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AN INTERDEPARTMENTAL CENTER OF
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September 22, 1995

U.S. Nuclear Regulatory Commission,
Washington, D.C. 20555

ATTN: Document Control Desk

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

Gentlemen:

Forwarded herewith is the Annual Report for the MIT Research Reactor for the period July 1, 1994 to June 30, 1995, in compliance with paragraph 7.13.5 of the Technical Specifications for Facility Operating License R-37.

Sincerely,

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MIT RESEARCH REACTOR
NUCLEAR REACTOR LABORATORY
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

ANNUAL REPORT

to

United States Nuclear Regulatory Commission for the Period July 1, 1994 – June 30, 1995

by

REACTOR STAFF

August 30, 1995

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MIT RESEARCH REACTOR

ANNUAL REPORT TO

U. S. NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1994 – JUNE 30, 1995

INTRODUCTION

This report has been prepared by the staff of the Massachusetts Institute of Technology Research Reactor for submission to the Administrator of Region I, 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 to 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 14, 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 twentieth annual report required by the Technical Specifications, and it covers the period July 1, 1994 through June 30, 1995. 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 eighteenth 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 Sections A-I of this report.

A. SUMMARY OF OPERATING EXPERIENCE

1. General

The MIT Research Reactor, MITR-II, has in recent years been operated on a routine, five days per week schedule, modified as necessary to facilitate the preoperational testing and installation of several in-core experiments. When operating, the reactor is normally at a nominal 5-MW. However, as was the case for the last five years, substantial departures were made from this schedule during the period covered by this report (July 1, 1994 - June 30, 1995). Specifically, for much of this reporting period, the reactor was run at full power almost continuously (160 hours/week). This schedule was followed in order to support a major experimental program concerning the development of methods to reduce the activation and transport of corrosion products in pressurized water reactor coolant. The period covered by this report is the eighteenth full year of normal operation for MITR-II.

The reactor averaged 107.6 hours per week at full power compared to 46.4 hours per week for the previous year and 61.2 hours per week two years ago. As was the case in FY 94 a lot of operation was conducted at low power in order to make measurements of the medical therapy facility beam. These measurements are for the purpose of maintaining an epithermal neutron beam for the treatment of brain cancer (glioblastoma multiforme) and possibly skin cancer (melanoma). When neither the corrosion reduction experiments nor the medical beam design was in progress, the reactor was usually operated from late Monday afternoon until late Friday afternoon, with maintenance scheduled for Monday mornings and, as necessary, for Saturdays.

The reactor was operated throughout the year with 24 elements in the core. The remaining three positions were used as follows: position A1 was occupied, during the early part of FY 95 by the Pressurized Water Reactor (PWR) Coolant Chemistry Loop (PCCL). This loop reproduces chemistry conditions in power reactors and is part of a major effort to identify methods for reducing radiation exposures in the nuclear industry. The second non-fueled position, B3, was occupied by the Irradiation-Assisted Stress Corrosion Cracking Facility (IASCC) during July-October 94 and later (December 94, February 95, and March-June 95) occupied by the SENSOR facility. This latter experiment, which complements the IASCC experiment, seeks to determine if a relation exists between crack propagation and electrochemical potential (ECP), and also whether hydrogen injection can arrect crack growth. The third core position, A3, was occupied by a solid aluminum dummy as were positions A1 and B3 whenever the PCCL or IASCC/SENSOR were not installed. Compensation for reactivity lost due to burnup was provided by six refuelings. These followed standard MITR practice which is to introduce fresh fuel to the inner portion of the core (the A- and B-Rings) where peaking is least and to place partially spent fuel in the outer portion of the core (the C-Ring). In addition, elements were inverted and rotated so as to achieve more uniform burnup gradients in those elements. Six other refuelings were performed for the purpose of making accurate reactivity measurements and trial fits of the various Coolant Chemistry Loop experimental facilities.

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%. (Note: One element was removed prematurely because of excess outgassing.) The maximum overall burnup achieved was 48%. Seventy-seven of the newer, higher loaded elements (506 grams

U-235) have been introduced to the core. Of these, thirty-four have attained the maximum allowed fission density. However, some of these may be reused if that limit is increased as would seem warranted based on metallurgical studies by DOE. Another seven have been identified as showing excess outgassing and two are suspected of this. All nine have been removed from service. The other thirty-four are either currently in the reactor core or have been partially depleted and are awaiting reuse in the C-ring.

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) 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 studies of the role of metallic and organo-metallic groups in the final properties of polymers.

c) Use of neutron activation analysis to determine the concentrations of heavy metals in sludge from sewage treatment plants.

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

e) Production of 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) Evaluation of various chemical additives for the suppression of nitrogen-16 activity in a boiling water reactor environment.

k) Use of trace analysis techniques to identify and monitor sources of acid deposition (rain).

l) Operation of an in-core slow strain rate testing rig to evaluate irradiation-assisted stress corrosion cracking of metals.

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.

n) Gamma irradiation of seeds for demonstration of radiation damage effects for high school students.

o) Experimental evaluation of flux synthesis methods as a means of estimating reactivity.

p) Evaluation of scintillation fluids for use in neutrino detectors.

q) Neutron activation analysis of serum samples in an effort to correlate mineral deficiencies with certain diseases.

r) Determination of uranium concentrations in samples of mica.

