SEVERE ACCIDENT SOURCE TERM REASSESSMENT

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ABSTRACT

This paper summarizes the status of the reassessment of severe reactor accident source terms, which are defined as the quantity, type, and timing of fission product releases from such accidents. Concentration is on the major results and conclusions of analyses with modern methods for both pressurized water reactors (PWRs) and boiling water reactors (BWRs), and the special case of containment bypass. Some distinctions are drawn between analyses for PWRs and BWRs. In general, the more the matter is examined, the consequences, or probability of serious consequences, seem to be less.

INTRODUCTION AND SUMMARY

Since the accident at Three Mile Island's Unit 2 (TMI-2) in March 1979, a major focus of research activity in the nuclear industry has been on the source term produced by severe accidents. The source term is defined as the quantity, type, and timing of fission product materials released to the environment during such accidents. In this context, severe accidents are defined as those events or sequences of events during which the reactor core suffers some degree of degradation, resulting in partial or complete release of fission products from the core. In most analytical investigations of severe accidents, the core state is analyzed through the point of fuel melting and subsequent breach of the reactor pressure vessel (RPV) by the molten core material.

The research devoted to this topic has been international in scope and covers a very broad range of analytical and experimental activities. The tools and techniques developed as a result of this work have revolutionized the manner in which severe accidents are analyzed. Prior to TMI-2, safety analyses of severe accidents in light water reactors (LWRs) were performed using highly conservative methods and assumptions for calculating the thermal/hydraulic behavior and fission product transport. The result was predictions of very high environmental release fractions of the fission product constituents which make up the source term. Many of these conservative predictions became embodied in current regulations governing emergency planning, equipment qualification, radiation protection, and other applications. Although the program of new research is not yet completed, much useful information has been produced. The work to date has yielded the following:

- Source terms for PWR in-containment accidents are generally agreed to be much lower than previously thought.
- Fission product behavior, for the most part, can be modeled as aerosol behavior.
- Containments are stronger than previously thought, and fail, if at all, at later times.
- Containment bypass sequences for both PWRs and BWRs are highly plant-specific.
- Fission product retention on, and subsequent revaporization and transport from, the various surfaces within the reactor coolant system (RCS) are important, especially for BWRs.
- There is substantial retention in containments for both PWRs and BWRs, and also in PWR auxiliary buildings and in BWR reactor buildings.

Although the accident at Chernobyl occurred in a fundamentally different type of reactor, opportunities to learn from it are being taken where possible. There is at present no indication that Chernobyl will reveal new physical or chemical phenomena relevant to LWR source terms or will show that the relevant phenomena have been misunderstood.

ANALYTICAL METHODS

The state of knowledge regarding severe accident progression, physical and chemical phenomena, and containment integrity has advanced significantly since the initial U.S. effort to analyze severe accidents in the NRC's Reactor Safety Study (WASH-1400).[1] The advancements have been most pronounced since TMI-2. The report of the Special Committee on Source Terms of the American Nuclear Society stated:

"The Committee concluded that the state of current knowledge and analytical methods and assumptions are sufficiently advanced to warrant the reduction of calculated source terms from estimates in WASH-1400 by more than an order of magnitude to several orders of magnitude. The noble gases are exceptions...."[2]

This conclusion was possible because of the following significant advancements in analytical capability which have evolved over approximately the past decade:

