THE CANARY, THE OSTRICH, AND THE BLACK SWAN: AN HISTORICAL PERSPECTIVE ON OUR UNDERSTANDING OF BWR SEVERE ACCIDENTS AND THEIR MITIGATION

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Between 1980 and 1995, Oak Ridge National Laboratory (ORNL) was engaged in an intense effort to understand commercial boiling water reactor (BWR) severe accident phenomenology, severe accident progression, and the potential role of the reactor operator in severe accident mitigation. This paper presents a summary of the major findings and conclusions from that period. Both detailed accident- and plant-specific results are discussed. The author, who was a member of the ORNL research team who performed the work, offers an historical perspective on lessons learned, lessons ignored, and lessons forgotten from that period. The relevancy of these findings in the post-Fukushima world is addressed. Finally, the author discusses the evolution of the current risk-informed regulatory framework; and identifies some key questions to be addressed, and critical steps to be taken to inform the development of the new nuclear safety construct required in the wake of the Fukushima Daiichi accident.

I. INTRODUCTION

The multi-unit accident that occurred at the Fukushima Daiichi nuclear power plant[1] in March 2011 is, like the 1979 accident at Three Mile Island[2], a major event in the history of commercial nuclear power. Both events have been described as "black swans" – events that share three characteristics[3]: (1) they are a surprise to virtually everyone at their time of occurrence; (2) they have a major impact; and (3) after their occurrence, the events are viewed as happenings that could have been expected because the relevant data was available in advance but not appropriately considered. Whether or not this characterization is valid, the accident at Fukushima and the events surrounding it call into question some traditional assumptions implicit in many probabilistic safety assessments and emergency response plans associated with external initiating events. Examples include:

1. Two simultaneous or nearly simultaneous beyond-design-basis external events (such as earthquakes and tsunamis) will not occur.

2. Accident events (such as hydrogen detonations) in one unit of a multi-unit site do not propagate to or significantly impact another unit at that site.

3. The availability of shared external emergency response equipment needed by a single unit is not compromised by events at other units on the site (or conversely, the ability of plant personnel to respond as needed to events at multiple units is not compromised by events at any individual unit).

4. Whatever happens, the world outside the plant boundary will not be so impacted by the initiating event as to render it incapable of delivering assistance to the plant within the time frame required to avert major external consequences.

5. The consequences to the public and to society, and the resultant "societal risk" of a major core damage accident in a commercial nuclear power plant are adequately captured by the traditional public health and safety risk metrics of prompt and latent cancer fatalities.

Clearly, this is a time for reflection and introspection by the nuclear power community – much like that which occurred in the wake of the TMI-2 accident:

- Will the community respond with only those actions that are obvious and easy, or will it search deeply and broadly for the lessons to be learned from Fukushima? *Put plainly, will the industry go beyond the expedient to do what is in its own self-interest and the long-term interest of society?*

- Will the industry fall victim to a common tendency during "crisis learning" events of focusing only on the "ills that brought on the crisis", rather than the "thinking that brought on the presenting ills"[4]?

- Will the industry be preoccupied with and paralyzed by the fear of the unintended consequences of its response, or will regulators and industry leaders move with discipline, determination, and diligence to ensure the events at Fukushima will not be repeated again?

It is with these questions in mind that this paper, an historical perspective of the early history of commercial boiling water reactor (BWR) severe accident analysis is presented. For the purposes of this paper, "early history" is defined to be the period between 1980 and ~ 1995. This period roughly corresponds to the period between the launch of the U.S. Nuclear Regulatory Commission's (NRC's) post-TMI severe accident research program, and its movement into the so-called "risk-informed" regulatory framework.

The author does not presume to provide a comprehensive review of all the relevant research performed in the federal and private sectors between 1980 and 1995. The author was a research engineer at Oak Ridge National Laboratory (ORNL) during that period and a member of the team that conducted some of the pioneering BWR severe accident sequence analyses that informed the development of BWR emergency operation procedures and Severe Accident Management Guidelines (SAMGs) used today. During this period, the ORNL team published ~ 250 reports, journal articles, and technical publications of various sorts in diverse venues dealing with BWR severe accident phenomenology; BWR severe accident sequence progression; BWR severe accident mitigation management studies; and BWR and PWR severe accident fission product release and transport phenomenology.[5]

As will be shown, the insights and findings from that work are very relevant today. In many ways these early findings can be viewed as the "canary in the cage" or early indicator of the potential for events such as those that occurred at Fukushima. Many insights and recommendations regarding BWR severe accident design vulnerabilities, operating procedure improvements, and accident mitigation system backfit options evolved from the work performed during that period. *Some lessons were learned. Some lessons were ignored. Other lessons may have been forgotten during the two decades since the work discussed here was performed.*

II. THINKING THE UNTHINKABLE BEFORE THE ACCIDENT AT TMI-2

The first major evaluation of the potential impacts of severe accidents in commercial nuclear power plants was published by the U.S. Atomic Energy Commission (AEC) in 1957 as the so-called, "WASH-740" report.[6] The analysis did not deal with risks per se, but examined the consequences of a hypothetical "maximum credible accident" in a 500 MWt nuclear plant located 50 km from a city of one million inhabitants. The accidents were assumed to occur near the end of an equilibrium 180-day cycle. No detailed accident sequence analysis was conducted. The analysis did not distinguish between a pressurized water reactor (PWR) and BWR, and assumed the reactor had no surrounding containment. Three different combinations of assumed core fission product release and weather conditions were evaluated. The results of the three cases where characterized as injuries (0 – 43,000), deaths (0 – 3400), and land contaminated (18 – 150,000 square miles). During this same period (late 1950s – 1960s), across the Atlantic in Great Britain, Reginald Farmer was pioneering the use of applied nuclear reactor risk assessment.[7]

In 1973, the AEC published the WASH-1250 report[8], which examined the state of reactor safety practice at that time. Among other things, WASH-1250 defined the "defense-in-depth" concept as it is known and practiced today, and endorsed the (then) emerging methodology of probabilistic safety assessment as a superior approach for synthesis of a risk-based perspective on reactor safety.

The Reactor Safety Study[9] ("RSS" or "WASH-1400" study) in 1975 was the first modern assessment of the public health and safety risks of commercial nuclear power plant operations. The RSS evaluated a wide range of potential "beyond design basis" accidents for both PWRs and BWRs, and identified a subset of accident categories that were considered "risk-dominate" for both plant types. The RSS methodology and results were initially greeted with a combination of skepticism, opposition, and (very limited and cautious) acceptance by regulators, industry, and the public.

The accident at TMI-2 in March 1979 was an event that changed the future of nuclear power. It jolted many into recognition that severe accidents of the type some in the industry had previously believed were impossible could actually occur. The TMI-2 accident (a small-break loss of coolant, or "SBLOCA" accident) was one identified by the RSS as a risk-dominant accident for PWRs. Thus the TMI-2 event served as a validation of sorts of the overall methodologies employed in the RSS.

III. TMI-2 CATALYZES BROAD RESPONSE

Following the accident at TMI-2, the NRC and the U.S. nuclear industry launched major long-term research and development (R&D) programs[10,11] aimed at developing a better understanding of severe accidents in commercial nuclear plants, and their public health and safety consequences. These programs included large experiments, computational code development efforts, and detailed accident analyses. The ultimate goal of these investigations was, of course, to inform the review and revision of the commercial nuclear power regulatory framework.

In 1980, ORNL was asked by the NRC to undertake what would become a multi-decade mission to understand severe accident phenomena, severe accident progression, and severe accident mitigation options in commercial BWR plants. During that period, the ORNL team sequentially analyzed representative accidents from the risk-dominant event groups identified by the RSS and the early Interim Reliability Evaluation Program (IREP).[12] ORNL developed unique BWR modeling and simulation capabilities, contributed to several computational code development programs, delved into the critical role of operator actions in the management and mitigation of severe accidents, and identified several BWR design features and operating procedures that play significant roles in BWR severe accident progression.

There is a common myth that the ostrich buries its head in the sand in the face of danger to avoid facing the obvious. If the early BWR severe accident studies can be viewed as the "canary in the cage" with regard to BWR severe accidents, and if the Fukushima Daiichi event can be accurately characterized as a "black swan", it is reasonable to ask whether the nuclear industry will act as the mythical ostrich in the presence of these realities.

IV. RESULTS AND OBSERVATIONS FROM EARLY BWR-4 / MARK I SEVERE ACCIDENT SEQUENCE PROGRESSION STUDIES

Some of the most important observations and lessons learned from ORNL's earlier studies of unmitigated BWR station blackout, small break loss of coolant accident, and loss of decay heat removal accidents are summarized here.

