The University of Utah TRIGA Reactor
Annual Operating Report
for the period
1 July 1987 through 30 June 1988

A. NARRATIVE.

1. Operating Experience.

The TRIGA Reactor was critical 76.4 hours and generated 3,726.1 kWh of thermal energy during this reporting year. The reactor was used for educational demonstrations, laboratory experiments, systems tests, power measurements, and sample irradiations. The reactor has not operated since 31 March 1988 as requested by NRC pending implementation of additional administrative support by the Licensee.

2. Changes in Facility Design.

On 7/2/87, the Startup Channel fission chamber was moved from its original position closer to the reactor core to increase the neutron count rate during reactor startup. The chamber signal was calibrated and the channel entered back into service.

The Intrusion Alarm system was rewired on 10/21/87 to provide a signal which is compatible with a new computerized console at the University of Utah Police Station. The modification causes a reversal in the signal voltage polarity upon activation which actuates a burglar alarm on the display console in the event of the following:

(1) intrusion,
(2) low water level in the TRIGA tank, or
(3) high-level radiation in the Nuclear Engineering Laboratory (NEL).

Interruption of the signal to the University Police because of loss of power to the alarm system or failure of the dedicated telephone line between NEL and the University Police Station results in the activation of a trouble light on the display console. The University Police are instructed to respond to both alarms in the same manner.

The linear power meter in the control console was disconnected on 1/27/88 to correct a grounding problem in the Linear Power Channel. This problem caused the initiation of a SCRAM at certain times while switching from one watt to higher power levels during reactor startup. Although disconnecting the meter temporarily corrects the switching problem, staff members will continue to check the Linear Power Channel circuitry to isolate the source of the grounding problem. The Linear Power Channel signal was routed from the console to the computerized display in February 1986 as reported in the Annual Operating Report for the period 1 July 1985 to 30 June 1986.

On 3/15/88, the high-voltage (HV) supply coaxial cable to the uncompensated ion chamber in the Linear Power Channel was disconnected from the HV power supply located in the IBM Series/I computer stack (see below in the Maintenance section). The cable was reconnected to the HV power supply in the control console thereby rectifying an unstable power signal to the Linear Power channel which was the result of a failed capacitor in the HV power supply of the computer stack.

A new emergency power supply was installed on 12 April 1988. The new system consists of two independent and redundant 12 volt DC batteries which supply power to the radiological monitoring and intrusion alarm systems in the event of a building power failure.

3. Surveillance Tests.

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(Documentation of all surveillance activities is retained and stored by the facility.)

a. Control Rod Worths.

Core Configuration #19	25 August 1987
Safety Rod	$1.81
Shim-safety Rod	$1.34
Regulating Rod	$0.44
Excess Reactivity	$1.17
Shutdown Margin	$0.61

Core Configuration #19	17 March 1988
Safety Rod	$1.82
Shim-safety Rod	$1.77
Regulating Rod	$0.39
Excess Reactivity	$1.05
Shutdown Margin	$1.11

b. Control Rod Inspection.

The Biennial Control Rod Inspection was performed December, 1987 through February, 1988. The control rods were sequentially removed from the reactor core for visual inspection and then reinstalled. All rods were found to be in good condition with no observable deterioration having occurred since the previous inspection. Rod drop times were measured on 8/25/87 and 3/17/88. All rod drop times were less than 0.7 second.

c. Reactor Power Level Instrumentation.

Calorimetric power calibrations were performed on 8/19/87 and 2/2/88. The following results were obtained.

Date	Meter Reading	Actual Power Level
8/19/87	90 kW	75.6 kW
2/02/88	90 kW	97.2 kW

d. Fuel Inspection.

The Biennial Fuel Inspection was conducted December, 1987 through March, 1988. The discovery of a potential leaking fuel element (see below in the section pursuant to 10 CFR 50.59) prompted a detailed examination of fuel for cladding integrity. Each TRIGA fuel element was visually inspected while keeping the element submerged in the reactor tank for shielding purposes. Fuel elements showing discolorations or other indications of corrosion were noted for exchange with fuel in storage. Operating the reactor for a short period following the fuel exchange and then monitoring a water sample from the reactor tank for the presence of fission products was performed to ascertain if the source of the leakage had been removed by the fuel exchange. Although several elements with evidence of corrosion were noted, no obviously damaged fuel was observed during the inspection and the source of the leakage has not yet been determined.

e. Fuel Temperature Calibration.

