MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1987 - JUNE 30, 1988

BY

REACTOR STAFF

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TABLE OF CONTENTS

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Introduction

This report has been prepared by the staff of the Massachusetts Institute of Technology Research Reactor for submission to the Administrator of Region 1, United States Nuclear Regulatory Commission, in compliance with the requirements of the Technical Specifications to Facility Operating License No. R-37 (Docket No. 50-20), Paragraph 7.13.5, which requires an annual report following the 30th of June of each year.

The MIT Research Reactor (MITR), as originally constructed, consisted of a core of MTR-type fuel, fully enriched in uranium-235 and cooled and moderated by heavy water in a four-foot diameter core tank, surrounded by a graphite reflector. After initial criticality on July 21, 1958, the first year was devoted to startup experiments, calibration and a gradual rise to one megawatt, the initially licensed maximum power. Routine three-shift operation (Monday-Friday) commenced in July 1959. The authorized power level was increased to two megawatts in 1962 and five megawatts (the design power level) in 1965.

Studies of an improved design were first undertaken in 1967. The concept which was finally adopted consisted of a more compact core, cooled by light water, and surrounded laterally and at the bottom by a heavy water reflector. It is undermoderated for the purpose of maximizing the peak of thermal neutrons in the heavy water at the ends of the beam port re-entrant thimbles and for enhancement of the neutron flux, particularly the fast component, at in-core irradiation facilities. The core is hexagonal in shape, 15 inches across, and utilizes fuel elements which are rhomboidal in cross section and which contain UAL_x intermetallic fuel in the form of plates clad in aluminum and fully enriched in uranium-235. Much of the original facility, e.g. graphite reflector, biological and thermal shields, secondary cooling systems, containment, etc., has been retained.

After Construction Permit No. CPRR-118 was issued by the former U.S. Atomic Energy Commission in April 1973, major components for the modified reactor were procured and the MITR-I was shut down on May 24, 1974, having logged 250,445 megawatt hours during nearly 16 years of operation.

The old core tank, associated piping, top shielding, control rods and drives, and some experimental facilities were disassembled, removed and subsequently replaced with new equipment. After preoperational tests were conducted on all systems, the U.S. Nuclear

Regulatory Commission issued Amendment No. 10 to Facility Operating License No. R-37 on July 23, 1975. After initial criticality for MITR-II on August 14th, 1975, and several months of startup testing, power was raised to 2.5 MW in December. Routine 5 MW operation was achieved in December 1976.

This is the thirteenth annual report required by the Technical Specifications, and it covers the period July 1, 1987 through June 30, 1988. Previous reports, along with the "MITR-II Startup Report" (Report No. MITNE-198, February 14, 1977) have covered the startup testing period and the transition to routine reactor operation. This report covers the eleventh full year of routine reactor operation at the 5 MW licensed power level. It was another year in which the safety and reliability of reactor operation met the requirements of reactor users.

A summary of operating experience and other activities and related statistical data are provided in the following Sections A-H of this report.

A. SUMMARY OF OPERATING EXPERIENCE

1. General

During the period covered by this report (July 1, 1987 - June 30, 1988), the MIT Research Reactor, MITR-II, was operated on a routine, four days per week schedule, normally at a nominal 5MW. It was the tenth full year of normal operation for MITR-II.

The reactor averaged 71.8 hours per week at full power compared to 80.1 hours per week for the previous year and 75.4 hours per week two years ago. The reactor is normally at power 90-100 hours/week, but holidays, major maintenance, long experiment changes, waste shipping, etc., reduce the average. During the past year it was reduced more than usual because of the planned installation of several major experiments concerning the production, activation, and transport of corrosion products in pressurized water reactors. Also, a lot of operation was conducted at low power for the purpose of making measurements on the medical therapy room beam. The reactor currently operates from late Tuesday afternoon until late Friday afternoon, with maintenance scheduled for Mondays/Tuesdays and, as necessary, for Saturdays. A return to Monday-Friday operation is anticipated starting in the fourth quarter of 1988.

The reactor was operated throughout the year with 25 elements in the core. The remaining positions were occupied by irradiation facilities used for materials testing and the production of medical isotopes and/or by a solid aluminum dummy. Compensation for reactivity lost due to burnup was achieved through two refuelings of three elements each. Each entailed the introduction of three new elements to the core's intermediate fuel ring (the B-ring) and the transfer of partially spent elements to the C-Ring to replace elements that were at the fission density limit. This policy was in keeping with the practice begun in previous years in which partially spent elements that had been originally removed from the B-ring were gradually introduced to the C-ring to replace fully spent elements. These procedures were combined with many element rotations/inversions, the objective of which was to minimize the effects of radial/axial flux gradients and thus achieve higher average burnups. Eight other refuelings were performed. One was to replace a C-Ring element that had attained the fission density limit. It was replaced by a partially-spent element. Three were for the purpose of trial fitting or making reactivity measurements of the thimble that will hold the Pressurized Coolant Corrosion Loop (PCCL) scheduled for insertion in the fourth quarter of 1988. (Refer to section E of this report.) Four were for the purpose of testing several suspect elements for possible excess outgassing. None was detected and the elements tested are now in routine use. This action resolves concerns originally raised in our letter of 17 August 1982 in which it was reported that several elements were suspected of possible excess outgassing.

