CONF—8610135--41

THE ROLE OF BWR MK I SECONDARY CONTAINMENTS IN SEVERE ACCIDENT MITIGATION

CONF-8610135--41

DE87 002180

Sherrell R. Greene
Oak Ridge National Laboratory
Oak Ridge, Tennessee

For Publication in the Proceedings of the 14th Water Reactor Safety Information Meeting at the National Bureau of Standards, Gaithersburg, Maryland, October 27—31, 1986.

Research sponsored by the Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission under Interagency Agreement DOE 40-551-75 with the U.S. Department of Energy under contract DE-AC05-84OR21400 with Martin Marietta Energy Systems, Inc.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

THE ROLE OF BWR MK I SECONDARY CONTAINMENTS IN SEVERE ACCIDENT MITIGATION

Sherrell R. Greene
Oak Ridge National Laboratory

ABSTRACT

The recent advent of detailed containment analysis codes such as CONTAIN and MELCOR has facilitated the development of the first large-scale, architectural-based BWR secondary containment models. During the past year ORNL has developed detailed, plant-specific models of the Browns Ferry and Peach Bottom secondary containments, and applied these models in a variety of studies designed to evaluate the role and effectiveness of BWR secondary containments in severe accident mitigation. The topology and basis for these models is discussed, together with some of the emerging insights from these studies.

1. INTRODUCTION

Commercial boiling water reactor (BWR) power plants incorporate primary containments of the pressure suppression type, together with a secondary containment which generally consists of a multi-level building that completely encloses the primary containment system. Traditionally, probabilistic risk assessments have employed extremely simplistic models of these secondary containments (Ref. 1). As a participant in the NRC-sponsored BWR Severe Accident Technology Program (BWRSAT), and its predecessor, the Severe Accident Sequence Analysis Program (SASA), Oak Ridge National Laboratory has been involved for several years in the development and application of advanced methods for analysis of severe accidents in BWRs (Ref. 2-24). While attempts have been made to assess the severe accident mitigation effectiveness of BWR secondary containments, it is only with the recent development of advanced containment simulation codes such as CONTAIN (Ref. 25) that reasonable representations of these containments could be formulated. During the past two years ORNL has pioneered the development and application of large-scale BWR secondary containment models for severe accident analysis. This work is yielding new insights into the role and effectiveness of BWR secondary containments in severe accident mitigation. Some of the more important insights from this work are described in this paper.

2. BWR SECONDARY CONTAINMENT DESIGN CONSIDERATIONS

In domestic BWRs of the Mark I and Mark II type, the secondary containment function is provided by a multi-floored reactor building and refueling bay which completely enclose the primary containment. Mark III BWRs employ two entirely different types of secondary containment structures. The standard

Mark III secondary containment design utilizes fuel and auxiliary buildings which completely surround the lower portion of a steel-shelled primary containment, and a cylindrical concrete shield building which surrounds the primary containment above the fuel and auxiliary building elevations. The alternative, or "Grand Gulf" design, consists of an auxiliary building which completely surrounds the lower portion of a concrete primary containment and an enclosure building which completely surrounds the containment above the auxiliary building roofline. Thirty-three of the forty-one BWRs currently operating or under construction in the United States are of the Mark I (24) or Mark II (9) design which employs the reactor building concept. It is this secondary containment concept that has been the focus of ORNL studies to date.

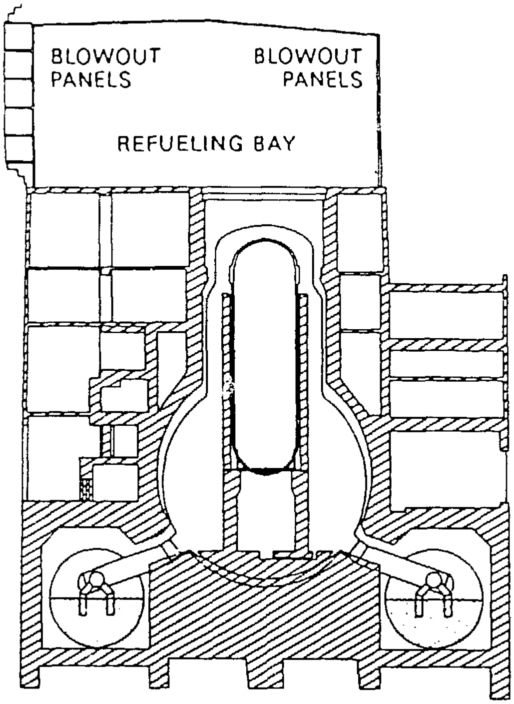
Exhibit 1 is a cross-sectional view of a typical BWR Mark I reactor building (Browns Ferry). The Browns Ferry reactor building is a massive, five-floored building with reinforced external concrete walls. Secondary containment above the top of the reactor building is provided by a refueling bay which is constructed of corrugated sheet metal walls that contain large blowout panels to provide protection from the effects of tornados and steam line breaks. Not shown in Exhibit 1 are details such as stairways, elevator shafts, and reactor building internal blowout panels which provide communication pathways between the various floors of the reactor building and between the reactor building and the turbine building.

Although Exhibit 1 may be viewed as a "typical" BWR MK I reactor building, detailed evaluations reveal a surprising frequency of plant-specific features in domestic plants. This is illustrated by Exhibit 2, which presents a comparison of the Browns Ferry Unit 1 secondary containment to that of Peach Bottom. Significant differences include the use of an enclosed fuel cask hatchway with three embedded blowout panels at Browns Ferry Unit 1, contrasted with a single (21 ft × 17 ft) open port in each reactor building floor at Peach Bottom. Additionally, the inter-floor stairways are open at Browns Ferry but enclosed at Peach Bottom. Finally, the total secondary containment volume of Browns Ferry is much larger than that of Peach Bottom, and the Browns Ferry reactor building has five floors, compared to only four floors at Peach Bottom. Not depicted in Exhibit 2 is an extensive fire protection spray system at Browns Ferry, which has no counterpart in Peach Bottom. Each of the differences outlined here have potentially significant ramifications for the transport of fission products through the secondary containment following a core melt-induced primary containment failure.

