

Mark I Containment Report

March 19, 2011

PURPOSE

This paper describes the Mark I containment design in use in the 23 U.S. reactors and its ability to fulfill its safety function in containing fission product releases under design basis conditions. It also offers, as an initial matter, some observations about the performance of the Mark I containment under many beyond-design-basis events experienced at Fukushima Daiichi in March 2011.

PERFORMANCE OF MARK I CONTAINMENTS AT FUKUSHIMA DAIICHI

Fukushima Daiichi Units 1-4 are boiling water reactors (“BWRs”) equipped with Mark I containments. Units 1-3 were operating; Unit 4 was in shutdown with fuel offloaded when the earthquake and tsunami occurred, which resulted in the loss all offsite and on-site power to all four units.

Early reports regarding Units 1-3 stated plant operators used safety relief valves to relieve pressure in the reactor pressure vessel. In addition, when the fuel rods became uncovered, hydrogen formed in the core (due to zirconium/water reaction) and was also transported into the wetwell when the reactor vessel safety relief valves opened. The combination of steam and hydrogen flowing into the wetwell increased the wetwell temperature and pressure. Since there was no on-site or off-site power available, there was no means available to cool the wetwell water. Over time, the pressure in the primary containment rose over the design pressure. To avoid containment breach, venting became necessary. Upon venting, it is believed that vented hydrogen gas caused explosions at these units.

The following should be noted:

- Coincident long-term loss of both on-site and off-site power for an extended period of time is a beyond-design-basis event for the primary containment on any operating nuclear power plant.
- The Mark I containment vessels appeared to have held pressure to well above the design pressure.
- The response of the reactor pressure vessel and reactor in general agree with severe accident management studies performed in the 1980s and early 1990s.

BACKGROUND

Description of the Mark I Containment System

Figure 1 shows a cutaway view of a typical Mark I Containment system. The major components shown in the figure include:

- The drywell, which surrounds the reactor pressure vessel (RPV) and recirculation loops. The drywell is light-bulb shaped steel-lined pressure vessel backed over most of its surface with reinforced concrete.

- A wetwell is situated beneath the drywell and connected to the drywell by a system of vent pipes. The wetwell is a toroidal-shaped (donut shaped) pressure vessel which is filled to about half of its height with water. The wetwell is often referred to as a torus (donut shape) or as a suppression pool (due to its function) and is made from either steel or concrete.
- An interconnecting vent network exists between the drywell and the wetwell. The vents are open on one end to the drywell and on the other end open into a header in the wetwell that has additional downcomer vents below the water level in the suppression pool.

The drywell, wetwell and vent system form the primary containment around the reactor pressure vessel and recirculation loops. The function of the containment system is to contain the energy released during a postulated design-basis loss-of-coolant accident of any size reactor coolant pipe and to protect the reactor from external events. The design-basis break is the largest reactor recirculation system pipe. The primary containment system is designed to withstand the combined seismic, pressure and temperature loads for this event and maintain integrity. The containment system accommodates this accident without exceeding the design leakage rate; in this way, the containment system limits the release of fission products during that event to offsite dose to levels significantly below the guideline values specified by regulation (10 CFR 100).

The primary containment is one of the three main barriers limiting release of fission products from the BWR nuclear fuel into the environment. Other barriers include the fuel rod cladding and the reactor pressure vessel together with its piping, which form the reactor coolant pressure boundary and the primary containment. In addition to the three fission product barriers, the secondary containment surrounds the primary containment and houses emergency core cooling systems and the spent fuel pool.

