Letter Report

Evaluation of the Data Base for SBWR LOCA Events During the High Pressure Period of the Blowdown Phase

Peter G. Kroeger
Applied Technologies Division

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Department of Advanced Technology, Brookhaven National Laboratory
Upton, New York, 11973

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1.0 Introduction

The Simplified Boiling Water Reactor (SBWR), introduced by General Electric (GE), includes various unique features. Different from other current designs, it relies on natural circulation in the reactor vessel, aided by a chimney region on top of the core, as compared to forced circulation via jet-pumps. Isolation Condensers (ICs) located above the reactor vessel can provide for decay heat removal under accident or shut-down conditions. Its ECCS uses a Gravity-Driven Cooling System (GDCS). Long term heat removal is effected by the Passive Containment Cooling System (PCCS), which by natural circulation takes in gas from the drywell (nitrogen and steam), returning condensate to the GDCS pool and noncondensibles to the suppression chamber.

There have been natural circulation BWRs in the early years of BWR development, including the experimental Vallecitos BWR, Humboldt Bay Unit 3, and the Dodewaard BWR/1. Both, the Humboldt Bay and the Dodewaard reactors used a chimney to enhance natural circulation. They also used isolation condensers, similar to the SBWR, but apparently with horizontal tubes. These were smaller reactors, and there are significant design differences between these early reactors and the current SBWR concept. A comprehensive comparison of early natural circulation reactors and the current SBWR is given by Heath et. al., 1992.

This report presents an assessment of the existing data base for modelling of SBWR LOCA transients, considering the initial blowdown phase with vessel pressures between approximately 72 and 10 bar. At lower pressures a significant data base of General Electric sponsored tests applicable to the SBWR exists, or is currently being established. Table 1.1 summarizes the major experimental efforts, specifically supporting the SBWR development. Even though not specifically designed for the SBWR development, the older FIST facility was included in the table, since that test program included specific high pressure natural convection tests. A more detailed summary of the experimental program can be found in Kullberg and Fletcher, 1992 and also in GE-93-2.

For the high pressure range the applicant's position has apparently been, that previous BWR experience is sufficient for a complete understanding of early blowdown events. This position appears to be restricted to in-vessel events, and does not include the Isolation Condensers, for which high pressure tests are currently in progress (Masoni, et al, 1993).

To establish the data base requirements for the high pressure range of large and small break LOCA scenarios, the following procedure will be applied:

1. Several sets of Phenomena Identification and Ranking Tables (PIRTs) are available for SBWR LOCA scenarios. Apparently the first set was developed by GE (GE-92 and GE-93-1, see also Shiralkar et al, 1992). Independently Brookhaven National Laboratory (BNL) has developed SBWR LOCA PIRTs, following a similar, but not identical format

(BNL-93). PIRTs have come into use with the development of the Code Scaling, Applicability, and Uncertainty (CSAU) methodology (Boyack, et al., 1989). They serve to establish the important phenomena for specific reactor transients, ranked in order of importance. The ranking is established in meeting(s) by a team of experts on reactor accident transients. While no theoretical proof of the completeness of PIRTs is possible, the manpower extensive direct discussions of the detailed event scenarios by experts provides the highest possible assurance, that all important aspects will be covered. The available PIRTs from GE and BNL will be reviewed here, and the phenomena ranked as important during the high pressure blowdown phase will be extracted.

2. Many important phenomena clearly are not unique to the SBWR and occur in essentially identical form in previous BWR designs. In general, these will not require any further consideration, since the data base for their evaluation was established during the development of previous designs. The SBWR unique phenomena will be identified and grouped by reactor component. As pointed out in detail below, in some cases individual phenomena may not be unique to the SBWR, but their system interaction with other phenomena is. Special consideration will be given to such cases.

3. For all phenomena identified as unique to the SBWR, the data range during the transient will be established and an assessment of the currently available data base will be conducted, to determine, whether any further data are desired, or whether the current state of knowledge is sufficient for reliable and high confidence simulations of LOCA scenarios.

Following the evaluation of the PIRTs in Chapter 2, the SBWR unique phenomena will be discussed and compared to the data base in Chapter 3. A summary of the evaluations will be presented in Chapter 4.

