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State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

Appendix A
Peach Bottom Integrated Analysis

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ABSTRACT

New analyses of severe accident progression and consequences were performed to assess the results of past analyses and help guide public policy. This study has focused on providing a realistic evaluation of accident progression, source term, and offsite consequences for the Peach Bottom Nuclear Power Station. By using the most current emergency preparedness (EP), plant capabilities, best-available modeling ~~and uncertainties~~, and recent security assessments, these analyses are more detailed, integrated, and realistic than past analyses. These analyses also consider all mitigative measures, contributing to a more realistic analysis.

really?

Paperwork Reduction Act Statement

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ACRONYMS

ATWS	Anticipated Transient Without Scram
CD	Core Damage
CDF	Core Damage Frequency
CRDHS	Control Rod Drive Hydraulic System
CRGT	Control Rod Guide Tube
CST	Condensate Storage Tank
EP	Emergency Preparedness
EPZ	Emergency Planning Zone
ETE	Evacuation Time Estimate
GE	General Emergency
HPCI	High Pressure Coolant Injection
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Event
LPCI	Low Pressure Coolant Injection
LPI	Low Pressure Injection
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
ORO	Offsite Response Organizations
PeCo	Philadelphia Electric Company
RAMCAP	Risk Analysis and Management for Critical Asset Protection
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SAE	Site Area Emergency
SECPOP	SECtor POPulation and Economic Estimator (Version 3.12.6)
SOARCA	State-of-the-Art Reactor Consequence Analysis Project
SPAR	Simplified Plant Analysis Risk
SRV	Safety/Relief Valve
TAF	Top of Active Fuel
UE	Unusual Event
VF	Vessel Failure

Many! many more to add! see eg pg 14

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1.0 INTRODUCTION

unrealistic

The evaluation of accident phenomena and the offsite consequences of severe reactor accidents has been the subject of considerable research by the U.S. Nuclear Regulatory Commission (NRC) over the last several decades. As a consequence of this research focus, analyses of severe accidents at nuclear power reactors is more detailed, integrated and realistic than at any time in the past. A desire to leverage this capability to address ~~excessively conservative~~ aspects of previous reactor accident analysis efforts was a major motivating factor in the genesis of the State-of-the-Art Reactor Consequence Analysis (SOARCA) project. By applying modern analysis tools and techniques, the SOARCA project seeks to provide a body of knowledge that will support an informed public understanding of the likely outcomes of severe nuclear reactor accidents.

There were also NON conservation aspects and those were addressed also!

The primary objective of the SOARCA project is to provide a best estimate evaluation of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective the SOARCA project ~~will~~ utilized integrated modeling of accident progression and off site consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective wisdom of the severe accident analysis community. This report documents the analysis of the Peach Bottom Atomic Power Station ~~to~~ for the dominant but extremely low likelihood accidents that could progress to radiological release.

1.1 Site Characteristics

This report describes results of the analysis of severe accident progression specific to the Peach Bottom Atomic Power Station -- a nuclear plant of the BWR/4 Mark I design located in southeast Pennsylvania. This station is located 17.9 miles south of Lancaster, Pennsylvania and 38 miles north-northeast of Baltimore, Maryland as shown in Figure 1. The site occupies 620 acres in the York and Lancaster counties of southeastern Pennsylvania and is 2.5 miles north of the Maryland-Pennsylvania state line. The plant is situated on the western shore of Conowingo Pond which is formed by the backwater of Conowingo dam, located 9 miles downstream on the Susquehanna River. The Holtwood dam is located 6 miles upstream.

The minimum exclusion distance from the center of the reactor to the site (for either Unit 2 or Unit 3) is about 2,700 ft. The minimum distance from the center of a reactor to the site boundary in a downstream direction is about 3,300 ft (from Unit 2) and in an inland direction about 3,100 ft (from Unit 2). The minimum distance across the pond from either the Unit 2 or Unit 3 reactor to the far shore of the pond (to the northeast) is 7,600 ft. The minimum distance from the stack to the site boundary is 2,350 ft. The "exclusion area," as defined in Section 100.3 of 10 CFR Part 100, includes the area within the minimum exclusion distance from the center of Unit 2 and Unit 3 reactors.

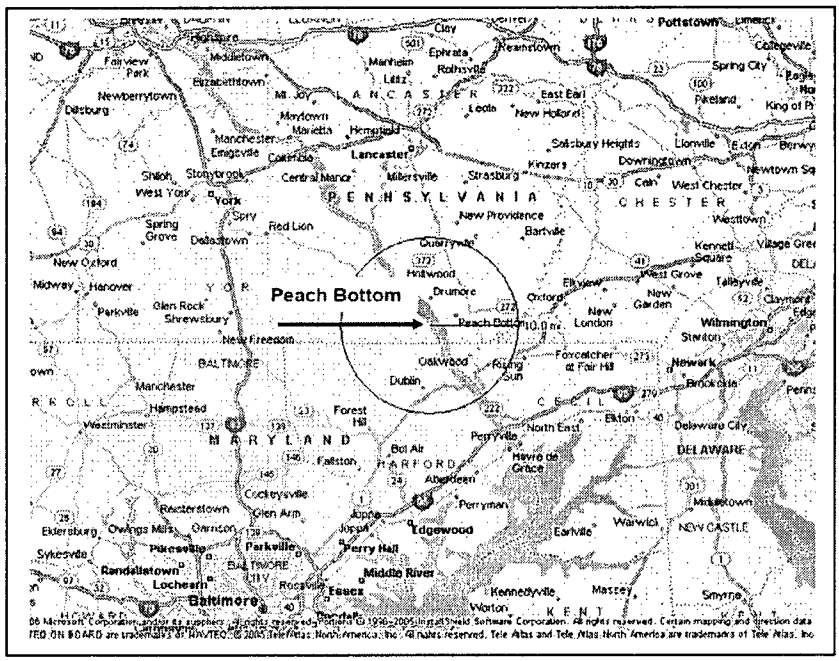

Figure 1 Site Location

Within a 1 mile radius of the plant and on both sides of Conowingo Pond, steep sloping hills rise directly up to about 300 ft above plant grade with outcroppings of rock apparent at many locations. Because of the relatively rough terrain, much of this area is desolate with wooded areas scattered throughout, although the more gentle sloping areas are cleared and cultivated. The site is located in a well-defined river valley, which in turn lies in rolling but not exceptionally rugged country. Maximum elevations in the immediate vicinity of the facility seldom exceed 300 ft above river level, although there are several plateau sections and hilltops reaching 500 to 800 ft above the river within 10 mi to the southwest, west, northwest, and north of the site [1] (see Figure 2).

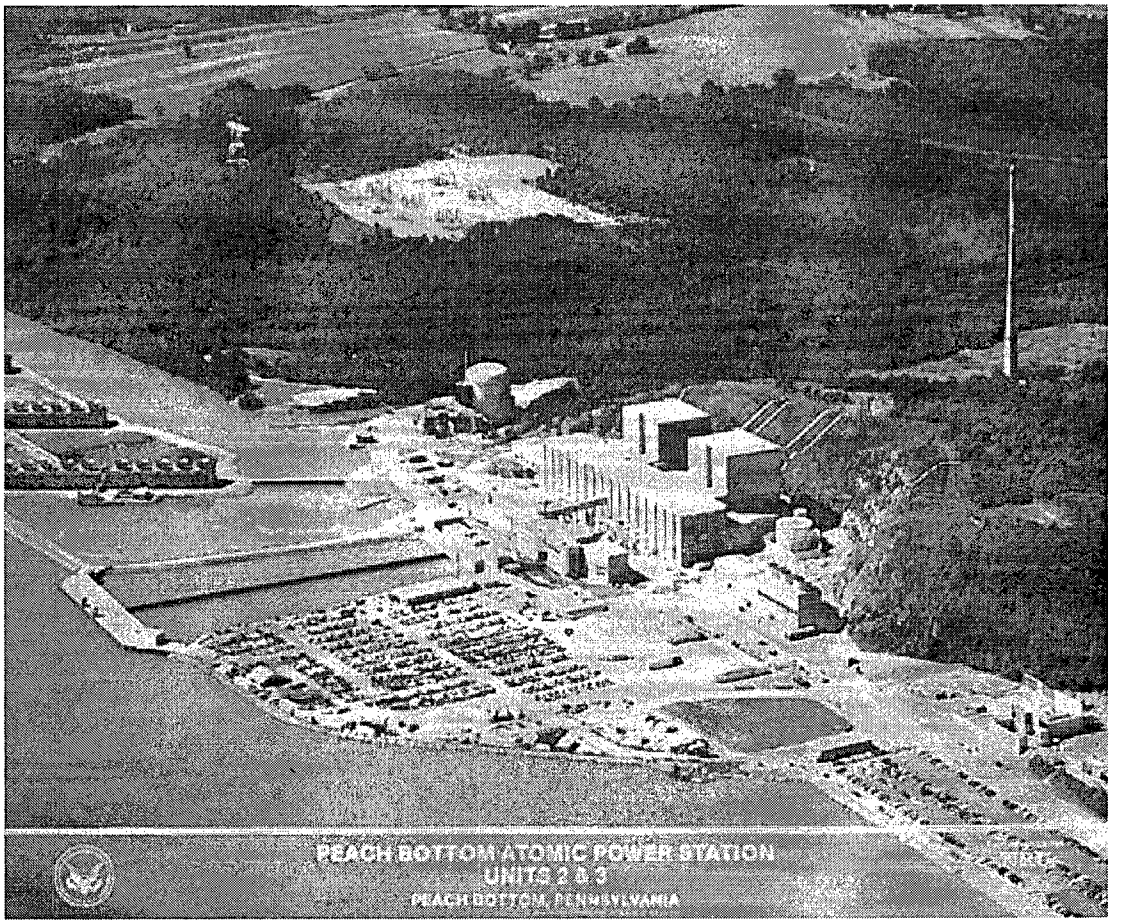

Figure 2 Site Photograph.

1.2 Outline of Report

Section 2 of this report briefly summarizes the method used to select the specific accident scenarios subjected to detailed computational analysis. Additional details of this method can be found in Volume 1 of this series of reports. Section 3 then describes the results of the accident scenario selection process when it was applied to Peach Bottom. Section 4 describes the key features of the MELCOR model of the Peach Bottom Atomic Power Station. Section 5 describes the results of MELCOR calculations of severe accident progression and radionuclide release to the environment for each accident scenario. Section 6 describes the way in which plant-specific emergency response actions were represented in the calculations of offsite consequences, and Section 7 describes the calculations of offsite consequences for each accident scenario. References cited in this report are listed in Section 8.

a CD is included with this report giving all the input values used for the analyses.

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2.0 ACCIDENT SCENARIO DEVELOPMENT

Specific radiological release scenarios were selected for detailed analysis of severe accident progression, radionuclide release to the environment and estimation of their impacts on public health and safety. Selection of a scenario begins by identifying sequences of events that lead to core damage, based on plant-specific models of plant systems developed for the purpose of Probabilistic Risk Assessment (PRA). In this context, an accident sequence begins with a postulated initiating event (for example, a major disruption in offsite power, a leak or rupture of reactor coolant system piping, or an earthquake) that perturbs the steady state operation of the nuclear power plant is such as way as to induce actuation of the plant's control and safety systems. If a sufficient number of control and/or safety systems fail to actuate or operate as needed to respond to the initiating event, damage to the reactor fuel and the release of radioactive fission products would result.

In the SOARCA Program, accident sequences that have an estimated frequency greater than 1×10^{-6} per year of reactor operation[1] are retained as candidate sequences for further evaluation. Section 2.1 summarizes the methods used to identify these sequences and the screening process for retaining candidate sequences. Additional information can also be found in Volume 1 of this report.

Once candidate accident sequences are identified, realistic opportunities for plant personnel to respond to the observed failures of control and safety systems are evaluated. Possibilities for mitigation included the licensee's emergency operating procedures (EOPs), severe accident management guidelines (SAMGs) and mitigation measures developed specifically for response to security concerns that arose from the events of September 11, 2001. The manner in which mitigation measures were evaluated for each accident sequence is described in Section 2.2.

The end result of this process was a list of accident scenarios (i.e., event sequence plus options for mitigation), which were subjected to detailed analysis of radionuclide release to the environment (described in Section 5) and offsite radiological consequence (Section 6).

2.1 Accident Sequence Analysis

An accident sequence is initiated by either an internal event (e.g., equipment failure[2], spurious control system signal or operator error) or external events (e.g., floods, fires, and seismic events). Sections 2.1.1 and 2.1.2 describe the method used to identify sequences initiated by internal and external events, respectively.

2.1.1 Sequences Initiated by Internal Events

The sequences generated by internal events and the availability of containment systems were identified using the NRC's plant-specific standardized plant analysis risk (SPAR) models,

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1 1×10^{-7} per reactor-year for sequences involving bypass of the containment pressure boundary or a perceived possibility of a large-early release.

2 For historical reasons, offsite equipment failures that result in failure of connections to the station's electric power grid are considered an 'internal event.'

licensee PRAs, and other risk information sources (see Figure 3). The core-damage frequencies calculated from the current SPAR models is similar to that calculated in utility PRAs. The following process was used to determine the scenarios for further analyses:

1. Candidate accident sequences were identified in analyses using plant-specific, SPAR models (Version 3.31).
   a. Initial Screening– Screened out initiating events with low CDFs ($<10^{-7}$) and sequences with a CDF $<10^{-8}$. This step eliminated 4% of the overall CDF.
   b. Sequence Evaluation– Identified and evaluated the dominant cutsets for the remaining sequences. Determined system and equipment availabilities and accident sequence timing.
   c. Sequence Grouping– Sequences determined to have similar equipment availabilities (i.e., details of individual component or support system failures might differ, but the functional capability of key systems was similar) and result in a similar time for the onset of core damage were aggregated into a single 'sequence group.'

2. Containment systems availabilities for each sequence were assessed using system dependency tables which delineate the support systems required for performance of the target front-line systems and from a review of existing SPAR model system fault trees.

3. Core-damage sequences from the licensee PRA model were reviewed and compared with the scenarios determined by using the SPAR models. Differences were resolved during meetings with licensee staff.

4. The screening criteria (CDF $< 10^{-6}$ for most scenarios, and $< 10^{-7}$ for containment bypass sequences) were applied to eliminate sequences from further analyses.

This process provides the basic characteristics of each scenario. However, it is necessary to have more detailed information about scenario than is contained in a PRA model. To capture the additional sequence details, further analysis of system descriptions and a review of the normal and emergency operating procedures (EOPs) were conducted.

The initial pass through this process identified only one sequence at Peach Bottom that survived the frequency threshold criteria. The sequence is initiated by the failure of vital AC Bus E12, which disables several (but not all) trains of safety equipment. The estimated frequency of this sequence was initially found be above the 1×10^{-6}/reactor-year threshold. As a result, the sequence was forwarded for an assessment of mitigative measures (see Section 2.2.1) and detailed analysis of accident progression and radiological release. However, later in time, the SPAR model was found to incorrectly represent certain features of this sequence and its frequency was reduced below the screening criterion. Further, the MELCOR analysis performed for this sequence determined that it would not, in fact, result in core damage. In spite of both of these late conclusions on the characteristics of this sequence, the analysis results provided unique insights into the effectiveness of small-capacity, non-safety related equipment in the plant to mitigate certain accident sequences, and it was retained in this report.

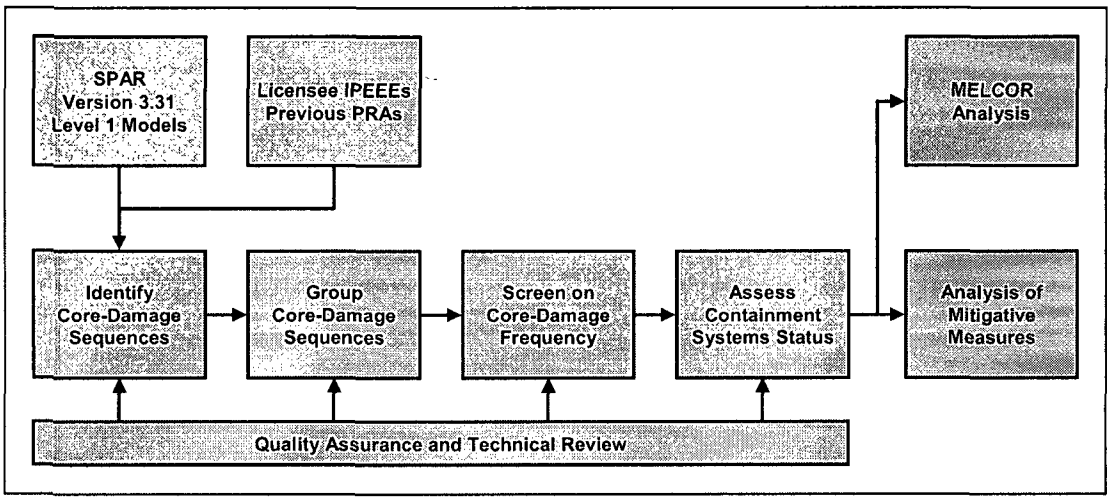

Figure 3 SOARCA Accident Scenario Selection Process.

2.1.2 Sequences Initiated by External Events

Detailed characteristics of the particular failures resulting from external events, such as fire, flooding or major seismic activity) are more difficult to systematically evaluate due to the lack of external event PRA models industry-wide. The external event sequences selected for analysis in the SOARCA project are representative of those that might have been observed to contribute to the core damage frequency in industry assessments of a wide spectrum of seismic, fire and internal flooding initiating events. Although these sequences were derived from a review of past studies such as NUREG-1150, individual plant examination for external event (IPEEE) submittals, and other relevant generic information, they do not represent particular accident sequences derived from any specific study.[3]

Various data sources and assessments were examined to identify the dominant contributors to core damage due to sequences initiated by external events. The dominant sequences were then reviewed to select a set of representative external event sequences that were deemed to be applicable to the Peach Bottom. The information from other studies was used to supplement the external event sequence descriptions with information about containment safeguards status and to estimate the sequence frequency.

No attempt was made to match the selected representative dominant external event sequences to actual sequence frequencies from one source; nor was any criterion used to capture a certain percentage of total external events CDF. Rather, the insights from other studies were used to select representative sequences. Special care was taken to preserve the (perceived) relative importance of external events CDF versus internal events CDF. In addition, dated or superseded

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[3] "External events" in this document refer to all other events at-power than those modeled routinely as internal events in a SPAR model. External events include internal flooding and fire, seismic events, extreme wind, tornado and hurricane related events, and other events that may be applicable to a specific site. The assessment is based on readily available information to NRC/RES analysts at the time of preparation of this report (such as NUREGs, SPAR-EE models, IPE and IPEEE submittals). The nature, vintage, and variety of the information may require a quantitative evaluation, supplemented with a suggested CDF for a representative dominant sequence.

required? If not, do we intend to do this?

information available (e.g., seismic hazard curves, internal fire frequencies and methodology, the internal flooding analyses) was updated to avoid undue conservatisms.

Seismic-initiated sequences were found to be the most restrictive in terms of the ability to successfully implement onsite mitigative measures and offsite protective actions. In addition, the seismic-initiated sequences were found to be dominant contributors to the external event core damage and release frequencies. As a result, representative external event sequences were assumed to be initiated by a moderate to large seismic event which leads to wide-spread damage to important plant support systems (primarily electric power sources).

This process identified two sequences groups which met the screening criteria of $1x10^{-6}$ per reactor-year for containment failure events and $1x10^{-7}$/reactor-year for events that have the potential to result in significant early releases to the environment:

- long-term station blackout – $1x10^{-6}$ to $5x10^{-6}$/reactor-year
- short-term station blackout – $1x10^{-7}$ to $5x10^{-7}$/reactor-year[4]

2.2 Mitigative Measures

The site-specific mitigation measures assessments were performed during visits to the Peach Bottom site in May 2007 and were supplemented by follow-up telephone conferences and emails with the licensee later in 2007. The licensee senior reactor operators, PRA analysts, and other licensee staff were provided the initial conditions and subsequence failures for each of the sequence groups being analyzed. The operator and plant response was subsequently evaluated to develop timelines for operator actions and equipment lineup or setup times for the implementation of the available mitigation measures. The resulting boundary conditions were used to develop the MELCOR boundary conditions that included operator actions and applicable mitigation measures.

Mitigation measures considered in the SOARCA analyses include the licensee's emergency operating procedures (EOPs), severe accident management guidelines (SAMGs), and mitigation measures and strategies incorporated into plant capabilities in response to the terrorist events of September 11, 2001.

2.2.1 Mitigation of Sequences Initiated by Internal Events

define

The Bus 1-E failure

As mentioned earlier, the MELCOR analysis described in Section 5.5 demonstrated that ~~this~~ sequence group did not result in core damage, even without crediting mitigation measures codified in 10CFR50.54(hh). The analysis included application of the standard plant procedures, including the operation of RCIC and the low-flow control rod drive hydraulic system (CRDHS) as an injection source. The SPAR model conservatively identified this sequence as having the potential for core damage. The MELCOR analysis in Section 5.5 provided a detailed evaluation of the effectiveness of the available plant systems and operating procedures without crediting additional mitigation capabilities.