Note: Secondary Containment may vary from site to site

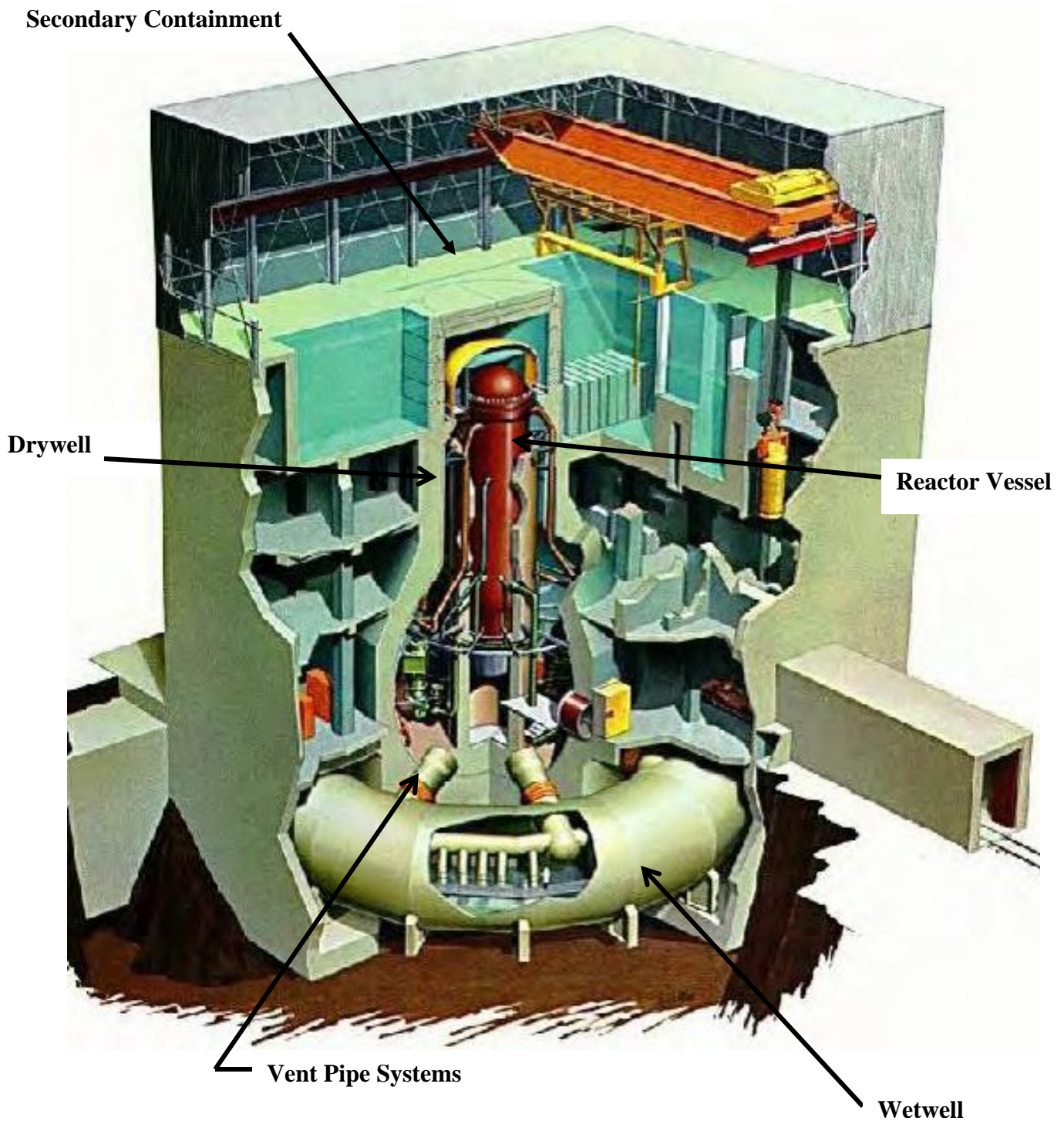


Figure 1 – Typical Mark I Containment System

CONTAINMENT OPERATION DURING A LOSS OF COOLANT ACCIDENT

During normal operation, the drywell atmosphere and the wetwell atmosphere is inerted (filled with nitrogen), and the wetwell water is at ambient temperature. In the event of a pipe break in the reactor coolant system inside the drywell, pressurized coolant escaping from inside the reactor coolant system will flash to steam and begin to pressurize and heat the drywell atmosphere. The reactor is automatically shut down. As the pressure rises in the drywell, the vent system will also pressurize, eventually forcing the steam into the wetwell below the water level. The steam contacting the water condenses in the wetwell. This reduces (suppresses) the pressure in primary containment following the loss of coolant accident by turning steam back into water.

