U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

NRC FOIA REQUEST NUMBER(S): FOIA-88-450

RESPONSE TYPE: X FINAL | PARTIAL

DATE: SEP 27 1988

DOCKET NUMBER(S) (if applicable):

REQUESTER: Ms. Catherine Walthers

PART I. – RECORDS RELEASED OR NOT LOCATED *(See checked boxes)*

	No agency records subject to the request have been located.
	No additional agency records subject to the request have been located.
	Agency records subject to the request that are identified in Appendix ______ are already available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC.
XX	Agency records subject to the request that are identified in Appendix A are being made available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and coying at the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
XX	Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC.
XX	Agency records subject to the request are enclosed. Any applicable charge for copies of the records provided and payment procedures are noted in the comments section.
	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
	In view of NRC's response to this request, no further action is being taken on appeal letter dated ______

PART II.A – INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

	Certain information in the requested records is being withheld from public disclosure pursuant to the FOIA exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.

Comments

Items 1 through 5 of Appendix A are enclosed.

Items 6 through 27 of Appendix A can be obtained from the PDR.

The new PDR is located at 2120 L Street, N. W., Lower-Level, Washington, D. C., 20555.

8810170293 880927
PDR FOIA
WALTHER88-450 PDR

SIGNATURE, DIRECTOR, DIVISION OF RULES AND RECORDS

Donnie H. Grimsley

NRC FORM 464 Part I
(9-86)

APPENDIX A

DOCUMENTS TO BE RELEASED

FOIA REQUEST NUMBER 88-450

	DATE	ORIGINATOR	RECIPIENT	DESCRIPTION
		Inspection Reports		
1.	3/21/85	T. Martin Region I, NRC	L. Clark, M.I.T.	Inspection No. 50-20/85-01 (9 pages)
2.	4/9/85	T. Martin Region I, NRC	L. Clark, M.I.T.	Acknowledgement of response letter dated 2/27/85 (copy attached) (6 pages)
3.	11/21/85	E. Wenzinger Region I, NRC	L. Clark, M.I.T.	Examination Report No. 50-20/85-03(OL) (77 pages)
4.	11/27/85	T. Martin Region I, NRC	L. Clark, M.I.T.	Inspection No. 50-20/85-02 (4 pages)
5.	4/25/86	E. Wenzinger Region I, NRC	L. Clark, M.I.T.	Inspection No. 50-20/86-01 (13 pages)
6.	8/21/86	T. Martin Region I, NRC	L. Clark, M.I.T.	Inspection No. 50-20/86-02 (5 pages)
7.	8/29/86	J. Kinneman Region I, NRC	A. Ducatman, M.I.T.	Inspection No. 86-01 (3 pages)
8.	10/27/86	E. Wenzinger Region I, NRC	L. Clark, M.I.T.	Examination Report No. 50-20/86-03(OL) (81 pages)
9.	10/28/86	J. Kinneman Region I, NRC	A. Ducatman, M.I.T.	Acknowledgement of 9/23/86 response (copy attached) to Inspection No. 86-01 (3 pages)
10.	9/9/87	T. Martin Region I, NRC	L. Clark, M.I.T.	Combined Inspection Nos. 50-20/87-02 and 70-938/87-02 (9 pages)
11.	9/9/87	S. Collins Region I, NRC	L. Clark, M.I.T.	Examination Report No. 50-20/87-01(OL) (44 pages)

APPENDIX A

DOCUMENTS TO BE RELEASED

FOIA REQUEST NUMBER 88-450

	DATE	ORIGINATOR	RECIPIENT	DESCRIPTION
12.	9/16/87	T. Martin Region I, NRC	F. X. Masse, M.I.T.	Combined Inspection Nos. 30-763/87-01 and 70-938/87-01 (12 pages)
13.	11/5/87	L. Clark, M.I.T.	USNRC	Letter Subject: Open Item No. 84-01-02, Facility Operating License R-37, Docket 50-20 (7 pages)
14.	11/27/87	T. Martin Region I, NRC	F. X. Masse, M.I.T.	Acknowledgement of 10/16/87 response (copy attached) to Inspection 30-763/87-01 and 70-938/87-01 (4 pages)
15.	12/30/87	T. Martin Region I, NRC	L. Clark, M.I.T.	Inspection No. 50-20/87-03 (8 pages)
16.	3/3/88	R. Bellamy Region I, NRC	L. Clark, M.I.T.	Inspection No. 50-20/87-03 (3 pages)
17.	3/16/88	R. Gallo Region I, NRC	L. Clark, M.I.T.	Examination Report No. 50-20/88-01(OL) (5 pages)
18.	6/8/88	G. Sjoblom Region I, NRC	P. Powell, M.I.T.	Inspection No. 30-763/88-01 (60 pages)
19.	7/18/88	J. Wiggins Region I, NRC	J. Bernard, M.I.T.	Inspection Report No. 50-20/88-02 (10 pages)
20.	8/30/88	R. Bellamy Region I, NRC	J. Bernard, M.I.T.	Inspection No 50-20/88-03 (6 pages)
21.	9/6/88	J. Bernard, M.I.T.	USNRC	Letter Subject: NRC Region I Inspection No. 50-20/88-02 (1 page)

APPENDIX A

DOCUMENTS TO BE RELEASED

FOIA REQUEST NUMBER 88-450

	DATE	ORIGINATOR	RECIPIENT	DESCRIPTION
		Annual Reports		
22.	8/30/85	L. Clark, M.I.T.	Dr. Murley, Region I, NRC	Letter Subject: Annual Report Period Covered: 7/1/84-6/30/85 (26 pages)
23.	8/29/86	L. Clark, M.I.T.	Dr. Murley, Region I, NRC	Letter Subject: Annual Report Period Covered: 7/1/85-6/30/86 (25 pages)
24.	8/29/87	L. Clark, M.I.T. & K. Kwok, M.I.T.	USNRC	Letter Subject: Annual Report Period Covered: 7/1/86-6/30/87 (26 pages)
25.	4/21/88	L. Clark, M.I.T.	USNRC	Letter Subject: Evaluation of Unresolved Safety Question, 10 CFR 50.59 (b)(2), MIT Reactor Lic. R-37, Docket 50-20 (119 pages)
26.	8/29/88	K. Kwok & J. Bernard, M.I.T.	USNRC	Letter Subject: Annual Report Period Covered: 7/1/87-6/30/88 (28 pages)
27.	8/30/88	K. Kwok & J. Bernard, M.I.T.	USNRC	Letter Subject: Revision to Annual Report Period Covered: 7/1/87-6/30/88 (29 pages)

MAR 21 1985

Docket No: 50-20

License No. R-37

Massachusetts Institute of Technology
Research Reactor
ATTN: Mr. Lincoln Clark, Jr.
Director of Reactor Operations
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

Subject: Inspection No. 50-20/85-01

A routine safety inspection was conducted on February 13-15, 1985 of the radiation protection program at the Massachusetts Institute of Technology Research reactor. Areas that were reviewed included implementation of radiation protection controls, equipment and instrumentation, and environmental monitoring.

This inspection indicated that one of your activities was conducted in violation of NRC requirements. Details are provided in enclosure Appendix A and in the accompanying inspection report. Your immediate corrective actions and actions to prevent recurrence have been provided in a letter to Dr. Thomas Murley, Regional Administrator, dated February 27, 1985. Therefore, no additional reply is required.

Your cooperation with us in this matter is appreciated.

Sincerely,

Original Signed By:
Walter J. Pasciak
for Thomas T. Martin, Director
Division of Radiation Safety
and Safeguards

Enclosures:
1. Appendix A, Notice of Violation
2. NRC Region I Inspection Report No. 85-01

cc w/encls:
Dr. O. K. Harling, Director of the Reactor Laboratory
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Massachusetts (2)

A/1

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OFFICIAL RECORD COPY DL50-20/85-01 - 0001.0.0

bcc w/encls:
Region I Docket Room (with concurrences)
~~Senior Operations Officer (w/o encls)~~

RI:DRSS	RI:DRSS	RI:DRSS	RI:DRSS	RI:DRSS
Cioffi/ms	White	Shanbaky	Bellamy	Martin
3/ /85	3/11/85	3/ /85	3/20/85	3/20/85

OFFICIAL RECORD COPY DL50-20/85-01 - 0002.0.0
03/07/85

APPENDIX A

NOTICE OF VIOLATION

Massachusetts Institute of Technology
Cambridge, Massachusetts

Docket No. 50-20
License No. R-37

As a result of the inspection conducted on February 13-15, 1985, and in accordance with the revised NRC Enforcement Policy (10 CFR 2, Appendix C), published in the Federal Register on March 8, 1984 (49 FR 8583), the following violation was identified:

10 CFR 71.5 requires each licensee, who transports licensed material outside of the confines of its plant, to comply with the applicable requirements of the regulations appropriate to the modes of transport of DOT in 49 CFR Parts 170 through 189. 49 CFR 172.203(d) requires that the description for a shipment of radioactive material must include: (i) the name of each radionuclide; (ii) a description of the physical and chemical form of the material; and (iii) the activity contained in each package in terms of curies, millicuries, or microcuries.

Contrary to the above, on September 13, 1984, a package containing 281 millicuries of rhenium-186 and 824 millicuries of rhenium-188 as rhenium wire was labeled and shipped with the incorrect description of radionuclide, physical and chemical form, and activity for the package. This material was shipped to Massachusetts General Hospital, identified as 8 millicuries of chlorine-38 in the chemical form of calcium chloride salt.

This is a Severity Level IV violation (Supplement V).

Pursuant to the provisions of 10 CFR 2.201, Massachusetts Institute of Technology is hereby required to submit to this office within thirty days of the date of the letter which transmitted this Notice, a written statement or explanation in reply, including: (1) the corrective steps which have been taken and the results achieved; (2) corrective steps which will be taken to avoid further violations; and (3) the date when full compliance will be achieved. Where good cause is shown, consideration will be given to extending this response time.

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-20/85-01

Docket No. 50-20

License No. R-37

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

Facility Name: MIT-R

Inspection At: Cambridge, Massachusetts

Inspection Conducted: February 13-15, 1985

Inspectors: for Jean A. Cioffi, Radiation Specialist, PWR Radiation Protection Section — 3/12/85 date

John R. White, Senior Radiation Specialist, PWR Radiation Protection Section — 3/12/85 date

Approved by: M. Shanbaky, Chief, PWR Radiation Protection Section — 3/18/85 date

Inspection Summary:
Inspection on February 13-15, 1984 (Report No. 50-20/85-01).

Areas Inspected: Routine, unannounced safety inspection of the radiation protection program, including: the status of previously identified items; radiological surveys, postings, material labeling, and controls; equipment, instrumentation, and leak tests; environmental monitoring.

The inspection involved 30 hours on-site by two region-based inspectors.

Results: Of the areas inspected, one violation of transportation requirements was identified, i.e., failure to properly label a radioactive shipment with respect to radionuclide identity, physical and chemical form, and correct activity as required by 49 CFR 172.203(d), paragraph 5.0.

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DETAILS

1.0 Persons Contacted

E. Karaian, Reactor Radiation Protection Officer
L. Clark, Jr., Associate Director, Nuclear Reactor Laboratory
P. Coggio, Reactor Radiation Protection Technician

2.0 Purpose

The purpose of this routine, unannounced, safety inspection was to review the licensee's radiation protection program with respect to the following elements:

- Status of Previously Identified Items
- Radiological Surveys, Posting, Material Labeling, and Controls
- Radioactive Material Identification
- Equipment, Instrumentation, and Leak Tests
- Environmental Monitoring

3.0 Status of Previously Identified Items

3.1 (Closed) Violation (83-02-01). Failure to post hot cell on reactor top as high radiation area and to control personnel access to the area. The licensee's corrective actions, as stated in Inspection Report 83-02, Section 5b, were reviewed to verify their implementation. Implementation of the corrective actions appeared to be adequate to prevent recurrence.

3.2 (Closed) Follow-up (83-02-02). Radiation protection to control use of radiation barricades and signs to avoid misuse. For other than radiological control purposes, the licensee has purchased white ropes for the researchers to use to enclose their equipment and experiments.

3.3 (Closed) Follow-up (83-02-03). Radioactive contamination control by individuals working in materials laboratory section of Engineering laboratory. The floors in front of Hoods 1 and 2 of NW12-139 are surveyed daily for contamination. Monthly surveys are performed for the entire laboratory area. Contaminated areas are cleaned immediately.

4.0 Radiological Surveys, Postings, Material Labeling, and Controls

The licensee's program for surveys, postings, labeling, and controls was reviewed against the criteria contained in 10 CFR 20.105, 20.106, 20.201, 20.203, 20.204, 20.207, and 20.401.

The licensee's performance relative to these criteria was evaluated by:

a. Examination of records of daily and monthly surveys for 1983 and 1984;
b. A tour of the facility;
c. Observation of signs and postings on equipment, in laboratories, in hallways, and on doors;
d. Direct radiological measurements of areas in the facility with a GM detector and a "Juno" ionization chamber;
e. Observations of access controls for the reactor building, and for monitoring activities within the reactor building; and
f. Discussions with licensee representatives.

The inspector noted the following:

Gamma surveys and smears are taken daily on floors and in common areas. Monthly surveys are performed in laboratories and near equipment and radioactive waste storage areas. The Radiation Protection Officer is informed daily of any contaminated areas through the use of a daily status sheet, which identifies the contaminated areas and states the corrective actions taken.

Access to the reactor building and auxiliary facilities, such as the radwaste storage areas and laboratories, is controlled with a key card. Personnel entering the reactor building are required to call the control room and notify the operator of their intent to enter. The entrance to the reactor building is observed in the control room by a closed circuit TV camera. The TV camera can also be moved to observe approximately half of the reactor floor and the top of the reactor where a hot cell is located.

There were no violations identified in review of this area.

5.0 Radioactive Material Identification

The inspector investigated two incidents, which occurred on September 4, 1984 and on September 13, 1984, in which rhenium wire samples were mistaken for other radioactive samples. On September 4, 1984, a reactor operator was directed to package and release a strontium chloride sample by an experimenter. The wrong sample was mistakenly packaged, but not shipped when it was fortuitously determined that the sample was erroneously identified.

In the incident occurring on September 13, 1984, a rhenium wire consisting of 281 millicuries of rhenium-186 and 824 millicuries of rhenium-188 was packaged, labeled, and shipped to Massachusetts General Hospital. The package was labeled and shipped as 8 millicuries of chlorine-38, in the form of calcium chloride salt. This incident constituted a violation of 49 CFR 172.203(d), which states that each package of radioactive material must be identified as to radionuclide identity, physical and chemical form, and amount of activity.

As a result of this occurrence, one Massachusetts General Hospital employee sustained minor unplanned exposure to the wrist and whole body, of 150 millirads and 25 millirads, respectively.

The licensee determined that the cause of this occurence was misidentification of the samples on the sample storage map, located on the outer wall of the hot cell. As a result, the licensee initiated the corrective actions listed below:

a. Two internal memos were circulated emphasizing the importance of accurately marking the identity of samples on the sample map located outside the hot cell. The memos also stated that beta surveys were to be performed on all samples in addition to gamma surveys. All reactor operators initialled the memo to verify that they read it.

b. A lead container was placed in the hot cell and labeled "Rhenium Only," for the placement of the rhenium wire samples. Rhenium wire samples are now placed in this container only.

The inspector interviewed one reactor operator to evaluate the effectiveness of this corrective action. The reactor operator stated that the beta survey was not being performed. All other corrective actions were being implemented.

The inspector discussed the failure to perform the beta survey with the Radiation Protection Officer (RPO). The Radiation Protection Officer stated that he disagreed with the newly instituted requirement to perform beta survey of the samples because this practice would increase exposure to the operators which would not be consistent with good ALARA practices.

On February 21, 1985, the Radiation Protection Officer and the Associate Director, Nuclear Reactor Laboratory, telephoned the inspector to present new corrective actions. The following actions were discussed with the inspector.

a. The requirement to beta survey would be eliminated;
b. A specific procedure will be written for work in the hot cell. The identity of the sample will be specified in one or more of the following ways:

    (1) Use of the sample reference number;
    (2) Use of any distinguishable marks on the sample and the mark recorded on the work form (Part II); and
    (3) Use of any unique shape of the samples and the shape recorded on the work form (Part II).

c. The gamma dose will be verified on the work form; and
d. The importance of confirming the identity of the sample with the work form (Part II) will be reemphasized.

The effectiveness of these corrective actions will be examined in a future inspection (85-01-01).

6. Equipment, Instrumentation, and Leak Tests

The licensee maintains logs of all instrument calibrations. Survey instruments and monitors are calibrated quarterly. Effluent radiation monitors are calibrated yearly and checked on a quarterly basis for response to a radioactive source.