IV.A. BWR-4 / Mark I Station Blackout Severe Accidents

The first severe accident sequence analyzed by the ORNL team was the station blackout (SBO) at Browns Ferry Nuclear Plant Unit 1 (BFNP-1) – an accident sequence and a plant (BWR-4 / Mark I primary containment) very similar in many respects to Fukushima Daiichi Units 2-4 (Fig. 1). The results of the ORNL analysis were documented in two reports[13,14] issued in 1981 and 1982, respectively. The first report focused on accident sequence progression, while the second focused on fission product release from the damaged core and its transport through the plant to the environment. This two-volume documentation model was retained as the reporting model for the next five years as ORNL sequentially analyzed each representative accident from the spectrum of risk-dominant scenarios. *(The discussion here and in the sections to follow focuses on the accident sequence progression, related operator actions, and plant design details bearing upon the plant's performance during this event. The reader is referred to the companion second volume of each accident report for the related fission product transport analysis results.)*

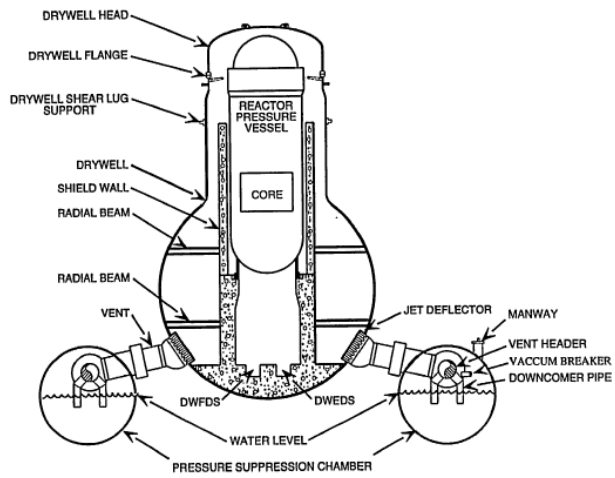

Fig. 1. Typical BWR Mark I primary containment (Ref. GE Technology Advanced Manual, NRC Technical Training Center, Rev. 1195).

The SBO accident is characterized by a complete loss of off-site power, concurrent with a failure of all onsite power sources other than main DC battery system – which was assumed to function for only 4 h based on analyses available at the time. ORNL examined six different "variants" of the SBO sequence involving different assumptions regarding High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC) system, and safety relief valve (SRV) operation (including a stuck-open SRV – a form of a SBLOCA). This was the first assessment of a BWR SBO severe accident that focused on detailed examination of the role of operator actions in the event sequence. ORNL enjoyed good cooperation with the Tennessee Valley Authority (TVA), and spent many hours in TVA's Browns Ferry plant simulator in Chattanooga investigating and understanding

plant procedures and what information reactor operators would have available to them as the accident unfolds.

It is important to note that unlike the actual Fukushima Daiichi event, no emergency diesel generator operation was assumed to occur at any time in the ORNL analysis. Some diesel generator operation apparently occurred during the 41-minute interval between the earthquake and the impact of the first massive tsunami wave[1] at Fukushima. Secondly, the ORNL analyses did not incorporate an isolation condenser such as that installed at Fukushima Daiichi Unit 1. In this sense, the ORNL analyses are more relevant to Fukushima Daiichi Units 2 and 3. Finally, it is also worth noting that other than the six combinations of HPCI/RCIC and SRV operability mentioned above, ORNL's SBO analysis assumed no damage to other plant systems. Thus all other plant systems were available and fully functional to the extent allowed by the limitations of DC power. (This was probably not the case at Fukushima Daiichi.)

The simulation tools available at the time (1980-1981) were primitive by today's standards. The accident analysis was conducted with a combination of a purpose-built computer code (BWR-LACP)[13] developed to facilitate a detailed analysis of operator actions and plant response prior to non-recoverable core uncovery, and the MARCH code[15] for the post-core-uncovery phase of the accident. MARCH had been forged by Battelle Columbus Laboratories during the height of the TMI-2 emergency, and began life as a form of "mechanized hand calculations" as termed at the time. The code had relatively few accommodations for BWR-specific design features, and was somewhat challenging to employ for such applications. Nevertheless, many useful observations evolved from the analysis. Among the more important observations and findings from the analysis were:

1. Providing there was no other plant damage from the initiating event that lead to loss-of-power, *and the plant operators responded as described below*, <u>the plant could cope well during the interval while DC power was available and would not be further damaged</u>. Thus, station battery life was the key determinant of accident sequence timing.

2. Events progressed relatively quickly following the assumed loss of DC power at 4 hours:

    a. For cases in which HPCI and/or RCIC were available at the start of the accident (*regardless of whether there was a stuck-open SRV*), core uncovery times ranged between 5.0 and 5.8 h; core melting began between 5.9 and 6.6 h; reactor vessel failure occurred between 7.0 and 7.6 h; and drywell failure (via degradation of electrical penetration assembles) occurred between 8.4 and 9.0 hours.

    b. The accident progression was dramatically accelerated for cases in which the HPCI and RCIC systems were unavailable throughout the accident. Core uncovery, core melt initiation, reactor vessel failure and drywell failure times were 0.5 h, 1.2 h, 2.2 h, and 3.2 h, respectively, for the case without a stuck-open SRV; and 0.3 h, 1.0 h, 1.4 h, 2.8 h, respectively, for the case with a stuck-open SRV.

3. The difficulty of accurately diagnosing plant damage states (reactor and containment) and accident progression with existing instrumentation and control room indicators was highlighted. It was clear there were a variety of plant system configurations, automatic system control actions, and operator actions (normal and emergency) that would compromise the BWR plant operator's ability to effectively manage an evolving severe accident.[13]

4. The operator's role in managing the event was highlighted as being extremely important. One particularly important action identified was to override the automatic actuation of the reactor safety relief valves (SRVs) to manually cycle different SRVs as necessary to ensure the pressure suppression pool (PSP) remained well mixed. This action was necessary to reduce the probability of steam blow-through that could lead to more rapid primary containment pressurization if a single low-set-point SRV were allowed to auto-cycle or stick open.

5. Manual depressurization of the reactor vessel prior to loss of DC power was identified as a critical accident management action for two important reasons:

    a. It reduces the drywell heat-up rate by decreasing the reactor primary system temperature.

    b. It removes significant stored energy from the reactor primary coolant system at a time when the RCIC system is available to reflood the reactor with

cool water from the condensate storage system. Thus, when DC power is finally exhausted (resulting in loss of injection capability and loss of remote manual SRV actuation capability even though instrument air reserves were deemed sufficient), the time required for repressurization of the reactor, boiloff of the primary water inventory, and core uncovery is lengthened – thus providing additional time for corrective and mitigative actions. (Later studies unveiled another advantage of depressurization: by reducing the water level below the bottom of the core, depressurization ensures the core is "steam-starved" in the event injection capability isn't recovered and the fuel cladding reaches run-away metal-water reaction temperatures).

6. The (then) automatic logic to shift HPCI system suction from the condensate storage tank to the PSP needed to be re-examined for the Long-Term Station Blackout (LTSBO) accident as it would likely lead to HPCI failure, because the HPCI lube oil was cooled by the PSP water – which would be at an elevated temperature at the time the shift would occur.

7. Drywell failure into the reactor building as a result of high-temperature / high-pressure failure of the drywell electrical penetration assemblies was identified as the most likely containment failure mode – occurring at a pressure approximately 30% lower than assumed in the RSS.

IV.B. BWR-4 / Mark I Small Break Loss of Coolant Accidents

The second BWR detailed severe accident sequence analysis conducted by ORNL was the small break loss of coolant accident or SBLOCA at BFNP-1. The ORNL studies were documented in reports issued in 1982 and 1983.[16,17]

The accident begins with a scram from full power, followed by a small break in the reactor primary coolant system. There are, of course, numerous options for assumed break locations in analyses of this type. ORNL's analysis assumed the break occurs in the scram discharge volume (SDV) piping following a reactor scram that cannot be reset. The SDV reservoirs are located outside primary containment in a mid-floor location in the reactor building. Thus a break in the piping in this area would provide a direct path for leakage of reactor coolant system water into the reactor building.

