1. INTRODUCTION

The Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research have cosponsored and jointly funded a cooperative containment research program at Sandia National Laboratories (SNL). NUPEC was founded in 1976 as the Nuclear Power Engineering Center under the initiative of academia and private corporations. Supported by the Agency for Natural Resources and Environment of the Ministry of Economy, Trade and Industry (METI), NUPEC is mandated to advance the performance and public acceptance of commercial nuclear power plants through engineering tests, safety analyses, information acquisition and analyses, and public relations activities. Within NUPEC, the Systems Safety Department is conducting research on the integrity of reactor containment vessels during severe accidents. Containment integrity tests include experiments and analyses of debris cooling phenomena, hydrogen combustion behavior, fission products transport behavior, and containment structural behavior. In addition, the department coordinates the cooperative containment program with the NRC and manages program activities with SNL and other subcontractors.

The Office of Nuclear Regulatory Research (RES) at U.S. NRC plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of safety issues for nuclear power plants and other facilities regulated by the NRC. Within RES, the Division of Engineering Technology (DET) plans, develops, and directs comprehensive research programs and standards development for nuclear and materials safety. In the nuclear safety area, there are programs for the design, qualification, construction, maintenance, inspection, and testing of current and advanced nuclear power plants. For materials safety, program activities include material characteristics, aging, and seismic and engineering aspects of these facilities and materials. Within DET, the Engineering Research Applications Branch has the lead for determining adequacy of structures and systems and for the coordinating and interfacing activities associated with the American Society of Mechanical Engineers (ASME) Code Section III. This branch coordinates the cooperative containment program with NUPEC and manages SNL activities.

SNL is a multi-program national security laboratory, operated by Sandia Corporation, a subsidiary of Lockheed Martin Company, for the National Nuclear Security Administration, U.S. Department of Energy (DOE). SNL's Nuclear Energy Technology Center has provided engineering and scientific support in the areas of reactor safety and safeguards to the NRC and the DOE for more than 20 years. A significant area of support has included analytical and experimental efforts to address issues related to severe accidents and containment integrity.

This cooperative containment program builds on the combined expertise of these organizations and continues to advance the understanding of nuclear containment structure's response to pressure loading beyond the design basis accident and the ability to predict, analytically, the structural behavior. This is accomplished by conducting static, pneumatic overpressurization tests at ambient temperature of scale models of actual containment vessels for nuclear power plants in Japan. NUPEC and the NRC formulated the overall scope of the program, and NUPEC, under contract with METI, is responsible for designing and constructing the models. SNL is funded by NUPEC to develop and operate a facility for conducting these tests, review the model designs and provide design support, instrument the models and collect data during the pressure tests, and report the results of the test. The NRC is funding SNL to perform pre- and posttest analyses of the models and to conduct the pressure tests. All funding is directed to SNL through agreements with the DOE's Work-for-Others Office in the Science and Technology Transfer Division.

Tests of two containment models were authorized under this program. The first model, a mixed-scale model of an Improved Mark-II type steel containment vessel (SCV) for a Boiling Water Reactor (BWR), was tested in December 1996. The results of the SCV tests and analyses have been published [1-5]. The second model tested was a 1:4-scale model of the prestressed concrete containment vessel (PCCV) of an actual nuclear power plant in Japan, Ohi-3 (Figure 1.1). Ohi-3 is an 1127 MWe Pressurized Water Reactor (PWR) unit, one of four units comprising the Ohi Nuclear Power station located in Fukui Prefecture and owned and operated by Kansai Electric Power Company.

This report describes the design, construction, and instrumentation of the PCCV model, the conduct of the pressure tests, and the results of those tests. The pre- and posttest analyses performed by ANATECH Corp (San Diego, CA) under contract to SNL are reported separately [6, 7]. Independent pretest analyses, conducted by a number of international organizations, were also conducted and presented in a summary report [8].

Figure 1.1. Ohi Nuclear Power Station, Ohi-cho, Fukui, Japan

1.1 Background

Containment vessels in nuclear power plants comprise, with the penetrations and other pressure boundary components, the final barrier between the environment and the nuclear steam supply system. The functions of the containment are to:

- contain any radioactive material that might be released from the primary system (reactor vessel, steam generators, piping) in the event of an accident;
- act as a supporting structure for operational equipment.

