SUPPLEMENT TO THE

SAFETY EVALUATION REPORT

FOR

THE PWR COOLANT CHEMISTRY LOOP (PCCL)

MITNRL-020

to be

installed and operated in the

MITR

April 19, 1988

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For Review by the
MITR Safeguards Committee
and by its
PCCL Subcommittee

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1. INTRODUCTION

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

2. SUMMARY OF DESIGN CHANGES

a) Changes to Loop and Support Systems Design

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

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

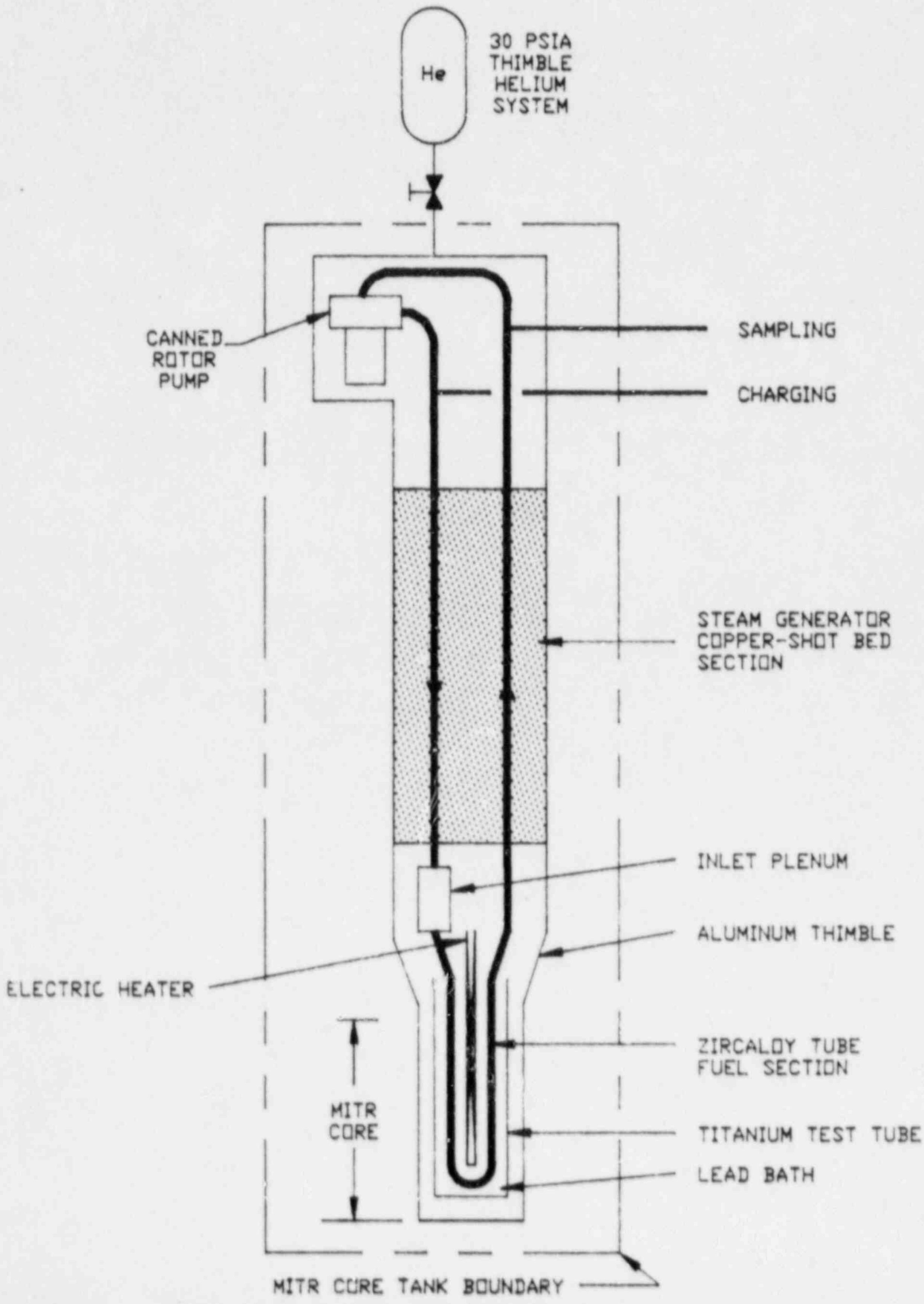
--a flowmeter will not be used; flow will be inferred from power and temperature data calibrated by out-of-pile testing (under normal circumstances, flow is directly related to the circulating pump input frequency, which is controlled),

