SAFETY EVALUATION REPORT

FOR

THE BWR COOLANT CHEMISTRY LOOP (BCCL)

MITNRL-031

to be

installed and operated in the

MITR

March 9, 1989

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For Review by the

MITR Safeguards Committee

1. INTRODUCTION

1.a. Foreword

This is the safety evaluation report for an in-pile loop facility designed to simulate core coolant chemistry of a boiling water reactor (BWR). The loop will be used to carry out research into the effects of radiolysis and chemical additions (hydrogen, for example) on coolant composition (oxygen and hydrogen peroxide concentration, for example), as part of a program to evaluate the effect of the coolant environment on the corrosion of materials. Loop construction and operation are funded by the Electric Power Research Institute and the Empire State Electric Energy Research Corporation.

The objective of this report is to present a summary description of the design and operating procedures of the BWR Coolant Chemistry Loop (BCCL) in sufficient detail, and with supporting analyses, to demonstrate that it can be operated safely within the envelope of applicable MIT Reactor Technical Specifications. Considerable reliance upon, and reference to, the SER for the companion PWR loop will be made (1), because of the many features shared in common (in-pile containment thimble, in-core heater bath). As will be seen, the major difference addressed in this report is the need for steady-state in-core boiling in the BCCL.

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(1) MITNRL-020 (2/13/87), and supplements thereto, in their most recent revisions, PCCL (4/19/88) and Large Circulating pump (10/24/88).

1.b. General Description of BWR Loop

1.b.1. Introduction

The design of the BWR loop has evolved through the operation of two full-scale laboratory mockups, by Oliviera (1) and by Baeza (2). Many of the components will be used as-is in the in-pile version, now under construction. Figure 1.1 is a schematic of the major features of the BWR Coolant Chemistry Loop (BCCL) which will be operated in the MIT Reactor. As can be seen, it is a once-through system: high-purity feedwater is pumped from out-of-pile through the in-pile Zircaloy core section, where approximately 10 weight percent boils; the effluent is then separated in an outlet plenum and taken out-of-pile for analysis, and ultimate re-use. It is similar in many respects to the by-now-familiar PWR loop shown in Fig. 1.2. In particular, the in-core section is physically identical. The major difference elsewhere is the substitution of titanium for stainless steel and Inconel as the principal construction material.

1.b.2. Design details

Figures (1.3) through (1.8) show additional details of the BCCL; Table 1.1 summarizes system parameters of interest. The overall layout is sketched in Fig. 1.3. Starting on the right: cold feedwater is

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(1) C. Oliveira, "Design and Proof-of-Principle Testing of an In-Pile Loop to simulate BWR Coolant Chemistry," Nuclear Engineer's Thesis, MIT Nuclear Engineering Department, December 1987.

(2) J. Baeza, "Refinement of an In-Pile Loop for BWR Chemistry Studies," S.M. Thesis, MIT Nuclear Engineering Department, January 1989.

