Research Plan to Assess the Dynamic Response of a Scaled B&W Reactor Model

Technical Brief

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I. Background

This document describes a research plan, to be undertaken by the Electric Power Research Institute and its Contractor SRI International to provide data regarding the dynamic response of a Babcock & Wilcox reactor from an integral test facility having two primary loops. The tests will address issues regarding the response of a B&W reactor following a small break loss-of-coolant accident.

Among the outstanding issues spelled out by the NRC (1, 2, 3,)*, the primary concern is the accumulation of steam at the top of a hot leg U-bend, sometimes referred to as the candy cane, such that the single-phase natural circulation around a primary loop is disrupted. At the same time, all the tubes in the steam generator are covered with water so that the secondary side of the steam generator cannot act as an effective heat sink. The postulated consequences of this steam bubble in the candy cane range from system refill, thus restoration of the single-phase natural circulation to system repressurization which can lead to, among other possibilities, many depressurization/repressurization cycles. For further details of these postulated consequences, see Figure 1.

There are many other concerns which will be listed below without elaboration.

- Two-phase flow regime in the hot leg
- The effect of internal vent valves
- Overcooling transient as a result of, for instance, steamline break
- Cold leg thermal shock
- Loop-to-loop instability perhaps as a result of asymmetric loop operations
- Tube rupture
- The effect of manual pump trip

* References are given at the end of this document.

II. Significance of Two-Phase Flow Regime in the Hot Leg

The type of flow in the hot leg when the primary system degenerates into two-phase is intimately related to the aforementioned primary concern and the various consequences of the small break scenerii shown in Figure 1. A special discussion of the two-phase flow regime in the hot leg is therefore warranted.

In general, the two-phase flow in a vertical pipe can be either bubbly, slug, or annular. These three types of flow, as they may appear in the hot leg, are schematically shown in Figure 2. If the flow is either bubbly or annular, the primary concern and its associated consequences may be a reality. On the other hand, if the flow is a slug flow, especially one being turbulent and unsteady, the constant agitation in the upper plenum of the steam generator by the carry-over water could result in heat removal near the upper tube sheet. Experimental evidence to that effect exists. In an EPRI/SRI test facility modeled after the Three Mile Island Unit 2 (4), a total of fifty two-phase natural circulation tests were performed. Test #4 was a high water inventory test in which the collapsed level in the steam generator was above the upper plenum but below the candy cane. The blockage of the single-phase natural circulation and the apparent absence of heat sink in the steam generator are obvious. Nevertheless, heat removal was accomplished and a steady state primary system condition was established. The slug flow detected in the model hot leg was apparently responsible for the heat transport.

In view of the importance of the hot leg flow regime, we will initiate a literature survey as part of a scaling analysis to be carried out in this research program.

III. Objective

The objective of this research program is to generate appropriate thermal-hydraulic data from an integral test facility modeled after a B&W reactor. The test facility will be designed with reference to the concerns enumerated in Section I and in accordance with the results of a scaling study which will precede the design.

IV. Method of Approach

The program will include the following tasks:

- Scaling analysis
- Conceptional design
- Facility design and fabrication
- Instrumentation calibration and installation
- Testing and data acquisition
- Reporting

A schedule for the performance of these tasks is provided in Section VIII. In order to complete this program atthe earliest possible date, contractual arrangement to initiate the task of scaling analysis has already begun. The results from the scaling analysis will serve as inputs to the conceptual design. In addition, technical consultation will be made with interested parties, such as B&W, to solicit constructive comments.

Design and fabrication of the test facility will draw upon the experience we have accumulated from similar programs in the past. As an example, when the existing 1/18-scale once-through steam generator (OTSG) two-loop test facility was assembled within weeks of the TMI accident, the facility suffered leak and heat loss problems. Since then, we have built and tested two U-tube steam generator (UTSG) integral facilities (7, 8), both of which performed satisfactorily with no leak and minimal heat loss. In the UTSG two-loop model (which operates at higher pressure and temperature, the heat loss is about 15% of the core power up to a vessel temperature of 300^{o}F. We believe that the heat loss can further be reduced by the replacement of exposed sight gages with differential pressure transducers.