3. FACTORS AFFECTING SECONDARY CONTAINMENT PERFORMANCE

Exhibit 3 is a listing of BWR secondary containment design factors which might impact the ability of the secondary containment to reduce and/or retard the release of fission products to the environment during a severe accident. Obviously, the secondary containment must maintain some degree of

REACTOR BUILDING AND REFUELING BAY PROVIDE BWR SECONDARY CONTAINMENT



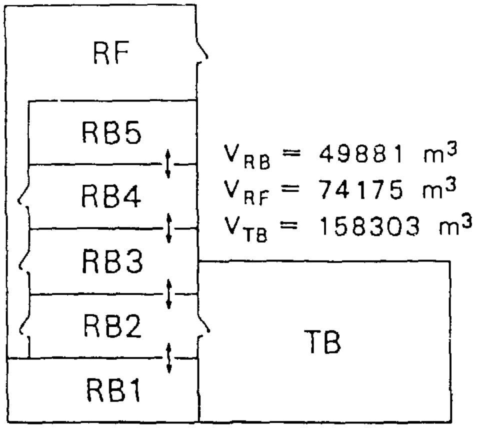
ornl

EXHIBIT 1

ORNL-DWG 86C-4779 ETD

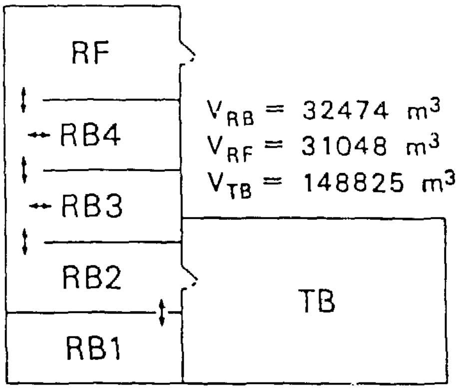
BWR SECONDARY CONTAINMENTS INCORPORATE MANY PLANT-SPECIFIC FEATURES

BROWNS FERRY NO. 1



$V_{RB} = 49881\ m^3$
$V_{RF} = 74175\ m^3$
$V_{TB} = 158303\ m^3$

PEACH BOTTOM



$V_{RB} = 32474\ m^3$
$V_{RF} = 31048\ m^3$
$V_{TB} = 148825\ m^3$

ornl

EXHIBIT 2

EXHIBIT 3

ORNL-WSC-45334 ETD

MANY FACTORS MAY IMPACT BWR SECONDARY CONTAINMENT EFFECTIVENESS

- Primary containment failure location
- Reactor building internal geometry
- Reactor building (RB) integrity
- Refueling bay (RF) integrity
- Turbine building (TB) integrity
- RB, RF, TB interaction
- Hydrogen burn dynamics
- Fire protection system spray dynamics
- Standby gas treatment system operation

integrity in order to perform its function. This integrity might be compromised via hydrogen or carbon monoxide (CO) burn-induced building failures. As seen from Exhibit 2, reactor building internal design features such as the size and placement of elevator shafts and stairwells, and the configuration of walls, partitions, and blowout panels have the potential to impact the flow of gases and aerosols within the reactor building as well as between the reactor building, refueling bay, and the turbine building.

The location of the primary containment failure and the mode of entry of primary containment blowdown to the secondary containment can impact secondary containment effectiveness by reducing or increasing the path length and volume through which the blowdown must flow prior to reaching the environment. This can be seen more clearly in Exhibit 4, which summarizes the six most probable primary containment failure modes and identifies the location at which the bulk of the blowdown would enter the secondary containment. Although failure of the drywell knuckle or the drywell head is currently believed to be less probable than failure of the vent bellows or the drywell liner, such failures could lead to circumstances in which the majority of primary containment blowdown enters the secondary containment at higher elevations in the reactor building, or flows directly into the refueling bay.

Another factor which has the capacity to impact secondary containment performance is hydrogen (or carbon monoxide) deflagration dynamics. As will be demonstrated in the following sections, factors such as burn initiation criteria, flame speed, total burn time, coincidence of multiple burns, and the location of the burn can have a great impact on burn-induced peak building pressures and the flow of fission products to the environment. All of these factors are actually indirect functions of internal building geometry and the primary containment failure location.

One plant-specific BWR secondary containment feature which was first identified by SASA studies (Refs. 8, 13) as having a potentially large impact on secondary containment fission product retention is the plant fire protection system. The Browns Ferry fire protection system utilizes fused-link sprinklers which auto-actuate (no external power required) when local temperatures exceed 165°F, providing a substantial spray which has the potential to remove aerosols and iodine from the internal reactor building atmosphere prior to their escape to the environment. Unfortunately, the design of the fire protection system is highly plant-specific. Browns Ferry has an extensive system, while Peach Bottom has only a skeletal fire protection spray system which employs no overhead sprays.

Finally, the plant standby gas treatment system (if operating) has the potential to maintain negative secondary containment pressures during substantial portions of the post-accident period (Refs. 8, 13). The fundamental design and capacity of this system is highly plant-specific, however, and its effectiveness during a severe accident will depend on factors such as: system power source, volumetric capacity, recirculation ratio, and the

EXHIBIT 4

ORNL-WSC-48888 ETD

CANDIDATE PRIMARY CONTAINMENT FAILURE MECHANISMS

COMPONENT	CRITERION	ADJACENT TO
DW penetrations	?	RB Floors 2, 3, 4
DW head	178 psig at $T_{bolt} = 700^{\circ}F$ 9 psig at $T_{bolt} = 1200^{\circ}F$	Refueling bay
DW knuckle	>117 psig	RB Floor 2 or 3
Vent/PSP bellows	>94 psig	Torus room
PSP	160 ± 25 psig	Torus room
DW liner	7/8 in.-1 1/8 in. steel ablation	Torus room

ornl

ability of the system HEPA and charcoal filters to withstand high aerosol loadings.

