TEXAS A&M UNIVERSITY
AGN-201M REACTOR FACILITY
LICENSE NO. R-023
DOCKET NO. 50-059

LICENSE RENEWAL APPLICATION

SAFETY ANALYSIS REPORT, TECHNICAL SPECIFICATIONS, ENVIRONMENTAL CONSIDERATIONS, AND OPERATOR REQUALIFICATION PROGRAM

REDACTED VERSION*

SECURITY-RELATED INFORMATIN REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUT OR DENOTED BY BRACKETS

APPENDIX G

SAFETY ANALYSIS REPORT

LICENSE RENEWAL APPLICATION
OF
FACILITY LICENSE R-23
DOCKET# 50-59
AGN201M (serial#106)

TEXAS A&M UNIVERSITY

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Emergency Procedure EA-4, Severe Natural Phenomena, has also been written for this facility in the case of such events and ensures the reactor is in a safe condition if the likelihood of such events is present or has already occurred..

The scram system for the AGN-201 is of a fail-safe design. Loss of power to the electromagnets will allow for the reactor to scram and be placed in a safe condition. The design basis for the scram system is to remove enough positive reactivity from the core to lesson the severity of any accident. The design basis for the control and safety systems for the AGN-201M reactor are presented in section 3.2 of the Technical Specifications.

Chapter 1: THE FACILITY

1.1 Introduction

This report is intended to be an a part of the application to the United States Nuclear Regulatory Commission (USNRC) by Texas A&M University (TAMU) at College Station, Texas for the renewal of the class 104, Facility License R-23, Docket # 50-59 for the University's AGN-201M research reactor. The reactor is primarily used for the teaching of classes in the Department of Nuclear Engineering and for the training of reactor operators. The University has operated this reactor public for a period of almost forty years and in its present location, the Zachry Engineering Center room [redacted] for nearly 25 years.

This application contains information on the local communities of Bryan and College Station are as well as the reactor facility located inside the Zachry Engineering Center. The local area information includes the meteorology, geology, seismology as well as other pertinent data required by NUREG-1537 Part 1.

This reactor has been rated for a maximum power level of 5 watts thermal since 1972. The AGN-201M reactor design has several inherent safety features, the most significant of these are the negative temperature coefficient. and the design limitation of a maximum of 0.65% excess reactivity above delayed critical. These designed safety features along with the core thermal fuse and the trips and interlocks associated with the nuclear instrumentation safety systems provide for protection of the reactor fuel even under the worst case accident conditions. If a sudden insertion of positive reactivity were to occur the reactor power rise would be attenuated by the negative temperature coefficient and then terminated by the separation of the reactor core due to the melting of the core thermal fuse. Other important reactor design considerations include, the type of fuel used, with its low enrichment, the core can, as well as the sealed fueled control rods. These features provide a complete barrier between the reactor fuel and the environment. While the core tank allows for thermal dispensation the heat generated due to the fissioning of the fuel, because of this no outside secondary cooling system is necessary. A feature of the TAMU reactor room is the thickness of the walls and ceiling, these components which act as a boundary to the reactor facility, are constructed of concrete. The walls are 3.5 feet thick and the ceiling is 4.0 feet thick. The AGN reactor design has proven to be a safe and effective training device for thousands of future engineers at this as well as other universities. The system has an established data and corroborated data base, so no research and development activities were required to evaluate the system.

1.2 Summary and Conclusions on Principal Safety Considerations

The safety criterion associated with the AGN-201M reactor at Texas A&M University is to limit as much as possible any adverse effect that reactor operations many have on the general public and students within the reactor facility.

It is believed that no adverse consequences have resulted from many years of reactor operations in this current facility. Radiation surveys conducted at 5 watts indicate very low doses outside the reactor room and none outside the areas controlled by the Department of Nuclear Engineering.

A complete description of the AGN-201M reactor is presented in Charter 4 of this report.

All the design bases for the AGN-201M reactor system and components are presented in the Technical Specifications for this AGN-201M reactor and are included in this report

The reactor facility in the Zachry Engineering Center is constructed in such a way that the reactor room and accelerator laboratory are in a cube. This cube has 3.5 foot thick reinforced concrete walls and 4 foot thick ceilings. This area of the building is isolated from the general public of as well as most personnel outside of the nuclear engineering students that have laboratories or offices in the area. Because of this design and the design of the reactor and its components, the reactor will remain in a safe, isolated condition even under the most severe hypothetical accident.

This reactor facility has been in its present location since the early 1970's and a renewal license was issued by the USNRC in August 1977, for this reactor in its present location. Some small changes have been made to the reactor facility in this time period.

1.3 General Description of the Facility

The AGN-201M reactor facility is located on the main campus of Texas A&M University in the Zachry Engineering Center. The location of the center on campus is presented in Figure 1.3-1. The facility is located in areas under the direct control of the Department of Nuclear Engineering on the 1st and ground floors of the center with the reactor room below unique the ground floor in a specially constructed reactor room. The floor plans for the facility are presented in Figures 1.3-2 and 1.3-3.

The location of the facility in the center allows for complete isolation of the reactor room , room 61B from the rest of the engineering center. The AGN-201M reactor is designed to operated in populated areas without undo risk to the general. This facility

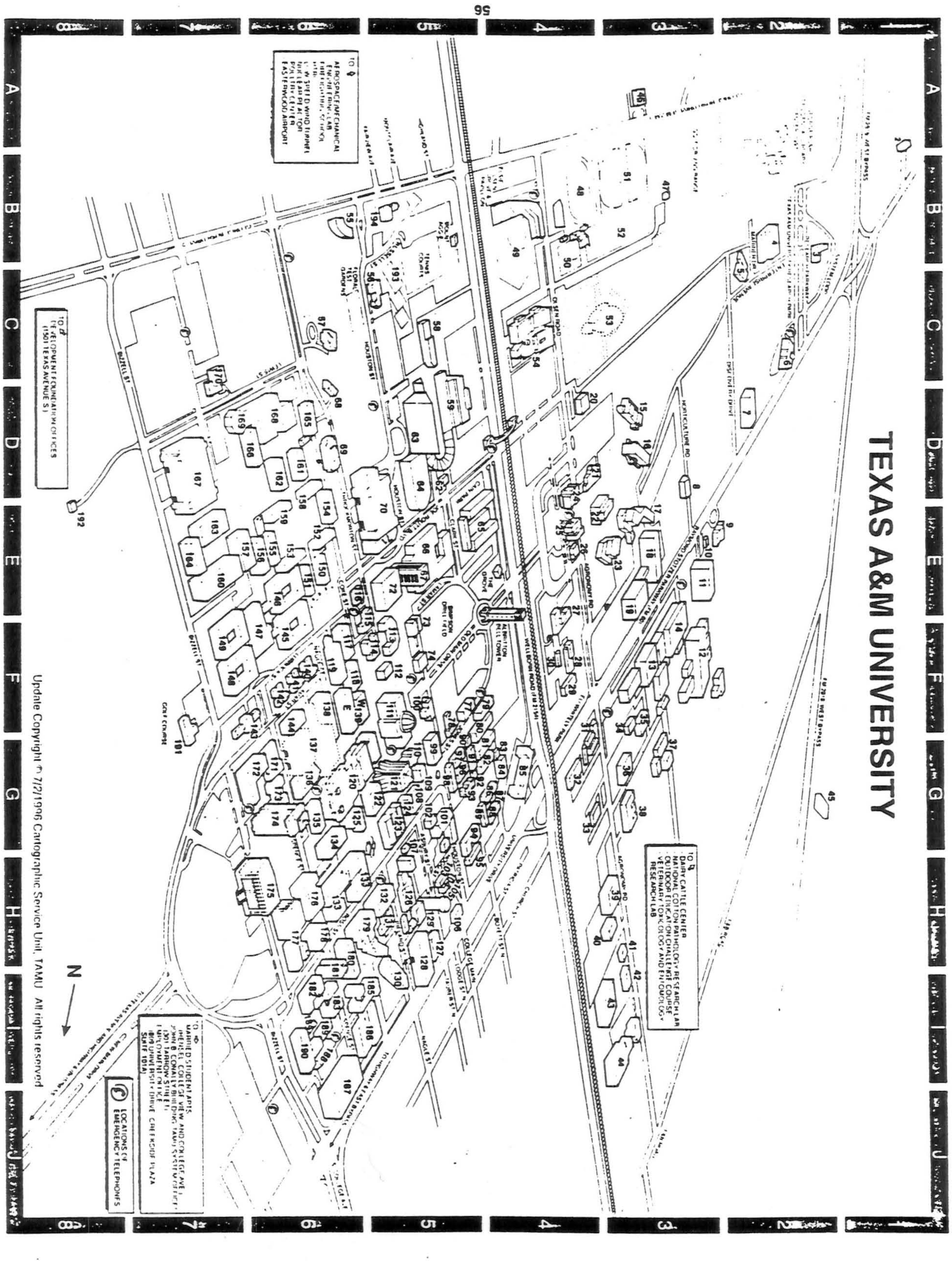


Figure 1.3-1 CAMPUS MAP OF TEXAS A&M UNIVERSITY

SITE BOUNDARY

EMERGENCY SHOWER

[redacted]

Figure 1.3-2 [redacted]

[redacted]

Figure 1.3-[redacted]OR

produces no effluents to the environment and is located in a room with thick reinforced concrete walls and below ground. The AGN-201M reactor system has been in service for many years and the University has had an operating license since 1957.

Specific site characteristics of the facility are addressed in Chapter 2 of this report. The site is in an area with a very low threat of earthquakes and flooding and the facility is designed to be fail safe. The reactor system has many safety features associated with it, some are passive systems while others are designed into the reactor protective systems. The reactor systems and safety features are described in greater detail in other chapters of this report.

A unique feature of this reactor facility is the design of the reactor room itself. The control console is shielded by an L-shaped walls to lower the dose to the operators and students and the walls of the room are 3.5 foot thick reinforced concrete and the ceiling is 4 foot thick reinforced concrete. This greatly enhances the ability of the operating staff to operate reactor with confidence that no member of the general public is being negatively affected.

1.4 Shared Facilities and Equipment

The only facilities that this reactor facility shares with the surrounding building is the electrical power supply and air supply. The reactor room has a ventilation unit inside and the discharge from the reactor room passes through a grate in the ceiling and into an exit filter unit located inside the accelerator room. A separate fan on the roof provides suction on this filter unit and air is drawn out of the reactor room and discharged on the roof. Power to the AGN-201M reactor console is supplied from the buildings power supply, but power can be remotely isolated by operation of two large breakers outside of room 60C. Inside the reactor room is a large graphite pile and a subcritical assembly

1.5 Comparisons to Similar Facilities

Two other AGN- 201M research reactors are currently licensed by the USNRC, these are both university reactors and have a similar training and teaching role as does the AGN-201M at Texas A&M University. But the roles are somewhat different, since TAMU has a much larger TRIGA reactor facility on the campus, this reactor is used for irradiation of materials for medical and commercial use while the AGN-201M is used as a teaching and training device only.

1.6 Summary of Operations

The AGN-201M operations have been limited in the past few years with the downturn in enrollment and more widespread use of the TRIGA reactor. The AGN-201M continues to be used for student reactor startups and for some limited experiments in two different nuclear engineering classes. The reactor has run for maintenance and operator proficiency more than anything else in the last four years, but it continues to be an important learning device for the students. With the presence of two reactors at one university the students get the unique opportunity to work with and operate both. The reactor should have critical run hours of 25 -35 hours per year at an average power level of 0.75 - 1.0 watts. As part of the ALARA program, the reactor operated at power levels which are as low as possible to achieve the desired results in order to keep doses to the student low..

In the future the AGN-201M will be operated a schedule similar to its current usage. The reactor is used in the first four weeks of both the fall and spring semesters for both graduate and undergraduate reactor laboratory classes.

1.7 Compliance With the Nuclear Waste Policy Act of 1982

Texas A&M University is in compliance with the Waste Policy Act. The University has entered into an agreement with Idaho National Engineering Laboratory for the final, no-cost disposal of the AGN-201M reactor fuel.

1.8 Facility Modifications and History

This AGN-201 reactor has been licensed by Texas A&M University since 1957. The initial maximum rated power was 100 mW but this was changed in 1973 to the reactors present 5 watt thermal power limit. The current reactor room was constructed in 1970 under construction permit CPRR-112 and the reactor moved to this room in 1972.

In September of 1984, the reactor control console was moved from room [redacted] to its present location in the reactor room, [redacted] behind the installed shielding. This was done so that the operator could be in the same room as the reactor during experiments. At the same time another modification was made to add a series of concrete blocks around the base of the reactor to lower general area radiation levels. Current radiation surveys indicate that the reactor produces very low levels of radiation outside the reactor room and in the accelerator room. The AGN-201M is used at Texas A&M University to train students about reactor operations and to allow the students some actual hands on reactor experience.

Chapter 2: <u>SITE CHARACTERISTICS</u>

2.1 Geography and Demography

2.1.1.1 Specification and Location

The AGN-201M reactor facility is located at the main campus of Texas A&M University, in College Station, Texas in the county of Brazos. [redacted]

A map, Figures 2.1.1.1-1A&B, is provided with the both Bryan and College Station clearly shown and the pin marker on the location of the Zachry Engineering Center. The next Figure 2.1.1.1-2 is a map of the main campus of the University with the Zachry Engineering Building clearly marked.

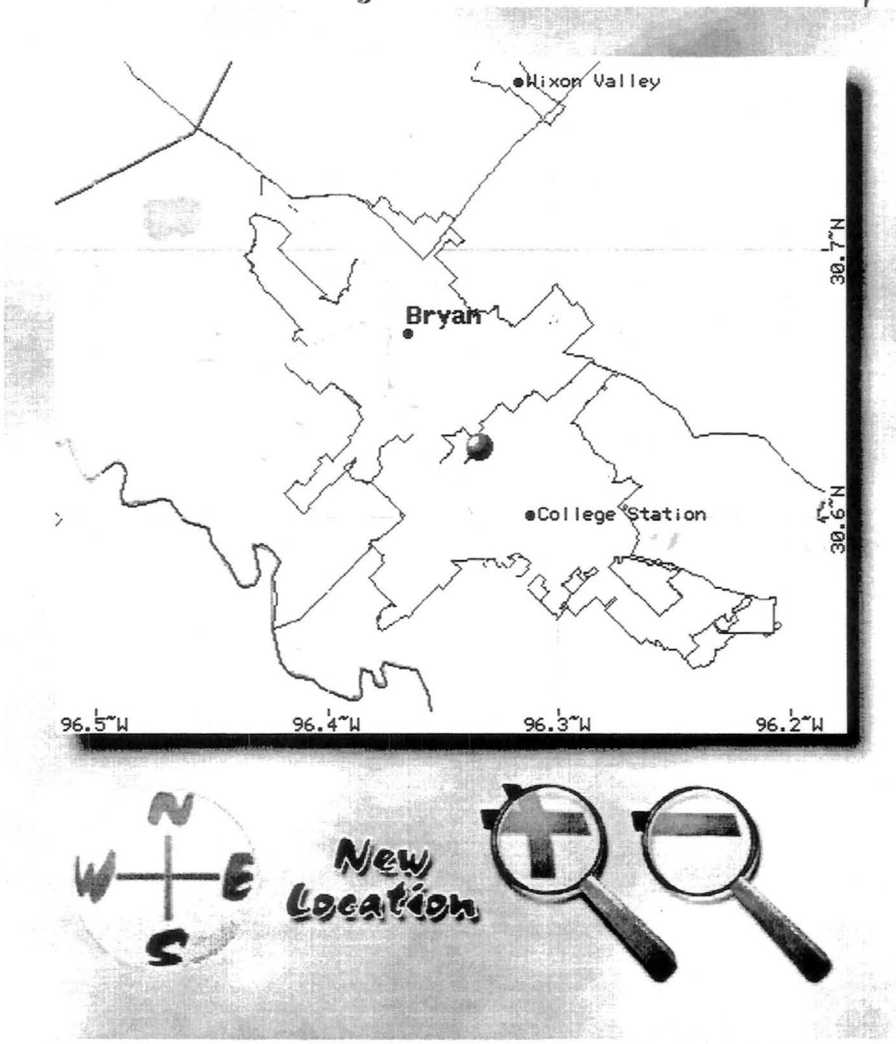
2.1.1.2 Boundary and Zone Area Maps

The reactor room and accelerator room areas are easily seen on Figures 2.1.1.2-1 and 2.1.1.2-2, the dark outer line indicates the boundary of the facility. The campus of Texas A&M University is bordered by Texas Highways 6 and 30. No emergency planing zones are needed for this facility due to its size and design.

2.1.2 Population Distribution

It can be seen from the population density on map 2.1.1.1-1A&B that the population density remains constant until you leave the towns of Bryan or College Station. and it falls off to near zero. The 1990 census recorded the population of the Bryan and College Station area at 107, 500 residents, many of which are associated with the University. Since the reactor is located on the main campus, up to 55,000 students and employees could be on campus during reactor operations. But since the reactor has

Figure 2.1.1.1-1A Map of Local Area



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Zoom in, factor:
Zoom out, factor:
Move to new center
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OFF/ON Layers	**OFF/ON Layers**
City labels	Interstate labels
Grid (lat/lon)	St Hwy labels
Cens bg points	State Bounds
Cens bg bounds	US Hwy labels
Congress dist	Water bodies
Counties	Zipcode points
Indian Resv	
Highways	
Parks and Other	
MSA/CMSA	
Cities/Towns	
Railroad	
Shoreline	
Streets	
Census Tracts	

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If your browser doesn't support client-side imagemaps, use the controls below to navigate the map.

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Zoom In	W	Pan	E	Zoom Out
	SW	S	SE	

Here is the FAQ and instructions on how to include these maps in your own web documents.
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Figure 2.1.1.1 - 1B Legend and Data for Local Area Map

LEGEND

County
City/Town
Pop Density (p/sq mi)
0.0
0.1 - 8268.0
8268.1 - 16536.0
16536.1 - 24804.0
24804.1 - 33072.0
33072.1 - 44624.6

Census Tract
Lake/Pond/Ocean
Expressway
Highway
Stream
County

Scale 1:228583 0 2 4 6 8 mi / 0 2 4 6 8 10 km
*average--true scale depends on monitor resolution

Click on the legend to download it as a GIF file.

Place a marker on this map:
Latitude(deg):
Longitude(deg):
Symbol:
Label:
Marker URL:

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Map Census Statistics:
Level:
Theme:
Classify Method:
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Enter precise coordinates:
Latitude(deg):
Longitude(deg):
Map Width(deg):
Map Height(deg):

Choose a color palette:

- **You can also search for a U.S. city or town:**

Name: State(optional):
or for a Zip Code:

- Or choose from the following preset values:

Washington, D.C. (default), The Mall, United States, Northeast U.S., New York City.

If you have feedback, please check out the Mapsurfer Feedback page.
If you have questions, please check out the service FAQ page.

TIGER Map Service Home Page.

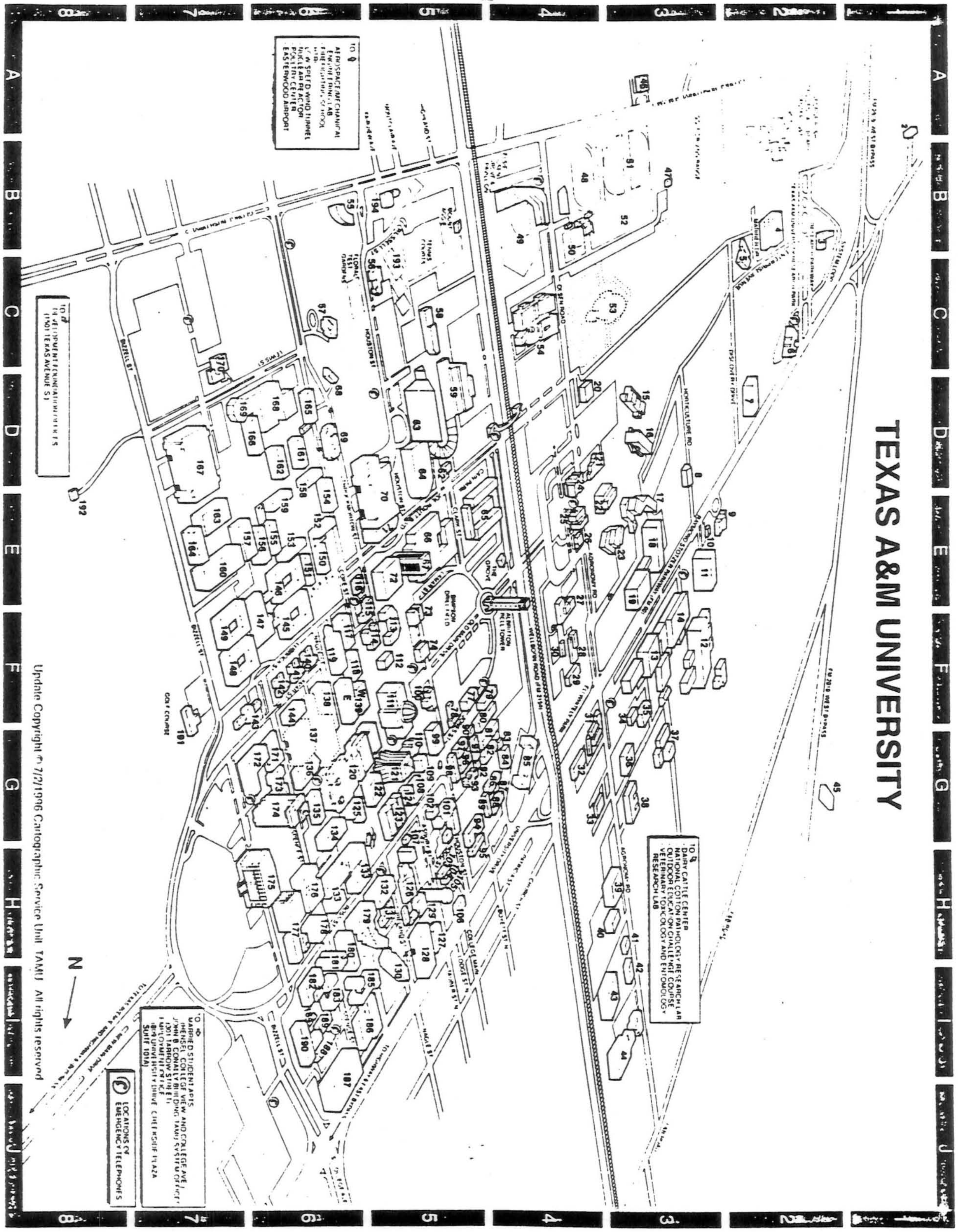


Figure 2.1.1.1-2 CAMPUS MAP OF TEXAS A&M UNIVERSITY

[redacted]

Figure 2.1.1.2-1 [redacted]

[redacted]

Figure 2.1.1.[redacted]

no effluents and has been proven safe under the most serve conditions this does not pose a safety hazard.

2.2 Nearby Industrial, Transportation and Military Facilities

Any potential accident associated with the AGN-201M reactor will have no effect upon the environment nor the surrounding area. No military installations are close the Bryan/College Station area, and industrial development is very limited within the 8 kilometer limit, since most of that area is still part of Texas A&M University or a residential area.

2.2.1 Locations and Routes

The only major route of concern is the railroad that cuts across campus, 1.8 kilometers southwest from the facility. Texas Highway 30 is located 0.6 kilometers due north of the facility. But in both cases the facility is located on the far side of the most direct point from the routes.

2.2.2 Air Traffic

The reactor facility is located 6.5 kilometers from a small commercial airport. The flight path for this airport is some 1.0 kilometers to the northwest of the facility. Since the facility is located on the first floor and the reactor room is in a steel reinforced concrete room the danger posed by passing planes is very small.

2.2.3 Analysis of Potential Accidents at Facilities.

No heavy industry nor chemical plants are located within 20 kilometers of the reactor facility. The worst possible accident deal with the trains that use the railroad tracks through the middle of campus. If an accident occurred that would require an evacuation of the facility and the surrounding building it is believed that the reactor facility would be in no real danger. If a large explosion occurred the shockwave would tear apart many other structure before getting to the Zachry Engineering Center and since the reactor room and accelerator room are located inside a steel reinforced cube the danger to the facility is very low. Even in the event of a highway accident the reactor is well protected from potential harm.

