Appendix-E

Severe Accident Analysis Report for Containment Performance

TABLE OF CONTENTS

LIST OF FIGURES

1.0 Introduction

The potential failure of a nuclear power plant containment due to pressurization caused by the generation of steam and non-condensable gases has been recognized as a possibility by the nuclear regulatory agencies and the nuclear industry worldwide. The concern is that decay heat generated in the core may not be able to be transferred to the Ultimate Heat Sink and may be deposited in containment. If containment heat removal is unavailable, the containment atmosphere will be heated and pressurized. The increased temperature and pressure can challenge the ability of containment to act as a barrier against the release of fission products.

United States Nuclear Regulatory Commission (USNRC) regulations establish requirements for containment performance as listed below:

Title 10 of the United States Code of Federal Regulations (CFR) Part 50 Appendix A General Design Criterion 50 states that the containment structure should accommodate “without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.” Containment structures designed in accordance with American Society of Mechanical Engineers (ASME) Code requirements have margin above the design pressure. Revision 3 of the Standard Review Plan (NUREG-0800) Sections 3.8.1 and 3.8.2 specify that for a license application to be accepted for review, it must include an explicit demonstration that the containment internal pressure capacity significantly exceeds the design-basis accident pressure.

The USNRC has established performance goals for containments under severe accident conditions. The Staff Requirements Memorandum (SRM) to SECY-90-16 (Reference 1) approved the use of a conditional containment failure probability of 0.1 as a basis for regulatory guidance on containment performance. Subsequently, the SRM to SECY-93-087 (Reference 2) stated:

> *The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) for approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.*

USNRC Regulatory Guide (RG) 1.216 (Reference 3) specifies the methodology for determining “more likely severe accident challenges” as selecting sequences or plant damage states that represent 90 percent or more of the core damage frequency and determining bounding pressure and temperature loadings based on the pressures and temperatures identified for these sequences. These pressure and temperature loadings can then be used to analyze the containment to determine whether the acceptance criteria have been met. Taken together, this imposes the requirement that concrete containments should meet the Factored Load Category requirements of ASME Code, Section III, Division 2.

The objective of this report is to determine if the APR1400 plant will satisfy the requirements related

to the containment integrity under the more likely severe accident challenges. Section 2.0 presents a review of past experimental and analytical investigations of the failure modes of large dry containment structures. Section 3.0 describes the containment performance criteria for the APR1400 containment. Section 4.0 presents the containment performance analyses during severe accidents. Section 5.0 presents the conclusions drawn in this report.

2.0 Review of Containment Failure Phenomenology and Experiments

2.1 Containment Failure Phenomenology

Containment overpressurization is a postulated event in which the pressure loads applied to the containment boundary during a severe accident eventually exceed the boundary's ultimate strength at its most vulnerable point(s). This event has been hypothesized as a means of containment failure through one or more of several potential physical mechanisms. The extent of pressurization, its timing, and the pressurization rate all depend on a number of factors, including the accident sequence characteristics involved, the containment geometric configuration, etc. At the heart of the matter however, are the needs to define the containment's pressure limits and determine how much pressure the containment will undergo during severe accidents.

2.1.1 Controlling Physical Processes

Several controlling physical processes have been postulated that can be considered relevant to containment overpressurization in severe accidents. Such processes result in either heating the gas and/or vapor mass in the containment's finite volume, or increasing the gas/vapor mass existing in the finite volume. These processes, or pressure sources, include potential ex-vessel vapor (steam) explosions, combustion, core-concrete interaction, direct containment heating (DCH) and steaming of primary system and water in the reactor cavity. Processes that exclusively involve heating of the existing containment volume include gas combustion and direct heating of the containment atmosphere by finely fragmented core debris. Over the course of a severe accident, pressure sources may participate individually or in combination.

Steam produced within or external to the primary system is another potentially influential pressure source in some severe accidents. Steam addition to the containment occurs due to either flashing of effluent from primary system ruptures or rapid vaporization of water in the reactor cavity by interactions with core debris ejected from the vessel. This phenomenon and ex-vessel steam explosions are pressurization mechanisms that originate as containment mass addition processes only. Following these initial stages, steam added to the containment through these sources becomes a containment volume heating mechanism. Concrete attack by molten core debris represents a situation where heat and mass are more or less simultaneously added to the containment volume. Chemical reactions that may occur in such cases directly release heat as well as contribute significant masses of aerosols and non-condensable gases, which in their turn also become containment volume heating mechanisms.

2.1.2 Containment Failure Mechanisms and Modes

Containment overpressurization can potentially result in early or late failure modes. Depending on the specific accident sequence characteristics, overpressurization failures may be observed across a wide range of event times, either substantially before or substantially after vessel failure. Apart from direct bypass or failure to isolate events (where containment pressure retention capability is assumed to fail by definition), the potential for containment overpressure failure exists in most severe accident scenarios where pressure suppression facilities, namely the normal containment sprays and/or the Emergency Containment Spray Backup System (ECSBS), are disabled.

The failure mechanism associated with overpressure of the containment is due to exceeding the ultimate strength of certain structural components or attachments. Efforts to precisely and confidently characterize this mechanism can be (and have been) extraordinarily complicated for severe accident purposes. There are four fundamental considerations; 1) failure flow sizes (areas), 2) failure locations, 3) failure timing, and 4) the pressure levels at which failures may occur. As discussed below, some aspects of the containment failure modes do not depend, to an important degree, on the physical process (or combination of processes) that causes overpressurization. Failure location and the pressure levels at which failure may occur depend upon the containment design, construction and materials, but not on the process creating the pressurization. Containment failure timing, i.e., early or late, and failure size, on the other hand, do depend on the physical process and the rapidity of the pressurization.

Regarding failure sizes, the ultimate strength of involved structures may be achieved rapidly in some postulated sequences, and the total energy delivered (to the containment) may be sufficient to result in large rupture areas. Conversely, this limit may be approached gradually, and the energy delivered may only be enough to induce relatively small rupture areas unlikely to become any larger than what is necessary to stabilize the containment pressure (i.e, create a choked flow condition). Based on the results of the APR1400 for DCH, steam explosions, and combustion, cases in this category are generally considered to be significantly more likely than rapid loading, high energy cases. Potential examples may include concrete attack events or steaming to the containment.