Argon sampling and monitoring are performed continuously using a GM detector which views a known volume of gas. A strip chart records all data. Counts are summed over one week. Additional air sampling equipment is mounted on carts and moved to various locations, such as port openings, when needed.

Leak tests are performed quarterly and semiannually, depending on the type of source. The licensee has determined that the lower level of detection for their leak tests is 1.8×10^{-6} microcuries alpha, and 9×10^{-6} microcuries beta. Accurate records are kept of leak tests with a clear description of the type of wipe (i.e., dry or wet).

Within the scope of this review, no violations were observed.

7. Environmental Monitoring

The licensee's program for environmental monitoring was reviewed against criteria contained in 10 CFR 20.106 and Appendix B, Table II.

The licensee's performance relative to these criteria was evaluated by:

a. Visual inspection of two separate environmental monitoring stations for working instrumentation, and location with reference to the reactor stack;
b. Discussions with the Reactor Health Physics technician and the Radiation Protection Officer on the calibration of the instrumentation and data collection and calculations; and
c. Review of the annual reports for 1983 and 1984.

The licensee uses GM tubes for their environmental monitoring stations. Each GM tube is connected to a count-rate meter located inside a sheltered area. The signal from the count rate meter is sent through telephone transmission lines to strip chart recorders located inside the Health Physics office at the Nuclear Reactor Laboratory. Data on the strip chart recorders is collected daily and summed monthly.

Within the scope of this review, no violations were observed.

8. Exit Interview

The inspector met with licensee management at the conclusion of the inspection of February 15, 1985 to discuss the scope and findings of the inspection as detailed in this report. At no time during this inspection effort was written material provided to the licensee by the NRC inspector.

APR 09 1985

Docket No. 50-20

Massachusetts Institute of Technology
ATTN: Lincoln Clark, Jr.
Director of Reactor Operations
138 Albany Street
Cambridge Massachusetts 02139

Gentlemen:

Subject: Inspection Report No. 50-20/85-01

This refers to your letter dated February 27, 1985, in response to our telephone conversation of February 21, 1985.

Thank you for informing us of the corrective and preventive actions documented in your letter. These actions will be examined during a future inspection of your licensed program.

Your cooperation with us is appreciated.

Sincerely,

Original Signed By:
Ronald R. Bellamy
for: Thomas T. Martin, Director
Division of Radiation Safety
and Safeguards

cc:
Public Document Room (PDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Massachusetts (2)
Dr. O. K. Harling, Director of the Reactor Laboratory
Local Public Document Room (LPDR)

bcc:
Region I Docket Room (with concurrences)

Coffin
Shanbaky
Bellamy

OFFICIAL RECORD COPY

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NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

138 Albany Street Cambridge, Mass. 02139
(617) 253-4202

L. CLARK, JR.
Director of Reactor Operations

February 27, 1985

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

Subject: Inspection No. 50-20/85-01

Dear Dr. Murley:

In connection with the above health physics inspection at the MIT Research Reactor on February 13-15, 1985 by Ms. Jean Cioffi and Mr. John White, it was requested that MIT provide by mail the following two items of information, since they were not available at the time of the inspection.

The first item is the paperwork associated with a radioisotope shipment that was made from the MITR to Massachusetts General Hospital (MGH) on September 13, 1985. The enclosed documents (Irradiation Information Form - Part II and Isotope Shipping Memo) relate to a shipment that was intended to be chlorine-38, but a rhenium-186 wire sample was inadvertently packaged and shipped instead. It should be recognized that the MITR has properly shipped more than three hundred chlorine samples to MGH and many thousands of other samples to other facilities. This instance is the only time that the wrong material has been released.

The second item is related to the preparation of a document covering the procedures that MITR operators are instructed to follow for the release of the irradiated materials for shipment. The procedure as now written specifies that:

1) Samples to be irradiated in the pneumatic tubes be identified by some distinctive feature or marking and, prior to the release of an irradiated sample, its identity be verified against this distinctive feature.

In addition, the procedure incorporates long-standing existing practice which was and is that:

2) A chart be maintained showing locations of irradiated samples stored in either of the pneumatic tube sample changing areas,

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3) Prior to release of any irradiated sample, it be identified by reference to the chart and to the Irradiation Information Form - Part II, its expected gamma dose be confirmed, the sample be packaged as required by DOT and/or NRC regulations, necessary paperwork be completed, and the storage chart be updated.

The written procedure covering the above was approved and issued on February 27, 1985.

Please contact me if any further information is required in this regard.

Sincerely,

Lincoln Clark, Jr.

Lincoln Clark, Jr.
Director of Reactor Operations

cc: J. Bernard, MIT
J. Cioffi, NRC Region 1
E. Karaian, MIT

Reference No. 50-42-141 A,B,C

MIT Proc. Ref. N/A

Name P. HOOP

Organ. M.G.H.

Phone 732-3732

MASSACHUSETTS INSTITUTE OF TECHNOLOGY
Cambridge, Massachusetts 02139

PART II IRRADIATION INFORMATION FORM

Sample: material CaCl2

weight 0.1gm/each

Time desired 30min / 5h

Heating ✓

Irrad. location 1PH1/2PH1 Power 4.9MW Flux 8x10^13/5x10^13 Reactivity ✓

Principal Nuclides	Cl36	Cl38m	Ca47				DOSE AT 100 M
Initial Activity	14mci	0.3mci	0.6mci				12mr/hr
Activity After 30MIN Decay	8mci	—	—				7mr/hr

Special provisions

Approval: (ROUTINE) (NON ROUTINE) Signature 2 required only if non routine.

1) Prior approval certified per. ref. 50-42 W McDermott date 3/8/84

2) Radiation Protection Officer date

Irradiation:

Date & Time in 13SEP84 1000 A Date & Time out 13SEP84 1030

13SEP84 1202 B 13SEP84 1232

13SEP84 1400 C 13SEP84 1430

MIT shipped a only sample shipped 22-85-300

Total Time 4 1/2 hr at 4.0MW MW Reactor down time

Activities:

Gamma activity 250/250/800 mr/hr at 10

Sample contamination: wipe activity contaminated (cpm / mr/hr) on 6"

Date 13SEP84 Time 1032 / 1232 / 1432 By [illegible]

Shipping:

Container radiation levels:

surface 25/25/35 mr/hr: at 1 meter 0.8/1.0/0.8

container wipe test NOA

label: principal isotopes Cl38 activity 8mCi index 0.8/

time date

Release: or other disposition, by

Under License No.

Waste Disposal

Other

Signature of person accepting sample Russell Amb. #239 9-13-84

Organization

Charges:

Facility $ 63 Handling $ 32.01 - 10.00 Total $ ~~126.50~~ 116.00

Account No. 116.00

NUCLEAR REACTOR LABORATORY
Massachusetts Institute of Technology
138 Albany Street Cambridge, Mass. 02139

№ 628

ISOTOPE SHIPPING MEMO

Invoice to:
DR. B. HOOP
MASS. GENERAL HOSPITAL

Ship to (if different from invoice to)

Customer Order No.	Customer License No.	Terms	MITR Irradiation Ref. No.
	20-03814-80		50-42-100C

Cask No. & Type	Seal	Routing	Shipped in	Weight/Pkg.	Total Weight
DOT 7A TYPE A	YES	☑ Medical ☐ Industrial	1 Packages	45#	45#

Isotope Name and Mass. No.	Lot No.	Product Weight or Volume	Physical Form	Chemical Form
Cl^{38}		0.1gm	SALT POWDER	$CaCl_2$

Total Activity Shipped	Specific Activity As Shipped	Cal. Date & Time	Calibrated Activity Ordered	Calibrated Activity Shipped	Unit Price	Amount
8mci	80mci/gm					

Other Charges (list)

Total

Special Instructions

☐ Exclusive Use Vehicle
Surface ______ Mr/Hr
At 6 feet ______ Mr/Hr

RADIATION SURVEY:
This survey is the result of readings taken after the material is completely packaged and ready for shipment.
☑ Smearable contamination meets U.S.D.O.T. requirements.
Maximum surface radiation: 35 Mr/Hr
Transport index (Mr/Hr @ 3 ft.) .8
Labels affixed: ☐ White I, ☑ Yellow II, ☐ Yellow III, ☐ NR
☐ Vehicle placarded
☑ This is to certify that the named articles are properly described, packaged, marked, and labeled and are in proper condition for transportation and in accordance with applicable regulations of the U.S.D.O.T. and IATA restricted article regulations.
☐ This shipment is within the limits prescribed for passenger carrying aircraft.

Radiation Survey by: [signature]
Caution: This material has not been sterilized or certified for medical use in its present form.

Packaging & Shipping Information

Description:
☑ Radioactive Material N.O.S. UN2982
☐ Radioactive Material Special Form
Transport Group ~~III or IV~~ 4-2

Shipment Released by: [signature] Date: 9-13-84 Time: 3:10 P.M.

Received by: Powell Amb # 239 Date: 9-13-84 Time: 3:15 PM

Copy 1: MITR OPS
White

MIT-NRL Shipping Checklist for DOT 7A-Type A Reusable Containers

This checklist and an MIT-NRL shipping memo should both be completed prior to releasing any non-liquid radioactive sample in a DOT 7A-Type A reusable container. These forms document the mechanical integrity of the container. Refer to the isotope shipping memo for radiation levels and required labels.

✓ MITR Irradiation Ref. No. 50-42-10α.

✓ Isotope container integrity satisfactory with product packaged in 2R inner container with packing material sufficient to prevent movement and/or damage and

✓ Shield in place and cap secured on 2R inner container.

✓ Threads of 2R container lubricated with graphite, teflon tape etc.

✓ 2R container cover screwed on with at least five turns.

✓ Packing material in place about 2R container. Top section of packing is within two inches of drum cover but not so close as to inhibit proper attachment of cover/bolt assembly.

✓ Drum cover and gasket intact.

✓ Drum cover and bolt assembly in position over cover and drum rims with bolt threaded through ring eyes with ring separation at ends less than one inch.

✓ Security seal wound around bolt eyes end to end and through bolt hole.

✓ Security seal sealed with crimper.

[signature] Signature

9-13-84 Date

File with Operations copy of the Irradiation Reference Form.

QA #-M-80-5

June 22, 1982

NOV 2 1 1985

Docket No. 50-020

Massachusetts Institute of Technology
Research Reactor
ATTN: Mr. Lincoln Clark, Jr.
Director of Reactor Operation
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

SUBJECT: EXAMINATION REPORT NO. 50-20/85-03 (OL)

This transmits the Examination Report of Operator Licensing Examinations conducted by USNRC Region I at the MIT Facility the week of September 30, 1985. At the exit interview held with Mr. J. Bernard and Mr. K. Kwok on October 1, 1985, the preliminary results of these examinations were discussed.

No reply to this letter is required. Your cooperation in this matter is appreciated.

Sincerely,

Original Signed By:

Edward C. Wenzinger, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosure:
Examination Report No. 50-20/85-03 w/attachments 1, 2

cc: w/enclosure and attachments 1, 2
Dr. O. K. Harling, Director, Reactor Laboratory
John A. Bernard, Training Coordinator
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
State of Massachusetts

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10/29/85

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bcc: w/enclosure, w/o attachments 1, 2
DRP Section Chief
D. Silk, Examiner
Chief, OLB/DHES,NRR
OL File 12.0
Region I Docket Room (w/concurrences)
Master Exam File

DRP:RI	DRP:RI	DRP:RI	DRP:RI	DRP:RI
DSilk/fi	RKeller	EMcCabe	HKister	EWenzinger
10/31/85	11/4/85	11/12	11/8/85	11/19/85

REPORT DETAILS

TYPE OF EXAMS: Initial___ Replacement X__ Requalification____

EXAM RESULTS:

	RO Pass/Fail	SRO Pass/Fail
Written Exam	2/0	1/0
Oral Exam	1/1	1/0
Overall	1/1	1/0

1. CHIEF EXAMINER AT SITE: David M. Silk

2. OTHER EXAMINER: Robert M. Keller

1. Summary of generic strengths of deficiencies noted on oral exams:

   Candidates displayed a good understanding of the plant. SRO candidate displayed a weakness in not assuming all responsibilities assigned to SRO by transferring responsibilities to plant management personnel who hold SRO licenses.

2. Summary of generic strengths or deficiencies noted from grading of written e ams:

   Candidates were not familiar with:

   - relationship of early xenon peaking to harder neutron spectrum
   - modes of operation for the 1-inch pneumatic tube system
   - the hazard of drying out charcoal filters
   - how to seal beam ports

3. Comments on availability of, and candidate familiarization with plant reference material in the control room:

   Candidates were familiar with plant reference material.

~~8512060062~~ 74 pp.

4. Personnel Present at Exit Interview.

   NRC Personnel

   David Silk

   Facility Personnel

   John Bernard
   Kwan Kwok

5. Summary of NRC Comments made at exit interview:

   - Two of the three candidates were clear passes on the oral examination.

   - Facility training material provided for examination preparation was well organized.

6. CHANGES MADE TO WRITTEN EXAM DURING EXAMINATION REVIEW:

Question No.	Change	Reason
A.2	Delete question.	The question called for a comparative knowledge of reactor types.
B.2	Include in answer "Verify system pressure".	Expands answer Key.
B.6	Delete from answer "the pitch of fan blades can be changed".	Inoperable at present.
C.1	Delete from question "The reactor has just been started".	Can mislead candidate. Clarifies question.
C.7	Also accept Answer a).	Unusual Occurrence Report #81-4 justifies answer a).
D.5	Delete question.	This experiment (FCE) has been out of the reactor for two years.

Question No.	Change	Reason
E.5	Also accept Answer c).	If some loads are shed, the battery could supply power for about 12 hours.
E.7	Include in answer - Weekend - Intrusion (Interior/ Exterior) - Fuel Vaults - Operator Incapacitated - Panic Button (In control room or receptionist desk)	These alarms will transmit a signal to the campus Patrol Alarm System.
G.1	Include in answer "Check radiation levels" "Order personnel out"	Expands answer Key.
G.3	Also accept Answer c).	Surface contamination includes beta radiation.
G.8	Include in answer "Gas monitor on reactor floor by main airlock".	Expands answer Key.
G.11	Include in answer "To prevent nitric acid formation from nitrous oxide".	Expands answer Key.
J.2	Include in answer a). - Reactor floor hot - 36V's if not sealed - A drop in building temperature Include in answer b). - Use helium gas - Seal ports	Expands answer Key.

Attachments:

1. Written Examination and Answer Key (RO)
2. Written Examination and Answer Key (SRO)

MASTER

U.S. NUCLEAR REGULATORY COMMISSION

REACTOR OPERATOR LICENSE EXAMINATION

Facility: MITR-II

Reactor Type: HWR/LWR Cooled/Moderated

Date Administered: October 1, 1985

Examiner: W. J. Apley/ J. C. Huenefeld

Candidate: Answer Key

INSTRUCTIONS TO CANDIDATE:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category. Examination papers will be picked up six (6) hours after the examination starts.

Category Value	% of Total	Candidate's Score	% of Cat. Value	Category
15.0	14.9			A. Principles of Reactor Operation
13.0 ~~14.0~~	13.9			B. Features of Facility Design
14.5	14.4			C. General Operating Characteristics
15.0	14.9			D. Instruments and Controls
15.0	14.9			E. Safety and Emergency Systems
13.5	13.4			F. Standard and Emergency Operating Procedures
14.0	13.9			G. Radiation Control and Safety
100 ~~101.0~~				TOTALS

Final Grade ________ %

All work done on this exam is my own. I have neither given nor received aid.

Candidate's Signature

A. PRINCIPLES OF REACTOR OPERATION (15.0)

Points Available

QUESTION A.1

When calculating an estimated critical position, the operator uses the previous week's position and corrects for five different delta K changes. List four (4) of those delta K changes. (2.0)

ANSWER A.1

Delta K due to temperature change
due to sample loading
due to Xenon
due to fuel loading
due to burnup

(4 of 5 for full credit)

REFERENCE A.1

PM 3.1.1.2, p.11

QUESTION A.2

The MITR-II reactor produces a relatively fast response to a given reactivity input. Explain that response in terms of what the values of neutron generation time and delayed neutron fraction are at MITR-II. (I.e., are both Beta and generation time small, one small and the other large, etc.) (2.0)

ANSWER A.2

The sensitive response is due to the short neutron generation time for the MITR-II, even though its delayed neutron fraction is large (beta-bar = 0.00786). The large Beta effective is predominately due to a large source of "slow born" photo neutrons developed in the reflector.

REFERENCE A.2

RSM 10.5

-Section A Continued on Next Page-

Points
Available

QUESTION A.3

Why isn't the MTR type elements cladding thicker or thinner? (1.5)

ANSWER A.3

It's thick enough to retain fission products (+0.5), and thin enough to not introduce a long delay time for heat removal in the event of a fast transient (+0.5).