The MITR-II fuel management program remains quite successful. All of the original MITR-II elements (445 grams U-235) have been permanently discharged. The average overall burnup for the discharged elements was 42%. (<u>Note</u>: One element was removed prematurely because of excess outgassing.) The maximum overall burnup achieved was 48%. Forty-two of the newer, higher loaded elements (506 grams U-235) have been introduced to the core. Of them, four have attained the maximum allowed fission density. However, these may be reused if that limit is increased as would seem warranted based on metallurgical studies by DOE. Another five have, as reported previously to the U.S. Nuclear Regulatory Commission, been identified as showing excess outgassing and have been removed from service. As for the other thirty-three new elements, they are either currently in the reactor core or have been partially depleted and are awaiting reuse in the C-ring.

The availability of a licensed spent fuel shipping cask from DOE is again delayed this year. The delay has thus far caused our total fuel inventory to approach the authorized possession limit and continues to force us to deviate from our normal fuel management practice in that:

(1) The inventory of partially spent elements is now substantially below normal. This is making it difficult to convert from one core configuration to another.

(2) Inability to bring in fresh fuel and to place it in the A and B-Rings of the core may necessitate premature C-Ring refuelings in order to obtain sufficient reactivity for continued operation. This will result in lower overall burnups and ultimately increase our need for additional fuel.

Finally, it should be recognized that if (as now appears likely) casks continue to be unavailable, we will have to request a reinstatement of part or all of our previous license limit for possession of U-235 in order to continue operation.

Protective system surveillance tests are conducted on Friday evenings after shutdown (about 1800), on Mondays, and on Saturdays as necessary.

As in previous years, the reactor was operated throughout the period without the fixed hafnium absorbers, which were designed to achieve a maximum peaking of the thermal neutron flux in the heavy water reflector beneath the core. These had been removed in November 1976 in order to gain the reactivity necessary to support more in-core facilities.

2. <u>Experiments</u>

The MITR-II was used throughout the year for experiments and irradiations in support of research and training programs at MIT and elsewhere.

Experiments and irradiations of the following types were conducted:

a) Prompt gamma activation analysis for the determination of boron-10 concentration in blood and tissue. This is being performed using one of the reactor's beam tubes. The analysis is to support our neutron capture therapy program.

b) Experimental measurements to determine the suitability of various materials to serve as a neutron filter in a medical therapy beam. These measurements are used to benchmark theoretical predictions.

c) The production of Mössbauer sources by the irradiation of Gd-160 and Pt-196 for studies of nuclear relaxation of Dy-161 in Gd and for the investigation of the chemistry and structure of gold compounds.

d) Irradiation of archaeological, environmental, engineering materials, biological, geological, oceanographic, and medical specimens for neutron activation analysis purposes.

e) Production of gold-198, dysprosium-165, and holmium-166 for medical research, diagnostic and therapeutic purposes.

f) Irradiation of tissue specimens on particle track detectors for plutonium radiobiology.

g) Irradiation of semi-conductors to determine resistance to high doses of fast neutrons.

h) Use of the facility for reactor operator training.

i) Irradiation of geological materials to determine quantities and distribution of fissile materials using solid state nuclear track detectors.

j) Fault detection analysis of the output of control and process channels from the MIT Reactor as part of a study leading to control of reactors by use of fault-tolerant, digital computers. This effort recently resulted in the demonstration of techniques for providing predictive information to reactor operators for their use in conducting transients.

k) Closed-loop direct digital control of reactor power using a shim blade as well as the regulating rod during some steady-state and transient conditions. Control laws for adjusting a reactor's neutronic power in minimum time were developed and demonstrated.

l) Experimental studies of various closed-loop control techniques including digital filters to reduce signal noise.

m) Development and experimental evaluation of a technique for the determination of subcriticality.

n) Measurements of the energy spectrum of leakage neutrons using a mechanical chopper in a radial beam port (4DH1). Measurements of the neutron wavelength by Bragg reflection then permits demonstration of the DeBroglie relationship for physics courses at MIT and other universities.

o) Studies of fast neutron damage to liquid crystal display materials using a delayed neutron detector.

