ENCLOSURE 2
BWR MARK I AND MARK II CONTAINMENT REGULATORY HISTORY

CONTENTS

1. INTRODUCTION

A key element of the design of nuclear power plants is the inclusion of multiple barriers to the potential release of radioactive materials created within the fuel by the fission process. In the United States, a containment barrier has always been included to confine the fission products within the plant should an accident lead to a compromise of the barriers provided by the fuel design and the reactor coolant pressure boundary. This philosophy was described in a report prepared in 1965 for the U.S. Atomic Energy Commission (AEC) by Oak Ridge National Laboratory (ORNL) that compiled the early practices and approaches for containment designs. The report provided the following summary:

> The need for a containment system in the large power reactor installation is well established by convention and precedent in the United States, and the specific design requirements are determined by the reactor safety analysis. Philosophically, containment is provided so that the risk that cannot be disassociated from the operation of a particular reactor can be reduced to acceptable proportions with respect to the corresponding gain that is expected to result from its operation. However, such a balance of gain versus risk is impossible to attain on a quantitative basis, and only the risk enters into the evaluation that is made in connection with every reactor safety analysis. The specific function of the containment system is to reduce the consequences of the maximum credible accident so that a particular facility may fulfill siting requirements as defined in the Code of Federal Regulations. On this basis, containment systems may be called upon to effect a reduction in the activity released in an accident by a factor of 10^2 to 10^5.
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> The accident that could occur and would have associated with it the most severe set of consequences as far as the radiation exposure of offsite personnel is termed the “maximum credible accident” (mca). Although this accident is a characteristic of a given plant, there are only two types of accidents that comprise the mca. The first is the loss of coolant accident, with subsequent core melting or possible nuclear excursion and release of fission products. The second is the fuel handling accident in which a fuel element, or assembly, is dropped or allowed to fail in such a way that its fission products are released. After these initiating events occur, the released fission products disperse through the system and leak to the environment at some rate determined by the containment vessel in question.

For currently operating plants, this barrier is provided by containments that include either (1) a large enough air volume to address the energy released from a design basis loss of coolant accident (LOCA) while not exceeding the design pressure for the containment, or (2) systems that include water or ice to absorb the energy released from a LOCA and thereby suppress the increase in pressure to values below the design limits for the containment. Boiling-water reactors (BWRs) employ such pressure suppression containment designs. Mark I and Mark II are specific containment configurations for BWRs that use water suppression pools to remove energy from the reactor following a LOCA or other plant transients or accidents. The pressure suppression designs were summarized as follows in the early ORNL report:

> In an effort to reduce the cost of containment, the concept of pressure suppression has been employed with water-cooled reactors. In principle, this technique is especially suited to water-cooled reactors, since the major portion of

> the energy released upon occurrence of an mca is in the form of saturated steam, which may be removed by condensation and thereby greatly reduce the final pressure to be withstood by the containment building. This scheme uses the "dry well" and vent piping to direct the steam that is released into the water of the suppression pool, where the steam is condensed and fission products may be partially removed.

As mentioned above, the primary focus of containment designs was, and largely remains, the demonstration that it addresses the "maximum credible accident" and limits the potential exposure of the public from radioactive materials. The maximum credible accident and its role in siting decisions and containment functions was described as follows in another early and key guidance document, TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," (AEC-1962):

> In evaluating proposed reactor sites, the basic safety questions involve the possibility of accidents which might cause radioactivity release to areas beyond the site, the possible magnitudes of such releases and the consequences these might have. Practically, there are two difficult aspects to the estimation of potential accidents in a proposed reactor which affect the problem of site evaluation.
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> (1) The necessity for site appraisal arises early in the life of a project when many of the detailed features of design which might affect the accident potential of a reactor are not settled.
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> (2) The inherent difficulty of postulating an accident representing a reasonable upper limit of potential hazard.
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> In practice, after systematic identification and evaluation of foreseeable types of accidents in a given facility, a nuclear accident is then postulated which would result in a potential hazard that would not be exceeded by any other accident considered credible during the lifetime of the facility. Such an accident has come to be known as the "maximum credible accident".
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> For pressurized and boiling water reactors, for example, the "maximum credible accident" has frequently been postulated as the complete loss of coolant upon complete rupture of a major pipe, with consequent expansion of the coolant as flashing steam, meltdown of the fuel and partial release of the fission product inventory to the atmosphere of the reactor building. There may be other combinations of events which could also release significant amounts of fission products to the environment, but in every case, for the events described above to remain the maximum credible accident the probability of their occurrence should be exceedingly small, and their consequences should be less than those of the maximum credible accident. In the analysis of any particular site-reactor combination, a realistic appraisal of the consequences of all significant and credible fission release possibilities is usually made to provide an estimate in each case of what actually constitutes the "maximum credible" accident. This estimated or postulated accident can then be evaluated to determine whether or not the criteria set out in 10 CFR 100 are met. As a further important benefit,

> such systematic analyses of potential accidents often lead to discovery of ways in which safeguards against particular accidents can be provided.
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> Since a number of analyses have indicated that the pipe rupture-meltdown sequence in certain types of water cooled reactors would result in the release of fission products not likely to be exceeded by any other "credible" accident, this accident was designated the "maximum credible accident" (MCA) for these reactors. The remainder of this discussion will refer chiefly to this type of reactor and this type of accident. Corresponding maximum credible accidents can by similar analyses be postulated for gas-cooled, liquid metal cooled, and other types of reactors.

The above discussion remains largely relevant today as the limits in Title10 of the *Code of Federal Regulations* (10 CFR) Part 100, "Reactor Site Criteria," are unchanged, and some plants continue to be evaluated using the estimates in TID-14844 to assess the adequacy of containment designs.[1] Other aspects of the containment design and evaluation are also derived from the establishment of a large pipe break as the maximum credible accident. Such design requirements include the ability of structures, systems, and components to withstand the pressures, temperatures, and hydrodynamic forces associated with pipe breaks within the containment, as well as withstanding external hazards such as seismic events.

There have been several significant issues related to the performance of BWR containments during design-basis accidents. These problems and their resolution are discussed in Section 4, "Other Design Issues," but are not related to the primary issue of this paper, which deals with beyond-design-basis accidents and the importance of containment venting during such scenarios.

In SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," dated May 25, 1988, the U.S. Nuclear Regulatory Commission (NRC) staff presented to the Commission its plan to evaluate potential generic severe accident containment vulnerabilities in a research effort entitled the containment performance improvement (CPI) program. This effort was predicated on the presumption that there are generic severe accident challenges to each light water reactor (LWR) containment type that should be assessed to determine whether additional regulatory guidance or requirements concerning needed containment features were warranted, and to confirm the adequacy of the existing Commission policy. These assessments were needed because of the uncertainty in the ability of LWR containments to successfully survive some severe accident challenges, as indicated by the results documented in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,". All LWR containment types were assessed in the CPI program, beginning with the boiling-water reactors (BWRs) with Mark I containments. The potential improvements for BWRs with Mark I containments were documented in NUREG/CR-5225 (including Addendum 1), "An Overview of BWR Mark-I Containment Venting Risk Implications," and SECY-89-017, "Mark I Containment Performance Improvement Program," dated January 23, 1989. The potential improvements for Mark II containments were published in NUREG/CR-5528, "An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements."

