Information on the Advanced Plant Experiment (APEX) Test Facility

Curtis Smith

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Information on the Advanced Plant Experiment (APEX) Test Facility

The purpose of this report provides information related to the design of the Oregon State University Advanced Plant Experiment (APEX) test facility. Information provided in this report have been pulled from the following information sources:


In 2011, the INL produce a technical report on attributes for a “next generation” safety analysis tool called R7. As part of this report, information was provided on the APEX facility since it was used to demonstrate the capabilities of the software. The background on this use is:

- Leveraging on a unique experimental facility APEX, at the Oregon State University
- APEX simulate PWR (W and CE), AP600, and AP1000
- Quarter-scale integral-effect test facility, with extensive diagnostics
- Cross-benchmarking with existing models and “legacy” STH codes
- Developing input decks compatible with R7 graphics-user interface (GUI)
- A coarse-grained 0D-1D STH-grade model of APEX
- A fine-grained 3D CFD-grade model of APEX
- Analyzing past APEX experiments for their (partial) relevance to validation data needs in support of F&B scenario model calibration


General information was provided on APEX, including the following:

- **STEAM GENERATOR (SG)**: Each SG is instrumented and can be a future test article for a compact steam generator (CSG) module. The SG is a pressurized water reactor with a strong heat source and a high-pressure, high-temperature reactor coolant water. Its heat is generated by steam turbines in the steam generator.
- **CORE MAKEUP TANKS (CMTs)**: The CMTs provide coolant in case of reactor shutdowns, with drains into the reactor coolant pump (RCP) and core makeup tank (CMT) systems. They are equipped with emergency core cooling systems (ECCS) to cool the reactor core in case of a loss of coolant accident (LOCA).
- **REACTOR COOLANT PUMPS (RCPs)**: The RCPs provide coolant to the core makeup systems, allowing for reactor coolant to be circulated through the core makeup systems. The RCPs are designed to operate under normal and emergency conditions.
- **ACCUMULATORS (ACCs)**: The ACCs store coolant in case of an LOCA, ensuring the safety of the reactor core by providing emergency core cooling. They are connected to the core makeup systems and are equipped with emergency core cooling systems.
- **HOT LEGS (HLs)**: The HLs collect steam from the steam generators and transfer it to the steam headers, which in turn transport it to the main steam turbine. The steam is then condensed back into water.
- **COLD LEGS (CLs)**: The CLs collect condensate water from the steam generators and transfer it to the condensers, where it is cooled and returned to the steam generators.
- **DIRECT VESSEL INJECTION (DVI)**: The DVI system injects water directly into the reactor vessel, providing an emergency core cooling capability in case of an LOCA.
- **ELECTRICAL HEATER RODS**: The EHRs provide a means to heat the reactor core to maintain its integrity in case of an LOCA, preventing the release of radioactive materials into the environment.

**AUTOMATIC DEPRESSURIZATION SYSTEM STAGES 1-3 (ADS1-3)**

- **PRESSURIZER (PZR)**: The PZR is a high-pressure vessel that maintains the reactor coolant pressure above atmospheric pressure. It is equipped with relief valves to protect the reactor against overpressure.
- **IN-CONTAINMENT REFUELING WATER STORAGE TANK (IRWST)**: The IRWST is a tank designed to store water for in-containment refueling operations. It is connected to the reactor coolant system and is equipped with emergency core cooling systems.
- **PASSIVE RESIDUAL HEAT REMOVAL SYSTEM (PRHR)**: The PRHR is a passive system designed to remove residual heat from the reactor core in case of a loss of coolant accident. It uses the natural circulation of water to cool the reactor core.
- **AUTOMATIC DEPRESSURIZATION SYSTEM STAGE 4 (ADS4)**: The ADS4 is a pressurizer that releases excess pressure from the reactor coolant system, allowing the reactor to depressurize in case of an LOCA.
- **REACTOR VESSEL (RPV)**: The RPV is the reactor vessel that houses the reactor core and contains the primary coolant system. It is designed to withstand the pressures and temperatures generated by the reactor core.
The R7 input deck (once visualized) looked like:
This input deck was based upon an older RELAP deck.
Key components in the R7 deck were identified:
The INL also included additional details – this information was contained in the following four chapters:
Chapter 1

APEX Test Facility

1.1 APEX Test Facility Description

The Advanced Plant Experiment (APEX) test facility at Oregon State University is maintained and operated by the Department of Nuclear Engineering and Radiation Health Physics. It was constructed in 1994 to perform integral system tests to simulate the important thermal hydraulic behavior of the Westinghouse AP600 reactor. The facility was then modified, through a grant from the U.S. Department of Energy, to simulate the Westinghouse AP1000 for the purpose of AP1000 plant certification through assessing system code capabilities and integral system behavior.