Two research projects that will make major use of the reactor in the next and subsequent years have been funded and are in various stages of design and development. The first project is a dose reduction study for the light water reactor industry which will involve the installation of pressurized loops in the reactor core to investigate the chemistry of corrosion and the transport of radioactive crud with systems that simulate PWRs and BWRs. The second project is an extension of previous research to develop the boron neutron capture method of therapy for brain cancer (glioblastoma). This is a collaborative effort with the Tufts University New England Medical Center. As noted above, this project made extensive use of the reactor during the past year.

3. Changes to Facility Design

Except for minor changes reported in Section E, no changes in the facility design were made during the year. As indicated in past reports the uranium loading of MITR-II fuel was increased from 29.7 grams of U-235 per plate and 445 grams per element (as made by Gulf United Nuclear Fuels, Inc., New Haven, Connecticut) to a nominal 34 and 510 grams respectively (made by the Atomics International Division of Rockwell International, Canoga Park, California). With the exception of six elements (one Gulf, five AI) that were found to be outgassing excessively, performance has been good. (Please see Reportable Occurrence Reports Nos. 50-20/79-4, 50-20/83-2, 50-20/85-2, 50-20/86-1, 50-20/86-2, and 50-20/88-1.) The heavier loading results in 41.2 w/o U in the core, based on 7% voids, and corresponds to the maximum loading in Advanced Test Reactor (ATR) fuel. Atomics International completed the production of 41 of the more highly loaded elements in 1982, 36 of which have been used to some degree. Four with about 37% burnup have been discharged because they have attained the fission density limit. Additional elements are now being fabricated by Babcock & Wilcox, Navy Nuclear Fuel Division, Lynchburg, Virginia. Six of these have been received at MIT and are now is use.

The MITR staff has been following with interest the work of the Reduced Enrichment for Research and Test Reactors (RERTR) Program at Argonne National Laboratory, particularly the development of advanced fuels that will permit uranium loadings up to several times the recent upper limit of 1.6 grams total uranium/cubic centimeter. Consideration of the thermal-hydraulics and reactor physics of the MITR-II core design show that conversion of MITR-II fuel to lower enrichment must await the successful demonstration of the proposed advanced fuels.

4. Changes in Performance Characteristics

Performance characteristics of the MITR-II were reported in the "MITR-II Startup Report". Minor changes have been described in previous reports. There were no changes during the past year.

5. Changes in Operating Procedures Related to Safety

There were no amendments to the Facility Operating License during the last year.

With respect to operating procedures subject only to MITR internal review and approval, a summary is given below of those changes implemented during the past year. Those changes related to safety are discussed in section E of this report.

a) Procedure 5.7.3, "High Level Equipment Room Sump," was revised to reflect a recent modification in which the sump itself was replaced and one high capacity pump installed in lieu of the two that had originally been in use. Also, a description of the interlocks associated with the sump pump was added to the procedure. (SR #0-87-14)

b) Procedures governing operation of the "Fatigue Cracking Experiment" were deleted reflecting the completion of this particular experimental program. (SR #0-87-15)

c) PM 3.3.1, "General Conduct of Refueling Operations," was revised to include an explicit requirement for the supervisor in charge of a refueling to visually inspect the core tank for foreign objects upon completion of the refueling. (SR #0-87-16)

d) PM 7.6.1.1, "Inspection of the Graphite Reflector Region," was revised prior to a follow-up visual inspection of that area. (Note: Inspection results were, as expected, normal.) (SR #0-87-18)

e) The administrative procedures, Chapter 1 of the Procedure Manual, were revised to update the list of names and committee memberships. Also, the requirement to give closed-book requalification exams was made explicit. (Note: MITR policy had always been to give closed-book requalification exams. This change merely formalized an existing practice.) (SR #0-88-1)

f) A special procedure for installation and removal of an aluminum plug in the medical therapy room beam was prepared. Measurements taken both with and without this plug in the beam were used to confirm theoretical studies concerning the design of a treatment beam for neutron capture therapy. (SR #0-88-2)

g) Procedure 1.4.5, "Safety Review Form" was revised to make explicit the requirement to document determinations of unreviewed safety questions. (Note: This change merely formalized an existing practice.) (SR #0-88-3)

h) The checklist used for procedure 6.1.2.1, "Building Pressure Test", was revised to reflect minor suggestions made during the previous year's test. (QA #0-88-1)

i) Miscellaneous minor changes to operating procedures and to equipment were approved and implemented throughout the year.

6. Surveillance Tests and Inspections

There are many written procedures in use for surveillance tests and inspections required by the Technical Specifications. These procedures provide a detailed method for conducting each test or inspection and specify an acceptance criterion which must be met in order for the equipment or system to comply with the requirements of the Technical Specifications. The tests and inspections are scheduled throughout the year with a frequency at least equal to that required by the Technical Specifications. Twenty-seven such tests and calibrations are conducted on an annual, semi-annual or quarterly basis.