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1 Licensees are allowed but not required by NRC regulations defined in 10 CFR 50.67. "Accident Source Term," to use revised accident source terms to take advantage of research and knowledge gained since the issuance of TID-14844.

2. BWR MARK I AND MARK II CONTAINMENTS

The key design attributes of Mark I and Mark II containments relevant to the need for containment venting during severe accidents such as Fukushima are: (1) the containment free gas volumes are relatively small compared to other light-water reactors, so gas and steam buildup in containment will cause the pressure to rise more dramatically, (2) BWR reactor cores have about three times the zirconium inventory compared to pressurized-water reactors (PWRs) with comparable power levels, so there is a greater potential to generate significant amounts of hydrogen gas which also will increase containment pressures. These design attributes, in comparison with other containment types, are illustrated in Figures 1 and 2.

2.1 Mark I Containment Designs

As shown in Figure 3, the Mark I containment design is a drywell in the shape of an inverted common incandescent light bulb containing the reactor vessel and primary piping attached with several large vent pipes to a torus shaped suppression chamber located below the drywell. The steam escaping from the break in the reactor coolant piping would vent, along with the drywell atmosphere, down into the suppression chamber. It would be distributed through a header to many downcomer pipes whose open ends were submerged in the suppression pool, which fills about half the suppression chamber.

Presently, worldwide a total of 37 commercial nuclear power units (reactors) use a Mark I-type (drywell /toroidal suppression pool) pressure suppression containment. Twenty-three—or roughly 60 percent—are licensed by the NRC to operate in the United States. All but one (Fermi 2) have been granted a license extension, with the earliest expiring in 2029 (Dresden 2) and the latest expiring in 2038 (Hatch 2). Twenty have been granted a power uprate between 1.5 percent (Pilgrim) and 20 percent (Brunswick 1, 2). Additional information is provided in Table 3.

Table 1. BWR Mark I Containments by Country

Country	Number	Name
US	23	See Table 3
Japan	8	Fukushima I 1-5 Hamaoka 1 Shimane Tsuruga
India	2	Tarapur 1,2
Taiwan	2	Chinsan 1,2
Spain	1	Santa Maria de Garona
Switzerland	1	Muehleberg

The General Electric (GE) BWR Mark I containment was an early design and evolutionary step in the development of the containment technology seen in the industry today. As knowledge and experience were acquired, shortcomings in the understood safety margins were identified and assessed. Over time, extensive improvement modifications have been made to restore those safety margins (See Section.4 in this Enclosure).

The Mark I pressure-suppression concept containment design was based on experimental information obtained from testing performed for the Humboldt Bay and Bodega Bay Power Plants. (The Humboldt Bay Nuclear Power Plant was rated for 63 megawatts electric (MWe) operated from August 1963 to July 1976 just south of Eureka, California. The Bodega Bay Power Plant was to be rated for 313 MWe, but construction at the site 50 miles north of San Francisco was cancelled about 1964.)

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 loss-of-coolant-accidents (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 for establishing containment design pressures. The data from these tests were the primary experimental bases for the design and the initial staff approval of the Mark I containment system. Dresden Generating Station (also known as Dresden Nuclear Power Plant or Dresden Nuclear Power Station) was the first privately financed nuclear power plant built in the United States. Dresden Unit 1, which had a Mark I type containment, received a construction permit in 1959, and was decommissioned in 1978.

Given that the primary function of this containment is to contain radioactive material following an accident, designers and regulators are faced with a challenge when it comes to maintaining the integrity of the containment when it is challenged by high pressures. Historically, primary containment pressure control to prevent structural failure, and thus unrecoverable loss of the primary function, was to be achieved by multiple, diverse active and passive systems (spray, fan-coolers, vents to suppression pools) and not by a simple relief valve or rupture disk discharging containment atmosphere directly to the environment as would be the practice for most other pressure vessels Thus, the American Society of Mechanical Engineers (ASME) created an exception to the general practice of requiring a passive relief device in the ASME Boiler and Pressure Code Section III Article NE-7000, which states:

> A containment vessel shall be protected from the consequence arising from the application of conditions of pressure and coincident temperature that would cause the Service or Test Limits specified in the Design Specification to be exceeded. Pressure relief devices are not required where the Service or Test Limits specified are not exceeded. It is recognized that the fundamental purpose of a containment vessel may be nullified by the incorporation of pressure relief valves discharging directly into the environment.

However, a controlled (and potentially filtered) release was identified as a favorable alternative to catastrophic failure of the containment. Subsequent to the Three Mile Island Unit 2 nuclear plant core melt event in 1979, NUREG-0585, “TMI-2 Lessons Learned Task Force Final Report,” October 1979, stated:

> Available studies indicate that controlled venting of the containment to prevent failure due to overpressure could be-an effective means of delaying ultimate containment failure by melting through. If appropriately filtered to partially decontaminate the gases that would be released in order to avoid over-pressurization, such venting may significantly reduce the consequences and risk from core-melt accidents… It appears to us that sufficient studies have been completed to support a preliminary conclusion that controlled filtered venting of containments is an effective and feasible means of mitigating the consequences of core-melting.

As probabilistic risk assessment (PRA) methods continued to mature, the Reactor Safety Study, "An Assessment of Risks in U.S. Commercial Nuclear Power Plants [NUREG-75/014 (WASH 1400)]," found that, for the Peach Bottom BWR Mark I nuclear plant, even though the core melt probability was relatively low, the containment could be severely challenged if a large core melt occurred. Based on this conclusion, and reinforced by the anticipation of similar findings (subsequently confirmed) in the draft Reactor Risk Reference Document (NUREG-1150, February 1987) a five element program was proposed in June 1986 to enhance the performance of the BWR Mark I containment. After the initial proposal, the staff held two separate meetings in early 1987 with researchers representing NRC contractors and industry. There was a wide range of views expressed regarding accident phenomenology as well as the efficacy of the various improvements. In view of the lack of technical consensus on the effectiveness of the proposed improvements, the staff decided to undertake additional efforts. In July 1987, the staff briefed the Commission on an integrated approach to resolve all severe accident issues, including matters relating to BWR Mark I containments. The integrated approach was to be comprised of four main programs: (1) the Individual Plant Evaluation Program (IPE), (2) the Containment Performance Program, (3) a program to improve plant performance, and (4) a program to implement guidance on Severe Accident Management Strategies.

The staff proposed a broad-based plan in December 1987 to address the performance issues of Mark I containments (SECY-87-297). The proposal listed several, relatively low-cost improvements whose purpose was to substantially mitigate potential offsite releases. This list of possible improvements included: hydrogen control, alternate water supplies for the containment spray system, venting, core debris control, enhancing reactor building fission product attenuation, basemat isolation, improving the automatic depressurization system, and improving existing emergency procedures and training to include coping with severe accidents.