- The dominant chemical form of iodine will be cesium iodine (CsI) and the dominant form of cesium will be cesium hydroxide (CsOH). These chemical forms are predominant because the iodine and cesium are released from the core into a chemical-reducing environment (steam and hydrogen) in the RCS and containment, and because the core inventory of cesium exceeds that of iodine by a factor of approximately 10 in LWRs.
- The ability to accurately assess the rate of aerosol deposition caused by physical phenomena has advanced significantly. Improved methodologies now exist to calculate the deposition in the RCS, containment, and adjacent structures due to agglomeration and gravitational settling of airborne aerosols, diffusiophoresis, thermophoresis, inertial impaction, pool scrubbing (the scrubbing of fission products by emission into submerged water pools), spray washout, and other processes.
- The ability to accurately predict the thermal/hydraulic profiles in each compartment of a plant during a severe accident has been greatly advanced. The models used in these calculations are well-founded and, for the most part, have been validated by experimental evidence. In particular, the capability to predict natural convection flow patterns of multi-component (steam, hydrogen, carbon dioxide, etc) flows in multi-compartment arrangements has advanced considerably. The effects of other severe accident thermal phenomena, such as the burning of combustible gases, also have been modeled.
- The robustness of containment structures is much greater than previously calculated. Ultimate capacities of 2.5 to 3 times design pressure capacities are typical for LWR containments of modern design. Because of this additional strength, time to containment breach is longer, resulting in longer aerosol residence times and increased depletion. Also, early containment challenges previously thought to be credible, such as those from steam explosions in the RPV (creating a missile from the RPV head), containment steam explosions, and hydrogen explosions, are no longer generally considered valid mechanisms for breaching of containment.

Major failure of containment is highly improbable because of the very large safety factors related to attainable peak pressures. What is somewhat credible is a degradation of containment function because of leakage at a penetration or other point of discontinuity. This would, of course, tend to propagate only to the point where pressure is limited or begins to decrease with escape of containment atmosphere.

The foregoing advancements have led to the following major observations:

- PWRs with large, dry containment systems will exhibit reduced source terms for accident sequences in which the fission product release from the RCS occurs inside the containment.
- Source terms for PWRs for in-containment sequences in which containment breach is late are much lower than for PWR in-containment sequences in which a pre-existing opening is assumed to exist. Source terms for PWR sequences with pre-existing openings are lower than previously analyzed PWR early containment breach sequences by approximately a factor of 10.
- Basemat penetration by molten core debris has a much lower likelihood than previously believed and results in very small releases, should it occur.

CONTAINMENT BYPASS

In the Reactor Safety Study, the dominant contributor to risk for the Surry PWR plant analyzed was an interfacing system loss of coolant accident (LOCA), also referred to as a "V Sequence" or "containment bypass." The scenario consisted of concurrent valve failures in a system connected directly to the RCS (an interfacing system). These valve failures allowed full reactor pressure to be applied to that portion of the interfacing system which was not designed to withstand reactor pressure. If failure of the piping occurred outside the containment, the discharge of coolant and possibly fission products would bypass the containment (but would, however, go into another building).

In the case of the U.S. Surry PWR plant, the interfacing system with the highest likelihood of producing such a sequence is the low pressure safety injection system. At Surry, the piping code class change occurred in a length of pipe located outside containment in the adjacent auxiliary building. In the Reactor Safety Study Analysis of Surry, a LOCA was assumed to occur at that point in the system, resulting in continued loss of RCS inventory outside containment. Since the LOCA effluent cannot be recirculated, as in the case of an in-containment LOCA sequence, when RCS makeup inventory is depleted, the remaining RPV inventory boils down and core uncovery and fission product release follow. The fission products take the same pathway as the RCS water inventory, directly into the auxiliary building via the interfacing system, thereby bypassing the containment volume completely. Reactor Safety Study source terms for this accident were very high, and all plants were assumed to be vulnerable to this accident, with differing degrees of likelihood, but with approximately the same source term release quantities. Because of this, and the generally low source terms calculated for sequences in which the containment participates, containment bypass sequences have held the close attention of all severe accident analysts and the NRC over the past decade.

Because of their important potential contribution to overall risk, S&W has undertaken several analytical efforts[3,4] in an attempt to provide better insights to various aspects of containment bypass sequences. These efforts have led to the general conclusion that containment bypass sequences are plant-specific. Engineering and construction details play an important role in determining the nature of such accidents in LWRs. Specifically, the following findings were made:

- Identifying the Flowpath

  Not only are valve configurations important, but the size, pipe schedule, piping routing, code class change locations, and pipe supports also must be examined.