Steam condensing in the wetwell gradually increases the wetwell temperature and pressure. As the accident progresses, plant instrumentation will sense the change in conditions in containment and in the reactor coolant system, and emergency systems will activate to cool the shut down reactor. Systems will also be activated to cool the water in the wetwell. These active emergency systems can be powered by off-site power or by on-site emergency diesel generators in the event of a loss of off-site power. With these emergency systems available, the Mark I containment system is designed to contain reactor water (and any fission products if present) without release during a range of pipe break scenarios, up to and including a full guillotine rupture of the largest pipe connected to the reactor vessel, such that the primary containment pressure does not exceed its design value (50 to 60 PSIG at expected accident temperature). This prevents discharge of any released water (and fission products if present) during the loss of coolant accident from escaping containment into the atmosphere.

Use of a wetwell for pressure suppression in primary containment is a feature of the General Electric BWR design.

DESIGN BASIS

Appendix A to 10 CFR 50 lists current general design criteria for nuclear power plants. Four General Design Criteria—2, 16, 50, and 51—are used for any reactor vendor containment design. Each of the criteria are provided below:

Criterion 2 – Design bases for protection against natural phenomena. Structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Criterion 3 – Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the

environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Criterion 50 – Containment design basis. The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Criterion 51 – Fracture prevention of containment pressure boundary. The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

The Mark I containment system has undergone extensive testing and analysis and in some cases has been modified to ensure that these criterion are satisfied, as described in the following sections.

ORIGINAL DESIGN OF THE MARK I CONTAINMENT

The original design of the Mark I containment system considered postulated accident loads associated with containment design. These included pressure and temperature loads associated with a loss-of-coolant accident (LOCA), seismic loads, dead loads, jet-impingement loads, and hydrostatic loads due to water in the suppression chamber.

The original design loads for the containment were based on large-scale tests performed by GE and other institutions. The purpose of these initial tests, performed from 1958 through 1962, was to demonstrate the viability of the pressure-suppression concept for reactor containment design. The tests were designed to simulate LOCAs with breaks in piping sized up to approximately twice the cross-sectional break area of the design-basis LOCA.

The tests were instrumented to obtain quantitative information associated with containment design pressures during these events. The data from these tests were the bases for the design and the initial NRC approval of the Mark I containment system.

EVOLUTION OF THE DESIGN

After the establishment of the original design criteria, additional loading conditions were identified which arise in the functioning of the pressure-suppression concept used in the Mark I containment system design. These conditions were noted in the course of performing large-scale testing of an advanced design pressure-suppression containment (Mark III). Specifically, during the large-scale testing of the Mark III containment system design in the period 1972 through 1974, new suppression pool hydrodynamic loads were identified for the postulated LOCAs. GE tested the Mark III containment concept in its Pressure Suppression Test Facility (PSTF). These tests were initiated for the Mark III concept because of configuration differences between the previous containment concept and the Mark III design. More sophisticated instrumentation was available for the Mark III tests, as were computerized methods for data analysis. It was from the PSTF testing that the short-term dynamic effects of drywell air being forced into the pool in the initial stage of the postulated LOCA were first identified. This air injection into the suppression pool water results in a pool swell event of short duration. In this event, a slug of water rises and impacts the underside of structural components within the suppression chamber.

In addition to the information obtained from the PSTF data, other LOCA-related dynamic load information were obtained from foreign testing programs for similar pressure-suppression containments. In these foreign tests, oscillatory condensation loads which occur during the later stages of a postulated LOCA were identified.

Also, experience at operating plants indicated that Safety Relief Valve (SRV) discharges to the suppression pool would cause oscillatory hydrodynamic loads within the suppression chamber. Both the LOCA and SRV discharge are characterized by an initial short period injection of air into the suppression pool, followed by a longer period of steam discharge into the suppression pool.

Further, during in-plant testing of Mark I containments, new suppression pool hydrodynamic loads which had not explicitly been included in the original Mark I containment design basis were identified. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA and from suppression pool response to safety relief valve (SRV) operation generally associated with plant transient conditions.

These hydrodynamic loads had not been considered in the original design of the Mark I containment, the NRC required that a detailed reevaluation of the Mark I containment system be made.