Fuel temperature circuits were calibrated on 8/20/87 and 2/18/88. The circuits were calibrated to less than a 5 °C error over the range 20 °C to 500 °C.

f. Reactor Safety Committee Audits.

Reactor Safety Committee (RSC) member J. M. Byrne audited the maintenance and operational activities of the facility during February, 1988.

University of Utah Radiation Safety Officer (RSO) and RSC member K. J. Schiager prepared an audit report and made recommendations concerning Monthly Inspections, Visitor Log, and Radiation Protection and Monitoring in January 1988.

g. Environmental Surveys.

RSO K. J. Schiager reported to the RSC a maximum net exposure of 10 millirem to environmental dosimeters located at various positions surrounding the NEL for the period July 1, 1987 through June 30, 1988.

h. Analysis of Water from TRIGA Fuel Storage Pits.

The University of Utah Radiological Health Department performed analyses of water samples taken from the three TRIGA Fuel Storage Pits on 10/28/87. Two of the tanks contained no detectable contamination and will be discharged to the sanitary sewer. The other tank, the North Storage Tank, was found to contain trace amounts of Cs-137 and Co-60 contamination. The volumetric activity of the Cs-137 and Co-60 was determined to be 1.6×10^{-6} µCi/ml and 9.5×10^{-6} µCi/ml, respectively. This contaminated water will be recirculated through the water purification system until the activity is well below 10 CFR 20 guidelines and then discharged.

B. ENERGY OUTPUT.

The reactor was critical for 76.4 hours and produced 0.156 megawatt-days (3726.2 kWh) of energy during this reporting period. Since initial criticality, the reactor has been critical a total of 1,702.0 hours with an accumulated total energy output of 4.24 megawatt-days (101,770.5 kWh).

C. INADVERTENT SCRAMS.

There were fifteen inadvertent SCRAMS while the reactor was critical during the current reporting period. The type, cause, and [illegible] taken by the operations staff of each SCRAM are outlined below:

Quantity	Type	Cause	Action
4	High Log Power Channel	Signal spike from bumping console (2). Noise and drift in channel calibration (2).	Cautioned personnel. Recalibrated channel.
6	Linear Power Channel	Signal spike during switching power level (5). Drift in power level at one watt during startup due to slight positive period (1).	Restart. Cautioned operator to monitor all power meters.

Quantity	Type	Cause	Action
2	Power Failure	Strong power fluctuation (2).	Restart.
1	Magnet Current Failure	Interruption of current while attempting to connect x-y plotter to Linear Power Channel during control rod worth determination (1).	Run terminated.
1	Area	Perimeter alarm activated by personnel opening rear door to perform maintenance (1).	Cautioned personnel.
1	High-level Radiation Alarm	Hot sample removed from reactor (1).	Cautioned experimenter.

D. MAJOR MAINTENANCE.

On November 10-12, 1987 the instrumented fuel element (IFE) was removed from the core to allow for visual inspection by the Reactor Supervisor. This element was suspected of being the source of the fission-product leakage during reactor operations because of the amount of corrosion observed on the element and because of the recent physical manipulations of the IFE that occurred between June 8-24, 1987 during thermocouple calibrations for verification of proper operation. Samples of the corrosion precipitate obtained from the element were analyzed and found to contain Cs-137 and other radionuclides suggesting the IFE was the source of the leakage. A replacement IFE was constructed from spare IFEs acquired from Northrup Corporation in 1986. Following the successful splicing of the three thermocouples to chromel/alumel thermocouple wire, the element was thoroughly tested for correct response to thermal stimuli. The replacement IFE was then inserted into the core adjacent to the suspect IFE to compare and calibrate its performance during reactor operation. The old IFE was removed from the core and stored at the side of the reactor tank after the calibration of the replacement. On December 22, 1987 the reactor was operated for a period of three hours during which time no appreciable increase in radiation level was observed on any of the area radiation monitors. An analysis of a tank water sample taken at the conclusion of the run showed that fission products continued to be released despite removal of the IFE. The replacement IFE continued service through the end of the reporting period.