- Potential Flooding of the Break

  Once the break location has been determined, the compartment into which the break effluent will emit must be analyzed to determine if the pipe in question might be covered by a water pool. During a reexamination by S&W of the Surry V sequence, it was discovered that the low-pressure safety injection (LPSI) line would break at a location well below grade. The low-pressure side of the piping was connected directly without any isolation valves to the refueling water storage tank (RWST), which is a 1.44 million-liter (380,000-gal) tank located at grade. Additionally, the pipe break was located in the LPSI pump cubicle, which is a flood-protected compartment. Therefore, the RWST inventory would empty into the LPSI pump cubicle via gravity flow, resulting in a 0.9-m (3-ft) standing water pool over the ruptured LPSI line. Passage through this pool would have a scrubbing effect on the break effluent, resulting in a significant reduction in the release of fission products. It should be noted that the arrangement of LPSI components and the RWST is a typical Emergency Core Cooling System (ECCS) layout for LWRs. Each plant must be examined individually to determine if the break is floodable.

- Auxiliary Building Layout

  A detailed evaluation of the auxiliary building fission product transport characteristics also will be necessary. In this case, it is important to identify which building openings to the environment might exist during a building overpressure condition and their location with respect to each other and the break location. When large openings exist at different elevations in a thermal/hydraulic environment dominated by buoyancy driven flow, the potential for free convective flow that enhances net building outflow is established.

- Auxiliary Building Sprays

  Some plants have passively actuated, dedicated fire protection spray systems. The capacities, power supplies, and available water volumes of these systems vary. Additionally, the coverage by the spray system in the auxiliary building varies. The time of available spray must be established to determine if that time period coincides with any portion of the fission product release period. If the two periods overlap, then credit for washout of fission products by the spray may be taken.

It is apparent from the foregoing description of key V sequence analysis topics that a detailed engineering evaluation of the possible interfacing LOCA systems, and of the receptor building are necessary to calculate realistic source terms for these types of accidents. Such details as floor drainage and curbing heights can be important design features in performing a realistic assessment. Since these details will vary widely from plant to plant, sequence analyses are truly plant-specific. Continuing work on the analysis of containment bypass sequences is considering alternate types of bypasses, such as PWR steam generator tube rupture and BWR control rod drive system valve failures.

PWR/BWR ANALYTICAL DIFFERENCES

PWRs and BWRs share many similarities as LWRs. However, in performing source term analysis, the design differences between the two types of plants can play important roles in thermal/hydraulic and fission product transport calculations. Figures 1 and 2 show simplified depictions of containments for two types of large, dry PWRs, an ice condenser PWR, a Mark I BWR, a Mark II BWR, and a Mark III BWR. The major design differences that affect severe accident analysis are shown in Table 1. This list is not meant to be exhaustive in nature and only addresses those differences which SWEC believes are important for source term considerations. In studies by SWEC of the Shoreham (BWR Mark II)[5] and Peach Bottom (BWR Mark I)[6] plants, the following aspects of the analyses required careful attention to ensure an accurate evaluation:

- The fission product retention characteristics of the RCS were vital analysis results, particularly the revaporization of fission products and the timing of their release from the RCS into the containment or suppression pool.

- Mass and energy inputs into a BWR drywell can follow two flowpaths if the containment breach is postulated to occur in the drywell, as in the case of a Mark I BWR. One pathway is through the opening in containment into the reactor building. The other path is into the suppression pool via the downcomer vents. Any fission products airborne at the time that the containment is pressurized will also follow these flowpaths. Therefore, careful consideration must be given to accurately defining the fraction of mass which flows into the pool where the fission products will be subject to the pool decontamination factor (DF).