4. SECONDARY CONTAINMENT MODELING - THE HISTORICAL APPROACH

The traditional approach employed in BWR secondary containment modeling is to treat the reactor building and refueling bay as one (or possibly two) compartment. This approach is shown pictorially in Exhibit 5. The appeal of this modeling approach includes: (1) it results in fast-running models, (2) no analysis of primary containment failure location is necessary, (3) all internal building structures may be collapsed into a few structures for heat-transfer modeling purposes and (4) it simplifies the analysis of hydrogen and CO deflagrations since inter-compartmental flame propagation questions need not be addressed. Unfortunately, as will be demonstrated in later sections, this approach often results in gross distortion of predicted secondary containment performance.

5. SECONDARY CONTAINMENT MODELING - THE ORNL APPROACH

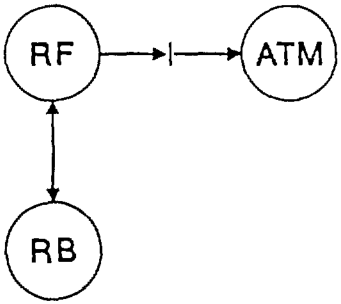
Approximately one and one-half years ago ORNL began an effort to develop large-scale, architectural-based BWR secondary containment models for use in the ORNL SASA program. The effort was originally oriented exclusively toward the development of a model for Browns Ferry Unit 1. Recently, however, a major effort was undertaken to develop a large-scale Peach Bottom secondary containment model.

The ORNL modeling approach begins with an onsite review of secondary containment construction, followed by a detailed review of the pertinent plant design drawings. Obviously, close cooperation between the analysts and the utility is a crucial element in the overall modeling effort, and ORNL has enjoyed close cooperation with both the Tennessee Valley Authority (Browns Ferry) and the Philadelphia Electric Company (Peach Bottom) throughout the course of the model development. Factors such as the number of floors in the reactor building, the size and location of blowout panels, the number and location of open doors and stairwells between floors, the design of the plant HVAC and fire protection systems, and many other such issues must be addressed during the formulation of the model. The end result of this modeling approach is a model topology which closely approximates the actual secondary containment topology and captures all significant plant-specific design features.

Exhibit 6 is a schematic representation of the current ORNL Browns Ferry Unit 1 and Peach Bottom Secondary Containment Models. Although several different model topologies have been investigated, it is believed that the current models achieve the best compromise between model detail and code run time. The Browns Ferry Secondary Containment Model is an 8 cell model (1 cell for each of the 5 reactor building floors, 1 cell for the enclosed reactor building fuel cask hatchway, 1 cell for the refueling bay, and 1 cell for the turbine building). Each of the three inter-zonal blowout

ORNL-DWG 86C-6164 ETD

TRADITIONAL BWR SECONDARY CONTAINMENT MODEL



ornl

EXHIBIT 5

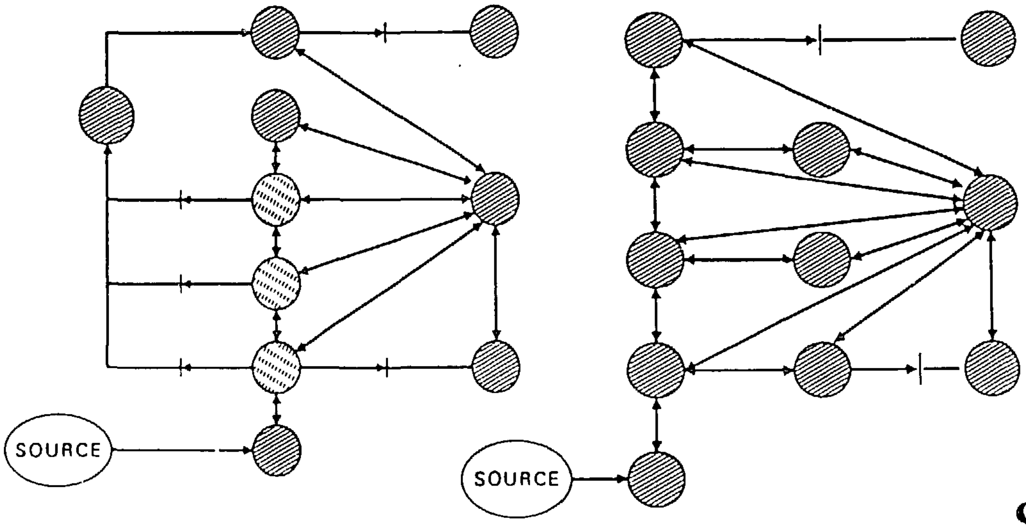
EXHIBIT 6

ORNL-DWG 86C-4816 ETD

ORNL STUDIES INDICATE PLANT-SPECIFIC SECONDARY CONTAINMENT MODELS ARE REQUIRED

ORNL BROWNS FERRY-1
CONTAIN MODEL

ORNL PEACH BOTTOM
CONTAIN MODEL



panels, the reactor building-to-turbine building steam vault blowout panels, and the refueling bay blowout panels are explicitly represented, as is the fire protection spray system. The Peach Bottom Secondary Containment Model is a 9 cell model (1 cell for reactor building basement, 2 cells for the second, third and fourth floors of the reactor building, 1 cell for the refueling bay, and 1 cell for the turbine building). The refueling bay and reactor building-to-turbine building blowout panels are explicitly represented. Both models are used in conjunction with either 2 environmental cells (CONTAIN - Exhibit 6), or 1 environmental cell (MELCOR). Normal secondary containment infiltration and exfiltration paths are incorporated in both models. Both of these models have been provided to the Sandia National Laboratories CONTAIN and MELCOR code development programs and are being exercised there in a variety of code testing and applications efforts. ORNL has exercised these models (CONTAIN format) in a variety of studies designed to identify the impact of historical (simplified) secondary containment modeling approaches, and to investigate the impact of various plant design features and accident events on BWR secondary containment severe accident mitigation effectiveness. Some insights from those studies are described below.

6. THE SINGLE CELL MODEL - AN INADEQUATE APPROACH

As previously stated, the historical approach to secondary containment modeling involved the use of single cell (reactor building and refueling bay combined) or two cell (1 cell reactor building and 1 cell refueling bay) models. ORNL studies indicate that the historical approach tends to produce a distorted view of the severe accident mitigation effectiveness of BWR secondary containments. This behavior is illustrated by Exhibit 7, which summarizes the results of a Peach Bottom station blackout calculation in which drywell blowdown data generated by ORNL MARCON 2.1B was employed to drive two CONTAIN 1.04 models of the Peach Bottom secondary containment.

