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Design, Construction and Testing of an
in-Pile Loop for PWR Simulation

Ocean Engineering Course 13A

by

LCDR James H. Wicks Jr.

June 1987

DESIGN, CONSTRUCTION AND TESTING OF AN
IN-PILE LOOP FOR PWR SIMULATION

by

James H. Wicks Jr.
B.S. Mechanical Engineering, Purdue University
(1974)

SUBMITTED IN PARTIAL FULFILLMENT
OF THE REQUIREMENTS FOR THE
DEGREES OF

MASTER OF SCIENCE
IN NUCLEAR ENGINEERING

N00228-85-G-3262   AND

MASTER OF SCIENCE IN NAVAL ARCHITECTURE
AND MARINE ENGINEERING
at the

MASSACHUSETTS INSTITUTE OF TECHNOLOGY
June 1987

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Signature of Author _____
Department of Nuclear Engineering
May 8, 1987

Certified by _____
Paul E. Sullivan Thesis Advisor

Certified by _____
Michael J. Driscoll Thesis Advisor

Certified by _____
Otto K. Harling Thesis Advisor

Accepted by _____
Ocean Engineering   Douglas A. Carmichael
Chairman, Department Committee on Graduate Students

Accepted by _____
Nuclear Engineering   Allan F. Henry
Chairman, Department Committee on Graduate Students

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LCDR James H. Wicks Jr., USN

Submitted to the Departments of Nuclear Engineering and Ocean Engineering on the 12th of May 1987, in partial fulfillment of the requirements for the Degrees of Master of Science in Nuclear Engineering and Master of Science in Naval Architecture and Marine Engineering.

Abstract

Corrosion product activation and transport in Light Water Reactors is a well recognized, but poorly understood phenomenon, despite studies conducted world-wide for the past 20 years. In an effort to determine the mechanistic parameters which produce, release and transport activated corrosion products throughout a typical Pressurized Water Reactor (PWR), MIT has undertaken the construction of a model pressurized water loop to simulate the thermal-hydraulic characteristics of a representative full scale PWR.

The present work is concerned with the development and construction of the final design for the subject facility, and represents the culmination of efforts to fabricate an inexpensive, simple to operate, reliable, flexible, safe research tool having a high degree of similitude. The immediate use of the PWR Coolant Chemistry Research Loop (PCCL) will be to research modes of corrosion product transport, as a function of key parameters such as pH. The ultimate goal of the project is to identify chemistry control methods which will reduce the production and release of activated corrosion products in light water reactors.

This report presents the final detailed design of the PCCL. As-built, the Loop duplicates the core and Steam Generator fluid surface film differential temperatures, bulk fluid temperatures, and wall fluid shear stress of an actual PWR. Included are proposed plans for "sea trial" testing of the initial Loop prior to research operation. The final design has been reviewed and authorized for construction by the PCCL Project Staff and the MIT Reactor Safeguards Committee.

Thesis Supervisor: Michael J. Driscoll,
          Title: Professor of Nuclear Engineering
Thesis Supervisor: Otto K. Harling
          Title: Director Nuclear Reactor Laboratory,
                  Professor of Nuclear Engineering
Thesis Supervisor: Paul E. Sullivan
          Title: Professor of Ocean Engineering

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<u>Acknowledgements</u>

The author wishes to express his <u>sincere gratitude</u> to Professors Michael J. Driscoll and Otto K. Harling, Doctors William T. Lindsay Jr. and Gordon E. Koshe and Mr. Regg McCullough, President, Delta <u>M</u> Corporation, for the technical guidance, assistance and understanding throughout the development of the MIT-PCCL. I also wish to thank the <u>artisans</u> of the MIT Nuclear Reactor Machine Shop who produced works of art from the back of many envelopes. Moreover, I am <u>indebted</u> to Professor W. James Healey of the New Hampshire Vocational Institute for his assistance and technical review of construction drawings, and for answering his phone no matter how late the hour.

I would also like to thank my daily mentors who provided daily encouragement to a weary graduate student:

Miss Dorothy K. Eichel

Mrs. Carolyn R. Hinds

Finally, I must thank the two most important people. Two people who waited patiently for two years for Dad to come out and play:

Master Eric Woodward Wicks

Miss Cynthia Diane Wicks

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Chapter 1

Introduction

1.1 Foreword

The problem of corrosion product activation and transport in Pressurized Water Reactors (PWRs) is well recognized but poorly understood. The radiation fields produced as a consequence of in-core activation and subsequent out-of-core deposition of corrosion product material released to the primary circuit can constitute a significant radiological problem, especially during reactor shutdown operations (B-1).

The resulting high doses cost utilities many millions of dollars in terms of prolonged and less-than-optimum maintenance operations and extra manpower requirements. Several approaches have been taken to mitigate the consequences of radionuclide deposition: better worker training, temporary shielding, elimination of cobalt from plant materials, use of robotics, and decontamination. However, careful control and optimization of plant coolant chemistry may well be the most effective means, as evidenced by the fact that nearly an order of magnitude difference in exposure dose exists between the best and worst of todays otherwise nearly identical units.

In recognition of the above situation, a team of researchers from the MIT Nuclear Reactor Laboratory and Nuclear Engineering Department proposed a research project to the electric power research community based upon the construction and operation of in-pile loops to study the radio-activation and transport of corrosion products. An initial in-house feasibility and conceptual design study of a PWR loop was completed by Burkholder (B-2) at MIT in June 1985. This was followed by local utility support under the Electric Utility Program of the MIT Energy Laboratory (Boston Edison, PSE & G and Duke Power) to refine the conceptual design and bring it to the engineering design stage. At this point sponsorship of the project was assumed by the Electric Power Research Institute (EPRI) and the Empire State Electric Energy Research Corporation (ESEERCO), who will jointly support the construction and operation phases of the project. It was also in the latter stages of the project that the present research effort was initiated, having as its objective the finalization of the design and the construction and out-of-pile proof testing of the subject PWR loop.

1.2 Background

Presently, the theoretical understanding of the radio-activation, transport and deposition of corrosion products in light water reactors (LWR) is inadequate.

Over the past three decades since the importance of the "CRUD" problem was first recognized there has been an evolution in the analytical models developed to describe the complex transport processes involved. Initially, out-of-core dose rates were attributed to the particulates, which could be filtered from the coolant and studied as visible crud concentrations. It was found that raising coolant pH would decrease visual crud concentrations, but steam generator plenum and primary coolant pipe wall radiation dose rates continued to increase. Subsequently, during the 1960s attention shifted to controlling coolant colloids. For example, coolant chemistry additives were proposed using double valence cations to neutralize the electrostatic repulsion of the colloid particles. However, through limited applications of this chemistry control system, it has been demonstrated that transport of corrosion products by the colloidal mechanism is probably not the primary transport phenomenon. As a result of continuous research over the last two decades, it is now felt that soluble corrosion products in PWR coolant are the primary factor in corrosion product transport. In parallel with experimental coolant chemistry research, the evolution of corrosion product activity transport models has progressed. CORA-II (S-1), PACTOLE (B-3), RAPTOR (K-1), and CRUDSIM (L-1) are examples of PWR activity transport models in use today. The CRUDSIM model, in particular, is based on the present day understanding of solubility-dominated transport theory, and

is the major tool being used in a parallel effort on the current project to plan and interpret experiments using the subject facility (M-1). However, these computer program models are still not capable of adequately describing or predicting changes in radiation build-up as a function of operational modes. At best, the models can only provide trends. Accordingly, there is a well recognized need for "clean" experiments which can contribute to model testing and improvement.

In addition to the shortcomings of analytical modeling there also are numerous problems with experimental PWR chemistry research. Past experiments have not adequately simulated the conditions found in a PWR. Consider, for example the following three options for PWR coolant research:

* Out-of-Pile experiments
* In-Pile loop simulation
* Use of operational reactors

Out-of-Pile experiments do not provide the necessary radiation chemistry environment to simulate the conditions experienced by circulating corrosion product ions, colloids, and particles. Most in-pile loops have out-of-pile to in-pile surface area ratios one or two orders of magnitude too large compared to the corresponding ratio in a PWR. Since

computer modeling remains at best semiempirical (C-1), this large variation in scaling factor makes extrapolation of data impossible. The DIDO Water Loop at Harwell, UK (B-1) is a good example. The DIDO loop has been used in the past to collect a large body of data, but because of large differences in surface area, volume, and flow ratios, as summarized in Table 1-1, not all data will correlate directly to the PWR. Finally, experiments conducted in a full scale PWR are not practical. The reactor plant is not controlled to tolerances necessary for research, and utilities are reluctant to vary operating conditions of a billion dollar plant without demonstrated proof of beneficial results. A power reactor is just too expensive to use as a research and testing facility.

In view of the situation just described, the need is evident for a facility to conduct "clean" plant experiments to obtain proof of principle data and information which can test and improve existing computer models. To this end MIT established the following goals in the design and construction of the MIT-PCCL:

* Match temperature fields, both coolant bulk
  temperatures and core heat transfer film differential
  temperatures. This temperature similitude accurately
  models what is believed to be the primary driving
  force of solubility and mass transfer. See Appendix

Table 1.1 Comparison of Westinghouse PWR's, DIDO Water Loop and MIT-PCCL

Parameter	DIDO(B-1)	PWR(B-1)	Westinghouse PWR(U-1)	MIT-PCCL
In-Core : Out-of-Core Coolant Volume Ratio	1:200	1:12	1:18.1	1:9.4
In-Core : Out-of Core Area Ratio	1:130	1:2.3	1:2.96 1:2.89	1:6.7
Material Area Ratios Zircaloy:Inconel:Stainless Steel	1:24:19	1:3.1:0.5	1:2.57:0.39	1:4.9:1.2(1) 1:3.2:0.8(2)
Fraction In-Core				
Zircaloy	0.3	1	1	0.64
Inconel	0.0	0.04	0.093	0.0
Stainless Steel	0.0	0.5	0.214	0.0
Coolant Transit Time (sec)	200	10	13	$Core_{1gpm}$ 0.7 $Core_{2gpm}$ 0.35 $Loop_{1gpm}$ 8.8 $Loop_{2gpm}$ 4.4
Flow Velocity (M/sec) DIDO Tubes/PWR S/G tubes	5-6 (S/G)	5.7-6.2	4.84 Core 5.74 S/G	$Core_{1gpm}$ 1.84 $Core_{2gpm}$ 3.68 S/G_{1gpm} 1.22 S/G_{2gpm} 2.45
Main Coolant piping	1	15		
Reynolds' Numbers	$1\text{-}5 \times 10^5$	$5\text{-}12 \times 10^5$	5.1×10^5 core 9.1×10^5 S/G	5.1×10^5 core 1.3×10^5 S/G

(1) Material ratio using the in-core area of zircaloy
(2) Material ratio using the total area of zircaloy

B for a more detailed discussion of loop scaling in
accordance with these premises.

* Match construction material area ratios as closely as
  possible within system layout and volume constraints
  imposed by the MITR core tank configuration.

* Devise a physically small system but one which is
  realistic in modeling ratios of importance. The MIT-
  PCCL simulates a single PWR flow cell: a channel
  between fuel pins in a PWR core and one steam
  generator tube. This modeling results in a MIT-PCCL
  to PWR flow cell scale of approximately 1/3 even
  though the power scale-down ratio is about 1/160,000.

* Make each experimental run with a "clean" well-
  characterized system. All wetted coolant surfaces
  should be replaced each run. Post mortem analysis of
  the entire loop, (with the exception of the main
  circulating pump which will be decontaminated after
  each run) can thus measure entire plant response to
  the maintained coolant environment.

* The loop must be designed with flexibility as a major
  goal to permit variation of all important parameters,
  scaling factors and procedures, to include a wide
  range of coolant chemistry studies.

With these considerations and goals in mind, a
preliminary conceptual design for an appropriate facility
was devised in 1985. Figures 1.1, 1.2, and 1.3 show the

Figure 1.1 Schematic of PWR Coolant Chemistry Loop

30 PSIA THIMBLE HELIUM SYSTEM

He

CANNED ROTOR PUMP

SAMPLING

CHARGING

TURBINE FLOW METER

STEAM GENERATOR COPPER-SHOT BED SECTION

INLET PLENUM

ALUMINUM THIMBLE

ELECTRIC HEATER

ZIRCALOY TUBE FUEL SECTION

MITR CORE

TITANIUM TEST TUBE

LEAD BATH

MITR CORE TANK BOUNDARY
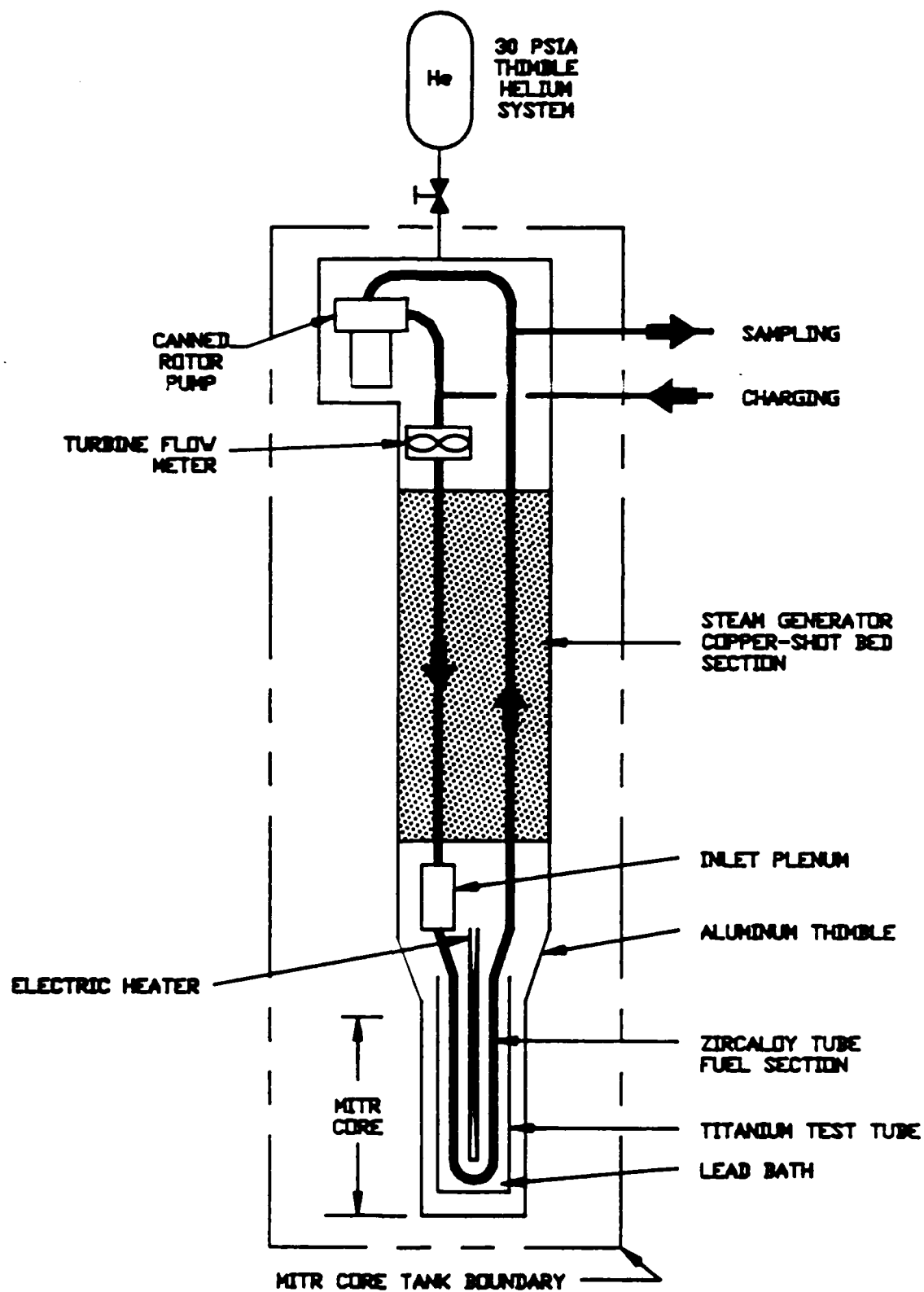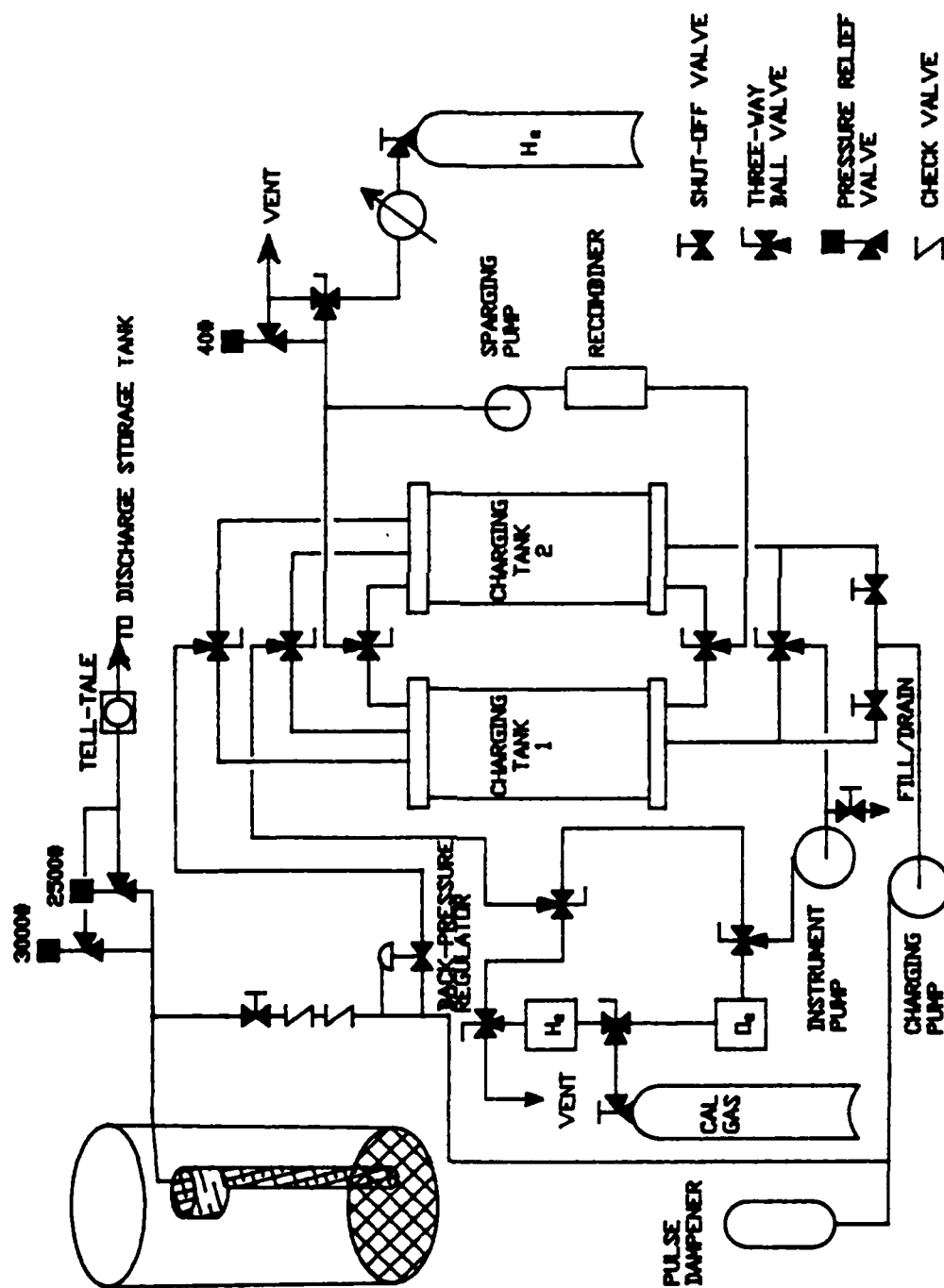
Figure 1.2. Out-of-Pile Charging System.
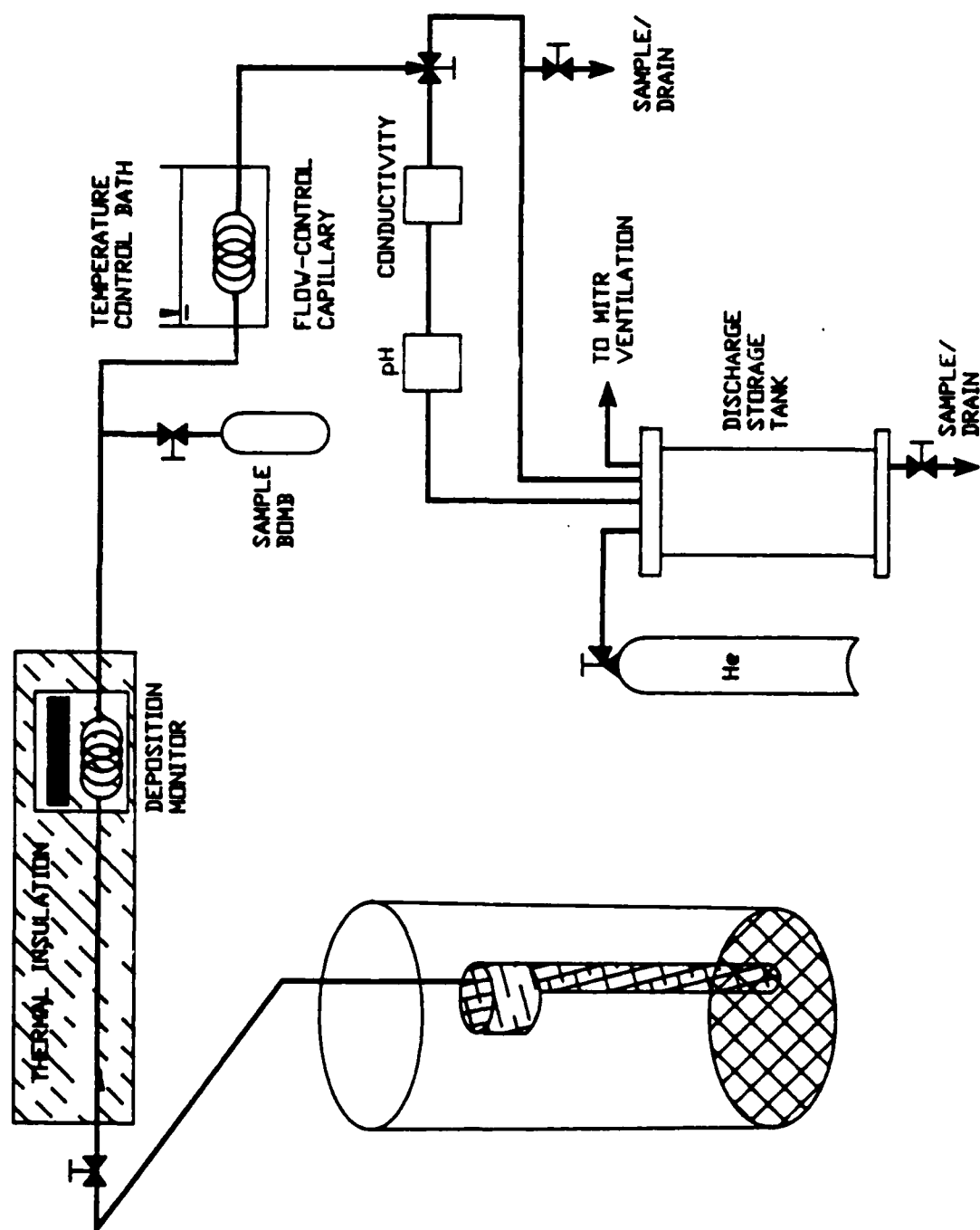
Figure 1.3. Sampling System.