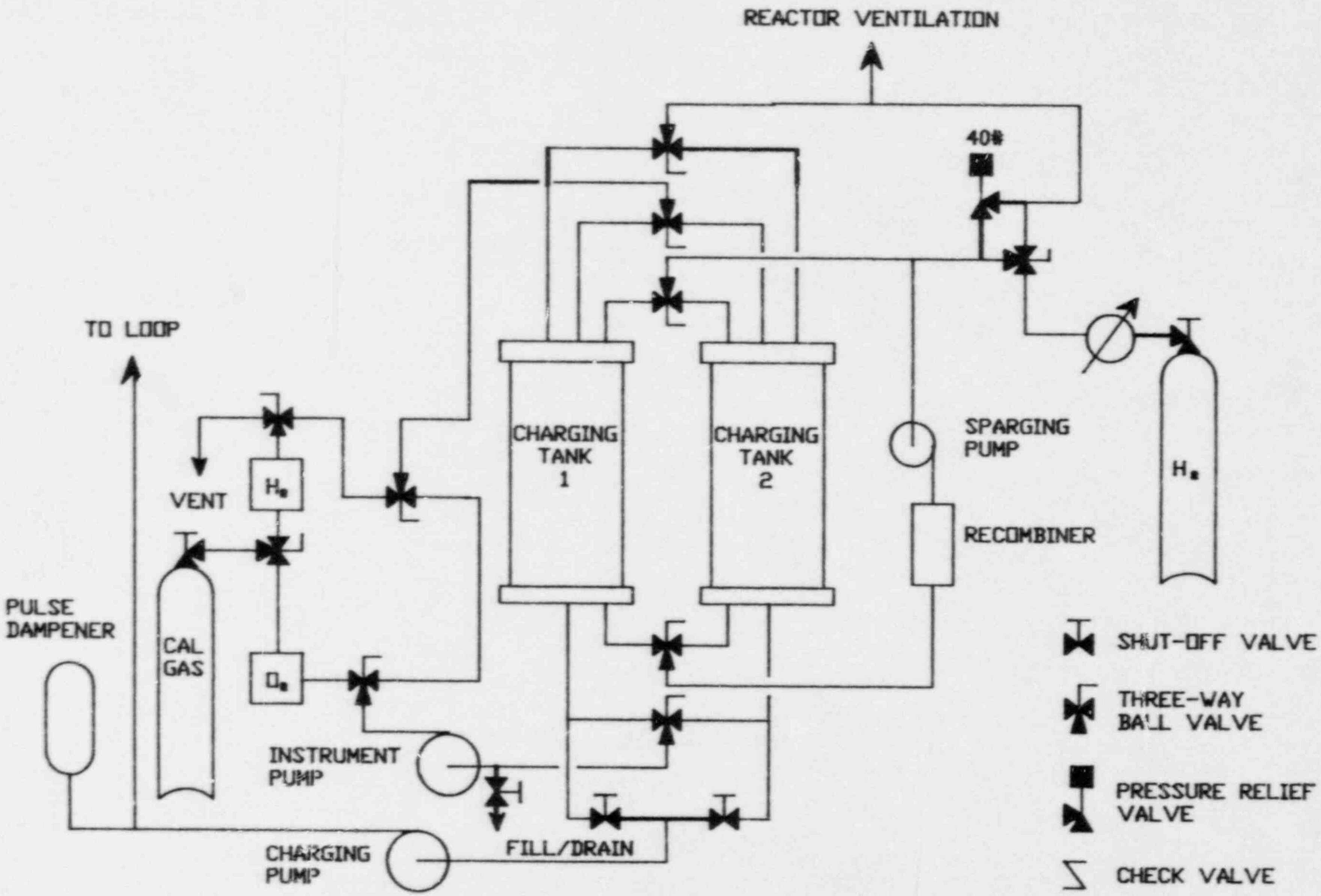
--the titanium lead bath container has been strengthened with weld beads to prevent deformation during operation at temperature,

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

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

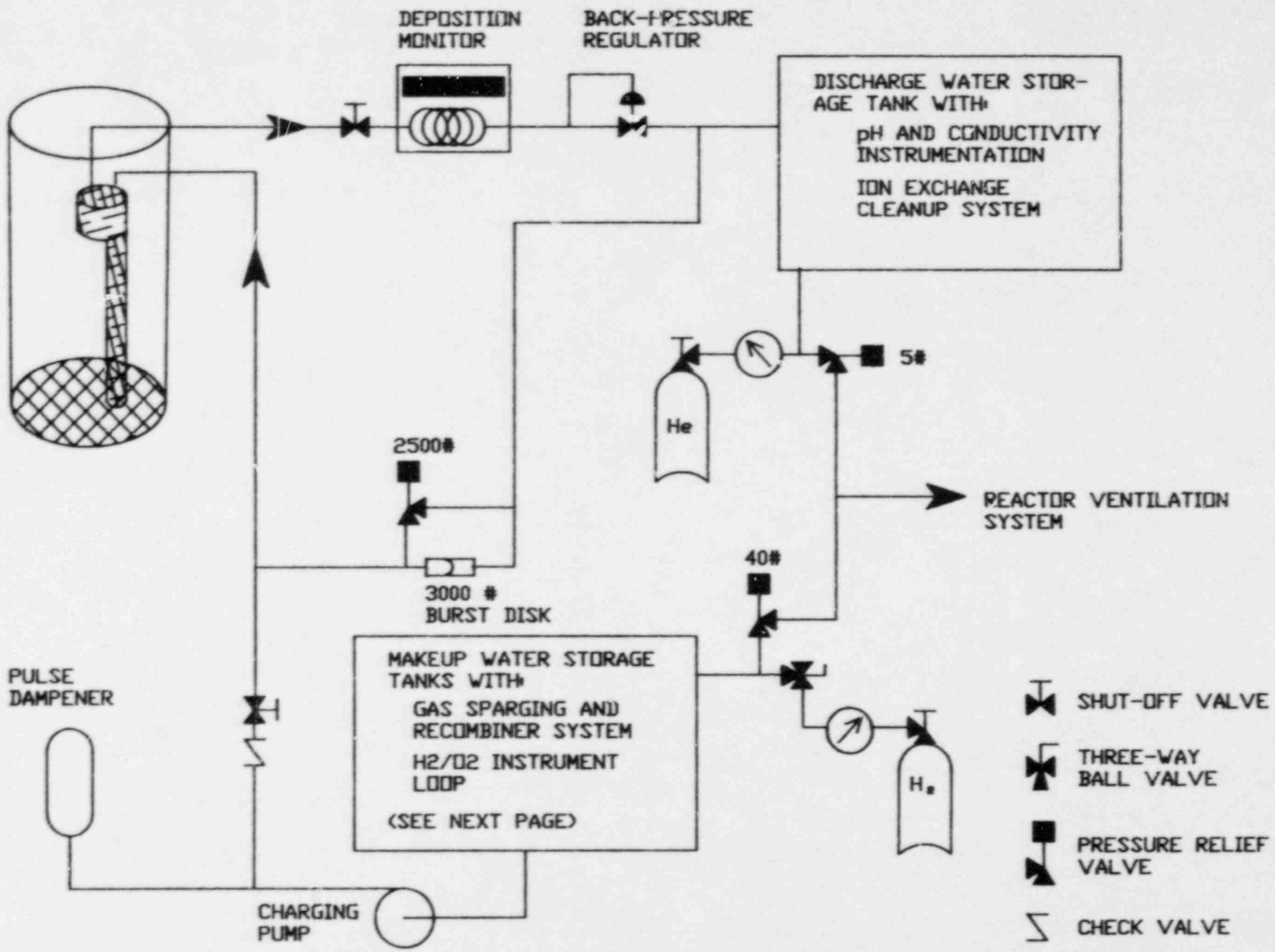
Figure 1.1a Schematic of PWR Coolant Chemistry Loop