REFERENCE A.3

Tech Spec 5-4

QUESTION A.4

Explain the two (2) ways that the control elements affect reactivity as they are moved in the core. (1.5)

ANSWER A.4

When inserted in the annular space between the core and the core housing assembly, these control elements decrease reactivity both by the direct absorption of neutrons and, to a lesser extent, by warping the core flux distribution, thereby increasing neutron leakage.

(+1.0 for absorbtion/+0.5 for increasing leakage)

REFERENCE A.4

RSM 10.5

-Section A Continued on Next Page-

Points
Available

QUESTION A.5

If the reactor is on a stable 25-second period, how long will it take to change power level 2 decades (show calculation)? (2.0)

ANSWER A.5

From equation sheet:

$$Sur = \frac{26.06}{T} = \frac{26.06}{25.00} = 1.0424$$

$$P = P_o 10^{sur\ t}$$

$$\frac{P}{P_o} = 100 = 10^{sur\ t}$$

$$2 = sur\ t$$

$$t = 2/1.0424 = 1.92 \text{ minutes}$$

If the candidate doesn't know about SUR (which is checked w/o calculation in A.1), then he can calculate using formula sheet.

$$P = P_o e\ t/T$$

$$P/P_o = 100 = e^{t/25\ sec}$$

$$\ln 100 = t/25\ sec$$

$$t = (25\ sec)(\ln 100)$$

$$= 115.13 \text{ seconds} = 1.92 \text{ minutes.}$$

REFERENCE A.5

Glasstone and Sesonske (MITR Trng Progr. Ref.)
PM 1.16.2, p.1

-Section A Continued on Next Page-

Points
Available

QUESTION A.6

TRUE or FALSE: Xenon peaks earlier in MITR-II after shutdown due to a harder neutron spectrum. (0.5)

ANSWER A.6

True

REFERENCE A.6

RSM 10.7

QUESTION A.7

Describe the two (2) phenomena that contribute to the temperature coefficient of reactivity for MITR-II. (2.0)

ANSWER A.7

The first is the temperature rise of the light water due to an increase in the thermal output of the reactor core. Any such temperature rise will insert negative reactivity by causing a hardening in the neutron spectrum. (This means that the average neutron takes longer to thermalize so there are fewer fissions.) The second phenomenon is the radiation heating of the heavy water reflector. Temperature rises of this type add negative reactivity by allowing more neutron leakage to increase. This second process lags the temperature rise of the light water in the core proper.

REFERENCE A.7

RSM 10.8

-Section A Continued on Next Page-

Points
Available

QUESTION A.8

If heavy water leaks into the light water system, what type of reactivity effect will it have if:

A. The leakage of pure, uncontaminated heavy water is into either the light water reflector above the top of the core, or the light water reflector below the top of the core that is formed by the annular space between the core and the sides and bottom of the core tank. (0.5)

B. Leakage of heavy water is into the core proper. (0.5)

C. The in-leaking D_2O progressively replaced the entire light water system. (0.5)

ANSWER A.8

A. Positive reactivity
B. Strong, negative reactivity
C. Strong, negative reactivity

REFERENCE A.8

RSM 10.11

-Section A Continued on Next Page-

Points
Available

QUESTION A.9

A nuclear reactor has a shutdown margin of 7% delta k/k and a neutron detector is recording 20 cpm. What will this detector read when $k_{eff} = 0.99$? (2.0)

ANSWER A.9

$$\frac{1 - K_1}{K_1} = 0.07$$

$$1 = K_1 + 0.07\ K_1$$

$$1 = 1.07\ K_1$$

$$K_1 = 1/1.07 = 0.93$$

$$\frac{1 - K_1}{1 - K_2} = \frac{C_{R2}}{CR_1}$$

$$\frac{0.07}{0.01} = \frac{CR_2}{20}$$

$$CR_2 = 140 \text{ cpm}$$

REFERENCE A.9

1. Generic: "Academic Program for Nuclear Power Plant Personnel," Volume II, pp. 5-6 through 5-13, General Physics corporation.
2. Glasstone and Sesonske (MITR Trng. Progr. Rev.) PM 1.16.2, p.1

-End of Section A-

B. FEATURES OF FACILITY DESIGN (14.0)

Points Availble

QUESTION B.1

Describe the four (4) modes of operation for the 1-inch pneumatic tube system. (2.0)

ANSWER B.1

A. Insertion and removal at the hot cell or primary chem room in the reactor basement.

B. Insertion at the hot cell and transfer of the irradiated sample to the NW-13 hot lab via the connecting pneumatic tube.

C. Insertion from the NW-13 hot lab, into the reactor, and transfer of the irradiated sample back to the NW-13 hot lab.

D. Transfer of a rabbit from the basement hot cell to the NW-13 hot lab.

REFERENCE B.1

PM 1.10, p. 7

QUESTION B.2

How does the operator verify that the secondary system is properly lined up to cooling tower basins? (1.5)

ANSWER B.2

Verify secondary system is propery lined up to cooling tower basins by either checking HV-14 of HV-14A open or by checking HM-1A running with flow through HF-3 at 60% of scale. (Either answer correct.)

REFERENCE B.2

PM 3.1.1.1, p. 2

-Section B Continued on Next Page-

Points
Available

QUESTION B.3

What design safety feature ensures that fuel loaded into the core will normally have access to only one core position at a time? (1.25)

ANSWER B.3

Hold-down grid latch must be released and the grid rotated to permit core access. Grid design prevents multiple position access.

REFERENCE B.3

PM 2.7, p.3

QUESTION B.4

If the pressure relief system's charcoal filters become submerged, what problems will exist during filter housing and exhaust dryout? (1.25)

ANSWER B.4

The charcoal generates heat while drying out and may cause spontaneous combustion.

REFERENCE B.4

PM 5.2.14, p. 2

-Section B Continued on Next Page-

Points
Available

QUESTION B.5

Explain how the anti-syphon valves work. (1.0)

ANSWER B.5

Ball float valves installed at the top of the core shroud. Inlet flow forces ball up closing outlet at top; w/o flow gravity forces ball down to break syphon.

REFERENCE B.5

RSM 1.7

QUESTION B.6

List three (3) ways to reduce the degree of cooling tower efficiency on cold days. (2.0)

ANSWER B.6

The yard booster pumps may be bypassed partially or completely, as may the towers themselves. One of the cooling tower fans may be operated at half-speed, the pitch of the fan blades can be changed, and the air admitted to the towers can be restricted by rearranging the external boards and flaps.

(Any three.)

REFERENCE B.6

RSM 3.12

-Section B Continued on Next Page-

Points
Available

QUESTION B.7

How are beam ports sealed? (1.5)

ANSWER B.7

A. A plug is placed in port
B. Gas seals
C. Gasketed cover bolted over beam port's opening

REFERENCE B.7

RSM 2.4

QUESTION B.8

Assume a loss of external electrical power feeders occurred. When normal power is later restored, what will happen to all the transfer switches and the motor generator set? (1.0)

ANSWER B.8

A. Transfer switches return to normal.

B. Relay at the motor-generator set is energized, thereby stopping the unit.

REFERENCE B.8

RSM 8.32

-Section B Continued on Next Page-

Points
Available

QUESTION B.9

Draw a top view of the core, including location of the: 1.5 ~~(1.0)~~

A. Regulating rod (.5)
B. Shim blades (.5)
C. Radial absorber plates (.25)
D. Hexagonal absorber plates (.25)

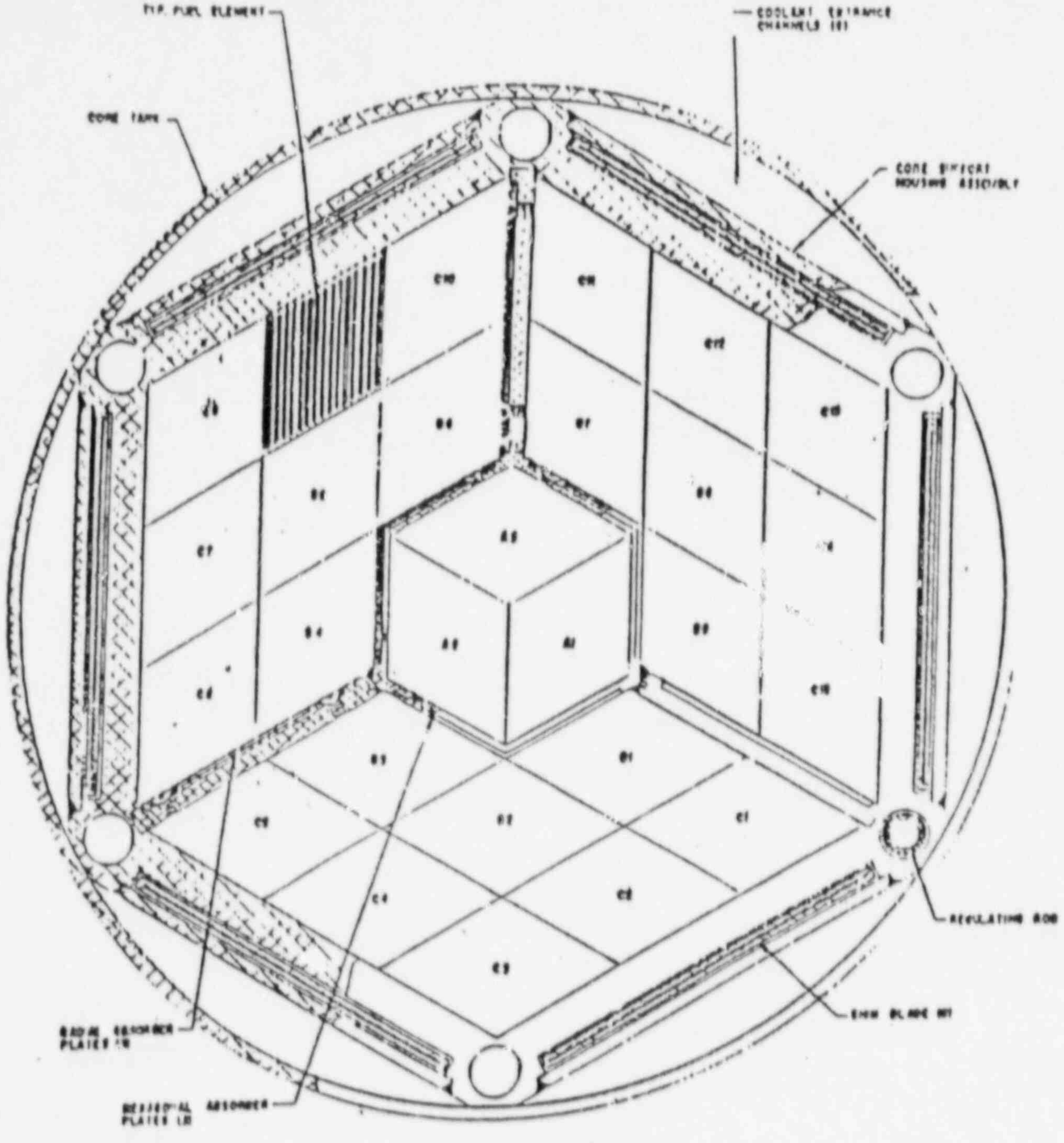
ANSWER B.9

See next page.

REFERENCE B.9

See attachment.

-Section B continued on next page-



Answer B.9

Core Section M.I.T.R. II

Reactor Systems Manual

-End of Section B-

MITR-II
October 1, 1985

C. GENERAL OPERATING CHARACTERISTICS (14.5)

Points Available

QUESTION C.1

The reactor has just been started up. Explain why nuclear instrumentation must be frequently calibrated in terms of thermal power as short lived fission product poisons (such as Xenon) build up in the reactor core. (3.0)

ANSWER C.1

Compensation for the negative reactivity associated with the building in of equilibrium xenon is achieved by withdrawing the shim blades. The out-motion of the shim bank causes the axial flux profile of the reactor to change with the point of maximum flux moving upward. That, in turn, alters the leakage flux which is what is viewed by the nuclear instrumentation. This affects reactor control in the following manner. The automatic control system controls the reactor by maintaining a constant flux at the location of the chamber that feeds the auto-control network. Hence, as the axial flux profile changes with shim bank height, the auto-control channel will detect a "power-change". In reality, of course, there is no net change in power, but a redistribution of power within the core. This is why it is essential to determine the thermal power output of the reactor by means of a heat balance which is not affected by flux distribution.

(+1.0 - change in axial flux profile)
(+1.0 - auto-control "sees" power change)
(+1.0 - need to re-calibrate to thermal, not distributed power)

REFERENCE C.1

PM 2.4, p.1

-Section C Continued on Next Page-

Points
Available

QUESTION C.2

What is the maximum amount of reactivity in percent of delta k/k that may be added to the critical reactor without causing damage to the fuel integrity by the resulting power transient? (1.0)

ANSWER C.2

1.8%

REFERENCE C.2

Tech Specs 3-8

QUESTION C.3

Why does it take 24 hours for the reactor to be in thermal equilibrium, such that a heat balance can be conducted? (1.0)

ANSWER C.3

Graphite reflector has a large heat capacity and is slow to attain an equilibrium temperature distribution.

REFERENCE C.3

RSM 6.4

-Section C Continued on Next Page-

Points
Available

QUESTION C.4

Why is "blowdown" of the water in the Forced Draft Cooling Towers required? (1.0)

ANSWER C.4

Forced draft cooling towers concentrate the solids in the makeup water and collect atmospheric dust. Hence, a feed-and-bleed purge is maintained while they are in operation in order to keep the level of dissolved solids within a factor of three to five times that of the makeup water. A small portion of the water is diverted through a flow accumulation meter directly to the sewer. This flow is called "blowdown".

REFERENCE C.4

RSM 3.12

-Section C Continued on Next Page-

Points
Available

QUESTION C.5

A. Explain how the reactivity effect of dumping the radial reflector varies with the position of the shim blades. (1.5)

B. Why is the radial heavy water reflector pumped up with the shim bank in the fully inserted position? (1.0)

ANSWER C.5

A. In as much as the shim blades also operate in the region between the core and the radial heavy water reflector, the reactivity worth of dumping this radial reflector is dependent on the position of the shim blade bank. This effect can be considered as being due to the shadowing influence that the blade bank exerts on the reflector. These results show that the reactivity worth of dumping the radial heavy water reflector when the shim bank is fully inserted is about two-thirds that of the corresponding value when the bank is at the top of the active core.

(+0.5 for reason, +1.0 for knowing more reactivity with rods at top.)

B. Safety considerations dictate that the radial heavy water reflector be pumped up with the shim bank in the fully-inserted position. This ensures that the reactivity insertion for this process will not occur when the reactor is or could go critical.

REFERENCE C.5

RSM 10.6

-Section C Continued on Next Page-

Points
Available

QUESTION C.6

You receive a high temperature shield coolant outlet alarm. The shield coolant outlet temperature is rising slowly, and there is no evidence of a loss of shield flow or level. Operationally, what is the probable cause? (1.5)

ANSWER C.6

The secondary side of the heat exchanger is probably clogged with mud. (Will accept other answers alluding to degraded HX performance.)

REFERENCE C.6

PM 5.4.8

QUESTION C.7

Approximately how long after a failure of the pneumatic blower (at full power), will the temperature in the pneumatic tubes reach 100 degrees C (select best answer)? (1.0)

A. Instantly
B. 5 Minutes
C. 30 Minutes
D. Never

ANSWER C.7

B. 5 Minutes

REFERENCE C.7

PM 5.5.1

-Section C Continued on Next Page-

Points
Available

QUESTION C.8

TRUE or FALSE: It does require bypassing a number of safety functions, but it is possible to operate in the 100 kw mode with no forced circulation of primary coolant. (0.5)

ANSWER C.8

True

REFERENCE C.8

PM 2.2

QUESTION C.9

Describe how to calculate the total thermal power output of the reactor. (3.0)

ANSWER C.9

Primary Power = (2.62×10^{-4})(Primary Flow)(Primary delta T)

Reflector Power = (2.91×10^{-4})(D_2O Flow)(D_2O delta T)

Shield Power = (2.62×10^{-4})(Shield Flow)(Shield delta T)

Total Power = Primary + Reflector + Shield Power

#s not important, just the parameters and three constituents of total power.

REFERENCE C.9

PM 2.4, p.5

-End of Section C-

MITR-II
October 1, 1985

D. INSTRUMENTS AND CONTROL (15.0)

Points Available

QUESTION D.1

What is the purpose of the AUTO TRANSFER ABORT switch in the reactor control room? (1.5)

ANSWER D.1

The AUTO TRANSFER ABORT switch in the reactor control room is used to eject a sample from the reactor, and cause it to exit into the reactor rabbit station, thus blocking its transfer to the NW-13 hot lab. The rabbit tube it controls (1PH1 or 2PH1) is determined by the position of the AUTO TRANSFER SELECTOR switch at the rabbit station. Also, in the case of 1PH1, a sample which had been previously ejected and was being monitored at the stop pin could be exited into the station.