The size of the break was assumed to be large enough such that the effective flow area was constrained only by the graphitar seals within the control rod drive (CRD) mechanism assemblies. This leakage area was assumed to increase at 1.5 h into the accident due to temperature-driven erosion of the seals. The end-effect of these assumptions was that the break flow rate was assumed to be 1.89E-04 m^3/s or 3 gpm (at normal reactor system pressure) per each of the 185 CRD mechanisms in the reactor from the start of the accident until 1.5 h later. (This would total 0.035 m^3/s or 550 gpm for all CRD mechanisms.) Thereafter, it was assumed to escalate linearly over 6.5 h to 6.31E-04 m^3/s (10 gpm) per CRD mechanism (the later value based on measurement data for CRD mechanism with their seals removed). In addition to the flow escaping the reactor vessel, the flow into the reactor building would include another 0.011 m^3/s or 170 gpm of room-temperature water from the CRD hydraulic system via the open scram inlet valves. The break could not be isolated by the operator because it was also assumed the scram was initiated by a reactor protection system (RPS) trip signal that remained in effect throughout the accident. (The detailed description presented above is intended both to clearly define the assumptions employed in the analysis and to provide an example of the level of detail employed in the analysis.)

The more important observations and recommendations from the analyses were:

1. It was noted that, while the control room operator would receive a variety of indications (such as high radiation in the reactor building) as a result of the small break loss-of-coolant (SBLOCA) outside of containment, the operators' ability to rapidly realize a break had occurred could be clouded by indications relating to the root cause of the initial scram (such as high main steam line radiation). Nevertheless, the probability of the operator recognizing the existence of the break outside of containment would increase strongly with time due to a variety of factors.

2. Once the operators recognized they were dealing with a SBLOCA outside of containment, the emergency operating procedures (EOPs) in place at the time would become controlling. However, the relevant emergency procedure in place at the time did not call for reactor vessel depressurization unless the reactor vessel water level could not be maintained by high pressure injection systems (effectively reserving

depressurization to breaks with flow rates greater than the 0.355 m^3/s or 5600 gpm combined flow rate of the HPCI and RCIC systems.)

3. The most effective operation action would be to depressurize the reactor as quickly as possible while maintaining reactor water level via HPCI operation. It was recommended that the EOPs be modified to direct the reactor be depressurized for any non-isolatable break outside of primary containment.

4. It was noted that much of the safety-related equipment important to accident diagnosis and mitigation (such as pump and valve motor control centers) were located in the immediate vicinity of the assumed break location. Thus, even if the scram were to be reset, it was possible the necessary relays and electrical equipment would not function as needed to isolate the break due to degraded environmental conditions in the vicinity.

5. HPCI/RCIC operation would be threatened in the absence of operator action outside the scope of the EOPs. Continued break flow, coupled with cyclical operation of the steam turbine driven HPCI system would eventually result in the reactor pressure dropping below the pressure required for HPCI operation. Since normal power is assumed available throughout the accident, the condensate pumps (CPs) and condensate booster pumps (CBPs) would continue to function. When the reactor pressure drops below the cutoff head of the CPs and CBPs, these systems would rapidly flood the reactor vessel, spilling water over into the main steam lines and flooding HPCI/RCIC steam supply lines as well. Thereafter, HPCI and RCIC would no longer be available even if reactor pressure were recovered.

6. CP and CBP operation would ultimately be lost due to loss of main condenser vacuum behind the closed main steam isolation valves (MSIVs). Thereafter, no injection flow would be available to the reactor, and the top of the core was predicted to uncover at 7.4 h into the accident.

7. It was observed that, depending on the accident initiator, a variety of low pressure systems (CPs, CBPs, residual heat removal (RHR), and core spray (CS) would have the ability to over-fill the reactor vessel and flood the HPCI/RCIC steam supply lines. This would effectively eliminate all high pressure injection capability should the reactor re-pressurize later in the sequence. It was recommended that a thorough review be conducted to identify all such systems and to take the steps necessary to preclude this outcome.

8. Following core uncovery at 7.4 h, the break flow changes to steam at ~ 7.6 h when the MARCH analyses predicted the reactor water level dropped below the core and the fuel bundle flow inlets were uncovered. The system was predicted to rapidly depressurize at this point through the SDV break, followed by initiation of the low pressure coolant injection (LPCI) mode of the RHR system when pressure differential between the reactor vessel and the PSP reached ~ 2 MPa (295 psi). Nevertheless, the models in use at the time predicted the core slumped into the bottom head of the reactor vessel, which was predicted to fail at ~ 10.9 h after inception of the accident. The primary containment was assumed to be breached at that point as the hot core debris melted through the CRD water supply lines, providing a direct flow route to the break in the SDV system. (Parametric analyses were conducted to understand the impact of various low pressure injection system operating modes on reactor vessel timing. For the limiting case of no low pressure injection, reactor vessel failure timing was predicted to accelerate ~ 40 minutes.

9. As will be noted in the discussion of accident mitigation studies, a companion analysis performed at the time indicated that drywell flooding, if accomplished within the necessary timeframe, could prevent reactor vessel failure.

10. Numerous undesirable simplifications in the MARCH code (such as its inability to model core sprays) were identified.

IV.C. BWR-4 / Mark I Loss of Decay Heat Removal Accidents

The third BWR detailed severe accident sequence analysis conducted by ORNL was the loss of decay heat removal (LDHR) accident.[18,19] ORNL's analyses of this accident were published in 1983 and 1984.

This accident sequence is defined to be one in which prolonged loss of the power conversion system (PCS) and both the PSP cooling and the reactor vessel shutdown cooling modes of the RHR system lead to core damage. This accident sequence was also one of the class of accidents identified by the RSS as a risk-dominant sequence, and had been further identified by TVA as a

component of six of the dominant accidents leading to core melt in their Browns Ferry Interim Reliability Evaluation Program.[12]

The basic initiating events for this sequence include a reactor scram, closure of the main steam isolation valves (MSIVs) so that the main condenser cannot function as a heat sink, and subsequent failure of the RHR system to provide either suppression pool cooling or reactor vessel shutdown cooling. *(It is important to note that BFNP-1 does not have an isolation condenser, as was the case in Fukushima Daiichi Unit 1.)* Given these assumptions, the reactor vessel injection systems inject into the reactor and the SRVs dump steam to the PSP in a "feed and bleed" mode, while the PSP functions as the "ultimate heat sink" during the accident.

ORNL employed an upgraded version of the BWR-LACP code and the MARCH code for the analysis. Following accident initiation, the reactor operator would control reactor vessel water level with the RCIC system (the higher-capacity HPCI system would not be required). At approximately one hour into the event, the reactor operator would begin a slow depressurization of the reactor (per the EOP extant at the time of the analysis), reaching the desired pressure level (~ 85 psig) required to maintain RCIC operation at 3.5 hours. At ~ 4 hours, reactor decay heat has decreased to the point that the CRD hydraulic system flow into the vessel is sufficient to maintain core coverage. All the while, the primary containment pressure is increasing. The primary containment pressure was predicted to exceed the design pressure (56 psig) at 21.5 h, and at 24 h the primary containment pressure would exceed 65 psig – at which point the SRVs could no longer be remote-manually operated. The reactor begins repressurizing, reaching the auto-action set point of the lowest-set SRV at 24 h. The reactor was predicted to remain at that pressure, being cooled by CRD hydraulic system flow and steam-dumping to the PSP, until the primary containment failed at the junction of the spherical and cylindrical sections of the drywell liner ~ 35 hours into the event when the containment pressure reached 117 psig.

Like the earlier RSS, the ORNL analysis assumed all injection capability to the reactor was lost at the time of primary containment failure, though this outcome is by no means certain. Under this assumption, the reactor core uncovers ~ 2.5 h after containment failure, and core melting begins ~ 1 h later, or ~ 38.5 h following accident initiation.

Among the more important observations and findings from the analysis were:

1. The role of the CRD hydraulic system as an alternative means of injecting water into the reactor from sources independent of the PSP was highlighted.

2. The value of maintaining a well-mixed PSP (even if PSP cooling is not available) as a means of delaying primary containment failure was highlighted. (*Maintenance of PSP circulation and mixing was predicted to delay primary containment failure by ~ 7 hours*.)

3. The rate of primary containment pressure increase in the BWR LDHR depends on the initiating event. If at least one RHR pump and its piping loop are available following reactor scram, the PSP will remain well-mixed, and containment heatup and pressurization will proceed more slowly than would otherwise be the case.

4. Availability of the HPCI and RCIC systems would be threatened during the accident due to a combination of automatic system actions, high reactor building torus room temperatures, and elevated containment temperatures.

5. The essential role of the reactor operator in aligning and throttling different systems to maintain reactor inventory and maintain PSP circulation (by maintaining RHR pump net positive suction head or NPSH) were identified.

6. Several different options were identified for cooling the PSP and/or maintaining PSP circulation and mixing in the event the normal RHR system and/or alignments were unavailable. It was noted that many of the options carried associated risks of inducing other failures, the necessary emergency procedures were not in place to facilitate most of the options, and that further analysis was required to judge the effectiveness and prudence of their use.