conceptual design, as revised to reflect refinements made over the past year and a half; and Table 1.1 shows the overall degree of similitude achievable. A conceptual design, however, lacks the practical detailing necessary to convert ideas into a working facility, and it is this task which is undertaken in the project described in this report.

1.3   Organization of this Report

This report is divided into chapters and sections describing the final design and construction of an in-pile loop for the study of corrosion product activation and transport in an environment closely simulating that of a pressurized water nuclear reactor.

Chapter 2 describes the physical construction and instrumentation of the PWR coolant loop. The in-pile loop is divided into four component sections: heated/in-pile section, plenum section, steam generator/heat rejection section, and pump section. Each corresponding section of the report will include a brief discussion supporting component selection and the choice of system configuration, and describing the details of manufacture of specific equipment.

Chapter 3 discusses the performance and safety analyses which were necessary to complete the final loop design. Proof-of-principle experiments were conducted to validate

the design features and meet the requirements of the MITR-II Reactor Technical Specifications and Safety Analysis Report. The Safety Evaluation Report (H-2) prepared using this information has been reviewed and conditionally approved by the MIT Reactor Safeguards Committee and the MITR-II Operating Staff.

Chapter 4 describes the operational tests, and the results, thereof, which have been been or must be performed to verify structural integrity and out-of-pile operability of the prototype loop .

Chapter 5 summarizes the design features and construction procedures for the PCCL, and suggests future alterations and improvements. The final design of the Loop is structurally flexible and robust enough that variations in the internal component orientation, material selection and experimental procedures can be accomplished easily.

Appendix A contains the results of experiments performed to determine the compatibility of MIT-PCCL materials and the liquid lead thermal bath. Appendix B presents the theory and assumptions use in the scaling of the MIT-PCCL to a full scale $3000MW_{th}$ PWR plant. Appendix C gives estimated dose rates based on activation experiments performed on either actual construction material samples or samples of materials that were proposed for the fabrication

of the PCCL. Appendix D presents the final design issues which will be presented to the MIT Reactor Safeguards Committee, in revision 1 to the PCCL Safety Evaluation Report (SER), for final approval to begin in-pile testing and operations of the MIT-PCCL. Revision 1 to the SER will also contain updated dose estimates based on irradiation studies performed on samples from the actual construction materials and components of the PCCL.

Chapter 2

As-Built Design

2.1 Introduction

In the sections which follow, specifications and supporting information is presented on component and material selection, and on fabrication and system arrangement of the first MIT-PCCL PWR loop. Where pertinent, design constraints imposed by MITR safety requirements or physical size constraints, are discussed.

Individual sections and sub-sections of the MIT-PCCL are shown in Figure 2.1 in schematic fashion. Each major feature shown will be discussed and described in more detail, together with isometric and section views, to provide an adequate level of understanding of form and function. Tables 2.1 and 2.2 document the existence and location of more detailed engineering drawings, material certifications, and operating manuals which provide supporting detail for our discussion.

2.2 Aluminum Thimble

The Aluminum Thimble, shown in Figure 2.1 , is the vessel which contains and supports the in-pile components of the PCCL. The thimble is constructed of 6061-T6 Aluminum. This conforms to the MITR-II Technical Specifications for
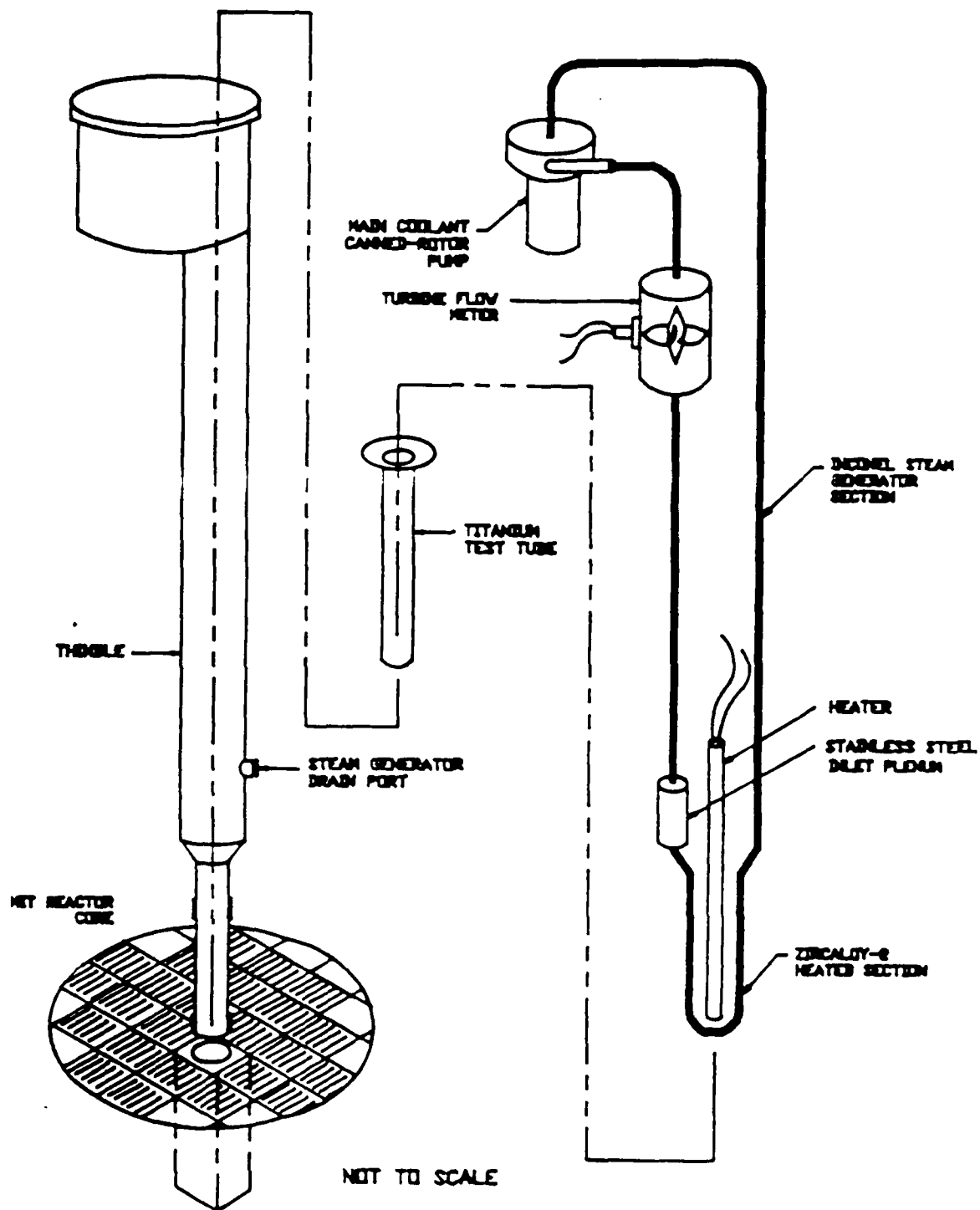
Figure 2.1 Major Components and Features of the MIT-PCCL

Table 2.1
Detailed Information on MIT Fabricated Components

	Component	File Drawing Number	QA File Number
1.	Dummy Element	R3L-1	M-86-2
2.	Thimble	R3L-2	M-86-2
3.	Titanium Test Tube	R3L-3	PCCL Project File
4.	Steam Generator Support Plate	R3L-4	PCCL Project File

NOTES:

1. Engineering drawings are filed in the MITR Operations Office, presently under the custody of the Nuclear Reactor Laboratory (NRL) Quality Assurance (QA) Officer   (617) 253-4211

2. Quality Assurance files are file in the MITR Operations Office, presently under the custody of the NRL QA Officer (617) 253-4211

Table 2.2

Manufacturers Information on Purchased Components

Component	Manufacturer	Location of Manual or Specs
Main Coolant Circulating Canned-Rotor Pump	Northern Research and Engineering Corp. 39 Olympia Ave. Woburn, MA 01801 Mr. A.M. Heitmann (617) 935-9050	PCCL Project File
Heater	Delta M Corp. 525 Warehouse Rd. Oak Ridge, TN 37830 Mr. R. McCullough (615) 483-1569	PCCL Project File
Turbine Flowmeter	Flow Technology, Inc. 4250 East Broadway Road P.O. Box 52103 Phoenix, AZ 85072 (602) 437-1315	PCCL Project File
Tubing (Stainless Steel and Inconel)	A.B. Murray Co. PO Box 1000 Sharon, MA (617) 668-9203	PCCL Project File
(Square and round seamless Al tube)	TUBESALES 23 Thompson Rd. E. Hartford, CT 06088 (800) 243-0173	PCCL Project File
Aluminum Plate	Industrial Aluminum Co, Inc. 341 Second Ave. Waltham, MA 02154 (617) 890-2410	PCCL Project File
Charging Pump	American LEWA ARMAC Co. 1 Wakefield, MA 01880 Mr. R. MacDonald (617) 245-2600	PCCL Project File

Pulsation Dampener	Liquid Dynamic Pulse Twin ARMAC Co. 1 Wakefield, MA 01880 Mr. R. MacDonald (617) 245-2600	PCCL Project File
Back-Pressure- Regulator	Tesom ARMAC Co. 1 Wakefield, MA 01880 Mr. R. MacDonald (617) 245-2600	PCCL Project File
Fittings	Parker Ultraseal Harbor Control 470 Wildwood St. Woburn, MA 01801 (617) 933-4430	PCCL Project File
Fittings	Hoke Gyrolock Joseph H. Bertram & Co. Four Britton Dr. Bloomfield, CT 06002 (800) 243-2340	PCCL Project File
Ball Valves	Parker CPI Valves Harbor Control 470 Wildwood St. Woburn, MA 01801 (617) 933-4430	PCCL Project File
High Pressure Stop Valves	NUPRO U Series Bellows Valves Cambridge Valve & Fitting INC 50 Manning Road Billerica, MA	PCCL Project File

the compatibility of construction materials exposed to the MITR-II core cooling water.

The Thimble is structurally divided into 3 sections: Heated Section, Steam Generator Section, and Pump Section. The Steam Generator Section is itself internally subdivided into 3 subsections: Plenum Subsection, Electrical Subsection, and Heat Rejection Subsection. The Heated Section is 36 inches (90.72 cm) in length, and has an 0.125 inch (3.15 mm) wall thickness, with an elliptical cross-section of 2-1/2 by 1-13/32 inches (6.35 x 3.57 cm) major/minor diameters. The elliptical cross-section was selected for the in-core section of the Thimble for two primary reasons. The first goal was to allow sufficient lateral clearance between the zircaloy U-tube and the major diameter of the titanium test tube to accommodate a sufficient thickness of liquid lead to insure uniform heat distribution. The second reason was to limit the total mass of lead contained in the Titanium Test Tube (TTT) to reduce the gamma heating contribution. The difference in gamma heating as a result of the use of an elliptical cross-section rather than an equivalent circular cross-section is approximately 3.3 kilowatts. This will allow the TTT to passively reject sufficient heat by radiation to the aluminum thimble wall to prevent a zirconium-water reaction during severe post-accident conditions (see section 3.2.3).

The weight of the Thimble (approximately 200 lbs (90.91 kgs) in air under operational conditions) is supported by the core lower support plate and upper thimble support bracket. The upper thimble support bracket is designed to bear 100 percent of the loop weight with a safety factor of 3.0. The support bracket is adjusted to allow the Thimble to positively seat on the core lower support grid without unnecessarily loading the core support grid. The force imposed by the PCCL on the upper and lower grids can be adjusted to eliminate vibration, if necessary. The Thimble is aligned in the core by the dummy fuel element and thimble alignment keys. Inadvertent upward motion of the thimble is prevented by the core upper grid locking tab. The Heated Section contains, aligns and supports the core simulator Zircaloy-4 tubing and lead bath.

The Heated Section transitions from the elliptical section to the 4 inch (10.16 cm), 0.25 inch (6.35 mm) wall thickness, circular Plenum region of the Steam Generator Section via a transition piece. The transition piece provides fairing to minimize the impact of the PCCL on MITR-II core cooling flow. The Steam Generator Section, containing the Plenum, Electrical, and Heat Rejection Subsections, is 101 inches (256.54 cm) in length. The Thimble wall thickness was selected to provide sufficient rigidity for handling in both the normal vertical configuration and horizontally (if necessary) during

installation of the internal components. The wall thickness is more than adequate to withstand postulated combustion of the amount of hydrogen in the water circulating in the loop. This is equivalent to approximately 20 mg of TNT, based on relative heats of combustion. The Thimble is hydrostatically tested with helium to 500 PSIG (3.45 MPa) annually (see section 3.2.7).

The Pump Section is an 8 inch diameter (20.32 cm) circular section with a 1/4 inch (6.35 mm) wall thickness, and is 12 inches (30.48 cm) in length. The Pump Section contains the Main Circulating Pump, and umbilical connections to instrumentation, heater power, charging and sampling systems, safety reliefs and the helium atmosphere charging flask, through vacuum and pressure tight connections in the Pump Section access cover. The Pump Section also connects and vertically aligns the thimble with a bridge across the MITR-II core tank.

2.3 In-Pile Heated Section

2.3.1 Zircaloy-4 Fuel Rod Simulator

The simulation of the PWR core flow cell consists of a 6.5 ft (198.12 cm) section of zircaloy-4 seamless tubing bent into a U tube. Zircaloy was chosen as the heated section material because of the virtually universal use of zircaloy as fuel cladding material in PWRs and BWRs. The zircaloy tubing is 5/16 inches (7.94 mm) in outside

diameter, with a 0.025 inch (0.64 mm) tube wall thickness. Zircaloy-4 tubing will be used for the PWR loop experiments. This tubing was provided by EXXON Nuclear with the cooperation of Sandvik Special Metals. It was rocked down to the dimensions required for our experiment from standard PWR fuel clad tubing. Approximately 48 inches (121.92 cm) of this tubing is within the MITR core in direct contact with the liquid lead bath contained in a titanium test tube. Appendix A presents the results of compatibility experiments run on zircaloy, low carbon steel and 316 stainless steel in a liquid lead bath at 750 degrees F (398.89 degrees C).

2.3.2 Titanium Test Tube

The Titanium Test Tube (TTT) is a crucible containing the molten lead, zircaloy tube, resistance heater and thermal interrupt. Titanium was selected for its compatibility with liquid lead (L-1), high strength at elevated temperature and low radionuclide activation. The TTT is filled with reagent grade lead to a depth equal with that of the top of the MITR core. The TTT is elliptical in cross section, major axis: 2 inches/5.08 cm, minor axis: 3/4 inch/1.9 cm, conforming to the inside wall of the Thimble. The TTT to Thimble gap is maintained by the use of a 0.035 inch (0.9 mm) titanium wire helically wrapped on the outside of the TTT. The TTT to Thimble clearance was kept to a practical minimum to limit the floodable volume (chapter 3,

subsection 3.2.4, discusses the design restriction on the floodable volume).

2.3.3 <u>Resistance Heater</u>

The design requirements for the "dip-stick" type resistance heater are as follows:

* 10 kilowatts/ft (164 watts/cm) maximum heat output
* 1/2 inch (12.7 mm) maximum outside diameter
* Ungrounded heater sheath
* Sheath material compatible with liquid lead
* Sheath material must have a low neutron capture cross-section to minimize the activation of the heater
* Boron free heater ceramic insulation
* Continuous duty at full power for an estimated three month period

The resistance heater selected for the PCCL project was purchased from the DELTA <u>M</u> Corporation of Oak Ridge Tennessee. The heater technology used by DELTA <u>M</u> originated in 1978 with the development of Fuel Rod Simulators (FRS) at Oak Ridge National Laboratory (ORNL). ORNL designed, developed and fabricated FRS for the Light Water Reactor (LWR), Liquid Metal Fast Breeder Reactor (LMFBR) and Gas Cooled Fast Breeder Reactors (GCFBR). The FRSs are special

heaters that are intended to operate with extremely high reliability at high heat flux and high temperature.

