Original w/CERs

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

MEDICAL DEPARTMENT

ENVIRONMENTAL MEDICAL SERVICE

77 MASSACHUSETTS AVENUE 20B-238
CAMBRIDGE, MASSACHUSETTS 02139

February 9, 1988

US Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia PA 19406

Attention: Ms. Betsy Ulrich

Dear Ms. Ulrich:

Enclosed is the summary report prepared by Mitch Galanek on all activities involved in the apparent extremity over-exposure initially reported by me on 12/3/87. Note that the operation that is most significant in this issue is that involving the rabbit studies in which 10 mCi of Ho-166 was injected into the rabbit's knee. Subsequent autopsy was performed such that the wrist badge was in close proximity to the knee. In my initial report I stated that "the wrist badge was often touching the knee and was between the wrist and the exposure source and may overestimate the wrist entry dose by a significant factor". This speculation appeared to be supported by the lower ring badge readings.

A reconstruction of the source-detector-extremity geometry now convinces us that the actual skin entry dose to the extremity from this high-energy beta emitter was overestimated by the wrist badge by 25 - 40%. While correction for this overestimate in effect reduces the actual extremity exposure to within regulatory limits, be assured that our efforts toward reduction of exposures in this operation and increased surveillance over this project will not be reduced.

Yours truly,

Frank Massé

Frank Massé
Director of Radiation
Protection Programs

FXM/nlj

enclosure

8806220053 XA4019
30608
20-01537-02 DCD

(INTERDEPARTMENTAL)

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

Medical Department, Environmental Medical Service

CAMBRIDGE, MASSACHUSETTS 02139

To: Francis X. Massé, Director of Radiation Protection Programs

From: Mitch Galanek, Associate R.P.O. Mitch

Subject: Report of Radiochemist Wrist Badge Overexposure

Date: February 5, 1988

During the month of September 1987, a radiochemist received a reported exposure to his G5 wrist badge dosimeter of 15,040 mrem (11,730 mrem-beta) which caused his 3d quarter cumulative extremity exposure to exceed the regulatory limit of 18,750 mrem. His total reported exposure for the period was 22,910 mrem (17,380 mrem-beta). The exposure was reported by R. S. Landauer on 10/28/87. This notification was made by telephone. R.P.O. requested that Landauer reread the badge to confirm the results. Written notification of the apparent badge overexposure was received by 11/05/87. As per contractual agreement, Landauer reports by telephone any exposure that exceeds the following limits:

G1	--deep------	400 mrem
	--shallow---	2500 mrem
G5	--wrist-----	6000 mrem
U3	--ring------	6000 mrem
U4	--ring------	6000 mrem

The project is involved in the development of a radiochemical method for the treatment of rheumatoid arthritis of the knee joint (radiation synovectomy). The radiochemist routinely handles approximately 80-90 Curies per month of Dysprosium-165. Dysprosium-165 decays by beta minus with an Emax=1.29MeV, and has a radiological half-life of 2.3 hours. To overcome the disadvantages of this short half-life, the project has begun to study alternate radionuclides with a longer half-life that will deliver the needed dose to the knee joint utilizing much less initial activity.

During September, the project began a 3-month study to measure the leakage rate of Holmium-166 from the knee joints of rabbits. Ho-166 decays by beta minus with an Emax=1.81MeV and has a radiological half-life of 27 hours. The <u>in vivo</u> work consisted of injecting up to 10 mCi of Ho-166 into a rabbit's knee joint and following the migration of the activity from the joint over a 72-hour period. At 72 hours, the rabbit is sacrificed and body organs are removed and analyzed for Ho-166 content. Similar experiments were performed earlier with Dy-165. Initial dose rates above the knee joints for both radioisotopes were 500mrem/hr at approximately 5 centimeters. Removal of body organs from the rabbits injected with Dy-165 was done at 18 hours after injection. The activity had decayed through approximately 8 half-lives; therefore at the time of sacrifice and removal of body organs, the dose rate above the knee joint was insignificant. The removal of body organs from rabbits injected with Ho-166 was done at 72 hours. The radioisotopes had gone through only 2.6 half-lives, so approximately 15% of the starting material was still in the knee joint. Also, more experimental time was spent removing additional organs from the Ho-166 rabbits, since it was important to study additional body sites for radionuclide migration due to the longer half-life of Ho-166. Although the calculated dose at 5 centimeters was only 75mrem/hr, the increased time of exposure and increased dose rate at contact were not properly considered with respect to radiation dose, and inadequate dose measurements of the knee were made at 72 hours. Operative procedures caused the wrist badge to be in close proximity to the knee joint, at times actually coming in contact with the knee.

On October 30, 1987, after notification of the apparent overexposure, R.P.O. met with the radiochemist to discuss the reported wrist badge exposure and to devise ways in which this important work could safely continue. R.P.O. monitored the continuing *in vivo* work in November to complete the series of runs and thus salvage the work to date. Plexiglass shielding was designed to shield beta exposures from the knee joint. November exposures reflect the use of this shielding since extremity exposure reduced from 15,040 mrem in September to 4600 mrem in November. Upon completion of these remaining, carefully monitored experiments, no further rabbit work was done using Ho-166 and all such approvals were suspended.

During November, the radiochemist was allowed to continue to process Dy-165 for use in the ongoing human treatment trials. A total of 80 Curies of Dy-165 was processed. Extremity doses were at levels typical of those seen in months before September and October when the higher doses, due to the animal work with Ho-166, were experienced (see attached 1987 summary of exposure history.)

All work with radioactive material for this individual was halted in early December when a second radiochemist joined the project. This allowed for a complete reassessment of the radioactive materials and handling procedures used by this project, and that review plus significant shielding retrofit was accomplished during the year-end holidays. Meetings between R.P.O. and the project were held almost daily to complete this task. New localized plexiglass shields surrounded by lead were developed to help reduce radiation exposures. Detailed procedures covering even the smallest processing step were developed and revised as we progressed toward a goal of maximum dose reduction. The project resumed processing Dy-165 during January 1988 after successfully demonstrating new techniques and procedures to R.P.O.

The following radiation protection requirements have been implemented with the resumption of this work:

1. Procedural changes allowing more steps to be done remotely. Although more time is needed to process the sample in the hot cell, dose reduction is assured.

2. Use of new localized plexiglass shields surrounded by lead to reduce beta and bremsstrahlung exposures.

3. Improvement in dispensing techniques utilizing more shielding and newer tools to allow for more distance from samples.

4. Film and TLD dosimeters will have a weekly exchange frequency to allow for faster turn-around times for exposure reporting, and to allow for continued amending of procedures and shielding to continue to reduce radiation exposures to as low as reasonably achievable.

5. Future animal work will require full review by R.P.O., and R.P.O. will observe initial experiments. This requirement will be in effect whenever changes are made in radionuclides used, amount of activity used, or significant changes in procedure.

6. Shielding will be fabricated to reduce exposures during animal work.

As in the past, R.P.O. and this project will continue to work closely together to enable the continued safe use of radioactive materials and keep radiation exposure as low as reasonably achievable.

Enclosures: 1. Month by month exposure results for radiochemists in project

2. Chronology of R.P.O. actions on this project

3. Most recently developled Dy-165 processing procedures

Chronology of R.P.O. Staff Review of Authorization R-D-9

9/1/86	Radiochemist registered as M.I.T. radiation worker. Meeting lasted 3--3.5 hours. Discussion of proposed project and requirement to get M.I.T. authorization.
9/5/86	R.P.O. met with project in NW13-243 to discuss hot cell layout and project development. Meeting lasted 2.5--3 hours. Discussed radiation instruments needed, shielding design, hot cell design, etc. (R.P.O. Director present.)
9/17/86	R.P.O. observed dry run of Sledge method. Dry run performed several times. (R.P.O. Director present.)
9/19/86	R.P.O. observed initial 1 Curie process for Sledge method.
9/27/86	R.P.O. met with project to review hot cell changes and discuss ventilation and filtration requirements.
10/15/86	Authorization R-D-9 approved by the Radiation Protection Committee.
10/20/86	R.P.O. observed dry run of Cadema method.
10/21/86	Project obtains 1 Curie target to make dose measurements around hot cell. Check shielding effectiveness. R.P.O. present.
10/86	Many informal meetings between R.P.O. and project in development of hot cell and review of procedures.
11/4/86	R.P.O. and project met to review paperwork and discuss transporation and licensing requirements.
11/14/86	R.P.O. and project met to finalize transportation and licensing requirements.
11/24/86 11/26/86 12/1/86	R.P.O. observed dry run of Cadema method for Dy-165 processing. Modification made to remote manipulators, localized shielding, and target crushing device. (R.P.O. Director present.)
12/2/86 12/3/86	R.P.O. and project make dose rate measurements around cell with 1 Curie of activity. Move target into various positions as required by experimental procedure to test shielding effectiveness and use of remote manipulators.

12/11/86	R.P.O. and project make dose rate measurements around hot cell with 20 Curie target.
12/23/86	R.P.O. received project request to perform in vivo rabbit experiments utilizing Dy-165. Request included detailed protocol submitted to M.I.T. Division of Comparitive Medicine.
1/87	R.P.O. discussed proposed in vivo work with D.C.M. D.C.M. approval given. R.P.O. discussed experiments with project. Authorization amendment approved 1/13/87.
1/27/87	Project begins in vivo rabbit work. R.P.O. present for initial work. Dose rate measurements of area and rabbit are performed.
1/27/87 through 2/5/87	Dy-165 rabbit work completed. R.P.O. and project in contact several times to discuss experiment. R.P.O. surveyed work area upon completion of work. No contamination found, R.P.O. transported material back to project laboratory.
10/29/86 through 3/9/87	Project processed weekly 1 Curie targets using Sledge method. Patient doses shipped to a broad medical licensee. R.P.O. packages all shipments. R.P.O. present intermittently as necessary.
4/22/87 to Present	Project processed weekly 20 Curie targets for clinical treatment trials. R.P.O. packages all shipments from M.I.T. and were present during a portion of all of the work.
5/19/87 through 8/13/87	Project began work with Ho-166 to check such details as particle size, filter efficiency, pyrogenicity, etc. prior to routine in vivo rabbit studies. R.P.O. observed some experiments. Target processing much less tedious than with Dy-165 due to lower initial activity used.
9/16/87 through 11/10/87	Project performed more than 30 in vivo rabbit studies to determine Ho-166 migration from knee joint.
10/28/87	R.P.O. notified by R. S. Landauer of wrist badge exposure in excess of contractual limits for phone reporting. Wrist badge exposure = 15,040mrem (11,730 beta).
10/30/87	R.P.O. met with radiochemist to discuss reported wrist exposure. Determined that large percentage of exposure due to Ho-166 rabbit experiment. R.P.O. reviews handling procedures for Ho-166 injected rabbits. Plexiglass shielding designed to reduce beta exposure.
11/5/87 through 11/10/87	R.P.O. monitors final series of in vivo rabbit experiments. Project utilizes plexiglass shielding to reduce wrist exposures. Upon completion of study, no further in vivo work is allowed without review of all procedures.

12/7/87 through 1/4/88	Revised handling techniques for Dy-165 work to include procedural changes allowing for more steps to be done remotely, developed new localized shields to reduce beta and bremsstrahlung exposures, improved dispensing techniques, etc. R.P.O. and project met several times per week during the month to complete the above.
1/6/88	Project performs one monitored processing run of Dy-165 under new procedures; 80 Curies processed for patient trials in January.
1/9/88	Project reviews step-by-step procedures for Dy-165 process, based on observations by R.P.O. on 1/6/88.
1/12/88	R.P.O. observes complete runthrough of revised procedures. (R.P.O. director present.)

DATE	G1 Deep	G1 Shallow	U3	U4	G5
October 1986:	10	10	300	380	M
November 1986:	10	10	180	70	60
December 1986:	M	M	80	70	50
January 1987:	50	180 (130)	1160	---	800 (690)
February 1987:	70	380 (300)	1330	1140	3070 (2170)
March 1987:	90	90	5260	3060	5450 (3650)
April 1987:	50	50	3100	2740	1030
May 1987:	40	150 (110)	--- (damaged TLD)	---	4250 (2980)
June 1987:	70	210 (140)	190	190	4030 (2750)
July 1987:	100	310 (210)	720	880	5970 (4590)
August 1987:	100	190 (90)	1070	940	1900 (1060)
September 1987:	70	70	7720	9620	15040 (11730)
October 1987:	50	190 (140)	7550	13340	12400 (10560)
November 1987: (by phone 12/4/87 4pm)	40	160 (120)	5840	4550	4680 (2860)
December 1987:	20	20	120	140	350 (230)
January 1988:	10	10	1210	2640	1600 (810)

DATE:	G1		U3	U4	G5
	Deep	Shallow			
September 1987:	70	[illegible]0	M	M	260 (190)
October 1987:	20	70 (50)	1820	750	2000 (1780)
November 1987:	50	50	2200	1820	2910 (2070)
December 1987:	20	20	4020	6950	2530 (1500)
January 1988:	10	10	780	2150	780

Calendar Quarter Totals:

DATE:	G1 Deep	G1 Shallow	U3	U4	G5
10/86--12/86:	20	20	560	520	110
1/87--3/87:	210	650 (440)	7750	4850	9320 (6510)
4/87--6/87:	160	410 (250)	3290	2930	9310 (6730)
7/87--9/87:	270	570 (300)	9510	11500	22910 (17380)
10/87--12/87:	110	370	13510	18030	17430 (14200)

Mitch G.
FYI

CADEMA

RADIOACTIVE MATERIAL TRANSFER & PROCESS

1. Monitor 2R transport container with GM monitor to determine if level of reading coincide with radiation shipping paper. If everything is in agreement, proceed to transfer primary shield into hot cell. If readings are higher than shipping paper figures, notify supervisor, do not open 2R container.
2. Position primary container into aluminum holder on transfer slide.
3. Using T-handle allen wrench, loosen three allen screws on primary DU shield cap.
4. Remove allen screws with forceps and place on floor of cell.
5. Remove cap and set aside.
6. Using forceps, remove lead plug and place on corner of aluminum holder.
7. With tongs in left hand, grasp ring of S.S. basket, raise basket to allow you to grasp poly tube containing target with forceps.
8. Grasp poly tube midway with tongs held in left hand.
9. Use forceps to open poly cap.
10. Grasp poly tube with forceps and lower opening of poly tube over top of crucible and gently slide target into zirconium crucible containing pre-measured 1N HCl.
11. Using forceps, immediately push platform with crucible and target into the right side of the hot cell.
12. Replace S.S. basket, poly vial, lead plug and cap on DU shield (no screws).
13. Using manips, transfer crucible with target onto preheated hot plate.
14. While target is heating (2-3 minutes), position crusher (using manips and rope raising device) over top of crucible, approx. a ½ inch.
15. * When target solution is heated, lower crusher bell into crucible aligning target in the center of the crucible.

1-9-88
4th Draft

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RADIOACTIVE MATERIAL TRANSFER & PROCESS

16. Using rope device , allow weight to free fall to break (crush) quartz target in solution inside crucible.
Note: A few attempts may be required to break target.

17. Once target is crushed, raise assembly just above solution and rinse with 1 to 2 ml. of 1$\underline{N}$HCl with remote pipet.

18. Position crusher in holder.

19. Using manip, pour dissolved target into roughing filter.

20. Rinse crucible with 0.5 ml. of 1$\underline{N}$HCl with remote pipet and pour into rough filter, place crucible on cell floor.

21 Eliminate residual liquid in the syringe by use of the syringe plunger rod.

* 22. Turn off heater.

23. Remove roughing filter and place in shielded holder in cell.

* 24. Measure volume in graduated cyclinder and record data.

* 25. Take 0.1 ml. assay sample with remote pipet device, touch drop off.

26. Dispense into 10cc vial with premeasured 1.0 ml. of 1$\underline{N}$HCl.

27. Place pipet unit back into its holder.

28. Transfer sample for assay into left side of cell.

29. Stopper and crimp sample.

30. Using remote tool, transfer sealed sample into shield "A" inside cell.

* 31. Transfer to counting station, count in copper, use tongs for transfer, assay sample, record data in batch record.

32. Return sample in shield "A" to left side of hot cell.

33. Transfer plexi liner "B" into right side of cell with vent needle.

34. Position plexi "B" under filter shield unit, set liner towards rear of shield.

35. Position new filter unit with needle (pre-wet) over sterile vented vial in plexi shield. Use remote tool & manip.

* 36. Using manip, push down on top of 10cc syringe barrel pushing needle into evacuated vial.

37. Place small funnel into top of 10cc syringe.

38. Recheck volume in grad. cylinder.

1-9-88
4th Draft

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RADIOACTIVE MATERIAL TRANSFER & PROCESS

39. Transfer bulk isotope from graduate cyclinder to 10cc final filter.
40. Remove funnel and place in lead storage shield.
41. With remote tool, position plunger in syringe barrel.
* 42. Position pressure device over plunger and apply steady pressure to empty contents into vial, use manip and remote tool.
* 43. Remove filter unit by lifting top of syringe barrel then remove vent needle.
44. Transfer plexi holder with bulk to left side of cell.
45. With remote tool, transfer plexi shield with target into shield "B" and cap, lock plexi into shield with lock screw.
46. Check handling tools for contamination.
47. Transfer "B" shield to pass thru, handle of shield towards glove box, open glove box window to facilate transfer.
48. Adjust dispensing syringe to correct volume settings. (Position bulk shield next to filling block)
* 49. Draw sample from "B" shield with right hand, transfer syringe to left hand. Replace cap with right hand, repeat for each vial.
50. Dispense in vial - record data.
51. Repeat procedure as required.
52. Cover each filled vial with plexi shield.
* 53. Remove bulk shield from glove box to hot cell via pass thru.
54. Place remote plexi shield in glove box.
* 55. Starting with vial # 1, with forceps, transfer to plexi hold - turn handle to lock in place.
56. Add appropriate amount of NaOH via syringe & needle.
57. Shake vial by hold handle.
58. Check pH via shielded syringe and record.
59. Repeat step 55-58 for each vial.
* 60. When finished, dispose of pH paper in hot waste.

1-9-88
4th Draft

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RADIOACTIVE MATERIAL TRANSFER & PROCESS

61. Using forceps, place each vial into water bath and heat for required time.

62. Using forceps, remove each vial from water bath and place in plexi shield.

63. Vortex vial.

64. Check pH with shielded syringe and record.

* 65. Place vial in "A" shield and transfer to assay station for final assay. After assay, place vial in primary shipping shield.

66. Repeat for all vials.

67. Follow packaging instructions for 2R shipper.

68. H.P. will determine transport index and apply security seal.

* Denotes a key step which must be performed as instructed requiring special attention to assure completion of the process.

1-9-88
4th Draft

REPORT NO. 30-00763/88-001

ATTACHMENT 2

LICENSEE AUTHORIZATION FOR DYSPROSIUM/HOLMIUM PROJECT

RP-01
Rev. 7/81

Authorization No. R-D-9

Expiration Date October 1988

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

APPLICATION FOR AUTHORIZATION TO POSSESS AND USE RADIOACTIVE MATERIAL

INSTRUCTIONS: Complete Section I **in triplicate** and forward to the Radiation Protection Office, Room 20B-238. When approved, a copy of the application, with a designated Authorization number, will be returned to the Project Supervisor. To place a purchase order, submit to your purchasing agency a purchase requisition on which is stated "Radioactive Material" and the designated Authorization number.

SECTION I

1. Identification of persons (a) who will use and (b) who will supervise use of radioactive material:
   (a) Name of person(s) who will use the material: (List principal user first)

Name	Department	M.I.T. Title	Room No.	Tel. No.
	Nuclear Reactor Laboratory (NRL)	Research	NW13-242	
Joseph Daniele	" " "	Research Affiliate	NW13-243	x2099
Timothy Lee	" " "	Research Affiliate	NW13-243	x2099

   (b) Name of person who will supervise the use of the material:

Name	Department	M.I.T. Title	Room No.	Tel. No.
Prof. O.K. Harling	NRL	Director	NW12-204	253-4201
Dr. Ilhan Olmez	NRL	Research Scientist	NW13-261	253-2995
W. Fecych	NRL	Supv. Reactor Utilization	NW12-112	253-4205

2. Rooms where the material will be handled:
   (a) Material stored in NW13-243
   (b) Material used in NW13-243

3. Description of material to be procured:

Radio nuclide	Amount of Activity: To be possessed*	Amount of Activity: In use per exp.	Chemical and physical form of material to be procured	Comments
165 Dy	20000 mCi	4000 mCi	Dysprosium-165 Ferric Hydroxide Macroaggreate	Ref. FDA Ind. #28,054
166 Ho	900 mCi	900 mCi	Holmium-166 Ferric Hydroxide Macroaggreate	
^{57}Co	5.67 mCi(7-22-81)	#4244MA	Sealed calib. source	Sources stored in NW13-245
^{137}Cs	98.2 μCi (5-26-81)	#90170481A24	"	"
^{60}Co	98.7 μCi (5-29-81)	#90230481A17	"	"

*Maximum amount to be possessed by project at any one time.

4. Is any of the radioactive material used as a label for potentially biohazardous material, toxic chemicals, or carcinogenic/mutaganic material? Yes ____ No X . If answer is "yes," explain on a supplementary page.

5. To be procured from: M.I.T. Reactor X Commercial Supplier X Other ____

6. Type of investigation for which the material will be used: See attached sheets.

SECTION II (This section to be completed by the Radiation Protection Office.)

A. Comments relating to the application:

Room NW13-243 is registered as a medium level laboratory.

The project will possess an area monitor for the hood area where the Dy-165 target is disolved, a portable survey instrument for contamination control, and a portal monitor for personnel monitoring before leaving the laboratory.

The ventillation system for NW13-243 has been equipped with an absolute filter. RPO will be monitoring post filters to determine any release of radioactivity.

A review of the sealed source wipe test records shows no leakage from any of the sources.

B. Following are the specific conditions of approval concerning work with radioactive materials under this authorization:

1. Radioactive material transported from NW13-243 must be in an approved DOT shipping container with authorization from the MIT Radiation Protection Office.
2. Radioactive material will be handled using appropriate handling tools (i.e., shielded syringe, tongs, lead transportation container) and appropriate shielding to keep personnel exposures as low as reasonably achievable.
3. All persons involved in handling of the radioactive materials at MIT will wear both body and wrist badges AND FINGER RING DOSIMETERS
4. A G.M. and ion chamber survey instrument must be available and operational in the laboratory when working with the radioactive material.
5. There will be no mouth pipetting of radioactive solutions.
6. The sealed sources, which are used for calibration purposes, will be stored in a locked, shielded safe in NW13-245. The sources will be wipe tested every six months by an RPO representative.
7. The authorization <u>does not</u> approve animal or human use studies at MIT.

C. This application is approved with the following general conditions:

1. The proposed work with radioactive material shall be performed in the manner specified in Sections I and II-B,C,D. There shall be no changes in the approved procedures without the prior approval of the Radiation Protection Committee. The Radiation Protection Office shall be notifed prior to a change in place of use or storage of radioactive material.
2. The use, storage and disposal of the radioactive material shall be in conformity with (a) the provisions of the Code of Federal Regulations Title 10, Part 20 "Standards for Protection Against Radiation" and (b) the provisions of "M.I.T. Required Procedures for Radiation Protection."

Only persons registered with RPO will be allowed in the project laboratories during the handling of the Dy-165 targets.

RPO will monitor the stack air discharge from NW13-243. If discharge concentrations approach the Maximum Permissible Concentrations listed in Appendix B Part 20 (Dy-165: $MPC_{air}=9\times10^{-8}$ uCi/ml) the project will cease work until appropriate engineering controls are in place.

RPO will monitor all outgoing shipments of Dy-165 to ensure compliance with all DOT and NRC regulations. The project will provide certified Type A containers for the transportation of the Dy-165.

The project will supply RPO the name of the carrier used for transportation and a copy of his insurance coverage.

To: Mitchell Galanek/Nickolas James
From:
Re: ^{166}Holmium FHMA
Date: 10/2/87

The arrangements are being finalized for phase II of the pre-clinical ^{166}Ho studies.

The purpose of the study is to acquire additional data to support human clinical studies.

We will follow a similar schedule as the one we used in June 87.

The product will be processed in NW 13-243/45 and transferred to Building 45 Comparative Medicine to conduct the animal studies.

The amount of radionuclide to be transferred or in possession at any one time will not exceed 150 mci. per study day.

The proposed study dates are:

DATE	DAY	PROCEDURE	# RABBITS
10/22	Thur.	1,2	1
10/23	Fri.	1,2,3	5
10/26	Mon.	2,3	4
10/27	Tues.	2,3	2

The above schedule will be repeated on the following dates:
10/29, 10/30, 11/2, 11/3, and 11/12, 11/13, 11/16, 11/17.

KEY: 1 = Inject, 2 = Blood, 3 = Bio-dist.

All organ weighing and counting will be performed in the assigned area. Portable counting equipment will be supplied.

At the end of each test date the animal carcasses and removed organs will be placed into plastic bags, securely closed, labeled and prepared for freezer storage or whatever protocol is prescribed.

All the details and necessary safety precautions to be reviewed with and approved by the Radiation Protection Committee and R.P.O. before proceeding with the study.

Investigator: ______
Project Supervisor Approval: OK Harling
Date: 10/5/87

cc: O.K. Harling
L. Clark
J. Bernard
W. Fecych
I. Olmez
J. Fox
F. Masse

SJK:rjk
87-89

To: Mitch Galanek
From:

Re: ^{166}Holmium FHMA Ferric Hydroxide Macroaggregate
Date: 6/1/87

I am requesting that the Nuclear Reactor Laboratory (NRL) license be amended to allow the transfer and possession of ^{166}Ho from NW 13-243/245 to Building 45, Division of Comparative Medicine.

The amount of radionuclide to be transferred or in possession at any one time will not exceed 150 mci. per study day.

The purpose of the study is to compare the ^{166}Ho FHMA process to the ^{165}Dy process. Leakage of radioactivity from the injected joint will be monitored over an 18 & 72 hour period post injection. The test animals will be sacrificed after 18 & 72 hours for bio-distribution.

The amount of ^{166}Ho [illegible] injected per rabbit is 20 mci. per 0.1-0.2 cc.

Bloods will be drawn and counted at 0,1,27, and 72 hours.

The proposed study dates are:

Date	Day	Procedure	# Rabbits
6/11	Thur.	1,2	2
6/12	Fri.	1,2,3	4
6/15	Mon.	2.3	2
6/18	Thur.	1,2	2
6/19	Fri.	1,2,3	4
6/22	Mon.	2,3	2

1 = Injection
2 = Blood
3 = Bio-dist.

Change in isotope use only. Same procedures as for Dy-165. Possession limit to remain as in original authorization.

All organ weighing and counting will be performed in the assigned area. Portable counting equipment will be supplied.

At the end of each test date the animal carcasses and removed organs will be placed into plastic bags, securely closed, labeled and prepared for freezer storage or whatever protocol is prescribed.

All the details and necessary safety precautions will be reviewed with and approved by the Radiation Protection Committee and R.P.O. before proceeding with the study.

Investigator: ______________________

Project Supervisor Approval: Otto K Harling

Date: 6/2/87

cc: O.K. Harling
L. Clark, Jr.
J. Bernard
W. Fecych
I. Olmez
J. Fox
N. James
F. Masse

To: Distribution
From:
Re: Status of Dysprosium Project
Date: 4/11/87

The final phase of the pre-clinical Dysprosium project will be completed the week of April 12th.

The completion of the pre-clinical phase will permit the project to proceed to phase one of the clinical trials.

Cadema will be providing patient doses of ^{165} Dy-FHMA to our principle investigators Dr. C. Sledge of Brigham & Womans Hospital and Dr. J. Zuckerman of Hospital for Joint Diseases.

It is planned to irradiate a twenty curie target with a backup on Wednesday's, start processing at 0730 hours and ship by 1000 hours.

The initial clinical test samples will be processed on Wednesdays, eventually it will include Fridays as the clinical trials expand. The dual weekly (Wed. - Fri.) processing is not expected to begin until June, 1987.

Arrangements have been made with commerical carriers to transport the material.

All appropriate state, federal and M.I.T. regulations will be adhered to in the transfer-ing of the material.

April Schedule

Date	Time	Event
4/21 (Tues.)	____	Target prep. (2) Del. to M.I.T.R.
4/22 (Wed.)	0730	Target Process
	1000	Ship to H.J.D. - N.Y.
	1030	Ship to B.W.H. - Boston
4/28 (Tues.)	____	Target prep. Del. to M.I.T.R
4/29 (Wed.)	0730	Target Process
	1000	Ship to H.J.D. - N.Y.
	1030	Ship to B.W.H. - Boston

Please contact me if you have any questions.

RP-05
Rev. 1/84

.T. RADIATION PROTECTION OFFICE

Amendment Number 1
Authorization # R-D-9

Review and Approval of a Request-For-Amendment of an Authorization To Possess and Use Radioactive Material

A. Identification of amendment-request:

Project Supervisor's Name O. Harling Authorization # R-D-9

B. Comments relating to review of amendment-request:

1. See attached request letter and in vivo protocol.
2. The procedures have been discussed with Chris Newcomer of DCM and all procedures meet with his approval.
3. Room 45-173 is classified as a low level laboratory.

C. Following are conditions of approval of this amendment-request:

1. Each condition of approval of the original authorization shall remain in effect unless specifically superseded by an approval condition specified below.
2. This amendment authorizes the use of up to 500 mCi of ^{165}Dy FHMA in animal studies at the DCM facilities in building 45.
3. The project will label the cages housing the injected rabbits with "Caution Radioactive Materials" labels. The radionuclide, amount, date, and person responsible will be entered on the labels.
4. The project will monitor the cages to ensure that the ^{165}Dy has decayed to background before the cages are released to the cage washers.
5. Sacrificed rabbits will be stored frozen for radioactive decay (1 week) and then released to DCM personnel for incineration.
6. RPO will transport the 500 mCi of ^{165}Dy from NW13-243 to 45-173.

D. Signature of Reviewer Mitchell Galanek Date 1/12/87

E. Approved by FKChrisgoe Date 1/12/87
(For the M.I.T. Radiation Protection Committee)

(INTER-DEPARTMENTAL)

MASSACHUSETTS INSTITUTE OF TECHNOLOGY

Medical Department, Environmental Medical Service

CAMBRIDGE, MASSACHUSETTS 02139

To: Frank Masse'

From: Mitch Galanek

Subject: Shielding of 20Ci Dy-165 Source

Date: December 12, 1986

On 12/11/86 and I made a series of radiation measurements associated with the handling of a 20 Ci Dy-165 source in NW13-243. The source was placed unshielded in the experimental set-up in such a manner as to give us the highest possible dose rates. During normal procedures, these dose rates would be lower due to some localized lead shielding employed during most of the handling procedures. The following are the results of our radiation survey:

Location of Measurement	Dose rate (mr/hr)
NW13-243	
Transport container	15 (at surface)
Inner container	50
Lead glass	15
Lead shield (Left and right of glass)	13
Lead bricks (below glass)	2
Wooden cabinets (knee high)	15
General area right of hood	5 (200 mr/hr hot spot)
General room area	2
Door to NW13-243	0.6-0.8
NW13-224	
Hood face	5
General room area	0.6
Corridor between NW13-224 and 226	0.3
General area left of hood	5 (20 mr/hr hot spot)
NW13-226	
Right side nearest NW13-243	<0.1

Several of the above measurements were repeated using a 600 mCi Dy-165 source. The following are the results of that survey:

Location of Measurement	Dose rate (mr/hr)
W13-243	
Lead glass	1.0
Lead shield (left and right of glass)	0.6
Lead bricks (below glass)	0.2
General room area	0.1
General area right of hood	1.0 (5.0 mr/hr hot spot)
W13-224	
Hood face	0.1
General room area	<0.1
Corridor between NW13-224 and 223	<0.1
General area right of hood	0.2 (1.0 mr/hr hot spot)
W13-226	
Right side nearest NW13-243	<0.1

All measurements were made with a Keithley ion chamber survey instrument.

cc. A. Ducatman
O. Harling
I. Olmez

NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

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

L. CLARK, JR
Director of Reactor Operations

To: Mitch Galanek
From: -

RE: ^{165}Dy-Dysprosium FHMA Ferric Hydroxide Macroaggregate

Date: 12/23/86

I am requesting that the Nuclear Reactor Laboratory (NRL) license be ammended to allow the transfer and possession of ^{165}Dy from NW13-243/245 to Building 45, Division of Comparative Medicine.

The amount of radionuclide to be transferred or in possession at any one time will not exceed 500 mci. per study day.

The purpose of the study is to compare the BWH ^{165}Dy FHMA process to the Cadema process. Leakage of radioactivity from the injected joint will be monitored over an 18 hour period post injection. The test animals will be sacrificed after 18 hours for biodistribution.

The amount of ^{165}Dy to be injected per rabbit is 50 mci. per 0.1 - 0.2 cc.

Bloods will be drawn and counted at 0,1,3,5 and 18 hours.

The proposed study dates are:

Date	Day	Procedure	# Rabbits
1/8	Thur.	1,2	5
1/9	Fri.	2,3	5
1/15	Thur.	1,2	5
1 16	Fri.	2,3	5
1/20	Tue.	1,2	5
1/21	Wed.	2,3	5
1/22	Thur.	1,2	5
1/23	Fri.	2,3	5

1 = Injection, 2 = Blood, 3 = Bio-dist.

NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

138 Albany Street, Cambridge, Mass. 02139

L. CLARK, JR.
Director of Reactor Operations

All organ weighing and counting will be performed in the assigned area. Portable counting equipment will be supplied.

At the end of each test date the animal carcasses and removed organs will be placed into plastic bags, securely closed, labeled and prepared for freezer storage or whatever protocol is prescribed.

All the details and necessary safety precautions will be reviewed with and approved by the Radiation Protection Committee and R.P.O. before proceeding with the study.

Investigator: ______________________

Project Supervisor Approval: O K Harling

Date: 12/23/86

cc: O. K. Harling
L. Clark, Jr.
J. Bernard
W. Fecych
I. Olmez
J. Fox
C. Newcomer
F. Masse

Massachusetts Institute of Technology

COMMITTEE ON ANIMAL CARE

PROTOCOL REVIEW FORM

Protocol Number ______
Date Submitted ____ / / ____
Date DCM Approved ____ / / ____
Date CAC Approved ____ / / ____
USDA Pain Category ______
(For office use only)

LEASE TYPE OR PRINT CLEARLY

I. GENERAL INFORMATION

A. PRINCIPAL INVESTIGATOR NAME: ______ DATE SUBMITTED: 12 / 23 / 86

B. OFFICE EXT: 2099 LAB EXT: ______ DEPT: Nuclear Reactor Labs

HOME PHONE NO. IN CASE OF EMERGENCY: ______

C. OTHER PERSONNEL INVOLVED: Sonya Shortkroff

D. TITLE OF PROPOSAL: ^{165}Dy-Dysprosium - FHMA

E. AGENCY FUNDING THIS PROJECT: Cadema #98379

F. PERIOD OF TIME TO BE COVERED BY GRANT: FROM: / / TO: / /

I. WRITE A BRIEF DESCRIPTION OF THE PURPOSE OF THE STUDY

A small volume of a radioactive drug will be injected into the ynovium of the rabbits knee joint. Leakage of radioactivity from the njected joint will be monitored in the course of the study over an 18 hour ime period. At the end of the 18 hour period the test animal will be acrificed with biodistribution to follow.

PLEASE NOTE: A SUBSTANTIAL CHANGE IN PROTOCOL, AN INCREASE IN THE NUMBER OF ANIMALS USED OR A CHANGE IN THE ANIMAL SPECIES USED, WILL NECESSITATE THAT THIS FORM BE RESUBMITTED.

II. SPECIES INFORMATION

A. ANIMAL SPECIES TO BE USED a: Rabbit b: ______ c: ______ d: ______

B. ESTIMATED NUMBER OF ANIMALS OF EACH SPECIES

TO BE USED IN THIS STUDY a: 20 b: ______ c: ______ d: ______

TO BE HOUSED PER DAY a: 5 b: ______ c: ______ d: ______

C. CHECK FOR EACH SPECIES WHEN APPLICABLE

a	b	c	d	
				1. RATIONALE FOR USE OF ANIMALS:
X	__	__	__	Studies involve analysis of a process/mechanism that cannot be reconstructed in vitro.
X	__	__	__	Studies cannot or should not be undertaken in humans.
__	__	__	__	Other, explain ______
				2. APPROPRIATENESS OF THE SPECIES:
X	__	__	__	Demonstrated similarity of the process under study to that in humans.
X	__	__	__	Large amount of relevant data already has been derived from species.
X	__	__	__	Manipulations required for the experiment (e.g., surgery) require an animal greater than 2 Kg in size.
__	__	__	__	Other, explain ______
				3. NUMBERS USED:
X	__	__	__	To establish statistical significance of experimental results.
X	__	__	__	To obtain sufficient biological materials to permit the studies proposed.
__	__	__	__	Other, explain ______

WRITE A BRIEF DESCRIPTION OF THE PROTOCOL (Include any particular husbandry requirements, such as special diet or housing.)

Rabbits fasted 24 hours prior to the test, Five (5) rabbits will be partically anesthetized and injected intra-articularly with the Dy^{165} FHMA drug. Blood samples will be taken 0,1,3,5 and 18 hours post injection. Test animals will be held in isolation during this time frame. The animals will be sacrificed at 18 hours by I.V. injection, opened up and organs removed for biodistribution of radioactivity. Organ assessment will include right and left inguinal fat, popliteal node, kidneys, right hepatic lobe of the liver, urine and feces specimen.

All organ weighing and radioactivity counting og same will be performed in the assigned necropsy room. Portable counting equipment will be supplied.

At the end of each test date, the animal carcasses and removed organs will be placed into plastic bags, securely closed, labeled and prepared for freezer storage or whatever protocol is in effect for handling same.

All necessary precautions will be used in the exercise of this test to minimize and control contamination.

SPECIAL QUESTIONS ER. Squibb- Mice, dogs, cats, rats, pigs, monkeys, rabbits
Cambridge Nuc.-Rabbits/mice
NEN-Rabbits/mice Diagnostic Isotopes- Rabbits, mice, dogs

A. What relevant training have the personnel of this project had in conducting animal experimentation?

Principal investigator: Angenics- cows

Associate investigator(s): Brigham & Womens Hospital = Rabbits, mice, rats

Technician(s):

Student(s):

B. Are any animal restraint devices used in this project? YES X NO ___

If "yes", what method and duration? Holding Stock - for restraint during blood collection - 30 minutes

C. Is surgery to be performed? YES ___ NO X

Is any other manipulation to be performed which would lead to pain or other discomfort which alters well being? YES ___ NO X

(If the answer to either of these is "yes", complete Surgical Report/Invasive Procedures section, pages 4 and 5)

Is more than one survival surgical procedure proposed for any animal in this study? YES ___ NO X

If "yes", provide justification for multiple survival surgery

D. How will the animal be euthanatized? I.V. with Phenobarbital

E. Are radionuclides to be used? YES X NO ___

If "yes", indicate Radiation Protection Office approval #

F. Are biohazards involved in this study? YES ___ NO X

If "yes", date of Committee on Assessment of Biohazards approval

G. Will any aspect of the animal experimentation in this study be conducted at another institution? YES ___ NO X

Describe the facility and the nature and duration of this activity

H. Signature of Principal investigator: 12/22/86

date

SURGICAL REPORT/INVASIVE PROCEDURE

(this form must be completed if answer to V-C is "yes")

GENERAL INFORMATION

A. INDIVIDUAL(S) RESPONSIBLE FOR SURGERY & POSTOPERATIVE CARE: [redacted]

B. LAB EXT: 2099 HOME PHONE NO. IN CASE OF EMERGENCY: [redacted]

C. CHECK LETTER CORRESPONDING TO SPECIES USED: X a ___ b ___ c ___ d

EXPERIMENTAL DESIGN

A. PREANESTHETICS: ______

B. ANESTHETIC AGENT(S): Acepromazine - Ketamine

C. DOSAGE: 0.1 - 0.2 cc

D. FREQUENCY OF DOSAGE (e.g., daily, weekly): Day of injection of isotope

E. ROUTE OF ADMINISTRATION: I.M.

I. **SURGICAL PROCEDURE:** (Please provide detailed information and attach additional pages if necessary.)

NONE

(continued on page 5)

VIII. SURGICAL PROCEDURE (continued):

N/A

IX. POSTOPERATIVE CARE:

N/A

X. DESCRIBE THE DAILY CARE AND MAINTENANCE PROCEDURES OF SURGICALLY IMPLANTED DEVICES:

N/A

XI. DESCRIBE ANY NONSURGICAL PROCEDURES THAT ARE PAINFUL OR STRESSFUL TO THE ANIMALS:

None

IX. ADDITIONAL COMMENTS

Rabbits will be anesthetized I.M. with Acepromazine/Ketamine

For office use only:

Massachusetts Institute of Technology
DI'''SION OF COMPARATIVE MEDICIN'
45-, • 37 Vassar Street • Cambridge, MA 021.
(617) 253-1733 or -1718

No. 87- 2305

REQUEST FOR LABORATORY ANIMAL PURCHASE AND CARE

PROJECT

Title (specify) ^{165}Dy-Dysprosium - FHMA			CAC Protocol Approval #
Investigator		Authorized Personnel Sonya Shortkroff	
Telephone # 253-2099	Room # NW 13-222	Department Nuclear Reactor Labs	

TO BE COMPLETED BY REQUISITIONER

A

1. Will radioisotopes be used? If yes, Radiation Protection Office Authorization # ______ — [ ] No [X] Yes*
2. Will any biohazardous agent be used? (Chemicals, carcinogens, toxic substances, infectious agents, etc.) — [X] No [ ] Yes*
3. Will a surgical procedure be performed? — [X] No [ ] Yes*

4a. Number of animals used where no pain, distress, or use of pain relieving drugs are involved: 0

4b. Number of animals involving pain or distress where appropriate anesthetic, analgesic, or tranquilizer will be used: 20

4c. Number of animals involving pain or distress without use of appropriate anesthetic, analgesic, or tranquilizer:** 0

5. Hazardous Precautions Required:
- [ ] None
- [ ] Biological
- [ ] Chemical
- [ ] Carcinogenic
- [X] Radioactive
- [ ] Other**

6. Required Protective Clothing:
- [ ] None
- [ ] Mask
- [X] Gloves
- [X] Lab Coat
- [ ] Footwear
- [ ] Head Gear
- [ ] Other**

B

1. Species	2. Strain	3. Sex	4. Age	5. Weight	6. Quantity	7. Est. Stay
Rabbit	New Zealand White	M or F		2 Kg	5/Study	42 Hrs./Study

8. Other specifications:

9. Special requirements for animal care Will hold food 24 Hrs.Pre & 18 Post Injection — 10. Est. #/cage 5/Study

11. Date needed 1/8,9--1/15,16-1/20,21-1/22,23 — 12. Vendor: Childrens Hospital

* If yes, complete DCM form(s), if not previously submitted for this project or if modification of procedure requires new form.
** Attach brief explanation.