Containment buildings have been an integral part of commercial nuclear power plants in Japan and the United States since the first units were constructed in the 1960s. For U.S. containments, the design loads and their combinations, as well as the response limits, are specified in the ASME Boiler and Pressure Vessel Code [9]. Initially, severe accidents were not part of the design basis due to their perceived low probability of occurrence, and pressure relief valves were not required. In Japan, METI Directives control the design of nuclear power plants, and the design standards for containments are specified in the METI Notification No. 501 and in JEAG4601.

After the accident at Three Mile Island in the United States in 1979, attention turned to the capacity of containment systems beyond their design basis. SNL conducted a preliminary study [10], commissioned by the NRC, to identify experiments conducted to investigate this issue, but concluded that the scope of the tests and the data did not provide sufficient insight into the problem. As a result, a program, including scale model tests coupled with detailed structural analysis, was formulated by the NRC to investigate the integrity of containment systems beyond their design basis. The primary objective of the NRC program was, and continues to be, the validation of analytical methods used to predict the performance of light water reactor containment systems when subjected to loads beyond those specified in the design codes. While some insights could be gained into structural response and failure mechanisms of actual containments, it was also recognized that the capacity of actual containments could not be determined solely from tests of simplified scale models. The results of this program, as summarized by Parks [11], concluded that there was significant reserve capacity in the containment vessels to resist loads above the design basis and that although the analytical efforts were encouraging, uncertainties remained about structural response and failure mechanisms.

Remaining uncertainties regarding the response of containment structures led to discussions among NUPEC, the NRC, and SNL that culminated in a 1991 agreement to start the NUPEC/NRC Cooperative Containment Program. In parallel with this cooperative program, there are independent efforts sponsored and conducted by both NRC and NUPEC. These efforts include investigating the response of penetrations [12,13], the effects of aging on containment structure capacity [14], and the seismic capacity of containment structures [15, 16].

1.2 Scope

Nuclear power plants in Japan and the U.S. generally utilize one of two types of light water reactor systems; BWR and PWR. The containment vessels for the pressurized water reactors in Japan and the U.S. are typically free-standing reinforced concrete shells with an integral steel liner. A few have only regular steel reinforcing bars (rebar); however, the majority use both regular and posttensioned reinforcing. (For this report, the terms prestressed and posttensioned are used synonymously, even though the reinforcing is, technically, posttensioned; i.e. tensioning of the reinforcing is conducted after the concrete has been placed and cured to the specified minimum strength.) A variety of prestressed reinforcing or tendon configurations are represented in the fleet of PWR containments. However, the evolution of prestressed containment designs has been toward the use of longer, continuous tendons, culminating in the two-buttress containment with meridonal 'hairpin' tendons and 360-degree hoop tendons, represented by the Ohi-3 design. No two-buttress prestressed concrete containments were constructed in the U.S. (although some were planned prior to the TMI-2 accident); however, many of the features of the Ohi-3 containment are similar to features in existing U.S. plants and the design philosophy is similar. As a result, NUPEC and the NRC agreed on a scale model of the Ohi-3 containment for the second test subject in the Cooperative Containment Program.

1.2.1 Model Features and Scale

The Ohi-3 containment is a thin prestressed concrete cylindrical shell with a hemispherical dome and a continuous steel liner anchored to a reinforced concrete basemat that extends beyond the containment to support other plant structures. Consistent with the objectives of the sponsoring organizations, the features and scale of the PCCV model were chosen so that the response of the model would mimic the global behavior of the prototype, and local details, particularly those around penetrations, would be represented. One of the primary considerations in determining the scale of the model was the desire to utilize nearly identical construction materials to the material used in the construction of the prototype. Preliminary design studies, conducted to determine the appropriate scale of the model, initially focused on a mixed scale model where the scale on the overall geometry would be 1:6, while the scale on the liner thickness would be 1:3. These preliminary studies indicated, however, that use of this mixed scale might upset the relationship between failure modes that might be expected in the prototype. In particular, the use of a steel liner, which was twice as thick, relative to the prestressed concrete shell, as the prototype, might retard the onset of liner tearing (leakage) failure modes and increase the likelihood of a structural failure mode occurring. As a result, it was decided that the scale of the model would be a uniform 1:4, with minor exceptions to accommodate fabrication and construction concerns. This was judged to be the minimum scale that would allow the steel liner to be constructed from prototypical materials and fabricated with details and procedures representative of the prototype. The overall geometry and dimensions of the PCCV model are shown in Figure 1.2.

