MITSUBISHI HEAVY INDUSTRIES, LTD.
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TOKYO, JAPAN

April 19, 2007

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. David B. Matthews

Project No.0751
MHI Ref: UAP-HF-07040
UAP-HF-07041

Subject: Transmittal of the US-APWR Test Plan Summaries.

With this letter, Mitsubishi Heavy Industries, LTD. (MHI) transmits to the U.S. Nuclear Regulatory Commission (NRC) the documents entitled "US-APWR Summary of Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Plan" and "US-APWR Summary of Neutron Reflector Reflooding Test Plan" in response to the NRC's request. For the NRC staff's information, MHI had previously transmitted the US-APWR test schedule by MNES's letter dated February 16, 2007 (MNES Ref: NS-USAPWR-07-01). In the enclosed documents, MHI provides the latest information concerning the tests planned for the US-APWR.

In the document entitled "US-APWR Summary of Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Plan", MHI provides the detail schedule for the three test items listed in the previously provided test information as they are performed successively using the common test model and equipment. The document entitled US-APWR Summary of Neutron Reflector Reflooding Test Plan" notes that the detail test schedule for this test will be fixed at the end of July 2007.

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is below.

Sincerely,

M. Kaneda

Masahiko Kaneda,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosures:
Enclosure1 – US-APWR Summary of Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Plan (UAP-HF-07040)
Enclosure2 – US-APWR Summary of Neutron Reflector Reflooding Test Plan (UAP-HF-07041)

CC: S. R. Monarque
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Enclosure 1

UAP-HF-07040, Rev.0

US-APWR

Summary of Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test Plan

April 2007

Enclosure 2

UAP-HF-07041, Rev.0

US-APWR
Summary of
Neutron Reflector Reflooding Test Plan

April 2007

US-APWR

Summary of
Neutron Reflector Reflooding Test Plan

April, 2007

MITSUBISHI HEAVY INDUSTRIES, LTD.

1. INTRODUCTION

This document is the summary of the plan for reflooding test of neutron reflector for the US-APWR.

2. TEST PURPOSE

A neutron reflector (NR) is located between the core barrel and core, and forms the core cavity. The purposes of the NR are to increase structural reliability by eliminating bolts in the high neutron fluence region, to reduce neutron irradiation of the reactor vessel, and to improve neutron utilization and thus the fuel cycle cost.

The NR consists of ten thick stainless steel blocks. The NR is cooled by coolant through the flow holes in the blocks to prevent excessive stress and thermal deflections of the blocks due to the heat generated inside this steel structure by absorbing gamma radiation.

The NR has a high heat capacity because of the large metal volume. Therefore, heat release from the NR may affect the reflooding rate in the reflooding process during LOCA. WCOBRA/TRAC which will be used for safety analysis is able to model the NR as a separate channel with heat structure which is similar to AP600 modeling and which was approved by NRC. The reflooding test of the NR for the US-APWR was planned to comfirm the thermal hydraulic modeling of the NR using WCOBRA/TRAC.

3. TEST EQUIPMENT

A schematic diagram of test loop is shown in Fig.1. Flooding water is supplied to the test section from the feed flooding water line and is boiled by heated metal in the test section. The two-phase flow flows out to the upper plenum where the liquid part of the two-phase flow is separeted and drained to the liquid storage tank. The mist flow discharged from the upper plenum is separated into droplets and steam flow at the separator in the gas-liquid separation tank. The steam flows out to the steam line through a steam flow meter measuring the steam flow rate.

3.1 Test loop

- The test loop consists of the following epuipment
  - -Test section simulating a flow hole of the NR
  - -Gas-liquid separation tank to separate droplets and measure the flow rate ejecting through the flow hole
  - -Liquid storage tank to measure the quantity of the liquid ejecting through the flow hole
  - -Feed water tank and feed water pump which supply flooding water
  - -Heaters in feed flooding water line and feed water tank to set the flooding water temperature
  - -Steam line and pressure control valve holding system pressure

3.2 Model of test section

- Diameter and height of NR flow hole is same as the US-APWR
- NR metal is heated by heater to set initial metal temperature condition

4. MEASURING

The major measurement items in test section are shown below.

-Outlet steam flow rate of the NR flow hole
-Outlet liquid flow rate of the NR flow hole
-Fluid temperature in the NR flow hole
-Metal temperature
-Differential pressure in the NR flow hole
-Pressure
-Flow regime at the outlet of the NR flow hole

5. TEST CONDITIONS

Test conditions including the range of test parameters will be determined appropriately based on evaluating the of large break LOCA analysis results with US-APWR plant parameter by WCOBRA/TRAC.

