IDAHO
STATE
UNIVERSITY

Vice President for Academic Affairs

Campus Box 8063
Pocatello, Idaho
83209-8063

10 July 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Re: Response to Request for Additional Information, Dated 29 May 2003: Idaho State University AGN-201 Renewal, TAC No. M94199
Docket No. 50-284, License No. R-110

Dear Madam/Sir:

Transmitted herewith is our formal response to your Request for Additional Information (RAI), dated 29 May 2003, regarding the renewal of the AGN-201 reactor operating License No. R-110 at Idaho State University (ISU).

In addition, we are submitting the revised Reactor Operator Requalification Plan at this time for NRC review and approval. Please find also enclosed: (1) a corrected copy of the facility Technical Specifications (TS), rev. 5; and (2) a corrected copy of pages 3 and 12 of the Facility Emergency Plan (EP), rev. 6. We request that you remove pages 3 and 12 of the Facility Emergency Plan (EP), rev. 6, and replace them with the corrected pages provided here.

Should you have any questions or require additional information, please call the Reactor Administrator, Dr. Jay Kunze at (208) 282-2902 or the Reactor Manager/Supervisor, Dr. John Bennion at (208) 282-3351.

Sincerely,

Jonathan Lawson

Dr. Jonathan Lawson
Vice President for Academic Affairs

A020
A035

Affirmation:
I certify under penalty of perjury that the foregoing is true and correct.

Executed on July 10, 2003 by Christine S. Salstrom
(Date) (Notary)
Expires: April 4, 2008

CHRISTINE S. SALSTROM NOTARY PUBLIC

Copy w/enclosures:
Mr. Daniel Hughes, Project Manager
Mr. Paul Doyle

Phone: (208) 282-2362
FAX: (208) 282-4487

Response to Request for Additional Information
July 10, 2003

This document provides the Idaho State University's response to the U.S. Nuclear Regulatory Commission's Request for Additional Information (RAI), dated May 29, 2003, regarding the renewal of the ISU AGN-201 reactor operating license (License No. R-110, Docket No. 50-284). Each question posed in the RAI is repeated below, followed by the response. The responses given below have been reviewed and approved by the ISU Reactor Safety Committee.

I. <u>*Technical Specifications*</u> *In general, the NRC found the revised Technical Specifications submitted by letter dated January 31, 2003, to be well written. However, the reviewer identified issues requiring clarification.*

A. The revised Technical Specifications included changes which were not accompanied by justification. Phone conversations with John Bennion of your staff assured the reviewer that these changes were based on bringing the Technical Specifications in line with the latest guidance associated with Research reactor licensing. The NRC lauds the effort to bring the Technical Specifications up to date, but requests justification for the following changes:

1. *The Limiting Safety System Setting for Nuclear Safety Channels 2 and 3 was changed from 10 watts to 6 watts. The NRC realizes that this is a change in the conservative direction.*

Response: Upon further review of our request to lower the LSSS (Limiting Safety System Setting) from 10 watts to 6 watts, we believe that the facility will be better served by keeping the LSSS at 10 watts as allowed in the original operating license. We therefore drop our request to change this technical specification and will retain the existing 10-watt LSSS technical specification for Nuclear Safety Channels 2 and 3.

2. *The definition for scram time was changed from a measurement of the time for the rod to eject out of the core, to a measurement of the time from the onset of a scram signal to the time that the rod is fully ejected from the core. Although the time limit for this has been increased, the NRC realizes that the overall change to this requirement is actually conservative, in that it also measures degradation of the scram circuitry, in addition to the measurement of the time for the rods to be ejected.*

Response: The justification for this change is twofold. First, the change conforms with the ANSI/ANS-15.1-1990 recommended definition of scram time as being "the elapsed time between the initiation of a scram signal and a specified movement of a control of safety device." Our current Technical Specifications require the annual measurement of rod withdrawal time, which only measures the time for the scrammable safety and control rods to be ejected from their most reactive positions. This measurement disregards any delay in the scram circuitry between the initiation of a scram signal and the complete withdrawal of a rod. Redefining the scram time to include this possible delay therefore gives us the ability to observe potential degradation in both the scram circuitry and the control rod drive mechanisms that effect the ejection.

Second, the one-second scram-time specification is not unreasonable or unacceptably conservative for two reasons: (1) NUREG 1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Form and Content," states "In most non-power reactors, for rods 2 to 3 feet long, full rod insertion time in the absence of excess mechanical friction or interference is less than 1 second," and (2) the one-second scram-time specification has been part of the Technical Specifications for the University of New Mexico's AGN-201 reactor facility for many years.