The dry-tube irradiator was removed from the heavy water reflector tank during a January 8, 1988 inspection of the assembly. Upon removal from the reactor tank, the polyethylene tubing was found to be severely damaged due to neutron embrittlement and was replaced. Supplementary ballast in the form of lead buckshot was added to the annular region of the canister to prevent it from accidentally floating out of the heavy water tank. The assembly was replaced in the heavy water tank and the top of the dry tube was securely fastened to a supporting beam at the top of the reactor tank.

The ± 18 volt DC power supply in unit J10 of the control console was damaged 1/13/88 while the reactor was shutdown when a high-voltage signal from a TENNELEC Counter-Timer was inadvertently connected to the fission chamber pre-amplifier to monitor the startup neutron count rate. The high voltage destroyed an integrated circuit chip in the pre-amplifier thereby short-circuiting the power supply outputs resulting in the burning of two 26.8 volt power transformers and two power transistors. The damaged components were replaced with equivalent parts which were reviewed by the Reactor Supervisor for quality assurance. The unit was returned to service after verifying the operability of the power supply. Steps taken to prevent recurrence of this event include: (1) the counter has been clearly labelled with a warning to not input a high-voltage signal while connected to the console, and (2) two fuses were installed in the ± 18 volt power supply to preclude drawing too much current from its output.

On February 5, 1988 the two uncompensated ion chambers (UIC) of the Linear and Percent Power Channels were disconnected from the Series/I computer and control console, respectively, for maintenance. During maintenance on the Linear Power Channel UIC in May, 1987, the insulation of the coaxial cables was observed to be severely damaged by the intense radiation field surrounding the reactor core. The condition of the cables was believed to be the cause of a very unstable signal from this channel which could not be used to calibrate the control rod worths. The decision was made to replace the signal and high-voltage supply coaxial cables of both UICs since the two detectors share a common support assembly which secures them to the core structure. The cables and connectors were replaced and the detectors returned to their original positions. Both detectors were verified to be operational and were returned to service. A subsequent attempt to calibrate the control rod worths proved unsuccessful, however, due to the continuance of the unstable signal. Further investigation showed the source of the problem was the superposition of an AC signal over the high-voltage bias supply to the detector due to a failed capacitor in the high-voltage power supply of the IBM computer stack. The bias supply was reconnected to the control console and the normal, stable signal from the Linear Power Channel was recovered, thereby allowing the successful calibration of the control rod worths.

A series of false intrusion alarms during January and February, 1988, was caused by the thermal expansion of the metallic foil taped around the perimeter of the exterior windows of the NEL. Insulated wire jumpers were installed across the window mullions to prevent the occurrence of more false alarms. The junctions between the old foil and the wires proved to be poor and resulted in further false alarms. NEL staff removed the existing foil from the exterior windows of the reactor room and the radiochemistry laboratories and applied new foil which improved the foil-wire junctions. The entire system was returned to service after the system was tested and verified to be operational.

During the NRC inspection of February, 1988, removable contamination was found in the exhaust duct of the NEL ventilation system. The duct was decontaminated by NEL staff on 2/26/88. The contamination consisted of Co-60 which is transported by moist air and condenses on the relatively cool interior surfaces of the exhaust duct. The Reactor Supervisor performed a survey of areas of the exhaust stack and found no evidence of removable contamination. Subsequent surveys of the exhaust duct by Radiological Health Department personnel during monthly inspections of the facility have not shown further contamination.

The Victoreen Area Radiation Monitor located above the entrance to classroom MEB 1205 was sent to the U. S. EPA in Las Vegas, Nevada for calibration on 3/30/88.