It is expected that the instrumentation and the data acquisition system will be similar to those used in the existing test facilities. With minor modifications, the instrumentation/data acquisition system can be readily adapted to the planned facility.

V. Model Configuration

The gross configuration under consideration is discussed in this Section. Details and modifications are expected as the program proceeds. For sake of simplicity, the commonly accepted PWR terminologies will be used without clarification. A schematic of the current configuration is shown on figure 3.

The model will have a reactor vessel with internal downcomer and vent valves. Electric heaters will be used to provide the core power. The flow resistance of the core will be simulated if the scaling study so dictates. So will the internal structures in the upperhead.

There will be two primary loops with one hot leg and two cold legs each. In view of the uncertainties about the effects of a steam bubble in the candy cane and the type of two-phase flow in a full-scale hot leg, a number of interchangeable configurations for the model hot leg may be built. The sizes and the configurations will be determined by the scaling analysis. Pumps will be installed in the cold legs to allow for forced circulation.

Two commercial once-through heat exchangers will be modified to serve as the model steam generators. Appropriate secondary water feeds will be provided. A controllable flow passage between the primary and the secondary will be designed to simulate a tube rupture.

A pressurizer with water level and pressure controls will be installed. Plumbing will be provided to simulate small breaks in cold legs as well as the high-pressure injection. Steamline break will be simulated with the provision of sudden depressurization of the secondary. After a leak check and some shakedown tests, the model will be fully insulated.

VI. Instrumentation

Detailed instrumentation will be specified after the scaling analysis. The parameters to be measured are:

Reactor vessel:	vertical temperature profile collapsed water level two-phase mixture level upperhead pressure
Downcomer annulus:	temperature distribution collapsed level vent valve opening
Hot leg:	temperature distribution pressure in the candy cane collapsed level two-phase mixture level
Cold leg:	temperature distribution primary flow rate pump head

Steam generator primary:	temperature distribution collapsed level in tubes
Steam generator secondary:	temperature distribution pressure collapsed level feedwater and steam flow rates

VII. Type of Tests

The categories of tests that are planned for this facility are indicated below. The test matrix will include:

- Small break LOCA with and without pump trip
- Tube rupture transient
- Forced-to-natural circulation transition
- Single-to two-phase natural circulation transition
- Steamline break transient
- Loop-to-loop instability as a result of asymmetric loop operation
- Effects of noncondensable gas

VIII. Schedule

The present schedule for the research plan outlined in this document is shown in Figure 4.

References

1. Letter and enclosure from Darrell G. Eisenhut, Director of Licensing of NRC, to John J. Mattimoe of B&W Owners Group, October 5, 1981.
2. Letter and enclosure from Eisenhut to Mattimoe, March 25, 1982.
3. Letter and enclosure from Harold R. Denton, Director of NRC, to Henry Myers, Congressman to the U.S. House of Representatives, May 7, 1982.
4. R. L. Kiang, "Two-Phase Natural Circulation Experiments in a Test Facility modeled after Three Mile Island Unit-2", EPRI Final Report NP-2069, October 1981.
5. A. E. Dukler and Y. Taitel, "Flow Regime Transition for Vertical Upward Gas-Liquid Flow; Preliminary Approach Through Physical Modeling", NUREG-0164, January 1977.
6. Graham B. Wallis, <u>One-dimensional Two-Phase Flow</u>, McGraw-Hill Book Company, 1969.
7. R. L. Kiang and J. S. Marks, "Two-Phase Natural Circulation Experiments on Small-Break Accident Heat Removal", EPRI Interim Report NP-2007, August 1981.
8. P. Jeuck, III, L. Lennert, and R. L. Kiang, "Single-Phase Natural Circulation Experiments on Small-Break Accident Heat Removal", EPRI Interim Report NP-2006, August 1981.