- BWR source term analyses require calculation of suppression pool DF. This factor generally varies with aerosol input size distribution and suppression pool temperature. Experimental data are usually referenced for this information.[7]

- Mark II BWRs conduct large fractions of the corium mass into the pool relatively quickly via the pedestal and/or drywell downcomer vents. Therefore, the core-concrete interaction is greatly reduced. This not only limits the gas and energy generation and its effect on containment temperature, but more importantly, less fission product aerosols are generated. Since this is the period when the majority of the nonvolatile fission products (principally strontium, barium, ruthenium, and lanthanum) are released, a limited core-concrete interaction will greatly

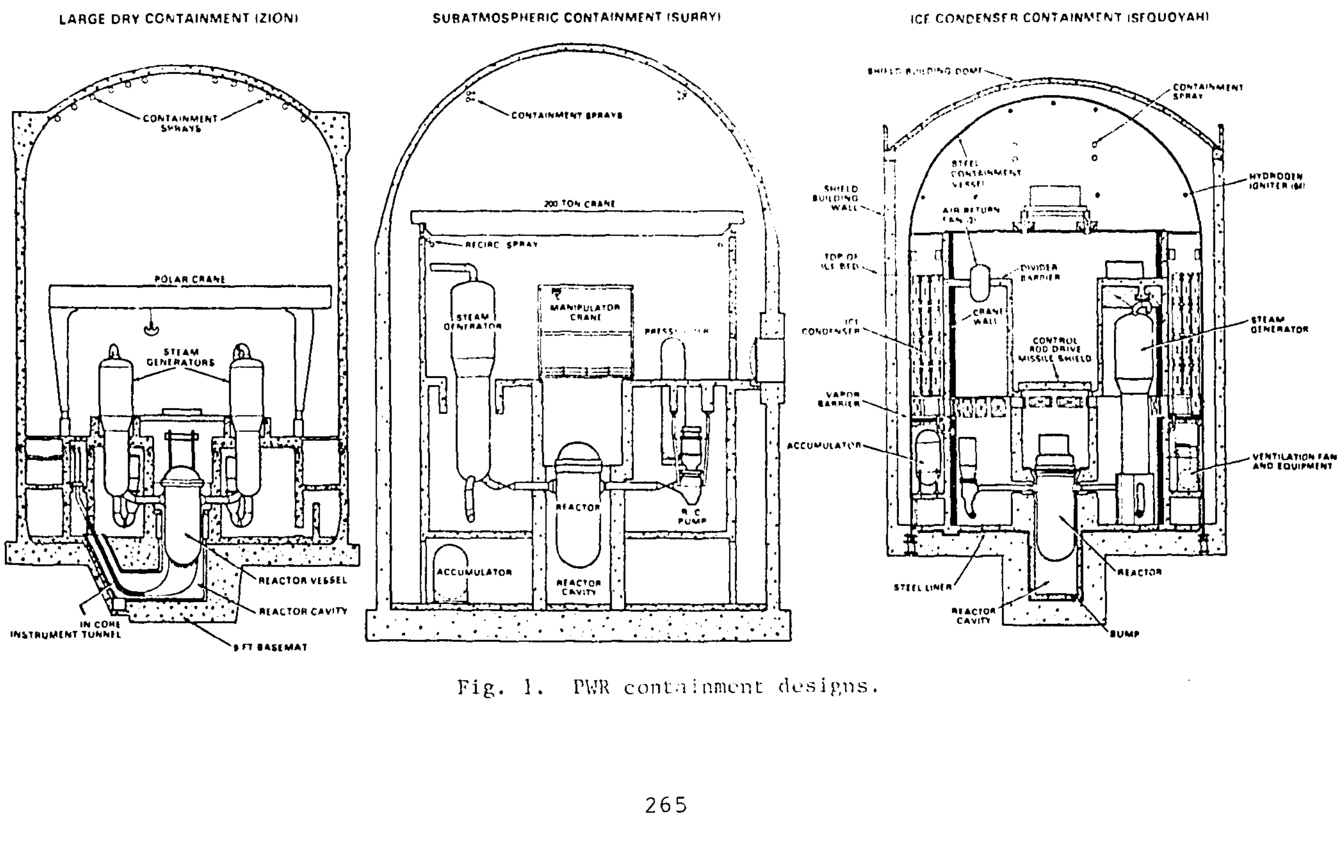

Fig. 1. PWR containment designs.