Although there are innumerable containment details that could fail under over-pressurization conditions, there are only a few possible overpressure failure locations that need consideration with respect to source term estimates. For a large dry containment, a failure location creates a flowpath to either the auxiliary building, the atmosphere or into the soil beneath the basemat. Each of these flowpaths has different fission product retention mechanisms, which results in varying source terms. For a direct release to the atmosphere, fission product retention occurs in the containment. In this case, most of the noble gases and some of the volatile fission products would escape into the environment. Some amount of fission product retention can be credited to the auxiliary building, depending upon its configuration and at what elevation the containment failure occurs. A failure of the containment that would result in a release into the soil (i.e., basemat failure) would have to consider soil retention mechanisms as well as effects of the failure occurring below the containment water line.

Factors that control overpressure failure timing are somewhat more defined than factors associated with size and location variables. In this area, the pressurization rates occurring in the containment regions during a severe accident exert the greatest influence on the time when a failure can be

expected. Pressurization rates are themselves controlled by the pressure source phenomena characteristics, the containment boundary conditions when the phenomena originate, and containment physical characteristics (region geometry in particular). A wide variety of efforts have been accomplished to analytically model the physical bases for how the phenomena actually generate pressure. For the most part, these models are sensitive to crucial boundary conditions and plant characteristics. Furthermore, the pressurization rates revealed in development of these models appear to fall within a relatively low range regardless of pressure source phenomena distinctions. This strongly suggests that loading of the containment shell and other pressure boundary components can be considered essentially static rather than dynamic. On this basis, overpressure failure timing is reduced to a matter of the rapidity at which actual pressures reach the pressure retention capacities (i.e., the pressure point corresponding to ultimate strength stress levels) of containment boundary components.

2.1.3 Relationship to Source Term

The three characteristics of containment overpressurization failures which most strongly influence the fission product source term are, 1) failure timing, 2) failure location, and 3) failure size.

A late containment failure provides ample time for natural fission product settling mechanisms to take effect, thus reducing the airborne fission product concentration and the source term.

The effect of different failure locations has been discussed in the previous section. Since the cylinder-basemat junction is located below ground level and would be submerged by water during a severe accident, fission product releases resulting from a failure at this location or in the base slab would be inhibited. The very effective soil and water pool scrubbing mechanisms which would be encountered along these failure paths would significantly reduce the fission product source term.

The remaining failure locations would not have the benefit of fission product retention in overlying water pools and soil beds but would be influenced by several other factors. For instance, failures at the access ways would create a path to the auxiliary building which would then act as an additional fission product barrier. Failures of the reinforced concrete cylinder, on the other hand, may not lead to the auxiliary building but directly to the environment. A failure path directly to the environment would have a minimal number of fission product retention mechanisms and an increased source term. Finally, fission product releases would be different depending on whether the containment failure occurred in the upper compartment wall or the annular compartment wall. Due to the containment gas circulation paths, the airborne fission product concentration would be larger in the upper compartment than it would be in the lower compartment. Thus, the most conservative containment failure location which can be postulated is one in the upper compartment leading directly to the environment.

Regardless of failure location, the failure size is based on “leak before break” considerations. Experimental results show that concrete containments subjected to slow pressurization will leak before they break. Thus, as a containment pressurizes, it is most likely that a leak will develop of sufficient magnitude to prevent further pressurization.

2.2 Containment Failure Experiments

A few experimental results relevant to containment overpressurization in large, dry PWR containments like Zion are highlighted here. Many of the experimental conclusions are applicable to the APR1400 because both Zion and the APR1400 have pre-stressed concrete containments. The Zion containment has a failure pressure of 150 psia and the APR1400 containment has a volume approximately 10% larger than that of Zion. Experimental programs that tested the overall response of 1:6 reinforced concrete containment model and the response of individual containment details (personnel airlocks, mechanical penetrations, etc.) are considered below.

2.2.1 Sandia 1:6 Scale Model Containment Pressure Test Program

This 1987 test (Reference 4) involved destructive testing of a 1:6 scale reinforced concrete model considered representative of large, dry PWR containment designs (see Figure 2-1). Conventional materials were used for the concrete aggregate and the #4 (1/2” dia.) primary rebar. Containment details such as a liner, equipment hatches, personnel airlocks and penetrations were represented in the model, which was built to ASME/ACI code. An integral basemat was included but the dead load internal to the containment was not. Tests were conducted at ambient, as opposed to elevated, temperatures, but this does not present a limitation for a large, reinforced concrete containment, because severe accident temperatures will not be high enough to affect material properties for this containment type. In brief, the test objectives were to determine the failure pressure and location for the device, and produce a wide spectrum of structural failure data for further analysis. The test was conducted by pressurizing the facility in 10 psi steps early on and 2 to 3 psi pressure steps at the end. Final test pressure was 145 psig.

A small leakage was noted near equipment hatch “A” at 125 psig and in equipment hatch “B” at 138 psig. Equipment hatch “A” began to ovalize at 128 psig, as the horizontal diameter increased nearly 1/2”. Leakage could not be quantified at this point. At 140 psig, however, leakage was measured to be 13% mass/day. Leakage became very large (over 200% mass/day) between 140 and 145 psig, suggesting that liner tears occur at these pressure levels. Leakage occurred due to strain concentrations in the vicinity of containment penetrations as shown by Figure 2-2.

Two important conclusions of the work are: (1) pre-test analyses gave good results for cylinder displacements, rebar strains, etc., and, (2) pre-test analyses of the bending areas of the containment, namely the cylinder-basemat junction, were not in good agreement with test results. Post-test analysis was performed on the cylinder basemat junction to make use of test results and improve on the original predictions. Revised predictions for liner tearing at basemat-cylinder junction show failure at 152 to 154 psig.