SECY-87-297 also laid out a two-stage strategy to attempt resolving such a large-scale set of technical issues. The first stage would consist of characterizing an issue and performing parametric studies and experimental assessments to assist in focusing on the most relevant technical aspects. After initial issue characterization, a meeting would be held with representatives from the staff, contractors, the industry, and other experts and interested members of the public on each issue. During the second stage, the staff would evaluate and sort each issue into one of three categories: (1) resolved or unimportant, (2) potentially resolvable by future research, or (3) candidates for regulatory initiatives.

The staff returned to the Commission in January 1989 to present recommendations on Mark I containment performance improvements and other safety enhancements (SECY-89-17, "Mark 1 Containment Performance Improvement Program"). In that paper, the staff described their findings associated with examining six areas of potential improvement for Mark I containments. These were: (1) hydrogen control, (2) alternate water supply for reactor vessel injection and containment drywell sprays, (3) containment pressure relief capability (venting), (4) enhanced reactor pressure vessel (RPV) depressurization system reliability, (5) core debris controls, and (6) procedures and training. Each area was evaluated to determine the potential benefits in terms of reducing the core melt frequency, containment failure probability, and offsite consequences.

The staff concluded there was no significant risk reduction associated with additional hydrogen control (beyond the existing rule, see Hydrogen Control section below for details). The primarily reason was because, during a severe accident, reactor pressure is anticipated to increase, releasing steam and noncondensable gases into the containment. This would increase containment pressure, preventing ingress of air. Therefore, the containment atmosphere would not become de-inerted for an extended period of time. Since offsite supplies of nitrogen could readily be obtained during this period, an onsite backup supply of nitrogen would not significantly reduce risk.

Additionally, the staff determined that more research was necessary to ensure the technical feasibility of core debris controls (e.g. curbs in the drywell or curbs or weir walls in the torus room under the wetwell). The design and installation costs, as well as the occupational exposure during installation, were also significant deterrents from pursuing further actions in this improvement area.

Aside from these two exceptions, the staff provided cost-justification for, and recommended implementation of, all the aforementioned improvements including: (1) improved hardened venting capability, (2) improved RPV depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) emergency procedures and training.

In the subsequent SRM, however, the Commission concluded that the majority of the staff's recommended safety improvements would be evaluated by licensees as part of the IPE Program. The only exception was the hardened vent capability recommendation. The Commission directed the staff to approve installation of hardened vents under the provisions of 10 CFR 50.59, "Changes, Tests, and Experiments," for licensees that would voluntarily implement this improvement and perform a back-fit analysis for requiring a hard vent installation at those plants who declined voluntary installation. Thus, Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," was issued in September 1989 providing an example of an acceptable design that used the suppression pool to achieve as much reduction in effluent radioactivity as possible without the cost of an external filter making the change more cost-beneficial.

In response to the issuance of the generic letter, all Mark I licensees installed a version of a hardened vent under 10 CFR 50.59. The Boiling Water Reactor Owners' Group (BWROG) developed a general design criteria document that was subsequently approved by the staff (with clarifications).

The hardened vent was specifically to provide an exhaust line from the wetwell vapor space to a suitable release point (e.g. stack, reactor building or turbine building roof). The basic design objective of the hardened vent was to mitigate the loss of decay heat removal accident sequence. As such, the piping was designed (sized) to accommodate a steam flow equivalent of 1 percent decay heat power assuming a pressure equal to the primary containment pressure limit (PCPL), and not designed for operation during a severe accident.

The staff requested that the capability for the initiation (although not termination) of utilizing the hardened vent be in the control room, and that radiation monitoring devices be required to alert control room operators of radioactive releases during venting. It was proposed in the staff recommendation in SECY 89-17 that the hardened vent isolation valves be capable of being opened from the control room under station blackout conditions beyond the then-established coping time; however, the generic letter only requested that the licensee include costs for electrical modifications in a plant-specific basis for why the vent was not cost beneficial if a vent was not voluntarily installed. The installed vents in most cases were dependent on alternating current power.

The newly installed hardened vents were subject to pre-existing technical specifications for containment isolation valves and containment integrity, but the system itself had no imposed limiting conditions of operation (LCO) or surveillance requirements. The valves were, however, subject to the local leak rate testing and inservice testing requirements (10 CFR 50.54(o) and 10 CFR 50.55a(f), respectively) of all containment penetrations and isolation devices.

2.2 Mark II Containment Designs

The Mark II containment concept (Figure 4) evolved the drywell and suppression pool to a simpler truncated cone over the cylindrical suppression chamber. Currently, there are a total of 17 commercial nuclear power units using a Mark II-type pressure suppression containment worldwide. The NRC has granted operating licenses to eight of these BWRs with Mark II containments on five different sites. Columbia, Nine Mile Point 2, and Susquehanna 1&2, have also been granted license extensions, and the application for license extension at Limerick was received by the NRC in June 2011.

The details of the design of the Mark II containment dry well floor directly below the reactor vessel, the in-pedestal region, greatly affects the accident progression, and thus the uncertainty in predicting consequences of a severe accident. The design of this in-pedestal region varies from plant to plant. The designs of the Shoreham and Nine Mile Point 2 containments include downcomers inside the pedestal region. At La Salle, Columbia and Nine Mile Point 2, the in-pedestal region is at a lower elevation than the ex-pedestal drywell floor. Columbia has two sumps cast into the in-pedestal floor. All Mark IIs, with the possible exception of the two Susquehanna units, have drain lines through the dry well floor in this area. Failure of a drywell floor penetration, or the floor itself (by core-concrete attack or from excessive differential pressure across the floor) would allow fission products in the dry well to bypass the wet well, thus resulting in no decontamination before release by a hardened vent from the wet well air space.

Table 2. BWR Mark II Containments by Country

Country	Number	Name
U.S.	8	Columbia LaSalle 1,2 Limerick 1,2 Nine Mile Point 2 Susquehanna 1,2
Japan	7	Fukushima I 6 Fukushima II 1-4 Hamaoka 2 Tokai 2
Mexico	2	Laguna Verde 1,2

In July 1990, the NRC published NUREG/CR-5528, “An Assessment of BWR Mark-II Containment Challenges, Failure Modes, and Potential Improvements.” The conclusions of this containment performance improvement program study, with respect to containment venting, are excerpted in the following paragraphs.