Although both NUPEC and SNL (under NRC sponsorship) had conducted component tests of both full-size and scaled penetrations [12-13, 17], the PCCV model included both a functional representation of the major penetrations, namely the equipment hatch (E/H) and the personnel air lock (A/L), and nonfunctional representation of the main steam (M/S) and feedwater (F/W) penetrations. The E/H and A/L penetrations were fully-functional, one-fourth scale models of the penetrations in the prototype, while only the penetration sleeves of the M/S and F/W penetrations, terminated with pressure seating blind flanges, were included in the model. The liner and concrete reinforcing details around these penetrations were also retained in the model.

During construction and instrumentation of the model, primary access to the interior was through the E/H, while the A/L was used to provide heating, cooling, and ventilation for personnel working inside the model. The M/S and F/W penetrations provided portals for interior instrumentation cabling, power and, during testing, the pressurization medium. Prior to testing, after the E/H cover was installed and sealed, the A/L provided the means for final egress and sealing of the model with a specially-designed pressure seating cover that could be closed from the outside.

Details of the design and fabrication of the PCCV model are described in Chapters 2 and 3.

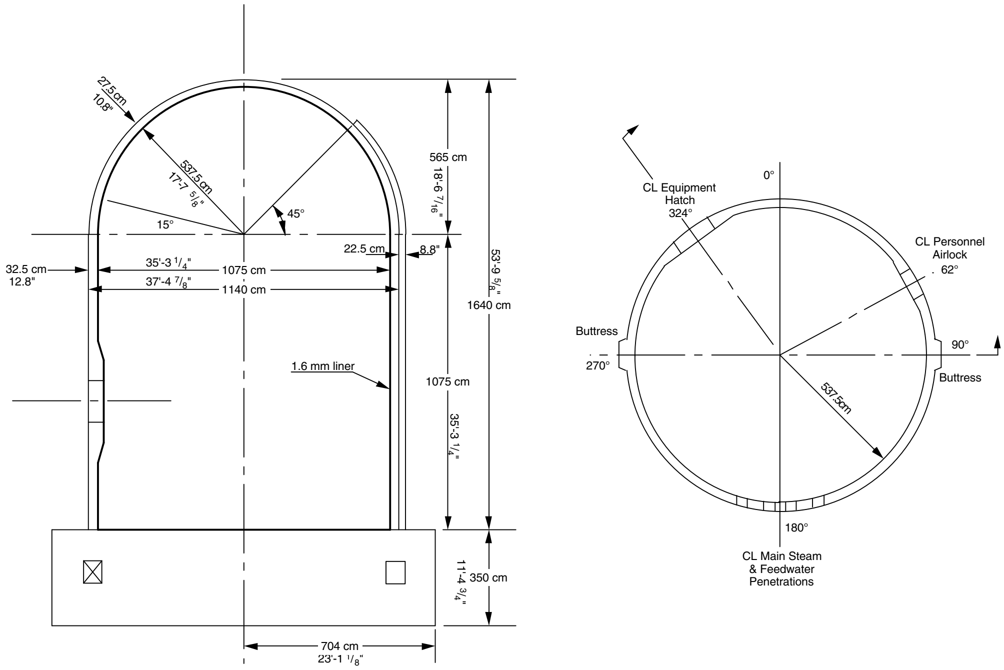

Figure 1.2 PCCV Model Elevation and Cross-Section

1.2.2 Loading

By definition, the scope of this program was limited to addressing the capacity of containment vessels to loads beyond the design basis, the so-called severe accident loads. Design accident loads for light water reactor containment vessels are typically based on the loss-of-coolant accident (LOCA) and are defined by a “bounding” pressure and temperature transients. The term “severe accidents” is used to describe an array of conditions that could result in loads exceeding the design basis on the containment. The definition of severe accident loads, which is not as rigorous as the design basis loads definition, results from considering of various postulated failure scenarios of the primary nuclear system, up to and including a complete core meltdown and breach of the reactor pressure vessel. The resulting pressure and thermal loading characteristics depend on the unique features of the nuclear steam supply (NSS) system and the containment structure, in addition to the postulated accident.