E. CHANGES, TESTS, AND EXPERIMENTS PURSUANT TO 10 CFR 50.59.

As of the end of the reporting period, the current membership of the Reactor Safety Committee (RSC) as designated by the Licensee is as follows:

Dietrich K. Gehmlich, Reactor Administrator
Gary M. Sandquist, Reactor Supervisor
Keith J. Schiager, Radiation Safety Officer
John S. Bennion

John R. Burton
James M. Byrne

The presence of Cs-137 was discovered on 10/20/87 during routine monitoring of the ion-exchange resin in the demineralizer tanks of the water purification system. The cesium indicated a possible leakage of fission products from a damaged or corroded fuel element. Confirmation of the leakage was achieved on or about 11/11/87 with the demonstration that the majority of the activity contained in reactor water samples obtained subsequent to a one-hour reactor run was due to relatively short-lived noble gases, i.e., Kr-85m, $t_{1/2}$ = 4.5 hours, and Xe-135, $t_{1/2}$ = 9.2 hours, as observed on an intrinsic germanium detector. The amount of activity released during short reactor operations was small -- well below levels of Co-60 contamination in the reactor tank incurred during the transfer of TRIGA fuel from a contaminated DOE fuel shipment cask in 1979; no abnormal increases in airborne activity had been observed. A program was devised to identify the source of the leakage and was reviewed and approved by the RSC. The program was designed to isolate the defective fuel element by exchanging incore with storage fuel and then operating the reactor for a short period of time. Water samples taken following the reactor operation were then counted to determine if the leaking fuel element had been removed. All reactor operations since the discovery of the cesium release were performed in conformance with TRIGA Technical Specification 4.4(3) which allows short-term operation of the reactor to assist in determining the source of the fission-product leakage.

During an inspection of the facility conducted by NRC on 16-19 February 1988, the Area Radiation Monitor (ARM) alarm setpoints were observed to be 1.0 and 10.0 millirem per hour for the low- and high-level alarm setpoints, respectively. As pointed out by NRC, the Licensee had apparently indicated in question 46 in the 1983-84 TRIGA license renewal application addendum that the setpoint of the low-level alarm was 0.1 millirem per hour and the setpoint of the high-level alarm was 1.0 millirem per hour. These values reported to NRC were typographical errors and are obviously too low to permit practical operations; the natural background radiation level in the area of the reactor facility is about 0.03 millirem per hour. The RSC reviewed this deviation and determined that 1.0 and 10.0 millirem per hour setpoints are correct and acceptable and are adequate to insure proper response to radiological emergencies.

The RSC has reviewed and approved several NEL procedures which were modified to update and correct deficiencies noted in RSC audit reports and NRC inspections. Copies of the procedures have been sent to NRC Region IV. The NEL staff continues to review and update facility documentation to assure compliance with applicable regulations.

F. RADIOACTIVE EFFLUENTS.

1. Liquid Waste - Negligible.

Decontamination of removable Co-60 contamination from the accessible areas of the exhaust duct yielded approximately 2.5 gallons of liquid waste. The liquid was analyzed by the University of Utah Radiological Health Department for activity content which was determined to be approximately 4.6×10^{-6} μCi/ml. Since the activity of the liquid was below 10 CFR 20 limits for the unrestricted release of water, the waste was transferred to the Radiological Health personnel for disposal on 5/2/88. The total amount of radioactivity released was estimated at 0.06 μCi.

2. Gaseous Waste - Negligible.

The TRIGA Reactor was operated for 76.4 hours at power levels up to approximately 90 kW. At this power level argon-41 production is negligible. The minimum detectable concentration of Ar-41 for the stack monitor has been found to be one-third of 10 CFR 20 appendix II limits for release to unrestricted areas. The average annual calculated concentration of Ar-41 generated during operations is estimated at 1.15×10^{-10} μCi/ml which is 0.3% of the MPC for this radionuclide. The total amount of radioactivity released was estimated at 26.0 μCi. A monthly summary of gaseous releases is given in Table I.

Table I.