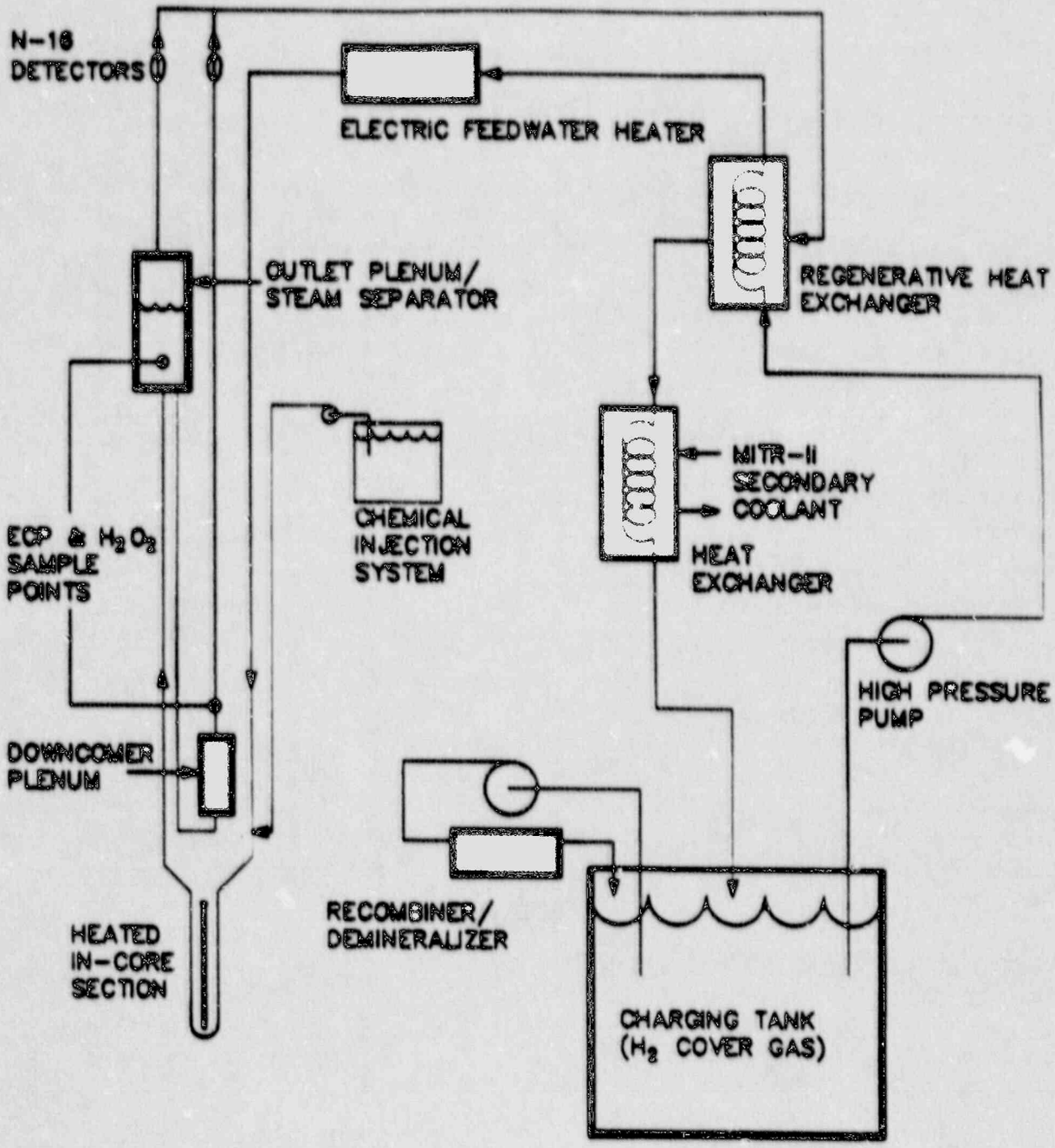

FIGURE 1.1: SCHEMATIC OF IN-PILE BWR COOLANT CHEMISTRY SIMULATION LOOP

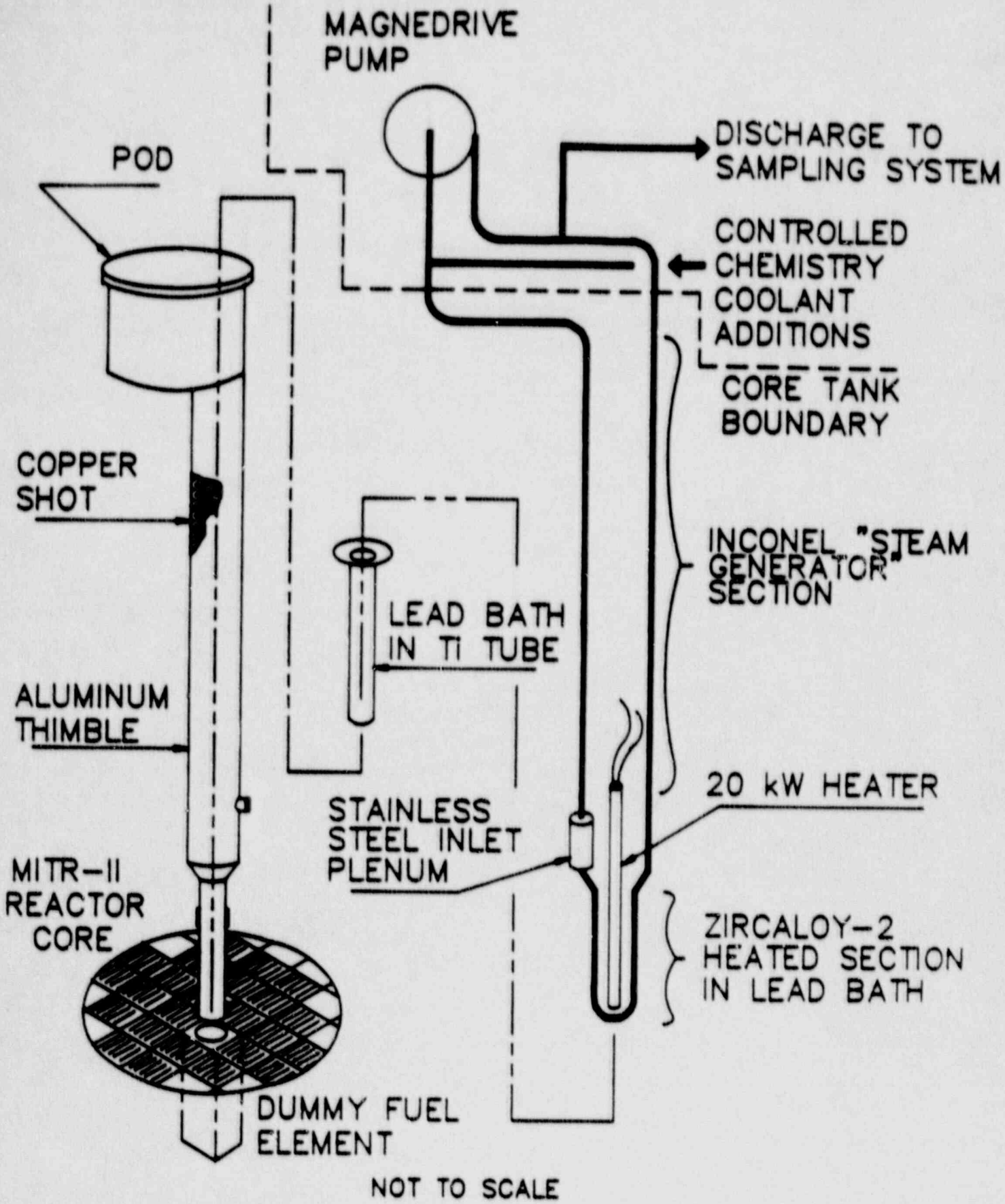

FIGURE 1.2: SCHEMATIC OF LOOP TO SIMULATE PWR COOLANT CHEMISTRY

TABLE 1.1

SELECTED BCCL DESIGN AND OPERATING PARAMETERS

Outlet Plenum Volume (water + steam)		1200 cc
Plenum Pressure and Temperature		1000 psia, 545°F
Downcomer Plenum Volume		55 cc
Feedwater Flow Rate		3000 cc/min
Core Section (Zircaloy U-Tube) Inlet T		530°F
Length of In-core U-Tube		48 in
Maximum Core Effluent Steam Quality		15 w/o
In-Pile Electric Heater Rating		20 kW
Feedwater Electric Heater Rating		20 kW
Maximum H_2 Content in Feedwater		2 cc/kg
In-core Dose Rates, (Mean, in H_2O)	neutron	4×10^8 R/hr = 1.6 w/g
	gammas	4×10^8 R/hr = 1.6 w/g
Operating Temperature of Hotwell Tank (max)		100°F
Rating of Heat Exchangers	Regenerative	36 kW
	Non-regenerative	7.5 kW
Void/Reflood Reactivity of In-Core U-Tube		$0.042\% \frac{\Delta k}{k}$ = 5.4 c

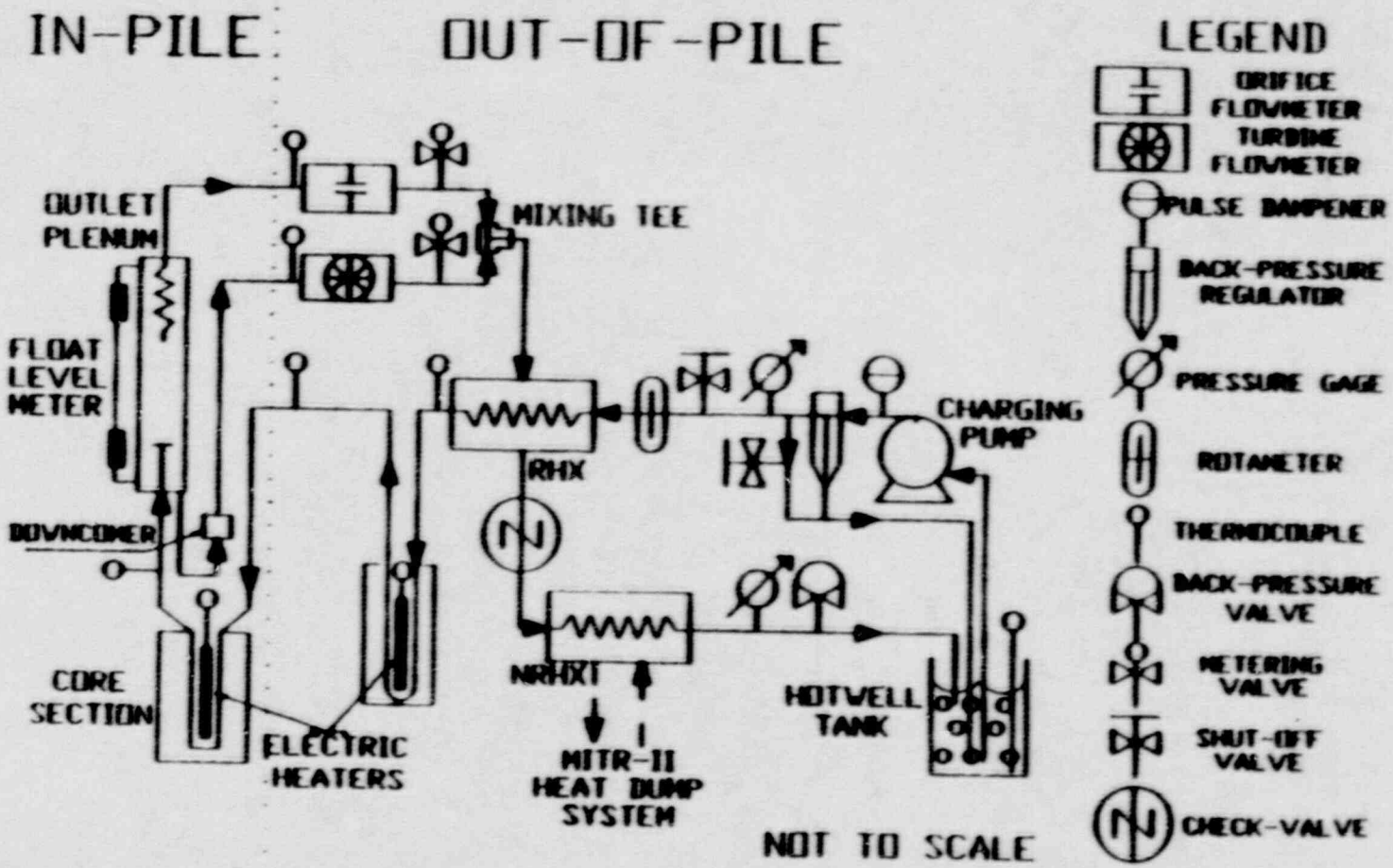

FIGURE 1.3: SCHEMATIC OF BCCL CONFIGURATION: OVERALL LAYOUT

pumped by a charging pump rated at $\geq$ 1200 psi at a rate of approximately 3 liters per minute through a regenerative heat exchanger, and electrically heated feedwater heater to the in-pile thimble. The same total mass of water, as steam and liquid at ≈ 300°C, is then brought back out of the MITR core tank, where the two effluent streams are combined, lose heat in the regenerative and non-regenerative heat exchangers, and are returned to the hotwell tank.

The similarity to the PWR system has already been noted, but there are differences: the main ones are summarized in Table 1.2. These differences and other details of the BCCL are elaborated upon in the paragraphs which follow.

1.b.3. Containment thimble

Of perhaps greatest interest are the differences in the containment thimble, since it is the actual interface between the MITR core and coolant and the subject experiment. The material of construction (aluminum) and all shapes/dimensions are the same as the PWR thimble except (see Fig. 1.4):

(a) the PWR shot port has been eliminated and replaced by a circumferential, (metal) gasketed, bolted flange.

(b) a second elliptical thimble section has been added to the bottom to increase the dose rate in the loops' ex-core "downcomer plenum."

The streaming snout terminates several inches above the fueled region of the MITR core, is beveled at its bottom, and positioned such that "shadowing" of adjacent core fuel assemblies, from the emergency cooling spray, is minimized: see Figs. 1.5 and 1.6. Thus no interference in either normal core coolant flow or emergency core spray reflood should take place.

TABLE 1.2

SUMMARY OF SIMILARITIES AND DIFFERENCES BETWEEN BWR AND PWR LOOPS

Item	Similarities	Differences
• Aluminum containment thimble	Same fill gas, pressure relief system, main structural components.	Flange in transition region, additional short elliptical section for enhanced streaming to plenum, no shot port.
• In-core heater bath	Identical; same heater. Ti sheath.	Zircaloy-2 U-tube in place of Zircaloy-4.
• Thimble internals		• Larger exit plenum in BWR, with steam/water interface and level measurement system. • No recirculating pump in BWR loop. • Insulating shot (quartz gravel) in place of conducting shot (copper). • Local sample extraction taps for out-of-pile analysis.
• Makeup/letdown system		Substantially different: • BWR flow rate ≈ 600 x higher, and ≈ 10 w/o of effluent is steam. • BWR has heat exchangers, feedwater heater.
• Charging, discharge tanks	Similar tank construction.	• Water is returned to charging tank and recycled.
• Operating conditions/procedures	≈ same temperature	Substantially different: • Steady-state boiling in-core • Substantially lower pressure (1000 psi vs. 2200 psi). • Shorter runs. • Attended runs.
• Handling facilities and equipment	Identical	