The FRSs were developed to support out-of-reactor core design experimentation and thermal hydraulic safety test programs. The design specification required the FRSs to perform reliably under postulated nuclear reactor core accident conditions. These conditions included surface heat fluxes as high as 1935.5 watts/in^2 (300 watts/cm^2) in liquid metals at temperatures as high as 1832 degrees F (1000 C) and rapid thermal transients to 122 degrees/sec (50 degrees C/sec). From 1978 to 1984 more than 300 FRSs were fabricated in support of four ORNL programs, other United States research institutions and the Canadian Atomic Energy Commission. Of the FRSs used in the aforementioned research, no unplanned FRS failures occurred in more than 700 out-of-reactor thermal hydraulic tests.

The heater purchased for our application meets or surpasses all design requirements. The heater is capable of delivering 22 kilowatts of power at 277 VAC single phase, with a constant linear heat flux, over a heated length of 21 inches (53.3 cm). To prevent stray currents, magnetic fields and induced electropotentials, the heater was designed with linear heating coils instead of spiral coils, and the heating elements are electrically isolated from the heater sheath. The heater is internally insulated with

Magnesium Oxide (MgO). The heater sheath is fabricated of low cobalt 1018 low carbon steel with an outside diameter of 0.44 inches (11.2 mm). The heater is internally instrumented with a small diameter thermocouple placed in the region of highest expected temperature. DELTA M̲ provided the PCCL group with samples and certifications of all materials use in the construction of the heater for neutron activation analysis (NAA). The NAA determined that the heater could be re-used for several loop experiments before the heater dose rates become a radiation handling problem for loop technicians.

The heater power supply is a single phase solid state zero voltage switching saturable current reactor (SCR) current regulator with demand oriented transfer (DOT). The heater output is controlled by a 0 to 100 degrees F (0 to 37.8 degrees C) differential temperature manual bias control.

2.3.4 Liquid Lead Heat Source

To reduce the total heating requirement of the Loop and to insure uniform heat transfer in the Heated Section's zircaloy tubing, the conceptual Loop design (B-2) used a liquid lead bath. The final detailed design retained the original use of liquid lead, but reduced the total volume of lead by the use of an elliptical cross-section for the TTT. The MITR core average in-core gamma dose rate of 1×10^9 RADS/hr (1×10^7 GRAY/hr) (H-1) produces approximately 2

watts/gm peak value, 1 watt/gm average, of deposited thermal energy in the lead bath. The volume of lead in the TTT is approximately 24.77 in^3 (406 cm^3), corresponding to a total gamma heat gain of 4.63 kilowatts.

To prevent the lead bath from over-heating, either from failure of the electric resistance heater controller, with or without a concurrent loss of PCCL cooling water flow, an aluminum Fusible Link shut-off device has been incorporated into the final design. The Safety Evaluation Report (SER) for the MIT-PCCL (H-1), discusses experimental data and calculations supporting the contention that there is sufficient passive cooling to prevent Loop temperatures from exceeding the NRC post-LOCA zircaloy-water reaction temperature limit (2,200 degrees F, 1,204.4 degrees C). A summary discussion is presented in Section 3.2.3 of Chapter 3 of this report.

2.3.5 Fusible Link

To provide a passive safety device to shut-off the electric heater in the event of a power surge or failure of the heater controller, an aluminum Fusible Link was designed jointly between the PCCL design group and DELTA M corporation. In this device an aluminum wire is placed in series with the heater and carries the full load current of the heater. At a temperature of approximately 1200 degrees F (649 degrees C), the aluminum in the link will melt and

open the circuit to the heater. Figure 2.2 shows the dimensions and construction of the Fusible Link. The link uses proven heater fabrication technology to produce a "dip-stick" type heater sheath with an aluminum conductor replacing the heater wire. The link sheath is immersed approximately 1 inch (2.54 cm) in the lead bath in the region at the top of the MITR core. In the heater fabrication process, the heater wires are welded together at the bottom of the heater sheath by first sand-blasting the MgO insulation from the bottom inch of the sheath, welding the connections together, and then back-filling the sheath with MgO to complete the heater prior to welding the sheath closure bottom in place. The Fusible Link was fabricated in the same manner except that the sheath was not back-filled with MgO to provide a gap at the bottom of the link. The gap was needed to accommodate the liquid aluminum, since aluminum expands on melting and a free volume was therefore needed to positively open the electric circuit

2.3.6 Heated Section Instrumentation

The Heated Section instrumentation consists of thermocouples to monitor the heater performance, and, at least on the initial operational Loop, establish the relationship between loop fluid temperatures, heater power, heater temperature, and lead bath temperature. As discussed in section 2.3.3, the heater contains an integral thermocouple in the expected hottest region of the heater.
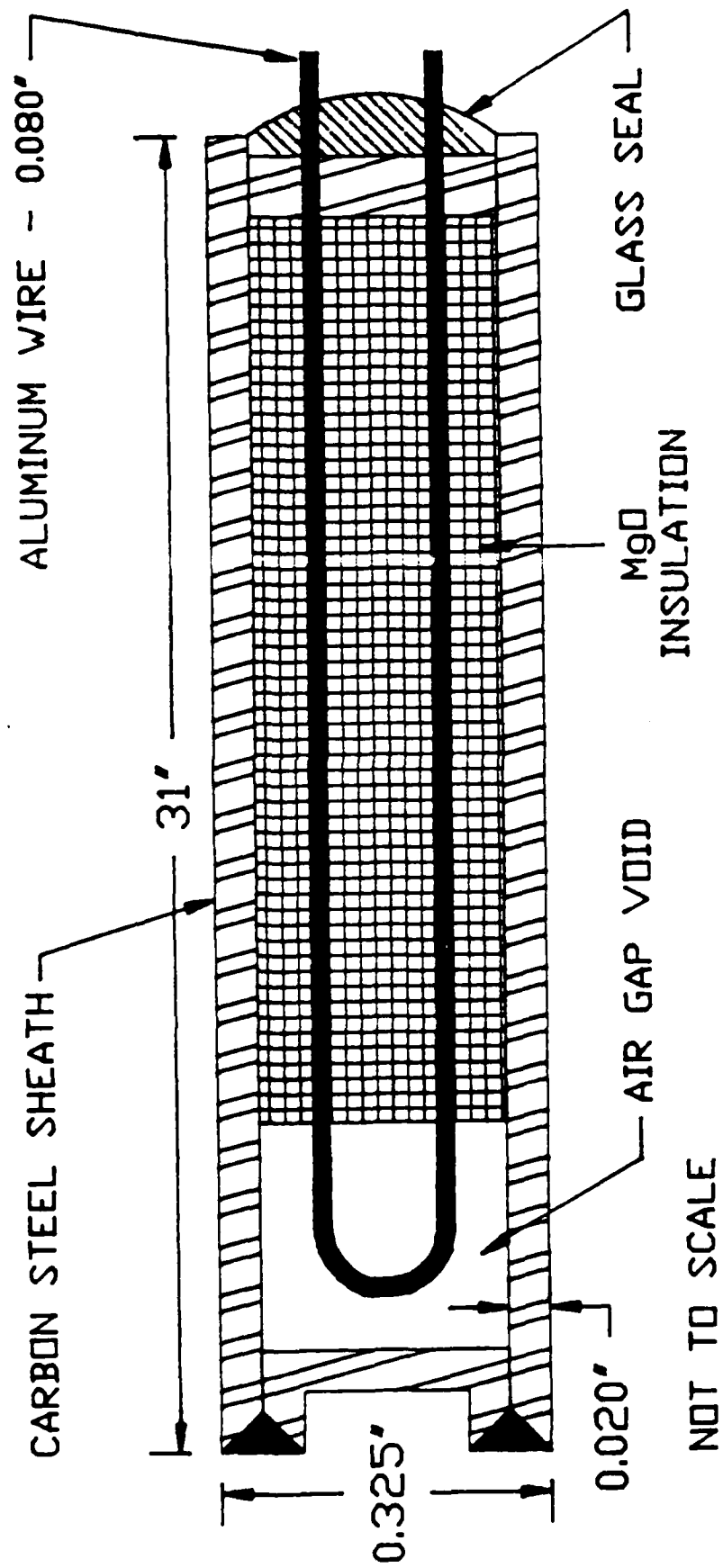
Figure 2.2 Fusible Link

Two thermocouples are installed in the lead bath. One thermocouple is located near the top of the lead bath, and the other is located deep in the lead bath. The use of two thermocouples provides redundant temperature monitoring and data on the temperature distribution in the lead bath. These temperatures will be read out on the data logger computer. The heater thermocouple output is wired in parallel to the heater power cabinet and provides a signal to either a high temperature alarm, low temperature alarm or heater high temperature shut-down bistables. The alarm/shutdown functions are selectable as operational experience determines the stability of the loop at full power.

2.4 Plenum Section

2.4.1 Basic Design Criteria

The Plenum Section is the bottom subsection of the Steam Generator/Heat Rejection section in the MIT-PCCL Thimble. The principle functions of the Plenum Section are:

* House the loop core inlet plenum
* Serve as a transition between the elliptical Heated Section and the circular Steam Generator Section
* House shielding to reduce vertical streaming of radiation from the MITR-core through the void created by the Thimble

* Provide helium gas passage to the Heated Section through channels milled in the Thimble transition piece

* Provide structural support/foundation for the TTT

* Provide room to vertically position the electrical heater and Fusible Link

* Provide room to connect loop differential temperature thermocouples at inlet and exit to core region

* Provide room for electrical connections between the Heater/Fusible Link and the power cabling.

The Plenum Section, shown in Figure 2.3 is contained within the circular section of the Thimble. The Thimble outside diameter is 4 inches (10.16 cm) with a wall thickness of 1/4 inch (6.35 mm).

2.4.2 Loop Inlet Plenum Design

The sizing of the loop Inlet Plenum was based on matching the relative surface area ratios of zircaloy to stainless steel (and zircaloy to Inconel) to preserve total mass transport ratios between the Loop and a PWR. Table 1.1 displays the range of material area ratios quoted for a "representative" PWR. Stainless steel has the largest variation, possibly due to the complex geometry (e.g. control rod drive mechanism extensions, etc.) in the inlet plenum of the reactor vessel. In the present work it is

INCONEL TUBE

4' CIRCULAR TUBE

PARKER ULTRA-SEAL UNION

TRANSITION PIECE

ELLIPTICAL IN-PILE TUBE

ZIRCALOY-4 TUBE

ZORAL PLATE

PLENUM

ZORAL PLATE

HEATER

FUSIBLE LINK

INLET PLENUM

Cavity of Plenum Section is Filled with a Boron Impregnated Ceramic Material

TTT

NOT TO SCALE
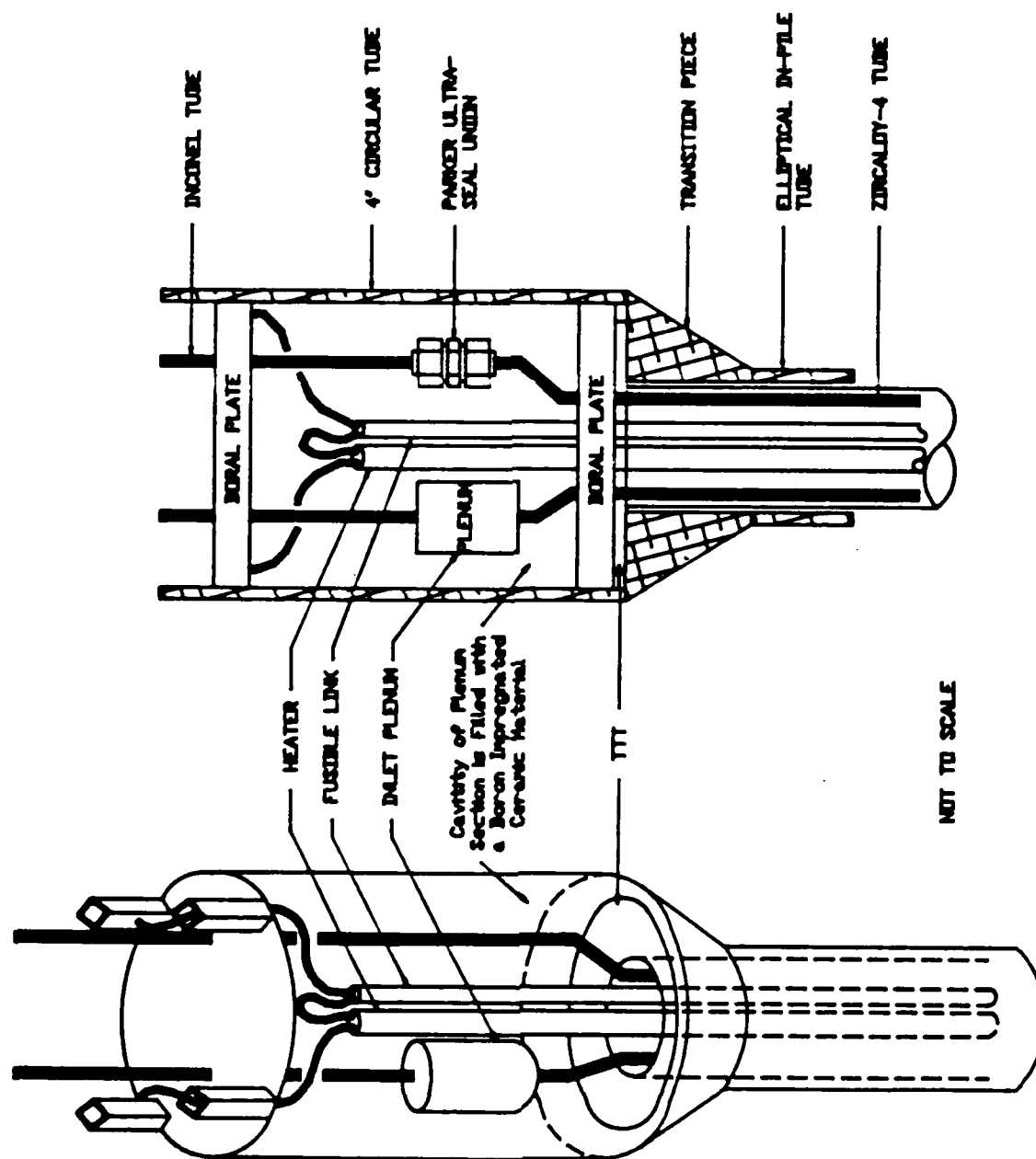
Figure 2.3 Plenum Section Isometric

assumed that the surface area of zircaloy comes entirely from the PWR core, and the Inconel surface area from the steam generators. The entire zircaloy surface of the core was used to calculate the surface area exposed to the coolant instead of the total heat transfer area of the core. For the Inconel surface area, the total steam generator heat transfer area was used (M-2),(P-1),(U-1). As can be seen from the following, the calculated values of surface area in the present work (PW) are close to published surface areas taken from the 1986 Bournemouth Conference papers 21 (P-1) and 22 (M-2). Calculated values are generated from parameters from a typical PWR used in the Nuclear Regulatory Commission's PWR fundamentals course, reference U-1.

Calculated Zr surface area (PW) $6,156 \ m^2$

Westinghouse PWR Zr surface area (P-1) $7,000 \ m^2$

French PWR Zr surface area (M-2) $5,530 \ m^2$

Calculated Inconel surface area (PW) $19,140 \ m^2$

Westinghouse PWR Inconel surface area $18,000 \ m^2$

French PWR Inconel surface area $13,600 \ m^2$

Calculated stainless steel surface area (PW) -- unable
        to calculate properly because of the complex
        geometry and insufficient information provided in
        standard power plant descriptions, such as FSAR's.

Westinghouse PWR stainless steel surface area   2,700 m^2

French PWR stainless steel surface area   2,372 m^2

Using the above data, the following material surface area ratios result:

Inconel to zircaloy surface area ratio:

    Calculated (PW)   3.11

    Westinghouse PWR   2.57

    French PWR   2.46

Stainless steel to zircaloy surface area ratio:

    Westinghouse PWR   0.386

    French PWR   0.64

Stainless steel to Inconel surface area ratio:

    Westinghouse PWR   0.15

    French PWR   0.26

The zircaloy surface in the MIT-PCCL is the fuel pin core simulator in the Heated Section of the loop. In the PCCL, approximately 50 inches (127 cm) of the 78 inch (198.12 cm) zircaloy-4 tubing is contained within the MITR core and exposed to the fast and thermal neutron fluxes. The Inconel inventory is in the "Steam Generator" Section. The sources of stainless steel in the loop are the Main Circulating Pump, The turbine flowmeter, and the Inlet Plenum. Using the surface area ratios generated above, a range of desirable Plenum surface areas result, leading to the following distribution of surface area in the Loop:

<u>Zircaloy-4</u> (OD - 0.315 inch/8 mm, ID - 0.26 inch/6.6 mm, tube length - 78.0 inches/198.12 cm) surface area

$$\underline{63.71 \text{ in}^2/411.04 \text{ cm}^2}$$

<u>Inconel Alloy 600</u> (OD - 0.375 inch/9.525 mm, ID - 0.319 inch/8.1 mm, tube length - 200 inches/508 cm) surface area

$$\underline{200.43 \text{ in}^2/1,293.12 \text{ cm}^2}$$

<u>Stainless steel (304/316)</u>

Main Circulating Pump	11 in^2/70.97 cm^2
Turbine Flowmeter	7.1 in^2/45.6 cm^2
Plenum	5.8 - 12 in^2/37.4 - 77.4 cm^2

Total stainless steel surface area:

$$\underline{23.88 \text{ in}^2 - 30.07 \text{ in}^2/154 - 194 \text{ cm}^2}$$

Hence the MIT-PCCL Ratios are as follows:

Inconel/zircaloy$_{(total)}$ = 3.15

Inconel/zircaloy$_{(in-core)}$ = 4.91

Stainless steel/zircaloy$_{(total)}$ = 0.785

Stainless steel/zircaloy$_{(in-core)}$ = 1.224

Stainless steel/Inconel = 0.249

The final design of the Heated Section stainless steel inlet plenum is a right circular cylinder whose inside dimensions are :

length: 6.6 inches/16.76 cm

diameter: 1.5 inches/3.81 cm

surface area: 32.74 in^2/211.23 cm^2

volume: 11.66 in^3/191 cm^3

Unfortunately, size constraints in the Plenum Section prevent expanding the plenum volume to a point where the loop in-core to out-of-core volume ratio is the same as the full scale PWR. The present design yields an in-core to out-of-core volume ratio of 1:9.36, whereas the PWR volume ratio is approximately 1:18. A future design might consider adding a small inconel plenum in the Steam Generator Section to increase the volume of the loop without increasing the stainless steel surface-area ratios.

2.4.3 Neutron Absorber Plug

The PCCL Thimble, if left relatively empty, would allow fast and thermal neutrons to stream vertically into the upper regions of the MITR core tank. To prevent this and to reduce the activation of PCCL components above the MITR core, the plenum section includes two 1/4 inch/6.35 mm thick Boral (B_4C in an Aluminum matrix) plates located at either end (see Figure 2.3). The free volume in between is filled with a boron impregnated ceramic material. The exact

composition of the ceramic matrix has yet to be finalized, but several machineable ceramic materials capable of binding sufficient boron or boron carbide to reduce neutron streaming are available.