Dose reduction studies for the light water reactor industry began reactor use on a regular basis in 1989. (Planning and out-of-core evaluations had been in progress for several years.) These studies entail installing loops in the reactor core to investigate the chemistry of corrosion and the transport of radioactive crud. Loops that replicate both pressurized and boiling water reactors have been built. The PWR loop has been operational since August 1989. The BWR loop became operational in October 1990. A third loop, one for the study of irradiation-assisted stress corrosion cracking, became operational in June 1994 and a fourth one, also for the study of crack growth, in April 1995.

Another major research project that is now making and will continue to make extensive use of the reactor is a program to design a facility for the treatment of glioblastomas (brain tumors) and melanomas (skin cancer) using neutron capture therapy. This is a collaborative effort with the Tufts – New England Medical Center.

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 seven elements (one Gulf, six 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, 50-20/88-1, and 50-20/91-1.) 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 forty-one of the more highly loaded elements in 1982, forty of which have been used to some degree. Twenty-eight with about 40% burnup have been discharged because they have attained the fission density limit. Four other AI elements remain in use. Of the other eight, six were, as previously reported to the U.S. Nuclear Regulatory Commission, removed from service because of excess outgassing and two were removed because of suspected excess outgassing. Additional elements are now being fabricated by Babcock & Wilcox, Navy Nuclear Fuel Division, Lynchburg, Virginia. Thirty-seven of these have been received at MIT, twenty-nine of which are in use. One has been removed because of suspected excess outgassing and seven have been discharged because they have attained the fission density limit.

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

One amendment to the facility operating license was received during the past year. It involved a request that Facility Operating License No. R-37, which was due to expire in May 1996, be extended in order to recover time during which the reactor was either under construction, shutdown for modification, engaged in low-power testing that was a prerequisite for operation at the authorized power level, or engaged in core modification found to be necessary as a result of the low-power testing. An amendment to modify the frequency of certain of the surveillances for the emergency battery was filed in February 1995. It is discussed in Secton E of this report.

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) PM 6.4.1, "Low Air Pressure Cooling Tower Sprinkler/Cooling Tower Sprinkler On" was revised to improve clarity and to incorporate tolerances on various system pressures. (SR#0-94-16)

b) The shadow shielding that runs along the perimeter of the facility's parking lot was modified to improve pedestrian visibility. This change had no effect whatsoever on reactor shielding requirements because the portion that was altered was not in any line-of-sight from a radiation source. (SR#0-94-17)

c) The core tank level probe was originally designed so that it vented to the exterior of the core tank. Such venting is necessary to preclude the buildup of gas that might cause spurious operation. This arrangement was modified so that the venting is now done to the air space within the reactor core tank. (SR#0-94-20)

d) The Administrative Procedures, Chapter One of the Procedure Manual, were revised to update the lists of names and committee memberships. (SR#0-95-1)

e) PM 6.1.3.10, "Emergency Battery Discharge Test," PM 6.1.3.11, "Emergency Power Transfer Test," and PM 3.5, "Daily Sur eillance Check" were revised to reflect the installation of a new emergency battery. (SR#0-95-2)

f) Several minor changes were made to PM 4.4.4.15," Escape of Radioactive Material from the Containment Building" and PM 5.6.2, "High Level Radiation Monitors." These changes reflected the installation of a new stack area monitor and other equipment changes. In addition, increased emphasis was placed on the early determination of the class of the emergency. (SR#0-95-3)

g) PM 3.11.2, "3GV Sample Procedure" was revised to improve ALARA considerations in the installation and removal of samples from the 3GV irradiation facilities. (SR#0-95-5)

h) PM 3.1.3, "Startup for Less than 100 kW Operation" was modified to include (1) an annual determination of the amplification factors for the nuclear safety system's low-range amplifiers and (2) updated values for the core purge and shield coolant system flows. (SR#0-95-6)

i) PM 6.1.3.14, "Determination of Low-Range Amplification Factor" was created as part of the corrective action for ROR #95-2. This procedure provides a standard method for verifying the amplification factor of the low-range amplifiers that are used in the nuclear safety system whenever operating in t' ; natural circulation mode at less than 100 kW. (SR#0-95-6)

j) The regulating rod, which is used for the fine control of reactor power, is made of cadmium. Slight changes in the design of this rod were necessitated by new OSHA regulations that govern the welding of heavy metals such as cadmium. (SR#M-95-1)

k) In FY93, a new reactor lid was designed for use with the Irradiation-Assisted Stress Corrosion Cracking Experiment (IASCC). A structural analysis of this lid was performed under the supervision of Prof. Griffith (MIT Dept. of Mechanical Engineering) during the past year. This analysis showed that this new lid can be used to support the fixtures needed for transfer of spent fuel from the core tank to the fuel storage pool. (SR#M-95-2)

l) An in-core thermocouple is used to calibrate various temperature instruments. During the past year, a high temperature scram (55°C) was installed in conjunction with this thermocouple. This provides a total of four high temperature scrams on the primary coolant, any two of which are legally required. (SR#E-95-1)

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 shutdown 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.