> Severe accident sequences at the Mark II plants can be grouped into two general categories: one where containment integrity is challenged before core

> degradation, the other where core damage precedes any threat to containment integrity. In the first category, which includes loss of long-term containment heat removal with reactor scram (TW) and anticipated transient without scram (ATWS) sequences, the challenge to containment is from overpressurization due to inadequate containment heat removal. In the second category, which includes station blackout (SBO) and other transients where reactor scram occurs, the challenge can be from either overpressurization at or near the time of reactor vessel failure or overpressure or overtemperature failure several hours after vessel failure. Potential improvements addressing the first category of containment challenges include containment pressure control. Examples could include venting from the wetwell through a hardened vent pipe, and containment pressure control and fission product scrubbing, such as the use of containment sprays with a backup water supply. A hardened vent line would allow excess energy in the containment to be rejected to the environment, while avoiding concerns associated with venting through existing "soft" heating, ventilating, and air conditioning (HVAC) ductwork. However, with the high estimated probability of suppression pool bypass in the base case via failure of in-pedestal drain lines shortly after vessel breach, the vent systems would need an external filter, such as the Swedish multiventuri scrubbing system, to prevent a severe offsite release of fission products. Containment sprays could be used to condense steam in the containment, thus delaying overpressurization failure.
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> For the second category of containment challenges (core melt before containment failure), potential improvements include: (a) containment pressure control, such as a hardened vent from the wetwell, (b) improved means of depressurizing the reactor, such as enhancements to the automatic depressurization system (ADS) and the safety/relief valves (SRVs), (c) containment temperature control and fission product scrubbing, such as containment sprays with a backup water supply and external cooling of the drywell head, and (d) mitigation of the fission product releases, such as the use of reactor building fire protection sprays to enhance fission product retention in the secondary containment. The hardened vent line (with or without an external filter) could be used to mitigate late overpressurization challenges.

The rationale for making an external filter optional for late containment failures at the time of the report was that the release would be less threatening than an early release and would likely not result in prompt fatalities if evacuation was not successful; the release to the environment would still be substantial. In summary, an external filter was indicated for the dominant failure modes of the Mark II containment.

The report summarized the benefits of a filtered containment venting system as:

(1) prevents overpressure failures for transients with scram
(2) delays overpressure failures for ATWS
(3) reduces base pressure through preemptive (early) venting before core damage
(4) mitigates hydrogen burns in secondary containment
(5) ensures scrubbing of aerosol releases
(6) is unaffected by suppression pool bypass

Concern about a large release from a severe accident was the key consideration in the decision to recommend a hardened vent for the Mark I containment, and not to recommend a hardened vent for the Mark II containment following completion of the CPIP. For the Mark I, where the wet well might provide some scrubbing of a release, a wet well vent was recommended, despite the potential for a low degree of decontamination in the wet well. For the Mark II, where risk was dominated by bypass of the wet well and thus no wet well decontamination at all, a vent was not recommended without an external filter. However, a filter was judged to not be cost effective based on published cost estimates at the time, e.g., multiventuri scrubber system (MVSS) approximately $5 million plus the cost of the vent.

3. HYDROGEN CONTROL INSIDE CONTAINMENT—MARK I AND MARK II

One of the key considerations associated with the protection of containment integrity is the control of the hydrogen which is produced by the coolant-zirconium reaction during a severe accident. Hydrogen gas can also be produced by radiolysis of the coolant and by core-concrete interaction; however, the main contributor to the production of hydrogen is the aforementioned coolant-zirconium reaction.

In October 1978, the NRC adopted a new rule, 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” specifying the standards for primary containment combustible gas control systems. The rule required the applicant or licensee to show that during the time period following a postulated LOCA, but prior to effective operation of the combustible gas control system, either: (1) an uncontrolled hydrogen-oxygen recombination would not take place in the containment, or (2) the plant could withstand the consequences of an uncontrolled hydrogen-oxygen recombination without loss of safety function. If neither of these conditions could be shown, the rule required that the containment be provided with an inerted atmosphere to provide protection against hydrogen burning and explosion. The rule assumed a release of hydrogen corresponding to 5 percent oxidation of the fuel cladding in determining compliance.

Subsequently, the NRC reassessed the vulnerability of various containment designs to hydrogen burning and adopted amendments to 10 CFR 50.44, including one in 1981 that added a requirement for an inerted atmosphere for BWR Mark I and Mark II containments. Results of research at the time were incorporated into various studies (e.g., NUREG–1150 and probabilistic risk assessments performed as part of the Individual Plant Examination (IPE) program) to quantify the risk posed by severe accidents for light-water reactors. The result of these studies was an improved understanding of combustible gas behavior during severe accidents and confirmation that the hydrogen release postulated from a design-basis LOCA was not risk significant because it was not large enough to lead to early containment failure, and that the risk associated with hydrogen combustion was from beyond-design-basis (e.g., severe) accidents. Combustible gas generated from design-basis accidents was not risk-significant for any containment type, given intrinsic design capabilities or installed mitigative features. The studies also concluded that combustible gas generated from severe accidents was not risk significant for Mark I and II primary containments, provided that the required inerted atmosphere was maintained.

A September 2003 amendment to 10 CFR 50.44 retained the requirement to inert Mark I and II type containments while removing the requirement for hydrogen recombiners or backup hydrogen purge systems. Given the large zirconium inventory in these reactors and their relatively small primary containment volumes, these containments, without inerting, would have a high likelihood of failure from hydrogen combustion due to the potentially large concentration of hydrogen that a severe accident could cause. The regulatory analysis found the cost of maintaining the recombiners exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late time frame, well beyond 24 hours, by the long-term generation of oxygen through radiolysis. The regulatory analysis for this rulemaking found the cost of maintaining the recombiners, and thus likely also the hydrogen purge systems, exceeded the benefit of retaining them to prevent containment failure sequences that progress to the very late timeframe. The rule retained existing requirements for ensuring a mixed atmosphere; inerting Mark I and II containments, and hydrogen control systems capable of accommodating an amount of hydrogen generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region in Mark III and ice condenser containments. The technical bases for the regulations were established from experience at

Three Mile Island along with bounding estimates for the amount of hydrogen likely to be generated by a severe core damage accident.

This rule also specified requirements for combustible gas control in future water-cooled reactors which are similar to the requirements specified for existing plants. However, a key difference is the need to accommodate an equivalent amount of hydrogen as would be generated from a 100 percent (active fuel) clad-coolant reaction. Particularly, if a containment does not have an inerted atmosphere, it must limit hydrogen concentrations in containment during and following an accident that releases hydrogen (equivalent to 100 percent fuel-coolant reaction) when uniformly distributed to less than 10 percent (by volume); and maintain containment structural integrity and appropriate accident mitigating features.

As stated in the rule, all BWRs with Mark I or Mark II type containments must have an inerted atmosphere. This concept reduces oxygen enough to suppress combustion; thereby, a hydrogen generation limit is not specified. The result of a hydrogen combustion event is characterized as a relatively sharp pressure pulse, and thus the intent of rule precludes this occurrence inside containment; but does not recognize the slow buildup of containment pressure as a result of the hydrogen gas generated by postulated severe core damage accidents. Therefore, containment pressure control is addressed in the severe accident management guidelines (SAMG). Essentially, pressure control for severe accidents in Mark I and Mark IIs are also related to hydrogen control for the containment.

4. OTHER DESIGN ISSUES—MARK I AND MARK II

The following issues deal primarily with design-basis issues, which are not directly relevant to the topic of containment venting. They do involve considerations of defense in depth and some early recognition that pressure suppression containments involved additional complexities compared to large dry containments and introduced concerns, such as bypassing the pressure suppression features. Such a bypass would lead to rapid over-pressurization given the smaller volumes of these containment designs (e.g., the relationships shown in Figure 1).

Hydrodynamic Forces

Between 1972 and 1974, the Mark III containment system design was undergoing large-scale testing of the new suppression pool hydrodynamic loads which were identified for the postulated LOCAs. GE was testing the Mark III containment concept at that time because of configurational differences between the previous containment concepts and the Mark III design.