CHARGING AND PRESSURIZATION SYSTEM

FIGURE 1.1.b-1



LOOP PRESSURIZATION AND CHEMISTRY CONTROL SYSTEM

FIGURE 1.1.b.-2

TABLE 2.1: PWR COOLANT CHEMISTRY LOOP DESIGN SPECIFICATIONS

IN-CORE SYSTEMS

Pump:

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

Heater:

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

Thimble:

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

(Simulated Fuel Pin)

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

Lead Bath Container:

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

Simulated Steam Generator Tube:

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

Out-of-Core System:

Charging/Pressurization Pump:

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

NOT TO SCALE

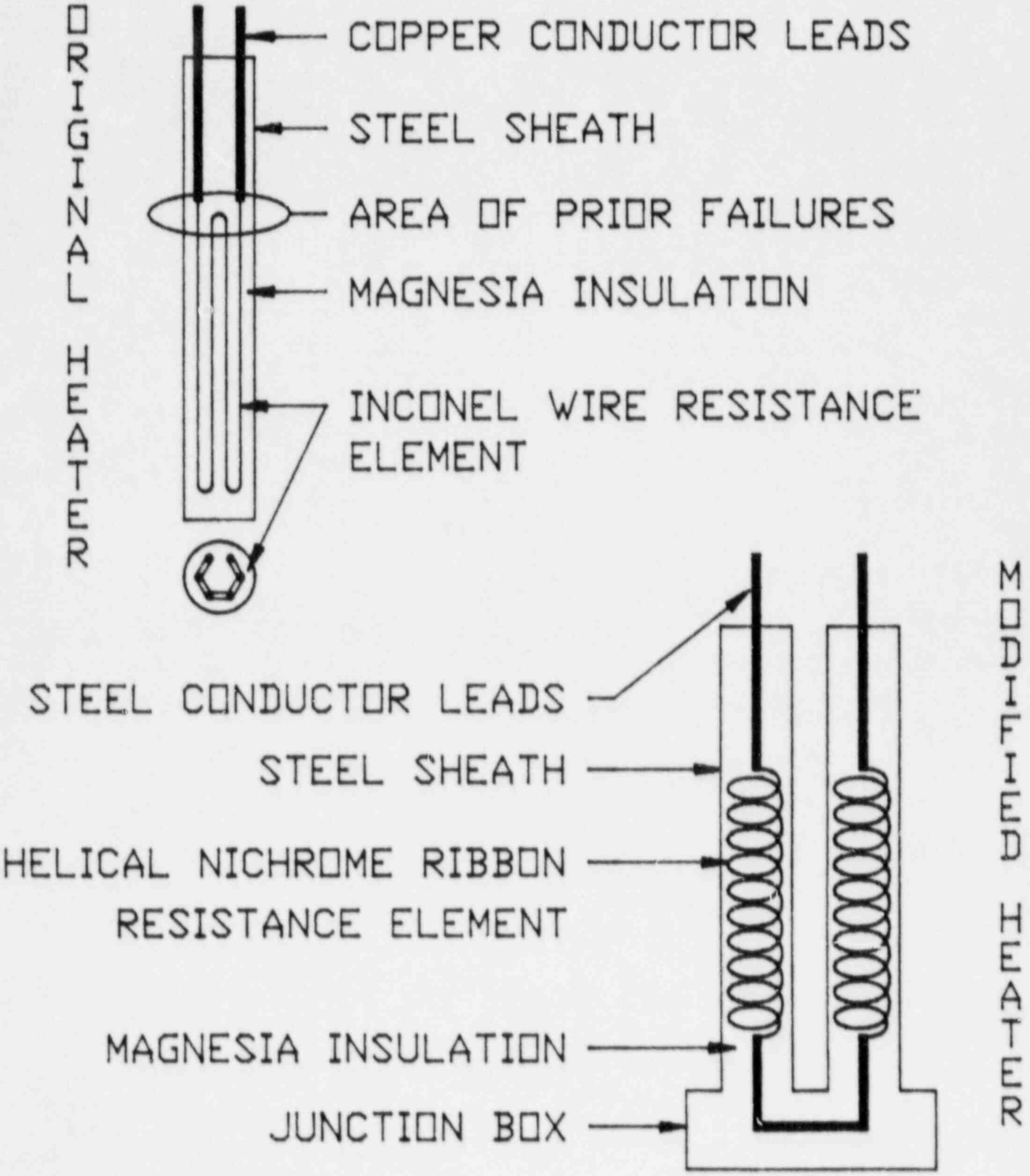


FIG. S1. SUMMARY OF HEATER MODIFICATIONS