7. Status of Spent Fuel Shipment

Pursuant to Amendment No. 25 to Facility Operating License No. R-37, paragraph 2.B.(2) subparagraph (b), reported herewith is the status of the establishment of a shipping capability for spent fuel and other activities relevant to the temporary increase in the possession limit.

MIT began efforts for spent fuel shipment as early as 1983. At that time, the plan was to use two MH-1A casks that had been acquired by DOE and which were being prepared for use by the non-power reactor community. After an MH-1A cask became unavailable, MIT made arrangements with General Electric to use the GE-700 cask for shipment of the MITR spent fuel. When the GE-700 cask was removed from service voluntarily by GE, the BMI-1 cask became the only one available that is approved for transportation of irradiated fuel elements.

The capability to ship spent MITR fuel was established by the end of 1992. Specifically, the following was accomplished:

(a) The Certificate of Compliance and the Safety Analysis Report of the BMI-1 cask were reviewed by MIT and the cask was determined to be acceptable for shipping MITR spent fuel. Arrangements have been made with DOE for MIT to use this cask.

(b) The University of Missouri Research Reactor (MURR) basket was reviewed and found to be suitable for use with the MIT fuel elements in the BMI-1 cask. MURR has agreed to make their basket available to MIT for the required shipments.

(c) A quality assurance program for MITR-II spent fuel shipment was prepared and approved under the MITR safety review program. This Q/A program was approved by NRC on July 23, 1991.

(d) The decay heat load of each spent element was determined by a member of the MITR staff and found to be within the limits specified in the Certificate of Compliance for the cask. Radiation shielding calculations were also performed and radiation levels associated with the loaded cask were estimated to be within allowed limits. Criticality calculations were performed using the Monte-Carlo Code KENO-V which was obtained from the Radiation Shielding Information Center of the Oak Ridge National Labora tory. Results show that the degree of subcriticality of a cask fully loaded with MIT fuel elements is within specification.

(e) In order to cross-check the cross sections used in the KENO-V code, criticality analyses were performed using a second Monte-Carlo code, MCNP. Results obtained were consistent with those obtained using KENO-V.

(f) Arrangements have been made with the fuel receiving organization at the Savannah River facility. Specific data on the MITR-II spent fuel elements were compiled. The Appendix A document and criticality study were prepared and reviewed by the spent fuel processing center.

(g) Spent fuel elements in the MITR spent fuel storage pool were arranged and grouped in accordance with our procedure for shipment preparation. A special structure for support of the BMI-1 cask was designed and fabricated.

(h) A third fuel storage rack, which has a capacity of twenty-five fuel elements, was built and installed in the spent fuel storage pool.

(i) License Amendment No. 25 which provided a temporary increase in the possession limit was extended to 31 December 1993.

(j) A criticality study of the BMI-1 cask with fresh MITR fuel was completed and approved by the U.S. Department of Energy.

(k) Funding was allocated by the U.S. Department of Energy for the return to a DOE facility of spent MITR fuel.

(l) Procedures for spent fuel shipment were prepared.

(m) A proposed route was reviewed and approved by NRC. All necessary State and City permits were obtained.

Six shipments of eight elements each were completed during the early part of 1993. In each case, the spent fuel was returned to the U.S. Department of Energy's facility at Savannah River, SC. As a result of these shipments, the on-site inventory of U-235 was sufficiently reduced so that there was no longer any need for the temporary increase in the possession limit that had been obtained under License Amendment No. 25. Accordingly, that amendment expired on 31 December 1993.

In early 1994, one shipment of eight elements was completed. No shipments have yet been made in 1995. Nor are any planned. At present, several additional shipments are needed in order to reduce the inventory of spent fuel at MIT to zero. However, it is currently unclear as to when or even if these shipments will occur. The problem is that the U.S. Department of Energy (DOE) has stopped the reprocessing of spent fuel and it has only limited storage space available. DOE is currently evaluating various options that would allow continued returns of spent fuel and MIT will notify NRC of the DOE decision as soon as it is known. DOE has indicated that a shipment may be possible in 1996.

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 FY95) (normally at 4.9 MW)	260.1	244.3	265.5	306.9	1076.8
b) MITR-II (MIT FY76-94)					13,185.3
c) MITR-I (MIT FY59-74)					10,435.2
d) Cumulative, MITR-I & MITR-II					24,697.3
2. MITR-II Operation (Hrs): (MIT FY95)					
a) At Power (>0.5-MW) for Research	1476.3	1224.6	1320.6	1572.9	5594.4
b) Low Power (<0.5-MW) for Training[1] and Test	56.7	45.4	58.9	45.4	206.4
c) Total Critical	1533.0	1270.0	1379.5	1618.3	5800.8

[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 the previous line.

C. SHUTDOWNS AND SCRAMS

During the period of this report there were 17 inadvertent scrams and 28 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.