The Mark I containment design is a drywell in the shape of an inverted common incandescent light bulb containing the reactor vessel and primary piping attached with several large vent pipes to a torus-shaped suppression chamber located below the drywell. The steam escaping from the break in the reactor coolant piping would vent, along with the drywell atmosphere, down into the suppression chamber where it would be distributed through a header to many downcomer pipes whose open ends were submerged in the suppression pool, which filled about half the suppression chamber. The Mark II containment concept evolved the drywell and suppression pool to a simpler truncated cone over the cylindrical suppression chamber. The Mark III containment concept involved more fundamental changes in the containment layout with the drywell being completely within the suppression chamber which formed the entire containment boundary.

More sophisticated instrumentation and data analysis was available for the Mark III tests and led to a better understanding of short-term dynamic effects of drywell air being forced into the suppression pool in the initial stage of the postulated LOCA. This air injection into the suppression pool water results in a pool swell event of short duration but with substantial forces associated with the water impacting the suppression chamber walls and internal structures. Additional LOCA-related dynamic load information was obtained from foreign testing programs for similar pressure-suppression containments, including the occurrence of oscillatory condensation loads during the later stages of a postulated LOCA blowdown. Actual experience at operating plants indicated that reactor vessel safety/relief valves (SRVs) discharging via tailpipes to the suppression pool would cause oscillatory hydrodynamic loads on the suppression chamber.

Consequently, in February and April 1975, the NRC transmitted letters to all 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.
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 an "ad hoc" Mark I Owners Group with the objective to determine the magnitude and significance of these dynamic loads and identify courses of action needed to resolve outstanding safety concerns. This task was divided into a short-term program (STP) to be completed in early 1977 and a long-term program (LTP). The STP objective was to verify that each Mark I containment system 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 licensed Mark I BWR facilities

could continue to operate safely, without endangering the health and safety of the public, while working on the comprehensive LTP. The STP acceptance criteria were based on providing adequate margins of safety, i.e., a safety-to-failure factor of 2, to justify continued interim operation of the plants.

The staff's conclusions relative to the STP are described in NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report," issued December 1977. The objective of the LTP was to establish design-basis (conservative) loads appropriate for the anticipated life of each Mark I BWR facility and restore the originally intended design-safety margins. The requirements resulting from the LTP (described in NUREG-0661 "Mark I Containment Long-Term Program," issued July 1980) were used by each BWR/Mark I licensee to perform a plant-specific analyses and identify plant modifications needed to restore margins of safety in the-containment design. Modifications included:

- Torus–Vent System—Considerable additional steel in the way of reinforcement for ring girders, miter joints, vent header and downcomers, internal catwalk and conduit. Torus temperature monitoring instrumentation system. Torus tie-downs and dynamic motion restraints (snubbers).

- Torus attached piping—Considerable additional steel in the way of reinforcement of torus attached piping at the penetration area and supports within the torus.

- SRVs—Added T-quencher spargers at the discharge point within the suppression pool, vacuum breakers for the discharge lines and control scheme circuitry to prevent immediate reopening of an SRV before the vacuum breakers function. Added SRV position monitoring instrumentation.

The Mark II containment suppression chamber dynamic load re-evaluation followed a similar course. NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," was issued in October 1978. NUREG-0487, Supplement 2, issued February 1981, completed the lead plant program after addressing the condensation–oscillation or chugging loads. NUREG-0808, "Mark II Containment Program Load Evaluation and Acceptance Criteria," issued August 1981, provides a discussion of LOCA-related suppression-pool hydrodynamic loads in the Mark II containment design and staff acceptance criteria for pool-swell loads from the lead-plant program and new criteria for steam loads developed in the LTP.

Emergency Core Cooling System Suction Strainers

In May 1996, NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," was issued requesting BWR operators to implement appropriate procedural measures and plant modifications to minimize the potential for clogging of emergency core cooling system (ECCS) suppression pool suction strainers by debris generated during a LOCA. The bulletin cited an event at a Swedish BWR, Barsebäck 2, which involved plugging of two containment spray system suction strainers with mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that spuriously opened.

Subsequent to this event, the NRC issued Information Notice (IN) 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," in September 1992, to alert addressees of the potential for loss of ECCS that was identified as a result of the Barsebäck 2 event. It was expected that recipients would review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems.

Two earlier events involving the clogging of ECCS strainers had occurred at the Perry Nuclear Power Plant, a domestic BWR in 1993. Based on these earlier happenings, the NRC issued Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," in May 1993. In it, the staff requested licensees to identify fibrous air filters or other temporary sources of fibrous material, not designed to withstand a LOCA, which were installed or stored in primary containment. The licensees were to take any immediate compensatory measures to assure the functional capability of the ECCS and promptly remove any such material.

Because of the apparent trend identified in these events, the staff conducted a detailed study of a reference BWR 4 plant with a Mark I containment and issued NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," in October 1995. A suction strainer debris plugging event at Limerick Unit 1 in September 1995, led to further evaluation. Eventually, all BWRs implemented programs to reduce potential strainer blockage debris in containment and improve suppression pool cleanliness and installed large capacity passive strainer designs by the mid-1990's to ensure ECCS pump net positive suction head available for emergency core cooling system during a LOCA.

GSI-191 Implications for BWRs

Because of the information the NRC learned during the assessment of BWR suction strainers and oversight of BWR plant-specific evaluations and modifications, the NRC sponsored a new research effort to study the accumulation of debris on PWR containment sump screens. Based on the most recent research study, "GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance," the NRC concluded that its guidance needed revision for PWRs. In November 2003, the NRC issued Revision 3 of Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant-Accident." Currently, the NRC is implementing its plan to have all PWR licensees perform a plant-specific evaluation for the potential for excessive head loss across the containment sump screen because of the accumulation of debris on the containment sump screen. The NRC also expects licensees to evaluate the effects of debris that might pass through the sump screens.

Based on the information available to date, continued operation of PWRs is justified until plant-specific evaluations are completed. To provide additional assurance regarding the continued operation of PWRs, the NRC asked the licensees of PWRs to implement compensatory measures. This was done through the issuance of Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors," in June 2003. If the results of ongoing NRC inspections and reviews or ongoing and planned studies indicate that unsafe conditions exist at any operating PWR, the NRC will take actions to ensure the continued health and safety of the public. Also, if a licensee discovers that it is not in compliance with the NRC regulations during the implementation of the requested actions in Bulletin 2003-01, it is required to take prompt corrective actions.

In 2007, the NRC did a preliminary area-by-area comparison of regulatory and technical treatment of BWRs vs. PWRs. The NRC's initial conclusion was that there were disparities in treatment, but there was not enough information to validate the issues or their significance. The NRC concluded additional evaluations were needed to determine the safety significance of these issues. The NRC's Office of Nuclear Regulatory Research and the BWROG have initiated additional work on BWR strainer performance.