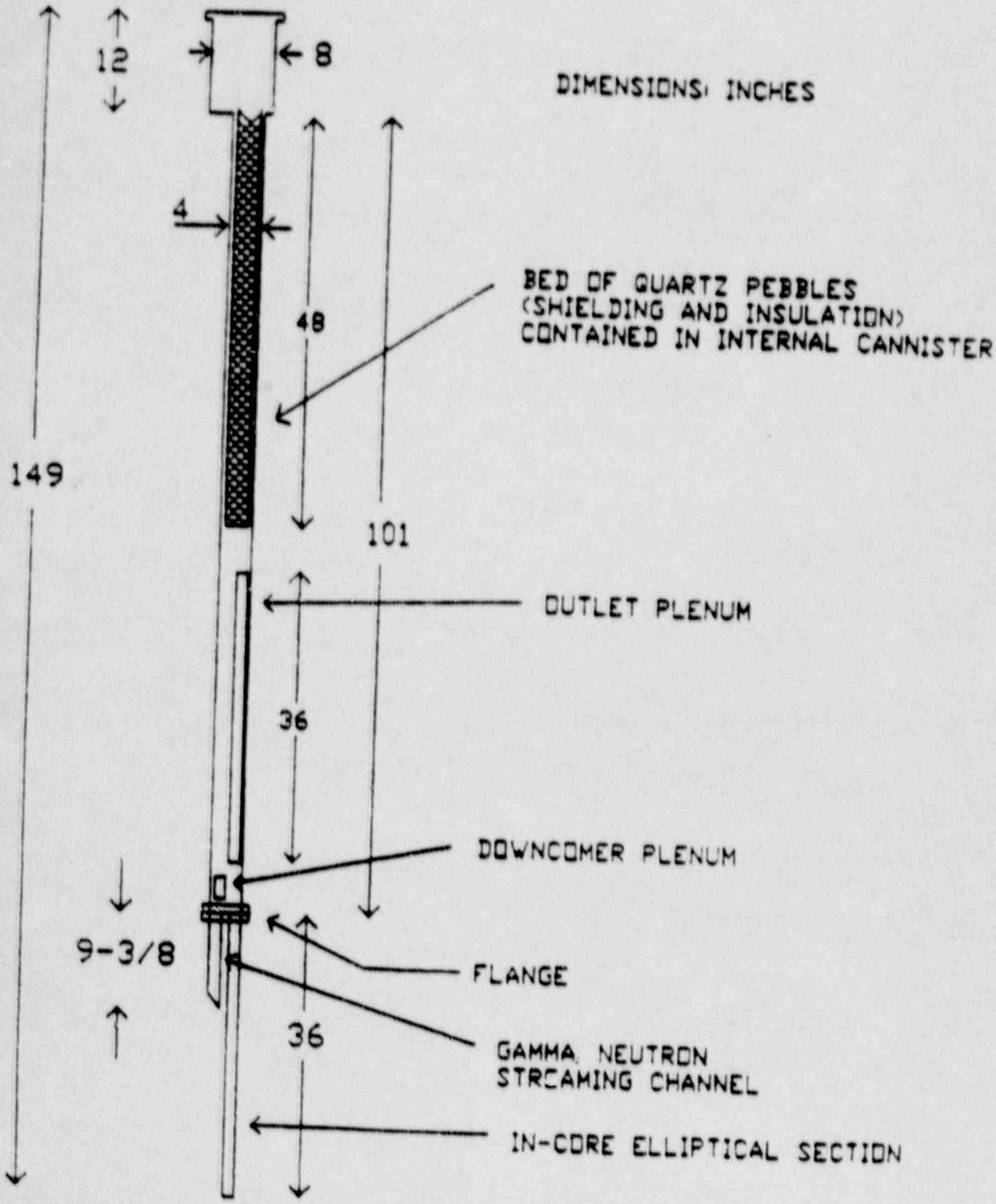

FIGURE 1.4: BCCL THIMBLE

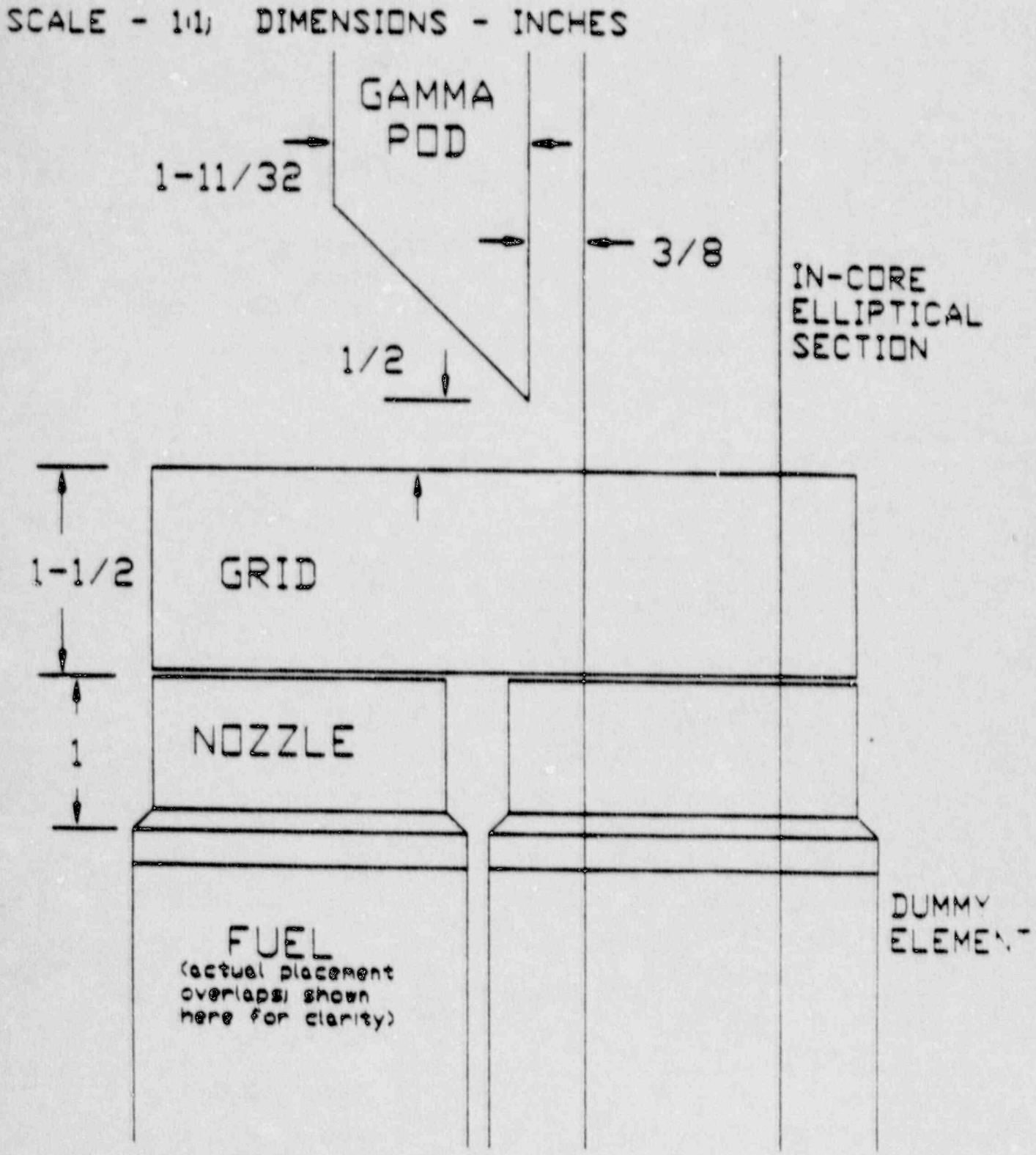

FIGURE 1.5: BCCL IN CORE TANK: RELATIVE PLACEMENTS

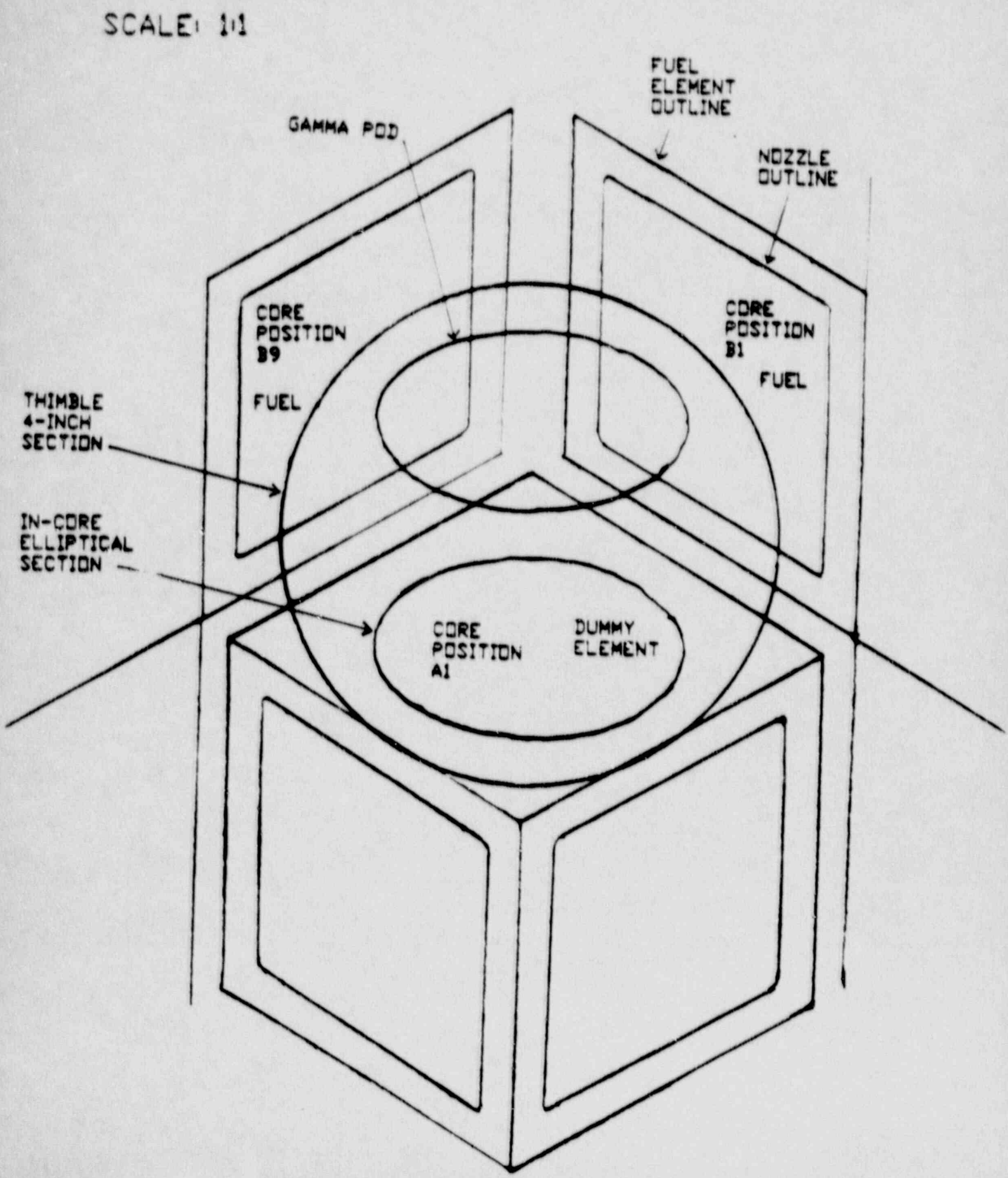

FIGURE 1.6: BCCL IN CORE TANK: TOP VIEW

1.b.4 Chemistry control features

Figure 1.7 shows the chemistry control system for the loop. Of main interest here is that for some tests a small concentration (≈ 0.2 ppm ≈ 2 cc/kg) of hydrogen gas (in a hydrogen-helium mixture to give a total hotwell tank overpressure of ≈ 15 psia) will be maintained. A system for addition of ppb-ppm levels of trace elements to the feedwater is also provided, but will probably not be used in the first round of tests. In place of the deposition monitor of the PWR loop, we instead have two NaI detectors to measure N-16 activity in the steam and water phases extracted from in-pile.

