NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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L. CLARK, JR.
Director of Reactor Operations

August 30, 1985

Dr. Thomas E. Murley, Administrator
U.S. Nuclear Regulatory Commission
Region #1
631 Park Avenue
King of Prussia, PA 19406

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

Dear Dr. Murley:

Forwarded herewith are two (2) copies of the Annual Report for the MIT Research Reactor for the period July 1, 1984 to June 30 1985, in compliance with paragraph 7.13.5 of the Technical Specifications for Facility Operating License R-37.

Sincerely,

Lincoln Clark, Jr.

Lincoln Clark, Jr.
Director of Reactor Operations

LC/gw
Enclosure: As stated

cc: MITRSC
USNRC-OI&E
USNRC-DMB
USNRC-OMIPC

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MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1984 - JUNE 30, 1985

BY

REACTOR STAFF

August 29, 1985

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MIT RESEARCH REACTOR

ANNUAL REPORT TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1984 - JUNE 30, 1985

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, rully 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 tenth annual report required by the Technical Specifications, and it covers the period July 1, 1984 through June 30, 1985. 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 eigth 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 fully 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, 1984 - June 30, 1985), the MIT Research Reactor, MITR-II, was operated on a routine, five days per week schedule, normally at a nominal 5MW. It was the eighth full year of normal operation for MITR-II.

The reactor averaged 86.3 hours per week at full power compared to 90.3 hours per week for the previous year and 85.2 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. The reactor routinely operates from late Monday afternoon until late Friday afternoon, with maintenance scheduled for Mondays and, as necessary, for Saturdays.

The reactor was operated throughout the year with 24 or 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 five refuelings of several elements each. These involved a continuation of the practice begun in previous years in which fresh fuel was introduced to the A and B-rings while 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.

The MITR-II fuel management program remains quite successful. All but seven of the original MITR-II elements (445 grams U-235) have been permanently discharged. The average overall burnup for the discharged elements was 42%. Of the remaining seven elements with the 445 gram loading, it is now projected that six will reach maximum depletion within the next six months. Thirty-six of the new elements (506 grams U-235) have been introduced to the core. Of them, three 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. 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 continued delays in the availability of a licensed cask from DOE are of increasing concern. Specifically, our inability to ship spent fuel is forcing us to deviate from our normal fuel cycle in that:

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

(2) Inability to place fresh fuel 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 casks continues 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. Experiments

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) Neutron diffraction spectrometer alignment and studies (3 ports).

b) 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.

c) Irradiation of biological, geological, oceanographic, and medical specimens for neutron activation analysis purposes.

d) Investigation of the technique of neutron-induced autoradiography for possible use in determining the history and authenticity of paintings.

e) Production of phosphorus-32, gold-198, dysprosium-165, fluorine-18, osmium-191, and chlorine-38 for medical research, diagnostic and therapeutic purposes.

f) Irradiation (i) of tissue specimens on particle track detectors for plutonium radiobiology, (ii) of agricultural specimens and animal tissue for boron location, and (iii) of geological samples for fissile element distribution.

g) Irradiation of amorphous hydrogenated silicon (a-Si:H) to produce some phosphorous in order to study the effect of such donor atoms on the properties of a-Si:H.

h) Use of the facility for reactor operator training.

i) Irradiation damage studies of candidate fusion reactor materials.

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.

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.

l) Experimental studies of various closed-loop control techniques including decision analysis, state-variable feedback, and the use of reactivity constraints.

m) 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) Detection of trace quantities of fissile nuclides using a delayed neutron detector.

3. Changes to Facility Design

As indicated in past reports the uranium loading of MITR-II fuel has been increased from 29.7 grams of U-235 per plate and 445 grams per element to a nominal 34 and 510 grams respectively. With the exception of two elements (plus a third that was found to be out-gassing excessively in July 1985 after the end of the report period), performance has been good. (Please see Reportable Occurrence Reports Nos. 50-20/79-4, 50-20/83-2 and 50-20/85-2.) 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. The most recent fuel fabricator, Atomics International Division of Rockwell International, has completed the production of 41 of the more highly loaded elements, 36 of which have been used to some degree. Three with about 37% burnup, had been in operation in the core since January 1980 and were discharged during the year, since they had attained the burnup limit. Additional elements are now being fabricated by Babcock & Wilcox, Navy Nuclear Fuel Division.