7. The value of extended/secure operation of the drywell cooling system in retarding primary containment pressurization was highlighted. (However, the coolers would not be available if the initiating event included a total station blackout.)

8. Primary containment venting was identified as a key LDHR mitigation measure. The existing 46 cm drywell and wetwell vent lines (which vented to the main ventilation system) were identified as the best option, though it was noted they were

not designed for high pressure situations (so the operators would be required to vent early in the sequence), and changes would be required in both the auto-operation logic and operator procedures to enable their use in this manner.

V. THE ROLE OF BWR SECONDARY CONTAINMENTS (REACTOR BUILDINGS) IN SEVERE ACCIDENT PROGRESSION AND MITIGATION

The established practice in the few BWR severe accident simulations conducted prior to the ORNL program, was to either ignore the reactor building (Fig. 2) entirely, or, at best, to treat the reactor building with a simple 2-volume nodalization (a well-mixed reactor building up to the refueling floor, and a well-mixed refueling floor).

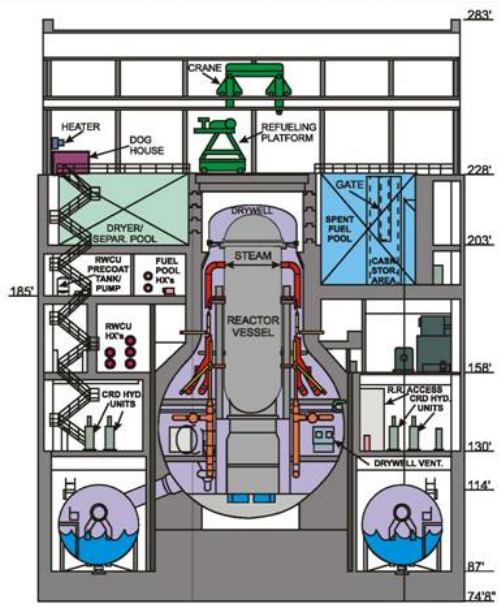

Fig. 2. Typical BWR Mark I primary containment and reactor building (secondary containment) arrangement.

However, it became increasingly clear as the ORNL studies progressed, that BWR secondary containment's intimate relationship with the primary containment ensures it has a role to play in severe accident progression and mitigation – both for accidents such as station blackout, and interfacing system loss of coolant accidents (LOCAs) where reactor building environmental conditions have the potential to lead to loss of reactor vessel injection capability. Each accident analysis conducted progressively probed more deeply into the interactions between the primary and secondary containments, and possible impact of secondary containment behavior and failure modes on severe accident progression.[20-23]

Beginning in ~ 1984, ORNL focused significant effort on the development of an improved understanding of BWR secondary containment severe accident behavior. Extensive effort was expended to understand the engineering and architectural design nuances of a number of different U.S. BWR Mark I (Mk-I), Mark II (Mk-II), and Mark III (Mk-III) reactor buildings. The effort included visits and walk-downs of a number of U.S. plants and in-depth reviews of plant design drawings. The CONTAIN code[24] and early versions of the MELCOR code[25], were applied to investigate plant-specific reactor building thermodynamics, fission product transport, and hydrogen deflagration dynamics during severe accidents. (To the author's knowledge, ORNL's porting of CONTAIN to the CRAY-I supercomputer during this period was the first successful application of a nuclear reactor safety simulation code in a supercomputing environment. The jump to the supercomputer provided an immediate and dramatic improvement in the degree of fidelity possible in the accident sequence analysis (Fig. 3).

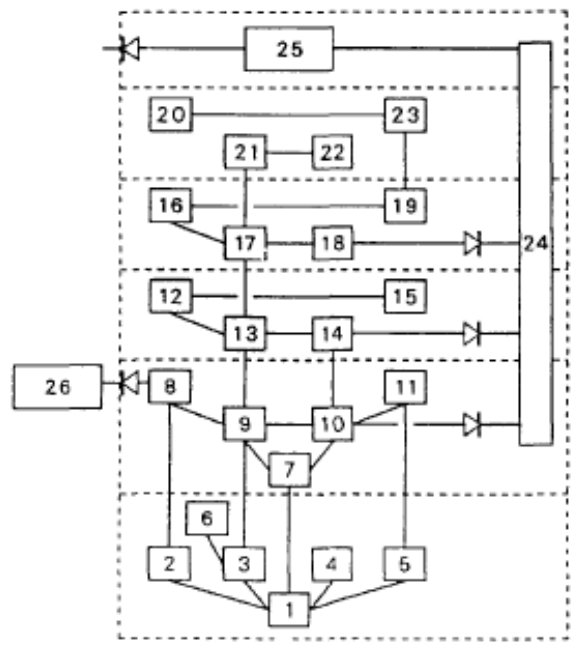

Fig. 3. ORNL Browns Ferry reactor building CONTAIN / MELCOR model nodalization ca. 1987.

The insights gained up to that time were summarized in a 1990 paper in Nuclear Engineering and Design[23] whose major findings are summarized here:

1. BWR secondary containment (reactor building) designs are highly plant-specific. (The paper identified ten different architect/engineer building designers for forty-five units in the U.S. alone, and widely differing design details such as building volumes, compartmentalization, equipment placement, fire suppression spray coverage, etc.)

2. Depending on plant design and accident sequence details, BWR secondary containments could retain as much as 90% of the radioactive aerosols released upon primary containment failure in severe accidents.

3. Primary containment failure mode and location is extremely important in determining secondary containment effectiveness. Several primary containment failure modes and locations were discussed and their impact on potential secondary containment performance examined.

4. Reactor building fire protection system sprays – even those with relatively small coverage – can significantly increase reactor building aerosol retention in a severe accident.

5. The role of the reactor building Standby Gas Treatment System (SGTS) and Reactor Building Standby Ventilation System (RBSVS) are potentially important in terms of their impact on both combustible gas concentrations in the reactor building, and the magnitude of fission product releases during and after the accident. However, due to plant-specific variations in system design specifications (principally flow capacity, suction/exhaust locations, and filtration technologies employed), and accident-specific considerations (whether power is available and for how long, and the aerosol loading on the system filters), the effectiveness of these systems would be highly variable.

6. Combustible gas (hydrogen and carbon monoxide) deflagrations or detonations in the reactor building have the potential for major damage to the building and reduction of its severe accident mitigation effectiveness. The severity of these deflagrations / detonations were found to be a function of primary containment failure location, reactor building compartmentalization, and reactor building ventilation design details. Hydrogen detonation-induced reactor building pressures of as much as 221 kPa (32 psia) were predicted in some ORNL simulations – a value exceeding reactor building design pressure differentials by almost a factor of four.

7. The observation was offered that the severe accident mitigation effectiveness of BWR reactor buildings could be improved by three generic steps: (1) reducing the probability of secondary containment bypass, (2) ensuring secondary containment integrity, and (3) improving secondary containment fission product retention capability. It was noted that chief among the options to be considered were the implementation of dedicated primary containment sprays, hardened (unfiltered or filtered) primary containment venting systems, and gravity-fed reactor building sprays.

8. It was pointed out that the blowout panels that isolate the turbine building from the reactor building in most plants would open in many severe accident sequences (either due to the initiating event, primary containment failure, or combustible gas deflagrations in the reactor building), thus providing a pathway for fission product release through the turbine building to the environment.

VI. BWR MARK II AND MARK III CONTAINMENT SEVERE ACCIDENT BEHAVIOR INSIGHTS

The BWR Mark II and Mark III Parametrics Program was established at ORNL in 1988, with the goal of providing best-estimate analyses of generic Mk-II and Mk-III severe accident behavior and assessing the potential value of hardware improvements that could impact severe accident containment performance. The major observations from the Mk-II and Mk-III work were published in 1991 (Ref. 26-28), with some work continuing through 1995 (Ref. 29). The analyses were conducted with the BWR-LTAS, BWRSAR, and MELCOR codes, and focused on the short-term station blackout (STSBO) sequence.

VI.A BWR Mark II Severe Accident Containment Response

Although the BWR Mk-II primary containment employs the same pressure suppression philosophy as the Mk-I design, the physical construction and arrangement of Mk-II containments is very different. The principal difference between Mk-I and Mk-II containments is that, unlike the Mk-I, which utilizes an “inverted light bulb” drywell and toroidal wetwell, the Mk-II employs a conical-shaped drywell sitting directly above a cylindrical wetwell (Fig. 4).