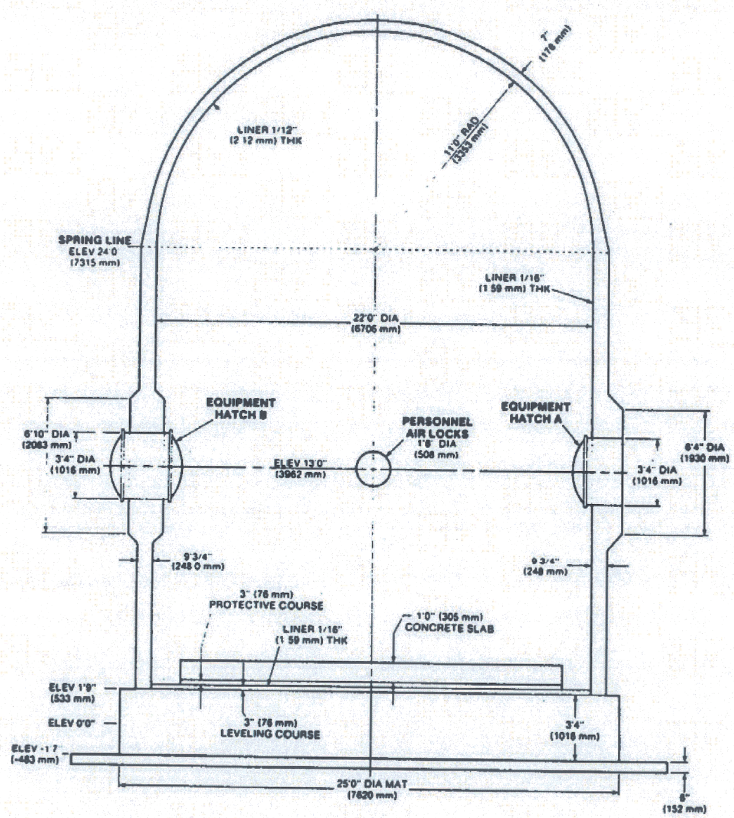

Figure 2-1 Sandia 1:6 Scale Reinforced Concrete Containment Model (Reference 5)

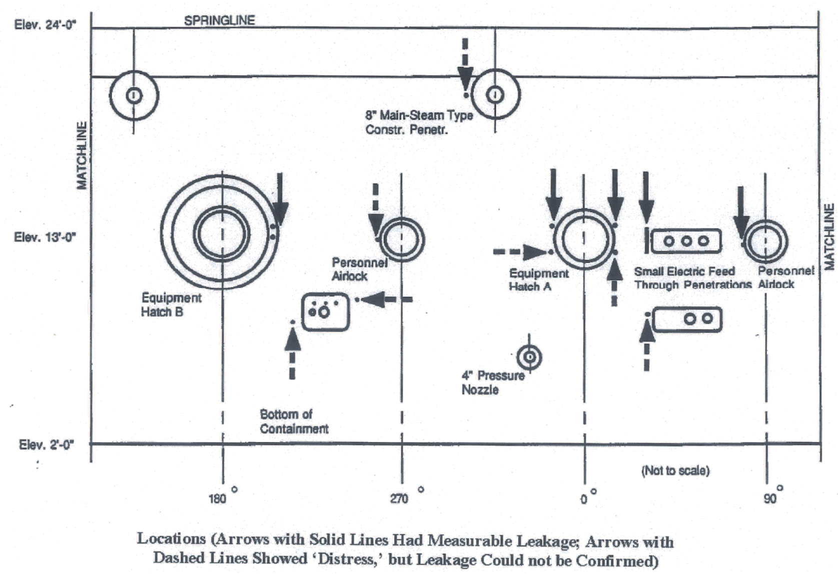

Figure 2-2 Sandia 1:6 Scale Model Liner Stretchout Showing Leakage (Reference 5)

2.2.2 Sandia/CBI Personnel Airlock Testing

In this test program, an actual full-scale airlock assembly (surplus from a canceled PWR; Callaway Unit 2) was subjected to environmental conditions considered applicable to certain design basis (LOCA) and severe accident events. In general terms, the overall objective of the program was to study potential adverse impacts on the pressure integrity of such devices attributable to these conditions. The conduct and results of these efforts are detailed in Reference 6. Test pressures as high as 300 psig were applied during this program.

In most of the individual tests performed, no pressure boundary integrity losses whatsoever were detected, structural response notwithstanding. For one case however, the test conditions did yield a significant degree of inner door seal leakage. In this specific case, the airlock inner door temperature was held at 650 °F while the atmosphere (air) temperature inside this door was raised to 800 °F. Pressure inside the door (i.e., corresponding to the containment side) was then increased from ambient to 150 psig, and at this point the leakage referred to was detected. Note that even at these elevated temperatures, which cannot be expected in a large, dry containment, the airlock would not begin to leak on overpressure until after other containment details, namely the equipment hatch and hoop rebars, (as demonstrated by the 1:6 scale test) had already failed.

2.2.3 EG&G Containment Penetration System Testing

Idaho National Engineering Laboratory performed a series of full-scale tests to determine the performance of mechanical, or piping, penetrations systems under design basis and severe accident conditions (Reference 7). In particular, three separate piping systems complete with valves, penetrations, supports, and piping were subjected to design basis conditions of 280 °F and 120 psig without any signs of failure for the duration of the test. These piping systems modeled the containment spray system (an 8” gate valve), the containment purge and vent system (an 8” butterfly valve), and a nominal small diameter (2”) globe valve piping system. These three systems were configured so that results “would be applicable to a high percentage of plants.”

Results indicate that there was significant plastic strain in the piping sections and many of the piping supports were badly deformed, but there was no buckling of the piping sections and the penetration assemblies showed no damage. The tested systems performed well and the program conclusion was that leak integrity and valve operability will most likely be maintained during severe accidents which challenge light water containments.

2.2.4 Sandia Electrical Penetration Assemblies (EPA) Program

The EPA program (Reference 8) tested electrical penetrations from three manufacturers: Conax, D. G. O’Brien, and Westinghouse. For a large, dry PWR containment, D. G. O’Brien assemblies were tested under a severe accident profile. This profile consisted of: (1) ramping the temperature and pressure from ambient to 293 °F and 60 psia, (2) then to 361 °F and 155 psia in 12 hours using saturated steam, and, (3) holding these conditions for the remainder of a 10 day test. There were no detectible leaks through the EPA during the severe accident test.