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 current 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.

Other changes in the facility are reported in Section E.

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

Amendment No. 24 to the Facility Operating License was issued on April 2, 1985. It revises Technical Specification 3.9 to authorize automatic control of shim blades and regulating rod with up to 1.8% $\Delta K/K$ in available positive reactivity. It also adds a new Technical Specification 6.4 that authorizes connection of the shim blades and/or regulating rod to a closed-loop controller provided that the overall controller is designed so that control of reactor power will always be feasible at either the desired termination point of any transient or at the maximum allowed operating power.

MIT is awaiting approval of its application for renewal of License No. SNM-986. This license covers kilogram quantities of slightly enriched U-235, gram quantities of Pu, normal and depleted U. Other licenses covering smaller quantities of similar material would be combined with License SNM-986. The MIT Reactor is involved, because most of the SNM is stored on the reactor site, and much of it is used on the reactor in accordance with authorized experiment review and approval procedures. Revision #1 of the renewal application was submitted to NRC on July 13, 1984.

Two SAR revisions were submitted during the year:

a) SAR Revision No. 31 was submitted to NRC in order to update the SAR so that it would reflect several minor changes that have been incorporated in procedures, related documents, and drawings. The following Table 1 (Enclosure 1 of Revision No. 31) summarizes the changes.

b) SAR Revision No. 32 changes the amount of available positive reactivity that may be connected to an automatic controller from 0.7% to 1.8% $\Delta K/K$ (Section 3.3.2.1.6) and describes closed-loop control of the reactor by means of digital controllers that incorporate the concept of "feasibility of control" which in turn, is based on constraining reactivity within manageable limits (Sections 10.1.6 and 10.2.5). Both of these changes have been authorized by the above Amendment No. 24 to the Facility Operating License.

   SAR Revision No. 32 also incorporates the use of variable speed motors in the regulating rod and shim blade drives, a 10 CFR 50.59 change described in Section E. Additional minor changes are described in Table 2 (Summary of SAR Changes - SR #0-84-12).

Table 1

Enclosure 1

Mr. C. Thomas, USNRC, Division of Licensing, (1/4/85)

SAR Revision No. 31

Remove Page	Insert Page	Description of Change
Fig. 9.3.1-1 (9/24/76)	Fig. 9.3.1-1 (9/16/83)	Figure updated to include valve DV-69 which is normally locked open but can be used to isolate the helium cover gas system when the blow off patch is serviced.
11.3-2 (3/17/72)	11.3-2 (10/26/84)	The name of the checklist used to document QA activities has been made more descriptive of its use, i.e. "Quality Assurance Approval Requirements Checklist".
11.3-3 (5/6/82)	11.3-3 (10/26/84)	A footnote has been added as a reminder that the uncertainty allowed in the fuel density tolerance was actually 1.10 rather than the 1.05 envisaged when the SAR was first written.
11.10-3 (3/17/72)	11.10-3 (10/26/84)	Same as 11.3-2.
11.17-2 (6/30/78)	11.17-2 (10/26/84) & 11.17-3 (10/26/84)	Table 11.17-1 revised to reflect records retention requirement of nuclear insurer (policy termination plus 10 years). Retention time for Items 2 and 5 reduced to reflect actual requirements. Name of checklist in Item 17 updated.
(none) (2/10/81)	11.17-4 (10/26/84)	Fig. 11.1-1 revised to show MIT Radiation Protection Committee (inadvertently omitted) and changed Environmental Medical Service reporting structure.

Table 2

Summary of SAR Changes - SR #0-84-12

SAR Revision No. 32

Page #	Changes
3.3.2-3	Section 3.3.2.1.6 rewritten to reflect change in Technical Specification #3.9-5
7.1.9-1	Section 7.2 change to reflect use of boron-impregnated stainless steel blades. (This completes a previous change made earlier.)
7.2-1	Modifies section 7.2 to include variable speed motors in the control system and shim blades in the automatic control system. Corrects typographical error in last line of page.
7.3-1	Changes wording to reflect use of constant or variable speed motors. Adds paragraph listing modes of control.
7.4-2	Changes frequency of building leak test from biannual to annual which is as specified in the Technical Specifications. Adds specification for damper leak test.
	Changes frequency of emergency cooling flow test to annual which is as specified in the Technical Specifications. Delineates specification for the test.
7.4-3	Changes frequency of calibration of reactor outlet temperature switches to annual which is as specified in the Technical Specifications. Deletes word bimetallic since capillary switches are used.
10.6c	New page.
10.6d	" "
10.8c	" "
10.8d	" "
10.31b	" "
10.32	Adds reference 10.1.6-1.

