ANNUAL REPORT
JANUARY 1, 1987-DECEMBER 31, 1987
ILLINOIS ADVANCED TRIGA
FACILITY LICENSE R-115

I. SUMMARY OF OPERATING EXPERIENCE

A. Summary of Usage

The reactor was scheduled for use 28.8 hours per week and was in operation 23.2 hours per week. The increases in scheduled time (52%) and operation time (74%) are primarily due to the addition of a faculty member who has a strong interest in Neutron Activation Analysis (NAA), increased use of the Thermal Column for neutron scatter research and increased use of the reactor as a result of the completion of re-organization of the Nuclear Engineering Laboratory Courses. In the following table, the per cent of time for different activities is given. Scheduled time is that time reserved for a given operation and it includes scheduling more than one reactor facility for use at the same time while operating time is from start-up to shutdown for all the scheduled activities.

CATEGORY	SCHEDULED	OPERATING
Research Projects	19.7%	11.3%
Irradiations (samples)	48.9%	59.2%
Education and Training	22.2%	21.8%
Maintenance and Measurements	9.2%	7.7%

Presently there are three individuals with Senior Operator Licenses and one individual with an Operators License. The facility operates with a 40 hour week schedule and a staff of two and a half full time equilalent operators and one full time reactor health physicist.

B. Performance Characteristics

1. Fuel Element Length and Diameter Measurements

These checks were made on the B & C Hexagonals during the month of March. The pulse number at the time of the checks was 8661. For the eighteen elements in this region, there was a slight decrease in the length (1.8 mils). The accuracy of a given measurement is estimated at ± 5 mils. There was no change in the diameter of the fuel elements checked.

These checks were made on the C through G Hexagonals in August. The Pulse number at the time of the checks was 8713. For the eighty three elements in this region, there was a slight decrease in the length (1.4 mils). The accuracy of a given measurement is as stated above. There was no change in the diameter of the fuel elements checked.

There were 301 pulses in 1987, bringing the total since 1969 to 8863. The values for pulse height, reactor period and fuel temperature were the same as measured in previous years.

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2. Reactivity

Control Rods: The measured reactivity values of the control rods have shown essentially no change. Variations between successive measurements are seldom greater than 5%.

Core Reactivity: The loss of reactivity, attributed to fuel burnup was $0.66 for the year. This value was determined by a comparison of the cold critical xenon-free control rod position at the beginning and at the end of the year. This determination takes into account the fact that one fuel element was added to the core in September to recover some reactivity lost due to burnup and to offset the reactivity lost due to the introduction of a Neutron Activation Tube into the core (see Section V). Based on an estimated 2 (±0.5) cents per MW-day of operation, the loss of reactivity would have bren $ 0.51.

II. TABULATION OF ENERGY AND PULSING

A. Hours Critical* and Energy

Type of Operation	Time (hrs)	Energy (MW-hrs)
0-10 kW	286.6	0.03
10kW-250kW	114.4	19.22
250kW-1.5MW	691.5	590.75
Pulsing	115.0	1.86
Total	1207.5	611.86

B. Pulsing

Pulse Size	Number
$1.00-1.70	4
1.71-2.00	46
2.01-2.30	0
2.31-2.90	19
2.91-3.19	232
Above $3.19	0
Total	301

*Because of the type of operation, the Hours Critical time includes instances where the reactor is not critical in the normal sense. These include the time to get critical during a start-up, the time between pulses during continuous pulsed operation and short periods of time during sample irradiations when samples may be removed or added.

III. REACTOR SCRAMS

There were 36 unplanned scrams and no emergency shutdowns during this time period. These scrams were attributed to Instrument Malfunction (14), Operator Error (17) and External Causes (5). This is about average for the facility over the years.

Linear Power (13)

This is a power level scram requirod by the Technical Specifications. It occurs when the signal on any power range exceeds about 108% of that range. Nine (9) of these scrams were due to electronic noise problems. In all cases the noise was generated when the Mode Switch was moved from the Automatic position to the Steady State position. The Mode switch is cleaned periodically and old electrolytic capacitors are replaced as they are identified. One scram was caused by too large a control rod movement while the operator was balancing control rods at full power. One scram occurred because the grounding strap on the pulsing power (nv) circuit had worked itself loose. Two (2) scrams occurred due to an operator or operator trainee turning the range switch the wrong direction.

Period Scram (9)

This scram is not required by the Technical Specifications. It occurs when the period is 3 seconds or less with the Mode Selector Switch in Automatic or Steady State position. All of these scrams (9) occurred either when the period circuit was placed in operation at too low a power level or when the true power level was masked by high gamma current and therefore the period limiter circuit could not effectively drive the Regulating Rod in to prevent the Period Scram. Adjustment of the Log-N Channel is not practical after every shutdown. This scram is usually caused by operator trainees or student operators.

Fuel Temperature (3)

These scrams are caused by RF signals from CB transmitters being used near the Nuclear Reactor Laboratory. When the reactor is at high power levels the RF signals are large enough to cause a fluctuation in the temperature indication circuit which can cause the channel reading to exceed the scram set point.