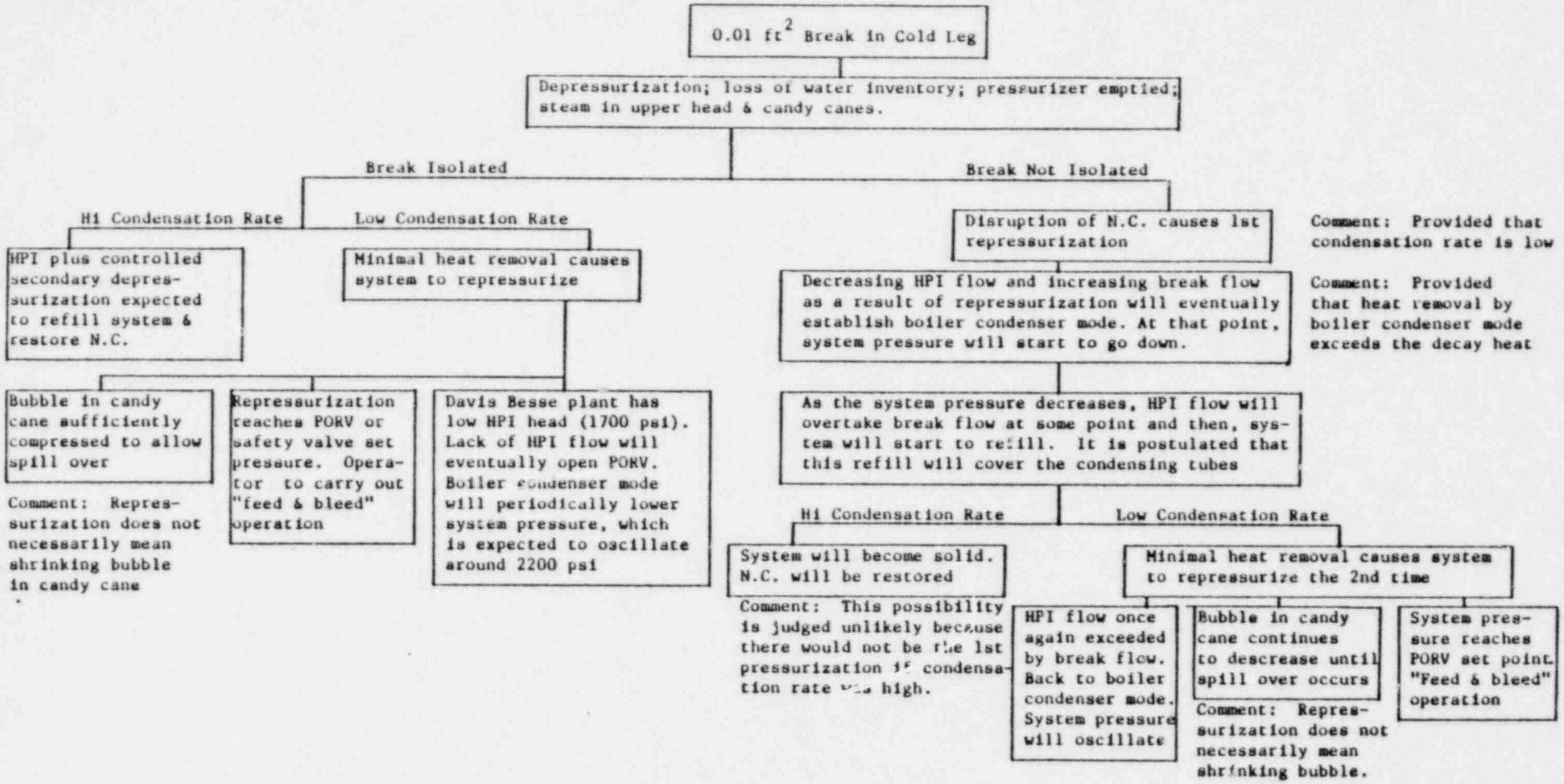

Figure 1 Small Break Scenarios