In February and April 1975, the NRC transmitted letters to all U.S. utilities owning BWR facilities with the Mark I containment system design, requesting that the owners quantify the hydrodynamic loads and assess the effect of these loads on the containment structure. The February 1975 letters reflected NRC concerns about the dynamic loads from SRV discharges, while the April 1975 letters indicated the need to evaluate the containment response to the newly identified dynamic loads associated with a postulated design basis LOCA.

As a result of these letters from the NRC, and recognizing that the additional evaluation effort would be very similar for all Mark I BWR plants, the affected utilities formed a Mark I Owners Group, and GE was designated as the Group's lead technical organization. The objectives of the group were to determine the magnitude and significance of these dynamic loads as quickly as possible and to identify courses of action needed to resolve any outstanding safety concerns. The Mark I Owners Group divided this task into two programs: A Short-Term Program and a Long-Term Program.

The objectives of the Short-Term Program (STP) were to verify that each Mark I containment system original configuration would maintain its integrity and functional capability when subjected to the most probable loads induced by a postulated design basis LOCA, and to verify that the licensed Mark I BWR facilities could continue to operate safely without endangering the health and safety of the public while a methodical, comprehensive Long-Term Program (LTP) was being conducted.

The STP structural acceptance criteria used to evaluate the design of the torus and related structures were based on providing adequate margins of safety; i.e., a safety-to-failure factor of 2, for continued operation of the plant's original configuration before the more detailed results of the LTP were available.

The NRC concluded that a sufficient margin of safety had been demonstrated in the STP to ensure the functional performance of the containment system and, therefore, any undue risk to the health and safety of the public was precluded. These conclusions were documented in the "Mark I Containment Short-Term Program Safety Evaluation Report," NUREG-0408, dated December 1977. The NRC granted the operating Mark I facilities an exemption relating to the structural factor of safety requirements of 10 CFR 50.55(a) for an interim period while the more comprehensive LTP was being conducted.

The objectives of the LTP were to establish conservative design basis loads that are appropriate for the anticipated life of each Mark I BWR facility, and to restore the originally intended design safety margins for each Mark I containment system. The plans for the LTP and the progress and results of the program were reviewed with the NRC throughout the performance of the program.

The LTP consisted of:

- The definition of loads for suppression pool hydrodynamic events
- The definition of structural assessment techniques
- The performance of a plant-unique analysis (PUA) for each Mark I facility in the U.S.

The generic aspects of the Mark I Owners Group effort were completed with the submittal of the "Mark I Containment Program Load Definition Report" (LDR) and the "Mark I Containment Program Structural Acceptance Criteria, Plant Unique Analysis Application Guide" (PUAAG). The NRC concluded that load definitions and structural acceptance criteria documented in these two reports were acceptable for use in the plant-unique analysis of each plant. The NRC conclusions and comments were presented in the "Mark I Containment Long-Term Program

Safety Evaluation Report, NUREG-0661," dated July 1980 and Supplement Number 1 to NUREG-0661.

GE provided reports to each Mark I BWR facility with plant unique LOCA hydrodynamic loads. GE also prepared and provided a supplementary generic load definition report and supporting application guides which provided the load definition procedures for the postulated LOCA and SRV actuation events for use in the structural re-evaluation of the pressure suppression chamber, vent system, SRV discharge piping, and other Mark I containment components.

The NRC reviewed the generic criteria and analysis techniques developed for the LTP for re-assessment of the Mark I containment. The NRC provided an acceptance criteria for application of the LTP loads assessment process in US NRC NUREG-0661, "Safety Evaluation Report, Mark I Containment Long-Term Program," July 1980 and NUREG-0661, Supplement 1, issued August 1982 (Refer to Attachment 4 for additional information regarding NUREG-0661)

Each BWR Mark I plant performed a plant-unique re-assessment of the Mark I containment, applying the load definition process developed by the LTP and implementing the NRC acceptance criteria of NUREG-0661. The objective of this reassessment was to either demonstrate that the existing plant design has the required safety margin or to identify any additional plant modifications that were necessary to restore the intended margins of safety in the containment design. The reports included, as appropriate, documentation of evaluation performed for modifications to the original design (the plant-unique assessments led to structural modifications to the torus and vent system at most US BWR's with Mark I containments to ensure compliance with the applicable criteria). The plant-unique analyses documented the efforts to address each of the applicable NUREG-0661 requirements and demonstrated with the NUREG-0661 acceptance criteria that the design of the containment is adequate and that the original design safety margins are either confirmed with the original design or restored with the modifications. Each Mark I BWR facility submitted its plant-unique analysis report to the NRC for approval, since each plant has an individual license with the NRC.