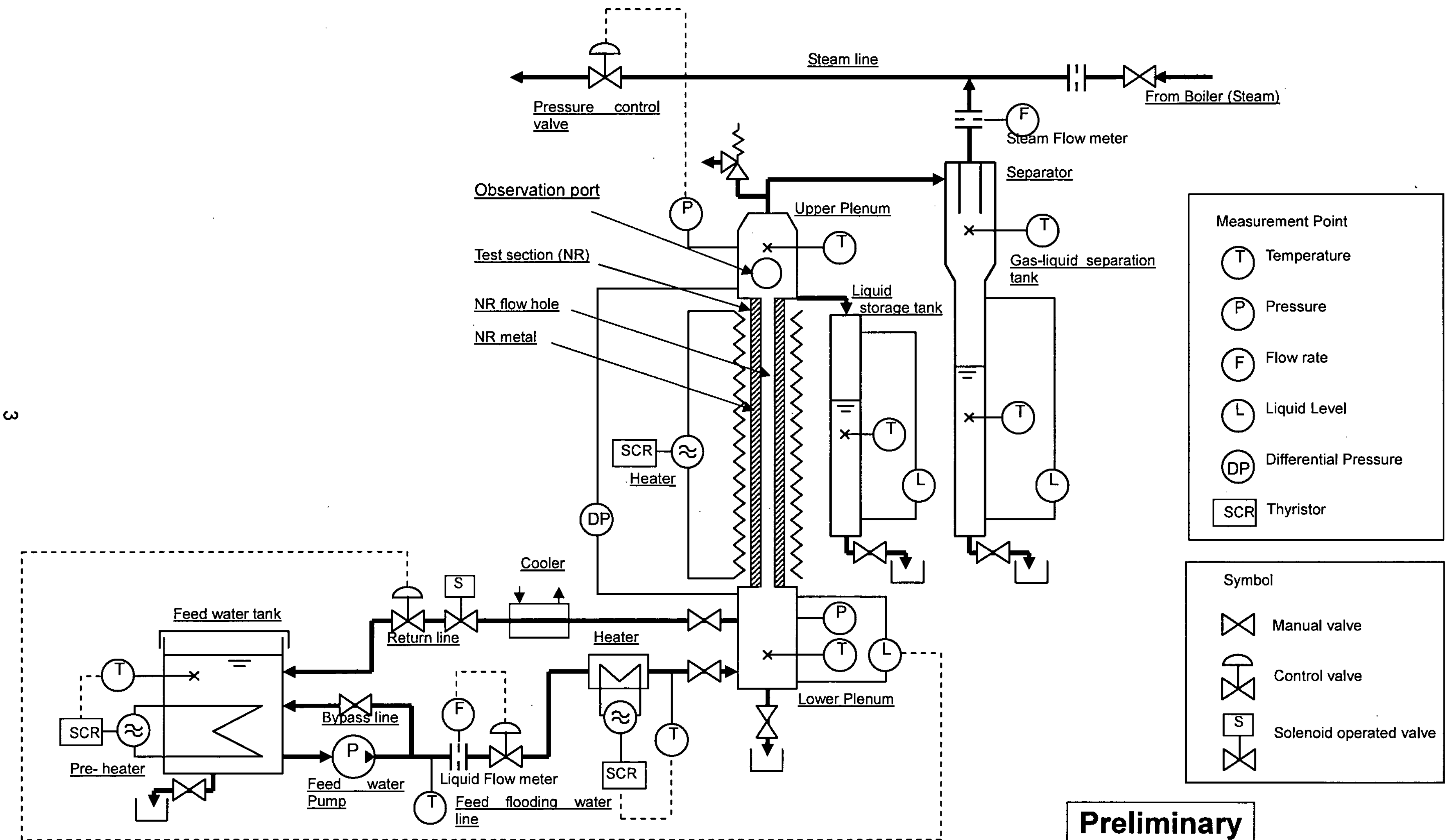
6. SCHEDULE

Table 1 shows the test schedule.

Table 1 Test Schedule

Year	2007								
Month	4	5	6	7	8	9	10	11	12
Test plan (2006/9-2007/4)									
Design of test equipment (2006/10-2007/5)									
Preparation/ Installation of the test equipment (2006/11-2007/7)									
Test									
Typical case test									
Parameter effect test									*

*: Detailed test schedule will be fixed at the end of July.



Preliminary

Figure 1 Schematic Diagram of Test Loop

US-APWR

Summary of
Reactor Vessel Lower Plenum
1/7 Scale Model Flow Test Plan

April, 2007

MITSUBISHI HEAVY INDUSTRIES, LTD.