Fundamentally, the change is appropriately conservative compared to the existing specification because it enables facility operating personnel to monitor the performance of the entire scram channel for potential degradation of both mechanical and electrical components.

3. *The k_{eff} for stored spent fuel was changed from 0.8 to 0.9.*

Response: The justification for this change is to bring the Technical Specifications into consonance with ANSI/ANS-15.1-1990 recommended limit on k_{eff} for fuel in storage. Section 5.9 of ANSI/ANS-15.1-1990, titled "Fissionable Material Storage," recommends: "Fuel, including fueled experiments and fuel devices not in the reactor, shall be stored in a geometrical array where k_{eff} is no greater that 0.9 for all conditions of moderation and reflection using light water except in cases where an approved fuel shipping container is used, then the k_{eff} for the container shall apply."

B. *In addition, the NRC reviewer noted that several paragraphs in Section 6 of the Technical Specifications refer to "an unreviewed safety question." This terminology is no longer part of 10 CFR 50.59. Please replace "an* *unreviewed safety question" with words relating the meeting the criteria contained in 10 CFR 50.59.(c)(2)(i) through (viii).*

Response: Section 6 of the Technical Specifications has been revised so that all occurrences of the phrase "unreviewed safety question" have been replaced by the phrase (modified appropriately to fit the specific context) "require a license amendment per the criteria listed in 10 CFR 50.59.(c)(2)(i) through (viii)." A corrected copy of Revision 5 of the Technical Specifications has been sent by email to the NRC reviewer.

II. *Emergency Plan* *Once again, the Emergency Plan has been upgraded to incorporate latest NRC and industry guidelines, and once again the upgrade was generally well written. Please submit additional information to clarify the following issues:*

A. § 7.3.2 Evacuation Procedure

1. *As a response to an emergency requiring an evacuation in § 7.3.1(a)(i) the licensed Reactor Operator is directed to sound the fire alarm either on the east wall near the entrance to the Reactor Laboratory or one of the additional fire-alarm pull stations located in the main hallway of the basement level of the Lillibridge Engineering Laboratory near the north and south stairwells. This is further emphasized by the Evacuation Procedure, C.6 within Attachment C to the Emergency Plan, which in step 5, directs the Reactor Operator to initiate building evacuation by tripping one of the building fire alarms.*

2. *§ 7.3.2(b)(i) has personnel gather at the "Hold Station" in the Southeast corner of the Machine Shop (Room 126 within the Lillibridge Engineering Building).*

Figure A.3, shows the Machine shop as Room 126 within the Lillibridge Engineering Laboratory. Please clarify whether you train people to leave the building, or to proceed to the holding area in the machine shop, upon hearing a fire alarm in the Lillibridge Engineering Laboratory. In addition, the first sentence of 8.1 refers to Figure A.2. This reference should refer to Figure A.3.

Response: Only personnel who are present in the EPZ (Emergency Planning Zone, i.e., the reactor room) at the time building evacuation is initiated are required to proceed to the "Hold Station" in the Machine Shop where they are checked for contamination before leaving the building. All other people occupying offices, classrooms, or laboratories within the LEL would evacuate the building as would be expected upon actuation of the building fire alarm. All personnel who are required to spend any significant time in the reactor room, either as employees who have responsibilities with the operation or maintenance of the reactor or as students using the reactor as part of their academic coursework or research activities, receive training in emergency procedures and are instructed to proceed to the Hold Station and wait until excused by either the Reactor Supervisor or the Radiation Safety Officer if building evacuation is ordered by the Reactor Operator. This training is also provided annually to all campus security officers. In addition, the erroneous reference to Figure A.2 in section 8.1 of the Emergency Plan has been corrected to Figure A.3.

III. *Requalification Program* *In your letter date January 30, 2003, responding to the Request of Additional Information, you committed to changing your Operator Requalification Program by June 30, 2003. Please submit the updated requalification plan as soon as possible. Please do not hold up your response to this letter to include the updated Operator Requalification Program.*

Response: The updated Operator Requalification Program is included in this package for NRC review and approval.