The test facility models a complete Westinghouse AP1000 2x4 loop containing two hot legs, 4 cold legs, 2 steam generators, pressurizer, reactor pressure vessel with an electrically heated rod bundle and upper plenum internals. It is a 1:4 length scale, 1:2 time scale, 1:192 volume scale of the prototype AP1000 and is of stainless steel construction. It completely models the passive safety systems of the AP1000 containing: 2 core makeup tanks (CMTs), 2 accumulators (ACCs), a passive residual heat removal (PRHR) heat exchanger, in-containment refueling water storage tank (IRWST), and a 4-Stage automatic depressurization system. The steady state operating conditions of the facility are with a core power at about 1 MW, steam generator shell side pressure at 2 MPa (290 psig), and pressurizer pressure at 2.55 MPa (370 psig). Past testing efforts utilizing APEX have include hot and cold leg SBLOCAs, MSLB, Inadvertent ADS, Double-Ended DVI Line Break, Station Blackout and Long Term Recirculation. Figure 1.1 show a
Fig. 1.1: APEX test facility at Oregon State University.
Chapter 2

APEX R7 Simulation Model

2.1 Introduction

The physical APEX test facility is transformed into working model within R7 as shown in Figure 2.1. The following sections work to describe the systems, sub-systems (groups), and components as they are modeled in R7 GUI.
Fig. 2.1 : Model of APEX in R7.
### 2.1.1 APEX R7 Model Groups

The APEX test facility is broken up into groups for modeling purposes. Table 2.1 gives each group with a list of the major components of that group along with the fidelity utilized in the APEX R7 model.

<table>
<thead>
<tr>
<th>System</th>
<th>Group</th>
<th>Component Name</th>
<th>Dimension</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Pressure Vessel</td>
<td>Downcomer Fluid</td>
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<td>1-D/3-D</td>
</tr>
<tr>
<td></td>
<td>Downcomer Inner Wall</td>
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<td>3-D</td>
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<td>Downcomer Outer Wall</td>
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<td>0-D</td>
</tr>
<tr>
<td></td>
<td>Lower Plenum</td>
<td></td>
<td>0-D/3-D</td>
</tr>
<tr>
<td>Primary Loop</td>
<td>Core</td>
<td>Lower Core Plate</td>
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</tr>
<tr>
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<td>Heater Rod Section</td>
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<td>1-D</td>
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<td></td>
<td>Upper Core Plate</td>
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<td>1-D</td>
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<tr>
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<td></td>
<td>Hot Leg 2</td>
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...continued...APEX R7 Model Components.

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<th>System</th>
<th>Group</th>
<th>Component Name</th>
<th>Dimension</th>
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<td>Steam Generator 1 U-Tubes</td>
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<td></td>
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<td>0-D/3-D</td>
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<td></td>
<td>Hot Side Lower Plenum</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Steam Generator 1 Primary</td>
<td>0-D/3-D</td>
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<td></td>
<td>Cold Side Lower Plenum</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Steam Generator 2 U-Tubes</td>
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<td>Steam Generator 2 Primary</td>
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<td>Hot Side Lower Plenum</td>
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<tr>
<td></td>
<td></td>
<td>Steam Generator 2 Primary</td>
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<td>Cold Side Lower Plenum</td>
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<td>ADS 1-3</td>
<td>ADS 1 Valve (RCS-601)</td>
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<td>Oriface (ORI-655)</td>
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<td>ADS 1-3</td>
<td>ADS 2 Valve (RCS-602)</td>
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<td>Oriface (ORI-656)</td>
<td>1-D</td>
</tr>
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<td>ADS 3 Valve (RCS-603)</td>
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<td>Oriface (ORI-657)</td>
<td>1-D</td>
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<td>Oriface (ORI-659)</td>
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<td>CVS Piping</td>
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<td>Air Operated Valve (RCS-808)</td>
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<td>IRWST Tank</td>
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</tbody>
</table>
Chapter 3

