U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

Region I

Report No. 50-20/78-05

Docket No. 50-20

License No. R-37 Priority -- Category F

Licensee: Massachusetts Institute of Technology
138 Albany Street
Cambridge, Massachusetts 02139

Facility Name: MITR-II Research Reactor

Inspection at: Cambridge, Massachusetts

Inspection conducted: September 13-15, 1978

Inspectors: Karl E. Plumlee 10/5/78
Karl E. Plumlee, Radiation Specialist — date signed

John A. Serabian 10/5/78
John A. Serabian, Radiation Specialist — date signed

date signed

Approved by: [signature] for 10/12/78
Peter J. Knapp, Chief, Radiation Support Section, FF&MS Branch — date signed

Inspection Summary:

Inspection on September 13-15, 1978 (Report No. 50-20/78-05)
Areas Inspected: Routine, unannounced inspection by regional based inspectors of the radiation protection program and effluent management including: radiation protection training; procedures; instruments and equipment; posting and control of hazardous areas; labeling of containers of radioactive materials; reactor coolant system water purity; effluent monitoring; exposure records; and reports to the NRC and to individuals. The initial inspection and area examination was conducted during nonregular hours (10 PM to 12 PM on September 13, 1978). This inspection involved 32 inspector-hours on site by two NRC regional based inspectors.
Results: Of the nine areas inspected, no items of noncompliance were identified in eight areas. One item of noncompliance was identified in one area (Deficiency - failure to submit a timely notice to NRC and failure to have a copy of a shipping container certificate - paragraph 4).

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DETAILS

1. Persons Contacted

   L. Andexler, Senior Reactor Operator
   J. Bernard, Senior Shift Supervisor
   M. Bolton, Associate Radiation Protection Officer, MIT
   H. Bondar, Administrative Assistant
   *P. Coggio, Project Technician
   *K. Collins, Reactor Superintendent
   *E. Karaian, Radiation Protection Officer - Reactor
   K. Kwok, Senior Reactor Operator
   B. McDermott, Shift Supervisor

   * denotes those present at the exit interview, 3:00 p.m., September 15, 1978.

2. Licensee Action on Previously Identified Items

   (Open) Unresolved Item (20/76-04): Release rate of 41-Ar. Review of this inspection did not identify any items of noncompliance - paragraph 9. This item is under review by licensing.

3. Licensee Action on IE Bulletins

   78-07 Airline respirator usage. The licensee's reply, dated June 19, 1978, stated that no allowances for the use of individual respiratory protection equipment are made at the MIT reactor. Review on this inspection did not identify any items of noncompliance - paragraph 7.

   78-08 Fuel element transfer tubes. None were installed at this reactor. Irradiated fuel is moved from the reactor to spent fuel storage inside a fuel transfer cask.

4. Shipments of Radioactive Material

   Part of the inspection effort was to review the licensee's shipping and receiving practices.

   10 CFR 71.3, "Requirement for License," requires that no licensee subject to regulations in this part shall: (a) deliver any licensed materials to a carrier for transport or (b) transport licensed material except as authorized in a general or specific license issued by the Commission or as exempted in this part. Subpart 71.12(b) requires the

person using a package pursuant to the general license provided by this paragraph to have a copy of the specific license, certificate of compliance, or other approval authorizing use of the package and prior to the first use of the package to submit in writing to the Director of Nuclear Material Safety and Safeguards his name and license number, the name and license or certificate number of the person to whom the package approval has been issued, and the package identification number specified in the package approval.

10 CFR 71.62(b) states that all records required by the part shall be made available to the Commission for inspection upon reasonable notice.

Review of the records of shipments and receipts of radioactive material between January 1, 1978 and September 15, 1978, indicated that the licensee had not obtained any exemption or kept available a copy of the certificate of compliance or authorization for the packaging (Identification No. 9094A and Model No. CNS 14-195) used to transport 210 Ci of Transport Group III LSA radioactive material on January 30, 1978; nor had MIT submitted such information to NRC prior to use.