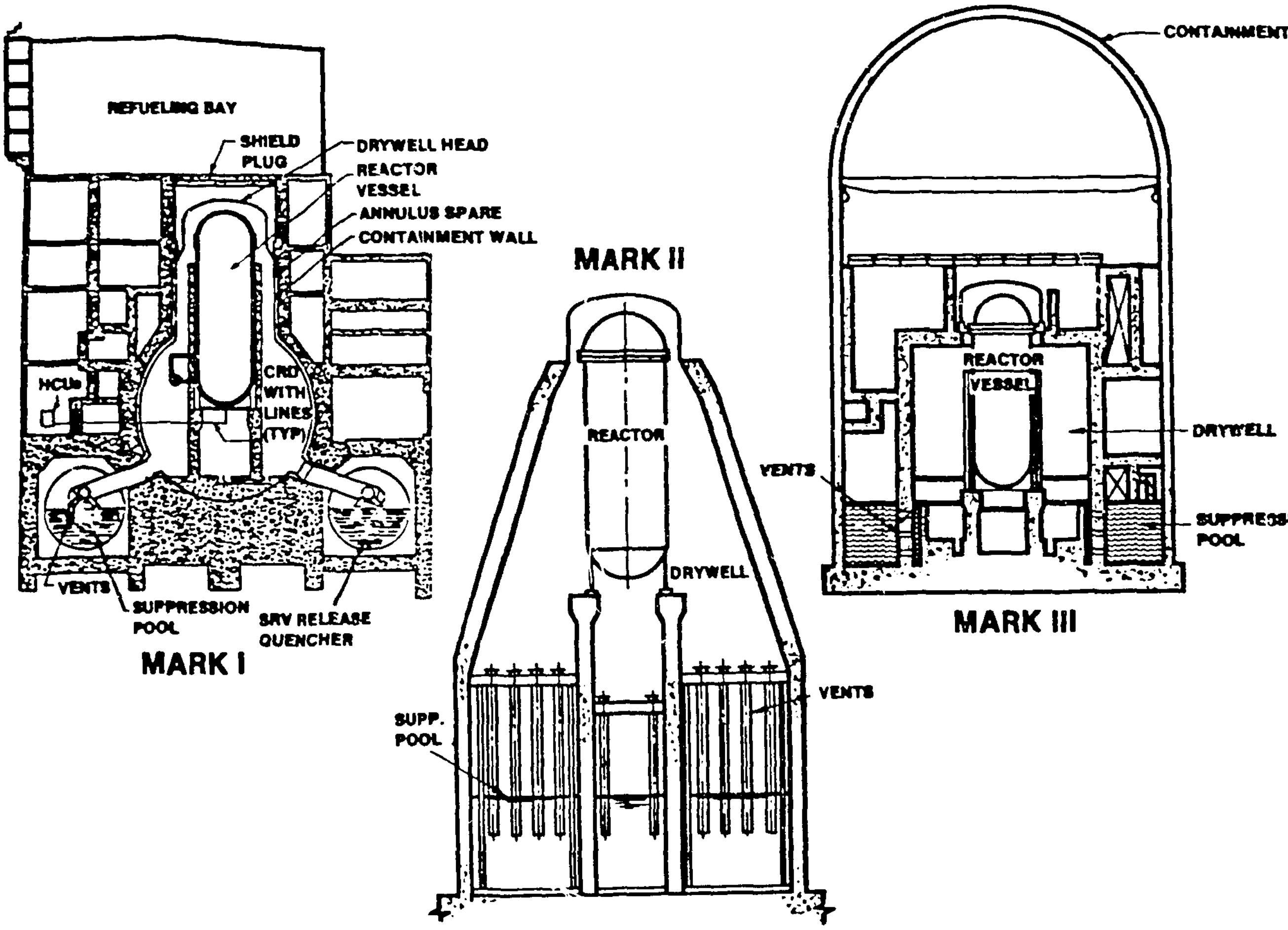

Fig. 2. BWR containment designs.

reduce the amount of these materials available for release to the environment. Also, when the vents are exposed to molten core debris, they are postulated to fail rapidly. This connects the drywell and wetwell airspaces directly, thereby bypassing the suppression pool.

- Designation of the location of the containment breach is also important. Drywell breaches will bypass the suppression pool after the breach has occurred. Wetwell airspace failures are less severe because all material must pass through the pool (via the safety/relief valves or downcomer vents) prior to emission into the reactor building. Containment analyses have shown that breaching of Mark I containments would most likely occur in the drywell,[8] while Mark II BWR containments are most vulnerable in the wetwell airspace region.[9]

IMPACT OF THE ACCIDENT AT CHERNOBYL

The accident at Chernobyl occurred in a graphite moderated pressure tube reactor without containment as it is applied to LWRs. Some phenomena important to the accident, such as the graphite fire which continued for 10 days, have no counterparts in LWRs. The International Nuclear Safety Advisory Group (INSAG) of the International Atomic Energy Agency (IAEA) included in its Summary Report on the Post Accident Review Meeting the following provisional conclusion relative to nuclear safety:

> "1. NO NEW PHYSICAL PHENOMENA HAVE BEEN IDENTIFIED
>
> No physical phenomena can be identified which have not been previously identified in safety analyses and the subject of some theoretical and/or experimental research."[10]

We intend to learn as much as we can from any additional data and details which may

Table 1. Design differences between PWRs and BWRs important to source term analysis.

REGION, Feature	PWR	BWR	Comment
CORE/RPV			
Fuel arrangement	Open lattice	Fuel assemblies enclosed by zircalloy channels	• More material to absorb decay heat in BWRs • Molten core contains more material in BWRs