Table 1.1 SBWR Specific Test Facilities

Test Facility	Location	Emphasis	Peak Pressure (bar)	Scale	Description	Status
FIST	GE USA	Reactor Vessel	72	full sized rod bundle	One full sized 8x8 rod bundle; downcomer with jet pump. Tests include natural convection test.	Completed. test results see Phase 1: NUREG/CR-3711, 1984. Phase 2: NUREG/CR-4128, 1985.
GIST	GE USA	GDCS, short term	10	Elevation 1:1 Volumes & Flow Areas 1:500	Reactor vessel with 45 simulated fuel rods; containment with drywell, wetwell, GDCS tank & injection line.	Completed. For results see NEDO-31680, 1989.
GIRAFFE	Toshiba Japan	PCCS, long term	4	Elevation 1:1 Volumes & Flow Areas 1:400	Reactor vessel, drywell, wetwell, PCCS, GDCS	Completed. For results see NEDC-32215P, 1993
PANDA	Paul Scherrer Institute Switzerland	PCCS (with multi-d effects), long term	10	Elevation 1:1 Volumes & Flow Areas 1:25	Reactor vessel, IC, drywell, wetwell, PCCS, GDCS	Testing pl[illegible]d for 1994/95
PUMA	Purdue University USA	Containment with PCCS & GDCS, all times	10	Elevation 1:4 Flow Areas 1:100 Volume 1:400	Reactor vessel, ICs, drywell, wetwell, PCCS, GDCS	Planning in progress; Ishii et al, 1994
PANTHERS	SIET Italy	PCCS & Isolation Condensers	PCCS 4 IC 95	Full size, flow, pressure & temperature	Full sized two-module PCCS & full sized single-module IC.[1]	Testing scheduled for 1994/95

[1] Actual SBWR IC & PCCS consists of two modules each

2.0 Evaluation of Available Phenomena Identification and Ranking Tables

2.1 General Electric PIRTs

Two apparently almost identical sets of PIRTs have been presented by General Electric (GE-92, GE-93-1, and Marquino and Munthe-Andersen, 1993). In general candidate phenomena ranked 7 and above, on a scale of 0 to 9, are considered to be important, and in case of doubt or disagreement a potentially lower ranked phenomenon is raised to 7. The GE PIRTs are quite detailed in the listing of phenomena in specific components, but divide the complete LOCA scenario into two time periods only, blowdown and refill. Since the current emphasis is on the high pressure range, only the blowdown phase of these PIRTs is of interest.

Table 2.1 lists the phenomena from the GE PIRTs ranked 7 or above for the blowdown phase of large and small break scenarios. The GE PIRTs also include other BWR designs (BWR/2, BWR/4, BWR/5-6 and ABWR). Therefore, the phenomena uniquely important in the SBWR can be readily identified, by comparing the SBWR rankings to those of the other designs. An exception was made in Table 2.1, in Category C12, Natural Circulation Flows. The ranking of natural convection in the core was high for all BWR designs. However, since the SBWR relies solely on natural circulation, and is designed for in-core natural circulation under accident conditions, it was felt, that this phenomenon should be identified as unique to the SBWR, and should be retained as candidate for further consideration.

The phenomena within the reactor vessel, which are identified as candidates for further consideration in Table 2.1, based on the GE PIRTs, are:

- Flashing in the Core
- Natural Circulation Flow in the Core
- Heat Conduction in Solid Structures in the Downcomer
- Flashing in the Downcomer
- Downcomer/Isolation Condenser Interactions
- Upper Plenum Void Distribution and Two-Phase Level.

The item "Natural Circulation Flow in the Core" here includes flow circulation through the reactor vessel, driven by buoyancy forces between the downcomer and the core with chimney, as well as core internal natural circulation, which can occur under LOCA conditions.

The containment items, identified as important and SBWR unique in the GE PIRTs of Table 2.1 are

- H_2O/NC Mixing in the Drywell and Suppression Chamber
- Flow Distribution with Low Flows at Small Pressure Drops

- Condensation Heat Transfer with Non-Condensibles (IC, PCCS & Walls)
- Conduction Heat Transfer in Composite Walls.