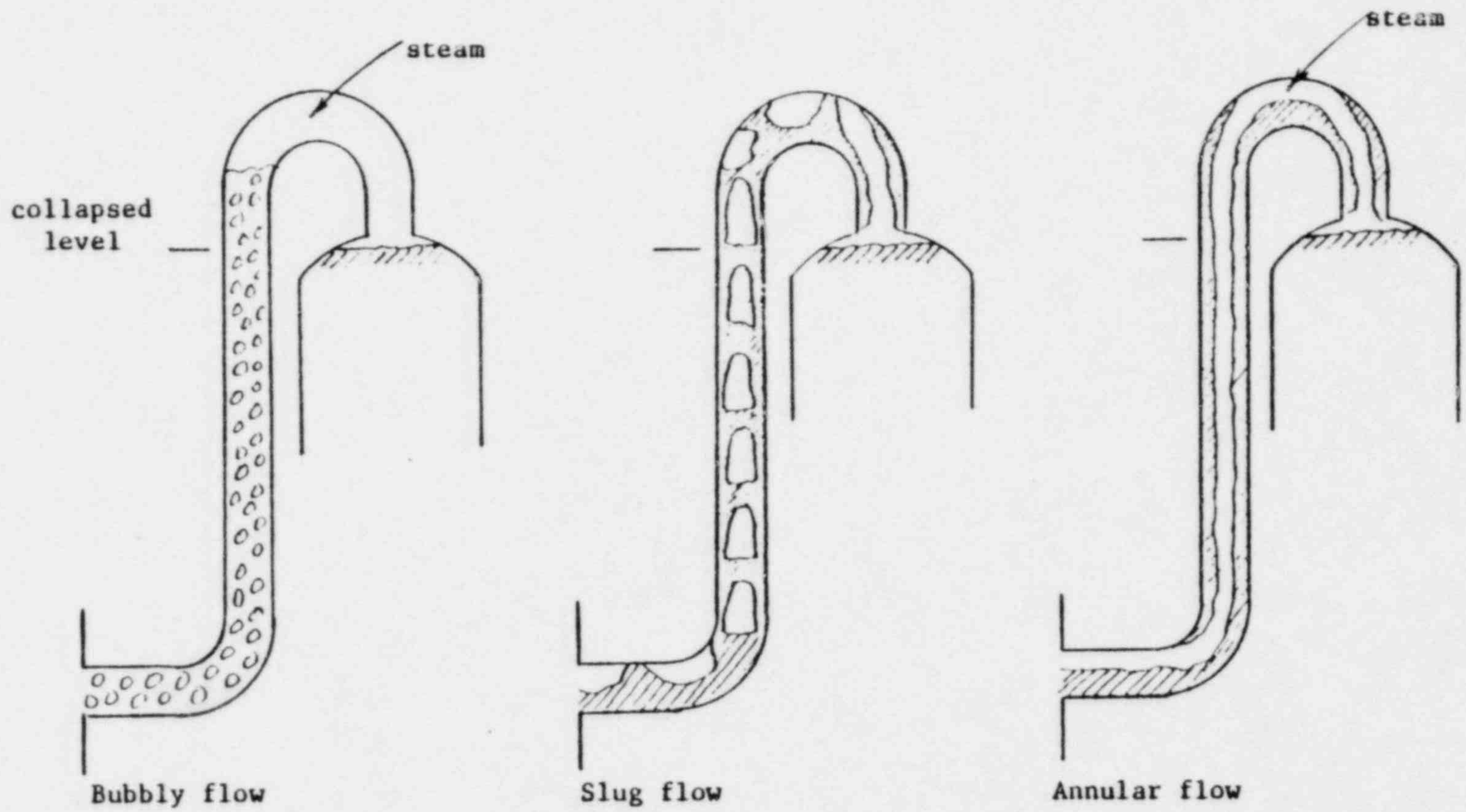

Figure 2 Two-Phase Flow Regimes in a Hot Leg

