 2.2×10^5 cm/s,
γ = 3.4×10^{-11} J/fission,
V = 1.2×10^4 cm^3, and
$V\delta C_p$ = 27,050 J/°C.

The value of α given above (-1.66×10^{-4} °C^{-1}) was calculated by Parker[4] and used in an early safety analysis of the ISU AGN-201 reactor. This value is somewhat smaller than the value reported by the manufacturer (-2.5×10^{-4}°C^{-1}) and therefore yields a more conservative solution to equations (5-3)-(5-6), since the calculated energy output is greater than what would actually result from the use of the larger magnitude for the negative temperature coefficient of reactivity. Parker's α was obtained from a DISNEL[5] calculation using five core temperature regions in the AGN reactor and accounts for the nonuniform temperature distribution in the core as a result of the flux distribution which is peaked at the center and near the reflector.

---

[4] Robert Eugene Parker, "Safety Analysis of the AGN-201 Reactor," Idaho State University, Pocatello, Idaho, Master of Science Thesis, 1978.

[5] Acronym for the Diffusion (equation) Iterative Solution, Nineteen Energy Levels computer code.

Equations (3)-(6) were solved numerically using the TUTSIM™[6] computer simulation package for three initial power levels at an initial uniform core temperature of 20° C: P_0 = 5 W, 0.1 W, and 0.0001 W. Higher initial power levels do not affect the consequences of the accident. The results of these computations are shown in Figures 5.5-1 through 5.5-3. Figure 5.5-1 gives the power of the reactor as a function of time after the instantaneous insertion of the 2% reactivity. As shown, the power level quickly rises to very high values, passes through a maximum of about 170 MW and then rapidly decreases to a power level of about 800 kW which then appears to slowly decay, all within a time window of about 300 milliseconds.

The height of the maxima is observed to be essentially independent of the initial power level, but as might be expected, the time at which the maxima occur increases with decreasing initial power. Following the excursion, the power decreases slowly until after several minutes the thermal power output attains a steady level of about 200 watts in which the reactivity addition is balanced by the compensating reactivity at the consequent elevated core temperature as a result of the large negative temperature coefficient. However, it is reasonable to assume that the thermal fuse has functioned as designed, thereby separating the core so that this equilibrium power is not maintained. The excursion simulation shown in Figures 5.5-1, 5.5-2, and 5.5-3 does not model core separation which would occur when the temperature of the thermal fuse located near the center of the core exceeds about 100° C, as discussed in Section 4.2.

The final power level is obtained by modifying Eq. (4) to include Newton's law of cooling and then equating the time derivatives with zero in Eqs. (1), (2) and (4). Equation (4) is then solved for the final power, P_f, which becomes

$$\frac{dT}{dt} = \frac{1}{V\delta C_p}(P - UA[T_f - T_0]) \qquad \textbf{(7)}$$

where

U = overall heat-transfer coefficient [W cm^{-2}], and
A = external core surface area [cm^{-2}].

The overall heat-transfer coefficient U is assumed not to be a function of temperature. The numerical value of the product UA was obtained from steady-state operation at 0.1 W. According to the manufacturer, the average temperature rise of the core at this power is 0.05° C, so that

---

[6] TUTSIM Products, 200 California Avenue, Palo Alto, CA 94306.