The Mk-II analyses consisted of detailed examinations of seven STSBO sequences and one LTSBO sequence at the Susquehanna Steam Electric Station. The following main conclusions and recommendations were offered[26]:

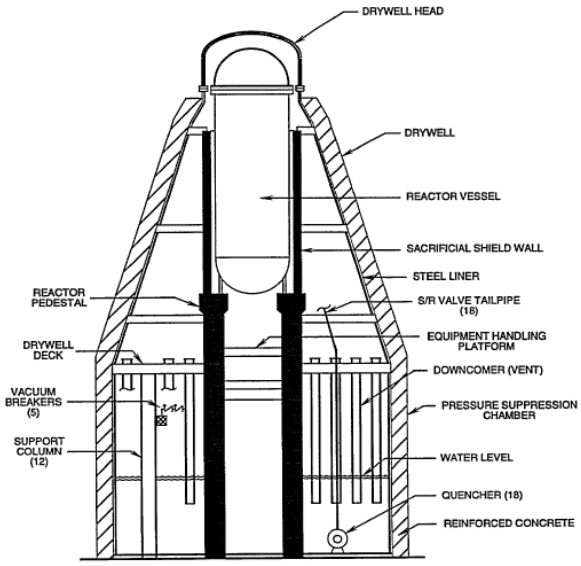

Fig. 4. Typical BWR Mark II primary containment (Ref. GE Technology Advanced Manual, NRC Technical Training Center, Rev. 1195).

1. Mk-II reactor pedestal and cavity designs in the six domestic Mk-II plants are very plant-specific. Both the dimensions of the reactor cavity underneath the reactor vessel and the presence of drywell vent downcomers within the pedestal are plant-specific. These differences have the potential to impact the effectiveness of emergency procedures such as drywell flooding and to speed the entry of core debris into the pressure suppression pool.

2. Mk-II drywell head flange seal designs are plant-specific design features. Since the head flange seal is a potential drywell containment failure pathway, it was not possible to generically evaluate this drywell failure model.

3. Reactor vessel Automatic Depressurization System (ADS) actuation significantly impacts the rate of core degradation. (Relocation of core debris was predicted to begin ~ 0.5 h later under Rev. 3 of the BWR Emergency Procedure Guidelines (EPGs) than Rev. 4 of the EPGs.)

4. Primary containment failure for the STSBO was predicted to occur (assuming 931 kPa or 135 psig failure pressure) in the depressurized reactor scenario at ~ 8.5 h when 100% of the core debris melted through the drywell floor and plunged into the wetwell below. This same failure mode was predicted to occur at ~ 10 h for the case in which the reactor is not depressurized prior to reactor vessel failure, but all of the debris remains inside the reactor cavity.

5. The predicted time for drywell floor melt-through ranged from 12.2 h to greater than 26.7 h for some sequences.

6. Failure to actuate the ADS would shorten the time to primary containment failure by as much as three hours.

7. The effectiveness of drywell flooding as an accident mitigation method is in question for Mk-II containments. The depth of the water pool that can be achieved in the drywell is limited both by the placement and height of the drywell downcomers (above the drywell floor), and the shut-off head of the pumps available for use in flooding the containment. The drywell pressure was predicted to exceed the cutoff capacity of the pumps at ~12 h into the sequence. The existing pool of water was predicted to boil away, and the drywell floor was predicted to collapse at ~ 16.7 h into the event – failing the containment on overpressure. However, it was noted the containment pressure was close to its failure limit at the time of drywell floor collapse, and so the value of the flooding would relate more to scrubbing of aerosols and fission products evolving from the melt debris bed, than to delay in primary containment failure timing.

8. The response of the Mk-II containment to the LTSBO is very similar to the STSBO in which the ADS is not actuated – except that the sequence is delayed due to the early availability of reactor vessel injection prior to battery exhaustion. Containment failure via overpressure was predicted to occur at ~ 20.8 h in this sequence compared to 10 h in the STSBO sequence without ADS actuation.

VI.B BWR Mark III Severe Accident Containment Response

ORNL's BWR Mk-III analyses also focused on the STSBO sequence, and proceeded in parallel with the Mk-II analyses.[27] The Grand Gulf Nuclear Station was employed as the reference design for the analyses. BWR Mk-III containments are much larger than Mk-I and Mk-II containments, and are not inerted (Fig. 5). They

employ hydrogen igniters inside the primary containment to control hydrogen concentrations resulting from cladding oxidation in a design basis loss of coolant accident.

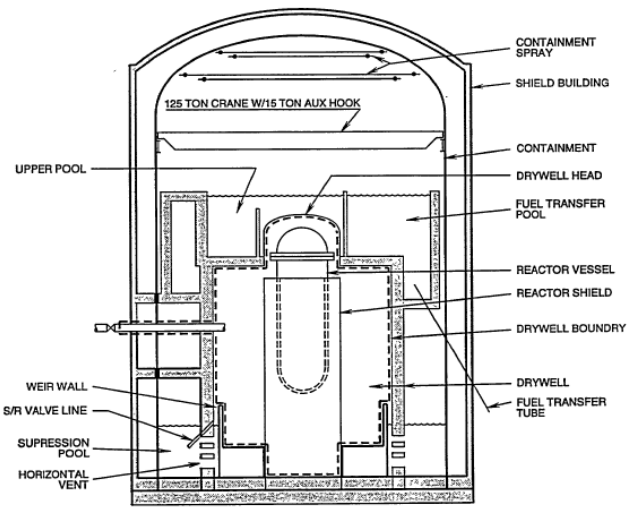

Fig. 5. Typical BWR Mark III primary containment (Ref. GE Technology Advanced Manual, NRC Technical Training Center, Rev. 1195).

Once again, the BWR-LTAS, BWRSAR, MELCOR code suite was employed. ORNL evaluated six cases involving different assumptions regarding operation actions: (1) no operator action, (2) only containment venting via existing purge lines, (3) only hydrogen igniters are active, (4) both hydrogen igniters and vacuum breakers are available, (5) both hydrogen igniters and containment venting are available, and (6) hydrogen igniters, containment vents, and vacuum breakers are available and functioning throughout the sequence. The following observations were presented:

1. As was the case in the Mk-II Susquehanna, it was determined that the timing of ADS actuation had a non-trivial impact on in-vessel core relocation time. Use of the Rev. 4 EPGs accelerated core degradation and relocation by ~ 0.5 h.

2. Three dominant challenges to Mk-III containment integrity were identified: (a) hydrogen-detonation-induced containment overpressure, (b) overpressure or missile-impingement-induced drywell failure due to ex-vessel steam explosions, and (c) static overpressure of the outer containment air locks and penetrations.

3. Static containment (wetwell) pressures reached their maximum value of 206.8 kPA (30 psig) at 24 h in cases 1 and 4. Since this did not exceed the estimated containment failure pressure of 386 kPA, it was concluded the containments would probably not fail on overpressure. Maximum containment pressures for all other sequences did not exceed 89.6 kPa (13 psig).

4. Hydrogen produced from in-vessel zirconium-steam reactions in Case 1 was predicted to reach detonable concentrations in the wetwell within 2 h of the initiation of the accident.

5. Following reactor vessel failure, ex-vessel debris interactions increase hydrogen concentrations and drive the drywell atmospheric temperature well above the "auto-deflagration" temperature. Thus the likelihood of hydrogen deflagrations and/or detonations in the primary containment was deemed to be very high in the absence of operable igniters.

6. Containment venting with the existing 20-in. diameter lines does not significantly reduce the probability of hydrogen detonations in the primary containment and possibly increases the probability of significant fission product release to the environment due to drywell wall penetration leakage into the wetwell airspace (bypasses the suppression pool).

7. If the hydrogen igniters are supplied with backup power supply to enable their operation during the accident, they will be very effective in controlling the primary containment hydrogen concentration below detonable limits in the absence of wetwell venting. The impact of wetwell venting is to temporarily render the drywell igniters ineffective for ~ 4 h due to a reduction in drywell oxygen concentration.

VII. BWR EMERGENCY PROCEDURE AND ACCIDENT MANAGEMENT INSIGHTS

Beginning in the mid 1980's, and continuing through the mid-1990's increasing emphasis was placed on the identification of effective severe accident mitigation options – particularly those aimed at preventing reactor vessel failure and escape of core debris from the reactor vessel. These studies provided valuable early insights used in the establishment of the EOP / SAMG / Extreme Damage Mitigation Guideline (EDMG) framework in place today. The examination of such options was always accompanied by the conviction that if water is available and the reactor vessel is intact, first priority should always

be given to injecting the water into the reactor vessel. Given that BWRs have such diverse systems (and interconnections between them) for injecting water into the reactor vessel, little attention had been given to exploring methods for injecting water into the BWR drywell – though ORNL had explored drywell flooding as a severe accident mitigation scheme[30] in 1982.