2.3 Meteorology

The Bryan-College Station area is located approximately 100 miles inland from the Texas Gulf Coast. The local weather is determined to a great extent by the high pressure areas which are predominant over the Gulf of Mexico. As a direct result of this condition, warm southeasterly winds occur a large majority of the time on an annual basis. It should be noted that in the winter months the prevailing winds come out of the north. The average wind speed is 7.34 knots. Hurricanes are very infrequent in this area do to the distance from the Texas Gulf Coast, in the last 75 years only two hurricanes have caused any damage in this area and that was due to the thunderstorms and gusty winds associated with the remnants of the system.

Tornadoes are fairly common in Texas, not is in this particular part of the state but not unheard here. Data collected on tornado frequency between 1950 and 1976 indicates that 17 tornadoes were reported within a 25 nautical mile radius of College Station.

2.4 Hydrology

The nearest large independent free flowing water the Brazos river located about 10 kilometers from the facility. Several small lakes and streams are located within 3 kilometers but pose no hazard to the facility. The facility is located 314 feet above seal level and not in any flood plains. The highest recorded crest of the Brazos river was recorded at 54 feet above flood stage or 246 feet above sea level. The nearest aquifer is the Bryan Sandstone and it is located under other geological at a depth beyond concern for his study.

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

The facility is located in a geological region known as the Gulf Coastal Plain. This region of Texas has only one fault zone associated with it and the fault lies of the western boundary of the coastal directly plain.

2.5.2 Site Geology

The local geology is dominant by a formation known as the Easterwood Shale. The thickness of this formation varies from 10-300 feet. This formation is located throughout the local area.

2.5.3 Seismicity

The State of Texas lies in a geological area that has minor seismic activity. Most of the activity that exist is limited to extreme West Texas. This region lies over 600 miles west from the local area but is the nearest active belt along the west coast of Mexico and the United States. No recorded earthquakes have ever occurred in the local area, the nearest fault zone is the known as the Balcones Escarpment and is located on the western boundary of the Gulf Coast Plain, 100 miles to the west of this facility.

2.5.4 Maximum Earthquake Potential

The largest earthquake ever recorded in the State of Texas occurred near El Paso and measured 6.4 on the Richter scale. No seismic activity has ever been recorded in the local area and the possibility of any activity occurring is very low.

2.5.5 Vibrotory Ground Motion

The area surrounding the facility has no geological activity associated with it. The area has been stable for as long as people have recorded such things and no threat of earthquakes or other seismic actives is concerned a possibility in this local region of Texas.

2.5.6 Surface Faulting

The local area is considered to have a very low potential for surface faulting. No earthquakes have ever been reported in the local area.

2.5.7 Liquefaction

The local foundation material have no potential for liquefaction. The facility is constructed in such a manner that this is not considered to be a potential problem.

Chapter 3: DESIGN OF STRUCTURES, SYSTEMS AND COMPONENTS.

3.1 Design Criteria

The AGN-201M reactor control and instrumentation system has two basic design criteria, the first is the ability to safely control the reactor under all normal and foreseeable accident possibilities and the second is the "fail-safe" design mode.

The fail-safe design mode is a concept that the reactor will be placed in a safe shutdown condition if any equipment or power failure were to occur. The nuclear instrumentation system is constructed with a series of trips and interlock systems that will accomplish this design criterion. A complete description of these trips and interlocks is presented in Chapter 7 of this SAR. The most critical passive design feature in this "fail-safe" mode, this mode is the present in the form of large springs on each of the safety rods and the coarse control rod. These springs ensure that the rods will be forcible removed from the core if power is interrupted to the scram magnets no matter the cause of this power interruption. The Technical Specification limit is less than 200 msecs , for the safety system rods to be removed from the core, this limit ensures that the reactor will be safety shutdown even under serve transient conditions.

The reactor control and instrumentation system is designed to monitor reactor power all ranges for the AGN-201M reactor system .

The construction of the reactor room and adjoining room provides for a safe isolated environment in which to conduct reactor operations. The facility is designed to have very limited interaction with the building systems. The reactor control console power and power to the local ventilation unit and exhaust fan come from the in building power system. No systems are associated with the reactor facility.

3.2 Meteorological Damage

The location of Texas A&M University provides some threat to severe weather conditions. Rare tornadoes have occurred in the Brazos county and have been within 10 miles of the facility they have done little damage to the University. Only the remnants of one hurricane has been detected in the county over the last 50 years and this lead to some damage in the local community.

The reactor facility in the Zachry Engineering Center is designed to be a large cube inside the northwest corner of the building. This cube extends from the first floor down to the reactor room and laboratory adjacent to it. The walls of this cube are 3.5 feet of reinforced concrete and the fail ceilings are four feet of reinforced concrete, the outer wall is a direct component of the building foundation. This system will stand up to any storms that may pass into the area.

In the event a tornado is sighted within a 5 mile radius of TAMU, the radio operator at the TAMU Communications Center will notify the first available person listed on the AGN-201M Emergency Response phone numbers. The communications center receives notification of tornadoes from both the TAMU weather radar and the Brazos County, Bryan-College Station Disaster Emergency Planning Organization. Tornadoes are detected by the use of TAMU radar and Doppler radar at a local television station.

3.3 Water Damage

The design of the reactor facility and the Zachry Engineering Center makes the likelihood of water damage very remote. The only route that floodwater could enter the area is through two sets of double doors. In the event of weather severe enough to warrant concern ,these doors have been sandbagged to eliminate any possibility of water entering the facility. A large drain is located in the parking lot adjacent to these double doors to drain off the water.

If water entered the facility it would have to rise up to the level of the double leading to room 60C. Then flood the floor in this room in order to reach the double doors leading to the laboratory outside the reactor room. If water were to enter the reactor room a drain system is set up so that the water would drain into a 1000 gallon holding tank located in the basement floor of the facility. This sump would have to be completely filled in order to raise the water level in the reactor room.

3.4 Seismic Damage

The possibility of seismic activity is extremely low in this region of Texas. This combined with the fact that the walls of this facility are constructed on 3.5 feet of steel reinforced concrete in the form of a cube give the facility the strength to resist any seismic activity in the area.

3.5 Systems and Components

The fail-safe AGN-201M control and instrumentation system and the earthquake switch are designed to place the reactor in a shutdown condition in the event of a natural disaster. The loss of electrical power would also scram the reactor, since this is a likely byproduct of such an event this also ensure the safe condition of the reactor. An

Chapter 4 REACTOR DESCRIPTION

4.1 Summary Description

The AGN-201M research reactor in operation at Texas A&M University is essentially identical to all the other AGN-201M reactors that have been operated since 1957. Therefore, the reactor description given in the original Aerojet-General Nucleonics AGN-201 reactor report (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 the other AGN-201M reactors on file with the USNRC.

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

The rated thermal power level for this AGN-201M reactor is 5 watts. This power level was approved by the USNRC in Amendment No. 10 to License R-23 issued on January 18,1973.

The critical mass of the reactor is approximately 665 grams of ^{235}U. The fuel itself is in a plate type configuration and is homogeneously mixed powder of polyethylene and UO_2 (61 mg $^{235}U/cm^3$) in the form of 20 micron diameter particles. The core disc have been formed by pressing under high pressure the homogeneously mixed powder.

The reactor is a free standing unit. the reactor core is surrounded by a graphite reflector, a lead shield and a large water shield tank for fast neutrons.

The reactor has a natural convection cooling system through the graphite reflector and the lead shield to the 1000 gallon tank of water the surrounds the reactor core. this removes any thermal heat due to the fission of the fuel.

The reactor has a hermetically core tank surrounding it , this tank is designed to contain fission products gases if any core damage were to occur.

The experimental facilities associated with this AGN-201M are the small 15/16 inch diameter glory hole that runs completely through the reactor and the 4 -inch diameter access ports that pass through the water shield and graphite moderator and lead gamma shield. A side view of the AGN-201M reactor is presented in Figure 4.1-1.

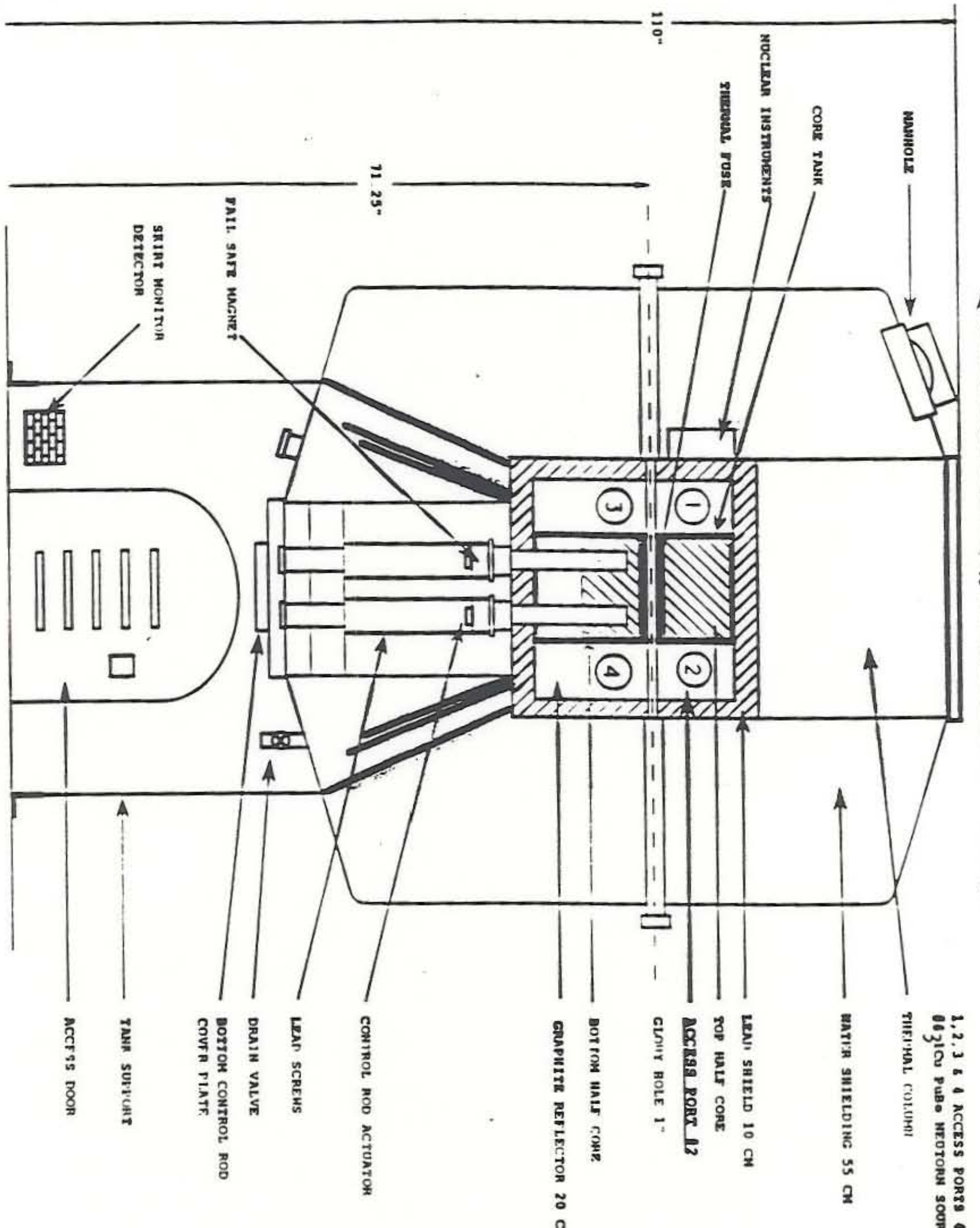
4.2 Reactor Core

4.2.1 Reactor Fuel

The reactor core of the AGN-201M reactor consist of nine separate fuel discs which are stacked into a right circular cylinder as shown in figure 4.2.1-1. This figure clearly depicts the physical relationship between the control rods, the core and which the core tank. The final core configuration is 10.15 inches (25.8 cm) in diameter and 9.34 inches (23.75 cm) in height. The fuel in the core is made up of approximately 665 grams of ^{235}U in 3330 grams of U for an enrichment of just less than 20%. Figure 4.2.1-1 shows the relationship between the core and the rods. The uranium is homogeneously mixed with polyethylene to form a series of fuel plates. A physical description of these fuel plates is included in the reactor characteristics table 4.2-1

The AGN-201M reactor core also contains four fueled control and safety rods. Three of these rods, Safety Rods 1&2 and the Coarse Control Rod, are constructed in a similar manner while the Fine Control Rod is much smaller in size. The fuel contained in these rods is exactly of the same type as the rest of the core. The reactor fuel has some limitations associated with it, in particular the breakdown of the fuel at high temperature. Several design features of the AGN-201M will prevent this temperature (200 °C) from

Figure 4.1-1 SIDE VIEW OF AGN-201M REACTOR



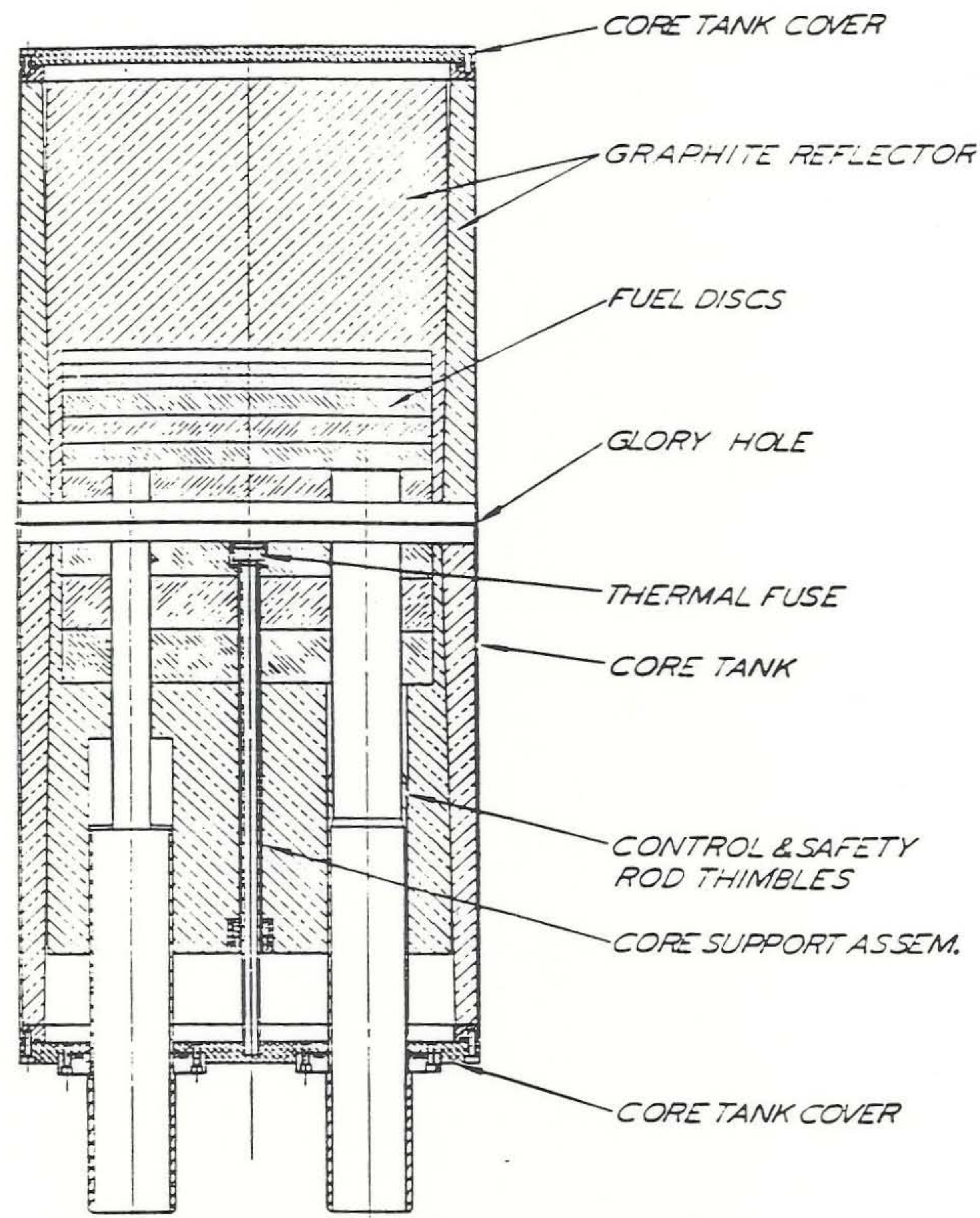


Figure 4.2.1-1 CORE TANK AND CORE

being reached and if some core damage were to occur the gases and other materials would be trapped inside the core tank.

4.2.2 Control Rods

The AGN-201M design contains four fueled control and safety rods. Three of these rods, the two safety rods and the coarse control rod, are identical in design but differ in their function. Each of these rods contains about 15 grams of ^{235}U and operate in such a manner that as the rod is driven into the core the reactivity in the core increases. The exact amount of fuel contained in each of the rods is listed on table 4.2-1. Figure 4.2.2-1 shows the design of the rod which fits into a 2 inch (5 cm) hole in the lower disc in the reactor core.

The coarse control rod and the two safety rods are designed to contain 1.25% of reactivity each, while the smaller fine control rod has a reactivity worth of 0.31%. The fine control rods is design to allow the reactor operator to make small adjustments in reactor power. The active length of these rods is 5.9 inches (15 cm) and consists of UO_2 embedded in radiation-stabilized polyethylene identical to that of the fuel discs in the reactor core. The active fuel material is enclosed in two aluminum containers, the outer containers provide fluid seals between the rods and their drive mechanisms and the shield tank. The innermost aluminum container are used to seal the active fuel material contained in the fuel slugs. These containers provide a double seal between the fuel and the outside environment. These rods are calibrated as part of the AGN maintenance program, the reactivity worth's of these control rods must be known to safety operate the reactor.

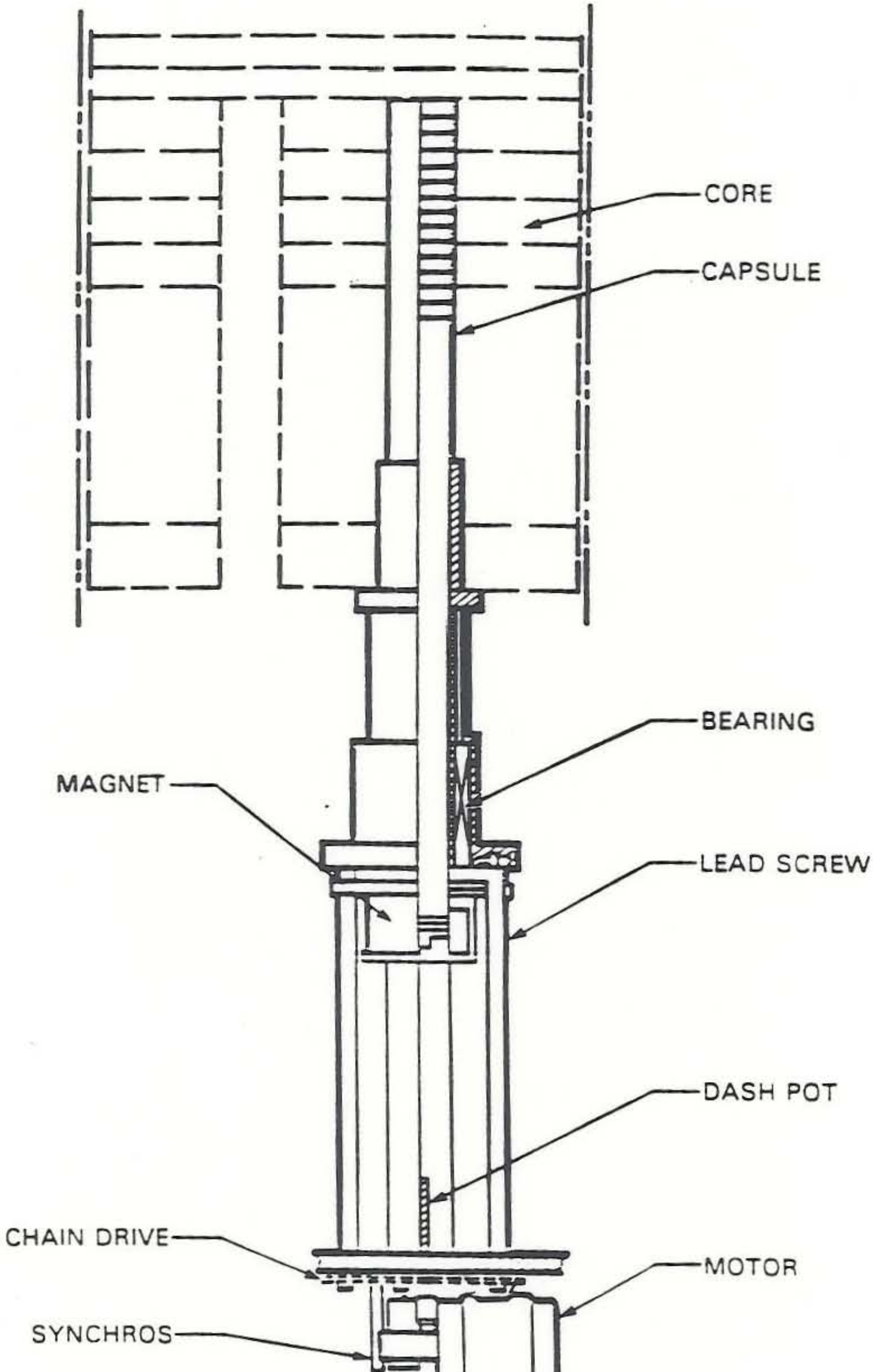


Figure 4.2.2-1 CONTROL ROD

Table 4.2-1

Reactor Characteristics

A. **General Descriptions**

Type	Homogeneous thermal reactor
Principle Uses	Education and Training
Maximum Rated Power	5 watts (thermal)
Core Design	UO2 in the form of < 20 % enriched ^{235}U particles homogeneously distributed in a solid polyethylene moderator. in nine fuel disc. in a cylindrical geometry
Reflector	Graphite
Shielding	Lead and water
Rods	Two safety and two control, all fueled

b) **Fuel**

Fuel Material	< 20%-enriched UO_2
Fuel Disc	Right circular cylinder fabricated from a compressed mixture of fuel material and polyethylene power Specifications: UO_2 ^{235}U enrichment < 20% Particle size - 15 $\pm$ 10 μm Polyethylene Powder Particle size- 100 μm Purity - commercial grade
Approximate Core Fuel Disc Size and ^{235}U content	9 separate fuel disc Four discs 1.57" (4 cm) high ([redacted] g ^{235}U each)

Three discs 0.79" (2 cm) high (58 g ^{235}U each)

Two discs 0.39" (1 cm) high (30 g ^{235}U each)

Approximate Safety and Coarse Control Rods fuel disc sizes : 4 per rod

1.85 " (4.7 cm) diameter and 1.57" (4 cm) in height

Approximate Safety and Coarse Control Rods fuel disc loading : 4 per rod

[redacted] g ^{235}U each

Approximate Fine Control fuel disc sizes : 4 per rod

0.9 " (2.3 cm) diameter and 1.57" (4 cm) in height

Total Fuel Loading	≈ [redacted] ^{235}U
^{235}U Density	[redacted] g $^{235}U/cm^3$
Core Thermal Fuse	Small, right circular cylinder. 0.87" (2.2 cm) in diameter a and 0.35" (0.9 cm) in height, enriched with 0.4 g of ^{235}U. Fabricated from a mixture of fuel materials and polystyrene powder.

c. **Reactor**

Core-containing Tank	Gas-tight, 0.026" (65 mil) aluminum cylindrical tank, 12.7" (32.2 cm) diameter and 30" (76 cm) high. The tank has a 1' glory hole on the horizontal tank centerline the appropriate control access areas
Core	9 fuel disc, separated at core midpoint by glory hole and thin aluminum baffle. Lower half of the core has appropriate access holes for control and safety rod thimbles and core thermal fuse.
Reflector	Heavy density (1.75 g/cm^3) graphite, approximately 7.87" (20 cm) on all sides of the core. Top and bottom reflector are contained in the core tank, side reflector surrounds the

	core tank and contains four 4" (10 cm) access ports running tangentially to the core tank
Gamma Shield	3.94" (10 cm) of lead shielding completely surrounding the graphite reflector
Reactor Tank	A 5/16" (80 mm) thick steel tank, 37.4" (95 cm) diameter and 58.3" (148 cm) in height tank. This tank contains core tank, reflector and lead shield and the appropriate access holes for the control and safety rods, the glory hole and access ports. This is a gas-tight vessel when all the seal are made. The upper portion of this tank contains the removable thermal column.
Water Shield Tank	Steel tank serves as the main structural tank as well as a fast neutron shield. The tank is 6.5 ' (198 cm) in diameter and 7' (213 cm) in height.
Reactor Dimensions	6.5 ' (198 cm) in diameter and 9.17' (280 cm) in height
Reactor Weight	20000 lbs. 11,400 lbs. without shield water
Reactor Control	Two Safety Rods Safety Rod #1[redacted] grams of ^{235}U Safety Rod #2 [redacted] grams of ^{235}U Coarse Control Rod [redacted] grams ^{235}U Fine Control Rod [redacted] grams of ^{235}U