For this test program, it was necessary to decide whether both thermal and pressure loads would be applied to the model, either separately or simultaneously; what the pressurization medium should be; and whether the transient characteristics of these loads should be considered. Programmatically, the decision to perform a *static, pneumatic* overpressurization test at *ambient temperature* was dictated by risk and cost considerations and previous experience.

The effects of severe accident *temperature* loads on the structural response of the containment building are primarily limited to (1) the effects of elevated temperatures on the mechanical properties of the materials and (2) the mechanical loads resulting from differential or constrained thermal expansion. The effects of temperature on the material properties can be determined from standard material tests methods. These test results could be incorporated into the evaluation of the prototypical containment vessels without adding this complexity and cost (in terms of generating the thermal environment and protecting the instrumentation) to the PCCV model test. Regarding the stresses imposed by differential thermal expansion, there are only a few locations in a steel and/or concrete containment building where these effects are significant, notably at the junction of the containment wall and the basemat or, in the case of the PCCV model, the differential thermal expansion between the steel liner and the concrete shell under non-steady-state thermal conditions. Again, the added complexity and cost of simulating the thermal environments to reproduce these local effects was judged not justified for the PCCV model. It was further concluded that the effects of temperature could be addressed using analytical methods that had been benchmarked against the pressure tests. Therefore, the decision was made to conduct the PCCV model test at *ambient temperature*.

The containment atmosphere during a severe accident consists of air, steam, and other by-products of the accident, including hydrogen and particulates (aerosols). The program's primary interest is in observing and measuring the structural response of the containment to pressure loads, and identifying failure modes. Containment failure (see Section 1.2.3) includes both functional failure, i.e. leakage, and structural failure, i.e., rupture of the pressure-resisting elements. There is not a rigorous distinction between functional and structural failure, and it is conceivable that they might occur simultaneously. Conventional wisdom holds, however, that local, limited structural failure (i.e. liner tearing) and leakage will occur prior to, and at pressures well below those required to cause extensive structural failure. As a result, *detection* of leakage, which indicates a tear in the steel liner or failure of a penetration seal, not *measurement* of actual leak rates for real containment atmospheres (see Section 1.2.3), is the objective of the test. Hence, there is no need to reproduce the containment atmosphere resulting from a severe accident. The choice of a pressurization medium, then, becomes somewhat arbitrary and is dictated by safety and operational considerations. Hydrostatic testing is preferable from a safety viewpoint; however, it raises operational problems and requires protection of sensitive electronics and wiring from the water under high pressure. *Pneumatic* testing, while more dangerous, does not present any risks that cannot be managed cost-effectively and does not require any unusual measures to protect the instrumentation. Nitrogen gas was chosen as the pressurization medium for the PCCV model tests primarily for operational considerations. Fairly large quantities could be delivered at the test site in liquid form with a limited amount of fixed equipment. Nitrogen gas also has the advantage of being dry for instrumentation considerations, and it allows simpler and more accurate calculations to detect a small leak.

The test plan and conduct of the pressure tests, along with the design and operation of the pressurization system, are described in Chapter 5.

It should be noted that the *pneumatic* Limit State Test (LST) was the final test in the original program plan. This test was terminated following a functional failure, i.e. a leak, in the PCCV model, with only limited structural damage occurring. Subsequently, it was decided to repressurize the PCCV model, prior to demolition, in an attempt to observe larger inelastic response and, possibly, a global structural failure. This test was a combined *pneumatic-hydrostatic* test, where the PCCV model was sealed inside with an elastomeric membrane and filled nearly full with water to reduce the volume of gas to be pressurized, and nitrogen gas was used to generate the overpressure. The rationale and design of this Structural Failure Mode Test (SFMT) are also described in Chapter 5.

1.2.3 Response

One important aspect of the PCCV model response in the high pressure tests is the concept of *failure*. In the U.S., the functional failure for the prototypical containment is defined in the regulations as containment leak rates exceeding 0.1 to 0.5% of the containment mass per day [18], considering maximum offsite dose rates due to fission product released to the environment. In Japan, the functional failure is defined in design specifications made by the utility company, not the regulations. (The specified leak rate for the PCCV prototype is 0.1% mass/day.) The functional failure criteria are not particularly useful to test the structural capacity of a containment vessel model, especially when one of the objectives is to generate large inelastic response modes for comparison with analytical predictions, which may be well beyond the levels required to cause functional failure; and secondly to gain some insight into design margins, i.e. the functional and structural capacity beyond the specified design load conditions. In the case of the PCCV model test, the pressurization system allows the model to be pressurized to levels significantly above those expected to cause local strains in the model to exceed the ultimate strain limits of the materials. The test(s) were terminated when the model and the pressurization system were incapable of maintaining or increasing the model pressure due to excessive leakage or gross rupture. In this report, the maximum pressure achieved prior to the termination of the tests will <u>*not*</u> be identified as the <u>*failure*</u> *pressure*, since failure is defined in terms of some acceptance criteria, not the operational inability to maintain pressure in the model.