The NRC and the BWROG have met on a number of occasions to discuss a path forward. The NRC staff has provided perspective to the Owners Group on some of the subject areas related to strainer performance based on lessons learned from evaluations of PWR sump performance. The BWROG continues to apply the lessons learned from GSI-191 and re-evaluate the modifications and analyses for BWRs completed in the 1990s. Final guidance for BWRs is scheduled for May 2016.

Generic Issue-193, "BWR ECCS Suction Concerns"

In May 2002, the staff opened Generic Issue (GI)-193, "BWR ECCS Suction Concerns," which evaluates possible failure of the ECCS pumps (or degraded performance) caused by unanticipated large quantities of entrained gas in the suction piping from suppression pools in BWR Mark I, II, and III containments during LOCA conditions that could cause gas binding, vapor locking, or cavitation. As a result of the initial screening, a task action plan (TAP) for the technical assessment of this issue was approved in May 2004. The staff completed a literature search for information on ECCS pump performance during intake conditions at high voiding in March 2005, and the staff also found experimental evidence that gas may reach the ECCS pumps during a LOCA. Although it appears the pumps can recover given a limited amount of void fraction, the impact of voiding on the operation of the pumps is a concern.

The TAP to resolve this GI involves an evaluation of suppression pool designs, the dynamics of air entrainment in the suppression pool, and the impact on ECCS pump performance. A review of wetwell and suppression pool designs was made to establish bounding parameters. Relevant experiments on pool dynamics were reviewed to identify pre-existing sources of data.

Completed portions of the TAP resulted in a basic understanding of the overall phenomena and a preliminary assessment that continued work on the GI is warranted. The next phase will involve a multi-step estimation of the maximum potential void fraction (MPVF) occurring at different stages of a large and medium LOCA and will attempt to quantify an upper bound for voids present at the ECCS pump suction strainer in the wetwell. The MPVF appears to be influenced by a number of phenomena, many of which overlap in time, such as the gas/liquid jet coming from the downcomer and noncondensable gas injection from the drywell. An estimate of the MPVF (based on a simplified, worst-case scenario for a generic containment) will be made. Ultimately, it is expected that this may provide licensees with insight on how to calculate the MPVF based on their plant-specific geometrical and operational characteristics. Initial emphasis will be placed on the calculations for the Mark I containment.

Based on a staff request, BWROG agreed to provide voluntary input that would provide insights into the characteristics of LOCA phenomena at the earliest stages of the postulated accidents, plus general information about wetwell geometries in relation to ECCS suction strainers. This proprietary input was received on October 29, 2009.

An experimental testing program was proposed in 2009 to help assess the complex phenomenology involved with bubble creation, injection, and transport into the containment wetwell. Modifications to the experimental facility at Purdue University began in fall 2009 in order to simulate the creation and behavior of voids following their injection into a BWR Mark I suppression pool. The testing program, underway during 2010, was completed at Purdue University to promote understanding of complex void-transport phenomena. The final report was received in March 2011. The results of the experimental program have shed light on the behavior of voids in the BWR Mark I wetwell design in regard to the potential transport of bubbles resulting from the LOCA blowdown. This information will be valuable in assessing the capability of bubbles to be transported to the suction strainer of ECCS pumps. The issue remains open.

Table 3. BWR Mark I Containments in the United States

BWR Mark I's Data
Sorted Alphabetically by Plant Name (Column A)

A	B	C	D	E						
Unit	License Renewal		Power Uprate		Power	Power	Containment Free Volume	Containment Free Volume	SP Water Volume	SP Water Volume
	Yes/No	Date of Expiration	Yes/No	%	MWth	MWe	Cubic Feet	Cubic Meters	Cubic Feet	Cubic Meters
Browns Ferry 1	Yes	12/20/2033	Yes	5	3458	1065	278,000	7872	85,000	2,407
Browns Ferry 2	Yes	06/28/2034	Yes	5	3458	1104	278,000	7872	85,000	2,407
Browns Ferry 3	Yes	07/02/2036	Yes	5	3458	1115	278,000	7872	85,000	2,407
Brunswick 1	Yes	09/08/2036	Yes	20	2923	938	288,100	8158	87,600	2,481
Brunswick 2	Yes	12/27/2034	Yes	20	2923	937	288,100	8158	87,600	2,481
Cooper	Yes	01/18/2034	No		2419	830	255,240	7228	87,660	2,482
Dresden 2	Yes	12/22/2029	Yes	17	2957	867	275,481	7801	112,203	3,177
Dresden 3	Yes	01/12/2031	Yes	17	2957	867	275,236	7801	112,203	3,177
Duane Arnold	Yes	02/21/2034	Yes	15.3	1912	640	225,560	6387	61,500	1,741
Fermi 2	No	01/23/2028	Yes	5	3430	1122	291,490	8254	121,080	3,429
Fitzpatrick	Yes	10/17/2034	No		2536	852	264,000	7476	105,000	2,973
Hatch 2	Yes	06/13/2038	Yes	15	2804	883	257,190	7283	87,660	2,482
Hatch 1	Yes	08/06/2034	Yes	15	2804	876	257,190	7283	87,660	2,482
Hope Creek 1	Yes	04/11/2046	Yes	16.6	3840	1061	302,500	8566	122,000	3,455
Monticello	Yes	08/08/2030	Yes	7	1775	572	242,450	6865	77,970	2,208
Nine Mile Point 1	Yes	08/22/2029	No		1850	621	300,000	8495	89,000	2,520
Oyster Creek	Yes	04/09/2029	No		1930	619	308,000	8722	77,970	2,208
Peach Bottom 2	Yes	08/08/2033	Yes	6.62	3514	1112	303,500	8594	123,000	3,483
Peach Bottom 3	Yes	07/02/2034	Yes	6.62	3514	1112	303,500	8594	123,000	3,483
Pilgrim	Yes	06/08/2032	Yes	1.5	2028	685	257,000	7277	84,000	2,379
Quad Cities 1	Yes	12/14/2032	Yes	17	2957	882	275,236	7794	115,600	3,273
Quad Cities 2	Yes	12/14/2032	Yes	17	2957	882	275,236	7794	115,600	3,273
Vermont Yankee	Yes	03/21/2032	Yes	20	1912	620	242,450	6865	77,970	2,208