REQUEST AND FINANCIAL APPROVAL

Authorized Agent (signature)		Date
Administrative Officer (signature)		Date
Purchase Account #	Care Account #	Expiration Date:

TO BE COMPLETED BY DCM

A *Arrangements*

1. Vendor	3. Other
2. Location	4. Approval (Facility Mgr) Date

B *Purchase*

1. Date confirmed ______ with	5. Charge	6. Purchasing Agent
2. Lot #	Vendor $ ______	7. Debit Acct # 5??
	Shipping $ ______	8. Credit Acct # 11449.146
3. Delivery date	Processing $ ______	9. Billing Agent
4. Tattoo	Total $ ______	10. Comments

WHITE: DCM COPY / CANARY: ANIMAL FACILITIES MANAGER / PINK: ACCOUNTING COPY / GOLD: DEPARTMENT COPY

7. Principal procedures involved in t' 'se of the material: Include procedures imp' nt to the consideration of contamination control, such as: evaporation, ..ansfer of powder, etc.

Nuclides	Activity mCi	Room Used	Exhaust Ventilation Used	Procedure description
^{165}Dy	20000	NW13-243	Yes	See attached sheets.
^{166}Ho	900mCi	"	"	
^{57}Co	*	"	No	Sealed sources used for CAPINTEC calibration
^{137}Cs	*	"	"	"
^{60}Co	*	"	"	"

If item 7 is continued on a supplementary page, check here ____

8. Radiation protection: Check special equipment that will be used to control external and internal radiation exposure. (If item 8 is continued on a supplementary page, check here ____)

X Glove-box
X Fume hood
X Shielding
X Handling tongs
X Shielded storage container
X Radiation signs and labels

X Transportation container
X Protective gloves
X Lab-coat
___ Shoe covers
X Trays
X Mechanical pipette

___ Scintillation-Survey Meter
X G.M. Survey Meter
X Ion-chamber Survey Meter
X Dosimeter
Monitoring
X Badges
Body X
Wrist X
Finger X

9. Radioactive-waste disposal:

Type of Waste	Method of Disposal
Solid	Into RPO collection container in room # NW13-243
Liquid	Into RPO collection container in room # NW13-243 Into sink of room # ________
Scintillation Fluid	Into RPO collection container in room # ________
Animal tissue	To be stored (for RPO collection) in freezer in room ____
Special waste*	Describe on attached sheet (check) ________

*Waste in form of gas, pyrophoric or pathogenic material are to be considered special wastes.

10. Name of person completing items 1 through 9 ________________

11. Project Supervisor's approval (Signature) Otto Harling Date 7/31/86

Project Supervisor's name (Please Print) Prof. O.K. Harling

D. In addition the following standard c[illegible]tions of approval are emphasized:

☑ The project supervisor's responsibilities must be transferred to another person (with supervisory qualifications) during any extended leaves from the Institute (a month or longer). RPO approval of any such changes must be secured in advance.

☑ No transfer of powdered radioactive material is allowed under this authorization. If powdered or crystalline material is purchased, it will be put into solution in the original shipping container.

☑ All procedures which may result in airborne contamination of radioactive materials will be performed in a hood which is approved for work with radioactive material.

☑ All unattended containers of radioactive material or any apparatus containing radioactive material must be labeled with properly completed "Caution Radioactive Material" sign or label.

☑ Radioactive material will be doubly contained in a shatterproof container when transported between laboratories and/or through corridors.

☑ There will be no mouth pipetting of radioactive solutions.

☑ Radioactive wastes will be disposed of in RPO approved radioactive waste containers and/or in RPO approved laboratory drains. A record of waste disposal will be kept by recording the experimenter's name, nuclide, amount, and date on the appropriate RPO forms.

☑ An appropriate functional survey instrument will be readily available for contamination control monitoring in all laboratories in which millicurie (mCi) quantities of β-γ emitters are being handled (purchase of 1 mCi or more of stock solution constitutes such handling.) An exception to the above will be for radioiodines that are not protein-bound. In such cases an appropriate functional survey instrument will be required if $\geq 100\ \mu$Ci is handled. Alpha emitters shall have the same 100 μCi limit.

☑ The project will ensure that their radiation survey instruments are in proper working condition. If repairs or calibrations are required, the project will deliver the instruments to the Radiation Protection Office. RPO will calibrate the instruments at six-month intervals.

☐ Liquid scintillation wastes and animal carcass wastes that are exempt from radioactive waste disposal regulations will be packaged in accordance with RPO regulations.

☐ Packages of radioactive material received by the project will be opened in accordance with RPO instructions that are attached to each package.

E. Signature of Reviewer Mitchell Galanek Date 10/14/86

Approved by: FX Massé Committee mtg of 10/15/86 Date: 10/15/86

For M.I.T. Radiation Protection Committee

F. Termination of this authorization:

1. Work with radioactive material terminated (date) ______

2. Disposition of:

Radioactive material ______

Waste containers ______

Survey meter(s) ______

3. Residual Contamination:

Hood(s) ______

Sink(s) ______

Laboratory surfaces ______

4. Lab. area checked out by: ______ Date: ______

5. Approved for "terminated file" ______ Date: ______

REPORT NO. 030-00763/88-001

ATTACHMENT 3

DOSES MEASURED BY WRIST AND FINGER RING DOSIMETERS

REPORT NO. 030-00763/88-001

ATTACHMENT 3

DOSES MEASURED BY WRIST AND FINGER RING DOSIMETERS

INDIVIDUAL A

1987	Right Wrist (mrem)	Finger Ring Dosimeter Right (mrem)	Finger Ring Dosimeter Left (mrem)
January	800	*	1,160
February	3,070	1,140	1,330
March	5,470	3,060	5,260
April	1,030	2,740	3,100
May	4,250	*	*
June	4,030	190	190
July	5,970	880	720
August	1,900	940	1,070
September	15,040	9,680	7,720
October	12,400	13,340	7,550
November	4,680	4,550	5,840
December	350	140	120
1988 (Monthly totals of weekly dosimeters)			
January	1,600	2,800	1,050
February	1,010	3,100	830*
March	880	2,060*	700*

QUARTERLY DOSES MEASURED BY WRIST AND FINGER RING DOSIMETERS

	Right Wrist	Right Ring	Left Ring
First Quarter, 1987	9,340	4,200*	7,750
Second Quarter, 1987	9,310	2,930*	3,290*
Third Quarter, 1987	22,910	11,500	9,510
Fourth Quarter, 1987	17,430	18,030	13,510
First Quarter, 1988	3,490	7,960*	2,580*

All doses are in mrem.

*Indicates that dosimetry data is missing for this time period.

REPORT NO. 30-00763/88-001

ATTACHMENT 4

MEASUREMENTS OF MOCKUP OF INCIDENT

REPORT NO. 30-00763/88-001

ATTACHMENT 4

MEASUREMENTS OF MOCKUP OF EXPOSURE

The licensee used a phosphorous-32 source on a planchet covered by plastic to simulate the rabbit knee injected with the holmium-166. These isotopes have similar energy beta particles emitted during decay. (1.8 MeV maximum energy for holmium-166 and 1.7 MeV maximum energy for phosphorous-32.) The licensee measured the count rate with a Ludlum model 44-3 scintillation counter on contact with the plastic which simulated the rabbit's knee and at a distance equal to the thickness of the wrist film badge. In addition, the count rate at three inches was also measured. The NRC inspectors observed these measurements on April 21, 1988. The counts were measured on a scaler after a ten-second count.

Location	Count	Relative Count Rate
A	14,456	1.00
B	9,536	0.66
C	1,728	0.12

A: Simulated surface of rabbit knee
B: Simulated location of wrist at distance of thickness of wrist badge
C: Three inches from simulated rabbit knee

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406

JUL 18 1988

Docket No. 50-20

Massachusetts Institute of Technology
ATTN: Dr. John Bernard
Director of Reactor Operations
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

Subject: NRC Region I Inspection Report No. 50-20/88-02

This refers to the routine safety inspection conducted by Messrs. F. Crescenzo and T. Kim of this office on June 20-23, 1988, of activities authorized by NRC License No. R-37 and to the discussions of our findings held by Mr. Crescenzo with Mr. Bernard 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 inspectors.

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,

James T. Wiggins

James T. Wiggins, Chief
Reactor Projects Branch No. 3
Division of Reactor Projects

Enclosure:
NRC Region I Inspection Report No. 50-20/88-02

8808250390 App.

A/19

JUL 18 1988

cc w/encl:
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)

bcc w/encl:
Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
J. Wiggins, DRP
R. Blough, DRP
D. Haverkamp, DRP
L. Doerflein, DRP
R. Barkley, DRP
M. Kohl, DRP
T. Kim, RI - Pilgrim
F. Crescenzo, SRI - Shoreham

U. S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No.: 88-02

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: June 20-23, 1988

Inspectors: F. Crescenzo, Senior Resident Inspector - Shoreham
T. Kim, Resident Inspector - Pilgrim

Approved by: Laurence T. Doerflein for 7/15/88
A. Randy Blough, Chief Date
Reactor Projects Section No. 3B
Division of Reactor Projects

Summary: Inspection on June 20-23, 1988 (Report No. 50-20/88-02)

Areas Inspected: A routine unannounced on-site inspection of licensee activities including: Action taken on Previous Inspection Findings, Facility Operations, Requalification Training, Surveillance, Experiments, Logs and Records, Procedures, Refueling Activities, and Review and Audit Functions. The inspectors also conducted tours of the facility and performed observations of control room and reactor floor activities.

Results: No violations were identified. Two minor concerns were identified. One concern regarded general housekeeping of the facility and the other regarded a recent personnel change at the facility. The facility appears to be in conformance with applicable requirements.

8808010392 8pp.

DETAILS

1. Key Persons Contacted

* J. Bernard, Director of Reactor Operations
  L. Clark, Jr., Director of Reactor Operations (retired)
* K. Kwok, Superintendent, MIT Research Reactor Operations and Maintenance
  F. McWilliams, MIT Reactor Radiation Protection Officer
  O. Harling, Director, MIT Nuclear Reactor Laboratory

*Denotes those personnel present at the exit interview.

2. Previous Inspection Items

(Closed) Unresolved Item (86-01-01). This item identified a concern that the licensee was not adequately documenting the basis for changes made to the facility as required by 10 CFR 50.59. Specifically, the licensee had defined a category of facility changes, "Category B", which involved systems or procedures described in the Safety Analysis Report (SAR) but did not involve Unreviewed Safety Questions (URSQ). In some instances Category B changes were made and Safety Reviews were performed, but documentation to support the determination that no URSQ existed was not maintained. Since identification of this concern to the licensee, all Category B changes are accompanied by a documentation package which includes an URSQ determination. The inspector reviewed several selected documentation packages for Category B changes which were completed since identification of the concern. All were found to contain safety reviews which provided adequate documentation to support the licensee's determination that no URSQ existed. The inspector noted that although proper documentation was available to support USRQ determinations, the licensee had not changed the administrative procedures to reflect this new requirement. This concern was brought to the attention of the facility Superintendent. The Superintendent agreed with the inspector's concern and procedure PM 1.4.5, "Safety Review" was changed effective June 22, 1988 to reflect that all Safety Reviews must contain documentation to support the determination of URSQ or lack thereof. The inspector had no further questions. This item is closed.

(Closed) Unresolved Item (86-01-02). This item addressed a concern that personnel pocket dosimeters were not being calibrated. The licensee's internal Quality Assurance Audit had also identified the same concern. Currently, the licensee has a program to calibrate personnel pocket dosimeters on a quarterly basis using a radium standard at the National Bureau of Standards. Review of dosimetry records and calibration of instruments indicated that the program was implemented in March, 1986. The inspector had no further questions. This item is closed.

(Update) Violation (87-03-01). During a previous inspection conducted during December, 1987, a violation of NRC requirements was identified concerning the posting of airborne radioactivity areas within the Equipment Room of the facility. During tours of the facility conducted in conjunction with this inspection, the inspector noted that a "Caution: Airborne Radioactivity Area" sign had been properly posted at the entrance to the Equipment Room. This item will remain open pending further specialist inspector review.

3. Facility Tours

The inspector arrived at the facility on June 20, 1988 and conducted a meeting with the Director of Reactor Operations and the facility Superintendent regarding the scope and purpose of the inspection. A tour of the facility was conducted shortly thereafter during which general observations of security, health physics controls, and housekeeping were conducted. The inspector noted the above areas of observation to be adequate with the exception that general housekeeping of certain areas was poor. This was particularly evident in the Equipment Room where it was noted that several radioactive sample bottles and other sampling apparatus were left astray near the sample station. No violations of regulatory requirements were noted but it was suggested by the inspector that housekeeping be improved. Additional tours of the facility were made during subsequent days of the inspection. No inadequacies were noted.

4. Facility Operations Review

The facility is used primarily by the MIT faculty and the reactor staff to perform a wide variety of experiments. The reactor is routinely operated continuously from 8:00 a.m. Monday through Friday on a three shift schedule. During the inspection, the facility was in a six-day maintenance/modification period and as such, no critical operations of the facility were observed by the inspector. The inspector witnessed maintenance activities involving corrosion removal and repainting of the upper shield access ring. The activity was directed by a licensed operator and the health physics practices appeared adequate. The inspector reviewed training records and logs for heavy equipment handling associated with lifting the reactor lid and upper shield access ring. The inspector also reviewed the inspection records on the containment crane for 1980-1987. The containment crane is equipped with a 20-ton main hoist and a 3-ton auxiliary hoist. Annual inspection of the crane is performed by Northeastern Electric Co. No discrepancies were noted.

The inspectors independently observed control room activities and held discussions with the operators. Shift staffing requirements were met and activities in the control room were observed to be conducted in a responsible and professional manner. The inspector walked down the control room panels and boards with the on-shift operators. The operators appeared to be knowledgeable of the plant conditions and system operability status. The control room log, maintenance logs (including tagouts), on-the-job training log, and surveillance logs were reviewed and no discrepancies were noted. The control room operating procedures, technical specifications, emergency plan procedures, and startup reactivity graphs were of the latest revision and were readily available to the control room staff.

The facility organization, as described in the Technical Specifications, was reviewed. It was noted that the Director of Reactor Operations had recently retired (June 1, 1988) and had subsequently been replaced by the former facility Superintendent. The Superintendent position was then filled by the former Assistant Superintendent leaving that position vacant. The Assistant Superintendent position is not required by the facility Technical Specifications. The inspector noted that although the licensee had appropriately notified the NRC of the above personnel changes effective June 1, 1988, transition of responsibilities appeared incomplete. On one occasion, a question raised by the inspector regarding the tracking of 10CFR50.59 issues had to be referred to the retired Director of Operations. Also, the physical transition of personnel within appropriate office spaces had not been completed. The inspector characterized this concern as minor but urged the licensee to complete the transition in a timely manner. No other concerns were identified and the licensee appears to maintain an adequately staffed facility.

The inspector reviewed details relating to the most recent Reportable Occurrences which occurred at the facility. These were 50-20/1987-2, "Improper Reactor Reshim Causing an Excessive Power Rise," dated November 19, 1987, and 50-20/1988-1, "Detection of an Incipient Fuel Element Failure," dated May 26, 1988. The inspector discussed the details of each report with the facility Superintendent. No inadequacies were identified. The inspector had no further questions.

5. <u>Technical Specification Surveillance Review</u>

The inspector reviewed the following surveillance test procedures and records to verify that requirements specified in the Technical Specifications were met:

T. S. No.

6.1.1	Emergency Cooling System
6.1.2	Containment Tests
6.1.2.1	Containment Building Pressure Test
6.1.2.3	New/Repaired Penetration Leak Test
6.1.2.5	Charcoal Filter Efficiency Test
6.3.4	Fan Interlocks and Alarms
6.5.9.1	Area Radiation Monitor Calibration

The licensee performed the annual integrated containment leak rate test (6.1.2.1) on April 14-15, 1988. Two tests were conducted with different configurations. The first configuration measured the leakage through the auxiliary dampers, inner main personnel and truck lock doors, and inner basement personnel door. The building was pressurized to 50 inches of water and data was taken every 30 minutes for 8 hours. The second test was conducted with change over of dampers and airlocks. The second test failed to meet the acceptance criteria due to excessive leakage through the outer truck lock expansion joint. The licensee re-performed the test with the inner truck lock door closed. The results were within the acceptance criteria. The licensee stated that they are planning to repair the outer truck lock door in the Fall of 1988. Currently the inner truck lock door is tagged out of service preventing inadvertent opening of the door.

Local leak rate testing is only required between integral tests when new penetrations are made or repairs of existing penetrations are necessary. Three penetrations were repaired and tested since 1982 and the test results indicated measured leakage well below specified acceptance criteria.

The inspector had no further questions.

6. Experiments

Experiments at the MIT Reactor appear to fall under two broad categories. The first involves routine irradiation of various materials via beam ports, incore apparatus, and the thermal column. Included in this category are experiments which are designed to study atomic particles. The experiments are performed on a routine basis and are requested by both the MIT Nuclear Laboratory staff and other organizations not affiliated with the laboratory or with MIT. The inspector reviewed documentation associated with several of these type experiments and verified that reviews, approvals, and safety evaluations were properly conducted and documented in accordance with facility procedures. Predicted experiment parameters were compared with actual parameters and found to be within tolerance and training of experimenters was adequate. The inspector also verified that by-product materials were properly dispositioned in accordance with 10 CFR 30 requirements. The majority of the experiments involved exempt quantities or concentrations of by-product materials; however, in those instances which involved larger quantities or concentrations, the licensee was found to be in compliance with transfer requirements where applicable.

The second category of experiments involve larger, more complicated, research oriented projects. Currently there are three such experiments in progress or being planned for use at the facility. The first involves medical therapy use of the facility to treat human brain cancers (BNCT). The facility was originally designed with this experiment in mind and similar experiments were conducted with little success during the 1960's. Recent successes with this therapy at other facilities has rekindled interest and currently the licensee is in the preparation/planning stages of this experiment. The inspector did not review documentation relating to this experiment. The second major experiment involves automatic digital control of the reactor reactivity control systems. This experiment has been on-going and required a license amendment which was approved in 1985. The inspector verified that the licensee's activities relating to this experiment were in compliance with Technical Specifications. The inspector also reviewed, in detail, the approval package and safety review for one portion of the experiment which provided automatic digital control of a reactor shim blade. No inadequacies were identified. The last experiment of this category involves the simulation of commercial nuclear facility operating loops to develop strategies to minimize occupational radiation exposure resulting from maintenance activities. This will involve operation of a scale model loop within the reactor core region while operating, and its subsequent removal and dissection. Although the physical size of the loop will be scaled down, pressures, temperatures and flow velocity will be equal to that of the loop being simulated. The licensee is close to approving the installation of the first experiment of this type which will simulate a Westinghouse Pressurized Water Reactor Chemistry loop (PCCL). The licensee has determined that this experiment does not involve an URSQ or require a Technical Specification change. The licensee further intends to submit a description of the experiment to the NRC per 10CFR50.59 prior to its installation rather than wait to include it in the annual report to the NRC. The inspector reviewed the safety analysis for this experiment and found it to be designed within the boundaries of the Technical Specifications with the following exception:

MIT Technical Specification 6.1.3 requires that experiment materials placed in the core region which could react to cause a pressure spike be encapsulated within a capsule prototype tested to 2 times the maximum expected pressure spike. The maximum pressure resulting from a rupture of the PCCL is calculated to be 485 psi. The aluminum thimble which will be used to encapsulate the PCCL was described in the Safety Analysis as having a "Proof Pressure" of 750 psi which is significantly lower than the required 970 psi.

This concern was identified to facility management. Although this specification is included in a section of the Technical Specifications involving chemical effects of experiments, the licensee agreed with the inspector's position that this specification should also be applied to direct mechanical effects. The licensee agreed to re-address this issue prior to final approval of the experiment. This item is unresolved pending final licensee disposition (88-02-01). No other inadequacies were identified.

7. Review of Refueling Activities

The inspector reviewed the licensee's records and procedures relating to movement of fuel within the facility. Several specific fuel movements were reviewed in detail. These included refueling movements, core reshuffling movements, and spent fuel transfers from the core storage ring to the spent fuel pool. All movements and associated activities were extremely well documented. Reactivity calculations were performed prior to fuel movement and verified through shutdown margin (SDM) demonstrations performed subsequent to the activity. During larger fuel movements, intermediate SDM demonstrations were performed although not required by facility procedures or Technical Specifications. In all cases reviewed by the inspector, reactivity calculations deviated from actual measured conditions within acceptable tolerances. The inspector had no further questions.

8. Requalification Training

The inspector reviewed the records associated with the requalification of licensed operators at the facility. The requalification examinations administered in 1987 were reviewed and found to be of high quality. Grading of the examinations was adequate and it was noted that the grading is independently performed by the Director of Reactor Operations, the Superintendent, and the Assistant Superintendent. One reactor operator did not meet the facility standards for passing grade and was administered retraining and was re-examined. The licensee maintains an on-the-job training log to document accumulated time and activities conducted while performing licensed duties. The licensee uses this to determine the active/inactive status of licenses. The inspector reviewed this log and compared several entries with the control room operations log to verify accuracy. No discrepancies were noted. Several individual operator files were reviewed for adequacy. No inadequacies were noted. The inspector had no further questions.

9. Review and Audit Functions

The inspector reviewed the previous years "Administrative Quarterly Audits," the "Independent Annual Audit," and selected Quality Assurance Audits. The audits appeared to have been conducted in accordance with the facility procedures. No significant findings were identified by the licensee audits but numerous minor findings and observations of a constructive nature were identified. It appeared that these findings were acted upon by the staff. The inspector had no further questions.

As described in previous paragraphs, reviews were conducted by the inspectors of several experiments, procedure changes, activities and facility modifications. The inspector found the process by which the licensee conducts and documents safety reviews relating to these activities to be adequate.

The inspector also reviewed the licensee's annual report to the NRC to verify that the facility activities conducted pursuant to 10 CFR 50.59 were being properly identified. The inspector found that all such activities were reported but that in some instances it was not clearly evident from reading the report that some activities had been conducted pursuant to 10 CFR 50.59. For example, the report to the NRC dated August 29, 1987 contained a specific section for 10 CFR 50.59 Changes, Tests and Experiments which appropriately described several of the more significant activities but omitted many others. These other activities were described in the "Summary of Operating Experience" section of the report and were listed under subtitles (Experiments, Changes to Facility Design, Changes to Operating Procedures Related to Safety). The inspector discussed this with the licensee and suggested that the procedures be modified to ensure future annual reports explicitly identify activities conducted pursuant to 10 CFR 50.59. The inspector had no further questions.

10. Exit Interview

At the conclusion of the inspection on June 23, 1988 the inspector met with the Director of the facility and reviewed the scope and findings of the inspection. Other attendees are listed in paragraph 1.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
475 ALLENDALE ROAD
KING OF PRUSSIA, PENNSYLVANIA 19406

AUG 3 0 1988

Docket No. 50-20

Massachusetts Institute of Technology
Research Reactor
ATTN: Dr. John A. Bernard
Director of Reactor Operations
138 Albany Street
Cambridge, Massachusetts 02139

Gentlemen:

Subject: Inspection No. 50-20/88-03

This refers to the routine safety inspection conducted by Mr. C. Z. Gordon on July 18-20, 1988, of activities authorized by NRC License No. R-37. The inspection findings were discussed with you and other 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 emergency procedures and representative records, interviews with personnel, and observations by the inspector.

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

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

Sincerely,

Ronald R. Bellamy

Ronald R. Bellamy, Chief
Facilities Radiation Safety
and Safeguards Branch

Enclosure: NRC Region I Inspection Report 50-20/88-03

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

bcc w/encl:

Region I Docket Room (with concurrences)
Management Assistant, DRMA (w/o encl)
Robert J. Bores, DRSS
L. Doerflein, Project Inspector, DRP
R. Blough, Chief, RPS, 3B

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U.S. NUCLEAR REGULATORY COMMISSION
REGION I

Report No. 50-20/88-03

Docket No. 50-20

License No. R - 37 Priority C Category F

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

Facility Name: MIT Research Reactor (MITR - II)

Inspection Conducted: July 18-20, 1988

Inspector: Craig Z. Gordon 8/25/88
C. Z. Gordon, date
Emergency Preparedness Specialist

Approved by: Craig Conklin for 8/29/88
W. J. Lazarus, Chief date
Emergency Preparedness Section

Inspection Summary: Inspection on July 18-20, 1988 (Report No. 50-20/88-03)

Areas Inspected: Routine, announced emergency preparedness inspection conducted by one NRC Region I based inspector of the facility organization, operations, notification, communication, equipment, and training.

Results: No violations were identified. The Emergency Plan and Procedures were found to be implemented in a manner to adequately protect public health and safety.

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DETAILS

1.0 Persons Contacted

*J. A. Bernard, Director, Reactor Operations
G. R. Elderd, Sergeant, Campus Security
*G. R. Hopkins, Assistant Director, Reactor Operations
*K. S. Kwok, Plant Superintendent
E. F. Mallove, Assistant Director, News Office
*F. F. McWilliams, Radiation Protection Officer
J. P. Reilly, Radiation Protection Officer

*Denotes attendance at the exit meeting.

2.0 Massachusetts Institute of Technology Reactor (MITR-II) Emergency Plan

The inspector reviewed the MITR-II Emergency Plan for the Cambridge, Massachusetts site. The Plant Superintendent is responsible for updating and implementing the Plan and administering emergency preparedness program functions. The Plan was submitted to the NRC in August 1982, in response to changes in Emergency Planning requirements for test and research reactors. The Plan was developed in accordance with ANSI/ANS 15.16, draft II, dated November 1981 and the criteria of NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors". The Plan appears to meet the NUREG guidance with regard to information on emergency organization and responsibilities, radiological assessment, emergency action levels, designated emergency equipment and facilities, and training. Formal NRC Plan approval was issued in 1983.

Controlled distribution is limited to Plan holders within the reactor building. NRC and other support groups are provided with "unofficial" copies only. Controlled distribution of the Emergency Plan to the NRC (2 copies) and other groups who may be involved in emergency response including updates and revisions should be provided (50-20/88-03-01).

A description of different accidents and corresponding emergency action levels for each classification are provided in the areas of fuel damage, radiological effluents, natural phenomena, fire, and security threat. There are three (3) implementing procedures for corrective and protective actions: "Action 1X- General Emergency", Action 1Y- General Emergency", and Action 2Y- Event/ Alert/Site Area Emergency". Procedures 1X and 1Y for General Emergencies state that integrated offsite doses could exceed 1 Rem whole body and 5 Rem thyroid to offsite populations resulting from a 1 hour exposure and that protective action recommendations to local authorities may be necessary. In reviewing the Plan's definition of the Emergency Planning Zone (EPZ) however, the inspector noted that during design basis accidents, offsite doses are not expected to exceed 60 mR whole body or 1 Rem thyroid. In this regard, the Plan and Procedures are not consistent (50-20-88-03-02).

Further review of Procedures 1X and 1Y indicate that they contain complex narratives of information which interrelate key response actions and do not outline in an orderly manner specific tasks either to be performed by the Emergency Director or delegated to other response organization members (50-20/88-03-03).

3.0 Facilities And Equipment

The Control Room and Emergency Support Center (ESC) are the designated emergency response facilities. The inspector toured these as well as the assembly areas and noted that facilities appear to be well maintained. Radiation detection devices for emergency use are available from the reactor health physics group. Inventories of emergency equipment are performed on a regular basis. The inspector observed lockers in the Control Room and Reactor Building containing protective clothing, supplies for contamination control, respiratory protection equipment, radiation survey meters, decontamination supplies, and other necessary safety equipment used for emergency response and determined that sufficient equipment is generally available and that inventories were up to date. The inspector noted that self-reading dosimeters provided in Control Room lockers are capable of detecting exposures in the range of 0-5 Rem only, while lower range SRD's were unavailable (50-20/88-03-04).

A tour of the Control Room identified a licensee change in the method for determining emergency classifications. In order to classify emergencies due to operational problems, the licensee provided emergency action levels and resultant classifications based upon increases in readouts from the auxiliary core purge monitor. It appears that the licensee developed these EAL's because no other symptomatic means is available to provide operators with direct monitor readings which correlate with emergency classifications. The inspector reviewed the EAL changes and licensee's safety review and noted that the prescribed action levels are not specific initiating conditions (trigger points), which, if exceeded, will result in classification. Instead, such levels are monitor readings which must be sustained over time. In the case of the Unusual Event classification, the emergency cannot be classified until elevated auxiliary core purge monitor readings continue for a period of 24 hours. Although the basis for selecting each action level adequately relates to MPC values, overall benefit to the licensee's emergency response program cannot be determined (50-20/88-03-05). Other emergency action levels for classification are related to measured site boundary radiation levels, offsite dose calculations based upon multiple increments of measured stack area monitors (and MPC values of I-131), or increases in the gas and particulate monitors. These EAL's are more appropriate for research reactor licensees and allow the Emergency Plan to be immediately implemented.

4.0 Notifications and Communications

An incident may be reported at any time (24 hours) by a caller dialing telephone extension 911 and being connected to the Campus Police. The MIT-public telephone communications network used for emergency notification by the licensee consists of commercial telephones located throughout the reactor building. Portable radios are available for use by Campus Police and emergency response staff for environmental monitoring. The on-duty Shift Supervisor becomes the Emergency Director and assumes the lead role for overall direction and control of the emergency. This includes interfaces with MIT support groups, the radiation protection officer, and upper-level licensee management.

Primary telephones and an intercom system are located in the Control Room and Emergency Support Center to make initial notifications to the emergency organization. Telephone numbers for NRC notification are in place at the Control Room desk. Notification messages to NRC and other groups are required to contain information about the description of the event, emergency classification, expected or actual radiation release, meteorological data, dose assessment, and protective action recommendations. Although the notification and communication capability is adequate, the Plan does not provide for 15 minute notification to the State of Massachusetts and City of Cambridge after declaration of an emergency (50-20-88-03-06).

5.0 Coordination With Offsite Groups

The inspector reviewed Section 4.3 of the Emergency Plan, "Organizations Responsible for MITR Emergency Response" and contacted representatives of site support groups in the hospital and medical facility, news office, and Campus Police Department to determine each group's understanding of the role and responsibilities it will fulfill in response to emergency incidents in the reactor building.

Representatives stated that full support would be provided to emergency personnel during emergencies. The Plant Superintendent indicated that arrangements are in place for local governmental support from the City of Cambridge to coordinate and assist with most emergencies at the MIT site. Individuals also stated that they were familiar with basic radiological hazards associated with reactor operation and had previously attended site tours. Based upon discussions with these individuals, the inspector determined that adequate outside assistance is available to support MITR staff in dealing with emergency response activities in the reactor building.

6.0 Drills and Exercises

Shift supervisors and operations staff are designated for Emergency Director positions and receive specialized emergency training from the Plant Superintendent. Classroom instruction covers a review of EAL's, corrective actions, radiological controls, communications capability, and emergency implementing procedures. This training is included as part of the preparation for operator licensing examination. For requalification, licensed operators are required to review the Emergency Plan and implementing procedures annually.

Qualification criteria for key emergency response personnel consists of participation in emergency drills and exercises, acting as scenario evaluators, performing Emergency Plan reviews, and helping in scenario development and planning. The inspector noted that a comprehensive exercise which simultaneously tests the major portions of the Plan is not conducted, but evacuation, medical, security, and fire drills are held at least once per year. Exercises and drills are critiqued (initiated in 1988), documented, and results are discussed with the Reactor Safeguards Committee for possible corrective action.

7.0 Exit Meeting

The inspector met with the licensee representatives listed in Section 1 of this report at the conclusion of the inspection and summarized the observations made during the inspection.

The licensee was informed that previously identified findings were adequately addressed and no violations were found.

Licensee management acknowledged the findings and indicated that appropriate action would be considered.

At no time during this inspection did the inspector provide any written information to the licensee.

MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

BY

REACTOR STAFF

August 29, 1985

8510170165 26pp.

TABLE OF CONTENTS

MIT RESEARCH REACTOR

ANNUAL REPORT TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

Introduction

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

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

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

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

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

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

This is the tenth annual report required by the Technical Specifications, and it covers the period July 1, 1984 through June 30, 1985. Previous reports, along with the "MITR-II Startup Report" (Report No. MITNE-198, February 14, 1977) have covered the startup testing period and the transition to routine reactor operation. This report covers the eigth full year of routine reactor operation at the 5 MW licensed power level. It was another year in which the safety and reliability of reactor operation fully met the requirements of reactor users.

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

A. SUMMARY OF OPERATING EXPERIENCE

1. General

During the period covered by this report (July 1, 1984 - June 30, 1985), the MIT Research Reactor, MITR-II, was operated on a routine, five days per week schedule, normally at a nominal 5MW. It was the eighth full year of normal operation for MITR-II.

The reactor averaged 86.3 hours per week at full power compared to 90.3 hours per week for the previous year and 85.2 hours per week two years ago. The reactor is normally at power 90-100 hours/week, but holidays, major maintenance, long experiment changes, waste shipping, etc., reduce the average. The reactor routinely operates from late Monday afternoon until late Friday afternoon, with maintenance scheduled for Mondays and, as necessary, for Saturdays.

The reactor was operated throughout the year with 24 or 25 elements in the core. The remaining positions were occupied by irradiation facilities used for materials testing and the production of medical isotopes and/or by a solid aluminum dummy. Compensation for reactivity lost due to burnup was achieved through five refuelings of several elements each. These involved a continuation of the practice begun in previous years in which fresh fuel was introduced to the A and B-rings while partially spent elements that had been originally removed from the B-ring were gradually introduced to the C-ring to replace fully spent elements. These procedures were combined with many element rotations/inversions, the objective of which was to minimize the effects of radial/axial flux gradients and thus achieve higher average burnups.

The MITR-II fuel management program remains quite successful. All but seven of the original MITR-II elements (445 grams U-235) have been permanently discharged. The average overall burnup for the discharged elements was 42%. Of the remaining seven elements with the 445 gram loading, it is now projected that six will reach maximum depletion within the next six months. Thirty-six of the new elements (506 grams U-235) have been introduced to the core. Of them, three have attained the maximum allowed fission density. However, these may be reused if that limit is increased as would seem warranted based on metallurgical studies by DOE. As for the other thirty-three new elements, they are either currently in the reactor core or have been partially depleted and are awaiting reuse in the C-ring.

The continued delays in the availability of a licensed cask from DOE are of increasing concern. Specifically, our inability to ship spent fuel is forcing us to deviate from our normal fuel cycle in that:

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

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

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

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

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

2. Experiments

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

Experiments and irradiations of the following types were conducted:

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

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

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

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

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

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

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

h) Use of the facility for reactor operator training.

i) Irradiation damage studies of candidate fusion reactor materials.

j) Fault detection analysis of the output of control and process channels from the MIT Reactor as part of a study leading to control of reactors by use of fault-tolerant, digital computers.

k) Closed-loop direct digital control of reactor power using a shim blade as well as the regulating rod during some steady-state and transient conditions.

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

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

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

3. Changes to Facility Design

As indicated in past reports the uranium loading of MITR-II fuel has been increased from 29.7 grams of U-235 per plate and 445 grams per element to a nominal 34 and 510 grams respectively. With the exception of two elements (plus a third that was found to be out-gassing excessively in July 1985 after the end of the report period), performance has been good. (Please see Reportable Occurrence Reports Nos. 50-20/79-4, 50-20/83-2 and 50-20/85-2.) The heavier loading results in 41.2 w/o U in the core, based on 7% voids, and corresponds to the maximum loading in Advanced Test Reactor (ATR) fuel. The most recent fuel fabricator, Atomics International Division of Rockwell International, has completed the production of 41 of the more highly loaded elements, 36 of which have been used to some degree. Three with about 37% burnup, had been in operation in the core since January 1980 and were discharged during the year, since they had attained the burnup limit. Additional elements are now being fabricated by Babcock & Wilcox, Navy Nuclear Fuel Division.

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

Other changes in the facility are reported in Section E.

4. Changes in Performance Characteristics

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

5. Changes in Operating Procedures Related to Safety

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

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

Two SAR revisions were submitted during the year:

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

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

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

Table 1

Enclosure 1

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

SAR Revision No. 31

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

Table 2

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

SAR Revision No. 32

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

SR# 0-84-12

JUL 16 1984

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

6. <u>Surveillance Tests and Inspections</u>

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

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

During the reporting period, the surveillance frequency has been at least equal to that required by the Technical Specifications, and the results of tests and inspections were satisfactory throughout the year for Facility Operating License No. R-37. During the containment building pressure test in April 1985, the seal on the truck lock outer door could not be tested due to a hydraulic line failure. The line has since been repaired and preparations are being made for a special pressure test of the truck lock.

B. REACTOR OPERATION

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

	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
1. Energy Generated (MWD):					
a) MITR-II (MIT FY85) (normally at 4.9 MW)	182.1	190.4	225.8	195.0	793.2
b) MITR-II (MIT FY76-84)					6,970.5
c) MITR-I (MIT FY59-74)					10,435.2
d) Cumulative, MITR-I & MITR-II					18,198.9
2. Hours of Operation MIT FY1985, MITR-II					
a) At Power (>0.5 MW) for research	1161.9	1061.1	1148.7	1114.7	4,486.5
b) Low Power (<0.5 MW) for training(1) and test	48.1	33.6	36.2	39.0	156.9
c) Total critical	1210.0	1094.7	1184.9	1153.7	4,643.4

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

C. SHUTDOWN AND SCRAMS

During the period of this report there were 7 inadvertent scrams and 3 unscheduled power reductions.

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

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

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

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

D. MAJOR MAINTENANCE

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

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

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

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

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

Many other routine maintenance and preventive maintenance jobs were done throughout the year.

E. SECTION 50.59 CHANGES, TESTS AND EXPERIMENTS

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

The review and approval of changes in the facility and in the procedures as described in the SAR are documented in the MITR records by means of "Safety Review Forms". These have been paraphrased for this report and are identified on the following pages for ready reference if further information should be required with regard to any item. Pertinent pages in the SAR have been or are being revised to reflect these changes, and they will be forwarded to the Chief, Standardization and Special Projects Branch, Division of Licensing, USNRC.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

F. ENVIRONMENTAL SURVEYS

Environmental surveys, outside the facility, were performed using area monitors. The systems (located approximately in a ¼-mile radius from the reactor site) consist of calibrated G. M. detectors with associated electronics and recorders.

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

Site	July 1, 1984 - June 30, 1985
North	1.1 mR/year
South	3.6 mR/year
East	4.7 mR/year
West	1.0 mR/year
Green (East)	0.4 mR/year

Fiscal Yearly Averages

1978	1.9 mR/year
1979	1.5 mR/year
1980	1.9 mR/year
1981	1.9 mR/year
1982	2.5 mR/year
1983	2.3 mR/year
1984	2.1 mR/year
1985	2.2 mR/year

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

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

Total Personnel - 145   Total Man Rem = 9.95

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

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

1. Collection and analysis of air samples taken within the reactor containment shell, and in the exhaust-ventilation system.
2. Collection and analysis of water samples taken from the reactor cooling towers, D_2O system, waste storage tanks, shield coolant, heat exchangers, fuel storage facility, and the primary system.
3. Performance of radiation and contamination surveys, radioactive waste collection, calibration of reactor radiation monitoring systems, and servicing of radiation survey meters.
4. The providing of radiation protection services for control rod removal, spent-fuel element transfers, ion column removal, etc.

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

H. RADIOACTIVE EFFLUENTS

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

1. Liquid Waste

Liquid radioactive wastes generated at the facility are discharged only to the sanitary sewer serving the facility. There were three sources of such wastes during the year: the cooling tower blowdowns; the liquid waste storage tanks; and laboratory drains. All of the liquid volumes are measured, by far the largest being the 4,213,000 liters discharged during FY 1983 from the cooling towers. (Larger quantities of non-radioactive waste water are discharged to the sanitary sewer system by other parts of MIT, but no credit for such dilution is taken since the volume is not routinely measured.)