REACTOR OPERATOR REQUALIFICATION PLAN

FOR

IDAHO STATE UNIVERSITY AGN-201 M REACTOR (SERIAL NO. 103)

DOCKET NO. 50-284

Reactor Operator Requalification Plan, Rev. 3
RSC Approval: January 30, 2003
NRC Approval: To Be Approved

TABLE OF CONTENTS

1.0 PURPOSE

This document sets forth the requirements for the Reactor Operator and Senior Reactor Operator Requalification Program for the Idaho State University (ISU) AGN-201 Nuclear Reactor (Docket No. 50-284) in accordance with the requirements provided in 10 CFR 55.59. 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 LECTURES

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

Topic	References
1. Nuclear Reactor Theory	Standard Nuclear Engineering Texts
2. Radiation Control and Safety	10 CFR 20 and 30, ISU Radiation Safety Manual
3. Governing Regulations	10 CFR 19, 50, 55, and 70
4. Reactor Design	Reactor Facilty Study Material
5. Reactor Control and Safety Systems	Reactor Facilty Study Material
6. Reactor Operating Characteristics	Reactor Facilty Study Material
7. Reactor Facility Procedures, Plans, Policies, and Rules	Operating, Maintenance, and Surveillance Procedures; Emergency Plan; Security Plan
8. 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

A. Each operator shall perform licensed functions for at least four hours per quarter to satisfy 10 CFR 55.53(e).

B. 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 a licensed operator.

C. As a minimum, to demonstrate proficiency at manipulating the reactor facility controls, each operator shall 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. Reactor 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.

A. 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 10 CFR 55.41 and may include requirements of 10 CFR 55.43. Generally, the Reactor Supervisor (RS) and/or the Reactor Administrator (RA) shall be responsible to prepare, administer, and grade the written examination. However, if the RA is not a licensed operator and if the RA is not able to prepare a valid requalification examination, then the RS may delegate the preparation, administration, and grading of the examination to either: (1) a currently licensed SRO, or (2) a previously licensed SRO who is familiar with the facility and operator licensing requirements. The RS and the RA, if the RA is licensed, shall each take the written examination at least once during the two-year requalification period, alternating between the two positions as to who prepares and takes the examination in a given year. The person who prepares, administers, and grades the written examination shall be exempt from taking the written examination during the year that they prepare the examination, but shall not be exempt from taking the written examination both years of a requalification period.

B. Console Examinations

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

1. Prestart checks,
2. Startup,
3. Operation at Power,
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.

C. Grading

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

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 enroll in an accelerated training program in the deficient area.

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. 10 CFR 55.53(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.

REQUALIFICATION/TRAINING LECTURE FORM

Topic_______________ Instructor_________ Date_________

Names of lecture attendees

__________________	__________________
__________________	__________________
__________________	__________________
__________________	__________________
__________________	__________________
__________________	__________________
__________________	__________________
__________________	__________________

Maintenance and Surveillance Log entries covered 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

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-JobTraining LICENSED FUNCTIONS

Quarter	Date	Hours	Date	Hours	Date	Hours	Total
1	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
2	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
3	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
4	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
5	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
6	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
7	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____
8	__/__/__	_____	__/__/__	_____	__/__/__	_____	
	__/__/__	_____	__/__/__	_____	__/__/__	_____	_____

Operator: ______________________________ License No.: ______________________

On-the-Job Training (continued)

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:__/__/__ Reactor Supervisor: ____________________

Quarter: _______ Date:__/__/__ Reactor Supervisor: ____________________

Quarter: _______ Date:__/__/__ Reactor Supervisor: ____________________

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

Topic:________________ Date:__/__/__ Reactor Supervisor: __________________

Topic:________________ Date:__/__/__ Reactor Supervisor: __________________

Topic:________________ Date:__/__/__ Reactor Supervisor: __________________

Medical Examination Completion Date:__/__/__

Medical Examination Due Date:__/__/__

APPENDIX A

TO FACILITY OPERATING

LICENSE NO. R-110

TECHNICAL SPECIFICATIONS

FOR

IDAHO STATE UNIVERSITY AGN-201 M REACTOR (SERIAL NO. 103)

DOCKET NO. 50-284

AGN-201 Technical Specifications, Rev. 5
RSC Approval: January 27, 2003
NRC Approval: To Be Approved

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1.0 DEFINITIONS

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

1.1 Authorized Operators – An authorized operator is an individual authorized by the Reactor Supervisor to operate the reactor controls and who does so with the knowledge of the Reactor Supervisor and under the direct supervision of a Reactor Operator.

1.2 Certified Observers – A certified observer is an individual certified by the Reactor Supervisor as qualified to activate manual scram and initiate emergency procedures.

1.3 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.

1.4 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.

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

1.6 Coarse Control Rod – The scrammable control rod that can be mechanically withdrawn/inserted at two possible speeds (25 to 55 seconds full insertion time or 75 to 125 seconds full insertion time).