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

For the charging and discharge systems, significant changes include:

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

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

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

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

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

b) Changes to Instrumentation and Control Design

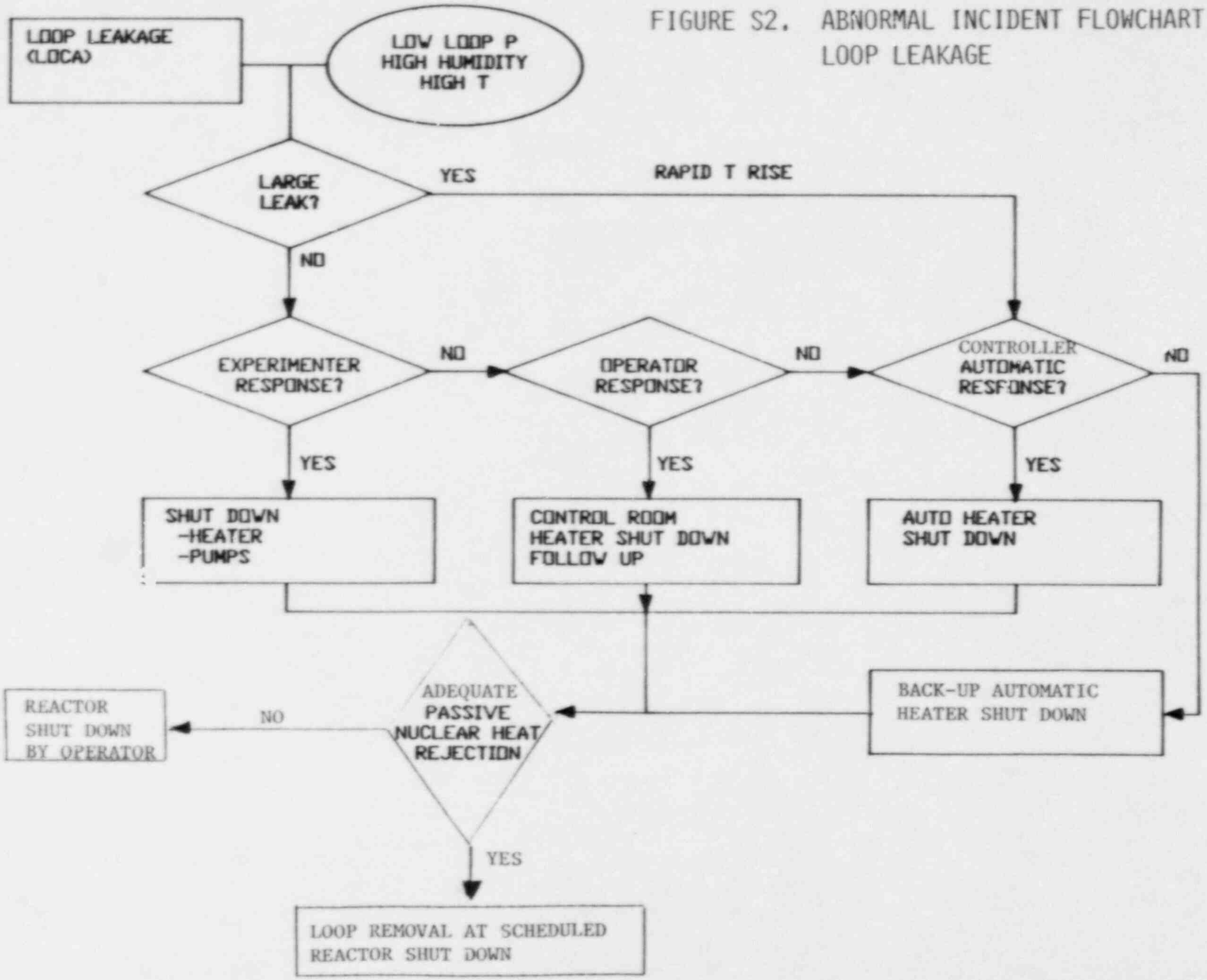
The basic philosophy of loop control and accident response remains the same as that outlined in the SER. It has been shown that all serious loop accidents eventually lead to overheating of the lead bath, and that consequences to the reactor are avoided if the electric heater is shut off. Therefore, the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