Power	3,200 MW_t for larger PWRs	3,200 MW_t for larger BWRs	Energy amounts and fission product inventories approximately the same for each type
Amount of zircalloy	20,210 kg	63,017 kg	More Zr available to oxidize in BWRs; more hydrogen generated
RPV volume	188 cu m	598 cu m	
RPV internal surface area and mass	1,000 sq m 850,000 kg	4,596 sq m 1,081,000 kg	• Fission product retention significant for BWRs • BWR steam conditioning equipment internal to RPV. PWR equipment located in steam generator. Thin-walled S/G U-tubes provide vulnerable bypass pathway.
Poison	Reactor coolant continually borated	Reactor coolant is demineralized water with no additives	Boron effect on fission product chemistry under investigation
Control rod drive	Electrically driven Located in top head	Hydraulically driven Located in bottom head	• BWR CRD system provides additional bypass pathway • BWR CRD system provides additional high-pressure injection capability • BWR RPV breach is via CRD penetrations, PWR RPV breach is via bottom head meltthrough for CE PWR penetrations for Westinghouse and B&W PWRs • BWR CRD hydraulic piping will add to core debris inventory.
CONTAINMENT			
Structure/design	Two designs: Large/dry Ice condenser	Three designs: - MKI (free-standing steel) - MKII (reinforced concrete) - MKIII (steel or concrete)	• MKI and MKII containments have higher ultimate capacity ratings than PWR or MKIII containments. • Steel containments are pressure vessels IAW ASME III code. Reinforced concrete containments have more complex structures.
Pressure suppression	Volume, heat sinks (except for ice condenser)	Large (1 million-gal.) water pool	• BWR suppression pool also acts as an efficient scrubbing medium for fission products. Suppression pool connected directly to RPV via SRVs. • Ice condenser also an efficient fission product scrubber, although DFs not as high as BWR suppression pool.