2.4.4 Plenum Section Instrumentation

The Plenum Section's instrumentation includes the primary control thermocouples which provide inputs to the electric resistance heater controller, and the electrical connections to the Heater and Fusible Link. The differential temperature thermocouples, T_{cold} and T_{hot}, are mechanically (hose clamps) attached to the core inlet plenum and core outlet tubing, respectively. The heater controller will, for the first loop's operation, be controlled using the delta T control scheme, in the same manner as a PWR. The flexibility of the heater controller is such that it is possible to operate the loop using any other differential temperature, e.g. lead bath to T_{cold} temperature, or lead bath to Steam Generator mean temperature. The final decision as to the preferable heater control scheme will be based on identification of the differential, or absolute, temperature which controls the loop with the highest degree of reliability and stability.

2.5 Steam Generator/Heat Rejection Section

The Steam Generator Section (S/G) is the largest of the loop sections, measuring 75 inches/190.5 cm in length with a

4 inch/10.16 cm outside diameter circular cross-section.
The wall thickness of the S/G Thimble is 1/4 inch/6.35 mm.
The major components contained within the S/G are:

* Steam Generator medium drain port

* Copper shot-bed heat transfer medium

* Coolant charging and coolant sampling/let-down
  systems connections to loop

* Thermal insulation separating the S/G from the Pump
  Section

* Support spider and conduits for power and
  instrumentation leads and steam generator tubing.

2.5.1 Packed Bed Heat Transfer Medium

The 20 kilowatts of thermal energy transferred into the
primary circuit in the Heated Section is rejected to the
MITR cooling water in the S/G section of the PCCL. Chapter
3, subsection 3.2.2 discusses the experiments performed to
measure the effective thermal conductivity coefficient for
the packed bed heat exchanger.

The use of lead shot was evaluated by Burkholder (B-2),
but rejected for the final design because of the weight of
the lead shot, the low value of k_{eff}, and the proximity of
the operational temperature limit of the loop to that of the
liquidus temperature of the lead. The use of aluminum shot
was considered but was deferred for the present, although

the predicted value of k_{eff} for an aluminum bed in helium was 1.9 BTU/hr-ft-F/3.29 W/m-K (using a thermal correlation from Batchelor and O'Brien (B-4)): a value very close to the measured copper in helium gas shot-bed k_{eff}. The most important factor in passing up this option was that the aluminum shot was not readily available.

The final choice of the copper-helium shot-bed heat transfer medium was based on the following:

* The larger effective thermal conductivity compared to other shot-gas combinations
* The chemical stability and good corrosion resistance of the copper shot at normal operating temperatures
* The inerting properties of the helium gas, particularly in the presence of a hydrogen leak and its potential combustion inside the Thimble (discussed in chapter 3 subsection 3.2.7 and references H-1 and R-2)
* The relatively low thermal neutron absorption and activation cross-sections of copper
* The commercial availability of high purity pre-screened (sized) copper shot
* The compatibility of copper with other construction materials at elevated temperatures.
* The copper shot has sufficient heat capacity to damp out variations in loop temperature, due, for

example, to changes in reactor power, and to slow down the temperature cooldown rate due to a reactor shutdown or loss of loop heater power.

Another criterion which was applicable to all aspirant shot-gas composites was that of the susceptibility to thermal ratcheting in the shot-bed. During the design phase, consideration was given to whether the selected shot-bed could self-compact by thermal cycling and subsequently over-pressurize/over-stress the aluminum Thimble. Examination of this phenomenon showed that the change in Thimble cross-sectional diameter during the maximum temperature transient (start-up to full power) was less than the diameter of a copper shot particle or the hole size between shot particles : a criterion considered sufficient to prevent shot-bed ratcheting. It was also imperative that the particle diameter be large enough to prevent passage of shot between fuel plates in the MITR-II core, but small enough to interpose several particles between the S/G tubes and the Thimble wall. The purchased copper shot was screened to a particle diameter between 0.09 inch/2.3 mm and 0.13 inch/3.3 mm.

Finally, the Thimble helium pressure specification was set at 30 psia/0.207 MPa to provide an inherent mechanism to deter water from entering the Thimble in the remote possibility of a small MITR coolant to Thimble leak. This

helium over-pressure will also provide a modest increase in the shot-bed conductivity.

2.5.2 Heat Transfer Medium Drain Port

The copper shot drain port is located at the very bottom of the S/G section. Figure 2.4 shows the details of construction of the drain port. The number one concern of the design was to insure, under all credible circumstances, that the copper shot would be retained in the Thimble. To meet this design requirement, the drain port plug was designed with a backing flange which physically prevents out-motion of the drain plug. Additionally, the backing-flange bolts are captured in the body of the drain-plug body by set screws which are seal-welded during construction of the Thimble. Another design feature is a 3 degree difference in the contact shoulders of the drain plug to drain plug-body. The off-set allows the drain plug to be torqued into the drain-plug body repeatedly without losing a positive seating force. Also, the 45 degree drain plug taper angle was chosen to give the maximum seating surface. These design features give adequate protection against inadvertent discharge of the Steam Generator heat transfer medium, while still insuring ease of operation when intentionally draining the shot from the Steam Generator.

2.5.3 Steam Generator Tube Support and Positioning

2-SET SCREWS, 1/4" X 8-32 SCREWS
COVER WELD SET SCREWS TO
PREVENT COVER PLATE BOLTS
FROM BACKING OUT

2-ALLEN HEAD 7/8" X 8-32 BOLTS

S/G
DRAIN
PORT
PLUG

S/G DRAIN PORT

COVER PLATE
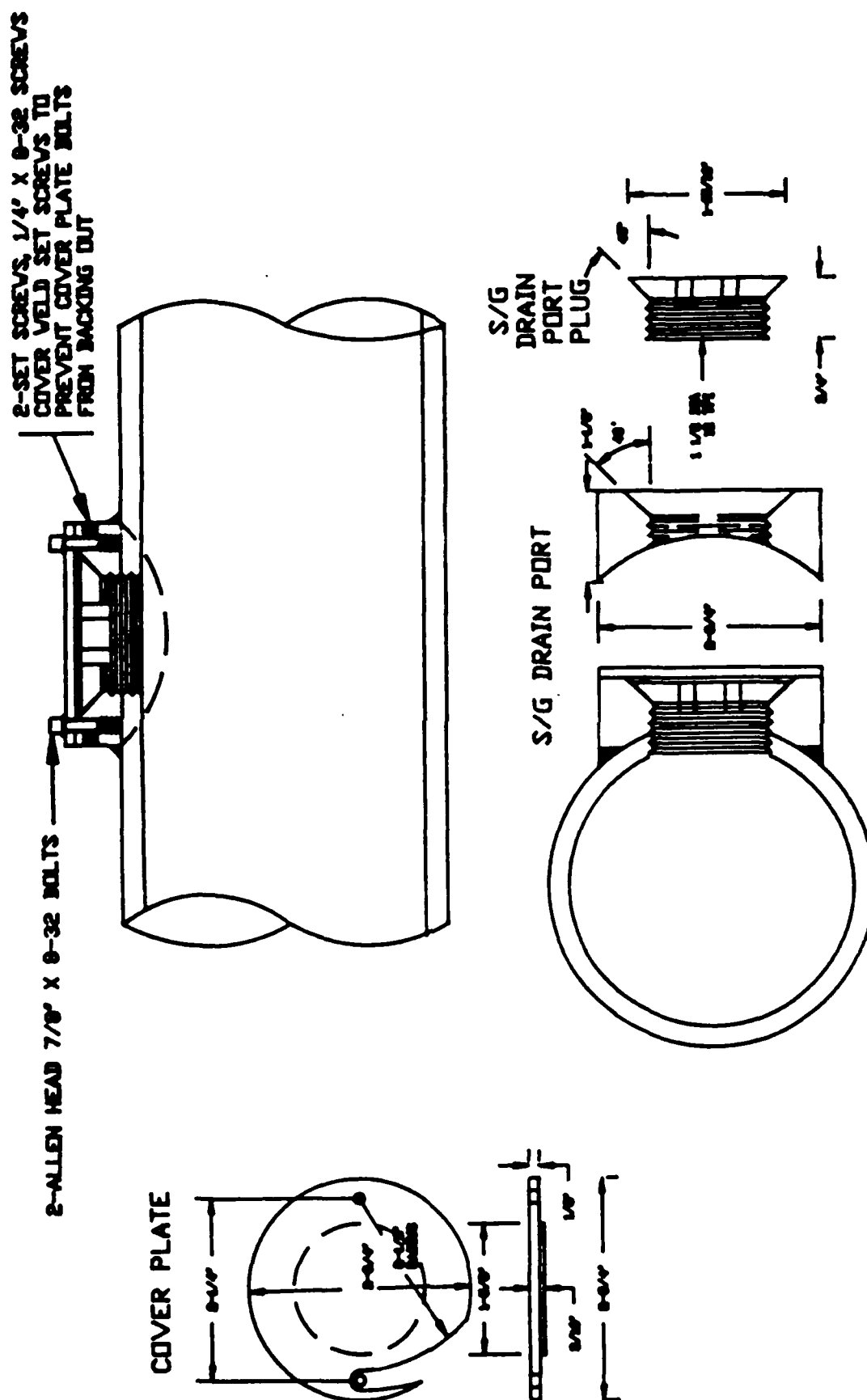
Figure 2.4 Steam Generator Shot Drain Port

The copper shot bed, discussed in subsection 2.5.1 of this report, is suspended from Pump Section bottom plate, see Figure 2.5. The S/G shot is loaded into the S/G section of the Thimble by pouring the shot through the Steam Generator Medium Fill Port (subsection 2.6.3). When loading the S/G medium, the Thimble is vertically oriented to insure the proper compaction of the copper shot. The Shot-Fill-Port is sealed by a 1 inch/2.54 cm stainless steel plug to prevent the shot from being ejected from the S/G section in the event of an accident to the loop. The 1/4 inch/6.35 mm thick aluminum lower S/G support plate is welded to four 1/2 inch/12.7 mm square aluminum conduit tubes. The other end of the conduit tubes are welded to the upper S/G support plate, which is bolted down to the Pump Section bottom plate. Square tubing was use to prevent shot particles from getting lodged between a circular tube and the Thimble wall.

The S/G conduit tubes support the S/G copper shot in the Thimble, prevent a pressure differential from building up between the upper and lower portions of the Thimble, and house the coaxial power leads for the electric heater and the thermocouple leads to the Heated and Plenum Sections. The high temperature coaxial cables are kept cool by the conduits being in physical contact with the Thimble wall. The Thimble wall is maintained at approximately 87 degrees F/100 degrees C by the MITR core tank coolant.

INCONEL TUBES

HELIUM PASSAGE HOLES

SHOT-BED FILL PORT

S/G UPPER SUPPORT PLATE

THERMAL INSULATION

NOT TO SCALE

COPPER SHOT

1/2″ X 1/2″ SQUARE
TUBE CONDUIT

SHOT-BED DRAIN PORT
(IN ALUMINUM THIMBLE WALL)

S/G LOWER SUPPORT PLATE
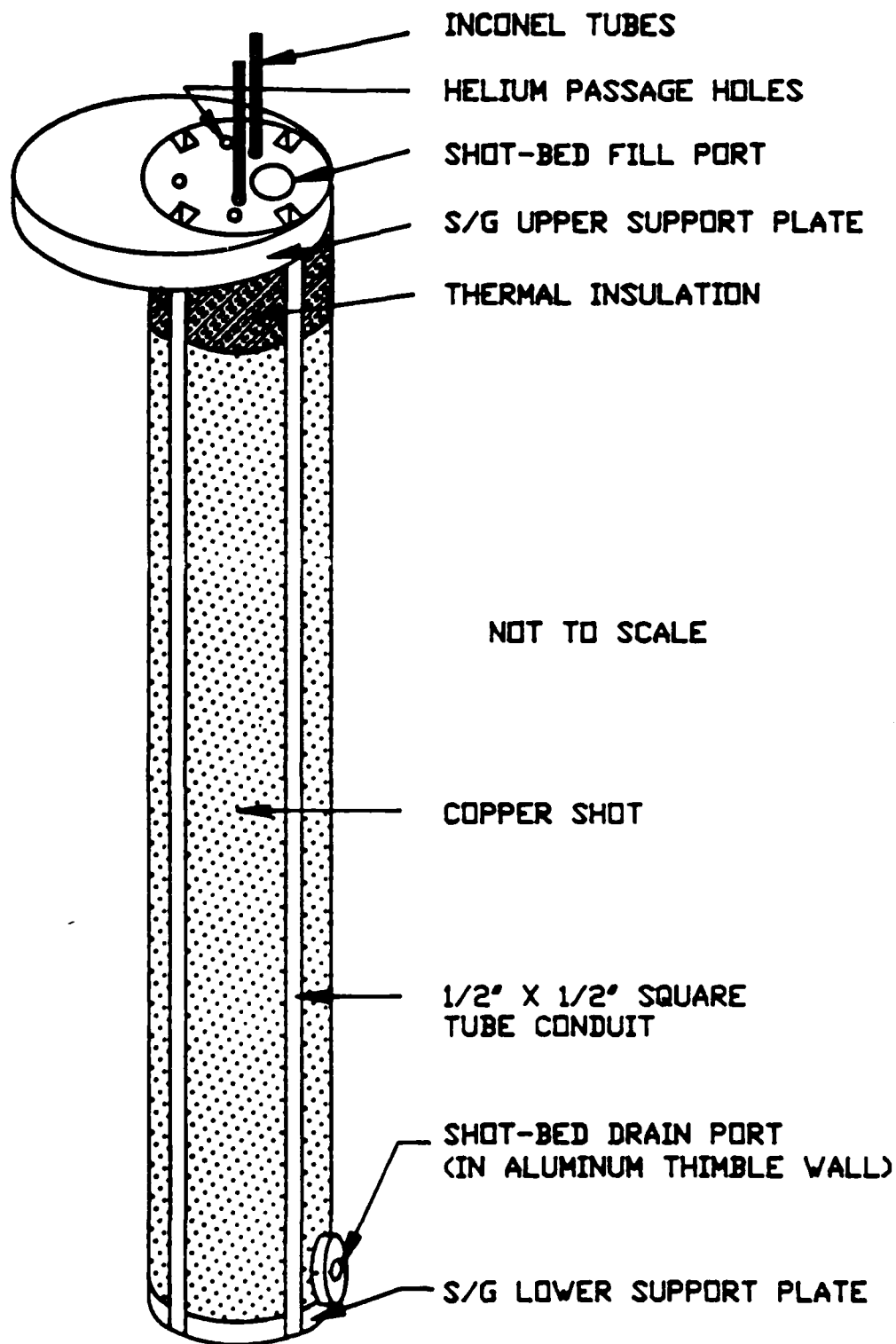
Figure 2.5 Steam Generator Support
Arrangement

The S/G support conduit tubes are forced outward against the Thimble wall by a "C" ring welded to the inner edges of the conduit tubes (see Figure 2.6). The "C" ring is positioned half-way up the S/G section. The "C" ring expands with temperature and insures that copper shot does not become lodged against the Thimble wall. Inside the "C" ring is welded a S/G tubing alignment block. This restricts movement of the S/G tubes while the S/G section is filled with copper shot and during normal Thimble handling. In order to change conductivity of the shot-bed, by adjusting the spacing of the S/G tubes, a new S/G upper and lower support plates and alignment block must be fabricated to the new specifications. The spacing of the S/G tubes for the initial operation of the loop divides the inside diameter of the S/G section Thimble into thirds (see Figure 2.6).

2.5.4 Charging and Sampling Connections

The Loop coolant inventory and system pressure are maintained at 2250 PSIG 50 PSIG/15.51 MPa 0.35 MPa by the out-of-pile charging system, (Figure 1.2 in Chapter 1 of this report). The Charging system is connected to the loop through 1/8 inch/3.18 mm OD 316 stainless steel tubing using a 1/8 inch X 3/8 inch X 3/8 inch (3.18 mm X 9.53 mm X 9.53 mm) 316 stainless steel Parker Ultra-Seal Tee fitting. This type fitting was selected for its ability to be made-up and broken several times without degrading the fittings ability to provide a positive seal. The Ultra-Seal uses stainless

54

1/16" WALL THICKNESS

5/8"

1 5/16"

9/16"

1/2"

Ø2 1/2"

Ø13/32"

2 5/32" 30

"C" RING CONDUIT SPREADER AND
TUBE ALIGNMENT BLOCK
ISOMETRIC SECTION
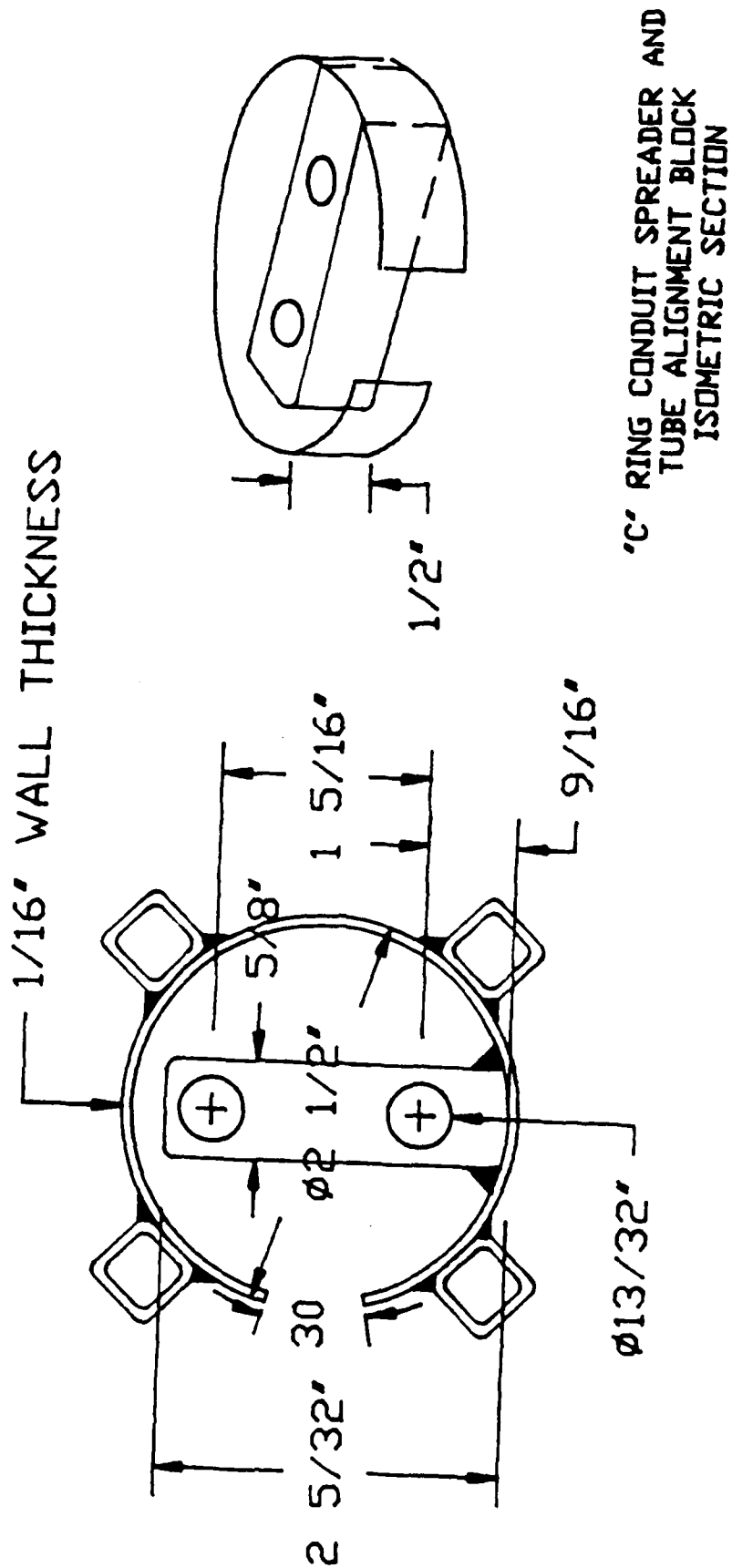
Figure 2.6 S/G Alignment Block
and "C" Ring Tube Spreader

to provide a positive seal. The Ultra-Seal uses stainless steel welded tube-end fittings with a nickel O-ring and stainless steel gland nut. The Parker Ultra-Seal and the standard Swadgelok fittings were evaluated for temperature cycling and pressure tightness during the liquid lead compatibility tests. The Swadgelok fitting did not consistently provide a first time seal. The Swadgelok fitting performance was related to the experience of the person making up the fitting. Whereas, the Parker fitting was virtually fool proof.