Full credit for answer 1; half-credit for 2 only.

REFERENCE D.1

PM 1.10, p. 11

-Section D Continued on Next Page-

Points
Available

QUESTION D.2

If automatic reactor operation is desired, the power-set is adjusted to bring the power-setpoint deviation indication to zero at the desired power level. Why must the scale be adjusted on channel #9 (the automatic control channel) so that its signal is reading mid-range on the indicating meter? (2.0)

ANSWER D.2

If this signal is at either the low or high end of the display meter, the automatic control will either not take control or be sluggish in its response.

REFERENCE D.2

PM 2.3, p. 5

QUESTION D.3

Small changes in power may be made through the automatic control system. This is done by slowly varying the setpoint of the power-set potentiometer and adjusting the scales of the other instruments as necessary.

What would happen if the operator moved the setting too rapidly? (1.5)

ANSWER D.3

The deviation meter trip would be exceeded and reactor control would trip off automatic.

REFERENCE D.3

PM 2.4, p. 4

-Section D Continued on Next Page-

Points
Available

QUESTION D.4

If a 3 GV hole that contains a Nuclear instrument detector is flooded, what will happen to the detector output? Explain why. (1.5)

ANSWER D.4

Output will decrease (+1.0) due to the increased attenuation of the neutrons (+0.5).

REFERENCE D.4

PM 5.4.11

QUESTION D.5

The fatique cracking experiment alarm is actuated. Name two (2) of the four (4) abnormal conditions which could cause such an alarm. (2.0)

ANSWER D.5

Two of the four needed.

a. A high sample temperature
b. A very high sample temperature
c. A GM counter alarm
d. Low air pressure

REFERENCE D.5

PM 5.7.9

-Section D Continued on Next Page-

Points
Available

QUESTION D.6

Once the reactor-ready lamp is on, the regulating rod can be moved to any position of travel. However, shim blade withdrawal motion is limited to 4 inches by the "sub-critical position" interlock circuit. What are the three (3) reasons for the sub-critical position interlock circuit? (1.5)

ANSWER D.6

1. To maintain the shim blade bank programmed at a uniform height during final approach to criticality.

2. To establish a level, below the critical position, to which the shim blades may be individually withdrawn in one step.

3. To provide a convenient reference point at which the operator can pause to make a complete instrument check before bringing the reactor to criticality.

REFERENCE D.6

RSM 4.3

QUESTION D.7

TRUE or FALSE: Channel 9 (automatic control) operates on a gamma-sensitive detector, not a compensated ion chamber. (0.5)

ANSWER D.7

True

REFERENCE D.7

RSM 5.9

-Section D Continued on Next Page-

Points
Available

QUESTION D.8

There are two (2) primary coolant conductivity cells: MC-1 and 2.
Why is MC-1 normally selected? (1.5)

ANSWER D.8

Conductivity cell MC-1, which is positioned in a filter line at the inlet to the ion exchange column, is normally selected. The other cell, MC-2, is positioned in the outlet filter return line. Obviously inlet measures highest and most conservative conductivity, unless the ion exchanger is leaching out.

REFERENCE D.8

RSM 6.1

QUESTION D.9

How are flows in the reflector secondary coolant and shield coolant measured? (1.0)

ANSWER D.9

Orifice plates and d/p cells.

REFERENCE D.9

RSM 6.6

-Section D Continued on Next Page-

Points
Available

QUESTION D.10

Explain how the reading on the linear N-16 monitor would change as reactor power increases. (2.0)

ANSWER D.10

N-16 production is directly proportional to the fast neutron flux, and therefore if the primary flow was constant, the radiation reading on this monitor would directly indicate reactor power.

REFERENCE D.10

RSM 7.3

-End of Section D-

E. SAFETY AND EMERGENCY SYSTEMS (15.0)

Points Available

QUESTION E.1

What are the three (3) major safety requirements associated with operating MITR-II (according to the Standard Operating Plan General Instructions)? (3.0)

ANSWER E.1

The first, and most important, is that the release of radioactive materials to the environment be restricted to the lowest practical amount. The second safety requirement is that on-site personnel be protected from contamination and that exposure to radiation be kept as low as is reasonably achievable. The third requirement is that equipment, especially the reactor itself, be operated and maintained properly and that nothing be done that would jeopardize future reactor operation.

REFERENCE E.1

PM 2.1, p. 1

QUESTION E.2

Why must the reactor be shut down if the compressed air system is lost? (2.0)

ANSWER E.2

If neither compressor is capable of maintaining system pressure, the dump valve will open, the pneumatic instrumentation will be lost, and all airlock gaskets will deflate once the air within them leaks out past system check valves. You'll eventually lose containment integrity.

REFERENCE E.2

PM 5.5.4

-Section E continued on next page-

Points
Available

QUESTION E.3

For each of the three (3) cases below, describe how emergency core cooling would be made available. (3.0)

a. Assumptions: 1. Loss of normal electric power supply from Cambridge Electric Company.

2. All process systems are normal except for the loss of power.

b. Assumptions: 1. Level in the core tank cannot be maintained at the overflow level, but it has been determined that it is not dropping below the reactor inlet penetration (inlet penetration at -52 inches).

c. Assumptions: 1. Level in the core tank cannot be maintained at the level of the reactor inlet penetration.

2. The lost water is being collected in the equipment room sump and/or a source of makeup other than city water is immediately available.

ANSWER E.3

a. The system will be aligned as per normal shutdown cooling except that MM-2 will be supplied power from the facility's emergency power supply and HE-2 will be cooled by city water.

b. The systems will be aligned as per modes 3 and 4, but these modes will not be initiated until required. As long as the conditions assumed for mode 2 prevail, natural circulation up through the core and down through the flow shroud check valves will suffice. Heat will be lost to ambient, the reflector tank, and the off-gas system.

c. MM-2 will be aligned to take a suction on either the equipment room sump through the portable hose and strainer, or the other source of makeup, and discharged directly to the 8 inch reactor inlet line through MV-60 or through the spray nozzles at the top of the core tank.

REFERENCE E.3

RSM 3.4.5

-Section E continued on next page-

Points
Available

QUESTION E.4

What two (2) mechanisms add negative reactivity to shut down the reactor when dump valve DV-4 is opened? (1.5)

ANSWER E.4

- When contents of reflector "dumps" to dump tank, negative reactivity added due to increased leakage (loss of reflector) (+1.0).

- There is a microswitch on the vale which provides a SCRAM when the dump valve is opened (+0.5).

REFERENCE E.4

RSM 3.8

QUESTION E.5

How long would the emergency batteries provide expected instrument and pump power following a loss of both external electrical power feeders? (Select best answer.) (0.5)

a. 40 minutes
b. 4 hours
c. 12 hours
d. 24 hours

ANSWER E.5

b. 4 hours

REFERENCE E.5

RSM 8.31

-Section E continued on next page-

Points
Available

QUESTION E.6

Explain the difference between a major and minor SCRAM. (2.5)

ANSWER E.6

All automatic reactor scrams cause the current to the magnets holding the shim blades to be interrupted. This causes the absorber sections to drop into the core and shut the reactor down. This action is defined as a minor scram. A major scram is initiated by depressing a major scram pushbutton. This action secures the ventilation system, seals the containment shell, dumps the top part of the D_2O reflector, and interrupts the withdraw permit circuit thereby dropping the shim blades.

(+0.5 for minor scram definition)
(+2.0 for major scram four parts, +0.5 each)

REFERENCE E.6

RSM 9.8

QUESTION E.7

There are eight safety and emergency related alarm conditions that will transmit a signal to the Campus Patrol Alarm System. Name five (5). (2.5)

ANSWER E.7

Any five of below

- High Temperature Reactor Outlet, MTS-1
- Low Level Core Tank
- Low Pressure HM-1A
- High Level Radiation Monitor
- Smoke Detector System
- Waste Tanks
- Low Pressure Helium Supply
- Leak Primary and D_2O System

REFERENCE E.7

RSM 9.15

-End of Section E-

F. STANDARD AND EMERGENCY OPERATING PROCEDURES (13.5)

Points Available

QUESTION F.1

Both shim blades and the regulating rod can be driven under automatic control provided the associated reactivity is less than ________% delta k/k. (1.0)

ANSWER F.1

1.8% delta k/k

REFERENCE F.1

Tech. Spec 3.9 (recent change)

QUESTION F.2

What increase in reactor power requires the authorization and witnessing by the duty shift supervisor? (0.75)

ANSWER F.2

>10%

REFERENCE F.2

PM 1.3, p. 2

QUESTION F.3

List five (5) entries made in the Reactor Console Log for criticality data during a startup. (2.5)

ANSWER F.3

i. time
ii. reactor power and period
iii. shim bank and regulating rod positions
iv. core outlet temperature
v. reflector outlet temperature

REFERENCE F.3

PM 1.8, p. 2

-Section F continued on next page-

Points
Available

QUESTION F.4

What maximum pH value in primary system water requires immediate corrective action? (0.75)

ANSWER F.4

7.0

REFERENCE F.4

PM 3.1.1.1, p. 12

QUESTION F.5

What three (3) requirements must be met for the reactor to be in a "SECURED CONDITION"? (3.0)

ANSWER F.5

1. The reactor is shutdown.

2. The console key switch is off with the key removed and in the proper custody.

3. No work is in progress within the main core tank involving fuel or experiments, or maintenance of the core structure, installed control blades, or installed control blade drives when not visibly decoupled from the control blade.

REFERENCE F.5

PM 2.2, p. 3

-Section F continued on next page-

Points
Available

QUESTION F.6

TRUE or FALSE: As defined in the MITR-II startup checklists, the ECP is actually not calculated for the infinite-period critical position, but for a supercritical position with a positive 50-second period. (0.5)

ANSWER F.6

True

REFERENCE F.6

PM 2.3, p. 2

QUESTION F.7

What are the four (4) emergency classifications addressed in your emergency plan (PM 4.4)? (2.0)

ANSWER F.7

1. Unusual Event
2. Alert
3. Site Area Emergency
4. General Emergency

REFERENCE F.7

PM 4.4, p. 1

-Section F continued on next page-

Points
Available

QUESTION F.8

The reactor is critical. You receive an alarm indicating that the primary coolant level has dropped 4.0" below the overflow point. List your required immediate actions. (3.0)

ANSWER F.8

1. Acknowledge the alarm. (+0.25)

2. Scram the reactor (minor) if it has not already scrammed. Verify that reactor power is decreasing. (+0.5)

3. Notify the reactor shift supervisor. (+0.5)

4. Check the core tank level indicators, ML-3A and ML-3B, both to determine the actual coolant level and to decide if it is dropping or remaining constant. (+0.25)

5. Prepare to initiate emergency cooling. Install the quick-connect hoses located in the control room and in the utility room between valves MV-69/MV-70 and city water lines. (+0.25)

6. Refer to Procedure 4.4.4.1 (Safety Limit Exceeded). (+0.5)

7. Notify the Assistant Reactor Superintendent, the Superintendent, and the Director of Operations. If a safety limit was exceeded, notify the Reactor Radiation Protection Officer. (+0.20)

REFERENCE F.8

PM 4.4.4.4, p. 1

-End of Section F-

G. RADIATION CONTROL AND SAFETY (14.0)

Points Available

QUESTION G.1

What action should the Operator-In-Charge take if the rabbit radiation monitor trips? (1.0)

ANSWER G.1

Inform the shift supervisor (before investigation and resolution).

REFERENCE G.1

PM 1.10

QUESTION G.2

What is the basis of the maximum irradiation time limit on the rabbit (60-megawatt hours at a neutron flux of 10^{13})? (1.5)

ANSWER G.2

Embrittlement of the polyethylene containers.

REFERENCE G.2

PM 1.10, p. 10

QUESTION G.3

There must be no direct contact with fingers on the irradiated container or samples because of: (Select best answer.) (1.0)

a. high probable gamma radiation
b. high probable beta radiation
c. high probable surface contamination
d. high probable alpha contamination

ANSWER G.3

b. beta

REFERENCE G.3

PM 1.10, p. 10

-Section G continued on next page-

Points
Available

QUESTION G.4

What two (2) types of dosimetry are all personnel working at the MIT reactor required to wear? (2.0)

ANSWER G.4

1. Beta-Gamma Monitoring Badge
2. Pocket Dosimeter (gamma)

REFERENCE G.4

PM 2.5, p. 1

QUESTION G.5

Why is a spill of heavy water a radiological concern? (1.0)

ANSWER G.5

Tritium content

REFERENCE G.5

PM 4.5, p. 4

QUESTION G.6

If the containment building's ventilation system fails, what is the principal radioactive gas that will buildup in containment? (1.0)

ANSWER G.6

Ar-41

REFERENCE G.6

PM 4.5, p. 5

-Section G continued on next page-

Points
Available

QUESTION G.7

TRUE or FALSE: When washing contaminated skin, it is important to use hot water to open and clean out potentially contaminated pores. (0.5)

ANSWER G.7

False

REFERENCE G.7

PM 4.4.4.10, p. 4

QUESTION G.8

Operation of the Blanket Test Facility (BTF) will cause certain radiation monitor detectors to read higher than normal. Which of the radiation monitors are most affected by use of the BTF? (1.0)

ANSWER G.8

Secondary Water Monitors

REFERENCE G.8

PM 5.6.2, p. 1

QUESTION G.9

Explain the difference in extent of qualification for blue, red, and yellow film badges. Which badged group(s) are permitted to escort members of the general public through the Reactor Building? (1.5)

ANSWER G.9

Blue - beginning experimental work, must be supervised
Red - allowed to operate experiment by themselves
Yellow - sufficiently knowledgeable to escort public

REFERENCE G.9

PM 1.12, p. 1

-Section G continued on next page-

Points
Available

QUESTION G.10

List three (3) independent measurements or indicators used to monitor or detect heavy water leakage into the secondary coolant. (2.0)

ANSWER G.10

1. The secondary water monitor is a gamma-sensitive scintillation detector. It cannot detect tritium but is sensitive to N^{16} and F-17, also present in the heavy water when the reactor is operating.

2. Daily sampling of the secondary water will allow detection of very small leaks.

3. Because of the nature of the reflector system, any loss of D_2O inventory will be reflected by a decrease in the D_2O level in the dump tank.

REFERENCE G.10

Tech Specs, p. 3-30

QUESTION G.11

Why is the Thermal Column Hohlraum maintained under a carbon dioxide purge? (1.0)

ANSWER G.11

To prevent activation of argon that would result if air entered the facility.

REFERENCE G.11

RSM 2.2

-Section G continued on next page-

Points
Available

QUESTION G.12

TRUE or FALSE: The purpose of the shield coolant system is to remove the heat deposited in the lead thermal shields by neutron radiation. (0.5)

ANSWER G.12

False (gamma)

REFERENCE G.12

RSM 3.13

-End of Section G-

MASTER

U. S. NUCLEAR REGULATORY COMMISSION
SENIOR REACTOR OPERATOR LICENSE EXAMINATION

FACILITY: MASS. INSTITUTE OF TECH.

REACTOR TYPE: TEST

DATE ADMINISTERED: 85/10/02

EXAMINER: SILK, D.

APPLICANT: Answer Key

INSTRUCTIONS TO APPLICANT:

Use separate paper for the answers. Write answers on one side only. Staple question sheet on top of the answer sheets. Points for each question are indicated in parentheses after the question. The passing grade requires at least 70% in each category and a final grade of at least 80%. Examination papers will be picked up six (6) hours after the examination starts.

CATEGORY VALUE	% OF TOTAL	APPLICANT'S SCORE	% OF CATEGORY VALUE	CATEGORY
20.00	20.00			H. REACTOR THEORY
20.00	20.00			I. RADIOACTIVE MATERIALS HANDLING DISPOSAL AND HAZARDS
20.00	20.00			J. SPECIFIC OPERATING CHARACTERISTICS
20.00	20.00			K. FUEL HANDLING AND CORE PARAMETERS
20.00	20.00			L. ADMINISTRATIVE PROCEDURES, CONDITIONS AND LIMITATIONS
100.00	100.00			TOTALS

FINAL GRADE _______________%

All work done on this examination is my own. I have neither given nor received aid.

____
APPLICANT'S SIGNATURE

QUESTION H.01 (3.00)

How much reactivity has been added to a subcritical reactor if the count rate has increased from 100 cps to 150 cps and if the initial value of Keff was .95? Show all calculations and assumptions.