2.2.5 NUPEC/NRC Pre-stressed Concrete Containment Vessel (PCCV) Model Test

As part of a Cooperative Containment Research Program that was co-sponsored and jointly funded by the Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC), Sandia National Laboratories (SNL) conducted a test of a 1:4 scale model of a large dry PWR pre-stressed concrete containment vessel. The PCCV model was a 1:4-scale model of the pre-stressed concrete containment vessel (PCCV) of an actual nuclear power plant in Japan, Ohi-3. Ohi-3 is an 1,127 MWe Pressurized Water Reactor (PWR) unit, one of four units comprising the Ohi Nuclear Power station located in Fukui Prefecture, owned and operated by Kansai Electric Power Company.

The design pressure for the model was 0.39 MPa (56.6 psig). 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. The model included a steel liner anchored to the concrete shell by semi-continuous structural shapes (T’s). Conventional reinforcing ratios match the prototype and pre-stressing tendons match 1-to-1 with the prototype. The un-bonded pre-stressing system consisted of three, seven-wire strands per hairpin tendon, anchored in the basemat and identical, 360° hoop tendons anchored in opposing vertical buttresses. Figure 2-3 shows a cross-section of the model and Figure 2-4 shows

an image of the completed model. Details of the design, including the design drawings, and construction are reported in the PCCV test report (Reference 9).

A Limit State Test (LST) was carried out to pressurize the model beyond design basis accident loads and to compare that data against calculations. During the test an audible event occurred and the subsequent leak test showed that the model had failed functionality between 2.4 and 2.5 times the design pressure. It was concluded that a leak path opened most likely due to a liner tear in the vicinity of the equipment hatch. The model was then further pressurized to collect data on the inelastic response of the structure and to observe, if possible, a structural failure mode. At a pressure of 3.3 times the design pressure large local liner strains (6.5%) were measured and the liner was torn in several locations. However, the remainder of the structure appeared to have suffered little damage. There was no indication of tendon or rebar failure and the tendon strains stayed within the elastic limit (1%). Since a structural failure mode was not seen it was decided to depressurize the system and seal the inside with an elastomeric membrane. The model was filled to 97% with water and the remaining gas space was pressurized with nitrogen for the Structural Failure Mode Test (SFMT). After exceeding 3.3 times the design pressure a high noise level event was registered, which was interpreted as the breaking of a tendon. Shortly thereafter, at a peak pressure of 3.63 times the design pressure (1.42 MPa or 206.4 psig), the model ruptured violently near the mid-height of the cylinder. Figure 2-5 and Figure 2-6 depict the rupture of the PCCV model during the SFMT.

The dominant containment failure mode in the PCCV test is leakage induced by tearing of the containment liner. Images of these failures are shown in Figure 2-7. These failures are caused by large local liner strains in the vicinity of containment penetrations and terminations of rebar patterns. Finite element analysis (FEA) of the PCCV model is discussed in Section 2.3.2.

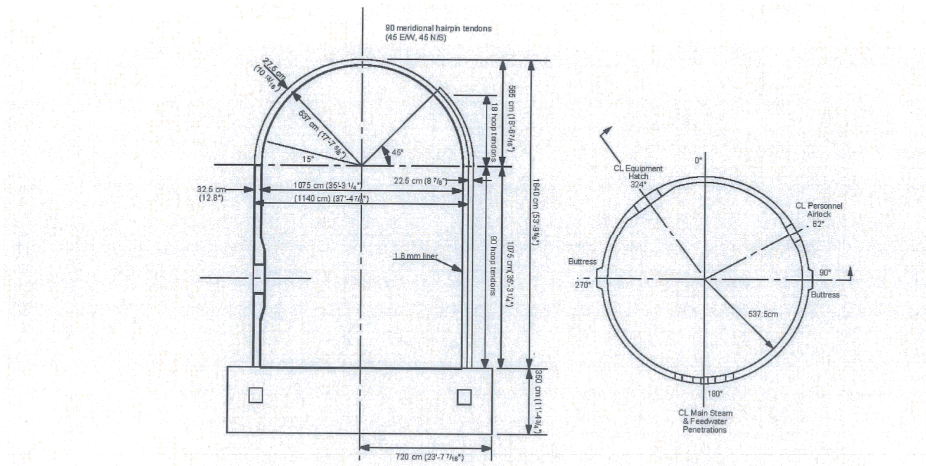

Figure 2-3 PCCV Model Elevation and Cross-Section (Reference 5)

Figure 2-4 Completed PCCV Model (Reference 5)

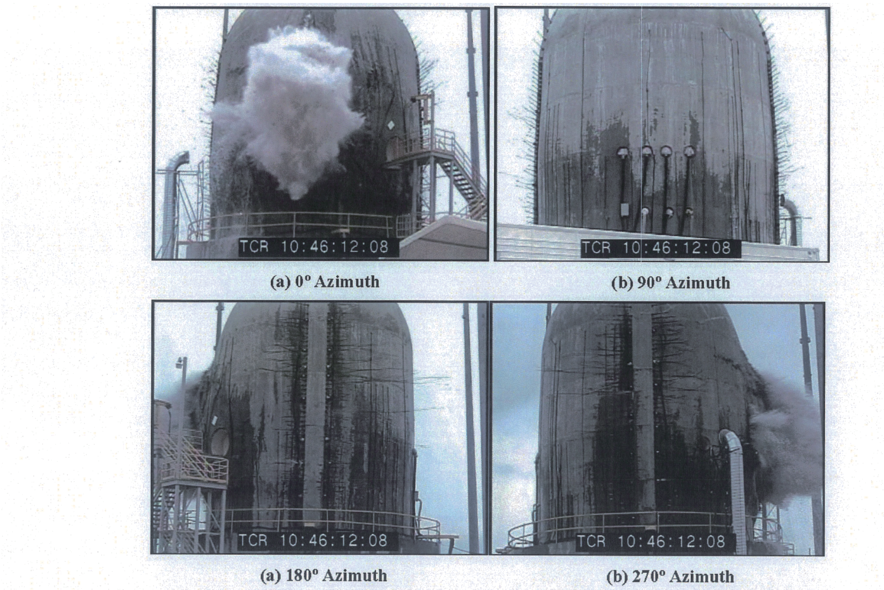

Figure 2-5 Images of the Eventual Rupture of the PCCV Model (Reference 5)