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

2. Gaseous Waste

Gaseous radioactivity is discharged to the atmosphere from the containment building exhaust stack and by evaporation from the cooling towers. All gaseous releases likewise were in accordance with the Technical Specifications and Part 20, and all nuclides were below the limits of 10 CFR 20.106 after the authorized dilution factor of 3000. Also, all were substantially below the limits of 10 CFR 20, Appendix B, Note 5, with the exception of argon-41, which is reported in the following Table H-1. The 4076 Ci of Ar-41 was released at an average concentration of 1.04×10^{-8} μCi/ml for the year. This represents 26% of MPC (4×10^{-8} μCi/ml) and is only about half the previous year's release. The improvement is the result of extensive studies of the sources generating Ar-41 in the reactor and of the inert gas blanket systems that minimize the generation.

3. Solid Waste

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

Table H-1

ARGON-41 STACK RELEASES

FISCAL YEAR 1985

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

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

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS FISCAL YEAR 1985

	UNITS	SHIPMENT #1	TOTAL
1. Solid waste packaged	Cubic Feet	120	120
2. Total activity (irradiated components, ion exchange resins, etc.) ^{60}Co, ^{51}Cr, $^{55-59}Fe$ ^{65}Zn, etc.	(Ci)	0.067	0.0[illegible]
3. (a) Dates of Shipment (b) Disposition to licensee for burial		3/26/85 Radiation Service Organization	

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Region #1
631 Park Avenue
King of Prussia, PA 19406

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

Dear Dr. Murley:

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

Sincerely,

Lincoln Clark, Jr.

Lincoln Clark, Jr.
Director of Reactor Operations

LC/gw
Enclosure: As stated

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

A/20

MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1985 - JUNE 30, 1986

BY

REACTOR STAFF

August 29, 1986

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TABLE OF CONTENTS

MIT RESEARCH REACTOR

ANNUAL REPORT TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1985 - JUNE 30, 1986

Introduction

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

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

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

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

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

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

This is the eleventh annual report required by the Technical Specifications, and it covers the period July 1, 1985 through June 30, 1986. Previous reports, along with the "MITR-II Startup Report" (Report No. MITNE-198, February 14, 1977) have covered the startup testing period and the transition to routine reactor operation. This report covers the ninth full year of routine reactor operation at the 5 MW licensed power level. It was another year in which the safety and reliability of reactor operation met the requirements of reactor users.

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

A. SUMMARY OF OPERATING EXPERIENCE

1. General

During the period covered by this report (July 1, 1985 - June 30, 1986), the MIT Research Reactor, MITR-II, was operated on a routine, five days per week schedule, normally at a nominal 5MW. It was the ninth full year of normal operation for MITR-II.

The reactor averaged 75.4 hours per week at full power compared to 86.3 hours per week for the previous year and 90.3 hours per week two years ago. The reactor is normally at power 90-100 hours/week, but holidays, major maintenance, long experiment changes, waste shipping, etc., reduce the average. During the past year it was reduced more than usual as the result of three week-long shutdowns for major maintenance activities (described later), two of them occurring at Christmas and New Year's. The reactor routinely operates from late Monday afternoon until late Friday afternoon, with maintenance scheduled for Mondays and, as necessary, for Saturdays.

The reactor was operated throughout the year with 24 or 25 elements in the core. The remaining positions were occupied by irradiation facilities used for materials testing and the production of medical isotopes and/or by a solid aluminum dummy. Compensation for reactivity lost due to burnup was achieved through five refuelings of several elements each. The first of these entailed the introduction of three fresh elements to the core's intermediate fuel ring (the B-ring). The others involved a continuation of the practice begun in previous years in which partially spent elements that had been originally removed from the B-ring were gradually introduced to the C-ring to replace fully spent elements. These procedures were combined with many element rotations/inversions, the objective of which was to minimize the effects of radial/axial flux gradients and thus achieve higher average burnups.

The MITR-II fuel management program remains quite successful. All but seven of the original MITR-II elements (445 grams U-235) have been permanently discharged. The average overall burnup for the discharged elements was 42%. Of the remaining seven elements with the 445 gram loading, six will reach maximum depletion within the next few months. Thirty-six of the new elements (506 grams U-235) have been introduced to the core. Of them, three have attained the maximum allowed fission density. However, these may be reused if that limit is increased as would seem warranted based on metallurgical studies by DOE. As for the other thirty-three new elements, they are either currently in the reactor core or have been partially depleted and are awaiting reuse in the C-ring.

The continued delays in the availability of a licensed cask from DOE are of increasing concern. Specifically, our inability to ship spent fuel is causing our total fuel inventory to approach our authorized possession limit and is forcing us to deviate from our normal fuel cycle in that:

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

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

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

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

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

2. Experiments

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

Experiments and irradiations of the following types were conducted:

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

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

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

d) Production of gold-198, dysprosium-165, and chlorine-38 for medical research, diagnostic and therapeutic purposes.

e) Irradiation (i) of tissue specimens on particle track detectors for plutonium radiobiology, and (ii) of geological samples for fissile element distribution.

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

g) Use of the facility for reactor operator training.

h) Irradiation damage studies of candidate fusion reactor materials.

i) Fault detection analysis of the output of control and process channels from the MIT Reactor as part of a study leading to control of reactors by use of fault-tolerant, digital computers. This effort recently resulted in the demonstration of techniques for reconfigurable control.

j) Closed-loop direct digital control of reactor power using a shim blade as well as the regulating rod during some steady-state and transient conditions. A new relation, the alternate dynamic period equation, was developed and used as the basis of a reactor controller.

k) Experimental studies of various closed-loop control techniques including rule-based control and the use of reactivity constraints.

l) Development and experimental evaluation of several new techniques for the measurement of reactivity.

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

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

3. Changes to Facility Design

As indicated in past reports the uranium loading of MITR-II fuel has been increased from 29.7 grams of U-235 per plate and 445 grams per element to a nominal 34 and 510 grams respectively. With the exception of three elements that were found to be out-gassing excessively performance has been good. (Please see Reportable Occurrence Reports Nos. 50-20/79-4, 50-20/83-2 and 50-20/85-2.) The heavier loading results in 41.2 w/o U in the core, based on 7% voids, and corresponds to the maximum loading in Advanced Test Reactor (ATR) fuel. The most recent fuel fabricator, Atomics International Division of Rockwell International, completed the production of 41 of the more highly loaded elements, 36 of which have been used to some degree. Three with about 37% burnup were in operation in the core starting in January 1980 and were discharged last year, since they had attained the burnup limit. Additional elements are now being fabricated by Babcock & Wilcox, Navy Nuclear Fuel Division.

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

Other changes in the facility are reported in Section E.

4. Changes in Performance Characteristics

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

5. Changes in Operating Procedures Related to Safety

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

MIT has received approval of its application for renewal of License No. SNM-986. This license covers kilogram quantities of slightly enriched U-235, normal and depleted U, and gram quantities of Pu. Other licenses covering smaller quantities of similar material have been combined with License SNM-986. The MIT Reactor is involved, because most of the SNM is stored on the reactor site, and much of it is used on the reactor in accordance with authorized experiment review and approval procedures.

Quality Assurance Program Approval for Radioactive Material Packages No. 164, Rev. 1, was renewed on June 20, 1986 (Rev. 3). This Approval is required for the shipment of (1) Type B quantities of radioactive material and (2) fissile material above exempt quantities. In order to renew the Approval, it was necessary to update SAR Chapter 11, "Quality Assurance Program", which supports Approval No. 164. Chapter 11 was changed as follows (SR #0-86-4):

1) References to paragraphs in 10 CFR, Part 71, "Packaging and Transportation of Radioactive Material, were updated to reflect changes in that Part since Chapter 11 was submitted five years earlier.

2) References to the Co-Director for Reactor Modification were deleted, since they are no longer applicable.

3) In paragraph 11.15, "Non-Conforming Material, Parts or Components", the granting of waivers from material or fabrication specifications is explicitly recognized.

4) Fig. 11.1-1 was updated to reflect the fact that the Machine Shop foreman now reports to the Director, Nuclear Reactor Laboratory.

5) Editorial improvements were made.

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

a) In order to satisfy the surveillance requirements established by the new Technical Specification #6.4 (Amendment No. 24 to Facility Operating License No. R-37, April 2, 1985), Procedure 6.7.1, "Shim Blade Digital Control System Surveillance", was prepared and approved. It provides for (1) the periodic measurement of differential and integral control blade worths, (2) functional check of the auxiliary period trip prior to use of the closed-loop digital control system for shim blades, (3) annual calibration of the digital system software's recording of blade position against the actual blade position, and (4) preparation of a quarterly procedure to check the speed limiter on a variable speed motor if one is installed and used on a shim blade as part of the digital control. (SR#0-85-15)

b) As the result of a recommendation contained in the independent audit of MITR operations and endorsed by the MITR Safeguards Committee, Procedure 1.14.3, "Equipment Tagout Procedure", was revised to incorporate a lockout procedure in a new "Equipment Tagout and Lockout Procedure". (SR#0-85-17)

c) The original conductance probe that provides a scram signal for a low level in the core tank was replaced by a float-type switch. A description of the switch was prepared for inclusion in section 6.5.1 of the Reactor Systems Manual, and a precedure was written for inclusion in PM 3.1.1.1.2, the reactor instrumentation startup checklist. (SR#0-85-18)

d) Procedure 7.1.5, "Damper Accumulators Charging and Actuator Inspection Procedure, Main Damper - Auxiliary Damper Inspection", was revised to require annual replacement of the hydraulic oil filter. This requirement was adopted following failure of the isolation valve to close on demand, as reported in Reportable Occurrence Report #85-3. (SR#0-85-19)

e) The "Schedule of Surveillance Tests and Calibrations", Procedure 7.3.1, was revised to include Procedure 6.5.13, "Shield Storage Tank Level Calibration" (established by SR#0-85-5 last year) and Procedure 6.7.1, "Shim Blade Digital Control System Surveillance" (item (a) above). (SR#0-85-21)

f) Procedure 4.4.4.16, "Instructions to the MIT Campus Police During MIT Reactor Radiological Emergencies", was revised to clarify the instructions (no substantive changes). (SR #0-85-22)

g) The emergency operating plans and procedures were updated to make explicit the requirement to identify the class of emergency and to delete the now redundant correlation in Table 4.7.3.4-1. In Plan Y, PM 4.4.4.14, the list of area occupants is updated. (SR#0-85-23)

h) Procedure 1.10, "Experiment Review and Approval", was revised to require explicitly the review of criticality considerations for in-reactor samples, and Procedure 1.18, "Audits", was revised to require an administrative audit of SNM criticality safety. (SR#0-86-1)

i) In Procedure 5.8.9, "Malfunction of a Shim Blade/Regulating Rod", the discussion of possible blade/rod malfunctions was clarified, and the erroneous operation of a blade drive motor was specifically listed as a reason for scramming the reactor along with a requirement to follow the approved restart procedure. (SR#0-86-2)

j) In Procedure 6.1.1, "Emergency Cooling System", the revalving necessary to check the operability of the system and to measure its discharge rate was clarified, and explicit precautions were added to preclude the possibility of flowing city waster into the core tank. (SR#0-86-3)

k) The Emergency Plan (PM 4.7.2), three Emergency Procedures (PM 4.4.4.14, PM 4.4.4.15 and PM 4.4.4.16) and an Abnormal Operating Procedure (PM 5.6.2) were revised to implement changes requested by the MIT Reactor Safeguards Committee at its December 1985 meeting (SR#0-86-6):

1) The effluent release criterion for determining action levels was clarified.

2) Cambridge Civil Authority titles were up-dated.

3) Operating instructions for the survey air sampler were clarified.

l) In accordance with a requirement of SNM License No. 986, as renewed on November 13, 1985, that the MIT Reactor Safeguards Committee membership collectively have the capability to review criticality safety including non-reactor applications, the MITRSC Charter was revised accordingly. (SR#0-86-7)

m) Procedure 2.7.1, "Receipt of Reactor Fuel", was clarified to make clear that the shipper of fuel is responsible for providing in-transit security. (SR#0-86-8)

n) The checklist used for Procedure 6.1.2.1, "Building Pressure Test", was revised so that both phases of the test were conducted with the truck lock inner door sealed. Both because of a suspected leak in the truck lock itself and because the outer door was therefore not tested, the inner door must be sealed during reactor operation until the leak is repaired and the lock and outer door tested. (QA#0-86-1)

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

6. Surveillance Tests and Inspections

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

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

During the reporting period, the surveillance frequency has been at least equal to that required by the Technical Specifications, and the results of tests and inspections were satisfactory throughout the year for Facility Operating License No. R-37. The truck lock is out of commission as of this writing due to a suspected leak and is isolated by the inner truck lock door from the reactor containment building.

B. REACTOR OPERATION

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

	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
1. Energy Generated (MWD):					
a) MITR-II (MIT FY86) (normally at 4.9 MW)	171.9	173.1	159.4	162.7	667.1
b) MITR-II (MIT FY76-85)					7,763.7
c) MITR-I (MIT FY59-74)					10,435.2
d) Cumulative, MITR-I & MITR-II					18,866.0
2. Hours of Operation MIT FY1986, MITR-II					
a) At Power (>0.5 MW) for research	1,021.9	932.9	978.2	985.7	3,918.7
b) Low Power (<0.5 MW) for training(1) and test	55.6	54.2	55.6	30.4	195.8
c) Total critical	1,077.5	987.1	1,033.8	1,016.1	4,114.5

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

C. SHUTDOWN AND SCRAMS

During the period of this report there were 14 inadvertent scrams and 13 unscheduled power reductions.

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

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

I. Nuclear Safety System Scrams	Total
a) Chan. 6 scram due to failure to raise trip before opening thermal column steel doors	1
b) Blade 6 dropped off while magnet current was being adjusted	1
c) Blade 3 dropped off while being calibrated due to dirty contacts on the period simulator receptacle	1
d) Withdraw permit open on relay failure	1
e) Withdraw permit open for no apparent cause	1
f) Dump valve relay failure	2
g) Low voltage on detector power supply	2
Subtotal	9

II. Process Systems Scrams	
a) High Temperature Reactor Outlet scram while operating too close to trip point	2
b) Failure of relay associated with secondary coolant pumps	1
c) Loose connection in secondary flow scram circuit	1
d) Simultaneous deflation of both gaskets on main personnel lock due to interlock malfuction	1
Subtotal	5

III. Unscheduled Shutdowns or Power Reductions

a)	Shutdowns due to Electric Company power loss	4
b)	Operator shut reactor down due to:	
i)	Imminent arrival of Hurricane Gloria	1
ii)	Failure of plenum gas #2 monitor	2
iii)	Reduction in compressed air pressure for main personnel lock	1
c)	Operator lowered power to investigate:	
i)	Low pressure in the helium supply at an irradiation thimble	2
ii)	Malfunction of plenum particulate monitor	1
iii)	Trip of secondary booster pump caused by defective controller contacts	1
d)	Operated at 2.5 MW to permit repair of broken drain line for cooling tower riser pipe	1
	Subtotal	13
	Total	27

The scrams and shutdowns during FY 86 compare with the 10, 19, 25, and 28 experienced in FY 85, FY 84, FY 83 and FY 82 respectively. The increase in FY 86 is due both to outside sources (power loses and hurricane) and to component aging. Relay failures have been a frequent cause, and selective replacements are now being made in an effort to reduce the number of such failures.

D. MAJOR MAINTENANCE

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

FY86 saw a continuation of the efforts in restoring the cooling towers closer to their original performance levels. The underground valves to the cooling tower basins had ceased to operate due to age. Replacement of these valves was initiated in FY85 and completed in FY86. There were a total of 2 eight inch, 2 six inch, and 2 three inch butterfly valves. New screens were fabricated and installed to cover openings at the bottom of the cooling towers and around the basins. The screens prevent debris and large objects from falling into the basins and fouling the system. The coupling on the shaft of cooling tower fan number one failed and was replaced. The sprinkler system for protection against fire on both cooling towers had developed an air leak. Due to old age, the entire system, including the piping and nozzles that are exposed to weather, was replaced. As a means of improving the heat rejection capability, both basins were drained and cleaned, the entire secondary system was drained, and all three main heat-exchangers were chemically flushed with hydrogen peroxide so as to remove the build-up of bio-deposits and other pollutants in the secondary system.

One of the main heat exchangers, HE-1A which developed a leak in FY85, was completely decoupled from the primary system and isolated on the secondary system side. Leak testing on the heat exchanger was initiated and is still in progress. Heat exchanger integrity is necessary to prevent potential fission products or other radioactivity from entering the secondary system.

The gaskets on the main intake damper, main exhaust damper, and the auxiliary exhaust damper were replaced as the result of conditions noted during a semi-annual damper inspection. The pillar block bearings on the main shaft of the exhaust fan developed noise and caused abnormal vibration. They were replaced. The intake fan motor was also replaced when it showed signs of bearing failure. The flexible expansion joints of the intake air plenum at both the suction and discharge of the intake fan had deteriorated and were replaced. An oil filter, similar to the one in the intake damper system for cleaning the hydraulic oil, was added to the exhaust damper hydraulic system so as to remove the debris which may have accumulated in the system over time and which might prevent closure of the damper, as happened with the intake damper (see Reportable Occurrence Report No. 85-3). The hydraulic system for the main personnel outer door was disassembled and rebuilt after seeing signs of sluggish performance during normal usage.

A compensated ion chamber (Channel 7 of the nuclear instrumentation) which is used for power level indication throughout the source range and the power range showed signs of breakdown at voltages above the operating value. The chamber was replaced with another three inch compensated ion chamber. The port plug for this chamber was redesigned to allow easy repositioning of the chamber within the shielding after installation. The new design also incorporates a lead shield in front of the chamber so as to compensate for some of the effects due to gamma radiation. Neutron shields were added to some of the beam ports on the reactor floor in an effort to reduce further the background level for the spectrometers used for physics experiments.

While performing the on-going preventive maintenance procedure on the control blade magnets, the magnet on control blade number 5 was identified as weak and was replaced. Preventive maintenance was also performed on the exterior of the containment shell. The paint on the the top section of the shell was mechanically removed and repainted with coats of red lead and the finishing paint. The wind speed and wind direction indicators were damaged in a hurricane during the year. The wind vane and the wind velocity indicator were replaced.

Many other routine maintenance and preventive maintenance jobs were done throughout the year.

E. SECTION 50.59 CHANGES, TESTS AND EXPERIMENTS

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

The review and approval of changes in the facility and in the procedures as described in the SAR are documented in the MITR records by means of "Safety Review Forms". These have been paraphrased for this report and are identified on the following pages for ready reference if further information should be required with regard to any item. Pertinent pages in the SAR have been or are being revised to reflect these changes, and they will be forwarded to the Chief, Standardization and Special Projects Branch, Division of Licensing, USNRC.

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

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

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

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

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

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

Initial tests of this digital closed-loop controller were conducted in 1983-1984 using the facility's regulating rod which was of relatively low reactivity worth (0.2% ΔK/K). Following the successful completion of these tests, a facility operating license amendment was submitted to NRC (January 11, 1985) that would:

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

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

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

A successful experimentation program is now in progress under the provisions of this license amendment. The test program that was originally performed using the regulating rod was repeated using a shim blade which was of considerably greater reactivity (1.4% $\Delta K/K$). Also a protocol has been developed in which this controller is used to monitor, and if necessary override, other novel controllers that are still in development. One series of tests is being conducted under the conditions of 10 CFR 50.59. It was found that the observation of controller performance could be maximized if the speed of the associated shim blade were reduced (i.e., made more conservative because the available rate of reactivity addition is reduced). Accordingly, the timing chain sprocket of one of the reactor's six shim blades was changed so that this one blade could only be moved at half-speed. This is a temporary change that will be removed once the test program is question is completed. It was concluded that no unreviewed safety questions were involved in this change.

F. ENVIRONMENTAL SURVEYS

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

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

Site	July 1, 1985 - June 30, 1986
North	0.4 mR/year
South	2.1 mR/year
East	6.0 mR/year
West	0.4 mR/year
Green (East)	0.2 mR/year

Fiscal Yearly Averages:

1978	1.9 mR/year
1979	1.5 mR/year
1980	1.9 mR/year
1981	1.9 mR/year
1982	2.5 mR/year
1933	2.3 mR/tear
1984	2.1 mR/year
1985	2.2 mR/year
1986	1.8 mR/year

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

Period 7/01/85 - 6/30/86

Whole Body Exposure Range (Rems)	No. of Personnel
No Measurable	73
Measurable - Exposure less than 0.1	68
0.1 - 0.25	6
0.25 - 0.5	10
0.5 - 0.75	2
0.75 - 1.0	2
Total Personnel - 161	Total Man Rem - 10.3

Summary of the results of radiation and contamination surveys from July 1985 to June 1986:

During the 1985-1986 period, the Reactor Radiation Protection Office continued to provide radiation protection services for full-power (5 megawatts) operation of the reactor. Such services (performed on a daily, weekly, or monthly schedule) include the following:

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

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

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

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

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

H. RADIOACTIVE EFFLUENTS

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

1. Liquid Waste

Liquid radioactive wastes generated at the facility are discharged only to the sanitary sewer serving the facility. There were three sources of such wastes during the year: the cooling tower blowdowns; the liquid waste storage tanks; and laboratory drains. All of the liquid volumes are measured, by far the largest being the 3,970,0C liters discharged during FY 1986 from the cooling towers. (Larger quantities of non-radioactive waste water are discharged to the sanitary sewer system by other parts of MIT, but no credit for such dilution is taken since the volume is not routinely measured.)

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

2. Gaseous Waste

Gaseous radioactivity is discharged to the atmosphere from the containment building exhaust stack and by evaporation from the cooling towers. All gaseous releases likewise were in accordance with the Technical Specifications and Part 20, and all nuclides were below the limits of 10 CFR 20.106 after the authorized dilution factor of 3000. Also, all were substantially below the limits of 10 CFR 20, Appendix B, Note 5, with the exception of argon-41, which is reported in the following Table H-1. The 3797 Ci of Ar-41 were released at an average concentration of 1.05×10^{-8} µCi/ml for the year. This represents 26% of MPC (4×10^{-8} µCi/ml) and is about the same as the previous year's release.

3. Solid Waste

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

TABLE H-1

ARGON-41 STACK RELEASES

FISCAL YEAR 1986

	Ar-41 Discharged (Curies)	Average Concentration(1) (μCi/ml)
July 1985	384	1.03×10^{-8}
August	314	1.05
September	337	1.19
October	477	1.28
November	306	1.03
December	274	1.23
January 1986	297	1.00
February	303	1.02
March	343	1.15
April	281	0.76
May	242	0.81
June	239	0.80
12 months	3797	1.05×10^{-8}
MPC (Table II, Column I)		4×10^{-8}
% MPC		26%

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

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS

FISCAL YEAR 1986

	Units	Shipment #1	Total
1. Solid waste packaged	Cubic Feet	75	75
2. Weight	Pounds	2336	2336
3. Total activity (irradiated components, ion exchange resins, etc.) ^{60}Co, ^{51}Cr, $^{55-59}Fe$ ^{65}Zn, etc.	(Ci)	0.097	0.097
4. (a) Dates of Shipment		5/22/86	
(b) Disposition to licensee for burial		Radiac	

NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

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L. CLARK, JR.
Director of Reactor Operations

August 29, 1986

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

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

Dear Dr. Murley:

Forwarded herewith are two (2) copies of the Annual Report for the MIT Research Reactor for the period July 1, 1985 to June 30 1986, in compliance with paragraph 7.13.5 of the Technical Specifications for Facility Operating License R-37.

Sincerely,

Lincoln Clark, Jr.

Lincoln Clark, Jr.
Director of Reactor Operations

LC/gw
Enclosure: As stated

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

A/23

MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1986 - JUNE 30, 1987

BY

REACTOR STAFF

August 29, 1987

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42521
282

TABLE OF CONTENTS

MIT RESEARCH REACTOR

ANNUAL REPORT TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

FOR THE PERIOD JULY 1, 1986 - JUNE 30, 1987

Introduction

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

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

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

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

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

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

This is the twelfth annual report required by the Technical Specifications, and it covers the period July 1, 1986 through June 30, 1987. Previous reports, along with the "MITR-II Startup Report" (Report No. MITNE-198, February 14, 1977) have covered the startup testing period and the transition to routine reactor operation. This report covers the tenth full year of routine reactor operation at the 5 MW licensed power level. It was another year in which the safety and reliability of reactor operation met the requirements of reactor users.

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

A. SUMMARY OF OPERATING EXPERIENCE

1. General

During the period covered by this report (July 1, 1986 - June 30, 1987), the MIT Research Reactor, MITR-II, was operated on a routine, five days per week schedule, normally at a nominal 5MW. It was the tenth full year of normal operation for MITR-II.

The reactor averaged 80.1 hours per week at full power compared to 75.4 hours per week for the previous year and 86.3 hours per week two years ago. The reactor is normally at power 90-100 hours/week, but holidays, major maintenance, long experiment changes, waste shipping, etc., reduce the average. During the past year it was reduced more than usual as the result of two week-long shutdowns for major maintenance activities (described later), one of them occurring at New Year's and one later in April. The reactor routinely operates from late Monday afternoon until late Friday afternoon, with maintenance scheduled for Mondays and, as necessary, for Saturdays.

The reactor was operated throughout the year with 25 elements in the core. The remaining positions were occupied by irradiation facilities used for materials testing and the production of medical isotopes and/or by a solid aluminum dummy. Compensation for reactivity lost due to burnup was achieved through seven refuelings of several elements each. The first of these entailed the introduction of three low burnup elements to the core's intermediate fuel ring (the B-ring). The others involved a continuation of the practice begun in previous years in which partially spent elements that had been originally removed from the B-ring were gradually introduced to the C-ring to replace fully spent elements. These procedures were combined with many element rotations/inversions, the objective of which was to minimize the effects of radial/axial flux gradients and thus achieve higher average burnups. An additional refueling was performed for removal of the one inch in-core facility so as to facilitate the design and installation of a new loop research project.

The MITR-II fuel management program remains quite successful. All of the original MITR-II elements (445 grams U-235) have been permanently discharged. The average overall burnup for the discharged elements was 42%. The maximum overall burnup achieved was 48%. Thirty-six of the newer, higher loaded elements (506 grams U-235) have been introduced to the core. Of them, three have attained the maximum allowed fission density. However, these may be reused if that limit is increased as would seem warranted based on metallurgical studies by DOE. As for the other thirty-three new elements, they are either currently in the reactor core or have been partially depleted and are awaiting reuse in the C-ring.

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

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

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

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

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

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

2. Experiments

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

Experiments and irradiations of the following types were conducted:

a) Neutron diffraction spectrometer alignment and studies (3 ports). In particular, the study of the use of pendellösung oscillations in the scattering of neutrons inside perfect crystals to greatly enhance the effect of spin-orbit contribution to high-energy neutron-nuclear scattering is being carried out by the neutron diffraction group.

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

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

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

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

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

g) Use of the facility for reactor operator training.

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

i) Fault detection analysis of the output of control and process channels from the MIT Reactor as part of a study leading to control of reactors by use of fault-tolerant, digital computers. This effort recently resulted in the demonstration of techniques for reconfigurable control.

j) Closed-loop direct digital control of reactor power using a shim blade as well as the regulating rod during some steady-state and transient conditions. A new relation, the alternate dynamic period equation, was developed and used as the basis of a reactor controller.

k) Experimental studies of various closed-loop control techniques including rule-based control and the use of reactivity constraints.

l) Development and experimental evaluation of several new techniques for the measurement of reactivity.

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

o) Detection of trace quantities of fissile nuclides in geological material using a delayed neutron detector.

Two research projects that will make major use of the reactor in the next and subsequent years have been funded and are in various stages of design and development. They did not actually make use of the reactor during the year, although reactor support services, e.g., electrical power supply, were augmented in preparation for installation of experiments on the reactor in the coming year. The first project is a dose reduction study for the light water reactor industry which will involve the installation of pressurized loops in the reactor core to investigate the chemistry of corrosion and the transport of radioactive crud with systems that simulate PWR's and BWR's. The second project is an extension of previous research to develop the boron neutron capture method of therapy for brain cancer (glioblastoma). This is a collaborative effort with the Tufts University New England Medical Center.

3. Changes to Facility Design

Except for minor changes reported in Section E, no changes in the facility design were made during the year. As indicated in past reports the uranium loading of MITR-II fuel was increased from 29.7 grams of U-235 per plate and 445 grams per element (as made by Gulf United Nuclear Fuels, Inc., New Haven, Connecticut) to a nominal 34 and 510 grams respectively (made by the Atomics International Division of Rockwell International, Canoga Park, California). With the exception of five elements that were found to be outgassing excessively, performance has been good. (Please see Reportable Occurrence Reports Nos. 50-20/79-4, 50-20/83-2, 50-20/85-2, 50-20/86-1, and 50-20/86-2.) The heavier loading results in 41.2 w/o U in the core, based on 7% voids, and corresponds to the maximum loading in Advanced Test Reactor (ATR) fuel. Atomics International completed the production of 41 of the more highly loaded elements in 1982, 36 of which have been used to some degree. Three with about 37% burnup were in operation in the core starting in January 1980 and were discharged in 1985, since they had attained the burnup limit. Additional elements are now being fabricated by Babcock & Wilcox, Navy Nuclear Fuel Division, Lynchburg, Virginia. Three of these have been received at MIT and are scheduled for use early in the coming year.

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

4. Changes in Performance Characteristics

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

5. Changes in Operating Procedures Related to Safety

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

Quality Assurance Program Approval for Radioactive Material Packages No. 164, Rev. 1, was renewed on June 20, 1986 (Rev. 3). This Approval is required for the shipment of (1) Type B quantities of radioactive material and (2) fissile material above exempt quantities. In order to renew the Approval, it was necessary to update SAR Chapter 11, "Quality Assurance Program", which supports Approval No. 164. The changes to Chapter 11 were minor and were described in last year's Annual Report. Chapter 11, as revised, was submitted to USNRC on April 7, 1987 as Safety Analysis Report Revision No. 33 for the purpose of updating that document.

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

a) Evacuation maps for the reactor office building (NW12) were updated and improved to include clearer and more direct evacuation paths as well as a description of the sound of evacuation signals. (SR #0-85-24)

b) In an effort to improve scheduling and bookkeeping of routine preventive maintenance, a set of thirteen checklists were established to cover the routine maintenance performed year round. The weekly checklist covers items that need to be checked frequently and the twelve monthly checklists cover items that need to be serviced at a longer but regular interval. (SR #0-86-10)

c) Procedure 4.4.5.1, "Instructions for Use of Utility Room Emergency Gauges", was revised to change the valve designation prefix from XTV to XV. This was done to simplify the valve numbers as well as matching those on the valve tags. (SR #0-86-12)

d) Procedures 4.4.4.3, "Reactor Fire", 5.7.8, "Smoke Detector System", and 6.6.2.2, "Self-Contained Breathing Devices", were revised to reflect recent changes to OSHA requirements. OSHA regulations (CPL2-2.20A) prohibit the use of the ten minute sling self-contained breathing devices for entry to a building. Accordingly, the storage location of the MITR's self-contained breathing devices were changed so that the thirty minute units are outside the reactor containment for reentry purposes and the ten minute units are kept in the containment for use in exiting. (SR #0-86-14)

e) A new combustible gas meter was purchased to replace an old unit which was used for detection of hydrogren buildup in both the medical water shutter system and the air space on top of the core tank under the lid (core purge), and deuterium buildup in the heavy water reflector system in the event of isolation in any of these systems. Procedures 6.5.20.1, "Calibration of Combustible Gas Meter", and 6.5.20.2, "Combustible Gas Meter Use", were modified to incorporate changes pertinent to the use of this new meter. Calibration and conversion charts were included in the procedures to preclude any ambiguity on interpretation of the meter readings. (SR #0-86-15)

f) The administrative procedures, Chapter 1 of the Procedure Manual, were revised to update the lists of names and committee memberships. This does not involve any change to the procedures. (SR #0-87-1)

g) Procedure 6.3.4, "Fan Interlocks and Alarms", was revised to include separate steps for opening the auxiliary dampers and stack base damper, and for starting the auxiliary fans and pneumatic blower. This change refined the existing procedure by making explicit several steps that had previously been implicitly assumed by the original procedure. (SR #0-87-2)

h) Procedures 3.1.1.1, "Full Power Startup Checklist - Mechanical", 3.1.1.2, "Full Power Startup Checklist - Instrumentation", 3.1.1.3, "Full Power Startup Checklist - Cooling Tower Operation", 3.2.1, "Shutdown from Operation at Power", 3.5, "Surveillance Check for Continuous Operation", were updated to incorporate changes to various systems. Mode of operation is changed from using all three heat-exchangers simultaneously to using two at a time. Provisions are incorporated to alternate the two on-line heat exchangers so as to preclude fouling resulting from stagnation of water. This new mode of operation provides better flow characteristics and standby assurance of a heat exchanger. (SR #0-87-3)

i) Procedures 1.16.2, "MITR Operations Qualification Program for Senior Operators/Shift Supervisors", and 1.16.3, "MITR Operations Qualification Program for Operators", were updated to reflect format changes of the page layout and correction of typographical errors. There were no changes to the procedure. (SR #0-87-4)

j) The graphite stringers in the graphite region of the reactor have been in use since startup of the original reactor, MITR-I. As a means of inspecting the conditions of the graphite reflector region and the outside surface of the reflector tank, three procedures were developed. The first procedure 7.6.1, "Special Procedure for Graphite Region Inspection", outlines the steps necessary for removal of a vertical irradiation facility (3GV2) in the graphite region, insertion of a periscope for visual inspection, acquisition of graphite specimens from the high flux region, acquisition of helium samples from the graphite region, and finally reassembly of the irradiation facility. The second procedure 7.6.2, "Measurement of Wigner Stored Energy in Graphite", outlines the steps necessary for determination of the Wigner stored energy, if any, in the specimens taken from the graphite reflector. The third procedure, 7.6.3, "Graphite Combustion Test", outlines a procedure for conducting combustion tests on samples of irradiated and unirradiated graphite. Precautionary information was incorporated whenever appropriate throughout these procedures. (SR # 0-87-5 and 0-87-6)

k) Procedures 3.2.4, "Response to Weekend Alarms", 3.7.1, "Weekly Security Checklist", 3.7.2, "Daily Security Checklist", were revised to reflect the installation of a remote surveillance system for the control room and remote indication panel for alarms and radiation levels. The daily and weekly security checklists were updated to implement new procedure to control access to the parking lot after normal business hours. (SR #0-87-7)

l) Procedures 5.6.1, "High Radiation Set-Up Area Vault", 5.6.4, "Trouble NW12 Gamma Monitor", were updated to reflect the location change of the reactor fuel vault which is no longer the small vault in the set-up area in NW12. (SR #0-87-8)

m) Nine sets of boron stainless steel control blades were procured in FY 1987. As part of the quality assurance program requirements, neutron transmission tests were necessary for final verification of the materials used for fabrication. Procedure 7.6.4, "Shim Blade Neutron Transmission Tests", was established for this purpose. This procedure represents a consolidation of all previously used procedures as documented by individual memos to the appropriate Q/A files. (SR #0-87-9)

n) The "Waste Storage Tank Dump Procedure", procedure 3.6, was revised (1) to add a step to require verification of the operability of the sewer monitor prior to use, (2) to add RRPO form 2005 which shows the details of the effluent calculation. This form had been in use since July 1985 and is incorporated here as part of the procedure. (SR #0-87-10)

o) Procedures 6.5.9.1, "Area Monitor Calibration Procedure", and 6.5.9.3, "Calibration Procedure for Fuel Vault Monitor", were revised to reflect the following changes: (1) the log N-16 monitor is no longer in use (because of the existence of the linear N-16 monitor which provides the same functions) and thus deleted from the procedure, (2) the new auxiliary core purge monitor has been installed and is added to the procedure, (3) the distances at which the calibration measurements are taken were changed to give better results. (SR #0-87-12)

p) The equipment room sump tank was replaced by a stainless unit. An "Equipment Room Sump Tank" procedure was established to outline the necessary steps in removing the old unit and installing and plumbing in the new one. (SR #M-86-1)

q) The checklist used for procedure 6.1.2.1, "Building Pressure Test", was revised to reflect the fact that the leak on the outer door of the truck lock had been repaired, and that both doors should be tested independently. (QA #0-87-1)

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

6. Surveillance Tests and Inspections

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

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

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

B. REACTOR OPERATION

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

	Quarter 1	Quarter 2	Quarter 3	Quarter 4	Total
1. Energy Generated (MWD):					
a) MITR-II (MIT FY87) (normally at 4.9 MW)	190.1	185.9	203.4	182.6	762.0
b) MITR-II (MIT FY76-86)					8,430.8
c) MITR-I (MIT FY59-74)					10,435.2
d) Cumulative, MITR-I & MITR-II					19,628.0
2. Hours of MITR-II Operation (MIT FY87)					
a) At Power (>0.5 MW) for research	1,097.7	1,000.5	1,087.7	977.0	4,162.9
b) Low Power (<0.5 MW) for training[1] and test	56.9	30.4	19.8	30.9	138.0
c) Total critical	1,154.6	1,030.9	1,107.5	1,007.9	4,300.9

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

C. SHUTDOWN AND SCRAMS

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

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

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

I. Nuclear Safety System Scrams	Total
a) Channel 3 noise resulting from technician failure to adequately secure chassis in rack	1
b) Channel 3 due to faulty coaxial connector	1
c) Channel 3 noise due to cold solder joint	1
d) Low voltage on detector power supply due to failure of the A.C. constant voltage power supply	1
Subtotal	4

II. Process System Scrams	
a) Primary pump circuit breaker opened on thermal overload due to a defective heater	2
b) High temperature reactor outlet scram due to operator inadvertently slamming closed the door of temperature indicator	1
c) High temperature reactor outlet scram due to technician investigating sticky recorder	1
d) Low flow secondary coolant due to operator tripping pump by mistake	1
e) Simultaneous deflation of both gaskets on main personnel lock due to trainee error	1
Subtotal	6

III. Unscheduled Shutdowns or Power Reductions

a) Shutdowns due to Electric Company power loss	5
b) Operator shut reactor down due to:	
i) High plenum particulate activity resulting from trainee failure to secure core purge filter housing adequately	1
ii) Low oil pressure in exhaust damper due to hydraulic pump failure	1
iii) Oil leak in intake damper	1
iv) Winds greater than 60 mph	1
c) Operator lowered power to investigate:	
i) Low pressure in the helium supply at an irradiation thimble	1
ii) Tripping of a cooling tower fan due to vibration switch	1
Subtotal	11
Total	21

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

Fiscal Year	Number
83	25
84	19
85	10
86	27
87	21

D. MAJOR MAINTENANCE

Major maintenance projects during FY87, including the effect, if any, on safe operation of the reactor are described in this section.

Many major maintenance items were performed in FY87 in anticipation of supporting the necessary requirements of the upcoming dose reduction project for light water reactors. In order to support the projected electrical loads for the loop experiments, a new three phase electrical penetration through the containment shell was installed and leak tested. Appropriately sized electrical cables were installed to bring the power from the penetration to the motor control center in the equipment room where the main switch gear is located. Additional cables were brought from this location to the reactor top where the instrumentation for the experiment is located. This new three phase system with a neutral ground is designed to operate at 480 V with a peak current of 400 A.

In addition to enhancing the high voltage electrical service capability, heat removal capability was also increased by installation of a new heat exchanger for experimental coolant. This heat exchanger is capable of rejecting 50 kW of heat at 120°F. This heat exchanger is located in the equipment room together with the other processing components. Piping was installed from the heat exchanger to the reactor floor where the out-of-pile experimental coolant tank will be located.

In FY87, nine sets of control blade absorbers and six electromagnets were fabricated in our own machine shop. These included the armature assembly and the pieces to form the connecting rod. All the installed control blade absorbers(6), magnets(6), and the regulating rod absorber(1) were replaced with new ones. The control rod drive mechanisms(7) were rebuilt and replaced. New calibration curves were generated with the standard calibration procedures. The old set of absorbers was in service since 1981 and had accumulated over 100,000 MWH of exposure.

The graphite region was last inspected during the modification in 1974 and 1975. A procedure and tools were developed to inspect the conditions of the graphite stringers in the graphite reflector. Access was gained by the removal of a graphite vertical facility (3GV2). A periscope was inserted into the graphite region and inspected both the graphite stringers in the vicinity and the surface of the reflector tank. Graphite samples were taken and tests were performed to determine their stored Wigner energy and combustibility. The results showed no signs of any stored energy in the graphite nor any ability to support combustion, even at temperatures in excess of 1200°C. The physical appearance of the graphite stringers was the same as when installed. A film presumably oxides, however, was found on the reflector tank. A subsequent visual inspection is scheduled to be performed early in the coming year.

The equipment room sump tank and pumps were replaced because of excessive corrosion accumulated over the years. The new sump tank was made of stainless steel so as to inhibit oxidation of the tank. The two old submersible pumps were replaced by a self-priming gear pump which has more than the combined capacity of the two submersible pumps.

The leak on the truck lock outer door was found and repaired. It passed the annual containment pressure test with adequate margin. An old PCB transformer, which developed a very small leak, was removed and disposed of in accordance with the environmental safety requirements. The transformer had been used for operating the helium refrigeration plant for the cryogenic system. This experiment has not been active for many years and will be dismantled; therefore, the transformer need not be replaced.