QUESTION H.02 (3.00)

If heavy water were mixed with light water cooling the core:

a. Would the neutron lifetime increase, decrease, or remain the same? (0.7)

b. Would the migration length increase, decrease, or remain the same? (0.7)

c. What is the overall reactivity effect? Explain. (1.6)

QUESTION H.03 (3.00)

Explain the different modes of heat transfer by which the heat of fission is removed from the fuel. Include major components involved in the heat removal process starting with the fuel and ending at the ultimate heat sink. (3.0)

QUESTION H.04 (1.00)

Why are delayed neutrons important?

QUESTION H.05 (3.00)

Explain the effect of the temperature coefficient on reactivity if the thermal power of the MITR II core increases. Include both light and heavy water effects.

(***** CATEGORY H CONTINUED ON NEXT PAGE *****)

QUESTION H.06 (3.00)

The reactor operator is conducting a routine reactor startup after it has been shutdown for several days. Prior to withdrawing a shim blade he reads a stable count of 50 cps on the startup channel. Immediately after withdrawing this blade he reads a count of 80 cps.

a. If he performed no blade motion for five minutes, would the count rate increase, decrease or remain the same? Explain, assuming the reactor is subcritical at 80 cps.

b. After 5 minutes he withdraws another blade the same distance but the reactor is still subcritical. Would the change in count rate (time and magnitude) be different than he saw in part (a) above? Explain.

c. What indications would the operator observe to determine when the reactor had gone critical?

QUESTION H.07 (4.00)

Xenon and Samarium are two poisons which have a significant effect on reactor operations. Discuss and compare these two poisons for the following:

a. Sources of the poisons in the core (1.0)

b. Means of removal from the core (1.0)

c. Effect on reactor operations after shutdown (2.0)

(***** END OF CATEGORY H *****)

QUESTION I.01 (4.00)

A 23 year old individual has accumulated a lifetime occupational dose of 24 rem of whole body exposure documented in accordance with 10CFR20 and has received no exposure during the present calender quarter.

a. How long may he work in a 3 mrem/hr area if he works an 8 hour day Monday through Friday? Show your work.

b. An individual in a restricted area may be allowed to receive a whole body dose in excess of the quarterly limit under certain conditions. Name three conditions.

QUESTION I.02 (2.00)

A mixed gamma and beta source in liquid form spills on the floor. Readings at 10 feet indicate 1.0 mrem/hr on a beta-gamma survey meter. If beta's are not detected further than six feet from the spill and if the combined beta-gamma dose rate at one foot is 120 mrem/hr, what is the beta to gamma ratio? Show your calculations.

QUESTION I.03 (3.00)

a. Does the biological effect of a 100 REM dose depend on whether it is a neutron or gamma dose? Explain.

b. Does the biological effect resulting from bodily intake of a given quantity (in terms of microcuries) of a radioactive material depend on which particular isotope is involved? Explain.

(***** CATEGORY I CONTINUED ON NEXT PAGE *****)

QUESTION I.04 (3.00)

A fuel element is suspended in the Reactor Pool approximately 1 meter under water. A radiation survey meter held at the surface of the water reads 100 mrem/hr.

a. Ignoring buildup, what radiation level would you expect if the fuel element broke the water? Assume an attenuation coefficient of 0.035 cm^-1. (1.0)

b. If the radioactive isotopes in the fuel element had an average half life of 30 minutes, how long would it take for the radiation level at the surface of a one inch lead shield cask to drop to 20 mrem/hr? Assume an initial contact dose of 2 R/hr for the fuel element and a tenth thickness of two inches for lead. (2.0)

QUESTION I.05 (3.00)

To assure that experiments in the reactor do not affect the safety of the reactor, Technical Specifications demand that all experiments within the reactor shall confirm to a set a conditions. List six of the seven conditions set forth in the Technical Specifications.

QUESTION I.06 (3.00)

For the case of a radiological emergency, list seven immediate actions that the on-shift supervisor must ensure have been completed. (Assume no medical assistance and no radiation surveys by Campus Police are required).

QUESTION I.07 (2.00)

Does the number of disintegrations per minute (dpm) from a radioactive source equal the counts per minute (cpm) obtained from a survey instrument? Briefly explain.

(***** END OF CATEGORY I *****)

QUESTION J.01 (3.00)

What three actions must be taken when 1 microcurie/liter of tritium is present in the secondary coolant water?

QUESTION J.02 (3.00)

a. If the Reactor Floor Ar-41 Monitor gives an 'High Level Radiation Monitor' alarm, where are five likely places for the Ar-41 to originate? (2.0)

b. What is done to prevent the production of Ar-41? (1.0)

QUESTION J.03 (2.00)

Briefly describe the natural convection valves, how they work, and what is their function?

QUESTION J.04 (3.00)

What does the 'subcritical position' interlock circuit do and give three reasons why it is incorporated into the shim blade control circuit.

QUESTION J.05 (3.00)

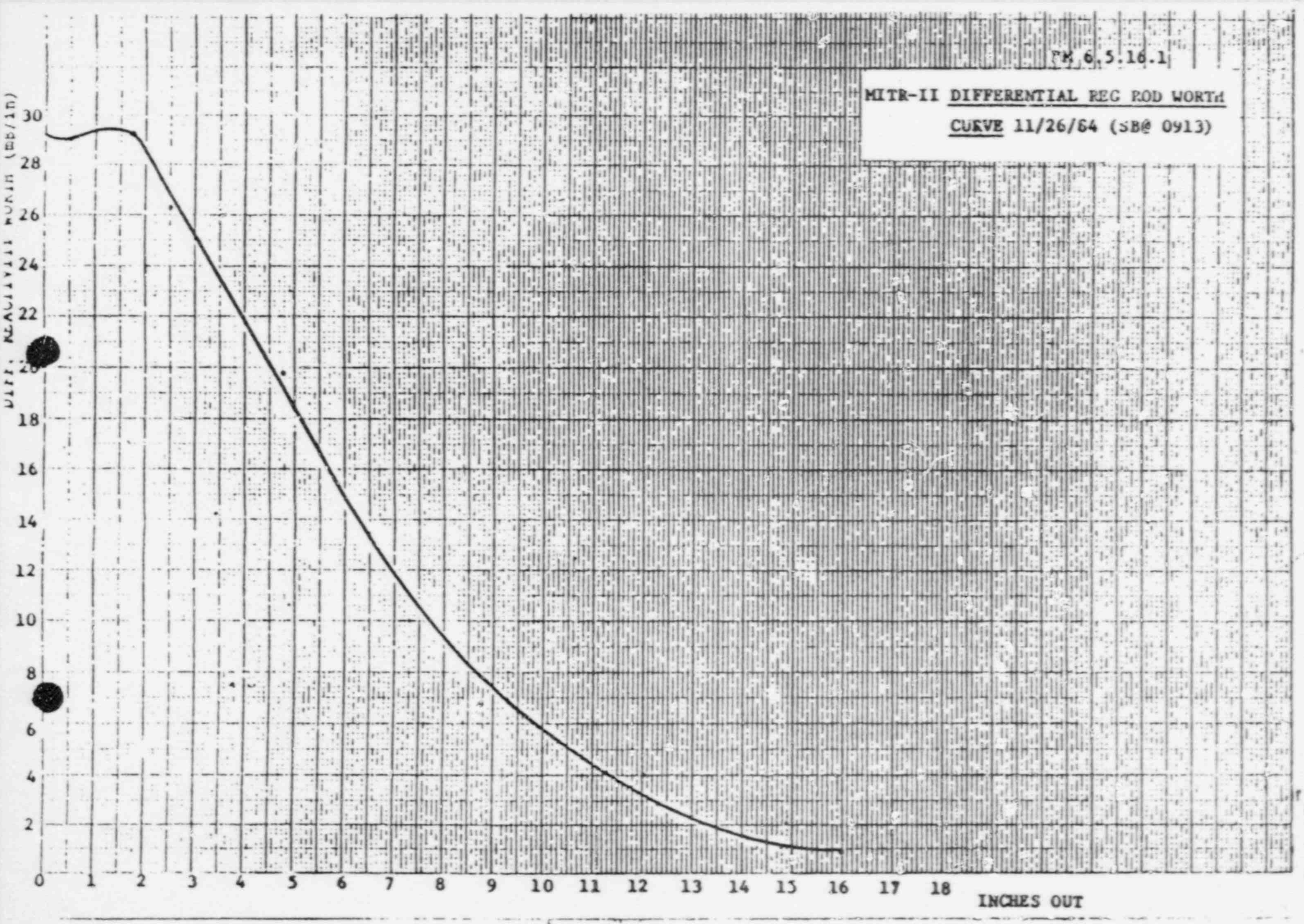
Figure 1 shows the differential regulating rod worth curve for your reactor. Give two reasons why the curve peaks at the location shown.

QUESTION J.06 (3.00)

a. Briefly explain why the reactivity worth of the D2O Reflector Dump is dependent on the position of the shim blade bank.

b. What is the required position of the shim bank when the radial heavy water reflector is pumped into place? Briefly explain why.

(***** CATEGORY J CONTINUED ON NEXT PAGE *****)


PM 6.5.16.1
MITR-II DIFFERENTIAL REG ROD WORTH
CURVE 11/26/64 (SB@ 0913)
DIFF. REACTIVITY WORTH (βB/in)
0
2
4
6
8
10
12
14
16
18
20
22
24
26
28
30
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
INCHES OUT

QUESTION J.07 (3.00)

Briefly explain the most reliable method of determining the steady state power at full power and when this method can be used..

(***** END OF CATEGORY J *****)

QUESTION K.01 (3.00)

After each refueling or change in core loading, the reactor shall not be operated above a power level of 1.0 KW unless an evaluation is made to ensure that two Technical Specifications are satisfied.

a. What are the two Technical Specifications? (2.0)

b. What persons shall complete and approve these evaluations? (1.0)

QUESTION K.02 (3.00)

Give the basis for the following specifications:

a. The reactivity worth of the regulating rod connected to the automatic control system is less than 0.7% delta k/k.

b. The maximum controlled reactivity addition rate is no more than 5x10^-4 delta k/k /sec.

c. The reactivity worth of the D2O reflector dump is greater than the reactivity worth of the most reactive shim blade.

QUESTION K.03 (4.00)

During refueling, what are two designed safety features associated with the hold-down grid plate and what do they prevent?

QUESTION K.04 (3.00)

a. Under what condition, during refueling, is the heavy water reflector not dumped? (2.0)

b. What Technical Specification requirement must be checked if the heavy water reflector is not dumped? (1.0)

QUESTION K.05 (2.00)

What two Technical Specifications requirements must be met before approval is given to remove the spent fuel from the reactor vessel to the transfer flask?

(***** CATEGORY K CONTINUED ON NEXT PAGE *****)

QUESTION K.06 (3.00)

According to your Technical Specifications what safety channels must be operable to move fuel in the core and what are the set points, if any?

QUESTION K.07 (2.00)

According to your Technical Specifications, when is your reactor considered secured?

(***** END OF CATEGORY K *****)

QUESTION L.01 (2.50)

In accordance with your Administration Procedures:

a. Briefly describe the administrative procedures followed if a safety function required by Technical Specifications as a Limiting Condition for Operation is to be temporarily bypassed (assume it is not a part of an approved procedure). Include in your answer who may authorize the bypass, condition of the reactor and recording requirements. (1.5)

b. What additional requirements are necessary if a jumper is used? (1.0)

QUESTION L.02 (4.50)

Indicate whether or not each of the following is a violation of procedures and/or Technical Specifications. Briefly explain why it is or it is not a violation.

a. Operating with five shim blades, the sixth shim blade is fully inserted

b. Operating at 2 MW with one primary pump and 1000 gpm primary coolant flow rate

c. Operating at 150 KW with the emergency cooling system inoperable

d. Operating at 100 KW without emergency power available

e. Operating at full power with one of the three reactor floor area radiation monitors inoperative

f. Increasing the reactor power from 200 KW to 300 KW with the duty shift supervisor in the Utilities Room.

(0.75 each)

QUESTION L.03 (3.00)

Any change to a component or system which involves an 'unreviewed safety question' is a 'Class A' proposal. A proposal change 'shall be deemed to involve an unreviewed safety question' if what three criteria are met?

(***** CATEGORY L CONTINUED ON NEXT PAGE *****)

QUESTION L.04 (1.50)

List five of the services that the Reactor Radiation Protection Office is responsible for providing for radiation protection and compliance with governmental regulations.

QUESTION L.05 (1.00)

Under what conditions may some one be authorized to incur radiation exposures in excess of the 10 CFR 20 limits?

QUESTION L.06 (3.00)

In regards to General Safety Rules, once permission is granted, what are three joint responsibilities of the operator-in-charge and the personnel entering either the reactor top, the medical therapy room, or the equipment room when the reactor is operating?

QUESTION L.07 (2.00)

a. What are four variables associated with the core thermal and hydraulic performance?

b. What is the objective of the Safety Limits?

QUESTION L.08 (2.50)

a. Given the events below, state which emergency classification should be declared. (0.5 pts each)

1. A large crowd of protesters marching around the reactor building.
2. A fire damaging an experiment which causes the release of radioactive materials.
3. A tornado damaging the containment building.
4. A slow and uncontrollable decrease in core tank level such that level remains above the anti-syphon valves.

b. What criteria is used for classifying emergency conditions? (0.5)

(***** END OF CATEGORY L *****)
(************* END OF EXAMINATION ***************)

Table 4.5.3-1: EALs for Notification of Unusual Events

1. Confirmed abnormal radiation levels leading to actual or projected radiological effluents at the site boundary exceeding 10 MPC for unrestricted areas when averaged over 24 hours. This level corresponds to an exposure of 15 mrem whole body accumulated over 24 hours. (PM 4.4.4.15b)
2. Report or observation that severe natural phenomena are either imminent or existing. These include storms with tornado or hurricane force winds that could strike the facility, earthquakes that could adversely affect the reactor's safety systems, and floods that could adversely affect the reactor's safety systems. (PM 4.4.4.2)
3. Threats to or breaches of security. (PM 4.4.4.5/4.4.4.6)
4. A reactor safety limit's being exceeded such that a fuel damage accident that could release radionuclides to the containment building is possible. (PM 4.4.4.1)
5. A fire within the containment building that lasts beyond the incipient stage or for more than ten minutes. (PM 4.4.4.3)
6. Receipt of a bomb threat. (PM 4.4.4.7)

Table 4.5.3-2: EALs for an Alert

1. Confirmed abnormal radiation levels leading to actual or projected radiological effluents at the site boundary exceeding 50 MPC for unrestricted areas when averaged over 24 hours. This level corresponds to an exposure of 75 mrem whole body accumulated over 24 hours. (PM 4.4.4.15b)
2. Same as #1 except the effluents could cause an integrated exposure of 100 mrem thyroid. (PM 4.4.4.15b)
3. Radiation levels at the site boundary of 20 mrem/hour sustained for one hour. (PM 4.4.4.14b/4.4.4.11)
4. Abnormal loss of primary coolant such that the core tank level remains at or above he anti-syphon valves. (PM 4.4.4.4)
5. Loss of radioactive material control that causes radiation dose rates or air-borne radionuclides to increase above permissible exposure levels by a factor of 1000 throughout the containment building. (PM 4.4.4.12)
6. Radiation dose rates throughout the containment building in excess of 100 mrem/hour sustained for one hour. These levels would necessitate evacuation of all personnel. (PM 4.4.4.12)
7. A fire leading to loss of radioactive material control within the containment building. (PM 4.4.4.3)
8. An imminent or existing hazard such as:
   (a) Missile(s) impacting on the containment building.
   (b) An explosion that affects facility operation.
   (c) An uncontrolled release of toxic or flammable gases into the containment building. (PM 4.4.4.9)

Table 4.5.3-3: EALs for a Site Area Emergency

1. Confirmed abnormal radiation levels leading to actual or projected radiological effluents at the site boundary exceeding 250 MPC for unrestricted areas when averaged over 24 hours. This level corresponds to an exposure of 375 mrem whole body accumulated over 24 hours. (PM 4.4.4.15b)
2. Same as #1 except the effluents could cause an integrated exposure of 500 mrem thyroid. (PM 4.4.4.15b)
3. Radiation levels at the site boundary of 100 mrem/hour sustained for one hour. (PM 4.4.4.14b/4.4.4.11)
4. Abnormal loss of primary coolant such that the core tank level drops below the anti-syphon valves. (Note: This accident is not considered credible, but procedures exist for coping with it.) (PM 4.4.4.4)
5. Imminent loss of physical control of the reactor. (PM 4.4.4.6)
6. Severe natural events being experienced. These include:
   - (a) An earthquake that is causing observable damage to the reactor safety equipment within the containment building.
   - (b) A flood that is affecting the operability of any reactor safety system.
   - (c) Tornado or hurricane force winds that are damaging the containment building. (PM 4.4.4.2)