Chem-Nuclear Systems, Incorporated, had delivered the empty packaging to MIT and subsequently transported the shipment to an out of state burial site. Review of the MIT file copies indicated that Chem-Nuclear Systems, Incorporated, typically had provided copies of certificates of compliance for packaging delivered to MIT, however, no copy was found for this particular packaging.

The MIT Director, Reactor Operations, informed the inspector by telephone on September 21, 1978, that a replacement copy of the certificate of compliance was obtained on September 21 and that the applicable information was submitted to the Director of Nuclear Material Safety and Safeguards by September 21, 1978.

The inspector reviewed an NRC copy of the Certificate of Compliance No. 9094, issued to Chem-Nuclear Systems, Incorporated, (DN 71-9094, for packaging Identification No. USA/9094/A, Model No. CNSI 14-195-H). This review of the authorized use of the packaging did not identify any misuse during the above shipments.

The inspector identified the apparent lack of required documentation as noncompliance with requirements of 10 CFR 71.12(b) which the licensee had promptly corrected following the inspection (78-05-01). The licensee has implemented a quality assurance program for shipping such packages (Revision No. 14 to the Safety Analysis Report, submitted June 30, 1978).

This quality assurance program should prevent recurrence of the above item.

5. Tours of the Facility

The inspectors observed the conduct of operations, including the following specific items throughout the parts of the facility that are accessible during reactor operation.

Adherence to procedures
Wearing of dosimeters and protective clothing
Calibration status, location and operability of survey instruments and friskers
Signs and labels
Control of access to radiation, high radiation and contaminated areas
Console indications of radiation levels, ventilation status, and water system purity
Posted information and instructions required by 10 CFR 19.11 and 19.12
Constant air monitors, status and locations

No items of noncompliance were identified.

6. Review of Records

The inspector reviewed records of the following information for the period January 1, 1978 to September 15, 1978, or as shown.

Radiation protection training of experimenters
Radiation protection staffing
Instrument and equipment calibrations
Area surveys
Gaseous and liquid effluent sampling
Primary and secondary coolant sampling
Spent fuel storage pool sampling
Iodine removal filter efficiency tests, dated November 14, 1977, and May 22, 1978
MIT Research Reactor Annual Report, dated September 11, 1978
MIT Research Reactor Annual Report, dated August 29, 1977

No items of noncompliance were identified.

7. Exposure Control

Part of the inspection effort was to review the licensee's practices, procedures and records involving the control of external and internal exposures to individuals.

No substantial changes in procedures were identified. The registration and authorization procedure (MITR Manual, Section E) excludes minors other than those on conducted tours of the facility. Whole body counts are performed of persons who work in contaminated areas.

The inspector reviewed selected file copies of Form NRC-4 and NRC-5 information, film badge reports for the period July 1, 1977 to June 30, 1978, and whole body count records for the period January, 1977 to August, 1978. Comparisons of selected annual report entries with these records did not identify any discrepancies.

These records did not indicate any uptake of radioactive material or any neutron doses to personnel. The maximum whole body exposure to any individual appeared to be less than 1.25 rem during any calendar quarter, and less than 2.0 rems during any four consecutive calendar quarters.

The licensee's records showed that neutron film badges were routinely issued to 6 or 7 individuals during each badge period from January 1, 1977, to September, 1978. These individuals either worked with isotopic sources that produced neutrons or entered areas where reactor neutrons were found. The badge documentation indicated a 20 mrem threshhold. None of these records indicated a detectible neutron dose to a film badge worn by any individual.

The licensee representative stated that one area was characterized by a thermal neutron dose rate up to 20 mrem/hr that the film badges would not detect; however, he had no measurements to verify this. He stated that the neutron film badges would be tried in this area which is above the reactor and is accessible to personnel during reactor operation. This measurement will be reviewed on a subsequent routine inspection (78-05-02).