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Table 1 (cont.)

REGION, Feature	PWR	BWR	Comment
Volume	Large/dry: 56,640 cu m (2×10^6 cu ft)	MKI: 7,080 cu m (250,000 cu ft) MKII: 12,744 cu m (450,000 cu ft) MKIII: 28,320 cu m (1×10^6 cu ft)	• MKI/II containments are very small; thermal/hydraulic effects amplified because reactor has same power level as PWRs. • MKIII combines the large volume of a large/dry PWR with a BWR suppression pool.
Heat sinks	Large/dry: 21,800 sq m	MKI: 8,300 sq m MKII: 8,300 sq m MKIII: 24,505 sq m	• Fission product retention due to diffusiophoresis lower for MKI/II than for PWR or MKIII.
Corium flow	Remains in cavity with some dispersion. Corium spread will be minimal.	MKI: Some overflow from pedestal will spread onto drywell floor; some corium may reach drywell wall. MKII: Pool directly beneath pedestal; corium will enter pool via pedestal or drywell downcomer vents. Corium spread will be minimal MKIII: Corium will accumulate in pedestal or drywell floor. Little or no corium will enter pool because of weir wall. Corium spread will be minimal.	• Core-concrete interaction mass, energy, and fission product will be less for those plants with less corium spread, because less concrete is contacted. • MKII suppression pools will receive the bulk of the core debris. The suppression pool will boil continuously, which will cause a steady volumetric turnover of the wetwell and/or drywell air spaces. • Prolonged core-concrete interactions cause very high temperatures in MKI containments.
Combustible gas	No special features	MKI/II: Nitrogen inerted containments. MKIII: Hydrogen ignition system used.	• PWR and BWR containments generally have the volume and strength to withstand all credible burning events, even without special systems. Most accidents also create steam inerted atmospheres. • If hydrogen burns do occur, the effects on containment outflow would be amplified for MKI/II BWRs because of the smaller volumes.
REACTOR/AUXILIARY BUILDING			
Configuration with respect to containment	Normally adjacent to containment	MKI/II: Normally encloses containment. MKIII: Normally adjacent to containment.	• Depending on location of containment breach, fission products may have to pass through reactor/auxiliary building before release to environment. • MKI/II reactor building will always be in fission product flowpath.

Table 1 (cont.)

REGION, Feature	PWR	BWR	Comment
Structure	Normally all concrete with low-strength openings at grade and at higher elevations (ventilation). Some vibration in design of openings and roofs.	MKI/II: Normally, concrete up to the refueling floor, with metal siding above floor. MKIII: Normally all concrete with low-strength openings at grade.	• MKI/II reactor buildings have lower pressure capacity because of small volume and low-strength siding • Buildings with reinforced openings generally have higher pressure capacity.
Arrangement	Various	MKI/II: Normally 5 to 6 elevations connected by large equipment hatch. Large volume above refueling floor. MKIII: Various.	• MKI/II elevation volumes relatively small; connected serially with little recirculation. • MKI/II refueling has very large volume, but very low strength (approx. 0.25 psig) due to siding. • Adjacent buildings have multiple openings to the environment, more recirculatory flow.
Volume	Large/dry: 56,000 cu m	MKI/II: 33,500 cu m MKIII: 32,000 cu m	• BWR reactor buildings have smaller volumes generally. Thermal/hydraulic effects are amplified. Fission product retention is generally lower due to lower residence times, higher temperatures, and less heat sinks.

emerge. We have not, however, seen anything yet which leads us to doubt the validity of the immense amount of careful experimentation and analysis which has been put into the source term reassessment for LWR plants.

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