Table 4.5.3-4: EALs for a General Emergency

1. Actual or projected doses at the site boundary in the exposure pathway of 1 rem whole body or 5 rem thyroid for unrestricted areas when averaged over one hour. Note: Figure 4.7.2.2-1 lists the conditions and instrument readings corresponding to a projected off-site dose of 1 rem/hour. (PM 4.4.4.15a)
2. Sustained actual or projected radiation levels at the site boundary of 500 mrem/hour whole body. (PM 4.4.4.14a/4.4.4.11/4.4.4.12)
3. Blockage of fuel element channels thereby causing a loss of coolant to the affected channels and a fuel melt. This is the design basis accident. (PM 4.4.4.15a)
4. Loss of physical control of either the containment building which includes the control room or of auxiliary areas that house vital equipment. (PM 4.4.4.5/4.4.4.6).
5. Events that have caused or will cause massive facility and/or reactor system damage that could lead to the melting of fuel. (PM 4.4.4.15a)

$f = ma$ | $v = s/t$ | Cycle efficiency = (Network out)/(Energy in)

$w = mg$ | $s = V_o t + 1/2\, at^2$

$E = mc^2$

$KE = 1/2\, mv^2$ | $a = (V_f - V_o)/t$ | $A = \lambda N$ | $A = A_o e^{-\lambda t}$

$PE = mgh$

$V_f = V_o + at$ | $w = \theta/t$ | $\lambda = \ln 2/t_{1/2} = 0.693/t_{1/2}$

$W = v\,\Delta P$

$t_{1/2}eff = \frac{[(t_{1/2})(t_b)]}{[(t_{1/2}) + (t_b)]}$

$\Delta E = 931\, \Delta m$

$I = I_o e^{-\Sigma x}$

$\dot{Q} = \dot{m} Cp \Delta t$

$\dot{Q} = UA \Delta t$

$I = I_o e^{-\mu x}$

$Pwr = W_f \Delta h$

$I = I_o\, 10^{-x/TVL}$

$TVL = 1.3/\mu$

$P = P_o 10^{sur(t)}$

$HVL = -0.693/\mu$

$P = P_o e^{t/T}$

$SUR = 26.06/T$

$SCR = S/(1 - K_{eff})$

$CR_x = S/(1 - K_{effx})$

$SUR = 26\rho/\ell^* + (\beta - \rho)T$

$CR_1(1 - K_{eff1}) = CR_2(1 - K_{eff2})$

$T = (\ell^*/\rho) + [(\beta - \rho)/\bar{\lambda}\rho]$

$M = 1/(1 - K_{eff}) = CR_1/CR_o$

$T = \ell/(\rho - \beta)$

$M = (1 - K_{effo})/(1 - K_{eff1})$

$T = (\beta - \rho)/(\lambda\rho)$

$SDM = (1 - K_{eff})/K_{eff}$

$\rho = (K_{eff} - 1)/K_{eff} = \Delta K_{eff}/K_{eff}$

$\ell^* = 10^{-5}$ seconds

$\bar{\lambda} = 0.1$ seconds^{-1}

$\rho = [(\ell^*/(T K_{eff})] + [\bar{\beta}_{eff}/(1 + \bar{\lambda} T)]$

$I_1 d_1 = I_2 d_2$

$I_1 d_1{}^2 = I_2 d_2{}^2$

$P = (\Sigma \phi V)/(3 \times 10^{10})$

$R/hr = (0.5\, CE)/d^2$ (meters)

$\Sigma = \sigma N$

$R/hr = 6\, CE/d^2$ (feet)

Water Parameters

1 gal. = 8.345 lbm.
1 gal. = 3.78 liters
1 ft^3 = 7.48 gal.
Density = 62.4 lbm/ft^3
Density = 1 gm/cm^3
Heat of vaporization = 970 Btu/lbm
Heat of fusion = 144 Btu/lbm
1 Atm = 14.7 psi = 29.9 in. Hg.
1 ft. H_2O = 0.4335 $lbf/in.^2$

Miscellaneous Conversions

1 curie = 3.7×10^{10} dps
1 kg = 2.21 lbm
1 hp = 2.54×10^3 Btu/hr
1 mw = 3.41×10^6 Btu/hr
1 in = 2.54 cm
°F = 9/5°C + 32
°C = 5/9 (°F-32)
1 BTU = 778 ft-lbf

ANSWERS -- MASS. INSTITUTE OF TECH. -85/10/02-SILK, D.

ANSWER H.01 (3.00)

```
cr1 /cr2   = (1-Keff2) / (1-Keff1)              [0.9]
100/150    = (1-Keff2) / (1-0.95)               [0.5]
1-Keff2    = 10/15 x 0.05
Keff2      = 0.967                              [0.1]

Change in reactivity = [1-Keff2/Keff2] - [1-Keff1/Keff1]
                     = Keff2 - Keff1 / Keff1 x Keff2        [0.9]
                     = 0.967 - 0.95 / 0.95 x 0.967          [0.5]
                     = 1.85 % delta K/K                      [0.1]
```

REFERENCE
Procedure Manual (PM) 2.3 pg. 1,2

ANSWER H.02 (3.00)

a. Increase (0.7)

b. Increase (0.7)

c. The increased migration length would tend to increase neutron lifetime and leakage and thus add negative reactivity. (1.6)

REFERENCE
Reactor Systems Manual (RSM) pg. 10.10

ANSWER H.03 (3.00)

Conduction through fuel.
Conduction transfer from fuel to coolant.
Forced convection to heat exchanger.
Conduction across heat exchanger.
Forced convection to cooling towers.
Evaporation to atmosphere. (0.5 pts each)

REFERENCE
Introduction to Nuclear Engineering, chapter 8; J R Lamarsh
RSM pgs. 3.1, 3.7, 3.10 to 3.12

ANSWER H.04 (1.00)

The delayed neutrons increase generation time which increases the period and thus the reactor can be controlled.

REFERENCE

Introduction to Nuclear Engineering, chapter 7 pg. 245; J R Lamarsh

ANSWER H.05 (3.00)

Increasing the temperature of the light water will insert nagative reactivity by causing the neutrons to take longer to thermalize so there are fewer fissions (1.5). Heating of the heavy water reflector will add negative reactivity by allowing neutron leakage to increase (1.5).

REFERENCE

RSM pg. 10.8

ANSWER H.06 (3.00)

a. Increase slightly then level out(0.6) due to subcritical multiplication (0.4).

b. Larger increase(0.3) and longer to level out(0.3) due to greater number of generations to reach equilibrium(0.4).

c. Steadily increasing count rate or slight positive period with no rod withdrawal. (1.0)

REFERENCE

PM 2.3 pg. 1,2

ANSWERS -- MASS. INSTITUTE OF TECH. -85/10/02-SILK, D.

ANSWER H.07 (4.00)

a. Both are produced directly from fission and from their respective decay chain. Te-135 decays to I-135 which decays to Xe-135. Nd-149 decays to Pm-149 which decays to Sm-149. (1.0)

b. Both can be removed from the core by neutron absorption. Xe-135 can also be removed by radioactive decay, whereas Sm-149 is stable. (1.0)

c. When the reactor is shutdown, both poisons increase in concentration due to production from their decay chains and because neither are being removed by neutron absorption. Sm-149's increase is relatively small and reaches a maximum and remains there until the reactor is operated again. Xe-135 will increase to a peak and then decrease slowly as more Xe-135 is decaying than is being produced by the decay of I-135. (2.0)

REFERENCE
RSM pg. 10.6 to 10.8

ANSWER   I.01   (4.00)

a. 5(N-18) = 5(23-18) = 25
   25 - 24 = 1.0 Rem = Max. Dose   (1.0)
   Max. Dose = Dose Rate X Time
   1.00 Rem = 0.003 Rem/hr X 8 hr/day X No. of Days
   No. of Days= 41.6 days   (1.0)

b. Provided that (1) He does not exceed 3 rem per quarter   (.66)

(2) His radiation history is known and recorded on the proper form (NRC Form 4)   (.67)

(3) The dose received when added to his radiation history does not exceed 5(N-18) rems where N = the person's age at his last birthday   (.67)

REFERENCE
10 CFR 20.101

ANSWER   I.02   (2.00)

d x (r)^2 = D x (R)^2
1 mr/hr x (10)^2 = D x (1)^2
D =100 mr/hr   [1.0]

Beta dose = 120mr/hr -100 mr/hr
          = 20 mr/hr

Beta to gamma ratio = 20/100 = 1/5   [1.0]

REFERENCE
Introduction to Nuclear Engineering, chapter 9 pg 409,410; J R Lamarsh

ANSWER I.03 (3.00)

a. No [0.5] A Rem dose accounts for the type and energy of radiation. [1.0]

b. Yes [0.5] Internal dose depends on biological and physical T 1/2, referred organ, type of radiation. [1.0]

REFERENCE
Introduction to Nuclear Engineering, chapter 9; J R Lamarsh

ANSWER I.04 (3.00)

a. I = Io e<-ux>
   100 mrem/hr = Io e<-0.035 cm(-1) 100 cm>
   Io = 3311 mrem

b. I = Io 10 <-x/TVL> (TVL ...
   Io = I 10 <x/TVL> = 63.25 mrem/hr

   Io = Ii e<-(.693/half life)t> (wher... ...e)
   t = -(half life/.693) ln(Io/Ii)
   t = -(30 min/.693) ln(63.25/2000) = 149.5 minutes ...]

REFERENCE
Introduction to Nuclear Engineering, pgs 22, 83; J R L...

ANSWER I.05 (3.00)

Reactivity Effects
Thermal-Hydraulic Effects
Chemical Effects
Radiolytic Decomposition
Experiment Scram
Prototype Testing
Radioactive Release (@ 0.5 pts, any ...ix ... 3.0)

REFERENCE
Technical Specifications (T.S.) 6.1, pg. 6-1 to 6-7

ANSWER I.06 (3.00)

a. The reactor is shut down
b. The containment building is isolated
c. Experimenters are evacuated
d. Off-duty licensed and radiation protection personnel are notified
e. The MIT Campus Police are requested to stand-by
f. Radiation levels are monitored on-site and tracked off-site using the MITR Radiation Protection Office's remote monitors
g. Off-duty personnel are briefed as they arrive

REFERENCE
Procedure Manual (PM) 4.3 pg 3

ANSWER I.07 (2.00)

No. The cpm must be corrected for efficiency of the detector and the geometry of the source in relation to the detector.

REFERENCE
RSM pgs 5.2, 7.1

ANSWERS -- MASS. INSTITUTE OF TECH. -85/10/02-SILK, D.

ANSWER J.01 (3.00)

The cooling tower spray shall be shut down

The secondary system water discharge shall be stopped

The D2O reflector heat exchanger shall be isolated

REFERENCE
T.S. 3.8, pg. 3-26

ANSWER J.02 (3.00)

a. High flux regions such as the thermal column, pipe tunnel, lid space, experimental port and instrument lead boxes. (2.0)

b. The high flux regions are sealed and/or flooded with carbon dioxide in order to exclude as much air as possible since Ar-40 is present in air. (1.0)

REFERENCE
RSM pg. 7.5

ANSWER J.03 (2.00)

Natural convection valves are ball type pressure-operated check valves located on the wall between the inlet and outlet of the core that are designed to open on a loss of primary pump pressure to allow natural convective flow around the core.

REFERENCE
SAR pg. 6.5 and 15.12

ANSWER J.04 (3.00)

It limits shim blade withdrawal motion to four inches.

1. It maintains shim blade bank programmed at a uniform height during final approach to criticality.
2. It establishes a level, below the critical position, to which the shim blades may be individually withdrawn in one step.
3. It provides a convenient reference point at which the operator can pause to make a complete instrument check before bringing the reactor to criticality

(.75 pts each)

REFERENCE
RSM pg. 4.3

ANSWER J.05 (3.00)

The peak in the differential regulating rod worth occurs at low rod height because the full in position for the regulating rod is six inches above the bottom of the fuel elements and once the regulating rod is withdrawn any appreciable amount, it is heavily shadowed by the adjacent shim blades.

REFERENCE
RSM pg. 10.6

ANSWER J.06 (3.00)

a. The blade bank exerts a shadowing influence on the reflector
b. Full in - this insures that the reactivity insertion for this process will not occur when the reactor is or could go critical

REFERENCE
RSM pg. 10.6

ANSWER J.07 (3.00)

A heat balance calculated from the primary, reflector, and shield system flows and temperature rises once these systems are in thermal equilibrium.

REFERENCE
PM 2.4, pg. 2

ANSWER K.01 (3.00)

a. The ratio $F_{HC} F_p / d_f F_f$ is predicted to be less than 2.9

The core is predicted to operate below incipient boiling at every point in the core.

b. Two Senior Reactor Operators.

REFERENCE

T.S. 3.1, pg. 3-1

ANSWER K.02 (3.00)

a. The total worth of the rod is to be limited such that the complete withdrawal of the rod will not make the reactor prompt critical

b. This value is conservatively within the range of reactivity insertion rates normally accepted for reactor operation. Control systems in this range give ample margin for proper human response during approach to critical and power operations.

c. The additional independent capability for reactivity control provided by the D2O reflector dump gives added assurance that the reactor can be made subcritical under an adverse condition of fuel loading or control blade malfunction.

REFERENCE

T.S. 3.9, pg. 3-32 to 3-35

ANSWER K.03 (4.00)

1. The grid is designed so that there is normally access to only one core position at a time (1.0). This limits the amount of water that can be in the core at any one time by making it difficult, though not impossible, for more than one core position to be defueled at time.(1.0)

2. The grid's latch is interlocked with the primary coolant pumps so that if the latch is released, the coolant pumps stop and remain off until the grid is latched again (1.0). This protects the fuel elements from damage and the reactor as a whole from inadvertent criticality(1.0)

REFERENCE
PM 2.7, pg. 3

ANSWER K.04 (3.00)

a. If dumping would cause the nuclear instrumentation startup channels to indicate less than 10 counts per minute. (2.0)

b. The shutdown margin would have to be checked. (1.0)

REFERENCE
PM 2.7, pg. 3

ANSWER K.05 (2.00)

1. The element to be moved cannot be moved unless it has not been operated in the core at a power level above 100 KW for at least four days.

2. The K-effective of any storage area outside of the reactor core shall be less than 0.90

REFERENCE
T.S. 3.10.4, pg. 3-37

ANSWER K.06 (3.00)

Safety channels operable	Set points
Period (2 channels)	> 3 sec
Neutron Flux Level (2 channels)	100 KW
D2O Dump Valve Selector Switch (1)	-
Manual major scram (2)	-

(0.5 each response)

REFERENCE
T.S. 3.7.2, pg. 3-21,22

ANSWER K.07 (2.00)

a. The reactor is shutdown (.66)

b. Console key switch off and key is in proper custody (.67)

c. No work in progress within the main core tank involving fuel or experiments, or maintenance of the core structure, installed control blades or installed control blade drives when not visibly decoupled from the control blade (.67)

REFERENCE
T.S. 1.1, pg 1-1

ANSWERS -- MASS. INSTITUTE OF TECH. -85/10/02-SILK, D.

ANSWER L.01 (2.50)

a. i. The reactor must be shutdown and the bypass must be removed before reactor startup (0.5)

ii. Must be approved by Duty-Shift-Supervisor or Reactor Superintendent (0.5)

iii. The bypass authorizor's initials must be recorded on the bypass log sheet (0.5)

b. If Jumpers are used, the jumper must be tagged; a warning tag placed on the shim blade control handle stating that the reactor is not to be started up until the bypass is removed. (1.0)

REFERENCE
PM 1.9, pg. 1

ANSWER L.02 (4.50)

a. Violation of Technical Specifications (T.S.)(.25), sixth shim blade must be at the operating position or higher (except if < 1 KW for blade calibration) (0.5)

b. No violation (.25), with one pump 3.0 MW allowed and minimum of 900 gpm (0.5)

c. Violation of T.S. (.25), power levels in excess of 100 KW require the emergency cooling system to be operable (0.5)

d. Violation of T.S. (.25), emergency power must be available whenever the reactor is operating (0.5)

e. No violation (.25), T.S. requires at least one area radition monitor on the reactor floor to be operating (0.5)

f. Violation of procedure (.25), the duty shift supervisor must authorize and witness both startups and increases in reactor power of greater than 10% (0.5)

REFERENCE
a. T.S. pg. 3-32
b. T.S. pg. 2-5
c. T.S. pg. 3-19
d. T.S. pg. 3-21

e. T.S. pg. 3-27
f. PM 1.3, pg. 2

ANSWER L.03 (3.00)

1. If the probability of occurence of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased.

2. If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created.

3. If the margin of safety as defined in the basis for any technical specification is reduced.

REFERENCE
PM 1.4, pg.2

ANSWER L.04 (1.50)

a. Registration and instruction of radiation workers
b. Personnel monitoring of radiation exposure
c. Radioisotope laboratory inspections, radiation surveys, and area monitoring
d. Radioactive waste collection
e. Calibration and repair of radiation protection instruments
f. Calibration of reactor radiation detection instruments
g. Environmental monitoring
h. Leak-testing of sealed radioactive sources
i. Advice in radiation emergencies, and special decontamination operations
j. Maintenance of radiation protection records (any five, .30 pts each)

REFERENCE
PM 1.11, pg. 1

ANSWER L.05 (1.00)

To save a human life (0.5) or to insure nuclear safety (0.5)

REFERENCE
PM 4.3, pg. 14

ANSWER L.06 (3.00)

1. To determine that normal radiation levels exist based on control room and/or local instrumentation.
2. To assess the need for a radiation survey with a portable detector.
3. To evaluate the potential for dose rate changes during occupancy.