1. INTRODUCTION

This document is the summary of the plan for the US-APWR Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test.

2. TEST PURPOSE

The US-APWR reactor internals are designed for the core with 14ft -257 fuel assemblies. Although the integrated lower core support design exists as the standard design of 14ft core reactor for Westinghouse 3XL/4XL, there is no operating experience for the 257 fuel assemblies. The elimination of the core instrumentation system from the bottom region simplifies the structures of the vessel lower plenum.

The purpose of this test is to confirm the following characteristics related to the lower plenum design configurations of the US-APWR as mentioned above.

(1) Hydraulic characteristics
- Core inlet flow distribution
- Inlet flow distribution for cooling of the Neutron Reflector (NR)
- Pressure loss of the vessel down-comer, lower plenum and lower core support

(2) Flow- Induced Vibration (FIV) of the structures in the lower plenum

(3) Core inlet temperature distribution in events with non-uniform cooling

3. TEST EQUIPMENT

3.1 Test loop

- The region to be simulated is identified in Fig. 1.
- Four sets of Inlet pipes for 4 loops are simulated as shown in Fig. 2.
- In the test configuration, the vessel and internals are placed in up-side down for easy access to the lower plenum.

3.2 Model of Vessel and Internals

- Vessel and lower internals are simulated in 1/7 scale as shown in Fig. 3.
- Lower head of the vessel is made of acryl to provide visualization of the flow.

4. TEST CONDITION

- The test will be performed at the room temperature.
- Reference flow rate will be determined to simulate the flow velocity in the actual plant.

5. MEASURING

5.1 Hydraulic Test

- Core and NR inlet flow distribution will be measured using venturi flow meters.
- Pressure loss of the vessel down-comer, the lower plenum and the lower core support will be measured by static pressure taps.

5.2 FIV Test

- Vibrations of the diffuser plates will be measured using accelerometers.
- Stress fluctuations on the support columns will be measured using strain gauges.
- Pressure fluctuations will be measured by pressure transducers.

5.3 Core Inlet Temperature Distribution Test

- Non-uniform cooling conditions will be simulated by pre-heated water injection from one of the inlet pipes.
- Temperature distributions at core inlet will be measured by thermo couples.

6. SCHEDULE

The schedule of this test is shown as Table 1.

Table 1 Schedule of Reactor Vessel Lower Plenum 1/7 Scale Model Flow Test

Year	2007								
Month	4	5	6	7	8	9	10	11	12
Test Plan (2006/9-10)									
Equipment Design (2006/11-2007/4)									
Manufacturing & Setting									
Hydraulic Test									
Measurement System Set up									
Test without Diffuser Plates									
Test with Diffuser Plates									
Test for Core Inlet Flow Holes Optimization									
FIV Test									
Measurement System Set up									
Test									
Core Inlet Temperature Distribution Test									
Measurement System Set up									
Test									

Note: The test schedule may be changed by detail test procedure preparations, inspection and maintenance work of the existing test facility.