All rods, with the exception of the FCR is mechanically coupled, are magnetically couple to the carriage which is driven into the reactor on a lead screw by a reversible DC motor. Total travel distance is about 25 cm

d. **Nuclear Data**

(1) Fuel Loading

(a) Approximate Critical Mass [redacted] grams ^{235}U

(b) Last measured Excess Reactivity corrected to 20 C with glory hole empty 0.137% Δk/k

(2) Neutron Flux

Average Thermal Flux	1.5×10^8 n/cm^2-sec at 5 watts
Peak Thermal Flux	2.4×10^8 n/cm^2 -sec at 5 watts

(3) Reactivity Worth of Reactor Components

(a)	Safety and Coarse Rods	≈ 1.25 % Δk/k (each)
(b)	Fine Control Rod	≈ 0.31 % Δk/k
(c)	Cadmium Shutdown Rod	2.35 % Δk/k
(d)	Polyethylene Rod in Glory Hole	
	New Poly Rod	0.29 % Δk/k
	Old Poly Rod	0.26 % Δk/k
e)	Temperature Coefficient of Reactivity	
		≈ - 0.024 % Δk/k per degree centigrade

(4) Important Figures

(a) Fine Control Rod Calibration Curve Figure 4.2.2-2

(b) Inhour Equation Figure 4.2.2-3

(c) Flux Profile Figure 4.2.2-4

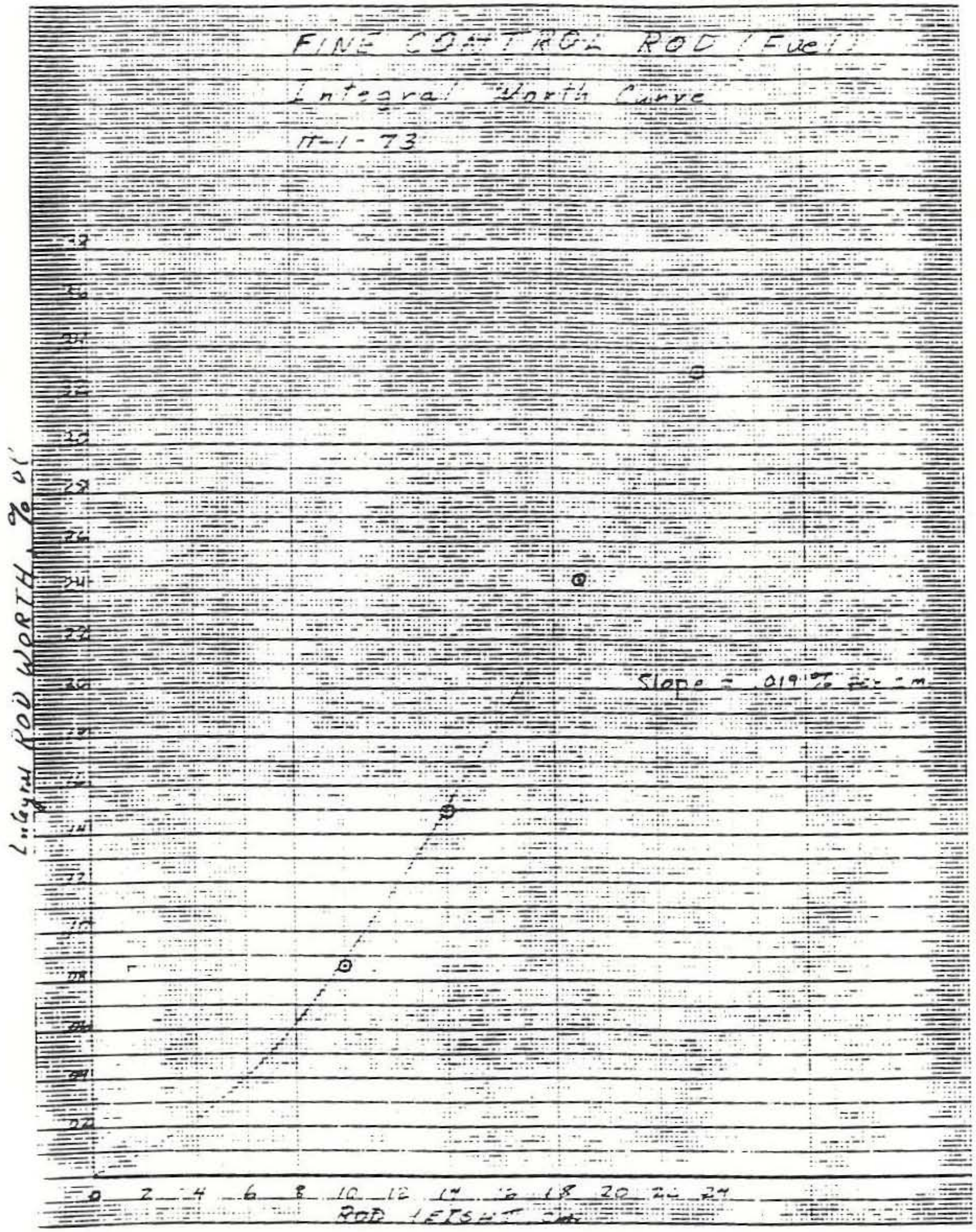


Figure 4.2.2-2 FINE CONTROL ROD CALIBRATION CURVE

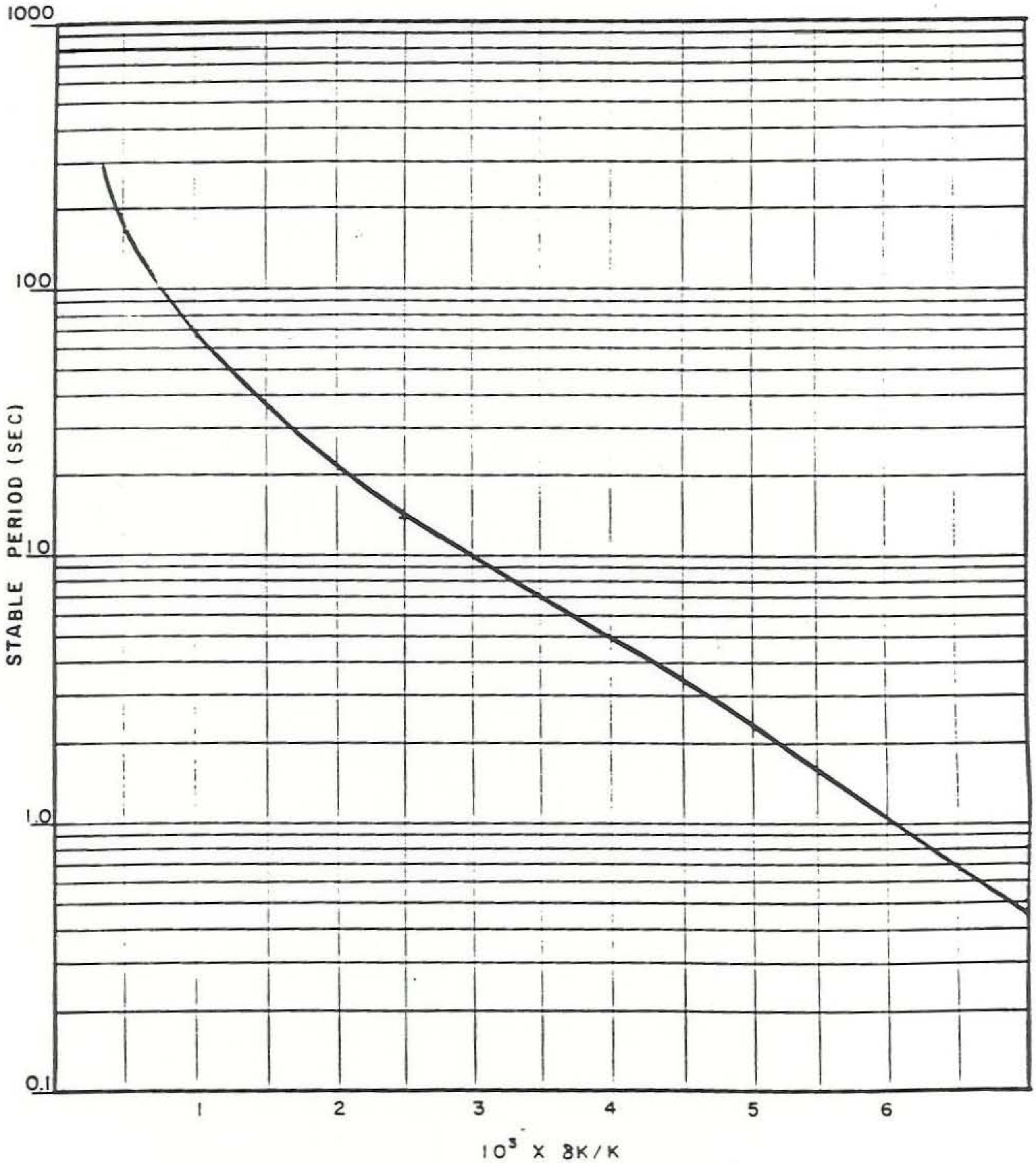


Figure 4.2.2-3 AGN-201 INHOUR EQUATION

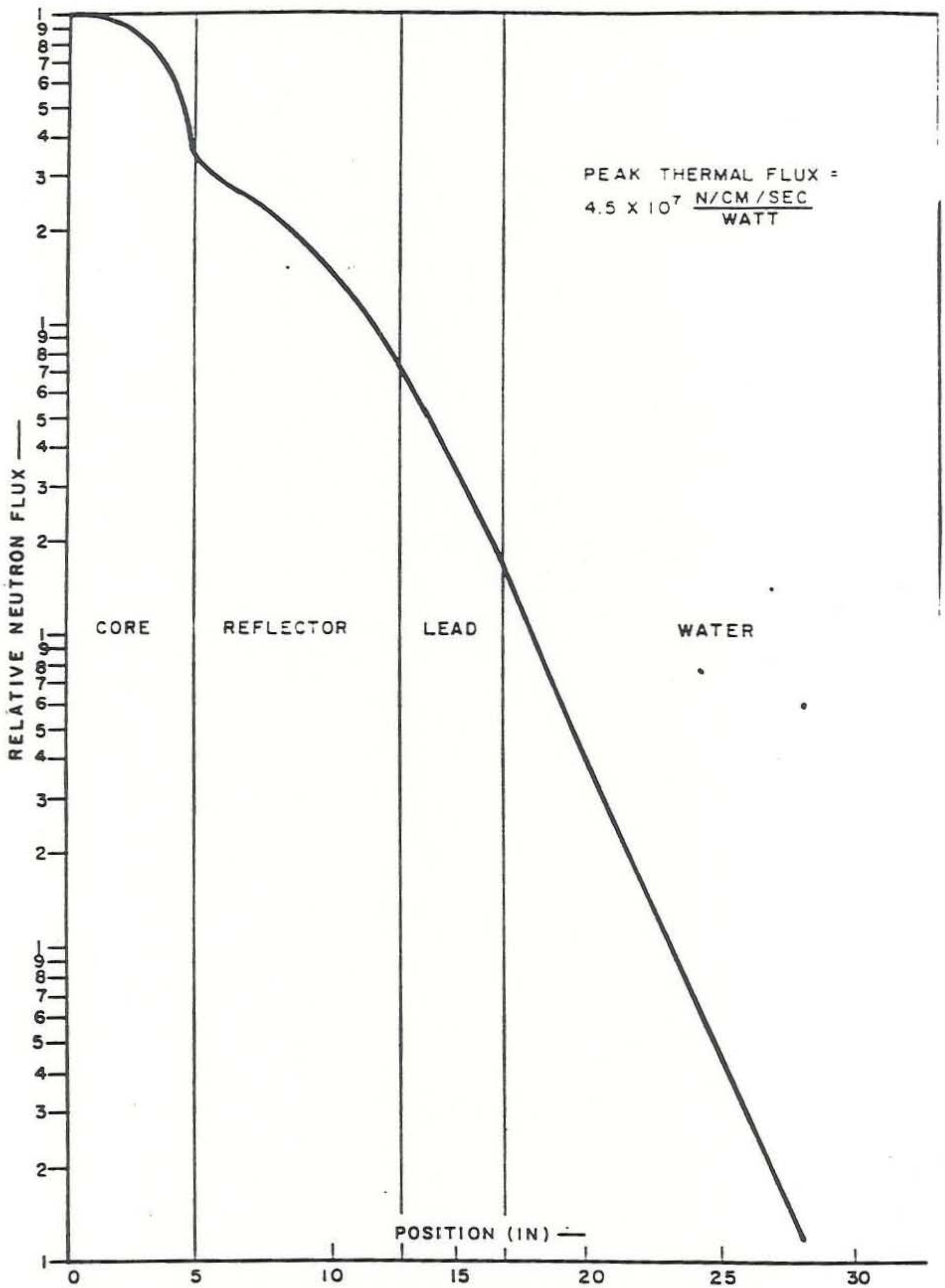


Figure 4.2.2-4 100 mW THERMAL NEUTRON FLUX PROFILE

When the reactor is operated the rods are inserted into the reactor in a specific sequence to ensure that sufficient shutdown margin is maintained at all times. This sequence is controlled by a series of limit switches associated with each of the safety rods and the coarse control rod. In order for rod drive motor power to be applied to Safety Rod #2 , Safety Rod #1 must be fully inserted in to the core, this feature is tested during the prestartup checkout in the operating procedures. A similar system is associated with Safety Rod #2, the Control Rod will not have rod drive motor power applied until Safety Rod #2 is in its fully inserted position in the reactor core.

All three of these rods are part of the reactor safety system and will be ejected from the core anytime a scram signal is received, even during rod insertion. The safety system is designed to be a "fail safe" system, this means that in the event of a scram signal the electromagnets holding the rods have their power interrupted. This will allow the rods to be accelerated out of the core downward by gravity and by the force of spring loading. The spring constant is such that these three rods are ejected from the core with a force of 5 g, with this amount of downward acceleration these rods are out of the core in $\approx$ 120 milliseconds. The safety rods insert a negative 0.7 % $\Delta k/k$ during the first 50 milliseconds following the scram signal. The rods are decelerated by dashpots during the last 10 cm of travel to avoid damage to reactor components. The Fine Control Rod due to its limited reactivity is not associated with the reactor safety system. This rod is driven out of the core at a rate of 0.5 cm sec.

Both the fine and coarse control rods are driven by reversible DC motors through a lead screw assemblies that are controlled by switches located on the control console.

The maximum speed associated with rod movement is 1.0 cm/sec, this speed will result is a maximum possible reactivity change of 2×10^{-4} % Δ k/k per second for the coarse control rod. Both the coarse and the fine control rods can move at two speeds, 0.5 cm/sec and 1.0 cm/sec, a switch for each rod is located on the control console to allow for the different control rod speeds.

The positions of all rods are indicated on the reactor control console but the safety rod position indications are limited to full out, fully inserted or somewhere in between. The fine and coarse control rods have numerical indicators of their position in respect to distance traveled.

The trips and interlocks associated with the scram and interlock systems for the control rods are summarized in the following table.

Table 4.2.2-1

Trips and Interlocks Leading to a Reactor Scram

Scram System	Interlock Relay , High Level Channel #3
	High Level Channel #2, Manual Scram
	Channel #2 Min. Period, Skirt Monitor High Level
Interlock System	Shield Water Temperature Switch, Earthquake Switch
	Shield Water Level Switch, Channel#1 Low Count Rate
	Channel #3 Low Level , Channel #2 Low Level
	Relay Chassis Interlock, Rod Drive System Plug

4.2.3 Neutron Moderator and Reflector

The neutron moderator associated with the AGN-201M is the polyethylene that fuel is part of the reactor fuel. The fuel is homogeneously mixed with 10, 900 grams of polyethylene, to form the fuel disc in the reactor core. Since the fuel and polyethylene are part of the same matrix the neutrons are able to be slowed down to a certain extent right after fission.

The reflector associated with the AGN-201M is made-up of high density graphite (1.75 g/cm^3) and is 20 cm thick. This reflector surrounds the reactor on all sides and is used to move neutrons back into the core after they scatter out of it.

4.2.4 Neutron Startup Source

A [redacted] gram plutonium-beryllium startup source is currently installed in the AGN-201M. This source was installed in access port #3 on May 1, 1972 after being approved as Amendment No.11 to License R-23 by the USNRC. The relative position of the source as it relates to the nuclear instrumentation is shown in Figure 4.2.4-1. The source is in a position in access port in the access port 11.25 in. from centerline based upon a study conducted by facility staff in 1972. It was determined at this location the source provide sufficient neutron flux to ensure the detectors are in their normal operating range (10^{-12} to 10^{-6} A).

4.3 Reactor Tank or Pool

A large 1000 gallon tank of water surrounds the AGN-201 reactor, while the thermal column at the top of the tank, itself is a separate tank. The bottom of the

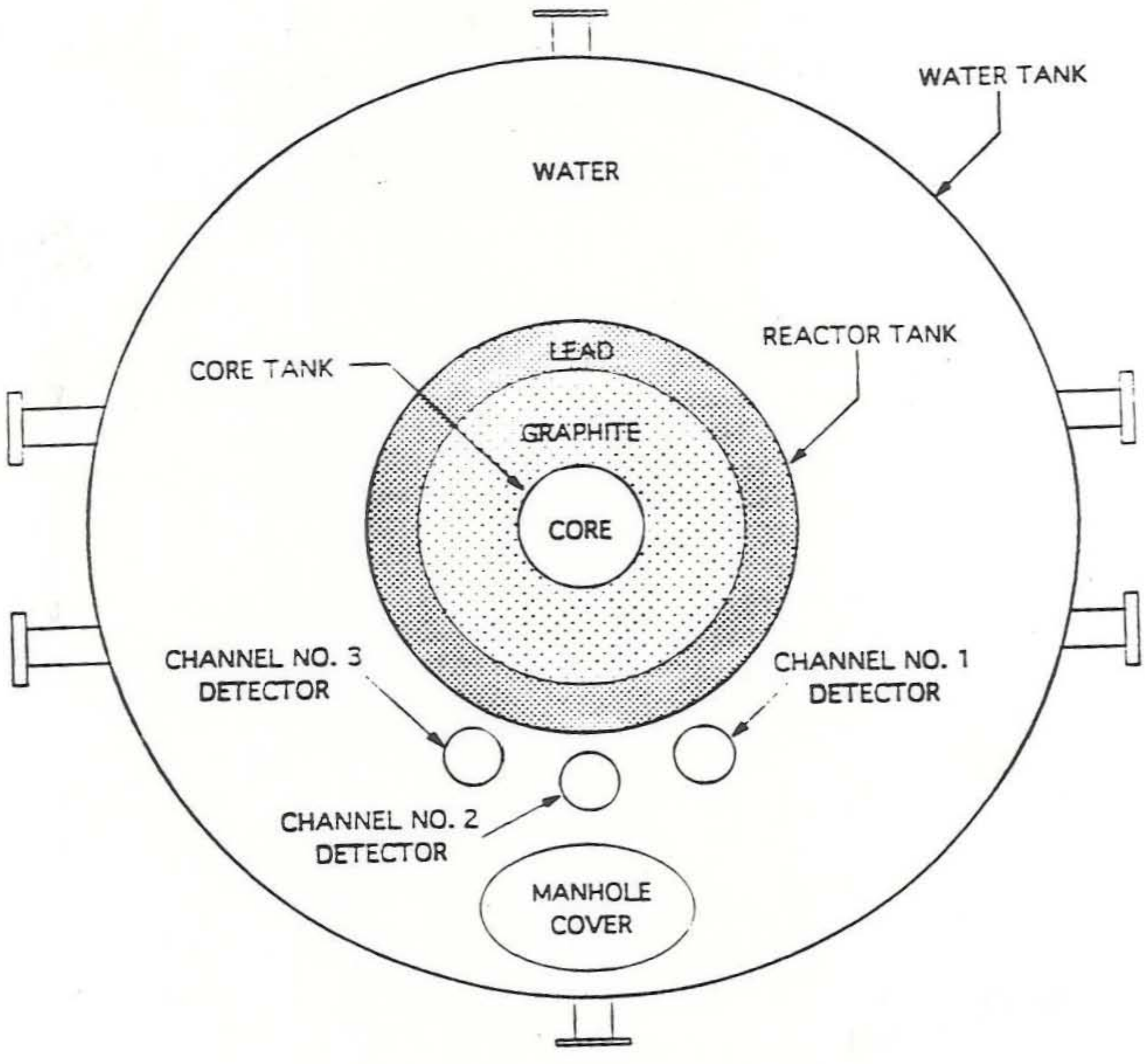


Figure 4.2.4-1 TOP VIEW OF REACTOR

AGN -201M is not shielded by this tank in order to provide access to the control drive mechanisms and the rods. The tank is filled with water and provides a fast neutron shield, scram features associated with this tank include a low-level trip and a low -temperature trip. This tank while providing fast neutron shielding is also the only cooling system associated with the AGN-201M reactor system. The tank is a solid unit and rest on the base of the AGN-201 system, the reactor core and lead shielding and graphite reflector are in the center of the tank but not directly associated with it.

4.4 Biological Shield

The biological shield is designed to limit total gamma and fast and thermal neutron dose at closest approach to the reactor is less than 100 mrem/hr at reactor power less than 1.0 watts. This dose rate should be as low as possible to allow access to the experimental areas of the reactor room and to surround the ALARA principle. The other requirement is that the dose rate due total gamma and thermal and fast neutron dose is less 15 mren/hr in the accelerator room. The accelerator room dose rate is easily achieved due to the construction of the floor on the accelerator room. A complete copy of a radiation survey has been included in the back of this section of the Safety Analysis Report. The biological shielding must also perform under emergency conditions.

4.5 Nuclear Design

The construction and description reactor core and fueled control rods for the AGN-201M have been described earlier in this chapter. The behavior of the AGN-201 reactor core has been well defined over many years of operations and its previous Safety

Analysis Reports and Hazard Safety Report from Aerojet-General Nucleonics. During normal and emergency operations no damage to the reactor core or fuel is expected due to the compact and conservative design of the reactor core as well as the presence of many passive safety features associated with this reactor system.

4.5.1 Normal Operating Conditions

During normal operating conditions the AGN-201M reactor will exhibit little fuel burnup or usage. The reactor core has only a single configuration for its control rods and operates in a very low and limited power range.

4.5.2 Reactor Core Physics Parameters

Table 4.2-1 Reactor Characteristics provides a listing of some core physics parameters. The reactivity worth's for the control and safety rods are determined annually using maintenance procedure for rod worth. The values for the polyethylene rods can be determined in the same manner. as well as the cadmium shutdown rod. This procedure has been used for many years and has proven an effective method for the determination of rod worth's.

4.5.3 Operating Limits

The limiting conditions for operations reactivity limits are designed so that the limitations on total core excess reactivity will help to 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 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. These reactivity limits must apply to the reactivity condition of the reactor and the reactivity worth's of control rods and experiments.

If these limit are followed it will ensure that the reactor can be shut down at all times and that the safety limits will not be exceeded.

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%Δ k/k.

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

The core excess reactivity limits are designed to ensure that the reactor will have reactor periods of sufficient length so as not exceed safety limits if the reactor protective system and/or operator action will be able to shutdown the reactor. These operating limits are verified during annual preventive maintenance procedure ROEX-3, Calculation of Shutdown Margin and Total Excess Reactivity.

When setting operating limits the maximum steady state power level and maximum core temperature during steady state and transient operating conditions one must always consider the integrity of the fuel matrix so that no fission products are allowed to escape the core matrix.

The polyethylene core material does not melt below a temperature of 200°C and is expected to maintain its integrity and retain essentially all of the fission products at temperatures below that point. The Hazards Summary Report dated February 1962 and 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.