REFERENCE
PM 1.14, pg. 6

ANSWER L.07 (2.00)

a. Total reactor thermal power

Reactor coolant total flow rate

Reactor coolant outlet temperature

Height of water above the outlet end of the heated section of the hottest fuel channel (.25 pts each)

b. To establish limits within which the integrity of the fuel clad is maintained (1.0)

REFERENCE
T.S. 2.1, pg. 2-1

ANSWER L.08 (2.50)

a. 1. Notification of Unusual Event

2. Alert

3. Site Area Emergency

4. Alert (0.5 pts each)

b. Potential radiological consequences (0.5)

REFERENCE

a. PM 4.5, pgs. 10 to 12

b. PM 4.4, pg. 2

9

NOV 27 1985

Docket No. 50-20

Massachusetts Institute of Technology
Research Reactor
ATTN: Mr. Lincoln Clark, Jr.
Director of Reactor Operations
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

Subject: Inspection No. 50-20/85-02

This refers to the routine, physical protection inspection conducted by Mr. William Madden of this office on September 25, 1985, at the Massachusetts Institute of Technology reactor facility of activities authorized by NRC License No. R-37 and to the discussions of our findings held by Mr. Madden with you at the conclusion of the inspection.

Areas examined during this inspection are described in the NRC Region I Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Within the scope of this inspection, no violations were observed.

No reply to this letter is required. Your cooperation with us in this matter is appreciated.

Sincerely,

Original Signed By:
James H. Joyner

Thomas T. Martin, Director
Division of Radiation Safety
and Safeguards

Enclosure: NRC Region I Inspection Report No. 50-20/85-02

cc:
Public Document Room (PDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Massachusetts (2)
Dr. O. K. Harling, Director of the
Reactor Laboratory

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11/08/85

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bcc:
Region I Docket Room (w/concurrences)
Management Assistant, DRMA (w/o enclosure)

RI:DRSS	RI:DRSS	RI:DRSS	RI:DRSS
Madden	Keimig	Joyner	Martin
11/15/85	11/ /85	11/2 /85	11/ /85

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MA INSTITUTE 50-20 - 0001.1.0
11/08/85

U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-20/85-02

Docket No. 50-20 License No. R-37

Licensee: Massachusetts Institute of Technology

138 Albany Street

Cambridge, Massachusetts 02139

Facility Name: MIT Nuclear Reactor Laboratory

Inspection At: Cambridge, Massachusetts

Inspection Conducted: September 25, 1985

Date of Last Physical Security Inspection: September 27 - 28, 1982

Type of Inspection: Routine, Unannounced, Physical Security

Inspector: William J. Madden — 11-18-85
William J. Madden, Physical Security Inspector — date

Approved by: R. R. Keimig — 11-18-85
R. R. Keimig, Chief, Safeguards Section, DRS&S — date

Inspection Summary: Routine, unannounced, physical protection inspection on September 25, 1985 (Report No. 50-20/85-02)

Areas Inspected: Implementation of the licensee's NRC approved physical security plan for the protection of special nuclear material (SNM) of moderate strategic significance. The inspection involved 4 hours onsite by one NRC inspector.

Results: The licensee was in compliance with NRC requirements in the areas examined.

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REPORT DETAILS

1. Key Persons Contacted

   *Lincoln Clark, Jr., Director of Reactor Operations
   *John Bernard, Superintendent, MIT Research Reactor Operations/Maintenance
   *Kwan Kwok, Assistant Superintendent, MIT Research Reactor
   Jerry McDade, Supervisor, Campus Security Systems

   *Denotes those present at the exit interview.

2. 30703 - Exit Interview

   The inspector met with the licensee representatives indicated in paragraph 1 at the conclusion of the inspection on September 25, 1985, and summarized the scope and findings of the inspection.

3. 81480 - Physical Protection of SNM of Moderate Strategic Significance

   The licensee's program for the physical protection of SNM of moderate strategic significance was reviewed by the inspector and was found to conform to NRC requirements and the licensee's implementing procedures. Specific components of the program that were inspected included: records and reports; security organization; alarm response; key control; detection aids; physical barriers; and written security procedures.

APR 25 1986

Docket No. 50-20

Massachusetts Institute of Technology
ATTN: Mr. Lincoln Clark, Jr.
Director of Reactor Operations
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

Subject: Inspection No. 50-20/86-01

This refers to the routine safety inspection conducted by Mr. T. Foley of this office on February 25-28, 1986 of activities authorized by NRC License No. R-37 and to the discussions of our findings held by Mr. Foley with Mr. Clark and members of your staff at the conclusion of the inspection.

Areas examined during this inspection are described in the NRC Region I Inspection Report which is enclosed with this letter. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observations by the inspector.

Our inspector also verified the steps you have taken to correct the violations brought to your attention in the enclosure to our letters dated June 13, 1982, October 25, 1983 and March 21, 1985. We have no further questions regarding your action at this time.

Within the scope of this inspection, no violations were observed.

No reply to this letter is required. Your cooperation with us in this matter is appreciated.

Sincerely,

Original Signed By:

Edward C. Wenzinger, Chief
Projects Branch No. 3
Division of Reactor Projects

Enclosure: NRC Region I Inspection Report No. 50-20/86-01

cc w/encl:
Dr. O. K. Harling, Director of Reactor Laboratory
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Massachusetts (2)

~~8605070407~~ 2pp.

IE01 A/S

bcc w/encl:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
DRP Section Chief
T. Foley (w/cy of encl)

RI:DRP
Foley/meo
4/23/86

RI:DRP
Elsasser
4/23/86

RI:DRP
Wenzinger
4/25/86

Elsasser called Wenzinger 4/24 am

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.: 86-01

Docket No.: 50-20

License No.: R-37

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

Facility Name: MIT Nuclear Laboratories

Inspection At: Cambridge, Massachusetts

Inspection Conducted: February 25-28, 1986

Inspectors: T. Foley, Senior Resident Inspector, Calvert Cliffs

C. Holden, Senior Resident Inspector, Maine Yankee

Approved by: T. C. Elsasser 4/23/86
T. C. Elsasser, Chief, Reactor Projects Section 3C Date

Summary: Inspection on February 25-28, 1986 (Report No. 50-20/86-01)

Areas Inspected: A routine unannounced on-site inspection of licensee activities including: Action taken on Previous Inspection Findings, Facility Tour, Facility Operations, Requalification Training, Surveillance, Experiments, Radiation Protections, Audits and Committees, and verification of reduced on-site storage of High Enriched Uranium (HEU).

Results: Although no violations were identified, two concerns were identified regarding documentation of the licensee's bases for changes, test and experiments determined not to involve an unreviewed safety question (Paragraph 5.a), and calibrations of dosimetry instruments (Paragraph 10). Stored quantities of HEU on site are minimal, and operation of the facility appears to be in conformance with applicable requirements.

DETAILS

1. Key Persons Contacted

*J. Bernard, Superintendent, MIT Research Reactor Operations and Maintenance
*L. Clark, Jr., Director of Reactor Operations
*O. Harling, Director, Nuclear Reactor Laboratory
*K. Kwok, Assistant Superintendent, MIT Research Reactor
*E. Karaian, MIT Radiation Protection Officer

*Denotes those present at the exit interview.

2. Licensee Action on Previously Identified Enforcement Items

(Closed) (82-01-01) The failure to maintain at least 12 inches edge-to-edge separation of packages containing SNM was corrected by moving the BTF subassembly containing 1.1% enriched UO_2 to a location not within 12 inches edge-to-edge of any other SNM. Additionally, within each storage location signs are posted with instructions specifically prohibiting storage within 12 inches of other SNM.

(Closed) (83-02-01) The corrective actions identified in Inspection Report 50-20/83-02, regarding the licensee's failure to adequately post the Hot Cell Area as a High Radiation Area, are still in place. The inspector verified the actions taken by the licensee identified in the above report.

(Closed) (83-02-02) The inspector verified that the licensee no longer uses yellow and magenta ropes for barriers where radiation areas do not exist, and that Radiation Protection controls the use of radiation area barrier ropes.

(Closed) Violation (85-01-01) The licensee's corrective actions to packaging 281 millicuries of Rhenium-186 and 824 millicuries of Rhenium-188 wire and incorrectly labeling the package as 8 millicuries of Chlorine-38 for shipment to Massachusetts General Hospital were as follows:

(1) a specific procedure for "Hot Cell" work was written,
(2) the control of work was re-emphasized to Hot Cell workers,
(3) specific references are now written on samples and pneumatic tube samples are identified,
(4) specifically-shaped containers as indicated on Part II of the work form are used,
(5) distinguishable markings on the samples are recorded on Part II of the work form, and
(6) the gamma dose rate on the work form is verified.

The inspector verified that these actions were performed and in effect during inspection of the facility.

(Closed) (84-01-01) Procedure should be clearly labeled with the title of the individual responsible for its implementation. The licensee has placed the responsibility on the console operator for all immediate actions of Abnormal Operating Procedures (AOP) and Emergency Operating Procedures (EOP). The shift supervisor is responsible for review of the immediate actions and for follow up action. The inspector reviewed AOPs and found the procedures contained the necessary direction to the licensed operators.

(Closed) (84-01-03) Incorporate Emergency Action Levels (EAL) into procedures such that classification of events is readily available. The licensee has incorporated into procedures the EALs listed for non-radiological emergencies. EALs for "Excess Radiation at the Site Boundary Resulting from a Contained Source" were incorporated into the appropriate procedure. EALs are covered in procedures either as a sub-part of major radiological emergencies or emergency procedures.

(Closed) (84-01-04) Accuracy of Procedural References. The licensee reviewed procedures and corrected the typographical errors which led to the inaccuracies.

(Closed) (84-01-05) Provide high range dosimeters within the Containment building emergency lockers. The licensee located two high range dosimeters in the emergency locker in Containment. In addition, other high range dosimeters are located outside of the Containment for use by other personnel assisting in emergency actions.

(Closed) (84-01-06) Provide guidance on supplying dosimetry to medical personnel. Dosimeters will be issued to responding medical personnel if the injury involves radiation exposure or contamination. This action has been proceduralized for medical emergencies.

(Open) (84-01-02) Develop EALS based on specific instrument readings for each of the four classification levels specified in the Emergency Plan. The licensee responded to this item in its reply to Inspection Report 84-01 dated July 25, 1984. This particular item was confusing since the licensee interpreted the action necessary to close this item as being a rewrite of the Emergency Plan. The licensee listed the actions it would need to accomplish a rewrite of the Emergency Plan and requested additional guidance. NRC Region I responded on September 14, 1984 and forwarded this item to Headquarters for review. The inspector discussed the issue with Headquarters personnel and determined that resolution of this item does not require a rewrite of the Emergency Plan. Additional discussions between the licensee and Headquarters were conducted. Documentation of the resolution of this item will be reviewed in subsequent inspections. This item is open.

3. Facility Tour

On February 25, 1986 at about 6:00 p.m. the inspector arrived on site. Observation of physical security controls appeared adequate. The inspector met the Assistant Superintendent and ascertained that shift staffing was in conformance with Technical Specifications (TS). Subsequently a meeting was conducted with the Director of Reactor Operations regarding the scope and purpose of the inspection. A tour of the facility was conducted immediately thereafter. General observations of security, health physics controls, housekeeping, staffing and back shift operations were noted. Control Room observations and Reactor Plant system parameters were monitored by the inspector and compared to Technical Specifications. No inadequacies were noted. Additional tours were made later during subsequent days of the inspection. Inspection tours included: Spent Fuel Pool, New Fuel Vault, Reactor Vessel Head area, Hot Cell, Rad Waste Storage areas, experimental laboratories, Blanket Testing Facility, and Administrative Offices. No inadequacies were identified.

4. <u>Facility Operation Review</u>

The facility is used primarily by MIT graduate students for a variety of neutron activation experiments. The licensee continues to operate the reactor continuously from 8:00 a.m. Monday until Friday evening using a three shift schedule. During the inspection the licensee performed various control rod manipulations and demonstrated the "automatic control of reactor power and reactivity constraints" experiment. The licensee demonstrated various reactivity limiting controls and safeguards associated with the reactivity control system. The inspectors reviewed shift staffing, Control Room logs and observed the operators' performance. Reactor coolant system parameters and system annunciators were discussed with the plant operators. General conditions as they applied to fire prevention and radiological cleanliness were observed. Although no discrepancies were noted in the above areas, the inspector had the following comments:

(a) A review of the reactor start up and shutdown checklists was conducted. The inspector noted that several start up checklists were not complete since some instrumentation was not checked. The inspector was able to verify, through other documentation, the exact status of the equipment. The instrumentation in question did not impact on Technical Specification requirements. The licensee agreed that a more thorough review of checklists was necessary.

(b) The inspector reviewed the hourly calorimetric calculation performed by the operators. Additionally, the Estimated Critical Position (ECP) calculations were reviewed. The October 21, 1985 and February 18, 1986 ECPs did not have all blanks completed. However, the inspector determined that the blanks did not apply to those startups. The licensee agreed that the ECPs should be annotated to show they are complete.

(c) The inspector also compared Technical Specification surveillance requirements with Operator Logs. The DF-1 flow recorder is bypassed during reactor start-up. The bypass is removed prior to increasing power above a pre-set level. The inspector reviewed the Bypass Log and determined that the operators were removing this bypass and signing for its removal, but the times and dates were not listed. This made verification of the reinstatement of the flow recorder difficult. The licensee agreed to study the problem.

Other operating documentation reviewed included the Job Workbook, Fuel Loading Permission, Shutdown Margin Calculations, and Operators' Logs. In general the licensee's record keeping was acceptable. The filing of data in logs was orderly, and data were easily retrievable.

5. Audits and Committees

A review of audit reports and committee activities was conducted.

a. Committees

The committee charged with the oversight of reactor safe operation is the Reactor Safeguards Committee. The committee meets at least once each year and is responsible to the Administration of MIT. The committee chairman establishes subcommittees to assist the committee in conducting its review functions. The committee or an active subcommittee reviews and approves all operating procedures, emergency plans, proposed modifications to the reactor, the use of reactor related experimental facilities and experiments, and all equipment and procedures involving the use of licensed radioactive material in the reactor building.

Through a review of committee activities, the inspector attempted to ascertain that the committee reviews abnormal occurrence and unusual occurrence reports, violations, categories of particular tests and experiments, Technical Specification changes, potential unreviewed safety questions (URSQ), emergency plans and security plans.

The inspector reviewed several unusual occurrence reports and associated corrective actions related to licensee experiments, logs, and emergency plans, and determined that each was properly documented by the Safeguards Committee. It was noted, however, that only "categories" of experiments are reviewed by the Safeguards Committee in order to determine whether an unreviewed safety question exists. The inspector further noted that within a "category", there are experiments which have no safety analysis. According to the licensee these other experiments do not require a review by the Safeguards Committee because they are considered "Class B" procedures, i.e., they are described in the Safety Analysis Report (SAR) and do not involve an URSQ. Instead, Class B procedures require a review by two licensed operators and the Director of Reactor Operations to determine, in part, whether a potential exists for an URSQ and consequently whether further review is required. The bases for this determination is not maintained. Similarly, bases are not maintained for other changes, tests and experiments, which have previously been reviewed, and determined not to involve a potential for an URSQ.

The inspector stated that 10 CFR 50.59 Paragraph (a) (1) is permissive in that it allows the licensee to make changes to the facility and its operation as described in the Safety Analysis Report (SAR) without prior approval, provided a change in Technical Specifications is not involved or an "unreviewed safety question" does not exist. Paragraph (b) requires that the licensee maintain records of changes made under the authority of Paragraph (a) (1). These records must include a written safety evaluation which provides the basis for determining whether an unreviewed safety question exists.