			Total
I.	Nuclear Safety System Scrams		
	a)	Channel #3 trip as result of electronic noise.	1
	b)	Channel #4 trip as a result of magnet current noise.	1
	c)	Channel #4 trip as result of overly conservative setting.	1
	d)	Channel #1 and/or #6 trip as result of #6 amplifier malfunction.	6
		Subtotal	9
II.	Process System Scrams		
	a)	ML-3 trip due to vent line blockage.	4
	b)	MTS-1 high temperature trip as result of instrument failure.	1
	c)	Main personnel lock gaskets deflated due to improper valving.	1
	d)	Low flow primary coolant trip as result of operator error while inspecting flow recorder.	1
	e)	Low flow primary coolant trip as result of flow recorder failure.	1
		Subtotal	8

III. Unscheduled Shutdowns or Power Reductions

a)	NTD Silicon machine malfunctions.	18
b)	Shutdown due to loss of offsite electricity.	6
c)	Blade #4 drop.	1
d)	Shutdown to repair main intake damper.	1
e)	Low power to fix primary leak at MM-1A.	1
f)	Bomb threat.	1
	Subtotal	28
	Total	45

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

Fiscal Year	Number		
	Scrams	Shutdowns	Total
91	11	9	20
92	5	12	17
93	6	14	20
94	13	32	45
95	17	28	45

D. MAJOR MAINTENANCE

Major maintenance projects performed during FY 95, including the effect, if any, on the safe and reliable operation of the MIT Research Reactor are described in this Section.

One of the FY 95 maintenance items was the preparation and installation of an experiment facility in the reactor core for the SENSOR project. This project, which tests response of sensors used in the previously completed irradiation-assisted stress corrosion cracking experiments, was initially installed in the reactor core on December 12, 1994.

Much maintenance was performed to support the ongoing research program to identify improved water chemistries that will result in reduced radiation exposure to workers in the nuclear industry. This project now involves four in-core experiments. These are the Pressurized Coolant Chemistry Loop (PCCL), the Boiling Coolant Chemistry Loop (BCCL), the Irradiation-Assisted Stress Corrosion Cracking Facility (IASCC), and the SENSOR facility.

The repair and maintenance of machinery for neutron transmutation doping of silicon also required support. This machinery, installed in two of the reactor throughports, includes two twenty-foot tubes for each port, rotating and pushing mechanisms, billet handling and storage conveyors, electronics, and associated microprocessor-based controllers.

Other major maintenance items performed in FY 95 were as follows:

(i) A new radiation monitor, in accordance with the recommendation of the MIT Reactor Safeguards Committee, was installed in the containment building ventilation exhaust stack area. This monitor is capable of registering abnormal radiation levels in the event of an accident leading to the release of radioactivity.

(ii) The emergency battery bank, consisting of sixty batteries capable of supplying electric power to essential reactor components in the event of a loss of off-site power, was replaced in its entirety.

(iii) New stairs leading to the top of the reactor cooling towers were erected and the old staircase was removed.

(iv) Additional lead shielding was installed in the medical therapy facility to decrease radiation levels for personnel involved in Boron Neutron Capture Therapy research.

(v) Shim blade drive No. 5 was replaced.

(vi) Ventilation intake filters and exhaust absolute filters were replaced.

(vii) The regulating rod was replaced.

(viii) One tube which houses proximity switches for "blade in" indication was replaced to prevent in-leakage of water.

(ix) The reactor floor hot cell blower was replaced.

(x) The parking lot shadow-shielding wall was modified to improve viewing of vehicular and pedestrian traffic.

(xi) A dimensional measurement was made in the thermal column area in support of a new BNCT facility. This involved extensive movement of equipment and shielding.

(xii) Piping and electrical equipment were installed in support of a new reactor compressed air supply, scheduled for installation in early FY96.

(xiii) The emergency decontamination showers were refurbished.

Many other routine maintenance and preventive maintenance items were performed throughout the fiscal 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 either have or will be forwarded to the Document Control Desk, 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 Chemistry Loop (PCCL)
SR#0-86-9 (04/21/88), #0-88-4 (07/28/88), #0-88-5 (09/09/88), #0-88-14 (12/07/88), #0-89-2 (01/06/89), #0-89-3 (01/19/89), #0-89-6 (01/24/89), #0-89-9 (06/02/89), #0-89-14 (06/19/89), #0-90-6 (03/20/90), #0-90-7 (03/20/90), #0-90-8 (03/20/90), #0-90-9 (03/20/90), #0-90-25 (12/10/90), #0-90-26 (12/18/90), #0-90-27 (12/18/90), #0-91-8 (05/21/91), #0-91-21 (12/27/91), #0-92-2 (01/27/92), #0-92-12 (08/19/92), #0-94-6 (06/30/94), #0-94-8 (03/10/94).

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 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 04/20/88. 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.

Subsequent to the determination that no unreviewed safety question existed, specific procedures for PCCL operation were prepared. These included:

- Procedure for Ex-Core Testing
- Supplement to the Safety Evaluation Report
- Preoperational Test Procedure
- Abnormal Operating Procedures for the PCCL
- Procedures for PCCL Startup/Shutdown
- Procedures for PCCL Installation/Removal
- Procedures for Transfer of Used PCCL Components to a Separate Storage Tank in the Spent Fuel Storage Pool.

Experiments using the PCCL began in April 1989 and have been quite successful. No design changes were made to the PCCL during the period covered by this report. For part of FY 95, this facility was again used to evaluate the passivating effect of zinc additives to PWR primary coolants.