The inspector observed a confirmatory measurement survey by a licensee representative who used a rem-meter type instrument, in the areas designated below by an asterisk. Comparison of these measurements with the records of a previous survey and with records of eight neutron film badges that were routinely used as area monitors did not identify any discrepancies. Detectible neutron levels were typically localized within the following areas:

* Neutron beam tubes, on experimental level (radiation areas)

Equipment room, a locked high radiation area(1)

* Top of reactor, a stepoff pad area (radiation and contamination area)

(1) Up to 60 mrem/hr neutron and 350 mr/hr gamma exposure rates were indicated in the equipment room.

* Individual sealed isotopic sources that produce neutrons - example Cf-252

Medical therapy room, a locked room controlled by procedures and interlocks

The licensee representative maintains closed circuit television surveillance of the experimental level and the reactor top, but no stay-time records. The neutron dose rate is taken to be a negligible contribution of less than 25% of the total whole body exposure to any individual.

No items of noncompliance were identified involving the control of personnel exposures.

8. Effluent Management

Part of the inspection effort was to review the effluent release records to determine compliance with the applicable limits. The Technical Specifications in Section 3.8.1 allow a stack dilution factor and specify a 10^{-3} uCi/ml tritium concentration limit on the normal discharge of the secondary coolant. Secondary coolant is normally circulated through a cooling tower and blown down at 6 gpm to the sanitary sewer. Otherwise, 10 CFR 20 release limits apply.

a. Secondary Coolant (H_2O)

The records of samples of secondary coolant for the period January, 1978 through August, 1978, did not indicate any detectible alpha, beta, or gamma activity except that a few results indicated tritium concentrations in excess of the minimum detectible activity (MDA), but none greater than $\sim 10^{-5}$ uCi/ml.

The tritium concentration indicated by samples of systems cooled by the secondary system was as great as 885 uCi/ml (D_2O reflector sample). The licensee representative stated that there is no known existing leakage of fluid containing tritium into the secondary system.

The inspector noted that the licensee's Annual Reports, dated August 29, 1977 and September 11, 1978, referenced the value stated in 10 CFR 20, Appendix B, Table I, Column 2, rather than the TS 3.8.1 cooling tower operating limit of 10^{-3} uCi/ml on secondary coolant tritium concentration , which is more restrictive than the 10 CFR 20, Appendix B limits for liquid releases by a factor of 3 (Table II) and 100 (Table I).

No releases exceeding the Technical Specification limits were identified.

b. Primary Coolant (H_2O)

Review of primary coolant sample records indicated no detectible alpha activity; typically 0.1 to 0.3 uCi/ml each of beta and gamma activity; and 16 $\pm$ 1 uCi/ml of tritium activity. Radioiodine activity was reported at concentrations less than 10^{-5} uCi/ml. Primary coolant specific conductivity was typically below a micromho/cm. The primary coolant circulates in a closed system that is vented to the waste system and has a potential for leakage through pump seals, valves and a heat exchanger. The licensee representative stated that there are no leaks.

No conditions constituting noncompliance were identified.

c. Releases to the Sanitary Sewerage System

In addition to the secondary cooling system blowdown (above) the licensee discharges suspect and contaminated water from a waste collection system to a sanitary sewer. Holdup tanks are used and sampling and evaluation is performed before each discharge. This release route is subject to concentration limits and a one Ci annual limit required by 10 CFR 20.303 except for tritium.

Review of records for selected periods of time, and a review of licensee's annual reports, dated August 29, 1977 and September 11, 1978, showed that the total annual release including tritium was less than 0.25 Ci and that most of the activity was tritium.

No items of noncompliance were identified.

9. Stack Release Rate

Previous inspection reports reviewed the history of the stack release rate during operation and noted that the release rate of 41-Ar was now greater than before the modification of the reactor. The licensee representative attributed this to an increase in the neutron flux in some air-filled regions.