The PCCV model instrumentation suite was designed to measure the global behavior in free-field locations of the model and the local structural response of the model near discontinuities. Global response measurements included both displacements referenced to a global or fixed reference, and strain measurements at a regular pattern of azimuths and elevations to characterize the overall shape of the model. Local response measurements consisted of individual structural element (i.e. liner, rebar, tendons, concrete) strain measurements to characterize the force distribution in the free field and near structural discontinuities. In areas without structural discontinuities or where membrane behavior was expected to dominate the response, relatively simple arrays of transducers were specified. Where structural discontinuities were judged to be significant, more complex arrays of strain gages were utilized. Both hoop and meridonal strains were measured.

Pressure measurement requirements included careful measurement of the PCCV interior pressure for leak detection (to a lesser extent); leak rate measurement; characterization of the mechanical response as a function of pressure; and controlling the pressurization rate. Note that while measurement of leak rates was not a primary objective, detecting the onset of leakage requires calculating very small leak rates with relatively high accuracy.

While there was no attempt to simulate severe accident temperature conditions, a fairly extensive set of thermal measurements were taken to measure both the interior and exterior atmospheric temperature for accurate leak rate calculation. Given the large volume of the PCCV model, gas temperatures inside the model could vary significantly and multiple measurements were required to limit errors resulting from nonuniform gas temperatures. During pressurization steps, large thermal gradients could occur as the gas inside the model was compressed. Furthermore, since the model was exposed to the environment, ambient thermal variations, both spatial and temporal, affected the interior gas temperature and could affect the accuracy of the leak rate calculations if not considered. Similarly, ambient thermal effects could affect the model response measurements. Multiple measurements of the model temperature using both embedded and surface mounted temperature transducers were employed to account for this effect.

As implied by the name, the unique feature of the PCCV model is the prestressing system, comprised of the vertical and hoop tendons and associated hardware. Special efforts were made to monitor the response of the prestressing system, both prior to and during pressure testing. An extensive effort was undertaken to develop and demonstrate the reliability of the tendon instrumentation. The resulting system was comprised of two types of gages to monitor the strain, and, by calculation, the force distribution along the length of selected tendons along with load cells to measure the forces at the tendon anchors. Since the behavior of the tendons and the overall response of the model to the pressure load would be directly affected by the initial prestressing forces, the response of the PCCV model was monitored continuously from the start of prestressing through the subsequent pressure tests.

While these force, strain, and displacement measurements provide accurate information on the response of the model at discrete locations, it is desirable to monitor the overall response of the model in the (likely) event that some significant response occurs at locations remote from any transducer. The displacement transducers reflect, to a greater extent than the strain or force transducers, the overall response of the model, but might still miss other local response modes. This deficiency was addressed by including an extensive array of acoustic and, to a lesser degree, video/photographic monitoring of the PCCV model. While more qualitative in nature than the discrete response measurements, some quantitative information could be obtained from these monitoring systems. The acoustic system, in particular, was designed to detect the onset of liner tearing and leakage, along with concrete cracking and rupture of tendon wires or rebar. Similarly, video and still photography were used to document the development and distribution of concrete cracking, detect liner tearing at discrete locations during pressure testing, and capture any unanticipated response modes.

The design and implementation of the model instrumentation suite are described in Chapter 3. Performance requirements and features of the data acquisition system and data management are summarized in Chapter 4. A summary and discussion of the high pressure tests and posttest inspections are provided in Chapter 5. The test results are also summarized in Chapter 5 and the corrected test data, including a description of the corrections applied to the raw data, are included in the appendices.