The safety limits shall be set to ensure the fuel matrix maintains it integrity at these levels

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.

Now that the limits exist reactor safety system setpoints must be determined which will limit maximum power and core temperature. These shall be the Limiting Safety System Settings, at these setting an automatic protective action will be initiated to scram the reactor and prevent a safety limit from being exceeded.

Based on instrumentation response times and scram tests, the AGN Hazards Report concluded that reactor periods in excess of 30-50 milli-seconds 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 milli-seconds. 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 characteristics due to the high negative temperature coefficient, and the melting of the thermal fuse at a temperature 120°C or below will assure safe shutdown without exceeding a core temperature of 200°C.

So the setting have to be ≤ 10 watts

Channel	Condition	LSSS
Nuclear Safety #2	High Power	≤ 10 watts
Nuclear Safety #3	High Power	≤ 10 watts

The core thermal fuse shall melt when heated to a temperature of 120°C or less resulting in core separation and a reactivity loss greater than 5% Δ k.

4.6 Thermal-Hydraulic Design

The thermal-hydraulic design of the AGN-201M reactor system is limited to the transfer of heat to the shield tank. The shield tank is so large, 1000 gallons and the heat load so low from the AGN-201M reactor that the reactors effect on the tank is not noticed. But the reverse is not true, the tank must be filled to within 10 inches of the top in order to meet shielding requirements. The shield tank is the primary fast neutron shield in the AGN-201M reactor design.

Procedure for AGN Reactor Radiation Survey

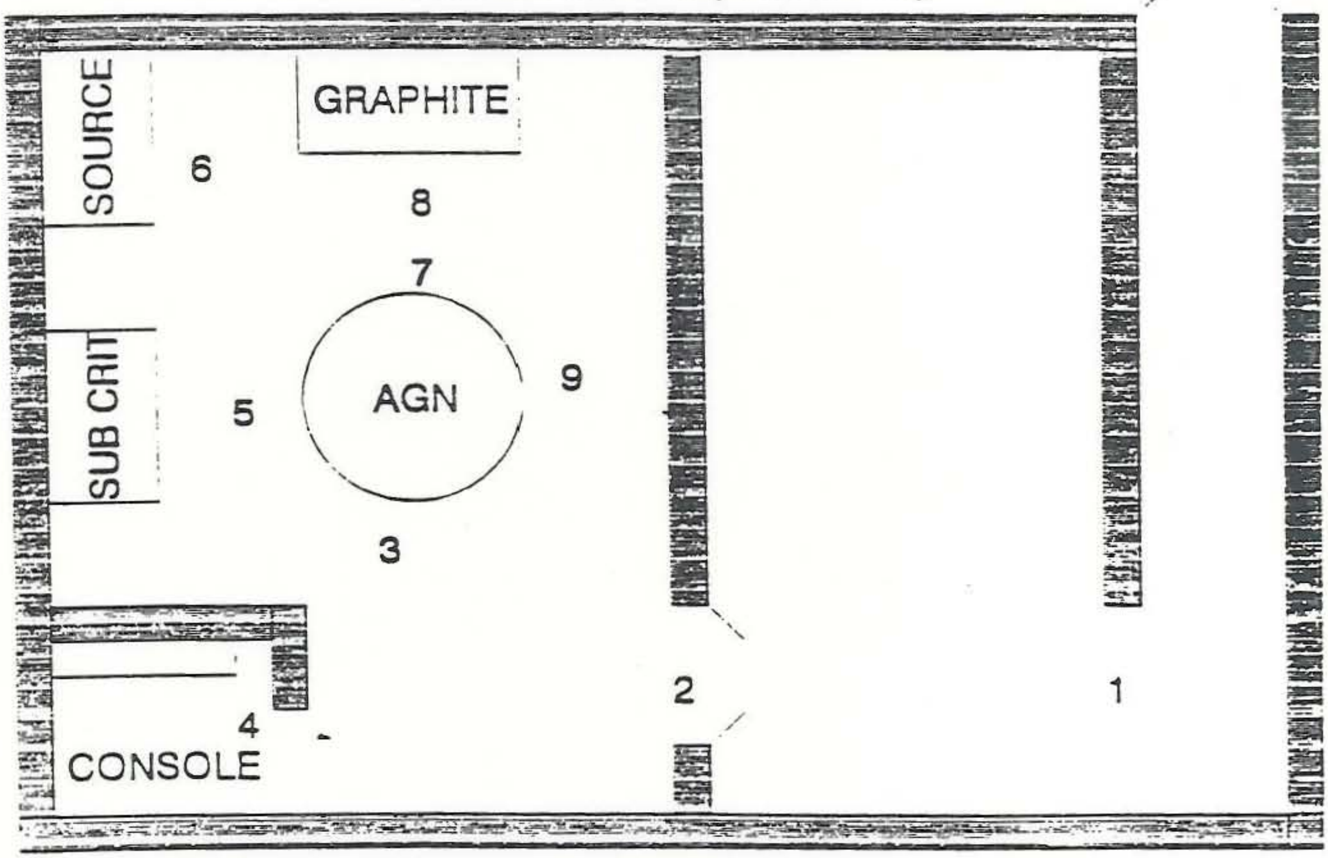
1. Fill out cover page including instruments to be used in the course of the survey.

2. Perform a background radiation survey in the AGN Room. Record the results as being performed at "0" (zero) power level. Measurements made in the AGN Room are performed with the detector placed on top of a meter stick and the bottom of the stick on the "X" on the floor.

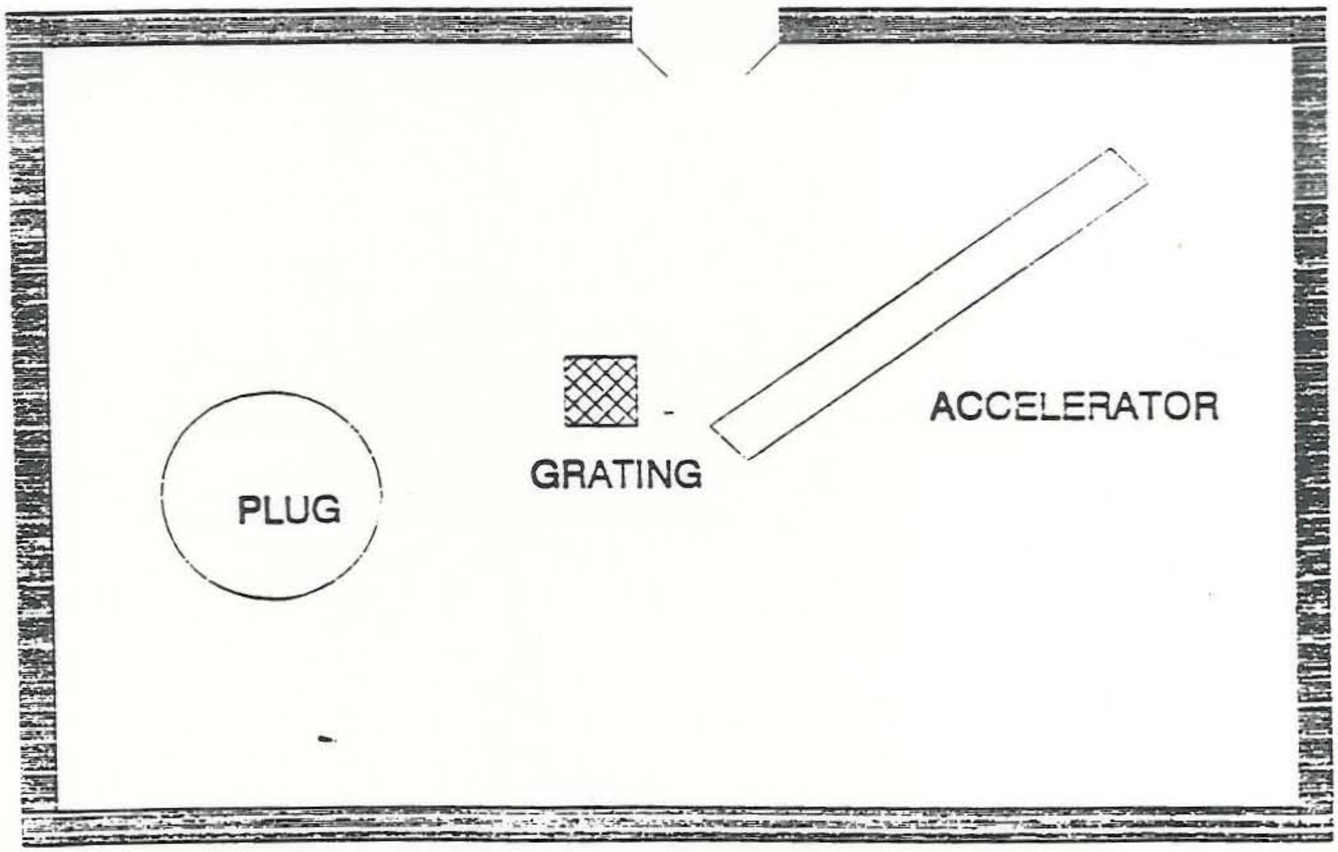
3. Perform surveys at various power levels. Although rated power is 5 watts, the reactor may not be able to achieve full power due to temperature constraints. At the highest power level achievable, survey in the Accelerator Room. These measurements are made with the detector placed on the floor.

DIM 7/6/92

AGN COMPLEX (ROOM 60L)



ACCELERATOR COMPLEX (ROOM 135)



Reactor Power Level: Ø Watts

Survey Point #	Gamma Dose mRem/Hr	Neutron Dose mRem/Hr	Total Dose mRem/Hr
AGN Room	—	—	—
1	0.12	0	0.12
2	0.15	0	0.15
3	0.15	0	0.15
4	0.13	0	0.13
5	0.35	0.6	0.95
6	3.5	6.5	10.0
7	1.0	0.4	1.4
8	3.5	0.14	3.64
9	0.3	0	0.3
skirting	0.6	0	0.6
Accelerator Room	—	—	—
Grating	0.2	0	0.2
Shield Plug	0.2	0	0.2

Reactor Power Level: 1 Watts 0955

Survey Point #	Gamma Dose mRem/Hr	Neutron Dose mRem/Hr	Total Dose mRem/Hr
AGN Room	—	—	—
1	0.45	0	0.45
2	3.0	0	3.0
3	10.0	3.8	13.8
4	1.0	0.2	1.2
5	10.0	3.0	13.0
6	4.5	3.9	8.4
7	10.0	10.2	20.2
8	5.0	0.25	5.25
9	12.0	5.0	17.0
Accelerator Room			
Grating			
Shield Plug			
Skirt.	20.0	22.4	42.4

Reactor Power Level: 3 Watts

Survey Point #	Gamma Dose mRem/Hr	Neutron Dose mRem/Hr	Total Dose mRem/Hr
AGN Room	—	—	—
1	0.6	0	0.6
2	8.0	0.6	8.6
3	28.0	9.9	37.9
4	2.5	0.1	2.6
5	25.0	3.9	28.9
6	9.0	5.1	14.1
7	28.0	7.2	35.2
8	16.0	8.3	24.3
9	25.0	6.4	31.4
KIRT	45.0	48.0	93.0
Accelerator Room			
Grating	—	—	—
Shield Plug	—	—	—

DIM 7/6/92

Reactor Power Level: 5 Watts

Survey Point #	Gamma Dose mRem/Hr	Neutron Dose mRem/Hr	Total Dose mRem/Hr
AGN Room	—	—	—
1	1.4	0	1.4
2	14.0	0.6	14.6
3	45.0	11.3	56.3
4	4.0	0.4	4.4
5	45.0	10.8	55.8
6	15.0	15.1	30.1
7	50.0	30.3	80.3
8	28.0	2.5	30.5
9	42.0	13.5	55.5
Skirting	80.0	84.0	164.0
Accelerator Room	—	—	—
Grating	0.7	0.1	0.8
Shield Plug	0.0	0.3	0.3

Chapter 5: REACTOR COOLANT SYSTEMS

The AGN-201M reactor is operated at a very low power levels for short periods of time. The 1000-gallons of water in the shield tank, whose design function is that of a fast neutron shield, will by natural convection flow dissipate any heat that is rejected from the core. The amount of heat transferred to the shield tank has a little effect on the temperature of the tank.

Chapter 6 ENGINEERED SAFETY FEATURES

Engineered safety features (ESFs) are systems provided to mitigate the radiological consequences of designed-basis accidents. Since the AGN-201M reactor system operates at such a low maximum power level, 5 watts thermal, the fission product inventory is very low. Evaluation of the maximum hypothetical accident has indicated no significant radiological releases. Therefore, no ESF systems are required or needed at an AGN-201M reactor facility.

Chapter 7: INSTRUMENTATION AND CONTROL SYSTEMS

7.1 Summary Description

The instrumentation and control (I&C) system of an AGN-201 reactor consists mostly of three nuclear instrument drawers that allow for operation of the reactor and, if necessary, send a variety of scram signals to the reactor control safety systems. The design basis for these trips will be discussed in detail later in this chapter.

The reactor control system is designed, through a series of interlocks and relays, to control movement of the safety and control rods. This is done to ensures that a reactor startup cannot commence unless both safety rods are fully withdrawn from the core. The interlocks ensure that only one safety rod can be inserted at a time and that the coarse control rod cannot be inserted unless both safety rods are fully inserted.

The rods are controlled at the control console by a series of four switches, one for each rod, and position indication is provided on the console. Indicator lights will illuminate to give the operator indication of the status of each rod. These lights are illuminated when the rod and the carriage reach certain positions in their travel. A series of eight relays (K10 -17) actually provide the controlling features to this system.

The reactor control safety systems receive inputs from the nuclear instruments and many other detectors and the sensing systems and voltage detectors on the interlock relay system itself.

The console and at its features are clearly presented in a series of five figures presented in section 7.6 of this chapter. The layout of the control console is presented in these diagrams and all detectors and instrumentation is clearly labeled.

The only area monitor device directly associated with the AGN-201M reactor is the skirt monitor under the reactor. This system is intended to scram the reactor if radiation levels greater than 2 times the last skirt monitor indication at 5 watts..

7.2 Design of Instrumentations and Control Systems

7.2.1 Design Criteria

All systems and components associated with this system are located in the reactor console, in the shield tank or under it. The reactor control system diagram is presented in Figure 7.2.1-1 , this figure shows both the interlock system as well as the scram system for the AGN-201M. In Figure 7.2.1-2 a block diagram of the nuclear instrumentation systems for the AGN-201M is presented. A complete description of each of the nuclear instrumentation channels is presented in section 7.2.3.

The design and operations of these detection systems is such that the Limiting Safety Systems Settings will not be exceeded due to a reactor scram initiated by these systems. The control and safety setpoints are presented in Table 3.1 in the Technical Specifications for the AGN-201M reactor and will be presented later in this chapter.

7.2.2 Design - Basis Requirements

The design basis of this system is to monitor the entire range of reactor power and to provide protective features that will safely shutdown the reactor in the event of an equipment failure, operator error, or accident. The three nuclear instrumentation channels constantly monitor the reactor at all times. The scram and interlock reactor control safety systems are directly associated with the nuclear instruments since many of the signals to these systems come directly from this instrumentation.

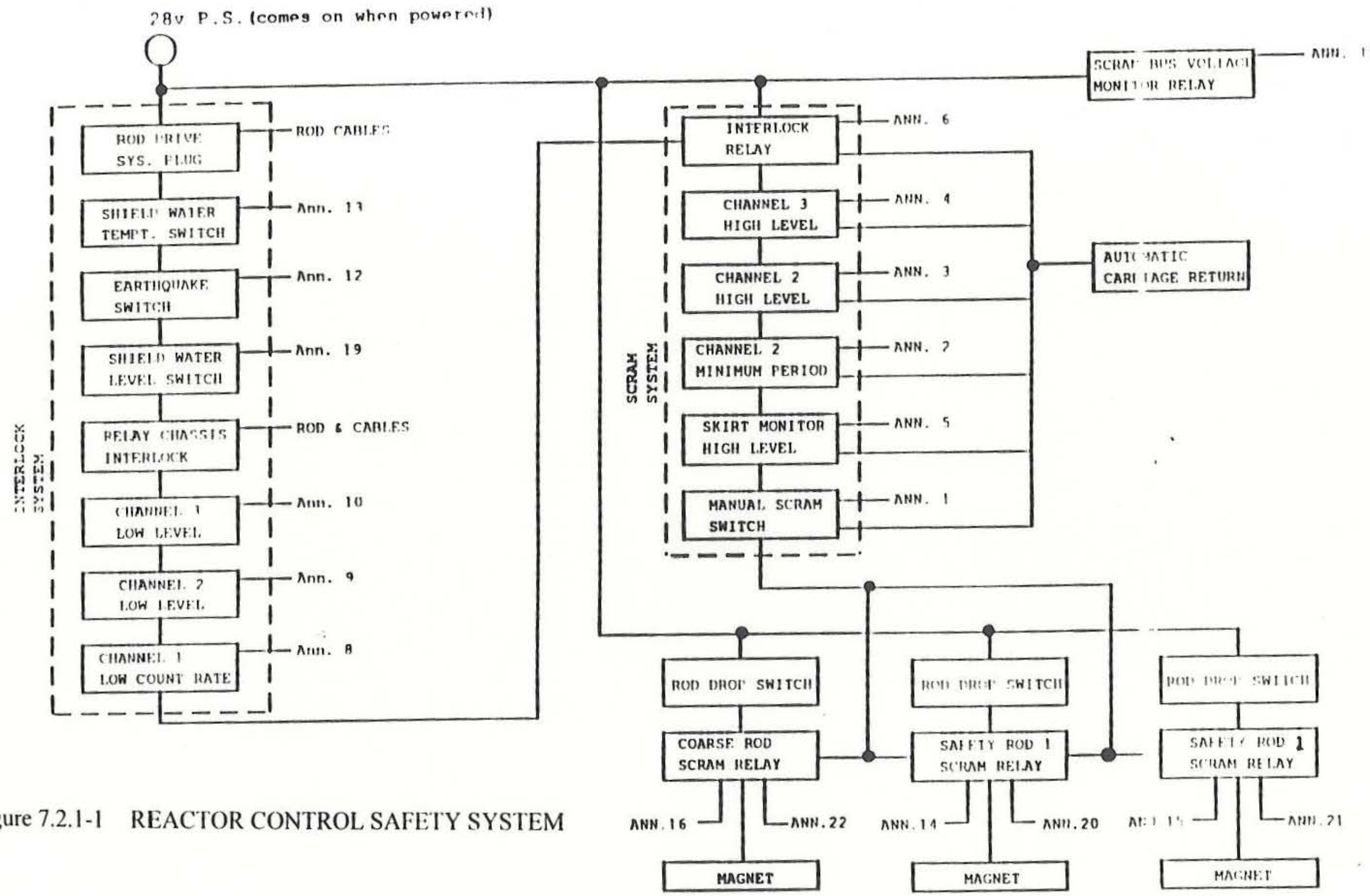


Figure 7.2.1-1 REACTOR CONTROL SAFETY SYSTEM

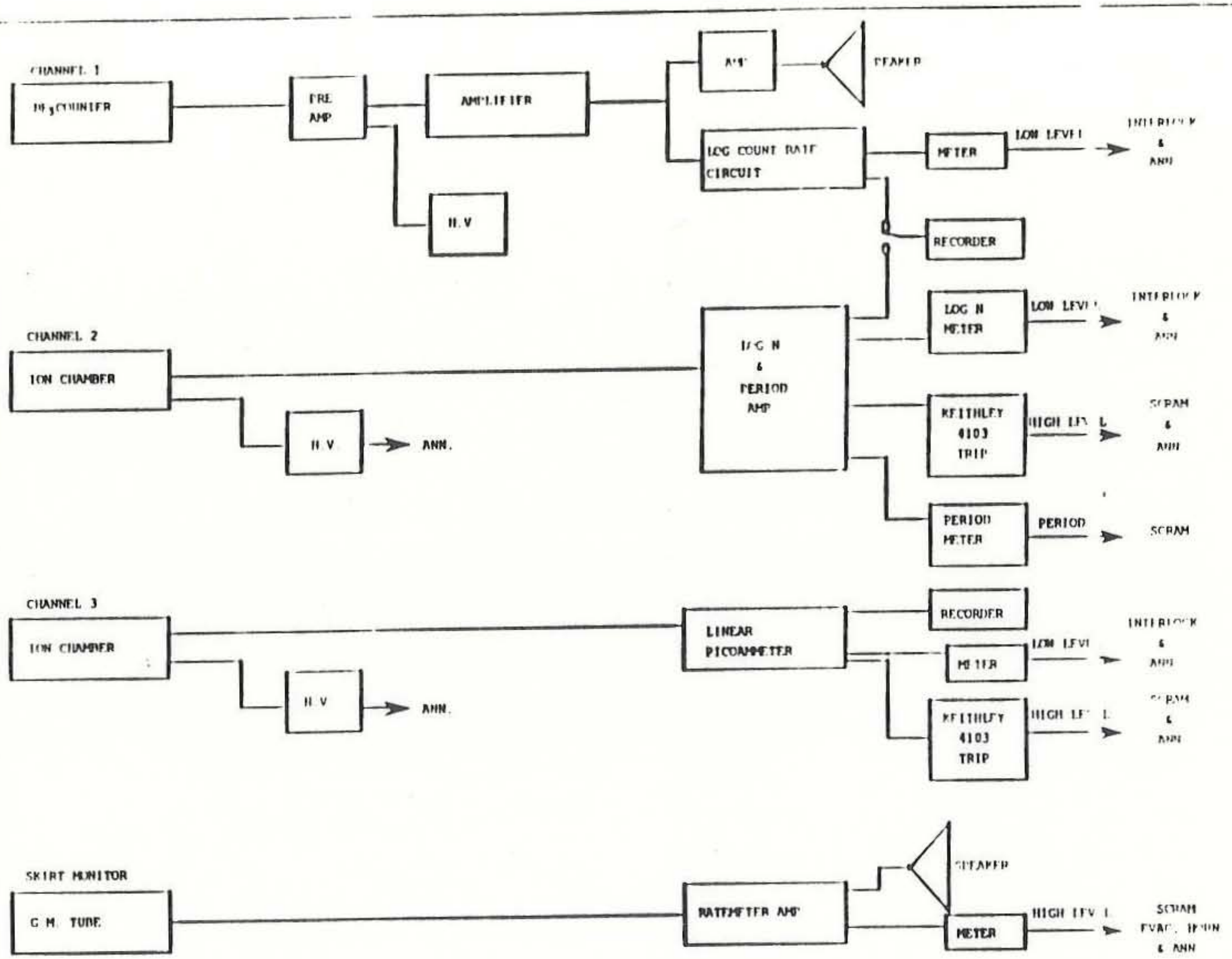


Figure 7.2.1-2 NUCLEAR INSTRUMENTATION SYSTEM

7.2.3 <u>System Description</u>

The nuclear instrumentation system consists of three channels and the skirt monitor.

<u>Channel No. 1</u> The components that comprise nuclear instrument Channel No. 1 are presented in block format in Figure 7.2.1-2. The detector is a BF_3- filled proportional counter with a range of 10 -10^6 counts per second. The detector is mounted in the shield tank and the relative position of all detector is presented in Figure 7.2.3-1. The channel drawer for this detector is located directly below channel No. 2 in the left-most side of the control console.

Channel No.1 is a low-range detector and its primary function is to ensure a neutron source is present in the reactor during startup. The detector system is made up of a high voltage power supply, a preamplifier and amplifier. The linear amplifier and power supply are located in the back of the control console in a NIM bin, the preamplifier is located on the top of the reactor in the cabling runs.