1.7 Control Rod – Any of the four moveable rods loaded with fuel that are manipulated by the reactor operator to change the reactivity of the reactor.

1.8 Excess Reactivity – The amount of reactivity above critical ($k_{eff} = 1$). The excess reactivity is the amount of reactivity that would exist if all control rods were moved to their maximum reactive positions from the point where the reactor is exactly critical.

1.9 Experiment –

a. An experiment is any of the following:

(1) An activity utilizing the reactor system or it 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 1.9.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. Removable 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.

1.10 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.

1.11 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, 7th ed., (1989), 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.

1.12 Fine Control Rod – A low worth, non-scrammable control rod used primarily to maintain an intended power level. Its position may be varied manually.

1.13 Measured Value – The measured value is the value of a parameter as it appears on the output of a channel.

1.14 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.

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

1.16 Operating – Operating means a component or system is performing its intended function in its normal manner.

1.17 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.

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

1.19 Reactor Operation – Reactor operation is any condition wherein the reactor is not shutdown.

1.20 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.

1.21 Reactor Secured – The reactor is secured when:

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.

c. No work is in progress involving core fuel, core structure, installed control rods, or control rod drives unless they are physically decoupled from the control rods, and

d. No experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment, or one dollar, whichever is smaller.

1.22 Reactor Shutdown – The reactor is shutdown if it is subcritical by at least one dollar in reactivity in the reference condition with the reactivity worth of all experiments included. 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.

1.23 Restricted Area – A restricted area is an area in which access to personnel is controlled for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

1.24 Safety Channel – A safety channel is a measuring channel in the reactor safety system.

1.25 Safety Control Rod – One of two scrammable control rods that can be mechanically withdrawn/inserted at only one speed (25 to 55 seconds full insertion time).

1.26 Scram Time – The time for the control rods acting under gravity and spring loading to change the reactor from a critical to a subcritical condition. In most cases, this time is less than or equal to the time it takes for the rod to fall from a full-in to full-out position.

1.27 Shall, Should and May – The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission—neither a requirement nor a recommendation.

1.28 Shutdown Margin – Shutdown margin shall mean the minimum shutdown reactivity necessary to provide confidence that the reactor can be made subcritical by means of the control and safety systems starting from any permissible operating condition assuming that the most reactive scrammable rod and the Fine Control Rod remain in their most reactive positions, and that the reactor will remain subcritical without any further operator action.

1.29 Static Reactivity Worth of Experiments - 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.

1.30 Surveillance Time – A surveillance time indicates the frequency of tests to demonstrate performance. Allowable surveillance intervals shall not exceed the following:

a. Two-year - interval not to exceed 30 months.
b. Annual - interval not to exceed 15 months.
c. Semiannual - interval not to exceed seven and one-half months (30 weeks).
d. Quarterly - interval not to exceed four months.
e. Monthly - interval not to exceed six weeks.

1.31 True Value – The true value is the actual value of a parameter.

1.32 Unscheduled Shutdown – An unscheduled shutdown is defined as any unplanned shutdown of the reactor caused by actuation of the reactor safety system, operator error, equipment malfunction, or a manual shutdown in response to conditions that could adversely affect safe operation, not including shutdowns that occur during testing or check-out operations.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Applicability

This specification applies to the maximum safe 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 0.44°C/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 (LSSS)

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 No. 2	High Power	≤ 10 watts
Nuclear Safety No. 3	High Power	≤ 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 operation 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 large 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% $\Delta k/k$ (\$0.878) referenced to 20°C.

b. The shutdown margin with the most reactive safety or control rod fully inserted and the fine control rod rod fully inserted shall be at least 1% $\Delta k/k$ (\$1.35).

c. The reactivity worth of the control and safety rods shall ensure subcriticality on the complete 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 scrammable rod fails to scram and the Fine Control Rod 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.

Specification

a. The total scram withdrawal time of the safety rods and coarse control rod shall be less than 1 second.

b. The average reactivity addition rate for each control or safety rod shall not exceed 0.065% $\Delta k/k$ per second ($0.00877 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, with the approval of the Reactor Supervisor, Safety Channel No. 1 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 manual scram shall be provided on the reactor console.

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

j. An operable installed area radiation monitor capable of detecting gamma radiation shall be immediately available to reactor operating personnel whenever the reactor is not secured. When required monitors are inoperable, portable instruments may be substituted for any installed monitor for periods up to two weeks, while the installed monitor is being repaired.

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 Nos. 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.