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4 This ~~following~~ scenario does not meet the SOARCA screening criterion of 1×10^{-6} per reactor-year; however, it was analyzed in order to assess the risk importance of a lower frequency, higher consequence scenario.

2.2.2 Sequence Groups Initiated by External Events

It was noted earlier that the initiating event for external event sequences was assumed to be a seismic event, because it was judged to be limiting in terms of how much equipment would be available to mitigate as well as constrain offsite response. Fewer mitigation measures are expected to be available for a seismic event than for an internal fire or flooding event. For these sequence groups, the seismic PRAs provided information on the initial availability of installed systems. Next, judgments were made concerning the general state of the plant to judge the availability of mitigation measures codified in 10CFR50.54(hh)and the additional time to implement mitigation measures and activate emergency response centers (e.g., Technical Support Center and Emergency Operations Facility).

Seismic events considered in SOARCA result in loss of offsite and onsite AC power, and, for the more severe seismic events, loss of DC power. Under these conditions, the turbine-driven RCIC system is an important mitigation measure. BWR severe accident mitigation guidelines (SAMGs) include starting of the RCIC without electricity to cope with station blackout conditions. This is known as RCIC 'black-start.' 10CFR50.54(hh) mitigation measures have taken this a step further and also include long-term operation of RCIC without electricity (RCIC black run), using a portable generator to supply indications such as reactor pressure vessel level indication to allow the operator to manually adjust RCIC flow to prevent RPV overfill and flooding of the RCIC turbine. For the long-term station blackout sequence, RCIC can be used to cool the core until battery exhaustion. After battery exhaustion, black run of RCIC can be used to continue to cool the core. MELCOR calculations are used to demonstrate core cooling under these conditions.

The external events PRA does not describe general plant damage and accessibility following a seismic event. The damage was assumed to be widespread and accessibility to be difficult, consistent with the unavailability of many plant systems. For the long-term station blackout, it was judged that the seismic event would fail the Condensate Storage Tank, which is the primary water reservoir for RCIC, and that RCIC would initially be fed from the torus. MELCOR calculations showed that several hours would be available before torus temperature and pressure conditions precluded this. It was judged that this would be sufficient time to identify or arrange for another water reservoir for RCIC, such as the cooling tower basin (a large low lying reinforced concrete structure). At the time of the Peach Bottom site visit, the licensee had not procured the required portable equipment. Furthermore, the associated mitigative procedures were still being developed.

Mitigation measures codified in 10CFR50.54(hh) include portable equipment such as portable power supplies to supply indication, portable diesel-driven pumps, and portable air bottles to open air-operated valves, together with procedures to implement these measures under severe accident conditions. At the time of the Peach Bottom site visit, the licensee had not purchased any portable injection equipment. However, their plans to address the 10CFR50.54(hh) mitigation measures were discussed and the functional requirements of the equipment. The mitigation measures include the possibility of bringing in equipment from offsite (e.g., fire trucks, pumps and power supplies from sister plants or from contractors, external spray systems), but it did not quantify the types, amounts, and timing of this equipment arriving and being implemented.

Finally, the following items are cited as relevant to the accident progression and mitigative measure response. First, using external water spray with conventional firefighting equipment to scrub an ongoing fission product release was not evaluated in SOARCA. This evaluation is being performed in a separate study. However, it should be noted that the licensee had not yet purchased equipment for spray mitigation. Second, no multi-unit accident sequences were selected for the SOARCA project. This was beyond the scope of the project and considered unrealistic (i.e., beyond the screen criteria) for internal event sequences. Therefore, the mitigation measures assessment for external events was performed assuming that the operators only had to mitigate an accident at one reactor, even though Peach Bottom is a two-unit site.

In the latest (2010) RTC there was a presentation by the French of a station blackout event at one of their 4-unit plants: 2 units had separate and flooding at the site itself because an "island." They noted that there were operational difficulties.

Just because it is beyond the screens criterion does not justify the term "unrealistic"!

Also, don't we now have information that the external water spray is of minimal/no use? If so, take that part out.

accident progression (MELCOR) what kind of consequences? Bus E-12 had no health consequence calc - or maybe the question is what is?

3.0 ACCIDENT SCENARIO DEFINITIONS

Only one scenario met the screening criteria. However, for reasons described in Section 2.0, three were examined with deterministic ~~consequence~~ calculations. These scenarios are listed in Table 1. It was noted in Section 2.1.1 that one of these scenarios, internal event: loss of a vital AC bus E-12, was determined to have a frequency below the screening criterion following a careful review of and update to the SPAR model for Peach Bottom. Further, preliminary MELCOR calculations for this scenario clearly demonstrated the sequence can be mitigated without crediting mitigative actions using equipment and procedures called for in 10CFR50.54(hh). Although this particular sequence does not result in offsite radiological consequences, results of the MELCOR calculations of accident progression offer useful information on the importance of small-capacity, non-safety related equipment in mitigating certain accident sequences. Therefore, the results of these calculations are retained and presented in this report although the scenario would not result in damage to the reactor core.

The long- and short-term station blackout scenarios can both result in core damage and radionuclide release to the environment. A detailed description of these scenarios is, therefore, provided in Sections 3.1 and 3.2, respectively. Both sections include a discussion of available mitigation measures. A description of the loss of AC Bus E-12 scenario is given in Section 3.3.

Table 1 Accident scenarios and their frequencies

Scenario Description	Frequency (per Reactor Year)
Long-Term Station Blackout	$1x10^{-6}$ to $5x10^{-6}$
Short-Tem Station Blackout	$1x10^{-7}$ to $5x10^{-7}$
Loss of Vital AC Bus E-12	$\sim 5x10^{-7}$

3.1 Long-Term Station Blackout

The long-term station blackout is initiated by a moderately large earthquake (0.3–0.5 pga). It has an estimated frequency of $1x10^{-6}$ to $5x10^{-6}$/reactor-year which meets the SOARCA screening criterion of $1x10^{-6}$/reactor-year.

Section 3.1.1 describes the initial status of the plant following the seismic event. The key system availabilities normally accessible during the course of the accident are summarized in Section 3.1.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.1.3. Section 3.1.4 describes various scenarios based on the success of the mitigative actions. In particular, mitigated scenarios are defined where the mitigative actions are successful. Unmitigated scenarios are also defined where certain key mitigate measures are not successfully implemented.

3.1.1 Initiating Event

The long term station blackout scenario is a composite of several similar sequences that differ only by their initiating event. The initiators can be a large seismic event or an internal fire or flood. The seismic event is the largest contributor to the composite frequency of this sequence,

and is used as the basis for defining consequential events and conditions at the plant. Damage caused by the earthquake is assumed to result in a total loss of offsite power. In addition, onsite AC power is unavailable, with all diesel generators failing to start or run as needed. The diesel generators have a shared configuration between the two units, which causes any power failure to affect both units. This analysis considers only the response of failures at one of the two units, however.

3.1.2 System Availabilities

Immediately following the initiating event, specifically the loss of vital AC power, reactor scram and containment isolation would occur. The 'station blackout line' from the hydroelectric station downstream of the plant site is also assumed to fail due to structural damage to the dam and electric station components. The station batteries are available for four hours following loss of AC power, allowing components and systems powered by DC power to operate for this four hour period. This duration of DC power assumes operators successfully follow procedural actions to shed non-essential loads from the emergency DC bus. As a result, high-pressure coolant injection from reactor core isolation cooling (RCIC) and/or high pressure coolant injection (HPCI) would be available for the first four hours following the loss of AC power. Additionally, manual control of the safety/relief valves (SRV) would be available. No other plant systems would function.

3.1.3 Mitigative Actions

An unmitigated MELCOR calculation was performed for the long-term station blackout scenario assuming manual actions to mitigate the loss of vital safety systems are limited to those currently implemented in emergency operating procedures. The effects of additional mitigative actions and equipment that are being installed at the plant were then examined in a separate "mitigated" calculation. Results of the unmitigated calculation are described in Section 5.1; results of the separate mitigated accident scenario are described in Section 5.2.

Three operator actions were credited in the unmitigated long-term station blackout calculation. First, operators are assumed to open one SRV to begin a controlled depressurization of the reactor vessel approximately one hour after the initiating event. This action is prescribed in station emergency procedures to prevent excessive cycles on the SRV. The target reactor vessel pressure is at, or above, 125 psi, which would permit continued operation of RCIC (or HPCI if necessary). Second, operators are assumed to take manual control of RCIC approximately two hours after the initiating event. This involves local manipulation of the position of the (steam) throttle valve at the inlet to the RCIC turbine to reduce and control turbine speed. This action flow reduces and stabilizes coolant flow from the RCIC pump to maintain reactor vessel level at within a prescribed range. The third action is manual opening of a containment vent path, when containment pressure reaches unacceptably high levels. In the current analysis, a 16-in (hard-pipe) vent path is assumed to be opened when containment pressure exceeds 24 psig. This value was selected based on the decision logic shown in plant emergency procedures. Local control of the vent line isolation valves would be accomplished by assembling necessary, portable air and electric power supplies.

The mitigated long-term station blackout calculation credits one additional manual action. First, a portable AC power supply is assumed to be connected (through an inverter) to the DC bus delivering power to at least one SRV and to essential control room instrumentation (primarily reactor vessel pressure and level indication.) The precise time this action is completed is not important, provided it occurs before power from station batteries is exhausted four hours after the initiating event. If, for some reason, this action is not successful, and the RCIC pump were to trip (off), coolant makeup could be provided through low-pressure injection lines by means of a portable diesel-drive pump.

3.1.4 Scenario Boundary Conditions

Section 3.1.4.1 lists the sequence of events to be prescribed in the unmitigated long-term station blackout calculation. Section 3.1.4.2 summarizes the sequence of events in the mitigated long-term station blackout calculation which credits one additional manual action.

3.1.4.1 Unmitigated Cases

One unmitigated case was considered. The first case did not credit the mitigation measures of a portable pump called for under 10CFR50.54(hh). However, the RCIC black run together with use of the portable power supply to provide level indication and SRV control were included. The second case was analyzed without the RCIC black run and the portable power supply to quantify the benefit of these 10CFR50.54(hh) mitigation measures. The times shown below are how long after the initiating event, which is a seismic event.

Unmitigated Case

Event Initiation and Initial Plant Response

- AC power fails (loss of offsite power, coupled with failure of all diesel generators)
- Reactor trips
- Reactor and containment isolate
- DC power (station batteries) functional
- RCIC auto-initiates when level drops to low-level setpoint (time to be predicted by MELCOR) (Water source: Torus)
- Operator takes manual control of RCIC at the end of its first cycle (which is after about an hour)

1 hour

- Initiate RPV depressurization by opening 1 SRV (target RCS pressure is 125 psi)

4 hours

- Battery power exhausted
- SRV re-closes
- RCIC continues to operate at a fixed (constant) flow rate until RCIC steam line floods

3.1.4.2 Mitigated Case

The following is the time-line for this sequence group. The mitigated case credits all the mitigation measures codified in 10CFR50.54(hh), including the portable pump. The times shown below are how long after the initiating event, which is a seismic event.

Event Initiation and Initial Plant Response

- LOOP and SBO occurs due to a seismic event, recovery of offsite power is not expected during the mission time
- Reactor shuts down. RCS and containment are isolated.
- CRD, ECCS, SLC, Condensate, Containment Cooling and Containment Spray Systems are not available.
- Loss of all AC power due to seismic event, DC power available without chargers, EDGs do not automatically start.
- HPCI and RCIC are both available initially. HPCI is secured early in the event, and RCIC is used to maintain RCS level and can be black started to provide continued use until steam supply is lost (75-50 psi of main steam pressure)
- Control Room receives indication that plant is in a SBO condition requiring operator to enter SE-11, Station Blackout Procedure
- Without any operator action, HPCI and RCIC auto start and operate to maintain RPV level.
- Cooling tower basin is assumed to be undamaged, contains ~3 billion gallons of water

15 minutes

- Initial Operations assessment of plant status complete
- HPCI might auto-start in response to initial transient, will be secured
- RCIC will be operated to makeup for boil-off and to maintain RPV level
- In accordance with SE-11, Operations initiates the following mitigation measures:
  - Attempt to lineup the Conowingo hydroelectric dam (SBO Line) as an alternative offsite power source
  - Attempt manual start of EDGs
  - DC load shedding initiated
  - Operation of SRVs using station battery for RCS pressure control (RCIC steam line drains can be used as an alternative)

50 minutes

- Emergency Operations Facility manned (The EOF is located in the Philadelphia area, far away from the plant. Therefore, the timing should not be affected by the seismic event.)

1 hour

- Hydroelectric dam power supply (SBO Line) assumed to be unavailable due to initiating event
- Manual start of EDGs assumed to fail due to initiating event

- DC Load shedding completed, battery life extended to an estimated 4 hours (Batteries typically last for approximate 2 to 8 hours under normal loading conditions depending on life cycle of battery. At the beginning of its life, the battery duration is 8 hours. At the end of its life, the battery duration is 2 hours. The licensee suggested that battery duration of 4 hours would be reasonable due to the minimal expected loading.
- RPV depressurization is initiated using 1 SRV. The target RCS pressure is 125 psi.

Where is closed power?

1.5 hours

- The Emergency Operations Facility (EOF) is operational. The EOF reviews actions taken by Operations and determines the availability of the remotely located pump and station pumper truck stored outside of the Protected Area. Actions recommended by the EOF include the following:
  - Use portable power supply for operating SRVs and for RPV level indication.
  - Perform RCIC black-start.
  - Use portable diesel driven pump (250 psi, 500 gpm) to provide makeup to RCS, Hotwell, CST, and other locations. However, no water source and no hotwell for CST or RHR to connect to RCS and containment.
  - Use portable air supply to manually operate containment vent valves (vent into SGTS).
  - Use portable pump available to provide spray to primary or secondary containment leakage pathway.
  - Use pumper truck in place of portable diesel driven pump.

How does the facility "far away" determine the availability of the pump?

Define

1.75 hours

- Operators assess and concur with EOF recommendations. Operators prioritize recommendations based on plant conditions and begin implementation.

Maybe they received the operations staff to determine the availability

what happens if they non-concur? do you mean rubber stamp?

2 hours

- Technical Support Center manned (Because of the magnitude of the event, loss of causeway and other potential infrastructure failures, and multiple emergency responders located on both sides of the river, a 1 hour delay in minimum manning of TSC was assumed.)

2.25 hours

- Technical Support Center operational

3.5 hours

- Portable DC power supply connected to continue operating SRV to depressurize RPV
- Portable air supply to manually operate containment vent valves (vent into SGTS) in place and ready for operation. Rupture disc on vent line set at ~ 30 psi.
- RCIC black-started to limit use of site batteries and to continue providing makeup to RCS

before 10 hours

- Portable diesel-driven pump available.

3.2 Short-term Station Blackout

The short-term station blackout is initiated by a large seismic event (0.5 – 1.0 pga). It is more severe than the long-term station blackout and has an estimated frequency of $1x10^{-7}$ to $5x10^{-7}$/reactor year. Although the scenario does not meet the SOARCA screening criterion of 1×10^{-6} per reactor-year, it was retained in order to assess the risk importance of a lower frequency, higher consequence scenario.

Section 3.2.1 describes the initial status of the plant following the seismic event. The key system availabilities normally accessible during the course of the accident are summarized in Section 3.2.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.2.3. Section 3.2.4 describes various scenarios based on the success of the mitigative actions. In particular, mitigated scenarios are defined where the mitigative actions are successful. Unmitigated scenarios are also defined where certain key mitigate measures are not successfully implemented.

3.2.1 Initiating Event

The short-term station blackout is initiated by the same spectrum of events that lead to the long-term station blackout. The most frequent initiators are large seismic event or an internal fire or flood. The seismic event is the largest single contributor to the composite frequency of this sequence, and is used as the basis for defining consequential events and conditions at the plant. Damage caused by the earthquake is assumed to result in a total loss of offsite power. In addition, all diesel generators failing to start or run as needed, rendering all onsite AC power unavailable. Again, the diesel generators have a shared configuration between the two units, which causes any power failure to affect both units. This analysis considers only the response of failures at one of the two units, however. Additionally, the earthquake results in failure of DC power.

3.2.2 System Availabilities

Immediately following the initiating event, specifically the loss of vital AC power, reactor scram and containment isolation would occur. The 'station blackout line' from the hydroelectric station downstream of the site is also assumed to fail due to structural damage to the dam and electric station components (1g is well beyond the design basis earthquake for the hydro-station.) The major difference between this scenario and the long-term station blackout (Section 3.2) is that vital DC power from station batteries is also not available. Thus a total loss of all onsite and offsite electrical power occurs immediately following the initiating event, rather than several hours later, thereby disabling all plant equipment that depends on control or motive power from normal or emergency electrical sources for start-up and operation. This includes steam-driven emergency coolant makeup systems (RCIC and HPCI) and manual control of reactor pressure relief valves, which were available for a few hours in the long-term station blackout.

An unmitigated MELCOR calculation was performed for the short-term station blackout scenario assuming actions to mitigate the event are not feasible. Results of this calculation are described in Section 5.3.

3.2.3 Mitigative Actions

A separate calculation was performed for a permutation of the short-term station blackout scenario in which operators are assumed to manually actuate ("black start") the steam-driven RCIC system. This action involves local, manual opening of normally-closed valves to admit steam from the main steam lines into the RCIC turbine and pump discharge valves to direct water into the reactor vessel. These actions are assumed to occur very soon after the initiating event (in 10 minutes), thereby preventing the reactor water level from decreasing below the top of active fuel. However, manual actions necessary to regulate steam flow into the RCIC turbine are not credited in this scenario. As a result, the system effectively operates at a constant flow rate equivalent to the rated capacity of the system. This flow rate is greater than the rate required to makeup for evaporative losses and, after an initial decrease, reactor water level gradually rises above nominal and eventually over-fills the reactor vessel[5]. In this context, 'over-fill' means the reactor water level rises to the elevation of the main steam line nozzles, allowing water to spill into the steam lines, causing them to flood with water. The steam extraction line for the RCIC turbine connects to the main steam line at a low elevation [adjacent to the inboard main steam isolation valves (MSIVs).] Therefore, water spilling over into the main steam lines blocks or flows toward the RCIC turbine, causing the system to cease functioning. Results of the short-term scenario with RCIC black start are described in Section 5.4.

3.2.4 Scenario Boundary Conditions

Section 3.2.4.1 lists the sequence of events to be prescribed for two the unmitigated short-term station blackout calculations. No mitigated cases were performed.

3.2.4.1 Unmitigated Cases

Two unmitigated cases were considered. The unmitigated cases case did not credit the mitigation measures of a portable pump called for under 10CFR50.54(hh). However, the RCIC black start was considered in case one and case two did not credit any injection.

<u>Unmitigated case one (RCIC black start)</u>

Event Initiation and Initial Plant Response

- AC power fails (loss of offsite power, coupled with failure of all diesel generators)
- Reactor trips
- Reactor and containment isolate
- DC power (station batteries) fails
- Operator black starts RCIC
- RCIC continues to operate at a fixed (constant) flow rate until RCIC steam line floods

[5] If electric (control) power was available, the RCIC system would cycle on/off to maintain reactor level between a minimum and maximum setpoint. Without these control signals, or an independent means of monitoring reactor water level and manually controlling coolant flow rate (i.e., turbine speed), the system is assumed to run at full capacity after it is started.

Unmitigated case two (no injection)

Event Initiation and Initial Plant Response

- AC power fails (loss of offsite power, coupled with failure of all diesel generators)
- Reactor trips
- Reactor and containment isolate
- DC power (station batteries) fails

3.3 Loss of Vital AC Bus E-12

The scenario is initiated by the loss of vital AC Bus E-12. It was initially estimated to have a frequency above the SOARCA screening criterion of $1x10^{-6}$/reactor-year. However, after further review of the SPAR model and comparison with the licensee's PRA, the scenario was determined to have a CDF below the screening criteria. Since the MELCOR analysis provided unique insights into the response of the plant to an internal event sequence, the MELCOR analysis was retained.

Section 3.3.1 describes the initial status of the plant following the initiating event. The key system availabilities normally accessible during the course of the accident are summarized in Section 3.3.2. The pertinent mitigative measures available to address the accident progression are described in Section 3.3.3. Section 3.3.4 describes various scenarios based on the success of the mitigative actions. In particular, mitigated scenarios are defined where the mitigative actions are successful. Unmitigated scenarios are also defined where certain key mitigate measures are not successfully implemented.

3.3.1 Initiating Event

The initiating event for this scenario is failure of vital AC bus E-12 to provide power to associated plant equipment.

3.3.2 System Availabilities

Loss of one vital AC bus disables some plant equipment, but not all. For example, power to the instrument and control air system would be lost, and the inverters that charge the station batteries would not function. However, other AC buses would direct motive power to the residual heat removal (RHR) and core spray pumps, permitting use of low-pressure coolant injection. One of the two control rod drive hydraulic pumps would also remain available.