1.3 Project Organization

As noted above, NUPEC and the NRC are the sponsoring organizations for this cooperative containment research program. Programmatic authorization to pursue this area of research is provided to these organizations by the ministerial or executive offices of their respective national governments, as dictated by statute. Technical guidance was provided by panels of expert advisers from academia and industry in each country. In Japan, the Structural Advisory Committee met regularly with NUPEC personnel to review the program plans and status, while in the U.S., a special Peer Review Panel provided the same support to NRC and SNL personnel.

Within the cooperative framework agreed to by NUPEC and the NRC, NUPEC and its Japanese contractors designed and constructed the PCCV model at SNL's Containment Technology Test Facility-West (CTTF-W). This test facility was specially constructed by SNL on land temporarily permitted for this purpose by Kirtland Air Force Base (KAFB), Albuquerque, New Mexico, USA. This 'West' facility is distinct from the CTTF used for the previous large-scale model tests conducted for the U.S. NRC in the 1980s. The 'East' facility was not considered suitable for continued large-scale model testing due to the identification of previous environmental contamination (not associated with the containment test operations) and subsequent clean-up operations that might interfere with the Cooperative program operations. The CTTF-West Land-Use Permit required NUPEC and the NRC, through their contracts with SNL, to remove all improvements within the permit boundaries and return the site to near its original condition at the conclusion of all test operations.

NUPEC and its Japanese contractors were authorized to construct the model at the CTTF-W under a specially negotiated Premise Access Agreement with SNL and the DOE. This agreement required NUPEC and its contractors to abide by all environmental health and safety regulations typically required for all capital construction activities managed by SNL, and authorized SNL to perform construction safety inspection to ensure that all requirements were being satisfied. The prime contractor to NUPEC for the construction of the PCCV model was Mitsubishi Heavy Industries (MHI), who also designed and constructed the prototype plant, Ohi-3. In addition to overall design and construction, MHI designed, fabricated, and erected the steel liner and all primary steel pressure-retaining components. Supporting MHI for the reinforced concrete portions of the model and ancillary structures were several subcontractors. Obayashi Corp., a large Japanese Architect/Engineer (A/E) and construction company, performed the detailed design of the PCCV model, and Taisei Corp, another large A/E/Contractor, was the construction manager. Taisei retained the U.S. construction firm, Hensel Phelps Construction Co., Greely, CO for general construction work and management of day-to-day construction operations. MHI prefabricated portions of the steel liner and the penetrations at their Kobe Shipyard and transported these components to the CTTF-W for final erection. The balance of the model was constructed on-site.

NUPEC also funded SNL to provide programmatic and model design support, instrument the model, and design and assemble the data acquisition system.

NRC funded SNL to perform preliminary, pre- and posttest analyses of the model. This analytical work was subcontracted by SNL to ANATECH Consulting Engineers, San Diego, CA. The decision to subcontract this work to ANATECH was based, in part, on a successful history of collaboration on previous containment model tests [19, 20] and ANATECH's experience in developing sophisticated concrete models and related efforts for the Electric Power Research Institute (EPRI), Palo Alto, CA [21]. The preliminary analyses supported design studies, identified critical response modes, and assisted in the locating instrumentation. The pretest analysis consisted of developing and analyzing detailed numerical models in an attempt to predict the response of the PCCV to the test pressures and predict the capacity and most probable failure mode. The posttest analysis compared the test results to the pretest predictions, investigated and demonstrated changes in the modeling methods to improve comparison with the test results, and provided insights into the response observed during the pressure tests. The pre- and posttest analyses are reported separately [6,7] and are not included in this report.

NRC also funded the planning and conduct of test operations.

NUPEC and NRC also jointly provided funding to share the costs associated with organizing and conducting a pretest Round Robin analysis. The Round Robin analysis euphemistically refers to an activity where a number of nuclear safety research organizations from government, industry, and academia in the U.S., Japan, and other countries, are provided with a common set of data on the model test (design drawings, material properties, test specifications, etc.) and complete independent predictions of the model response, failure mode, and pressure capacity. SNL was the focal point for this effort in terms of disseminating and consolidating the work of the participating organizations. Seventeen independent organizations, including NUPEC and SNL, participated in this effort, performing pretest analyses and meeting before and after the PCCV model test to discuss and compare analysis results. The efforts of these Round Robin participants are documented in separate NUREG Contractor Reports [8]. While a formal posttest Round Robin exercise was not conducted for the PCCV, most of the participants attended a posttest workshop and have reported the results of their posttest analyses independently.