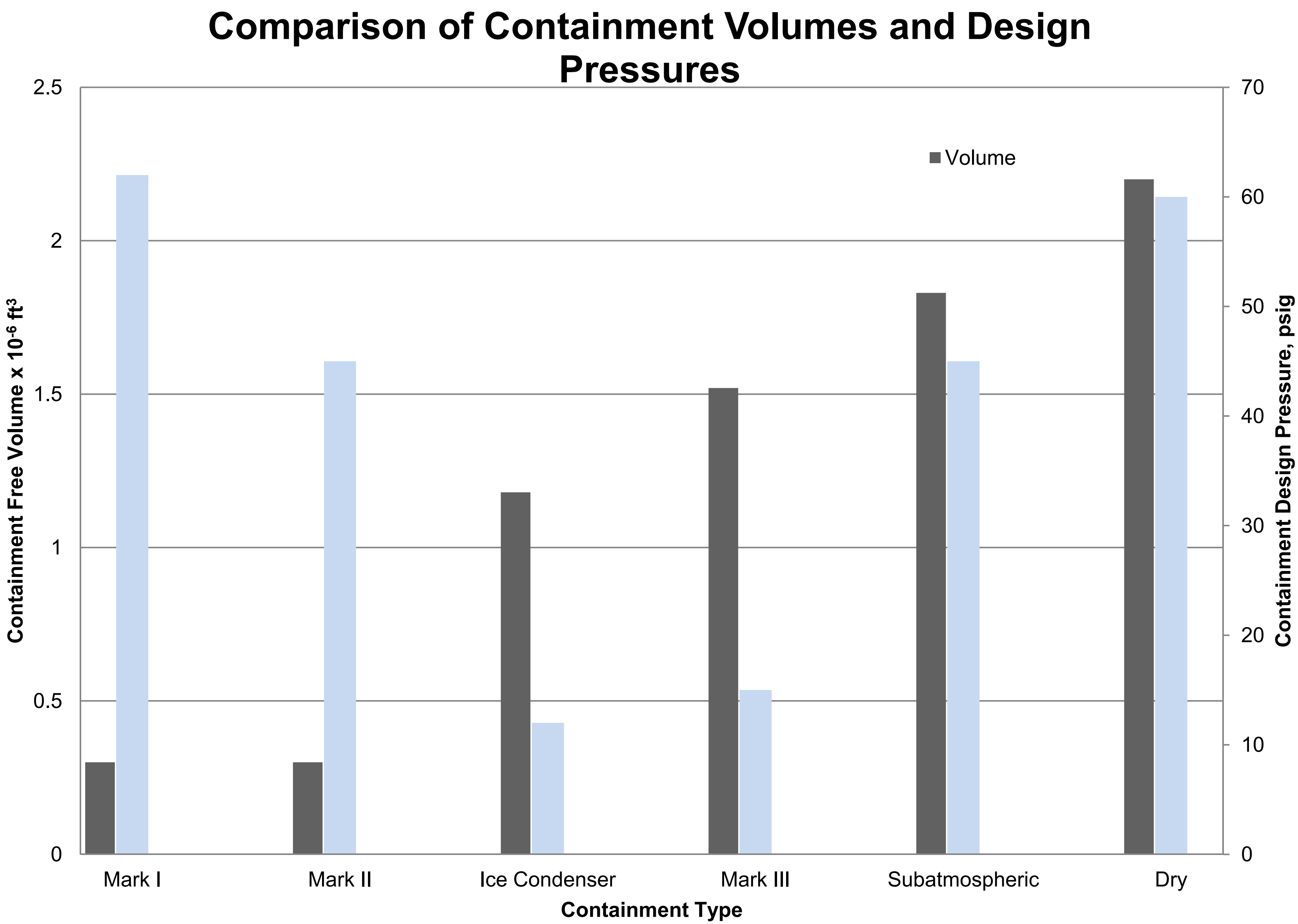

Figure 1. Comparison of containment volumes and design pressures

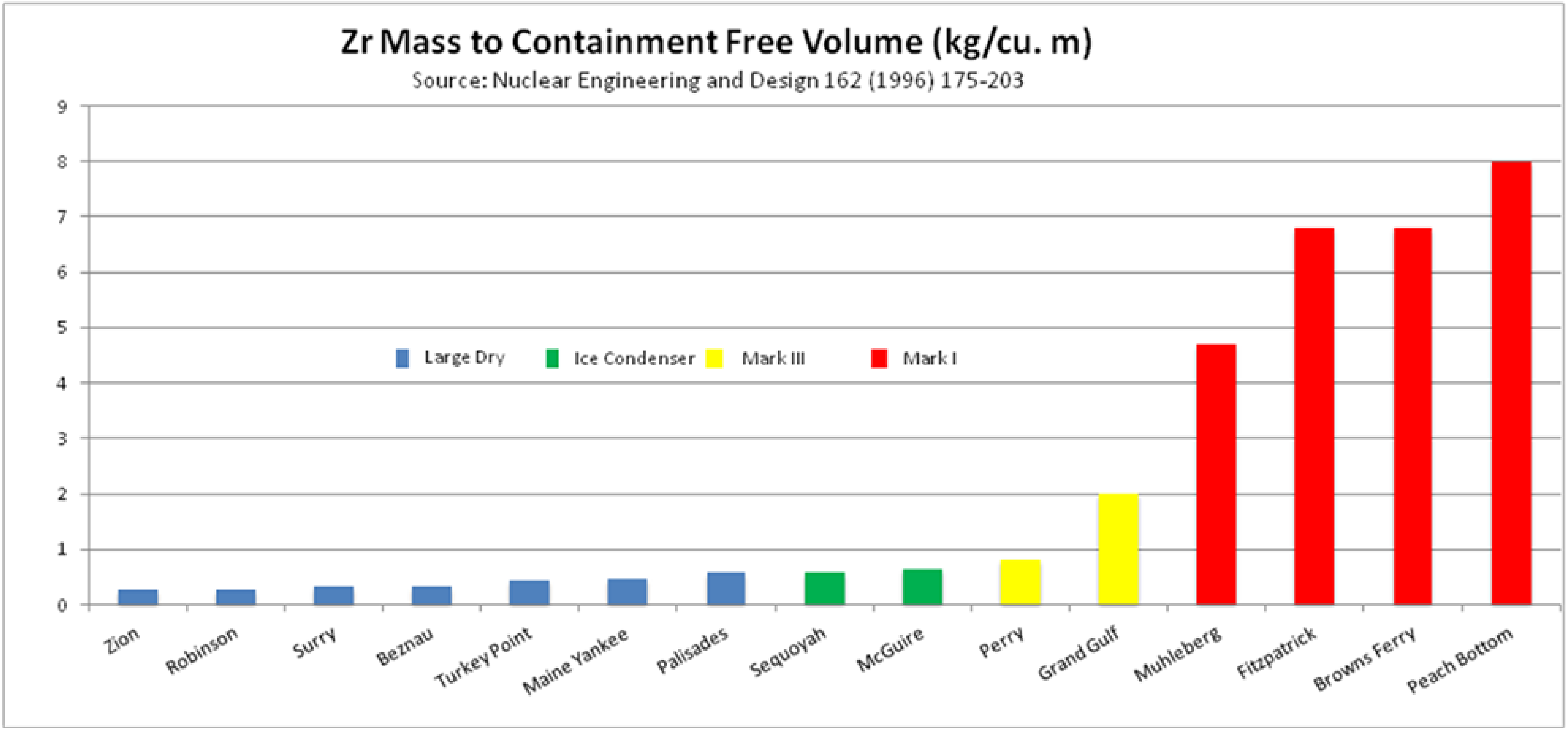

Figure 2. Zirconium mass to containment free volume