Finally, Fig. 1.8 shows the special chemical sample extraction features built into the BCCL. Here small side streams (roughly equivalent to the PWR loop let-down stream) are cooled in situ (by cold-sinking them to the thimble wall) and then led out of pile for analysis. Table 1.3 summarizes the initial campaign planned for the loop: a succession of short runs (on the order of hours) to study the radiolysis chemistry of BWR coolant - particularly H_2O_2 concentrations, since this constituent is thought to aggravate stress corrosion cracking.

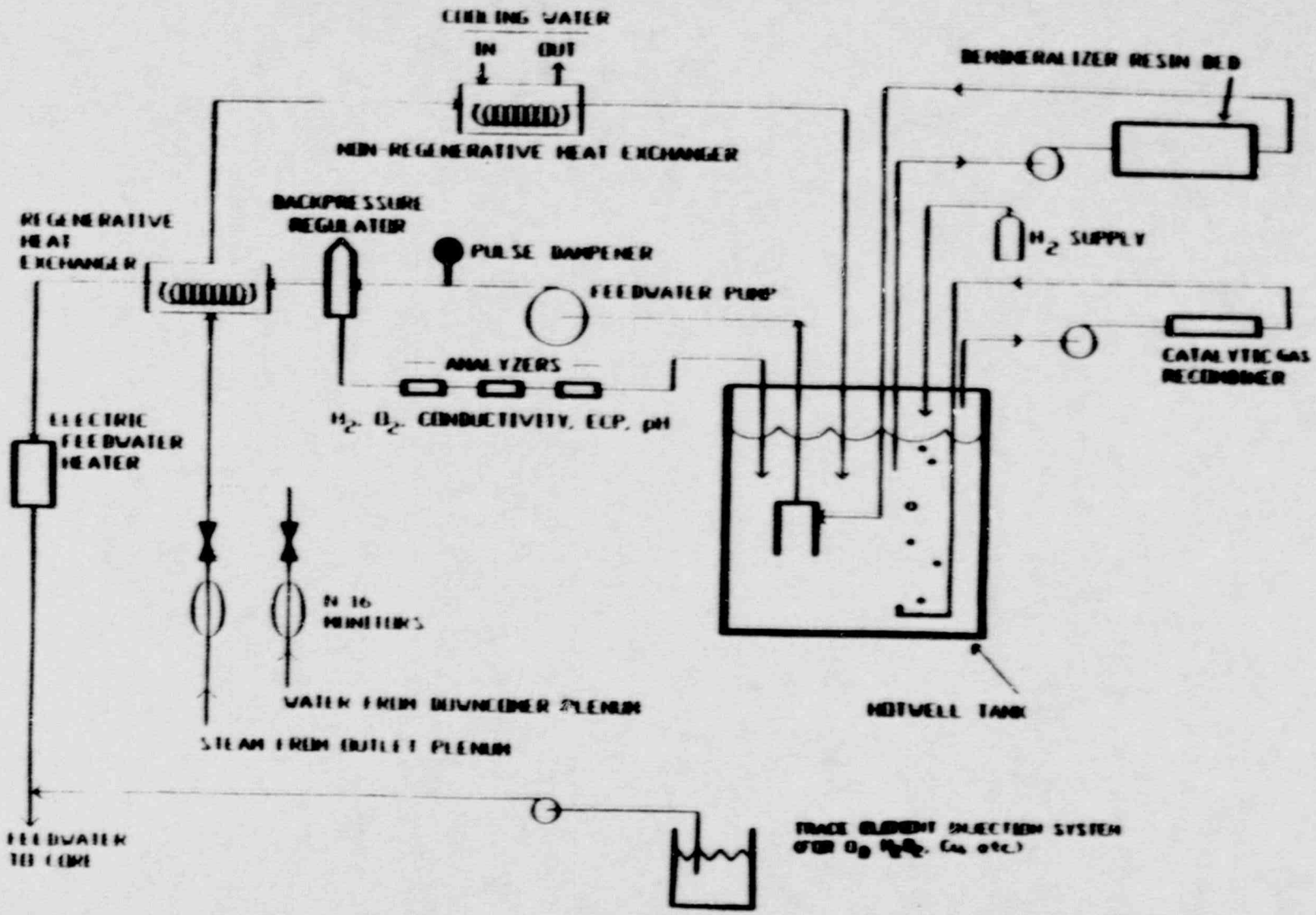

FIGURE 1.7 SCHEMATIC OF BCCL CONFIGURATION: DETAILS OF CHEMISTRY CONTROL SYSTEM

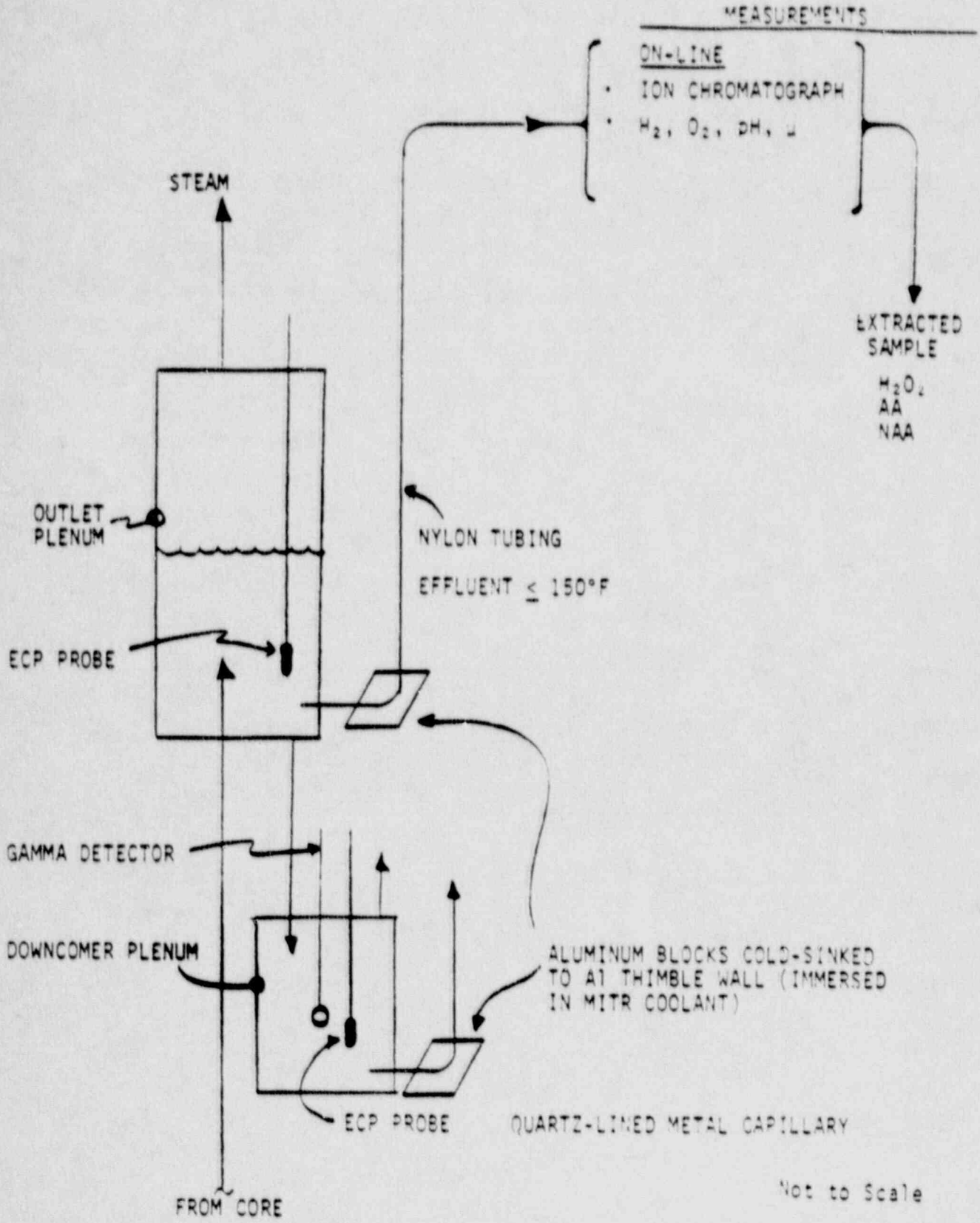

FIGURE 1.8: FEATURES RELATED TO CHEMICAL ANALYSIS

TABLE 1.3

PROPOSED BWR LOOP TEST MATRIX

WATER CHEMISTRY →

Runs ↓ / ppm H_2: (Loop feedwater)	A 1st 2-wk sequence of runs	B 2nd sequence of runs	C 3rd sequence of runs	D 4th sequence of runs
	Normal WC 10-20 ppb H_2	FULL HWC (≈0.15 ppm) to give approx. -230 mV ECP	≈1/4 H_2 to give specified interme-diate ECP	≈1/2 H_2 to give specified interme-diate ECP
I **Downcomer** exit H_2O				
Core effluent H_2O				
(a) Full Flow (transit time)	X	X	X	X
(b) 1/2 Flow	X	X	X	X
(c) 1/4 Flow	X	X	X	X
II **Bypass** (i.e. core at very low power & flow)				
(a) Full Flow	X	X	X	X
(b) 1/2 Flow	X	X	X	X
(c) 1/4 Flow	X	X	X	X
III Rerun I (core) but at				
(a) Lower heat flux	X	X	X	X
(b) Lower dose rate (MITR power)	X	X	X	X
IV **Downcomer**				
(a) 1/2 Dose Rate	X	X	X	X
(b) 1/4 Dose Rate	X	X	X	X

MEASUREMENTS: ECP, H_2O_2, H_2, O_2, N-16 in steam and water, for all runs; also pH, conductivity and cations (by AA) and anions (by Ion Chromatography)

NOTE: The two-week-long BWR experimental campaigns are planned to run in alternation with approximately one-month-long PWR loop runs. Thus, this set of BWR tests could be completed in about six months under such a scenario.