CONTAINMENT OPERATION DURING A STATION BLACKOUT

In the late 1980s and early 1990s, BWR operators made procedure changes and modifications to cope with events which involved the loss of the normal offsite power and normally available emergency diesel generators as discussed in NRC Regulatory Guide 1.155 (Station Blackout).

To support safe operation in a variety of circumstances, the plants have been designed and have developed procedures to address a wide range of potential events. The Emergency Operating Procedures provide instructions for maintaining adequate core cooling and protection of the reactor vessel and containment under a variety of prescribed emergency conditions. If adequate core cooling cannot be maintained, radiological emergency response procedures provide instructions for plant staff to take actions to mitigate the consequences of an event that could lead to radioactive material release to the public and provides for making recommendations to state and local agencies to take action to protect the health of the public such as evacuation or sheltering. The industry has also implemented Severe Accident Management Guidelines (SAMG) to diagnose and mitigate severe accidents. These operating guidelines include steps for

dealing with challenges to containment integrity and reactor coolant loss beyond the original plant design basis. This includes methodology to use auxiliary equipment that is not driven by normal plant power sources to provide makeup water to the reactor vessel/containment. These guidelines interface with Emergency Operating Procedures to mitigate a loss of large areas of the plant and with state and local radiological emergency response procedures.

As a result of the September 11, 2001 terrorist attacks, additional actions and equipment were put in place at certain U.S. plants to allow water makeup to the reactor and the fuel pools should significant damage occur to the reactor buildings. These changes include pre-staged diesel-driven pumps, piping, and procedures that would support water makeup from various water supplies without the need for electrical power.

Also, as a result of the Industry Degraded Core Rulemaking (IDCOR) and NRC programs for Severe Accident Closure, recommended that all Mark I U.S. nuclear power plants add a containment venting capability. This containment vent was designed as a hard pipe that would discharge from the containment in the case of a BWR from the wetwell or drywell, and discharge to an elevated release, such as the plant stack. All U.S. Mark I nuclear reactors have installed this containment venting modification.

In summary, this vent allows operators to protect the integrity of the primary containment as well as preventing a ground-level release for the severe accident scenarios beyond the design and licensing basis (Refer to Attachment 5 for additional modification information specific to BWR Mark I containment).

CONCLUSION

The Mark I containments in currently operating BWRs have been designed to meet the specific provisions of 10 CFR 50 Appendix A, General Design Criteria 2, 16, 50 and 51 for containment design or the applicable equivalent regulation at the time of licensing.

The GE Mark I containment systems in U.S. BWRs have undergone extensive testing and analysis and have been modified to meet NRC regulations. The Mark I pressure suppression containment is a proven technology that has been enhanced with confirmatory testing, enhanced knowledge and advanced analysis over time. It meets all regulations and has been certified by review of the NRC through a Safety Evaluation Report (SER) at each Mark I plant under comprehensive, NRC-mandated Mark I Containment Program re-analyses performed to address the evolving design loading conditions.

The Mark I containment also has many features inherent in its design that make the probability of a severe accident extremely low. They have been modified throughout their operation to provide additional features and response capabilities to further reduce this probability.

Also, to ensure containment integrity, the drywell or primary containment, as called, is tested at established intervals (every 10 to 15 years) in accordance with 10 CFR 50 Appendix J Program (Primary Reactor Containment Leakage Testing). This test is used to demonstrate containment integrity and to demonstrate it will perform its safety function by verifying that leakage through

the containment, and systems and components penetrating the primary containment, shall not exceed established limits. The containment and associated systems and components penetrating containment are designed to provide the final barrier in preventing the release of quantities of radioactive material that would have a significant radiological effect on the health of the public. This program also uses periodic surveillance testing to demonstrate the leak tightness.