Steam-driven injection systems (high pressure coolant injection and reactor core isolation cooling)[6] operate as long as station batteries deliver DC power to control system components. Station batteries also facilitate manual control of SRVs. When battery power is depleted, high

[6] Although RCIC is available in all the standard plant analysis risk cut sets for this sequence, high pressure coolant injection (HPCI) is disabled due to independent failures in some of them. Availability of HPCI is not important in this sequence and is neglected.

pressure coolant injection, RCIC, and SRV controls are assumed to be lost. Injection flow from these sources terminates coincident with the loss of DC power, and any open SRV re-closes.

The shutdown cooling mode of residual heat removal would not be available due to loss of power to certain valves needed to align the system for that configuration. However, the system can be aligned to operate suppression pool cooling and/or drywell sprays.

Duration of DC power is treated as an uncertain parameter in this scenario. The licensee probabilistic risk assessment (PRA) uses a value of two hours, which is the minimum (tech spec) value and represents the worst possible condition: i.e., 'old' batteries (maximum tolerable voltage degradation) and no load shedding. New batteries (maximum voltage) are expected to have an eight hour lifetime without loading shedding. A reasonable estimate for the 'average' value of battery duration (taking into account battery age and the effectiveness of actions to shed non-essential DC loads) is four hours. As described in Section 5.5.3, a precise value is not particularly important, provided battery duration is greater than three hours.

3.3.3 Mitigative Actions

This event was shown to be satisfactorily mitigated without crediting any of the security-related mitigative actions mentioned in Section 3.1.3. As such, no additional mitigative analysis was performed.

See next page

3.3.4 Scenario Boundary Conditions

Section 3.3.4.1lists the sequence of events to be prescribed in the unmitigated loss-of-vital AC Bus E-12 accident scenario. Section 3.3.4.2 summarizes the sequence of events in the mitigated case.

3.3.4.1 Unmitigated Cases

A set of parametric unmitigated cases was considered. The unmitigated cases did not credit the mitigation measures of a portable pump called for under 10CFR50.54(hh). However, controlled RCIC operation until the station battery exhaustion and 1-pump of CRDHS injection were credited. The parametric cases varied the station battery life and other critical cooling functions.

Unmitigated Cases

Event Initiation and Initial Plant Response

- Loss of all AC-powered injection except 1 CRDHS pump
- Reactor trips
- Reactor and containment isolate
- DC power (station batteries) functional
- RCIC auto-initiates when level drops to low-level setpoint (Water source: CST)
- When level rises to operating range, operator takes manual control of RCIC to maintain RPV level

1 hour

- Initiate RPV depressurization by opening 1 SRV (target RCS pressure is 125 psi)

4 hours

- Battery power exhausted
- SRV re-closes
- RCIC continues to operate at a fixed (constant) flow rate until RCIC steam line floods

Parameters Varied in Sensitivity Calculations

- Not opening SRV
- Not taking manual control of CRDHS
- Maximize CRDHS flow by opening valve
- Include SLC injection flow
- Battery life of 2, 3, 4, and 6 hours

3.3.4.2 Mitigated Case

section 3.3.3 last page says that mitigation was satisfactory without credit for security-related mitigative action

The following is the time-line for this sequence group. The mitigated case credits all the 10CFR50.54(hh) mitigation measures, including the portable pump. The times shown below are how long after the initiating event, which is a seismic event.

Event Initiation

- Division IV DC power lost
- Nitrogen supply to Containment Isolation lost
- MSIVs close on loss of Instrument Air
- RCIC starts on low level and operates for period of time batteries are available
- 1 CRD pump operating at 110 gpm
- Control Room receives alarm that DC chargers are not available requiring operator to enter SE-13, Loss of DC power
- Without any operator action, CRD and RCIC are operating maintaining the core covered
- Drywell spray is available, but neglected because it is not necessary
- Shutdown cooling mode of RHR is not available because the needed valve alignment could not be done due to the power failure
- SLC is available, but neglected because its cooling injection flow of 50 gpm is not necessary

15 minutes

- Initial Operation=s assessment of plant status complete
- RCIC operating, maintaining RCS level
- In accordance with SE-13, DC load shed initiated

50 minutes

- Technical Support Center manned (Primary function would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigative measures.)
- EOF manned (Primary function would be to review initiating event, plant status, and operator action to provide guidance on alternative mitigative measures. The primary

Has this been defined?

users of SAMGs and EDMGs are the TSC supervisors who are trained on SAMGs and EDMGs.)

1 hour

- DC load shedding complete extending battery life to 4 hours. (Batteries typically last for approximate 2 to 8 hours under normal loading conditions depending on life cycle of battery. At the beginning of its life, the battery duration is 8 hours. At the end of its life, the battery duration is 2 hours.)
- Also available, opening CRD throttle valve to increase flow from 110 gpm to ~140 gpm without depressurization. (The increased flow rate of 140 gpm is an estimate provided by the licensee.)

This is about the 5th time saying this –

1.25 hours

- Technical Support Center operational

1.5 hours

- Manual controlled depressurization using 1 SRV
- TSC and/or EOF reviews actions taken by operations and determined the availability of the remotely located equipment. Recommend the following actions:
  - portable power supply to ensure long-term DC to hold SRV open and provide level indication (allows management of RCIC)
  - RCIC blackstart
  - portable diesel driven pump (250 psi, 500 gpm) to makeup to RCS, Hotwell, CST, etc
  - portable air supply to manually operate containment vent valves (vent into SGTS)
  - portable diesel driven pump to inject into drywell via RHR and RCS
  - portable pump to provide spray to primary or secondary containment leakage pathway
  - pumper truck can be used in place of portable diesel driven pump

Really? at 1.5 hours what does the spray accomplish? and even if it could be implemented I thought our experts showed little/no effectiveness!

1.75 hours

- Operations staff assesses and concurs with TSC and/or EOF's recommendations. Operations staff prioritizes recommendation based on plant conditions and begin implementation.

2.5 hours

- Manual operation of RCIC to sustain RCS level after battery depletion
- Use of a portable DC power supply to operate SRV to depressurize RPV and to allow makeup with the portable pump @ ~ 500 gpm.

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4.0 MELCOR MODEL OF THE PEACH BOTTOM PLANT

This section provides a summary of the MELCOR model of the Peach Bottom Atomic Power Station. A comprehensive description of the model is available in separate documentation [3].

The MELCOR Peach Bottom model was originally generated for code assessment applications with code version 1.8.0 at Brookhaven National Laboratories. The model was subsequently adopted by J. Carbajo at Oak Ridge National Laboratories to study differences in fission product source terms predicted by MELCOR 1.8.1 and those generated for use in NUREG-1150 using the Source Term Code Package (STCP) [4]. In 2001, considerable refinements to the BWR/4 core nodalization were made by Sandia National Laboratories to support the developmental assessment and release of MELCOR 1.8.5. These refinements concentrated on the spatial nodalization of the reactor core (both in terms of fuel/structural material and hydrodynamic volumes) used to calculate in-vessel melt progression.

These developments culminated in the re-assessment of radiological source terms for high burnup core designs and a comparison of their release characteristics to the regulatory prescription outlined in NUREG-1465 [5]. These calculations addressed a wide spectrum of postulated accident sequences, which required new models to represent diverse plant design features, such as:

- modifications of modeling features needed to achieve steady-state reactor conditions (recirculation loops, jet pumps, steam separators, steam dryers, feedwater flow, CRDHS, main steam lines, turbine/hotwell, core power profile),
- new models and control logic to represent coolant injection systems (RCIC, HPCI, RHR, LPCS) and supporting water resources (e.g., CST with switchover), and
- new models to simulate reactor vessel pressure management (safety relief valves, safety valves, ADS, and logic for manual actions to affect a controlled depressurization if torus water temperatures exceed the heat capacity temperature limit).

Subsequent work in support of other U.S. NRC research programs motivated further refinement and expansion of the model in two broad areas. The first area focused on the spatial representation of primary and secondary containment. The drywell portion of primary containment has been sub-divided to distinguish thermodynamic conditions internal to the pedestal from those within the drywell itself. More importantly, considerable refinements have been added to the spatial representation and flow paths within the reactor building (i.e., secondary containment). The second area has focused on bringing the model up to current "best practice" standards for MELCOR 1.8.6.

4.1 Reactor Vessel and Coolant System

Excluding the core region, the reactor pressure vessel is represented by seven control volumes, nine flow paths and 24 heat structures. Nodalization for the core region between the core top guide and the bottom of active fuel are described in detail in Section 4.2. Figure 4 provides a

reactor vessel nodalization detail comparing MELCOR modeling features to actual vessel design. Control volumes are indicated by "CV" followed by the three-digit control volume number, and flow paths are indicated by "FL" followed by the three-digit flow path number.

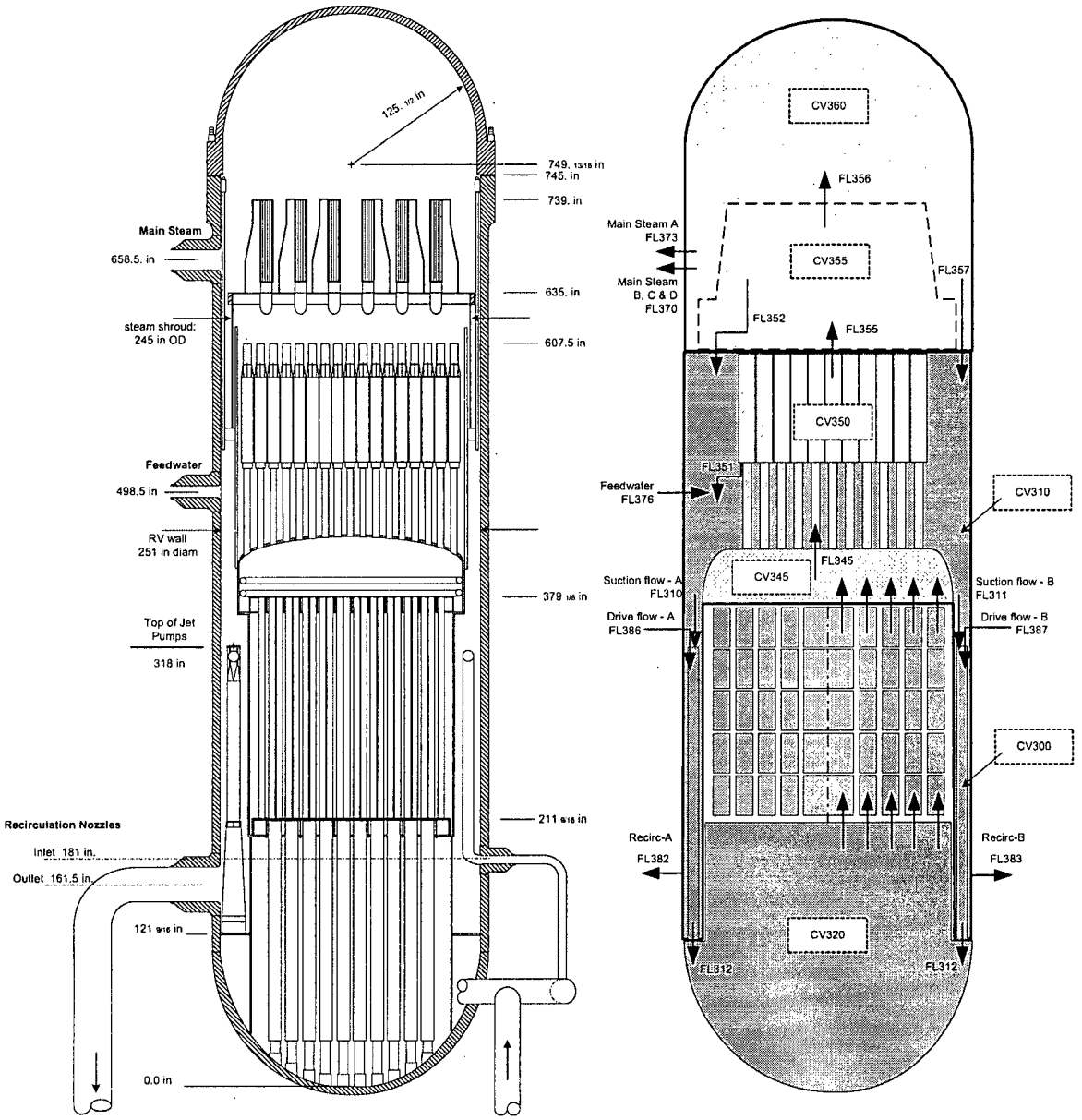


Figure 4 Reactor Vessel Cross-Section Detail and MELCOR Hydrodynamic Nodalization

Figure 5 is a schematic representation of the MELCOR control volumes and flow paths for the reactor coolant system, including:

- reactor recirculation piping,
- main feedwater and steam lines, and
- connections to emergency coolant injection and heat removal systems.

Collectively, these ancillary systems permit the model to properly calculate steady state, as well as a wide variety of transient conditions. To optimize numerical performance of this model, some consolidation of parallel lines or trains of certain systems has been made. For example, the four main steam lines have been represented by two parallel "lines," one of which represents the single steam line containing the lead (i.e., lowest set point) SRV, the second represents the composite geometry of the remaining three lines. Isolating the steam line with the lead SRV permits the proper geometry (internal volume, structural surface area, etc.) to be represented for fission product transport from the reactor to the suppression pool during accident sequences in which fuel damage begins while the reactor vessel is at high pressure and pressure relief is accomplished by SRV operation.

4.2 Reactor Core

In MELCOR, the region tracked directly by the COR Package model includes a cylindrical space extending vertically downward along the inner surface of the core shroud, from the core top guide to the reactor vessel lower head. It also extends radially outward from the core shroud to the hemispherical lower head in the region of the lower plenum below the base of the downcomer, preserving the curvature of the lower head from this point back to the vessel centerline.

The core and lower plenum regions are divided into concentric radial rings and axial levels. Each core cell may contain one or more core components, including fuel pellets, cladding, canister walls, supporting structures (such as the lower core plate and control rod guide tubes), non-supporting structures (such as control blades, the upper tie plate and core top guide) and (once fuel damage begins) particulate and molten debris.

The spatial nodalization of the core is shown in Figure 6. The entire core and lower plenum regions are divided into six radial rings. As shown in Figure 7, rings one, two, three, four and five represent 112, 160, 200, 168 and 124 fuel assemblies, respectively. The radial distance between each of the five rings is not uniform. The radius of each ring was defined in a way that preserves the radial power distribution in the Unit 2 core, based on plant operating data from four recent and consecutive operating cycles. Radial ring 6 represents the region in the lower plenum outside of the core shroud and below the downcomer. Ring 6 exists only at the lowest axial 'levels' in the core model.