In 1991, ORNL embarked on an NRC-funded, two-year study to evaluate the BWR Owner's Group Emergency Procedure Guidelines (EPGs) then in place. These guidelines focused on preventative measures to avoid core damage. The primary purpose of the review was to examine the extent to which the EPGs implemented the intent of the BWR accident management strategies proposed by Brookhaven National Laboratory.[31] The second objective was to examine the effectiveness of the EPGs during the "late-phase in-vessel" period of a severe accident and, if possible, to identify candidate strategies that could enhance or extend the EPGs for the management of severe accidents. The results of this study were published in 1991 and 1992 and are summarized here.[32,33] The ground rule for ORNL's analysis was the constraint that the identification of any new accident mitigation strategies "should not require major equipment modifications or additions, but rather should be capable of implementation using only the existing equipment and water resources of the BWR facilities."[33]

The Anticipated Transient Without Scram (ATWS) had consistently been identified in prior BWR risk assessments as second only to station blackout in risk importance. Containment events were/are the basic cause of loss-of-injection capability to the reactor and subsequent core damage. ORNL's recommendations with regard to EPG ATWS procedures *prior to core damage* were focused (a) on simplifying the procedures, (b) taking steps to ensure the reactor operators do not attempt to depressurize a critical reactor, (c) changing the method of reactor power control during ATWS from one keyed to reactor vessel water level to one tied to reactor vessel injection rate, (d) overriding the rod sequence control system to enable manual insertion of control blades under ATWS conditions, and (e) delaying the timing of automatic depressurization system actuation until lower water levels are reached, in order to enhance the thermal quenching value of the operation.

ORNL examined four candidate generic late-phase in-vessel accident mitigation strategies:

1. Ensuring that, once depressurized, the reactor vessel can be maintained in a depressurized state, by providing an alternate means of reactor vessel venting if the SRVs become inoperable due to loss of control air or DC power. This option was not recommended for implementation because it was considered more practical to improve the reliability of the control air and DC power supplies to the SRVs.

2. Strategies to restore reactor vessel water injection in a controlled manner following core damage. These strategies basically revolved around ensuring high-capacity, low pressure injection systems do not respond in a manner that complicates – rather than enables – accident recovery.

3. Strategies to ensure boron injection (particularly important if the control blades have been damaged) following the onset of core damage. One method emphasized was to implement a means to mix boron-10 directly into the plant condensate storage tank (CST) water, and provide a means for low pressure injection systems to draw suction from the CST – thus combining the response to Strategies 2 and 3.

4. Containment flooding to maintain the core and structural debris within the reactor vessel.

The fourth strategy (containment flooding) warrants further discussion, as it was the principal focus of the report and the analysis remains one of the most detailed evaluations of the issue. The focus of the analysis of this issue was the Mk-I drywell system such as that employed at Browns Ferry, Peach Bottom, and Fukushima Daiichi Units 1-3. The immediate goal of the strategy was to evaluate the effectiveness of, and potential for raising the water level in the drywell to a point sufficiently high on the reactor vessel external wall to cool the vessel and avoid reactor vessel failure (via melting of the vessel drain, instrument tubes, control rod drive guide tube penetrations, or overall vessel head itself) – even in cases where the bulk of the core debris was resting on the bottom head of the vessel.

Based on detailed multi-dimensional analysis of in-vessel melt progression, likely in-vessel lower head debris bed configurations, and reactor vessel through-wall heat transfer, the following observations and conclusions were reached:

1. No portion of the reactor vessel pressure boundary (including drains, instrument tubes, etc.) in contact with water on its external surface would fail.

2. Most of the upper portion of the reactor vessel could not be submerged by water due to the location of the drywell vents – which would have

to remain open to facilitate the flooding procedure.

3. Full submergence of the reactor vessel lower head would be impeded by the development of a trapped gas pocket between the reactor vessel support skirt and the lower head. This problem could be partially remedied by intentionally leaving the manhole access door in the skirt open. However, it would be necessary to modify the skirt by drilling holes in it or adding a siphon tube to ensure water would contact the entire lower head. The report states, "Because of the associated personnel radiation exposure penalty and the predicted low core melt frequencies for the existing plants, this is not considered to be a practical suggestion for the existing BWR facility, but provision for complete venting might be easily implemented for the advanced BWR designs."[33]

4. Based on # 3, and analysis of the Browns Ferry BWR-4 / Mk-I system response to an unmitigated short-term station blackout (one in which all reactor vessel injection capability is lost at the inception of the accident), it appeared drywell flooding would preclude early over-temperature failure of the reactor vessel drain and instrument tubes. However, if penetration failures did not occur, the lower reactor vessel head would be expected to fail via creep rupture at ~ 10 h for the case in which the drywell is not flooded, and ~ 13 h for the case in which the drywell is flooded.

5. Even if reactor vessel lower head failure could be avoided by drywell flooding, it appeared upward radiation from the top surface of the lower plenum core debris bed, if left unchecked, would eventually lead to failure and downward relocation of the upper reactor vessel internal structures (steam dome, steam separators, and steam dryers). Depending on the nature of the upper internals relocation, the reactor vessel could be assumed to fail at that point or sometime thereafter as the upper debris bed surface continued to heat the upper reactor vessel wall and upper head.

6. An important disadvantage of drywell flooding is the requirement for venting of the drywell atmosphere while the containment is being filled by existing low pressure pumping systems. Depending on the accident-specific, and plant-specific details, this venting could result in early releases of radioactivity to the external environment if non-hardened, non-filtered vent paths were employed.

7. Containment flooding was highly likely to protect the BWR Mk-I drywell liner from failing due to direct contact with hot core debris in the event of reactor vessel failure and core debris escape from the reactor.

8. Existing plant systems were inadequate for accomplishing containment flooding on the time frame necessary to preclude reactor vessel failure. Because it was considered unrealistic to expect the emergency procedures to call for containment flooding until core damage had occurred, plant operators would have only ~ 150 minutes (the length of time predicted between core damage and lower plenum debris bed dry-out) to flood the drywell.

9. Finally, it was noted that a few simple modifications to future BWR designs (such as locating the reactor vessel in a cavity and adopting a vented skirt design) would greatly facilitate the use of drywell flooding as a severe accident management procedure.

Most of these conclusions are still considered valid today.

VIII. BWRS ARE DIFFERENT – THE CHALLENGES OF SIMULATING SEVERE ACCIDENTS IN BWRs

From the time of ORNL's first BWR station blackout analysis in 1981, and continuing to this day, BWR severe accident simulation capabilities have generally lagged those of PWRs. This is true both because BWRs comprise only ~ one third of the commercial nuclear fleet (and therefore naturally tend to hold second priority), and because BWRs are in some respects more complex machines than their PWR cousins. Work began in 1981 to adapt and improve the MARCH code for BWR applications.[34,35] ORNL published a comprehensive analysis of BWR severe accident simulation requirements in 1984. The report[36] presented a detailed analysis of in-vessel, ex-vessel (primary containment), and secondary containment structures, components, and systems relevant to severe accident progression in BWRs. The dozens of design details identified can be grouped loosely into five categories of BWR-specific features:

1. The multiplicity of diverse options for injecting water into the reactor vessel.

2. In-vessel structures (steam separators and dryers in the upper reactor vessel; Zircaloy channel

boxes and cruciform control blades in the core; and control rod drive guide tubes and instrument tubes in the lower regions of the reactor vessel).

3. The multitude of penetrations (for instrument tubes, control rod drive guide tubes, and the reactor vessel drain) in the reactor vessel bottom head.

4. Mk-I, Mk-II, and Mk-III primary containment configuration (reactor vessel pedestal and support flange, drywell liners, the PSP, distributed drywell vent downcomers and safety relief valve exhausts into the PSP, vacuum breakers, etc.).

5. The intimate relationship between the BWR primary containment and the surrounding reactor building (secondary containment), and the complexity of the reactor building and its embedded structures, components, and systems.

As time passed and severe accident codes such as CONTAIN, MELCOR and RELAP/SCDAP[37] continued to evolve, the challenge of simulating the phenomenological impacts of some of the design features noted above proved to be more problematic than others. However, two areas of simulation proved so challenging ORNL found it necessary to develop BWR simulation tools with capabilities beyond those available at the time in the NRC's other simulation codes.

The first area of focus had to do with accurately simulating automatic plant system actions, operator actions, and the diversity of reactor vessel injection system options from initiation of the accident to the point in time in which core damage occurs. This challenge was addressed via the development of the BWR-LACP code (previously mentioned) and the BWR-LTAS code.[38] These tools provided analysts a simulation capability similar in some respects to the software behind physical plant simulators of the day, coupled with plant operator logic models and in-vessel thermodynamic analysis capabilities unavailable in standard plant simulators.