Other surveillance tests are done each time before startup of the reactor if shut down for more than 16 hours, before startup if a channel has been repaired or de-energized, and at least monthly; a few are on different schedules. Procedures for such surveillance are incorporated into daily or weekly startup, shutdown or other checklists.

During the reporting period, the surveillance frequency has been at least equal to that required by the Technical Specifications, and the results of tests and inspections were satisfactory throughout the year for Facility Operating License No. R-37.

B. REACTOR OPERATION

Information on energy generated and on reactor operating hours is tabulated below:

	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
1. Energy Generated (MWD):					
a) MITR-II (MIT FY88) (normally at 4.9 MW)	137.5	140.4	89.3	120.7	487.9
b) MITR-II (MIT FY76-87)					9,192.8
c) MITR-I (MIT FY59-74)					10,435.2
d) Cumulative, MITR-I & MITR-II					20,115.9
2. MITR-II Operation (Hrs): (MIT FY88)					
a) At Power (>0.5 MW) for Research	742.4	843.7	585.6	709.8	2,881.5
b) Low Power (<0.5 MW) for Training[1] and Test	146.9	137.5	87.5	94.1	466.0
c) Total Critical	889.3	981.2	673.1	803.9	3,347.5

[1] Note: These hours do not include reactor operator and other training conducted while the reactor is at full power for research purposes (spectrometer, etc.) or for isotope production. Such hours are included in the previous line.

C. SHUTDOWNS AND SCRAMS

During the period of this report there were 11 inadvertent scrams and 10 unscheduled power reductions.

The term "scram" refers to shutting down of the reactor through protective system action when the reactor is at power or at least critical, while the term "reduction" or "shutdown" refers to an unscheduled power reduction to low power or to subcritical by the reactor operator in response to an abnormal condition indication. Rod drops and electric power loss without protective system action are included in shutdowns.

The following summary of scrams and shutdowns is provided in approximately the same format as for previous years in order to facilitate a comparison.

I. Nuclear Safety System Scrams	Total
a) Improper setting of power safety channel #6	2
b) Improper reshim (refer to ROR #87-2)	1
c) Channel #3 reset prematurely during startup	1
d) Channel #1 and #3 off scale low while operating in ion chamber mode	1
e) Channel #1 noise due to cold solder joint	1
f) Channel #3 low voltage chamber power supply	1
Subtotal	7

II. Process System Scrams	
a) Diaphragm failure of reflector tank dump valve	1
b) Primary coolant flow channel scram set too conservatively	3
Subtotal	4

III. Unscheduled Shutdowns or Power Reductions

a)	Shutdown due to Electric Company power loss	1
b)	Shutdown due to loss of offsite power caused by incorrect servicing of MIT distribution network	1
c)	Blade #2 dropped due to a faulty potentiometer	1
d)	Operator shut reactor down to:	
	i) Inspect interior of core tank	1
e)	Operator lowered power to investigate or correct:	
	i) Temporary loss of cooling tower fans	1
	ii) Low oil pressure in exhaust damper due to hydraulic pump failure	3
	iii) Cause of abnormal pH levels in primary coolant. (Note: Cause identified as excess air entrainment in coolant. Repaired pump shaft seals.)	1
	iv) Clogged filter in ion column	1
	Subtotal	10
	Total	21

Experience during recent years has been as follows for scrams and unscheduled shutdowns:

Fiscal Year	Number
84	19
85	10
86	27
87	21
88	21

D. MAJOR MAINTENANCE

Major maintenance projects during FY88, including the effect, if any, on safe operation of the reactor are described in this section.

Major maintenance items were continued to be performed in FY88 in anticipation of supporting the necessary requirements of the upcoming dose reduction projects for light water reactors. These projects are the Pressurized Water Reactor Coolant Corrosion Loop (PCCL), the Boiling Water Reactor Coolant Corrosion Loop, (BCCL), and the Intragranular Radiation Assisted Stress Corrosion Cracking (IASCC) Loop. Construction of the 480 V three phase AC power supply system, the cables for which were installed in FY87, was completed early this fiscal year. The final tie-in to the existing 480 V three phase bus was installed and the system is capable of delivering three phase power up to 400 A at 480 V.

In addition to enhancing the high voltage electrical services, the Emergency Core Cooling System (ECCS) was reevaluated for proper distribution of the spray pattern after the installation of these in-core experimental loops. The actual spray nozzles were removed from the reactor core tank (with the reactor shutdown) and tested in a full-scale mock-up of the reactor core and its associated components. The base case was a replica of the set-up used for the pre-operational tests of the reactor in 1975. The results were found to be in agreement with those performed in 1975. The tests were then repeated with the various in-core experimental loops simulated in the mock-up. The existing ECCS system was found to be more than adequate. These results were documented in a safety review which is described in section E of this report. A copy of these results was forwarded to the NRC on 8 March 1988.