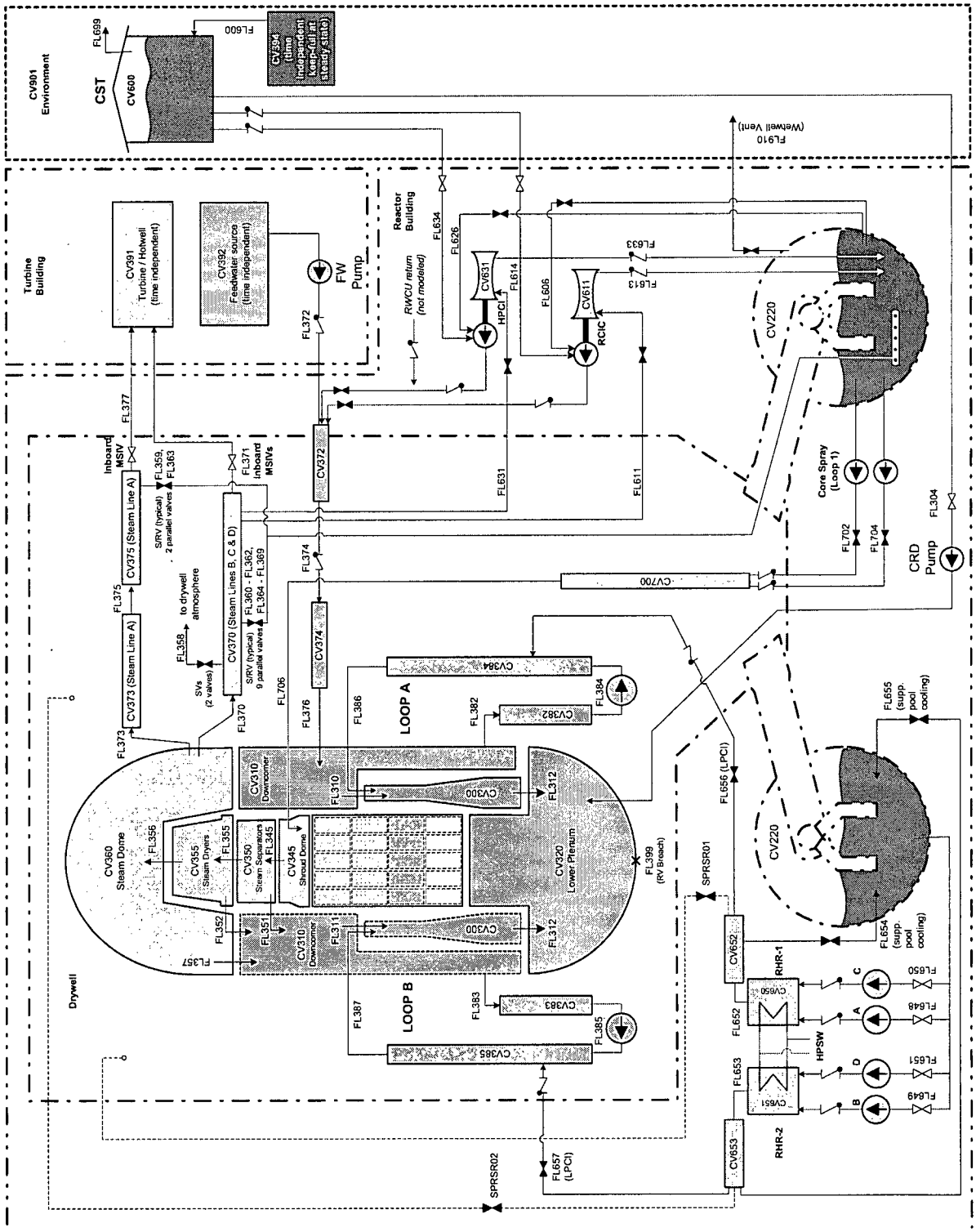


Figure 5 **Spatial Nodalization of Reactor Pressure Vessel and Coolant System**

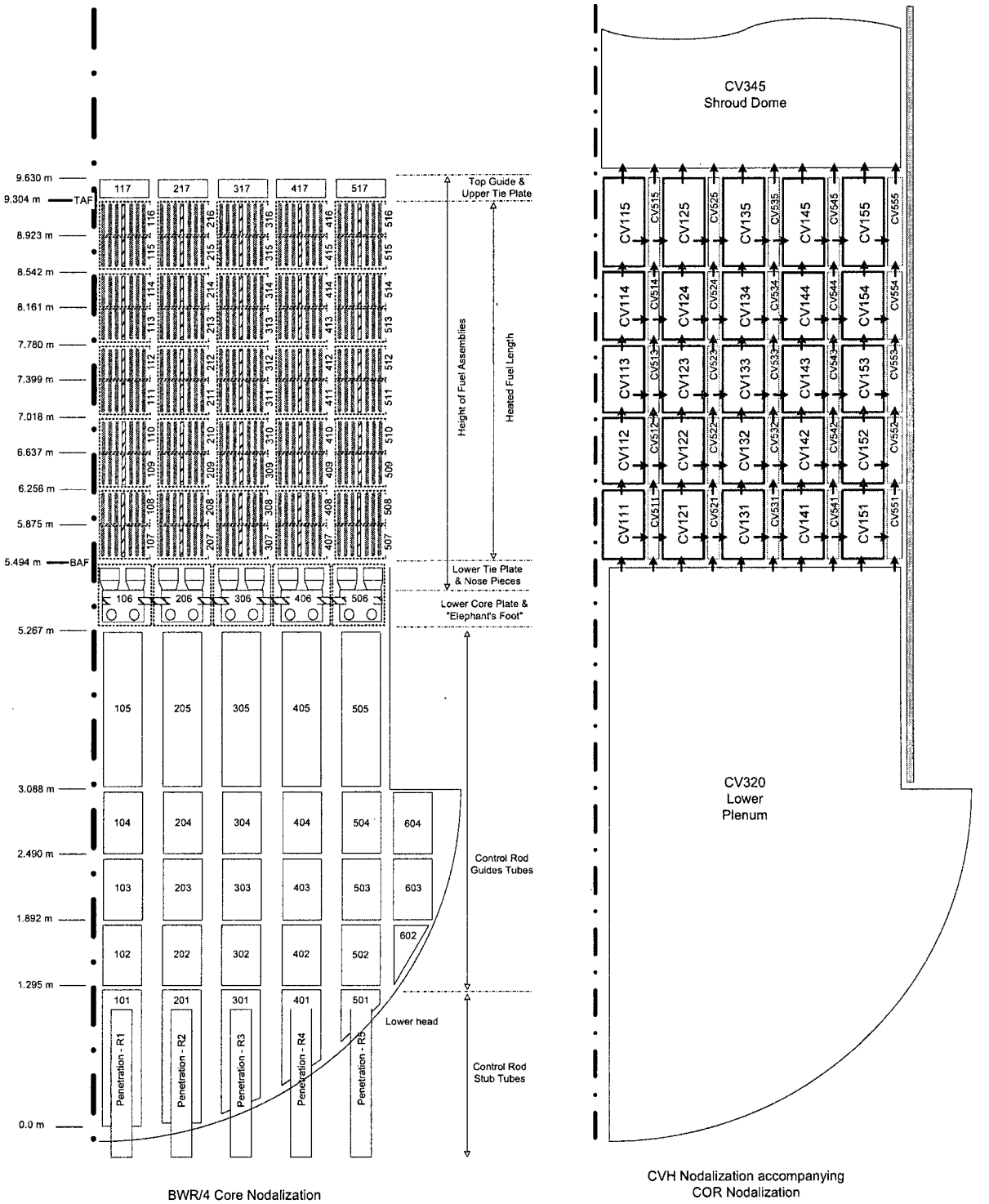


Figure 6 Spatial Nodalization of the Core and Lower Plenum

radius > / v / 0	0 center	0.153 16	0.305 17	0.458 18	0.610 19	0.763 20	0.915 21	1.068 22	1.220 23	1.373 24	1.525 25	1.678 26	1.830 27	1.983 28	2.135 29	2.288 30 (m)
0.153	16	0.928	1.306	1.084	1.258	1.045	1.413	1.094	1.344	1.069	1.440	1.090	1.162	0.727	0.528	0.270
0.305	17	1.305	1.100	1.374	1.065	1.395	1.207	1.466	1.209	1.453	1.303	1.421	1.225	0.818	0.536	0.272
0.458	18	1.084	1.367	1.155	1.339	1.162	1.471	1.216	1.487	1.203	1.496	1.236	1.273	0.791	0.531	0.268
0.610	19	1.253	1.044	1.365	1.064	1.341	1.134	1.482	1.215	1.454	1.268	1.403	1.156	0.756	0.510	0.257
0.763	20	1.040	1.368	1.159	1.335	1.120	1.453	1.242	1.448	1.203	1.465	1.098	1.110	0.704	0.484	0.241
0.915	21	1.410	1.199	1.460	1.120	1.459	1.246	1.500	1.163	1.480	1.323	1.337	1.121	0.700	0.453	0.215
1.068	22	1.092	1.452	1.240	1.473	1.233	1.489	1.235	1.455	1.250	1.378	0.995	0.980	0.620	0.336	0.168
1.220	23	1.343	1.205	1.479	1.206	1.431	1.129	1.417	1.145	1.258	1.139	1.014	0.724	0.450	0.246	
1.373	24	1.069	1.459	1.201	1.443	1.186	1.432	1.203	1.245	0.954	0.945	0.720	0.522	0.295		
1.525	25	1.439	1.299	1.489	1.257	1.447	1.297	1.352	1.128	0.942	0.775	0.498	0.306	0.172		
1.678	26	1.089	1.417	1.231	1.394	1.087	1.320	0.981	1.007	0.716	0.497	0.314				
1.830	27	1.161	1.222	1.268	1.152	1.104	1.112	0.970	0.707	0.517	0.304					
1.983	28	0.727	0.613	0.788	0.754	0.701	0.695	0.615	0.445	0.291	0.171					
2.135	29	0.528	0.532	0.530	0.508	0.482	0.450	0.333	0.244							
2.288	30	0.270	0.271	0.268	0.256	0.240	0.214	0.166								
(m)																

	# assemblies	3-cycle average RPF
RING 1	112	1.223
RING 2	160	1.313
RING 3	200	1.275
RING 4	168	0.746
RING 5	124	0.295
	764	

Figure 7 Local Relative Power Fraction (RPF) and 5-Ring Radial Boundaries of Core

The core and lower plenum are divided into 17 axially-stacked levels. The height of a given level varies, but generally corresponds to the vertical distance between major changes in flow area, structural material(s) or other physical features of core (and below core) structures. Axial levels 1 through 5 represent the open space and structures within the lower plenum. Initially, this region has no fuel and no internal heat source, but contains a considerable mass of steel associated with the control rod guide and in-core instrument tubes. During the core degradation process, the fuel, cladding and other core components displace the free volume within the lower plenum as they relocate downward in the form of particulate or molten debris.

Axial level 6 represents the steel associated with fuel assembly lower tie plates, fuel nose pieces, the lower core plate and its associated support structures. Particulate debris formed by destroyed fuel, canister and control blades above the lower core plate will be supported at this level until the lower core plate yields. Axial levels 7 through 16 represent the active fuel region. All fuel is initially in this region and generates the fission and decay power. Axial level 17 represents the non-fuel region above the core, including the top of the canisters, the upper tie plate and the core top guide.

4.3 Primary Containment and Reactor Building

The primary containment of the BWR Mark I design consists of two separate regions: a 'drywell' and 'wetwell.' As shown in Figure 8, each region is explicitly represented in the

MELCOR model with distinct hydrodynamic control volumes, flow paths and heat structures to preserve the geometric configuration and major functional features of the Mark I design; e.g., steam pressure suppression, fission product scrubbing and surface deposition. The drywell is further divided into four connected volumes to account for non-uniformities in the temperature and composition of the atmosphere during late phases of a severe accident.

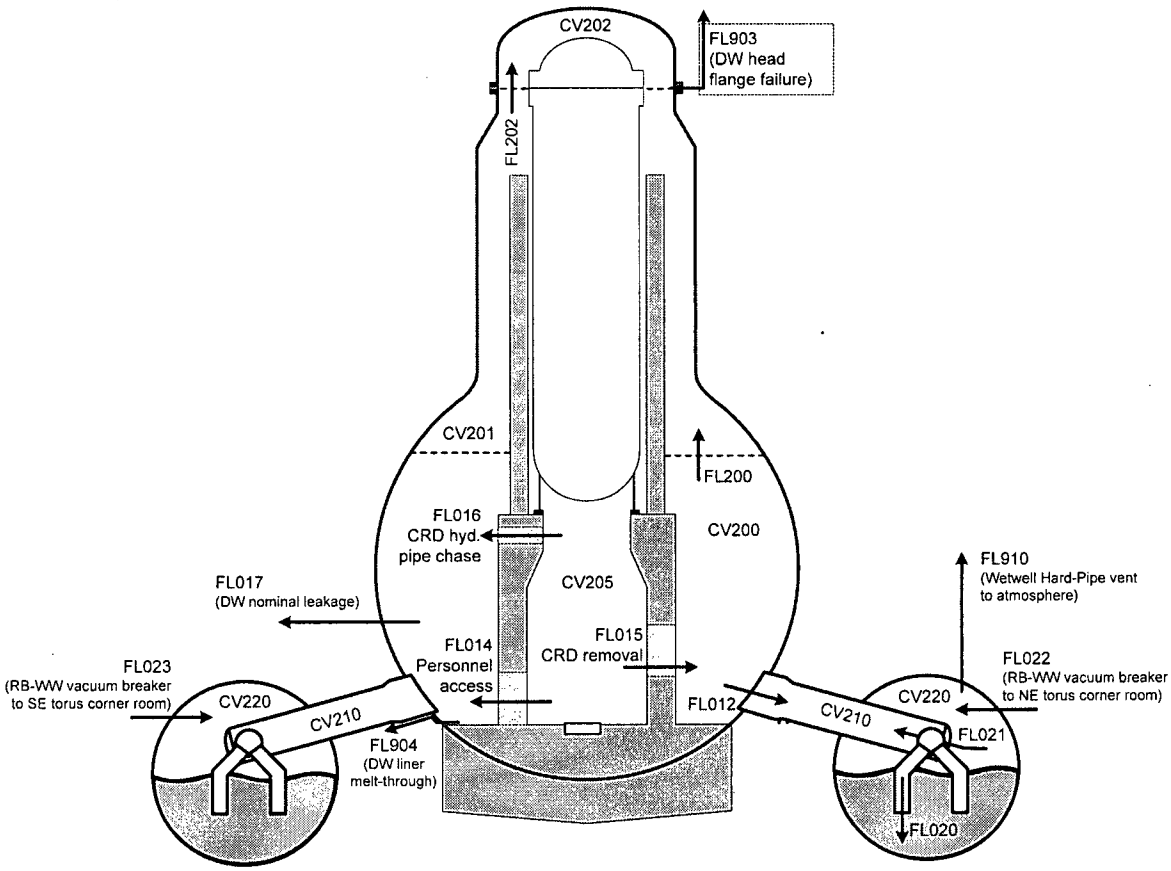


Figure 8 Hydrodynamic Nodalization of the Primary Containment

The internal volume, airflow flow pathways and structures of the reactor building are modeled in considerable detail as illustrated in Figure 9 and Figure 10. The reactor building fully encloses the primary containment, and participates in the release pathway of fission products to the environment released from the containment, by offering a large volume within which an airborne radionuclide concentration can be diluted by expansion into and mixing with the building atmosphere.

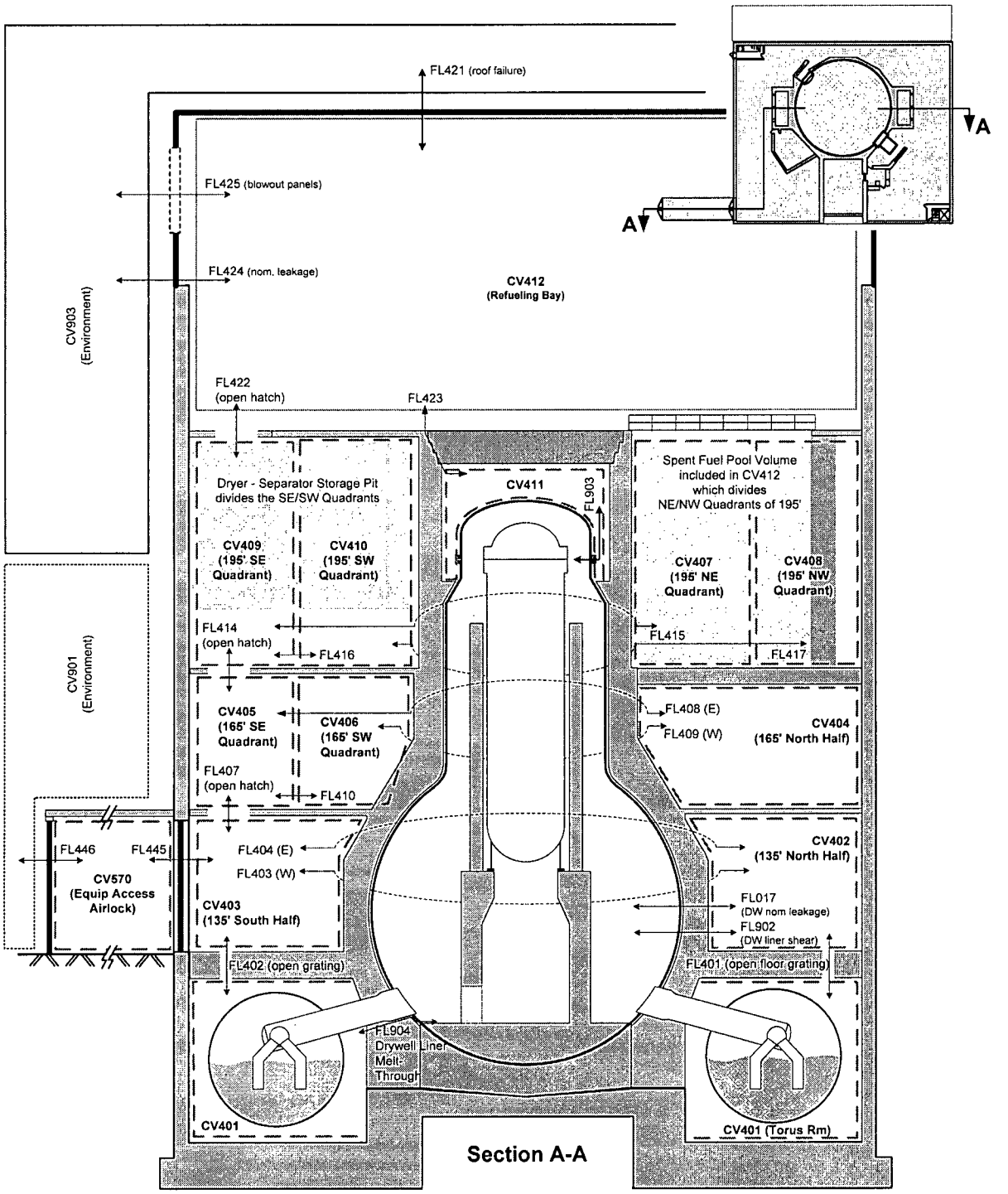


Figure 9 Hydrodynamic Nodalization of the Reactor Building (a)

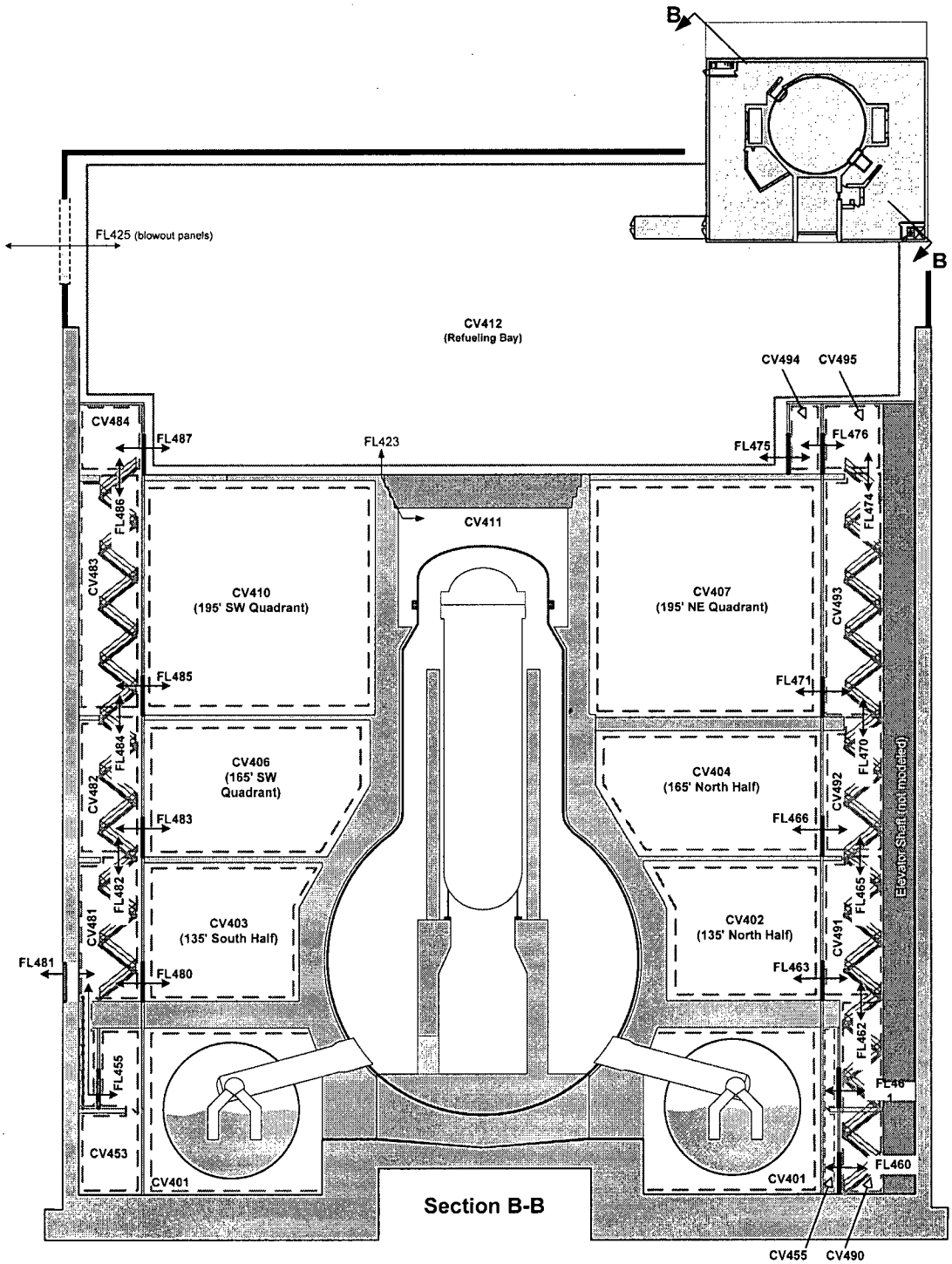


Figure 10 Hydrodynamic Nodalization of the Reactor Building (b)

The airborne concentration of fission product aerosols within the reactor building is attenuated by gravitational settling and other natural deposition mechanisms[7]. The building is, therefore, occasionally referred to as a secondary containment, in spite of the fact that it has a negligible capacity for internal pressure.

4.4 Ex-vessel Drywell Floor Debris Behavior

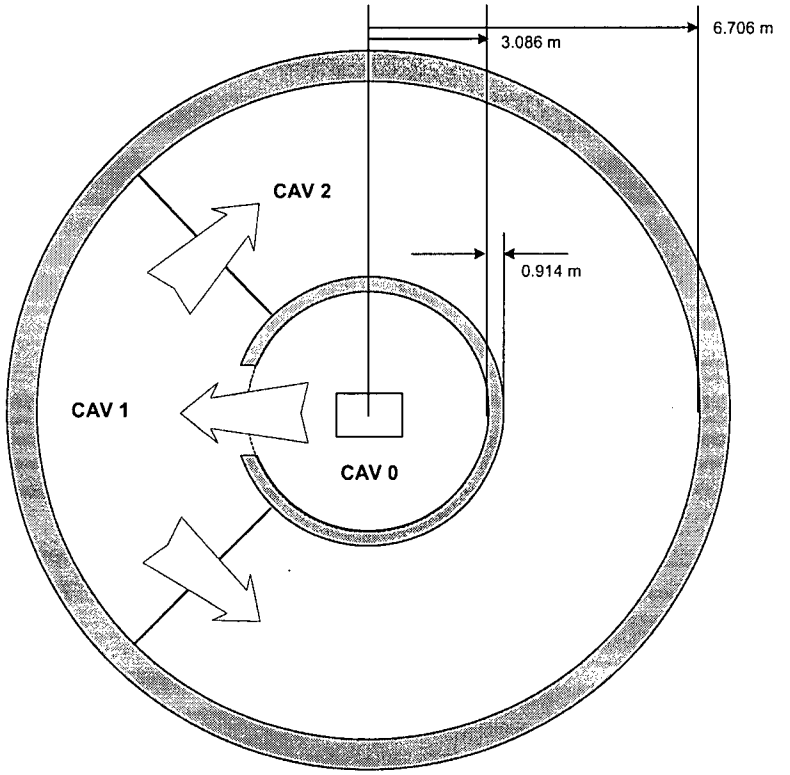
The drywell floor is sub-divided into three regions for the purposes of modeling molten-core/concrete interactions. The first region (which receives core debris exiting the reactor vessel) corresponds to the reactor pedestal floor and sump areas (CAV 0). Debris that accumulates in CAV0 can flow out through an open doorway in the pedestal wall to a second region representing a 90° sector of the drywell floor (CAV 1). If debris accumulates in this region to a sufficient depth, it can spread further around the annular drywell floor into the third region (CAV2). This discrete representation of debris spreading is illustrated in Figure 11.

Two features of debris relocation within the three regions are modeled. The first represents bulk debris 'spill over' or movement from one region to another. A control system monitors the debris elevation and temperature within each region, both of which must satisfy user-defined threshold values for debris to move from one region to its neighbor. More specifically, when debris in a cavity is at or above the liquidus temperature of concrete, all material that exceeds a predefined elevation above the floor/debris surface in the adjoining cavity is relocated (6 inches for CAV 0 to CAV 1, and 4 inches for CAV 1 to CAV 2). When debris in a cavity is at or below the solidus temperature of concrete, no flow is permitted. Between these two debris temperatures, restricted debris flow is permitted by increasing the required elevation difference in debris between the two cavities (more debris 'head' required to flow).

The second control system manages debris spreading radius across the drywell floor within CAV1 and 2. Debris entering CAV 1 and CAV 2 are not immediately permitted to cover the entire surface area of the cavity floor. The maximum allowable debris spreading radius is defined as a function of time. If the debris temperature is at or above the concrete liquidus temperature, then the maximum transit velocity of the debris front to the cavity wall is calculated (i.e., results in 10 minutes to ~~transverse~~ CAV 1 and 30 minutes ~~to transverse~~ CAV 2). When the debris temperature is at or below the concrete solidus, the debris front is assumed to be frozen and lateral movement is precluded (i.e., debris velocity is 0 m/s). A linear interpolation is performed to determine the debris front velocity at temperatures between these two values.

traverse

[7] The building is also equipped with a ventilation system with aerosol and charcoal filters, which would greatly aid in reducing an airborne radioactive release. However, these systems would not be available during the particular accident scenarios examined in this work, due to loss of electrical power or other equipment failures.