The output of this detector could be displayed on one of the chart recorders but normally this channel is not recorded. The high voltage for this detection system is determined annually as part of the preventive maintenance program . The system is also checked prior to reactor startup during the prestartup checklist. The 'push to calibrate" button is depressed and the meter should have a 60 counts per second input signal, if not the system can be adjusted with the calibrate to 60 CPS knob. Once this adjustment is completed the push to calibrate button is pressed a second time and the system comes back on line. Another check performed as part of the pre-startup checklist is the low count rate channel #1 interlock. This check is performed to ensure that if the count-rate

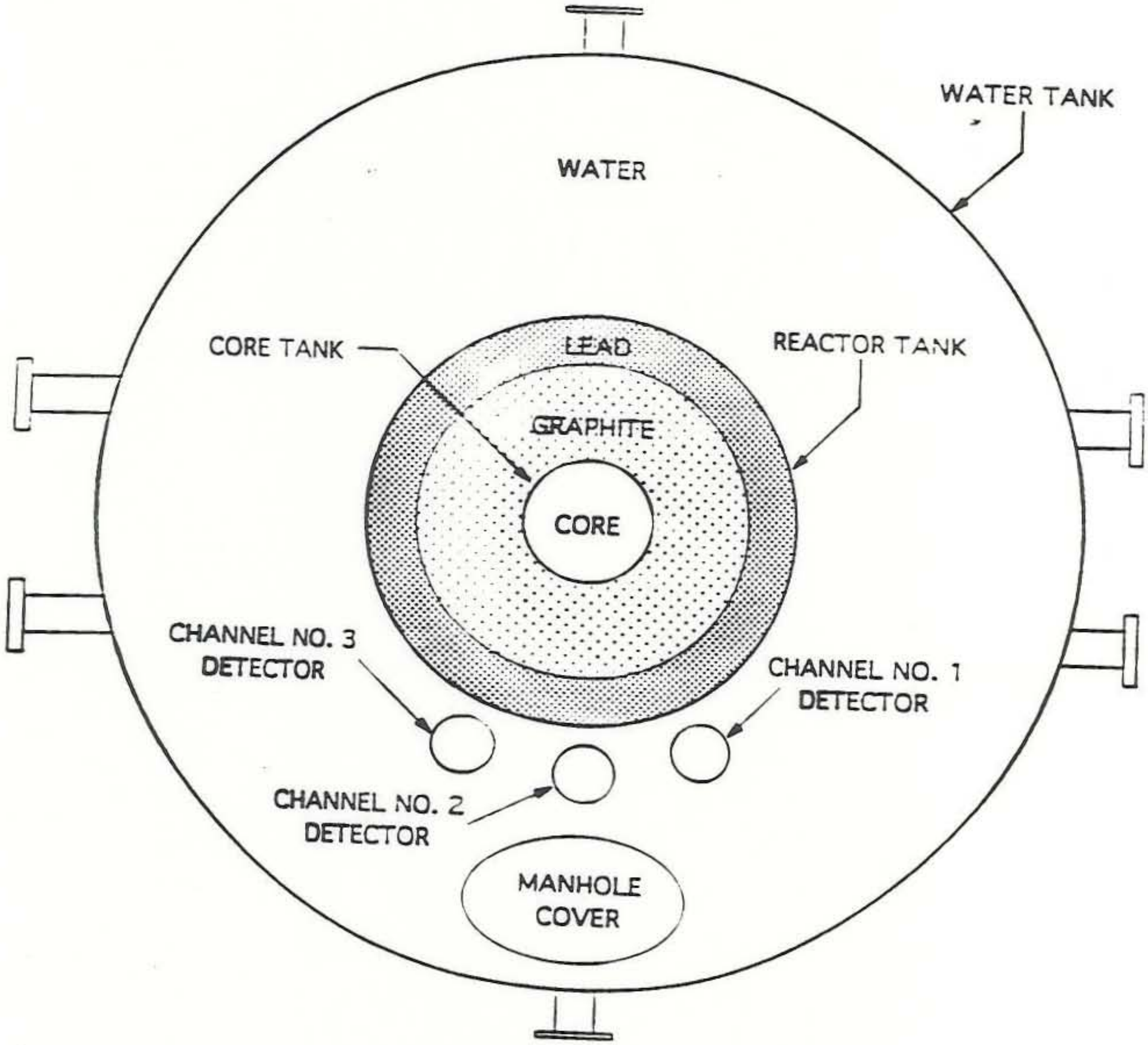


Figure 7.2.3-1   POSITION OF NEUTRON DETECTORS

drops below 10 counts per second the reactor will scram. An annual maintenance procedure also test this interlock to ensure that it is functioning correctly.

Channel No. 2 The block diagram for Channel No.2 is presented in Figure 7.2.1-2. The function of this channel is to provide both high and low-level trips as well as an input signal to the period meter. This system present output in a logarithmic fashion on a ammeter that reads from 10^{-13} - 10^{-6} amps allowing the power to be monitored over the complete range of reactor operations without switching scales.

The detector associated with Channel No. 2 is a BF_3-filled ionization chamber. The detector is physically located in the shield tank as shown in Figure 7.2.3-1 in a waterproof housing. The high voltage power supply for the detector is located in a NIM bin in the back of the control console.

Once the signal leaves the detector it goes to a Keithley Model 420 Log n Amplifier and Period Meter for processing. This processed signal is fed into the ammeter on the face of the Channel No.2 drawer for a logarithmic indication of power. A five - position switch is located on the front of the channel No.2 drawer that allows testing of the drawer and the associated trips and interlock relays.

These tests are performed during the prestartup checkout procedure and are required to be done prior to a reactor startup. This procedure test the operation of Channel No.2 drawer and the period meter as well.

The first test of the drawer is to ensure that it will operate correctly over the full range of reactor power. In order accomplish this, a series of test signals are feed into the system depending upon the switch position. Test signals of 10^{-13}, 10^{-11} and 10^{-7} are selected and an individual potentiometers adjusted for each test signal, this process is repeated until all three input signals are read on the meter face.

A low level Channel No.2 interlock is also associated with the meter on the front of the drawer, the interlock needle is moved to contact the meter needle and the interlock should activate. A high-level trip is also checked and the switch on front of the drawer is positioned to do this. This setting is calculated annually during reactor maintenance and is used to ensure the reactor will automatically scram at about 150% of the five watt reading.

Operation of the period meter is also checked during this procedure and a test signal is delivered through the check ∞ switch on front of the drawer. Once this is check a test signal is delivered to the period meter since it responds directly to this input. Once this check in complete the switch on the Channel No.2 drawer is moved to the 10^{-7} input position and the operation of full the minimum period scram is verified with the trip needle positioned at the 5-second position. An annual maintenance requirement also is in place to check the proper operation of the period meter.

The output signal from the Channel No.2 ammeter has a number of very important functions, it provides a signal to both the interlock system and scram system directly as well as to the period meter which also has a direct input to the scram system. The signal also is fed into one of the recorders so a record of Channel No.2 output is permanently recorded.

Channel No.3 The block diagram for Channel No.3 is presented in Figure 7.2.1-2. The primary function of this channel is to provide high power level scram protection and to ensure that the LSSS and LCO are not exceeded. This channel is the primary indication of reactor power over all ranges of operation.

The Channel No. 3 detector is located inside the shield tank as shown is Figure 7.2.3-1. This detector is fitted into a waterproof container and is a BF_3-filled ionization chamber. The Channel No.3 drawer has a Keithley 410A picoammeter installed in it with reads the output signal from the detector directly into watts. The lowest scale of detection of the selector switch is 0 -3 mircowatts and the highest scale reads from 0 - 10 watts.

The automatic trips associated with this drawer come directly from the picoammeter and are set by annual maintenance procedure to trip at about 5% and 95% of full scale. The 95% of scale trip will activate a scram system relay and the 5% of scale will activate an interlock relay, both will lead to reactor scrams. These scram features are tested prior to a reactor startup and the picoammeter is also tested on the 0 -10 watt scale.

The output 3 of this drawer is also directed to a recorder which records Channel No.3 output on a 0 -10 scale.

Skirt monitor The skirt monitor is a G.M. tube placed directly under the skirt of the reactor and inside a series of lead blocks. This system diagram is presented in Figure 7.2.1-2. The function of this system is too provide a high area radiation level scram signal if the detector indicates a reading that is 2 times a previous 5 watt reading or comes in contact with the needle on the meter face. The signal from this system goes directly to the scram relays and will also sound the evacuation alarm.

Recorders Two recorders are present in the control console and in operation during all reactor operations. These recorders are checked for proper operation during the prestartup checklist. One recorder is always recording Channel No.3 while the other maybe selected between the remaining channels. As per the operating procedures. Channel No.2 is always selected and this is verified by the prestartup checklist. Each chart is stamped with the time and date of both reactor startup and shutdown to maintain a permanent record.

The Technical Specifications clearly state all the required testing for all channels of the nuclear instrumentation system. The LSSS and LCO are clearly described in this document.

7.3&4 Reactor Control and Protective Systems

The AGN-201M reactor system only has a manual control system for control of the safety and fine and coarse control rods. The first part of this section will discuss a system design to limit rod movement and the second part will discuss the reactor protective system. The only interlocks associated with this system are directly related to rod movement and the maintenance of adequate shutdown margin.

The rods in the AGN-201M reactor system which are directly related to reactor safety. the two safety rods and coarse control rod. have limitations on their respective movement and cannot be moved unless certain conditions are meet.

The first rod to be moved is Safety Rod #1. this rod must be driven into its fully inserted position prior to the K-10 relay activating. cutting power to Safety Rod #1 and allowing movement of Safety Rod #2. The K-relays are designed to activate only when the appropriate safety system rod is in its fully inserted position or in the fully withdrawn position depending upon their function. Once Safety Rod #2 is fully inserted and the K-12 relay operates and then the Coarse Control Rod can be moved. A set of these K relays also are activated when the carriages are returned to the down position so that rod movement maybe started again after a reactor scram.

The reactor protective system, known as the Reactor Control Safety Systems consists of the Interlock System as well as the Scram System. The Interlock System was designed as an add-on to the original Scram System when it was found over the years of AGN operations that more automatic protective devices need to be added to ensure that none of the LSSS and LCO were exceeded as the reactor power levels were raised.

The Reactor Control Safety Systems rely upon signals from the nuclear instruments as well as other detection systems to perform their designated function, that is, to scram the reactor. The features directly associated with the nuclear instrumentation system were discussed in previous section, so the other parts of the Interlock Relay and Scram System will be addressed in this section.

The manual scram switch is located in the middle of the control console and maybe activated in the event of any emergency condition that exists in or around the reactor facility. This switch is also used to scram the reactor after completion of normal reactor operations. A diagram of the Scram System is provided in Figure 7.2.1-1.

The Interlock Relay in the Scram System will activate anytime one of the Interlock System trips is activated, just as any trip circuit in the Scram System.

The Interlock System consists of some trips that are associated with the shield water tank, a low-temperature trip that activates at 20 ± 0.5 °C and a low level trip that activates when the level in the shield tank falls below 10 inches from the highest point.

These trips are important maintaining shielding requirements in the reactor room. Both of these trips are checked as part of the annual maintenance program and as directed by the Technical Specifications.

The other trips will active if the scram bus voltage is sensed to be low a trip will occur or if the rod drive system plug is removed from any of the safety system rods.

7.6 Control Console and Display Instruments

The reactor control console is presented in Figures 7.6-1-5. These figures provide a clear idea of the particular console design for the AGN-201M reactor. The console is designed so that the operator has within easy reach all control switches for the nuclear instrument channels as well as a clear view of all indications including the alarm panel. Figure 7.6-1 depicts the alarm panel as well as the skirt monitor. The indicator lights on the panel are different colors and give the operator a visual as well as audio indication of the reactor status.

The rod energized lights are green and the scram indicator lights are red, these are just two examples of how the alarm panel was designed with the operator in mind. The rod position indicator lights are different colors which makes the identification very simple as well as providing a visual indication of rod position. Figure 7.6-2 displays the center of the control console and the position of one the recorders and the rod position indicators and well as the manual scram button. Figure 7.6-3 displays the other recorder, the one selected to Channel No. 3, as well as the drawer for Channel No. 3. Figure 7.6-4, the last of the upper section of the control console indicates the relative positions of

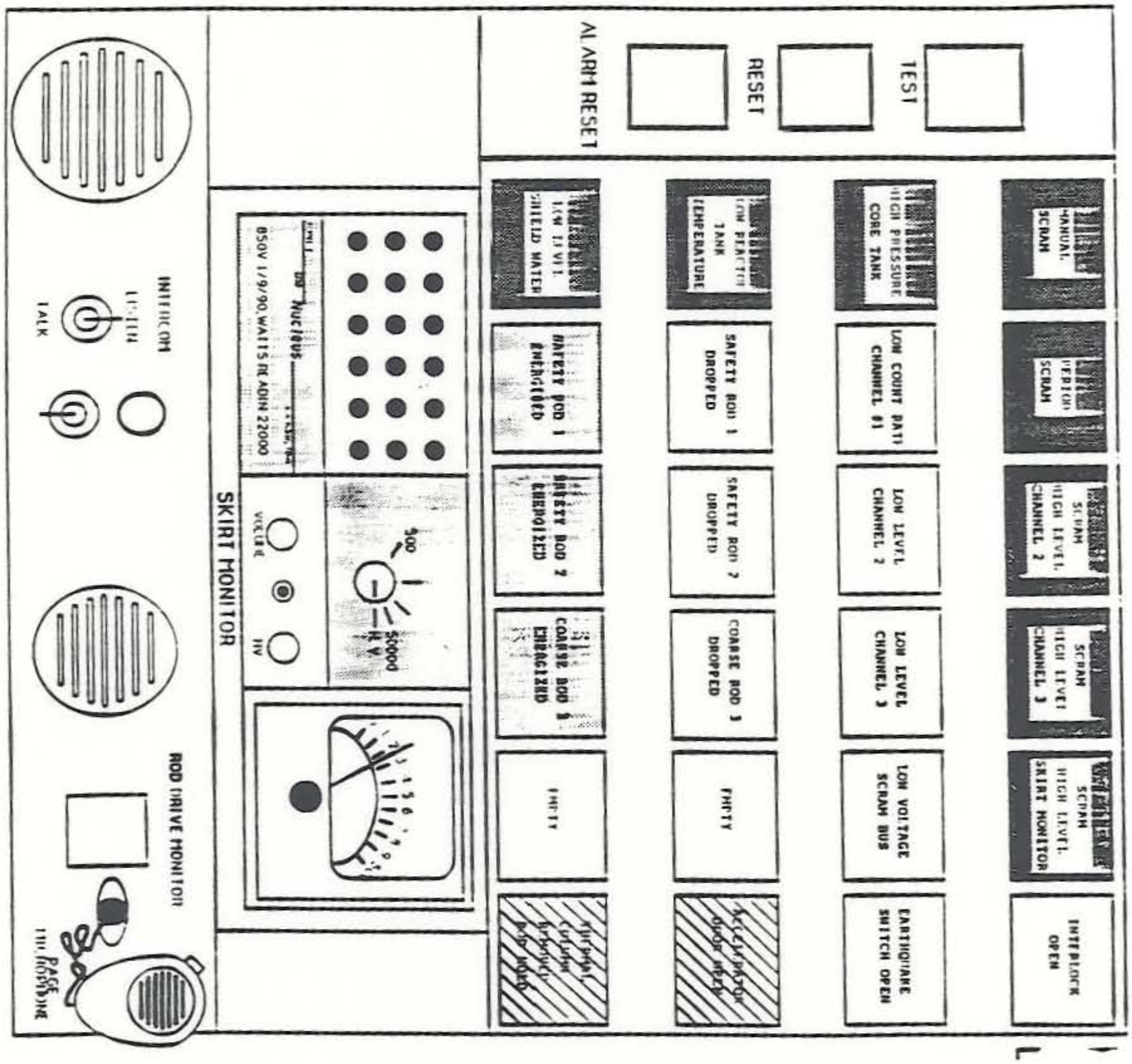


Figure 7.6-1 CONTROL CONSOLE ALARM PANEL

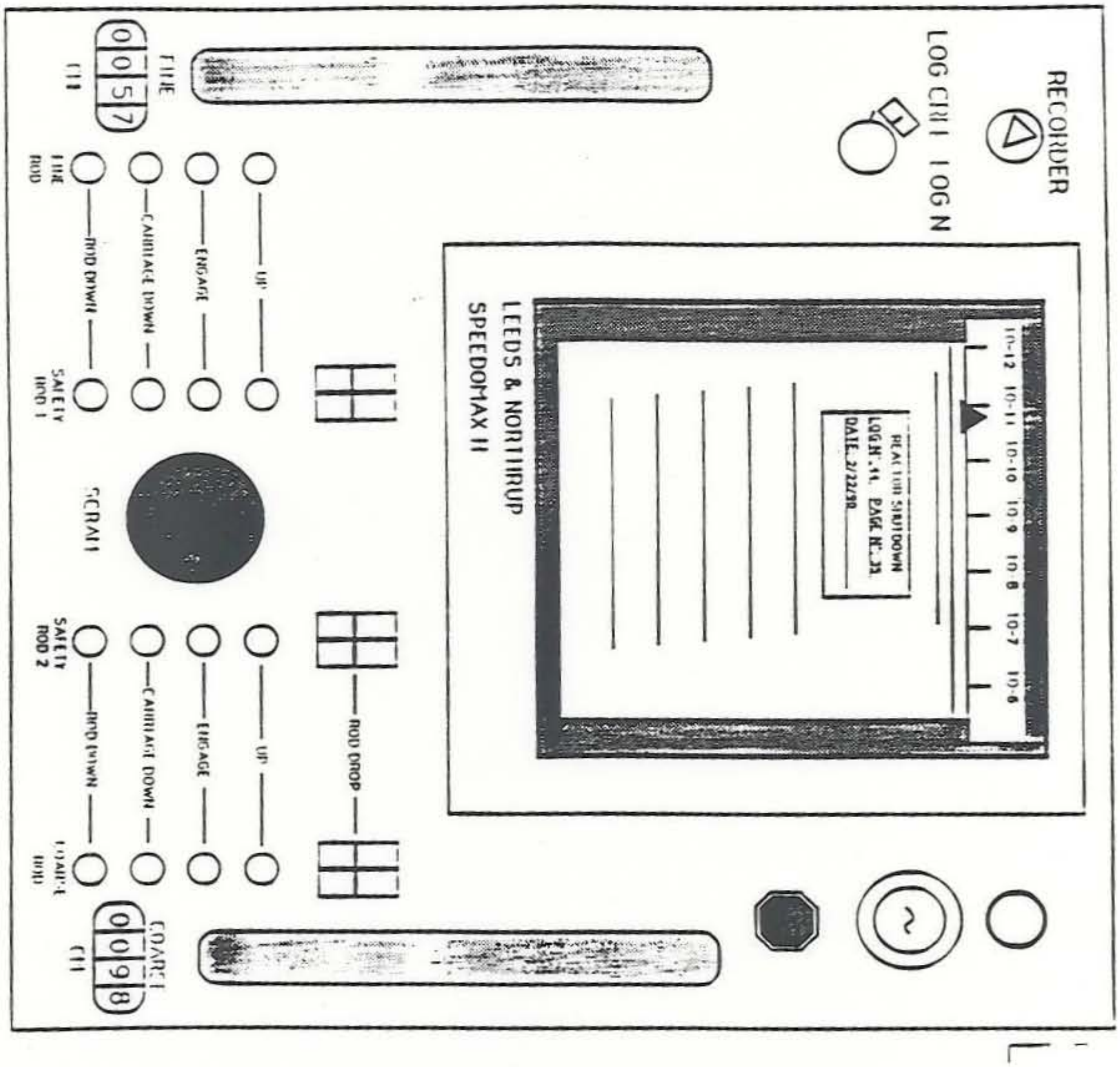


Figure 7.6-2 ROD POSITION INDICATORS AND RECORDER

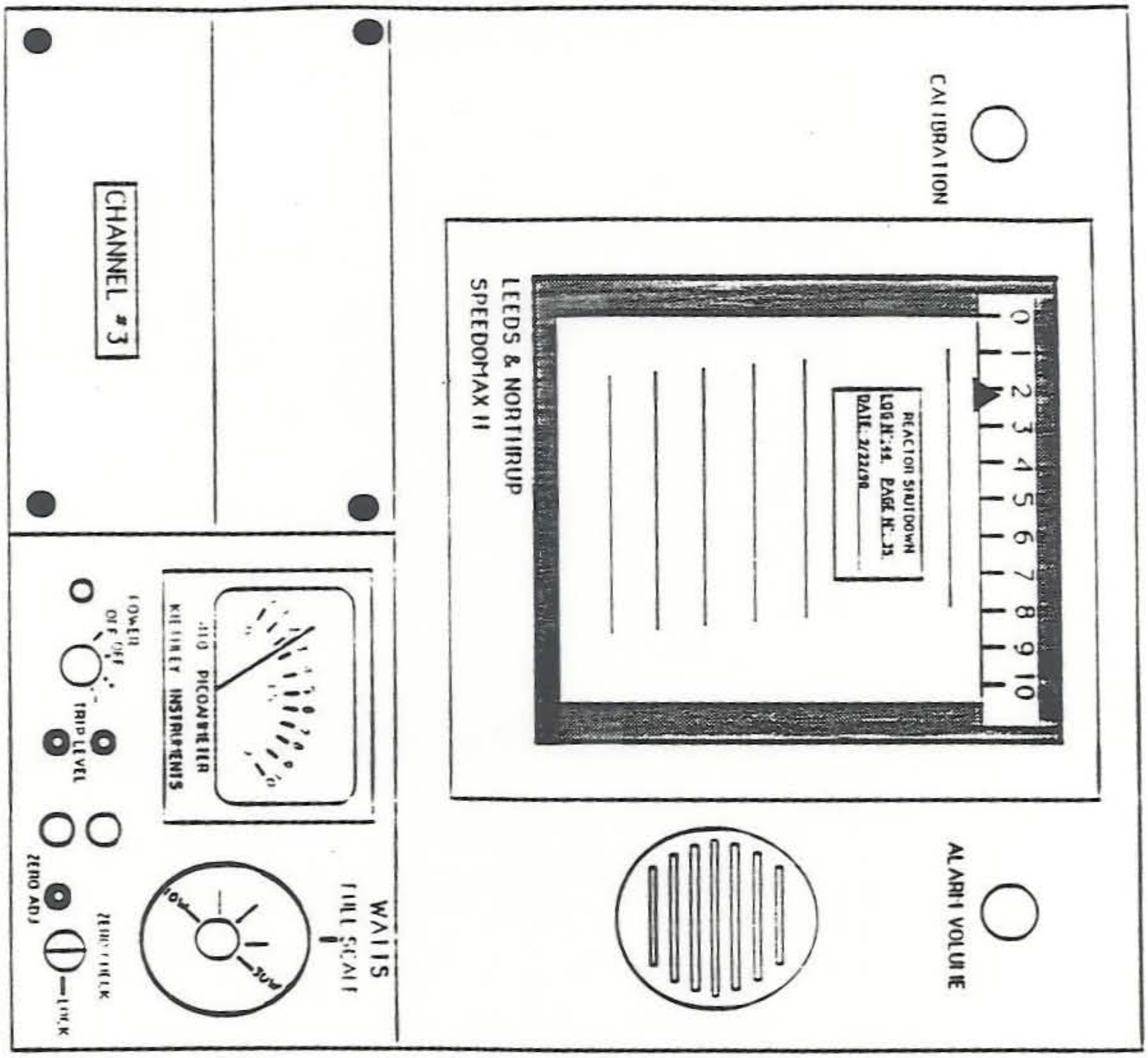


Figure 7.6-3   CHANNEL No.3 DRAWER AND RECORDER

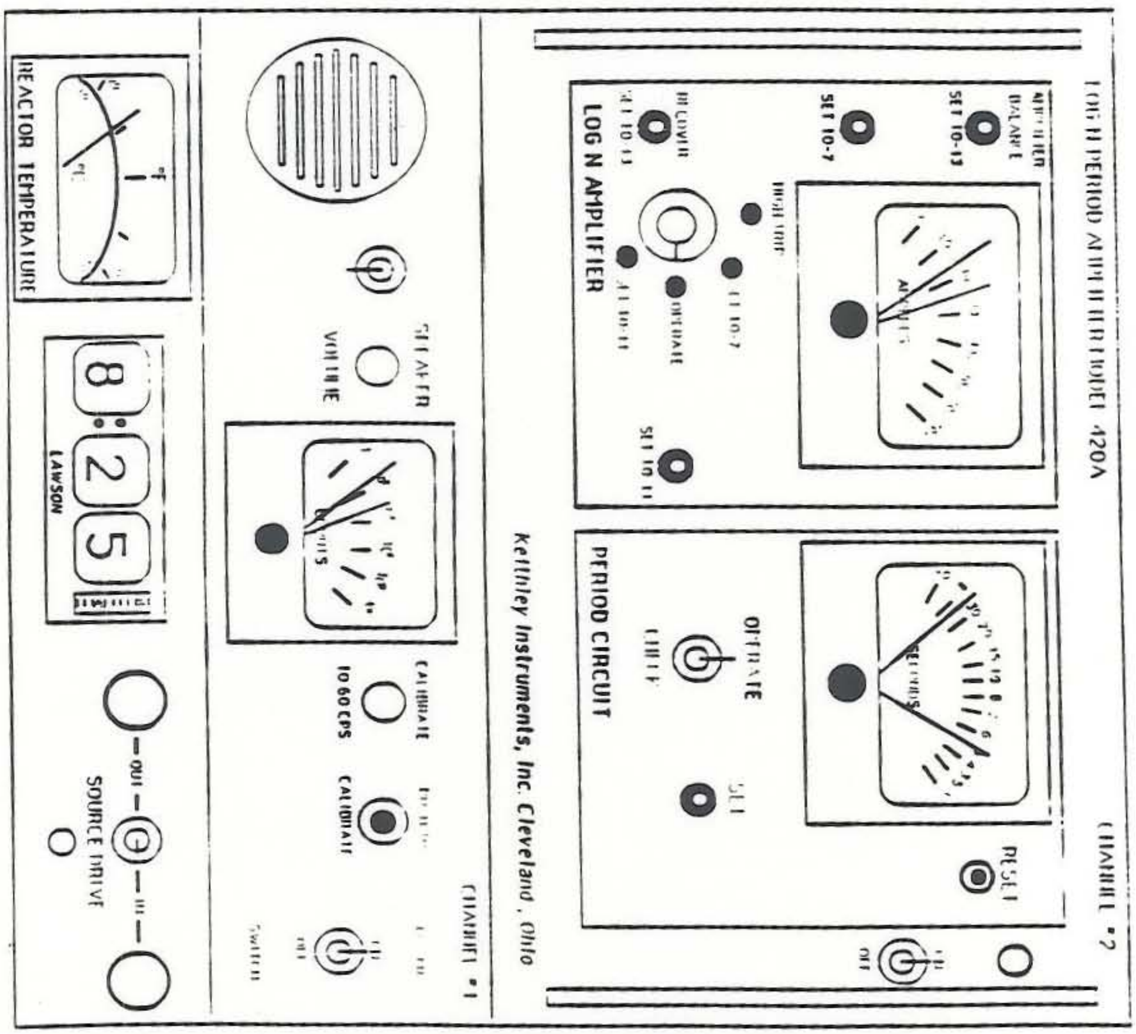


Figure 7.6-4 CHANNEL 1 & CHANNEL 2 DRAWERS

nuclear instrument Channels No. 1&2. The location of the control panel for the source drive is also shown in this figure. These figures clearly show the ease with which the reactor operator can observe and monitor reactor operations. In Figure 7.6-5, the switches that control the rods are shown, this horizontal panel is located directly below the recorder for Channel No. 2.