Primary Loop Inputs

3.1 Core Downcomer

3.1.1 $R_7$ Model

Fig. 3.1: Core Downcomer.
3.2 Core Lower Plenum

3.2.1 R7 Model

Fig. 3.2: Core Lower Plenum.
3.3 | Core

3.3.1 | $R_7$ Model

Fig. 3.3: Core.
Fig. 3.4: Core.
3.4 Core Upper Plenum

3.4.1 $R_7$ Model

![Fig. 3.5: Core Upper Plenum.](image)

3.5 Cold Leg 1

3.5.1 $R_7$ Model

![Fig. 3.6: Cold Leg 1.](image)
3.6 Cold Leg 2

3.6.1 R7 Model

Fig. 3.7: Cold Leg 2.
3.7 Cold Leg 3

3.7.1 R7 Model

Fig. 3.8: Cold Leg 3.
3.8 Cold Leg 4

3.8.1 $R_7$ Model

Fig. 3.9: Cold Leg 4.
3.9  Hot Leg 1

3.9.1  R7 Model

Fig. 3.10:  Hot Leg 1.
3.10  Hot Leg 2

3.10.1  R7 Model

Fig. 3.11:  Hot Leg 2.
3.11 Steam Generator 1 U-Tubes

3.11.1 $R_7$ Model

Fig. 3.12: Steam Generator U-Tubes.
3.12 Steam Generator 1 Hot Lower Plenum

3.12.1 R7 Model

Fig. 3.13: Steam Generator 1 Hot Lower Plenum.
3.13  Steam Generator 1 Cold Lower Plenum

3.13.1  R7 Model

Fig. 3.14:  Steam Generator 1 Cold Lower Plenum.
3.14 Steam Generator 2 U-Tubes

3.14.1 R7 Model

Fig. 3.15: Steam Generator U-Tubes.
3.15  Steam Generator 2 Hot Lower Plenum

3.15.1  R7 Model

Fig. 3.16:  Steam Generator 2 Hot Lower Plenum.
3.16  Steam Generator 2 Cold Lower Plenum

3.16.1  R7 Model

Fig. 3.17:  Steam Generator 2 Cold Lower Plenum.
3.17 Pressurizer Surge Line

3.17.1 $R_7$ Model

Fig. 3.18: Pressurizer Surge Line.
3.18 Pressurizer

3.18.1 R7 Model
Chapter 4

Safety System Inputs

4.1 Automatic Depressurization System 1/2/3

4.1.1 R7 Model

Fig. 4.1 : Automatic Depressurization System 1/2/3
As noted in reference 2:

“The OSU Radiation Center (the location of the Oregon State University Department of Nuclear Engineering) houses a one quarter scale model of the Westinghouse Electric Corporation advanced light-water nuclear reactor design called AP600.

The AP-600 reactor design incorporates many passive safety features for reactor core cooling. In this case, passive means that the systems are capable of core cooling using only the phenomena of gravity driven flow and natural convection of heated fluids. The model of the AP-600 (APEX) was built to perform the testing necessary for design certification.

APEX operates at 2.76 MPa (400 Psia) and has been formally scaled to simulate the important thermal hydraulic behavior of the AP-600. APEX is electrically heated and simulates the nuclear steam supply system (NSSS) and all of the AP-600 safety systems. The systems modeled include the primary system, passive safety systems, the non-safety grade chemical and volume control system, and the residual heat removal system (PRHR).”

Additional details on the system shown above are found in the Reference 2 report.

Reference 3 explains some of the concepts behind the scaling approach used by the APEX researchers. While much of the technical information in Reference 3 has been redacted from the public document, some information is still presented, including:

![Graph showing comparison of AP1000 and APEX Axial Void Fraction Profiles for an Average Subchannel](image)

**Figure 18** Comparison of AP1000 and APEX Axial Void Fraction Profiles for an Average Subchannel
Figure 21 APEX Upper Core Plate Geometry

Reference 4 summarizes the research effort performed using APEX at Oregon State University. Some facility information is shown in the report, however the report notes:

“Three proprietary compact disks are available as a supplement to this report. They include APEX Facility Drawings, APEX Test Reports, and the APEX Database. These three compact disks include all of the information generated in support of NRC’s test program at OSU.”

The information from the compact disks is not publically available. Summary of information show in this reference includes:

**Layout of the APEX Test Facility**

Photos of key parts of the system shown above are included in the report. Information related to the testing initial conditions are also included.

This reference describes the NRC-sponsored tests that were run using the APEX facility during the 2003-2004 time-period. Limited dimensional information is provided.