The first model employs a single cell representation for the entire reactor building. The second model employs a four cell representation of the reactor building in which each floor of the building was represented by a single cell. The drywell blowdown included approximately 168 kg of CO_2 (which was produced by the in-vessel reaction of steam and boron carbide powder from the control blades), and a 20-kg aerosol tracer which was injected at the rate of 0.1 kg/s for the first 200 seconds of the analysis. Although unimportant in itself, the CO_2 serves as a reasonable noble gas surrogate, allowing one to assess the behavior of noble gases under similar conditions.

Exhibit 7 summarizes the results of the two calculations at one hour after drywell failure. As can be seen from Exhibit 7, single cell models tend to under-predict the number of hydrogen burns (because enough hydrogen must enter the secondary containment to bring the entire volume to combustible conditions) and over-predict peak building pressures relative to multi-cell treatments. Experience has shown no distinct trend in the relative

EXHIBIT 7

ORNL-WSC-48864 ETD

TRADITIONAL SINGLE CELL REACTOR BUILDING MODELS PROVIDE DISTORTED VIEW OF SECONDARY CONTAINMENT PERFORMANCE

	1 CELL PB	4 CELL PB
No. RB burns	2	8
Peak RB pres (psia)	15.7	15.4
Peak RB temp (°F)	1695	1112
No. RF burns	2	1
Peak RF pres (psia)	15.5	15.5
Peak RF temp (°F)	672	1187
% CO_2 release	55	35
% Aerosol release	57	31

predictions of peak reactor building atmospheric temperatures - much depends on the interaction of hydrogen burn dynamics and reactor building geometry in the multi-cell model. The most important difference between the predictions of the two models is in the area of fission product release (gas and aerosol). The results of numerous studies have shown that single cell representations consistently over-predict (by factors of 2 to 4) the magnitude of aerosol release to the environment.

A second area in which single cell secondary containment models yield distorted results is in the area of hydrogen burn-induced reactor building pressures. Extensive studies at ORNL have indicated that single cell models predict a simple relationship between assumed flame speed and peak induced pressure. This behavior is demonstrated in Exhibit 8 which indicates that as the assumed flame speed is increased from 1.8 m/s to 6.5 m/s, the predicted peak building pressure increases from 15.7 psia to 20.3 psia. Not shown in Exhibit 8 are the results of several intermediate runs (flame speeds between 1.8 and 6.5 m/s) which further demonstrated the consistent relationship between flame speed and peak induced pressure. The results presented in Exhibit 8 (and Exhibit 9 in the following section) are based on assumed deflagration initiation limits of 8 mole % minimum hydrogen, 5 mole % minimum oxygen, and 55 mole % maximum steam concentrations in the local cell. As will be demonstrated below, multi-cell treatments reveal a much more complex relationship between burn dynamics and building geometry. Since reactor buildings are typically designed for differential pressures of 2 psig, such distortions have obvious implications for assessment of building survivability.

In summary, the results of numerous comparisons between single and multi-cell secondary containment models reveal that single cell reactor building models generally yield distorted results, and their use should be avoided where possible.

7. MULTI-CELL MODELS - THE KEY TO ASSESSMENT OF SECONDARY CONTAINMENT SURVIVABILITY

As previously stated, single cell reactor building models generally predict a simple relationship between assumed flame front propagation speed and peak induced reactor building pressure. ORNL's studies with multi-cell models have revealed, however, that a much more complex relationship exists between hydrogen burn dynamics and reactor building geometry. Specifically, our studies indicate that the peak burn-induced reactor building pressure is a complex (and not completely understood) function of (a) burn initiation criteria, (b) the location at which the burn occurs, (c) the assumed (or calculated) flame speed, (d) burn coincidence (the occurence of simultaneous or over-lapping burns in different compartments), and (e) building geometric factors such as inter-floor flow path charactèristics. Exhibit 9 compares the results of a series of Browns Ferry calculations (8 cell model) in which the flame speed was varied from 2.7 m/s to 6.0 m/s. It is readily apparent that the simple, monotonic relationship between flame speed and peak

EXHIBIT 8

ORNL-WSC-48863 ETD

SINGLE CELL REACTOR BUILDING MODELS PREDICT SIMPLE RELATIONSHIP BETWEEN FLAME SPEED AND BURN-INDUCED PRESSURE

	1 CELL PB $V_F = 1.8$	1 CELL PB $V_F = 6.5$
No. RB burns	2	3
Peak RB pres (psia)	15.7	20.3
Peak RB temp (°F)	1695	1720
No. RF burns	2	2
Peak RF pres (psia)	15.5	19.4
Peak RF temp (°F)	672	714

ornl

EXHIBIT 9

ORNL-WSC-48862 ETD

MULTI-CELL REACTOR BUILDING MODELS REVEAL COMPLEX BURN-INDUCED PRESSURE SENSITIVITY

	6 CELL BF $V_F = 2.7$	6 CELL BF $V_F = 3.0$	6 CELL BF $V_F = 6.0$
No. RB burns	21	18	18
Peak RB pres (psia)	18.42	15.89	17.31
Peak RB temp (°F)	1749	1009	1157
No. RF burns	0	0	0
Peak RF pres (psia)	15.23	15.05	15.37
Peak RF temp (°F)	316	233	242

ornl

building pressure predicted by single cell models is not predicted by multi-cell models. It is therefore clear that realistic assessments of secondary containment survivability and integrity cannot be conducted with traditional one or two cell secondary containment models.

8. SECONDARY CONTAINMENT DECONTAMINATION FACTORS

Historically, probabilistic risk assessments have given little credit for fission product retention in the secondary containment. Recent studies at ORNL in support of on-going NUREG 1150 assessments indicate that secondary containments can function as effective fission product traps - even in cases where the standby gas treatment system is not functioning and all secondary containment blowout panels are open. This effect is illustrated by Exhibits 10 and 11, which present the results of an SNL CONTAIN 1.06 calculation in which the current ORNL 11 cell (9 cells + 2 atmospheric cells) Peach Bottom model was exercised in conjunction with Peach Bottom station blackout drywell blowdown data (steam, gases, aerosols) provided by Battelle Columbus Laboratories.