Two new electrodes for cathodic protection of the steel containment shell were installed to increase protection coverage near the truck lock and cooling tower area where the secondary pipes are located. These two additional electrodes replenish the decrease of electrical potential in the soil due to deterioration of the older units. The condensers in the air conditioning units for the containment building were completely flushed and cleaned. Good size leaks in the cooling coils in the intake plenum were found and repaired. This restored the capacities of these two units.

The leak checking on one of the main heat exchangers, HE-1A, is still in progress. Various methods of locating leaks in the approximately 1000 tubes have been tried. The helium leak detection method was found to be the most sensitive and is being used. One of the cooling tower booster pumps developed a leak on the shaft seal which was subsequently replaced.

To provide improved surveillance of reactor status prior to entry under emergency conditions, a TV camera system complete with pan and tilt capabilities was installed in the control room for remote monitoring the reactor instrumentation. The TV monitor and the movement controls for the camera are located in the operation office. A remote zoom lens allows reading of almost all indications in the control room from the operations office. An annunciator alarm on the alarm panel in the control room is actuated whenever the remote viewing system is turned on.

Facility security has been enhanced by the installation of additional closed circuit surveillance equipment.

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

E. SECTION 50.59 CHANGES, TESTS AND EXPERIMENTS

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

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

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

In recent annual reports, the only facility changes and experiments carried out under Section 50.59 were in connection with the digital closed-loop computer control project. No change made in FY87 in connection with this experiment involves an unreviewed safety question. The current status of this computer control project and the tests performed during the reporting period are as follows:

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

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

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

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

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

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

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

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

a) Completion of the tests in which one of the reactor's shim blades was moved at half-speed. (Note: The initiation of these tests was described in last year's report.) The timing-chain sprocket of the blade used for these tests has now been returned to its normal configuration.

b) Tests of a controller designed using state analysis methods.

In addition, the MIT Reactor Safeguards Committee approved use of reactivity constraints derived from the alternate dynamic period equation as satisfying the provisions of Technical Specification #6.4. This approval did not involve an unreviewed safety question because the alternate constraints are always bounded by the standard ones on which the reactivity constraint concept was originally based.

F. ENVIRONMENTAL SURVEYS

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

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

Site	July 1, 1986 - June 30, 1987
North	0.4 mR/year
South	0.7 mR/year
East	4.3 mR/year
West	0.3 mR/year
Green (East)	0.2 mR/year

Fiscal Yearly Averages:

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

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

Period 7/01/86 - 6/30/87

Whole Body Exposure Range (Rems)	No. of Personnel
No Measurable	31
Measurable - Exposure less than 0.1	91
0.1 - 0.25	9
0.25 - 0.5	7
0.5 - 0.75	6
0.75 - 1.0	1

Total Personnel = 145 Total Man Rem = 11.4

Summary of the results of radiation and contamination surveys from July 1986 to June 1987:

During the 1986-1987 period, the Reactor Radiation Protection Office continued to provide radiation protection services necessary for full-power (5 megawatt) operation of the reactor. Such services (performed on a daily, weekly, or monthly schedule) include, but are not limited to, the following:

1. Collection and analysis of air samples taken within the containment shell, and in the exhaust-ventilation system.
2. Collection and analysis of air samples taken from the cooling towers, D_2O system, waste storage tanks, shield coolant, heat exchangers, fuel storage facility, and the primary system.
3. Performance of radiation and contamination surveys, radioactive waste collection, calibration of reactor radiation monitoring systems, and servicing of radiation survey meters.
4. Providing of radiation protection services for control rod removal, spent-fuel element transfers, ion column removal, etc.

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

H. RADIOACTIVE EFFLUENTS

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

1. Liquid Waste

Liquid radioactive wastes generated at the facility are discharged only to the sanitary sewer serving the facility. There were two sources of such wastes during the year: the cooling tower blowdowns and the liquid waste storage tanks. All of the liquid volumes are measured, by far the largest being the 6,208,000 liters discharged during FY 1987 from the cooling towers. (Larger quantities of non-radioactive waste water are discharged to the sanitary sewer system by other parts of MIT, but no credit for such dilution is taken since the volume is not routinely measured.)

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

2. Gaseous Waste

Gaseous radioactivity is discharged to the atmosphere from the containment building exhaust stack and by evaporation from the cooling towers. All gaseous releases likewise were in accordance with the Technical Specifications and Part 20, and all nuclides were below the limits of 10 CFR 20.106 after the authorized dilution factor of 3000. Also, all were substantially below the limits of 10 CFR 20, Appendix B, Note 5, with the exception of argon-41, which is reported in the following Table H-1. The 4223 Ci of Ar-41 were released at an average concentration of 1.20×10^{-8} μCi/ml for the year. This represents 30% of MPC (4×10^{-8} μCi/ml) and is slightly more than the previous year's release of 3797 Ci. The increase is due to an imbalance in one component of the reactor building ventilation system that existed for a week during December 1986.

3. Solid Waste

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

TABLE H-1

ARGON-41 STACK RELEASES

FISCAL YEAR 1987

	Ar-41 Discharged (Curies)	Average Concentration(1) (μCi/ml)
July 1986	304	0.90×10^{-8}
August	320	1.16
September	203	0.75
October	329	0.96
November	313	1.13
December	653	2.40
January 1987	394	1.14
February	394	1.43
March	396	1.43
April	317	0.92
May	293	1.06
June	297	1.08
12 months	4223	1.20×10^{-8}
MPC (Table II, Column I)		4×10^{-8}
% MPC		30%

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

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS

FISCAL YEAR 1987

	Units	Shipment #1	Total
1. Solid waste packaged	Cubic Feet	112.5	112.5
2. Weight	Pounds	3489	3489
3. Total activity (irradiated components, ion exchange resins, etc.) ^{60}Co, ^{51}Cr, $^{55-59}Fe$ ^{65}Zn, etc.	Ci	0.082	0.082
4. (a) Date of shipment		06/09/87	
(b) Disposition to licensee for burial		U.S. Ecology, Inc.	

NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

138 Albany Street Cambridge, Mass. 02139
(617) 253-4211

L. CLARK, JR.
Director of Reactor Operations

August 29, 1987

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

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

Dear Sirs:

Forwarded herewith is the Annual Report for the MIT Research Reactor for the period July 1, 1986 to June 30 1987, in compliance with paragraph 7.13.5 of the Technical Specifications for Facility Operating License R-37.

Sincerely,

Kwan S. Kwok
Kwan S. Kwok
Assistant Superintendent,
Reactor Operations

Lincoln Clark, Jr.
Lincoln Clark, Jr.
Director of Reactor Operations

LC/ga
Enclosure: As stated

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

A/24

SUPPLEMENT TO THE

SAFETY EVALUATION REPORT

FOR

THE PWR COOLANT CHEMISTRY LOOP (PCCL)

MITNRL-020

to be

installed and operated in the

MITR

April 19, 1988

Prepared by:

M. R. Ames
M. J. Driscoll
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For Review by the
MITR Safeguards Committee
and by its
PCCL Subcommittee

TABLE OF CONTENTS

1. INTRODUCTION

A draft supplement to the PCCL Safety Evaluation Report (MITNRL-020, February 13, 1987) was issued in August 1987 and revised after review by the PCCL Subcommittee of the Reactor Safeguards Committee. Since that time, further design changes have been made, some of which were discussed at the Reactor Safeguards Committee meeting of December 10, 1987. This document replaces the Supplement of August 1987; it describes all changes made since the SER was issued and gives a complete description of the current design of the loop (which is nearing completion).

2. SUMMARY OF DESIGN CHANGES

a) Changes to Loop and Support Systems Design

Revised versions of Figure 1.1 a) and b) and Table 2.1 are provided. Important changes to note for portions of the loop inside the MITR-II Core tank:

--only one plenum (on the core inlet) is now provided in the loop tubing,

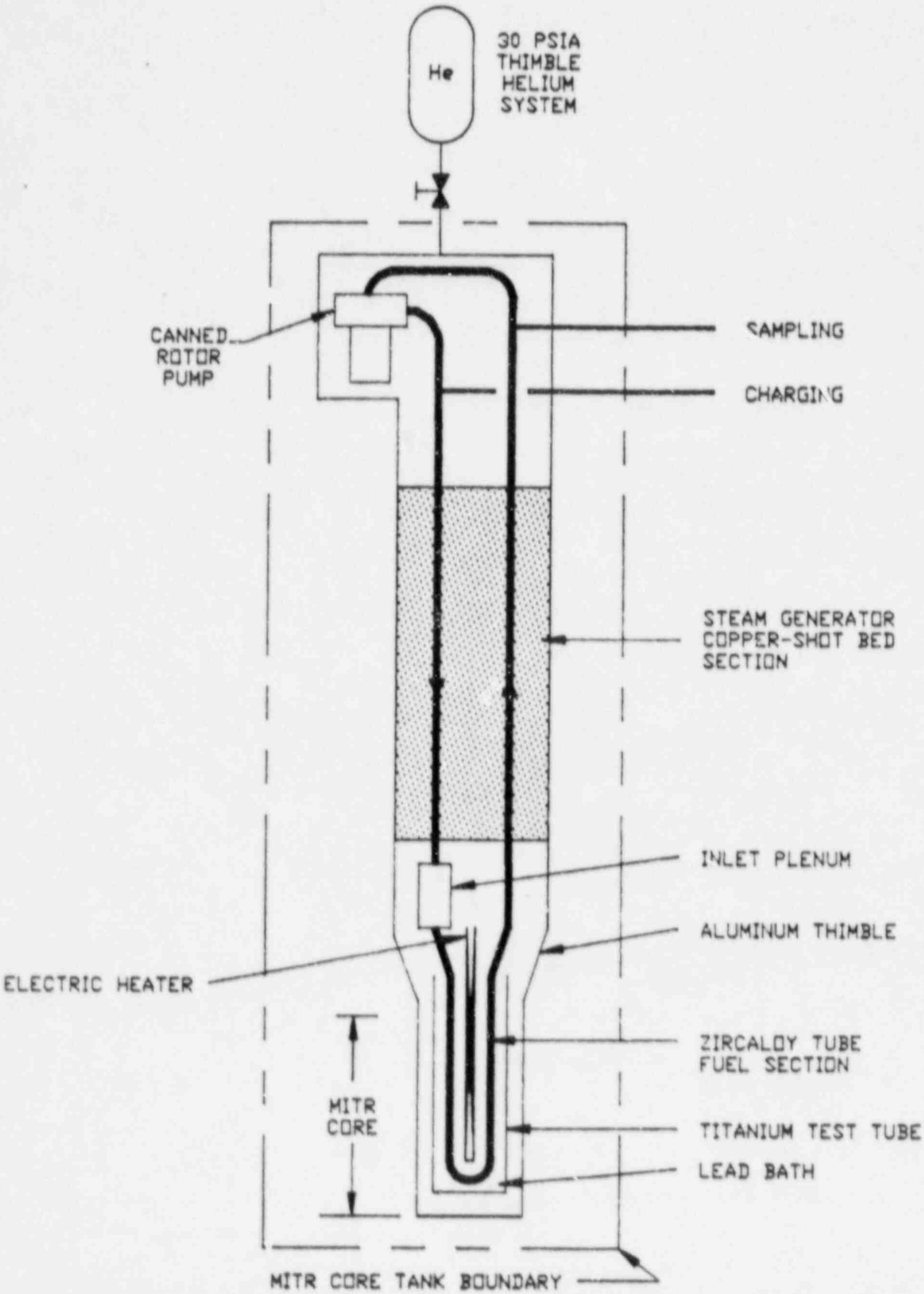
--a flowmeter will not be used; flow will be inferred from power and temperature data calibrated by out-of-pile testing (under normal circumstances, flow is directly related to the circulating pump input frequency, which is controlled),

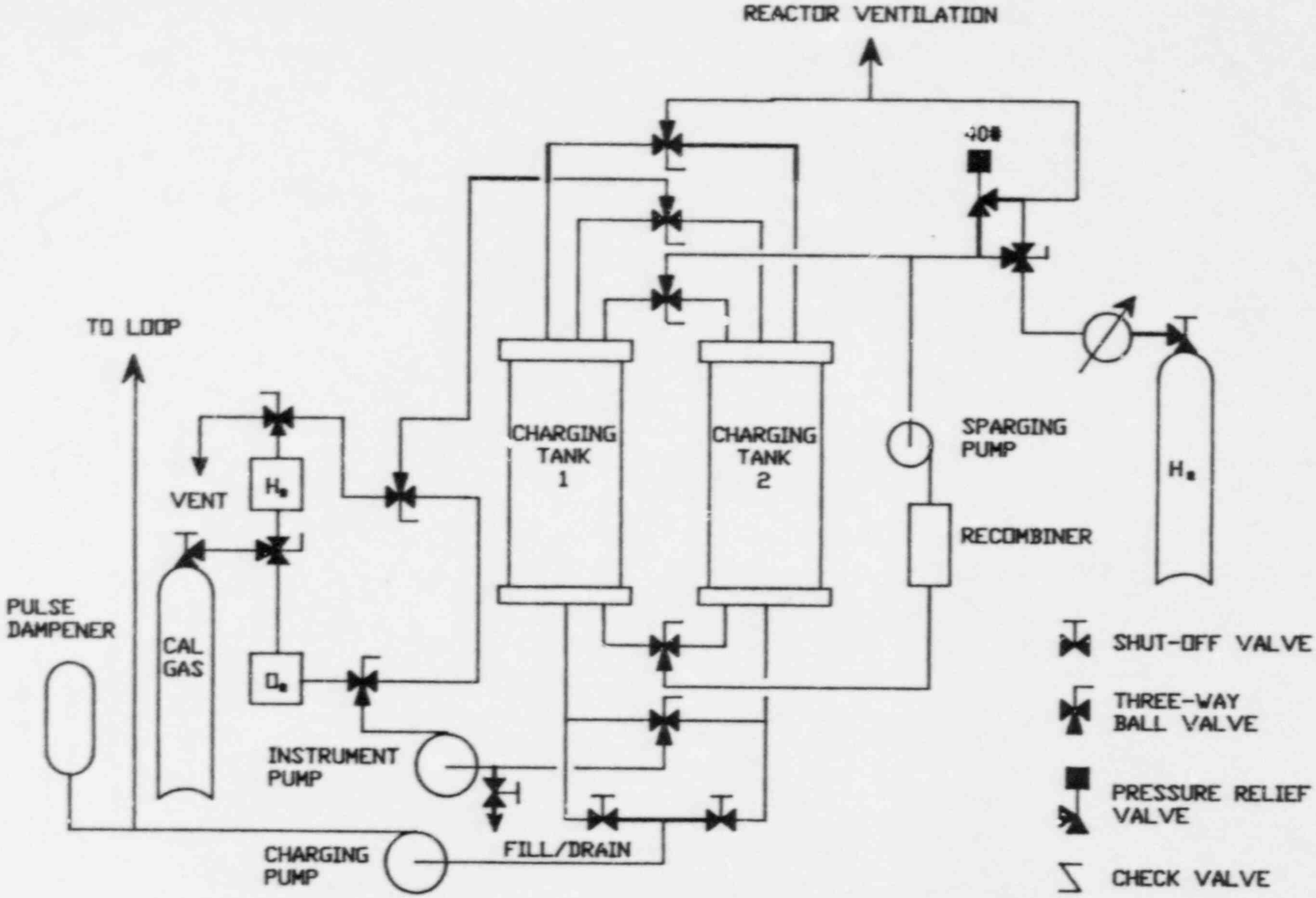
--the titanium lead bath container has been strengthened with weld beads to prevent deformation during operation at temperature,

--the aluminum fusible link has been eliminated and passive failure of the heater as a safety feature has been de-emphasized in favor of redundant trips without common mode failure possibilities,

--the most recent heater design was a two-element "U" configuration in place of the former single sheath, multi-pass heater, see Fig. S1 and further discussion below,

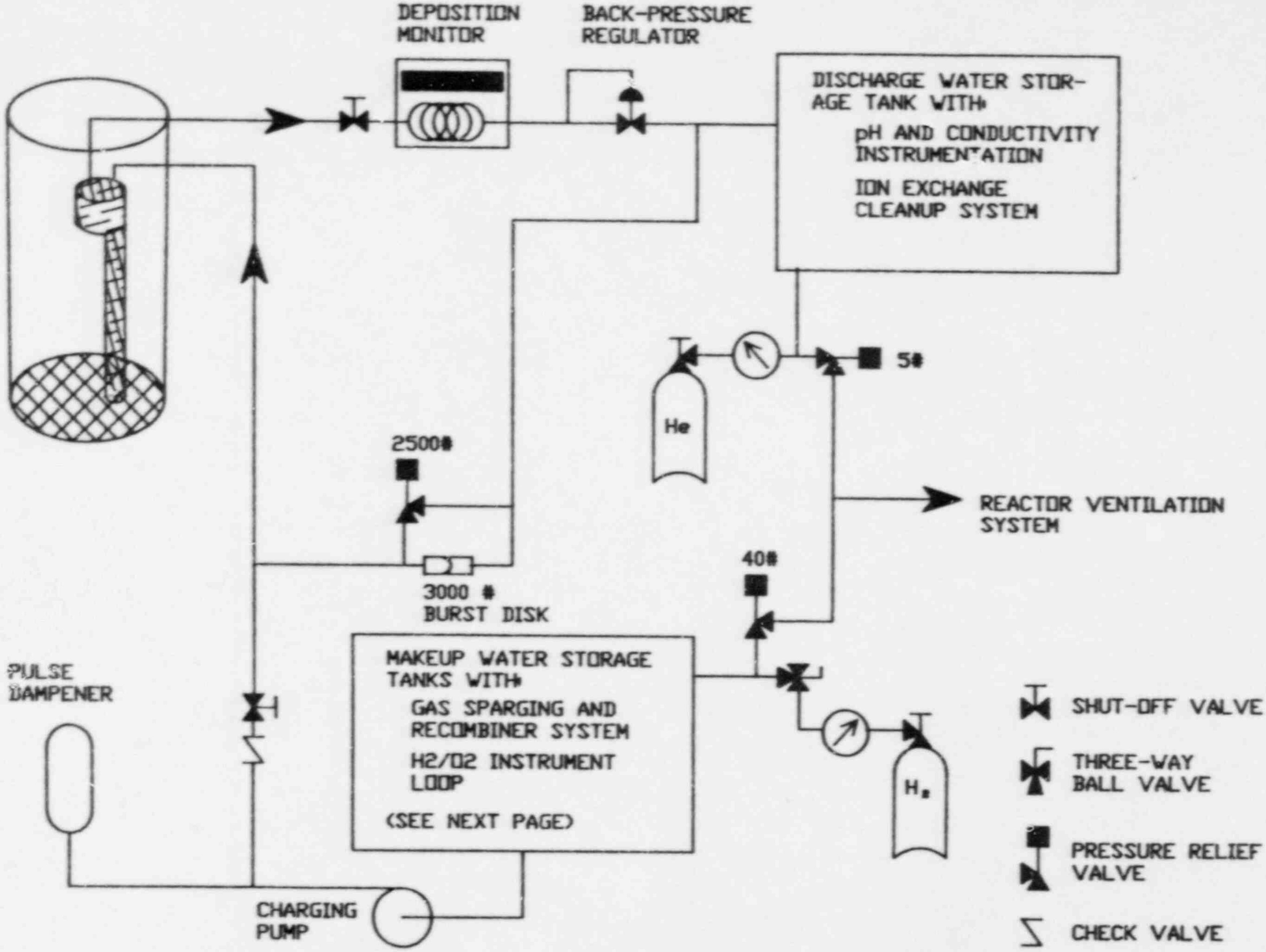
Figure 1.1a Schematic of PWR Coolant Chemistry Loop





CHARGING AND PRESSURIZATION SYSTEM

FIGURE 1.1.b-1



LOOP PRESSURIZATION AND CHEMISTRY CONTROL SYSTEM

FIGURE 1.1.b.-2

TABLE 2.1: PWR COOLANT CHEMISTRY LOOP DESIGN SPECIFICATIONS

IN-CORE SYSTEMS

Pump:

Capacity (GPM)	1-2
Design	Canned-rotor
Temperature (F/C)	603/317
Normal Operating Pressure (PSI/Bars)	2200/152
Maximum Design Pressure (PSI/Bars)	3000/207
Loop Differential Pressure (PSID/Bars)	15-25/1-1.7
Material	Inconel/Stainless Steel
Power Supply	220 VAC, 500 watts Variable Speed

Heater:

Power (variable)	0-20 kW
Power Distribution	Linear
Length (heated section)(in/cm)	21.5/55) X 2 segments
Diameter (in/cm)	0.440/1.12) X 2 segments
Sheath Material	Carbon Steel
Voltage (VAC)	275
Shutoff systems	- Automatic shutoff initiated by one of two independent thermocouple temperature signals. - Manual shutoff by experimenter/reactor operator under loop operating procedures.

Thimble:

Material	6061 Aluminum
Wall Thickness (in/mm)	0.125/3.2
Design Pressure (PSI/Bars)	30/2.1
Maximum Pressure- Loop Leak Accident	100 PSI (relief valve)/ <500 PSI (no relief)
Proof Pressure	750 PSI

(Simulated Fuel Pin)

Material	Zircaloy 2 or 4
Diameter OD (in/min)	0.312/7.9
ID (in/min)	0.26/6.6
Configuration	"U" Tube
Heated Length (approx)(in/cm)	50/127

Lead Bath Container:

Material	Titanium
Wall Thickness (in/mm)	0.032/0.79

Simulated Steam Generator Tube:

Shot-Bed Heat Transfer Medium	Copper Shot
Tubing	Inconel
Diameter OD (in/mm)	0.312/7.9
ID (in/mm)	0.26/6.6

Out-of-Core System:

Charging/Pressurization Pump:

Metering Pump	Positive Displacement
Maximum Flow Rate (cc/h)	2500 (Normal make-up flow rate 100-300 cc/h)
Maximum Pressure (PSI/Bars)	3000/207
Back-Pressure Valve	Spring Loaded
Check Valves	Dual Ball-Type to prevent back flow and depressurization

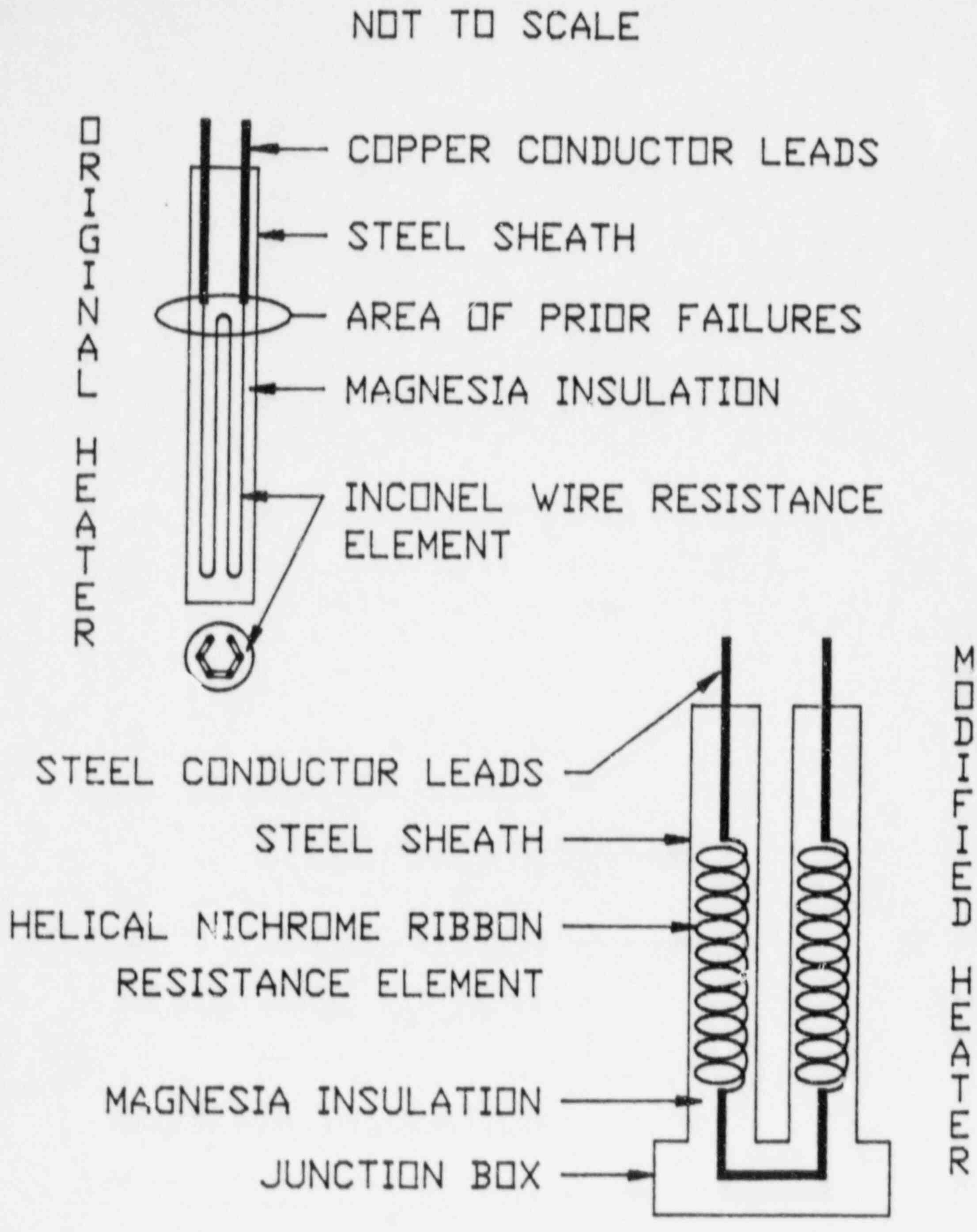


FIG. 51. SUMMARY OF HEATER MODIFICATIONS

--the burst disk on the thimble lid has been eliminated on the grounds that the redundant relief valves will provide adequate protection, since a hydrogen explosion has been determined to be extremely unlikely.

For the charging and discharge systems, significant changes include:

--the 3000 psig pressure relief valve has been replaced by a burst disk,

--two 1.75 $ft.^3$ pyrex tanks will be used for charging water storage to allow flexibility in run length and/or charging rate without interrupting charging flow to the loop; only one charging tank will be valved in at any time, except during manual tank changeover operations; the charging system has been fully tested in a 1000-hour-long run, supplying make-up water to prefilm a set of ten loop segments being prepared for later in-pile experiments,

--The back-pressure regulator has been moved to the discharge of the loop, replacing the let-down flow control capillary (see discussion below).

--since the charging pump incorporates check valves only one additional check valve is planned for the loop.

Adjustments to in-core loop component dimensions have been made during fabrication. These changes affect the reactivity and nuclear heating analyses and are discussed in Section 3.

b) Changes to Instrumentation and Control Design

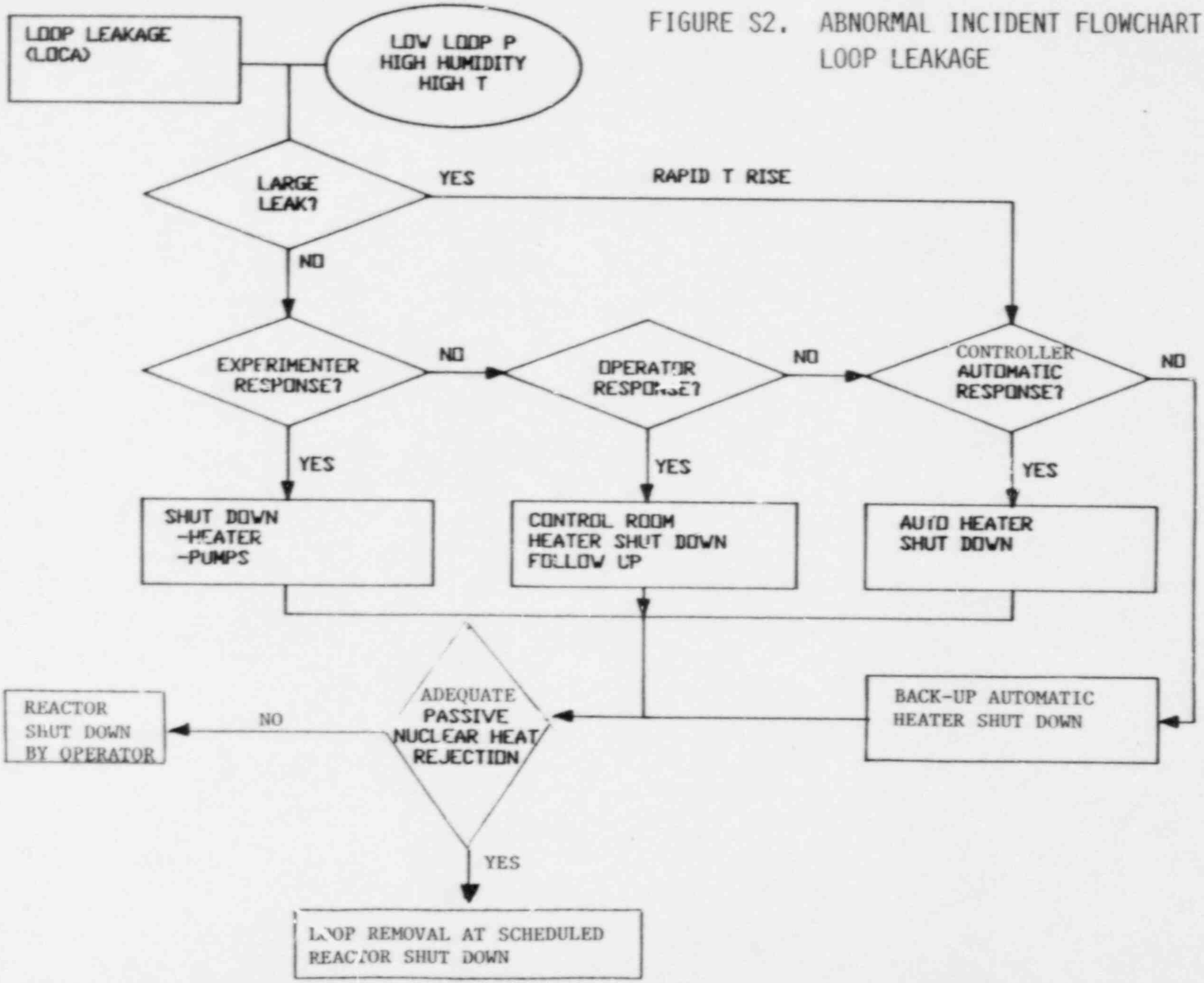
The basic philosophy of loop control and accident response remains the same as that outlined in the SER. It has been shown that all serious loop accidents eventually lead to overheating of the lead bath, and that consequences to the reactor are avoided if the electric heater is shut off. Therefore, the only automatic shut-down systems provided are triggered by high temperature in the lead bath. In order to clarify the sequence of events arising from potential accident scenarios, a sample flowchart (Fig. S2) which details the predicted pathway arising from a loop leakage accident is provided to replace Table 3.1 of the SER. Alarms at the experiment panel and in the control room are provided for abnormal conditions of temperature, loop pressure, thimble humidity and charging tank level. See Appendix 1 for a description of the alarm circuits.

3. DETAILED DISCUSSION OF IMPORTANT DESIGN CHANGES

a) Over-temperature Protection and Heater Design

The original SER and subsequent experimentation have established that nuclear heating of the in-core components can be safely rejected by conduction and radiation to the cooled thimble wall (see Section 4.b). Highly reliable shut-off of the electric heaters has therefore been a key element in the design of the loop. A "fusible link" of low melting point material was originally proposed to back up the thermocouple/relay system which is intended to shut off heater power on over-temperature. Experience with heater failures (eight) led to the conclusion that the heaters themselves would in all likelihood fail passively before Zircaloy temperatures reached the point where rapid Zircaloy-steam reaction takes place. It is still the belief of the PCCI group that the heaters will act in this manner. However, in light of the fact that heater design modifications are continuing in order to produce a heater that is capable of the service required, and the wide variety of possible abnormal conditions which can be postulated, it is difficult to prove this contention satisfactorily. Taking this into account, and following a suggestion made by the Safeguards Committee at its December 10, 1987 meeting, an additional, separate heater shut-off with its own thermocouple has been added. The active heater shut-offs now available are 1) a relay driven

FIGURE S2. ABNORMAL INCIDENT FLOWCHART - LOOP LEAKAGE



by a heater bath thermocouple which trips heater power at the controller of the SCR power supply and 2) a relay driven by another heater bath thermocouple which trips input power to the heater power supply. These automatic trips, together with high temperature alarms at the experiment control panel and in the control room, and the control room heater shut-off switch, should provide reliable heater shut-off without relying on passive failure. Note that this change has been incorporated into the abnormal occurrence flow chart of the previous section.

The current heater design is illustrated schematically in Fig. S1. It is expected that this heater will be considerably more rugged than previous ones: power per foot of length has been halved and the helical element is more compliant under thermal stresses. This heater is currently undergoing endurance testing and this design is likely to become the standard model for future applications.

b) Loop Pressure and Flow Regulation

The charging/let-down system as shown in the original SER relied upon a back-pressure regulator at the inlet of the loop to control loop pressure and a temperature controlled let-down capillary to control the let-down flow rate. In this design, most of the charging pump flow was bypassed back to the charging tank. Since dissolved hydrogen and oxygen measurement flow is now provided by an auxiliary low-pressure pump in a separate circuit, bypass flow is no longer experimentally necessary. Experience with operating a system for pre-filming the loop tubing led to the decision to place the back-pressure regulator at the outlet of the loop. Since the charging pump provides positive displacement flow at an adjustable rate, this arrangement provides an easily controllable, stable let-down flow rate without the complication introduced by use of a temperature controlled capillary.

The safety considerations associated with this change are related to loop behavior during heating and cooling and the response to charging pump failure. Heating and cooling the loop produces volume changes in the contained water. With the back-pressure regulator on the outlet, increased water volume during heating is automatically let down through the back-pressure regulator, and loop over-pressure leading to relief valve operation cannot occur (as it can in the let-down capillary case if the let-down capacity is exceeded by too rapid heating or by capillary obstruction during heating). Conversely, however, too rapid a temperature drop could lead to a volume reduction (≈70 cc/100°F for the loop inventory of ≈500 cc) in the loop which cannot be made up rapidly enough by the steady state charging flow, potentially leading to a drop in pressure and possible boiling. However, this effect is compensated for by the pulse dampener in the charging line, which consists of a gas volume (of ≈300 cc) pressurized to ≈70% of the working pressure, separated from the liquid stream by a heavy neoprene diaphragm. When the loop water volume decreases, the gas will expand preventing the pressure from dropping below the gas charge pressure until the volume capacity of the pulse dampener is exceeded - a highly unlikely circumstance. Preliminary calculations, which will be verified during loop shakedown testing, show that this mechanism will prevent boiling in the loop for any possible temperature drop. Furthermore, depressurization of the loop on charging pump failure occurs very slowly with the back-pressure regulator at the outlet, since outlet flow is driven by inlet flow. This change is therefore considered to enhance the safety of the loop, since depressurization will not occur during momentary power interruptions. The charging pump (and possibly the circulating pump) will be connected to the reactor emergency power supply system which should restore power within 30-60 seconds of loss of Cambridge Electric power.

4. REACTIVITY, POWER PEAKING AND NUCLEAR HEATING BASED ON AS-BUILT IN-CORE DIMENSIONS

The as-built dimensions of the titanium tube containing the lead bath, the electrical heater and the elliptical section of the aluminum thimble differ from the design values used in the SER. This results in different values for some of the potential reactivity effects, and for the nuclear heat which must be dissipated under LOCA conditions. Reactivity measurements have been made using actual loop components, described below. The new values do not change the conclusions which were made in the SER.

a) Reactivity and Power Peaking due to the PCCL

Appendix 2 contains the data which were collected during low power operation with various loop configurations, for reactivity worth and power peaking (using uranium foils). The data most relevant to the safety evaluation are those which indicate the effect of water flooding/voiding incidents (see Section 4.1.2 of the SER). For $\bar{\bar{\beta}}$ = .00786, the maximum reactivity change for:

PCCL tube flooding	0.042%Δk/k (measured)
Total in-core free volume flooding	0.14%Δk/k (measured)
Dummy/thimble channel reflood	0.17%Δk/k (computer evaluation based on measured data)

Note that these values are all within the <0.2% Δk/k limit for movable experiments, and well within the <0.5%Δk/k for nonsecured experiments, which is the controlling limit.

The actual power peaking limits will be evaluated and documented at the time of the installation for the core configuration that will exist at that time. Present studies indicated that the safety and operating limits as specified in the MITR-II technical specifications will be met.

b) Nuclear Heating of In-Core Components

In Appendix 1.a of the SER the total nuclear heating of the titanium lead bath can and its contents is estimated to be 7.2 kW, based on a total mass of 6.5 kg. The as-built value is 4.8 kg (lead bath, heater, Zircaloy U-tube, water, titanium can). However, a more conservative value for the core average heating rate is now being used. If an average value of 2 W/g (equal to the peak value measured in aluminum at reactor power of 5 MW) is used, the total power generated will be 9.6 kW. The true average value should be somewhat lower than this, and axial conduction will tend to reduce temperature peaking in the bath. However, the value of 9.6 kW is used in the discussion of passive cooling based on experimental results which are presented in the next section. Note that the actual value will be obtained by an energy balance on the in-core section when the experiment is first operated in the reactor.

5. SAFETY EXPERIMENT RESULTS AND OPERATIONAL EXPERIENCE

a) Heater Operating Experience and Passive Rejection of Nuclear Heating

As is evident from the flowcharts of postulated accident scenarios, the passive rejection of gamma and neutron heating when water flow through the loop tubing is interrupted is a key element in PCCL safety. In order to demonstrate that such rejection will occur at temperatures which do not result in damage to MITR-II core components, an experiment was conducted using an actual PCCL elliptical thimble section, titanium tube, lead bath, heater and Zircaloy U-tube. The thimble section was immersed in a tank of uncirculated water, and the gap between the titanium and aluminum was moderately well sealed so that it could be partially evacuated and back-filled with gas.

Data was obtained to determine heat transfer rates from the titanium can to the aluminum thimble. Table 5.1 gives steady state temperature data for the interior of the heater hot zone, the axial maximum heater sheath temperature and the axial maximum of the temperature inside the Zircaloy U-bend at various heater power levels. In the original SER, the analysis of passive cooling considered only radiative heat transfer. However, comparing the temperatures resulting when argon was used in place of helium in the titanium/aluminum gap shows that strong contributions are made by conduction, particularly for the helium fill case.

As discussed in the SER, the criterion for loop safety in a LOCA is that the Zircaloy tubing temperature does not exceed 2200 °F, above which Zircaloy/steam reactions can occur with large release of energy. This temperature limit must be met while dissipating the 9.6 kW of nuclear

Table 5.1: ZIRCALOY AND HEATER TEMPERATURES UNDER LOCA CONDITIONS

Heater Power (W)	Heater* Internal T (°F)	Maximum* Sheath T (°F)	Maximum Zircaloy T (°F)
2470	1243	873	860
3140	1370	960	873
3650	1489	1040	966
4060	1610	1112	1024
4510	1745	1179	1085

*Note that the data were obtained using an earlier heater design and therefore not relevant to the current case. The Zircaloy temperature, however, is still a significant parameter.

heating which is conservatively estimated to be generated at 5 MW reactor thermal power. The Zircaloy temperature measured in the experiment is linear with power from ≈ 3.0 to 4.5 kW (higher power is precluded to avoid damage to the heater). A conservative estimate of the temperature at 9.6 kW can therefore be made by linear extrapolation, essentially ignoring the contribution of T^4 radiant heat transfer which is expected to increase sharply at higher temperatures. Linear extrapolation gives a maximum Zircaloy temperature of 1845 °F, well below the 2200 °F limit. It has been noted that at such elevated temperatures if the Zircaloy were to remain pressurized it would be beyond its yield strength. If this should lead to rupture, however, it would place the loop in the LOCA situation discussed above.

b) Experience Relevant to Loop Electrical Safety

There is concern about the effect of possible shorts or accidental grounding of the heater electrical leads within the thimble. Protection in such a case consists of a 150 A semiconductor fuse in the heater power controller, connected on one leg of the power output. This is backed up by 100 A circuit breakers in the box which feeds the power controller. There are also 200 A fuses at the safety disconnect where the connection from the insulated pothead to the CCL heater bus is made. The aluminum thimble and the power controllers will be grounded to a heavy copper bus connected by 4/0 copper cable to the reactor electrical equipment ground bus. Since the cross section of the aluminum thimble is large and its conductivity is high, it could also act as an effective shield for MITR-II components in the event of electrical accidents.

Several incidents which have occurred during operation of the loop heaters confirm that the protection devices operate effectively in minimizing the consequences of grounding and shorting the heater power leads:

1) Short between a power lead and a grounded thermocouple wire - This accident resulted from inadequate insulation on the power lead and resulted in the vaporization of fractions of an inch of both the wires involved (#12 copper lead and small gauge chromel wire). A 50 A semiconductor fuse in the heater power supply blew rapidly and prevented any further damage.

2) First heater failure - In a heater failure resulting from overheating it is common for a short circuit to be produced by melting together of the heater wires at the overheating point. In this case the heater resistance was reduced from 4 Ω to 0.6 Ω, blowing a 150 A semiconductor fuse in the heater power controller. No damage to any wiring or to the heater sheath was observed.