2. SAFETY-RELATED ASPECTS OF DESIGN AND OPERATION

2.a. Conventional Considerations

2.a.1. Introduction

Accident scenarios for the BWR loop parallel those of its PWR counterpart in most respects, differing primarily in details which have no significant impact on ultimate safety. In the subsections which follow, four such categories are evaluated:

(1) Loss of coolant flow

(2) In-thimble leakage

(3) Hydrogen release

(4) Radiation exposure

By inference, items not addressed here are judged not to be of safety significance - with the exception of reactivity-related aspects which raise sufficient questions to deserve attention in a separate section of their own (See Section 2.b).

2.a.2. Loss of coolant flow

As with the PWR loop, this is the type of incident having the highest probability of occurrence. The end result is the same: the in-core lead heater bath temperature increases until the redundant over-temperature trip cuts off electric heater power, following which gamma heat is safely rejected by passive means (radiation and conduction to the thimble wall, and thus to the MITR-II coolant). The intervening sequence differs somewhat in detail, however.

In the PWR loop loss of flow refers to the canned-rotor circulating pump, since its out-of-pile charging/pressurization pump cannot provide enough coolant by feed and bleed to remove significant heat by forced circulation. In the BWR loop, on the other hand, the charging

pressurization pump supplies all of the coolant flow. So long as this pump operates, even if the loop is fully depressurized, some 15 kW of heat removal capacity will be available. If interrupted for more than roughly one minute, however, the loop will boil dry, and overtemperature will follow shortly. While the pump is accessible, the short time interval before restart is required limits the nature of corrective actions to those which can be automated, such as restoration or substitution of power following an outage.

Since the BWR loop is made up of insulated components inside the in-pile thimble, its internal heat rejection capacity is limited - unlike the PWR loop, whose "steam generator" section can easily and passively reject 15 kW at 600°F. Thus for all heat loads above about 5 kW, the BWR loop relies upon the regenerative and (especially) the non-regenerative heat exchangers in its out-of-pile auxiliary systems, for which the ultimate heat sink (shared with the out-of-pile test tank) is a coolant system linked to the MITR-II secondary cooling system. Failure of the circulating pump in this heat sink loop will therefore require shutting down the loop. Since several minutes' warning will be available, this can easily be done by manually shutting off power to the in-pile electric heater using the switch provided in the reactor control room.

Although the final line of defense is automatic heater cutoff triggered by lead bath overtemperature, it is worth noting that the first line of defense - loop operator action - will be enhanced for the BWR loop, since it will be staffed by a crew of attendants during most operations. Unlike the PWR loop, which is designed to operate at steady state unattended for month-long runs, the BWR loop will be run in a

succession of short runs during which a number of on-the-spot chemistry analyses will be carried out. Thus it is quite likely that corrective action can be initiated to interrupt potential accident scenarios before any automatic protective systems are required to operate.

2.a.3. In-thimble leakage

The concern here is whether a rapid large leak could overpressurize the aluminum containment thimble. It is concluded that this will not occur for several reasons:

(a) During normal operation the BWR loop contains approximately the same inventory of liquid water (600 g) as the PWR loop; its maximum inventory, when the outlet plenum is flooded (no steam/water interface) is 1000 g. The BWR loop, however, operates at a much lower pressure (1000 psia vs. 2200 psia).

(b) The cold thimble wall can rapidly quench any steam it contacts: the entire loop inventory, if need be, within a few seconds (see appended estimates).

(c) As with the PWR loop, ultimate protection is provided by the thimble relief valve (set to lift at 20 psia) and rupture disk (designed to break at 65 psia). The large area lid on the top of the thimble would also leak through its gasketed seal at several hundred pounds overpressure.

Note that we cannot rely upon the attemperating effect of the copper shot bed used in the PWR loop. The quartz pebbles have a much longer time constant for heat absorption (≈0.5 sec), and may be at a higher initial bed-average temperature (because of the insulating gap between the liner and thimble). Nevertheless, the other features of the BWR design more than compensate for this difference.

Following the initial transient of a large leakage incident, the BWR loop response will differ significantly from that of its PWR counterpart. The normal feedwater charging rate for the BWR loop is approximately 3000 cc/min, about 600 times the feed and bleed rate for the PWR loop. Hence the BWR thimble internal void space can flood

within several minutes after initiation of leakage. No additional safety problems would ensue, but the consequences would be undesirable (e.g., shorting out of the in-pile electric heater). Thus consideration is being given to augmentation of the alarm system indication of this incident (currently high humidity), by addition of an internal leak tape and a low loop pressure alarm, perhaps including an automatic heater shutoff. Note also that the thimble PRV/burst disk system will be designed to accommodate thimble overflow at the maximum charging rate.

2.a.4. Hydrogen Combustion

Hydrogen has not traditionally been added to the coolant in BWR's as it has in PWR's. Recently, however, hydrogen water chemistry has been implemented in an attempt to suppress oxygen, and thereby reduce the susceptibility to stress corrosion cracking of materials exposed to BWR coolant. It is in fact the evaluation of the efficacy of this fix (and confirmation of the scientific basis for it in terms of aqueous radiochemistry) which is the central focus of BWR loop design and operations.

We see no problems not already addressed in the PWR loop SER, for the following reasons:

(a) The nominal required concentration of H_2 in BWR coolant is ≈ 2 cc/kg, an order of magnitude less than that in PWR coolant.

(b) The makeup water system for the BWR loop is quite similar to that for the PWR loop, including the hydrogen cover gas/sparging/recombiner subsystems.

(c) The same in-containment limit on total hydrogen inventory per loop ($\leq$ 20 SCF) will be observed. (Note that in the future the PWR and BWR and/or IASCC loops may be operated simultaneously).

The most significant difference between the BWR and PWR loops is that the former has a makeup/discharge rate some 600 times the latter. This means that a large leak in the charging or letdown systems outside of the MITR-II core tank would vent hydrogen into the containment volume at a correspondingly higher rate. Even so that release would amount to only 360 cc of H_2 (STP) per hour - a negligible amount which would be readily dissipated, and considerably less hazardous than, for example, a butane-filled cigarette lighter (compared on the basis of total energy release).

2.a.5. Radiation Exposure

In this category differences between the BWR and PWR loops will be negligible. Specific issues are as follows:

The single largest dose contributor is the in-core section of the loop and thimble. Here the BWR and PWR assemblies are virtually identical (with the only, and trivial, difference being the substitution of Zircaloy-2 for Zircaloy-4 for the heater bath U-tube). Since BWR loop runs are to be two weeks in duration (compared to five weeks for the PWR), end-of-run dose rates should be considerably less.

Handling doses for the remainder of the loop's components should be less because of the following considerations:

(a) The BWR loop will be all-titanium construction (to avoid catalytic decomposition of peroxide formed during radiolysis).

(b) Crud deposition studies are not currently planned; hence the in-core section will not be disassembled for segmentation, decontamination and assay of Zircaloy U-tube deposits.

(c) copper shot is replaced by quartz (SiO_2), which has much less radio-activation; and the quartz shot bed will not be drained after each run.

Here again the main difference comes from the higher letdown rate in the BWR loop. The increased flow rate and shorter transit time means that N-16 decay out of pile will comprise a more significant source (except when the version of the PWR loop employing an ex-tank circulating pump is used). Even so, dose rates are not significant. Extensive calculations have been made in the course of designing a system to measure N-16 in the steam and water effluents* and while count rates are more than adequate for measurement purposes, the corresponding addition to local exposure dose rate is less than ≈10 mr/hr at 1 meter. By comparison, the background dose rate at this location is approximately 15 mr/hr. In addition, the subject discharge lines are heavily shielded in the detector assembly to reduce undesirable background from the MITR-II: this coincidentally reduces the contribution of loop N-16 to passersby.

Finally, since neither boron or lithium are added to BWR coolant, unlike the PWR situation, we will not have (a small amount of) tritium in the effluent.

2.a.5. Section Summary

In this section, a number of safety concerns for which there is a one-to-one parallel with comparable PWR loop features have been reviewed. Although differing in detail, none of the issues addressed - loss of flow, in-thimble leakage, hydrogen escape, and radiation exposure - appear to differ significantly as regards consequences. We conclude that the BWR loop is at least as safe as its PWR counterpart.

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* Enhanced N-16 carryover in steam is a consequence of hydrogen water chemistry (thought to be in the form of ammonia, NH_3). Hence measurement of this phenomenon is a major subtask in the planned research program.

In the following section the major difference between the two loops will be addressed: boiling in core - a low probability transient for the PWR loop, but a steady-state feature for the BWR loop.

2.b In-Pile Boiling

2.b.1. Introduction

The obvious and major difference between the PWR and BWR loops is the intent to operate without steady-state boiling in the former, and with steady-state boiling in the latter.

While MITR-II technical specifications preclude boiling in core, it is clear [and has been formally so-interpreted (1)] that this refers to boiling in fueled assemblies, and not to boiling inside experiments. Moreover the total void/reflood reactivity difference for the PCCL and BCCL in-core coolant U-tubes is only about 0.042% $\Delta k/k$ (5.4 cents), as experimentally determined (2). This is well below technical specification limits for either "non-secured" $\Delta k/k = \leq 0.5\%$ or "movable" $\Delta k/k \leq 0.2\%$ experiments. Thus boiling is not a reactor safety issue per se. It is, rather, an issue related to the effect of boiling on ease of reactor operations in two respects:

(a) reactivity instrumentation noise

(b) spurious scrams due to intermittent short period indications.

Table 2.1 outlines the range of reactivity oscillations which boiling in the BCCL might encompass, during steady state operating conditions: background information essential to understanding these two issues.

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(1) J. Bernard memo to MITRSC Subcommittee for the Coolant Loop dated 8/31/88, "Conduct of Experiments Involving In-Core Boiling."

(2) K. Kwok memo dated 1/28/88, "PCCL" Reactivity Measurement Results - Revision I."

TABLE 2.1

PARAMETER ENVELOPE FOR BCCL BOILING EFFECTS

Feedwater Flow Rate*: 3000 cc/min = 50 cc/sec

Steam Quality: 10 wt%

Steam/Water Specific Volume Ratio: 20/1 (SAT @ 1000 psia)

Volume of In-Pile Zircaloy U-Tube: 50 cc

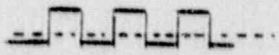
Thus one has:

2 cc steam/1 cc H_2O exiting core

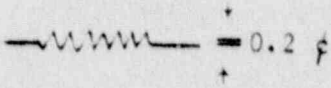
1 second core transit time

Limiting bubble frequencies:

(a) 100 cc steam, then 50 cc H_2O every second for step reactivity of ± 2.7 ¢ (5.4 ¢ peak to valley ≡ 0.042% Δk/k).