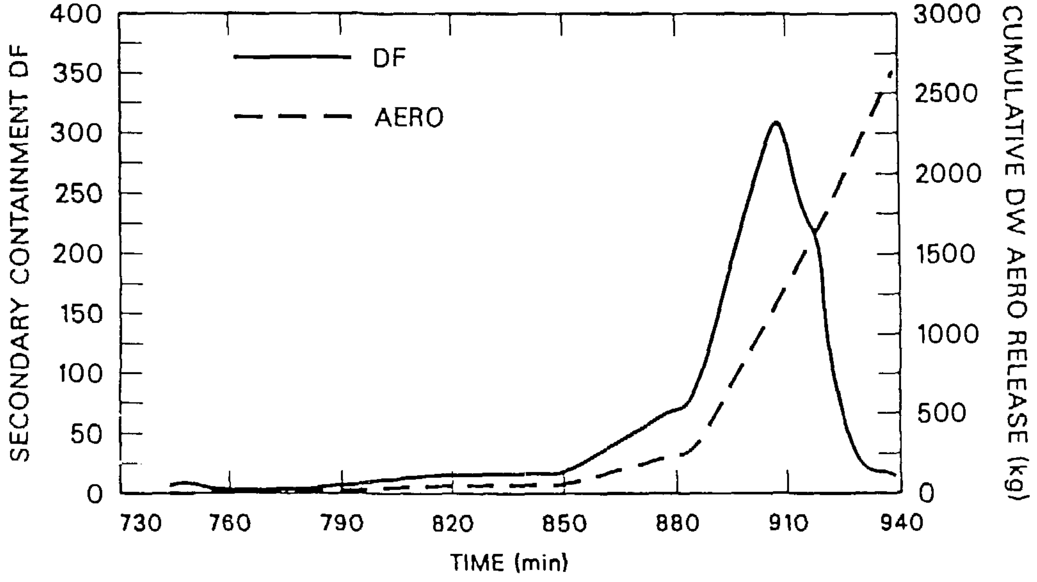
Exhibit 10 displays the cumulative drywell aerosol release and the instantaneous secondary containment aerosol decontamination factor (DF) as a function of time for the first 200 minutes after drywell failure (which occurs in this calculation at 738 minutes into the accident). Decontamination factors are defined here as the integrated drywell aerosol leakage from the moment of drywell failure, divided by the total mass of aerosols that has escaped to the environment over the same interval. Decontamination factors as high as 300 are predicted, with a final predicted value (at 940 minutes) of 14. It is clear that the analyst must exercise caution in quoting DFs, since grossly misleading results can be obtained if the DF is calculated just after a large "puff" of aerosols have exited the drywell. A true secondary containment DF can only be defined at the point in time at which no active aerosol sources remain (i.e., the core/concrete reaction has ceased) and no aerosols remain suspended in any primary or secondary containment compartment. Unfortunately, such calculations require several hours of transient simulation and have not yet been completed.

The aerosol trapping function of the secondary containment is further demonstrated by Exhibit 11, which displays plots of the time-dependent cumulative drywell and secondary containment aerosol release. While significant quantities of aerosols have escaped the drywell (primary containment) by the 820 minute point, significant secondary containment aerosol releases do not occur until approximately 925 minutes into the accident - a 105 minute delay. The secondary containment not only reduces the total fission product release, but also delays the initial release to the environment by more than one and one-half hours.

EXHIBIT 10

ORNL-DWG 86C-4777 ETD

SECONDARY CONTAINMENT DECONTAMINATION FACTORS ARE TIME DEPENDENT

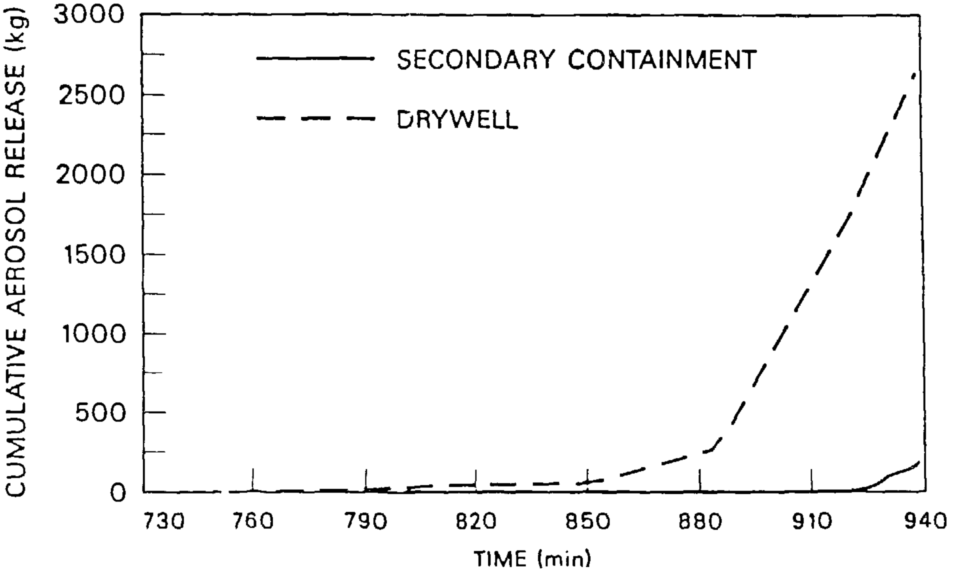


ornl

EXHIBIT 11

ORNL-DWG 86C-4778 ETD

SECONDARY CONTAINMENT CAN SIGNIFICANTLY DELAY RELEASE OF FISSION PRODUCTS TO ENVIRONMENT



ornl

9. BWR SECONDARY CONTAINMENT FIRE PROTECTION SYSTEMS AND THEIR ROLE IN SEVERE ACCIDENT MITIGATION

As mentioned in Section 3, some BWR secondary containments incorporate comprehensive fire protection systems which utilize fused-link water sprinklers for fire suppression. Exhibit 12 is a simplified schematic representation of the Browns Ferry Fire Protection System. The Browns Ferry plant utilizes a system that employs fused-link sprinklers, which cover the first three floors of the reactor building (soon to be upgraded to coverage of all floors). The system consists of two 10000 gallon raw service water (RSW) storage tanks (located atop the reactor building), four RSW pumps which maintain the tank inventory during normal operation, four fire system pumps (one of which is diesel-driven), and the sprinkler system. The RSW storage tanks provide a 20000 gallon, gravity-fed sprinkler supply reservoir, and no power is required for actuation of the fused-link sprinklers. Additionally, and very importantly, the one diesel-driven pump provides a highly reliable supply of water to sprinklers located in the first two levels of the reactor building.