7.7 Radiation Monitoring Systems

The only radiation monitoring system directly associated with the AGN-201M reactor is the skirt monitor. A G.M. tube, inside a lead shield, is in place under the reactor inside the skirt. The skirt monitor panel on the control console provides an indication of the radiation level under the reactor. As part of the annual maintenance program the radiation levels at a reactor power of 5 watts is recorded and a voltage plateau performed on the detector. The skirt monitor is then set to alarm at two times this 5-watt reading. It is assumed that, if the radiation levels reach these levels in the local area, some accident or loss of shielding has taken place. When this monitor reads its setpoint the reactor will scram and the evacuation horn will sound in the areas outside and above the reactor room. Should this occur Emergency Procedures will be initiated, and the cause of the problem will be investigated, and corrective actions will be taken. Another monitor is present as a criticality monitor for the source locker, this system is designed to measure radiation levels above the locker an ensure that conditions don't exist that a k_{eff} of 0.8 is possible.

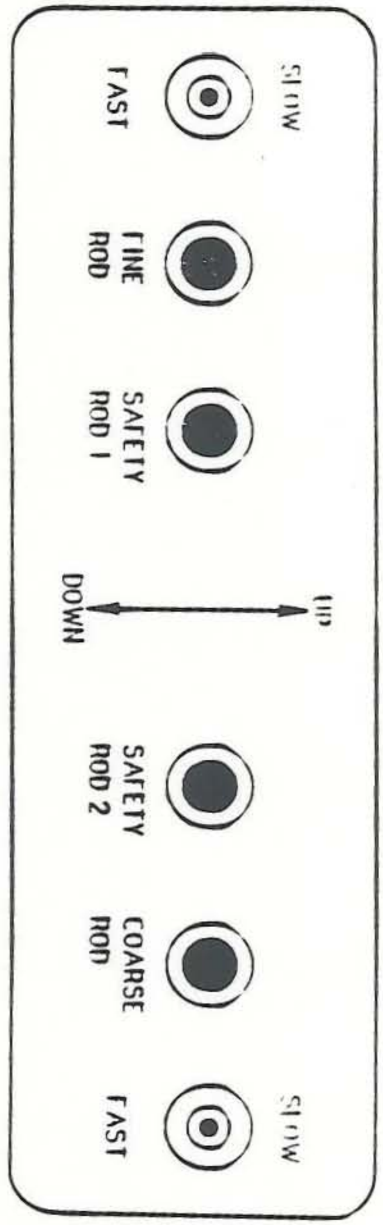


Figure 7.6-5 ROD CONTROL SWITCHES

Chapter 8: ELECTRICAL POWER SYSTEMS

8.1 Normal Electrical Power Systems

The AGN-201 reactor system has an electrical requirements of 2kw of 110 V AC. This power is supplied directly to the reactor room from the Zachry Engineering Center power network. The actual power to the reactor room is both 220 and 110 volt AC. Both of these power systems can be remotely disconnected using two breakers located in the hallway outside of room 60C. The AGN-201M reactor system has no emergency power requirements. If a loss of power occurs during operations the rods are forced from the core and the reactor is considered shutdown. The cadmium rod is added to the core to ensure the reactor is shutdown since without instrumentation this is hard to verify. A radiation survey can also be performed to ensure the reactor is shutdown.

The AGN-201M reactor system uses a variety of voltages and currents in normal operations. It would be impossible to present all that information even in this context. The major power distribution electrical block diagram for the Texas A&M University AGN-201M reactor is Drawing Number 2-C-100631. This diagram breaks down the regulated 110 volt AC to all the detector high voltage power supplies and to each of the channel drawers. This regulated AC passed through two sola type voltage transformers, these transformer have been replaced by modern line conditioners to better handle the power from the Zachry Engineering Center.

8.2 Emergency Electrical Power Systems

The AGN-201M reactor does not require an emergency power system due to its design and lack of external systems. The reactor facility has an emergency lighting system that will energize when power is lost. This will ensure some lighting and allow for safe departure from the reactor room.

Chapter 9: AUXILIARY SYSTEMS

AGN-201M has few auxiliary systems associated with it and none have any safety significance expect for the ventilation system.

9.1 Heating, Ventilation, and Air Conditioning Systems

The ventilation system for the AGN-201M reactor room consists of one circulating unit by the reactor console, a supply area in the accelerator laboratory and an exit fan on the roof of the Zachry Engineering Center. The inlet plenum for the fan suction is located in room 135, the accelerator laboratory and the air flow is designed to pass from the reactor room up through the grate in the ceiling and into this inlet plenum. Measurements have confirmed that air passes from outside both the accelerator laboratory and the reactor room in this manner. The supply header in the accelerator consist of three outlets supplying air at 500 cfm each and the exhaust fan drawing off air at 1500 cfm. The unit in the reactor room recirculates air at 1100 cfm.

9.2 Handling and Storage of Reactor Fuel

The AGN-201M reactor has such a low fuel burnup rate that it will never have to be refueled in its useful lifetime. Some approved experiments can require the removal of fuel from the core but these experiments are no longer performed at Texas A&M University.

9.3 Fire Protection Systems and Programs

No permanent of fixed fire suppression system exists in the Zachry Engineering Center. Several CO_2 fire extinguishers are present in the reactor facility including one by the reactor control console and another outside the reactor room in room 61B. The design of the facility limits the effects of fire on the facility. The reactor console is in the corner of the room, a fire starting in the area it could be put out before spreading. The reactor room itself is large and without much equipment in it so everything is fairly spread out. If a fire were to occur outside the facility, access to the facility would be limited and the fire would have to pass the thick concrete walls and ceiling of the accelerator laboratory and the reactor room and adjoining laboratory. The doors into both the reactor room and accelerator room are metal so the fire would have little chance of getting through.

Chemicals used by the department are stored inside explosion- proof lockers. During the biannual reactor emergency drill involving outside agencies, one of the local fire department companies responds at the scene. After the drill is over, fireman are given a tour of the facility and the different aspects of the facility are pointed out to them..

One of the Emergency Procedures directly addresses a fire in the facility and the Reactor Emergency Procedure (RE-1) addresses a fire that could be in a nuclear system. With the core surrounded by 1000 gallons of water and the isolated nature of the reactor room in the Zachry Engineering Center the chances of any damage to the reactor or any reactor components is very small.

9.4 Communication Systems

An public address system is used by the reactor operator to make important announcements in the facility. The microphone is located on the reactor console and the speakers for the system are located in the hallway outside of room 60C and room 135.

9.5 Possession and Use of Byproduct, Source and Special Nuclear Material

The only special nuclear material directly associated with the AGN-201M reactor is the startup source located in access port #3. This is a 1 Ci Pu-Be sealed source and is 1 used to provide neutrons in the reactor. Several other sealed sources are located within the reactor room but these are just stored in the source locker and are under the control of Radiological Safety. Two sealed reactor fuel plates are also stored in the source locker, one from an AGN-211 and one from an AGN-201. These plates will not be used and will be returned with the rest of the AGN-201M fuel when the reactor is decommissioned.

Chapter 10: EXPERIMENTAL FACILITIES AND UTILIZATION

The experimental facilities associated with the AGN-201M reactor system consist of the glory hole, which passes directly through the center of the reactor and the access ports which pass outside the reactor in the graphite reflector the lead and water shields.

This particular facility has very little experimental use. The TRIGA reactor is primarily used by researchers at the University for experimental research. The AGN-201M reactor is used for experiments in nuclear engineering classes. The only experiments run using the reactor during these classes are from the set of approved experiments for the AGN-201M. Section 3.3 of the Technical Specifications clearly listed all the limitations on experiments at the AGN-201M reactor facility.

If an experiment were proposed to be performed using the AGN-201M, the experimental procedure first would have to be approved by the reactor supervisor. Then, this approved proposed experiment would be presented to the Head of the Department of Nuclear Engineering. Once reviewed and approved by the Department Head, the proposed experiment must be presented to the Reactor Safety Board, if a majority on the Board approve the experiment only then can it be performed. It may only be performed under the direct supervision of the reactor supervisor and with the approval of the Department Head.

10.1 Summary Description

The AGN-201M reactor design has both the in-core facility, the glory hole, and in-reflector facility with the access ports. A thermal column may be installed in the AGN-201M. However, this feature has not been used and this area is filled with water.

10.2 Experimental Facilities

The experimental facilities in an AGN-201M reactor design are limited to the glory hole and the access ports. While the center of the glory hole provides a relatively large neutron flux, the remaining area of the glory hole will be filled by either an experiment holder or a poly rod so the generation of radioactive gases is very limited.

10.3 Experiment Review

The experiment review process is a three-level system at the AGN-201M reactor facility. The proposed experiment must be presented to the reactor supervisor for review and consideration. His responsibilities include ensuring all requirements are meet for limitations on experiments and reactivity as per the Technical Specifications. This review includes a verification of all reactivity values of the materials and experimental equipment that are exposed to the reactor neutron flux. Special care must be and taken to ensure that if any material becomes activated during the experiment that proper radiological controls are taken and that personnel from Radiological Safety are directly involved in evaluating the experiment. The reactor supervisor must ensure that no part of the experiment can lead to damage to reactor systems or injury of personnel in the event of a failure. If the reactor supervisor is satisfied that the experiment can be conducted safely and poses no danger to the general public, he must present the proposed experiment to the Head of the Department of Nuclear Engineering.

The Department Head must then complete an evaluation of the proposed experiment and determine if it is in fact worth the resources to conduct and has no unresolved safety considerations. All the aspects of the experiment will be discussed with the reactor supervisor to ensure no questions remain. The Department Head must then present the proposed experiment to the Reactor Safety Board for consideration. If a majority of the Board approves then the experiment can be conducted.

The experiment can only be conducted with the direct approval of the Department Head and under the direct supervision of the reactor supervisor.

11 RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1 Radiation Protection

The radiation protection program at Texas A&M University is the responsibility of the Radiological Safety personnel which is part of the University's Environmental Health and Safety organization. The AGN-201M reactor room is part of the nuclear engineering area but all of these laboratories lie within the physical area of the reactor facility as defined by the Security Plan. The personnel directly involved in the radiation protection program for the AGN-201 are staff of the facility. The reactor operators control access to and control the actions of all personnel in the reactor room.

Radiological Safety personnel conduct an annual radiation survey in association with the AGN's preventive maintenance program. This survey verifies the extremely low levels of radiation present in uncontrolled areas outside the reactor room.

The main concern of the program is to limit exposure to both the students and the general public. The design of the reactor facility and its location in the Zachry Engineering Center has isolated the reactor room from unrestricted access by the general public. The Physical Security Plan does not allow unrestricted access to the reactor facility by the general public. The operators do not allow any personnel in the reactor during reactor operations unless they have some form of dosimetry. The reactor operating procedures do not allow for a reactor startup without verification of proper personal dosimetry.

An important design feature of the facility is the presence of the C-2 warning device on the door to the reactor room. This device will sound a horn if anyone opens the room door whenever the reactor console is energized. As a further limitations on access, the door to laboratory 61A is always locked closed and only personnel assigned to the facility or doing research in that laboratory have a key to the door.

11.2 Radiation Sources

The only radioactive source that is directly associated with license R-23 is the 1 curie Pu-Be startup neutron source located inside access port #3. This sealed source is swiped quarterly and these records are maintained by Radiological Safety personnel. A radioactive source storage locker is located in the corner of the reactor room 61B. All sources contained in this storage locker are under the control of Radiological Safety personnel. Three 2-Ci and two 1-Ci sealed Pu-Be sources are stored in this locker, along with one 2 mCi Co-60 source. These sealed sources are stored inside containers in the source locker. Inside the source locker are two fuel plates one from an AGN-211, and the other from an AGN-201, these plates are inside sealed aluminum storage holders and will remain inside these holders until a time when the AGN fuel is removed.

Another source locker is located inside the counting laboratory, this locker and laboratory are secured at all times unless in use these sources are used as check souces in nuclear engineering laboratories and consist mostly of source sets. All of these sources remain inside the locked storage cabinet, inside a locker storage container unless they are required for a laboratory. The teaching assistant or professor must obtain a key to the source locker from the laboratory manager and be briefed on precautions regarding their use.

11.1.1 Airborne Radiation Sources

The only possible airborne radioactive source associated with the AGN-201 could be due to the generation of Ar-41 from a neutron interaction with Ar-40. Analysis of this interaction has shows a minial production of Ar-41 at a this particular type of reactor facility.

11.1.1.2 Liquid Radioactive Sources

The design of this reactor facility and the experiments conducted do not lead to the production of any liquid radioactive sources. The only possible source is the 1000 gallons of water that surround the reactor in a sealed tank. This water will be analyzed and level of possible contamination determined as part of a decommissioning plan for the facility.

11.1.1.3 Solid Radioactive Sources

No solid radioactive sources exist at this facility other than those described in section 11.1 on this chapter. No solid radioactive waste is present at the facility other than small amounts laboratory waste generated by students during experiments. This radioactive reactor waste is removed by Radiological Safety personnel, but by its very nature is a very small about of waste. No waste other than laboratory trash-type waste is generated by experiments directly associated with AGN-201M reactor operations, and maintenance.

11.1.2 Radiation Protection Program

The radiation protection program for the AGN-201M is cooperative program between Radiological Safety personnel and the facility staff. The Radiological Safety Officer directs the radiological safety personnel in the workings of the program. Under emergency conditions , the Head of the Department of Nuclear Engineering or his designated alternate is the emergency director for the emergency response personnel at the AGN but gets direct input from radiological safety personnel.

The main function of this program to protect the general public under normal and emergency conditions from radiation exposure and ensure that staff personnel and students use the ALARA principle to fullest extend possible to keep exposures low.

The dose to the students is directly controlled and limited by the actions of the facility staff. Any student that is working with the reactor will be issued a dosimeter. If they are classified as radiation workers in the department, Radiological Safety will issue them a suitable personnel monitoring device. If these individuals are nuclear engineering students, the reactor operator will issue pocket ion chambers, obtained from Radiological Safety personnel, prior to reactor operations.

When a nuclear engineering class is scheduled to use the AGN-201M for experiments in the laboratory , the class is lectured about reactor operations and construction by a qualified operator. During these lecture, a tour of the reactor room is performed and through this effort the student will become familiar with the reactor room layout. It is during this tour that the reactor operator points out the location of specific items (i.e. the source locker) of ALARA concern. This lecture will point out that avoidance of these areas will lower the radiation exposure of the students. When the student is to perform their reactor startup, one of the first steps of the pre-startup checklist is to conduct a radiation survey of the room. It is during this time that the student is trained on the location of all survey points and on the practice of ALARA during reactor operations.

Reactor operations during laboratory classes is limited to the lowest power level practical for the experiment. Student access to the reactor room is limited to only personnel necessary to successfully complete the experiment and to operate the reactor. All other students involved in the experiment, are allowed access only to room 61B, this way the students level of exposure is greatly limited.

11.1.3 ALARA Program

The ALARA program in place at Texas A&M University comes directly from procedures created by the Office of Radiological Safety. All of these procedures as well detailed instructions are presented in "PROCEDURE MANUAL FOR THE USE OF RADIOISOTOPES AND RADIATION PRODUCING DEVICES" which was last revised in July, 1990. The ALARA Program is addressed in section 4 of the manual

11.1.4 Radiation Monitoring and Surveying

At the University personnel from Radiological Safety perform the required radiation surveys and instrument calibrations. The basis for the radiation survey is detailed in the Technical Specifications section 3.4. and specifications and basis for radiation monitoring and control are contained in section 4.4 of the Technical

Specifications. Section 4.4 also details the requirements for calibration of all portable radiation survey instruments assigned to the reactor facility by personnel for Radiological safety.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure control is a very important concern at the AGN-201M facility. The reactor room is considered to be a radiation area whenever the reactor is operating at a power level of less than 0.9 watts and a high radiation area whenever the reactor is operated at power levels greater than 0.9 watts. These radiation levels raise many ALARA concerns but the reactor is only operated at power levels greater than 0.9 watts when required by maintenance procedures. When students operate the reactor for laboratory reactor startups, the power level rarely exceeds 50 milliwatts.

If irradiations are required for a laboratory only the reactor operator and one student will be in the reactor room and they are shielded by the shielding around the control console. If the activation foils are to be taken to the counting laboratory they are moved in a lead container and handled with large tweezers and remain in the custody of the reactor operator. When the laboratory is completed, the foils are placed into the source locker inside the reactor room for decay.

All students performing reactor startups are issued pocket ion chambers and students performing experiments in the area around the reactor are also issued pocket dosimeters. These dosimeters are provided by Radiological Safety personnel and issued by the reactor operator just prior to reactor startup.

With the reactor operating only for about 30-40 hours per year mostly for reactor maintenance exposure, levels in the reactor room over the course of a year due to reactor operations are relatively low. The reactor room has also been used for calibration of portable survey instruments by Radiological Safety personnel. These exposures have added to the dose measured by the local area badges. During years of consistent reactor usage and instrument calibration, the area badges read between 180 and 830 mrem per year of gamma and neutron radiation. Since the calibration of instruments has been moved to another location, the area badge inside the reactor room still reads about 750 mrem per year but the area badge outside the reactor room read only 10 mrem.

The only anticontamination clothing worn at this facility are gloves that are worn to perform swipe surveys on samples, no respiratory protective equipment is used at the AGN-201M reactor.

The area badges and personnel dosimetry issued to facility personnel are evaluated monthly. If problems were to occur in this facility personnel from Radiological Safety would be called in to assist facility personnel in accessing the problem and help in the restoration efforts.

Doses at this facility have been very low even to personnel assigned to the facility. In calendar year 1996 only one person assigned to the facility had any recorded radiation exposure.. Great care is taken by the facility staff to ensure that all personnel associated with the AGN-201M reactor receive doses which are as low as possible.

11.1.6 <u>Contamination Control</u>

The reactor facility is swiped monthly by personnel from Radiological Safety. A radiation and contamination surveys are performed on all samples after they are removed from the reactor. With a sealed reactor system, no loose contamination is expected to found in the AGN-201M reactor facility during normal operations. When the control rods are removed from the for reactor maintenance or repair, they are surveyed to ensure no loose contamination is present. All the sources stored in the reactor room are sealed.

11.1.7 Environmental Monitoring

No environmental monitoring, expect for area badges is conducted at this facility

11.2 Radioactive Waste Management

The only waste generated directly from reactor operations is simple laboratory trash. Efforts are made by facility staff to limit the amount of waste generated. The total amount is very small and Radiological Safety personnel dispose of this laboratory waste.

Chapter 12: CONDUCT OF OPERATIONS

12.1 Organization

The AGN-20M reactor is owned by Texas A&M University and has been operated by the University since 1957 under USNRC license R-23, Docket 50-59. Since the reactor is at a university the organization surrounding it is quite unique to this particular university and will be discussed in the following chapter. A detailed description of all duties responsibilities and requirements are presented in the Technical Specification, section 6.0

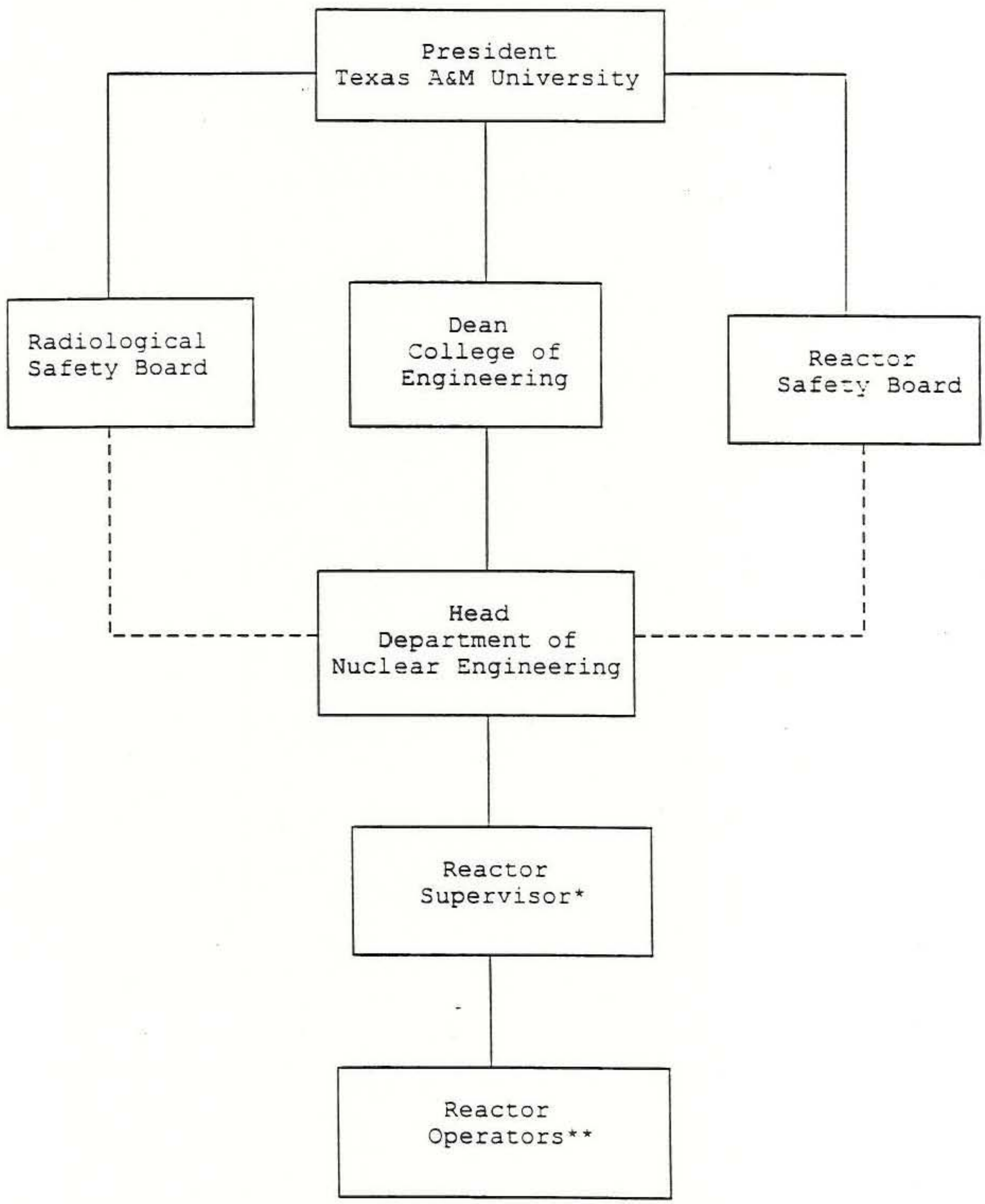
12.1.1 Structure & Responsibility

The organizational structure at Texas A&M University is intended to fulfill the requirement setforth by the USNRC. The organizational structure is presented in flowchart format in Figure 12.1.1-1.