Instrumentation necessary for operation of the PCCL was designed and constructed. Also, control panels in the control room have been modified to accommodate the PCCL instrument panel. In order to obtain the needed space, the electronics for the now completed Fatigue Cracking Experiment were disconnected and removed. The electronics for the PCCL, which are located in the control room, were installed but remain unconnected to the reactor alarm circuit and to the loop itself. Final connections will be made and intended functions tested as part of the planned pre-operational testing when the PCCL experiment is installed in the fourth quarter of 1988.

In addition to the above changes to the control room instrumentation, a new auto-ranging pico-ammeter which has a digital interface was installed in parallel with the existing nuclear instrumentation. This pico-ammeter is designed to change range automatically based on the input signal level. It is intended to be used in place of the existing selectable fixed-range pico-ammeter when performing closed-loop computer control experiments. The detector that is connected to the auto-ranging pico-ammeter when performing the control experiments will be a compensated ion chamber capable of covering the entire range of the reactor power from shutdown source levels to full power.

The graphite region was inspected in FY87. No indications of any swelling, distortion, or stored energy were found in the graphite stringers. The reflector tank outer wall, however, was found to have a thin layer of loose material which appeared to be an oxide. Samples of this layer were taken and a small area on the reflector tank wall was polished to have the oxide layer removed. X-ray diffraction studies on this loose material showed that it was aluminum oxide. A second visual inspection of the graphite region and reflector tank was performed in FY88 about one year following the initial inspection. The oxide layer on the reflector tank was found to be unchanged. No visible change or any new oxide build-up was found on the small polished area on the tank. It is believed that this oxide layer formed shortly after the inital operation of the MITR-II in 1975. A small amount of moisture may have been present in the graphite reflector region and it would have condensed on the outer wall of the reflector tank. In any event, the layer is stable and not growing. Also, it is only a few mils thick. Photographs of the graphite stringers and the reflector tank were taken. Follow-up inspections of the graphite region, reflector tank, and the polished area will be performed in the next few years so as to confirm that, as is now evident, the oxide layer is stable and no new material is forming.

As part of the graphite region and reflector tank study, an effort was made to verify that an inert gas atmosphere is being maintained in the graphite region. Doing so both inhibits any chemical reactions that could occur and prevents the formation of Ar-41 gas. The purge gas of the graphite region was changed from helium to CO_2 in FY88 so as to allow easier detection of any leakage from the graphite region. The conversion from using helium to CO_2 was documented in a safety review as described in section E of this report. Another purpose of the helium to CO_2 conversion study was to reduce the overall production of Ar-41 in the biological shields of the reactor. A CO_2 detector was used to locate any gas leaks which may have developed over the years in the graphite region and the biological shields. Several leaks were found and sealed. This effort to locate and seal possible sources of argon production is continuing.

A leak in one of the main heat exchangers, HE-1A, was located and plugged. The heat exchanger has since been returned to service. The shaft seals on the two main primary coolant pumps were replaced because a minor leak at the rate of about a drop per day was found on one of the pump shafts when the temperature of the primary system was below 15 °C.

Many exterior panels on the cooling towers were replaced and leaks on the panels were sealed with a water-proof compound. The ice formation on the cooling tower panels in the winter following the repairs was much reduced.

The interior portion of the cryogenic facility was removed from the thermal column of the MITR-I in the 60's. The exterior portion of the cryogenic facility had since been placed in a stand-by condition. The Lewis Research Center of NASA expressed interest in transferring the refrigeration plant of this system from the MIT Reactor to their facility in Cleveland, Ohio. Preparatory work was completed and the refrigeration plant of the cryogenic facility was shipped out of MIT in FY88. The refrigeration plant consisted of (1) an Ingersoll-Rand TVH compressor which was driven by a 2400 V 300 HP GE synchronous motor with a separate motor-generator exciter, (2) a three stage Joule-Thompson expansion engine made by Linde, (3) a 1000 liter liquid helium Linde Dewar, and (4) the associated breakers, instrumentation, valves, piping, and evaporators. The lines that penetrated the containment were cut, blank-flanged, and leak tested in accordance with the standard leak testing procedures and requirements.

The containment vehicle (truck) lock was painted and the outer door gasket replaced. The results of the annual containment pressure test showed that there is a leak along the expansion joint which is located in the middle section of the truck lock. The repair of the expansion joint is non-standard and involves special materials and equipment. Repair work is expected to extend into the upcoming fiscal years. The vehicle lock in the meantime is tagged out-of-service and its use is prohibited whenever the reactor is in a non-secured condition. (Note: At no time during the preceeding year had containment integrity depended upon the truck lock's expansion joint.)

Many other routine maintenance and preventive maintenance items were performed throughout the year.