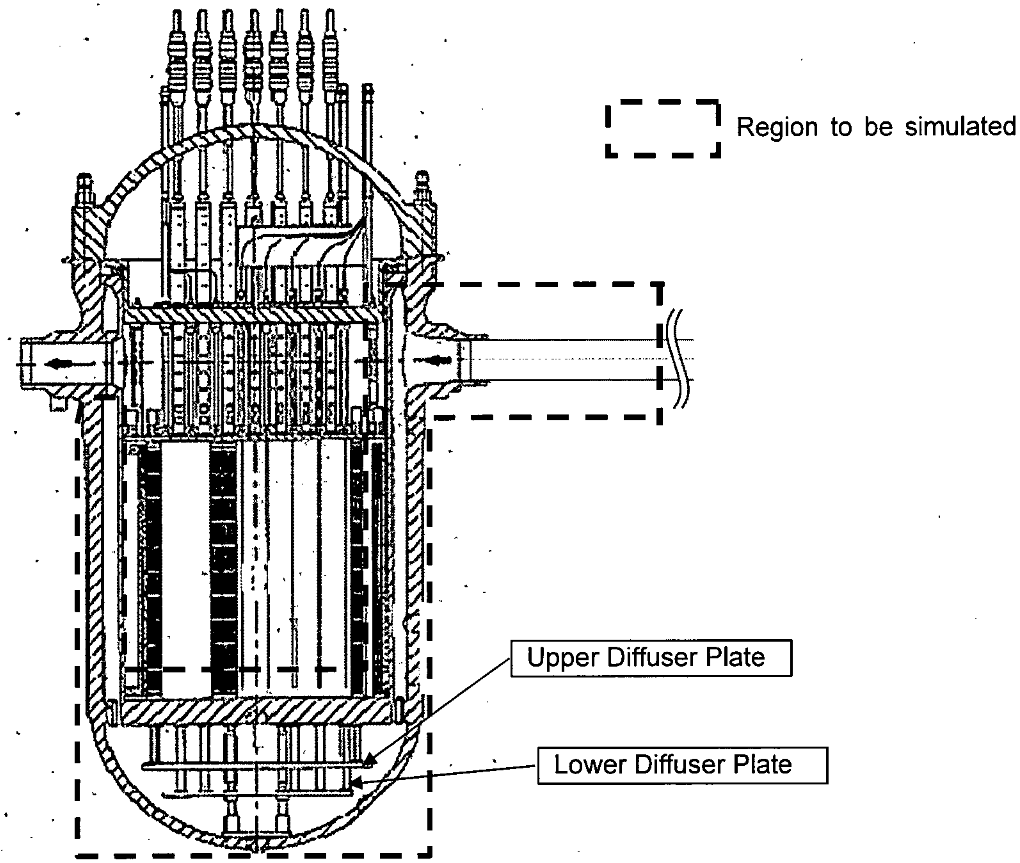


Figure 1 Scope of Simulation

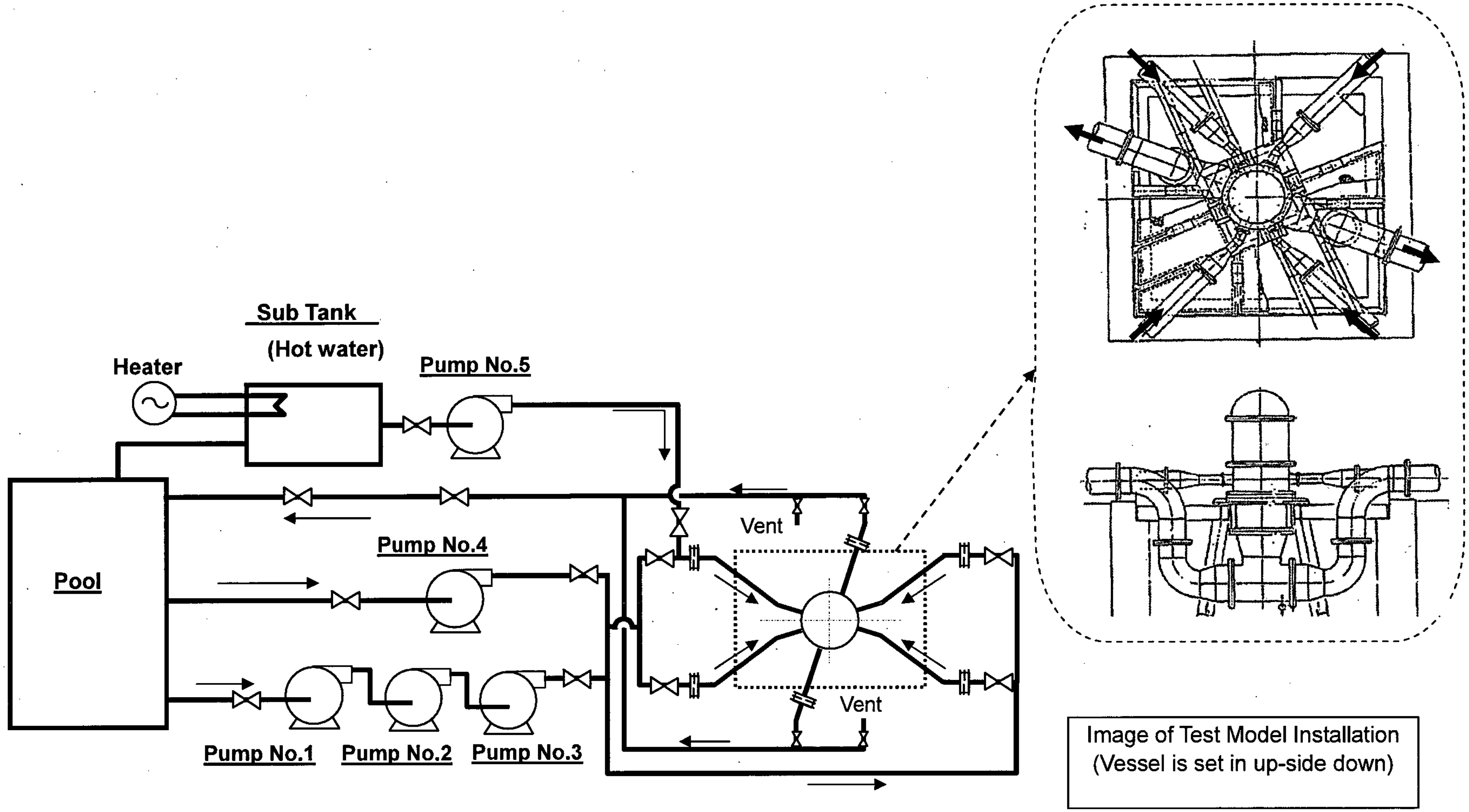