Additional procedures, hardware and resources have been planned and prepared for the beyond-design-basis scenarios to assure protection of the safety and health of the public. In addition, 10 CFR 50, Appendix B, also assures that any conditions adverse to quality be identified and resolved. It requires assuring that the cause of the condition is determined and corrective actions taken to preclude repetition.

Attachments

U.S. Nuclear Plants With Mark I Containment

Reactor Name	State
Browns Ferry 1	Alabama
Browns Ferry 2	Alabama
Browns Ferry 3	Alabama
Brunswick 1	North Carolina
Brunswick 2	North Carolina
Cooper	Nebraska
Dresden 2	Illinois
Dresden 3	Illinois
Duane Arnold	Iowa
Edwin I. Hatch 1	Georgia
Edwin I. Hatch 2	Georgia
Fermi 2	Michigan
Hope Creek 1	New Jersey
James A. Fitzpatrick	New York
Monticello	Minnesota
Nine Mile Point 1	New York
Oyster Creek 1	New Jersey
Peach Bottom 2	Pennsylvania
Peach Bottom 3	Pennsylvania
Pilgrim 1	Massachusetts
Quad Cities 1	Illinois
Quad Cities 2	Illinois
Vermont Yankee 1	Vermont

Analyzed Loading Conditions For Mark I Containments

Hydrodynamic Loads evaluated a spectrum of postulated pipe breaks to determine the worst loading condition for each structural element. For the long-term program, an intermediate liquid break accident (IBA) and a small steam break accident (SBA) were specified in addition to the Design Basis Accident DBA. All LOCA and seismic loads are added together as appropriate for the load combination scenario. Not all of the suppression pool hydrodynamic loads can occur at the same time. In addition, the load magnitudes and timing will vary, depending on the accident scenario under consideration. Therefore, combinations of loading conditions have been determined from typical plant primary system and containment response analyses, with considerations for automatic actuation, manual actuation, and single active failures of the various systems in each event. The typical new loads analyzed were:

- Pressure and temperature time histories for the suppression chamber wetwell and drywell
- Vent system pressurization and thrust loads
- Net vertical pool swell loads and average submerged pressures on the suppression chamber
- Pool swell impact and drag loads on the vent system
- Pool swell froth impingement loads
- Pool fallback loads
- Vent header deflector loads
- Condensation oscillation loads and chugging loads
- Fluid structure interaction
- Safety-relief valve discharge loads
- Submerged structure drag loads
- Secondary effects loads
- Seismic slosh/loads, which occurs due to horizontal seismic motion on the pool
- Post-pool-swell waves/loads, due to the wave action associated with continued flow through the downcomers
- Asymmetric vent system flow, resulting from asymmetric flowrates due to vent blockage
- Downcomer gas-clearing loads, resulting from the rapid clearing of gas from the vent system causing lateral loads as bubbles are being formed in the pool
- Sonic and compression wave loads, due to the shock wave propagating from the break location
- Safety-relief valve steam discharge loads

Original loads included pressure and temperature loads associated with a LOCA, seismic loads, dead weight loads, jet impingement loads, hydrostatic loads due to water in the suppression chamber, overload pressure test loads, and construction loads.

The generic aspects of the Mark I Owners Group LTP were completed with the submittal of the Mark I Containment Program Load Definition Report (LDR), and the Mark I Containment Program Structural Acceptance Guide (PUAAG), as well as supporting reports on the LTP experimental and analytical tasks. The generic analysis techniques were used to perform a plant unique analysis to confirm the adequacy of the modifications made to the containment structures and related piping. This analysis was documented in the Plant Unique Analysis Report (PUAR), which shows that the original margins of safety in the containment design have been restored.