E. SECTION 50.59 CHANGES, TESTS, AND EXPERIMENTS

This section contains a description of each change to the facility or procedures and of the conduct of tests and experiments carried out under the conditions of Section 50.59 of 10 CFR 50, together with a summary of the safety evaluation in each case.

The review and approval of changes in the facility and in the procedures as described in the SAR are documented in the MITR records by means of "Safety Review Forms". These have been paraphrased for this report and are identified on the following pages for ready reference if further information should be required with regard to any item. Pertinent pages in the SAR have been or are being revised to reflect these changes, and they will be forwarded to the Director, Standardization and Non-Power Reactor Project Directorate, Office of Nuclear Reactor Regulation, USNRC.

The conduct of tests and experiments on the reactor are normally documented in the experiments and irradiation files. For experiments carried out under the provisions of 10 CFR 50.59, the review and approval is documented by means of the Safety Review Form. All other experiments have been done in accordance with the descriptions provided in Section 10 of the SAR, "Experimental Facilities".

Pressurized Coolant Corrosion Loop (PCCL)
SR #0-86-9 (04/21/88)

This project involves the design, installation, and operation of a pressurized light-water loop in the MITR core for the purpose of studying the production, activation, and transport of corrosion products. The effect of various water chemistries will be examined to determine the optimum method for reducing the creation of activated corrosion products (crud) and thereby reducing radiation fields associated with pressurized water reactors (PWRs). The ultimate goal is to reduce radiation exposures to PWR maintenance personnel.

Approval for the PCCL was given by the MITR Staff and the MIT Reactor Safeguards Committee on 04/21/88. It was determined at that time that no unreviewed safety question existed because no failure or accident associated with the PCCL could lead to an accident or failure involving reactor components. Details of that determination, together with safety review #0-86-9, were submitted to the U.S. Nuclear Regulatory Commission on 04/21/88 and are therefore not further discussed here.

Actual installation of the PCCL is now scheduled for the fourth quarter of 1988.

Analysis of the Emergency Core Cooling System (ECCS)
SR #0-87-19 (03/08/88)

As discussed in section D, "Major Maintenance" of this report, the capability of the emergency core cooling system was reanalyzed in order to verify proper spray pattern distribution following installation of the proposed in-core corrosion loop experimental facilities. Both theoretical calculations and a full-scale, ex-core mock-up were used. It was found that the existing system would be more than adequate even with the corrosion loops installed.

This reanalysis of the ECCS was reviewed and approved by the MIT Reactor Safeguards Committee on 03/08/88. It was concluded that no unreviewed safety question existed. The new analysis has been provided to the U.S. Nuclear Regulatory Commission as Revision No. 34 to the Safety Analysis Report for the MIT Research Reactor and is therefore not further discussed here.

Change of Graphite Reflector Cover Gas from Helium to Carbon Dioxide
SR #0-87-20 (03/08/88)

The cover gas used in the reactor's graphite reflector region was changed from helium to carbon dioxide. This was done as part of the program to reduce percentage of air in the graphite region and hence reduce argon emissions. An evaluation of the effectiveness of this conversion is currently in progress. The appropriate sections of the MITR Safety Analysis Report will be revised once this evaluation is complete.

Approval for this change was received from the MIT Reactor Safeguards Committee on 8 March 1988. It was determined that no unreviewed safety question exists because carbon dioxide is essentially an inert gas as is helium. The principal concerns associated with use of CO_2 instead of helium appear to be the possible production of minute quantities of carbonic acid and carbon monoxide. Neither could form in significant quantity. Accordingly, as is documented in SR #0-87-20, there is no increase in probability of an analyzed accident (i.e., increased corrosion of the reflector tank), no possibility of a new type of accident (i.e., a health hazard caused by carbon monoxide), and no decrease in any margin of safety as defined in the basis of a technical specification.

Evaluation of carbon dioxide as a reflector cover gas is on-going and will probably continue until mid-1989.

Digital Computer Control of Reactors Under Steady-State and Transient Conditions

SR#-M-81-3 (11/17/81), M-81-4 (12/10/81), E-82-2 (01/08/82), E-82-3 (02/24/82), E-82-4 (03/03/82), E-82-5 (04/14/82), E-82-6 (07/13/82), 0-83-5 (02/03/83), E-83-1 (02/08/83), 0-83-12 (04/23/83), 0-83-20 (07/20/83), 0-84-11 (06/25/84), 0-84-12 (07/12/84), 0-84-16 (12/6/84), 0-84-21 (11/1/84), 0-85-11 (5/9/85), 0-85-13 (6/28/85), 0-85-16 (7/12/85), 0-85-20 (8/16/85), 0-85-25 (12/1/85), 0-85-26 12/1/85), 0-86-11 (10/17/86), 0-86-13 (11/28/86), 0-87-11 (6/1/87), 0-87-17 (12/24/87).