3) Second heater failure - Again the heater resistance was reduced from 4 Ω to 0.6 Ω. In this case a 30 A fuse in the safety disconnect switch which was feeding the heater controller blew. Wiring and heater sheath damage was not observed.

Based on these experiences, it seems likely that the precautions taken against electrical accidents are adequate to protect the reactor.

c) Molten Lead Compatibility and Lead Bath Leak Testing

The SER discusses the possible consequences of a leak in the titanium can which contains the lead bath. Testing has been done to rule out the possibility of local boiling on the thimble surface in the event of a small lead leak producing a frozen "bridge" between the lead bath and the aluminum can wall. A 2 in. O.D. x 1/8 in. wall aluminum tube was immersed in a circulating water bath and locally heated using an insulated 175 W soldering iron connected by a drop of solder ≈ 1/8 in. in diameter. Temperatures produced at the contact point ranged as high as 750 °F without exceeding an exterior tube temperature opposite the contact point of 105 °F. Furthermore, the exterior did not rise more than 2° F above the adjacent water temperature. Since the postulated frozen lead bridge could not exceed the melting temperature of lead, 627 °F, it seems unlikely that local boiling could result from such an incident.

An experiment to determine the resistance to corrosive metal of various loop materials which will be exposed directly or indirectly to molten lead has also been carried out. An extract from a master's thesis by J. Wicks describing this work is provided in Appendix 3. No deleterious effects of the lead exposure for times relevant to our loop experiment were detected.

6. REQUIRED SAFETY PARAMETERS AND LIMITS

The SER and this Supplement represent the current best projection of loop design features, operating conditions and safety parameters. As experience is gained in operating the loops, new information will become available and some design and operating changes will most likely be necessary. Any significant changes will be reviewed by Reactor Operations and the Reactor Safeguards Committee or its Subcommittees as appropriate. In particular, the following safety features and limits will not be changed without prior approval from the Safeguards Committee:

1) Redundant high temperature alarms and trips will be provided for electric heater shutoff under abnormal conditions. These alarms and trips will be tested prior to each startup of the loop system in-core.

2, Pressure in the loop tubing will be limited to 2500-3000 psia by redundant pressure relief devices. Pressure in the thimble will be limited in the range 100-500 psia by redundant pressure relief devices. These devices will be tested periodically to verify their operation.

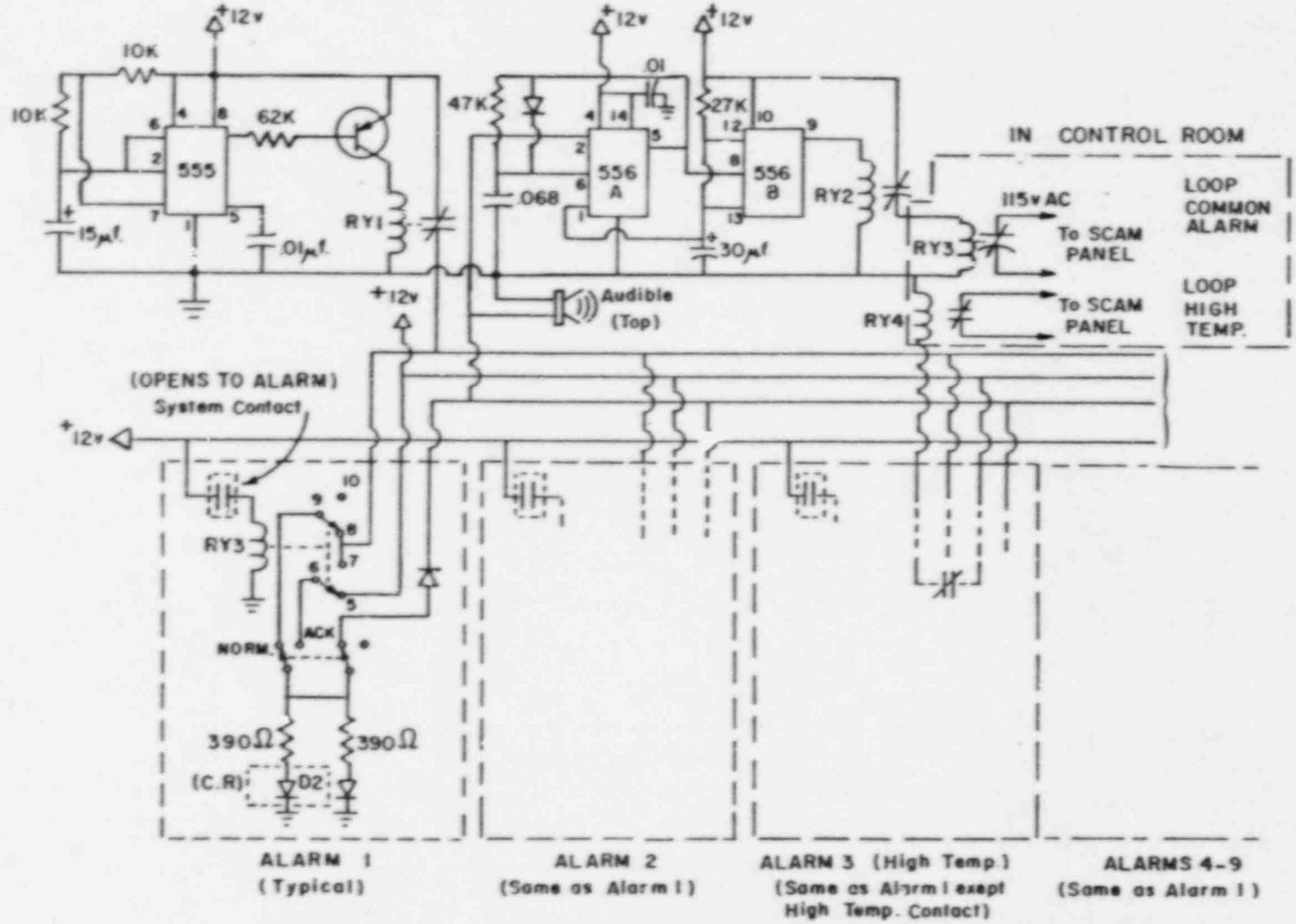
3) The hydrogen inventory in the containment building will be limited to 30 SCF, the combined maximum inventories of the transfer flask and the charging and discharge tanks.

APPENDICES

Appendix 1 - PCCL Alarm Circuit

Appendix 2 - PCCL Reactivity Measurement Results

Appendix 3 - Compatibility of Liquid Lead at 750 Degrees Fahrenheit with Zircaloy-2, Inconel, and 316 Stainless Steel



PCCL ALARM CIRCUIT

APPENDIX 1

NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

138 Albany Street Cambridge, Mass. 02139
(617) 253- 4211

L. CLARK, JR.
Director of Reactor Operations

January 28, 1988

Memorandum

To: Distribution

From: K. Kwok

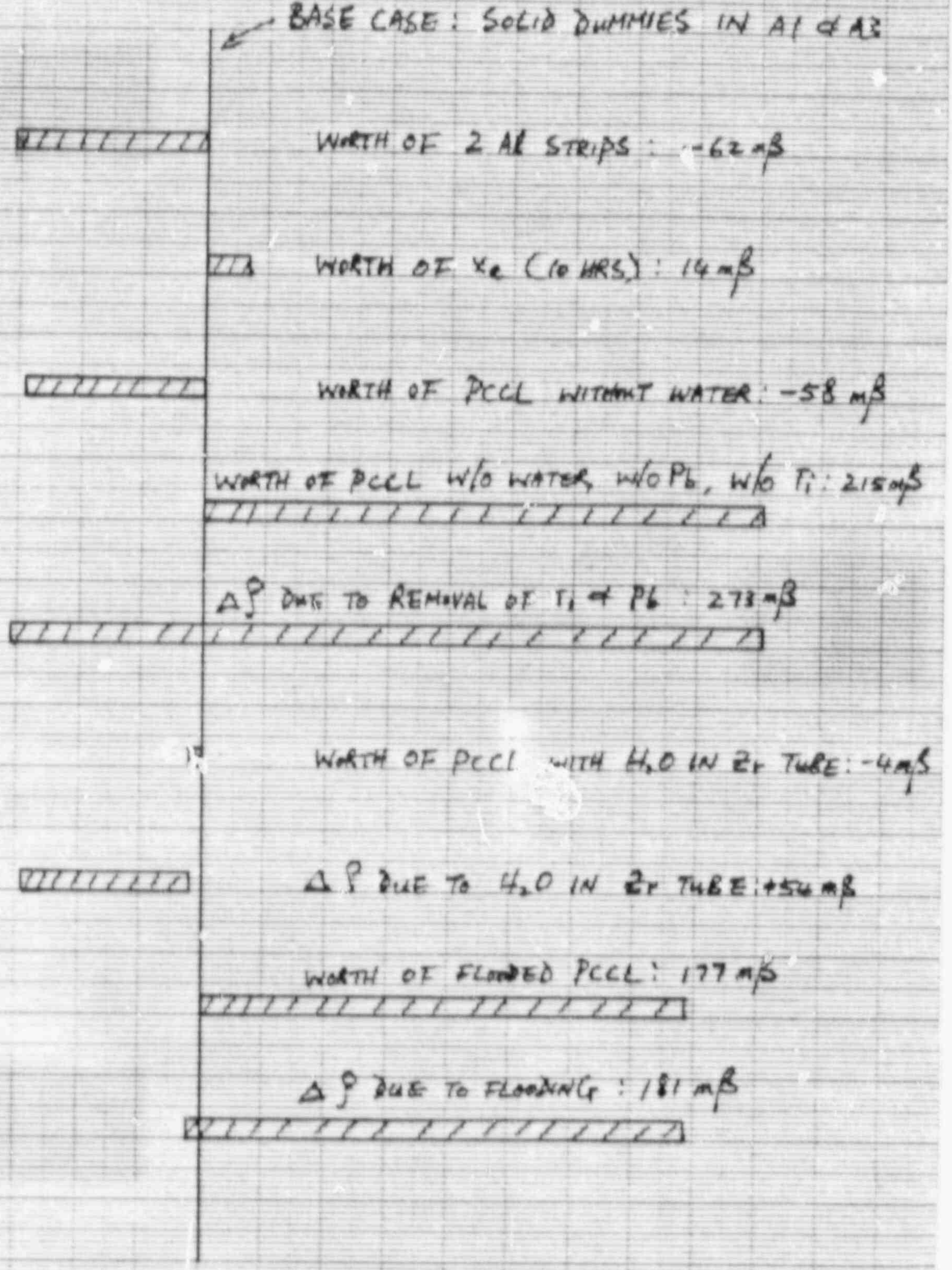
Subj: PCCL Reactivity Measurement Results - Revision 1

1. A series of 6 criticals were performed at 100 watts during the in-core trial fit procedure on 19 January 1988. The base case for these measurements was core configuration #87 with solid dummies in both core positions A1 and A3. The results are as follows:

    - Worth of two aluminum strips which held the Uranium foils: -62 mβ
    - Worth of Xenon change during the ten hour period for reactivity measurements: 14 mβ
    - Worth of PCCL without water: -58 mβ
    - Worth of PCCL witho water, titanium, and lead: 215 mβ
    - Change of worth due to removal of titanium and lead: 273 mβ
    - Worth of PCCL with water in the Zircaloy tube: -4 mβ
    - Change of worth due to addition of water in the Zircaloy tube: 54 mβ
    - Worth of PCCL with all available space flooded with water: 177 mβ
    - Change of worth due to flooding of all available space: 181 mβ

2. The aluminum strips are 0.04" thick and 0.83" wide. These give an in-core volume of 25 cm^3 for the two strips and a reactivity coefficient of 2.5 $m\beta/cm^3$. This is consistent with previously measured data. (Note: The active core length is 23".)

3. Attached is a bar diagram showing the above results.



PCCL REACTIVITY MEASUREMENT RESULTS 1/19/88

Appendix 3

Compatibility of Liquid Lead at 750 Degrees Fahrenheit with Zircaloy-2, Inconel, and 316 Stainless Steel

Zircaloy-2, Inconel alloy 600, austenitic 316 stainless steel and low-carbon content mild steel were tested for liquid-lead corrosion under conditions which were more severe than the Loop's normal operating conditions. Titanium was not investigated in this experiment. Reference L-1 thoroughly documents titanium's excellent resistance to liquid-lead in the range of temperatures of the MIT-PCCL. Additionally, the TTT is not pressurized, and its thin wall thickness (0.03125 inch, 0.79 mm) virtually eliminates thermal stresses across the TTT wall even with titanium's low thermal conductivity. The 316 stainless steel was examined because of the proximity of this metal to the liquid-lead bath. Inconel alloy 600 was examined as a quasi control. Opinions varied, and a literature search was inconclusive in eliminating a theory that the high nickel content of Inconel (76% Ni) would lead to intergranular cracking of the Inconel when exposed to the liquid-lead.

Zircaloy-2 was an alloy developed to improve the swell and creep characteristics of early nuclear fuel cladding. Zircaloy-2 was found to have excellent corrosion resistance in a steam environment. For this reason zircaloy-2 is used as the primary cladding material in Boiling Water Reactors (BWR). Unfortunately, zircaloy-2 was found to have a high affinity for monoatomic hydrogen, which formed an intermetallic compound of zirconium-hydride. The "zirc-hydride" is very brittle and contributes to brittle fracture of zircaloy cladding. The zircaloy-4 alloy has half the thermodynamic affinity for hydrogen and reduced levels of zirc-hydride formation. For this reason zircaloy-4 is the cladding of choice in todays Pressurized Water Reactors.

The alloys of zirconium have apparently not been tested for liquid-metal corrosion to any substantial extent. The following outlines the evolution of zircaloy cladding used in the nuclear industry:

Alloy	Composition
Zirconium	Pure Zr (used on the earliest reactors)
Zircaloy-1	2.5% Sn
Zircaloy-2	1.5% Sn, 0.15% Fe, 0.1% Cr, 0.05% Ni
Zircaloy-3	0.25% Sn, 0.25% Fe
Zircaloy-4	1.5% Sn, 0.21% Fe, 0.1% Cr

LIQUID LEAD COMPATIBILITY EXPERIMENT

Figure A.1 illustrates the set up of the compatibility experiment. The apparatus was set up in a closed cabinet having a ventilated exhaust hood to insure personnel safety. To simulate the thermal and mechanical stresses to which the

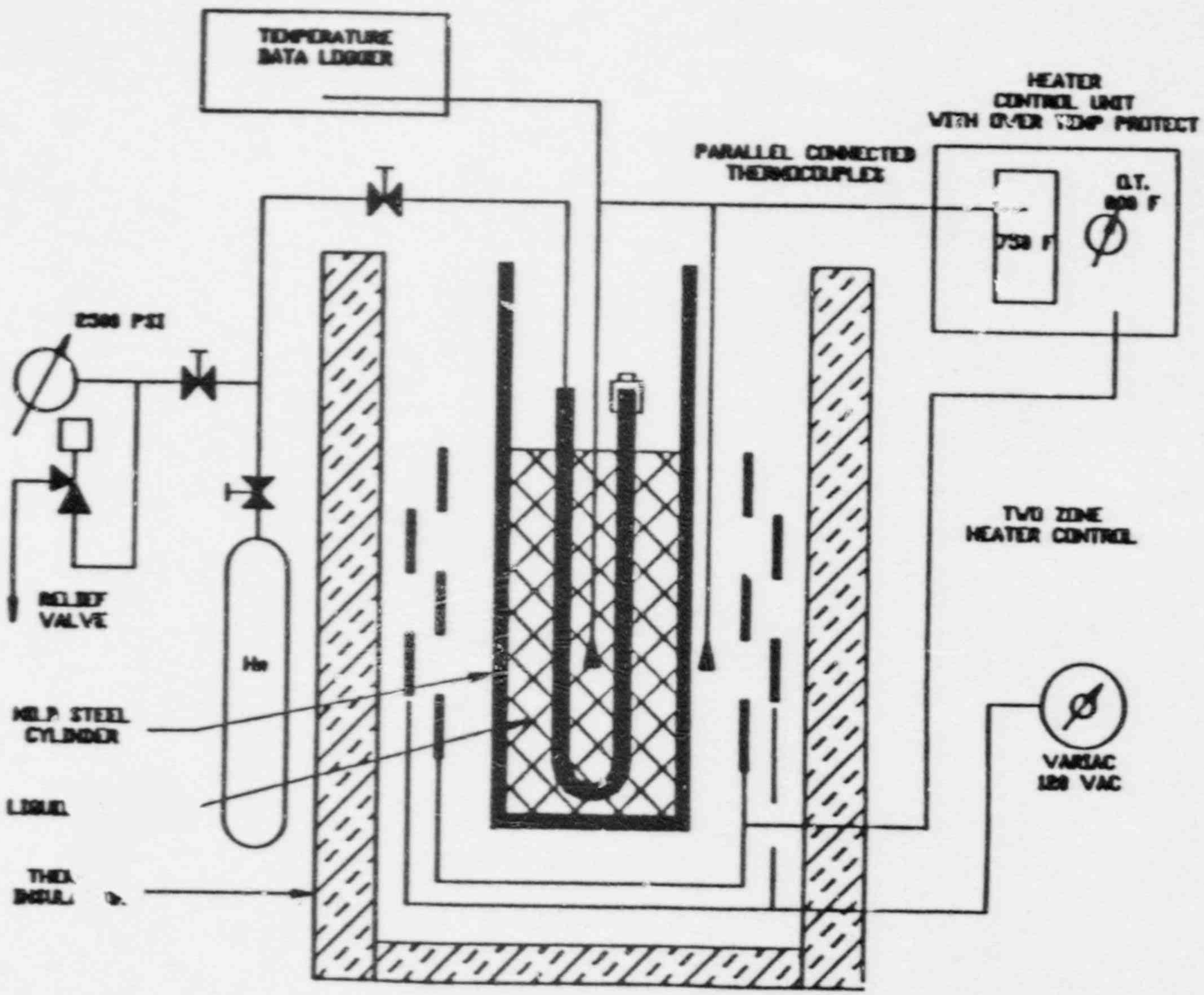


Figure A.1 Liquid Lead Compatibility Experiment tubing would be exposed, the tubes were bent into a "U" tube and internally pressurized to 2500 PSIG with helium gas. A temperature of 750 degrees F (398.89 C) was selected on the basis that it is very close to the actual maximum expected temperature of the lead bath, and the proximity of this temperature to available data from the Liquid-Metal Handbook. The pressurized tubing samples were exposed to the lead bath for 120 hours. Reagent grade lead powder was used for this experiment. In the opinion of Professor Ballinger of the MIT Nuclear Engineering Department, in the corrosion of materials by liquid metals, impurities may in fact play a major role in intergranular cracking corrosion. Table A.1 lists the percent impurities as taken from the lead manufacturer and as determined by neutron activation analysis.

Experimental Results:

Zircaloy 2, 316 stainless steel, and Inconel alloy 600 were immersed in the lead bath, pressurized to 2500 PSIG with helium, and maintained at 398 +/- 2 degrees C for 120 hours. The bent "U" tubes were removed from the bath hot in an attempt to limit the amount of lead clinging to the tube surface. In comparison to the stainless steel and Inconel, the surface of the Zr-2 was not wetted by the molten lead. There was no indication of cracking, preferential attack of the base metal, or a general liquid metal corrosion of the Zr-2 surface. The small amount of lead present on the

Table A.1

ANALYSIS OF LEAD PURITY

Maximum impurities and specifications from Manufacture		Impurities as determined by Neutron Activation Analysis
Lead	99.9%	99.9%
Antimony & Tin (as Sn)	approx 0.005%	
As	1 ppm	
Bi	5 ppm	5 ppm
Cu	3 ppm	
Fe	0.001%	
Ni	0.001%	
Ag	2 ppm	

surface of the Zr-2 tube was removed with a 50% solution of nitric acid. No additional information or indications were observed after removing the surface oxide layer with the acid. The results of the experiment on the 316 stainless steel and Inconel alloy 600 were consistent with the results for the Zr-2. The tubes were hydrostatically tested to 3000 PSIG prior to the experiment, and again following the experiment with no observable loss in tube wall strength.

This experiment was more aggressive than the actual loop application for the following reasons:

1. The lead bath was maintained at a higher temperature then expected in the loop
2. The pressure was maintained 300 PSIG higher then the normal operating pressure of the loop
3. The molten lead was exposed to an oxygen rich atmosphere instead of the Loop's helium atmosphere.

A one month long compatibility experiment was subsequently conducted on the zircaloy-2 tubing to verify the initial findings.

Conclusions:

Zircaloy 2, 316 stainless steel, Inconel alloy 600, and mild steel will be unaffected by the molten lead for the anticipated 2 month time that the loop internals will be exposed to the molten lead bath.

Documentation on liquid lead and its affect on engineering alloys is scarce. A fairly thorough search has been made of literature looking for answers to hear-say problems with liquid-lead. Reference L-2 is the only true handbook on the properties and corrosion of materials by liquid-metals. In our experiment, the concentration of Polonium, from the neutron activation of Bismuth, is of major concern because of the long half-life of the Polonium. References H-2 and B-5 provide some information on the process of removing bismuth and the importance of lead purity on liquid-lead corrosion.

The subject experiment is written up in more detail as a term project paper (MIT course 3.54 - Corrosion/ Professor Ronald M. Latanison - "Compatibility of Liquid Lead at 750 Degrees Fahrenheit with Zircaloy-2, Inconel, and 316 Stainless Steel"), a copy of which is in the PCCL project files.

When this experiment was performed, the project team did not have a sample of the zircaloy-4 tubing which is used in the construction of the first operational loop. It is the conjecture of the project team that the results of the zircaloy-2 compatibility experiment will accurately predict the compatibility of the zircaloy-4 tubing. This decision is based on the fact that the zircaloy-4 alloy does not contain any nickel, and the presence of nickel is believed to be a necessary ingredient in the susceptibility of alloys to liquid metal cracking.

Item: PWR Coolant Chemistry Loop (PCCL)

Submitted by L. Clark, Jr. Date April 19, 1988

Q/A number if required M-86-2

Does the item change or contradict the

Technical Specifications? ______ Yes* X No

SAR? ______ Yes* X No

* Attach explanation

Description of Change (Attach extra pages if necessary):

A pressurized coolant chemistry loop (PCCL) is to be installed in the MITR core. The PCCL is described and its safety evaluated in a Safety Evaluation Report (SER), MITNRL-020, dated February 13, 1987 and a Supplement dated April 19, 1988.

Safety Evaluation (Attach extra pages if necessary):

The MITR Staff's safety evaluation is contained in the attached pp. 1 - 11. It concurs with the PCCL Project Staff that operation and experimentation with the loop will fully satisfy the MITR-II Technical Specifications and that no unacceptable safety hazards will result.

Summary of Review:

a) Does the proposal:	Yes	No
i) involve an unreviewed safety question (10CFR50.59(a)(2))		X
ii) decrease scope of requalification program (10CFR50.54(i-1))		X
iii) decrease effectiveness of security plan (10CFR50.54(p))		X
iv) decrease effectiveness of emergency plan (10CFR50.54(q))		X

b) Reviewer's Comments: Recommend approval. MITRSC approval required.

RRPO F. McWilliams Date 20 April 88

Recommend Approval ✓ Yes ______ No

Reviewer [signature] Date 4/20/88

Reviewer [signature] Date 20 April 88

Approved L. Clark, Jr. Date 4/21/88
(Director of Reactor Operations)

10CFR50.59 & 50.54(p and q) changes logged for reporting to NRC, Date Reported by letter 4/21/88

Copy to Director for Operations
Copies circulated to and initialled by all Licensed Personnel
Original to Safety Review File

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L. CLARK, JR.
Director of Reactor Operations

April 19, 1988

Safety Review #-0-86-9: PWR Coolant Chemistry Loop (PCCL)

1. Description of Change

An in-pile loop designed to simulate the primary coolant system of a pressurized water reactor (PWR) is to be installed in the MITR-II core. The facility is described in detail in Reference 1, "Safety Evaluation Report (SER) for the PWR Coolant Chemistry Loop (PCCL)", Report No. MITNRL-020, February 13, 1987, plus a supplement dated March 22, 1988, both prepared for review by the MIT Reactor Safeguards Committee and attached hereto.

2. Safety Evaluation

The SER addresses the following topics:

- PCCL loop design
- Operational and experimental procedures
- Maximum effects of reactivity, pressure, and temperature
- Radiation levels and ALARA considerations
- PCCL safety evaluation
- Waste handling and disposal
- Future work

It concludes that operation and experimentation with the PCCL loop will fully satisfy the MITR-II Technical Specifications[2] and that no significant health or safety hazards will result from such activities. The MITR Staff has worked with the PCCL Staff on the design of the facility and on preparation of the SER, and it concurs with the above conclusions.

Attachment of the SER and its Supplement to this Safety Review is only for the purpose of providing a description of the experiment and the Project Group's evaluation of its safety. It is expected that experience gained from installing and operating the loop will require changes in the facility and its operation as now described in the SER and Supplement. In accordance with internal MITR procedures, such changes will be documented and will be reviewed to assure compliance with the MITR Technical Specifications. As required by 10 CFR 59(b)(1), such changes will also be evaluated to determine whether or not an unreviewed safety question (USQ) exists. As a minimum, any change that is predicted to permit any one or more of the following parameters to exceed the limit stated in the SER (and again below) will be submitted to the MIT Reactor Safeguards Committee for its opinion concerning whether or not an USQ exists and, if one is deemed to exist, the change will be submitted to USNRC for approval before implementation:

1. Pressure limits:

   a) 2500 psia in the Zircaloy loop, with redundant protective devices designed to relieve at 2500 to 3000 psia.

   b) 100 psia in the aluminum thimble, with redundant protective devices designated to relieve at 100 to 500 psia.

2. Temperature limit: assurance that the Zircaloy tubing temperature will not exceed 2200°F under any conditions, including abnormal, shall be achieved by utilizing redundant electric heater shut-offs, each having its own thermocouple sensing the lead bath temperature.

3. Hydrogen inventory limit: 30 SCF, which is the combined maximum inventories of the transfer flask, the charging water and discharge water storage tanks, and the dissolved hydrogen. Assurance that this quantity will not be exceeded is provided by the limited capacities of the tanks and by administrative controls that will restrict the hydrogen charged into the transfer flask to a maximum of 10 SCF.

Surveillance procedures will provide for periodic functional testing of the pressure relief valves and heater shut-off circuits that assure compliance with the above limits.

3. Unreviewed Safety Question Determination

The loop (0.26" ID Zircalloy in core and 0.26" ID Inconel out of core, 0.026" wall thickness in both cases) will operate at 2200 psi and 600°F. It (along with a heater, lead bath and instrumentation) will be enclosed in an oval-shaped aluminum thimble having a 0.125" wall thickness. The lower end of the thimble, the in-core section, fits into a solid, aluminum dummy fuel element in the same manner as do in-core sample thimbles. Details are provided in the SER and its Supplement.

Among the functions of the above components is protection of the fuel, core structure and other components of the reactor important to safety from damage or malfunctions regardless of credible failures of or within the thimble. It must be shown that such failures cannot credibly interact with the above reactor components in such a way as to create the potential for an unreviewed safety question as defined below:

10CFR50.59(2) - A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

Failures or accidents that originate with the experimental equipment are evaluated to see if they can lead to accidents or failures involving reactor components. If they cannot, an unreviewed safety question does not exist. If they can, then the accident to or failure of the affected reactor component must be evaluated with respect to the three parts of the USQ definition. The following methods of interaction between the loop and the reactor will occur or may be postulated:

3.1 Reactivity Effects

MITR-II Technical Specification 6.1-1 limits the reactivity worth of experiments to the following values:

	Single Experiment Worth	Total Worth
Movable	0.2% $\Delta K/K$	0.5% $\Delta K/K$
Non-secured	0.5% $\Delta K/K$	1.0% $\Delta K/K$
Total of the above	---	1.5% $\Delta K/K$
Secured	1.8% $\Delta K/K$	---

The three types of experiments are defined in Section 1 of the Technical Specifications as follows:

1.23 Secured Experiment

A secured experiment is an experiment or experimental facility held firmly in place by a mechanical device or by gravity, such that the restraining forces are substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment or by forces which can arise as a result of credible malfunctions.

1.24 Movable Experiment

A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into or out of the reactor while the reactor is operating.

1.25 Non-Secured Experiments

Experiments where it is intended that the experiment should not move while the reactor is operating, but is held in place with less restraint than secured experiment.

Potential reactivity effects associated with the PCCL have been addressed in the Safety Evaluation Report for the facility[1]. Credible effects arise from the use of boron for water chemistry and from possible flooding/reflooding incidents.

Water and contained chemicals in the loop are classified as a non-secured experiment, and so their reactivity worth is limited to 0.5% ΔK/K. Conservative calculations given in the SER show that ejection of all in-pile boron, even if it were first concentrated in the core region of the loop, would not exceed +0.02% ΔK/K. This reactivity effect is minimized by use of boron enriched in the B-11 isotope.

For the reactivity effects of flooding/reflooding scenarios for the in-core loop, the void volumes in the thimble and the coolant channel annulus between the thimble and the dummy element are such that effects are measured or calculated to be well within the 0.5% ΔK/K limit for non-secured experiments. Flooding the void volume in the thimble has been measured to cause a +0.14% ΔK/K reactivity effect. The 0.050" cooling annulus between the thimble and the dummy element, if voided, is calculated to produce a +0.17% ΔK/K reactivity effect on reflooding.

The total worth of all non-secured experiments must not exceed 1.0% ΔK/K. The only other non-secured experiment presently in the core is a 1.75 inch I.D. irradiation thimble whose non-secured reactivity (due to flooding accidents) is limited to 0.5% ΔK/K by the insertion of sample capsules or solid spacers. Even though there are no conceivable accidents during reactor operation that could lead to rapid flooding of more than one thimble at a time, the total non-secured worth of all such experiments that might be installed at any one time will be limited as required by the Technical Specifications, using measurements or conservative calculations of the flooding reactivity worth.

The titanium can and contents (lead, loop, heater, and fixtures) are classified as a secured experiment, because they are mechanically held in position by the loop tubing and other structural components. Their complete ejection from the thimble followed by flooding must not exceed 1.8% ΔK/K. Complete ejection, such as by sudden rupture of the loop at its lowest point (the U-bend), is difficult to envision, because of limited void volume between the top of the in-core section and the bottom of the steam generator (shot bed) section, which will limit ejectable lead to no more than one-third of the amount in the bath. Also, materials thrust upwards by the steam would tend to fall back into the thimble, thereby limiting the floodable volume in-core.

At worst, flooding one-third of the thimble volume would approximate one-third of the 1.0% ΔK/K positive reactivity effect that has been measured for flooding of an empty 1.75 inch I.D. irradiation thimble[3], because the volumes are comparable. The reactivity of the ejectable lead is not expected to be large, because the combined reactivity effects of removing the lead, the titanium can and the water-filled zircalloy loop is only +0.17% ΔK/K. This assures compliance with the 1.8% ΔK/K limit.

No limit is imposed by the Technical Specifications on the total worth of all secured experiments. In the future, if more than one loop is installed in the core, it must be shown that there is no credible coupling between them that could lead to a positive reactivity effect exceeding 1.8% ΔK/K. This is the limiting value, as shown in Chapter 15 of the MITR-II Safety Analysis Report[4], for a step insertion of reactivity below which the reactor can be safely shut down without damage to the core. Measurements made at startup of the MITR-II[5] confirmed the value.

Because the various components (whether classified as non-secured or secured) meet applicable technical specifications and because there are no conceivable reactivity events that could exceed the limiting value of 1.8% ΔK/K, there are no unreviewed safety questions related to reactivity insertions.

3.2 Pressure Effects

The PCCL will operate at approximately 2200 psia and 600°F. The system contains about 0.5 liters of water at these conditions, 40 ml of which are in the core region. The loop itself, circulating pump, charging pump, and associated equipment have design pressures of 3000 psi or higher, and the system is protected by a relief valve set to open at 2500 psi, backed up by a burst disc designed for 3000 psi.

The loop itself is contained in an elliptical Type 6061 aluminum thimble (major axis 2.5 in., minor axis 1.4 in., thickness 0.125 in.) in the core and in a cylindrical jacket (diameter 4 in.) above the core. The thimble and jacket are designed for, and will be hydrostatically tested at, 750 psia. They are protected by redundant pressure relief valves set at 30 to 100 psia.

In the event of a loop rupture allowing the 0.5 liters of pressurized water to flash to steam, calculation in the SER of the maximum steam pressure at 350°F (average temperature of the shot bed surrounding the steam generator section of the loop) shows that it will not exceed 481 psia, ignoring the pressure suppression effect of condensation of steam on the cold (≈100°F) aluminum walls of the thimble.

Hence, there can be no effect on components outside the thimble and no unreviewed safety question.

3.3 Temperature Effects

The in-pile loop asssembly will be heated both by a 0 to 20 kW heater and by a combination of gamma and fast neutron radiation. The normal combined heat load will be less than 20 kW. The radiation heating is estimated in the SER Supplement to be 9.6 kW at a reactor power of 5 MW, so that 29.6 kW would be the maximum heat load potentially available under malfunction conditions. This is not much more than the hottest running fuel plate in the MIT Reactor, and most of the heat will be dumped to the reactor primary coolant via the shot bed in the steam generator section above the core. Hence, the thimble is easily cooled by the flow of primary coolant through the 0 050 inch thick channel between the thimble and the dummy fuel element that surrounds it.

The SER addresses the potential for a Zircaloy-water reaction and shows that cooling by conduction and radiation will prevent temperatures in the thimble from exceeding 1845°F for the maximum radiation heating, which is estimated not to exceed 9.6 kW. This is based on very conservative extrapolation of temperatures measured out of core in a test mock-up of the loop assembly under LOCA conditions at heater powers in the range of 2470-4510 watts. The 1845°F is significantly below the 2200°F post-LOCA limit on Zircaloy temperature imposed for PWR units by NRC[4].

Assurance that electrical heating will be stopped, so that total heating will not exceed the 9.6 kW which might result from gamma and fast neutron heating with the reactor at full power, is achieved by redundant heater shut-offs that are activated by high lead bath temperatures. The sensors and relays that interrupt power to the heaters are completely independent, thus avoiding compromise by a single failure.

Elevated lead bath temperatures are not a threat to the aluminum thimble, because there is no contact between the thimble and the titanium can holding the lead except at occasional small points of contact with high spots on the weld bead stiffener on the outer surface of the titanium can and at the support ring which is at the top of the titanium can extension about 12 inches above the lead bath.

In view of the above active and passive safety features, it is not credible that temperature effects within the thimble can affect the fuel, core structure or other components important to safety and, hence, there is no unreviewed safety question in this regard.

3.4 Hydrogen Leak and Combustion

The SER demonstrated that the hydrogen combustion hazard in the thimble is minor. The hydrogen, except in the charging tank and transfer flask, both of which are outside the biological shield, is dissolved in water. The hydrogen within the biological shield is almost all in the water circulating in the loop and amounts to about 25 cc at standard temperature and pressure. This is approximately equivalent to 9 mg of TNT, less than the 25 mg permitted by Technical Specification 6.1-3b without a documented safety analysis.

The maximum hydrogen in service will be about 3 ft^3 (STP) in one of the charging tanks, located outside the biological shield, when nearly all of the charging water has been emptied from the tank. The SER demonstrated that only through highly improbable scenarios can this gas, along with the oxygen necessary for combustion, get into the loop thimble. Even if it does, it will be mixed with helium and with water vapor or steam, and the void geometry is small (about 1 ft^3), dispersed, and very unfavorable for a detonation'. Calculations show that the reaction of a stoichiometric quantity of hydrogen (0.5 mols) with the oxygen in one cubic foot of air (equal to the void volume) at atmospheric pressure would produce only 135 BTU, a negligible amount (equivalent to burning 3.4 grams of fuel oil). Pressure buildup from any deflagration is readily relieved by redundant relief valves. It is, therefore, not credible that the thimble integrity can be breached by hydrogen combustion. The thimble itself is contained within the dummy fuel element, which presents a further barrier for protection of the fuel, core structure, and other components important to safety. Since these cannot be damaged or caused to malfunction by the highly unlikely combustion of hydrogen within the thimble, there is no unreviewed safety question in this regard.

The transfer flask and the charging and discharge tanks, containing no more than 10 SCF of hydrogen each, are not a hazard in the containment, because discharge of their entire contents, even simultaneously, into the containment atmosphere (200,000 ft^3) will result in a concentration far below the lower explosive limit, and an explosion-proof fan mounted near the transfer flask and the charging and discharge tanks will prevent local accumulation of a combustible gas mixture.

3.5 Loss of Loop Pumping Power or Loss of Flow

Without circulation, the coolant in the loop will overheat and escape via a relief valve to the discharge tank, allowing the loop to boil dry. Again, there can be no effect on components outside the thimble.

3.6 Leak in the Lead-Bath Can

The SER addresses the questions of large and small leaks of lead from the titanium can and concludes that there are no credible mechanisms by which the loop can adversely affect MITR safety. This conclusion is supported by successful results in experiments designed to simulate such failures.

The SER analysis is conservative in that it does not take credit for the additional protective barriers provided by the coolant flow outside the thimble and the aluminum dummy element in which the thimble is contained. Both would protect the fuel and other components in the hypothetical event that molten lead should penetrate the thimble wall.

3.7 Emergency Core Cooling System (ECCS)

The steam generator (shot bed) section of the system is enclosed in a 4 inch diameter aluminum tube extending from near the top of the reactor primary coolant tank to about a foot above the core. It is thus large enough to create the potential for the shadowing of some fuel elements from the water sprayed onto the top of the core by the ECCS system in the event that the reactor core should not be covered by water.

Tests were made using a mock-up of the core top, primary coolant flow guide, PCCL and a 2-inch diameter in-core sample irradiation facility. It was found that sprayed ECCS water splashed randomly from the experimental facilities and from the interior surfaces of the flow guide so that any shadowing effect was minimized, and each fuel position received at least one-third of the average flow per element. Adding a second 4-inch diameter experimental facility increased the shadowing effect slightly but each fuel position received at least one-quarter of the average flow per element.

The ECCS for MITR-II is described in the Safety Analysis Report, Section 6.1, Emergency Cooling(4). The analysis in that section has been revised to account for the installation of experimental facilities, such as the PCCl, in the core, and it demonstrates very conservatively that the ECCS system will adequately cool the core containing such facilities (even those fuel positions that receive only one-quarter of the average flow) in the event, considered incredible, that the core should become uncovered.

3.8 Electrical Short Circuit within the Thimble

The electrical heating system has been analyzed to determine whether a short circuit can cause damage to or malfunctions of the fuel, core structure or other components important to safety, either by arcing or by current surges.

The electrical heater is rated for 20 kW. In the design presently being tested, the heat output is distributed unformly over the two legs of a U-shaped heater, each leg being 21.5" long. Each leg is sheathed in carbon steel with an O.D. of 0.440 inches, and the insulation in ceramic. The power source is 270-Volt A.C.

Electrical protection consists of a 150-A semiconductor fuse in the heater power controller, connected on one leg of the power output. This is backed up by 100-A circuit breakers in the box which feeds the power controller. There are also 200-A fuses at the safety disconnect where the connection from the insulated pothead (which penetrates the containment) to the CCL heater bus is made. (The designation CCL is used because this bus eventually will power other loops, not just the PCCL.) The aluminum thimble and the power controller will be grounded to a heavy copper bus connected by 4/0 copper cable to the reactor electrical equipment ground bus.

A failure of the ceramic insulation could result in a short circuit between the heater leads or between one lead and grounded components in the core. Characteristics of the protection devices are such that energy deposited in the materials subject to damage by the short circuit can melt and/or vaporize only small amounts of materials. This melting would involve only materials inside the thimble, such as the cable sheath, lead or support brackets. Additional barriers protecting the fuel are the titanium can, the aluminum thimble, a water gap, the aluminum dummy element and either another water gap or the side plate of a fuel element. Because the thimble is well grounded and the titanium can makes good contact with the thimble, there should be no arcing at the thimble wall. The grounded thimble is a very effective shield against voltages being applied to in-core components, due to its large cross-sectional area and high conductivity.

The ability of the heater fuse to protect the system was observed during an unplanned demonstration when failure of the insulation on the heater power leads led to a short circuit at the point where the leads are sealed into the heater. The heater was connected at the time to 220 V, a little less than the 270 V to be used in service, but the only damage before the fuse opened was vaporization of a small fraction of an inch of lead-in wire and a little adjacent thermocouple wire. The failure was related to the method of insulating the leads near where they are attached to the heater, and redesign should prevent a repetition. The event, however, demonstrates that this hazard is easily contained within the thimble.

There have also been two failures of the Inconel heating elements near the top of the heater during ts to measure the system's capability for dissipating gamma and neutron heating upon loss of loop coolant during reactor operation. The tests have served this purpose (see Section 3.3), but, in addition, they have shown (1) that the effects of short-circuits in the heating elements are confined within the heater sheath and (2) that overheating will cause failure of the heating elements and cut-off of power before the bath temperature significantly exceeds the normal operating range.

In view of these considerations, it is not conceivable that fuel, the core structure or other components important to safety can be affected by destructive heating at the point of insulation failure.