CAV	FLOOR AREA	EQUIV RADIUS	PERIMETER RATIO
0	29.92	3.086	0.95
			--
1	22.75	2.691	0.94
			0.62
2	68.25	4.661	1.72
			1.08

Figure 11 Drywell Floor Regions for Modeling Molten-Core/Concrete Interactions.

Full mixing of all debris into a single mixed layer is assumed in each of these debris regions. The specific properties for concrete composition, ablation temperature, density, solidus temperature and liquidus temperature are specified. The concrete composition represented in the MELCOR model is listed in Table 2. The drywell floor concrete includes 13.5% rebar.

Other key user-defined concrete properties are selected to match defaults for limestone common sand concrete and include:

- initial temperature of 300 K
- ablation temperature of 1500 K
- solidus temperature of 1420 K
- liquidus temperature of 1670 K
- density of 2340 kg/m^3
- emissivity of 0.6

Table 2 Concrete Composition

Species	Mass Fraction
Al_2O_3	0.0091
Fe_2O_3	0.0063
CaO	0.3383
MgO	0.0044
CO_2	0.2060
SiO_2	0.3645
H_2O_{evap}	0.0449
H_2O_{chem}	0.0265

4.5 Containment Failure Model

Peach Bottom has a Mark I containment (Figure 8) that consists of a drywell and a toroidal-shaped wetwell, which is half-full of water (i.e., the pressure suppression pool.) The drywell has the shape of an inverted light bulb. The drywell head is removed during refueling operations to gain access to the reactor vessel. The drywell head flange is connected to the drywell shell with 68 bolts of 2 ½" diameter (Figure 12). The flanged connection also has two ¾" wide and ½" thick Ethylene Propylene Diene Methylene (EPDM) gaskets. The torque in the 2 ½" diameter bolts range from 817 to 887 ft-lbs [18][19]. An average bolt toque of 850 ft-lbs was used in this study.

The 68 drywell head flange bolts (see Figure 12) are pre-tensioned during reassembly of the head. This pretension also compresses the EPDM gaskets located in the head flange. During an accident condition, the containment vessel may be pressurized internally. The internal pressure would counteract the pre-stress in the bolts. At a certain internal pressure, all the pre-stressing force from the bolts would be eliminated, and the EPDM gaskets would be decompressed. Further increase in the internal pressure would result in leakage at the flanged connection.

The EPDM gasket manufacturers recommend a maximum squeeze (compression) of 30 percent for a static-seal joint. The gaskets recover about 15 percent of the total thickness after the compressive load is removed from the flange. However, the licensee engineers informed the SOARCA personnel that the gaskets for the reactor vessel head flange are squeezed to 50 percent to have a metal to metal contact to ensure no leakage at design pressure of 56 psi. In addition, the gaskets are exposed to constant temperature and radiation which contribute to early degradation. For this reason, the gaskets are replaced during each reassembly of the reactor vessel head. Based on this information and actual observations, the PBAPS licensee engineers recommended a gasket recovery of 0.03 inch.

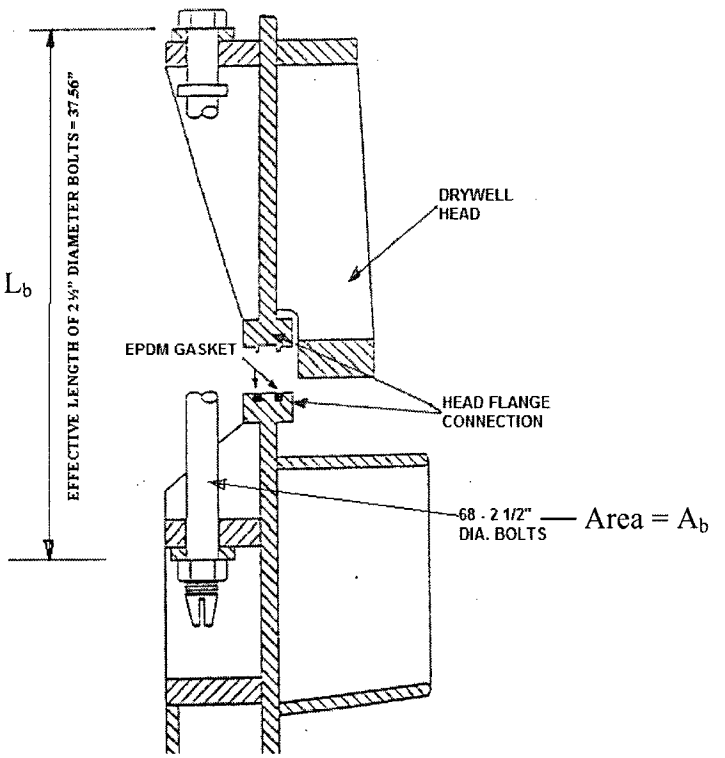


Figure 12 Drywell Head Flange Connection Details.

Based on the gasket recovery of 0.03 inches, the actual gap was determined at various internal pressures as:

Elongation in the bolt = $\Delta L_b = \Delta L_{bl} - 0.03$ inch

where: implies that these define terms in the above equation but S doesn't see A_b or E or L_b explain

L_b = length between the bolt head and nut (Figure 12) = 37.56 inches
A_b = Tensile stress area of the bolt [16] = 4.0 in^2
$E = 28.0 \times 10^6$ psi
$\Delta L_{bl} = 0.0054$ inch

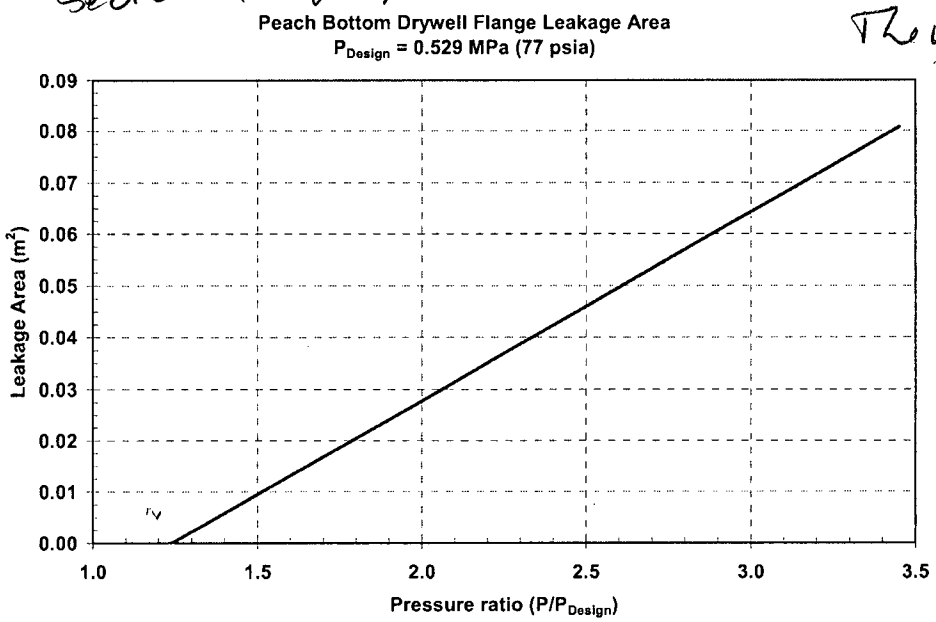
Leakage areas for different internal pressures are shown in Figure 13. The reactor vessel head flange does not leak until the internal accident pressure is 0.660 MPa (i.e., $P/P_D = 1.25$ or 81 psig). Thereafter there is a gradual increase in the leakage area.

At high temperatures (>755 K, or >900°F), upward and radial thermal growth of the drywell would lead to binding of small and large penetrations against the biological shield wall and failure. In addition, radial growth of the containment may also cause the seismic stabilizers to punch through the upper portion of the drywell at high temperatures [17]. This observation is consistent with the results of the previous studies which show that the drywell is likely to fail at

the low pressure range of 0-65 psig [17]. Therefore, it can be concluded that the drywell is likely to fail under any appreciable pressure load at temperatures of 900°F or greater.

Finally, the containment can fail by drywell liner melt-through containment failure (see relevant discussions in Sections 4.4 and 4.7.2).

Section 4.4 only discusses transport of debris. The word "failure" is never used.



Now in 4.7.2 which only says that water around debris would not produce failure.

Figure 13 Drywell Flange Leakage Model versus Containment Pressure

How large is the "failure?"

4.6 Radionuclide Inventories and Decay Heat

One important input to MELCOR is the initial ~~concentration of radionuclides~~ masses of fission products in the fuel and their associated decay heat. The values are important to the timing of initial core damage and the location and concentration of the initial radioactive source. The radionuclides in a nuclear reactor come from three primary sources: (1) "fission products" are the result of fissions in either fissile or fissionable material in the reactor core; (2) actinides are the product of neutron capture in the initial heavy metal isotopes in the fuel; and (3) other radio-isotopes are formed from the radioactive decay of these fission products and actinides. Integrated computer models such as the TRITON sequence in SCALE exist to capture all of these inter-related physical processes, but they are intended primarily as reactor physics tools [15]. As such, their standard output does not provide the type of information needed for MELCOR [8]. It is important to note that changes to the TRITON sequence in SCALE were not needed for this analysis. The BLEND3 post-processing software extracts output from the TRITON sequence and combines it in a way which makes it useful for MELCOR [8].

You talk about melt through high T - and high P failures - So is area? How does one the other mechanisms.

A Global Nuclear Fuel (GNF) 10x10 (GE-14C) fuel assembly was used as a typical fuel element for Peach Bottom analysis. Information regarding assembly dimensions, enrichments and operating characteristics were obtained from the licensee (with permission from the fuel vendor)

Would that be off-site release implications?

and used for a realistic evaluation. Twenty-seven different TRITON runs were performed to model three different cycles of fuel at 9 specific power histories. The specific power histories ranged from 2 MW/MTU to 45 MW/MTU which bounded all expected BWR operational conditions. For times before the cycle of interest, an average specific power of 25.5 MW/MTU was used. For example, for the second cycle fuel, the fuel was burned for its first cycle using 25.5 MW/MTU, allowed to decay for an assumed 30 day refueling outage and then 9 different TRITON calculations were performed with specific powers ranging from 2 to 45 MW/MTU. The BLEND3 code was applied to each of the 50 core nodes[8] in the MELCOR model using average specific powers derived from data for three consecutive operating cycles and appropriate nodal volume fractions. Once new libraries for each of the 50 nodes in the model were generated, the final step in the procedure was to deplete each node for 48 hours. The decay heats, masses and specific activities as a function of time were processed and applied as input data to MELCOR to define decay heat and the radionuclide inventory.

See comments on Vol 1 relative to this discussion

4.7 Modeling Uncertainties

The primary objective of the SOARCA project is to provide a best estimate prediction of the likely consequences of important severe accident events at reactor sites in the U.S. civilian nuclear power reactor fleet. To accomplish this objective, the SOARCA project utilizes integrated modeling of the accident progression and off-site consequences using both state-of-the-art computational analysis tools as well as best modeling practices drawn from the collective body of knowledge on severe accident behavior generated over the past 25 years of research.

The MELCOR 1.8.6 computer code [8] embodies much of this knowledge and was used for the accident and source term analysis. MELCOR includes capabilities to model the two-phase thermal-hydraulics, core degradation, fission product release, transport, and deposition, and the containment response. The SOARCA analyses include operator actions and equipment performance issues as prescribed by the sequence definition and mitigative actions. The MELCOR models are constructed using plant data and the operator actions were developed based on discussions with operators during site visits. The code models and user-specified modeling practices represent the current best practices.

[10]

Uncertainties remain in our understanding of the phenomena that govern severe accident progression and radionuclide transport. Consistent with the best-estimate approach in SOARCA, all phenomena were modeled using 'best estimate' characterization of uncertain phenomena and events. Important severe accident phenomena and the proposed approach to modeling them in the SOARCA calculations were presented to an external expert panel during a public meeting sponsored by the NRC on August 21-22, 2006 in Albuquerque, New Mexico. A summary of this approach is described in Section 4.7.1. These phenomena are singled out because they are important contributors to calculated results and have uncertainty.

The two other topics, steam explosions and drywell liner melt-through on a 'wet' drywell floor have been previously included in lists of highly uncertain phenomena. Section 4.7.1 briefly

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[8] 5 radial rings by 10 axial levels

describes them and offers a summary of the significant research that led the SOARCA program to neglect their inclusion.

Finally, a systematic evaluation of phenomenological uncertainties for a particular sequence is a separate task and not discussed in this report. The task will evaluate the importance and impact of alternative settings or approaches for key uncertainties.

4.7.1 Base Case Approach on Important Phenomena

of severe accident progression modeling [10]

A ~~review of severe accident progression modeling for the State-of-the-Art Reactor Consequence Analyses (SOARCA) project was~~ conducted ~~at a public meeting in Albuquerque, New Mexico on August 21-22, 2006 [10]~~. This review focused primarily on best modeling practices for the application of the severe nuclear reactor accident analysis code MELCOR for realistic evaluation of accident progression, source term, ~~and offsite consequences~~. The scope of the meeting also included consideration of potential enhancements to the MELCOR code as well as consideration of the SOARCA project in general.

and

said last page

The review was conducted by five panelists with demonstrated expertise in the analysis of severe accidents at commercial nuclear power plants. The panelists were drawn from private industry, the Department of Energy national laboratory complex, and a company working on behalf of German Ministries. The review was coordinated by Sandia National Laboratories and attended by Nuclear Regulatory Commission staff. A discussion of the important uncertain modeling practices and their baseline approach are further discussed in ~~Volume II, "Best Modeling Practices."~~ A separate task in the SOARCA program is planned to address the importance of uncertainties in these modeling parameters.

NUREG/CR

4.7.2 Early Containment Failure Phenomena

The objective of SOARCA is to perform best-estimate evaluations of the accident progression and consequences from the most likely severe accident sequences for specific plants. Two phenomenological issues not included in the best-estimate approach used in SOARCA include (1) alpha-mode containment failure and (2) drywell liner melt-through in the presence of water leading to containment failure. These severe phenomena leading to an early failure of the containment were included in some of the NUREG-1150 analyses to quantify the risks from nuclear reactors.

The alpha-mode event is characterized by the supposition that an in-vessel steam explosion might be initiated during core meltdown by molten core material falling into the water-filled lower plenum of the reactor vessel. The concern was that the resulting steam explosion could impart sufficient energy to separate the upper vessel head from the vessel itself and form a missile with sufficient energy to penetrate the reactor containment. This of course would produce an early failure of the containment building at a time when the largest mass of fission products is released from the reactor fuel. In the following years, significant research was focused on characterizing and quantifying this hypothesized response in order to attempt to reduce the significant uncertainty. A group of leading experts ultimately concluded in a position paper published by the Nuclear Energy Agency's Committee on the Safety of Nuclear Installations that the alpha-mode failure issue for Western-style reactor containment buildings

yes, but if the PB upwhead separates, the stuff is just on the reactor bldg hg, no capacity

fix up this discussion to fix PB.

can be considered resolved from a risk perspective, posing little or no significance to the overall risk from a nuclear power plant.

The issue of Mark I drywell shell (liner) melt-through at Peach Bottom was assessed by the NUREG-1150 molten core-containment interaction panel. The results of expert panel elicitation are reported in Reference [12]. There were two schools of thought on this issue and hence the response was uncertain. Since the completion of NUREG-1150, the NRC has sponsored analytical and experimental programs to address and resolve this so-called "Mark I Liner Attack" issue. The results of an assessment of the probability of Mark I containment failure by melt attack of the liner were published in NUREG/CR-5423 [13] and NUREG/CR-6025 [14]. It was concluded that, in the presence of water, the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable.

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5.0 ACCIDENT PROGRESSION AND RADIOLOGICAL RELEASE ANALYSIS

This section describes the MELCOR accident progression analysis for the internal and external event scenarios described in Section 3.0. Version 1.8.6 of the MELCOR severe accident analysis code was used the accident progression and radiological release calculations.

5.1 Long-Term Station Blackout – Unmitigated Response

The unmitigated scenario event progression for the LTSBO accident progression analysis assumes that the operators follow the actions dictated in Special Event Procedure SE-11 [4]. This document provides guidelines for managing the plant with degraded AC power sources. Initial operator actions would concentrate on assessing plant status. Successful reactor scram, containment isolation and automatic actuation of RCIC for reactor level control would be verified. These checks would take approximately fifteen minutes. Additionally, one or more SRVs would cycle to control the reactor pressure vessel (RPV) pressure.

Special Event Procedure SE-11 requires the immediate alignment of the 'station blackout line' from Conowingo Dam in the event of failure of offsite power combined with the failure of all diesel generators to start. When this fails to provide AC power to the plant, which is what was assumed to occur for the MELCOR analysis, the operators are directed to de-energize all unnecessary DC loads. By removing as many unnecessary loads as possible from the DC bus, the station battery lifetime is extended. This load shedding would not affect or disable control logic to the RCIC, high pressure injection cooling, main control room instrumentation, or SRV control.

The load shedding is expected to begin fifteen minutes into the event, and take approximately fifteen minutes to complete. Plant system engineers estimate the effect of load shedding would be to extend station battery duration from two to four hours.

One consequence of station blackout is the loss of cooling to the RCIC and HPCI corner rooms. Heat losses from system piping and equipment to the room atmosphere would cause these areas to overheat. In such an event, step H-5 in the Special Event Procedure SE-11 is applicable. It directs operators to block open doors to these rooms, and facilitate cross ventilation which would slow the rate of room heat up. These actions are assumed to successfully prevent system isolation from high temperature for the maximum four hour period of system operation[9]. The Special Event Procedure SE-11, step H-7, directs the operators to monitor the inventory in the Condensate Storage Tank (CST) and take actions to refill the tank via gravity feed from other sources if necessary. Long-term viability of the Condensate Storage Tank (CST) is therefore assumed in the MELCOR calculations.

The calculated timing of key events that follow from all these actions is listed in Table 3. The time at which core damage begins strongly depends on the duration of station batteries. The

[9] Heat loss from RCIC (or HPCI) systems to their enclosure corner rooms is not explicitly represented in the MELCOR model.

difference in time between loss of DC power and the onset of core damage increases as battery lifetime increases due to gradual reductions in decay heat levels with time. In the absence of effective manual intervention, core damage eventually proceeds to melting and relocation of core material into the reactor vessel lower head, reactor vessel lower head failure, and release of molten core debris to the drywell floor.

Table 3 Timing of Key Events for Long-Term Station Blackout

Event (Time in hours unless noted otherwise)	LTSBO with 4 hr DC power (hours)
Station blackout – loss of all onsite and offsite AC power	0.0
Low-level 2 and RCIC actuation signal	10 minutes
Operators manually open SRV to depressurize the reactor vessel	1.0
RPV pressure first drops below LPI setpoint (400 psig)	1.2
Battery depletion leads immediate SRV re-closure	4.0
RCIC steam line floods with water – RCIC flow terminates	5.2
Downcomer water level reaches top of active fuel (TAF)	9.0
First hydrogen production	9.2
First fuel-cladding gap release	10.1
First channel box failure	10.6
First core cell collapse due to time at temperature	11.0
Reactor vessel water level reaches bottom of lower core plate	11.6
SRV sticks open due to cycling at high temperatures	11.7
First core support plate localized failure in supporting debris	13.4
Lower head dries out	14.9
Ring 2 CRGT Column Collapse [failed at axial level 1]	17.5
Ring 1 CRGT Column Collapse [failed at axial level 1]	17.6
Ring 5 CRGT Column Collapse [failed at axial level 2]	17.7
Ring 3 CRGT Column Collapse [failed at axial level 1]	18.1
Ring 4 CRGT Column Collapse [failed at axial level 2]	19.0
Lower head failure	19.5
Drywell head flange leakage begins	19.5
Hydrogen burns initiated in drywell enclosure region of reactor building	19.5
Refueling bay to environment blowout panels open	19.5
Hydrogen burns initiated in reactor building refueling bay	19.7
Refueling bay roof overpressure failure	19.7
Drywell liner melt-through initiated and drywell head flange re-closure	19.7
Hydrogen burns initiated in lower reactor building	19.7
Door to environment through railroad access opens due to overpressure	19.7
Time Iodine release to environment exceeds 1% of initial core inventory	20.0
Calculation terminated	96

Next of the calc change the main vol to reflect

I thought the ORIGEN calc was changed to 48 hrs when did you get the decay heat for the

Figure 20 indicates smaller release to the environment when head bolt stretching begins - leaking at the flange explosion

The absence of water on the drywell floor in a transient scenario like station blackout[10] allows core debris ejected from the reactor vessel after lower head failure to spread laterally across the floor and contact the drywell wall. Past calculations have predicted drywell liner melt-through to occur relatively soon after vessel failure (within 30 minutes.) Fission product release from the containment to the reactor building and (with a very short delay) to the environment will begin at this point in time. Several release points to the environment are possible, depending on the response of the reactor building. Past calculations have shown hydrogen combustion leads to near-simultaneous opening of the refueling bay blow-out panels and the railroad doorway at grade level. Blowout panels into the turbine building and personnel access doorways out of the reactor building might also open. The dominant flow path for fission products to the environment, however, is expected to be through the refueling bay blowout panels[11].