The coolant sampling/coolant let-down system (Figure 1.3) is also connected to the loop in the same manner as the charging system, through 1/8 inch/3.18 mm OD 316 stainless steel tubing. The sampling line is lagged with thermal insulation to insure that the deposition monitors and wet chemistry samples taken in the out-of-pile system are representative. The lagging also reduces additional heat loss to the Pump Section ambient atmosphere (30 PSIG/0.21 MPa helium).

2.5.5 Thermal Insulation

The top of the S/G section is insulated from the Pump Section by the use of a standard thermal fiber-glass insulation. The upper section of the PCCL loop itself is also insulated to prevent heat loss from the loop, and to reduce the ambient temperature of the Pump Section.

2.5.6 Steam Generator Section Instrumentation

The only instrument associated with the S/G section is one thermocouple, which will be located in the Shot-Bed medium to provide performance data on the Shot-Bed heat exchanger during the initial operations of the loop. Subsequent loop S/G sections will probably not need any instrumentation.

2.6 Pump Section

2.6.1 Main Coolant Circulating Pump

A world-wide search was conducted for the Main Coolant Circulating Pump. Ideally, the goal of the PCCL committee was to find a small canned-rotor centrifugal pump to closely simulate a PWR reactor coolant pump. The design specifications for the pump are as follows:

* Fluid - water with LiOH (1-10 ppm), Boric Acid (0-1000 ppm Boron), Hydrogen (3-100 cc/kg)
* Normal operating fluid temperature - 610-550 degrees F (321-288 degrees C)
* Normal operating pressure - 2250 psig (15.51 MPa)
* Hydrostatic test pressure - 3000 psig (20.68 MPa)
* Flow rate - variable 1 to 2 gpm (0.227 m^3/hr to 0.454 m^3/hr)
* Pump type - centrifugal

* Material limitations - no cobalt or cobalt alloys
  exposed to pumped fluid - 304 or 316 stainless steel
  or Inconel (625) preferred, bearing material -
  graphite

* Pump motor ambient temperature - 150-200 degrees F
  (65.6-93.3 degrees C) - Helium atmosphere at 3
  atmospheres)

* Leakage - none

* Pump differential pressure - 17 psid @ 1.5 gpm (0.01
  MPa @ 0.34 m^3/hr), 30 psid @ 2.0 gpm (0.014 MPa @
  0.45 m^3/hr)

* Ideally, to simplify the design of the system, the
  pump should operate without additional ,outside,
  cooling

* Pump instrumentation is limited to one stator
  winding thermocouple

* Continuous operation at full power for 1 year
  minimum

A mock-up, full scale, loop was constructed and operated at
low temperature to determine the loop differential pressure
for the pump specifications.

After a year-long search among pump manufacturers in
the United States, Europe, the United Kingdom and Japan, the
Northern Research and Engineering Corporation of Woburn,
Massachusetts, USA, proposed a canned rotor centrifugal pump

based on a pump Northern had developed for spacecraft air conditioning/refrigerant systems. One difficult aspect of the sought-for pump design was a desire to minimize the total internal surface area and volume of the PCCL pump. The pump had to be constrained in size because of the dimensional constraints imposed on the loop's Thimble, particularly in view of the desire to eventually install 2 loops in the MITR core tank simultaneously. The PWR reactor coolant pump volume to total reactor coolant system volume ratio is 4.41 X 10^{-3} (56 ft^3 12,710 ft^3). Because of the small volume of the PCCL (approximately 410 cm^3 , excluding the pump), modeling by equivalent volume ratios would require a pump volume of only 1.8 cm^3. Since any proposed pump would be significantly larger than the volume-similitude pump, the decision was made to fabricate the pump from 316 Stainless Steel. This would increase the inventory of stainless steel in the loop, and decrease the required stainless steel surface area in the Plenum. This design decision reduces the physical size of the Plenum, yet maintains the equivalent stainless steel to Inconel and stainless steel to zircaloy surface area ratios of a full scale PWR.

The Northern Research pump specifications meet or exceed all design requirements. The pump motor is a 400 Hz single phase, 120 VAC unit manufactured by Hitachi. The motor's small size and low power requirement eliminates the

need to cool the motor through an external secondary cooling system. The pump delivers 0.5 to 2 gpm (0.11 to 0.45 m^3/hr) in speed steps to a maximum speed of approximately 22,000 rpm. The total surface area of the pump exposed to the pumped fluid, suction to discharge, is 10.19 sq-in (65.74 cm^2). Pump fluid volume is 0.61 in^3 / 10 cm^3

The Main Circulating Pump is mounted vertically in the Pump Section of the Thimble with the volute above the stator/rotor-motor cavity. This orientation minimizes the possibility of dissolved gases coming out of solution in the low pressure suction of the pump and migrating into the motor-cavity, thereby vapor-binding the motor bearings (running the pump bearings with insufficient lubrication). Loop vacuum fill procedures also insure that the motor-cavity will be liquid-filled prior to starting the pump.

2.6.2 Main Coolant Pump Foundation

The Main Coolant Pump foundation is fabricated of aluminum, and is concurrently in physical contact with the pump motor stator frame and the cool Pump Section Thimble wall. This design adequately removes the relatively small amount of heat generated by the pump, and provides a rigid foundation for the pump, which operates at approximately 22,000 rpm.

2.6.3 Turbine Flowmeter

The flowmeter selected for the loop is a turbine flowmeter developed by Flow Technology, Incorporated, of Phoenix, Arizona. This flowmeter is capable of measuring low flow rates with a high degree of precision and reliability at the normal operating conditions of the loop. The body of the flowmeter directs flow tangentially past the underside of a paddle-bladed rotor. The rotor spins in a plane in line with the coolant flow in a manner similar to an under-shot water wheel. A modulated carrier RF pick-off, located above the rotor, senses rotor motion. The pick-off in conjunction with a signal amplifier provides a pulse output which is proportional to flow-rate. This signal can be fed to digital totalizers and recording devices.

The maximum temperature limit for this device is 750 degrees F/398.9 degrees C, and its maximum operating pressure is 6000 psia/41.4 MPa. The flowmeter is located in the pump discharge line, and requires approximately 5 inches/12.7 cm of straight tubing at the flowmeter inlet.

A venturi type detector was also evaluated, and a manufacturer identified (Fox Valve Development Company, East Hanover, New Jersey), but the cost, and physical size of the associated differential pressure detector was prohibitive.

2.6.4   <u>Steam Generator Medium Fill Port</u>

The copper shot is loaded into the S/G section through a 1 inch/2.54 cm diameter threaded hole in the S/G upper support plate (Figure 2.5 subsection 2.5.3 of this report). The fill port is normally sealed with a stainless steel plug to prevent the copper shot from being ejected from the S/G section in the event of an accident to the loop or during normal handling of the Thimble.

2.6.5 Charging, Sampling, Instrumentation, and Power Cable Connections

The systems internal to the Thimble are connected in an umbilical fashion to either side of the Thimble pressure envelope on the Pump Section closure cover. Penetration through the Pump Section closure cover is made through commercially available vacuum tight and pressure tight pass-throughs. A terminal board is mounted in the Pump Section for the thermocouple leads. This prevents excess lengths of thermocouple wire looping uselessly in the pump housing. In addition to the subject system connections there is a 3/4 inch/1.9 cm pipe which is also connected to the top of the closure cover. This pipe is routed through the biological shield of the MITR core to the vacuum draw-down and helium supply system.

2.6.6 Thimble Over-Pressure Protection

The Thimble is hydrostatically tested to 500 PSIG/3.45 MPa annually using helium. While at test pressure all

Thimble welds, joints and fittings are scanned with a helium leak detector to verify a gas tight envelope. During normal operation the Thimble is pressurized to 30 PSIG/0.21 MPa with helium gas. The Thimble is initially evacuated using a rotary vacuum pump, and then back filled with dry helium through the same 3/4 inch/1.9 cm helium charging line. The helium supply line has two redundant relief valves set at 50 PSIA/0.35 MPa. These relief valves are installed to relieve the initial expansion of the helium gas during startup and to relieve Thimble pressure in the event of a loop leak. Leakage from the loop will normally be detected by a Thimble humidity detector. The location of the humidity detector sample point on the Thimble, and the required flow rate to accurately detect a loop leak is being investigated at this time. In the event of a rapid pressure transient a rupture disk is installed in the helium supply line set to rupture before the 500 PSIA/3.5 MPa pressure is reached. Several rupture disk materials, with rupture pressures between 100 PSIA/0.69 MPa and 490 PSIA/3.38 MPa, are commercially available. The selection of the specific rupture disk is presently under investigation pending an estimate of the time to peak pressure following a credible hydrogen combustion accident.

2.6.7 Pump Housing Closure Cover

The Pump Section closure cover is an 11 inch/27.94 cm diameter circular section made of 1/4 inch/6.35 mm aluminum

plate. The closure cover is bolted to the Pump Section upper flange. A pressure tight seal is maintained by a stainless steel O-ring which is retained in the Pump Section upper flange in a machined square O-ring groove. The cover receives added stiffness from the presence of the pressure-tight pass-throughs. The closure cover bolts screw into the Pump Section upper flange. The upper flange holes are tapped to eliminate the chance of a nut dropping into the core tank. The access cover bolts are captured on top of the cover by the use of lock wires.

2.6.8 Pump Section Instrumentation

Instrumentation in the Pump Section includes a thermocouple which is embedded in the stator windings of the Main Circulating Pump, and a thermocouple which, for the first operations at power, will detect the ambient temperature of the Pump Section, to determine the adequacy of the cooling of the Pump Section and the Main Circulating Pump. The Pump Section is also the termination point for all Thimble instrumentation.

2.7 Thimble Alignment and Retention in the MITR

The final design of the alignment and structural support structure for the PCCL in the MITR-II core tank will be completed by the PCCL Project Staff prior to in-core testing of the first loop in September 1987. The basic design features of the in-pile support have been agreed

upon. A number of the design requirements are easily predicted from constraints imposed by the MITR core and core tank geometry. The PCCL Thimble has also been configured to accommodate installation in the MITR core tank.

The Heated Section of the Thimble is aligned in the MITR core by the use of a Dummy Fuel Element. The Dummy Element is fitted with vertical keyways which accept alignment keys that are welded to the periphery of the major axis of the Thimble's elliptical Heated Section (see Figure 2.1 in subsection 2.1 of this report). The bottom of the Thimble rests on the base plate of the Dummy Element, which is supported by the lower core grid plate. The base plate of the Dummy Element has milled channels which provide cooling water flow up past the in-core elliptical section of the Thimble. The Thimble-to-Dummy Element clearances were designed to allow roughly the same mass flow rate of MITR core coolant past the PCCL as passes through a fuel element.

Lateral movement of the Thimble is constrained by a support bridge which spans the core tank top. The support bridge is designed to align the Thimble laterally and to support the weight of the Thimble. The weight of the operational Thimble is approximately 200 lbs/90.91 Kgs in air, and approximately 50 lbs/23 Kgs less due to the liquid volume displaced by the Thimble (buoyancy) in the MITR core tank. The Thimble foundation and support bracket is

adjustable, and will support 100% of the Thimble's weight with a minimum safety factor of 3.0. The consideration here was to limit the loading on the lower core grid plate, but allow the option to apply some positive downward force to the lower grid plate in the event the MITR cooling water flow induces vibration of the Thimble.

The Thimble is also captured in the core and core tank. The MITR-II fuel elements are retained in the core basket by the use of an interlocking rotating grid plate. The elliptical section of the Thimble has a core upper grid locking tab which is positioned to prevent upward motion of the Thimble when the grid plate is in the normally locked position. Only when the grid plate is rotated to the refueling position, for that cell, can the Thimble be removed from the core. The vertical clearance between the Thimble Pump Section top, and the bottom of the core tank shield plug is approximately 6 inches/15 cm. This ultimately prevents the Thimble from being ejected from the MITR core, should all other restraints somehow fail.

2.8 Out-of-Pile Charging and Sampling Systems

The PCCL Coolant Charging System and Coolant Sampling/Let-down System diagrams, Figures 1.2 and 1.3, respectively, are repeated on the following pages for convenience.
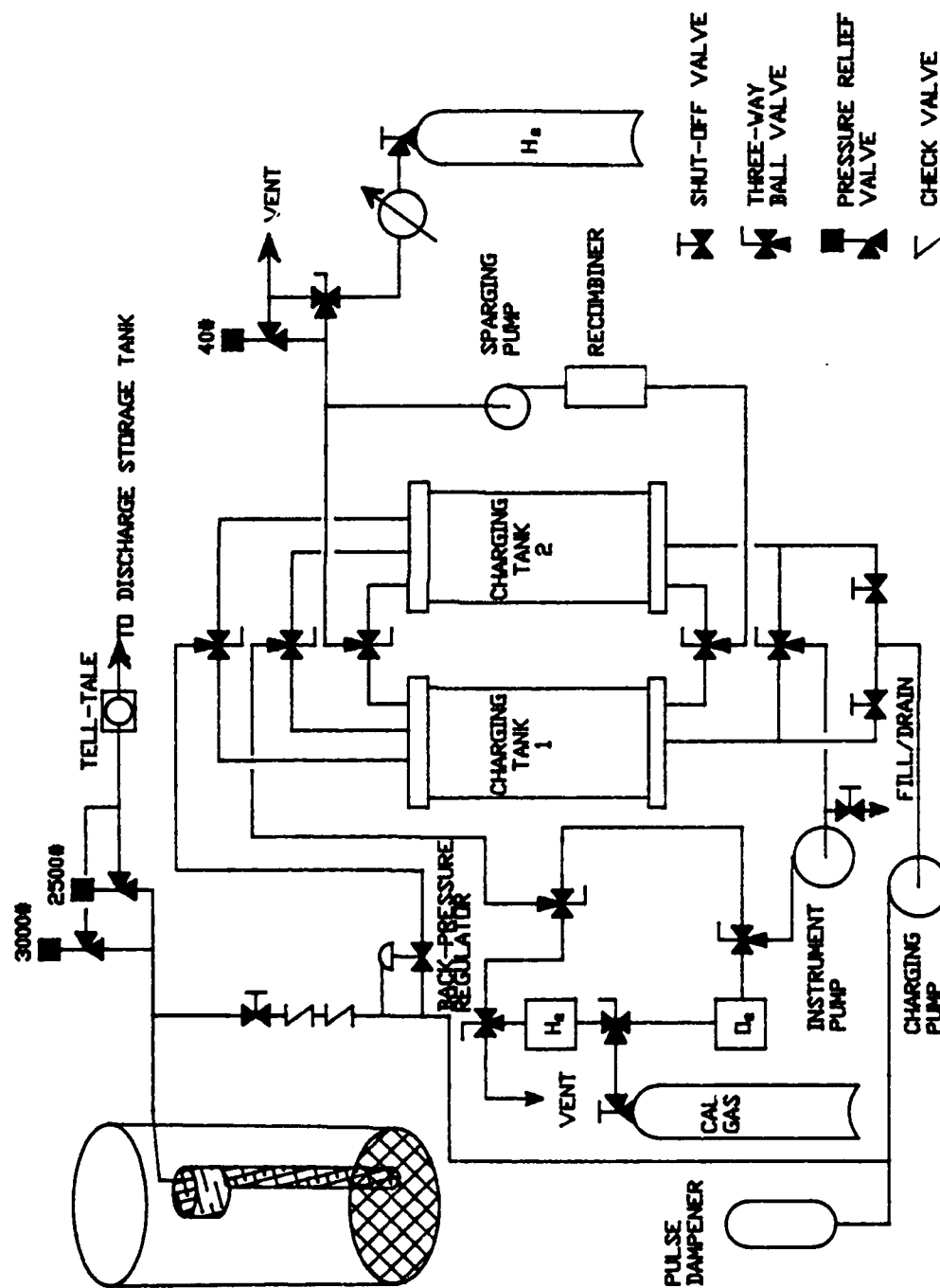
Figure 1.2. Out-of-Pile Charging System.
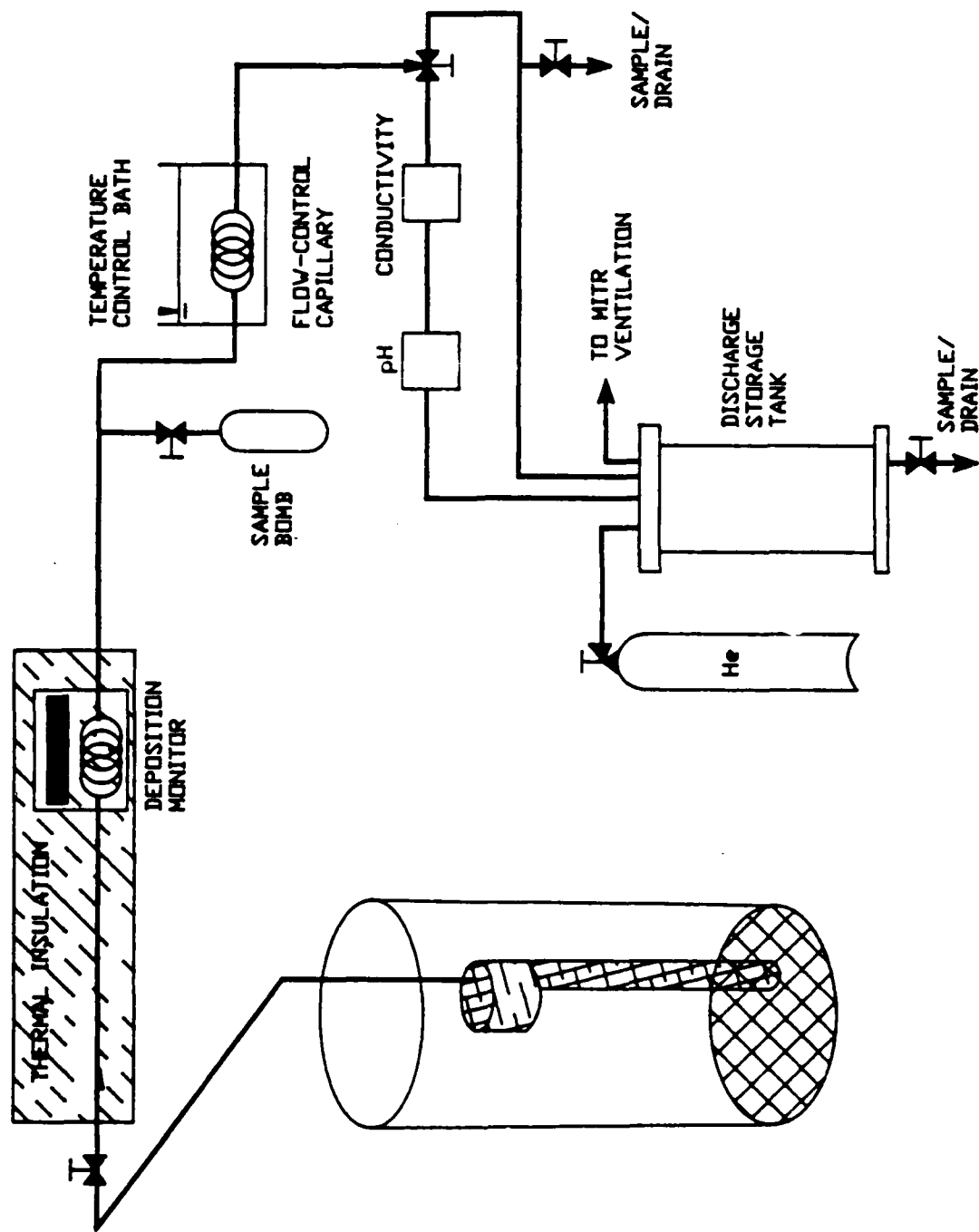
Figure 1.3. Sampling System.