Recommended Modifications For Mark I Containments

The Mark I containment was originally designed based on large-scale experimental tests in 1958 through 1962. More advanced large-scale tests in 1972 through 1974 and actual plant operations identified some new phenomena and issues needing resolution. The Mark I containment program began in 1975 when NRC sent letters to Mark I owners requiring Reevaluation of Containment Response to Hydrodynamic Loads. The BWR Owners Group embarked on a program to resolve the issues. The issues were highly scrutinized and reviewed for plant-specific applicability on operations and structural capability. In summary, dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA and from suppression pool response to SRV operation generally associated with plant transient operating conditions were addressed for:

- Loss-of-Coolant-Accident-Related Hydrodynamic Loads including Pool Swell Phenomena
- Loss-of-Coolant-Accident Steam Condensation Phenomena
- Safety-Relief Valve Discharge-Related Hydrodynamic Loads

The new experimental data and new analytical models extensively studied the issues. Immediate operability of the existing structures was demonstrated for the short-term then more robust structural improvements were implemented. Operational changes and strengthened structural supports at specific locations were implemented to meet ASME and other industry acceptance criteria. A typical list of hardware changes made:

- Torus: Additional ring girder reinforcement, Miter joint support saddles and saddle extension plates, Additional ring-girder-to-torus weld, Torus Temperature monitoring instrumentation, Torus tie-downs, Dynamic restraint snubbers
- Vent System: Downcomer/vent header stiffeners, Downcomer lateral bracing, Downcomer longitudinal bracing, Vent header deflector, Vent line drain reinforcement, Torus-to-drywell vacuum breakers, and Vacuum breaker header support
- Internal Structures: Catwalk midbay supports, Catwalk lateral bracing, Catwalk supports at ring girders, Conduit rerouted
- Wetwell Piping Modifications (Internal): Spray header supports, HPCI turbine drain pot support, HPCI turbine exhaust line support, ECCS suction strainer reinforcement, LPCI full-flow test line supports, modify external supports
- Relief Valve Discharge Line Piping: Reinforced vent line penetration, Added T-quencher, Added T-quencher supports, Added SRV line support, SRVDL vacuum breaker

- Torus Hardened Vent
- Torus Vacuum Breaker orientation changes
- SRV operating – recommendations to minimize loading, discharge piping pressure switch, instrumentation to allow SRV position monitoring

NUREG - 0661 ABSTRACT July 1980

“This Safety Evaluation Report prepared by the staff of the Office of Nuclear Reactor Regulation discusses suppression pool hydrodynamic loads in boiling water reactor (BWR) facilities with the Mark I pressure-suppression containment design. The report finishes the NRC’s Generic Technical Activity A-7 (Mark I Containment Long-Term Program), which has been designated an “Unresolved Safety Issue.” The report describes the generic techniques for the definition of suppression pool hydrodynamic loads in a Mark I system and the related structural acceptance criteria.

On the basis of a review of the experimental and analytical programs conducted by the Mark I Owners Group, the staff has concluded that, with one exception, the proposed suppression pool hydrodynamic load definition procedures (as modified by the staff’s requirements in Appendix A of this report) will provide conservative estimates of these loading conditions. The exception is the lack of an acceptable specification for the downcomer “condensation oscillation” loads. In addition, requirements for confirmatory analyses and testing have been identified. The resolution of these issues will be described in a supplement to this report.

The staff also has concluded that the proposed structural acceptance criteria are consistent with the requirements of the applicable codes and standards. In conjunction with the general structural analysis techniques, these criteria will provide an acceptable basis for establishing the margins of safety in the Mark I containment design.”

NUREG-0661 SUPPLEMENT 1 ABSTRACT August 1982

“When the NRC staff published "Safety Evaluation Report, Mark I Containment Long-Term Program" (NUREG-0661) in July 1980 four areas were identified where the technical issues had not been fully resolved. These were:

1. Specification for condensation oscillation loads acting on the down comers
2. Adequacy of the data base for specifying torus wall pressures during condensation oscillations,
3. Possibility of asymmetric torus loading during condensation oscillations, and
4. Effect of fluid compressibility in the vent system on pool swell loads.

The first item, downcomer condensation oscillation loads, lacked an acceptable load definition. The remaining three items had acceptable specifications; however, the NRC requested additional confirmatory information to justify the adequacy of the load specifications.