The project involving computer analysis, signal validation of data from reactor instruments, and closed-loop control of the MIT Reactor by digital computer was continued. A non-linear supervisory algorithm has been developed and demonstrated. It functions by restricting the net reactivity so that the reactor period can be rapidly made infinite by reversing the direction of control rod motion. It, combined with the signal validation procedures, insures that there will not be any challenge to the reactor safety system while testing closed-loop control methods. Several such methods, including decision analysis, rule-based control, and modern control theory, continue to be experimentally evaluated. The eventual goal of this program is to use fault-tolerant computers coupled with closed-loop digital control and signal validation methods to demonstrate the improvements that can be achieved in reactor control.

Each new step in the program is evaluated for safety in accordance with standard review procedures (Safety Review numbers listed above) and approved as necessary by the MIT Reactor Safeguards Committee.

Initial tests of this digital closed-loop controller were conducted in 1983-1984 using the facility's regulating rod which was of relatively low reactivity worth (0.2% ΔK/K). Following the successful completion of these tests, facility operating license amendment No. 24 was obtained from NRC (April 2, 1985). It permits:

(1) closed-loop control of one or more shim blades and/or the regulating rod provided that no more than 1.8% ΔK/K could be inserted were all the connected control elements to be withdrawn,

(2) closed-loop control of one or more shim blades and/or the regulating rod provided that the overall controller is designed so that reactivity is constrained sufficiently to permit control of reactor power within desired or authorized limits.

A successful experimentation program is now continuing under the provisions of this license amendment. A protocol is observed in which this controller is used to monitor, and if necessary override, other novel controllers that are still in development. Tests performed during this reporting period include:

a) Tests in which predictive displays were provided to the reactor operator as an aid in the conduct of reactor transients. These experiments were approved by both the MIT Reactor Safeguards Committee and the MIT Committee on the Use of Humans as Experimental Subjects. Operators participating in the tests were restricted to use of the regulating rod which is of low reactivity worth, were limited to transients of 1-2 MW, and were under the continuous supervision of a licensed senior operator who was not a test participant. The tests were successful, showing that predictive information was of direct benefit. (SR #0-87-11 dtd 06/01/88)

b) Evaluations of the MIT-SNL Period Generated Minimum Time Control Laws. These laws adjust the rate of change of reactivity so that the reactor period is maintained constant at a specified value. As a result, power changes are accomplished in minimum time for a reactor limited to the specified period. These tests were conducted under the provisions of technical specification #6.4 using our now standard protocol in which reactivity is constrained by a supervisory controller that maintains "feasibility of control". Signal implementation is accomplished using a variable speed stepping motor. This motor is installed prior to the tests and removed upon their completion. An independent hard-wired circuit is used to monitor motor speed and preclude an overspeed condition. The conduct of these tests was approved by the MIT Reactor Safeguards Committee. The tests were very successful. (SR #0-87-17 dtd 12/24/87)

For both of the above changes it was concluded that no unreviewed safety question existed. Relative to the use of the predictive displays, control remained at all times under the direction of a licensed senior operator. The operator using the displays utlized them as he or she would any instrument. Relative to the minimum time laws, the experiments were also conducted under the supervision of a licensed senior operator. Also, use of the now-standard supervisory algorithm and the independent hard-wired circuits limited the possible envelope of operating conditions. In particular, there was no possibility of control mechanism withdrawal such that the allowed rate of insertion of positive reactivity could be exceeded. Hence, for neither experiment was there an increase in the probability of an analyzed accident, a possibility of a new type of accident, or a decrease in a safety margin defined in the basis of any technical specification.

F. ENVIRONMENTAL SURVEYS

Environmental surveys, outside the facility, were performed using area monitors. The systems (located approximately in a 1/4-mile radius from the reactor site) consist of calibrated G.M. detectors with associated electronics and recorders.

The detectable radiation levels due to argon-41 are listed below:

Site	July 1, 1987 - June 30, 1988
North	0.10 mR/year
South	0.17 mR/year
East	0.43 mR/year
West	0.34 mR/year
Green (East)	0.11 mR/year

Fiscal Yearly Averages:

1978	1.9 mR/year
1979	1.5 mR/year
1980	1.9 mR/year
1981	1.9 mR/year
1982	2.5 mR/year
1983	2.3 mR/tear
1984	2.1 mR/year
1985	2.2 mR/year
1986	1.8 mR/year
1987	1.2 mR/year
1988	1.2 mR/year

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

A summary of radiation exposures received by facility personnel and experimenters is given below:

Period 7/01/87 - 6/30/88

Whole Body Exposure Range (Rems)	No. of Personnel
No Measurable	100
Measurable - Exposure less than 0.1	44
0.1 - 0.25	13
0.25 - 0.5	6
0.5 - 0.75	3
0.75 - 1.0	1
Total Personnel = 167	Total Man Rem = 7.44

Summary of the results of radiation and contamination surveys from July 1987 to June 1988:

During the 1987-1988 period, the Reactor Radiation Protection Office continued to provide radiation protection services necessary for full-power (5 megawatt) operation of the reactor. Such services (performed on a daily, weekly, or monthly schedule) include, but are not limited to, the following:

1. Collection and analysis of air samples taken within the containment shell, and in the exhaust-ventilation system.

2. Collection and analysis of air samples taken from the cooling towers, D_2O system, waste storage tanks, shield coolant, heat exchangers, fuel storage facility, and the primary system.