5.1.1 Thermal Hydraulic Response

When plant conditions are stabilized, Special Event Procedure SE-11 calls for a controlled depressurization of the reactor pressure vessel (RPV) to 125 psig using the instructions in the RC/P leg of Trip Procedure T-101. Depressurization would be accomplished by opening one or more SRVs or, if necessary, by manually opening other steam vent pathways, such as main steam line drains. The cooldown rate would be limited to less than 100°F/hr. A controlled depressurization is initiated at one hour by opening a single SRV. As shown in Figure 14, this results in a stable pressure of approximately 125 psig[12]. Reactor vessel pressure remains near this pressure for approximately two hours, while active DC power permits an SRV to hold in the open position. Four hours into the scenario, however, DC power from the station batteries is exhausted and the solenoid valve regulating control air to the SRV operator closes, causing the SRV itself to reclose[13]. SRV closure causes reactor vessel pressure to gradually increase back to its automatic (safety) lift setpoint. Reactor vessel pressure subsequently cycles about its lift setpoint for the next 5 hours.

During this same time frame (i.e., the first 12 hours of the accident scenario) reactor vessel water level is also undergoing significant changes (refer to Figure 15.) The hydraulic transient immediately following reactor scram and isolation results in a gradual decrease in water level due to coolant evaporation and discharge through a cycling SRV to the suppression pool. RCIC automatically starts 10 minutes after the initiating event and begins to restore reactor water level. Two hours into the scenario, operators take manual control of RCIC and maintain level within the indicated range of +5 to +35 inches, i.e., 16 ft above top of active fuel (TAF).

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[10] As opposed to a loss-of-coolant accident (LOCA), where reactor coolant effluent accumulates on the drywell floor.

[11] A stable flow of air into the building is expected through the open railroad doorway, upward through the open equipment hatches from grade level to the refueling bay and into the environment through the open blowout panels.

[12] The target value of RPV pressure provides some margin above the RCIC isolation pressure of 75 psig.

[13] Loss of control air pressure to the valve operator might take a few minutes to effect valve position, but this short time is ignored in this analysis.

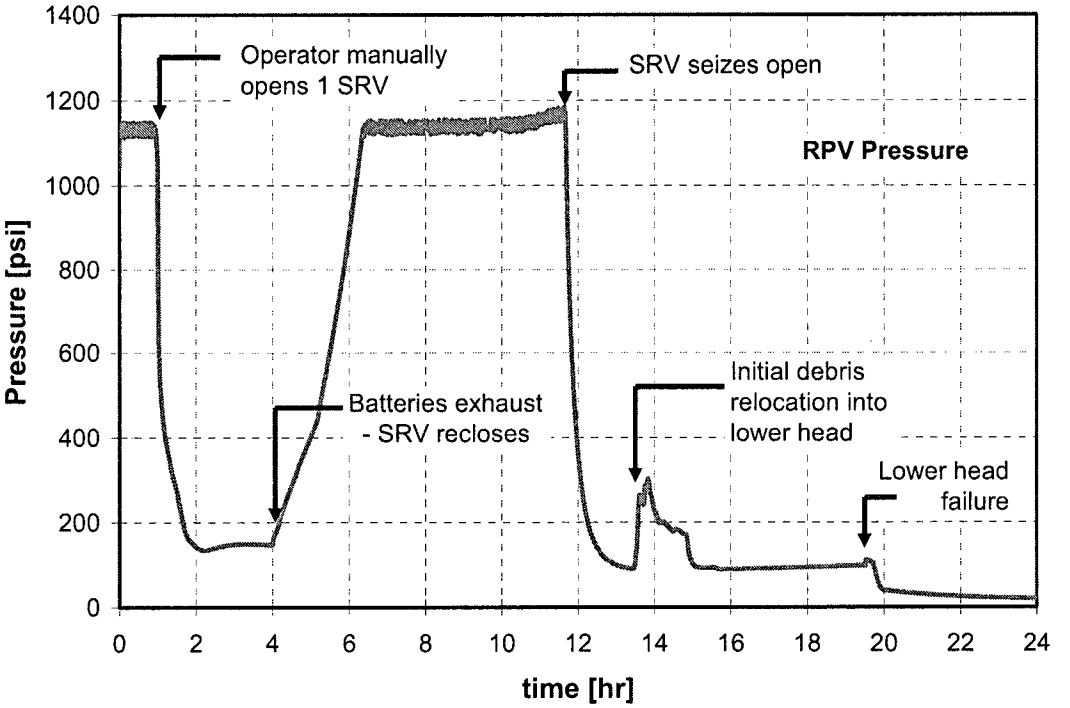

Figure 14 LTSBO Vessel Pressure

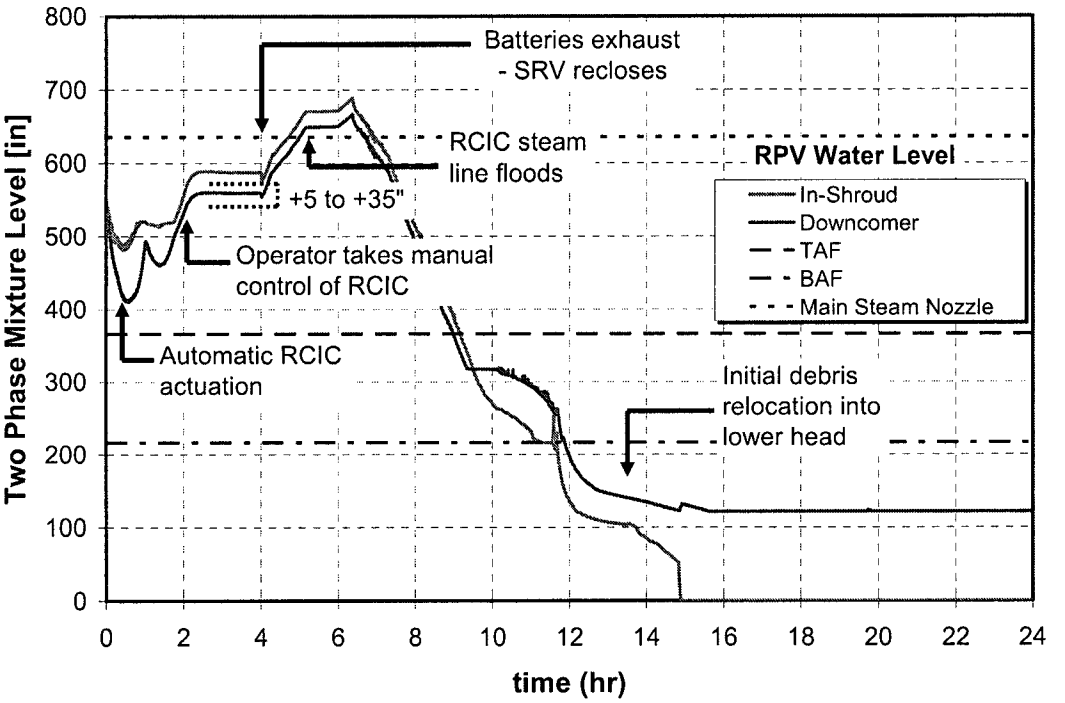

Figure 15 LTSBO Coolant Level

When DC power from the station batteries expires four hours into the scenario, RCIC turbine speed is assumed to remain fixed at its last position. [Electric (DC) power is required to move the turbine inlet throttle valve (open or close), and the loss of power simply leaves the valve in its last controlled position.] As a result, RCIC continues to deliver coolant flow at approximately the same flow rate it had at the time DC power expired. However, closure of the SRV at 4 hours means coolant losses from the reactor vessel are temporarily terminated. Therefore, the reactor vessel level begins to rise (i.e., coolant injection continues, but losses are terminated.) A continuous rise in level is evident in Figure 15 between 4 hours and approx. 5.2 hours.

At 5.1 hours, the water level in the reactor vessel increases above the elevation of the main steam line nozzles. Water subsequently spills over into the main steam lines causing the steam line to the RCIC turbine to flood within a few minutes. The resulting termination of RCIC operation at 5.2 hours causes the reactor water level to stabilize. Approx. 50 minutes later the average water temperature in the reactor vessel increases to saturation. When that occurs (6.0 hours), the reactor vessel pressure is 900 psia and increasing. Increasing reactor vessel pressure causes a slight increase in the effective level of water in due to decreasing average coolant density[14]. At 6.4 hours, reactor vessel pressure returns to the SRV lift pressure and coolant losses through the cycling SRV resume. Without any form of coolant makeup, the reactor water level continuously decreases at a rate of 10 ft/hr. Nine hours into the scenario, the reactor water level reaches TAF. At approximately twelve hours, the level decreases below the bottom of the lower core plate. By the time the plant has been without power for fifteen hours, the entire inventory of water in the reactor vessel evaporated (see Figure 15 and Table 3).

The thermal response of fuel in the core is illustrated in Figure 16, which shows the calculated temperature of fuel cladding across the core mid-plane. Cladding temperatures begin to rise at the top of the core when the mixture level decreases below approximately two-thirds of the core height. Within two hours, the mixture level is approaching the bottom of the core and fuel temperatures, and the extent of Zircaloy cladding oxidation, are sufficiently high to cause fuel at the top of the core to fragment and relocate toward the lower core plate as rubble.

In the midst of the core damage process, the cycling SRV is discharging a mixture of steam and hydrogen (from clad oxidation) to the suppression pool. The temperature of these gases increases along with the average temperature of fuel and debris near the top of the core. By 11.5 hours, the temperature of gases discharged through the SRV exceeds 1000 K. Thermal expansion of valve internal components above this temperature results in valve seizure. The valve is assumed to seize in the open position after 10 cycles above 1000 K. This occurs at 11.7 hours, and results in rapid depressurization of the RPV (see Figure 14) and a sharp decrease in mixture level (see Figure 15.)

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[14] The density of saturated water decreases by 4-5% as pressure increases from 900 psia to 1150 psia. This causes the entire body of water within the core shroud to expand slightly, resulting in an increase in effective (swollen) water level.

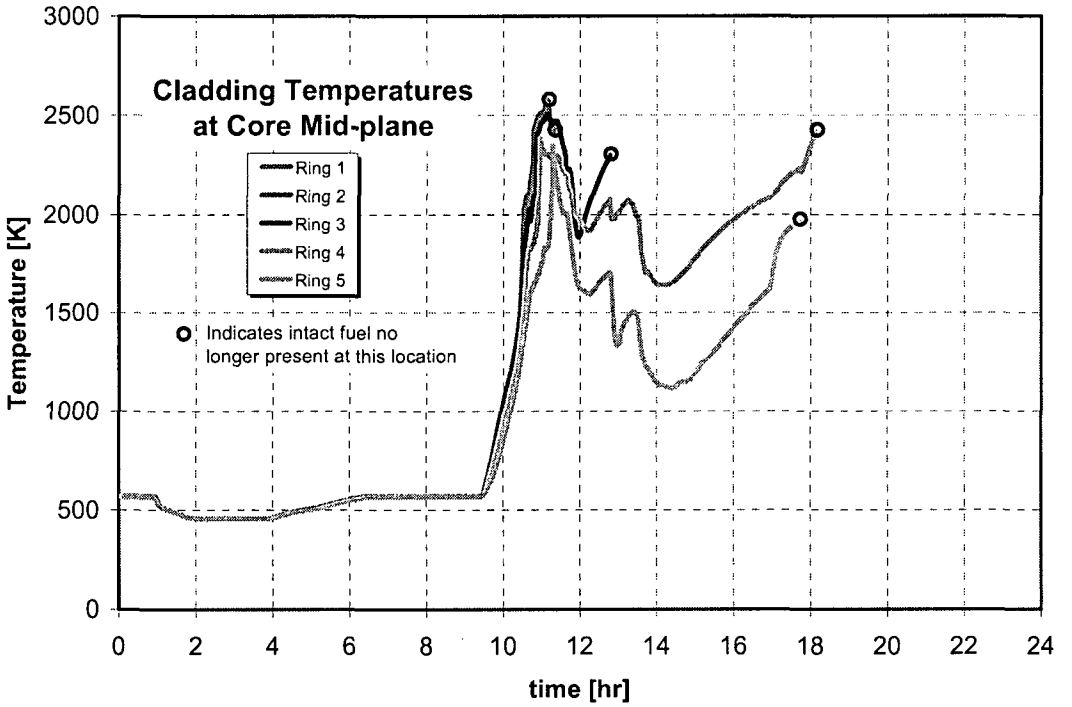

Figure 16 LTSBO Fuel Cladding Temperatures at Core Mid-plane

Particulate and molten debris accumulate near the bottom of the core until 13.5 hours, when the lower core plate yields, releasing core debris into the reactor vessel lower head. The interaction between hot debris and residual water in the lower head increases the rate of coolant evaporation, as indicated in Figure 15 by the increase in (negative) slope of the "in-shroud" water level. It also causes the temperature of debris submerged below the lower plenum mixture level to decrease to near-equilibrium conditions. This is evident in Figure 17, which shows the calculated temperature of debris along the inner surface of the lower head. When residual water in the lower plenum is completely evaporated at 15 hours, debris temperatures begin to increase. Heat transfer from debris to the inner surface of the lower head causes the lower head temperature to increase as well. This is illustrated in Figure 18, which depicts the calculated temperature on the inner and outer surfaces of the lower head across all five rings of the MELCOR model. Because reactor vessel pressure is relatively low during this heat up, the failure of the lower head is more strongly influenced by thermal rather than mechanical stresses.[15]

Failure of the lower head (at 19.5 hours) results in the rapid ejection of over 100 metric tons of core debris onto the floor of the reactor pedestal in the drywell. The composition of this debris

[15] The inner surface temperature of the nadir of the lower head (MELCOR rings 1-3) is above the melting point of steel at the time failure occurs.

(at the time of head failure) is a mixture of molten stainless steel (~30% by mass), unoxidized Zirconium (~11%) and particulate debris containing UO_2 and metallic oxides (remainder).

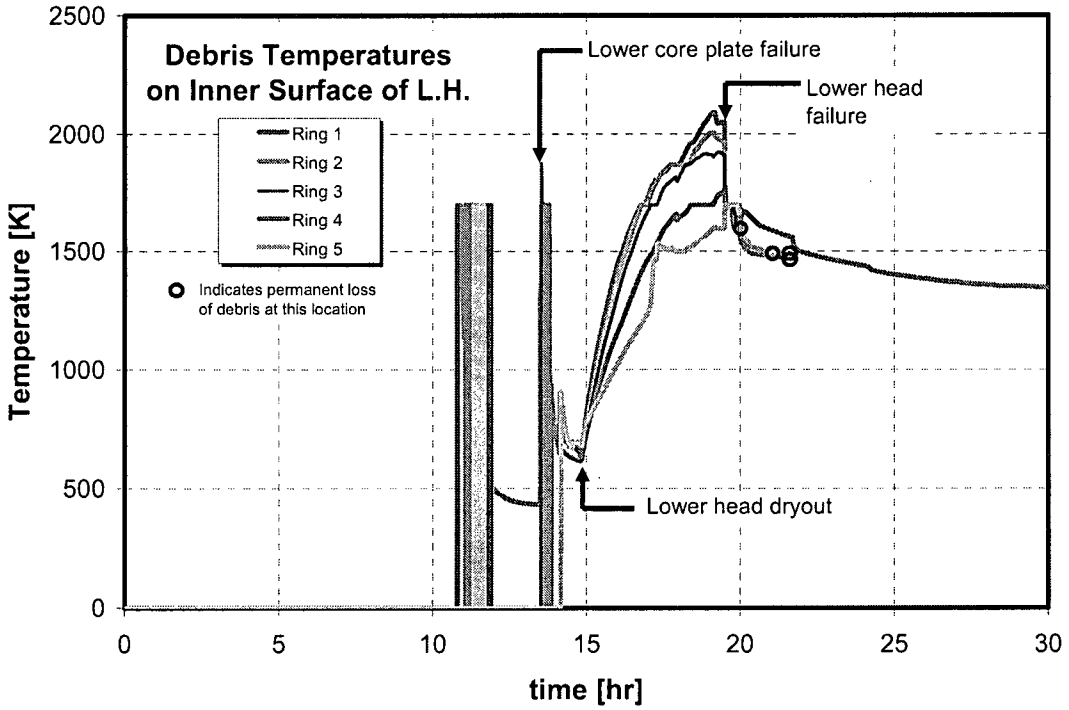

Figure 17 LTSBO Temperature of Particulate Debris on Inner Surface of Lower Head

Prior to the time at which the reactor vessel lower head fails, thermodynamic conditions in the containment are governed by the gradual release of hydrogen through the SRV to the torus. The large quantity of hydrogen (over 1300 kg between 10 and 19 hours), combined with the small free volume of the containment, result in significant increases in pressure. The containment pressure history is shown in Figure 19. Thirteen hours after the initiating event (8 hours after the loss of all coolant injection), the containment pressure increases above the design pressure of 56 psig. Immediately prior to lower head failure (19.5 hours), containment pressure exceeds 76 psig.

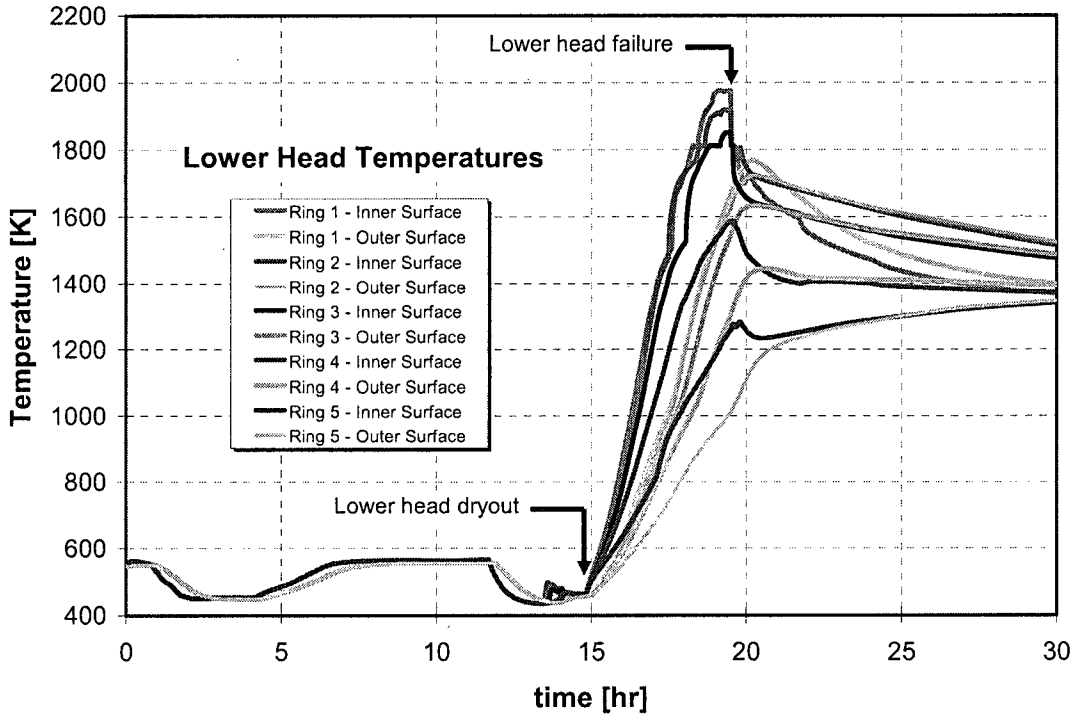

Figure 18 LTSBO Lower Head Temperature

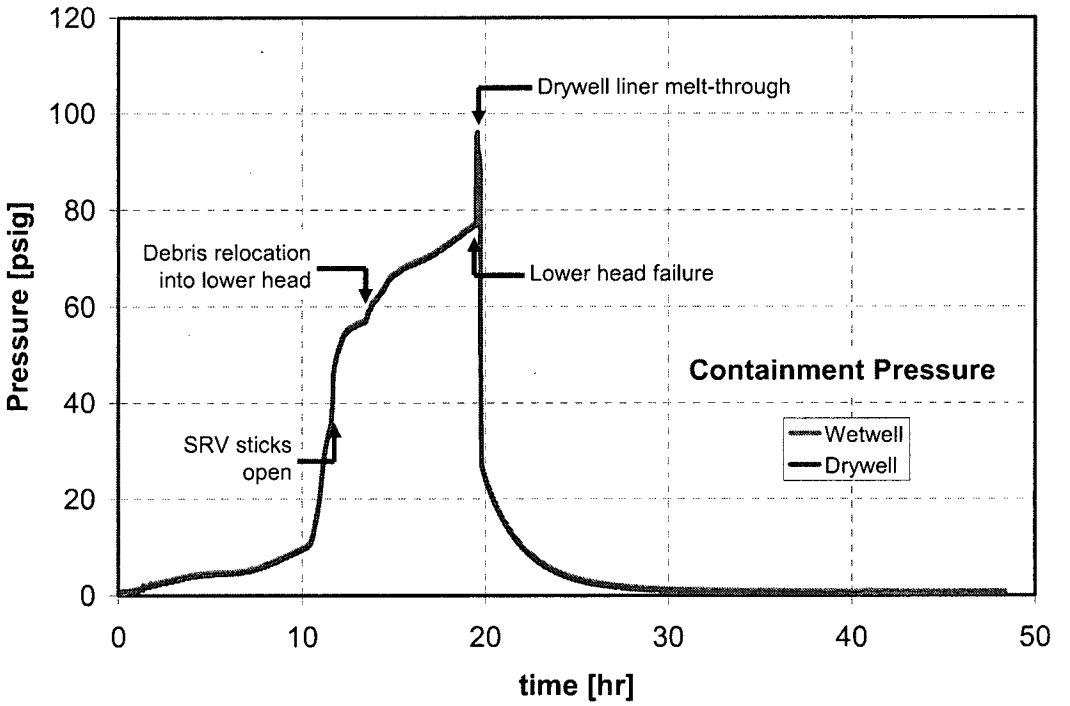

Figure 19 LTSBO Containment Pressure

Soon after debris is released onto the reactor pedestal floor, it flows laterally out of the cavity through the open personnel access doorway and spreads out across the main drywell floor. Lateral movement and spreading of debris across the drywell floor allows debris to reach the steel liner at the outer perimeter of the drywell within 10 minutes. Five minutes later, thermal attack of the molten debris against the steel liner results in liner penetration and opening of a release pathway for fission products into the basement (torus room) of the reactor building. The combined leakage through the drywell head flange and the ruptured drywell liner results in a rapid depressurization of the containment to approximately 25 psig, then a gradual long term depressurization, primarily through the opening in the drywell liner[16]. Before drywell liner melt-through occurs, hydrogen leaks through the drywell head flange and accumulates in the reactor building refueling bay[17]. Within a few minutes, a flammable mixture develops and is assumed to ignite. The resulting increase in pressure within the building causes the blowout panels in the side walls of the refueling bay to open, creating a release pathway to the environment.