The President of the University is considered to be the chief administrative officer for the facility. The application for licensing of the reactor is made in his name. The Dean of the College of Engineering is administrative officer responsible for the day to day operations of the College of Engineering and, since the Department of Nuclear Engineering is among those departments, he has an administrative role in the operations of the AGN-201M reactor.

The Head of the Department of Nuclear Engineering is the person administratively responsible for the department. This includes the AGN-201M reactor. The Department Head shall have final authority and ultimate responsibility for the operation, maintenance and safety of the reactor facility. The Department Head shall appoint personnel to the position of Reactor Supervisor and shall select qualified

Figure 12.1.1-1 TEXAS A&M UNIVERSITY ORGANIZATIONAL STRUCTURE



* Requires NRC Senior Operators License
** Requires NRC Operators License except where exempt per 10 CFR 55 paragraph 55.13

candidates as Reactor Operators. He shall work with members of the Reactor and Radiological Safety Boards to ensure that no unresolved safety questions are present. He shall present new experiments and procedures and facility modifications to the Boards for their review and approval.

The Reactor Supervisor shall be a licensed SRO and shall be responsible for the preparation, promulgation and enforcement of administrative controls including all the rules, regulations, instructions and operating procedures to ensure that the reactor facility is operated in a safe competent and authorized manner. He shall direct the activities of operators and technicians in the daily operation and maintenance of the reactor; schedule reactor operations, maintenance; be responsible for the preparation, authentication and storage of all prescribed logs and operating records: authorize all experiment and procedures and changes thereto which received the approval of the Reactor Safety Board and/or Radiological Safety Board and the Head of the Department of Nuclear Engineering. He shall be responsible for the preparation of experimental procedures involving the use of the AGN-201M reactor.

Reactor Operators shall be responsible for the safe manipulation of the reactor controls, monitoring of instrumentation, operation of reactor related equipment, and maintenance of complete and current records during the operation of the facility.

The Reactor Safety Board is responsible for, but not limited to, reviewing and approving safety standards associated with the use of the reactor facility. The Board shall review and approve of all proposed experiments and procedures and changes to either.

The Board must also approve all reactor facility modifications which might affect its safe operation. A complete listing of the Reactor Safety Boards duties and responsibilities is listed in Section 6.1.6 of the Technical Specifications.

The Radiological Safety Board shall advise the University administration and the Radiological Safety Officer on all matters concerning radiological safety at the university facilities.

12.1.3 Staffing

The staffing requirements for the AGN-201M are as follows

a. The minimum operating staff during any time in which the reactor is not shutdown shall 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 the 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 performing that duties stated in paragraph 1 or 2 above or be 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 modifications which could affect the reactivity of the reactor.

12.1.4 Selection and Training of Personnel

The selection of personnel to operate the AGN-201M reactor typically has been restricted to former nuclear navy personnel or graduate students at Texas A&M University. These individuals require less basic training as operators than would other less-qualified personnel. All personnel that operate the AGN-201M reactor, even students, undergo some form of training. The students are lectured on the construction and operations of the reactor prior to student startups.

12.1.5 Radiation Safety

The Radiological Safety Officer at the University is responsible for all aspects of radiological safety. In this facility, the staff will train the student operators on the importance of the ALARA program and what particular aspects of that program can be used at the facility. As started above, all questions and reviews concerning radiological safety are presented directly to the Radiological Safety Board. At Texas A&M University, Radiological Safety personnel have the authority to stop any experiment or activity they believe to be unsafe.

12.2. Review and Audit Activities

The Reactor Safety Board is responsible for reviews and audits at this facility. The review and audit activities of the Board are clearly outlined is Sections 6.4.2 and 6.4.3 of the AGN-201M Technical Specifications. The authority of the Reactor Safety Board comes directly from the office of the President of the University.

The Reactor Safety Board shall review:

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

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

c. Proposed test or experiments which are significantly different from previously approved tests of experiments, or those that involve an unreviewed safety question as defined in 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 effect nuclear safety.

g. Reportable occurrences.

h. Audit reports.

Audits of facility activities shall be performed at least quarterly under the cognizance of the Reactor Safety Board but in no case shall personnel responsible for the item conduct the audit. These audits shall examine the operating records and encompass, but shall not be limited to the following:

a. The conformance of the facility operation to the Technical Specifications and applicable license conditions, at least annually.

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

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

12.2.1 Composition and Qualifications

Reactor Safety Board members shall have a minimum of five (5) years experience in their profession or a baccalaureate degree and two (2) years of professional experience. Members will generally be University faculty with considerable experience in their own field of expertise.

12.2.2 Charter and Rules

The Reactor Safety Broad shall meet at least once per calendar year, or more often as deemed necessary by the Chairman. A quorum for the conduct of official business shall be the chairman, or his designated alternate, and two other regular members. At no time shall the operating organization comprise a voting majority of the members at any Reactor Safety Board meeting.

12.2.3 Review Function

Reviews are performed when either the facility staff has a proposed change to a standing item or when some sort of violation has occurred. The review of standing procedures, Technical Specifications, security and emergency plans, etc. is not required but encouraged since same minor changes and updates occur have all the time.

12.2.4 Audit Function

The audit function at Texas A&M University is limited to members of the Reactor Safety Board, in special cases, audits may be conducted by others, e.g. qualified individuals from the University, consultants ,etc..

12.3 Procedures

There shall be written procedures for the following;

a. Startup, operation and shutdown of the reactor.

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

c. Conduct of irradiations and experiments that could affect the operation or 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 the Technical Specifications.

f. Implementation of the Security and the Emergency Plans.

This list is taken directly from section 6.6 of the Technical Specifications for the AGN-201M.

Any procedure must be submitted to the reactor supervisor for approval and then to the Head of the Department of Nuclear Engineering . Finally it must be submitted to the Reactor Safety Board for approval and signature of the Chairman.

12.4 Required Actions

Section 6.9.2 of the facility Technical Specifications clearly defines what are considered to be reportable occurrences. In all cases of reportable occurrences, the reactor shall be shutdown.

Reportable occurrences, including causes, probable consequences, corrective actions and measures to prevent recurrence, shall be reported to the USNRC. Supplemental reports may be required to fully describe final resolution of the occurrence. In case of corrected or supplemental reports, an amended license 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 shall be reported as expeditiously as possible by telephone and confirmed by mailgram, or facsimile transmission to the Director of the appropriate NRC Regional Office, or his designated representative no later than the first work day following the event, with a written follow-up report within two weeks. Information provided shall contain narrative material to provide a complete explanation of the circumstances surrounding the event.

1. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter has reached the set point specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

2. Operation of the reactor or affected systems when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the technical summary specifications without taking permitted remedial action.
3. Abnormal degradation discovered in a fission product barrier.
4. Reactivity balance anomalies involving:

   (a) Disagreement between expected and actual critical rod positions of approximately 0.3% $\Delta k/k$.

   (b) Exceeding excess reactivity limit.

   (c) Shutdown margin less conservative than specified in technical specifications.

   (d) If sub-critical, an unplanned reactivity insertion of more than approximately 0.5 % $\Delta k/k$ or any unplanned criticality.
5. Failure or malfunction of one (or more) component(s) which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) 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 Safety Analysis Report.
7. Unscheduled conditions arising from natural or man-made events that, as a direct result of the event require reactor shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

8. 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 bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

9. 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 analysis in the Safety Analysis Report or technical specifications bases; 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.

12.5 Reports

Routine annual operating reports shall be submitted no later than ninety (90) days following the end of the operating 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 related to reactor safety that occurred during the reporting period.

(b) Results of major surveillance tests and inspections.

2. A 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 for the shutdowns and corrective actions 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, paragraph 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety question existed and that no technical specification change 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 (summarized for each release)

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

(b) Solid Waste (summarized for each release)

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

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

(3) 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.

If a Safety Limit is violated then the appropriate USNRC Regional Office of Inspection and Enforcement must be contacted, the Director of the USNRC, and the Reactor Safety Board not later than the next working day. In addition a Safety Limit Violation Report shall be prepared for review by the Reactor Safety Board. This report shall be submitted to the USNRC and the Reactor Safety Board within 14 days of the violation.

12.6 Records

Generally record retention falls into two broad categories, records that need to be retained for a period of at least five years and records that are required to be retained for the life of the facility. The records that are retained in this facility are contained in a series of file cabinets, one located inside the reactor room and one next to the reactor supervisors desk.

The following is a list of records to be retained for a period of at least five years, taken from the Technical Specifications section 6.10

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 requirement may be satisfied by the normal operations log book plus,

1. Records of radioactive material transferred from the facility as required by the license.
2. Records required by the Reactor Safety Board for the performance of new or special experiments.

g. Changes to operating procedures.

Records to be Retained for the Life of the Facility:

a. Records of liquid and solid 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 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 Board.

12.7 **Emergency Planning**

The main function of the Emergency Plan is to provide an guide by which the facility staff is able to contain and overcome emergency situations. The plan clearly states the role and response of area emergency services and with the procedures in place the severity of incidents associated with the AGN-201M will be limited .

12.8 Security Planning

The Security Plan associated with the AGN-201M is intended to take full advantage of the isolated nature of the facility inside the Zachry Engineering. The location of the facility in the building is over two floors but with restricted access to the outer doors, access to the reactor room itself is extremely limited.

12.10 Operator Training and Requalification

A copy of the Requalification Program for Licensed Reactor Operators and Senior Reactor Operators for the AGN-201M has been included in this application for license renewal as Appendix C.

12.12 Environmental Reports

An environmental report has been submitted with this license renewal application and is included as Appendix E.

Chapter 13: Accident Analyses

13.1 Accident-Initiating Events and Scenarios

13.1.1 Maximum Hypothetical Accident

Studies have demonstrated the Maximum Hypothetical Accident (MHA) involving an AGN-201M reactor system is a large reactivity addition. This analysis is based upon the assumption of an instantaneous increase of 2%Δk/k reactivity above delayed critical with the reactor at a power level $\leq$ 5 watts. This situation would not occur under normal situations but for the MHA it is being considered. This instantaneous 2% positive reactivity addition is considered to be the maximum creditable accident for the AGN-201M reactor system.

In the past, this scenario has been analyzed using one group theory with one group of delayed neutrons. This analysis was first performed by Aerojet-General Nucleonics and was published in August 1957. This simple analysis assumed that: (1) at time zero a 2% step increase in reactivity was inserted with the reactor at a power level of 100 milliwatt and (2) the energy in the core at time zero was negligible in comparison with the energy liberated during the ensuing excursion. Further, this analysis assumed that there was no energy transfer from the core during the excursion. A complete description of the analysis can be found in the Hazard Summary Report, the results are summarized below.

Numerical solution of the governing equations by finite-difference methods yielded a total energy release of 1.7 MJ and raised the average core temperature by 71 C. This excursion produced a peak power excursion of 54.4 MW at 204 msec after the reactivity insertion. These results demonstrated 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.

A study completed by the University of New Mexico using the same parameters resulted in a total energy release of 2.41 MJ causing a temperature rise of 100.7 C with the temperature at the center of the core rising to a level of 150 C, total dose to a person next to the reactor would be about 1 rem.

Another study completed by the Idaho State University using the same parameters resulted in a total energy release of 5.8 MJ causing a temperature rise of 170 C with the temperature at the center of the core only rising to a level of 120 C. Since this is the point the core thermal fuse would separate and insert a large amount of negative reactivity. Total dose to a person next to the reactor would be about 3.2 rem.

Based upon these data and using the highest area radiation level next to the reactor at 5 watts, the dose would be 18.14 rem using the New Mexico data and 7.54 rem using the Idaho State data. These exposures are quite high but these conditions are not realistic and will never occur.

13.1.2 Insertion of Excess Reactivity

Consider in 13.1.1.

13.1.3 Loss of Coolant

Not considered in the AGN-201M system

13.1.4 Loss of Coolant Flow

Not considered in the AGN-201M system

13.1.5 Mishandling or Malfunction of Fuel

Not considered in the AGN-201M system

13.16 Experiment Malfunction

No explosives or corrosive are permitted in the AGN-201M.

13.1.7 Loss of Normal Electrical Power

In the event of a loss of normal electrical power the reactor will scram and be placed in a shutdown condition.

13.1.8 External Events

A large scale meteorological disturbance would certainly interrupt reactor operations. The reactor would be shutdown and placed in a safe condition, if an unforeseen event occurred the construction of the reactor facility would ensure the safety of personnel and the reactor would be shutdown in a controlled manner.

In the event of a large seismic event, the earthquake switch would activate and shutdown the reactor.

The walls of the facility at ground level are constructed of 3.5 feet of steel reinforced Type II concrete. The rest of the structure in the reactor facility around the reactor room and accelerator laboratory is not, if any large impact were to occur the weaker structure would bear the burnt of the damage.

If a large explosion were to occur the construction of the facility would mitigate most of the damage.

13.1.9 Mishandling or Malfunction of Equipment.

Operator error is a very important consideration during reactor operation. Since the design of the AGN-201M reactor system with all the scrams and associated interlock the likely result of operator error would be a scram. The goal of proper operations is not to let an engineered system scram three reactor because the operator has made a mistake.

The most dangerous error could occur if the operator exceeded 5 watts, on a low period. A scram would occur at 95% of scale on Channel No. 3 so no core damage would result. Since the only 5 watt operations of the AGN-201M reactor occur during reactor maintenance procedures which require controlled reactor operation this event is unlikely to occur.

If reactor period were to go to less than 5 seconds, due to a failure, the reactor interlock system would scram at 95% scale on Channel No. 3, unless the operator changed scales but to keep pace but eventually the reactor would scram would trip on the 0-10 watt scale.

When students operate the reactor, the operation of the Channel No. 3 scale selector switch and how to correctly read the picoammeter seem to be one of the most challenging aspects of reactor operations. Some students have inadvertently scrammed the reactor due to misoperation of the scale selector switch.

If an equipment malfunction were to occur, such as failure of a nuclear instrument the reactor would scram on the interlock relay or the scram system..

13.2 Accident Analysis and Determination of Consequences

(1) In the case of the MHA, initial reactor conditions would be, reactor critical, operating just below 5 watts, shield tank temperature at 21 C. Glory hole empty

(2) The cause would be an instantaneous addition of 2% reactivity to the center of the reactor in the glory hole

(3) Reactor would scram on high power, 95% of 10 watt scale, Channels No. 1 & 2 would be off scale high, all the scram lights would be on but with the release of so much energy in such a short amount of time it is possible the core thermal fuse would melt. If this happens the core will separate adding a negative 5% reactivity to the core which will control the reactivity addition

(4) It has been should that the reactor fuel would not be damaged but since the core separated some major repairs would be required.

(5) With the presence of the core tank to trap any of the gases released from the fuel and the separation of the core no releases to the environment would occur. The radiation levels in the reactor room would be very high, but will decrease rapidly.

13.3 Summary and Conclusions

Many studies have demonstrated with USNRC approval that the MHA for the AGN-201M reactor will not melt the polyethylene which surround the fuel. Consequently an insignificant amount of fission products would be released and the direct radiation would only be a small fraction of the 10 CFR 20 limits.

Chapter 14: TECHNICAL SPECIFICATIONS

The Texas A&M University AGN-201M Technical Specifications in their present format were approved by the NRC on April 25, 1979 as Amendment 12 to NRC License R-23. version submitted with this license renewal is the same set of Technical Specifications with a few minor revisions. At this time, these revisions are being submitted to the NRC for approval.

Chapter 15: FINANCIAL QUALIFICATION

15.2 Financial Ability to Operate a Non-Power Reactor

Texas A&M University for the last 40 years has demonstrated the financial ability to own and operate an AGN-201M reactor. The overhead cost of operation of this facility is small. The reactor supervisor is only supported, 1/2 time in this capacity and the other operating cost of the reactor are quite minimal. Based upon Department of Nuclear Engineering records the operating cost for the AGN-201M in 1996 was about fifteen thousand dollars.

These cost include salary, repair parts and some new equipment for the facility. In the next five years it is estimated that this figure will vary between fifteen and twenty thousand dollars. All operating capital associated with the AGN-201M reactor is provided by the Department of Nuclear Engineering.

The reactor is not used for any commercial work and no plans have been made to change the operating schedule of the reactor.

15.3 Financial Ability to Decommission the Facility

Texas A&M University is a state supported educational institution that receives the majority of its financial support directly from the state. Texas A&M University has pledged to apply to the State of Texas for whatever financial resources are necessary to decommission the AGN-201M. Texas A&M University has filled a letter of intent, dated July 19,1990, with the USNRC concern the decommission funding.

Chapter 16: OTHER LICENSE CONSIDERATIONS

The AGN-201M reactor at Texas A&M University is not used for medical research and no future considerations are being made for this type of work. The components that presently made up all the systems associated with the AGN-201M contain no prior use components from any other reactors since the license was last renewed in August of 1977.

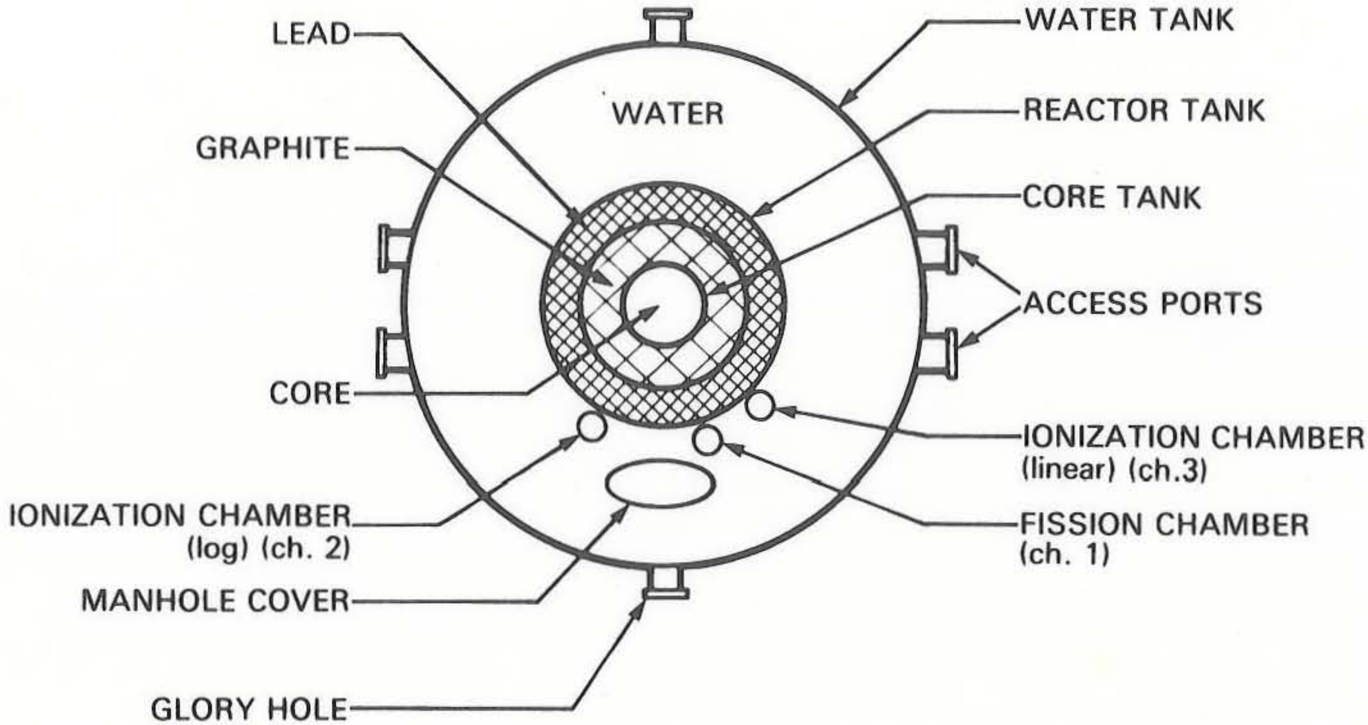


Figure 1.2 Schematic of the reactor (looking from above)

Chapter 17: DECOMMISSIONING AND POSSESSION ONLY LICENSE AMENDMENTS

At the current time Texas A&M University has no plans to decommission to the AGN-201M reactor. When necessary, facility staff will prepare a preliminary decommissioning plan which will be followed by a complete decommissioning plan for the facility. The University responded to the NRC INFORMATION NOTICE N0. 90-16 with a letter to the Document Control Desk stating that since the University is a state institution that, at the time a decommissioning decision is made, the proper funds would be requested from the State of Texas.

APPENDIX B

TECHNICAL SPECIFICATIONS

LICENSE RENEWAL APPLICATION
OF
FACILITY LICENSE R-23
DOCKET# 50-59
AGN201M (serial # 106)

TEXAS A&M UNIVERSITY

APPENDIX A

LICENSE NO. R-23

TECHNICAL SPECIFICATIONS

FOR

TEXAS A&M UNIVERSITY AGN-201M REACTOR (SERIAL #106)

DOCKET NO. 50-59

DATE: April 25, 1979

AS MODIFIED TO INCLUDE ANSI N378-1974 AND

REGULATORY GUIDE 2.2 GUIDANCE

Amendment No. 12 to NRC License R-23
(This copy includes Editorial Corrections as agreed to by NRC, May 24, 1979).

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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 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 - An experiment is any of the following:

a. An activity utilizing the reactor system or its components or the neutrons or radiation generated therein;

b. An evaluation or test of a reactor system operational, surveillance, or maintenance technique;

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

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 exist 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 Co.

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

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

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

1.11 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.

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

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

1.14 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.

1.15 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.

1.16 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.

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

1.18 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, bouyant, or other forces which are normal to the operating environment of the experiment or which might arise as a result of credible malfunctions.

1.19 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.

1.20 Unsecured Experiment - Any experiment or component of an experiment is deemed to be unsecured whenever it is not secured as defined in 1.18 above. Moving parts of experiments are deemed to be unsecured when they are in motion.

2.0 SAFETY LIMITS AND LIMITING 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.

Bases

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

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 120°C or less resulting in core separation and a reactivity loss greater than 5% Δ k.

Bases

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 characteristics 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% Δ 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% Δ k/k.

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

Bases

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 positions.

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 200 milliseconds.

b. The average reactivity addition rate for each control or safety rod shall not exceed 0.065% Δ 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. Nuclear safety channel instrumentation shall be operable in accordance with Table 3.1 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 reactor startup and scram the reactor if the shield water level falls 9.5 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.

TABLE 3.1

Safety Channel	Set Point	Function
Nuclear Safety #1		
Low count rate	$\geq$ 10 cps	scram below 10 cps
Nuclear Safety #2		
High power	$\leq$ 10 watt	scram at power >10 watt
Low power	$\geq 1.0 \times 10^{-12}$ amps	scram at source levels $< 1.0 \times 10^{-12}$ amps
Reactor period	$\geq$ 5 sec	scram at periods <5 sec
Nuclear Safety #3 (Linear Power)		
High Power	$\leq$ 10 watt	scram at power > 10 watt
Low power	$\geq$ 5% full scale	scram at source levels <5% of full scale
Manual scram		scram at operator option

3.4 RADIATION MONITORING, CONTROL, AND SHIELDING

Applicability

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

Objective

The objective is to protect facility personnel and the public from radiation exposure.