ORNL first identified the secondary containment fire protection system as a potential severe accident mitigation system during a previous SASA study (Refs. 8, 13). Although not designed as a severe accident mitigation system, the fused-link sprinklers would actuate following primary containment failure due to localized reactor building temperature increases. The current ORNL Browns Ferry Secondary Containment Model incorporates a detailed simulation of the sprinkler system. This model has been exercised in a variety of studies at ORNL to assess the impact of sprinkler system operation on reactor building pressurization and fission product transport within the reactor building.

Exhibit 13 displays the results of a study in which the current ORNL Browns Ferry Secondary Containment Model (CONTAIN 1.04 version) was driven by drywell blowdown data from a MARCON 2.1B station blackout calculation. An artificial "puff" aerosol source of 20 kg (0.1 kg/s for 200 s) was added to the MARCON blowdown data during the first 200 seconds following drywell failure, and all blowdown was assumed to enter the reactor building in the torus room (RB1). During single unit station blackout conditions, the spray system would be fed only by the roof-top RSW storage tanks and the single diesel-driven fire pump. Under these circumstances, water is available to the third floor (RB3) sprinklers only until the RSW tanks are depleted (about 12 minutes in this example). The second floor sprinklers are continuously fed, however, by the single diesel-driven pump. Exhibit 13 depicts the cell by cell distribution of aerosols at one hour after drywell failure for the case in which spray system actuation was inhibited, and the case in which the spray system was allowed to function in its normal manner.

The ordinate in Exhibit 13 is the fraction of total aerosol inventory (i.e., 20 kg) present in each cell of the secondary containment and in the outside atmosphere. The impact of the spray system can clearly be seen by noting

EXHIBIT 12

ORNL-DWG 86C-4830 ETD

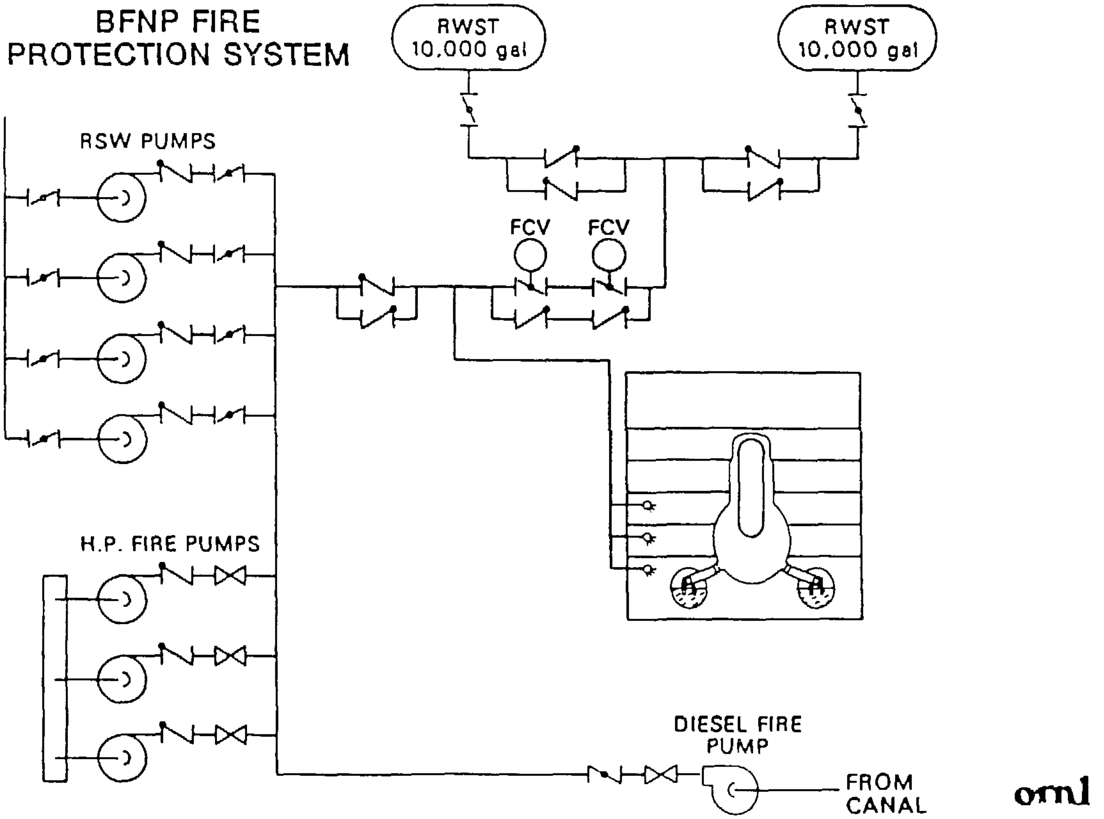
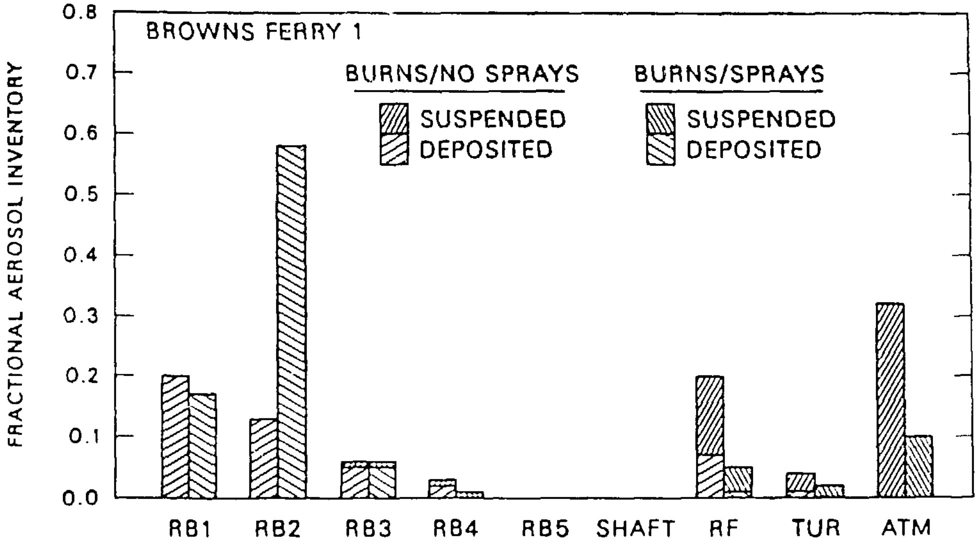