(b) ≈ 100 ± 10 one cc steam bubles exiting the core each second with a core-average steam content of 20 ± 2 cc, to give noise of ± 0.2 ¢ at ≈ 100 Hertz (±0.0016% Δk/k).



*NOTE: All values are approximate full flow, full power quantities.

2.b.2. Possibility of causing reactor scram

The nature of the false scram concern is explained in Fig. 2.1. As shown there, step reactivity insertions of any magnitude involve extremely short transient periods (in a purely mathematical sense) before reaching their asymptotic value. Thus an instantaneously responding period scram circuit would conceivably trip the reactor during virtually any and all power maneuvers. The actual circuit, however, has its own built-in time lag to avoid just such occurrences. The question, then, is whether loop boiling transients can somehow excite detectable periods of just the right type to dupe the period scram system. While this would not compromise reactor safety, it is clear that spurious challenges to safety shutdown systems are undesirable, and if too frequent, would effectively prevent meaningful operation of the BWR loop for the intended purpose.

2.b.3. Reactor noise

With respect to the issue of steady state boiling noise, it is more difficult to establish a specific criterion for judging it to be innocuous. Clearly both amplitude and frequency are of interest. Table 2.1 defines one such scenario: ± 0.2 c @ 100 Hz. Higher frequencies cannot be ruled out, but they would in general be associated with smaller amplitudes. Amplitude will surely be less than in a BWR reactor, where neutron flux variation related to boiling noise is ≈ 5%.

While both extremes discussed above appear tolerable, experimental verification is considered desirable. Hence a boiling experiment will be carried out using the PWR loop prior to BWR loop operations, and the results reported to the MITRSC.

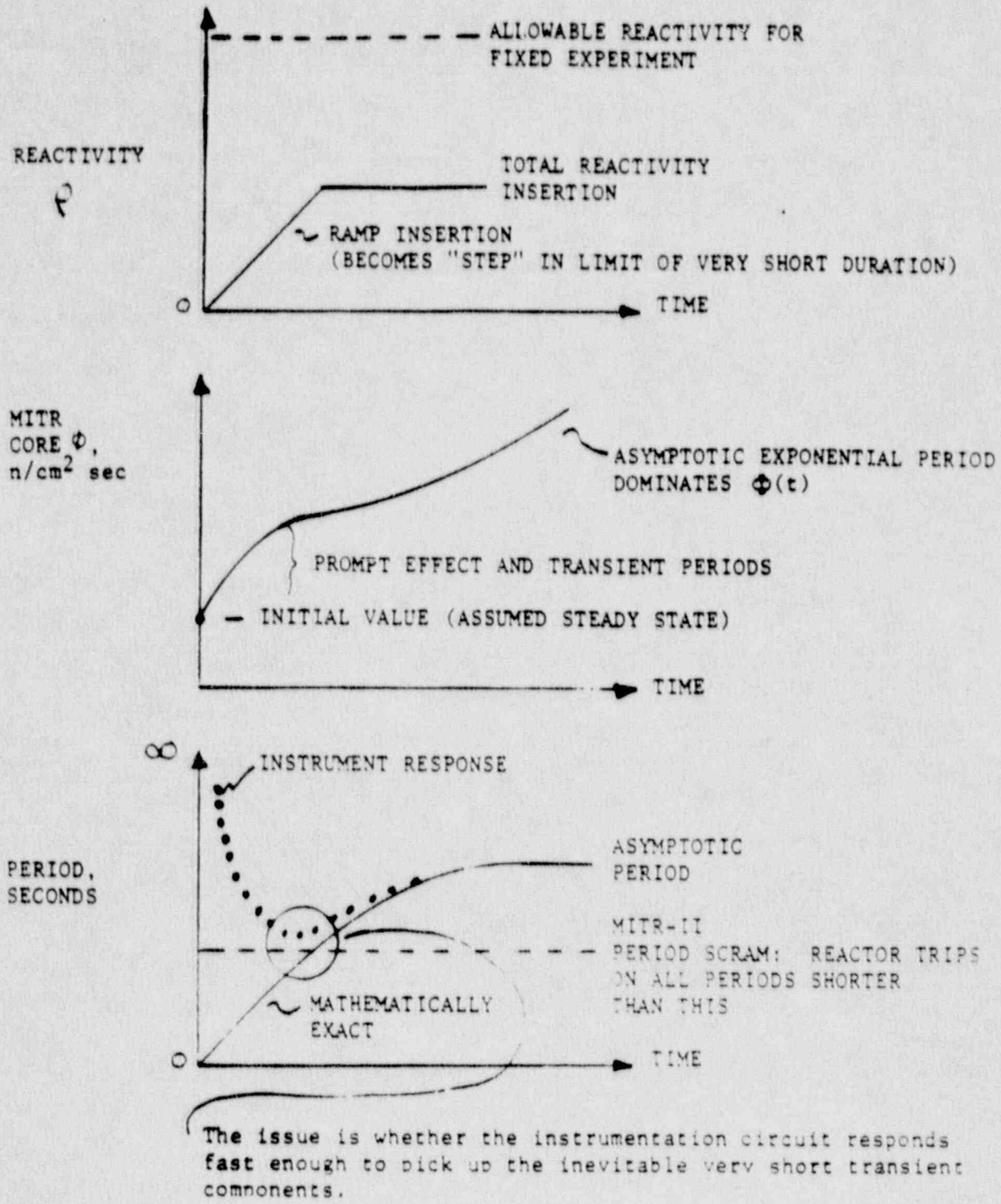


FIGURE 2.1: SCHEMATIC SEQUENCES SHOWING PERIOD SCRAM DILEMMA

A final note is of interest. From a purely neutronic viewpoint, operation of the BCCL will be quite analogous to a classical rod oscillator experiment (1,2) in which a neutron absorber is used to measure a reactor's transfer function by subjecting it to a variable frequency reactivity perturbation (10^{-3}-10^{1} cycles/sec). This is often done one frequency at a time, but even more relevant here are experiments in which random sequences are used, capable of exciting many frequencies simultaneously (3).

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(1) A. E. Profio, "Experimental Reactor Physics," John Wiley & Sons, NY (1976).

(2) J. M. Harrer, "Nuclear Reactor Control Engineering," D. Van Nostrand Co., NY (1963).

(3) T. W. Kerlin, "Methods for Frequency Response Measurements in Power Reactors," in Dynamics of Nuclear Systems, D. L. Hetrick, ed., University of Arizona Press, Tucson (1972).

3. SUMMARY AND CONCLUSIONS

The project staff has completed its safety evaluation of the Boiling Water Reactor Coolant Chemistry Loop (BCCL), as documented in this report.

As summarized in Table 3.1, in most safety-related aspects, the BWR loop differs only moderately in degree compared to its PWR counterpart. Moreover, differences often favor the BWR: e.g., its operating pressure is approximately 1200 psia lower.

The major noteworthy difference between the two loops is the presence of steady-state boiling in the in-core Zircaloy U-tube of the ~~BWR.~~ BCCL While measurements of the void/reflood reactivity change show that it is well within allowable limits (i.e. only 0.042% $\Delta k/k$), the uncertain frequency of the boiling noise and precisely how this will interact with MITR-II instrumentation has led us to recommend a proof-by-test approach to this aspect. Accordingly a low power boiling experiment is currently being planned for execution using the PWR loop. We anticipate that this will confirm the benign nature of BCCL operations. A report on this test will be prepared and circulated prior to requesting final permission for in-pile operations of the BWR loop, currently scheduled for the week beginning April 3, 1989.

In this report we have not gone into detail on the safety instrumentation and control systems because they are virtually identical to those on the PWR loop. Fundamentally, all high-severity scenarios, if not checked by intervening actions or events, lead to overtemperature of the in-pile lead heater bath - which then triggers (through redundant detection and control circuits) electric heater shutoff. The loop then need only reject approximately 6 kW of gamma heat - which can be accom-

TABLE 3.1

SUMMARY OF SAFETY REALTED ASPECTS OF BWR LOOP OPERATIONS

Aspect	Comments
Hydrogen release	• External inventory: same as PWR. • Internal inventory (dissolved in coolant) less than PWR. • Discharged H_2 is recycled, not vented. Consensus: as safe or safer than PWR.
Leak inside thimble	• Loop pressure is much lower than PWR. • Quartz shot has slower heat transfer, hence quenching must rely upon condensation by cold thimble wall. Consensus: Thimble rupture disk should protect against overpressure damage to thimble.
Loss of coolant or coolant flow	• Loop will boil dry, overtemperature will shut off electric heater, conduction and radiation will safely dissipate gamma heating. • Since there is no internal heat sink analogous to steam generator tube section in PWR, wet heat rejection will not be effective. Consensus: Ultimate passive safety mode is exactly same as PWR - proven effective by demonstration. Note: This mode may be used as the normal dry layup mode between runs if the loop is left in-core.
Steady-state and/or transient boiling in-core	• A total void/reflood $\Delta\rho \approx 5.4$ ¢ and a steady state $\Delta\rho$ variation of $\approx \pm 0.2$ ¢ must be accom modated (values to be established by in-pile tests with PWR loop). • Max step $\Delta\rho$ will not cause period scram. • Boiling noise shown tolerable in experiments using PWR loop. Consensus: Operation will not lead to challenges of MITR-II safety system.

plished by passive conduction and radiation without exceeding roughly 1700°F in the heater bath and its contents.

Baeza has drafted a preliminary list of incident response actions, reproduced here as Table 3.2. More comprehensive operating and emergency procedures, paralleling those drafted for the PWR loop, will be prepared prior to in-pile operations.

As a result of our review, it is concluded that there are no unreviewed safety questions involved in operating the BCCL in the MITR-II.