Boiling Coolant Chemistry Loop (BCCL)
SR#0-89-14 (06/19/89), #0-89-20 (12/20/89), #0-90-17 (09/17/90), #0-90-18 (09/14/90), #0-90-20 (10/15/90), #0-91-20 (01/30/92), #0-92-11 (08/15/92), #0-92-16 (09/25/92), #0-93-10 (09/03/93), #0-94-5 (01/24/94), #0-94-6 (06/30/94).

This project involves the design, installation, and operation of a boiling 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 is being examined to determine the optimum method for reducing the creation of activated corrosion products (crud) and thereby reducing radiation fields associated with boiling water reactors (BWRs). The ultimate goal is to reduce radiation exposures to BWR maintenance personnel.

In 1988 and 1989, the Reactor Staff made a determination that boiling within an in-core facility is not contrary to the technical specifications provided that reactivity limits for movable experiments are not exceeded. It was also concluded that boiling in the proposed experiment volume would not significantly affect reactor operation. Accordingly, a carefully controlled experiment was proposed to demonstrate that boiling within an in-core facility would not adversely affect reactor operation. Following both a determination that no unreviewed safety question was involved and approval by the MIT Reactor Safeguards Committee, this experiment was conducted. The results were as expected.

The final safety evaluation report for the BCCL was completed on 8 March 1989 and approved by the MITR Staff. On 12/20/89, the MIT Reactor Safeguards Committee determined that there was no unreviewed safety question involved in the conduct of the BCCL experiment and approved the BCCL SER. On 9 March 1990, a copy of the BCCL SER together with the safety analysis prepared by the MITR Staff were forwarded to the U.S. Nuclear Regulatory Commission pursuant to 10 CFR 50.59(b)(2).

Subsequent to the determination that no unreviewed safety question existed, specific procedures for BCCL operation were prepared. These included:

- Preoperational Test Procedure.
- Abnormal Operating Proc es for the BCCL.
- Procedure for BCCL Startup.

Other necessary procedures such as BCCL shutdown and installation/removal are the same as those previously developed and approved for the PCCL. Experiments using the BCCL began in October 1990 and have been successful in that many theories concerning the transport of nitrogen-16 in boiling water reactors have been disproven. No changes were made to the BCCL experimental protocol during this reporting period.

Experiments that make use of the BCCL facility were conducted during portions of this reporting period.

Experiments Related to Neutron Capture Therapy
SR#0-89-4 (01/23/89), #0-89-8 (03/01/89), #0-91-7 (05/06/91), #0-91-17 (03/06/92), #0-92-3 (03/06/92), #0-92-4 (03/02/92), #M-92-2 (05/14/92), #0-93-5 (05/28/93), #0-93-9 (07/13/93), #0-93-20 (11/30/93), #0-94-19 (12/02/94)

In conjunction with the Tufts – New England Medical Center and with the support of the U.S. Department of Energy, MIT has designed an epithermal neutron beam for the treatment of brain cancer (glioblastoma). Thermal beams have been used successfully for this treatment in Japan. The reason for designing an epithermal beam is to allow tumor treatment without having to subject the patient to surgery involving removal of a portion of the skull. Also, an epithermal beam gives greater penetration. In October 1991, MIT hosted an international workshop for the purpose of reviewing proposed beam designs and dosimetry. Subsequent to the receipt of advice from the workshop panel members, a final design was selected for the epithermal filter for the MIT Research Reactor's Medical Therapy Facility beam. That design, which was one of many that had been previously constructed and evaluated, is No. M-62. It has now been installed permanently. Approvals of the protocol for the conduct of patient trials have now been received from all requisite MIT and NEMC Committees as well as from the U.S. Food and Drug Administration. Also, a license amendment and quality management plan for use of the MIT Research Reactor's Medical Therapy Facility was issued by the U.S. Nuclear Regulatory Commission as License Amendment No. 27 on February 16, 1993.

Subsequent to the receipt of that license amendment and a similar one in August 1993 for our medical partner, the Tufts – New England Medical Center, both procedures for performing BNCT and a preoperational test package were prepared. The latter was completed during FY 94.

Patient trials were initiated in September 1994 as part of a Phase I effort that is required by the FDA. In December 1994, changes were issued to certain of the procedures that had been prepared for conduct of the patient irradiations. These changes were intended to reduce the signature burden on senior personnel during the trials so that their full attention could be given to the patient. Additional information is given in Section I of this report.

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 (12/06/84), #0-84-21 (11/01/84), #0-85-11 (05/09/85), #0-85-13 (06/28/85), #0-85-16 (07/12/85), #0-85-20 (08/16/85), #0-85-25 (12/01/85), #0-85-26 (12/01/85), #0-86-11 (10/17/86), #0-86-13 (11/28/86), #0-87-11 (06/01/87), #0-87-17 (12/24/87), #0-88-10 (12/01/88), #0-90-28 (12/27/90), #0-91-2 (05/14/91), #0-91-3 (06/06/91), #0-91-10 (07/15/91), #0-91-11 (06/25/91), #0-92-6 (06/09/92), #0-93-3 (03/29/93), #0-93-23 (12/06/93), #0-94-12 (04/25/94).