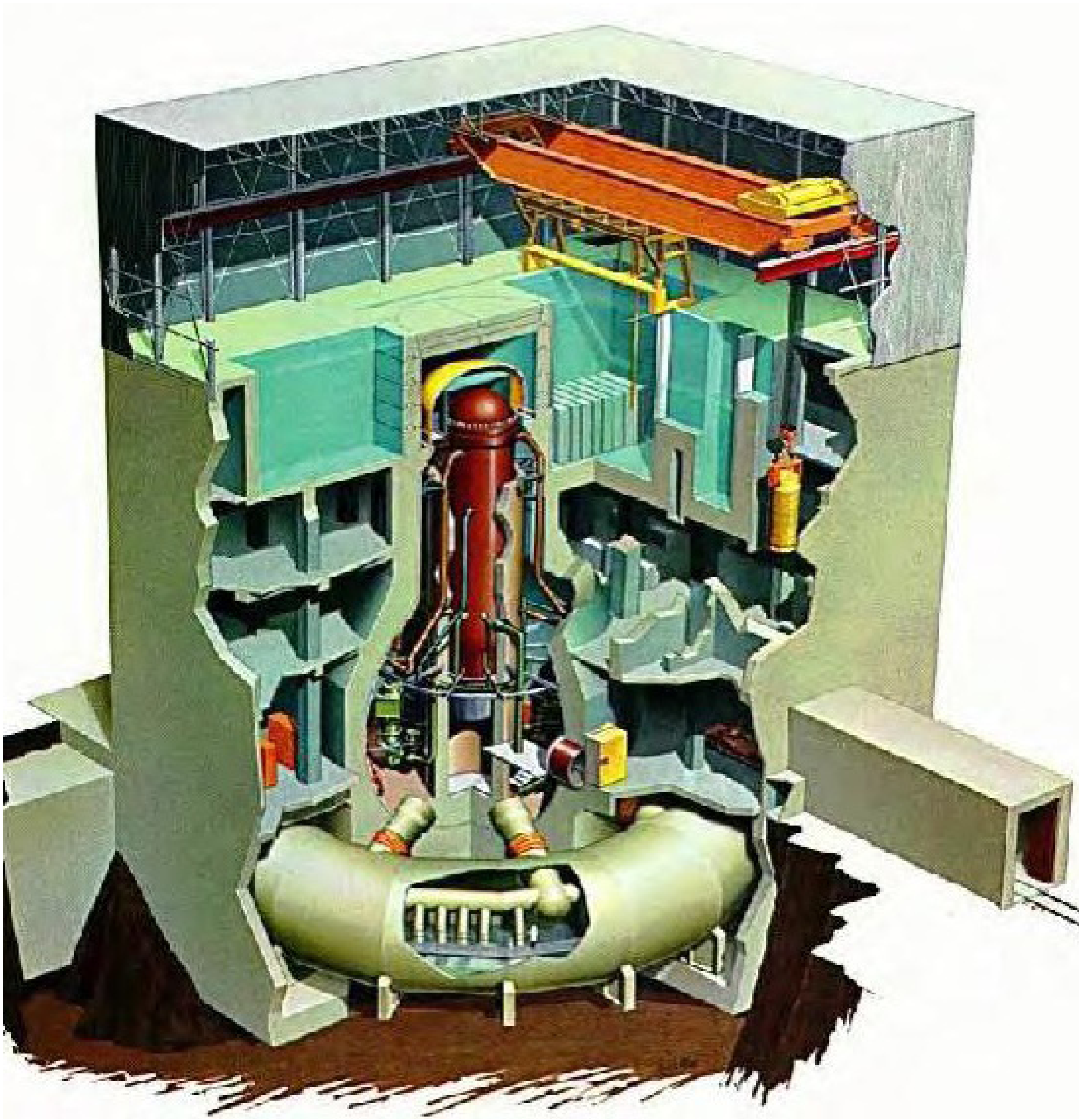

Figure 3. BWR Mark I containment cross section

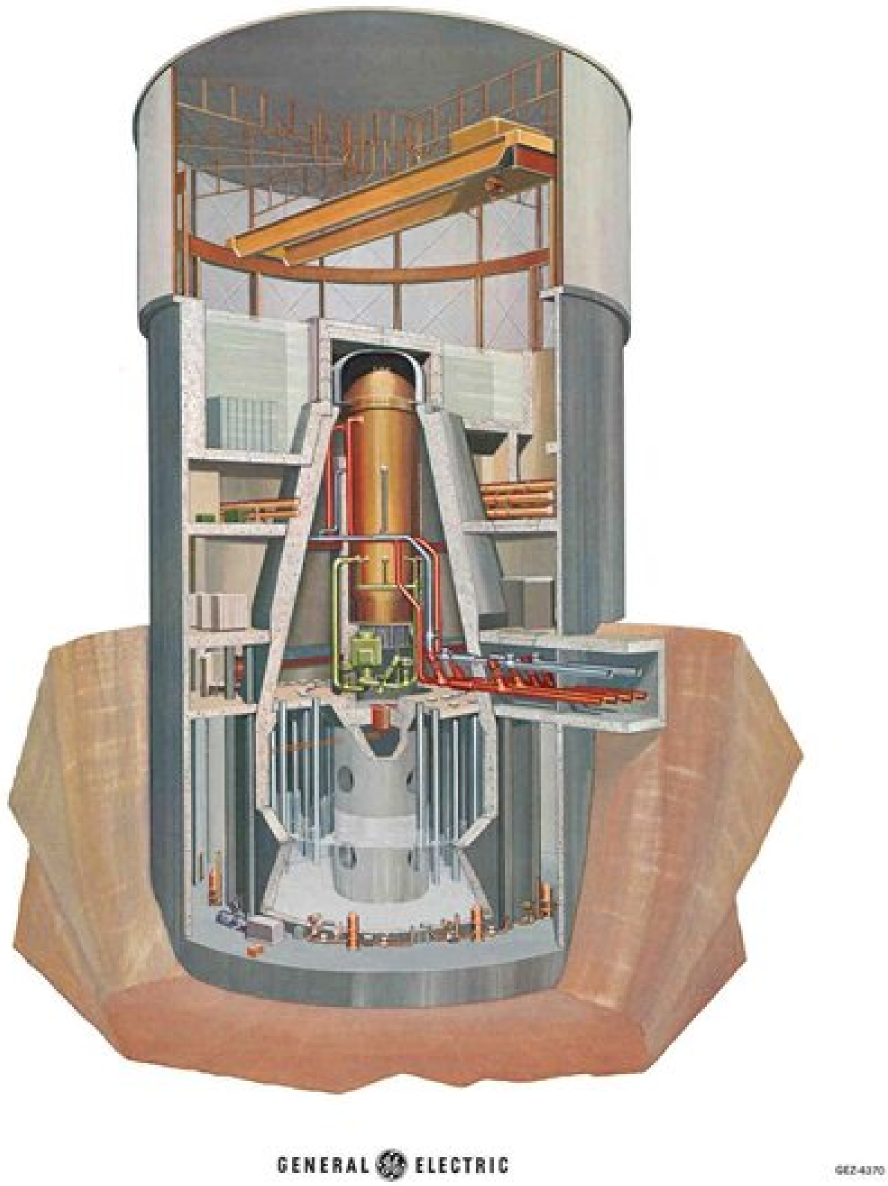

Figure 4. BWR Mark II containment cross section