Summary of Monthly Gaseous Radioactive Effluent
1 July 1987 through 30 June 1988

Month	Estimated Release (μCi)
July	1.4
August	7.4
September	1.2
October	0.5
November	2.0
December	1.7
January	5.4
February	1.8
March	4.6
April	0.0
May	0.0
June	0.0
Total activity of gaseous effluent:	26.0

3. Solid Waste - None.

Approximately 0.5 cubic meter of solid waste was generated by the facility during the reporting period. This waste consists of low-level decontamination materials, debris removed from the reactor tank during cleaning, and radioactive reactor components which have been replaced during maintenance -- such as the ends of the UIC coaxial cables which were proximal to the reactor core. The waste is being stored in the Controlled Access Area of the facility pending transfer to the Radiological Health Department for disposal.

G. RADIATION EXPOSURES.

Personnel with duties in the reactor laboratory on either a regular or occasional basis have been issued a film-badge dosimeter by the University of Utah Radiological Health Department. The duty category and monitoring period of personnel are summarized below:

Name	Monitoring Period	Duty Category
G. M. Sandquist	1/87-12/87;1/88-6/88	regular
T. C. Gansauge	1/87-12/87;1/88-6/88	regular
J. S. Bennion	1/87-12/87;1/88-6/88	regular
V. Tang	1/88-6/88	regular
K. C. Crawford	1/87-6/87	regular
C. Henderson	3/88-6/88	occasional
M. Tolle	1/87-12/87;1/88-6/88	terminated
R. Deadman	1/87-9/87	terminated

Dose Equivalent summary for Reporting Period:

Measured Doses

1987 Annual Doses: 10 mrem average; 30 mrem highest measured.
January - June, 1988: 10 mrem highest measured.

Dose Equivalent Limits

Maximum Permissible Dose Equivalent = 5000 mrem/year (1250/quarter).
Minimum Detectable per Monthly Badge = 10 mrem.

Of the 639 visitors to the facility under the DOE Reactor Sharing Program for the reporting year, no visitor received a measurable dose. Therefore, the average and maximum doses are all within NRC guidelines. A summary of whole body exposures is presented in Table II.

Table II.

Summary of Whole Body Exposures
1 July 1987 through 30 June 1988

Estimated whole body exposure range (rems):	Number of individuals in each range:
No Measurable Dose	2
Less than 0.10	6
0.10 to 0.25	0
0.25 to 0.50	0
0.50 to 0.75	0
0.75 to 1.00	0
1.00 to 2.00	0
2.00 to 3.00	0
3.00 to 4.00	0
4.00 to 5.00	0
Greater than 5	0
Total number of individuals reported:	8

H. LABORATORY SURVEYS.

Monthly surveys of the facility were conducted by the University of Utah Radiological Health Department during the reporting period. Some of these surveys have identified removable contamination sources which were immediately cleaned. The surveys have indicated no increase in radiation levels over previous years. Records of surveys are retained by the facility.

I. ENVIRONMENTAL SURVEYS.

The Air Monitoring Station, operated by the Environmental Protection Agency and located outside the reactor building, has indicated no unusual changes in radiation or radioactive material concentrations during the reporting period.

Environmental surveys conducted quarterly by the University of Utah Radiological Health Department indicated no unusual dose rates in the areas surrounding the Merrill Engineering Building, which houses the reactor facility, or anywhere on the University of Utah campus.

Prepared by: Gary M. Sandquist Date: 31 Aug 1988
Reactor Supervisor

Approved by: Dietrich K. Gehmlich Date: 31 Aug '88
Reactor Administrator

50-407

THE
UNIVERSITY
OF UTAH

August 30, 1988

Alexander Adams, Jr.
Project Manager
Standardization and Non-Power Reactor
  Project Directorate
Division of Reactor Projects - III, IV,
  V and Special Projects
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Adams:

Enclosed you will find a copy of the Annual Operating Report for the University of Utah TRIGA Reactor, Docket No. 50-407, for the period 1 July 1987 through 30 June 1988. This report fulfills the requirements of TRIGA Technical Specification 6.10(5).

Please contact the operations staff at (801) 581-4188 with any questions you may have concerning the report.

Sincerely,

Dietrich K. Gehmlich

Dietrich K. Gehmlich, PhD.
Reactor Administrator

cy: Regional Administrator,
    NRC Region IV

College of Engineering
Office of the Dean

2202 Merrill Engineering Building
Salt Lake City, Utah 84112
Telephone (801) 581-6911

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