Power for the reactor facility is provided by a 13,800/480-277V, 1000-KVA transformer. Power for the PCCL equipment is fed directly via a 360-A breaker, a new pothead at the containment wall, a 200-A fused safety disconnect switch to the CCL heater bus, a 100-A breaker off the bus, a heater power controller and finally a 100-A fuse. Power for the reactor is supplied over entirely separate lines from the 1000-KVA transformer, so that current surges due to loop malfunctions will have negligible effect on the power supply for reactor instrumentation and controls. Cables for the CCL power are run in conduits distinct from those for reactor control circuits. The PCCL thimble will be gounded by heavy cable to the reactor ground system, again avoiding proximity to reactor control circuits. Consequently, current surges due to short circuits or otherwise are expected to have no significant effect on reactor operation.

4. Conclusion

It is concluded that failures or accidents originating with the PCCL loop cannot interact with the reactor fuel, core structure or other components important to safety, except through reactivity effects. In this case loop failures or accidents will not cause reactivity changes exceeding those authorized by the Technical Specifications. For equipment important to safety (i) the probability of an accident or malfunction is not increased, (ii) the possibility for an accident or malfunction of a different type than that previously evaluated in the SAR is not created, and (iii) no margin of safety in any technical specification is reduced. Consequently, the PCCL experiment does not involve an unreviewed safety question.

References

1. Safety Evaluation Report for the PWR Coolant Chemistry Loop (PCCL), Report No. MITNRL-020, February 13, 1987, plus Supplement dated April 19, 1988.

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5. MITR-II Start-Up Report, Report No. MITNE-198, February 14, 1977.

6. 10 CFR 50.46(b)(1).

7. Berman, M., "A Critical Review of Recent Large-Scale Experiments on Hydrogen - Air Detonations", Nuclear Science and Engineering, 93, 321-347 (1986).

Item: PWR Coolant Chemistry Loop (PCCL)

Submitted by L. Clark, Jr. Date April 19, 1988

Q/A number if required M-86-2

Does the item change or contradict the

Technical Specifications?		Yes*	X	No
SAR?		Yes*	X	No

* Attach explanation

Description of Change (Attach extra pages if necessary):

A pressurized coolant chemistry loop (PCCL) is to be installed in the MITR core. The PCCL is described and its safety evaluated in a Safety Evaluation Report (SER), MITNRL-020, dated February 13, 1987 and a Supplement dated April 19, 1988.

Safety Evaluation (Attach extra pages if necessary):

The MITR Staff's safety evaluation is contained in the attached pp. 1 - 11. It concurs with the PCCL Project Staff that operation and experimentation with the loop will fully satisfy the MITR-II Technical Specifications and that no unacceptable safety hazards will result.

Summary of Review:

a) Does the proposal:	Yes	No
i) involve an unreviewed safety question (10CFR50.59(a)(2))		X
ii) decrease scope of requalification program (10CFR50.54(i-1))		X
iii) decrease effectiveness of security plan (10CFR50.54(p))		X
iv) decrease effectiveness of emergency plan (10CFR50.54(q))		X

b) Reviewer's Comments: Recommend approval. MITRSC approval required.

RRPO F.M. Williams Date 20 April 88

Recommend Approval ✓ Yes ____ No

Reviewer [signature] Date 4/20/88

Reviewer [signature] Date 20 April 88

Approved L. Clark, Jr. Date 4/21/88
(Director of Reactor Operations)

10CFR50.59 & 50.54(p and q) changes logged for reporting to NRC, Date Reported by letter 4/21/88

Copy to Director for Operations
Copies circulated to and initialled by all Licensed Personnel
Original to Safety Review File

NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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L. CLARK, JR.
Director of Reactor Operations

April 19, 1988

Safety Review #-0-86-9: PWR Coolant Chemistry Loop (PCCL)

1. Description of Change

An in-pile loop designed to simulate the primary coolant system of a pressurized water reactor (PWR) is to be installed in the MITR-II core. The facility is described in detail in Reference 1, "Safety Evaluation Report (SER) for the PWR Coolant Chemistry Loop (PCCL)", Report No. MITNRL-020, February 13, 1987, plus a supplement dated March 22, 1988, both prepared for review by the MIT Reactor Safeguards Committee and attached hereto.

2. Safety Evaluation

The SER addresses the following topics:

PCCL loop design
Operational and experimental procedures
Maximum effects of reactivity, pressure, and temperature
Radiation levels and ALARA considerations
PCCL safety evaluation
Waste handling and disposal
Future work

It concludes that operation and experimentation with the PCCL loop will fully satisfy the MITR-II Technical Specifications[2] and that no significant health or safety hazards will result from such activities. The MITR Staff has worked with the PCCL Staff on the design of the facility and on preparation of the SER, and it concurs with the above conclusions.

Attachment of the SER and its Supplement to this Safety Review is only for the purpose of providing a description of the experiment and the Project Group's evaluation of its safety. It is expected that experience gained from installing and operating the loop will require changes in the facility and its operation as now described in the SER and Supplement. In accordance with internal MITR procedures, such changes will be documented and will be reviewed to assure compliance with the MITR Technical Specifications. As required by 10 CFR 59(b) (1), such changes will also be evaluated to determine whether or not an unreviewed safety question (USQ) exists. As a minimum, any change that is predicted to permit any one or more of the following parameters to exceed the limit stated in the SER (and again below) will be submitted to the MIT Reactor Safeguards Committee for its opinion concerning whether or not an USQ exists and, if one is deemed to exist, the change will be submitted to USNRC for approval before implementation:

1. Pressure limits:

   a) 2500 psia in the Zircaloy loop, with redundant protective devices designed to relieve at 2500 to 3000 psia.

   b) 100 psia in the aluminum thimble, with redundant protective devices designated to relieve at 100 to 500 psia.

2. Temperature limit: assurance that the Zircaloy tubing temperature will not exceed 2200°F under any conditions, including abnormal, shall be achieved by utilizing redundant electric heater shut-offs, each having its own thermocouple sensing the lead bath temperature.

3. Hydrogen inventory limit: 30 SCF, which is the combined maximum inventories of the transfer flask, the charging water and discharge water storage tanks, and the dissolved hydrogen. Assurance that this quantity will not be exceeded is provided by the limited capacities of the tanks and by administrative controls that will restrict the hydrogen charged into the transfer flask to a maximum of 10 SCF.

Surveillance procedures will provide for periodic functional testing of the pressure relief valves and heater shut-off circuits that assure compliance with the above limits.

3. Unreviewed Safety Question Determination

The loop (0.26" ID Zircalloy in core and 0.26" ID Inconel out of core, 0.026" wall thickness in both cases) will operate at 2200 psi and 600°F. It (along with a heater, lead bath and instrumentation) will be enclosed in an oval-shaped aluminum thimble having a 0.125" wall thickness. The lower end of the thimble, the in-core section, fits into a solid, aluminum dummy fuel element in the same manner as do in-core sample thimbles. Details are provided in the SER and its Supplement.

Among the functions of the above components is protection of the fuel, core structure and other components of the reactor important to safety from damage or malfunctions regardless of credible failures of or within the thimble. It must be shown that such failures cannot credibly interact with the above reactor components in such a way as to create the potential for an unreviewed safety question as defined below:

10CFR50.59(2) - A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

Failures or accidents that originate with the experimental equipment are evaluated to see if they can lead to accidents or failures involving reactor components. If they cannot, an unreviewed safety question does not exist. If they can, then the accident to or failure of the affected reactor component must be evaluated with respect to the three parts of the USQ definition. The following methods of interaction between the loop and the reactor will occur or may be postulated:

3.1 Reactivity Effects

MITR-II Technical Specification 6.1-1 limits the reactivity worth of experiments to the following values:

	Single Experiment Worth	Total Worth
Movable	0.2% ΔK/K	0.5% ΔK/K
Non-secured	0.5% ΔK/K	1.0% ΔK/K
Total of the above	---	1.5% ΔK/K
Secured	1.8% ΔK/K	---

The three types of experiments are defined in Section 1 of the Technical Specifications as follows:

1.23 Secured Experiment

A secured experiment is an experiment or experimental facility held firmly in place by a mechanical device or by gravity, such that the restraining forces are substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment or by forces which can arise as a result of credible malfunctions.

1.24 Movable Experiment

A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into or out of the reactor while the reactor is operating.

1.25 Non-Secured Experiments

Experiments where it is intended that the experiment should not move while the reactor is operating, but is held in place with less restraint than secured experiment.

Potential reactivity effects associated with the PCCL have been addressed in the Safety Evaluation Report for the facility[1]. Credible effects arise from the use of boron for water chemistry and from possible flooding/reflooding incidents.

Water and contained chemicals in the loop are classified as a non-secured experiment, and so their reactivity worth is limited to 0.5% ΔK/K. Conservative calculations given in the SER show that ejection of all in-pile boron, even if it were first concentrated in the core region of the loop, would not exceed +0.02% ΔK/K. This reactivity effect is minimized by use of boron enriched in the B-11 isotope.

For the reactivity effects of flooding/reflooding scenarios for the in-core loop, the void volumes in the thimble and the coolant channel annulus between the thimble and the dummy element are such that effects are measured or calculated to be well within the 0.5% ΔK/K limit for non-secured experiments. Flooding the void volume in the thimble has been measured to cause a +0.14% ΔK/K reactivity effect. The 0.050" cooling annulus between the thimble and the dummy element, if voided, is calculated to produce a +0.17% ΔK/K reactivity effect on reflooding.

The total worth of all non-secured experiments must not exceed 1.0% ΔK/K. The only other non-secured experiment presently in the core is a 1.75 inch I.D. irradiation thimble whose non-secured reactivity (due to flooding accidents) is limited to 0.5% ΔK/K by the insertion of sample capsules or solid spacers. Even though there are no conceivable accidents during reactor operation that could lead to rapid flooding of more than one thimble at a time, the total non-secured worth of all such experiments that might be installed at any one time will be limited as required by the Technical Specifications, using measurements or conservative calculations of the flooding reactivity worth.

The titanium can and contents (lead, loop, heater, and fixtures) are classified as a secured experiment, because they are mechanically held in position by the loop tubing and other structural components. Their complete ejection from the thimble followed by flooding must not exceed 1.8% ΔK/K. Complete ejection, such as by sudden rupture of the loop at its lowest point (the U-bend), is difficult to envision, because of limited void volume between the top of the in-core section and the bottom of the steam generator (shot bed) section, which will limit ejectable lead to no more than one-third of the amount in the bath. Also, materials thrust upwards by the steam would tend to fall back into the thimble, thereby limiting the floodable volume in-core.

At worst, flooding one-third of the thimble volume would approximate one-third of the 1.0% ΔK/K positive reactivity effect that has been measured for flooding of an empty 1.75 inch I.D. irradiation thimble[3], because the volumes are comparable. The reactivity of the ejectable lead is not expected to be large, because the combined reactivity effects of removing the lead, the titanium can and the water-filled zircalloy loop is only +0.17% ΔK/K. This assures compliance with the 1.8% ΔK/K limit.

No limit is imposed by the Technical Specifications on the total worth of all secured experiments. In the future, if more than one loop is installed in the core, it must be shown that there is no credible coupling between them that could lead to a positive reactivity effect exceeding 1.8% ΔK/K. This is the limiting value, as shown in Chapter 15 of the MITR-II Safety Analysis Report[4], for a step insertion of reactivity below which the reactor can be safely shut down without damage to the core. Measurements made at startup of the MITR-II[5] confirmed the value.

Because the various components (whether classified as non-secured or secured) meet applicable technical specifications and because there are no conceivable reactivity events that could exceed the limiting value of 1.8% ΔK/K, there are no unreviewed safety questions related to reactivity insertions.

3.2 Pressure Effects

The PCCL will operate at approximately 2200 psia and 600°F. The system contains about 0.5 liters of water at these conditions, 40 ml of which are in the core region. The loop itself, circulating pump, charging pump, and associated equipment have design pressures of 3000 psi or higher, and the system is protected by a relief valve set to open at 2500 psi, backed up by a burst disc designed for 3000 psi.

The loop itself is contained in an elliptical Type 6061 aluminum thimble (major axis 2.5 in., minor axis 1.4 in., thickness 0.125 in.) in the core and in a cylindrical jacket (diameter 4 in.) above the core. The thimble and jacket are designed for, and will be hydrostatically tested at, 750 psia. They are protected by redundant pressure relief valves set at 30 to 100 psia.

In the event of a loop rupture allowing the 0.5 liters of pressurized water to flash to steam, calculation in the SER of the maximum steam pressure at 350°F (average temperature of the shot bed surrounding the steam generator section of the loop) shows that it will not exceed 481 psia, ignoring the pressure suppression effect of condensation of steam on the cold (≈100°F) aluminum walls of the thimble.

Hence, there can be no effect on components outside the thimble and no unreviewed safety question.

3.3 Temperature Effects

The in-pile loop asssembly will be heated both by a 0 to 20 kW heater and by a combination of gamma and fast neutron radiation. The normal combined heat load will be less than 20 kW. The radiation heating is estimated in the SER Supplement to be 9.6 kW at a reactor power of 5 MW, so that 29.6 kW would be the maximum heat load potentially available under malfunction conditions. This is not much more than the hottest running fuel plate in the MIT Reactor, and most of the heat will be dumped to the reactor primary coolant via the shot bed in the steam generator section above the core. Hence, the thimble is easily cooled by the flow of primary coolant through the 0.050 inch thick channel between the thimble and the dummy fuel element that surrounds it.

The SER addresses the potential for a Zircaloy-water reaction and shows that cooling by conduction and radiation will prevent temperatures in the thimble from exceeding 1845°F for the maximum radiation heating, which is estimated not to exceed 9.6 kW. This is based on very conservative extrapolation of temperatures measured out of core in a test mock-up of the loop assembly under LOCA conditions at heater powers in the range of 2470-4510 watts. The 1845°F is significantly below the 2200°F post-LOCA limit on Zircaloy temperature imposed for PWR units by NRC*.

Assurance that electrical heating will be stopped, so that total heating will not exceed the 9.6 kW which might result from gamma and fast neutron heating with the reactor at full power, is achieved by redundant heater shut-offs that are activated by high lead bath temperatures. The sensors and relays that interrupt power to the heaters are completely independent, thus avoiding compromise by a single failure.

Elevated lead bath temperatures are not a threat to the aluminum thimble, because there is no contact between the thimble and the titanium can holding the lead except at occasional small points of contact with high spots on the weld bead stiffener on the outer surface of the titanium can and at the support ring which is at the top of the titanium can extension about 12 inches above the lead bath.

In view of the above active and passive safety features, it is not credible that temperature effects within the thimble can affect the fuel, core structure or other components important to safety and, hence, there is no unreviewed safety question in this regard.

3.4 Hydrogen Leak and Combustion

The SER demonstrated that the hydrogen combustion hazard in the thimble is minor. The hydrogen, except in the charging tank and transfer flask, both of which are outside the biological shield, is dissolved in water. The hydrogen within the biological shield is almost all in the water circulating in the loop and amounts to about 25 cc at standard temperature and pressure. This is approximately equivalent to 9 mg of TNT, less than the 25 mg permitted by Technical Specification 6.1-3b without a documented safety analysis.

The maximum hydrogen in service will be about 3 ft^3 (STP) in one of the charging tanks, located outside the biological shield, when nearly all of the charging water has been emptied from the tank. The SER demonstrated that only through highly improbable scenarios can this gas, along with the oxygen necessary for combustion, get into the loop thimble. Even if it does, it will be mixed with helium and with water vapor or steam, and the void geometry is small (about 1 ft^3), dispersed, and very unfavorable for a detonation[7]. Calculations show that the reaction of a stoichiometric quantity of hydrogen (0.5 mols) with the oxygen in one cubic foot of air (equal to the void volume) at atmospheric pressure would produce only 135 BTU, a negligible amount (equivalent to burning 3.4 grams of fuel oil). Pressure buildup from any deflagration is readily relieved by redundant relief valves. It is, therefore, not credible that the thimble integrity can be breached by hydrogen combustion. The thimble itself is contained within the dummy fuel element, which presents a further barrier for protection of the fuel, core structure, and other components important to safety. Since these cannot be damaged or caused to malfunction by the highly unlikely combustion of hydrogen within the thimble, there is no unreviewed safety question in this regard.

The transfer flask and the charging and discharge tanks, containing no more than 10 SCF of hydrogen each, are not a hazard in the containment, because discharge of their entire contents, even simultaneously, into the containment atmosphere (200,000 ft^3) will result in a concentration far below the lower explosive limit, and an explosion-proof fan mounted near the transfer flask and the charging and discharge tanks will prevent local accumulation of a combustible gas mixture.

3.5 Loss of Loop Pumping Power or Loss of Flow

Without circulation, the coolant in the loop will overheat and escape via a relief valve to the discharge tank, allowing the loop to boil dry. Again, there can be no effect on components outside the thimble.

3.6 Leak in the Lead-Bath Can

The SER addresses the questions of large and small leaks of lead from the titanium can and concludes that there are no credible mechanisms by which the loop can adversely affect MITR safety. This conclusion is supported by successful results in experiments designed to simulate such failures.

The SER analysis is conservative in that it does not take credit for the additional protective barriers prov[illegible]he coolant flow outside the thimble and the aluminum dummy [illegible] which the thimble is contained. Both would protect the fue[illegible]er components in the hypothetical event that molten lead should p[illegible]rate the thimble wall.

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Tests were made using a mock-up of the core top, primary coolant flow guide, PCCL and a 2-inch diameter in-core sample irradiation facility. It was found that sprayed ECCS water splashed randomly from the experimental facilities and from the interior surfaces of the flow guide so that any shadowing effect was minimized, and each fuel position received at least one-third of the average flow per element. Adding a second 4-inch diameter experimental facility increased the shadowing effect slightly but each fuel position received at least one-quarter of the average flow per element.

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A failure of the ceramic insulation could result in a short circuit between the heater leads or between one lead and grounded components in the core. Characteristics of the protection devices are such that energy deposited in the materials subject to damage by the short circuit can melt and/or vaporize only small amounts of materials. This melting would involve only materials inside the thimble, such as the cable sheath, lead or support brackets. Additional barriers protecting the fuel are the titanium can, the aluminum thimble, a water gap, the aluminum dummy element and either another water gap or the side plate of a fuel element. Because the thimble is well grounded and the titanium can makes good contact with the thimble, there should be no arcing at the thimble wall. The grounded thimble is a very effective shield against voltages being applied to in-core components, due to its large cross-sectional area and high conductivity.

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Power for the reactor facility is provided by a 13,800/480-277V, 1000-KVA transformer. Power for the PCCL equipment is fed directly via a 260-A breaker, a new pothead at the containment wall, a 200-A fused safety disconnect switch to the CCL heater bus, a 100-A breaker off the bus, a heater power controller and finally a 100-A fuse. Power for the reactor is supplied over entirely separate lines from the 1000-KVA transformer, so that current surges due to loop malfunctions will have negligible effect on the power supply for reactor instrumentation and controls. Cables for the CCL power are run in conduits distinct from those for reactor control circuits. The PCCL thimble will be gounded by heavy cable to the reactor ground system, again avoiding proximity to reactor control circuits. Consequently, current surges due to short circuits or otherwise are expected to have no significant effect on reactor operation.

4. Conclusion

It is concluded that failures or accidents originating with the PCCL loop cannot interact with the reactor fuel, core structure or other components important to safety, except through reactivity effects. In this case loop failures or accidents will not cause reactivity changes exceeding those authorized by the Technical Specifications. For equipment important to safety (i) the probability of an accident or malfunction is not increased, (ii) the possibility for an accident or malfunction of a different type than that previously evaluated in the SAR is not created, and (iii) no margin of safety in any technical specification is reduced. Consequently, the PCCL experiment does not involve an unreviewed safety question.

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SAFETY EVALUATION REPORT

FOR

THE PWR COOLANT CHEMISTRY LOOP (PCCL)

MITNRL-020

to be

installed and operated in the

MITR

February 13, 1987

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For Review by the

MITR Safeguards Committee

~~8804280023~~ 75pp.

TABLE OF CONTENTS

1. INTRODUCTION

a) Foreword

This is the safety evaluation report for an in-pile loop facility designed to simulate the primary coolant system of a pressurized water reactor (PWR). The loop will be used to carry out research into the effects of coolant chemistry on the transport and deposition of corrosion product radionuclides as part of a program to develop means for the reduction of maintenance doses in the nuclear utility industry. Loop construction and operation are funded by the Electric Power Research Institute and the Empire State Electric Energy Research Corporation; loop conceptual design has been funded in-house and by a research project supported by regional utilities (Boston Edison, PSE&G, and Duke Power) under the Electric Utility Program of the MIT Energy Laboratory. This program also calls for the design and operation of a loop simulating BWR conditions; however, the BWR loop will be covered by a separate submission.

The objective of this report is to present a summary description of the design and operating procedure of the PWR Coolant Chemistry Loop (PCCL) in sufficient detail, and with supporting analyses, to demonstrate that it can be operated safely within the envelope of applicable MIT Reactor Technical Specifications. To this end, a number of topics will be emphasized, including the effect of the PCCL on core reactivity, energy dissipation following loss of normal energy removal capability, the consequences of leakage, and hydrogen handling.

b) General Description

1) In-pile loop design

The guiding design philosophy in the development of the PCCL concept has been to simulate all important PWR primary coolant system parameters (e.g., pressure, temperature, velocity, materials, surface area ratios, etc.) as closely as possible, as summarized in Table 1.1, but at a greatly reduced scale: on the order of 10^5 smaller. Despite being quite small on a macroscopic basis, good simulation of a PWR coolant flow unit cell (steam generator tube--inter-fuel-pin channel) is achieved on approximately a one-third scale. The resulting design is depicted schematically in Fig. 1.1. It is a relatively simple layout consisting mainly of 0.25 inch ID tubing, containing less than 0.5 liter of coolant circulated at approximately 1-2 gpm with a canned rotor pump. An electric heater supplies 10-20 kW of energy to the Zircaloy in-pile segment. The system is externally pressurized using a positive displacement diaphragm-type pump plus backpressure valve--a practice proven in many years of out-of-pile autoclave experiments operated at MIT, Westinghouse, and General Electric.

Features which are particularly significant from a safety viewpoint are as follows:

- The entire loop is encapsulated by an aluminum thimble of 2 inches diameter in-core (where it is housed in a dummy fuel element), increasing to 4 inches above the core, topped by a pod containing the pump. The thimble atmosphere is helium gas at $\leq$ 50 psi, and the thimble is protected from overpressurization by a pair of 100 psi relief valves. Relief valves also protect the loop itself against $\geq$ 2,500 psia (see Section 2.a).
- Energy is added to the in-core section of the loop, a Zircaloy U-tube, by electric resistance heaters immersed in a small lead bath surrounding the U-tube. Calculations and confirmatory experiments have shown that when electrtic heat is shut off,

TABLE 1.1: COMPARISON OF PWR AND PCCL PARAMETERS

Parameter	Representative PWR	MIT PCCL*
1. Designed to Match Virtually Exactly		
Pressure, psia	2,240	hence same $\overline{T}$, H_2O density, viscosity
Temperature: high/low, °F	605/547	
Flow velocity in core, ft/s	14	hence same fluid shear
Thermal neutron flux, n/cm^2-s	$2x10^{13}$	hence real-time activation
Core heat flux, Btu/h-ft^2	180,000	hence same film ΔT
Steam generator heat flux, Btu/h-ft^2	67,000	
Purification rate, sys. vols/m	10^{-3}	hence, same exogenous sink strength
2. Other Comparisons of Significance		
Steam generator:		
Tube velocity, ft/s	21	15
Reynolds Number	$7.4x10^5$	$2.4x10^5$
Nusselt Number	1,100	450
Length/diameter ratio	940	550
Axial T gradient °F/in	0.08	0.38
Area ratio SG/core	2.7	3.3
Core:		
Reynolds Number	$5x10^5$	$2.4x10^5$
Nusselt Number	770	450
Length/diameter ratio	270	170
Axial T gradient, °F/in	0.4	1.2
Fast neutron flux, n/cm^2-s	$2x10^{14}$	$1x10^{14}$
Loop transit time, s	15	5

*At maximum power (20kW) maximum flow (2 gpm); operating conditions may be changed for specific experiments.

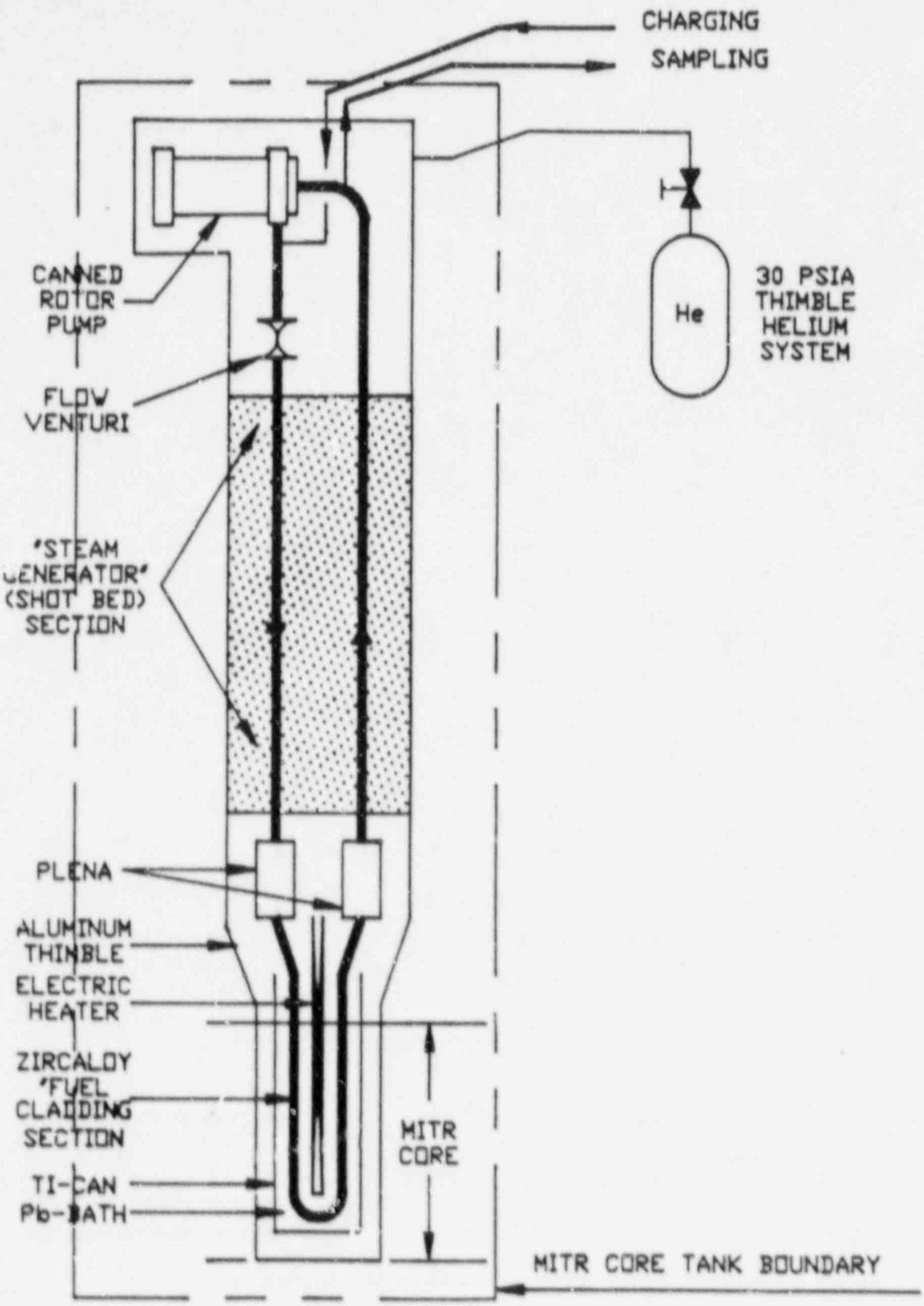

FIGURE 1.1: a) Schematic of PWR coolant chemistry loop: Portion inside MITR core tank.

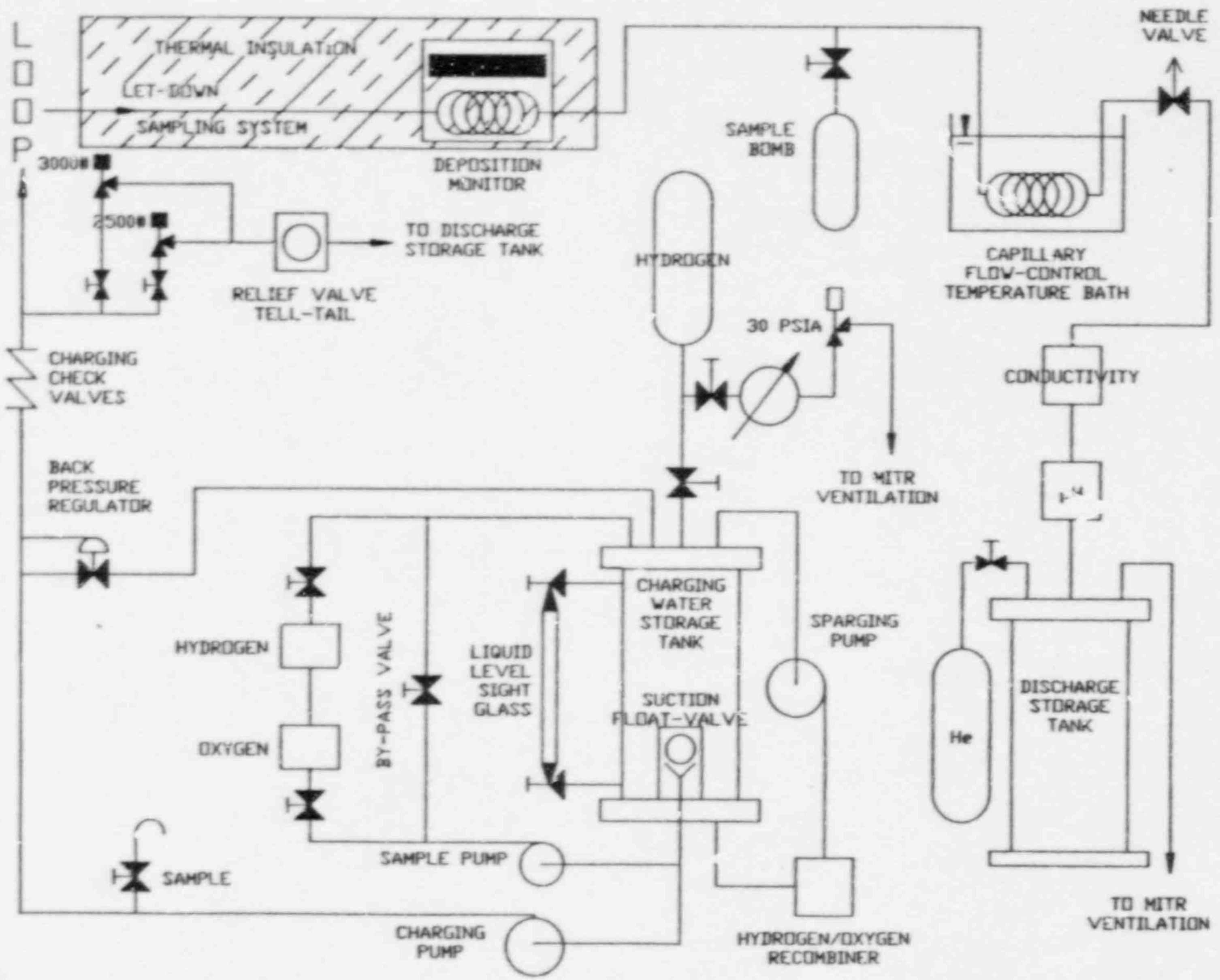

FIGURE 1.1: b) Schematic of PWR coolant chemistry loop: Portion outside MITR core tank.

gamma heating by the MITR at full power can be safely dissipated by radiation from the wall of the titanium bath can to the cold aluminum thimble wall (see Appendix 1.a).

- Energy ($\leq$ 20 kW) is normally rejected by conduction through a bed of copper shot to the thimble wall, and thence to the pool of MITR coolant in the region well above the core (see Section 3.a).
- As in a full-scale PWR, water decomposition is suppressed by maintenance of a small amount [$\leq$ 50 cc (STP)/kg] of H_2 dissolved in the coolant. A catalytic recombiner on the PCCL makeup tank atmosphere provides assurance that a combustible mixture will not exist within the tank, and the discharge tank is vented to the MITR off-gas system through a flame arrestor. Total in-containment H_2 inventory is limited by use of a small, low pressure, transfer flask as the only source of this combustible gas (see Section 6.b).
- Total in-core H_2O inventory in the loop is $\leq$ 100 cc, hence void/reflood reactivity is well within MITR experiment limits. Up to 2,000 ppm of boron (as boric acid) may be added during experiments, but the total boron inventory is inconsequential, and, in any event, 98% B-11 is used (to reduce tritium production; Li-7 is used for LiOH treatment for similar reasons) (see Section 4).
- In-core materials have been selected (and screened using test irradiations) to insure that even unshielded dose rates during loop handling could not exceed several R/h--a value easily reduced by two orders of magnitude using a shielded transfer flask/storage container (see Sections 2.b and 3.b).

* The loop support arrangement is conservatively designed to preclude dropping an unsupported loop into the core, and to insure that excessive weight is not borne by the MITR core grid (see Section 2.c).

The above considerations, and others, are described in the various sections of the body of the report, where supporting calculations and references are also documented.

2) Experimental protocol

An appreciation of the type of experiments to be conducted using the PCCL is essential to the understanding of its various design features, and the safety implications of loop operations.

Research contracted with EPRI and ESEERCO for the first several years of operation is devoted to measurement of the effects of coordinated $LiOH/H_3BO_3$ treatment (i.e., effective operating pH) on the production, transport, and deposition of corrosion product radionuclides on ex-core surfaces in a PWR environment. Thus, the experimental procedures are relatively straightforward:

* operate the PCCL for approximately one month out-of-pile to precondition the corrosion film on all loop surfaces;
* move the loop into the MITR core tank for another one- to two-month run under steady state conditions (temperature, heat flux, flow rate) in the presence of neutron and gamma irradiation;
* remove and disassemble the loop to assay the amount and spatial distribution of important radionuclides such as Co-60 and Co-58 on loop surfaces (amounts measured in microcuries are to be expected);

- repeat otherwise identical runs, varying only the $LiOH/H_3BO_3$ ratio, and compare results to activity transport models; use the results as the basis of recommendations to PWR operators for a regimen of coolant chemistry control which will reduce exposure doses. (Improvements by as much as a factor of 10 can be anticipated, based upon the current level of understanding.)

Section 3 of this report provides appropriate detail on the proposed operational and experimental procedures, and Section 7 discusses the subsequent disposal of radioactive waste products.

2. DETAILS OF THE PCCL LOOP DESIGN

a) Design Specifications, Tolerances and Safety Margins

Figure 1.1 above, indicates the layout of the loop components which will be within the MITR-II core tank. Specifications for the major components are summarized in Table 2.1. Specifications most relevant to the safety of the loop and the reactor are the thimble and loop maximum pressure specifications and the heater shut-off mechanisms. Dimensional tolerances are standard machine shop practice except in the case of the thimble-dummy element mating surfaces. For this fit, the dummy element will be measured after fabrication and the critical thimble dimensions specified to allow 0.005 in. clearance, +0.000-0.005.

b) Loop Materials - Compatibility with MITR-II Core and Coolant

Several types of materials issues must be considered in evaluating PCCL safety: compatibility with reactor primary coolant, reactivity, and activation. Reactivity issues are dealt with in Sections 2.d and 4, and activation is covered in Sections 5 and 7. This section deals with the compatibility of loop materials with the reactor primary coolant.

As discussed above, the loop components are encapsulated in an aluminum thimble, which will constitute the major surface in contact with the reactor coolant. The material used will be a certified reactor grade aluminum and is thus within the envelope of materials approved for use in the core tank as specified in the MITR-II Technical Specifications Section 5.3. Apart from the aluminum, small amounts of gasketing material will be used. A Viton O-ring will be used to seal the upper flange at the top of the thimble (above core tank water level), and a pure lead gasket or other metal-to-metal seal will be used to seal a 2" port at the

TABLE 2.1: PWR COOLANT CHEMISTRY LOOP DESIGN SPECIFICATIONS

IN-CORE SYSTEMS

Pump:

Capacity (GPM)	1-2
Design	Canned-rotor
Temperature (F/C)	603/317
Normal Operating Pressure (PSI/Bars)	2200/152
Maximum Design Pressure (PSI/Bars)	3000/207
Loop Differential Pressure (PSID/Bars)	15-25/1-1.7
Material	Inconel/Stainless Steel
Power Supply	220 or 440 VAC, 100 watts Variable Speed

Heater:

Power (variable)	0-20 kW
Power Distribution	Linear
Length (heated section)(in/cm)	24/61
Diameter (in/cm)	0.316/0.803
Sheath Material	Carbon Steel
Voltage (VAC)	220
Shutoff systems	- Automatic shutoff initiated by one of two independent thermo-couple temperature signals. - Manual shutoff by experimenter/reactor operator under loop operating procedures. - Passive shut-off by melting of aluminum link in power line if other systems fail and bath temperature reaches ≈ 1100 °F

Thimble:

Material	6061 Aluminum
Wall Thickness (in/mm)	0.125/3.2
Design Pressure (PSI/Bars)	30/2.1
Maximum Pressure-Loop Leak Accident	100 PSI (relief valve)/ <500 PSI (no relief)
Proof Pressure	750 PSI

Heated Section:
(Simulated Fuel Pin)

Material	Zircaloy 2 or 4
Diameter OD (in/min)	0.312/7.9
ID (in/min)	0.26/6.6
Configuration	"U" Tube
Heated Length (approx)(in/cm)	50/127

Lead Bath Container:

Material	TI 6Al-4V
Wall Thickness (in/mm)	0.032/0.79

Simulated Steam Generator Tube:

Shot-Bed Heat Transfer Medium	Copper Shot
Tubing	Inconel
Diameter OD (in/mm)	0.312/7.9
ID (in/mm)	0.26/6.6

Out-of-Core System.

Charging/Pressurization Pump:

Metering Pump	Positive Displacement
Flow Rate (cc/min)	<1000
Maximum Pressure (PSI/Bars)	3000/207
Back-Pressure Valve	Gas or Spring Loaded
Check Valves	Dual Ball-Type to prevent back flow and depressurization

bottom of the shot bed heat exchanger section (approximately 18" above the top of the MITR-II core). Note that all screws or bolts on the thimble, which could potentially fall into the core tank, will be captured or held so that they cannot fall if inadvertently dropped. Small amounts of gasketing material are permitted under the existing approval, and it is not expected that they will produce any significant impact on the primary coolant. Inconel tubing for feed/bleed flow, and instrumentation and heater and pump power wiring will be fed through the top flange of the thimble and through the core tank wall. These feedthroughs will be above the core tank water level and will be isolated from the core tank environment in polyethylene tubing or stainless steel conduit (as is currently done with other experimental and operational facilities). These components will be subject to splash and humidity from the primary coolant, but will not be continuously exposed. Again, no significant impact on the primary coolant is expected, and the proposed PCCL falls within the envelope of previously approved procedures from the standpoint of coolant compatibility.