The inspector stated that this meant that any proposed change to a system or procedure, as described in the SAR, either by test or drawings should be reviewed by the licensee to determine whether it involves an unreviewed safety question, and in all cases, the safety evaluation must provide the basis for determination that the proposed change, test or experiment does not involve an unreviewed safety question.

The inspector determined that the licensee complies with the above for those changes, tests and experiments which have been reviewed and determined to have a potential for an URSQ, but not for those that have been determined not to involve an URSQ, in that the bases or reasoning for the "sorting out" (determination of why a potential for an URSQ does not exist) is not documented.

The licensee questioned the inspector as to what constituted "a change" and how other licensees resolve documenting the basis for changes which occur to system and procedures or drawings described in the SAR. The inspector discussed various acceptable alternatives and subsequently forwarded to the licensee the NRC Policy, Part 9800 of Inspection and Enforcement Manual "CFR Discussions" 10 CFR 50.59.

The licensee agreed to further evaluate the requirement in light of the provided NRC interpretation/policy. This matter is unresolved pending the licensee's action to provide the documented bases or rational for those changes, tests or experiments which do not involve an unreviewed safety question (50-20/86-01-01).

b. Audits

Audits of facility operations are performed primarily by the Reactor Superintendent. These audits are quite thorough and comprehensive. However, corrective action, recommendations and implementation are largely the responsibility of the Reactor Superintendent. The Superintendent completed audits of the following, during October through December 1985:

(1) Reactor Console Log Unusual or Abnormal Entries
(2) Changes to procedures/checklists/manuals
(3) Job Workbook Records
(4) Test and calibrations

(5) Radiation Surveys and Environmental Monitoring Radioactive Effluent Records
(6) Refueling and Excess Reactivity
(7) Recommendation of Reportable Occurrence Reports and Unusual Occurrence Reports
(8) Training Files
(9) QA Program/Tagouts/License R-37.

The Reactor Superintendent performs these audits repetitively on three month cycles throughout the year in addition to his normal duties. The inspector reviewed the above audits for July through December 1985, and noted that there were no substantive findings.

However, the inspector questioned the lack of independence and organizational freedom provided by this method of auditing one's own work. The licensee had previously been concerned about this matter and subsequently initiated an annual independent audit by Mr. W. Fecych, a licensee consultant. Audits by Mr. Fecych for the 1984 and 1985 period were reviewed by the inspector and found to encompass outstanding items, operating logs, and dosimetry calibrations.

The inspector stated that although this independence provided more objectivity, the scope and depth of the audits was limiting and should be more comprehensive.

The licensee's Safety Analysis Report which described the Quality Assurance Plan, dated October 1970, provides justification for not requiring the independence and organizational freedom required by 10 CFR 50, Appendix B; however, Section II.2.2 provides a list of activities which fall under the licensee's Quality Assurance Program, and as such should be included in a schedule to be audited on a periodic frequency. Although a clear requirement for audits addressing all aspects of the Quality Assurance Program is not evident, current regulations and industry standards do place more emphasis in this area. The inspector recommended that the licensee consider evaluating current requirements and provide additional independence to those areas within the defined Quality Assurance Program.

6. Technical Specification Surveillance

The inspector verified by review of plant surveillance and other records that the following TS surveillance requirements met frequency and acceptance criteria:

TS No.	Requirement
6.4.1.3	Helium Gas Holder Alarm
6.4.1.4	D_2O Helium System Alarm

TS No.	Requirement
6.4.15	Reflector Tank D_2O Level Scram
6.2.4	Period Level Indication Off Scale Scram
6.1.4.1	Nuclear Safety System Response Time
6.1.4.2	D_2O Reflector Dump Time
6.1.4.4	Primary Coolant Flow Scram Time

No inadequacies were identified.

7. Emergency Planning

The Massachusetts Institute of Technology Reactor Emergency Plan was reviewed. Drills and lectures are periodically (at least annually) performed. Training records, changes in the plan and audits of emergency planning activities were reviewed. The November 19, 1985 Emergency Plan Exercise consisted of MIT Reactor Operators, Radiation Protection Personnel and MIT campus police. Local police, hospital and fire department agreements were verified to be up-to-date. The Emergency Plan is up-to-date and being effectively implemented except as noted in paragraph 2, "Licensee Action on Previously Identified Items," Item 84-01-02, which remains open.

No inadequacies were identified.

8. Experiments

Experiments performed at the MIT Reactor are varied. Currently, neutron activation and analysis and automatic reactivity control experiments are in progress. Experiments are divided into the following categories: reactor operation experiments, Beam Port experiments, incore experiments, thermal column experiments and medical therapy experiments. The licensee uses a "Proposed Experiment Review and Approval Form" in order to control the approval process. The inspector reviewed the following experiments for approvals and safety analysis:

-- Use of Dry Ice in Pneumatic Tubes

-- Sodium metal filled subassembly in the Blanket Test Facility

-- Closed Loop Control of Reactor Power using Shim Blades and Regulating Rods simultaneously

The use-of-dry-ice experiment and use-of-sodium experiments were not accompanied by safety evaluations, however, they were reviewed and approved. The acceptability of these experiments was based on similarity to the other experiments which had previously been approved and which were accompanied by a written safety evaluation. The inspector verified that in addition to reviews, approvals, and safety evaluations, predicted parameters were determined and ascertained within tolerance, irradiated items were properly controlled, and individuals conducting the experiments were trained prior to using the facility (see Training, paragraph 9.b).

No inadequacies were identified.

9. Training Review

a. Requalification Training

A review was conducted of licensed operator training, examinations and reactivity manipulation records. Schedules of lectures and samples of lesson plans were also reviewed. The inspector ascertained that required records were maintained and that the licensee requalification training program was current and fully implemented.

A review of the 1984 and 1985 records indicated that five senior reactor operators had passed their requalification examinations. One reactor operator was upgraded by virtue of passing the SRO examination. One reactor operator's license duties were suspended by the licensee for failure to take the requalification examination.

No inadequacies were identified.

b. Experiments and Student Training

The inspector reviewed documents and discussed with various department staff the training of individuals who conduct experiments. Personnel are trained in the following areas:

-- 10 CFR Part 19
-- 10 CFR Part 20
-- Tables from 10 CFR Parts 20 and 30
-- USNRC Regulatory Guide 8.13
-- Procedures for Radiation Protection
-- Facility Emergency Evaluation Procedure
-- Film Badge Classification Procedure
-- Radiation Exposure Record Application
-- Exclusion Area Entry Permit
-- Maximum Permissible Dose

Each person is given approximately three days to read the above material. A one and one-half hour lecture is given on the same material followed by a question/answer session to determine students' knowledge of exposure limits and restrictions. Twenty hours of classroom instruction is provided on the use and handling precautions associated with the experimental facility and equipment prior to allowing each person to work or attend classes in the building.

Retraining is given annually to persons who handle or receive radioactive materials. This retraining includes but is not limited to the following topics:

-- Permissible Radiation Doses
-- Facility Organization
-- Biological Effects of Radiation
-- Facility Evacuation Plan

No inadequacies were identified.

10. Radiation Protection Controls

The inspector noted radiation postings and controls throughout the facility. Radiation instruments were noted to be calibrated and source checked regularly. Reviews were conducted of radiation surveys, contamination surveys, exposure records of experiments and MIT staff. (Generally, the radiation levels are less than 5 mr/hr in most accessible areas.) Hot Cells were adequately posted as High Radiation areas. Some small areas around the Beam Ports had higher radiation intensities (as high as 15-25 mr/hr) whereas other areas around the Beam Ports were 1-2 mr/hr. The inspector indicated that 10 CFR 50, Appendix I provides guides for maintaining dose to individuals as low as reasonably achievable. The licensee agreed to consider placing controls/signs in or around those areas where higher than normal (5 mr/hr) radiation levels could exist to make personnel aware of the potentially higher intensities and to aid personnel in minimizing their dose.

During review of dosimetry records and calibrations of instruments, the inspector determined that personnel pocket dosimeters were not being calibrated. The inspector noted that 10 CFR 50, Appendix B requires that all devices used to ensure quality should be properly calibrated. The licensee provided a quality assurance audit that previously had identified this same issue. The licensee stated that programs would be established to calibrate all dosimetry. The inspector indicated that pending licensee action on the Quality Assurance Audit, dated November 18, 1985, this item is unresolved (86-01-02).

11. Stored Quantities of High Enriched Uranium (HEU) On Site

In accordance with NRC Inspection and Enforcement Temporary Instruction 2545/1, the inspector examined the quantity, storage and controls associated with HEU on site.

The inspector observed the new fuel vault contents to physically ascertain what new fuel was accessible. Only one fuel element and a few miscellaneous components totalling less than 1 kg of HEU were in the new fuel vault. Safeguard controls associated with the vault are described in Safeguard Inspection Report (50-20/84-02).

Through discussions with the licensee, review of operation history, and observation of the Spent Fuel Pool, the inspector determined that the quantity of material exempt from the licensee's inventory of accessible HEU was greater than 100 Rem/hr at three feet. The current inventory of accessible fuel is of Low Strategic Significance. The current MIT policy is to maintain 'hundreds" of grams of accessible HEU on site versus the "thousands" of grams permitted, excluding the self-protecting fuel, except just prior to fuel transfer. This was documented in a letter to the Secretary of the Commission from L. Clark, October 19, 1984.

The licensee is currently awaiting a fuel cask from DOE in order to reduce its inventory of spent fuel.

12. Exit Interview

At the conclusion of the inspection on February 28, 1986 the inspector met with the director of the facility and reviewed the scope and findings (i.e., unresolved items in paragraphs 5 and 10). The inspector noted the licensee's candor and good cooperation. At no time during this inspection was written material provided to the licensee by the inspector.

AUG 21 1986

Docket No: 50-20

License No. R-37

Massachusetts Institute of Technology
Research Reactor
ATTN: Mr. Lincoln Clark, Jr.
Director of Reactor Operations
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

Subject: Inspection Report No. 50-20/86-02

A routine, unannounced inspection was conducted on July 21-22, 1986 by Ms. Jean A. Cioffi of the Massachusetts Institute of Technology Research Reactor radiation protection program. The elements of the program reviewed are described in the enclosed inspection report.

Within the scope of this inspection, no violations or deviations were observed.

No reply to this letter is required. Your cooperation with us in this matter is appreciated.

Sincerely,

Original Signed By:

for Thomas T. Martin, Director
Division of Radiation Safety
and Safeguards

Enclosure: NRC Region I Inspection Report No. 50-20/86-02

cc w/encl:
Dr. O. K. Harling, Director of the Reactor Laboratory
Dr. Alan Ducatman, Director, Environmental Medical Service
Public Document Room (PDR)
Local Public Document Room (LPDR)
Nuclear Safety Information Center (NSIC)
Commonwealth of Massachusetts (2)

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The licensee determined that the cost-benefit ratio for all Argon reduction work was in the range of $550-1100 per man-rem. These figures compared favorably to the guidelines specified in 10 CFR 50, Appendix I, section II.D. Additional Argon reduction work took place in 1985 (see paragraph 7.0). This item is considered closed.

3.2 (Closed) 86-01-02 (Inspector Follow-up) Calibration of personnel pocket dosimeters. The licensee initiated the calibration of their pocket ion chambers. The dosimeters will be calibrated semi-annually using a 5 curie Cesium-137 source. This item is considered closed.

4.0 Training and Qualification of Personnel

The licensee's program for training and qualification of personnel was reviewed with respect to criteria contained in

- 10 CFR 19.12, "Instructions to Workers";
- Technical Specification 7.10, "Radiation Protection Program."

The licensee's performance with respect to the above criteria was determined by:

- review of the "Massachusetts Institute of Technology Required Procedures for Radiation Protection,"
- discussions with licensee personnel.

Within the scope of this review, no violations were identified. The licensee appeared to be training and qualifying radiation workers in accordance with regulatory requirements and the conditions of their license.

5.0 Implementation of the Radiation Protection Program

The licensee's program for controlling radioactive materials and contamination, providing surveillance and monitoring, and establishing and maintaining administrative radiological work controls was reviewed relative to criteria and commitments in:

- 10 CFR 19.11, 19.12, 20.201, 20.203 and 20.401;
- Technical Specification 3.8, "Radioactive Effluents and Radiation Monitors";
- Technical Specification 4.3, "Reactor Control, Safety, and Radiation Monitoring System Surveillance"; and
- Technical Specification 7.10, "Radiation Protection Program."

The licensee's performance related to the above criteria was determined by:

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- facility tour on July 21, 1986 to observe work in progress; postings, signs, and labels; and radiation monitoring instrumentation;

- review of calibration records for hand and foot monitors, survey instrumentation, area radiation monitors;

- review of survey records for radiation, contamination, and airborne radioactivity; and

- discussions with licensee personnel.

Within the scope of this review, no violations or deviations were identified. The inspector witnessed the lifting of the reactor head and noted that licensee personnel observed the proper industrial safety precautions, and efficient contamination control techniques. The inspector also observed the addition of signs to remind experimenters to survey their work areas for possible higher radiation intensities around beam ports.

Two areas for improvement were identified for licensee attention.

- The licensee maintained no implementing procedures for the reactor radiation protection program. For instance, there were no procedures for calibration of survey instruments and pocket dosimeters, when to read pocket dosimeters and log the reading before re-zeroing, nor how to resolve discrepancies between pocket dosimeters and film badge results. The licensee stated that due to the long employment of all health physics personnel, such procedures were not necessary. The inspector stated that such procedures were necessary for the program to be implemented consistently if the staff were replaced due to illness or retirement. The licensee stated that because of the upcoming retirement of the Reactor Radiation Safety Officer, such implementing procedures would be developed and established. This item will be reviewed in a future inspection (86-02-01).

- The licensee uses a 5 curie Cesium-137 source for their instrument and pocket dosimeter calibrations. However, the source is used in a room without interlocks, warning lights or alarming devices at the entrances to indicate when the source is exposed. The inspector discussed this practice with licensee representatives, who stated that all calibrations were performed when the staff and experimenters were not present, and the individual performing the calibration remained in the vicinity to provide positive control over the area. The inspectors stated that while the controls being used met minimum regulatory requirements, they may not be sufficient to prevent an unplanned exposure should the individual leave the area, or a guard inadvertently enter the room. Following this discussion, the licensee stated that they would: 1) set up a barrier to prevent personnel from inadvertently wandering near the calibration area, and 2) modify the calibration facility with warning lights, alarming devices, and/or interlocks to prevent inadvertent personnel entry. This item will be reviewed in a future inspection (86-02-02).

6.0 Internal and External Exposure Controls

The licensee's internal and external exposure control program was reviewed against criteria provided in:

- 10 CFR 20.101, 20.102, 20.103, 20.104, 20.105, 20.201, 20.202, 20.203 and 20.

The licensee's performance relative to the criteria above was determined by:

- a review of exposure records for 6 radiation workers;
- tour of the counting laboratory and whole body counter in Building 20; and
- discussions with licensee personnel.

Within the scope of this review, no violations were identified. The licensee uses Landauer film badges for dosimetry of record. Visitors to the reactor are issued pocket dosimeters. Internal exposures are monitored by urinalysis and whole body counting.

7.0 Effluent and Environmental Monitoring

The licensee's program for monitoring liquid and gaseous effluents was reviewed with respect to criteria contained in:

- 10 CFR 20.106;
- Technical Specifications 3.8, 4.3 and 7.13.5.

The licensee's performance related to the above criteria was determined by:

- tour and observation of control room effluent radiation monitor indicators;
- review of effluent monitor logs;
- review of the following effluent monitor calibrations procedures:
  - P.M. 6.1.3.9.1, "Water Monitor Calibration Procedure"
  - P.M. 6.1.3.9.2, "Particulate Monitor Calibration Procedure"
  - P.M. 6.1.3.9.3, "Gaseous Monitor Calibration Procedure"
  - P.M. 6.5.9.2, "Environmental Monitor Calibration Procedure"

- review of the 1984 and 1985 Annual Reports;

- discussions with licensee personnel.

Within the scope of this review, there were no violations or deviations identified. The licensee was calibrating all effluent and environmental monitors in accordance with license conditions. Environmental surveys indicated that there were no inconsistencies for the monitoring periods during 1984 and 1985. Furthermore, the licensee was able to further reduce gaseous releases in 1985 by additional studies of the sources generating the Argon-41 in the reactor and by the use of an inert gas blanket system for the reactor. In previous years, the licensee discharged 7000 to 8000 Curies of Argon-41 per year. In 1985, the licensee was able to reduce the gaseous discharge to about 4000 Curies for the year. (See additional information on Argon-41 released in paragraph 3.0.).

8.0 Exit Interview

The inspector met with the licensee's representatives (denoted in Paragraph 1) at the conclusion of the Inspection on July 22, 1986. The inspector summarized the purpose and scope of the inspection and findings as described in this report.