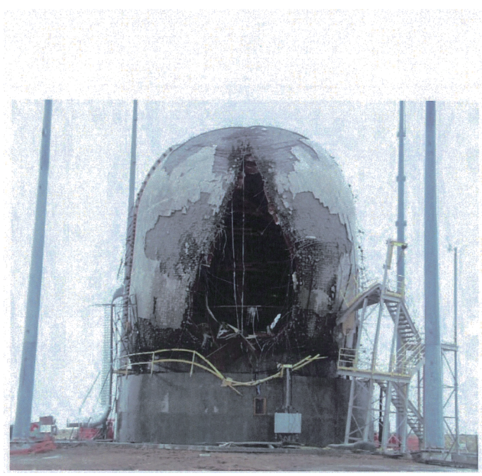

Figure 2-6 Image of the PCCV Model After the SFMT (Reference 5)

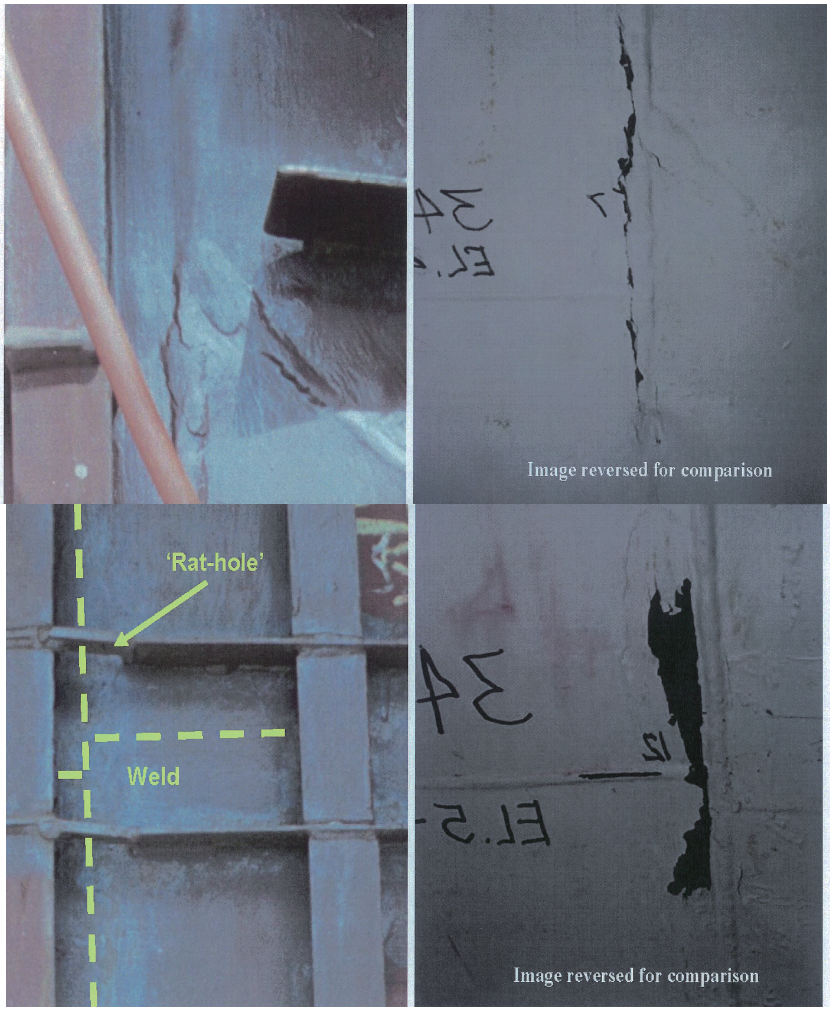

Figure 2-7 Images of the Equipment Hatch where Liner Tear Occurred in the PCCV Test (Reference 5)

2.3 Past Analytical Studies of Containment Failure

2.3.1 Large Dry Containment Types

Three structural assessments of the Zion containment have been performed by: Sargent & Lundy (S&L) for the IDCOR program (Reference 10), Brookhaven National Laboratory (BNL) for the NRC (Reference 11), and Los Alamos National Laboratory for the NRC (Reference 12). Because of the similarities between the APR1400 and Zion containment types, some of the analyses' conclusions are applicable to the APR1400. For this reason, the Zion analyses will be reviewed briefly below.

A 1983, S&L study utilized equilibrium hand calculations for membrane stresses in the Zion cylinder and dome, and a non-linear finite element analysis for overall response of the containment building, i.e., the stresses and strains in the post-tensioning tendons, steel reinforcement, concrete, steel liner and the soil. Both the hand calculations and the finite element analysis were in excellent agreement predicting failure at the hoop tendons near 134 psig.

Conclusions from this study which are pertinent to the current APR1400 analysis are that the ultimate containment failure pressure and location are not affected by the physical process that causes the pressurization. That is, the containment failure pressure and location will be the same regardless of whether the pressurization is caused by DCH, steaming, combustion, etc.

A 1985 BNL study using the NFAP computer code stands in contrast to the S&L study in that it predicted a shear failure at the basemat-cylinder intersection at 111 psig. To obtain this result, a finite element model with 268 elements and 954 nodes was used. To model the containment wall thickness with the liner and various rebars and tendons, six layers of elements were used. A more refined grid was used at the basemat-cylinder intersection.

One of the major conclusions of the study is that although hoop yielding can be calculated accurately even by hand, shear failure must be handled with complicated FEA models that capture the influence of rebars and tendons. BNL concluded that the S&L finite element model overestimated shear capacity in the basemat because the shell elements used were inadequate. BNL's report states that "models in which the steel members are lumped on the inside and outside faces of the containment wall, overestimate the shear capacity of the containment structures." This is precisely the model used by S&L for its study. However, the results of Sandia's 1:6 scale model test contradict the findings of the BNL study. In the scale model tests, failure occurred in the equipment hatch and the hoops tendons at roughly the same time rather than in the basemat-cylinder junction. There is no mention of modeling the load bearing capability of the soil in the BNL study, which might also explain some of the differences between the BNL and S&L study.

Los Alamos work performed in 1982 predicts shear failure near the basemat- cylinder intersection at 125 psig. However, since analysis above 125 psig could not be carried out due to numerical instabilities, this value seems dubious. This work also predicts hoop tendon yielding at 136 psig, which is in good agreement with the S&L study and the 1:6 scale model test.

2.3.2 Finite Element Analysis of the PCCV Model Test

The PCCV Model Test described in Section 2.2.5 consisted of three phases: pretest analysis, testing, and posttest analysis. These phases are described in NUREG/CR-6906 (Reference 5) and the analysis phases are discussed in this section.

The pretest analysis was completed several months prior to the execution of the Limit State Test so that analysis predictions could be published ahead of test results. The purpose of the pretest analysis was to validate structural models of pre-stressed concrete containments, gain insight into probable failure modes, and support planning of test procedures. The bulk of the analysis was performed using several finite element models which covered both global structural response as well as local response around key containment details, some of these are shown in Figure 2-8 through Figure 2-10. Analysis was performed using the ABAQUS and ANACAP-U computer programs.

The key results of the pretest analysis were:

- The largest hoop expansion occurs at the Equipment Hatch, and the "free-field displacement" (displacement at 0º and 180º) are slightly less and are approximately equal to each other.
- There is significant local circumferential bending adjacent to each buttress.
- There are significant strain concentrations at terminations or step-downs in rebar patterns.
- There are significant strain concentrations near hatches and near the edges of wall embossments.
- Using a strain-based failure criteria which considers the triaxiality of stress and a reduction in ductility in the vicinity of a weld, the liner failure strain was 0.16. The failure pressure at which a local analysis computed effective plastic strain that reached the failure strain, was 3.2 times the design pressure or 1.3 MPa (188.5 psi). The location for this liner-tearing failure was near the Equipment Hatch (E/H), adjacent to a vertical liner anchor that terminated near the liner insert plate transition.

These results show good general agreement between analysis and experiment, but show that the pretest analysis overpredicted the experimentally determined pressure capacity somewhat.

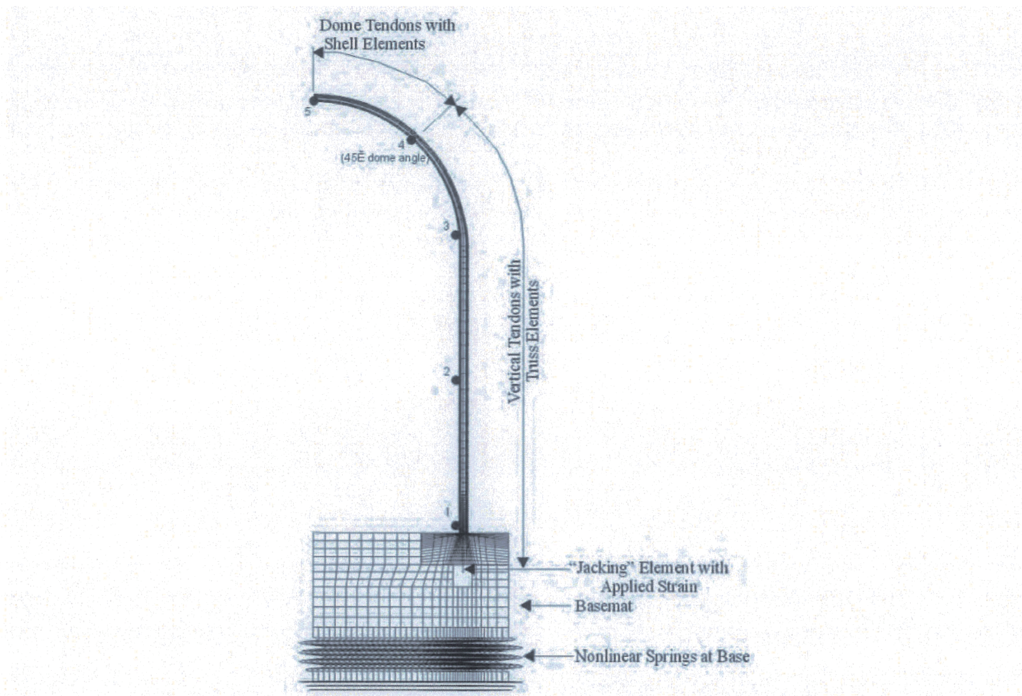

Figure 2-8 Axisymmetric Model of 1:4-scale PCCV (Reference 5)

]

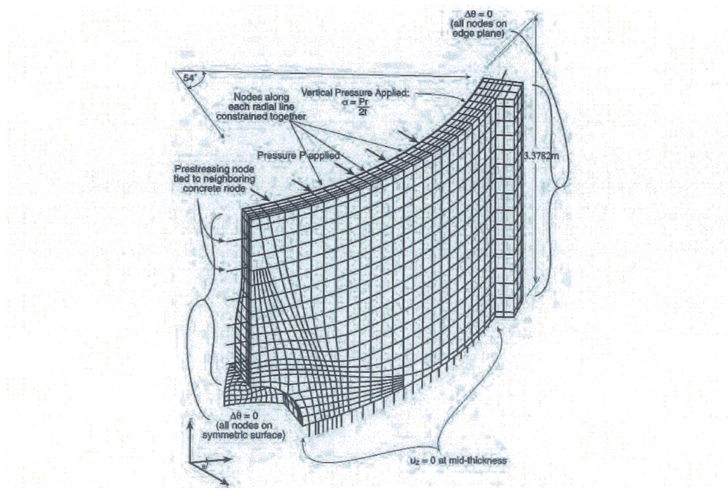

Figure 2-9 Local Model of PCCV Equipment Hatch (Reference 5)

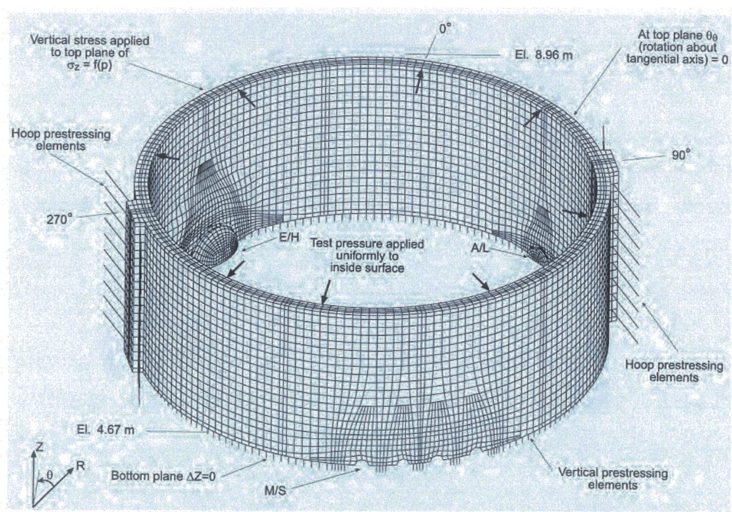

Figure 2-10 PCCV Three-Dimensional Cylinder Mid-Height Model (Reference 5)

The final phase of the PCCV Model Test involved a posttest analysis with a goal toward improving the analytical methods for predicting the structural response of a pre-stressed concrete containment and analysis of any phenomena or failure mode observed during the test that had not been predicted by analysis. The posttest analysis involved performing several additional studies using the models developed for the pretest analysis to identify where key model assumptions deviated from experimental results.

One key result of the posttest analysis was that the representation of friction in the tendon modeling of the analytical model was not in good agreement with the experiment. Because of this, liner strains in the vicinity of the Equipment Hatch penetration collar measured in the experiment had been found to be significantly lower than those determined in the pretest model. Since this was the failure location in the experiment, this difference was analyzed more closely. It was determined that preventing slip between the liner and concrete prevented prediction of the erroneously high strains around Equipment Hatch collar while leaving the results of the rest of the model relatively unchanged.

In addition to the change in slip modeling, a new model was also created for the liner “rat-hole” detail. Analysis of this detail showed that the elevated strain in this location was sufficient to cause liner tearing at as early as 2.8 times the design pressure. This prediction is in fairly good agreement with the experimental result.

Overall, the posttest analysis showed that a main factor in determining the limit state of the vessel was the radial expansion of the containment cylinder. This response must be predicted correctly in order to approximate the local mechanisms of liner tearing and penetration ovalization. The posttest analysis confirmed that the analytical model developed in the pretest analysis was able to predict the strain and radial expansion of the cylinder very accurately. Based on this, it is determined that the minimum requirement for a containment overpressurization evaluation is an accurate representation of the structural elements and material properties in a numerically-stable nonlinear finite element model that is insensitive to numerical solution strategy and control parameters.

3.0 APR1400 Containment Performance Criteria

This section contains the description of the containment design for the APR1400 and determination of containment performance criteria. Containment performance criteria are the criteria used to judge if the containment would maintain its role as an effective fission product barrier in the analyses presented in Section 4.0.

3.1 Description of the Containment

The APR1400 containment is a prestressed concrete structure composed of a right circular cylinder with a hemispherical dome and is founded on safety-related common basemat. The APR1400 containment encloses the reactor vessel, steam generators, reactor coolant loops, and portions of the auxiliary and engineered safety features systems. The containment provides reasonable assurance that leakage of radioactive material to the environment does not exceed the acceptable dose limit as defined in 10 CFR 50.34 even if a loss of coolant accident (LOCA) occurred.

The cylindrical containment shell has a constant thickness of 1.37 m (4 ft 6 in) from the top of the foundation basemat to the springline. The shell is thickened locally around the equipment hatch, two personnel airlocks, feedwater, and main steam line penetrations. The containment reinforcing consists primarily of hoop and meridional steel. Prestressing tendons are also arranged in hoop and meridional directions.

The roof of the containment is a hemispherical dome. The inside of the dome is lined with a steel liner plate to provide leak-tightness. The buttresses are extended up to 48 degrees into the dome to provide anchorage for the dome hoop tendons.

The containment provides a minimum net free volume of 0.088 million cubic meters (3.128 million cubic feet) with its internal structures arranged in a manner to (1) protect the inner containment from missile threats, (2) promote mixing throughout the containment atmosphere, and (3) accommodate condensable and non-condensable gas releases from design basis and severe accidents. The internal structures, which are made of reinforced concrete, enclose the reactor vessel and other primary system components. The internal structures provide radiation shielding for the containment interior and missile protection for the reactor vessel and containment shell.

3.2 Containment Performance Criteria regarding FLC requirement

As stated in SECY 93-087, the containment stress should not exceed Factored Load Category (FLC) presented in ASME Code, Section III, Division 2, Subarticle CC-3720, approximately 24 hours following the onset of core damage. In addition, USNRC RG 1.216 requires the containment integrity analysis to consider the FLC requirement presented in requirement position 5 of USNRC RG 1.7.

Based on an adiabatic isochoric complete combustion (AICC) analysis with the assumption of the unavailability of hydrogen mitigation features, the pressure resulting from 100-percent metal water reaction of fuel cladding and uncontrolled hydrogen burning was determined as 8.7 kg/cm^2 (123.7 psia) including the safety margin of the APR1400 containment. For the pressure of 8.7 kg/cm^2 (123.7 psia), the structural analysis for the APR1400 containment showed that the strain of containment liner plate did not reach the limit of ASME Code, Section III, Division 2, Subarticle CC-3720. Therefore, the criterion for the containment performance was determined as 8.7 kg/cm^2 (123.7 psia) with regard to the FLC requirement.

4.0 APR1400 Containment Performance Analysis

The response of the APR1400 containment to gradual pressurization under severe accident conditions was investigated by using the MAAP 4.0.8 code for a range of possible sequences. The containment pressurization sequences considered in the analysis included the overpressurization by steam and non-condensable gases.

The sequences analyzed covered the most likely severe accident initiators for the APR1400. In accordance with the design of the APR1400, Cavity Flooding System (CFS) and ECSBS were included in the sequences. The ECSBS is designed to protect the containment integrity against

overpressure and prevent the uncontrollable release of radioactive materials into the environment. The ECSBS flow path is from external water sources (the reactor makeup water tank, demineralized water storage tank, fresh water tank, or the raw water tank), through the fire protection system line via the diesel-driven fire pump, to the ECSBS line emergency connection located at ground level near the auxiliary building. The ECSBS flow rate provides sufficient heat removal to prevent containment pressure from exceeding 8.7 kg/cm^2 (123.7 psia).

Among the sequences, a large break LOCA resulted in the highest pressure in containment at 24 hours following the onset of core damage. The break was approximately equivalent to a 0.24 m (9.5 inch) diameter hole in one of the hot legs. Four safety injection tanks (SITs) and motor-driven auxiliary feed water (AFW) pumps were available. Safety injection pumps (SIPs), charging pumps (CPs) and containment spray pumps (CSPs) were assumed to be unavailable. RCS depressurization using the POSRV was not assumed for this sequence. The CFS actuation and alignment of the three-way valves to the steam generator compartment were assumed to start 30 minutes after the onset of core damage.

Due to the break, primary system pressure rapidly drops and causes the reactor to scram. The break flow from the RCS rapidly pressurizes containment and the core is quickly uncovered. Due to the low primary system pressure, the coolant in SITs is injected temporarily recovering and cooling the core. Once the inventory in SITs has been depleted, the core uncovers again and begins to heat up. Core exit temperature soon reaches 922.04 K (1,200˚F), signaling the onset of core damage.

30 minutes after the onset of core damage, it is assumed that operators align the three-way valves to the steam generator compartment. At the same time, the actuation of the CFS, which allows the gravity-driven water flow from the IRWST into the reactor cavity, is assumed. The core begins to melt and relocate to the reactor vessel lower plenum. This produces a momentary pressure spike in the RCS due to the rapid steaming of the water pool that exists in the lower plenum.

The molten corium in the reactor vessel lower plenum eventually caused the vessel to fail at low pressure. The core debris slumped into the flooded reactor cavity. The corium caused the water pool in containment to heat up and boil. The core debris slumps into the flooded reactor cavity. The corium causes the water pool in containment to heat up and boil. Due to the gradual steam generation in the cavity and non-condensable gas generation during MCCI, containment pressure steadily increases. Heat removal by the overlying water pool quenches the corium and eventually halts concrete ablation. At the time of complete quenching of the corium, the concrete ablation depth is much less than the depth of the steel containment liner in the reactor cavity.

Considering the further generation of steam after the concrete ablation stopped, the containment pressure does not reach the 8.7 kg/cm^2 (123.7 psia) within 24 hours after the onset of core damage. The ECSBS is actuated 24 hours after the onset of core damage at a flow rate of 2.84 m^3/min (750 gpm). It begins to cool and depressurize the containment atmosphere as shown in Figure 4-1. The result shows that the ECSBS is capable of controlling containment pressure for a period of 48 hours after 24 hours following the onset of core damage. The maximum pressure and temperature following the initial 24 hour period are enveloped by the maximum pressure and temperature during the initial 24 hour period.

TS

Figure 4-1 Containment Pressure for Large Break LOCA with ECSBS Actuated 24 hours after Onset of Core Damage

In summary, the most likely severe accident initiators for the APR1400 were analyzed in the analysis. For initial 24 hours following the onset of core damage, the pressure of the APR1400 containment does not exceed 8.7 kg/cm^2 (123.7 psia), so that the strain of containment liner plate does not reach the limit of ASME Code, Section III, Division 2, Subarticle CC-3720. Following the initial 24 hour period, the actuation of ECSBS enables to maintain the containment pressure lower than 8.7 kg/cm^2 (123.7 psia). It can be concluded that the APR1400 containment is prevented from the uncontrolled release of fission products into the environment.

5.0 Conclusion

Based on the containment performance criteria established in Section 3.0 and the severe accident analyses discussed in Section 4.0, it is determined that failure of containment integrity will not occur for the more likely severe accident challenges. In a view point of regulatory requirements, it is determined that the APR1400 containment design satisfies the regulatory criteria established by the USNRC such as SECY 93-087 and RG 1.216.

6.0 References

1. "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," SECY-90-016, U.S. Nuclear Regulatory Commission, Washington, DC, June 1990.

2. "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," SECY-93-087, U.S. Nuclear Regulatory Commission, Washington, DC, April 1993.

3. "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design Basis Pressure," NRC RG 1.216, U.S.NRC, August 2010.

4. Horschel, D. S. "Experimental Results from Pressure Testing a 1:6-Scale Nuclear Power Plant Containment". Sandia National Laboratories. Albuquerque NM : s.n., NUREG/CR-5121, SAND88-0906, 1992.

5. Hessheimer, M. F. and Dameron, R. A. "Containment Integrity Research at Sandia National Laboratories. Sandia National Laboratories". Albuquerque NM : s.n., NUREG/CR-6906, SAND2006-2274P, 2006.

6. Julien, J. T. and Peters, S. W. "Leak and Structural Test of Personnel Airlock for LWR Containments Subjected to Pressures and Temperatures Beyond Design Limits", CBI Research Corporation. Plainfield IL : s.n., NUREG/CR-5118, SAND88-7155, 1989.

7. MacDonald, P. E. "Containment Penetration System Behavior During Design Basis and Severe Accidents", Idaho National Engineering Laboratory. Idaho Falls ID : s.n.

8. Clauss, D. B. "Severe Accident Testing of Electrical Penetration Assemblies", Sandia National Laboratories. Albuquerque NM : s.n., NUREG/CR-5334, SAND89-0327, 1989.

9. Hessheimer, M. F., et al. "Overpressurization Test of a 1:4-Scale Prestressed Concrete Containment Vessel Model", Sandia National Laboratories. Albuquerque NM : s.n., NUREG/CR-6810, SAND2003-0840P, 2003.

10. IDCOR. "Containment Capability of Light Water Nuclear Power Plants", Technical Report 10.1., 1983.

11. Sharma, S. et. al. Ultimate Pressure Capacity of Reinforced and Prestressed Concrete Containments. Brookhaven National Laboratory. NUREG/CR-4149, BNL-NUREG-51857, 1985.

12. Butler, T. A. and Fugelso, L. E. Response of the Zion and Indian Point Containment Buildings to Severe Accident Pressures. Los Alamos Scientific Laboratory. NUREG/CR-2569, LA-9301-MS., 1982.