2.8.1 Coolant Charging System

The PCCL Charging System maintains the loop pressure at the normal operating pressure (2250 PSIG/15.51 MPa), supplies make-up water through the let-down Sampling System, recirculates water through hydrogen and oxygen analyzers, removes dissolved oxygen from the charging water storage tanks, and provides over-pressure protection for the high pressure section of the primary loop and charging system.

The loop coolant inventory and pressure is maintained by the charging pump. The out-of-pile system charging pump is an American LEWA positive displacement metering pump capable of delivering from 0 to 2420 milliliters/hour at a maximum operating pressure of 4600 PSIA/31.72 MPa. The high discharge pressure capability allows the loop to be hydrostatically tested with the installed equipment, thereby saving the significant investment for a separate hydrostatic test pump and additional test gauges. The wide range of pump capacity is needed to provide flexibility in the sampling/let-down rate. In scaling the PCCL sampling rate to that of a PWR the coolant let-down rate could be as low as 20 to 50 cc/hr, whereas to allow a greater buildup of activated corrosion products on the out-of-pile deposition monitors a higher sampling rate would be needed. Discussions within the PCCL project committee, with EPRI and other consultants has supported this point of view, particularly since higher flow rates will also exert a

higher degree of control over loop chemistry. Establishing the final sampling rate is a trade off between similitude and data acquisition, and is the subject of an ongoing study by Morillon (M-1). The system pressure is adjustable from 0 PSIG/0 MPa to 4600 PSIG/31.72 MPa by the use of a TESOM back pressure regulator. The regulator bypasses coolant flow, in excess of that needed to maintain system pressure and fluid inventory, back to the on-service charging water storage tank (one of two parallel units). A Liquid Dynamic Pulse Twin Pulsation Dampener reduces charging pump pulsations to 1.0% (22 PSIG/0.15 MPa at normal operating pressure).

The design of the charging water storage tanks was based on creation of a tank which could contain approximately 3 atmospheres of hydrogen gas over-pressure to maintain a coolant dissolved hydrogen gas concentration within the concentration specified by the test in progress (normally in a range of approximately 3 to 100 cc/Kg H_2). The tank volume was selected assuming a charging/sampling rate of 250 ml/hr, and a desire that the tank last a week on service. Also the tank needed to have a simple, and reliable level indication. The charging water tanks shown in Figure 1.2 are 13.2 gallons/50 liter Pyrex glass tanks, for a total inventory of 26.4 gallons/100 liters. The use of 2 tanks allows the flexibility to adjust coolant chemistry prior to placing a standby tank on service, and to meet the demand for make-up water if the decision is made to

operate the loop at a higher coolant sampling rate. The hydrogen sparging system can be selected to serve either tank. This allows the off-service tank to be filled and its oxygen depleted prior to being placed on-service.

The hydrogen sparging circuit consists of a stainless steel, valveless piston (ceramic piston), metering pump, and a Atomic Energy of Cananda Limited (AECL) wet catalyst oxygen/hydrogen recombiner. The wet catalyst is capable of maintaining the charging water at less than 1 ppb oxygen.

The chemistry analysis loop is used daily to monitor the concentrations of hydrogen and oxygen in the charging water storage tanks, and to adjust the hydrogen concentration in the standby charging tank. A bottle of calibration gas is installed in the loop to calibrate the hydrogen meter. The calibration gas consists of 10% H_2 and 90% N_2. The hydrogen and oxygen meters are manufactured by Orbisphere. The instrument pump is a teflon gear pump. The sample pump not only circulates charging water through the O_2 and H_2 detectors, but also is used to fill the stand-by, off-service, depressurized charging water storage tank.

Over-pressure protection of the loop and charging system is provide by redundant high pressure Parker relief valves. The first relief is set to lift at 2500 PSIG/17.24 MPa, and the second relief is set to limit the maximum

pressure to the hydrostatic design pressure of 3000 PSIG/20.68 MPa. The relief valves relieve to the discharge storage tank through a visual, "bulls eye" leak detector.

2.8.2 Coolant Sampling/Let-Down System

The coolant sampling system (Figure 1.3) provides for the collection of discharged liquid, the analysis of pressurized and depressurized coolant, analysis of depressurized coolant for pH and conductivity, the collection of an atmospheric sample, and the collection of corrosion products in 2 capillary tube "deposition monitors".

The discharge storage tank collects coolant from either the pH and conductivity sample stream or directly from the sample line. The discharge storage tank also collects coolant from the Charging System relief valves. The storage tank is vented to the MITR ventilation system and is purged with helium cover gas to inert the tank atmosphere in the presence of hydrogen gas in solution.

The sample line contains two capillary tubes to collect and monitor activated corrosion products from the primary loop. The sample line is lagged with thermal insulation until after the coolant passes through the first capillary which is scanned by a NaI detector. The second capillary tube is immersed in a temperature controlled water bath to

control the flow rate of coolant let-down through the sampling system. The water bath will use the temperature-dependent viscous properties of water to vary the coolant flow through the capillary. Estimates are that the flow rate can be varied by a factor of 5 over a reasonable range of water bath temperatures. If the required temperature of the water bath exceeds 212 degrees F/100 degrees C at the desired flow rate , other liquid bath fluids can be used, e.g. oil. The geometry of the temperature control bath will be designed to provide the option to employ a NaI detector to scan the second capillary if deemed necessary.

2.9 Chapter Summary

This chapter has presented the structures and components which comprise the final design of the MIT-PCCL. The construction phase of the first loop is well underway at this writing and all major components for the in-pile and out-of-pile systems are either in-hand or under contract for delivery in the near future.

In conclusion, the design presented herein represents a simple, safe, flexible, and inexpensive research tool having a high degree of similitude for the study of Pressurized Water Reactor coolant chemistry.

Chapter 3

Performance and Safety Analysis

3.1 <u>Introduction</u>

This chapter presents an abridged review of some of the experiments and research conducted to support the final design presented herein. Also discussed is the design philosophy necessary in the development of a research tool which will reside within the physical boundaries of a NRC licensed 5 MW_{th} research reactor. The commitment to reactor safety has been the overriding consideration in selecting among numerous design alternatives to arrive at the design presented in this report. The detailed documentation of the safety issues involved in the design and construction of the PCCL is contained in the MIT-PCCL Safety Evaluation Report (H-1), and the reports and minutes of the MITR Safeguards Committee.

3.2 <u>Proof of Technology Studies</u>

3.2.1 <u>Liquid Lead Compatibility Experiment</u>

Experimental research was conducted to verify the compatibility of materials which would either come into direct contact with the liquid phase of the thermal bath's lead in the TTT, or which could experience indirect exposure to lead vapor. The design of the Plenum Section filler

block, and the thermal mass of the loop above the Heated Section, virtually insures that the lead vapors from the TTT will not migrate above the Heated Section. Appendix A discusses the experimental procedure used and the conclusions drawn from the research conducted.

The failure of the TTT wall and subsequent leakage of the liquid lead have been discussed at length and documented in the Safety Evaluation Report for the PCCL (H-1). A leak from the TTT will permit molten lead to contact the inner Thimble wall, where the lead will solidify on contact with the cool (100 degrees F/37.8 degrees C) wall. The resulting thermal short circuit will not create a sufficient hot spot to induce boiling - as demonstrated using a high powered soldering iron to deposit energy inside a 1/8 inch/3.18 mm aluminum tube immersed in a stagnant water bath.

3.2.2 Shot-Bed Conductivity Experiments

A series of experiments were conducted to determine the final selection of heat transfer medium and the effective bed thermal conductivity coefficient, k_{eff}. Experiments determined the k_{eff} for two types of shot (copper coated steel "bee-bee shot",and solid copper shot) and three quasi-static fill/cover gases (air, carbon dioxide, and helium). A "dip-stick" type heater was placed in the center of a 2.5 inch/6.35 cm, OD aluminum tube with a 3/16 inch/4.8 mm, wall thickness (the cylinderical section of a regulation size and

weight aluminum softball bat). The diameter of the heater was 5/16 inch/7.9 mm. The shot-bed was immersed in an ice bath to establish a reference heat sink temperature. Thermocouples measured the heater wall temperature, inner wall temperature of the aluminum tube at the mid plane, and the ice bath temperature. In the case of helium and carbon dioxide the gas was purged through the shot-bed to flush the air from the shot. The shot bed exhaust tube was immersed in water to isolate the bed from in-leakage of air and to monitor the amount of positive pressure in the shot-bed. The cover gas did not flow through the shot-bed, creating a stagnant gas shot-bed heat transfer medium closely resembling the Steam Generator section of the loop. Heater power was determined by separate voltage and current measurements. The resistance of the heater was periodically checked to correct the total heater power for temperature.

The shot-bed effective conductivity coefficient was determined using the following equation:

$$k_{eff} = [q'/2 \ T] * \ln(b/a) \qquad\qquad 3-1$$

where: q' = linear heat rate (heater power   heater length)

$T = T_{heater} - T_{wall}$

a = heater diameter

b = inner aluminum tube wall temperature.

The results were as follows:

SHOT	GAS	k_{eff} (BTU/hr-ft-F)
Cu coated steel	air	0.32
Cu coated steel	He	1.22
Cu coated steel	CO_2	0.37
Copper	He	1.67
Copper	CO_2	0.54

As discussed in Chapter 2, subsection 2.5.1, the copper shot in a helium atmosphere was selected for the shot bed heat exchanger for the loop.

3.2.3 Radiation Cooling Experiment

The philosophy underlying the design of the PCCL is to insure the safety of the MITR-II core and the PCCL during extreme accident scenarios. The major concern has been to design the loop to possess sufficient passive cooling to prevent material temperatures from reaching or exceeding recognizable safe limits, e.g. 2,200 degrees F post LOCA limit to preclude zircaloy-water reaction. Appendix 1.a of the Safety Evaluation Report (SER) (H-1) presents the detailed results of an experiment to verify calculations of the radiative heat transfer rate between the lead-filled titanium can and the cooled aluminum Thimble. The results of this experiment indicate that the maximum temperature (from a TTT and lead mass of 14.3 lbs/6.5 Kg giving a total

nuclear heating of   7.2 KW) is below the zircaloy-steam reaction temperature.   The margin to this limit has been improved by the reduction in the mass of lead in the TTT by the elliptical design of the TTT.   A full scale operational test , using the actual PCCL heater, Fusible Link and TTT is scheduled to be conducted prior to hot operational testing of the first loop.

3.2.4 Reactivity Effects of the Loop

Section 4, subsection a.2, of the SER discusses the potential reactivity effects on the MITR core caused by the reactivity worth of the loop and Thimble during water flooding or voiding incidents.   Three scenarios were examined:

* Sudden flooding of the entire void space between the TTT and Thimble walls
* Primary loop rupture, and draining of water into the in-pile section of the Thimble
* Undetected voiding of the volume between the Thimble and the Dummy Element, followed by a sudden reflooding.

The results of these calculations support the conclusion that in all the above cases the reactivity change is within the allowable reactivity restriction for movable MITR-II experiments.   Various design features and planned

proof-testing of the PCCL insures large margins for avoidance or mitigation of these potential accidents. Design features include:

* The Thimble is hydrostatically tested to 500 PSIG/3.5 MPa annually.

* The Thimble is leak tested with helium to 50 PSIG/0.35 MPa prior to going in-pile following pre-conditioning.

* The humidity detector, connected to the Thimble, can provide indications of water ingress either from the primary loop or from a Thimble envelope leak.

* The helium atmosphere of the Thimble, maintained at 30 PSIA/0.21 MPa, will insure that helium out-leakage would precede water in-leakage.

* The TTT upper flange fits snugly on top of the Thimble transition piece, and the only communication between the upper Thimble sections and the Heated Section is through three small channels milled in the transition piece to allow helium flow to the Heated Section. Thus, rapid draining of water from either a primary loop or upper Thimble wall leak into the lower section is unlikely. Moreover, the high surface temperature of the TTT (570 degrees F/300 degrees C) will insure that small amounts of water entering the void space are vaporized.

A confirmatory measurement of the actual reactivity difference with and without water in the PCCL primary loop will be made as part of the PCCL initial in-pile checkout.

3.2.5 Reactivity Effects of PWR Coolant Chemistry

The other reactivity effect of safety concern is associated with the presence of boron (B^{10} in the form of dissolved boric acid H_3BO_3) in the simulated PWR coolant contained in the primary loop of the PCCL. Section 4, subsection a.1, of the SER (H-1) presents a detailed analysis of the reactivity effect of the B^{10} on the MITR-II core. The B^{10} analysis considers two hypothetical scenarios:

* The sudden voiding of the in-pile primary loop water at a boron concentration of 2000 ppm.
* A non-mechanistic incident in which all the in-pile boron is first concentrated in the in-pile section of the primary loop and then ejected.

Analysis of both postulated accidents indicate that the reactivity worth of the B^{10} falls well within the allowable limit for movable experiments in the MITR-II core. Moreover to minimize tritium production and reactivity effects the, boric acid employed will be enriched in B^{11}. The B^{11}

isotope will be enriched to 98%, such that the B^{10} content is an order of magnitude less than in natural boron.

3.2.6 Loop Pressure Drop Experiment

In order to provide reasonably accurate specifications to prospective pump manufacturers and to determine initial flow velocities for the scaling of the PCCL, a loop pressure drop experiment was conducted. The experiment was conducted using a mock-up loop using tubing of the approximate length of that in the final loop. A small centrifugal pump was instrumented with a differential pressure detector across the pump, and flow rates were recorded. The diameter of the tubing was varied to estimate the friction factors and head losses expected from the loop. The result, as presented to prospective pump suppliers, was 30 PSID at 2 gpm (0.21 MPA at 0.91 M^3/sec).

3.2.7 Hydrogen Combustion

The possible combustion of hydrogen in the Thimble either from a zircaloy-steam reaction or from a leak from the primary loop of the PCCL are important considerations in the design of the PCCL.

In the design, every effort was made to limit the number and size of free volume voids in the Thimble. The helium over-pressure cover gas system is pressed into service as a stable inerting gas to sufficiently dilute the

Thimble atmosphere to prevent the transition to detonation of any hydrogen present in the Thimble atmosphere (R-2). The PCCL also includes a large cool surface area (Thimble wall and immediately adjacent copper shot). The large surface area to volume ratio of the shot-bed S/G section, exploits the same concept as commercially available flame arrestors. The Thimble section is sufficiently baffled to prevent the formation of a flame front which could travel the length of the Thimble and rupture the Thimble pressure boundary. Finally, the Thimble helium supply line has the multiple protection feature of redundant relief valves and a rupture disk.

The out-of-pile combustion of the available hydrogen gas was also considered. The SER (H-1) details the analysis of igniting all the available hydrogen in the hydrogen supply cylinder for the over-pressure cover gas for the Charging Water Storage Tanks. The conclusion of this analysis is that the maximum energy release is 3,400 BTUs/1 KW - about the same as the energy generated by a cup (250 ml) of fuel oil. Hydrogen accumulation in the event of leakage is prevented by the use of a ventilation fan.

3.2.8 Estimated Radiation Levels and ALARA Considerations

The PCCL components located in the Heated Section of the Thimble will be exposed to a significant neutron flux from the MITR-II core. These materials include:

Material	Component
Aluminum T6061	Thimble
Inconel 600 series	Heater wire
Lead	Conduction Bath
Magnesium Dioxide	Heater Insulation
Steel 1018	Heater Sheath
Titanium	TTT
Zircaloy-4	U-Tube primary loop

The in-core materials were chosen to minimize the activation of the components within the constraints imposed by the functional requirements of the components. In several cases, the activation of the components is due largely to impurity elements. Appendix C gives an inventory of the elements present in the in-core section of the PCCL, and estimated unshielded gamma dose rates for the PCCL components. The dose rate results are presented for components exposed for 20 full power days followed by a 60 hour decay. The irradiations were performed, when available, on samples which are the actual fabrication materials for the PCCL. In several cases, manufacturers cooperated in supplying samples of the materials which were used to fabricate the actual components. In some cases a

similar material had to be substituted to complete these initial tests. The results of the irradiations showed the dose rate levels are within the experience level of the MITR operational staff and radiation protection personnel. The results also confirm that the post-irradiation disassembly of the PCCL and post-mortem analyses can be carried out using standard, on-site equipment and procedures.

3.3 <u>Safety Analysis</u>

Figure 3.1 summarizes the PCCL safety features which have been incorporated into the final design and operational planning. The following list summarizes the accidents which have been carefully considered by the PCCL Project staff and have been analyzed in detail in the SER (H-1), and the mitigating factors which reduce the effects of incidents involving the PCCL:

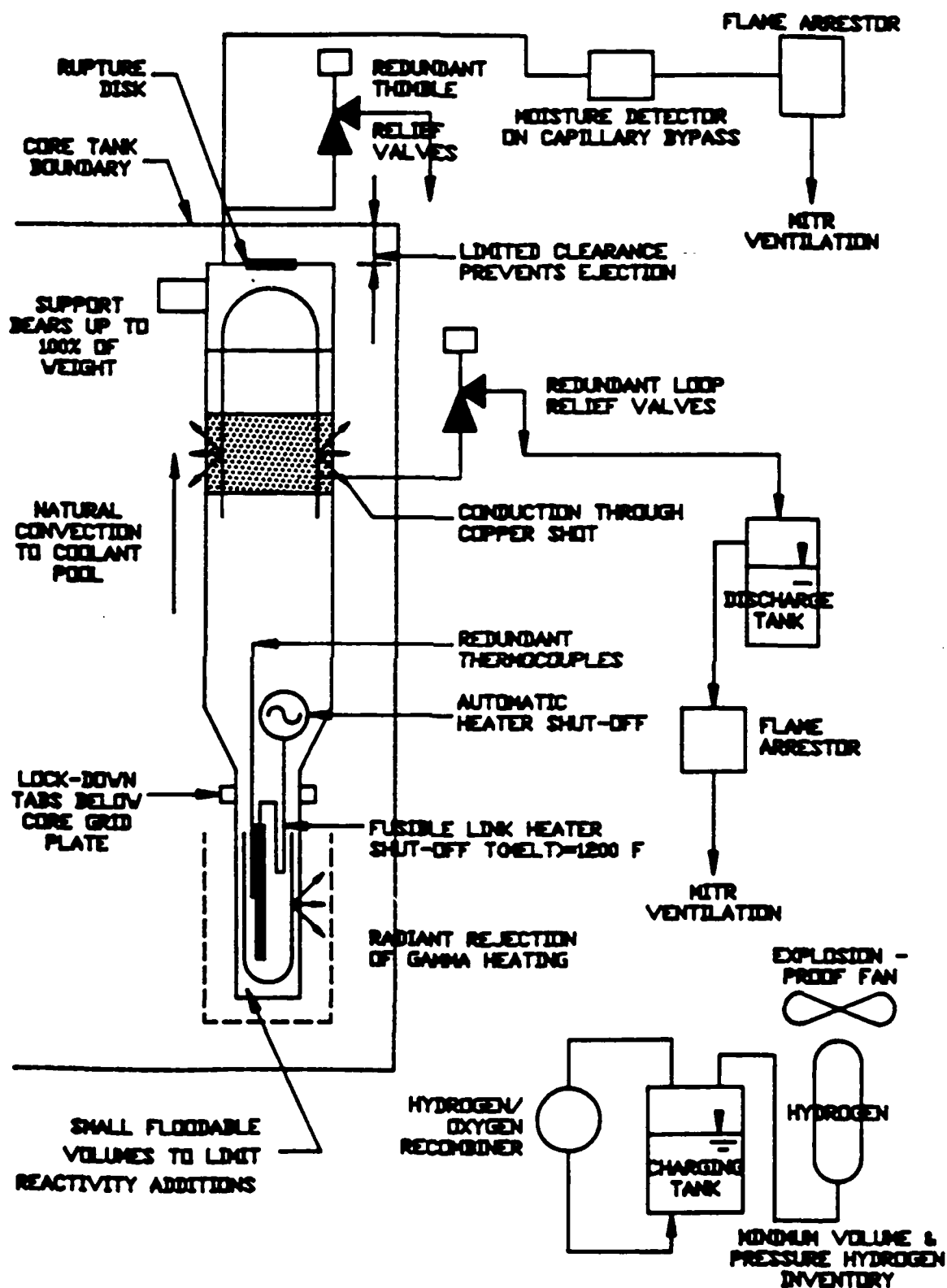| <u>Event</u>	<u>Mitigating Factor</u>
Loop rupture/Loss of coolant/Relief valve fail open or fail to reseat	Turn off Heater/ Turn off charging pump
Severe Thimble Leak	Turn off Heater/ Initiate Cooldown
Failure of Charging Pump	Turn off Heater

Figure 3.1 Summary of PCCL Safety Features

Heater Control Failure   Manually turn off heater
(fails to turn off)   power on high temperature
                                          alarm. If high temperature
                                          alarm and heater cut-off
                                          fail simultaneously, the
                                          Fusible link will melt and
                                          cut off heater power.