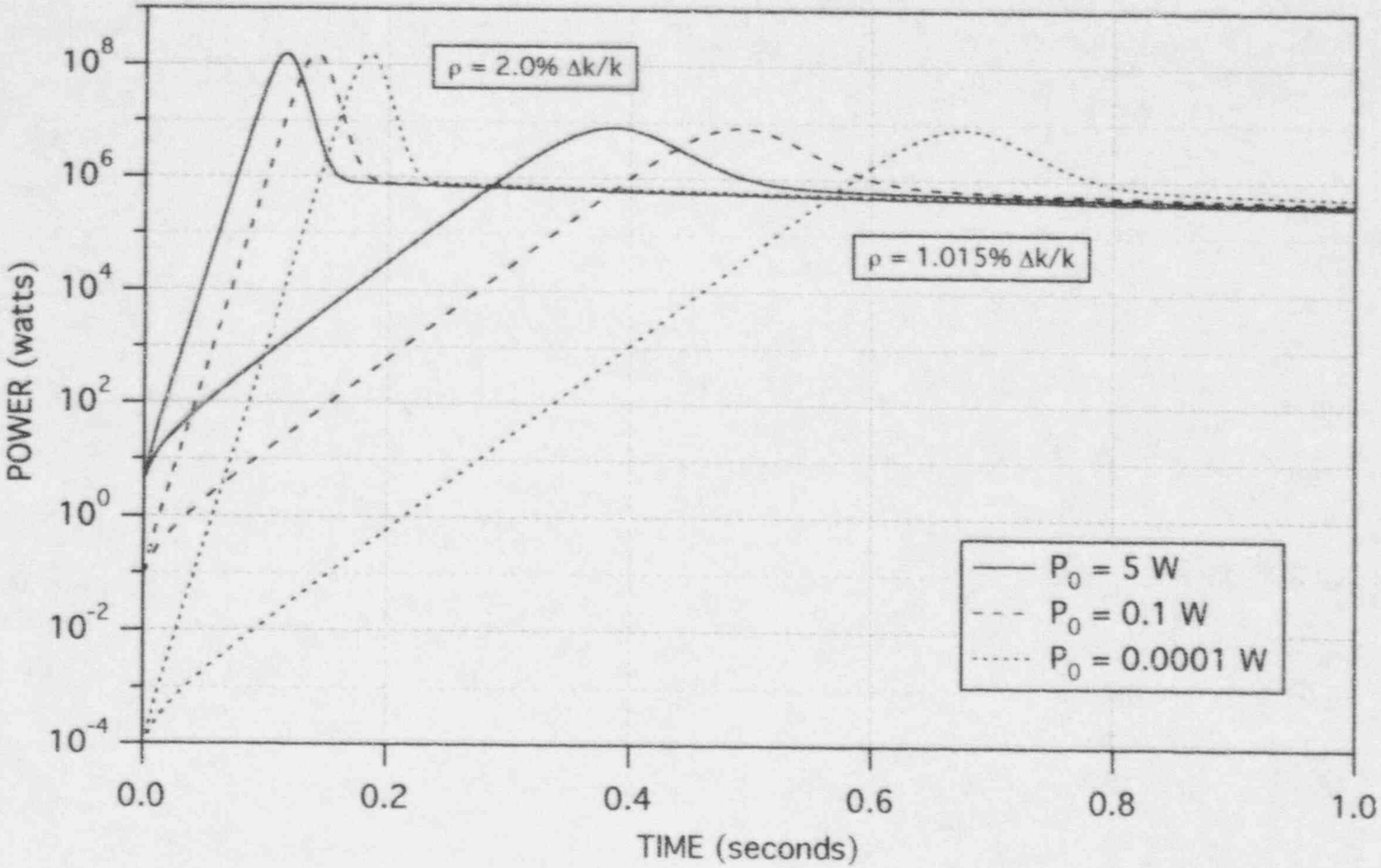


Figure 5.5-1 Reactor power during maximum credible accident.

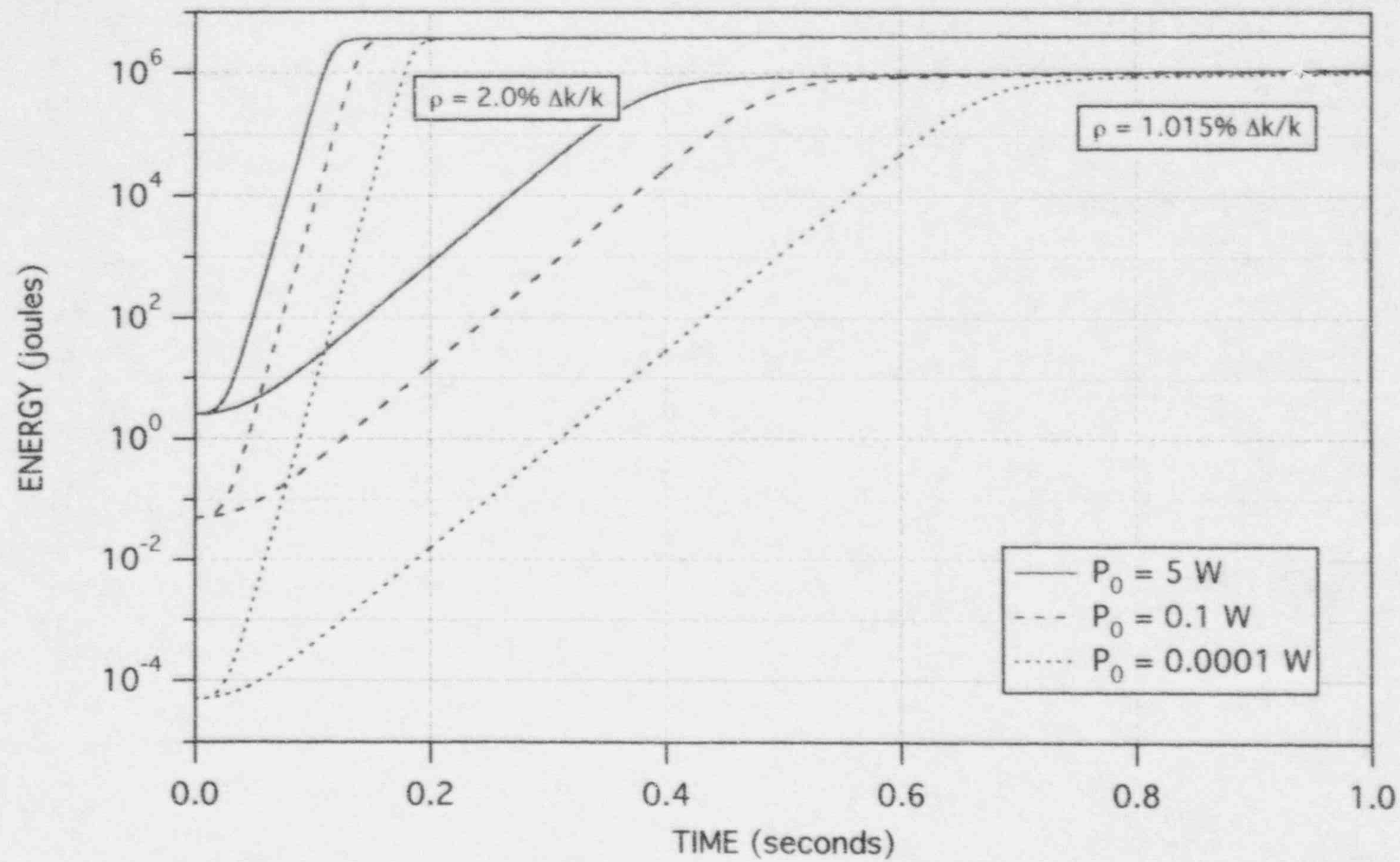


Figure 5.5-2 Energy release during maximum credible accident.

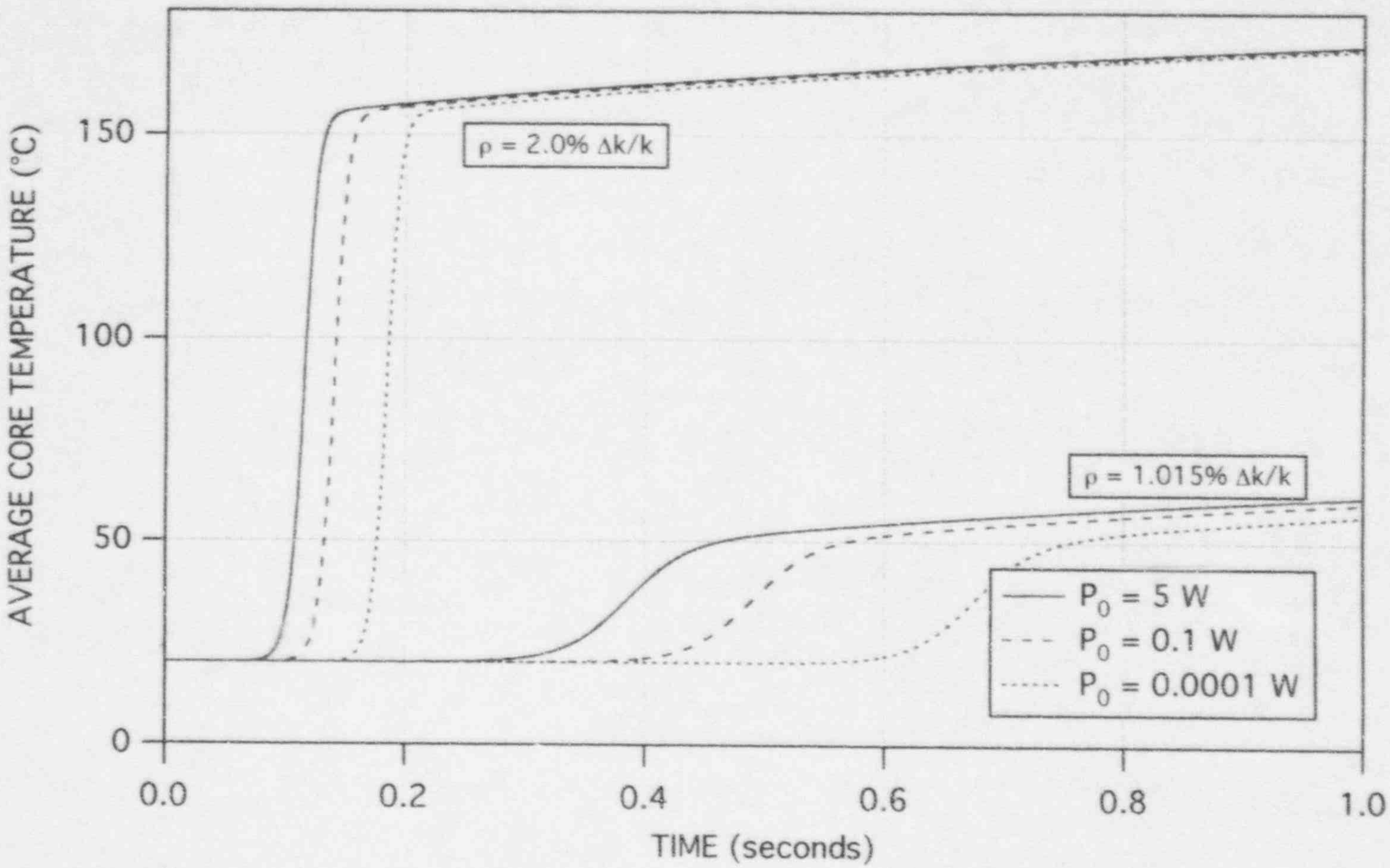


Figure 5.5-3 Core temperature rise during maximum credible accident.

$$UA = \frac{P}{\Delta T} = 2\ W/^{\circ}C \tag{8}$$

The value of UA has little influence on the total energy released during the accident, but it does determine the final power level which the reactor ultimately attains. Setting the time derivatives to zero gives

$$P_f = UA\left(\frac{\rho}{\alpha} - T_0\right), \tag{9}$$

where the substitution $T_f = \rho/\alpha$ has been made.

The energy released in the excursion is shown as a function of time in Figure 5.5-2. These curves are simply the time-integrated curves shown in Figure 5.5-1. The average core temperature is shown as a function of time in Figure 5.5-3. Assuming that the initial excursion is an adiabatic process, the maximum temperature is greater than 170° C for all three initial conditions. The maximum temperature, however, will be less than 150° C because of core separation once the thermal safety fuse deploys as designed. Core separation will result in an approximately 5% decrease in reactivity, thereby shutting down the reactor and preventing recurrence of the excursion. The final temperature in all cases is about 120° C.

The radiation dose received by personnel as the result of the maximum credible accident can be computed from the known radiation doses for 5 watt operation and the total energy released during the accident. Assuming that a person is in contact with the surface of the concrete shield (40.6cm) when the incident occurs, s/he would ordinarily receive a dose which is less than 10 mrem/hr (0.1 mSv/hr) at 5 watt operation. This is equivalent to a dose of about ((10 mrem/hr)/(5 J/s))/(3600 s/hr) = 5.56 x 10^{-4} mrem/J (5.56 x 10^{-3} μSv/J). From Fig. 5-7 the total energy released in the accident is about 5.8 MJ. It follows, therefore, that such an individual would receive a dose of (5.8 x 10^{6} J)(5.56 x 10^{-4} mrem/J) = 3.2 rem (32 mSv) as the result of the accident.

As unlikely as a 2% increase in reactivity is, such a reactivity change with either of the first two initial conditions is extremely remote. This is because, as already noted, the aforementioned reactivity increase can only occur by the insertion of extraneous materials in the glory hole or some other port, and this would be very difficult to accomplish while the reactor is operating. A step reactivity insertion of 1.015% Δk/k ($1.37) is more reasonable. This value of reactivity was determined by Parker[7] as the maximum reactivity that could be inserted into the AGN core through the glory hole based on the volume of the glory hole and the availability of fuel material. The response of the AGN-201 operating at the same initial power levels for this reactivity step has been analyzed and is also shown in Figures 5.5-1 through 5.5-3.

The maximum power, which is attained at about 400 ms for an initial power level of 5 W, is about 80 MW for this reactivity step. The total energy generated is about 2.9 MJ, and the maximum average temperature is about 128° C, again assuming no energy losses during the excursion. From the results, it may be assumed that the thermal fuse will function as designed and the core will separate. The resulting dose to an individual standing next to the concrete shielding is about one-half of the dose for the 2% reactivity insertion, or about 1.6 rem (16 mSv).

5.6 Loss of Water Shield from AGN Tank

If the reactor is operated without water in the shielding tank at 0.1 watt power, the radiation level just outside the reactor tank will be about 10 mrem/hr (0.1 mSv/hr) of gamma rays, and about 50 mrem/hr (0.5 mSv/hr) of fast neutrons. These levels are six and eleven times as much, respectively, as they would be through the water shield. At the outside surface of the concrete shield, radiation levels would be about 0.14 mrem/hr 1.4 μSv/hr) of gamma rays and 3.4 mrem/hr (34 μSv/hr) of fast neutrons at 0.1 watt without the water shield. At 5 watts the radiation levels would be about 7 mrem/hr (0.07 mSv/hr) for gamma rays and 170 mrem/hr (1.7 mSv/hr) for fast neutrons. This radiation level would trip the high-level radiation alarm mounted on the reactor console and initiate laboratory evacuation.

While it is extremely unlikely for an excursion to occur without the water shield in place, the maximum acute dosage a person might receive at the surface of the concrete shield would be 18 rem (0.18 Sv) of gamma rays and fast neutrons. This would of course be a high amount of radiation but is within the guidelines for emergency doses.

Another potentially hazardous condition which can be envisioned is the case where the control and safety rods are fully inserted and the scram mechanisms are made to be inoperative. Under such circumstances the reactor power would continue to rise until the negative temperature coefficient reduced the reactor to a delayed-critical state at some high

[7] Robert Eugene Parker, op cit.

power. Under this circumstance, equilibrium is determined by the condition that the rate of energy conducted away from the core be exactly equal to the fission energy generation rate. Since the temperature coefficient of reactivity is approximately $-1.66 \times 10^{-4}\ ^{\circ}C^{-1}$, and the heat conductivity rate from the core may be estimated, it may be readily calculated that with 0.2% excess reactivity the equilibrium temperature is approximately 10°C above ambient. This corresponds to a fission rate of approximately 10 watts. Postulating these conditions of the reactor operating at a continuous power level of 10 watts, the radiation received by a person outside the concrete shield would be approximately 12 mrem/hr (0.12 mSv/hr).

The above postulated exposure, although constituting a slight hazard, is considered improbable since it is doubtful that anyone would stay in such a position under reasonable administrative control for more than a few hours. It is interesting to try to predict whether or not operation at this high power level would cause the fuse to melt, and, accordingly, shut down the reactor. Unfortunately this is a very difficult heat transfer calculation, due to the complicated geometries and, although it is believed there is a reasonable expectation that the fuse would function, no claim is made to this effect.

5.7 Energy Released

At 5 watts, the total fission rate is about 1.6×10^{11} fissions/sec. Each fission produces approximately 0.6 neutrons that may leak out, 5 MeV of prompt gamma rays, 6 MeV of delayed gamma rays, and a small number of delayed neutrons. By far the largest source of radiation is due to the radioactive fission products. If the reactor has been operated at this level for a long time, the activity in MeV-curies equivalent at a time *t* seconds after shutdown is given by $0.4t^{-0.2}$,[8]: the activity which produces 3.7×10^{10} MeV of ionizing radiation is defined as one MeV-curie equivalent. Table 5.7-1 shows these values for various times after shutdown. In the event that the reactor is operated on an eight-hour-per-day schedule, the figures for one day and one month may be reduced by a factor of one-fourth. Five-watt operation leads to 50 times the radiation fields generated at a power level of 100 mW, as shown in Table 5.7-1.

5.7.1 Operational containment of fission products

The one significant difference resulting from 5 watt operation is the

---

[8] A.T. Biehl, R.P. Geckler, S. Kahn, and R. Mainhardt, Elementary Reactor Experimentation, Aerojet-General Nucleonics, San Ramon, CA, October 1957.

Table 5.7-1
Activity Contained in the Reactor Core for Various Times After Shutdown

t	Activity in MeV-curies Equivalent 5-W Operation	100-mW Operation
0 (reactor operating)	50.	1.0
1 second	20.	0.40
1 hour	3.75	0.075
1 day	2.0	0.040
1 month	1.0	0.020

increased fission product inventory. Concerning the levels of activity in the core following 5 watt operation, reference is made to Biehl, Geckler, Kahn, and Mainhardt, Elementary Reactor Experimentation, Aerojet-General Nucleonics, San Ramon, Calif., October 1957, pp. 19-21. Further, it is noted that operation of Aerojet-General Nucleonics AGN-201 at 5 watt levels resulted in no detectable release of radioactive effluents due to the retention of fission products by the fuel matrix material. Even if there were gaseous effluents released they would be contained in the core tank. If, following recent 5-watt operation, it becomes necessary to open the core tank, samples of the gas within the core tank will first be taken and analyzed to assure that there has been no hazardous release of radioactive effluents from the fuel material. If any significant levels do exist, appropriate radiological safety procedures will be followed prior to and during subsequent opening of the core tank. These procedures are under the direct supervision of the Reactor Supervisor as authorized by the Idaho State University Radiation Safety Officer.

The core tank is vented to purge radiolytic hydrogen and noble gas fission products as part of a biennial surveillance procedure. Only small quantities of radioxenon and radiokrypton have been detected in the performance of these surveillances. Thus, significant fission product leakage from the fuel core is assumed not to occur. At power levels of 5 watts or less, leakage is insignificant and measurable amounts have not been found in other similar facilities. If leakage is experienced under any conditions of operation within the scope or the authorization requested, procedures for the safe and authorized disposal of fission gas that may be formed together with procedures to assure containment will be formulated.

Two assumptions are used as a basis for calculating the power generated in an accident:

a. At time zero, a 2% step increase in reactivity is inserted with the reactor at low power (≤ 5 watts).

b. At time zero, the thermal energy in the core is negligible in comparison with the energy liberated during the accident and there is no heat removed from the core during the excursion.

The time-dependent behavior of the neutron density, including one average group of delayed neutrons is considered. A numerical finite-difference solution of the three coupled nonlinear differential equations yields a value of 54.4MW for the peak power at t = 204 milliseconds and a total energy release of 2.61 MJ, resulting in a temperature rise of 71.3° C. There will be about 1.45×10^{17} Mev of prompt gamma radiation produced in this excursion. Table 5.7-2 presents the residual activity formed in the core as a function of time after this excursion.

Table 5.7-2
Activity Contained in the Reactor Core following 2% Step Increase in Reactivity

t	Activity MeV-curies Equivalent
1 second	10^6
1 hour	50
1 day	1
1 month	0.02

5.8 Gaseous Fission Product Release

For the purpose of analysis, the gaseous fission products have been divided into two groups as shown in Table 5.8-1. The first group comprises those radionuclides that will remain in the tank water should the release occur when the tank is filled with water. This group includes the bromines and iodines. In the incredible event that no tank water is present, these isotopes would be added to the radioactive cloud and add to the hazard.

The second group comprises the insoluble volatiles, the krypton and xenon isotopes. They are the major source of potential radioactivity in the room (and outside) if tank water is present.

TABLE 5.8-1

GASEOUS FISSION PRODUCTS IN AGN FUEL AT 5-W OPERATION FOR 30 DAYS

Nuclide	Decay Constant (hr^{-1})	Inventory (mCi)
Group I (soluble volatiles)		
Br-83	3.02×10^{-1}	20.0
Br-84	1.31×10^{0}	46.1
Br-84m	6.95×10^{0}	0.95
Br-85	1.39×10^{1}	62.5
Br-87	4.49×10^{1}	112
I-129	4.60×10^{-12}	41.8
I-131	3.58×10^{-3}	121
I-132	3.07×10^{-1}	184
I-133	3.34×10^{-2}	271
I-134	7.93×10^{-1}	318
I-135	1.04×10^{-1}	247
I-136	2.90×10^{1}	130
	Total Iodines	1320 mCi
	Total Group I	1560 mCi
Group II (insoluble volatiles)		
Kr-83m	3.66×10^{-1}	20.1
Kr-85m	1.59×10^{-1}	62.5
Kr-85	7.67×10^{-6}	12.6
Kr-87	5.35×10^{-1}	113
Kr-88	2.50×10^{-1}	155
Kr-89	1.31×10^{1}	193
Kr-90	7.55×10^{1}	216
Kr-91	2.54×10^{2}	129
Xe-131m	2.41×10^{-3}	1.20
Xe-133m	1.26×10^{-2}	6.55
Xe-133	5.50×10^{-3}	2.71
Xe-135m	2.67×10^{0}	74.0
Xe-135	7.60×10^{-2}	200
Xe-137	1.07×10^{1}	24.7
Xe-138	2.45×10^{0}	229
Xe-139	6.08×10^{1}	237
Xe-140	1.56×10^{2}	249
	Total Group II	2190 mCi

5.8.1 Water activity

In the fission product release accident in which the tank water remains in situ, the fraction of the total soluble fission products which are released from the element is distributed in the water. Thus about 1.5 x 10^{-5} times the Group I fission products or 2.34 x 10^{-5} Ci remains in the tank. Since the volume of water in the reactor tank is 1000 gallons or 3.57 x 10 cm^3, the activity concentration is 6.55 x 10^{-6} μCi/cm^3. In 24 hours the activity would decrease to 1.17 x 10^{-6} μCi/cm^3. The activity then remains moderately constant because of the small decay constants for I-129, I-131, and I-133. However, under normal conditions the polyethylene moderator will prevent release of the gas, even to the inside of the core tank. It is therefore expected that the radioactive gas will not constitute a hazard to personnel.

The 5.8 MJ excursion will form 34 kCi of radioactive gas in the fuel particles. Gas diffusing out of these particles should be contained by the polyethylene fuel matrix. However, if any leakage from the core does occur, the presence of the two additional fluid-tight metal tanks insures that the gas will be contained. In the implausible event that the core leaks 0.l% of the activity and both tanks rupture, allowing 1.0% of this free radioactive gas to leak out of each tank, there would be 82 mCi of free gas outside the reactor l minute after the accident.

5.8.2 Exposure inside the reactor room

The maximum exposure to a person in the Reactor Laboratory will occur if all fission product gases from the core tank without water are distributed instantaneously within the room and no air change occurs after 30 days of continuous operation at 5 watts.

The total release of radioactive gases is 82 mCi and the volume of the reactor room is 5.11 x 10^8 cm^3 so that the concentration is 1.6 x 10^{-4} μCi/cm^3, or 5.9 Bq/cm^3. If the room is assumed to be equivalent to a hemisphere of radius R = 496 cm, the exposure rate at the center is given by

$$D = \frac{S_v}{2K\mu}\left(1 - e^{-\mu R}\right) \qquad \textbf{(10)}$$

where S_V = volumetric source strength [Bq/cm^3],
K = conversion factor for flux-dose rate
= 4.2×10^4 γ/cm^2-sec per mR/min for photons of 0.7 MeV (average energy of fission product decay)
μ = attenuation coefficient for air
= 3.5×10^{-5} cm^{-1} for E_γ = 0.7 MeV

Thus, the maximum exposure rate is 0.03 mR/min or 2.1 mR/hr (20.5 µSv/hr). Thus, an individual could remain in the room for about 47 hours before exceeding a dose of l00 mrem (1 mSv). The controlling factor, however, will be the dose to the thyroid from radioiodine. In any event the exposure rate from the plume added to the normal exposure due to reactor operation is sufficient actuate the radiation alarms so that the reactor room will be promptly evacuated. In addition, with the ventilation system in operation during such an accident, venting air at the rate of 1660 cfm, the dose rate to any person in the reactor room will be significantly reduced. Also, diffusion and escape of the gases through the concrete shield would delay and reduced the release.

The dose to the thyroid can be calculated at any time t from the following equation[9]

$$D_t = \frac{(5.92 \times 10^2)\, A_\gamma f_a \bar{E}\, (1 - e^{-\lambda_e t})}{m\lambda_e} \quad (rads) \tag{11}$$

where A_γ = inhaled iodine [Ci];
f_a = fraction which is deposited in critical organ;
$\bar{E}$ = effeciive energy absorbed by thyrcid per disintegration [MeV];
λ_e = effective decay constant, including both radioactive decay and biological elimination [s]; and
m = mass of the thyroid [gm].
For a long time, $t \to \infty$, and using $T_e = 0.693/\lambda_e$, Eq (11) becomes:

---

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

$$D_\infty = \frac{(8.54 \times 10^2) A_\gamma f_e \bar{E} T_e}{m} \quad (rads)$$
$$= constant \times \sum_i A_i \bar{E}_i T_{ei} \tag{12}$$

where the parameters are summed over the *i* isotopes the amount of iodine inhaled is given by

$$A_{\gamma i} = R\,\tau\left(\frac{A_i}{V}\right) \tag{13}$$

where A_i/V is the activity concentration and *R* the mean breathing rate over interval τ.
If τ is made equal to 1 second, then

$$D_\infty = \frac{(8.54 \times 10^2) f_e R}{m} \sum_i \frac{A_i \bar{E}_i T_{ei}}{V} \quad (rads/sec).$$

For the standard man,[10]

$f_e = 0.23$,
$m = 20$ gm,
$R = 10\ m^3/8\ hr = 3.47 \times 10^{-4}\ m^3/s$,

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

The data necessary for the summation are contained in Health Physics, 3, June 1960, and the necessary activity concentrations are calculated from data given in Table 5.8-1. The summation for I-131 through 1-136 yields a value of about D_∞ = 1.4mrem/sec (14.4 μSv/sec).

Assuming that 300 rem (3 Sv) to the thyroid is a limiting dose,[11] a person will have approximately 60 hours to evacuate from the reactor room. Actually the time will be longer than this since the room exhaust will be in operation and will reduce the dose. Further the leakage of the soluble iodine through the moderator and two containment tanks and the assumption no water in the shield tank is an extremely improbable scenario.

---

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

[11] J. J. DiNunno et al., loc. cit.

5.8.3 Exposure outside the building

On release of radioactive fission products into the reactor room, the facility Emergency Plan requires the reactor operator to initiate the evacuation of the Lillibridge Engineering Building by the actuation of the building fire alarm. The reactor operator will also shut the doors to the reactor room and trip the ventilation cut-out switch to prevent air exchange with the remainder of the building. Reentry to the laboratory will be allowed by the Reactor Supervisor in consultation with the ISU Radiation Safety Officer only after thorough radiation monitoring and when radiation levels permit entry. The room air may then be exhausted by normal operation of the ventilation system.

At the penthouse roof level of the Lillibridge Engineering Laboratory Building, the ventilation system exhausts with a discharge flow rate of about 4.25×10^6 cm^3 s^{-1} (9,000 cfm). It is reasonable to assume a dilution of the reactor room effluent by a minimum factor of 27. Using this, and assuming no further dilution outside the building the maximum dose to the thyroid may be calculated.

The calculation assumes: (1) complete mixing in the reactor room at all times, and (2) the person is immersed in the effluent from the building in which the concentration of radioactivity, X_t, at any time t is equal to $C_t/70$ where C_t is the concentration in the exhaust and in the room.

With assumption (1), C_t is given by a simple first order rate law:

$$C_t = C_o \exp\left[- \frac{a}{V} t\right]$$

where

Q = exhaust rate [cm^3/sec],
V = room volume [cm^3],
C_0 = initial concentration, at t = 0

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

$$A_t = \frac{RC_o}{70} \int_o^t \exp\left[- \frac{a}{V} t\right] dt$$

$$= \frac{RC_oV}{70a} \quad [1 - \exp(-\frac{a}{V} t)]$$

If the exposure is for an infinite time,

$$A_t = \frac{RC_oV}{a(70)}$$

ar .. comparing with the previous calculation, the total dose

$$int\ D \approx \frac{D\ (rads/sec) \cdot V(m^3)}{70\ a\ (m^3/sec)}$$

$$= 180 rad\ (1.8\ mGy)$$

Thus we may conclude that for any location outside of the building, the maximum possible dose to the thyroid from a fission product release will be considerably less than 0.07% of a maximum permissible dose, even when no credit is taken for radioactive decay.

5.9 Emergency Procedures

It is important to realize that it may not be obvious that an excursion occurred. For example, if the 2% step increase were caused by an insertion of fissionable material into the glory hole, the effect seen by the operator would be similar to that of an intentional scram with all of the reactor control console neutron monitoring and radiation surfeillance meters pegged off-scale. There would probably be no external evidence of an accident having occurred. The first action of the operator would be to observe the gamma-ray background measured by the portable monitoring instrument provided at the control console. If the scram were caused by a true excursion, the background would be up by a factor of about ten thousand after one second.

If the operator decided that an accident might have occurred, the operator would evacuate the reactor area and initiate building evacuation. The operator would then inform the appropriate administrative personnel, who would see that reentry to the area was not made until conditions were tolerable in regard to both radioactivity from the reactor as well as any airborne radioactivity that might have escaped. Complete emergency procedures are given in the facility Emergency Plan.

Any person close to, or exposed to radiation from the reactor excursion would be placed under observation at the Bannock Regional Medical Center which is prepared to handle and treat such persons. Similarly, an early attempt would be made to evaluate exposures by pocket dosimeters, film badges, etc., and to determine if any fission product leakage has occurred so that appropriate action could be taken.

After reentry, an estimation of the energy release would be made by gamma-ray flux measurements. When the radioactive hazards had been sufficiently reduced to permit working near the reactor another effort would be made to detect any radioactive gas leakage and the reactor would be secured by inserting cadmium poison rods into the glory hole and access ports. After a period of about a week, the control and safety rods would be permanently removed. At this time the core activity should be down to the millicurie range where the normal radiological handling procedures could be used. To place the reactor back in operation, the core would be assembled as a new critical assembly with a new safety fuse.

5.10 Safety Devices

Every effort has been made to make the reactor safe against all foreseeable circumstances. In the event of an electric power failure, the control system is designed to "fail safe" and scram the reactor. The terminology of "fail safe" for a power failure means that energy from the power system is used to hold the control rods in a critical position in the reactor, i.e., spring and gravity forces acting on the safety rods are held in check with an electromagnet. Loss of power de-energizes the holding magnet and the rods containing fuel are accelerated out of the reactor to their safe, stable positions.

Although every effort has been made to make the standard instruments used in the reactor "fail safe", the very nature of their electronic operation makes this quality intrinsically imperfect. However, instrument failure as a potential danger in the reactor operation is decreased by having three independent systems.

One of the scram systems in the reactor is connected to a water-level indicator so that the reactor shuts down if the water level decreases by a small amount. This safety device protects personnel from inadvertently operating the reactor with a low water level, as well as stopping operation if a significant leak develops.

A thermal safety fuse is provided which, in the event of a 2% reactivity excursion, will shut down the reactor in a manner such that the accident cannot recur. The basic concept of the fuse involves having a safety device that prevents large overloads, is foolproof (not dependent on the electronic circuits), and is not easily circumvented. The core fuse is a polystyrene plug containing 150 mg/cm^3 of ^{235}U which supports the bottom half of the core and a section of the reflector.

The higher uranium density loading results in heat being generated at a higher rate in the fuse than in the core. The temperature in the fuse rises about twice as fast as it does in the core. At a core temperature of about 120° C, the fuse melts and the core separates, thereby completely shutting down the reactor. This temperature is not reached except in cases of large energy releases, such as an excursion. Polystyrene is used as the fuse material because of its resistance to changes in physical properties induced by radiation. The melting point is unaffected below doses of 100 megarad, and therefore the properties of the fuse are not affected by normal operation for many years. Even if the safety and control rods remain in the core, the separation of the two halves of the core would cause the reactor to go subcritical.

In the event of an earthquake of horizontal amplitude greater than 0.16 cm (1/16 inch), the earthquake switch is designed to shut down the reactor[12]. The safety rods are spring loaded and operate very nearly independently of the gravitational field orientation. Thus this safety system makes the reactor safe even in a major tremor.

The reactor itself is designed to be structurally earthquake proof: continual horizontal accelerations of 0.6 g will not overturn or otherwise damage the system. Vertical oscillations of comparable magnitudes are allowed for in the design safety factor. If the support does break and the reactor overturns, the control and safety rods, which are housed within a compartment that is enclosed by the main water shield tank, will not be pushed into the core. Finally the concrete shielding walls should prevent severe tilting of the reactor.

Some additional accidents or events that might conceivably result in the release of radioactive materials from the reactor are considered below. In the event of fire, damage to the reactor is not considered likely, since the AGN-201 reactor superstructure is intrinsically fireproof, as is the reactor room. The reactor may be sprayed with water, CO_2, or any other fire-quenching material without damage to the reactor tank or fear of inadvertent criticality. The reactor will be shut down and the reactor room

---

[12] A.T. Biehl, R.P. Geckler, S. Kahn, and R. Mainhardt, Elementary Reactor Experimentation, Aerojet-General Nucleonics, San Ramon, CA, October 1957.

locked in the case of fire, and the Reactor Supervisor or designated alternate will be notified. If a fire involving the reactor or laboratory does occur, the reactor will be thoroughly inspected for damage before operation resumes.

In the extremely improbable event of a flood, no special precautions are necessary other than those normally taken in the event of a flood at an industrial site. The reactor will be secured and not operated at this time. The radiological hazards are not severe as the reactor is built to withstand minor flooding (one-foot of water). In the event of a major flood where the reactor might be overturned or carried away, there is no serious problem since the self-contained reactor has been designed to withstand such an emergency. Further, because the core resides within a water-tight shield tank, there is no risk of inadvertent criticality in the event of complete immersion of the reactor.

It is extremely unlikely that a storm could damage the AGN-201 reactor. However, in the event of a severe storm, the reactor will be shut down and secured. In the event of civil disturbance such as a strike or riot, the reactor will be shut down and secured and guards will be posted at the entrance of the reactor laboratory to prevent unauthorized entry.

In addition to all the above safety features and administrative controls, there exists a negative temperature coefficient in the reactor core. The temperature of the reactor will vary during normal operating conditions as well as during an excursion. In both cases, the change in temperature will cause a change in reactivity. The temperature equilibrium rise of the core can be shown to be on the order of 2° C when the reactor is operated at 5 watts. Thus, under this condition, the steady state temperature is essentially ambient temperature.