EXHIBIT 13

ORNL-DWG 86C-4817 ETD

FIRE PROTECTION SYSTEM SPRAYS SIGNIFICANTLY INCREASE SECONDARY CONTAINMENT DFs



ornl

the difference in the aerosol inventory in cell 2 (RB2) for the case with and without sprays. The overall impact of spray operation in this case is to reduce the fractional aerosol release from 33 percent to 11 percent (ATM cell). It should be noted that this reduction is achieved with only one of the four fire system pumps operating. It is apparent, therefore, that secondary containment fire protection systems can significantly enhance the secondary containment aerosol decontamination factor. Installation and enhancement of these systems for severe accident mitigation should be investigated.

10. STANDBY GAS TREATMENT SYSTEM EFFECTIVENESS UNDER SEVERE ACCIDENT CONDITIONS

The purpose of the plant standby gas treatment system (SGTS) is to process exhaust air from the secondary containment boundary under design basis accident conditions. As seen in Exhibit 14, the SGTS (Browns Ferry design) draws suction on the reactor building and refueling bay, passes the exhaust air through charcoal and high efficiency particulate absolute (HEPA) filters, and exhausts the air through the plant stack. The design and capacity of the SGTS is highly plant specific (8000-25000 scfm). Many plants utilize systems which recirculate a portion of the exhaust air back to the secondary containment after filtering. The SGTS would normally be available during any accident in which offsite power or station diesels are available.

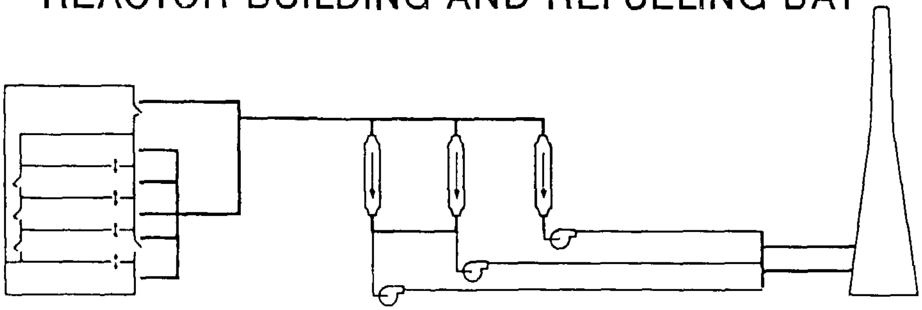
The plot in Exhibit 14 depicts the time-dependent reactor building pressure through the early stage of a severe accident (loss of decay heat removal) and demonstrates that the SGTS is capable of maintaining negative pressures within the secondary containment under severe accident conditions. The continued operation of this system during a severe accident could, therefore, significantly reduce total fission product release to the environment, as well as delay the timing of the initial release. Although not depicted in Exhibit 14, clogging of the HEPA filters due to aerosol deposition would result in gradual reduction in SGTS flow and an increase in secondary containment pressure. ORNL has sponsored tests at New Mexico State University to investigate the ability of the HEPA filters to function in circumstances involving high levels of deposited aerosols. These tests have demonstrated that the HEPA filters would remain intact throughout the course of the accident, although the flow through the SGTS would asymptotically approach zero as a result of continually increasing aerosol loading of the HEPA filter surfaces.

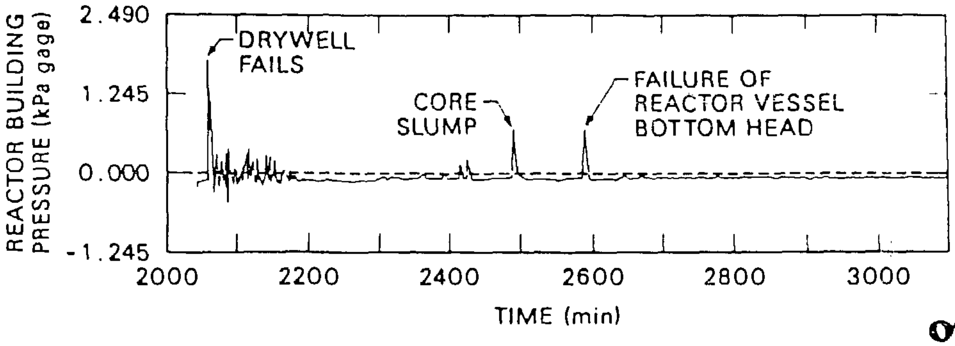
The results depicted in Exhibit 14 were produced with an early version of a dedicated secondary containment analysis code developed at ORNL. Efforts are underway to integrate a detailed plant-specific simulation of the SGTS in the current ORNL Peach Bottom and Browns Ferry Secondary Containment Models.