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 signal validation procedures, ensures 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% $\Delta K/K$). 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% $\Delta K/K$ could be inserted were all the connected control elements to be withdrawn, and

(2) Closed-loop control of one of the 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 the NRC-licensed supervisory controller is used to monitor, and if necessary override, other novel controllers that are still in development. Tests of novel controllers are conducted under the provisions of technical specification #6.4 which requires that reactivity be constrained to ensure "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. This arrangement for the conduct of these tests has been approved by the MIT Reactor Safeguards Committee.

An extensive upgrade to the digital control system's hardware was performed in 1991. The present system consists of five interconnected computers and has been designated as the Advanced Control Computer System. The five computers are:

(i) Rack-Mount 80386: This is an IBM-AT computer that is used for data acquisition, execution of software essential to safety such as the code to implement the requirements of MITR Technical Specification #6.4, and the writing of data to disk. Software on this computer is normally invariant.

(ii) MicroVAX-II: This machine is dedicated for intensive floating-point computations such as are required to implement the various control concepts. This machine receives validated sensor information from the IBM-AT and returns the demanded actuator signal to that computer. Software changes on this computer are expected to be frequent.

(iii) IBM-Compatible 80386: This is a high-speed machine on which programs are first edited, compiled, and finally linked to form an executable module. This machine is capable of supporting automated reasoning using PROLOG, LISP, or C.

(iv) IBM-XT 8088: This computer's role is to receive validated signals from the data acquisition computer and to display model-based predictive information or a safety parameter display on its screen.

(v) LSI-11/23: This unit was the original MITR digital control computer. It is now connected to the MicroVAX-II for the purpose of providing an independent machine on which a model of the reactor can be run. This improves simulation studies because signals must be passed between two computers as is done for actual implementations.

Both the MITR Staff and the MIT Reactor Safeguards Committee concluded that this upgraded digital control system was within the envelope of conditions prescribed in the 1985 license amendment issued by NRC for digital control experiments at MIT and that no unreviewed safety question was involved. As part of the installation of this new system, several preoperational test packages were prepared and performed. Included were tests to verify signal transmission, to compare software performance on both the original and upgraded systems, and to test all software and hardware interlocks.

In addition to this upgraded hardware, an auto-ranging digital picoammeter has been installed to measure reactor neutronic power. This instrument provides both the level and range of the power signal. Moreover, it switches scales automatically and thus facilitates the development of control strategies for automated startups in which operation over many decades of power is required. This instrument was subjected to a preoperational test in which its accuracy was verified.

No new experimental research on the closed-loop digital control of nuclear reactors was conducted in FY 95 because of the demands placed on the reactor for steady-state operation by other experiments. Open-loop experiments that had begun in previous years to demonstrate the practicality of flux synthesis methods for the estimation of reactivity were completed.

Irradiation-Assisted Stress Corrosion Cracking (IASCC) Experiment
SR#0-89-15 (06/19/89), #0-90-21 (10/22/90), #0-90-22 (10/22/90), #0-90-23 (11/05/90), #0-91-21 (12/27/91), #0-92-5 (04/02/92), #0-92-17 (09/28/92), #0-92-21 (01/21/94) #0-93-6 (05/26/93), #M-93-1 (05/24/93), #0-93-14 (09/09/93), #0-94-4 (01/28/94), #0-94-7 (06/13/94).

In the past several years a variety of austenitic stainless steel components in boiling water reactor (BWR) cores have failed by an intergranular cracking mechanism called irradiation-assisted stress corrosion cracking (IASCC). Characteristics of such failures are that the component was exposed to a fast neutron fluence under tensile stress and in an oxidizing water environment.

A facility to study IASCC in typical BWR water and radiation environments has been designed, built, and put into in-core service. This facility positions a pre-irradiated test specimen in the core of the MIT Research Reactor, circulates water with controlled temperature and chemistry past the specimen, and applies a tensile load to the specimen to maintain a constant slow strain rate until specimen failure. A DC potential drop (DCPD) technique was developed to measure specimen strain during in-core testing. Electrodes are incorporated to measure the specimen's electrochemical corrosion potential (ECP) under test, and for initial analysis, the sensitivity of the specimen's ECP to varying water chemistry, flowrate, in-core position, and reactor power level. A chemistry control system was designed and built to measure and control the water chemistry. Remote specimen handling tools and procedures were developed to allow the fracture surface to be analyzed by scanning electron microscopy (SEM). The facility and operating procedures were designed to minimize radiation exposure of personnel during facility operation and transfer to a hot cell for specimen removal and replacement.

Initial in-core tests, which measured the ECP of stainless steel in in-flux sections of the testing rig have been completed successfully. These tests showed that the desired oxidizing environment can be established and monitored during in-core SSRT testing. Initial in-core SSRT testing is presently underway. Results of these tests will be used to investigate the effects of neutron fluence and materials variables on IASCC.

As part of the preparations for this experiment, a new reactor top lid was designed and installed in FY 93. This lid, which provides an additional four inches of vertical clearance for in-core experiments, meets or exceeds the specifications for the original lid. Radiation levels directly above the reactor were reduced as a result of the installation of this new lid.

The IASCC experiment was operated in-core from July 5 to October 31, 1994. Further runs are anticipated in academic year 1995-96.