Failure of Main Circulating   Turn off Heater
Pump

Computer Failure   Heater cut off on loss of
                                                         computer

Loss of site electrical power   Heater control
                                          circuit designed with low
                                                 voltage release.

Emergency electric power is not required in the event
of loss of site power. Over-pressure protection (relief
valve) is spring actuated, and passive cooling of the PCCL
is provided by radiation to the MITR core tank.

Appendix D presents a preliminary outline of the MIT-
PCCL SER revision/update to be presented to the MITR
Safeguards Committee in August 1987. This revision

incorporates changes to the loop design which affect reactor safety since the initial safeguards committee meeting.

3.4 Chapter Summary

This chapter has presented a summary description of the design features and operating procedures of the PCCL in sufficient detail, and supported by the detailed analysis contained in the SER and PCCL project files, to demonstrate that the PCCL can be operated safely within the MITR Technical Specifications by the MITR operating staff and the PCCL project engineers.

Chapter 4

Loop Pre-Operational and Operational Tests

4.1 Introduction

The work reported here has advanced the facility status
to the point where all major components have been purchased
or fabricated. Numerous details, including a series of
shakedown tests remain to be accomplished, however. The
sections which follow in this chapter document requirements
in this area as projected from the perspective attained over
the past fifteen months of design and construction.

4.2 Loop Pre-Operational Tests and Inspections

This section specifies tests and inspections which must
be accomplished on each component prior to cold operations
testing. Pre-operational tests include quality assurance
checks which need to be performed on components fabricated
by the MIT Nuclear Reactor Laboratory Machine Shop, and
receipt inspections performed on equipment received under
contract from manufacturers.

Dummy Fuel Element - Complete surface finish, structural and
        dimensional checks, and certify the quality assurance
        prior to installation into the MITR II core.

- Fabricate and certify dummy element handling tools, and install dummy element in the MITR II core.

Thimble: - Accurately determine correct alignment of the Thimble in core tank, and properly position the Pump Section.

- Determine correct back out distance of the S/G shot port cover plate, tighten retaining set screws, and tack weld set screws.

- Weld Pump Section to Steam Generator Section.

- Helium leak check the Thimble.

- NDT, dye Penetrant checks of Thimble welding to be completed by Artisan Industries Inc.

- Hydrostatically test the Thimble to 500 PSIG with helium. Leak check at pressure for 30 minutes.

- Complete quality assurance package, and certify Thimble for use in MITR II core.

- Align the S/G support plate in the Thimble, and weld upper support plate in the correct position. Insure

that the S/G support conduit does not interfere with the operation of the S/G drain port.

- Complete the Thimble access cover design, and fabricate. Vacuum check and pressure check access cover pass-throughs.

- Select the humidity detector required for the Thimble and make final decision on the correct installation in the Thimble access cover to monitor for water leakage into the Thimble void space.

- Complete final design of Thimble access cover O-ring groove and select O-ring material.

Loop support Bridge - Design, fabricate and test loop support bridge. Verify correct alignment, strength, and operation of the bridge and the loop load adjustable foundation.

Canned-Rotor Pump - Complete receipt inspection of the Main Circulating Pump when received from the manufacturer.

- Establish procedures to prevent foreign material intrusion into the loop during construction and fill procedures to prevent damage to the pump bearings.

- Install the pump in a test loop, and conduct a 24 hour test run of pump at low temperature (because of the narrow tolerances of the pump design, initial testing should be coordinated with the manufacturer to prevent violating any guarantees). Determine the system characteristics curve and pump curve for the loop at low temperature. Determine the loop friction factor.

- Conduct a 24 hour test run of the pump at operating temperature. Determine system characteristic and pump curves. Determine loop friction factors at temperature.

- Complete the design, fabrication and fit-up of the pump foundation.
- During cold and hot operation tests, determine the Pump Section enclosure temperature as a function of loop power, and the resulting pump stator temperature. Based upon the measured stator operational temperature, determine, after consulting with the manufacturer, if additional pump cooling is required.

Heater - Complete receipt inspections required by the manufacturer and report finding by letter to Delta M̲.

- Design and fabricate heater/loop spacer/retainer clip.

- Design and fabricate clamp to adjust the depth of the heater in the TTT.

- Complete material selection and final design on the boron impregnated ceramic filler block and boral plate and machine to accommodate heater and fusible link.

- Make-up connectors between the heater leads and Fusible link and power coax cables.

Fusible Link - Conduct experiment to determine the temperature at which the fusible link will open circuit under load. Report results to Delta $\underline{M}$.

- design and fabricate heater/link/loop spacer/retainer clip.

TTT - Perform a vacuum-helium leak check of TTT.

- Determine correct lead height in TTT, install heater, fusible link, lead bath thermocouples, and zircaloy-4 tubing, and melt/fill the TTT with the correct quantity of lead.

Flow Meter - Perform receipt inspection of the flow meter as required by the manufacturer.

- Determine location of flow meter as required to achieve proper inlet flow conditions.

Loop - Assemble loop. Install heater power coax cable (bend coax with tube bender as required). Install redundant thermocouples on inlet plenum, S/G Section, Pump Section. Install thermocouple terminal connector board in Pump Section. Load copper shot, and thermal insulation in S/G Section. Lag primary loop tubing in the Pump Section with fiberglass insulation.

- Complete final construction design and fabricate the inlet plenum

Out-of-Core Test Tank - Complete construction of the test tank and necessary cooling systems in preparation for cold test operations.

Handling Equipment - Determine the NRC/MIT requirements for the certification of Thimble handling equipment in the vicinity of the MITR II core.

Miscellaneous Items - Calibrate all test equipment and gauges which will be used in the loop or in support of

loop testing. Determine calibration periodicity for equipment not specified by MIT QA procedures. Complete material and quality assurance certifications on all PCCL components and equipment. Set relief valves and proof test.

4.3 Loop Cold Operational Test Plan

- Conduct Grid plate clearance checks, and document in QA file.

- Check for vibration of the loop installed in the MITR core. Adjust the loop load support springs as required.

- Conduct Main Circulating Pump vibration checks. Determine if pump requires a sound isolated sound mount.

- Determine and record the stator temperature as a function of operating speed and loop temperature. Determine alarm set point of the stator high temperature alarm with the assistance of the manufacturer.

- Conduct loop reactivity tests with and without water in the loop, and with the maximum allowed boron concentration expected.

- Pressurize the loop to normal operating pressure and establish the correct surge volume in the pulsation dampener

to provide the least pressure ripple. Determine the optimum pressure high-to-low variation and establish limits for alarm functions in the data-logger.

- Determine minimum allowable pressure for Main Circulating Pump as a function of temperature and hydrogen concentration. Consider drawing a cold, pressurized water sample and determine the total gas concentration of the coolant (H_2 & N_2).

- Verify the proper operation and clearance of in core experiments and equipment with the Thimble installed.

- Determine the hydrogen inventory needed on the reactor top to maintain the desired loop H_2 concentration.

- Review the results of Cold Operations Testing and submit standard operating procedures for approval in preparation for hot operations.

- Conduct loop disassembly rehearsal, and resolve discrepancies identified.

- Install off-design diameter S/G tubing to determine, for future use in modeling various fluid parameters, the maximum fluid velocity attainable from the canned rotor pump. Test will determine if pump will achieve fluid velocities which
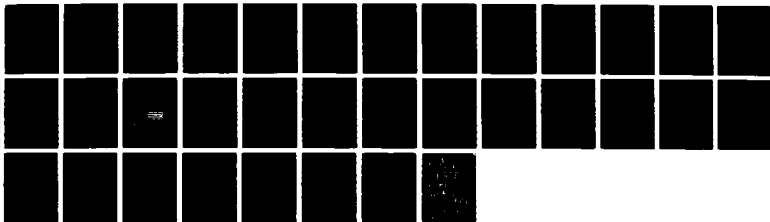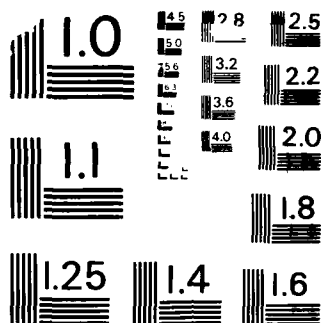
MICROCOPY RESOLUTION TEST CHART
NATIONAL BUREAU OF STANDARDS-1963-

will more acceptably model other important scaling parameters, i.e. boundary layer thickness, in future experiments.

4.4 Loop Hot Operational Test Plan

- Determine shot bed average temperature while loop is in preconditioning tank. Also, see section 5.3 of this report for thermal-hydraulic tests required to verify design performance.

-    Determine Thimble cooling/coolant differential temperatures.

- Determine final radial spacing of S/G tubing to achieve desired S/G temperature under reference full power conditions.

-    Determine pre-conditioning and in-core experiment operating parameters and normal operating ranges.

- Repeat Main Circulating Pump stator temperature test performed during cold operation tests, and establish normal operating limits of the pump stator and Pump Section ambient temperatures.

- Verify that the SCR heater controller, and pump controller do not interfere with MITR nuclear instruments.

- Determine maximum flow rate and heater power attainable from the pump and heater, respectively.

- Determine heat load on pre-conditioning tank cooling system in order to establish the operating conditions of the loop in pre-conditioning, and establish the loop operating power level required during weekend in-pile experiments.

4.5 Chapter Summary

This chapter summarizes some of the tests which must be performed to certify the PCCL equipment, gain operational experience, and establish the limits of normal operations and the capabilities of the loop. The results of these tests will determine areas where the design could be improved, and will establish reference operating conditions which will be used during the first in-core experiments. Additional tests may be specified by either the MIT reactor staff to familiarize the reactor operators with the loop or the MIT Reactor Safeguards Committee to certify the acceptability of the loop as an in-pile experiment.

Chapter 5

Summary, Conclusions and Recommendations for Future Work

5.1 Introduction

The work summarized in this chapter deals with an evolving design which has matured significantly during the fifteen month period covered here. The focus is on the significant efforts which have been made to anticipate and avoid design alternatives which adversely affect safety, and detract from the simplicity of the basic loop design. This summary chapter concludes with recommendations on a limited number of aspects which need to be reexamined, and a summary of work remaining to be completed prior to the in-pile operation of the PCCL.

5.2 Summary and Conclusions

The initial design goal of the PCCL project was to design a safe, inexpensive research tool to study simulated PWR coolant chemistry. Furthermore, the design had to facilitate analysis of the results. This has been achieved, in large part, by combining state-of-the-art technology for a high power density heater from LMFBR fuel rod research, and a canned rotor pump from aerospace technology to create a loop simulating a PWR unit flow cell (one steam generator tube coupled to one intra-fuel-pin coolant channel) at approximately 1/3 scale, which is capable of duplicating a

host of parameters. Figure 3.1 is a schematic of the overall design concept. Table 5.1 presents a comparison of a representative Westinghouse PWR and the MIT-PCCL design parameters. As can be seen, the parameters considered most important to the CRUD transport process - temperatures in particular - are either exactly or closely reproduced.

5.3 Recommendations for Future Work

The following comments address areas of the Loop design and operations which, at this writing, deserve further attention.

1. Evaluation of aluminum shot as a possible heat transfer medium. This will provide more flexibility, reduce the thermal mass of the S/G section, and reduce the weight of the Thimble.

2. Based on the results of Hot Operational Tests, evaluate the addition of cooling fins to the S/G section of the Thimble. Also consider adding fins to the Pump section if the ambient temperature cannot be maintained low enough to cool the Main Circulating Pump.

3. Devise weekend operating schemes which will limit the impact of the PCCL on the MITR decay heat removal system and still allow the Loop to operate at a power level sufficient to remain within the similitude operating envelope.

Table 5.1 Comparison of PWR and MITR-PCCL Design Parameters

Parameter	Typical Westinghouse PWR 4 Loop – 3400 MW	MIT-PCCL
(1) Closely Matched Parameters:		
Pressure (PSIA/MPa)	2240/15.44	2250/15.51
T_{high} (deg F/deg C)	615/324	615/324
T_{low} (deg F/deg C)	550/288	550/288
Temperature Rise in Core (deg F/deg C)	65/18.3	65/18.3
Density (Lbm/Ft^3/Kg/M^3)	44.88/718.9	44.88/718
Core Film Differential Temperature (deg F/deg C)	36/2.22	36/2.22
Thermal Neutron Flux (n_{th}/cm^2-sec)	2 X 10^{13}	2 X 10^{13}
Core Surface Heat Flux, q" (KW/M^2)	598.5	758$
Steam Generator Surface Heat Flux, q" (KW/M^2)	178.8	154.66$
Core Average Fluid Velocity (Ft/sec / M/sec)	15.7/4.77	12.1/3.68$
(2) Other Parameter Comparisons:		
Core Reynold's Number	5.13 X 10^5	2.1 x 10^5$
Steam Generator Reynold's Number	9.09 X 10^5	1.65 x 10^5$
Core Length/Diameter Ratio (17 x 17)	401.1	198

	(15 x 15)	
	355.5	
Steam Generator Fluid Velocity (M/sec)	5.87	2.45$
Steam Generator/Core Surface Area Ratio	3.11	4.91
Core Nusselt Number*	864.2	423
Steam Generator Nusselt Number	1,365.8	357.2
Core Length/Hydraulic Diameter Ratio (17 x 17)	310.9	192.3
(15 x 15)	269.7	
Steam Generator Length/Tube Diameter Ratio	943.8	626.9
Core Linear Heat Generation Rate, q'(KW/M)	18.8	15.8$
Steam Generator LHGR**, q' (KW/M)	11.0	6.2$
Fast Neutron Flux (n_f/cm^2-sec)(>0.1 Mev)	2×10^{14}	1×10^{14}
Core Hydraulic Diameter (in/mm)	0.464/11.8 (17 X 17)	0.26/6.6
	0.534/13.6 (15 X 15)	
Steam Generator Tube ID (in/mm)	0.775/19.7	0.319/8.1
Loop Transit Time sec	13	3.58$

Nu = QD_e/A(delta T)*k
** LHGR = Linear Heat Generation Rate
$ Data based on 20 KW, and 2 gpm

4.   Continued work on the modeling is needed, not only to further the design of the Loop, but also to correlate the data from the loop to what the present computer models are predicting, and thereby create a better computer model.

5.   Reevaluate the size (volume and surface area) of the loop's stainless steel Plenum.   It is, at present, not entirely clear as to what the effective stainless steel surface area is in a full scale PWR, or how best to go about matching it.

6. Design, build and test the in-core handling system. The equipment must be robust enough to insure personnel safety, must permit ease of operation, and maintain structural rigidity of the Thimble in all orientations.

7.   Evaluate the out-of-pile let down flow control system. The presently proposed thermal bath control scheme may be subject to flow control and pressure oscillation problems, especially during long unattended operating periods.

8. The PCCL project staff should develop a training course and establish a level of proficiency required by the MITR operators to conduct normal operation with the PCCL installed in the MITR-II core.   The staff should also eventually develop a training course and qualify PCCL

operators to prepare and handle the loop during preconditioning and normal operations.

9. The staff, with the involvement of Health Physics, needs to develop procedures to handle the loop during disassembly and post mortem analysis. These procedures should look closely at ways to eliminate the unnecessary generation of radioactive waste from the operation of the loop, and methods to reduce personnel exposures.

10. Consider the use of a gamma thermometer in the Heated Section of a future Thimble to measure the actual gamma flux.

11. Prepare loop assembly and disassembly procedures, with quality assurance checks by PCCL operators and MITR Operations staff quality assurance inspectors.

12. Recommend the project install a small generator or battery pack/static inverter to back-up the PCCL in the event of a loss of site power. The Main Circulating Pump and the out-of-pile charging pump should be supplied from the emergency generator.

13. The following heat transfer studies must be completed during the Summer of 1987:

* Measure the radiation-only heat loss at high temperature to confirm safe rejection of estimated gamma heating.

* Measure the effective shot bed k and adjust inconel S/G tube spacing to achieve the desired operating temperatures under reference full power conditions.

* Measure shot bed k under helium, nitrogen, and carbon dioxide filled, and evacuated conditions, for verification of analytical models.

* Measure the heat rejection capability of the Thimble to water in the out-of-pile tank; test the efficiency of a helically wound wire wrap to increase heat transfer (or perhaps a coarse wire mesh wrap).

* Conduct an experiment to evaluate the effectiveness of cooling fins if bare or mesh wrapped transfer is inadequate.

* Upgrade analytical and computer models of the shot bed conduction problem and apply them to evaluate experimental results

* Measure the shot bed mean temperature.

* If radiation heat transfer is inadequate, investigate blackening of the Thimble and TTT surfaces.