2.2 Brookhaven National Laboratory PIRTs

The BNL PIRTs for LOCA (BNL-93)[1] have a finer division into the following five separate time phases:

- Pre-Isolation
- Isolation
- Depressurization
- Refill from GDCS
- Long Term Cooling.

However, these early BNL PIRTs are in some cases less detailed in the treatment of phenomena and do not consider these in the same detail as GE, component by component. For instance, "Flashing" is considered as one phenomenon for the reactor vessel, while GE treated it in two separate entries, for the downcomer and for the core. Since the BNL PIRTs were done for the SBWR only, one cannot deduce directly from the tables, which items are unique to the SBWR. Judgement had to be applied, which was done in conjunction with other members of the BNL SBWR team. The BNL PIRTs were done for a main steam line break, a bottom drain line break, and a GDCS line break. From the results of these three PIRTs, a Composite PIRT was accumulated. A summary of this composite PIRT, giving the results for the time range of interest here, i.e. the first three time zones, is presented in Table 2.2 for the Reactor System and in Table 2.3 for the Containment System.

In both tables, the items ranked high, have been shaded, to highlight areas of importance. In assessing the uniqueness to the SBWR, it was felt, that, for instance, in-core flashing or pressure drop in themselves are by no means unique to the SBWR. However, with the high emphasis on natural circulation flows, between downcomer and core, as well as core internal recirculation, it was felt that the potential interaction between these phenomena may require further attention. Therefore, these items are labelled in the tables "syst int", for "system interaction". With tube side condensation in the isolation condensers it was felt that system interactions as well as condensation at high pressure require further attention.

The phenomena identified as candidates for further attention in the BNL PIRTs of Table 2.2 for the Reactor System can be summarized as

- Natural Circulation through the Vessel (Downcomer/Core Interaction)
- Core Internal Natural Circulation

[1]More detailed PIRTs are currently being developed at BNL; However, it is anticipated, that their results will not impact on our current evaluations.

- Isolation Condenser Tube Side Heat Transfer at High Pressure
- Isolation Condenser and Downcomer Flow Interactions.

The phenomena identified as candidates for further attention in the BNL PIRTs of Table 2.3 for the Containment System can be summarized as

- Non-Condensibles Distribution in the Dry Well
- Condensation with Non-Condensibles on Containment Structures
- Horizontal Vent Clearing
- Condensation with Non-Condensibles in Flow through Short L/D Horizontal Vent

The phenomena identified in the GE and BNL PIRTS, are generally identical, even though expressed in slightly different terms and maybe identified at different locations. All these items will be discussed further in Chapter 3.

Table 2.1 Abstract from GE PIRT for Blowdown Phase of LOCA (Items Ranked 7 and Above)

Category	Description	Accident Scenario: Small Break	Accident Scenario: Large Break	Uniquely Important in SBWR	Neglected in SSAR short-term LOCA analyses	Candidate for Further Investigation
A	**Lower Plenum**					
A1	Flashing / Redistribution		√	no		
A3	Two-Phase Level (SEO Uncovery Time)		√	no		
B	**Bypass**					
B1	Flashing	√	√	no		
B4	CCFL/CCFL Breakdown (Guide Tube-Bypass)	√	√	no		
B5	CCFL/CCFL Breakdown (Top of Bypass)		√	no		
B6	Channel-Bypass Leakage Flow	√		no		
C	**Core/Bundle**					
C4	Flashing	√	√	yes		√
C5	SEO Inlet Uncovery Vapor Flow Split	√	√	no		
C6	CCFL / CCFL Breakdown (SEO)	√	√	no		
C10	Void Distribution	√	√	no		
C11	Bundle-Bypass Leakage Flow	√		no		
C12	Natural Circulation Flows	√	√	yes (see text)		√
C13	Dryout / Boiling Transition		√	no		
C14	Film Boiling / Low Flow		√	no		
C26	Stored Energy	√	√	no		
D	**Guide Tube**					
D1	Flashing / Redistribution		√	no	yes	
D2	CCFL: Top of Guide Tube		√	no		