The licensee has carried out a program to alleviate the stack release rate. This included filling some areas with aluminum blocks and flushing other areas with carbon dioxide to displace air.

The licensee reported 941 Megawatt-days (MWD) of operation during the period July 1, 1977 to June 30, 1978, with a stack release of 9,659 Ci of 41-Ar and 7 Ci of tritium, as compared to 702 MWD, with a stack release of 7,795 Ci and 7 Ci respectively during the preceding 12 month period. The licensee estimated that the effort to alleviate releases reduced the release rate by 7.6% (including the adjustment for the increase in MWD of operation).

The annual average release rate was reported as 70.3% of the Technical Specifications limit.

No items of noncompliance were identified.

10. Confirmatory Measurements

In a previous inspection, conducted on September 26-28, 1977, Inspection Report Number 50-20/77-06, a liquid effluent sample was split with the licensee and NRC:I. Analyses were performed by the licensee using his normal methods and procedures, and the NRC:I analyses were performed by the Department of Energy's Radiological and Environmental Services Laboratory (RESL). The comparison of the analyses results indicated that one of the measurements was in agreement under the criteria used for comparing results (See Attachment 1), and one measurement was in disagreement.

The measurement in disagreement was a gross beta measurement. The licensee's value is higher than the NRC value and may be due to the presence of short lived radionuclides as the NRC sample was analyzed approximately 30 days after the sample was taken. This area will be reviewed during a future inspection. The inspector had no further questions in this area. No items of noncompliance were identified. The results of the comparisons are presented in Table I.

11. Exit Interview

The inspector met with licensee representatives (denoted in paragraph 1) at the conclusion of the inspection.

The inspector reviewed the scope and findings of the inspection.

The inspector stated that he would contact the licensee to confirm that the documentation that was described in paragraph 4 had been obtained. (This contact was made by telephone on September 21, 1978, with Mr. L. Clark).

Table I

M.I.T. - Verification Test Results

Sample	Isotope	NRC Value	Licensee Value	Comparison
		Results in Microcuries Per Milliliter		
Waste Storage Tank 10/25/77	H-3	(6.78 ± 0.03)E-4	(6.55 ± ?)E-4	Agreement
	Gross Beta	(6.4 ± 0.4)E-7	(2.76 ± ?)E-6	Disagreement

Attachment 1

Criteria for Comparing Analytical Measurements

This attachment provides criteria for comparing results of capability tests and verification measurements. The criteria are based on an empirical relationship which combines prior experience and the accuracy needs of this program.

In these criteria, the judgement limits are variable in relation to the comparison of the NRC Reference Laboratory's value to its associated uncertainty. As that ratio, referred to in this program as "resolution", increases the acceptability of a licensee's measurement should be more selective. Conversely, poorer agreement must be considered acceptable as the resolution decreases.

$$\text{RATIO} = \frac{\text{LICENSEE VALUE}}{\text{NRC REFERENCE VALUE}}$$

Resolution	Agreement	Possible Agreement A	Possible Agreement B
<3	0.4 - 2.5	0.3 - 3.0	No Comparison
4 - 7	0.5 - 2.0	0.4 - 2.5	0.3 - 3.0
8 - 15	0.6 - 1.66	0.5 - 2.0	0.4 - 2.5
16 - 50	0.75- 1.33	0.75- 1.66	0.6 - 2.0
51 - 200	0.80- 1.25	0.75- 1.33	0.6 - 1.66
> 200	0.85- 1.18	0.80- 1.25	0.75 -1.33

"A" criteria are applied for the following analyses:

Gamma Spectrometry where principal gamma energy used for identification is greater than 250 Kev.

Tritium analyses of liquid samples.

"B" criteria are applied to the following analyses:

Gamma Spectrometry where principal gamma energy used for identification is less than 250 Kev.

89Sr and 90Sr Determinations.

Gross Beta where samples are counted on the same date using the same reference nuclide.