An area 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 down the reactor and assess the hazards to personnel.

Table 3.1 Reactor control and safety systems set-point specifications.

SAFETY CHANNEL	SET POINT	FUNCTION
Nuclear Safety Channel No. 1 (Startup Count Rate Channel) Low Power	5% Full Scale	Scram at levels < 5% of Full Scale
Nuclear Safety Channel No. 2 (Log Power Channel) High Power	10 watts	Scram at power > 10 watts
Nuclear Safety Channel No. 2 (Log Power Channel) Low Power	3.0×10^{-13} amps	Scram at source levels $< 3 \times 10^{-13}$ amps
Reactor Period	5 sec	Scram at periods < 5 sec
Nuclear Safety Channel No. 3 (Linear Power Channel) High Power	10 watts	Scram at power > 10 watts
Nuclear Safety Channel No. 3 (Linear Power Channel) Low Power	5% Full Scale	Scram at levels < 5% of Full Scale
Manual Scram	-----	Scram at operator option
Area Radiation Monitor	≤ 10 mR/hr	Alarm at or below level set to meet requirements of 10 CFR 20

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:

(1) A total effective dose equivalent to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 0.1 mSv (10 mrem) as a result of any airborne pathway, or

(2) A total effective dose equivalent to any person occupying an unrestricted area continuously for a period of two hours starting at the time of release in excess of 1 mSv (100 mrem) as a result of all pathways, or

(3) A total effective dose equivalent to any radiation worker occupying a restricted area during the length of time required to evacuate the restricted area in excess of 50 mSv (5 rem).

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. Specification 3.3c conforms to the regulatory position put forth in 10 CFR 20, issued January 1993.

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 an installed radiation survey instrument capable of detecting gamma radiation shall be immediately available to reactor operating personnel whenever the reactor is not secured.

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

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, or other approved experiments which require the thermal column to be empty.

(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 250 μSv/hr (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.

b. Total excess reactivity and shutdown margin shall be determined annually.

c. The reactivity worth 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 15-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.

b. Safety and control rods and drives 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 Channel No. 1, No. 2, and No. 3
   Manual Scram

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

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

   Nuclear Safety Channel No. 1, No. 2, and No. 3

f. Prior to each day's operation or prior to each operation extending more than one day, Safety Rod No. 1 shall be inserted and scrammed to verify operability of the manual scram system.

g. The period, count rate, and power level measuring channels shall be calibrated and set points verified annually.

h. The shield 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.

i. The radiation monitoring instrumentation shall be calibrated annually.

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 prior to each startup. Leakage sufficient to leave a puddle on the floor 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 annually under the supervision of the Radiation Safety Officer.

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, to determine the location of radiation and high radiation areas corresponding to reactor operating power levels.

Basis

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 ^{235}U 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 upper 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/cm^3) 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-product 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 or "thermal column tank" may serve as a shield tank when filled with water or as a thermal column when filled with graphite.

d. The 198-cm-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 3785 liters 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 gm/cm^3). Results of shielding calculations are summarized in the ISU AGN-201 Reactor Safety Analysis Report.

f. Two safety rods and one control rod (identical in size) contain up to 20 grams of ^{235}U 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 polyethylene with or without fuel.

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 0.9 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. 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