Regular Technical Working Group meetings were held in both Japan and the U.S., involving program personnel from NUPEC, (including its contractors), the NRC, and SNL. These meetings planned and coordinated program activities and resolve technical issues. Separate meetings were held to discuss administrative issues related to cost and schedule.

1.4 Project Schedule

The NUPEC/NRC Cooperative Containment Research Program commenced in June 1991. The tests were conducted at the CTTF-W at SNL. Figure 1.3 illustrates the layout of the test site. A safety zone consisting of a circular area with

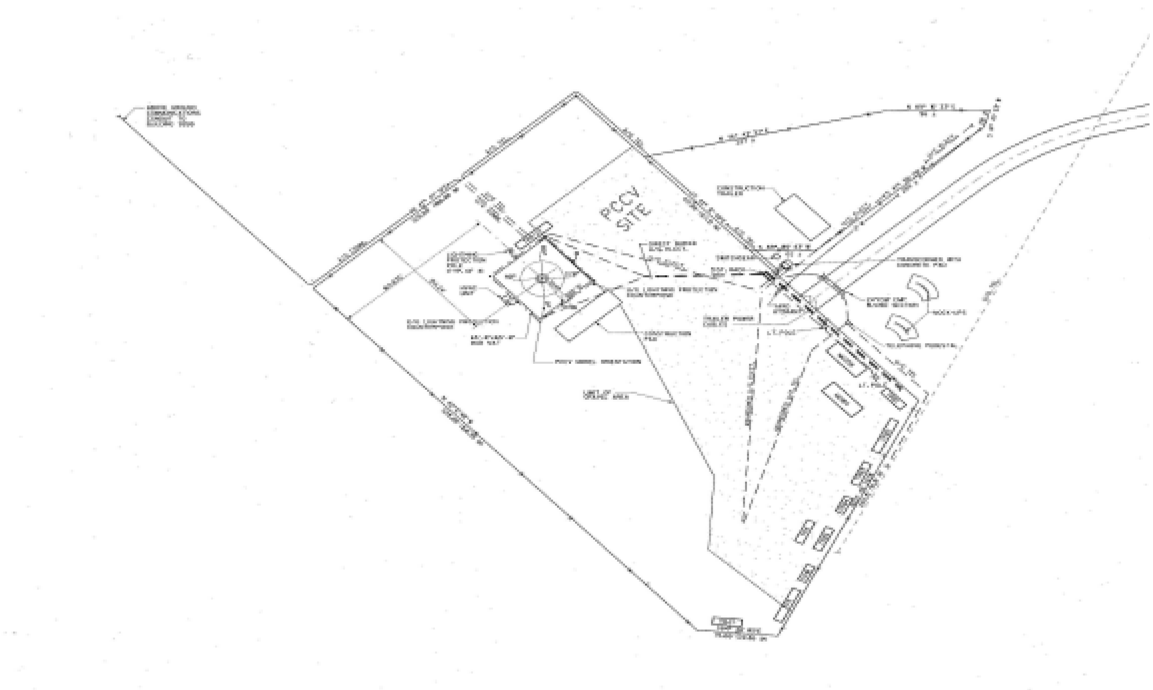

Figure 1.3 Plan of Containment Technology Test Facility-West

radius of 600 m (2000 ft) was maintained and monitored throughout the high-pressure test. The command center in Building 9950, located outside the exclusion zone, served as headquarters for conducting the high-pressure tests.

The high-pressure test of the SCV was completed on December 12, 1996. Construction of the PCCV model commenced January 3, 1997 with initial site preparation. Milestones in the construction and testing of the PCCV model included the following:

- 12 February 1997; First Basemat Pour (F1)
- 19 June 1997; First Liner Panel Installed
- 15 April 1999; Final Dome Pour (D3)
- 12-14 October 1999; Pretest Round Robin Meeting
- 8 March-3 May 2000; Prestressing
- 25 June 2000; PCCV Construction Completed
- 12-14 September 2000; Structural Integrity and Integrated Leak Rate Test
- 27-28 September 2000; Limit State Test
- 22 August 2001; Posttest Round Robin Meeting
- 14 November 2001; Structural Failure Mode Test
- 3 May 2002; PCCV Demolition and Site Restoration Completed

With the completion of the PCCV tests, restoration of the test site, and publication of the test reports, the NUPEC/NRC Cooperative Containment Research Program was formally concluded on December 31, 2002.