3. DETAILED DISCUSSION OF IMPORTANT DESIGN CHANGES

a) Over-temperature Protection and Heater Design

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

FIGURE S2. ABNORMAL INCIDENT FLOWCHART - LOOP LEAKAGE



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

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

b) Loop Pressure and Flow Regulation

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

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

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

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

a) Reactivity and Power Peaking due to the PCCL

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

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

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

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

b) Nuclear Heating of In-Core Components

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

5. SAFETY EXPERIMENT RESULTS AND OPERATIONAL EXPERIENCE

a) Heater Operating Experience and Passive Rejection of Nuclear Heating

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

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

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

Table 5.1: ZIRCALOY AND HEATER TEMPERATURES UNDER LOCA CONDITIONS

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

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

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

b) Experience Relevant to Loop Electrical Safety

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

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

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

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

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

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

c) Molten Lead Compatibility and Lead Bath Leak Testing

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

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

6. REQUIRED SAFETY PARAMETERS AND LIMITS

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

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

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

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

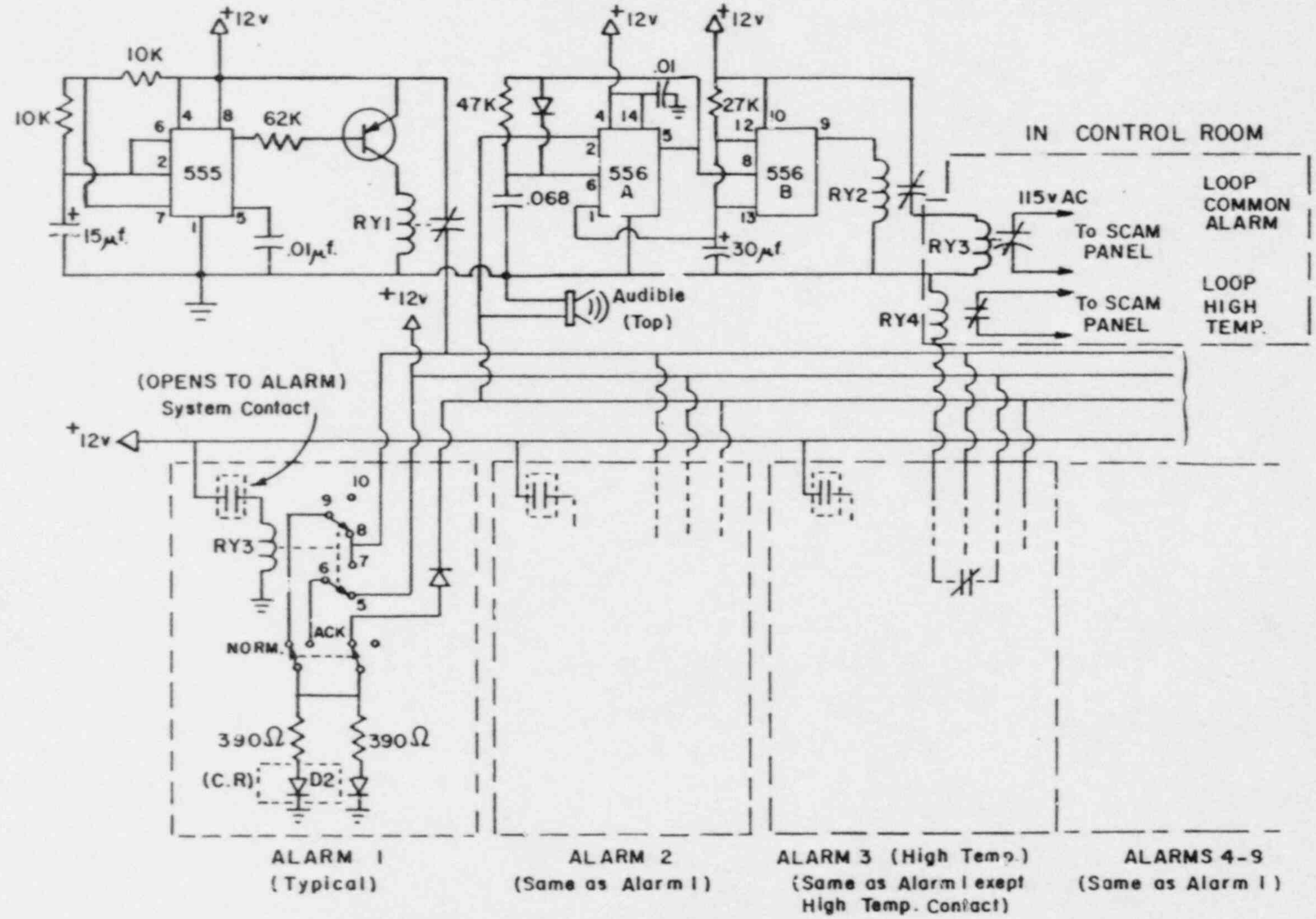
APPENDICES

Appendix 1 - PCCL Alarm Circuit

Appendix 2 - PCCL Reactivity Measurement Results

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

APPENDIX 1



PCCL ALARM CIRCUIT

APPENDIX 2

NUCLEAR REACTOR LABORATORY

AN INTERDEPARTMENTAL CENTER OF
MASSACHUSETTS INSTITUTE OF TECHNOLOGY

O.K. HARLING
Director

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

L. CLARK, JR.
Director of Reactor Operations

January 28, 1988

Memorandum

To: Distribution

From: K. Kwok

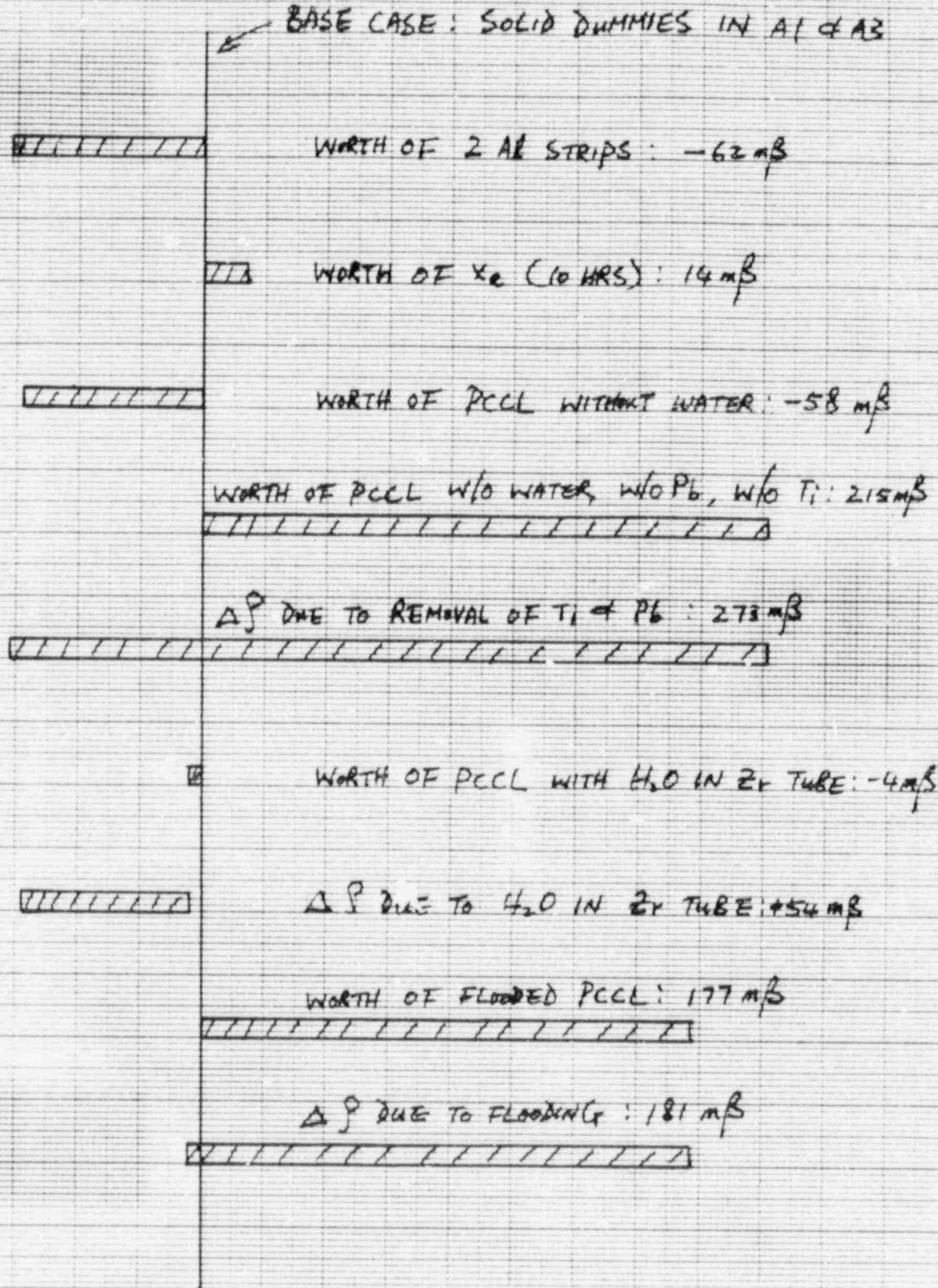
Subj: PCCL Reactivity Measurement Results - Revision 1