<u>SENSOR Facility</u>
SR#0-94-18 (11/29/94), #0-95-8 (06/30/95)

The SENSOR facility complements the Irradiation-Assisted Stress Corrosion Cracking (IASCC) experiment. The objective of the SENSOR experiment is to place sensors in a loop that replicates the water chemistry of a Boiling Water Reactor (BWR) and then to place that loop in the core of th MIT Research Reactor (MITR-II). The sensors are to measure crack growth in situ and simultaneously monitor the electrochemical potential (ECP) with the objective of determining if a relation exists between crack propagation and ECP. Also, the experiment seeks to determine if hydrogen injection can arrest crack growth. Several types of specimens will be used. Some will be thermally sensitized and are expected to crack almost at once. Others will not have been presensitized. (<u>Note</u>: Sensitization is achieved by causing chromium depletion at the grain boundaries. This can be done either thermally or via neturon irradiation.) The facility is described in detail in the "Final Safety Evaluation Report (SER) for the Sensor irradiation Facility."

In addition to the safety evaluation report, procedures for the installation and removal of the SENSOR were prepared. These were similar to those developed earlier for the PCCL, BCCL, and IASCC in-core experiments. An ALARA plan was also prepared for the SENSOR experiment.

The SENSOR experiment was installed in the MITR core in March 1995 and it ran almost continuously until late June when all planned experiments were complete. Data analysis remains on-going. The experiment appears to have been succesful in all of its major objectives. In particular, the capability to affect crack growth through a change in water chemistry was shown.

Revision of the Surveillance Requirement on the Emergency Battery
SR#0-95-4 (02/22/95)

The emergency battery for the MIT Research Reactor was replaced in its entirety during FY95. Subsequent to that replacement, a change on the surveillance requirements for measurement of the battery's voltage and specific gravity was identified as being desirable. Accordingly, a safety analysis was prepared and, following approval by the MIT Reactor Safeguards Committee, submitted to the U.S. Nuclear Regulatory Commission (NRC) on 02/22/95. A request for additional information was received on 03/13/95 and a reply submitted on 04/10/95.

F. ENVIRONMENTAL SURVEYS

Environmental monitoring is performed using continuous radiation monitors and dosimetry devices. The radiation monitoring system consists of G-M detectors and associated electronics at each remote site with data transmitted continuously to the Reactor Radiation Protection Office and recorded on strip chart recorders. The remote sites are located within a quarter mile radius of the facility. The detectable radiation levels per sector due primarily to Ar-41 are presented below.

Site	Exposure (07/01/94-06/30/95)
North	0.317 mrem
East	1.220 mrem
South	0.087 mrem
West	0.475 mrem
Green (east)	0.039 mrem

Fiscal Year Averages

1995	0.4 mrem
1994	0.4 mrem
1993	0.5 mrem
1992	0.2 mrem
1991	0.1 mrem
1990	0.1 mrem

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

July 1, 1994 - June 30, 1995

Whole Body Exposure Range (rems)	Number of Personnel
No measurable	122
Measurable – < 0.1	44
0.1 – 0.25	14
0.25 – 0.5	10
0.5 – 0.75	4

Total Person Rem = 9.81   Total Number of Personnel = 194

From July 1, 1994 through June 30, 1995, the Reactor Radiation Protection Office provided radiation protection services for the facility which included power and non-power operational surveillance (performed on daily, weekly, monthly, quarterly, and other frequencies as required), maintenance activities, and experimental project support. Specific examples of these activities include, but are not limited to, the following:

1. Collection and analysis of air samples taken within the containment building and in the exhaust/ventilation systems.

2. Collection and analysis of water samples taken from the secondary, D_2O, primary, shield coolant, liquid waste, and experimental systems, and fuel storage pool.

3. Performance of radiation and contamination surveys, radioactive waste collection and shipping, calibration of area radiation monitors, calibration of effluent and process radiation monitors, calibration of radiation protection/survey instrumentation, and establishing/posting radiological control areas.

4. Provision of radiation protection services during fuel movements, in-core experiments, sample irradiations, beam port use, ion column removal, etc.

The results of all surveys and surveillances conducted 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 blowdown and the liquid waste storage tanks. All of the liquid volumes are measured, by far the largest being the 15,617,000 liters discharged during FY 95 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 because the volume is not routinely measured.)

Total activity less tritium in the liquid effluents (cooling tower blowdown, waste storage tank discharges, and engineering lab sink discharges) amounted to 0.000059 Ci for FY 95. The total tritium was 0.0247 Ci. The total effluent water volume was 2.37×10^7 liters, giving an average tritium concentration of 1.04×10^{-6} μCi/ml.

The above liquid waste discharges are provided on a monthly basis in the following Table H-3.

All releases were in accordance with Technical Specification 3.8-1, including Part 20, Title 20, Code of Federal Regulations. All activities were substantially below the limits specified in 10 CFR 20.2003. Nevertheless, the monthly tritium releases are reported in Table H-3.