TABLE 3.2

SUMMARY OF MAIN BCCL SAFETY EVENTS

EVENT	SYMPTOMS	EXPERIMENTER ACTIONS	AUTOMATIC ACTIONS
Charging pump failure/ Loss of coolant	--Reduction or no flow readings in any of 3 flowmeters --Low loop pressure --Eventual high lead bath temperature	--Shut off heaters and charging pump	--High T heater shut off
Backpressure regulator failure	--Low loop pressure	--Shut off heaters. Place loop in standby mode.*	
Loop leak (small)	--Humidity increase	--Normal shutdown procedure	
Thimble leak (severe)	--Helium pressure decrease --Humidity increase	--Shut off heaters and open heater power breaker --Notify operators and consider requesting reactor shutdown	
NRHX cold side failure	--Loop temperature increase	--Shut off heaters. --Place loop in standby mode*	
Float level meter failure	--No output signal --Turbine flowmeter overspeed or orifice flowmeter differential pressure decrease	--Normal shutdown procedure	

*In standby mode cooling water is circulated through the loop by the main charging pump at low temperature to remove nuclear heat (electric heaters off).

APPENDIX A

Condensation on Thimble Wall

Steady state condensation relations are of the form (1):

$$Nu_L = 1.13 \left[\frac{Gr_L \; Pr}{Ja} \right]^{1/4}$$

where

Ja = Jakob number = $\Delta H\ell / \Delta Hv$, ratio of enthalpy rise of liquid water (between wall and saturation temperatures) to enthalpy of evaporation.

Pr = Prandtl number for liquid water at T_{FILM}

Gr_L = Grashof number = $g\rho^2 L^3/N^2$, also for liquid water at T_{FILM} (this form valid because $\beta\rho\Delta T \equiv \Delta\rho = \rho$)

$T_{FILM} = 1/2\ (T_{WALL} + T_{SAT})$

For T_{WALL} = 100°F and T_{SAT} at 1000 psia, the above expressions yield

$Gr_L = 1.66 \times 10^{15}$ $\qquad$ $Nu_L = 7800$

$Pr = 1.1$ $\qquad$ $h = 520$ BTU/hr ft^2 °F

$Ja = 0.73$

and for a 3.5 inch ID, 6 ft long thimble section the heat removal accounts for condensation at a rate of 0.54 lb/sec.

This estimate should be taken as a rough approximation only, because of several approximations:

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(1) C. J. Geankoplis, Transport Processes: Momentum, Heat, and Mass, Allyn and Bacon, Inc., Boston (1983).

(a) It applies in the laminar regime; turbulence will lead to substantial increases.

(b) It applies to the steady state; condensation will be much more rapid in the initial transient period.

(c) On the other hand, the retarding effect of non-condensible gas (here = 2 atm. of He) is neglected.

Nevertheless, the presence of a fairly rapidly acting and effective heat sink is evident.

Safety Review #-0-89-20: BWR Coolant Chemistry Loop (BCCL)

1. Description of Change

An in-pile loop designed to simulate the primary coolant system of a boiling water reactor (BWR) is to be installed in the MITR-II core. The facility is described in detail in [1], "Safety Evaluation Report (SER) for the BWR Coolant Chemistry Loop (BCCL)," Report No. MITNRL-031, March 9, 1989, prepared for review by the MIT Reactor Safeguards Committee and attached hereto.

2. BCCL Safety Evaluation Report

The BCCL SER provides both a description of the BCCL and an evaluation of the safety of its design and operation. Aspects of the BCCL which are virtually identical to those of a previously installed loop, the PCCL, are not discussed here in detail. This includes the safety instrumentation and control systems. Both the BCCL and PCCL can be placed in a safe condition by deenergizing the electric heaters. (Note: The PCCL is a 'Pressurized Coolant Corrosion Loop' used to investigate the formation, transport, and deposition of radioactive crud in Pressurized Water Reactors. The PCCL has been operational in the MITR-II since August 1989. It has functioned without posing a problem to the reactor.)

The BCCL SER concludes that the 'most noteworthy difference between the two loops (BCCL and PCCL) is the presence of steady-state boiling in the in-core Zircaloy U-tube of the BCCL.' The Reactor Staff agrees with this finding. In addition, the Reactor Staff has concluded that the potential for the BCCL's shadowing the reactor fuel during use of the emergency core cooling system (ECCS) is another major difference. Accordingly, this safety review examines these two issues (boiling and ECCS) in some detail. Other issues such as the presence of a high pressure system in-core and the use of H_2 gas are given less consideration because, for these issues, the BCCL parameters fall well within the envelope of conditions established by the PCCL. Refer to the PCCL SER, the PCCL SER Supplement, and SR #0-86-9 for details [2-3].

3. Compliance of Boiling Coolant Chemistry Loop (BCCL) with MITR-II Technical Specifications

A description of the BCCL is given in [1] and is therefore not repeated here except to note that the in-core portions of the BCCL and PCCL are identical. Also, the BCCL will be inserted in the A-Ring, as was the case with the PCCL.

3.1 Technical Specification Design Criteria

MITR Technical Specification #5.2.2 specifies that in-core experiments must comply with four criteria. These are:

a. They shall be positively secured in the core to prevent movement during reactor operation.

b. Materials of construction shall be radiation resistant and compatible with those used in the reactor core and primary coolant system.

c. Sufficient cooling shall be provided to ensure structural integrity of the assembly and to preclude any boiling of the primary coolant.

d. The size of the irradiation thimble shall be less than sixteen square inches in cross section.

The proposed BCCL design meets these criteria as noted below:

3.1.1 Lack of Movement During Operation

The BCCL, in the same manner as the PCCL and all previous in-core sample assemblies, is secured via the upper grid plate.

3.1.2 Materials of Construction

The exterior of the BCCL is aluminum and hence is compatible with the MITR-II's fuel, core vessel, and water chemistry.

3.1.3 Heat Removal

Energy is generated by means of a 20 kW electric heater and a lead bath (gamma-ray attenuation). In this respect, the BCCL is identical to the PCCL. Energy removal is different in that the PCCL uses a copper shot bed in the riser section of the loop. The BCCL rejects heat outside the reactor tank. The BCCL is protected by redundant sensors that will shut off the electric heater if the temperature becomes excessive. Should that cutoff fail, an alarm will sound in the control room and the operator can deenergize the system. This arrangement satisfies Technical Specification #5.2.2(c).

3.1.4 Cross-Sectional Area

The BCCL occupies only one fuel element position. Its cross-sectional area is less than sixteen square inches.

3.2 Reactivity of the BCCL Experiment

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:

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.

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.

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.

No reactivity measurements of the BCCL have as yet been made. However, given that the BCCL's in-core section is identical to that of the PCCL, the figures obtained for the PCCL are germane. The principal difference between the two is that soluble boron is not used in the BCCL coolant. This difference is minimal because of the small amount of boron present. (Note: For documentation of PCCL reactivity measurements, refer to MITR core configuration #87, dated 01/19/88. It should be noted that the more important reactivity numbers were measured on several occasions to be certain of accuracy.)

3.2.1 BCCL Water and Contained Chemicals

For the PCCL, these were classed as 'non-secured.' However, for the BCCL a change of phase is intended. Hence, the water and contained chemicals are classed as 'movable.' The limit is therefore 0.2% $\Delta K/K$ or 254 mbeta. Measurements for the PCCL put the reactivity worth of the coolant at +54 mbeta.

3.2.2 Flooding/Reflooding

Flooding/reflooding scenarios were examined for the accessible volume of the BCCL. This includes the void volume in the thimble and the coolant channel annulus between the thimble and the dummy element. For the PCCL, the effect of flooding these spaces was classed as 'non-secured' because it was not intended that these spaces be flooded or voided during normal operation. The same classification applies to the BCCL. The measured reactivity (PCCL data) was +181 mbeta. Hence, the flooding/reflooding scenarios meet the 'non-secured' limit which corresponds to 0.5% $\Delta K/K$ or 636 mbeta.

3.2.3 BCCL Components

The BCCL components (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% $\Delta K/K$. The measured reactivity worth for the PCCL was -407 mbeta. The absolute value of this figure is well below the limit of 1.8% $\Delta K/K$ or 2290 mbeta.

3.2.4 Experiment Interactions

As noted at the outset of section 3.2 of this report, Technical Specification #6.1.1 imposes limits on the total amounts of reactivity associated with both movable and non-secured experiments. Initially, the BCCL will be the only in-core experiment. However, at some future time, two loop experiments might be run simultaneously. It is noted here that the MITR-II core could contain three such experiments and not exceed either the movable or the non-secured limit.

3.3 Pressure Effects

Technical Specification #6.1 requires that experiments be designed to withstand twice the anticipated pressure. The loop portion of the BCCL will operate at approximately 1000 psi. Surrounding the actual loop is an elliptical thimble made of 6061 aluminum. It would be pressurized in the event of failure of the BCCL internals. The thimble is protected by a pressure relief valve set at 15 psi and a burst disk set at 65 psi. The maximum expected pressure is therefore 65 psi. The thimble will therefore be hydrostatically tested to 150 psi. (Note: The same approach was used for the PCCL following discussions with the U.S. Nuclear Regulatory Commission.)

3.4 In-Core Boiling

The BCCL experiment will involve boiling of the loop coolant which will, of course, be physically within the core. The MITR-II's technical specifications were carefully examined to determine if there was any prohibition of such an activity. It was concluded that boiling within an experiment (as opposed to boiling of the primary coolant itself) is permitted. The relevant technical specifications were:

3.4.1 Technical Specification #2.2 – Limiting Safety System Settings

The objective of this specification is: "To assure that automatic protective action will prevent incipient boiling in the reactor core and will prevent conditions from exceeding a safety limit." The specification itself establishes limiting conditions for flow, power, core tank level, and temperature to "... prevent incipient boiling ... which is initiated prior to the initiation of flow instability."

As to the applicability of this specification, the specification section itself contains no prohibition against boiling in an experimental facility. Its purpose is to restrict certain reactor parameters so that there will be no problem with flow in the fuel element channels. Accordingly, it is concluded that this specification does not pertain to in-core experiments that are isolated from the core.

3.4.2 Technical Specification #5.2 – Reactor Core

Paragraph 2a of this specification states that the "Design of in-core sample assemblies shall conform to the following criteria," one of which is that "sufficient cooling shall be provided to ensure structural integrity of the assembly and to preclude any boiling of the primary coolant."