SR# O-84-12

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With respect to operating procedures subject only to MITR internal review and approval, a summary of those related to safety is given below:

a) An annual independent audit of reactor activities was initiated in 1983 to supplement the internal audits previously conducted by the reactor staff. In 1984 Procedure 1.18.2, "Independent Audit", was written to formalize the audit procedure, and Procedure 1.18.1 was revised to include therein the several internal audits routinely being carried out. (SR #0-84-15)

b) Procedure 6.1.3.4B, "Reactor H_2O Outlet Temperature - MTS-1 and MTS-1A", formerly required the placement of one thermocouple in the core tank and a second in an ice bath as a reference while the primary coolant temperature (and consequently the probes for MTS-1 and 1A) was varied in the range 25°-55°C. The procedure has been revised to permit the use of any calibrated temperature measuring instrument. A Fluke Data Logger calibrated by an independent testing laboratory is now used. (SR #0-84-17)

c) Procedures 1.13.7, "QA Records", and 1.20, "Records Preservation and Retention", were clarified in response to a finding in the 1983 independent audit. Some records retention times were increased in order to meet nuclear insurer requirements. (SR #0-84-18)

d) New paragraph 1.13.9 was added to Procedure 1.13, "Quality Assurance Program", to state explicitly that the QA Program is applicable to packaging for Type B quantities of RAM and to fissile materials not otherwise exempted. New paragraph 1.13.10 was added to specify that radioactive wastes at the reactor are subject to the classification and other requirements of 10 CFR 20.311(d) for disposal by land burial. (SR #0-84-18)

e) The drill scenario used in Procedure 6.6.1.1, "Radiological Emergency Exercise", was updated to reflect the revised action levels contained in the Radiation Emergency Plan as most recently approved by NRC. Other parts of the procedure were revised for purposes of clarification and to add a number of precautions to be observed in the conduct of the drill. (SR #0-84-19)

f) The readouts for Core Tank Level meter ML-3A originally gave the level in terms of inches below overflow. The scales have been redesigned to give the level in terms of feet above the top of the core. Procedure 4.4.5.1, "Instructions for Use of Utility Room Emergency Gauges", was revised accordingly. (SR #0-84-22)

g) A review of Procedure Manual Chapter 1, "Administrative Procedures", resulted in updates to name lists and other non-substantive revisions. Section 1.4.6, "Procedure Manuals", was revised to specify where official copies of individual procedures are posted, e.g. the perchloric acid hood, the reactor floor hot cell. Section 1.19, "Receiving, Storing and Issuing of NRL Materials", was revised to eliminate duplication with Chapter 2 regarding procedures for receipt of reactor fuel. References to NRC licenses held by MIT, pertinent Parts of 10 CFR, and related written plans that might need to be consulted regarding the receipt, storage and issuing of nuclear materials were added. (SR #0-85-1)

h) Procedure 2.7.1, "Procedure for Receipt of Reactor Fuel", was revised to incorporate those items previously contained in Procedure 1.19.1. Other minor revisions, mostly editorial, were made in the remaining sections of Procedure 2.7, "Fuel Handling". (SR #0-85-2)

i) As the result of errors in sample identification that resulted, in one case, in the shipping of the wrong sample (see Unusual Occurrence Report #84-3 and NRC Inspection No. 50-20/85-01), a new Procedure 3.11.5, "Identification of Pneumatic Tube Samples", was prepared that consolidated existing memos and verbal directives and contained new requirements regarding identification of samples. (SR #0-85-3)

j) An editorial correction in the chart specifying conditions leading to radiological emergency Actions 1X or 2X or AOP 5.6.2 was made in Procedures 4.7.2 and 4.4.4.15 and in AOP 5.6.2. (SR #0-85-4)

k) The scale on the remote primary storage tank level indicator, ML-4, has been redesigned to give a direct reading of the storage tank contents in inches, instead of a psig reading that had to be converted. Procedure 6.5.12, "Primary Storage Tank Level Calibration", was revised accordingly. (SR #0-85-5)