Figure 2 Schematic of Test Loop and Test Model

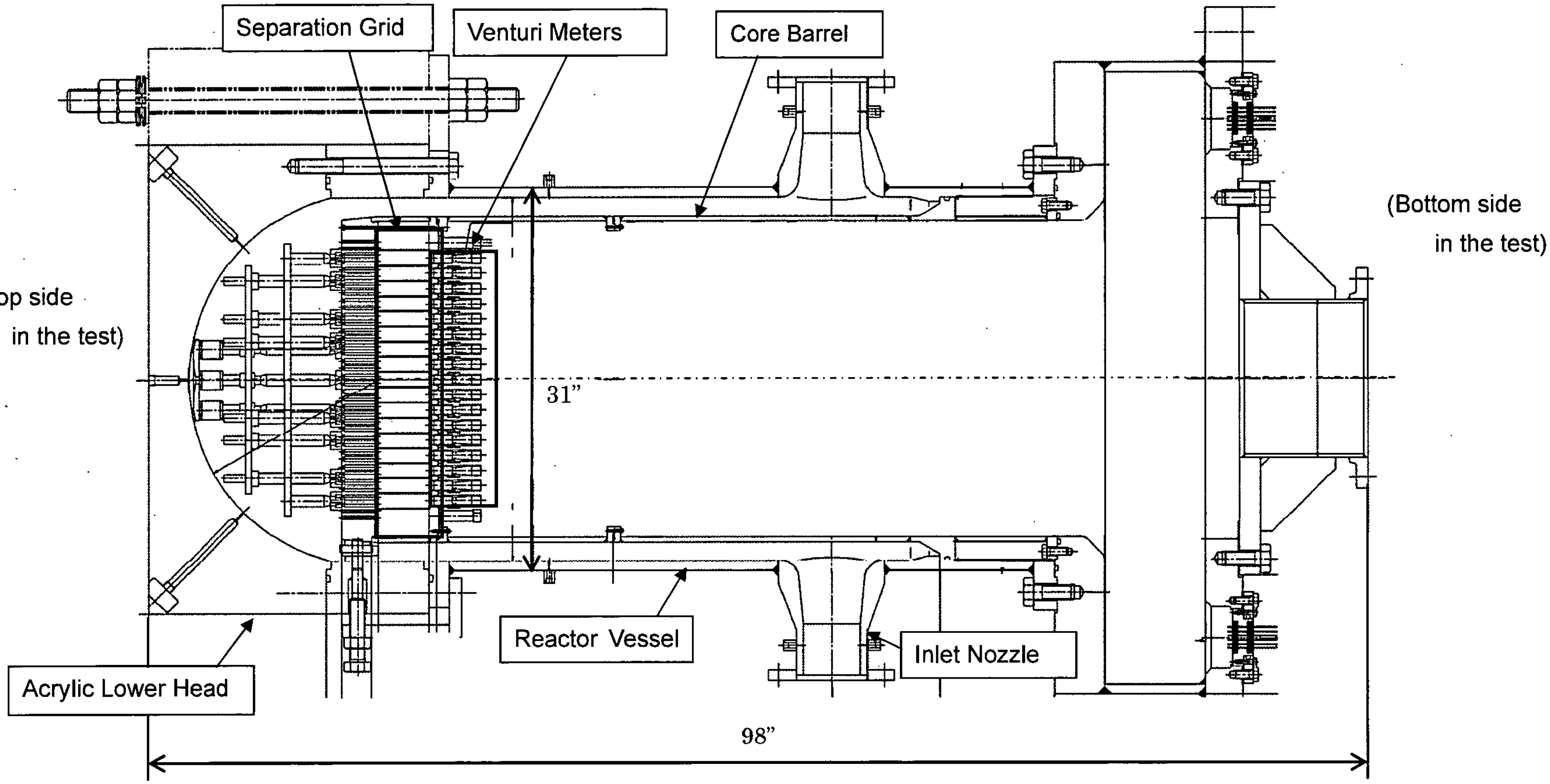


Figure 3 Model of Reactor Vessel and Lower Internals (1/7 Scale)