2. Gaseous Waste

Gaseous radioactivity is discharged to the atmosphere from the containment building exhaust stack. All gaseous releases likewise were in accordance with the Technical Specifications and 10 CFR 20.1302, and all nuclides were below the limits after the authorized dilution factor of 3000 with the exception of Ar-41, which is reported in the following Table H-1. The 1279.8 Ci of Ar-41 was released at an average concentration of 0.324×10^{-8} μCi/ml. This represents 32.4% of EC (Effluent Concentration (1×10^{-8} μCi/ml)). Direct comparison with the FY 94's figures is not meaningful because of the change in 10 CFR 20 during that fiscal year.

3. Solid Waste

No shipments of solid waste were made during the year, as shown in the following Table H-2.

TABLE H-1 – Part A

ARGON-41 STACK RELEASES

FISCAL YEAR 1995

	Ar-41 Discharged (Curies)	Average Concentration[1] (μCi/ml)
July 1994	62.83	2.12 E-9
August	63.84	2.16 E-9
September	119.33	3.22 E-9
October	124.12	4.19 E-9
November	54.03	1.82 E-9
December	108.16	2.92 E-9
January 1995	89.94	3.04 E-9
February	55.13	1.86 E-9
March	76.48	2.58 E-9
April	150.45	4.06 E-9
May	108.00	3.65 E-9
June	267.51	7.22 E-9
Totals (12 Months)	1,279.82	3.24 E-9
EC (Table II, Column I)		1×10^{-8}
% EC		32.4%

[1]After authorized dilution factor (3000). (Note: Average concentrations do not vary linearly with curies discharged because of differing monthly dilution volumes.)

TABLE H-2

SUMMARY OF MITR-II RADIOACTIVE SOLID WASTE SHIPMENTS

FISCAL YEAR 1995

Description	
Volume	0 ft^3
Weight	0 lbs.
Activity	0 Ci
Date of shipment	N/A
Disposition to licensee for burial	N/A
Waste broker	N/A

No dry active waste was shipped off-site during FY 95 due in part to closure of the waste sites.

TABLE H-3

LIQUID EFFLUENT DISCHARGES
FISCAL YEAR 1995

	Total Activity Less Tritium	Total Tritium Activity	Volume of Effluent Water(1)	Average Tritium Concentration
	($x10^{-6}$ Ci)	($x10^{-3}$ Ci)	($x10^{4}$ liters)	($x10^{-6}$ µCi/ml)
July 1994	8.9	0.905	143	0.372
Aug.	NDA	0.226	183	0.123
Sept.	4.43	2.41	171	1.41
Oct.	NDA	1.73	155	1.12
Nov.	2.35	1.15	93.9	1.22
Dec.	NDA	3.09	147	2.10
Jan. 1995	9.71	4.97	153	3.25
Feb.	NDA	1.36	111	1.23
Mar.	17.4	0.233	71.8	0.325
Apr.	NDA	0.134	134	0.100
May	8.86	1.03	898	0.114
June	7.20	0.716	110	0.651
12 months	58.85	17.95	2,370.70	12.02

(1) Volume of effluent from cooling towers, waste tanks, and NW12-139 Engineering Lab sink. Does not include other diluent from MIT estimated at 2.7 million gallons/day.

(2) No Detectable Activity (NDA); less than $1.26x10^{-6}$ µCi/ml beta for each sample.

I. SUMMARY OF USE OF MEDICAL FACILITY FOR HUMAN THERAPY

The use of the medical therapy facility for human therapy is summarized here pursuant to Technical Specification No. 7.13.5(i):

1. Investigative Studies

A Phase I study to investigate the toxicity (or lack thereof) of neutron capture therapy is required by the U.S. Food and Drug Administration. This trial was begun in September 1994 and three subjects have participated to date.

The first step in the Phase I protocol's implementation is for the subject to be given a test dose (400 mg/kg) of the boron-containing drug (BPA). Blood and punch biopsy samples are then taken in order to determine the biodistribution of the boron in both healthy tissue and tumor over time. This is necessary because the uptake of boron in tumor varies markedly from one person to another. The irradiations themselves are done in four fractions. For each, the subject is given 400 mg/kg of BPA and a limited number of blood/biopsy samples are taken to confirm the previously measured uptake curve. The starting point in the Phase I protocol was a total dose to healthy tissue of 1000 RBE-cGy. This equates to a dose per fraction of 250 RBE-cGy. Three subjects have thus far completed irradiation at the lowest dose level. No problems were experienced. A summary of the subject responses follows:

Subject 94-1 — 66 year old male, initial treatment September 6, 1994

Location of irradiation: plantar surface of right foot

Follow up: normal tissue graded as "0." Minimal tumor response.

Subject 94-2 — 61 year old male, initial treatment October 24, 1994

Location of irradiation: medial surface of lower left leg

Follow up: normal tissue response graded as "0." Tumor response regression observed.

Subject 94-3 — 81 year old female, initial treatment December 5, 1994

Location of irradiation: outer surface of left lower leg

Follow up: normal tissue response graded as "0." Visible tumor response one day after final fraction. First follow up on 01/19/95 showed tumor nodules in radiation field markedly reduced in size while nodule outside of radiation field somewhat increased in size.

The second and third subjects' positive response (marked reduction of tumor nodule in irradiation field and no normal tissue response) is very promising, especially in consideration of the low doses used in these initial experimental irradiations.

2. Human Therapy

None.