It is recognized that the materials which will contact the primary coolant as described above are subject to certification requirements. Procedures for procurement and quality assurance of such materials are described in Section 2.f. The consequences of a thimble leak followed by re-release of primary coolant to the core are discussed in Section 6.b.4.

c) Structural Supports and Loading

Figure 2.1 shows the thimble support system. It consists of a stainless steel bridge bolted to the core tank wall and capable of supporting the full weight of two loops (in air) with appropriate safety margin. Each loop will be attached to this support using two spring-

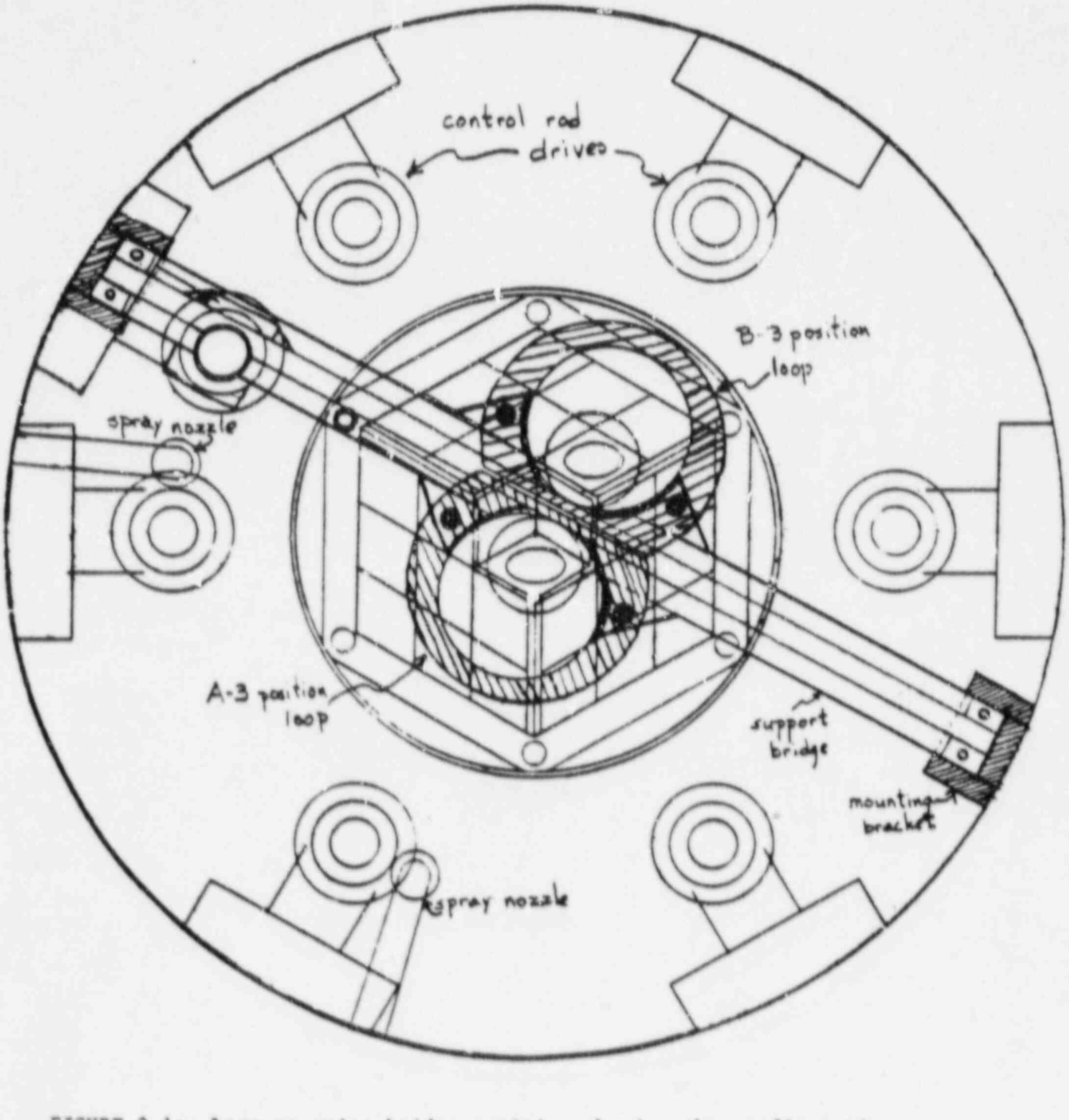

FIGURE 2.1: Loop mounting bridge position showing the configuration for operating the loops in the A-3 and B-3 element positions.

loaded bolts designed to accommodate the thermal expansion of the loop at the operating condition. These supports will be designed to carry up to the full 200 lb. weight of the loop in water. To prevent vibration of the thimble in the flowing primary coolant, a small fraction of the loop weight could be carried on the lower grid plate, or an upward force could be exerted by the hold-down tabs against the upper grid plate as in the case of the ICSA's.

The loop thimble will be secured against ejection from the core by locking of the upper grid plate over a latch tab on the thimble. This is equivalent to the fuel element securement. (Note also that the clearance between the top of the loop thimble and the bottom of the core tank lid will be only several inches, and complete ejection of the loop from the core is therefore impossible even if the locking system fails.)

Provisions against dropping the loop and/or shield structures into the core tank are discussed in Section 3.b.

d) Power Peaking in the MITR-II Core

Computer calculations of the effect of the loop experiment under various operating and accident scenarios are being carried out by Operations staff. Experience with previous experiments such as the FCE suggests that the PCCL will meet the Technical Specification requirements in this regard. A report on the results of the calculations will be made available to MITRSC members as soon as it is completed.

e) <u>Instrumentation and Control</u>

The instrumentation and control requirements for the PCCL are rather simple. Redundant thermocouples on the loop's inlet and outlet plena are used to generate a loop average temperature signal, and power to the heater is varied to maintain constant T-average: much the same approach as used on actual PWR units. Control and safety instrumentation both come in two categories: essential and supplementary. The former category consists of redundant thermocouples which measure loop hot and cold plenum temperatures and heater bath temperature. The PCCL will be cooled down and depressurized unless one thermocouple of each pair is functional. Other instrumentation (such as the flow meter and humidity detector) is used to provide supplementary information, and loop operation may continue should items in this category become inoperable.

The principal safety system consists of redundant thermocouples measuring the temperature of the lead heater bath; an overtemperature signal is programmed to cause the interruption of power to the heater, since this is indicative of a serious accident in progress (see Section 6.b). All severe accidents ultimately lead down this path, and very little damage is done if heater power is cut off.

Most of the other instrumentation on the loop itself is diagnostic in function (e.g., to measure flow or pressure), or for the purpose of logging data pertinent to the interpretation of experimental results (e.g., H_2 and O_2 concentration, pH and conductivity). Readings (and for certain signals, alarms) from this instrumentation can help loop operators identify the specific nature of an incident which disables the loop, but heater bath overtemeprature alone is sufficient to satisfy all safety protection needs.

A parallel set of safety-related diagnostic instrumentation is provided on the thimble gas space: over and under pressure alarms to indicate loop or thimble leakage, respectively, and a humidity gauge which will respond to water ingress in either scenario. The humidity detector (solid state moisture detector) will be mounted out-of-core on a thimble vent stream obtained through a capillary bypassing the thimble pressure relief valves. This allows for periodic testing and replacement of the detector without interrupting loop operations, which is important given the relatively low reliability of such detectors. The long response time of this system and the reliability question preclude automatic loop shutdown from this signal. Considerable time is available for deliberate action by the operator subsequent to most malfunctions; those rare sequences that are more serious will be interdicted by the heater bath overtemperature protection system.

All alarms for the PCCL which require reactor operator response will be brought to a common panel in the control room. Breakers for emergency shut-down of the loop heaters and pumps will also be provided in the control room. (See Section 3 for typical response sequences.) It should be emphasized again, however, that operator response is not required for reactor safety, and in most cases is aimed only at minimizing consequences to the loop equipment.

f) Quality Assurance Requirements

The PCCL will be designed, constructed and installed in conformance with the MITR quality assurance (QA) program. A copy of the relevant parts of this program is appended to this report. Under this program, a QA file will be maintained in the Reactor Operations office incorporating: material certifications, design and construction drawings, safety experiment and proof testing documentation and procedure approvals.

It should be noted that a hierarchy of PCCL systems with varying impact on reactor operation and safety can be established. Materials which contact the primary coolant or which are exposed to significant neutron flux are the most critical and must be the most stringently controlled. Materials within the MITR-II core tank but not in the above two categories will have somewhat less stringent certification requirements. Safety-related equipment such as pressure relief valves and the overtemperature protection system (including thermocouples, relays and the fusible link) will be subject to testing and calibration requirements.

Authorization of personnel to carry out critical loop operations, and to carry out or approve quality assurance procedures, will consist of a letter signed by one of the project co-principal investigators and by the Director of Reactor Operations.

3. OPERATIONAL AND EXPERIMENTAL PROCEDURES

In this section the general procedures for system operation and the conduct of experiments are outlined. Since the PCCL has been designed to achieve and maintain an invariant steady state status for up to several months non-stop, these procedures are fairly simple. Except for a few hours during startup and shutdown, the loop is under automatic control, and those responsible for its operation need only make occasional (less than daily) small adjustments in power level, pump speed (i.e., flow rate), and feed-and-bleed rate. Even less frequent changes will be made in thimble helium pressure and makeup tank hydrogen overpressure. Most data of significance is measured by built-in instrumentation and logged by computer, and makeup/discharge tank samples need only be drawn for supplementary analyses on a weekly basis. The most significant data from the point of view of the experimentalist will be recorded by the on-line deposition monitor, and measured by post-mortem dissection and gamma scanning of the loop's tubing.

In what follows, the procedures are outlined in narrative form; detailed step-by-step checklists will be prepared for use by the loop operators and the MITR-II operating staff.

a) Loop Operations

Each run will begin with reassembly of the loop, usually using new in-core Zircaloy and out-of-core Inconel tubing. The loop will then be installed in an out-of-pile tank for its preconditioning run. The objective of this phase is to establish a significant corrosion product film on all internal surfaces. Operation will be virtually identical to subsequent in-pile operation (i.e., at full pressure and temperature, hooked up to the same control and auxiliary systems as will be used to support its in-pile operation), except that the thimble helium space will be evacuated to permit running at very low power, simulating "isothermal" operation. Full power operational tests will also be undertaken before transfer of the loop to its in-core position.

Since the PCCL will then be moved into the MITR-II core tank with as little perturbation as possible, the preconditioning run--about one month long--is an excellent "shakedown cruise," which should help insure that all systems are operating flawlessly before each in-pile run in initiated. Transfer from out-of-pile will be effected with the loop disconnected (and sealed off) from the feed and bleed train, in a cooled-down and depressurized state. The insertion procedure will include the use of mechanical stops to prevent the application of "missile forces" to core structures if the thimble is dropped. The thimble helium pressure of 30 psia will be maintained during transfer operations, since out-of-pile operations at power serve to verify that air was not left in the thimble prior to helium back-fill (see Section 6.b.3). Thimble outside diameter measurements will be made at this point ot ensure that shot-bed ratcheting is not occurring. After it is emplaced and secured in the core tank (see Section 2.c), all fluid, power and instrumentation lines will be reconnected, and the MITR-II button-up/startup can then proceed as normal.

In parallel with MITR-II startup the PCCL is then pressurized cold using its feed-and-bleed system. Next the PCCL's in-pile heater is turned on and power is ramped up, at rates set in the operating procedures, to its normal full power rating (10-20 kW) as specified for the particular experimental run which is scheduled. During heatup, thimble helium is vented to keep its pressure at about 30±5 psia.

At this point loop heater power/loop temperature control is put on automatic. As the MITR-II itself comes up to power, gamma heating will gradually assume about 10% of the PCCL heat load, and electric power will automatically be reduced to keep the loop at steady state with respect to coolant temperature. From this point on the objective is to hold the established conditions until the PCCL is shut down for removal (in 1-2 months). Over weekends, when the MITR-II is shut down, the PCCL control system will compensate for the reduction in gamma heating.

Of more interest as regards a safety analysis is the operator response in the event of an accident which disrupts normal loop operation. Sections 4 and 6.b discuss such sequences in more detail. Here we confine the discussion to operator action. Table 3.1 summarizes the responses appropriate to a variety of incidents. Note that in each case the loop control system is programmed to execute a sufficient action (usually cut-off of heater power) to put the loop in a safe state, and hence operator action is of the nature of confirmation and backup. Heater bath temperature will indicate and alarm in the control room. In the event of heater control (automatic power cutoff) failure, operator control is important, but a passive ("fusible link") heater shut-off is provided to avoid serious overtemperature incidents without any intervention.

b) <u>Post-Operation Handling</u>

Following completion of a PCCL irradiation the loop will be removed from the reactor and transferred to a shielded test stand where it can be disassembled for analysis. This section outlines the procedures to be followed for removal, disassembly and analysis, with

TABLE 3.1: INCIDENT RESPONSES

Event	Symptom/Alarm	Automatic Control	Operator Response
Loop rupture/ loss of coolant Relief valve open	- High humidity - Low loop pressure - High heater bath T	Cut-off power to heater	Open circuit breakers for heaters Turn off charging and circulating pump
Severe thimble leak	- Low He P - High humidity	Cut off power to heater	Open circuit breaker for heater; execute cooldown
Failure of makeup/ pressurization pump	- Low loop pressure	Cut off power to heater	Open circuit breakers for heaters and pump
Heater control failure (overheat)	- High Pb bath T - High loop T	(Heater control failure postulated) but T alarms work*	Open circuit breaker for heater
Failure of circulating pump	- Low loop flow - High Pb bath T	Cut off power to heater*	Open circuit breakers for heaters and pump
Computer Failure	- Loss of all signals and control capability	Fail-safe feature: heater power cutoff	Execute cooldown
Loss of all Cambridge electric power	- MITR control room procedures relied upon	Heater goes off; system is designed to prevent heater restart when power is restored	Open circuit breakers for entire system; consider restart in conjunction with experimenter.

NOTES: Cutting off power to the pump, where indicated, is to protect the pump against damage and is not a required safety procedure per se.

Emergency electric power is not required in the event of a local or global power failure. Relief valves are all spring-loaded, and energy removal in extremis is by radiation.

*In the event of failure of both automatic and manual shut-off of the electric heaters, the fusible link in the heater power supply line will melt and cut off heater power.

emphasis on the measures used to prevent possible damage to the reactor and to minimize irradiation dose to personnel.

1) Loop removal

Loop removal will take place on or after Monday, allowing about sixty hours of decay time from Friday evening shutdown. (See Section 5 for an estimate of the radioactive inventory at this time.) The heater power will be ramped down and loop depressurization will follow when the temperature has dropped sufficiently. With the loop cold and depressurized and the core tank lid removed, the power and instrumentation leads will be disconnected, freed from the core tank wall feedthrough and secured to the thimble head. The feed and bleed and helium pressurization lines will be disconnected at the thimble feedthrough and capped.

A lifting harness will then be fastened to the eyebolts on the thimble head, and the harness placed on top of the thimble lid so that it may be reached later. The lifting harness has three lines, with two of them adjustable, so that the loop hangs vertically. The nuts which hold the thimble down to its support bridge are now removed. At this point a special core tank lid is to be lifted into place by the overhead reactor crane. This lid is larger than the maximum opening to the core tank, and fits over the studs protruding from the top of the core tank. A single port in the lid is positioned directly over the thimble to be removed so that in lifting the loop from the reactor nothing may be dropped into the core tank. Two three foot long alignment rods are then screwed into the pins on the support bridge so that in lifting the loop from the core (and more importantly when putting it in) the loop does not rotate or tip, which might cause it to wedge into the dummy fuel element. The rods (and all other hardware over the core tank) will be secured to prevent their falling into the reactor. The loop lifting wires will then be threaded through a shielded cask (similar to the cask used by Operations for control blade removal) which is then positioned on the lid to allow the thimble to be drawn through.

Loop removal will be performed using the 3-ton hoist. Once the pump pod at the top of the thimble has cleared the top of the cask, additional shielding with a 4 in diameter central hole (to fit closely around the 4 in. thimble section) will be attached to the top of the cask. This structure also serves as a stop by supporting the pump "pod" at the top of the thimble if the thimble is dropped. When the lower thimble section is in place in the cask, a support frame for the upper thimble will be bolted to the cask.

Having secured the loop in this frame the operator will detach the loop from the lifting hoist and connect cask lifting cables to the large reactor crane. The cask containing the loop will then be moved over and down to the reactor floor (just to the right of the hot cells, against the containment building wall).

The thimble lid may now be removed and all the electrical and fluid lines to the loop disconnected. The pump will be drained through a tee fitting and the water will be collected for analysis. Once the pump is empty it should be completely disconnected and removed from the thimble head. The open end of the tee should be capped and a line connected to the open end of the loop. The water in the loop may then be pumped out through the tee using helium pressure. Both lines may then be removed and the loop ends capped.

A small hand hoist fixed to the wall above the loop is then attached to the loop itself. At this point all operations in the line of the radiation beam have been completed and so the copper shot which has been serving as shielding may be drained through the shot drainage port at the bottom of the heat transfer section of the thimble. Care will be taken to contain the shot. The loop is now ready to be removed from the thimble. Since the in-core section of the loop will not be immediately surrounded by shielding during this procedure all non-essential personnel should be away from the area and the operator will be shielded by a concrete wall. The operation may be observed through a video camera and monitor. The loop is to be removed from the thimble to a position over a disassembly rack next to the loop stand. The hoist will move by pivoting its bracket between two well-defined stops. Cables will run from the bracket through the stops and down to the operator position where they will be secured. When the loop is over the disassembly rack, it is lowered into a plastic sock (for contamination control) with the in-core section going into a transportation cask. Once within the cask the in-core section may be disconnected from the rest of the loop at its SWAGELOK® fitting and heater connection; this will be done using tools designed to avoid personnel exposure in the radiation beam from the in-core sections. The cask containing the activated in-core section is then rolled out from under the rest of the loop and a lid may be placed over the top.

2) Loop disassembly

Disassembly and sectioning of the in-core section of the loop is to be carried out in the reactor floor hot cell. The transport cask containing the activated section will be lifted to the top of the hot cell and positioned over an access port. A small power supply will be connected to the heater and a lowering bracket attached to the top of the Zircaloy tubes. By opening a sliding door in the bottom of the cask the in-core loop section is then lowered down into a holder in the hot cell. The lead in the titanium can is then melted by the heater and the Zircaloy lifted out. Sectioning and characterization is then carried out using the hot cell remote manipulators and tools prepared for this purpose.

Activities in the loop water and the out-of-core loop sections are expected to be easily manageable, because the activities involved are small. Contamination control is critical to the quality of experimental data acquired, and such control can be achieved through careful application of standard decontamination procedures. Analysis of the tubing will be performed within the exclusion area, avoiding the removal of this activity.

4. MAXIMUM EFFECTS OF REACTIVITY, PRESSURE AND TEMPERATURE

In this section a conservative set of maximum effects will be estimated. In many cases a highly improbable sequence of events would be required to create a situation leading to the effect in question, but our interest here is more in establishing an envelope of limits rather than examining a variety of more plausible scenarios.

a) Reactivity Effects

As with all other in-core experiments, it is essential that the PCCL meet the Technical Specifications (Section 6.1, see Appendix 2) with respect to its potential effect on MITR core reactivity during postulated accident scenarios. In the case of the PCCL two phenomena are of particular concern: the presence of B-10 (in the form of dissolved boric acid) in the simulated PWR coolant contained within the PCCL circuit; and the reactivity worth of loop and thimble water contents during void/reflood incidents.

1) Reactivity effect of B-10

Two hypothetical scenarios of increasing severity are postulated, involving:

a) sudden voiding of the in-pile water and its contained maximum (2000 ppm) boron content

b) a non-mechanistic incident in which all in-pile boron is first concentrated in-pile and then ejected

To minimize tritium production and reactivity effects, boron enriched in B-11 will be used in experiments; the B-10 content is 2 w/o instead of the natural value of 20 w/o. Strict administrative controls will be imposed to prevent inadvertent substitution of natural boric acid. In addition, the tritium concentration of the PCCL coolant inventory will be assayed weekly; among other things, it will serve as a positive indication of B-10 concentration via the $^{10}B(n,^{8}Be)^{3}T$ reaction.

We then have for our three scenarios:

Case	H_2O Involved	Total B	B-10
(a)	40 g	8.0×10^{-2} g	1.6×10^{-3} g
(b)	500 g	1.0 g	2.0×10^{-2} g

Appendix 1.b develops an estimate for the reactivity worth of B-10 in the MITR-II core:

$$\left[\frac{\Delta k}{g}\right]_{B-10} \approx 0.998\%$$

Thus, we have:

Case	Scenario	Max. % $\Delta k/k$
(a)	Eject all in-core B-10	1.6×10^{-3}
(b)	Eject all PCCL B-10	2.0×10^{-2}

As can be seen, cases (a) and (b) fall well within the allowable limit for a movable in-pile experiment, namely $\Delta k/k \leq 0.2\%$. Since we will assay PCCL discharge tank water weekly for boron content and since the feed/bleed rate is only 5 kg H_2O/wk, it is not conceivable that hideout in-pile of all of the charging tank inventory would go undetected.

Also, note that the effect of voiding boron and water are opposite in sign; since boric acid is highly soluble, realistic scenarios which would decouple these compensatory effects are hard to imagine.

2) Water flooding/voiding incidents

There are three scenarios of interest here:

(c) sudden flooding (preceded by undetected voiding) of all in-core PCCL inventory (40 cc = 40 g under cold conditions)

(d) loop rupture and drainage into the in-pile thimble ($\leq$ 75 cc = 75 g, again cold) or thimble rupture

(e) undetected voiding of the volume between the thimble and the dummy element, followed by sudden reflooding ($\leq$ 100 cc = 100 g)

For small water voids, we have:[1]

$$\left[\frac{\Delta\rho}{g}\right]_{H_2O} \equiv \left[\frac{\Delta k/k}{g}\right]_{H_2O} \leq 2 \text{ milli } \beta/g$$

Thus, we find for $\bar{\beta}$ = .00786, $\%\frac{\Delta k}{k}/g \leq 1.57\times10^{-3}$

Case	Scenario	$\%\Delta k/k$
(c)	PCCL tube flooding	0.062*
(d)	Loop drainage into thimble	0.12*
(e)	Dummy/thimble channel reflood	0.16

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[1]Personal communication, J. Bernard, October 6, 1986, and his memorandum to O. Harling dated October 7, 1986. See also MITR-II Start-up Report for Core IV.

*Note that all values exclude the opposing effect of any B-10 dissolved in the water involved.

All these cases are within the allowable reactivity restriction for a movable MITR-II experiment, namely $\Delta k/k \leq 0.2\%$. Note that measurements show that flooding a 1.75 inch ID ICSA would add 0.982% $\Delta k/k$, for an effective <u>large</u> void coefficient of ≈1.51 milli β/g.[1] This figure is less than the <u>small</u> void coefficient of 2.0 milli β/g used above.

The following additional qualifiers to the preceding analyses should be noted:

1) The potential events (sudden, undetected flooding) can only occur in their most severe version when the PCCL is in a cold startup/shutdown mode; otherwise, the in-pile heaters (and energy stored in their molten lead bath at $\geq$ 600°F) would immediately boil any water in the (low-pressure) thimble's in-pile gas space. Therefore, it will be normal practice (but not an absolute requirement) that the PCCL power level exceed 10 kW whenever the MITR is critical. This is not a severe imposition, since the anticipated experimental protocol involves non-stop steady state runs of a month or more in duration with loop insertion/removal over weekends.

2) The helium atmosphere in the PCCL thimble will be maintained at 2 atm (30 psia), so that out-leakage of helium would take precedence over in-leakage of MITR cooling water (maximum pressure at the bottom of the core--atmospheric plus hydrostatic of ≈ 22 psia) if there is thimble failure.

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[1]Memo from L. Clark to MITRSC dated July 5,1978.

3) The thimble will be tested for leakage (visual indication of gas bubbles) at 50 psia internal pressure at the end of its out-of-pile preconditioning run, prior to being transferred into the MITR core tank. Post-manufacture and periodic testing with a helium leak detector will also be performed.

4) A humidity detector is installed in the thimble's vent line to provide indication of the presence of small amounts of water ingress. An alarm will be sounded in the MITR control room to warn of a potential problem.

5) Most importantly, the titanium can housing the in-pile heater bath fits snugly into the aluminum thimble at its top (except for small grooves to admit helium to this region). Thus, rapid drainage of either PCCL or MITR water into this region from above is unlikely. Hence, only sudden massive failure of the conservatively designed lower thimble is a plausible maximum accident initiator.

Based upon the very conservative analyses documented above, there does not appear to be any way in which installation and operation of the PCCL can exceed MITR-II Technical Specifications with respect to reactivity effects. A confirmatory measurement of the reactivity difference with and without water in the PCCL's Zircaloy in-pile tube will be made as part of its initial checkout.

b) Pressure Effects

In its normal operating state, the PCCL consists of roughly 1/2 liter of hot (≈600°F) water under ≈ 2,200 psia pressure inside a loop comprised mainly of 0.25 inch ID tubing, all surrounded by an aluminum thimble with approximately 1 ft^3 free volume containing helium gas at ≈ 30 psia.

The high pressure part of the system is, because of its configuration (small diameter), small volume, and encapsulation, less hazardous than common corrosion test autoclaves; and a small amount of helium at two atmospheres is also not a safety problem. Thus, normal operation does not create any situations of concern, and one must turn to low-probability accident scenarios to postulate consequences of potential interest: the most severe being an instantaneous loop rupture or LOCA (loss-of-coolant accident).

A particularly simple, but quite realistic, calculation of an upper limit on the pressure attainable in the thimble following instantaneous release of the entire loop water inventory can be made when one recognizes that the thermal balance is dominated by the large mass of copper shot (≈200 lbs), which completely overwhelms the small amount of water involved (≈1 lb) and the even smaller mass of helium present (≈ 3 ft^3 at STP) and even the ≈ 3 lbs of molten lead at ≈ 700°F in the in-pile heater bath.

The average temperature of the shot bed remains virtually unaltered throughout at ≈ 350°F. Enough stored energy is available to vaporize all water not immediately flashed to steam upon loop depressurization. Thus, we need only compute the pressure of the helium-steam mixture at 350°F.

At 350°F saturated steam has a vapor pressure of 135 psia and a specific volume of 3.342 ft^3/lb; hence, compression of 1 lb into 1 ft^3 (loop-free volume) would require a pressure of approximately 135 x 3/342 ≈ 451 psi; to this, add the initial (and final) helium pressure of 30 psi ≈ (at 350°F) to obtain a post-blowdown pressure of 481 psia in the thimble.

This value is conservative because it ignores condensation of steam on the cold (≈100°F) aluminum walls of the thimble. Even so, it is a quite tolerable pressure. Moreover, in the interests of design conservatism, the loop is protected by a relief valve which will vent the thimble at $\leq$ 100 psi, and a back-up rupture disk which will burst at ≈ 500 psi.

Note that the thimble and all of its contents are designed to withstand a full vacuum, hence cold conditions in which steam condenses after expelling the helium fill gas will not lead to additional problems. In fact, when operated in its "isothermal" pre-conditioning mode out-of-pile, the loop gas space is evacuated to increase the thermal resistance of the shot bed, and permit high temperature operation at "zero" (actually very low) power.

c) <u>Temperature Effects</u>

The design philosophy underlying our approach to PCCL safety during extreme accident scenarios has been to provide for sufficient passive (i.e., radiation) cooling to prevent material temperatures from ever exceeding recognizably safe values: for example, conforming to the same $\leq$ 2,100°F post-LOCA limit on Zircaloy temperature as is imposed on actual PWR units by the NRC. Appendix 1.a gives an estimate of the gamma and fast neutron heating of the in-core section, with experimental data and calculations to show that the total radiation heating can be dissipated at maximum temperatures < 1500°F.

The combination of active and passive (aluminum fusible link exposed to the lead bath) shut-off mechanisms for the electric heaters is expected to prevent heater power being applied at lead bath temperatures above ≈ 1500°F. The nuclear heating analysis is therefore sufficient to show that Zircaloy-water reactions will not occur.

5. RADIATION LEVELS AND ALARA CONSIDERATIONS

The PCCL components which will be exposed to significant neutron flux in-core are: Zircaloy tubing (Zircaloy-2 or Zircaloy-4), pure lead for the conductive bath, electric heaters consisting of Nichrome heating elements with magnesium oxide insulation and carbon steel sheaths, a titanium-aluminum-vanadium alloy can containing the lead bath and the lower section of the aluminum thimble. Stainless steel sheated thermocouples (chromal/alumel) in the lead bath and heaters will also be exposed to neutron flux. Table 5.1 gives an inventory of the elements present in-core in the proposed PCCL.

Apart from the Zircaloy tubing, which is necessary for the simulation of the PWR flow loop, all in-core component materials were chosen to minimize activation within the constraints imposed by the functional requirements of a given component. In several cases, the activation of the components is due largely to impurity elements, leading to some uncertainty in prediction of the activity levels. Activation experiments have been performed in an equivalent core position to that proposed for the PCCL to assist in estimating post-irradiation activity levels. Table 5.2 gives estimated dose rates after a one-month irradiation and 60 hours' decay.

The activities present lead to a total unshielded gamma dose rate of less than 10 R/h after a typical one-month irradiation and decay from reactor shutdown at 1800 hours on Friday to Monday morning loop removal. This dose level is within the range of experience of MITR operators and radiation protection, and the loop handling, shielding and procedures discussed in Section 3.b will keep personnel doses small for transfer and disassembly operations. A small amount of the α-emitter polonium-210

TABLE 5.1: IN-CORE MATERIALS INVENTORY FOR THE PCCL

Component	Material	Composition (wt.%)	Wt. (g)
U-tube	Zircaloy-4	1.45Sn, 0.21Fe, 0.10Cr, bal. Zr and impurities	110
Heater	1018 Steel	0.20C, 0.90Mn, bal. Fe and impurities	85
	Inconel 600	76Ni, 0.25Cu, 8Fe, 15.5Cr, 0.25Si, 0.5 Mn	35
	Magnesium oxide	60.3Mg, 39.70 (high purity)	180
Conduction bath	Lead	99.9 Pb (<5ppm Bi)	6000
Containment tube	Titanium	99.5 Ti, bal. impurities	240
Thimble	6061 Aluminum	0.8Si, 0.7Fe, 0.4Cu, 0.15Mn, 1.2Mg, 0.35Cr, 0.25Zn, 0.15Ti	670

TABLE 5.2: ESTIMATED UNSHIELDED GAMMA DOSE RATES FOR PCCL IN-CORE COMPONENTS AFTER 20 FULL POWER DAYS AND 60 HOURS' DECAY

Component	Principal Activities	γ Dose Rate (R/h at 1M)
U-tube	^{95}Zr, ^{51}Cr	1.5
Heater	^{58}Co, ^{60}Co, ^{59}Fe, ^{51}Cr	0.3
Conduction bath	^{203}Pb, ^{203}Hg	4.0
Containment tube	^{46}Sc	0.2
Thimble	^{24}Na, ^{58}Co, ^{60}Co, ^{59}Fe, ^{51}Cr	1.0
		7.0

NOTE: The dose rates expected are strongly dependent on impurity content in several components. It will therefore be necessary to recheck the activation data with the actual loop materials when these become available.

will also be generated in the lead bath, principally from bismuth impurity. Polonium production in reagent grade lead has been estimated from experiments at ≈ 1 μCi/kg lead for a one-month irradiation. The plug which isolates the lead bath from the upper thimble is expected to be an effective barrier (by condensation and adsorption) to polonium, and all lead bath disassembly operations will be performed in a hot cell. Personnel exposure to polonium-210 is therefore not expected, but care must be taken in handling the Zircaloy tubing section, which may take up some polonium. Component handling procedures used will call for routine wipe tests and alpha counting.

It should be noted that the out-of-core sections and the loop water are expected to have only μCi levels of activity. Doses from water chemistry analysis and post-irradiation inspection and gamma counting of the Inconel sections will be insignificant once these sections are disassembled from the in-core section in the shielded test stand. Tritium production in the loop will be minimized by the use of lithium hydroxide and boric acid enriched to 99.9% Li^7 and 98% B^{11}, respectively. Since the flow rate through the bleed capillary is slow (≈30 cc/h), virtually no N^{16} activity will be transported outside the reactor core tank during loop operation, and no effect on core top radiation levels is expected. Personnel exposure during loop operations will be limited. Routine operation will require experimenters to be present on the reactor top and loop platform for approximately one hour daily, resulting doses far below the allowable limits for each worker.

Procedures for all loop operations will be developed and implemented with shielding, ventilation and appropriate controls to insure that radiation exposure to all personnel is as low as reasonably achievable.

6. PCCL SAFETY EVALUATION

a) Safety and Operational Envelopes

To facilitate the review of the safety considerations for the PCCL, it is helpful to make specific reference to the MITR-II Technical Specifications that where written to provide an envelope within which the MITR Safeguards Committee and MITR staff can approve experiments. Pages 6-1 to 6-7 from Section 6 and Section 5.3 of the Technical Specifications are provided as Appendix 2. The PCCL design and operation as described in this Safety Evaluation Report is to be in conformance with these specifications.

In considering the effect of normal PCCL operations on MITR operation, the following points should be noted:

--The gamma energy deposited by the MITR in the PCCL would otherwise be deposited elsewhere in the MITR structure; hence there is no increase in total heat load from this phenomenon--merely a redistribution (which is usually beneficial).

--In incidents severe enough to warrant scramming of the MITR, gamma heating of the PCCL is reduced proportionally. Hence the PCCL does not aggravate the consequences for its host reactor.

--Accordingly, only the energy added by the electric heater (PCCL pumping energy is negligible) need be considered in assessing the impact of the PCCL during steady state, transient or accident scenarios.

Furthermore, investigation has confirmed that insertion and operation of the PCCL will not interfere with proper operation of the MITR-II cooling system under normal conditions. In particular:

-- The transition between the 2 in. diameter lower thimble and the 4 in. upper thimble is gradual, and ≈12 in. above the core, so as not to perturb in-core flow patterns. Pre-operational tests will attempt to verify that core flows are not significantly altered.

-- Total PCCL power input, 10-20 kW, is comparable to the reactor decay heat by Saturday morning following Friday evening shutdown (≈25 kW). Provision for additional decay heat removal will therefore be necessary.

-- Mechanical interference with control rod drives is ruled out because the PCCL is firmly captured, both in the core grid plate and at the PCCL support bridge across the reactor top, in a position with clearance all around.

-- Interference with the emergency core cooling sprays (ECCS) will be prevented by relocating one spray nozzle to ensure that no areas of the core are shadowed by the PCCL loop or loops. Operations staff is investigating the necessary physical and procedural changes.

b) Malfunction Sequences and Consequences

As noted in the preceding sections of this report, design features have been incorporated in the PCCL to either preclude, limit, or mitigate the consequences of severe malfunctions or misoperation. Nevertheless, there are plausible sequences which cannot be ruled out and which therefore merit detailed discussion and analysis here.

In this category are the following:

(1) loop leak

(2) loss of PCCL pump

(3) leak in hydrogen cover gas system

(4) thimble leak

(5) lead bath can leak

Each of these scenarios is examined in some detail in the subsections below.

1) Loop leak

The most likely site of leakage in the PCCL is at one of the several SWAGELOK® fittings, and leakage is likely to be at a slow rate. Some of the hot (600°F), high pressure (2200 psia) water will flash to steam immediately and the remainder will be vaporized if the water contacts hot metal in the thimble shot bed or the in-core heater lead bath. No undue safety consequences will result for several reasons:

(1) If called upon, the thimble's pressure relief valve will relieve pressure at < 100 psia and the back-up rupture disk will burst at ≈ 500 psia. (Failure to relieve could result in ultimate thimble pressure ≤ 500 psia, see Section 4). The thimble will withstand 500 psia with adequate safety margin.

(2) Steam will condense (thereby reducing temperature and pressure) on the cold thimble wall.

(3) Leakage will normally be detected by the humidity detector, located on a gas bleed line from the thimble's gas space.

(4) The reactivity consequences of either losing or adding the maximum amo nt of water and boron in the in-core thimble are acceptable (see Section 4).

Although loop leakage is not serious from a reactor safety viewpoint, continued operation of the experiment is inadvisable. Hence, following positive confirmation of significant leakage, an orderly shutdown of the PCCL will be carried out. Electric power to the in-core heater will be cut off and the temperature allowed to fall to its equilibrium value--which is expected to be about 250°F when only gamma heating has to be removed.

At this point, the loop can be depressurized and allowed to boil dry. After this, gamma heating is dissipated by radiation to the thimble wall. This mode is maintained until the next regularly scheduled weekly MITR shutdown, at which time the PCCL is removed. Boiling dry is preferable to continued, wet operation, to avoid either flooding the thimble solid or damaging the loop's canned rotor pump by cavitation.

Small, steady leaks cannot cause any damage of consequence in the upper thimble because the loop is surrounded by a shot bed which will absorb the energy in water or steam jets; similarly, a cast ceramic insulator surrounds the inlet and outlet plena. Leakage into the lead heater bath is more serious since it will result in instantaneous flashing into steam. While the energy is absorbed by the lead, it will churn and spatter the bath. To allow for this, the upper foot-long section of the titanium can housing the lead bath is empty, and a baffle/plug is used to cap this container. Note that water injection into molten lead is a well studied and benign process which forms the basis of liquid-metal-type MHD generators now under development for solar and nuclear applications.[1]

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[1]L. Blumenau <u>et al</u>., "Liquid Metal MHD Power Conversion Systems with Conventional and Nuclear Heat Sources," 24th Symposium on Engineering Aspects of Magnetohydrodynamics (SEAM), June 24-27, 1986.

2) Loss of pumping power

Without circulation, the coolant in the PCCL will overheat, even after the coincident cut-off of the heater power by the loop control system. The loop pressure will rise to 2,500 psia, at which point the relief valve will lift and depressurize the system, following which it will boil dry. Loop removal can again be postponed until the next MITR shutdown.

3) Hydrogen leakage and combusion

It is standard practice on all PWR units to maintain a small concentration of dissolved hydrogen ($\leq$ 50 cc @ STP/kg H_2O) in the primary coolant to suppress water dissociation. Since the PCCL will, in general, be operated under representative PWR conditions, it will be necessary to adhere to this practice. To accomplish this, a small-volume, low-pressure, hydrogen flask is used to provide H_2 cover gas for the makeup water storage tank. This will maintain 50 cc/kg of H_2 in the $\leq$ 30 kg of makeup water (and in the < 1 kg of water in the loop circuit it serves); hence, a total water-borne inventory of approximately 1,500 cc (STP) will be present--a virtually negligible amount.

The maximum H_2 in service, however, will occur when the makeup tank is in a near-empty status, in which case its volume (30,000 cc = ft^3) will, at 3 atm, contain some 3 ft^3 (STP) of H_2. Accordingly, we will restrict the hydrogen inventory of the transfer flask to $\leq$ 10 SCF by design and administrative controls. These controls will include locking the transfer flask in place, restricting access to the parent bottle to authorized loop operators, and maintaining hydrogen inventory records.

Thus, the maximum instantaneous H_2 combustion incident, which would occur outside the reactor biological shield, could conceivably involve a maximum of 10 SCF; the heat of combustion of H_2 is 343 Btu/ft^3. Hence, the maximum energy release is ≈ 3,400 Btu (a kWh)--about the same as would be generated in combustion of one cup of fuel oil: a tolerable amount as long as not suddenly released in a confined space.

Various safety features are provided either to help preclude or to mitigate the consequences of even such small-scale events:

. The charging water makeup tank has a hydrogen-oxygen recombiner circuit attached since we need to reduce feedwater O_2 to $\leq$ 1 ppb. Hence, except during initial fill operations, there will not be enough oxygen to support significant, confined combustion. During filling, the gas space will be << 1 ft^3, and partially evacuated before the first addition of hydrogen.

. The discharge tank is vented through a flame-arrestor to the MITR ventilation system. The rate of water (hence hydrogen) discharge is extremely low: $\leq$ 30 cc/h H_2O, thus $\leq$ 50 cc H_2 per day.

. A small fan powered by a spark-proof electric motor will be mounted near the hydrogen transfer flask and the makeup/discharge tanks to prevent the local accumulation of a combustible gas mixture.

. A helium atmosphere is maintained in the thimble gas space.

In view of the above considerations and precautions, the combustion of hydrogen in the systems associated with the feedwater and H_2 supply tanks, and all other connected components outside the reactor biological shield, is not considered a credible hazard to either personnel or equipment. The amount of hydrogen in the water circulating in the loop (i.e. the normal hydrogen inventory within the biological shield) is equivalent to approximately 20 mg of TNT, based on relative heats of combustion.

In reviewing possible failure or accident scenarios, we considered another scenario which postulates a water leak in the high pressure PCCL located inside the thimble which is located in the core tank. If this leak goes undetected, water from the makeup tank and eventually hydrogen gas might be pumped into the thimble. If at the same time oxygen were available from some source, e.g., if the thimble had not been purged of air and filled with helium, as would normally be done, a combustible gas mixture could result.

The above scenario postulates simultaneous occurrence of a loop leak, failure of several sensors and experimenter inattentiveness, as well as the availability of oxygen in a space normally purged with helium. This scenario is very unlikely. However, since the energy released during an optimal rapid burn of a significant fraction of the hydrogen inventory of the supply tank could be significant, it will be necessary to show that it does not compromise thimble integrity, or that by suitable design and instrumentation of the PCL system this scenario can be made incredible.

Our approach to designing the PCCL system so that a significant fraction of the inventory from the hydrogen flask cannot enter the thimble is as follows:

a. PCCL experimental thimbles will not be inserted into the core tank until a preconditioning run has taken place outside the core tank. This will insure that air has been properly purged from the thimble because shot bed conductivity and loop temperatures will be sensitive to relatively small amounts of air in the helium cover gas.

b. A high quality instrumented level indicator will be installed in the makeup or charging tank. This will assure that the water level in the tank never approaches the bottom of the tank. It is only after this tank has been emptied of water that significant amounts of hydrogen could be pumped into the thimble by the charging pump.

c. The amount of water in the charging tank, even at the end of an experiment, will be more than the in-core thimble volume. This assures that a loop leak can conceivably only result in filling the thimble with water and not with gaseous hydrogen.

d. A float operated check valve can be installed in the bottom of the charging tank to prevent pumping hydrogen when the tank is empty. This will be added if the other measures cited here are not deemed adequate.

Note that these measures are independent of the instrumentation and procedures which would normally detect a loop leak, which must precede or accompany hydrogen ingress to the thimble. The humidity detector and the loop temperature would respond rapidly to loop leakage.

Finally, a burst disk will be installed in the top of the in-core thimble to guard against overpressurization from whatever source.

4) Thimble leak

The only potentially troublesome leakage event would involve a sudden large rupture in the thimble below the water level in the MITR pool. Since the helium gas in the thimble will be maintained at 30 psia, small leaks will involve egress of helium rather than ingress of water.

A large rupture is highly unlikely--no credible mechanistic sequence has been identified leading to this event. If it did occur, however, the consequent loss of helium would permit flooding by water. Initially the hot metal in the shot bed and lead bath would cause some of the water to flash to steam. Eventually cooldown would progress to the point where flooding of the thimble would persist. As discussed in Section 4, flooding of the fill space between the heater bath and the thimble does not add a consequential amount of reactivity. Similarly, the volume between the thimble and dummy fuel element is insufficient to permit the addition of unacceptable reactivity upon reflooding after helium or steam is expelled into this gap.

Upon detection of significant thimble leakage, PCCL heater power will be cut off and the loop operated in a cooled-down mode, but at its normal pressure, until its removal at the next regular MITR shutdown.