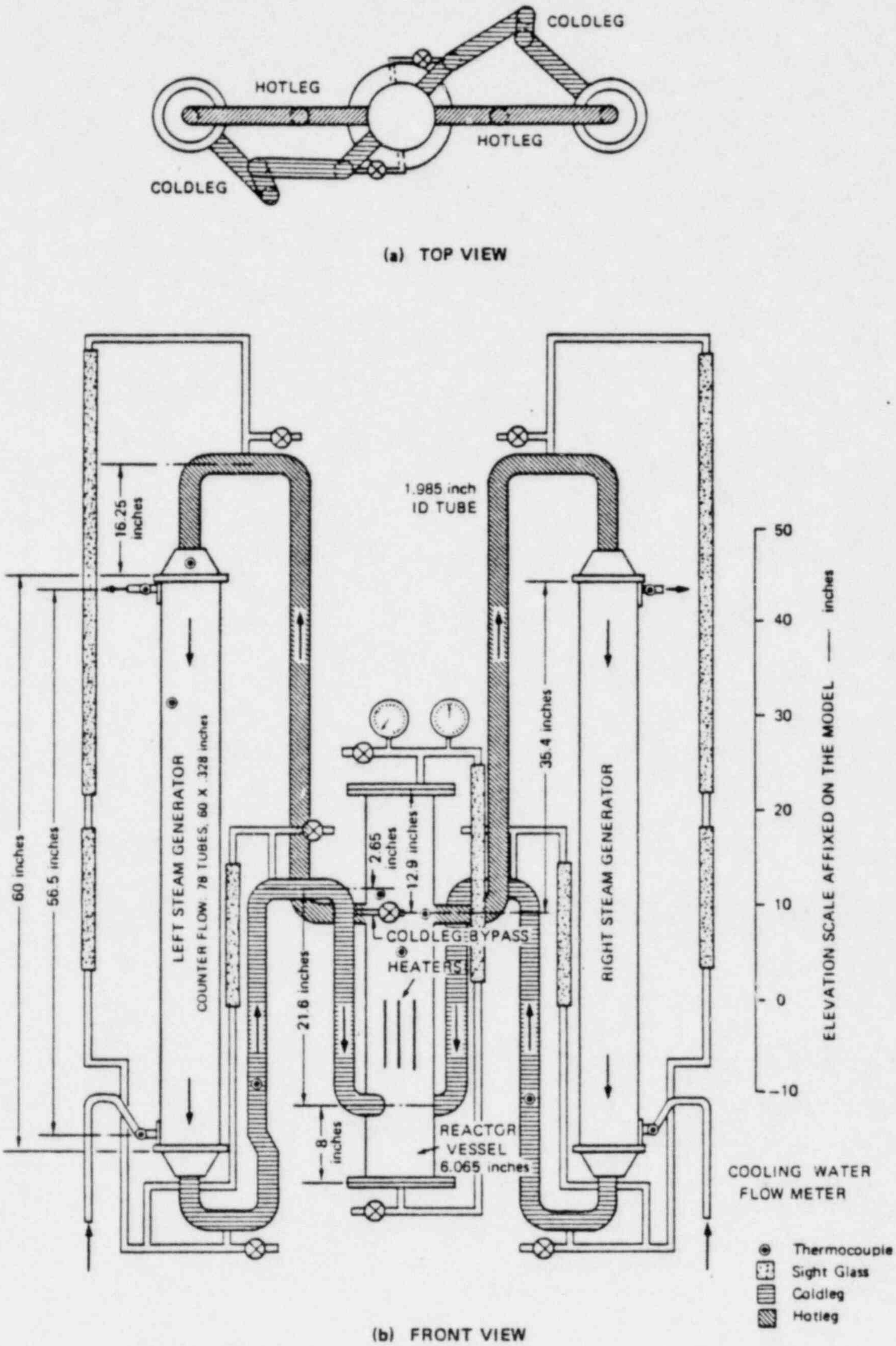

Figure 3-1. TMI-2 Cooling System Model, 1/18-Scale