Following drywell liner melt-through (several minutes later), hydrogen is released from the drywell into the basement of the building (i.e., torus room), and is transported upward through open floor gratings into the ground level of the reactor building. Flammable mixtures quickly develop in these regions, which are assumed to ignite. The pressure rise within the building at this lower location causes several doorways within the building to open, including the large equipment access doorway. This large opening at grade level, coupled with the open blowout panels in the refueling bay (at the top of the building) create an efficient transport pathway for material released from containment to the environment. That is, a vertical column of airflow is created within the building, whereby fresh air from outside the building enters through the open equipment doors at grade level, rises upward through the open equipment hatches at every intermediate floor within the building, and exits through the blowout panels at the top of the building. As will be shown in the next section, this 'chimney effect' reduces the effectiveness of the reactor building as an area for fission product retention.

5.1.2 Radionuclide Release

The release of radionuclides that immediately accompanies containment failure is shown in Figure 20. This release occurs in two steps, due to sequential breaches in the containment boundary by two distinct failure modes. The first appearance of significant release to the environment begins at 19.5 hours, when leakage through the drywell head flange begins. The leak area associated with this failure mode is relatively small. Therefore, the leak rate is low and the initial radionuclide release to the environment is relatively slow. Within 15 minutes, however, a larger leak area develops due to melt-through of the drywell liner. A sharp increase in the release rate is shown in Figure 21 (at 19.7 hours), when this second failure mode occurs.

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[16] Reduction in drywell internal pressure cause the drywell head flange leak pathway to reclose.

[17] The precise leak pathway includes intermediate transport through the drywell head flange to the drywell head enclosure. Leakage from the enclosure into the refueling bay occurs through gaps in the concrete shield blocks on the refueling bay floor. This complex leak pathway is explicitly represented in the MELCOR model.

The long-term release of radionuclides to the environment is shown in Figure 21. Following the 'puff' release that accompanies containment failure, a steady and gradual increase in the total quantity of radionuclides released to the environment is observed. The gradual, long-term increase in release is caused by two processes. First, molten corium-concrete interactions (MCCI) on the drywell floor drive the residual quantity of volatile fission products from fuel debris, and release a relatively small fraction of all non-volatile species. Second, the combination of high drywell atmosphere temperatures generated as a byproduct of MCCI and heating of reactor vessel internal structures due to decay heating of deposited radionuclides results in a late revaporization release of volatile species from within the containment and reactor coolant system. The latter is described in greater detail below.

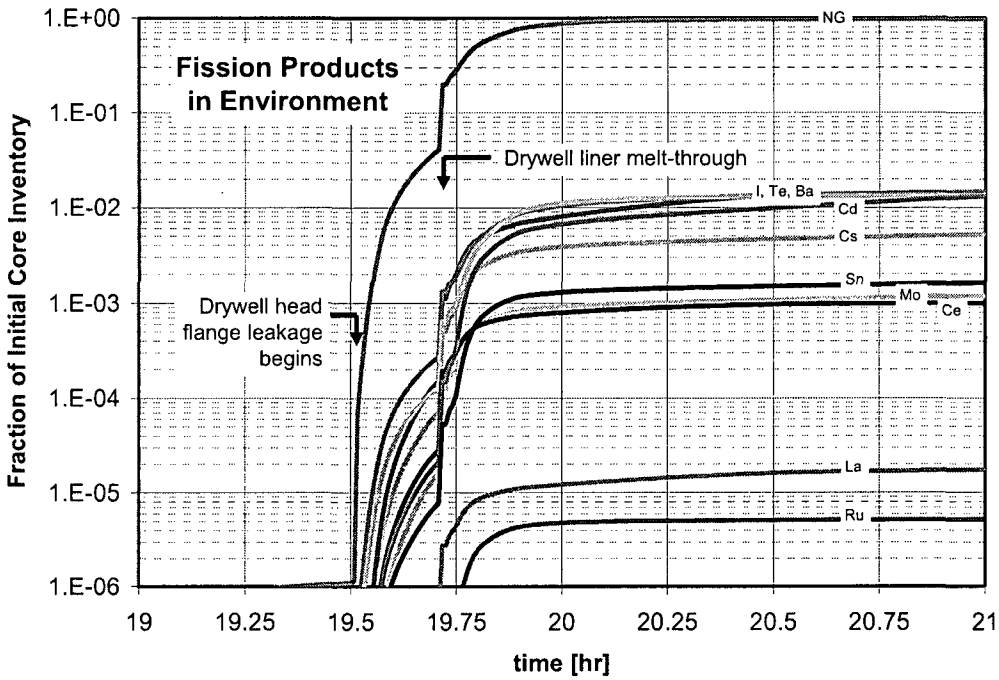
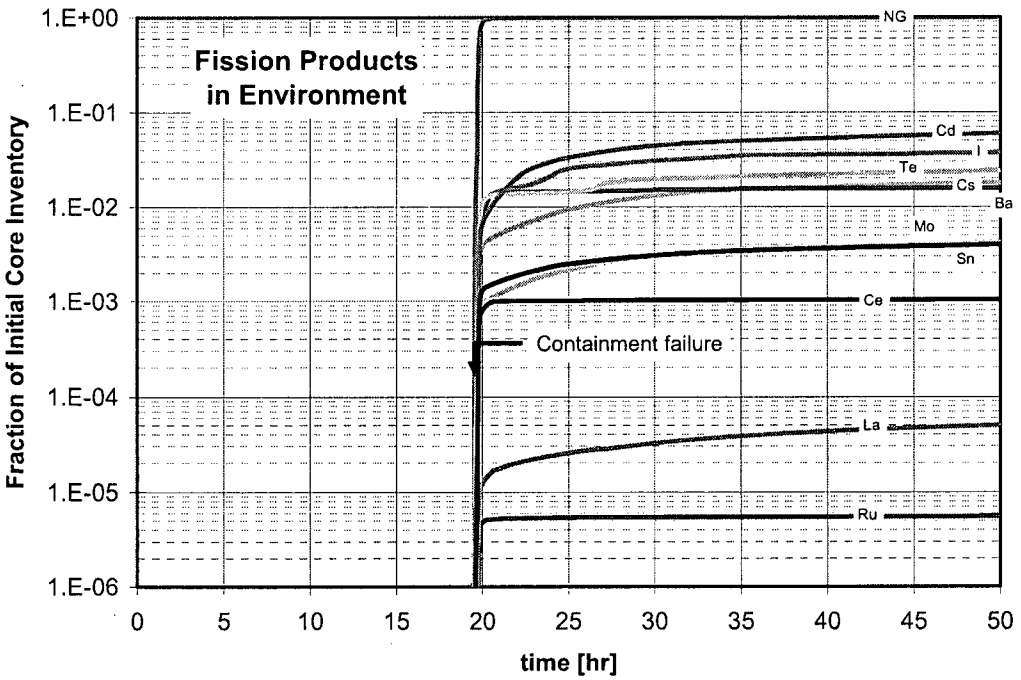


Figure 20 LTSBO Environmental Source Term: Detail at Time of Containment Failure



If you carried out the calc to 96 hours (Table 3) plot it all

Figure 21 LTSBO Environmental Source Term: Long term

Figure 22 depicts the fraction of the initial iodine inventory that is captured in the suppression pool, deposited or airborne within the reactor pressure vessel (RPV), in the drywell, and released to the environment as a function of time. Similar information is shown in Figure 23 for cesium, Figure 24 for tellurium, and Figure 25 for non-volatile cerium.

Collectively, these figures provide useful information about the mobility of different radionuclide species and temporal changes in their spatial distribution. For example, next to noble gases, iodine is the most volatile radionuclide group. In the SOARCA calculations, iodine is assumed to be transported in the form of CsI, which vaporizes at relatively modest temperatures for a severe accident. As a result, CsI is released from fuel during the early phases of in-vessel core damage progression and a significant fraction remains airborne due to relatively high temperatures of structures within the reactor vessel. Airborne iodine is efficiently transported to the wetwell through the operating SRV. In particular (see Figure 22), approximately 60% of the initial core inventory of iodine is discharged to the suppression pool during the blowdown of the reactor vessel that accompanies SRV seizure at 11.7 hours. During the succeeding eight hours, the majority of CsI that remains deposited on reactor vessel internal structures after RPV blowdown evaporates from their surfaces due to decay heating, and is also carried into the suppression pool.

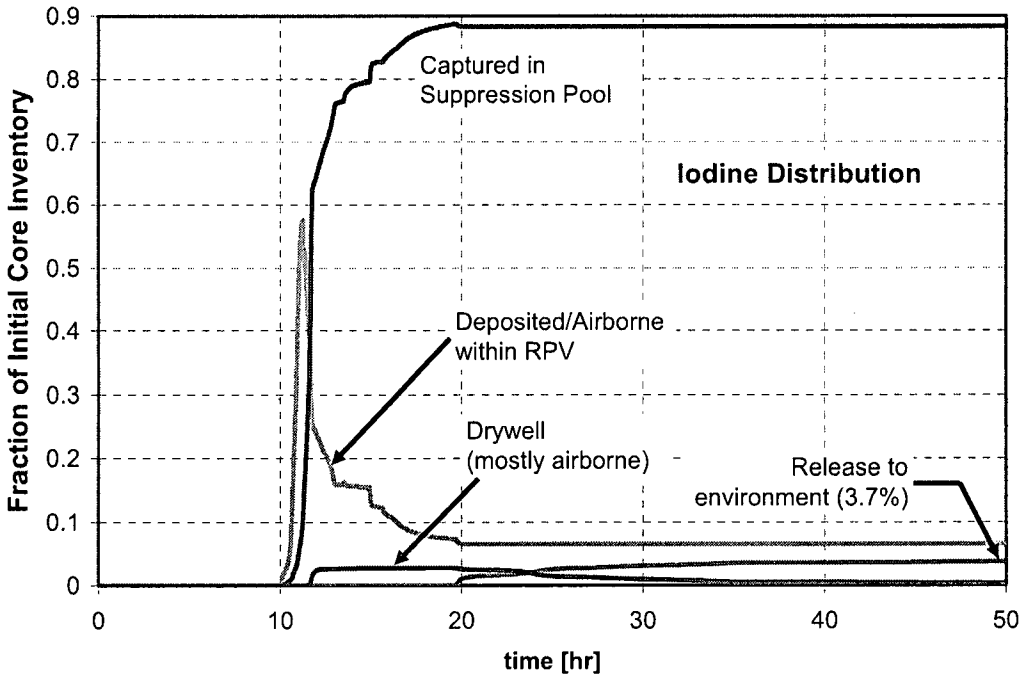


Figure 22 LTSBO Iodine Fission Product Distribution

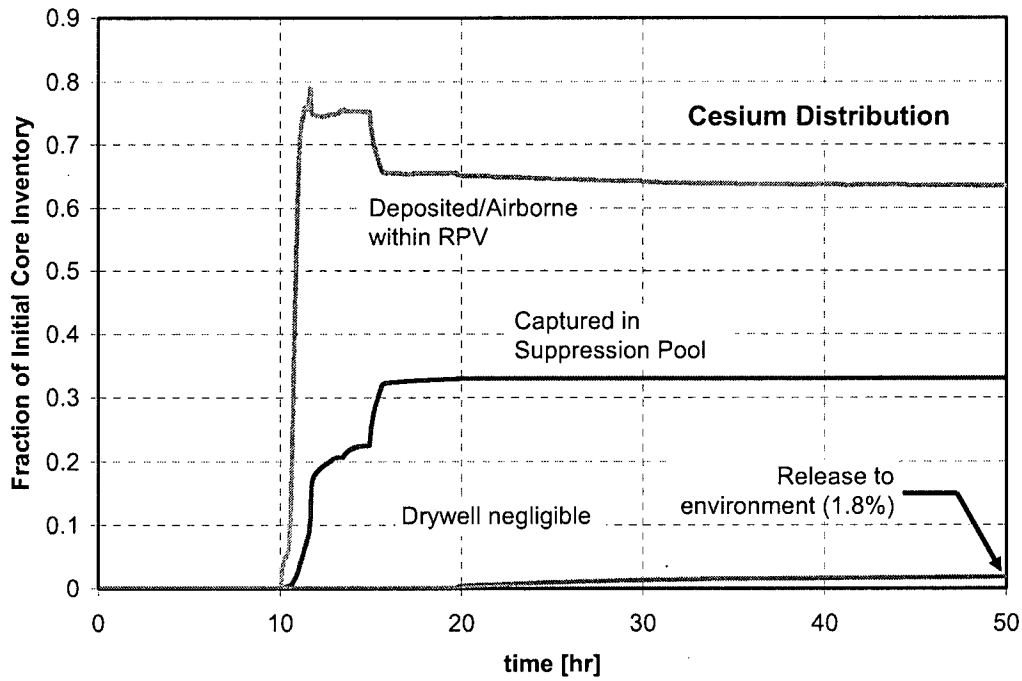


Figure 23 LTSBO Cesium Fission Product Distribution

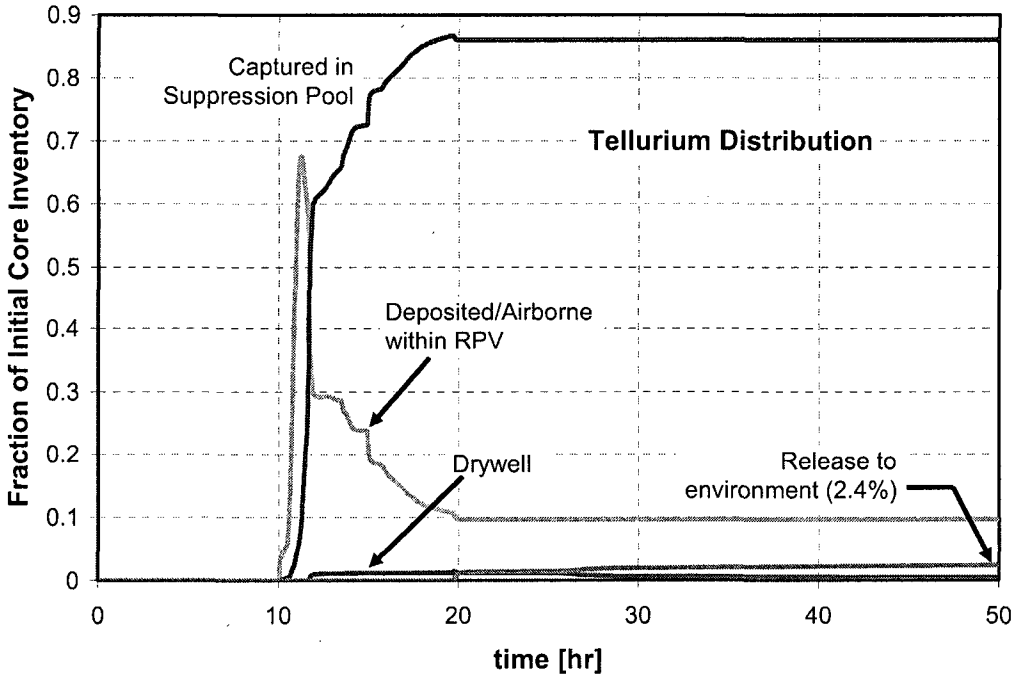


Figure 24 LTSBO Tellurium Fission Product Distribution

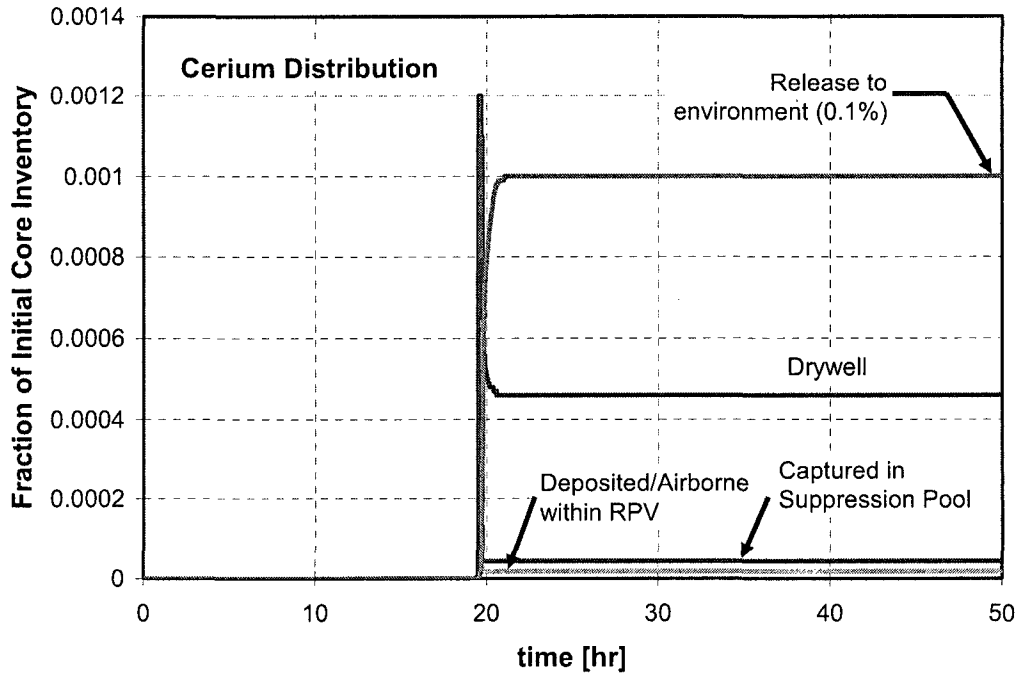


Figure 25 LTSBO Cerium Fission Product Distribution

A small fraction (few percent) of iodine enters the drywell atmosphere at 11.7 hours (i.e., during RPV blowdown) due to incomplete scrubbing in the suppression pool. The high flow rate, combined with the high non-condensable (hydrogen) fraction of carrier gas, reduces scrubbing efficiency during this brief period of iodine transport to containment. This iodine initially deposits on drywell surfaces, but revaporizes when corium-concrete interactions begin after lower head failure. Late revaporization of the small amount of iodine in the drywell is the primary source of iodine to the environment.

Temporal changes in the spatial distribution of cesium (Figure 23) differ from those observed for iodine. First, a much larger fraction of the cesium inventory remains deposited on in-vessel structures during the early phase of in-vessel damage progression than is observed for iodine. When reactor vessel blowdown occurs at 11.7 hours, a significant, but smaller, fraction of cesium is airborne in the vessel atmosphere. Therefore, a smaller quantity is promptly swept into the wetwell following SRV seizure. In contrast to iodine, for which nearly 60% of the initial core inventory is swept into the suppression pool during RPV blowdown, less than 20% of the cesium inventory is transported to the torus at the same time. Revolatilzation and transport of deposited cesium to the suppression pool prior to vessel breach is also less than that observed for iodine. Approximately 33% of the cesium is transport to the pool prior to lower head failure, whereas nearly 90% is observed for iodine.