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

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

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

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



PCCL REACTIVITY MEASUREMENT RESULTS 1/19/88

Appendix 3

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

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

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

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

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

LIQUID LEAD COMPATIBILITY EXPERIMENT

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

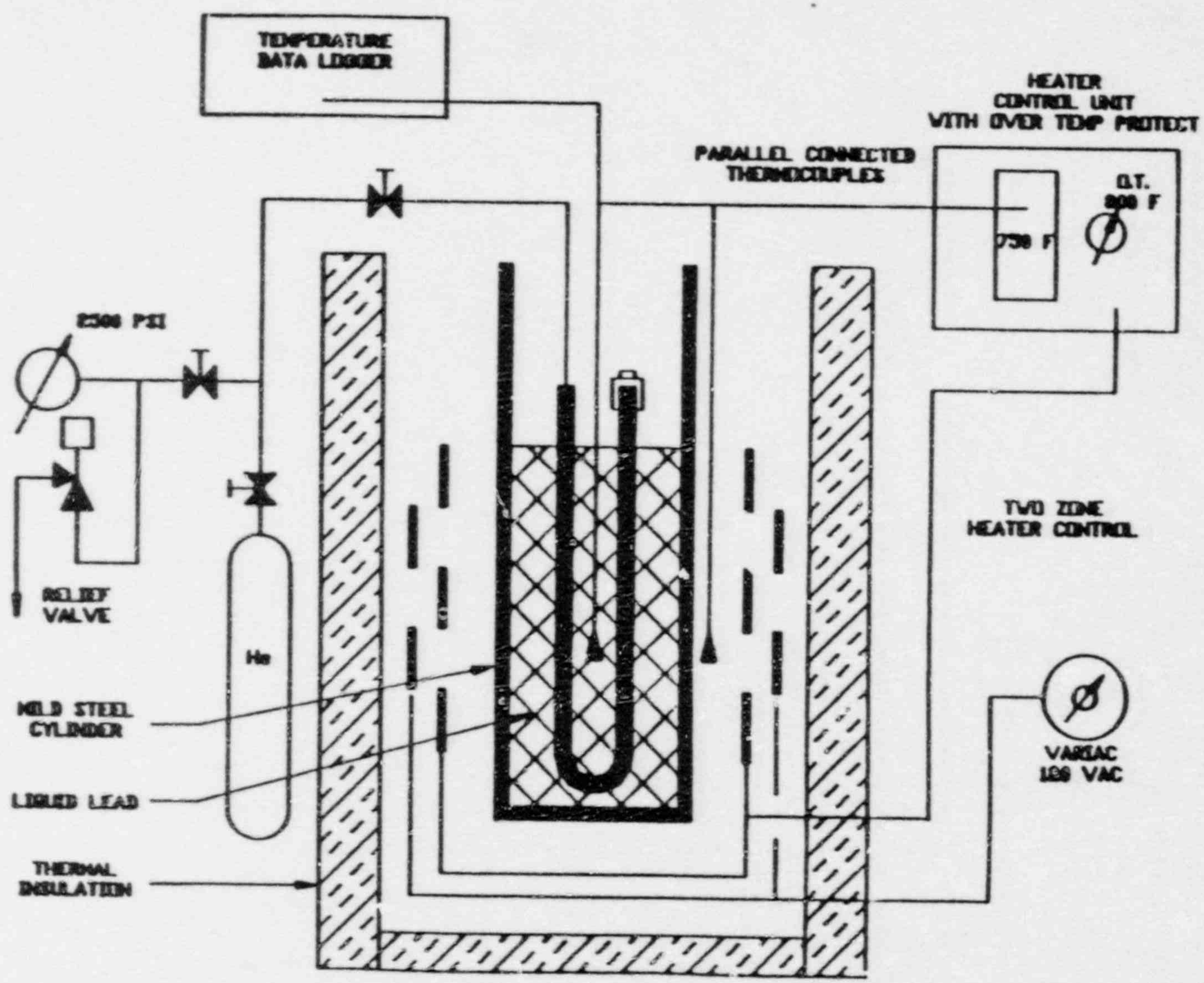


Figure A.1 Liquid Lead Compatibility Experiment tubing would be exposed, the tubes were bent into a "U" tube and internally pressurized to 2500 PSIG with helium gas. A temperature of 750 degrees F (398.89 C) was selected on the basis that it is very close to the actual maximum expected temperature of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

<u>Experimental Results:</u>

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

Table A.1

ANALYSIS OF LEAD PURITY

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

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

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

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

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

Conclusions:

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

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

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

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

Item: PWR Coolant Chemistry Loop (PCCL)

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

Q/A number if required M-86-2

Does the item change or contradict the

Technical Specifications? ______Yes* X No

SAR? ______Yes* X No

* Attach explanation

Description of Change (Attach extra pages if necessary):

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

Safety Evaluation (Attach extra pages if necessary):

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

Summary of Review:

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

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

RRPO F. McWilliams Date 20 April 88

Recommend Approval ✓ Yes ______ No

Reviewer [signature] Date 4/20/88

Reviewer [signature] Date 20 April 88

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

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

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

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April 19, 1988

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

1. Description of Change

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

2. Safety Evaluation

The SER addresses the following topics:

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

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

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

1. Pressure limits:

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

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

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

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

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

3. <u>Unreviewed Safety Question Determination</u>

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

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

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

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

3.1 Reactivity Effects

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

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

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

1.23 Secured Experiment

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

1.24 Movable Experiment

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

1.25 Non-Secured Experiments

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

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

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

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

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

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

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

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

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

3.2 Pressure Effects

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

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

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

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

3.3 Temperature Effects

The in-pile loop asssembly will be heated both by a 0 to 20 kW heater and by a combination of gamma and fast neutron radiation. The normal combined heat load will be less than 20 kW. The radiation heating is estimated in the SER Supplement to be 9.6 kW at a reactor power of 5 MW, so that 29.6 kW would be the maximum heat load potentially available under malfunction conditions. This is not much more than the hottest running fuel plate in the MIT Reactor, and most of the heat will be dumped to the reactor primary coolant via the shot bed in the steam generator section above the core. Hence, the thimble is easily cooled by the flow of primary coolant through the 0.050 inch thick channel between the thimble and the dummy fuel element that surrounds it.