The second area of focus, and one that continues to be a topic of investigation and debate today, was (is) lower in-vessel BWR melt progression and bottom head failure phenomenology. Following its identification in the 1984 report[36] as a major challenge to accurate analysis of BWR severe accidents, ORNL embarked on a sustained effort to improve the state-of-the-art in BWR in-core, lower-vessel, and bottom head failure melt progression simulation.[39-42] This effort, which combined detailed knowledge of BWR design details and emerging data from U.S. and international core melt progression experiments, resulted in the BWRSAR code.[43] BWRSAR provided the first detailed ability to differentiate the oxidation and melt progression of fuel rods, channel boxes, control rods, and control rod drive guide tubes in a BWR – and thus spawned many insights into BWR accident sequence progression. Subsequent to BWRSARs development, ORNL's typical analysis approach employed BWR-LTAS, BWRSAR, and either CONTAIN or MELCOR. For a time, BWRSAR's in-vessel accident progression models were incorporated in MELCOR.[44] *However, the current version of MELCOR (MELCOR 2.1) no longer employs the detailed BWRSAR-derived lower plenum melt progression and lower head failure models. The BWRSAR control blade melt progression models were also integrated into SCDAP*[45,46] *at one point, and the author understands they are still resident in the current commercial version of the code (RELSIM).*

IX. MEGA-LESSONS FROM 1980 - 1995: WHAT WE KNEW, AND WHEN WE KNEW IT

The previous sections have identified, in some detail, major accident-specific lessons and observations derived from over a decade of focused BWR severe accident sequence analysis between 1980 and 1995. Here we move to a higher-level summary the "mega-lessons" from that period of inquiry:

1. Virtually from the outset of the earliest severe accident analyses in 1981, it became clear that neither standard plant instrumentation, nor the control room indicators were capable of providing the desired quantity and quality of information once a severe accident progresses to the point of major core damage.[47] Little has transpired to change this conclusion in the thirty years that have passed since these first observations were made.

2. The evolution of "symptom-based" emergency and severe accident management procedures has a pedigree tracing back to 1980. There is no question the industry's focus on the development of these procedures has improved the safety of nuclear power. However, one of the reoccurring lessons of the historical studies discussed here is that the path one followed to arrive at a point in the event sequence sets very real limits on one's options for response. Thus, *while the symptoms presented may be sufficient to diagnose a plant's present damage state, knowledge of how one arrived at the current damage state can be critical to predicting future event progression, and identifying the best intercession/mitigation options to retard or halt the progression of the*

accident. This may require real-time knowledge of plant parameters not normally of interest to the operator and not conveniently available to them.

3. Plant operators are the key to effective severe accident response. It is not possible to have a canned "recipe" for every possible severe accident condition. However, *operators should not be expected to deal effectively with accidents the engineers have not studied in advance, and which the operators have not trained for in realistic simulated conditions.*

4. The majority of previous accident mitigation studies focused on the creative use of existing plant systems and components, rather than on identification and development of more costly system backfits or major hardware improvements. The justification for this limited span of consideration can be traced (at least in part) to the belief that severe accidents will not occur in "my plant", and the use of societal risk/reward metrics that relate exclusively to human health impacts.

X. THE EVOLUTION OF A SEVERE ACCIDENT REGULATORY FRAMEWORK

Following the RSS in 1975, the NRC funded Sandia National Laboratories to conduct an assessment of the potential socio-economic impacts of severe accidents at 91 existing or proposed commercial reactor sites around the U.S. The study,[48] commonly referred to as "the Sandia Siting Study", or simply "the Siting Study" was released in 1982, and utilized five generic source terms derived from the RSS results. The study did not attempt to integrate the results into a more holistic assessment of public risk. The Siting Study results are widely considered today to be overly conservative with regard to severe accident radiological source term magnitudes and frequencies of release. However, at the time the study was performed (soon after the accident at TMI-2), it attracted considerable attention both within and outside the reactor safety community. The Siting Study has remained one of the most widely quoted sources of information regarding the potential consequences of severe accidents at commercial nuclear power plants to this day.

NRC's 1985 Severe Accident Policy Statement[49] concluded "existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants." However, the Commission recognized the value of systematic plant examinations for identification of plant-specific vulnerabilities to severe accidents that could be corrected with "low cost changes in procedures or minor design modifications."

The NRC announced its adoption of quantitative safety goals for operating nuclear power plants in 1986. In addition to promoting "societal risk goals" for individuals and populations based strictly on expected cancer fatalities, the Commission announced its adoption of a "large release fraction" or "LRF" goal specified as "the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."[50] The release of the safety goal statement was a catalyst for a massive increase in efforts within the commercial nuclear industry directed at understanding severe accidents and quantifying the risk associated with them.

Following on the heels of the Severe Accident Policy Statement and issuance of its new safety goal, the NRC issued Generic Letter 88-20 (GL 88-20), in November 1988. GL 88-20 requested each licensee conduct an individual plant examination (IPE) "to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission.[51]

Originally published for comment in 1987, the NRC's NUREG-1150 report[52] provided a systematic analysis of the risks from severe accidents in five commercial nuclear power plants in the U.S. – including the Peach Bottom BWR-4 / Mk-I plant and the Grand Gulf BWR-6 / Mk-III plant. The detailed BWR accident progression and risk analyses were more fully documented in NUREG/CR-4551, Volume 4 (Peach Bottom)[53] and Volume 6 (Grand Gulf).[54] The Peach Bottom analyses examined overall core damage frequencies and risks from both internally-initiated and externally-initiated events. The analysis indicated the major contributors to total mean core damage frequency (estimated to be 4.5E-6 per reactor-year) from internally initiated events were station blackout and ATWS – with smaller contributions from loss of coolant accidents (LOCAs) and transients (Fig. 6).

The predicted core damage frequency from externally initiated events (estimated to be 9.7E-5 per reactor-year) was dominated by seismically-induced transient loss of off-site power, seismically-induced loss of coolant accidents, and fire-induced station blackouts (Fig. 7).

The NUREG-1150 Peach Bottom accident analyses confirmed and extended many of the earlier ORNL BWR severe accident sequence analysis insights. In particular, the NUREG-1150 authors noted:

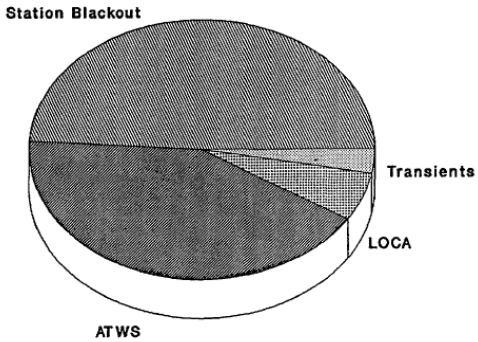

Fig. 6. NUREG-1150 Peach Bottom results: contributions to core damage frequency from internally initiated events.

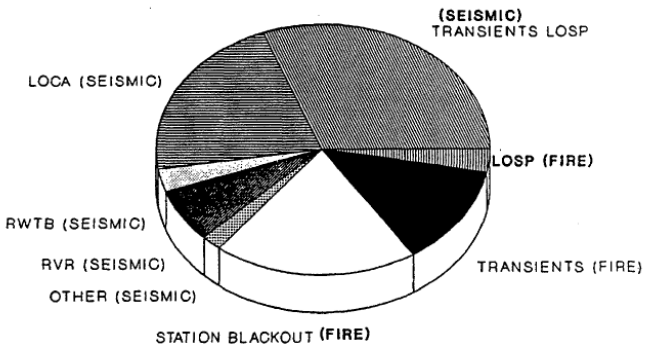

Fig. 7. NUREG-1150 results for external event contributors to total mean core damage frequency at Peach Bottom.

1. The value and importance of the many diverse BWR water supply and heat removal systems.

2. The redundancy and diversity of water supply systems.

3. The importance of diesel generators and battery capacity.

4. The potential value of diverse primary containment venting systems.

5. The importance of the location of the control rod drive hydraulic pumps (unlike Browns Ferry, Peach Bottom's pumps were located in the turbine building).

6. The importance of automatic and manual reactor vessel depressurization systems and approaches.

7. The critical role of operator actions in managing the response to the accident.

8. The importance of containment inerting and drywell sprays in impacting the response of the drywell during both pre- and post-reactor vessel failure phases of the severe accident.

9. The important role of the pressure suppression pool, containment venting, and the reactor building in impacting environmental releases of radioactivity following the accident.

NUREG-1150's Grand Gulf analysis considered only the risks from internally initiated events, and concluded the mean core damage frequency (4.0E-6 per reactor-year) was almost completely dominated by internally initiated station blackouts, with a very minor contribution from ATWS events.