These differences in iodine and cesium behavior can be attributed to differences in the physical properties of their dominant chemical forms. As mentioned earlier, iodine is transported as CsI. The cesium contribution to CsI represents only 6% of the total cesium inventory. The vast majority (approx. 90%)[18] of the cesium inventory is transported in the form of cesium molybdate (Cs_2MoO_4). Cesium molybdate is less volatile than the iodide and remains deposited on in-vessel structures at significantly higher temperatures. The in-vessel temperature history calculated for the long-term station blackout creates a thermal environment that promotes the evaporation of CsI relative to that of Cs_2MoO_4. Therefore, iodine is preferentially transported to the torus, but cesium remains deposited on in-vessel structures.

The suppressed mobility of cesium compared to iodine also affects the ultimate quantity transported to environment. Because the amount of cesium swept into the suppression pool during reactor vessel blowdown at 11.7 hours is a small fraction of the total core inventory, carry-over into the drywell atmosphere (due to inefficient pool scrubbing) is negligible. Therefore, the amount of cesium in the drywell atmosphere at the time of containment failure (19.5 hours) is also very small. In contrast to the iodine release, which is dominated by an early 'puff' release immediately accompanying containment failure, the cesium release is characterized by a small, protracted release that begins after containment failure. The primary mechanism for this long-term release is the slow revolatilization of cesium deposited on RPV internal surfaces.

The behavior of tellurium (Figure 24) is similar to that described above for iodine, and is not described in further detail here. Release of the heavy non-volatile species (cerium, for example) differs substantially from the trends described above for volatile species. As indicated in

[18] The remaining fraction is cesium located in the fuel-cladding gap.

Figure 25, the release of these radio-elements does not begin until after vessel breach, when MCCI occurs on the drywell floor. Release of cerium and other non-volatile species (La and Ru, for example) from fuel debris begin soon after vessel breach when MCCI is most aggressive. As indicated in Figure 26, the temperature of ex-vessel debris decreases significantly as it spreads across the drywell floor from its initial point of arrival in the reactor pedestal. This greatly reduces the rate at which the non-volatile species are released.

The release of Ru is also governed by the oxidation state. The drywell is inerted, Ru release is low and when Pedestal, a hole in the drywell that means let in oxygen, the increase the release, the temperature is lower.

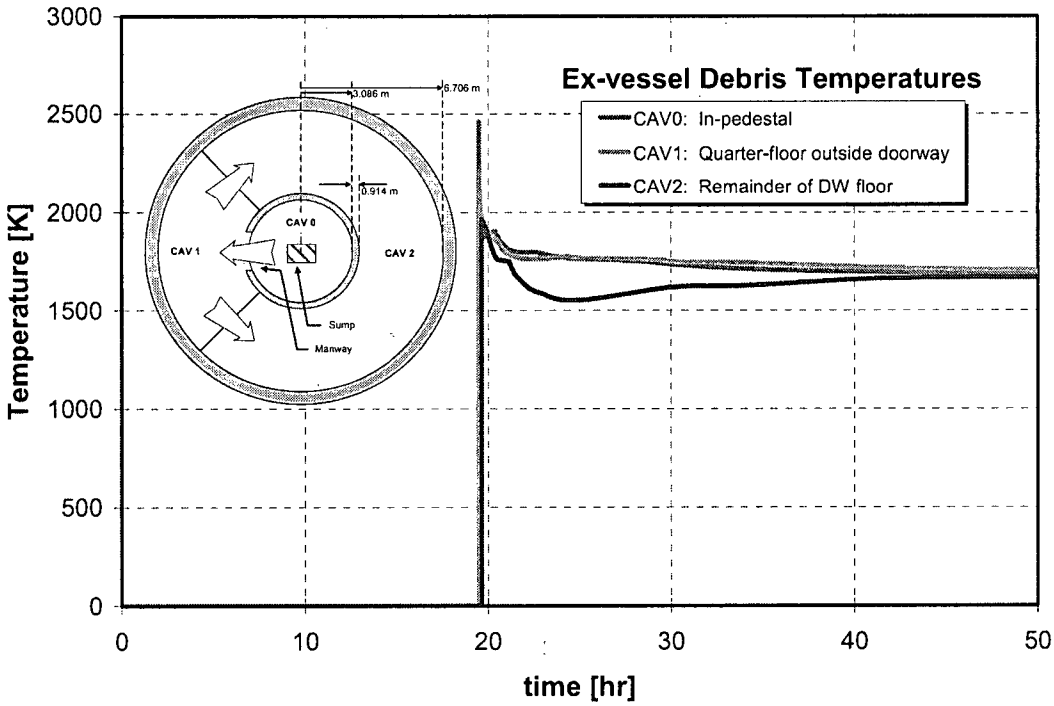


Figure 26 LTSBO Ex-vessel Debris Temperatures

5.2 Long-Term Station Blackout – Mitigated Response

The key events for LTSBO with mitigative actions (discussed in Section 3.1.3) are listed in Table 4.

Table 4 Timing of Key Events for Mitigated Long-Term Station Blackout

Event (Time in hours unless noted otherwise)	**Mitigated LTSBO with 4 hr dc power**
Station blackout – loss of all onsite and offsite AC power.	0.0
Automatic reactor scram and containment isolation	0.0+
Low-level 2 and RCIC actuation signal	10 minutes
Operators manually open SRV to depressurize the reactor vessel	1.0
RPV pressure first drops below LPI setpoint (400 psig)	1.2
Operators take manual control of RCIC; flow throttled to maintain level within range (+5 to +35 in)	2.0
Portable electric generated positioned, started and connected to remote panel	1.0 to 4.0
Station batteries depleted	4.0
Operators position, align and start portable pump to replace RCIC as injection source	4.0 to 10.0
High suppression pool temperature isolation signal for RCIC	10.0
Calculation terminated	24

5.2.1 Thermal Hydraulic Response

Like the unmitigated case, the operator manually opens a safety/release valve (SRV) to reduce pressure in the reactor pressure vessel (RPV). When the station batteries are exhausted, a portable power supply is engaged to sustain the open safety/release valve (SRV) in the mitigated case. This maintains the reactor pressure vessel (RPV) at a stable pressure at or above 125 psig as directed in the Special Event Procedure SE-11. This is shown in Figure 27.

The coolant level history for the mitigated long-term station blackout is plotted in Figure 28. The core temperature history for the mitigated long-term station blackout is shown in Figure 29. No plot was included for the long-term station blackout lower head temperature history because the mitigated case does not result in core damage. The curve would be 'flat-line' at nominal shutdown conditions. The containment pressure history for the mitigated long-term station blackout is shown in Figure 30. The operator actions are labeled in the plot.

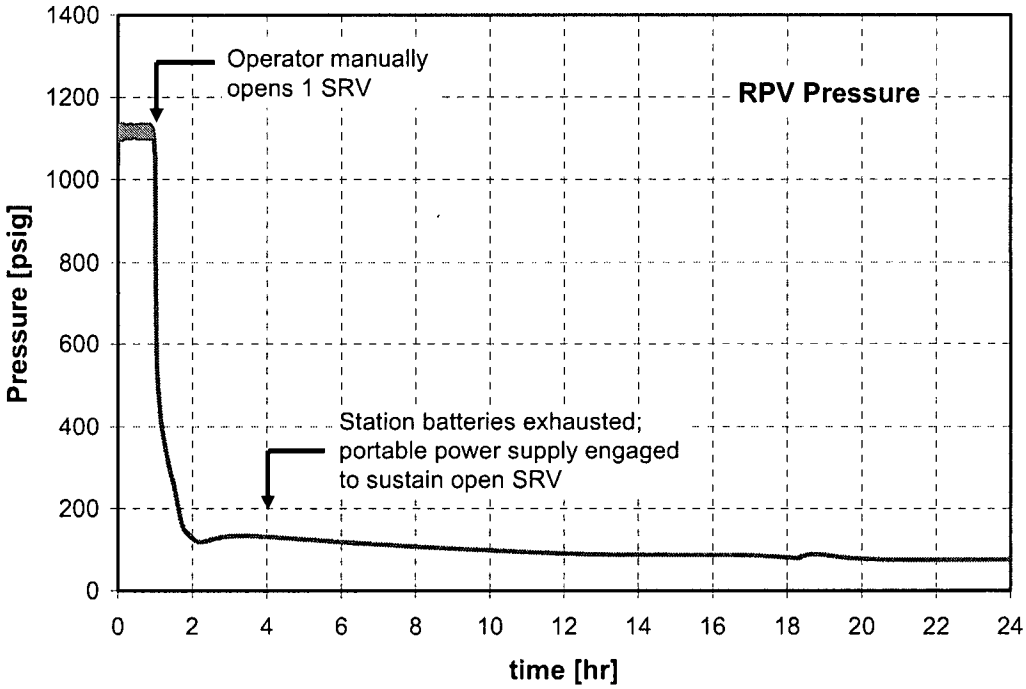


Figure 27 Mitigated LTSBO Vessel Pressure

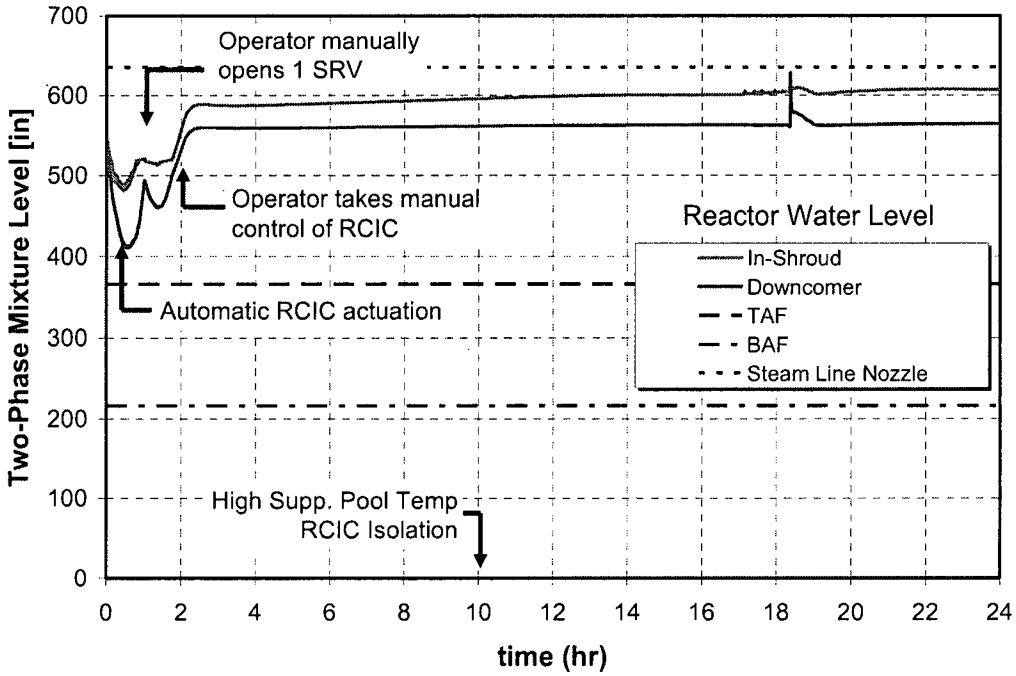


Figure 28 Mitigated LTSBO Coolant Level

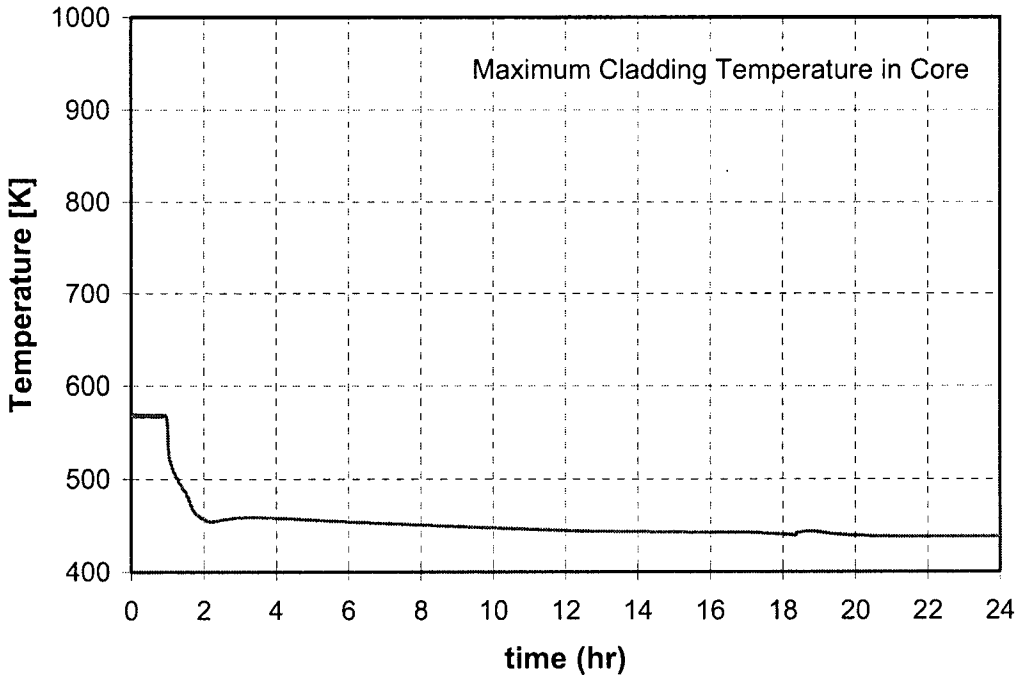


Figure 29 Mitigated LTSBO Core Temperature

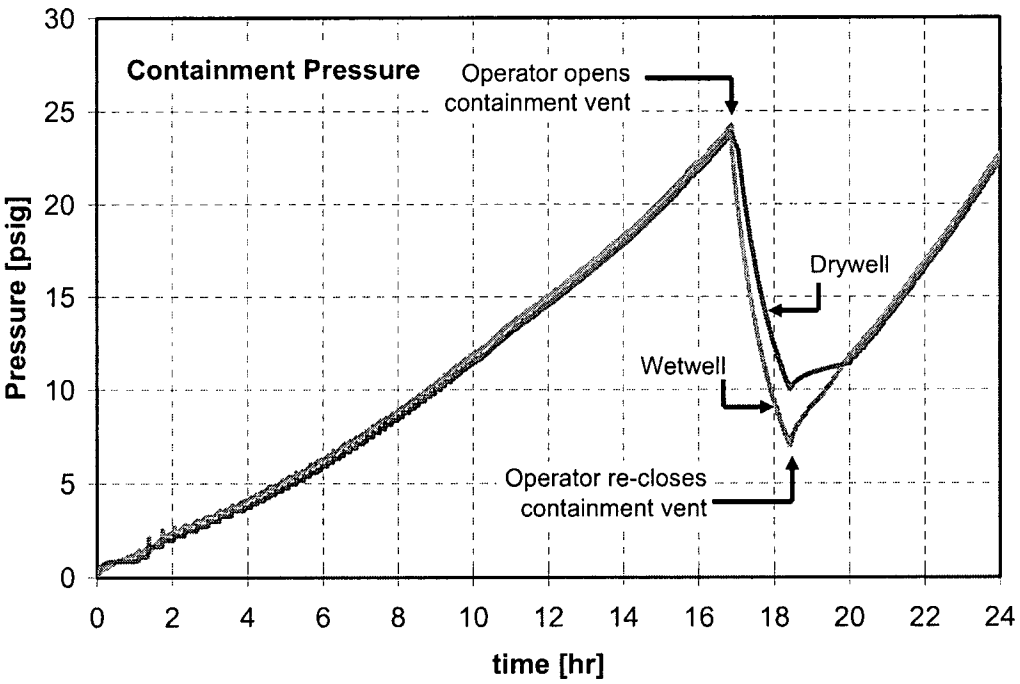


Figure 30 Mitigated LTSBO Containment Pressure

5.2.2 Radionuclide Release

No plots were included for the iodine fission product distribution history, cesium fission product distribution history, barium fission product distribution history, cerium fission product distribution history, or environmental release history of all fission products resulting from mitigated long-term station blackout because the mitigated case does not result in core damage. All of the curves would be 'flat-lines' at nominal shutdown conditions.

5.3 Short-Term Station Blackout – Unmitigated Response

The general response of plant equipment and operating personnel to the STSBO closely resembles the 'unmitigated' LTSBO scenario. Therefore, the reader is referred to Section 5.1 for a description of the actions plant personnel would take in response to this type of event. A key difference, however, is the early failure of DC power, which significantly reduces the time available for intervention, and accelerates the timeline of damage progression.

The accelerated event chronology is evident in Table 5, which indicates the onset of core damage (measured as the first time at which fuel cladding fails) occurs approximately one hour after the initiating event in the short-term scenario, whereas the same condition occurs eight hours later in the long-term scenario where station batteries (DC power) ensure coolant makeup for four hours[19]. Late phases of in-vessel damage progression also proceed at a relatively rapid pace due to the higher levels of energy retained in the core. For example, reactor vessel dryout occurs approximately one hour after core debris relocates into the lower plenum in the short-term scenario, whereas it takes nearly five hours in the long-term scenario. As noted later, these differences have a relatively minor impact on the quantity of activity released to the environment; but it does impact the time at which a release begins, and therefore may impact the assessment of offsite consequences.

5.3.1 Thermal Hydraulic Response

The initiating event causes a prompt failure of all AC and DC power supplies to plant equipment and instrumentation. Reactor control blades, MSIVs and containment isolation valves would all move to their fail-safe positions (inserted and closed). Isolation of the reactor coolant system causes reactor pressure to rise to the set point of the SRVs, which open, directing coolant to the pressure suppression pool. As shown in Figure 31, reactor pressure is maintained at approximately 1120 psia, as the SRV with the lowest set point cycles open/closed for approximately two hours[20]. Actions taken by plant operations personnel to manually reduce reactor pressure and prevent frequent cycling of the SRVs are assumed to not be successful. This is because control power to necessary equipment (e.g., SRV solenoid control valves) would not be available and manual actions to open alternative steam relief paths are assumed to be inhibited by obstacles preventing access to plant equipment (a result of the severity of the initiating event.)

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19 The delayed time to the onset of core damage in the long-term station blackout is not proportional to the duration of DC power or coolant makeup due to the non-linear change in core decay heat with time.

20 A second SRV periodically opens during the first 45 minutes of the transient, when decay heat levels remain high. However, after this point in time, only one valve is cycling.

Table 5 Timing of Key Events for the Unmitigated Short-term Station Blackout

Event	Time (hr)
Station blackout – loss of all onsite and offsite AC power	0.0
Low-level 2 and RCIC actuation signal	10 minutes
Downcomer water level reaches top of active fuel	0.5
First hydrogen production	1.0
First fuel-cladding gap release	1.0
First channel box failure	1.2
Reactor vessel water level reaches bottom of lower core plate	2.0
SRV sticks open due to excessive cycling	2.0
RPV pressure decreases below LPI set point (400 psi)	2.3
First core support plate localized failure in supporting debris	2.6
Lower head dries out	3.5
Ring 5 CRGT Column Collapse [failed at axial level 2]	5.5
Ring 3 CRGT Column Collapse [failed at axial level 2]	5.8
Ring 1 CRGT Column Collapse [failed at axial level 1]	5.9
Ring 4 CRGT Column Collapse [failed at axial level 1]	6.1
Ring 2 CRGT Column Collapse [failed at axial level 1]	6.1
Lower head failure (yield from creep rupture)	7.9
Drywell liner melt-through (leakage into torus room of reactor building)	8.2
Refueling bay to environment blowout panels open	8.2
Hydrogen burns initiated in torus room (basement) of reactor building	8.2
Door to environment through railroad access opens due to overpressure	8.2
Blowout panels from RB steam tunnel to turbine building open	8.2
Steel roof of ~~reactor building fails~~ due to over-pressure	8.4
Reactor Pedestal through-wall erosion	11.1
Time Iodine release to environment exceeds 1%	8.5
~~Total In-vessel H_2 production (kg)~~	~~1142.~~
Calculation terminated	48.0

what is this? The pedestal is eroded? Never discussed. Is it really drywell failure? Why include KG in an HR table? Yikes so important Pattern Tag! Jeff!

Two hours after the initiating event, the (single) cycling SRV sticks in the open position, causing a rapid depressurization of the reactor coolant system[21]. The continuous discharge of steam

[21] The time (or cycle) at which an SRV would fail to reclose is determined by calculating the cumulative probability of failure, based on the total number of cycles and the probability of failure on demand. The latter is taken from the Individual Plant Examination (IPE) for Peach Bottom, which reports a value of 3.7E-3 per demand. This value is larger than the industry average value of 8E-4/demand reported in NUREG/CR-6928, and is assumed to be representative of plant-specific performance. In the MELCOR