Table 2.1 Abstract from GE PIRT for Blowdown Phase of LOCA (cont'd) (Items Ranked 7 and Above)

Category	Description		Accident Scenario: Small Break	Accident Scenario: Large Break	Uniquely Important in SBWR	Neglected in SSAR short-term LOCA analyses	Candidate for Further Investigation
E	**Downcomer/Annulus**						
E1	Break Uncovery / Two-Phase Break Flow			√	no		
E2	Void Profile/Two-Phase Level		√		no		
E3	GDCS Interaction / Condensation		√	√	Apparent Error, no GDCS interaction during blowdown phase		
E5	Heat Slabs		√	√	yes		√
E6	Flashing		√	√	yes		√
E7	Isolation Condenser Interaction		√	√	yes	yes	√
F	**Upper Plenum**						
F1	Void Distribution & Two Phase Level		√	√	yes		√
N	**Containment**						
N2	Mixing	Drywell	√	√	yes		√
		Suppression Pool	√	√	yes		√
N6	Pressure Drop (flows at small Δp)		√	√	yes		√
N7	Condensation Heat Transfer with Non-Condensibles	IC/PCCS	√	√	yes	yes	√
		Free Surfaces & Walls	√	√	yes		√
		Wetwell	√	√	yes		√
N10	Conduction Heat Transfer in Composite Walls		√	√	yes		√

Legend for Table 2-1:

- SEO side entry orifice
- UTP upper tie plate
- BT boiling transition
- GT guide tube
- IC isolation condenser
- GDCS gravity driven cooling system
- PCCS passive containment cooling system

Table 2.2 Summary of BNL Composite PIRT for SBWR LOCA Reactor System

Component	Phenomenon	Accident Phase			Uniquely Important in SBWR	Neglected in SSAR short-term LOCA analyses	Candidate for Further Investigation
		Pre-Isoln	Isoln	Deprs			
Break	Break Flow	H	H	H	no		
Steam Line	Flow	H	-	-	no		
ADS Valves	Valve Flow	-	-	H	no		
Separator	Phase Separation	L	L	M			
	Friction	L	L	M			
Core	Parallel Channel Flow Distribution	L	L	H	syst int		√
	Pressure Drop	H	M	M	syst int		√
	Flashing	H	L	H	syst int		√
	Level Swell	H	L	H	syst int		√
	Heat Transfer	H	H	H	syst int		√
Fuel Rods	Core Power	H	H	H	no		
	Heat Transfer	H	H	H	no		
	Stored Energy	L	L	H	no		
RV Structures	Heat Transfer	L	L	H	no		
	Stored Energy	L	L	H	no		
Lower Plenum	Mixing	L	L	L			
	Flashing	H	L	H	syst int		√
Downcomer	Mixing	L	L	L	syst int		√
	Flashing	H	L	H	syst int		√
	Level Swell	H	L	H	syst int		√
	Condensation	-	-	-			
Feedwater Supply	Pump Coastdown/Flashing	M	M	M		yes	
Isolation Condenser	Condensation Tube Side	-	H	M	syst & p	yes	√
	Pool Heat Transfer	-	H	M	no	yes	

Table 2.3 Summary of BNL Composite PIRT for SBWR LOCA Containment System

Component	Phenomenon	Accident Phase			Uniquely Important in SBWR	Neglected in SSAR short-term LOCA analyses	Candidates for Further Investigation
		Pre-Isoln	Isoln	Deprs			
Dry Well	Non-Condensible Distribution	M	M	H	yes		
Structures	Condensation with NCs	H	M	H	yes		
Wet Well	Non-Condensible Distribution	-	-	L			
Horizontal Vent	Vent Clearing	H	H	H	no		
Suppression Pool Horizontal Vents	Condensation with NCs Flow from Horizontal Pipe	L	-	H	yes		
Suppression Pool SRV Sparger	Condensation	-	M	H	no		
Suppression Pool Purge Line	Condensation with NCs	-	-	H	no		
GDCS Line	Friction	-	-	-	yes (gravity)		
PCCS	Tube-Side Condensation with NCs	-	-	M			
	Pool Heat Transfer	-	-	M			
PCCS Purge Line	Friction	-	-	M			
PCCS Condensate Return Line	Friction (2[illegible])	-	-	M			
Vacuum Breaker	Friction (Steam & NCs)	-	-	-			

3.0 SBWR-Unique Phenomena

The items identified in the PIRTs of Chapter 2, can conveniently be arranged in three subsections, for further discussion here. These are reactor system phenomena, isolation condenser phenomena, including IC/RV interactions, and containment system phenomena.