Loss of Power (2)

One loss of power scram was caused by a momentary interruption of electrical power to the building due to an electrical storm. The other loss of power scram resulted from an unrelated piece of equipment experiencing a short circuit and tripping the breaker which supplies electrical power to the TRIGA Control Console.

Primary Flow (7)

Three of these scrams occurred due to the operator leaving the fill switch to the Cooling Towers closed. For long operations this can lead to a low water level in the Cooling Towers which will trip the Secondary Pump, which in turn will trip the Primary Pump. If Log-N power has exceeded 1 MW (1 MW Bistable has actuated) and the Primary Pump trips, then the reactor will scram. Since the 1 MW Bistable does not reset until power drops below 100 kW, it is possible for this scram to occur between 100 kW and 1 MW if, the Primary Pump is tripped. Two scrams occurred due to the operator failing to establish Primary Coolant flow before going to

1 MW. Two scrams occurred when the Secondary Flow switch sensed a loss of flow due to an air slug entering the Secondary System in the Cooling Towers.

Percent Power (2)

Two Percent Power scrams occurred this year. Both were caused by electronic noise spikes similar to those discussed under Linear Power scrams.

IV. Maintenance

It is estimated that about 480 hours (40 hours per month) were spent on maintenance-related activities. However only 71 of these hours are reflected in the Summary of Operations. These hours account for time when normally scheduled activities could not be carried out due to the need to make necessary repairs to the reactor system and for that time when the reactor was needed to perform surveillance activities. The significant items of maintenance are given below.

Bulk Water Temperature Channel: A new bistable was designed and built for the Bulk Water Temperature Alarm circuit. This design utilizes more dependable electronic components. In addition a sonalert alarm was substituted for the electric alarm buzzer. During testing, it was found that this buzzer generated noise spikes which would cause Linear Power Scrams upon actuating. Power supplies for the channel were re-configured, the Water Temperature display switch was replaced and a capacitor upgrade was performed on associated circuits. These items were reviewed by the Nuclear Reactor Committee. This item is not specified in the facility Technical Specifications. The channel still functions as described in the Safety Analysis Report (SAR), but its performance is considered superior.

Count Rate Channel: The Count Rate Switch was replaced. The old switch was worn and would open intermittently.

Automatic Control Circuit: The Automatic Control Circuit was overhauled. This overhaul was prompted by the need to eliminate electronic noise from the control console. A capacitor upgrade reduced the problem but did not eliminate it.

High Voltage Power Supply: The High Voltage Power Supply for the TRIGA neutron monitoring chambers was overhauled. This included a capacitor and diode up-grade. This corrected the electronic noise problem in the control console.

Stack Exhaust Monitor: The power supply for the Stack Exhaust Monitor was modified so that counts would not be lost if the monitor transferred from a.c. power to the battery back-up in the event of an a.c. power failure.

V. Conditions Under Section 50.59 of 10 CFR

In August USNRC granted the amendment to the facility Technical Specifications that was described in last years annual report. This amendment updated the Administrative Section of the Technical Specifications to include current reporting requirements, deleted references to USAEC and inserted USNRC in their place and altered the Administrative Section to correspond to changes brought about by the elevation of the Nuclear Engineering Program to Departmental status. These changes were then incorporated into the Physical Security Plan and forwarded to USNRC appropriately. The changes had been previously incorporated into the Emergency Plan and forwarded to USNRC. Both submissions were accepted on the basis that the changes did not weaken either Plan.

One new experiment was approved this year. This experiment is similar to other Nuclear Pumped Laser experiments which have been approved in the past. The experiment involves placing a gas filled cell in the Thru Beam Port, using neutrons from the reactor to cause the gas to fluoresce. In turn, this light is used to drive a Nd:YAG Laser located outside of the Thru Beam Port. The light from the Nd:YAG Laser is then investigated with an optical multichannel analyzer.

With this experiment, evaluations of radiation hazards, procedures for making changes to the experiment, and shielding requirements and personnel safety involving the use of lasers were reviewed and found to be satisfactory.

One change to the facility was reviewed under the provisions of 10CFR50.59. This change involved the design, construction and testing of a new irradiation facility which can be inserted into the reactor core in one of the two removable three element clusters near the edge of the reactor core. These removable three element clusters were included in the original design to provide for additional irradiation facilities, as needed.

This irradiation facility, called the Neutron Activation Tube (NAT) occupies part of a position in the F hexagonal and part of two adjacent positions in the G hexagonal. The NAT consists of an aluminum tube 2 1/8 inches in diameter and 5 1/2 feet long. The internal volume of the NAT is about 3.8 liters. It is weighted with lead so that it is negatively buoyant. The NAT is equipped with a vent connection and a relief valve, which relieves at a differential pressure of 1/2 psi. Before the relief valve was installed the NAT was pressurized to slightly greater than one atmosphere gauge without leakage.

The NAT is inserted about 2/3 of the way into the core and is supported vertically from above and radially by a carriage that is bolted to the upper grid plate. The carriage is equipped with a "trap door" which is closed when the reactor in operated with the NAT removed. This "trap door" minimizes perturbation of the coolant flow through the reactor core. The NAT is locked in place and slowly rotated to assure uniform fluence to samples. The reactivity worth of the NAT is less than $0.05 when compared to water. When the NAT is rotated no perturbation of power level is observed.