Another possible consequence of thimble leakage is related to the exposure of the primary coolant water to the loop internals, with subsequent re-release of some of this water to the MITR coolant. Since internal and external pressures are equalized, no large driving force exists for this process. The solubility product of $Cu(OH)_2$ is approximately $10^{-20.5}$; thus in pure water only about 20 μg Cu/liter would be expected to solubilize. A completely flooded PCCL shot bed would contain less than ten liters of water. Hence considerably less than one milligram of copper should be introduced into the MITR coolant if thimble in-leakage were to drain back out--during removal of the damaged loop, for example. (The extremely small solubility of copper in water is attested to by its extensive use as a durable anti-fouling sheathing for wooden ships in the 19th century.) Diffusive release would be many orders of magnitude smaller. Note that this small amount of copper is the only contaminant of concern in a back leakage incident. No long-lived radionuclide production would result, corrosion of core tank structures would not be significantly accelerated, and the MITR ion exchange purification system would be capable of removing the small quantities of material involved.

Of greater concern would be escape of the copper shot itself. This is guarded against by the monolithic structure of the loop thimble, its conservative design and low operating pressure. The port used to drain the shot from the thimble is securely and redundantly bolted in place, and the outer seal plate is backed up by an internal plug which is threaded into the drain hole, and capable by itself of retaining the shot bed. The diameter of the shot has been selected to be too large to pass down between MITR-II fuel plates, and the high settling velocity of the shot (≈2 ft/s) makes extensive entrainment unlikely. The shot could, however, pass through the circulation valves and get under the core, resulting in blocked fuel channels. Care will be taken in all shot handling procedures to insure that none escapes to become an uncontrolled source of in-containment debris.

5) Lead bath can leak

Titanium can failure will permit contact of molten lead with the cold aluminum thimble wall, which will cause virtually instantaneous freezing of the lead. Laboratory tests are planned which are anticipated to show that the freezing process seals the leak and that the resulting thermal shock will not cause the aluminum to fail. However, the thermal short circuit will create a hot spot on the thimble wall, and local boiling cannot be ruled out at the present time. Again, experiments are planned to explore the consequences of such an event. Note that if this boiling occurred, it would not be on the surface of a fuel plate.

A large can failure would be indicated by a large decrease in loop average temperature (with the heater on), and would be cause for loop shutdown. If operation continued, for a short time, and maximum loop power (20 kW) were shunted through the thimble wall, a lower thimble average heat flux of approximately 68,000 $Btu/h\text{-}ft^2$ would result. This is a quite tolerable result considering that heat transfer coefficients in excess of 10^4 $Btu/h\text{-}ft^2\text{-}°F$ are to be anticipated.

Small leaks will be difficult to detect. If the leak eventually has significant effect on the heat losses in the lead bath, this would probably be detected by thermocouples or heater power level and the experiment would be shut down if necessary. Another indication might be an increase in reactivity noise if local boiling were to occur.

It should be noted that the lead bath can will be subject to vacuum leak testing and pre-operational hot testing. It is unlikely that leaks will develop in a can which has passed this testing, since no significant stresses are applied, and the liquid lead and helium environments are benign. A thermocouple may be installed to monitor the temperature of the thimble bottom and provide an indication of small lead leaks.

The overall conclusion of the preceding review is that there are no credible mechanisms by which the PCCL can adversely affect MITR safety or otherwise create a situation hazardous to operators or experimenters. The most serious outcome projected is the loss of valuable experimental data, since large changes in the internal chemical environment of the PCCL would probably invalidate any subsequent analysis of corrosion product characteristics.

Some concluding remarks are also in order on the inverse question as to whether there are any normal transient or accident situations of the

MITR itself which could induce a hazardous response by the PCCL. In brief, none have been identified. The MITR interacts significantly with the PCCL only by 1) provision of approximately 10% of PCCL energy input via gamma heating, and 2) serving as a heat sink for the PCCL. Since electrical power to the PCCL in-core heat can be cut off in an emergency, thereby greatly reducing the need for a heat sink, a loss of MITR capabilities sufficient to severely inconvenience the PCCL would unquestionably be a far greater threat to the MITR itself.

7. WASTE HANDLING AND DISPOSAL

Two major types of radioactive waste will be produced by the PCCL experiments: loop coolant water contaminated by activated corrosion products, and activated in-core components. As with other aspects of the loop operation, no new issues are raised in this regard.

Approximately 22 liters of water will accumulate in the discharge tank during a one-month run. The activity level of this water is expected to be very low, since most of the corrosion products in the effluent (a few precent of those generated in the loop itself) should deposit on the bleed capillary surfaces before reaching the tank. The activity levels will be sufficiently low to permit disposal of the discharge water to the drain after counting. Although particulates are expected to deposit on the capillary surfaces, tests will be run on initial discharges to determine if filtration is necessary. The approximately 0.5 liters of water present in the loop at the end of a run will be collected, counted and disposed to the drain after verification that it is within acceptable limits.

In-core loop components will be stored for decay and eventually shipped as solid waste. The Po^{210} produced in the lead bath has a half-life of 138 days and can therefore be decayed to insignificant levels before disposal.

Note that the PCCL is designed for 1:1 simulation of an actual PWR, hence coolant activity levels will be comparable. Table 7.1 gives representative radionuclide concentrations expected after one month of irradiation. Using values for a typical PWR given by Benedict et al.,[1]

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[1]M. Benedict, T.H. Pigford and H.W. Levi, Nuclear Chemical Engineering, Second Edition, McGraw-Hill, New York (1981) p. 396.

scaling by water volume and taking into account the use of ^{11}B and ^{7}Li, tritium concentrations are expected to be ≈ 5 μCi/l.

TABLE 7.1: REPRESENTATIVE ACTIVITY VALUES ESTIMATED FOR A ONE-MONTH PCCL IRRADIATION (from typical PWR values)

	"Steam Generator" Activity (μCi/cm^2)	Loop Water Activity (μCi/liter)
^{58}Co	3.0	no data
^{60}Co	.2	0.02
^{59}Fe, ^{51}Cr, ^{54}Mn	<<^{60}Co	<<^{60}Co

8. FUTURE WORK

This section summarizes the requirements which remain to be fulfilled before loop installation in the reactor. It also gives a complete tabulation of the safety-related experiments and pre-operational tests which will be carried out, see Table 8.1. Note that the results of these experiments and tests will be reported by PCCL personnel, certified by senior PCCL personnel (Professors M. J. Driscoll and O. K. Harling) and incorporated into the QA file.

The following items are required before experiment irradiation:

1) Standard operating procedures
2) Abnormal operating procedures
3) Calibration procedures and test schedule for safety-related instrumentation
4) An Irradiation Request Form (Part I)
5) A final signed safety review by the MITR staff
6) Pre-operational test procedures and results
7) Material certifications (see Section 2.f on QA procedures)

Development of the required procedures and checklists for operating the loop will be done in conjunction with MITR staff.

TABLE 8.1: SAFETY-RELATED EXPERIMENTS AND PRE-OPERATIONAL TESTS TO BE PERFORMED BEFORE PCCL IRRADIATION

a) Safety-related experiments

- Shot bed conductivity measurements
- Radiative heat transfer measurement
- Shot bed average temperature determination
- Liquid lead compatibility experiment
- Thimble cooling/coolant ΔT measurement
- Lead bath can leak/break simulations

b) Proof testing and pre-operational tests

- Circulation pump low T/P performance testing
- Loop pressure proof test (3000 psi)
- Thimble vacuum/pressure proof test
- Pressure relief valve proof test
- Out-of-core conditioning, "shakedown" run
- Grid plate clearance test
- Low reactor power reactivity tests (empty dummy fuel element, unfilled loop, cold loop, hot loop)
- Loop disassembly rehearsal

9. SUMMARY AND CONCLUSIONS

a) Summary

This safety evaluation report pertains to the operation and research use of a small pressurized water loop in the MITR-II core, core-tank and containment building. The preceding sections and the appendices which follow document the design features, as they relate to safety, of an in-pile facility which is intended to closely simulate the primary system of a PWR. Figure 9.1 is a schematic illustration and summary of the safety features of the loop.

Although designed to operate at PWR temperatures and pressures, the small size, the design of the facility and the control instrumentation limit the potential hazards. In this SER we have addressed issues associated with 1) reactivity changes, 2) thermal-hydraulic effects, 3) chemical effects, 4) radiolytic decomposition, 5) experiment scrams, 6) prototype testing and proof testing, 7) radioactive releases, waste handling and disposal, as well as other concerns such as operational and experimental procedures, radiation levels and ALARA considerations. Our conclusions in these areas are summarized below:

1) Reactivity changes

The loop is designed so that the maximum reactivity changes which are conceivable are less than the 0.20% $\Delta k/k$ allowed for an experiment which would be movable during reactor operation.

2) Thermal-hydraulic effects

The most serious concern overall is assurance of the heat input to the loop: 10-15 kW of electric heater power and 2-4 kW of gamma heat. Radiation to the thimble wall is sufficient to remove the latter in a purely passive mode of

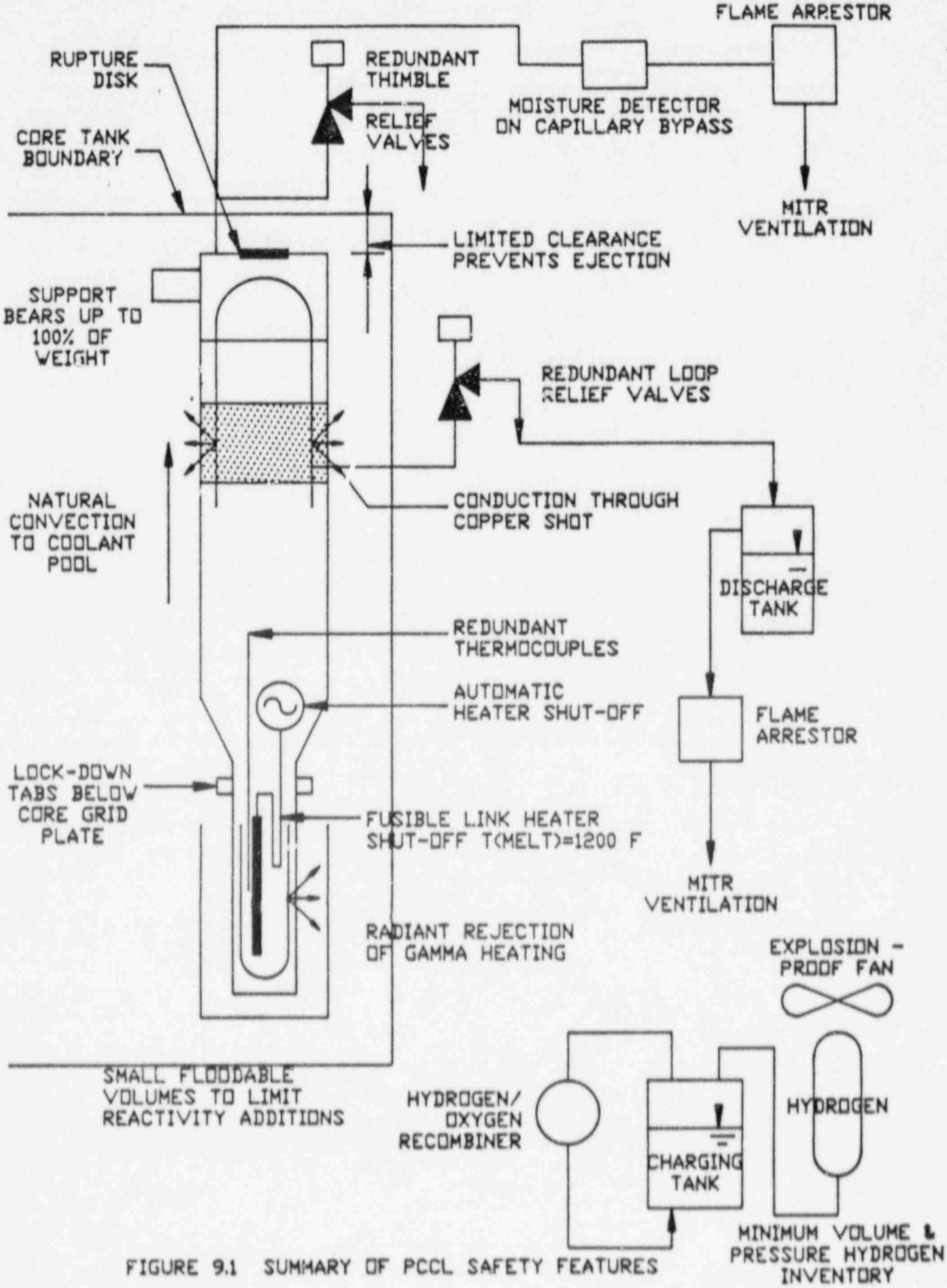

FIGURE 9.1 SUMMARY OF PCCL SAFETY FEATURES

operation; and redundant means for interruption of electric power are provided--an active overtemperature control system based upon thermocouples which monitor heater bath temperature, and a passive fusible link which melts to sever the electrical connection to the resistance heaters in the lead bath.

3) Chemical effects

A moderate amount of hydrogen gas ($\leq$ 10 scf) will be used inside the reactor containment. The systems outlined in the body of the report assure safe use of hydrogen through two basic approaches avoiding the contact of hydrogen with oxygen or high dilution levels to prevent accumulation of an explosive mixture in the event of hydrogen leakage.

A scenario which involves the potential for hydrogen entry into the in-pile thimble has been analyzed and is discussed in 6.b.c. The operational, instrumental and design features which we have incorporated into the PCCL are deemed adequate to assure that hydrogen and oxygen can never be present in significant amounts in the thimble when it is in-core.

4) Radiolytic decomposition

No significant unresolved issues have been identified. Basically the PCCL operates in the mode and with materials identical to a PWR.

5) Experiment scrams

This is addressed in the body of this report, and the protection of the reactor systems as well as the PCCL can be assured.

6) Prototype testing and proof testing

A considerable amount of testing of this type is planned to assure the safe conduct of PCCL experiments. Section 8 outlines the testing which will be performed in this category prior to routine in-pile operation.

7) Radioactive releases, waste handling and disposal, radiation levels, ALARA considerations and related operational procedures

These are discussed in various sections of the SER. For the most part the PCCL breaks no new ground in this area. Previous experiments reviewed and authorized for operation in the MITR-I and MITR-II, such as the Fatigue Cracking Experiment, have involved comparable or higher levels of activities, radiation fields and radioactive material handling procedures.

b) Conclusion

The overall conclusion of this SER is that operation and experimentation with the PCCL in-pile loop, having the design features outlined above, can be made to fully satisfy the MITR-II Technical Specifications. Furthermore, we feel that no significant health or safety hazards will result from these activities.

APPENDICES

Appendix 1: Safety Related Experiments and Calculations

a) Radiative dissipation of gamma and neutron heating of loop components.

b) Reactivity worth of B-10.

c) Thimble stresses.

Appendix 2: Extracts from the MITR-II Technical Specifications

APPENDIX 1.a: RADIATIVE DISSIPATION OF GAMMA AND NEUTRON HEATING OF LOOP COMPONENTS

As discussed above, one of the primary safety features of the loop is the passive rejection of nuclear heating of in-core components at a temperature below the zirconium-steam reaction temperature of 2100°F. An experiment to verify calculations of the radiative heat transfer rate between the lead-filled titanium tube and the cooled aluminum thimble has been completed. A simulation at approximately full scale with respect to radial dimensions and half the length of the actual in-core section was used. A one kilowatt electric heater was used at varying power levels, and the temperature difference from the titanium tube wall to the aluminum tube (which was maintained at constant temperature by a water bath) was measured.

The experimental data was fit to the expression:

$$Q = \frac{\sigma(T_2{}^4 - T_1{}^4)}{\frac{1}{\varepsilon_1 A_1} + \frac{1}{A_2}\left(\frac{1}{\varepsilon_2} - 1\right)}$$

where: Q = total heat transfer rate

σ = Stefan-Boltzmann constant

ε = emissivity

A = surface area

1 - Al thimble, 2 - Ti can

The denominator of this expression can be estimated from the experimental data, and should be approximately constant since the change of emissivity with temperature is small over the range encountered. The calculated values, however, varied by a factor of five, indicating that convection was contributing significantly to the heat transfer. An approximate value for the denominator at $T_2 = 2100°F$ was nevertheless obtained by extrapolation, and the heat rejection at that temperature was estimated to be 8.5 kW.

At full power, the core average γ-heating rate is ≈1 w/g. Wood[1] and Brown[2] have shown that neutron heating adjacent to a thermal neutron fission spectrum is one tenth the gamma heating. Thus, using a total nuclear heating rate of 1.1 W/g, and with 6.5 kg total mass in the titanium tube and contents, the total nuclear heating is ≈7.2 kW. The analysis above indicates that this will result in temperatures below the zirconium-steam reaction temperature, although the margin is not large. When the loop is constructed, tests of the actual passive heat rejection will be performed. If necessary, the maximum temperature attained in passive cooling mode can be reduced by reducing the mass of lead in the conductive bath and/or by treating the titanium and aluminum surfaces to increase their emissivity. The experimental results described above indicate that adequate safety margins should be attainable.

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[1] P.J. Wood and M.J. Driscoll, "Assessment of Thorium Blankets for Fast Breeder Reactors," MITNE-148, July 1973.

[2] G.J. Brown and M. J. Driscoll, "Evaluation of High Performance LMFBR Blanket Configurations," MITNE-150, May 1974.

APPENDIX 1.b: ESTIMATION OF BORON WORTH

The (unknown) worth of Boron-10 can be es mated from the (known) worth of Uranium-235. Perturbation theory yields the simple ratio:

$$\frac{(\Delta\rho/g)\ B\text{-}10}{(\Delta\rho/g)\ U\text{-}235} = \frac{\bar{\sigma}_{a,\ B\text{-}10}}{(\bar{\eta}-1)\ \bar{\sigma}_{a},\ U\text{-}235} \cdot \frac{235}{10} \qquad \text{(A-1)}$$

For cross-sections, the Maxwellian-averaged thermal values can be used, since over 80% of the neutron absorption in both B-10 and U-235 are in the thermal region. Cross-section values are a follows:

	$\bar{\sigma}_a$	$\bar{\eta}$
B-10	3,400	0
U-235	588	2.07

Finally, we have for the MITR-II core:

$$\frac{\Delta\rho}{g_{U\text{-}235}} \leq 10 \text{ milli } \beta/g^{1} \text{ (worst-case value - A-ring, 445 g element)}$$

Thus, Eq. (A.1) gives:

$$\left[\frac{\Delta\rho}{g}\right]_{B\text{-}10} = 1270 \text{ milli } \beta/g$$

and if $\bar{\beta}_{U\text{-}235} = 0.00786$

$$\left[\frac{\Delta\rho}{g}\right]_{B\text{-}10} \equiv \left[\frac{\Delta k/k}{g}\right]_{B\text{-}10} \approx \left[\frac{\Delta k}{g}\right]_{B\text{-}10} = 9.93\text{x}10^{-3} = 0.998\%$$

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[1] Personal communication, J. Bernard, October 6, 1986, and his memorandum to O. Harling dated October 7, 1986.

APPENDIX 1.c: THIMBLE STRESSES

The thimble is designed to withstand the stresses expected under both normal and accident conditions. The cases of interest which have been identified are:

1) internal pressurization - normal helium pressure
   - shot bed hydrostatic pressure
   - loop leak pressurization accident
   - hydrogen deflagration accident

2) external pressure on evacuated loop

3) thermal stresses

The thimble consists of three sections: an approximately elliptical section 2.50 in. x 1.40 in. x 0.125 in. wall, 3 ft. long; a cylindrical section 4.0 in. OD x 0.250 in. wall (possibly with 0.125 in. deep grooves for added heat transfer), 8 ft. long, and a cylindrical section 8.0 in. OD x 0.250 in. wall. All these components are constructed from 6061-T6 aluminum tubing with minimum yield strength of 40,000 psi.

For the case of internal pressurization in a long, thin-walled cylinder, the hoop stress is given by:

$$\sigma = \frac{Pr}{t}$$

where P = pressure

r = tube radius

t = tube thickness.

For the cylindrical sections, the pressure at which the hoop stress is one-half the yield stress can readily be derived: P = 1250 psi (equal for both sections if a minimum thickness of 0.125 in. is assumed for the 4 in. section, as would be the case for the grooved version). For the elliptical section, maximum stress due to internal pressure occurs at the minimum radius point where hoop stress is:

$$\sigma = \frac{Pb}{t}$$

where b = semi-major axis, and the bending moment is given by:[1]

$$M = \frac{P}{2}\left(b^2 - \frac{I_x}{S} - \frac{I_y}{S}\right)$$

where I_x = moment of inertia of a quadrant of the ellipse about the x-axis

I_y = moment of inertia of a quadrant of the ellipse about the y-axis

S = arc length of a quadrant of the ellipse

Using these relationships, an internal pressure of ≈850 psi can be shown to produce a maximum stress of one-half the yield stress. It is evident from this analysis that the normal helium operating pressure of 20-30 psig can easily be withstood, and the ≈20 psi additional pressure due to the weight of the shot bed is also inconsequential. (Note that a simple

[1]Baumeister, Avallone and Baumeister, eds., Marks' Standard Handbook for Mechanical Engineers (8th Ed.), McGraw-Hill, New York (1978) pp. 5-51.

analysis indicates that shot-bed ratchetting, which could exert significant forces on the thimble, is unlikely to occur. Dimensional monitoring of the thimble before and after its out-of-pile runs will be used to verify the absence of ratchetting forces.) The loop leak pressurization accident was shown above to result in a maximum pressure of less than 400 psi if the two thimble pressure relief valves fail to open. Pressures in a hydrogen deflagration accident are much more difficult to assess, but this accident has been shown to be extremely unlikely, and a burst disk set at 500 psi or lower will be provided to gaurd against thimble failure in this scenario. Considerable margin to yield is provided, and the ductility of the aluminum (min. ≈ 10% elongation) gives further assurance that thimble leakage is a remote possibility even in extreme pressurization accidents.

The stresses produced on the thimble by evacuation (plus submergence to about 12 ft. in water) are negligible, and the ability of the thimble to withstand this loading will be amply demonstrated during the conditioning run, which will be made with the thimble at vacuum. The worst case for thermal stresses will occur on the in-core thimble section during a loss-of-coolant accident. Using a conservative value of 10 kW total heat transfer to this 150 in.^{2} area gives $\Delta T \approx 3°F$;, and a corresponding thermal stress of ≈300 psi, which is again negligible.

APPENDIX 2

EXTRACTS FROM THE MITR-II TECHNICAL SPECIFICATIONS

5.3 Primary Coolant System

Applicability

This specification applies to the design of the primary coolant system.

Objective

To assure compatibility of the primary coolant system with the safety analysis.

Specification

The reactor coolant system shall consist of a reactor vessel, a single cooling loop containing three heat exchangers, and appropriate pumps and valves. All materials, including those of the reactor vessel, in contact with primary coolant (H_2O), shall be aluminum alloys or stainless steel, except small non-corrosive components such as gaskets, filters and valve diaphragms. The reactor vessel shall be designed in accordance with the ASME Code for Unfired Pressure Vessels. It shall be designed for a working pressure of 24 psig and 150°F. Heat exchangers shall be designed for 75 psig and a temperature of 150°F. The connecting piping shall be designed to withstand a 60 psig hydro test.

Basis

The reactor coolant system has been described and analyzed in the Safety Analysis Report as a single loop system containing two heat exchangers. Additional analysis based on the use of three heat exchangers, has been described in the NRC staff's Safety Evaluation of Amendment No. 14 to these Technical Specifications. Materials of construction, being primarily stainless steel, are chemically compatible with the H_2O coolant. The stainless steel pumps are heavy-walled members in areas of low stress and should not be susceptible to chemical attack or stress corrosion failures. The failure of the gaskets and valve bellows, although undesirable, would not result in catastrophic failure of the primary system; hence, strict material limitations are not required for technical specifications. The design, temperature, and pressure of the reactor vessel and other primary system components provide adequate margins over operating temperatures and pressures, and it is believed prudent to retain these margins in order to further reduce the probability of a primary system failure. The reactor vessel was designed to Section VIII, 1968 edition, of the ASME Code for Unfired Pressure Vessels. Subsequent design changes should be made in accordance with the most recent edition of this code.

Since the safety analysis is based on the reactor coolant system as presently designed and with the present margins, it is considered necessary to retain this design and these margins or to redo the analysis.

6. EXPERIMENTS

6.1 General Experiment Criteria

Applicability

This specification applies to experiments using the reactor.

Objective

To assure that experiments in the reactor do not affect the safety of the reactor.

Specifications

All experiments within the reactor shall conform to the following conditions.

1. Reactivity Effects

The reactivity worth of experiments shall not exceed the values indicated in the following table:

	Single Experiment Worth	Total Worth
Movable	0.2% ΔK/K	0.5% ΔK/K
Non-secured	0.5% ΔK/K	1.0% ΔK/K
Total of the above	______	1.5% ΔK/K
Secured	1.8% ΔK/K	______

2. Thermal-Hydraulic Effects

   a. All experimental capsules shall be designed against failure from internal and external heating at a reactor power level or process variable corresponding to the Limiting Safety System Setting associated with that power level or process variable.

b. The outside surface temperature of a submerged experiment or capsule shall not cause nucleate boiling of the reactor coolant during operation of the reactor.

c. The insertion of an experiment into the core should not cause a coolant flow redistribution that could negate the safety consideration implicit in the Safety Limits.

3. Chemical Effects

a. Metastable or other materials that could react to create a rapid pressure rise shall be encapsulated. The capsule shall be prototype tested under experimental conditions to demonstrate that it can contain without failure an energy release equivalent to at least twice the material to be irradiated or at least twice the pressure that could be expected from any reaction of these materials. These tests must also include effects of any fragments which may be generated. If a change in experimental conditions could result in a greater potential for failure than design experimental conditions, the capsule shall also be tested under these changed conditions. In addition, the quantity of material should be limited such that if the maximum calculated energy release should occur, significant damage to the reactor core will not result, assuming the material is not encapsulated.

b. No explosive materials (defined to include all materials that would constitute Class A, Class B and Class C explosives as described in Title 49, Parts 172 and 173 of the

Code of Federal Regulations) shall be placed in the reactor core or within the primary biological shield, which, if completely detonated, could cause any rearrangement or damage to the reactor core. Proposed quantities of explosive materials greater than the equivalent of 25 milligrams of TNT shall require a documented safety analysis and approval by the MIT Reactor Safeguards Committee. Capsule design for explosive material shall be prototype tested to demonstrate that it can contain at least twice the pressure produced inside the capsule as a result of detonation of the material or the pressure produced by the detonation of twice the amount of material.

c. Corrosive materials that could affect or react with another material present in the reactor system will be doubly encapsulated. If the material can adversely affect the reactor core or any of its component parts or auxiliary systems or the building containment to cause loss of function of the affected component or system, means shall be provided to monitor the integrity of the material container.

4. Radiolytic Decomposition

a. Compounds subject to radiolytic decomposition shall be irradiated in containers which can withstand the maximum gas pressure produced as a result of the decomposition under irradiation including the effect of any temperature rise. This pressure will be determined by previous experience or by testing as described in Specification 6.1, subsection 6.

b. Consideration shall also be given to any pressure buildup resulting from the decomposition of the sample container, such as might occur with a polyethylene vial.

c. Compounds subject to radiolytic decomposition may be irradiated in a capsule that is vented, provided that the vented release is less than 1.0% of the limits of 10 CFR 20 at any point of possible exposure.

5. Experiment Scrams

Provisions have been made in the Reactor Safety System for the addition of experiment scrams. These scrams may be added for the protection of the experimental equipment and/or reactor components in the event of some malfunction. If malfunction of the experiment can adversely affect the reactor core or any of its component parts or auxiliary systems or the building containment to cause loss of function of the affected component or system, the experiment scrams shall be redundant so as to satisfy the single mode failure protection discussed in Section 7 of the SAR.

6. Prototype Testing

Materials whose properties (composition, heating, radiolytic decomposition, etc.) are uncertain must be prototype tested. These tests will be designed to give a stepwise approach to final operating conditions. The tests may either be stepwise time or flux irradiations with proper instrumentation to determine temperature, pressure and radioactivity for each step as required.

7. Radioactive Releases

a. Experiments shall be designed so that operation or malfunction is not predicted to result in exposures or releases of radioactivity in excess of the limits of 10 CFR 20 to either onsite or offsite personnel.

b. The total radioactive materials inventory of an experiment or credibly coupled experiments shall be limited such that the dose in unrestricted areas resulting from release of this inventory at its calculated maximum value shall not exceed that of the Design Basis Accident (Section ?.3.1 of the SAR).

Bases

Accidents resulting from the step insertion of reactivity have been discussed in the SAR. It was determined that following a step increase of 1.8% ΔK/K, fuel plate temperatures would be below the clad melting temperature and significant core damage would not result. The 0.2% ΔK/K limit for movable experiments corresponds to a 20-second period, one which can be easily controlled by the reactor operator with little effect on reactor power. The limiting value for a single non-secured experiment, 0.5% ΔK/K is set conservatively below the prompt critical value for reactivity insertion and below the minimum shutdown margin. The sum of the magnitudes of the static reactivity worths of all non-secured experiments, 1.0% ΔK/K, does not exceed the minimum shutdown margin. The total worth of all movable and non-secured experiments will not reduce the minimum shutdown margin as the shutdown margin is determined with all movable experiments in the most positive reactive state.

Specifications 2, 3, 4, 5 and 6 are intended to minimize the probability of experiment failures. Experiment capsules should be designed to withstand expected temperatures, pressures, chemical and radiochemical effects. The requirement for testing containers at twice the pressure or with twice the amount of explosive or metastable material to be irradiated provides a factor of 2 safety margin as allowance for experimental uncertainties. Table 6.1-1 gives a summary of the requirements for specimen irradiations for ease of review and classification of the specifications.

The radiological consequences of experiment malfunctions must be considered as stated in Specification 7. Consistent with the Commission's regulations, predicted onsite personnel exposures or offsite concentrations resulting from these malfunctions must not be in excess of those permitted by 10 CFR Part 20.

Table 6.1-1

Summary of Requirements for Specimen Irradiations

Requirement	Stable	Metastable	Explosive	Corrosive	Radiolytically Decomposible
Consideration of reactivity effects, induced activity, heating and temperature distribution	X	X	X	X	X
Estimation of pressure buildup		X	X		X
Single encapsulation		X	X		X
Double encapsulation				X	
Container pressure test		X	X		
Other		d	a, d	b	c

a. Amounts above the equivalent of 25 mg TNT require safety analysis and approval of MITRSC.

b. If corrosion can cause loss of functions of the reactor core or any of its component parts or auxiliary systems or the building containment, integrity of container must be monitored during irradiation.

c. Container may be vented if release is less than 1.0% of 10 CFR 20.

d. Amounts limited such that reaction will not damage reactor core.

NUCLEAR REACTOR LABORATORY
AN INTERDEPARTMENTAL CENTER OF
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O.K. HARLING
Director

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

L. CLARK, JR.
Director of Reactor Operations

September 6, 1988

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

Gentlemen:

Subject: NRC Region I Inspection Report No. 50-20/88-02

As noted in NRC Region I Inspection Report No. 50-20/88-02, no reply to that report was required. Accordingly, this letter is submitted as an item of information. Under Section 6 (Experiments) of the aforesaid report, a concern was raised that a proposed experiment be prototype tested to a pressure of at least 970 psi. Please be advised that the proposed experiment has now been prototype tested to a pressure of 1000 psi. Also, certain changes have now been made to the proposed experiment which significantly reduce the necessary hydrostatic test pressures. These are described in a safety evaluation report that is being forwarded under separate cover.

Sincerely,

John A. Bernard

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

JAB/crh

cc: USNRC - Region I
Chief, Reactor Projects Section 1B
USNRC - Region I
L.T. Doerflein, Project Inspection, Section 1B
USNRC - Resident Inspector, Pilgrim Nuclear Station
USNRC - Region I
Chief, James T. Wiggins, Reactor Projects Branch No. 3

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A/21

NUCLEAR REACTOR LABORATORY

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

April 21, 1988

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk, M.S. OWFN 11-H-3
Washington, D.C. 20555

Subject: Evaluation of Unreviewed Safety Question, 10 CFR 50.59(b)(2)
MIT Reactor License No. R-37, Docket No. 50-20

Gentlemen:

The Massachusetts Institute of Technology Nuclear Reactor Laboratory is forwarding herewith a safety evaluation by the MIT Research Reactor Staff providing the bases for the determination that there are no unreviewed safety questions concerning the installation and operation of an experiment on the MITR-II.

The experiment is a pressurized coolant chemistry loop which will be installed in the reactor core and which is described in the enclosed "Safety Evaluation Report for the PWR Coolant Chemistry Loop (PCCL)", Report no. MITNRL-020, dated February 13, 1987 and its Supplement dated April 19, 1988. The purpose of the experiment is to investigate the formation, transport and deposition of radioactive crud in a carefully controlled loop that will simulate pressurized water reactor conditions, all with the objective of learning how to reduce radiation exposures to personnel during maintenance work on these reactors.

Since experiments of this type are not described in the "Safety Analysis Report for the MIT Research Reactor (MITR-II)", Report No. MITNE-115, October 22, 1970, as amended, a safety review of the experiment has been conducted by the project personnel, the reactor staff, and the MIT Reactor Safeguards Committee, including a safety evaluation as to the existence of any unreviewed safety questions (Safety Review #0-86-9, dated April 19, 1988 enclosed). No unreviewed safety questions have been indentified.

In accordance with 10 CFR 50.59(b)(2), evaluations of unreviewed safety questions have routinely been reported to NRC in the annual report required by paragraph 7.13.5 of the MITR-II Technical Specifications, Facility License No. R-37. Because of the unusual nature of this experiment, we are reporting the safety evaluation required by 10 CFR 50.59 (b)(1) at this time rather than wait until the next MITR-II annual report, which will be issued in July or August 1988.

A/25

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Our schedule calls for the initial installation on the reactor to occur in May.

If you should have any questions regarding the evaluation or any of the information furnished, we request that you contact us as soon as possible.

Sincerely,

Lincoln Clark, jr.

Lincoln Clark, Jr.
Director of Reactor Operations
Associate Director,
Nuclear Reactor Laboratory

Enclosures:

Safety Evaluation Report, February 13, 1987
Supplement to the Safety Evaluation Report, April 19, 1988
Safety Review #0-86-9, April 19, 1988

LC/crh

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

NUCLEAR REACTOR LABORATORY

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L. CLARK, JR.
Director of Reactor Operations

August 30, 1988

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

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

Dear Sirs:

Information on liquid discharges to the sanitary sewerage system was inadvertently omitted from the Annual Report of the MIT Research Reactor for the period July 1, 1987 to June 30, 1988, submitted on 29 August 1988 in compliance with paragraph 7.13.5 of the Technical Specifications for Facility Operating License R-37. Enclosed herewith is a revised copy of the entire report containing that information. The affected sections are the last paragraph on page 23 and the table on page 26. We regret any inconvenience that this may have caused.

Sincerely,

Kwan S. Kwok
Superintendent

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

JAB/gw

Enclosure: As stated

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

A/27

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MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

BY

REACTOR STAFF

August 29, 1988

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TABLE OF CONTENTS

MIT RESEARCH REACTOR

ANNUAL REPORT TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

Introduction

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

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

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

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

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

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

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

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

A. SUMMARY OF OPERATING EXPERIENCE

1. General

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

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

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

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

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

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

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

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

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

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

2. Experiments

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

Experiments and irradiations of the following types were conducted:

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

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

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

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

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

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

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

h) Use of the facility for reactor operator training.

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

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

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

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

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

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

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

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

3. Changes to Facility Design

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

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

4. Changes in Performance Characteristics

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

5. <u>Changes in Operating Procedures Related to Safety</u>

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

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

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

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

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

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

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

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

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

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

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

6. Surveillance Tests and Inspections

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

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

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

B. REACTOR OPERATION

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

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

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

C. SHUTDOWNS AND SCRAMS

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

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

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

I. Nuclear Safety System Scrams

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

II. Process System Scrams

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

III. Unscheduled Shutdowns or Power Reductions

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

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

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

D. MAJOR MAINTENANCE

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

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

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

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

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

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

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

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

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

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

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

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

E. SECTION 50.59 CHANGES, TESTS, AND EXPERIMENTS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

F. ENVIRONMENTAL SURVEYS

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

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

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

Fiscal Yearly Averages:

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

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

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

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

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

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

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

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

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

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

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

H. RADIOACTIVE EFFLUENTS

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

1. Liquid Waste

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

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

2. Gaseous Waste

Gaseous radioactivity is discharged to the atmosphere from the containment building exhaust stack and by evaporation from the cooling towers. All gaseous releases likewise were in accordance with the Technical Specifications and Part 20, and all nuclides were below the limits of 10 CFR 20.106 after the authorized dilution factor of 3000. Also, all were substantially below the limits of 10 CFR 20, Appendix B, Note 5, with the exception of argon-41, which is reported in the following Table H-1. The 2627 Ci of Ar-41 were released at an average concentration of 0.67×10^{-8} μCi/ml for the year. This represents 17% of MPC (4×10^{-8} μCi/ml) and is significantly less than the previous year's release of 6223 Ci. The decrease is due to a combination of factors including the sealing of leaks and the temporary reduction in operating hours.

3. Solid Waste

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

4. Liquid Discharge to the Sanitary Sewerage System

Total gross beta activity in the liquid effluents (cooling tower blowdowns and waste storage tank discharges) amounted to 0.00011 Ci for FY1988. The total tritium was 0.071 Ci. The total effluent water volume was 4,188,000 liters, giving an average tritium concentration of 17.0×10^{-6} μCi/ml.

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

TABLE H-1

ARGON-41 STACK RELEASES

FISCAL YEAR 1988

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

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

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS

FISCAL YEAR 1988

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

TABLE H-3

LIQUID WASTE DISCHARGES
FISCAL YEAR 1988

	Total Gross Beta Less Tritium	Total Tritium	Volume of Effluent Water[1]	Average Tritium Concentration
	($x10^{-6}$ Ci)	($x10^{-3}$ Ci)	($x10^{4}$ liters)	($x10^{-6}$ μCi/ml)
July 1987	189.0	20.0	57.8	34.6
Aug.	NDA[2]	2.12	36.1	5.9
Sept.	499.0	13.0	65.0	20.0
Oct.	111.0	2.63	33.5	7.9
Nov.	NDA	3.60	45.4	7.9
Dec.	NDA	2.26	34.9	6.5
Jan. 1988	NDA	4.54	32.1	14.1
Feb.	NDA	1.00	15.0	6.7
Mar.	56.1	2.88	19.8	14.5
Apr.	NDA	2.18	27.5	7.9
May	164.0	13.0	29.9	43.5
June	31.0	4.08	21.8	18.7
12 months	1050.1	71.29	418.8	17.0

Notes:

(1) Volume of effluent from cooling towers and waste tanks. Does not include other diluent from MIT estimated at 2.7 million gallons/day.

(2) No Detectable Activity; less than 1.26×10^{-6} μCi/ml beta for each sample.

NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

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L. CLARK, JR.
Director of Reactor Operations

August 29, 1988

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

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

Dear Sirs:

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

Sincerely,

Kwan S. Kwok
Superintendent,
Reactor Operations

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

JAB/crh
Enclosure: As stated

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

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MIT RESEARCH REACTOR

ANNUAL REPORT

TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

BY

REACTOR STAFF

August 29, 1988

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TABLE OF CONTENTS

MIT RESEARCH REACTOR

ANNUAL REPORT TO

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

Introduction

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

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

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

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

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

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

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

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

A. SUMMARY OF OPERATING EXPERIENCE

1. General

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

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

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

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

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

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

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

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

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

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

2. Experiments

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

Experiments and irradiations of the following types were conducted:

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

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

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

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

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

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

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

h) Use of the facility for reactor operator training.

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

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

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

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

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

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

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

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

3. Changes to Facility Design

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

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

4. Changes in Performance Characteristics

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

5. Changes in Operating Procedures Related to Safety

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

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

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

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

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

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

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

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

g) Procedure 1.4.3, "Safety Review Form" was revised to make explicit the requirement to document determinations of unreviewed safety questions. (Note: This change merely formalized an existing practice.) (SR #0-88-3)

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

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

6. Surveillance Tests and Inspections

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

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

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

B. REACTOR OPERATION

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

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

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

C. SHUTDOWNS AND SCRAMS

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

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

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

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

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

III. Unscheduled Shutdowns or Power Reductions

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

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

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

D. MAJOR MAINTENANCE

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

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

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

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

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

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

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

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

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

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

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

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

E. SECTION 5C 59 CHANGES, TESTS, AND EXPERIMENTS

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

F. ENVIRONMENTAL SURVEYS

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

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

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

Fiscal Yearly Averages:

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

G. RADIATION EXPOSURES AND SURVEYS WITHIN THE FACILITY

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

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

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

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

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

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

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

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

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

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

H. RADIOACTIVE EFFLUENTS

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

1. Liquid Waste

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

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

2. Gaseous Waste

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

3. Solid Waste

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

TABLE H-1

ARGON-41 STACK RELEASES

FISCAL YEAR 1988

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

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

TABLE H-2

SUMMARY OF MITR RADIOACTIVE SOLID WASTE SHIPMENTS

FISCAL YEAR 1988

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