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Appendix A

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
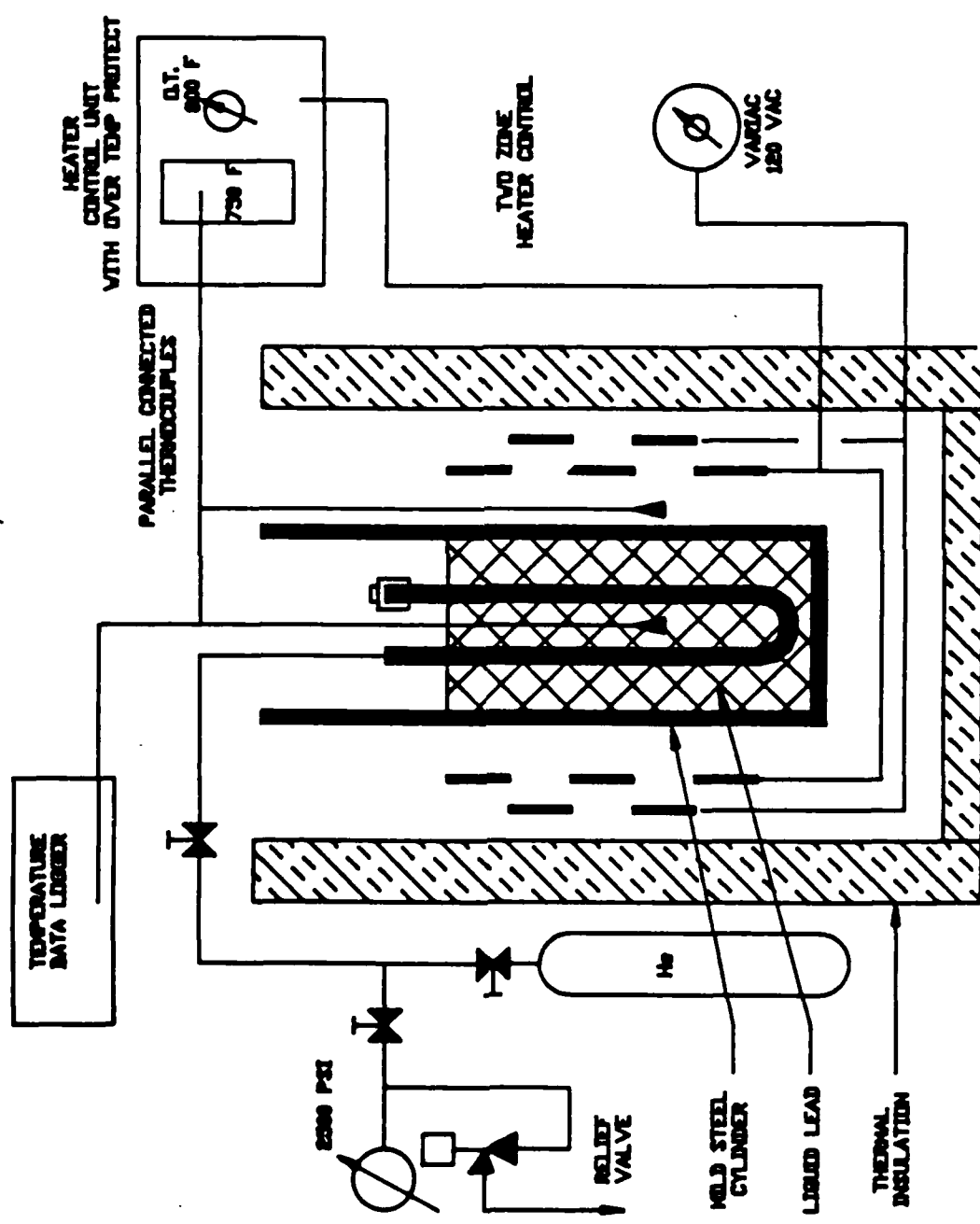
Figure A.1 Liquid Lead Compatibility Experiment

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.

Appendix B

PWR to MIT-PCCL Scaling

In developing the analytical model to scale the principal parameters for the MIT-PCCL, the decision was made to attempt to establish similitude based, initially, on core and steam generator fluid film differential temperatures. Not all parameters of interest can be matched exactly, but as will be shown below the flexibility of the MIT-PCCL has allowed the near duplication of full scale PWR conditions; especially those conditions which model the corrosion product transport process. For a more detailed analysis of the analytical model of the PCCL, refer to Morillon's thesis (M-1).

For the PWR model the convective heat transfer coefficient, h, is generally correlated from the Dittus-Boelter equation (given by McAdams) (R-1) in terms of the Nusselt Number:

$$Nu = hD_e/k = 0.023 \ (R_e)^{0.8}(Pr)^{0.4}$$

or

$$h = 0.023 \ (R_e)^{0.8}(Pr)^{0.4} \ k/D_e \qquad \text{B-1}$$

where D_e is the Hydraulic Diameter

When correlating the PWR core to the Heated Section of the PCCL, the PWR hydraulic diameter is correlated to the inside diameter of the PCCL zircaloy-4 tubing. The use of the hydraulic diameter reduces the analysis of the PWR core to a

single flow channel consisting of four tubes in a square
lattice (or the equivalent flow cell consisting of a single
fuel rod centered on a square of area [rod pitch]^2). The
PWR unit flow cell hydraulic diameter for representative 17
x 17 and 15 x 15 assembly lattices is as follows:

$$PWR\ D_{e(core)}\ =\ 0.464\ inches,\ 11.77\ mm\quad(17\ x\ 17)$$
$$=\ 0.534\ inches,\ 13.57\ mm\quad(15\ x\ 15)$$

The PCCL core hydraulic diameter = Zircaloy-4

inside diameter: $D_{e(core)}$= 0.26 inches, 6.6 mm ID

define: subscript L = PCCL parameters, and

P = PWR parameters.

Correlation of the PWR and PCCL Steam Generators is based on
the inside diameter of the steam generator tubes.

$$PWR\ D_{eP(S/G)}\ \ =\ 0.775\ inches,\ 19.7\ mm\ ID$$
$$PCCL\ D_{eL(S/G)}\ =\ 0.319\ inches, 8.1\ mm\ ID$$

Since all fluid temperatures are matched, the density,
absolute viscosity, thermal conductivity, Prandtl Number,
and specific heat are the same for both the PWR and the PCCL
and cancel out of subsequent analysis.

The heat transfer film differential temperature is given by:

$$\text{delta } T_{film} = q''/h \qquad\qquad \text{B-2}$$

Combining equations B-1 and B-2, and ratioing the fluid film differential temperature for the PWR to that of the PCCL, yields a relationship for the differential film temperature as a function of the core surface heat flux, hydraulic diameter, and core average fluid velocity:

$$dT_{f(pccl)}/dT_{f(pwr)} = (q''_{pccl}/q''_{pwr}) \times (h_{pwr}/h_{pccl}) \quad \text{B-3}$$

Substituting for h (equation B-1), B-3 becomes:

$$dT_{fL}/dT_{fP} = (q''_L/q''_P)(D_L/D_{eP})^{0.2}(V_P/V_L)^{0.8} \qquad \text{B-4}$$

This equation is valid for the analysis of both core and steam generator film differential temperatures.

$$q''_{P(core)} = (MW_{th})/nL(PI)D_{clad} \qquad\qquad \text{B-5}$$

where: n = number of fuel rods in core

(17 x 17) n = 50,952   (15 x 15) n = 39,372

L = core length = 3.66 M

D_{clad} = outside diameter of fuel cladding

= 1.07 cm (15 x 15),   0.95 cm (17 x 17)

MW_{th} = 3400 MW

therefore:

$$q''_{P(core)} = 190,000 \text{ BTU/hr ft}^2 = 598.98 \text{ KW/M}^2 \quad \text{(U-1)}$$

$$q''_{P(S/G)} = 56{,}700 \text{ BTU/hr ft}^2 = 178 \text{ KW/M}^2 \quad (U-1)$$

$q''_L = $ heater power $Q_L/A_{surface\ area}$

$A_{surface\ area} = PI\ D_P\ L_{(core)}$

$L_{L(core)} = 50$ inches/127 cm

$L_{L(S/G)} = 200$ inches/508 cm

$$q''_{L(core)} = 3.53\ Q_L \text{ KW/ft}^2 = 37.95\ Q_L \text{ KW/M}^2 \qquad B-6$$

$$q''_{L(S/G)} = 0.72\ Q_L \text{ KW/ft}^2 = 7.73\ Q_L \text{ KW/M}^2 \qquad B-7$$

The PCCL velocity is a function of volumetric flow rate G(gallons/minute).

$$G \text{ (gpm)} = V(m/s)\ [\text{gals}/3.785 \times 10^{-3}\ M^3]\ A_{xsection}\ 60 \text{ sec/min}$$

$A_{xsectionL(core)} = 0.053 \text{ in}^2/0.343 \text{cm}^2$

$A_{xsectionL(S/G)} = 0.08 \text{ in}^2/0.52 \text{ cm}^2$

$A_{xsectionP(core)} = 53 \text{ ft}^2/4.93 \text{ M}^2$

$A_{xsectionP(S/G)} = 0.472 \text{in}^2/3.04 \text{ cm}^2$

$V_{L(core)}(M/s) = 1.854\ G\text{(gpm)}$

$V_{L(S/G)}(M/s) = 1.223\ G\text{(gpm)}$

$V_{P(core)} = $ core average fluid velocity $= 15.7$ ft/sec $\quad (U-1)$

$$= 4.77 \text{ M/sec}$$

$V_{P(S/G)} = $ S/G average fluid velocity $= 19.3$ ft/sec $(U-1)$

$$= 5.87 \text{ M/sec}$$

Substituting the above parameters into equation B-4 for both the core and steam generator relationships, results in relationships for the PCCL total power, in watts, as a function of PCCL volumetric flow rate in gpm:

$$G_{L(core)} (gpm) = [Q_L]^{1.25} \, 1.266 \times 10^{-5} \quad (Q \text{ in watts}) \qquad B-8$$
$$G_{L(S/G)} (gpm) = [Q_L]^{1.25} \, 1.351 \times 10^{-5} \quad (Q \text{ in watts}) \qquad B-9$$

Next, we wish to investigate the result of matching the bulk fluid temperature rise across the core $(T_{hot} - T_{cold})$. The total heat added across the core is given by:

$$q = MC_p (T_{hot} - T_{cold}) \qquad\qquad B-10$$

For $(T_{hot} - T_{cold})_P = (T_{hot} - T_{cold})_L$:

$$M_{pccl} \, q_{pwr} = M_{pwr} \, q_{pccl} \qquad\qquad B-11$$

$$M = A_{xsection} \, V \text{ (average fluid density)}$$
$$M_P = 127 \times 10^6 \text{ lbm/hr} = 279 \times 10^6 \text{ Kg/hr}$$

Substituting representative values for a 3400 MW_{th} PWR, and using parameters developed above, the following relationship is obtained:

$$G_L(gpm) = Q_L \; 1.0558 \times 10^{-4} \qquad (Q \text{ in watts}) \qquad \text{B-12}$$

Equations B-8, B-9 and B-12 are plotted in Figure B.1. The close agreement between the two proposed modeling arguments is evident. From Figure B.1, the Loop can be operated at low heater power and fluid flow where all three correlations agree closely, or operated at higher power and flow with only a small introduced error. In other words, Figure B.1 illustrates that the loop can be operated within a broad envelope of heater powers and fluid flow rates and still closely satisfy basic similitude goals.

The close agreement of the above relationships, and the resulting flexibility in the choice of operating conditions without introducing a large error, directed efforts to establish the operating point based on matching another criterion of similitude. Loop similitude based on equal wall shear stress (T) was considered appropriate since this parameter may control particulate transport processes. The wall shear stress in a tube is given by:

$$T = (dP/L) * D/4 \qquad \text{B-13}$$
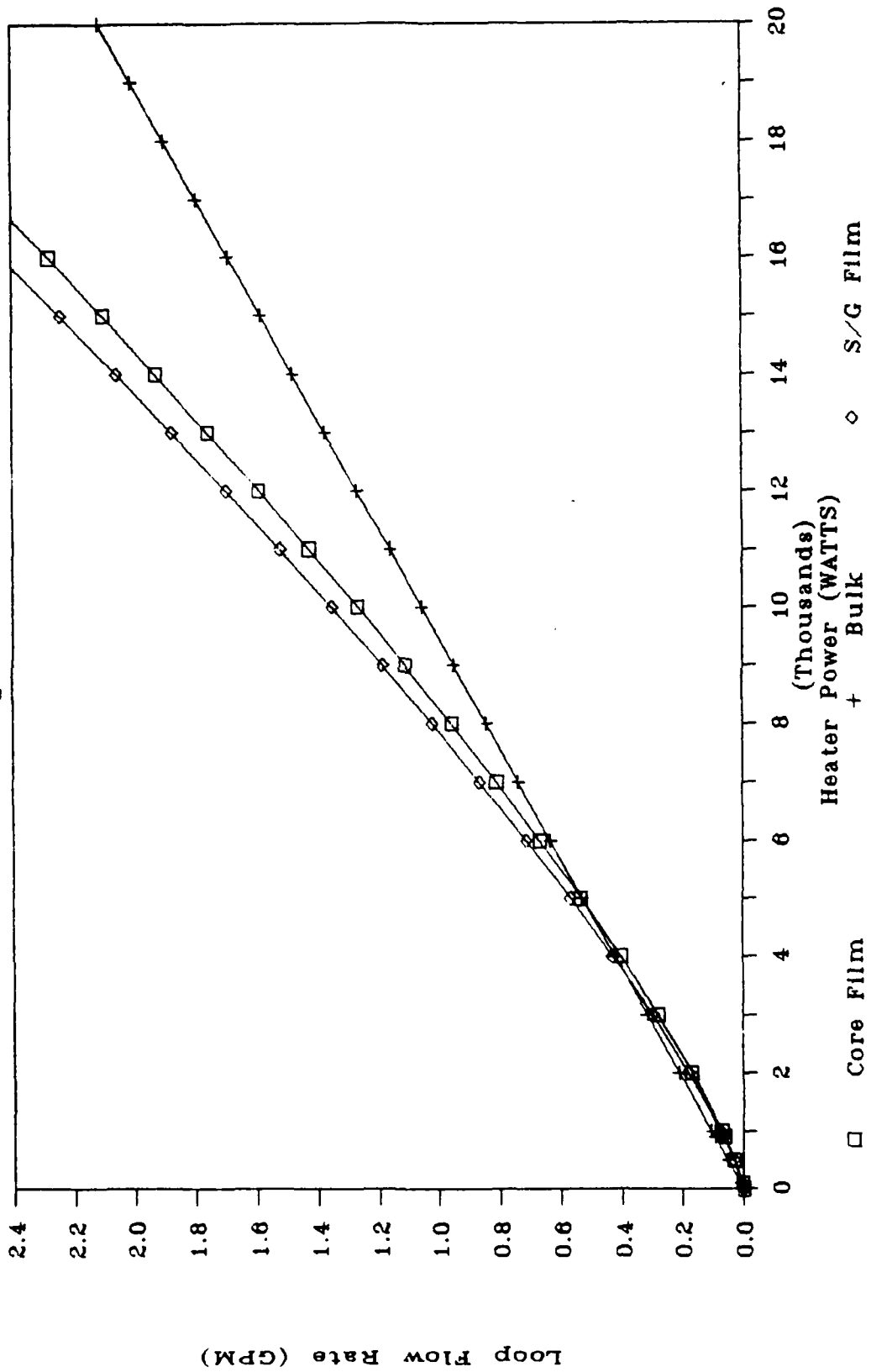$$dP = (f/D)(V^2/2)$$

thus: $\qquad T = fV^2 \qquad$ B-14

Since: $\qquad f = R_e^{-\frac{1}{4}} \quad$ for turbulent flow, substituting for the Reynold's Number we have:

$$f = (DV)^{-\frac{1}{4}}$$

PCCL Flow vs Power Operating Envelope

Figure B.1

Heater Power (WATTS)
(Thousands)

□ Core Film   + Bulk   ◇ S/G Film

Loop Flow Rate (GPM)

Therefore: $\quad T = V^{7/4}/D^{1/4}$ B-15

Matching T gives:

$$V_L/V_P = (D_L/D_P)^{1/7}$$ B-16

$D_L/D_{eP(core)}$ = 0.561 $\quad$ (17 x 17) core

$\qquad\qquad$ = 0.487 $\quad$ (15 x 15) core

$D_L/D_{eP(S/G)}$ = 0.412 $\quad$ Steam Generator

thus: $\quad V_L/V_{P(core)}$ = 0.921 $\quad$ (17 x 17) core

$\qquad\qquad\quad$ = 0.90 $\quad$ (15 x 15) core

$V_L/V_{P(S/G)}$ = 0.881 $\quad$ Steam Generator

This indicates that for similitude the loop heated section velocity should be in the range of 14.3 to 14.6 ft/sec and steam generator velocity: 16 ft/sec. The PCCL fluid velocity can be varied from 3 ft/sec (0.5 GPM) to 12 ft/sec (2 GPM) in the zircaloy tubing and 2 ft/sec (0.5 GPM) to 8.5 ft/sec (2 GPM) in the Inconel section. Hence, at 2 GPM, the loop has approximately 85% of the full scale PWR core fluid wall shear stress, while in the steam generator section the loop develops 53% of the wall shear stress. Therefore, to match film and bulk differential temperatures, and to approach full scale plant wall shear stress, the loop should be operated at the high end of the operating envelope. It should be noted that the actual "full pump power" flow rate

achievable in the as-built loop will not be truly known until hot test operations have been completed.

Appendix C

Estimated Radiation Levels and ALARA Considerations
in the Design of the MIT-PCCL

I. In-core Materials Inventory for the PCCL

Component	Material	Composition (WT.%)	Weight (gms)
U-tube	Zircaloy-4	1.45 Sn, 0.21 Fe, 0.1 Cr, balance Zr and impurities.	110
Heater	1018 Steel	0.2 C, 0.9 Mn, balance Fe and impurities.	85
	Inconel 600	76 Ni, 0.25 Cu, 8 Fe, 15.5 Cr, 0.25 Si, 0.5 Mn.	35
	MgO	60.3 Mg, 39.7 O_2	180
Conduction bath	Lead	99.9 Pb (5 ppm Bi)	4650
Containment Tube	Titanium	99.5 Ti, balance impurities.	240
Thimble	6061 Aluminum	0.8 Si, 0.7 Fe, 0.4 Cu, 0.15 Mn, 1.2 Mg, 0.35 Cr, 0.25 Zn, 0.15 Ti.	670

II. Estimated Unshielded Gamma Dose for PCCL In-Core

Components After 20 Full Power Days and 60 Hours' Decay

Component	Principal Activities	Gamma Dose Rate (unshielded) Rad/hr @ 1 M
U-Tube	Zr^{95}, Cr^{51}	1.5
Heater	Co^{58}, Co^{60}, Fe^{59}, Cr^{51}	0.3
Conduction Bath	Pb^{203}, Hg^{203}	4.0
Containment Tube	Sc^{46}	0.2
Thimble	Na^{24}, Co^{58}, Co^{60}, Fe^{59}, Cr^{51}	1.0
		total 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.

Appendix D

Preliminary Outline of the Safety Evaluation Report (SER) Supplement

NOTE: An addendum to the SER as issued in February 1987 (H-1) (conditionally approved by the MIT Reactor Safeguards Committee) will be submitted to the PCCL sub-committee for final approval in August 1987. The outline below is intended to provide guidance on the content and organization of this report.

I. Summary of significant design changes incorporated into the as-built loop, compared to the February 1987 SER design.

    a. Thimble

    b. Thimble internals

    c. Charging, pressurization and sampling systems

    d. Loop handling and disassembly equipment

II. Review of loop operations testing, including:

    * Thimble and loop proof testing

    * actual loop operating parameters (power, temperature, flow, pressure etc.) including Thimble external surface

temperatures for various flow
conditions

* radiative cooling measurements (LOCA
simulation at low power)

* reactivity testing (voided vs flooded)

III. Synopsis of loop standard and abnormal operating
procedures, including:

* loop installation and start-up

* loop removal and disassembly

* summary of automatic responses to abnormal
conditions

* summary of alarm conditions and required
experimenter and reactor operator actions

IV. Final review of safety-related experiments and
calculations, including:

* activation data for actual loop materials

* liquid lead compatibility experiments

* lead bath can (TTT) leak/break simulations

V. Overall summary and evaluation of PCCL safety.

END

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