As to the applicability of this specification, it clearly requires that the experiment be designed so that there will be no boiling on the outer surface of the experiment thimble and tube. The type of boiling, nucleate or bulk, is not mentioned. Presumably both are prohibited. There is no prohibition of boiling within the thimble.

3.4.3 Technical Specification #6.1 – General Experiment Criteria

Paragraph 2b of the specification states that, "The outside surface temperature of a submerged experiment or capsule shall not cause nucleate boiling of the reactor coolant during operation of the reactor."

As to the applicability of this specification, it clearly requires that the experiment be designed so that there will be no nucleate boiling of the coolant. There is no prohibition of boiling within the thimble.

3.5 Explosion Hazard

Technical Specification #6.1.3(b) limits explosive materials to the equivalent of 25 mg of TNT. The BCCL will have a hydrogen gas concentration of 2 cc/kg. Assuming this concentration is in all of the BCCL coolant that is within the reactor's biological shield, the total amount of hydrogen present will be about 2.5 cc at standard temperature and pressure. This corresponds to less than a milligram equivalent of TNT.

4. Safety Analysis

In the previous section of this safety review, the compliance of the BCCL experiment with the MITR-II's technical specifications was shown. In this section, specific safety issues are examined with the objective of demonstrating that an 'unreviewed safety question' or URSQ, does not exist. For the record, an URSQ is defined in 10 CFR 50.59(2) to exist if:

> 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.

Certain of the analyses given here closely follow those for the PCCL [2].

4.1 Temperature Effects

The in-pile loop assembly will be heated both by a 0-20 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 to be 9.6 kW at 5 MW [2]. Hence, 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. 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 potential for a Zircaloy-water reaction was analyzed in the PCCL SER [2] where it was shown that cooling by conduction and radiation should prevent temperatures in the thimble from exceeding 1845 °F for the maximum radiation heating i.e. 9.6 kW. This was based on a very conservative extrapolation of temperatures measured in a test mock-up of the PCCL assembly operating out of core at heater powers in the range of 2470-4510 watts. The 1845 °F temperature is significantly below the 2200 °F post-LOCA limit on Zircaloy temperature imposed for PWR units by NRC [4].

Assurance 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.

4.2 Hydrogen Leak and Combustion

This issue was thoroughly reviewed as part of the PCCL SER[3]. The concentration of hydrogen gas in the PCCL coolant is approximately 20 cc/kg. That in the BCCL's coolant will be 2 cc/kg, a factor of ten less. Hence, the BCCL's use of hydrogen is well within the envelope of previously established conditions.

The hydrogen gas used for the BCCL will be stored and handled using the same procedure as is now followed for the PCCL. The only difference is that the total inventory of hydrogen gas present in the containment will be doubled. Specifically, a storage limit of 20 SCF per loop not to exceed a total of 80 SCF will be observed. This does not present a hazard because discharge of 80 cubic feet of hydrogen into the containment building volume of 200,000 ft^3 will result in a concentration far below the lower explosive limit of 4.1%. Moreover, there is a fan mounted near the hydrogen station that ensures rapid dispersion of the gas. The maximum amount stored in any one container will not exceed 20 SCF.

4.3 Loss of Loop Pumping Power or Loss of Flow

As noted earlier, the heat sources for the BCCL are the same as those for the existing PCCL. Namely, a 0-20 kW electric heater and a lead bath for gamma ray attenuation. The heat sinks are quite different. The PCCL rejects heat to a copper shot bed located within the reactor vessel. The BCCL uses a non-regenerative heat exchanger located outside of the reactor vessel.

The BCCL SER makes several arguments to show that under most circumstances, a loss of flow would be more easily rectified for the BCCL than for the PCCL. The principal reasons are the external location of the circulating pump and the likelihood of operating in a continuously attended mode. However, these arguments are not germane because for some of the operating cycle, the BCCL may well be unattended. The important consideration is that, in the absence of any human interaction following a loss of BCCL flow, the in-core lead heater bath temperature will increase until the redundant over-temperature trips cut off electric heater power, following which gamma heat will be safely rejected by passive means, radiation and conduction to the thimble wall, and thus to the MITR-II coolant.

4.4 In-Thimble Leakage

Section 2.A.3 of the BCCL SER addressed the issue of whether a rapid, large leak could result in an overpressurization of the thimble. The MITR Staff agrees with the analysis given and with the conclusion that the relief valve and burst disks installed on the thimble are adequate.

4.5 Lead Bath Can Leak

Section 3.6 of SR #0-86-9 (PCCL Safety Review [3]) addressed the issue of large and small leaks of lead from the titanium can and concluded that there was no credible mechanism by which the loop could adversely affect MITR Safety. That conclusion applies here because the can construction is identical.

4.6 Electrical System Malfunction

The electric power supply for the BCCL is essentially identical to that for the PCCL. Section 3.8 of SR #0-86-9 (PCCL Safety Review [3]) examined the possibility of an electrical short and concluded that the loop was properly designed to prevent such an occurrence. That same analysis applies to the BCCL and is therefore not repeated here.

4.7 Effect of Boiling on Reactor Operation

The concern here was that the presence of boiling in the BCCL might cause reactivity fluctuations which, while within the magnitude allowed by the technical specifications, would cause operational problems such as excess movement of the regulating rod. Following discussions with the experimenter, an experiment was proposed to resolve this issue. Known as the 'In-Pile Boiling Experiment,' it involved performing boiling tests in the PCCL while at low power. Safety aspects of the proposed experiment were documented in SR #0-88-6 and approval for the experiment was given by the Special Subcommittee of the MITRSC on 09/15/88 and by the full MITRSC on 12/20/88.

The 'In-Pile Boiling Experiment' was conducted on 6/19/89 and it was found that the presence of boiling in the PCCL in-core section did not have any discernible effect on MITR-II operation. Additional information is given in Appendix A to this safety review which provides the experimenter's documentation of the test results. There is no credible mechanism by which the BCCL can adversely affect MITR safety on this issue.

4.8 Emergency Core Cooling System (ECCS)

The concern here was that the BCCL might shadow some of the 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. This shadowing could occur either as the result of the BCCL's riser section or as the result of the gamma ray pod.

Tests were made using a mock-up of the core top, primary coolant flow guide, and BCCL. 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 40% of the average flow per element. Five tests were performed and the results averaged. The final set of figures is given in Table I. The tests were conducted according to PM 6.1.1 which is used to verify compliance of the installed ECCS system with the MITR Technical Specification criteria on an annual basis. Technical Specification #3.6 requires that a total of 10 gpm be delivered via the ECCS system. Section 6.1 of the MITR-II's SAR further requires that each fuel

Table I

BCCL Test of Emergency Core Cooling Spray Distribution

Position	Grams H_2O	% of Average
A1 (not fueled)	146.5	35.5
2	591.3	143.4
3 (not fueled)	939.9	227.8
B1	356.1	86.3
2	184.9	44.8
3	268.4	65.1
4	359.8	87.2
5	349.5	84.7
6	558.6	135.4
7	533.3	129.2
8	419.7	101.7
9	506.2	122.7
C1	450.0	109.1
2	194.5	47.1
3	174.7	42.3
4	239.8	58.1
5	319.5	77.4
6	261.6	63.4
7	284.7	69.0
8	503.7	122.0
9	374.4	90.7
10	463.8	112.4
11	513.6	124.5
12	425.5	103.1
13	644.2	156.1
14	568.0	137.7
15	508.8	123.3

element receive at least 20% of the average flow per element. As Table I shows, there was significant variation in the flow distribution. However, all fueled element positions received at least 42% of the average, which is well above the 20% required. Accordingly, the installation of the BCCL will not interfere with the proper functioning of the emergency core cooling system. (Note: The mock-up tests were performed for the BCCL alone. Should the PCCL and BCCL be run simultaneously, a new set of tests would be required.)

5. Conclusion

It is concluded that failures or accidents originating with the BCCL 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 BCCL experiment does not involve an unreviewed safety question.

References

1. Safety Evaluation Report for the BWR Coolant Chemistry Loop (BCCL), Report No. MITNRL-031, March 9, 1989.

2. Safety Evaluation Report for the PWR Coolant Chemistry Loop (PCCL), Report No. MITNRL-020, February 13, 1987 plus supplement dated March 22, 1988.

3. SR #0-86-9, "PWR Coolant Chemistry Loop," April 21, 1988.

4. 10 CFR 50.46(b)(1)

Appendix A to SR #0-89-20

Results of In-Pile Boiling Experiment

Filed as part of Q/A program #M-89-2 are copies of the notebook entries which describe the thermal/hydraulic parameters established during a test on 6/19/89 to determine if reactivity noise generated by controlled boiling in an in-core loop poses a problem for reactor operation. The experiment was carried out using the PCCL loop with the circulating pump bypassed. The charging pump was used to provide forced once-through flow at a low flowrate and pressure was adjusted using the outlet line back pressure regulator. Note that the PCCL and BCCL in-core sections are identical except that the PCCL tubing is Zircaloy 4 while the BCCL tubing is Zircaloy 2.

Two reactor criticalities were established at low power. During the first of these the coolant was solid and the loop outlet temperature was maintained at approximately 515 °F. Once a baseline reactivity was determined under these conditions, the reactor was shut down and boiling was established in the loop. This was accomplished by lowering pressure to ~1000 psia (the planned operating point for the BCCL). Constant loop outlet temperature with increasing power indicated that boiling was occurring in the in-core tubing. The conditions achieved at the two operating points were as follows:

	Flowrate (lb/h)	Pressure (psia)	Electric Power (kW)	Loop Inlet T (°F)	Loop Outlet T (°F)
Solid	6.3	1515	2.8	107	500-530
Boiling	6.3	1005-1035	3.4	120	543

Note that T_{SAT} at 1020 psia is 547 °F. This agrees well with the measured loop outlet temperature. Using the solid run to calibrate the energy losses from the lead bath through the thimble walls, and energy balance gives a loop outlet quality of 50-60%.

No effect of the boiling was observed by the operators at reactor power levels up to 10 kW. The planned maximum quality in the BCCL is 10-15%. The higher flowrates to be used in the actual loop should result in less spatial and time variation of bubble sizes.