l) There previously had been no formal procedure for calibration of the shield storage tank level remote indication and no formal test for the Shield Tank Low Level alarm, PL-1. A new Procedure 6.5.13, "Shield Storage Tank Level Calibration", similar to that for the primary storage tank, was prepared and instituted. (SR #0-85-5)

m) Formerly, the D_2O dump tank remote level indicator was calibrated by varying the dump tank level over its range, and the Low Level Dump Tank Alarm and the Transfer Pump interlock were tested by pumping the dump tank down until the level probe, DL-2, was uncovered. Considerable valving and pumping is avoided by isolating the dump tank sight glass, DL-1, and attaching a movable D_2O reservoir that can be raised or lowered to vary the D_2O level in DL-1. This is the same method that is now used for the primary coolant storage tank and the shield coolant storage tank. Procedure 6.3.8, "Reflector Dump Tank Level Alarm, Interlock and Calibration", was rewritten to reflect this new method. (SR #0-85-6)

n) Formerly, calibration of the core tank level scram point and the level indications was accomplished, using Procedure 6.1.3.7, by observing meter indications as the tank level was lowered in steps of about 2" each to just above the outlet pipe (radiation levels permitting). Differential pressure transmitters, ML-3A and ML-3B, respond to coolant level in the core tank through helium backpressure that depends on the coolant level. By isolating the helium and substituting air whose pressure can be varied with a regulator over

120 inches H_2O (10 feet), as determined by a precision manometer, the control room and utility room level indicators can be calibrated and the Low Level Core Tank scram point verified. The advantage is that the reactor top lid need not be removed, and radiation exposures are reduced. Procedure 6.1.3.7A, "Calibration of Core Tank Level Indications ML-3A, ML-3B, and Verification of Low Level Scram Point", was issued as an alternate to Procedure 6.1.3.7. (SR #0-85-7)

o) A one-time procedure was reviewed and approved for determining the neutron transmissin characteristics of an irradiated MITR-II shim blade. Measurements were made in the spent fuel storage tank. (SR #0-85-8)

p) Procedure 6.1.2.2, "Main Ventilation Damper Inspection", which called for a light transmission test of the damper gasket seal, was revised to include also and explicitly a visual inspection of the condition of the gasket. This constitutes one of the corrective actions described in Reportable Occurrence Report #50-20/85-01. (SR #0-85-9)

q) A new Procedure 7.3.5, "Procedure for Replacement of the D_2O Recombiner Blower", was prepared and approved for the above operation. (SR #0-85-10)

r) Several emergency procedures in Section 4.4.4 were revised at the direction of the MIT Reactor Safeguards Committee (page 6 of minutes of meeting dated 12/19/84) to (1) clarify the wording of tables used to determine the class of an emergency, (2) delete the procedure step to open all air locks in the event that a tornado is anticipated and (3) make minor changes to clarify other wording. (SR #0-85-12)

s) A new Procedure 7.4.3.7, "Flushing of Heat Exchangers HM-1 & HM-1A", was prepared and approved to flush the above two heat exchangers with maximum secondary flow without exceeding the primary coolant pressure. (SR #0-85-14)

t) 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. During the containment building pressure test in April 1985, the seal on the truck lock outer door could not be tested due to a hydraulic line failure. The line has since been repaired and preparations are being made for a special pressure test of the truck lock.

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 FY85) (normally at 4.9 MW)	182.1	190.4	225.8	195.0	793.2
b) MITR-II (MIT FY76-84)					6,970.5
c) MITR-I (MIT FY59-74)					10,435.2
d) Cumulative, MITR-I & MITR-II					18,198.9
2. Hours of Operation MIT FY1985, MITR-II					
a) At Power (>0.5 MW) for research	1161.9	1061.1	1148.7	1114.7	4,486.5
b) Low Power (<0.5 MW) for training(1) and test	48.1	33.6	36.2	39.0	156.9
c) Total critical	1210.0	1094.7	1184.9	1153.7	4,643.4

Note (1): 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 previous line.

C. SHUTDOWN AND SCRAMS

During the period of this report there were 7 inadvertent scrams and 3 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 last year in order to facilitate a comparison.