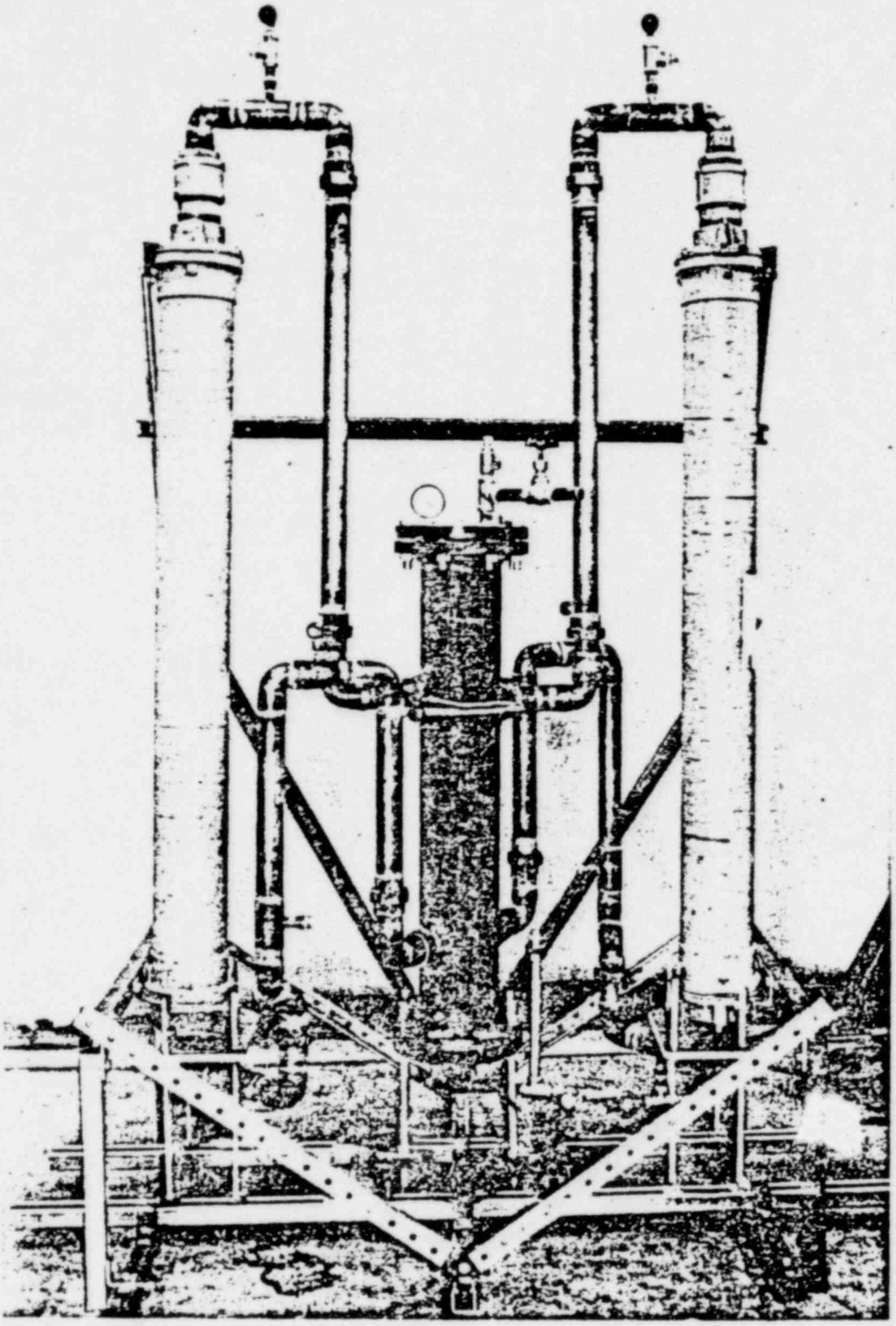

Figure 3-2. TMI-2 Model Test Facility