3. Performance of radiation and contamination surveys, radioactive waste collection, calibration of reactor radiation monitoring systems, and servicing of radiation survey meters.

4. Providing of radiation protection services for control rod removal, spent-fuel element transfers, ion column removal, etc.

The results of all surveys described above have been within the guidelines established for the facility.

H. RADIOACTIVE EFFLUENTS

This section summarizes the nature and amount of liquid, gaseous and solid radioactive wastes released or discharged from the facility.

1. Liquid Waste

Liquid radioactive wastes generated at the facility are discharged only to the sanitary sewer serving the facility. There were two sources of such wastes during the year: the cooling tower blowdowns and the liquid waste storage tanks. All of the liquid volumes are measured, by far the largest being the 4,161,000 liters discharged during FY 1988 from the cooling towers. (Larger quantities of non-radioactive waste water are discharged to the sanitary sewer system by other parts of MIT, but no credit for such dilution is taken since the volume is not routinely measured.)

All releases were in accordance with Technical Specification 3.8-1, including Part 20, Title 10, Code of Federal Regulations. There are no reportable radionuclides inasmuch as all activities were substantially below the limits specified in 10 CFR 20.303 and 10 CFR 20, Appendix B, Note 5.

2. Gaseous Waste

Gaseous radioactivity is discharged to the atmosphere from the containment building exhaust stack and by evaporation from the cooling towers. All gaseous releases likewise were in accordance with the Technical Specifications and Part 20, and all nuclides were below the limits of 10 CFR 20.106 after the authorized dilution factor of 3000. Also, all were substantially below the limits of 10 CFR 20, Appendix B, Note 5, with the exception of argon-41, which is reported in the following Table H-1. The 2627 Ci of Ar-41 were released at an average concentration of 0.67×10^{-8} µCi/ml for the year. This represents 17% of MPC (4×10^{-8} µCi/ml) and is significantly less than the previous year's release of 4223 Ci. The decrease is due to a combination of factors including the sealing of leaks and the temporary reduction in operating hours.

3. Solid Waste

Only one shipment of solid waste was made during the year, information on which is provided in the following Table H-2.

TABLE H-1

ARGON-41 STACK RELEASES

FISCAL YEAR 1988

	Ar-41 Discharged (Curies)	Average Concentration[1] (µCi/ml)
July 1987	249	0.68 x 10^{-8}
August	119	0.40
September	231	0.78
October	318	0.86
November	146	0.50
December	129	0.44
January 1988	161	0.44
February	82	0.28
March	159	0.43
April	376	1.28
May	251	0.85
June	406	1.10
Totals (12 Months)	2627	0.67 x 10^{-8}
MPC (Table II, Column I)		4 x 10^{-8}
% MPC		17%

(1) Note: After authorized dilution factor (3000).

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS

FISCAL YEAR 1988

	Units	Shipment #1	Total
1. Solid waste packaged	Cubic Feet	60.5	60.5
2. Weight	Pounds	1459	1459
3. Total activity (irradiated components, ion exchange resins, etc.) ^{60}Co, ^{51}Cr, $^{55-59}Fe$ ^{65}Zn, etc.	Ci	0.0034	0.0034
4. (a) Date of shipment	10/27/87		
(b) Disposition to licensee for burial	U.S. Ecology, Inc.		

NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
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L. CLARK, JR.
Director of Reactor Operations

August 29, 1988

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: Annual Report, Docket No. 50-20, License R-37,
Technical Specification 7.13.5

Dear Sirs:

Forwarded herewith is the Annual Report for the MIT Research Reactor for the period July 1, 1987 to June 30 1988, in compliance with paragraph 7.13.5 of the Technical Specifications for Facility Operating License R-37.

Sincerely,

Kwan S. Kwok
Superintendent,
Reactor Operations

John A. Bernard, Ph.D
Director of Reactor Operations

JAB/crh
Enclosure: As stated

cc: MITRSC
USNRC - Region I
Chief, Reactor Projects Section 1B
USNRC - Region I
L.T. Doerflein, Project Inspector, Section 1B
USNRC - Resident Inspector, Pilgrim Nuclear Station

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