Specification

a. An operable portable 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 and accelerator room shall be considered restricted areas whenever the reactor is not shutdown.

c. The accelerator room shall be considered a radiation area whenever the reactor is not shutdown.

d. The reactor [redacted] shall be considered a radiation area whenever the reactor is operated at a power level less than 0.9 watts.

e. Whenever the reactor is operated at a power level equal to or greater than 0.9 watts the reactor room shall be considered a high radiation area and the reactor room entrance shall be equipped with a control device which shall energize a conspicuous visible and audible alarm signal in such manner that the individual entering the reactor room and the reactor operator are made aware of the entry, or the reactor room entrance shall be maintained locked except during periods' when access to the area is required, with positive control over each entry.

f. 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 other approved experiments requiring the thermal column to be empty.

Bases

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, is less than 100 mrem/hr at reactor power levels less than 1.0 watt, and that the total gamma, thermal neutron, and fast neutron dose rate in the accelerator room is less than 15 mrem/hr at reactor power levels less than or equal to 5.0 watts and the thermal column filled with water.

The facility shielding in conjunction with radiation monitoring, control, and restricted 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

This 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 worth of an experiment shall be estimated or measured, as appropriate, before or during the first startup subsequent to the experiment's insertion.

Bases

The control and safety rod reactivity worths are 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 that 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 Systems Applicability

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

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 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 reactor operation extending more than one day:

Nuclear Safety #1, #2, and #3

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 reactor operation or prior to each reactor operation extending more than one day, safety rod #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, but at intervals not to exceed 16 months.

h. The shield water level interlock and shield water temperature interlock shall be calibrated by perturbing the sensing element to the appropriate set point. These calibrations shall be performed annually, but at intervals riot to exceed 16 months.

Bases

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

The objective is 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.

Bases

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 and high radiation areas within the reactor facility are identified and controlled as required by Specification 3.4.

Specification

a. All portable radiation survey instruments assigned to the reactor facility shall be calibrated under the supervision of the Radiological Safety Office 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 area alarm (Ref. 3,4e) shall be verified to be operable.

c. A radiation survey of the reactor room, reactor control room, and accelerator room shall be performed under the supervision of the Radiological Safety Office 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.

Bases

The periodic calibration of radiation monitoring equipment and the surveillance of the reactor room high radiation area alarm (Ref. 3.4e) 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 660 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 a fuse temperature of about 120°C causing the lower core section to fall away from the upper section reducing reactivity by at least 5 Δ 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 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 a thermal column when filled with graphite.

d. The 6½ 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 personnel in unrestricted areas to levels less than permitted by 10 CFR 20 under operating conditions.

e. Two safety rods and one control rod (identical in size) contain less than 15 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. Deenergizing 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 nuclear engineering department 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, Reactor Control Room, Accelerator Room

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

b. The reactor control room houses the reactor control console.

c. The accelerator room is directly above the reactor room and a hole in the accelerator room floor provides access to the thermal column.

d. The reactor room, reactor control room, and accelerator room are separate rooms in the Zachry Engineering Center, 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.

e. The access doors to the reactor room, reactor control room, and accelerator room shall contain locks.

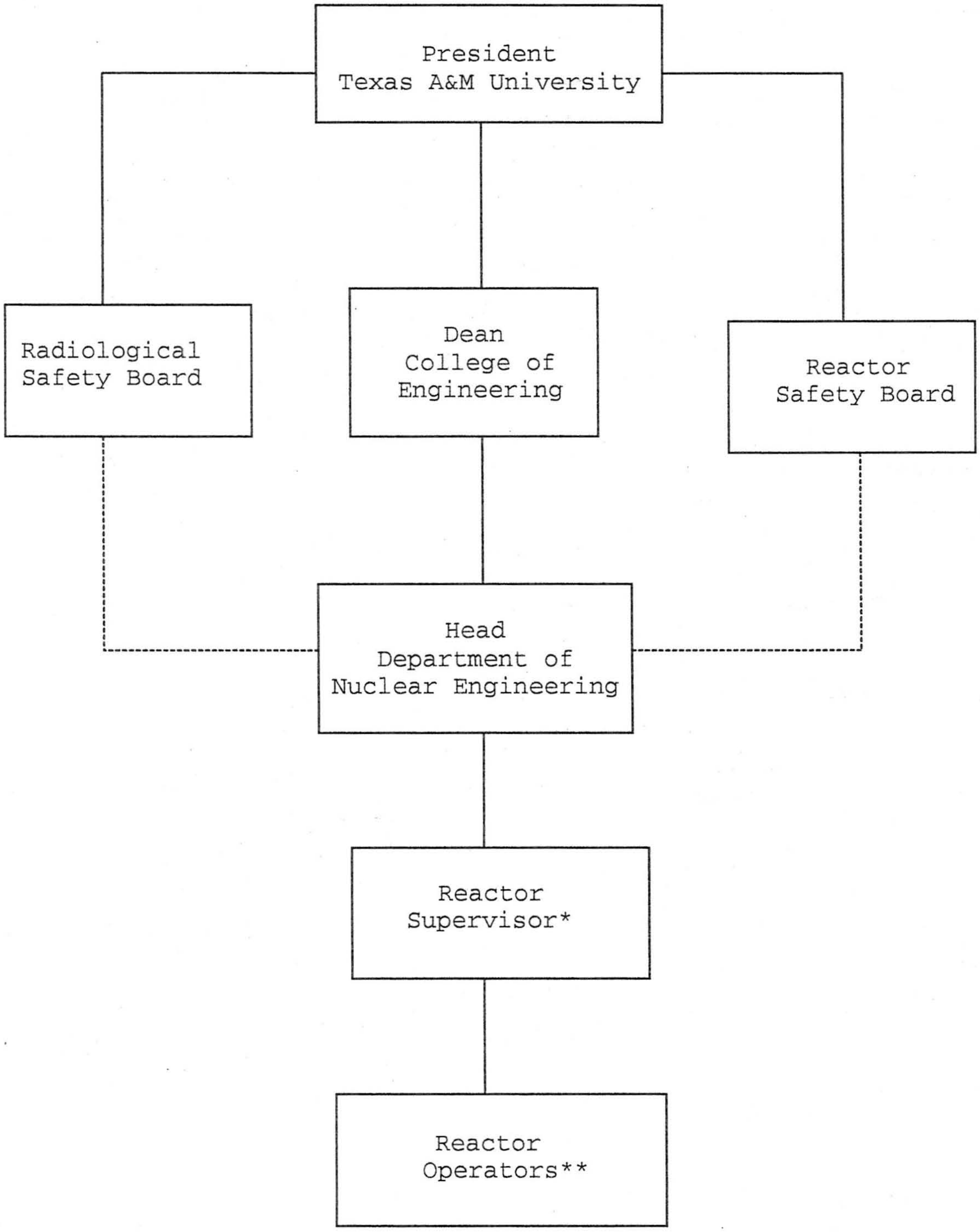
6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

The administrative organization for control of the reactor facility and its operation shall be as set forth in Figure 1 attached hereto. 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.

FIGURE 1

Administrative Organization of the Texas A&M University AGN-201M Reactor Facility NRC License R-23



* Requires NRC Senior Operators License

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

6.1.1 President

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

6.1.3 Head, Department of Nuclear Engineering

The Head of the Department of Nuclear Engineering is the administrative officer responsible for the operation of the Department of Nuclear Engineering, including the AGN-201M Reactor Facility. In this capacity be 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 Radiological Safety Board and/or the Reactor Safety Board 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 Board.

6.1.4 Reactor Supervisor

The Reactor Supervisor shall be a licensed SRO and shall be responsible for the preparation, promulgation, and enforcement of administration 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 Board and/or the Radiological Safety Committee and the Head of the Department of Nuclear Engineering; and be responsible for the preparation of experimental procedures involving use of the reactor.

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 Board

The Reactor Safety Board 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 Head of the Department of Nuclear Engineering.

6.1.7 Radiological Safety Board

The Radiological Safety Board shall advise the University administration and the Radiological Safety Officer on all matters concerning radiological safety at university facilities.

6.1.8 Radiological Safety Officer

The Radiological 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 Head of the Department of Nuclear Engineering. He shall provide personnel dosimetry and keep records of personnel radiation exposure. He shall advise the Head of the Department of Nuclear Engineering on all matters concerning radiological safety at the reactor facility. The Radiological Safety Officer shall be an ex officio member of the Reactor Safety Board.

6.1.9 Operating Staff

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

6.2 Staff Qualifications

The Head of the Department of Nuclear Engineering, 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 Board members shall have a minimum of five (5) years experience in their profession or a baccalaureate degree and two (2) years of professional experience. Reactor Safety Board members will generally be University faculty members with considerable experience in their area of expertise. The Radiological Safety Officer shall have a baccalaureate degree in biological or physical science and have at least two (2) years experience in health physics.

6.3 Training

The Head of the Department of Nuclear Engineering shall be responsible for directing training as set forth in 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 Board

6.4.1 Meetings and Quorum

The Reactor Safety Board shall meet as often as deemed necessary by the Reactor Safety Board Chairman 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 Board meeting.

6.4.2 Reviews

The Reactor Safety Board 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 previous approved tests or experiments, or those that involve an unreviewed safety question as defined in 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 at least quarterly under the cognizance of the Reactor Safety Board 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 the facility operation to the Technical Specifications and applicable license conditions, at least annually.

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

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

6.4.4 Authority

The Reactor Safety Board shall report to the President and shall advise the Head of the Department of Nuclear Engineering on those areas of responsibility outlined in section 6.1.6 of these Technical Specifications.

6.4.5 Minutes of the Reactor Safety Board

The Chairman of the Reactor Safety Board 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 Approvals

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

a. The Reactor Supervisor shall prepare the proposal for review and approval by the Head of the Department of Nuclear Engineering.

b. The Head of the Department of Nuclear Engineering shall submit the proposal to the Chairman of the Reactor Safety Board.

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

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

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 operation or 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 Head of the Department of Nuclear Engineering and the Reactor Safety Board. Temporary procedures which do not change the intent of previously approved procedures and which do not involve any unreviewed safety question may be employed t 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 Head of the Department of Nuclear Engineering and the Reactor Safety Board

b. Approved experiments shall only be performed under the cognizance of the Head of the Department of Nuclear Engineering and 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 Board not later than the next work day.

c. A Safety Limit Violation Report shall be prepared for review by the Reactor Safety Board. 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 Board within 14 days of the violation.

6.9 Reporting Requirements

In addition 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 annual operating reports shall be submitted no later than ninety (90) days following the end of the operating 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 related to reactor safety that occurred during the reporting period.

   (b) Results of major surveillance tests and inspections.

2. A 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, paragraph 50.59 which clearly shows the reason leading to the conclusion that no unreviewed safety question existed and that no technical specification change 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 (summarized for each release)

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

   (b) Solid Waste (summarized for each release)

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

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

      (3) 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. 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 shall be reported as expeditiously as possible by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the appropriate NRC Regional Office, or his designated representative no later than the first work 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 or other systems 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 or failure to complete the required protective function.

2. Operation of the reactor or affected systems when any parameter or operation subject to a limiting condition is less conservative than the limiting condition for operation established in the technical specifications without taking permitted remedial action.

3. Abnormal degradation discovered in a fission product barrier.

4. Reactivity balance anomalies involving:

   (a) Disagreement between expected and actual critical rod positions of approximately 0.3% Δ k/k.

   (b) Exceeding excess reactivity limit,

   (c) Shutdown margin less conservative than specified in technical specifications.

   (d) If sub-critical, an unplanned reactivity insertion of more than approximately 0.5% Δ k/k or any unplanned criticality.

5. Failure or malfunction of one (or more) component(s) which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) 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 Safety Analysis Report.

7. Unscheduled conditions arising from natural or man-made events that, as a direct result of the event require reactor shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

8. 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 bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

9. 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 analysis in the Safety Analysis Report or technical specifications bases; 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.

6.9.3 Special Reports

Special reports which may be required by the Nuclear Regulatory Commission shall be submitted to the Director of the appropriate NRC Regional Office within the time period specified for each report.

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 requirement 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 Board 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. Records of liquid and solid 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.

C 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 Board.

APPENDIX F

ENVIRONMENTAL REPORT

LICENSE RENEWAL APPLICATION
OF
FACILITY LICENSE R-23
DOCKET# 50-59
AGN201M (serial#106)

TEXAS A&M UNIVERSITY

ENVIRONMENTAL REPORT

CONTENTS

1.0 INTRODUCTION

The Texas University AGN-201M has been operated by the Department of Nuclear Engineering in the Collage of Engineering for nearly 40 years. The reactors operating license R-23 was first issued by the United States Atomic Energy Commission in 1957. The AGN-201M has been operating in the reactor facility in the Zachry Engineering Center since 1972. The reactor is currently used both as a research device and as a training center for students in the Department of Nuclear Engineering and for facility reactor operators. The reactor is used in laboratories which are directly associated with classes taught in the Department of Nuclear Engineering, the reactor is used to irradiate small samples for later analysis in counting laboratories and for conducting for reactor operations for the training of students.

The reactor room is located in room 61B of the Zachry Engineering Center. The reactor room is designed so that it can only be entered through a double set of doors that are built in a concrete block wall. This door has an electromechanical lock with combination and is located in a concrete block wall in the middle of what once was one large room. The exterior walls and ceiling of the reactor room and room 61A are constructed of 3.5 and 4.0 thick concrete, respectively. The reactor room, 61B has the following dimensions , 12 foot high walls and a floor that measures 29 feet by 25 feet.

The AGN-201M reactor is a self-contained, thermal reactor with a compact core of <20% enriched uranium oxide homogeneously dispersed in a matrix of low-density polyethylene. The reactor core is contained in a hermetically sealed aluminum can that isolates the reactor fuel and prevents the escape of fission-product gases to the environment. Since the reactor has such a low thermal power no external means are required to cool 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 1000 gallons of water contained in the shield tank. In addition to serving as a fast neutron shield, the water acts as a massive thermal sink. Hence, there are no thermal or radioactive liquid effluents released from the facility.

Since 1973 this reactor has been operated with a maximum rated power level of 5 watts. All shielding surveys in the area of the reactor and in the accelerator laboratory above indicate extremely low radiation levels except when operating at higher power levels. The highest radiation levels have been noted in the area around the rear skirt door, this local area has been posted as a high radiation level during 5 watt reactor operations and all personnel in the reactor facility are briefed on the areas of highest possible exposure. It should be noted that the reactor is only operated at higher power levels during required reactor maintenance procedures and not during normal reactor operations.

A radiation protection program is administered by Radiological Safety Division of Texas A&M University's Environmental Health and Safety Office. The University Radiation Safety Officer heads up the office of Radiological Safety. This office provides administrative and technical support for all users of radioactive materials at TAMU. All personnel assigned to the reactor facility and the Head of the Department of Nuclear Engineering have been issued individual radiation dosimeters which measure equivalent dose from beta, gamma and neutron radiation. No visitors are allowed in the reactor facility during reactor operations and students working in the reactor facility are issued self-reading pocket dosimeters. Two environmental monitoring film badges are placed in areas inside and outside the reactor room to measure cumulative exposure to ionizing radiation. These environmental monitors, as well as personnel dosimeter devices, are analyzed on a monthly basis.

2.0 RECENT POWER HISTORY

The AGN-201M reactor has had limited power history in the last few years. This has been primarily due to material problems associated with the reactor channel #2 drawer. The reactor has been operated for many years in direct support of several Nuclear Engineering classes and for operator proficiency. The reactor is mostly operated during the start of both the fall and spring semesters. Students perform reactor startups for both graduate and undergraduate reactor laboratories and other experiments are conducted during these times using the reactor facility.

Table 1. Average annual values for AGN-201 reactor operations.*

Annual Operations	28.92 Hours
Energy Generation	10.25 watt-hours
Mean Power Level	0.666 watts

* Data from Annual Operating Reports from May 90 - May 96, May 94-May 95 not used due to very limited reactor operations during this time period.

3.0 GASEOUS EFFLUENT: ARGON-41 PRODUCTION

With the design of the AGN-201M reactor core and fuel, the production of argon-41 is considered to be the only gaseous effluent of concern. Since argon-40 is a minor constituent of dry air, 0.93 percent by volume or 1.3 percent by mass, the production Ar-41 should be at least considered. Argon-41 is produced by the neutron capture in argon-40. The only air volumes that might be effected are those in the glory hole or inside one of the access ports. The access ports are filled with wood , lead and graphite under normal operating conditions. Air is also contained in solution in shield tank but this tank is sealed from the outside environment. Other considerations are that the flux level in the shield tank is much lower than in the areas closer to the core and that the argon concentration in the tank will be very small. In order to reach saturation activity at the highest possible concentration levels the air in question would have to be irradiated for at least seven half-lives, about 13 hours at maximum rated power. The AGN-201M reactor has not been operated at full power for this length of time in the past and such reactor operations are very unlikely in the future.

With the small amount of air actually exposed in the sealed glory hole, the low reactor power history and the large amount of airflow thought the laboratory the production levels of argon-41 and release levels are very minuscule. The reactor facility has its own discharge flow path through the accelerator laboratory and out a discharge vent on the roof of the four-story Engineering Building.

4.0 OTHER EFFLUENTS

No liquid or solid radioactive effluents are generated by the AGN-201M reactor facility. The only radioactive material produced is that used in irradiation experiments and all this material remains within the Department of Nuclear Engineering laboratories or is disposed of by Radiological Safety as solid waste. If any material were irradiated for use other than in the department, Radiological Safety would have to transfer the material to a sub-licensee. With the presence of a TRIGA reactor at the University, no outside departmental irradiation work takes place in the AGN reactor facility.

Some levels of reactor structure and component activation have been experienced during decommissioning of this type of reactor but that has been limited to components that have been in direct contact with the uranium-impregnated polyethylene fuel disks. Some contaminated surfaces were also found but were limited to the inner surface of the core, in-core graphite reflector and the interior surfaces of the aluminum clads of the safety and control rods. Since this will not be of concern until decommissioning and is already a recognized problem, measures will be taken at that time to prevent any possible spread of contamination.

5.0 ENVIRONMENTAL MONITORING

All environmental monitoring is done by the Radiological Safety personnel. A radiation survey of the reactor facility and surround laboratories is performed an annual basis. These annual survey are done at the same points every year and are compared to previous radiation surveys. With the thickness of both the ceiling and the surrounding walls of the reactor room and the accelerator room, only very low levels of radiation can be detected outside the reactor room even during full power operations. Contamination surveys of the reactor room and adjoining laboratory are currently performed on a monthly basis by Radiological Safety personnel.

Two environmental dosimeters are currently in place in areas in and around the reactor room to measure accumulated equivalent doses. One is located in the reactor room itself, another is in the adjoining laboratory. These dosimeters are film badges and plastic tract-etch (CR-39 or equivalent) for detection of beta, gamma and neutron radiation.

6.0 CONCLUSIONS

No events have occurred or maintenance items performed which might in any way have affected the physical barriers between the reactor core and fuel and the local environment. All contamination surveys have indicated no loose surface contamination in the reactor room and the annual radiation surveys have verified the effectiveness of the shielding at all reactor power levels. Any and all material associated with experiments performed with the AGN-201 remain within the confines of the departmental laboratories, none are transferred to other facilities. If material were to be transferred the personnel of Radiological Safety would make all the necessary arrangements and transfer the material, since departmental personnel are not authorized to make such transfers of material.

The benefits of continuing operation of the TAMU AGN-201 to the current and future students in the Department of Nuclear Engineering far outweigh any minor negative environmental effects. The reactor by its design considerations and current location will pose no significant environmental impact.

APPENDIX C

OPERATOR LICENSING AND REQUALIFICATION PROGRAM

LICENSE RENEWAL APPLICATION OF FACILITY LICENSE R-23 DOCKET# 50-59 AGN201M (serial#106)

TEXAS A&M UNIVERSITY

REQUALIFICATION PROGRAM FOR LICENSED REACTOR OPERATORS

AND SENIOR REACTOR OPERATORS

TEXAS A&M UNIVERSITY

AGN-201M REACTOR - LICENSE R-23

Approved: 1 June 1976
(Modified 20 May 1988 to incorporate changes to 10 CFR part 55 effective 26 May 1987)

Evelyn M. Breiner

Evelyn M. Breiner
Reactor Supervisor

John W. Poston

John W. Poston
Department Head

AGN-201M BIENNIAL REQUALIFICATION PROGRAM

A. Each licensed operator and senior operator will be required to operate the reactor or otherwise perform his duties as a licensed operator or senior operator for a minimum of four hours per calendar quarter. Each licensed operator must perform the following control manipulations or demonstrate knowledge of actions to be taken, as appropriate, annually: reactor startup, reactor shutdown, reactor power change of 10 percent or more, and a loss of electrical power. Each licensed operator must demonstrate knowledge of actions to be taken during the following transients biennially: inability to drive control rods, reactor trip, and a nuclear instrumentation failure. Senior operators may be credited with these activities if they direct control manipulations as they are performed.

B. Each licensed operator and senior operator will be required to take an annual operating examination to establish the operator's ability in the following categories: manipulations of console controls; identification of annunciators and any remedial actions required; identify instrumentation systems and their significance; observe and control the operating characteristics of the facility; perform control manipulations during normal, abnormal, and emergency situations; describe the use of the facility's radiation monitoring systems; demonstrate knowledge of radiation hazards; demonstrate knowledge of the facility emergency plan; demonstrate ability to perform his function in the emergency plan; and demonstrate ability to perform his function as a member of the control room team.

C. Each licensed operator and senior operator will be required to take an annual written examination over the elements listed in 10 CFR 55.41 and 55.43. Any licensed person scoring less than 70% on the annual written examination will be removed from his licensed duties until he has corrected the deficiency. Any licensed person scoring less than 80% in any category on the annual written examination must be tutored to bring his performance up to a minimum of 80% overall.

D. Each licensed operator and senior operator shall participate in an accelerated requalification program where performance evaluations conducted pursuant to A, B or C above clearly indicate the need. A licensed individual preparing and grading any portion of the annual written examination is exempt from taking the written examination in that category.

E. All operators and senior operators must have a physical examination performed once every two years. The results of the physical examinations will be maintained in the operator's training file.

F. Individuals who maintain operator or senior operator licenses for the purpose of providing backup capability to the operating staff shall participate in this requalification program except to the extent that their normal duties preclude the need for retraining in particular areas. All licensed operators who do not participate in the Requalification Program, because of other duties, will have their names removed from the list of authorized operators. An operator will then be required to satisfactorily complete an accelerated Requalification Program in order to be listed on the authorized operator's list. The accelerated program will include at least six hours of experience with the reactor. Prior to placing an operator back on an authorized list, the NRC will be notified by letter indicating that the operator has satisfactorily completed a retraining program in accordance with 10 CFR 55.

G. A training file shall be maintained on each licensed operator and senior operator to document his participation in the Requalification Program. Each file shall include copies of written examinations administered, the answers given by the licensees (as corrected), results of evaluations, documentation of any additional or accelerated training administered in areas in which an operator or senior operator has exhibited deficiencies, and a copy of the reactor operating log documenting the required reactivity manipulations. Documentation of any accelerated training must be signed by the Head of the Department of Nuclear Engineering.

Requalification Record Year _____ Quarter _____ Page # ___

Operator License #					
Date Log # Time					
Date Log # Time					
Date Log # Time					
Date Log # Time					
Date Log # Time					
Date Log # Time					
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TOTAL					

AGN-201M REACTOR FACILITY

QUARTERLY REACTOR MANIPULATIONS

19___

		JAN-MAR	APR-JUN	JUL-SEP	OCT-DEC
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg
Name Lic #	Eff Date	Date Log Pg	Date Log Pg	Date Log Pg	Date Log Pg

COMMENTS:

Part 3 REQUALIFICATION EXAM SCORES

Date of Exam: / /

OPERATOR

TITLE								
A) Theory and Principles of Operation								
B) General and Specific Plant Operating Characteristics								
C) Plant Instrumentation and Control Systems								
D) Plant Protection Systems								
E) Engineered Safety Systems								
F) Normal, Abnormal and Emergency Operating Procedures								
G) Radiation Control and Safety								
H) Technical Specifications								
I) Applicable Portions of Title 10, Chapter I Code of Federal Regulations								

COMMENTS:

Part 2 REQUALIFICATION LECTURE ATTENDANCE

TOPIC / OPERATOR								
A) Theory and Principles of Operation								
B) General and Specific Plant Operating Characteristics								
C) Plant Instrumentation and Control Systems								
D) Plant Protection Systems								
E) Engineered Safety Systems								
F) Normal, Abnormal and Emergency Operating Procedures								
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COMMENTS: