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Overpressurization Test of a 1:4-Scale Prestressed Concrete Containment Vessel Model

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ABSTRACT

The Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research, cosponsored and jointly funded a Cooperative Containment Research Program at Sandia National Laboratories (SNL) from July, 1991 through December, 2002. As part of this program, a 1:4 scale model of a prestressed concrete containment vessel (PCCV) was constructed and pressure tested to failure. The prototype for the model is the containment building of Unit 3 of the Ohi Nuclear Power Station in Japan. The design accident pressure, P_d, of both the prototype and the model is 0.39 MPa (57 psi). The objectives of the PCCV model test were to simulate some aspects of the severe accident loads on containment vessels, observe the model failure mechanisms, and obtain structural response data up to failure for comparison with analytical models.

The PCCV model was designed and constructed by NUPEC and its Japanese contractors, Mitsubishi Heavy Industries, Obayashi Corp., and Taisei Corp. SNL designed and installed the instrumentation and data acquisitions systems and conducted the overpressurization tests. ANATECH Consulting Engineers conducted the pre- and posttest analyses of the model under contract to SNL.

Nearly 1500 transducers were installed on the PCCV model to monitor displacements, liner, rebar, concrete and tendon strains and tendon anchor forces. This instrumentation suite was augmented by the Soundprint® acoustic monitoring system, video, and still photography.

Low pressure testing, including a Structural Integrity Test to 1.125 P_d, and an Integrated Leak Rate Test at 0.9 P_d, was conducted in September, 2000. The Limit State Test (LST) of the model was conducted on September 27-28, 2000 by slowly pressurizing the model using nitrogen gas. A leak, presumably through a tear in the liner, was first detected at a pressure of 2.5 P_d and a leak rate of 1.5% mass/day was estimated. The test was terminated when the model reached a pressure of 3.3 P_d. At this pressure, the leak rate was nearly 1000% mass/day, exceeding the capacity of the pressurization system. Posttest inspections revealed 26 tears in the 1.6mm (1/16") steel liner as the source of the leaks.

Since only limited damage and inelastic response occurred during the LST, the interior was resealed with an elastomeric membrane. The PCCV was then filled nearly full with water and repressurized on November 14, 2001. This Structural Failure Mode Test reached a maximum pressure of 3.6 P_d when the model ruptured violently by failure of the prestressing tendons and then the reinforcing steel.

The resulting data from all the tests are provided for comparison with pretest and posttest analyses.

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EXECUTIVE SUMMARY

Introduction

The Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research co-sponsored and jointly funded a cooperative containment research program at Sandia National Laboratories[1] (SNL). 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 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. 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, owned and operated by Kansai Electric Power Company. The scale of the PCCV model was 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 that were representative of the prototype.

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 bounding pressure and temperature transients. The design accident pressure, P_d, of both the prototype and the model is 0.39 MPa (57 psi). The term "severe accidents" is used to describe an array of conditions that could result in loads, in excess of the design basis loads, on the containment. The definition of severe accident loads, which is not as rigorous as the design basis loads definition, results from a consideration 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.

Design and Construction

Within the cooperative framework agreed on 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 on Kirtland Air Force Base (KAFB), Albuquerque, New Mexico, USA. 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., Greeley, CO for general construction work and management of day-to-day construction operations. MHI pre-fabricated 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.

[1] This work is jointly sponsored by the Nuclear Power Engineering Corporation and the U.S. Nuclear Regulatory Commission. The work of the Nuclear Power Engineering Corporation is performed under the auspices of the Ministry of Economy, Trade and Industry, Japan. Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the U.S. Department of Energy under Contract Number DE-AC04-94AL85000

Instrumentation and Data Acquisition

NUPEC funded SNL to provide programmatic and model design support, instrument the model, and design and assemble the data acquisition system. 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 strain measurements of individual structural elements (i.e. liner, rebar, tendons, concrete) to characterize the force distribution near structural discontinuities. In areas absent of 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 meridional strains were measured.

Pressure measurement requirements included careful measurement of the PCCV interior pressure for purposes of leak detection, and to a lesser extent, leak rate measurement, characterization of the mechanical response as a function of pressure and to control the pressurization rate. It should be noted, that while measurement of leak rates was not a primary objective, detection of the onset of leakage requires the calculation of very small leak rates with relatively high accuracy.

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 strain 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 was desirable to have some method 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 was 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.

Analysis

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 preliminary analyses supported design studies, identified critical response modes and assisted in locating instrumentation. The pretest analysis consisted of the development and analysis of 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 the comparison with the test results and provided insights into the response observed during the pressure tests. The pre- and posttest analyses have been reported separately and are not included in this report.

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 United States, Japan and other countries are provided with a common set of data on the model test (design drawings, material properties, test specifications, etc.) and then 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. 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.

Testing

NRC funded the planning and conduct of test operations. After extensive discussions between NUPEC, the NRC and SNL, a detailed Test Plan was developed by SNL to describe the conduct of the pressurization tests of the PCCV model. A final series of three tests were agreed upon:

- A leak check and System Functionality Test (SFT) @ 0.5 P_d (2.0 kg_f/cm^2 or 28.4 psig)
- A Structural Integrity Test (SIT) @1.125 P_d followed by an Integrated Leak Rate Test (ILRT) @ 0.9 P_d
- A Limit State Test (LST) to the static pressure capacity of the PCCV model (or the pressurization system, whichever comes first)

The *pneumatic* Limit State Test 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 re-pressurize the PCCV model, prior to demolition, in an attempt to observe larger inelastic response and, possibly, a global structural failure. This Structural Failure Mode Test (SFMT) was a combined *pneumatic-hydrostatic* test, where the PCCV model was filled nearly full with water, to reduce the volume of gas to be pressurized, and nitrogen gas was used to generate the overpressure.

The SFT was conducted beginning approximately 9:00 AM, July 18, 2000. The model was pressurized using nitrogen to 0.5 P_d (0.2 MPa or 28.4 psig) in three increments holding pressure for one hour or longer at each step, depending on the duration needed to perform all system functionality and leak checks. The model was then isolated and a leak rate check was performed by monitoring the model pressure and temperature for approximately 18 hours. After 18 hours, the calculated leak rate was 0.15% mass/day, which was interpreted as confirming that the model was leak-tight. After the model leak rate check, the model was allowed to depressurize through a pair of orifice plates calibrated to leak rates of 1% and 10% mass/day to perform a calibration test on the leak rate measurement instrumentation. The calculated leak rates for each test were 0.87% and 7.86%, respectively, indicating that the leak rate instrumentation was capable of accurately detecting a leak of 1% mass per day, which is the goal specified for the ILRT. The SFT was concluded on July 20 by opening the vent valve, allowing the model to depressurize.

The Structural Integrity Test and the Integrated Leak Rate Test were conducted on September 12-14, 2002 as a combined test, with the ILRT following immediately after the SIT. The SIT/ILRT reproduced the pre-operational tests conducted at the prototype plant and allows for a comparison of the model's elastic response characteristics and leak behavior with the prototype and pretest analyses. The SIT test pressure, P_{SIT}, was 1.125 P_d. After the SIT pressure was maintained for one hour, the PCCV model was depressurized to the ILRT pressure, 0.9 P_d. The calculated leakage rate at P_{ILRT}, L_{tm}, after 24 hours at 0.9 P_d, was 0.06% mass/day.

The Limit State Test (LST) was designed to fulfill the primary objectives of the PCCV test program, i.e. to investigate the response of representative models of nuclear containment structures to pressure loading beyond the design basis accident and to compare analytical predictions to measured behavior. The LST was conducted after the SIT and ILRT were completed and the data from these tests evaluated. The PCCV model was depressurized between the SIT/ILRT and the LST. The LST began at 10:00 AM, Tuesday, September, 26, 2000 and continued, without depressurization, until the test was terminated just before 5:00 PM on Wednesday, September 27. The model was pressurized in increments of approximately 0.2P_d to 1.5 P_d when a leak check was conducted yielding a leak rate of 0.48% mass/day. Pressurization of the model continued in increments of approximately 0.1P_d to 2.0P_d when a second leak check resulted in a calculated leak rate of 0.003%, i.e. essentially zero. Pressurization of the model resumed in increments of 0.1P_d to 2.5P_d. At 2.4$P_{d,}$ the acoustic system operator reported hearing a change in the acoustic output which might indicate that "something had happened". The model was isolated for a third leak check and after approximately 1-1/2 hours, a fairly stable leak rate of 1.63% mass per day was calculated, indicating that the model was leaking, most likely from a tear in the liner in the vicinity of the E/H. The average hoop strain at $2.5P_d$, coinciding with the onset of liner tearing and leakage was 0.18%.

After concluding that the model had functionally failed between 2.4 and 2.5 P_d, the next goal was to continue to pressurize the model as high as possible to collect data on the inelastic response of the structure and to observe, if possible, a structural failure mode. Pressurization continued in increments of 0.05 P_d. The pressure was increased to slightly over 3.3 P_d before the leak rate exceeded the capacity of the pressurization system and the test was terminated. After the model had completely depressurized, it was purged with fresh air, the E/H was removed and a detailed inspection of the inside of the model revealed 26 discrete tears in the liner, all located at vertical field welds. Extensive examination and metallurgical analysis of the liner after the test revealed that fabrication defects contributed to nearly all of the liner tears.

Almost immediately after the completion of the LST, there was a recognition that while the PCCV model had demonstrated it's capacity to resist pressures well above the design pressure and had exhibited liner tearing and leaking as the functional failure mode, the test objectives were not fully met with respect to observing large inelastic deformations, for comparison with analyses. NUPEC and NRC approved a concept proposed by SNL to seal the interior surface of the liner with an elastomeric membrane, fill the model with water to 1.5m (5') from the dome apex, approximately 97% of the interior, and repressurize the remaining gas pocket with nitrogen until the model failed or pressure could not be maintained.

The Structural Failure Mode Test (SFMT) began shortly after 10:00 AM on Wednesday, November 14, 2001. The model was continuously pressurized at a rate of approximately 0.035 MPa/min (5 psi/min). All active sensors were continuously scanned at intervals of approximately 30 seconds and the video cameras were continuously recording the response of the model. As the pressure was increased, evidence of leakage was visible by increasing wetting of the concrete surface. At 10:38 AM, the effective pressure in the model equaled the peak pressure achieved during the LST, 3.3 P_d. At approximately 10:39 AM, the acoustic system recorded a very high noise level event which was interpreted as the breaking of a tendon wire. At this point in the test, events occurred very quickly. Shortly after detecting the wire break, a small spray of water was observed at approximately 0º azimuth and additional tendon wire breaks were detected by the acoustic system with increasing frequency. The rate of pressurization was decreasing and the nitrogen flow rate was increased to maintain the pressurization rate. Pressurization of the model continued until a second spray of water was observed and then, suddenly, at 10:46:12.3, at an effective pressure of 3.63 P_d (1.42 MPa or 206.4 psig) the PCCV model ruptured violently at ~6º azimuth near the mid-height of the cylinder. The maximum average hoop strain at the peak pressure of 3.63 P_d was 1.02%. The model continued to expand after reaching the peak pressure and the maximum hoop strain recorded just prior to rupture was 1.65%.

Conclusions

The over-pressurization tests of the 1:4-scale PCCV model represent a significant advance in understanding the capacity of nuclear power plant containments to loads associated with severe accidents. The data collected during the tests, as well as the response and failure modes exhibited, will be used for many years to come to benchmark numerical simulation methods used to predict the response of concrete containment structures. While lessons for actual plants can and should be drawn from this and previous large scale containment model tests, these insights are beyond the scope of this report and will be addressed in a future effort. The reader is cautioned ***not*** to draw direct conclusions regarding the pressure capacity of actual plants from these tests or interpret these results as a demonstration of the prototype capacity. The PCCV model tests have demonstrated the importance of the unique details and as-built characteristics of the model on the ultimate capacity. Any efforts to estimate the capacity of an actual containment must address the unique features of the plant under consideration.

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

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No one person was responsible for the success of the PCCV Test Project. Without the dedicated efforts of a team including the project sponsors, partners and contractors and supporting organizations, the outcome of this project would have been far less successful than it has proven to be. It is, however, difficult to acknowledge everyone who contributed to this project without unconsciously omitting some individuals. I would like to begin therefore by apologizing to those anonymous, but no less important contributors I may have overlooked.

But the contributions of some individuals are too significant to overlook. I would like to begin, therefore, by acknowledging the program managers of the sponsoring organizations, NUPEC and the U.S. NRC. Dr. James F. Costello, U.S. NRC, has been the guiding force behind the containment integrity research conducted at SNL for over 25 years, including this project. His perseverance, support, and guidance has been invaluable and it is no overstatement to say that this project may never happened without his involvement. Similarly, the NUPEC project directors: Dr. Kenji Takumi, Dr. Hideo Ogasawara and Dr. Takshi Kiguchi; and the project managers: Mr. Akira Nonaka, Mr. Tomoyuki Matsumoto, Mr. Masaki Iriyama and Mr. Satoru Shibata ensured that this program had the financial and technical resources to meet the program objectives in order to make a significant contribution to the international nuclear power industry.

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It is very difficult to adequately describe the contribution of all these individuals in this short space, but to each of them, I would like to express my heartfelt thanks.

Thank you.

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ABBREVIATIONS

A/L	air lock
A/E	Architect/Engineer
AO	Acoustic System Operator
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CE	fiber optic gages
CONV	converted data
COR	corrected data
CPOT	Cable Potentiometer
CTTF	Containment Technology Test Facility
CVDT	converted
DA	data analyst
DAS	Data Acquisition System
DET	Division of Engineering Technology
DISP	displacement
DO	Data Acquisition System Operator
DOE	Department of Energy
DOR	data of record
DT	displacement
DYN	dynamic
E/H	equipment hatch
EPRI	Electric Power Research Institute
ES&H	Environmental Safety and Health
F/W	feedwater
GBST	gage bar strains
GFAC	gage-specific factors
ILRT	Integrated Leak Rate Test
KAFB	Kirtland Air Force Base
LI	liner strain gage
LINST	liner strains
LOCA	loss-of-coolant accident
LST	Limit State Test
LVDT	Linear Variable Differential Transformer
M/S	main steam
METI	Ministry of Economy, Trade and Industry
MHI	Mitsubishi Heavy Industries
NO	Nitrogen Supply Operator
NRC	U.S. Nuclear Regulatory Commission
NSS	nuclear steam supply
NUPEC	Nuclear Power Engineering Corporation
PCCV	prestressed concrete containment vessel
PLC	Programmable logic controller
PRES	pressure
PWR	Pressurized Water Reactor
REBST	rebar strain
RES	Office of Nuclear Regulatory Research
RS	rebar strain gages
RTD	Resistance Temperature Detectors
SCV	steel containment vessel
SFMT	Structural Failure Mode Test
SFT	System Functionality Test
SIT	Structural Integrity Test
SNL	Sandia National Laboratories
SOL	Standard Output Location

T/C	thermocouple
TC	test conductor
TEMP	temperature
UTS	ultimate strength
YS	yield strength