Upon review of the design of the NAT and a cadmium lined NAT (CLNAT), the Nuclear Reactor Committee approved testing of the NAT on the basis that it did not change the facility Technical Specifications nor did it constitute an unreviewed safety question as defined in 10CFR50.59(2).

The Nuclear Reactor Committee approved a testing procedure for the NAT but tabled testing of the CLNAT until the NAT was fully tested. Testing of the NAT is in it final stages. When it is complete, a full report of the design and testing of the NAT will be forwarded to USNRC.

VI. Release of Radioactive Materials

Argon-41:	Average concentration to environs via exhaust= 1.04 E-7 uCi/ml Total release= 3360 mCi Monthly range= 190-371 mCi
Tritium:	Estimation of 1.4 mCi release from the evaporation of water in the reactor tank. This is based on the measured concentration of H-3 and water usage for the year. Note: Water usage was high in 1987 because water was allowed to evaporate from the LOPRA tank on several occasions to facilitate some maintenance in the LOPRA System.
Effluent (sanitary sewer):	Less than 5.2 uCi.

VII. Environmental Surveys

There were no environmental surveys performed in 1987 other than routine radiological monitoring. Contamination surveys are performed in the Laboratory. See Section VIII.

VIII. Personnel Radiation Exposure and Surveys Within the Facility

A. Personnel Exposure

Twenty-two persons were assigned film badges at the facility. Three were full time employees, while the others averaged less than twenty hours per week in the facility. The badges were sent to Radiation Detection Company of Sunnyvale California on a monthly basis through June 30.1987. Effective July 1, 1987 they are being sent to R.S. Landauer of Glenwood Illinois. The table below gives the whole body dose equivalent received by those who were assigned film badges during 1987.

Dose Equivalent (REMS)		Number of Individuals
No Measurable Exposure		3
0.01-0.10		11
0.10-0.25		6
Above 0.25		2
	Total	22

The highest individual dose equivalent was 370 millirems. This was received by the Reactor Health Physicist who handles radioisotopes which are produced in the reactor and calibrates radiation monitoring equipment. Seven other individuals received a dose equivalent above 100 millirems. They received this dose equivalent as a result of handling radioisotopes and/or special experimental apparatus. Students and visitors doses are recorded on self-reading dosimeters and were less than 10 millirems.

B.Contamination Surveys

Smear samples from various locations around the laboratory are taken at periodic intervals. The removable contamination is determined by counting the smears with a gas flow proportional counter.

The maximum contamination is usually found in the vicinity where the irradiated sample containers are handled. There were 6,642 samples irradiated during the year. In the sample area the contamination varied from 2-47,600 dpm/100 cm^2 or 9.0E-09 to 2.1E-4 uCi/cm^2. In the control room area the maximum was 100 dpm/100 cm^2 or 4.5E-07 uCi/cm^2. Smears from other areas of the laboratory showed a maximum of 2,714 dpm/100 cm^2 or 1.2E-05 uCi/cm^2. Clean up of these areas always results in levels of less than 1000 dpm/100 cm^2 or 4.5E-6 uCi/cm^2.

IX. Nuclear Reactor Committee

Dr. Bradley J. Micklich continues as Chairman of the Nuclear Reactor Committee for the 1986-1988 term. He is an Assistant Professor of Nuclear Engineering and Bio Engineering and has been a member of this committee previously. The following members remain on the Nuclear Reactor Committee: Dr. John G. Williams, Associate Professor of Nuclear Engineering; Dr.Abderrafi M. Ougouag, Assistant Professor of Nuclear Engineering; Mr. Hector Mandel, Campus Radiation Safety Officer; Mr. Craig Pohlod, Reactor Supervisor (ex-officio); Mr. Neil Barss, Reactor Health Physicist (ex-officio). All are previous members of this committee. The following change to the committee was made in 1987:

1. Dr. Sheldon Landsberger replaced Dr. James F. Stubbins as a member of the committee. Dr. Landsberger is an Assistant Professor in the Department of Nuclear Engineering. His background is in nuclear chemistry and his specific area of interest is Neutron Activation Analysis. He comes to the University of Illinois from McMaster University in Canada.

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February 29, 1988

Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Attention: Document Control Desk

Dear Sir:

SUBJECT: ANNUAL REPORT: Illinois Advanced TRIGA Reactor
License No. R-115
Docket No. 50-151

The following is written to comply with the requirements of Section 6.7.f. of the Technical Specifications and the conditions of Section 50.59 of 10 CFR. The outline of the report follows the numbered sequence of section 6.7.f of the Technical Specifications.

Yours truly,

Craig S Pohlod

Craig S. Pohlod
Reactor Supervisor

J G Williams

John G. Williams
Reactor Laboratory Director

Bradley J. Micklich

Bradley J. Micklich, Chairman
Nuclear Reactor Committee

Barclay G. Jones

Barclay G. Jones, Head
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Engineering

cc: Regional Administrator, Region III, USNRC

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