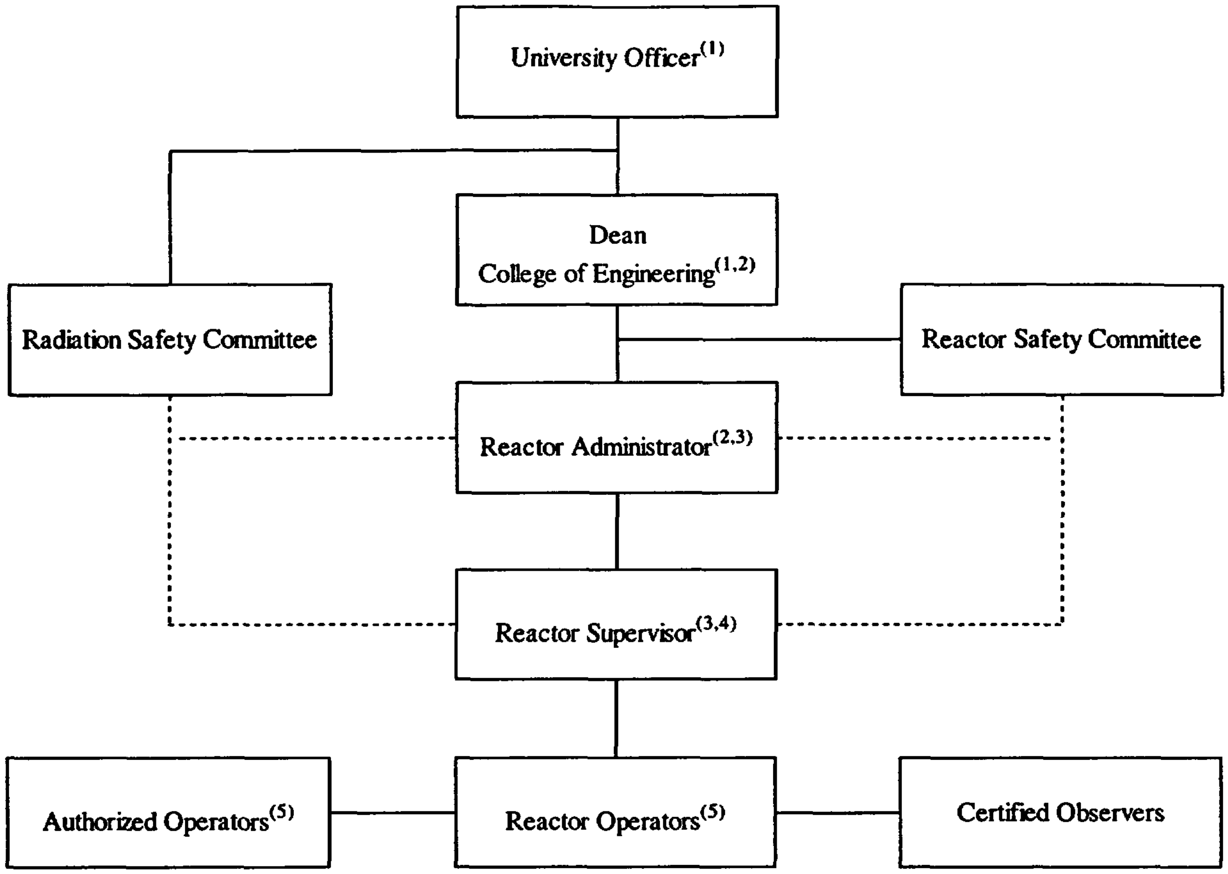
The dean of the College of Engineering is the administrative officer responsible for the operation of the College of Engineering.

6.1.3 Reactor Administrator

The Reactor Administrator (RA) is the administrative officer responsible for the operation of the AGN-201M Reactor Facility. In this capacity the RA 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. The Reactor Administrator shall be responsible for appointing the Reactor Supervisor, who reports to the Reactor Administrator. The RA 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. The RA shall be an ex officio member of the Reactor Safety Committee.

6.1.4 Reactor Supervisor

The Reactor Supervisor (RS) 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. The RS 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 RA; and be responsible for the preparation of experimental procedures involving use of the reactor. The RS shall hold a valid Senior Reactor Operator's license issued by the U.S. Nuclear Regulatory Commission.



(1) University Officer and Dean of the College of Engineering may be same individual.
(2) Dean of the College of Engineering and Reactor Administrator may be same individual.
(3) Reactor Administrator and Reactor Supervisor may be same individual.
(4) Requires NRC Senior Reactor Operators License.
(5) Requires NRC Reactor Operators License except where exempt per 10 CFR 55.13.

Figure 1. Administrative Organization of the ISU AGN-201M Reactor Facility, NRC License R-110.

Persons holding positions on the Administrative organization shall meet or exceed the qualification requirements of ANSI/ANS-15.4-1988, "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 shall hold a valid Reactor Operator's license issued by the U.S. Nuclear Regulatory Commission.

6.1.6 <u>Authorized Operators</u>

Individuals authorized by the Reactor Supervisor to operate the reactor controls and who do so with the knowledge of the Reactor Supervisor and under the direct supervision of a Reactor Operator.

6.1.7 <u>Certified Observers</u>

Individuals certified by the Reactor Supervisor as qualified to activate manual scram and initiate emergency procedures in the event of an emergency situation during reactor operation.

6.1.8 <u>Reactor Safety Committee</u>

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 do not require a license amendment per the criteria listed in 10 CFR 50.59(c)(2)(i) through (viii), 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.

6.1.9 <u>Radiation Safety Committee</u>

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

6.1.10 <u>Radiation Safety Officer</u>

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 <u>ex officio</u> member of the Reactor Safety Committee.

6.1.11 Operating Staff

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

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

(2) One Certified Observer in the reactor control room.

(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 reactivity of the reactor.

c. A listing of reactor facility personnel by name and phone number shall be conspicuously posted in the reactor control room.

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 ANSI/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 will be sought in areas where additional experience is considered necessary by the Reactor Administrator.

6.3 Training

The Reactor Supervisor shall be responsible for directing training as set forth in ANSI/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 (RSC) shall meet as often as deemed necessary by the RSC Chair but shall meet at least annually. A quorum for the conduct of official business shall consist of not less than one-half of the current RSC membership and shall include the Chair or designated alternate. At no time shall the operating organization comprise a voting majority of the members at any RSC meeting.

6.4.2 Reviews

The RSC 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 50.59, to verify that such actions do not require a license amendment per the criteria listed in 10 CFR 50.59(c)(2)(i) through (viii).

b. Proposed changes to procedures, equipment or systems that change the original intent or use, and are non-conservative, or those that require a license amendment per the criteria listed in 10 CFR 50.59(c)(2)(i) through (viii).

c. Proposed tests or experiments which are significantly different from previously approved tests or experiments, or those that require a license amendment per the criteria listed in 10 CFR 50.59(c)(2)(i) through (viii).

d. Proposed changes in Technical Specifications or other license documents.

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 RSC 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 annually.

b. The performance, training, and qualifications of the entire facility staff, at least every two years.

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 annually.

d. The Facility Emergency Plan and implementing procedures at least every two years.

e. The Facility Security Plan and implementing procedures, at least every two years.

6.4.4 Authority

The RSC shall report to the dean of the College of Engineering and shall advise the Reactor Administrator on those areas of responsibility outlined in Section 6.1.6 of these Technical Specifications.

6.4.5 Minutes of the Reactor Safety Committee

The RSC Chair 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. Minutes should be distributed to all administrative levels and RSC members within 3 months after each meeting.

6.5 Approvals

The procedure for obtaining approval for any change, modification or procedure which requires approval of the RSC 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 RSC for review, comment, and possible approval.

c. The RSC shall approve the proposal by majority vote.

d. The Reactor Administrator shall provide final approval after receiving the approval of the RSC.

6.6 Procedures

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.

g. Radiation Safety Protection for all reactor-related personnel.

The above listed procedures shall be approved by the Reactor Administrator and the RSC. Temporary procedures which do not change the intent of previously approved procedures and which do not involve a license amendment per 10 CFR 50.59 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 RSC.

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 Reactor Administrator immediately. The violation shall be reported to the NRC and the RSC Chair or designated alternate not later than the next working day.

c. A Safety Limit Violation Report shall be prepared for review by the RSC. This report shall describe the applicable circumstances leading to the violation including, when known, the cause and contributing factors; the effects of the violation upon facility components, systems, or structures and on the health and safety of personnel and the public; and corrective action to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the NRC and RSC 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, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Attention: Document Control Desk, Washington, D.C., 20555.

6.9.1 Annual Operating Report

Routine operating reports covering the operation of the reactor 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 safety that occurred during the reporting period.

   b. Results of major surveillance tests and inspections.

(2) A monthly tabulation showing the hours the reactor was operated and the energy produced by the reactor in watt-hours.

(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 neither a change to the Technical Specifications nor a license amendment per 10 CFR 50.59(c)(2)(i) through (viii) 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 received during the reporting period by facility personnel and 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. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, an amended licensee event report shall be completed and reference shall be made to the original report date.

a. Prompt Notification with Written Follow-up

The types of events listed below are considered reportable occurrences and shall be reported as expeditiously as possible by telephone and confirmed by overnight mail, mailgram, or facsimile transmission to the NRC Document Control Desk no later than the first working day following the event, with a written follow-up 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 set point 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 found to be less conservative than the limiting condition for operation established in the technical specifications, without evaluation and permitted remedial action.

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

(4) Reactivity balance anomalies involving:

(i) Disagreement between expected and actual critical positions exceeding 0.3% $\Delta k/k$;

(ii) Exceeding excess reactivity limits;

(iii) 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 the 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 the 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 Analysis 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.

(9) Deployment of the thermal fuse.

6.9.3 <u>Special Reports</u>

Special reports which may be required by the NRC shall be submitted to the U.S. NRC Document Control Desk within the time period specified for each report. These reports include notification of changes in Level 1, 2, or 3 administration, as defined in ANSI/ANS-15.4 and shown in Figure 1, which shall be reported within 45 days of such a change.

6.10 Record Retention

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 RSC for the performance of new or special experiments.

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

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

h. Records of meetings of the RSC.

1.0 INTRODUCTION

This Emergency Plan shall be used as a plan of action to follow in the event of an emergency situation at the nuclear facility located at Idaho State University, Pocatello, Idaho.

The nuclear facility consists of an AGN-201 nuclear reactor manufactured by Aerojet General Nucleonics (AGN) in 1956 and a Subcritical Assembly. The AGN reactor and Subcritical Assembly are owned by Idaho State University and operated under U.S. NRC License Nos. R-110 and SNM-1373, respectively. The reactor is licensed to operate at a maximum power of 5 W. The fuel for both facilities consists of uranium enriched to 19.9% 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 UO_2 uniformly 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 that provide safe and efficient operation of the reactor. The weight of the reactor unit, with the water shield, is approximately 20,000 pounds; the weight of the reactor control console is about 800 pounds. The AGN-201 reactor is located in Room 20 of the basement of the Lillibridge Engineering Laboratory (LEL) building at Idaho State University (ISU). Refer to Appendix A for the floor plans of the laboratory.

The Subcritical Assembly is located in Room 23. The Subcritical Assembly fuel consists of 150 aluminum-clad plates containing an aluminum-uranium mixture enriched to 19.9% U-235. The fuel plates measure 26-in long by 3-in wide by 0.08-in thick. The aluminum cladding is 0.020 inches thick. The fuel is stored in a locked cabinet when the Subcritical Assembly is not in operation.

The Emergency Plan has the following purposes:

(1) To describe provisions made through advanced planning to cope with an emergency situation not normally expected from routine operations of the nuclear facilities; and

(2) To provide assurances that appropriate measures can and will be taken to mitigate the consequences of such an emergency, should it occur, and thereby further assure the protection of the public health and safety, as well as the safety of radiation workers at the facility.

This Emergency Plan was prepared to be in compliance with ANSI/ANS-15.16-1982, "Emergency Planning for Research Reactors," and NUREG-0849, "Standard review Plan for the review and Evaluation of Emergency Plans for Research and Test Reactors." Differences and variation of this document from those that are used for other reactor facilities realistically reflect the characteristics unique to the ISU Nuclear Facility.

10 CFR 20. If warranted by the situation, emergency doses in accordance with 10 CFR 20.1206 may be authorized by the DEO for volunteers but shall be consistent with the Environmental Protection Agency (EPA) Emergency Workers and Lifesaving Activity Protective Action Guides.

7.3.4 Doors leading to and from the Operations Boundary and other onsite areas shall remain shut and locked to minimize exposures to radiation and the spread of contamination. Restricted areas shall be posted and access shall be controlled as directed by the DEO consistent with the nature of the emergency.

7.3.5 Radiation dose rates shall be continuously monitored with survey meters and airborne particulate samplers. Those monitors permanently installed within the Operations Boundary shall be used, if accessible, otherwise portable units available to radiation safety personnel shall be used.

7.3.6 Personnel exposures shall be monitored by TLD badge and/or self-reading pocket dosimeters. In the event that unmonitored personnel may have been exposed to radiation, an estimate of exposures shall be made by the DAC based upon surveys and air particulate samples for areas that were occupied, the potential for exposure from the emergency situation that existed during the time such areas were occupied, and standard dose assessment practices.

7.4 Corrective Actions

The type of actions that could mitigate or correct the problems for each emergency class listed in this plan shall be specified in the Implementing Procedures in section 11.

8.0 EMERGENCY FACILITIES AND EQUIPMENT

8.1 Emergency Support Center (ESC)

The Emergency Support Center will normally be in the northeast corner of the Machine Shop (Room 126) near the large overhead door, as shown in Appendix A, Figure A3. Emergency control directions will be given from this area. In the event that this location is inaccessible or otherwise considered inadequate for the emergency at hand, an alternate location should be established in the Technical Safety Office located in the Physical Science Building, as directed by the DEO.

Selection of the normal and alternate ESC is based upon the proximity of locations for portable radiation monitoring and sampling equipment, fixed systems for determining specific radionuclide identification and analyses, decontamination equipment, and availability of telephone communications.

8.2 Assessment Facilities

8.2.1 Portable survey instruments of sufficient range to determine alpha, beta, gamma and neutron radiation levels commensurate with potential radiological consequences of credible emergency situations shall normally be stored, except when in use, in Room 20 (Reactor Laboratory room) of the LEL building. Additional portable alpha, beta, and gamma survey instruments are stored for emergency use in the Emergency Locker in