This supplement addresses the resolution of the four issues listed above. In response to NRC concerns expressed in NUREG-0661, the Mark I Owners Group conducted additional experimental and analytical studies. The experimental studies consisted basically of two additional condensation oscillation tests in the Full-Test Facility (Norco, California). The staff has reviewed these efforts and has concluded that all technical issues connected with the generic Mark I Long-Term Program have been resolved.”

Summary:

NUREG 661 Contains a safety evaluation of loading determination methods and acceptance criteria for the evaluations that resulted in the modifications made as part of the Long Term Program.

NUREG 661 Supplement 1 contains a safety evaluation of the four unresolved issues described in NUREG 0661.

Summary of Critical Modifications

The following modifications are the more significant examples of improvements to the Mark I containment design as enhancements to design margins or increased capability to address “beyond design basis conditions.”

Hardened Vent

In the 1980s, the NRC staff reviewed the potential for accidents more severe than those the plants were licensed and designed to mitigate. In order to enhance the ability of all containments to prevent and mitigate the consequences of severe accidents beyond the design basis accidents, the NRC requested that all plants install a hardened vent. In the event of a core damage accident, the hardened vent would allow reduction in containment post-accident pressure. Hardened meant that the vent would transport hydrogen, steam and other accident products and release them outside the reactor building. This would preclude damage to the reactor building and equipment from steam and the possible hydrogen explosions.

In response to the GL 89-16 and the results of their Individual Plant Examinations (IPEs), all of the nuclear power plants installed a hardened vent or demonstrated to the NRC’s satisfaction. This hardened vent capability would allow BWR plant operators to vent post-accident airborne materials by passing them through the water in the wetwell, thereby removing a large amount of the radioactive material contained in the vent stream before venting. This added capability increases the ability of the Mark I containment to mitigate the consequences of design-basis and beyond-design-basis accidents.

SRV Modifications

Mark I containments were designed for hydrostatic loads as part of the original design basis of the plants. During subsequent testing of the Mark III containment design, additional hydrodynamic loads were identified. The industry established a Mark I Owners’ Group and developed test regimes to define the hydrodynamic loads and establish design criteria for evaluation of the Mark I containment.

The hydrodynamic loads are defined in Plant Unique Load Definition Reports and each Mark I containment was evaluated against its own report. As a result of this effort, Mark I plants modified their piping and/or pipe supports inside containment to reduce the loads an SRV discharge would impose on the torus and to withstand post-LOCA loads on SRV discharge piping. These modifications included pipe supports, discharge line vacuum breakers, “rams head” discharge pipe fittings and T-quenchers. The modifications provide additional margin to ensure that SRV actuation will not damage the wetwell pressure boundary, any of the wetwell internal components or piping, and also ensure that the SRV piping will remain functional during postulated transients and accidents.

Torus Attached Piping

In addition to the SRV piping modifications discussed above, the plant-unique analysis identified new loads on torus attached piping (TAP). In order to ensure the structural integrity of the torus for all postulated loads, all TAP was re-evaluated and either removed or modified as necessary. The piping was evaluated, and upgrades were made to torus attachments and pipe supports to ensure that they would withstand all postulated loads with a significant factor of safety between the design load and the actual load. These modifications ensure that the Mark I containment will continue to perform its safety function during plant transients and postulated accidents as well as improve the capability to mitigate beyond-design-basis events.

ECCS Suction Strainers

NRC Bulletin 96-03 identified NRC concerns with the potential for debris plugging of Emergency Core Cooling System (ECCS) suction strainers inside the containment wetwell. These concerns resulted from an event in Sweden and two events in the U. S. that indicated the possibility that fibrous insulation material dislodged by a design-basis accident and particulate material suspended in the wetwell water could be entrained on strainer surfaces causing loss of the ECCS pumps. These pumps are used for normal reactor shutdown and to inject cooling water into the reactor following an accident that depressurizes it.

The plant owners employed the BWR Owners' Group (BWROG) and the Electric Power Research Institute (EPRI) to conduct testing to determine criteria for designing and, particularly, sizing the strainers to ensure ECCS pumps capability if needed. Each plant owner then designed and installed new larger strainers. These strainers provide additional assurance that ECCS pumps will perform their intended function in the unlikely event of an accident.