Based in part on the NUREG-1150 results, the NRC issued Generic Letter 88-20, Supplement 4, in June 1991. This notice expanded prior IPE guidance by requesting each licensee perform "an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvement and corrective actions to the Commission."[55] These analyses were known as Individual Plant Examinations of External Events, or IPEEE.

The results of the IPE and IPEEE analyses yielded a continuing stream of insights regarding plant-specific and plant-type-specific severe accident vulnerabilities and risk-significant structures, components, and system design details.[56-60] The findings of these studies have generally been consistent with the earlier studies, but have provided significant additional insight into the importance of plant design features and their impact on BWR plants' vulnerabilities to, and response capabilities for, externally-initiated seismic and fire events. Plant owners and operators proceeded cautiously during the mid-1990s to the present day to address high-value plant upgrades identified as worthy of implementation based on the IPE and IPEEE results.

Most recently, the NRC released the results of its long-awaited, and much peer-reviewed, "State-Of-The Art Reactor Consequence Analysis (SOARCA)".[61] A principal goal of the SOARCA project was to update the results of the 1982 Sandia Siting Study[62] to reflect changes and improvements in technologies, systems, procedures, and emergency response strategies since the time the Siting Study was performed. A second priority of SOARCA was to place the accident consequences in an overall risk context (something the original Siting Study

did not do). The SOARCA project analyzed three severe accidents for the Peach Bottom BWR-4 / Mk-I plant: LTSBO (the most probable event), STSBO, and a "loss of vital alternative current Bus E-12" event. The general conclusions of the Peach Bottom SOARCA analyses are that radiological releases are smaller, and occur later than those predicted in the Siting Study. Even the unmitigated versions of the SOARCA scenarios present lower risk of early fatalities than those derived when the Siting Study source terms are applied under similar assumptions.

XI. SEVERE ACCIDENTS, RISK, AND THE CURRENT REGULATORY FRAMEWORK

The long series of events described have led to the current U.S. nuclear safety framework. This framework incorporates two *qualitative* safety goals, and three *quantitative* goals for new reactors.[63] The two *qualitative* goals are:

1. Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

2. Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The three *quantitative* goals are:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

- The LRF shall occur with a frequency of less than 1.E-6 per reactor year of operation.

Based on dialog with the Advisory Committee on Reactor Safeguards (ACRS) and others, the NRC has adopted two additional *subsidiary* quantitative safety goals:

1. The core damage frequency (CDF) shall be less than 1.E-4 per reactor year.

2. The conditional containment failure probability (CCFP) for advanced LWRs shall be less than 0.1.

Given the events at Fukushima Daiichi, and their aftermath, it is reasonable to ask if these criteria and the current regulatory framework are truly adequate to protect modern society from the range of public health and safety consequences resulting from such accidents.

XII. PUTTING IT ALL TOGETHER – WHERE'S OUR HEAD?

The American Society of Mechanical Engineers (ASME) has recently issued a call to forge "a new nuclear safety construct".[64] As noted in the ASME report, "*the major consequences of severe accidents at nuclear power plants have been socio-political and economic disruptions inflicting enormous cost to society*." The report goes on to state, "*Socio-political and economic consequences such as experienced in Japan after the Fukushima accident, even if caused by extreme natural disaster, are unacceptable*." The report calls for a "*new nuclear safety construct*" – "*the set of planned, coordinated, and implemented systems ensuring that nuclear plants are designed, constructed, operated and managed to prevent extensive societal disruption caused by radioactive releases from accidents, using an all-risk approach*."

Thus, it is in the interest of furthering the dialog regarding "a new safety construct", the author presents these final observations:

- Based on historical BWR station blackout studies, and given the hybrid short- / long-term station blackout sequence that occurred at Fukushima Daiichi, we have little reason to be surprised about the course and timing of events that occurred in Fukushima Daiichi Units 1-3.

- Little solace should be taken in the (very real) fact that the direct consequences of the earthquake and tsunami that occurred in Japan on March 11, 2011 far exceed those resulting from the ensuing accident at the Fukushima Daiichi plant. It is sobering to consider the likely impact of such an event, had it occurred in the U.S. – even in the absence of the causal earthquake and tsunami.

- Neither the nuclear industry nor the public should settle for a reoccurrence of events such as those that transpired at Fukushima. The investment community is unlikely to settle for such events, even if the industry and the public are not roused to action by the accident. *We have the knowledge and tools to understand such accidents, to identify prudent plant modifications and improvements, and to implement those improvements. The question is: do we have the vision and will to do so?*

- Strictly speaking, the NRC's current U.S. societal risk goals were not violated by the events at Fukushima Daiichi. Thus, from the perspective of one who feels Fukushima-like accidents are unacceptable (regardless of their cause), it is reasonable to question whether our current risk goals provide the public adequate protection against major accidents. *A robust dialog regarding the metrics and values for quantitative societal risk goals is in order.*

- Today's Severe Accident Management Guidelines are symptom-based. Symptoms indicate current status, but reveal little about the history of how one arrived at a point. A plethora of historical severe accident sequence studies such as those discussed here have revealed that the manner in which one arrives at a point can strongly influence the future progression of the accident from that point, and the options available to deal with the situation at that point in time. Thus, it is reasonable to ask whether the symptom-oriented severe accident management guideline approach is optimal (or even sufficient). *It would be prudent to review current severe accident management guidelines in this light*.

- The overlaps and interfaces between EOPs, SAMGs, and EDMGs remain undesirably vague in some instances, despite years of earnest effort by the industry to resolve these interfaces. *A holistic look at the interfacial dynamics involved in applying the EOPs, SAMGs, and EDMGs in the real-time environment of an evolving severe accident is in order to ensure maximum value is gained from the evolving lessons-learned from the Fukushima Daiichi event.*

- Finally, (and as noted by the NRC's Near-Term Task Force[65]), *the coupling and integration of the EOP / SAMG / EDMG framework, and the broader emergency preparedness / emergency response / protective action framework should be re-examined in light of evolving Fukushima lessons-learned.*

The late Dr. Alvin Weinberg was a passionate advocate for nuclear safety, and often stated his conviction that the nuclear power industry has an obligation to the public that borders on a sacred trust.[66] One can only imagine what his views would be regarding the events of March 2011 and our responsibilities in their wake. As noted in the beginning of this paper, the widespread belief that the ostrich sticks its head in the sand to avoid stressful circumstances and stark realities is a myth. Nevertheless, millions around the world believe this myth to be true. *The industry must go beyond the expedient in its response to the Fukushima Daiichi disaster* if we are to avoid the image problem of the ostrich. But the stakes are much higher than simply our image. The confidence of the public and the investment community hang in the balance – and perhaps the future of the nuclear power option as well.

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NOMENCLATURE

ACRS – Advisory Committee on Reactor Safeguards
AEC – Atomic Energy Commission
ASME – American Society of Mechanical Engineers
ATWS – Anticipated Transient Without Scram
BFNP-1 – Browns Ferry Nuclear Plant Unit 1
BWR – Boiling Water Reactor
CDF – Core Damage Frequency
CBP – Condensate Booster Pump
CCFP – Conditional containment failure probability
CP – Condensate pump
CRD – Control Rod Drive
CS – Core Spray
CST – Condensate Storage Tank
DC – Direct Current
EDMG – Extensive Damage Mitigation Guideline

EOP – Emergency Operating Procedure
EPG – Emergency Procedure Guideline
IPE – Individual Plan Examinations
IPEEE – Individual Plant Examinations of External Events
IREP – Interim Reliability Evaluation Program
HPCI – High Pressure Coolant Injection
LDHR – Loss of Decay Heat Removal
LOCA – Loss of Coolant Accident
LRF – Large Release Fraction
LTSBO – Long-term Station Blackout
Mk-I – Mark I
Mk-II – Mark II
Mk-III – Mark III
MSIV – Main Steam Isolation Valve
MWt – Megawatts thermal
NRC – U.S. Nuclear Regulatory Commission
ORNL – Oak Ridge National Laboratory
PSP – Pressure Suppression Pool
PWR – Pressurized Water Reactor
R&D – Research and Development
RBSVS – Reactor Building Standby Ventilation System
RCIC – Reactor Core Isolation Cooling
RHR – Residual Heat Removal
RPS – Reactor Protection System
RSS – Reactor Safety Study
SAMG – Severe Accident Management Guideline
SBLOCA – Small Break Loss of Coolant Accident
SBO – Station Blackout
SDV – Scram Discharge Volume
SGTS – Standby Gas Treatment System
SOARCA – State-Of-the-Art Reactor Consequence Analysis
SRV – Safety / Relief Valve
STSBO – Short-term Station Blackout
TMI-2 – Three Mile Island Unit 2
TVA – Tennessee Valley Authority

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