EXHIBIT 14

ORNL-DWG 86C-4831 ETD

SGTS CAN SIGNIFICANTLY IMPROVE SECONDARY CONTAINMENT DF BY MAINTAINING VACUUM IN REACTOR BUILDING AND REFUELING BAY





ornl

11. CONCLUSIONS

The recent advent of detailed containment analysis codes such as CONTAIN and MELCOR has facilitated the development of the first large-scale, architectural-based BWR secondary containment models. Ongoing research with these new tools is providing analysts with new insights into the role of BWR secondary containments in severe accident mitigation. Close utility/analyst cooperation is essential to the success of these modeling efforts.

Preliminary studies with these new models indicate that traditional one or two cell secondary containment models yield distorted and overly conservative pictures of the effectiveness of BWR secondary containments. ORNL studies indicat that plant-specific features must be incorporated in secondary containment models and that availability of such large-scale models is a prerequisite to understanding secondary containment behavior and importance. The results of preliminary analyses indicate that secondary containment aerosol decontamination factors of 10 to 40 appear credible, but that extreme caution must be exercised in calculating decontamination factors since they tend to be highly time dependent. Secondary containment survival of hydrogen burn-induced pressure spikes seems probable, but not certain. Many uncertainties remain regarding minimal hydrogen concentration burn thresholds and flame propagation speeds. Additional research is necessary to further characterize these burn parameters.

Reactor building fire protection system sprays significantly enhance secondary containment aerosol retention, and their installation and enhancement for such purposes should be further explored. Although not designed for severe accident applications, the plant standby gas treatment system can significantly reduce and retard the release of fission products from the secondary containment.

Finally, although great advances have been made in our ability to perform realistic secondary containment analysis, much remains to be done. Significant modeling improvements in the areas of intra-building plume rise dynamics and fission product chemistry are badly needed. ORNL is currently performing the first BWRSAT severe accident analysis in which the complete impact of recent modeling developments will be demonstrated.

REFERENCES

1. Reactor Safety Study, WASH-1400, Appendix VII, U.S. Nuclear Regulatory Commission, October 1975.

2. M. H. Fontana, "Severe Accident Sequence Assessment and Time-Line Charts for Boiling Water Reactors - Program Overview," NUREG/CR-2008, May 1981.

3. D. H. Cook et al., "Station Blackout at Browns Ferry Unit One - Accident Sequence Analysis," Vol. 1, NUREG/CR-2182, November 1981.

4. R. P. Wichner et al., "Station Blackout at Browns Ferry Unit One - Iodine and Noble Gas Distribution and Release," Vol. 2, NUREG/CR-2182, August 1982.

5. W. A. Condon et al., "SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis," Vol. 1, NUREG/CR-2672, October 1982.

6. D. D. Yue et al., "BWR4/MARK I Accident Sequences Assessment," NUREG/CR-2825, October 1982.

7. D. H. Cook et al., "Loss of DHR Sequences at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-2973, May 1983.

8. R. P. Wichner et al., "SBLOCA Outside Containment at Browns Ferry Unit One - Iodine, Cesium, and Noble Gas Distribution and Release," Vol. 2, NUREG/CR-2672, September 1983.

9. R. M. Harrington and L. J. Ott, "The Effect of Small-Capacity, High-Pressure Injection Systems on TQUV Sequences at Browns Ferry Unit One," NUREG/CR-3179, September 1983.

10. M. Z. Podowski et al., "Mechanistic Core-Wide Meltdown and Relocation Modeling for BWR Applications," NUREG/CR-3525, December 1983.

11. S. R. Greene, "Realistic Simulation of Severe Accidents in BWRs - Computer Modeling Requirements," NUREG/CR-2940, April 1984.

12. R. M. Harrington and S. A. Hodge, "ATWS at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-3470, July 1984.

13. R. P. Wichner et al., "Noble Gas, Iodine, and Cesium Transport in a Postulated Loss of Decay Heat Removal Accident at Browns Ferry," NUREG/CR-3617, August 1984.

14. D. H. Cook, "Pressure Suppression Pool Thermal Mixing," NUREG/CR-3471, September 1984.

15. R. M. Harrington and L. C. Fuller, "BWR-LTAS: A Boiling Water Reactor Long-Term Accident Simulation Code," NUREG/CR-3764, February 1985.

16. B. R. Koh et al., "The Modeling of BWR Core Meltdown Accidents - For Application in the MELRPI.MOD2 Computer Code," NUREG/CR-3889, April 1985.

17. R. M. Harrington, "Modifications to BWR-LTAS Since December 1984," letter report to Dr. Thomas J. Walker, Division of Accident Evaluation, RES, U.S. NRC, June 21, 1985.

18. M. F. Albert, "The Absorption of Gaseous Iodine by Water Droplets," NUREG/CR-4081, July 1985.

19. R. M. Harrington, "Evaluation of Operator Action Strategies for Mitigation of MSIV Closure Initiated ATWS," letter report to Dr. Thomas J. Walker, Division of Accident Evaluation, RES, USNRC, November 11, 1985.

20. R. M. Harrington, "The Effect of Reactor Vessel Pressure and Water Level on Equilibrium BWR Core Thermal Power During MSIV-Closure-Initiated ATWS," letter report to Dr. Thomas J. Walker, Division of Accident Evaluation, RES, USNRC, January 10, 1986.

21. R. M. Harrington and S. A. Hodge, "Containment Venting as a Severe Accident Mitigation Technique for BWR Plants with Mark I Containment," letter report to Dr. Thomas J. Walker, Division of Accident Evaluation, RES, USNRC, June 26, 1986.

22. R. M. Harrington and S. A. Hodge, "Loss of Control Air at Browns Ferry Unit One - Accident Sequence Analysis," NUREG/CR-4413, April 1986.

23. C. R. Hyman and C. F. Weber, "Effects of Lateral Separation of Oxidic and Metallic Core Debris on the BWR MK I Containment Drywell Floor," NUREG/CR-4610, pending publication.

24. G. E. Mueller and A. Sozer, "Thermal/Hydraulic and Characteristic Models for Packed Debris Beds," NUREG/CR-4689, pending publication.

25. K. D. Bergeron et al., "User's Manual for Containment 1.0, A Computer Code for Severe Nuclear Reactor Accident Containment Analysis," NUREG/CR-4085, Sandia National Laboratories, May 1985.