		Total
I.	Nuclear Safety System Scrams	
a)	Chan. 6 scram at 4.2 MW due to trip malfunction while reshimming	1
b)	Blade 6 dropped off while meter adjusted	1
c)	Withdraw permit open on relay failure	1
	Subtotal	3
II.	Process Systems Scrams	
a)	Low Flow Primary Coolant scram during servicing of flow recorder	1
	Subtotal	1
III.	Unscheduled Shutdowns or Power Reductions	
a)	Shutdowns due to Electric Company power loss	3
b)	Operator lowered power to investigate:	
i)	Medical shutter operation	1
ii)	Low pressure in the helium supply to an irradiation thimble	2
	Subtotal	6
	Total	10

The 10 scrams and shutdowns during FY 85 compare with the 19, 25, and 28 experienced in FY 84, FY 83, and FY 82 respectively.

D. MAJOR MAINTENANCE

Major maintenance projects during FY86, including the effect, if any, on safe operation of the reactor, are described below in this section.

FY85 saw a continuation of the efforts in repairing the cooling towers which have been deteriorating due to age and ice forming on the outside panels. Exterior panels on both cooling towers were repaired and steam cleaned. Drain holes were drilled on the north-east side of cooling tower #1 where the majority of the ice had built up in the winter. Installation of the drain holes allows water to return to the interior of the cooling tower thereby reducing water accumulation on the outside of the tower where ice could form. Interior poly-grid filling in cooling tower #1 was repaired so as to restore the water distribution capacity of the tower. Spray rings and nozzles on both towers were inspected and cleaned. The underground valves to the cooling tower basins had ceased to operate due to age. Replacement of these valves was initiated in FY85 and will be completed in FY86. There are a total of 2 eight inch, 2 six inch, and 2 three inch butterfly valves to be replaced.

Diaphragms in the pneumatic operator of the automatic solenoid operated valves in the primary, core purge, and medical shutter systems were replaced as a preventive maintenance measure. The transmitter for the reflector level indication system was replaced with a new unit because repair parts for the old unit are no longer supplied by the manufacturer.

The piping for the dump tank level indication was modified so as to allow calibration of the remote level indication without actual inventory change in the dump tank. New stainless steel clad thermocouples were obtained for calibration of the two temperature sensors in the primary core tank. The new calibration procedure and set-up allows calibration of the temperature probes without removing the reactor top shield lid. These two improvements reduce much of the radiation exposure to the operators who are performing these calibrations.

The hydraulic lines for the truck lock of the reactor containment building ruptured during routine use of the lock. The steel hydraulic lines were replaced. The hydraulic hoses on the ventilation intake damper showed signs of deterioration and were replaced. The gaskets which form an air tight seal in the intake damper were found to be worn and were replaced with a new set. The accumulator in the hydraulic system of the ventilation exhaust damper developed a nitrogen leak. The diaphragm of the accumulator was replaced. The core purge blower in the primary system was replaced due to excessive wear inside the housing on which the graphite vanes of the rotor ride. The timing of the lobes in the D_2O helium recombiner blower became out-of-synchronization due to normal wear, and the blower was replaced with a new unit.

Many other routine maintenance and preventive maintenance jobs were done 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 Chief, Standardization and Special Projects Branch, Division of Licensing, 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".

For the past year the only facility changes and experiments carried out under Section 50.59 were in connection with the closed-loop computer control project described on the following pages:

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), O-83-5 (02/03/83), E-83-1 (02/08/83), O-83-12 (04/23/83), O-83-20 (07/20/83), O-84-11 (06/25/84), O-84-12 (07/12/84), O-84-16 (12/6/84), O-84-21 (11/1/84), O-85-11 (5/9/85), and O-85-13 (6/28/85).

A joint project involving computer analysis, signal validation of data from reactor instruments, and closed-loop control of the MIT Reactor by digital computer was continued with the Charles Stark Draper Laboratory in Cambridge. 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, "fuzzy" logic, 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.

In initial tests, the digital controller was designed to control the reactor's regulating rod. That rod, whose worth is limited to 0.7% $\Delta K/K$ and is actually worth 0.2% $\Delta K/K$, is normally positioned by an analog controller. The digital controller has been shown to be equal to the analog one during near steady-state conditions while transients such as those due to xenon or temperature are in progress.