The SER addresses the potential for a Zircaloy-water reaction and shows that cooling by conduction and radiation will prevent temperatures in the thimble from exceeding 1845°F for the maximum radiation heating, which is estimated not to exceed 9.6 kW. This is based on very conservative extrapolation of temperatures measured out of core in a test mock-up of the loop assembly under LOCA conditions at heater powers in the range of 2470-4510 watts. The 1845°F is significantly below the 2200°F post-LOCA limit on Zircaloy temperature imposed for PWR units by NRC[6].

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

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

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

3.4 Hydrogen Leak and Combustion

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

The maximum hydrogen in service will be about 3 ft^3 (STP) in one of the charging tanks, located outside the biological shield, when nearly all of the charging water has been emptied from the tank. The SER demonstrated that only through highly improbable scenarios can this gas, along with the oxygen necessary for combustion, get into the loop thimble. Even if it does, it will be mixed with helium and with water vapor or steam, and the void geometry is small (about 1 ft^3), dispersed, and very unfavorable for a detonation[7]. Calculations show that the reaction of a stoichiometric quantity of hydrogen (0.5 mols) with the oxygen in one cubic foot of air (equal to the void volume) at atmospheric pressure would produce only 135 BTU, a negligible amount (equivalent to burning 3.4 grams of fuel oil). Pressure buildup from any deflagration is readily relieved by redundant relief valves. It is, therefore, not credible that the thimble integrity can be breached by hydrogen combustion. The thimble itself is contained within the dummy fuel element, which presents a further barrier for protection of the fuel, core structure, and other components important to safety. Since these cannot be damaged or caused to malfunction by the highly unlikely combustion of hydrogen within the thimble, there is no unreviewed safety question in this regard.

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

3.5 Loss of Loop Pumping Power or Loss of Flow

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

3.6 Leak in the Lead-Bath Can

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

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

3.7 Emergency Core Cooling System (ECCS)

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

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

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

3.8 Electrical Short Circuit within the Thimble

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

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

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

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

The ability of the heater fuse to protect the system was observed during an unplanned demonstration when failure of the insulation on the heater power leads led to a short circuit at the point where the leads are sealed into the heater. The heater was connected at the time to 220 V, a little less than the 270 V to be used in service, but the only damage before the fuse opened was vaporization of a small fraction of an inch of lead-in wire and a little adjacent thermocouple wire. The failure was related to the method of insulating the leads near where they are attached to the heater, and redesign should prevent a repetition. The event, however, demonstrates that this hazard is easily contained within the thimble.

There have also been two failures of the Inconel heating elements near the top of the heater during tests to measure the system's capability for dissipating gamma and neutron heating upon loss of loop coolant during reactor operation. The tests have served this purpose (see Section 3.3), but, in addition, they have shown (1) that the effects of short-circuits in the heating elements are confined within the heater sheath and (2) that overheating will cause failure of the heating elements and cut-off of power before the bath temperature significantly exceeds the normal operating range.

In view of these considerations, it is not conceivable that fuel, the core structure or other components important to safety can be affected by destructive heating at the point of insulation failure.

Power for the reactor facility is provided by a 13,800/480-277V, 1000-KVA transformer. Power for the PCCL equipment is fed directly via a 360-A breaker, a new pothead at the containment wall, a 200-A fused safety disconnect switch to the CCL heater bus, a 100-A breaker off the bus, a heater power controller and finally a 100-A fuse. Power for the reactor is supplied over entirely separate lines from the 1000-KVA transformer, so that current surges due to loop malfunctions will have negligible effect on the power supply for reactor instrumentation and controls. Cables for the CCL power are run in conduits distinct from those for reactor control circuits. The PCCL thimble will be gounded by heavy cable to the reactor ground system, again avoiding proximity to reactor control circuits. Consequently, current surges due to short circuits or otherwise are expected to have no significant effect on reactor operation.

4. Conclusion

It is concluded that failures or accidents originating with the PCCL loop cannot interact with the reactor fuel, core structure or other components important to safety, except through reactivity effects. In this case loop failures or accidents will not cause reactivity changes exceeding those authorized by the Technical Specifications. For equipment important to safety, (i) the probability of an accident or malfunction is not increased, (ii) the possibility for an accident or malfunction of a different type than that previously evaluated in the SAR is not created, and (iii) no margin of safety in any technical specification is reduced. Consequently, the PCCL experiment does not involve an unreviewed safety question.

References

1. Safety Evaluation Report for the PWR Coolant Chemistry Loop (PCCL), Report No. MITNRL-020, February 13, 1987, plus Supplement dated April 19, 1988.

2. Technical Specifications for the MIT Research Reactor, Appendix A to Facility License No. R-37, July 23, 1975 as amended.

3. Memo dated July 7, 1978, L. Clark to MITRSC.

4. Safety Analysis Report for the MIT Research Reactor (MITR-II), Report No. MITNE-115, October 22, 1970, as amended.

5. MITR-II Start-Up Report, Report No. MITNE-198, February 14, 1977.

6. 10 CFR 50.46(b)(1).

7. Berman, M., "A Critical Review of Recent Large-Scale Experiments on Hydrogen - Air Detonations", Nuclear Science and Engineering, 93, 321-347 (1986).