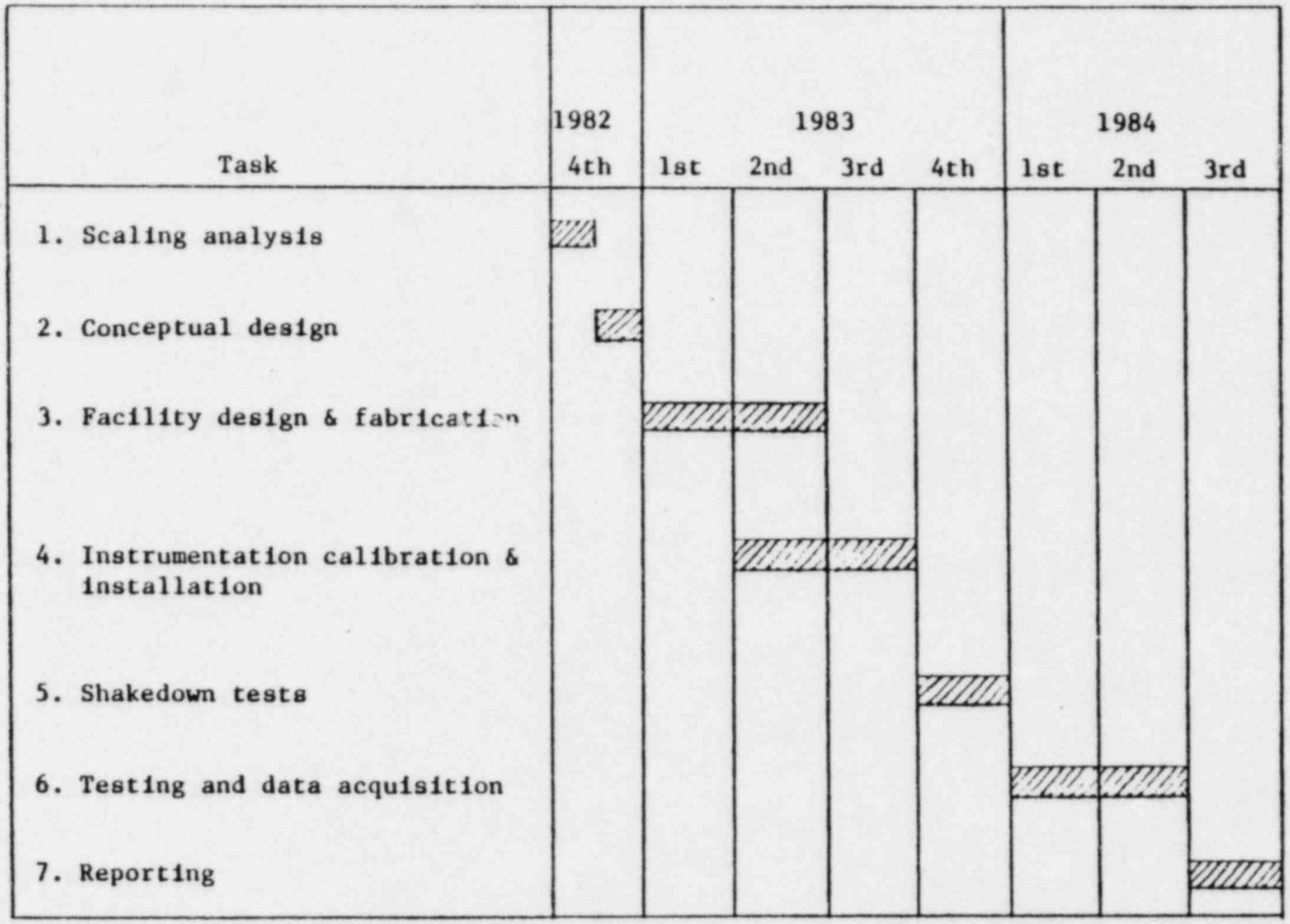

Figure 4 'Tentative Schedule of the Research Plan