Some of the subsequent tests were or will be conducted under the conditions of 10 CFR 50.59, i.e. consideration was given to the possible existence of unreviewed safety questions:

1) Experiments were conducted in which the digital controller was connected to a shim blade drive (instead of the regulating rod, as described above). This required initially that the available positive reactivity that could be inserted if the blade were to be fully withdrawn be limited to 0.7% $\Delta K/K$ or less. It was concluded that no unreviewed safety questions were involved, including consideration of any increased probability of continuous blade withdrawal and any increased probability of excessive positive reactivity insertion. See also Safety Review #0-84-11, submitted to NRC, Standaridization and Special Projects Branch, Division of Licensing, on January 11, 1985.

2) Further experiments will involve the use of a variable speed motor on a shim blade in place of the usual constant speed motor that can insert reactivity at a maximum rate of $1x10^{-4} \Delta K/K$ per second. The variable speed motor, in out-of-core tests, could not exceed an insertion rate of $4x10^{-4} \Delta K/K$ per second, less than the Technical Specification limit of $5x10^{-4}$ $\Delta K/K$ per second. Based on this and other considerations detailed in safety review #0-84-11, no unreviewed safety questions have been identified.

In order to conduct further tests with available positive reactivities exceeding 0.7 $\Delta K/K$, a facility operating license amendment was submitted to NRC (January 11, 1985) that would:

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

(2) permit 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.

Amendment No. 24 authorizing the above activities was issued by NRC on April 2, 1985. Pertinent pages of the SAR that concern the reactor control system have been updated through submission with the January 11, 1985 letter of SAR Revision No. 32.

F. ENVIRONMENTAL SURVEYS

Environmental surveys, outside the facility, were performed using area monitors. The systems (located approximately in a ½-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, 1984 - June 30, 1985
North	1.1 mR/year
South	3.6 mR/year
East	4.7 mR/year
West	1.0 mR/year
Green (East)	0.4 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/year
1984	2.1 mR/year
1985	2.2 mR/year

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

Whole Body Exposure Range (Rems)	Period 7/01/84 - 6/30/85 No. of Personnel
No Measurable	89
Measurable - Exposure less than 0.1	30
0.1 - 0.25	8
0.25 - 0.5	14
0.5 - 0.75	3
0.75 - 1.0	1

Total Personnel - 145 Total Man Rem = 9.95

Summary of the results of radiation and contamination surveys from July 1984 to June 1985:

During the 1984-1985 period, the Reactor Radiation Protection Office continued to provide radiation protection services necessary for full-power (5 megawatts) operation of the reactor. Such services (performed on a daily, weekly, or monthly schedule) include the following:

1. Collection and analysis of air samples taken within the reactor containment shell, and in the exhaust-ventilation system.
2. Collection and analysis of water samples taken from the reactor 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. The 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 three sources of such wastes during the year: the cooling tower blowdowns; the liquid waste storage tanks; and laboratory drains. All of the liquid volumes are measured, by far the largest being the 4,213,000 liters discharged during FY 1985 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 4076 Ci of Ar-41 was released at an average concentration of 1.04×10^{-8} µCi/ml for the year. This represents 26% of MPC (4×10^{-8} µCi/ml) and is only about half the previous year's release. The improvement is the result of extensive studies of the sources generating Ar-41 in the reactor and of the inert gas blanket systems that minimize the generation.

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 1985

	Ar-41 Discharged (Curies)	Average Concentration[1] (μCi/ml)
July 1984	381	1.28×10^{-8}
August	689	1.85
September	442	1.49
October	453	1.22
November	173	0.58
December	217	0.73
January 1985	280	0.75
February	268	0.90
March	265	0.89
April	182	0.61
May	363	0.97
June	363	1.22
12 months	4076	1.04×10^{-8}
MPC (Table II, Column I)		4×10^{-8}
% MPC		26%

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

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS FISCAL YEAR 1985

	UNITS	SHIPMENT #1	TOTAL
1. Solid waste packaged	Cubic Feet	120	120
2. Total activity (irradiated components, ion exchange resins, etc.) ^{60}Co, ^{51}Cr, $^{55-59}Fe$ ^{65}Zn, etc.	(Ci)	0.067	0.067
3. (a) Dates of Shipment		3/26/85	
(b) Disposition to licensee for burial		Radiation Service Organization	