3.1 Reactor System Phenomena

The composite list of items identified in the GE and BNL PIRTs, which occur in the reactor vessel, are

- Natural Circulation through the Vessel (Downcomer/Core Interaction)
- Core Internal Natural Circulation
- Flashing in the Core and in the Downcomer
- Upper Plenum Void Distribution and Two-Phase Level
- Heat Conduction in Solid Structures in the Downcomer.

Clearly, both PIRTs expressed a predominant concern with natural convection through the reactor loop, as well as potential core-internal recirculation.

Flashing, as a general phenomenon, occurs in numerous applications and during many reactor accident transients. In the SBWR with a design core coolant inlet temperature of 279 ^{0}C, bulk flashing in the core can begin at pressures of about 63 bar, which is generally reached early in most accident transients. Flashing is important in its contributions to level swell and natural circulation driving head. It is hard to understand, why it should not be ranked equal to natural circulation flow in all BWR designs. It was apparently ranked higher in the SBWR, since natural circulation is a particularly important phenomenon here. However, flashing in itself is a well researched and well understood phenomenon, which occurs in all BWRs and does not require further study. This phenomenon can, therefore, here be lumped into the natural circulation flow items.

The upper plenum of the SBWR is the chimney region on top of the core. The void distribution and two-phase level in this region are of great importance, since they determine the driving head for natural circulation in the reactor vessel. This phenomenon can, therefore, also be treated together with the natural circulation flow items.

Heat conduction in the downcomer region can be particularly important in the SBWR, but, we are dealing here with a well understood phenomenon, and with material properties which are not significantly different from other reactors and are sufficiently well known. Therefore, the available data base for downcomer heat conduction is sufficient. However, the code nodalization detail might have to be reviewed, to ascertain proper modelling of this phenomenon.

This leaves as the major phenomenon, to be addressed here for further consideration, the natural circulation flows in the reactor, considering

- natural circulation through the reactor vessel (downcomer/core interactions), as well as
- in-core natural recirculation flows.

The major possible safety concern here would be that, under extended flow stagnation or due to parallel channel instabilities at low flow rates, the critical heat flux could be exceeded locally, resulting in excessive peak clad temperatures and fuel damage.

A summary of the event scenarios during LOCAs is presented in Table 3.1, which is mainly based on Section 6.3 of the SSAR. Unfortunately, the SSAR remains vague concerning many details of the event scenarios. Assumption, apparently generally made in these evaluations are loss of power at time zero with loss of feed water, no CRD flow, and no IC operation (SSAR, Kim et al, 1993, and Paradiso et al, 1993). Apparently initial water level at L_3 and scram at time zero are also to be assumed, but some of the figures in the SSAR seem to contradict this. In particular, neglecting the effects of the ICs can lead to drastically different transients, than the ones presented in the SSAR, as will be shown below.

The steam line break outside the containment, as presented in the SSAR, leads to scram and isolation within seconds after the break, followed by about 1 hr of a pressurized state in the RV with repeated steam release through the SRVs, with gradually decreasing water level, followed finally by depressurization after about 1 hr. This is a very artificial event scenario, completely dominated by the lack of IC operation. We understand that GE will present revised LOCA scenarios, to include the effects of the ICs. Therefore, the currently presented scenario is not of interest. Since isolation occurs within a few seconds, a more realistic assessment of this scenario would be similar to the case of spurious Main Steam Line Isolation Valve (MSIV) closure. That case was analyzed with the INEL RELAP5 model (Ghan et al, 1992), showing that within less than 100 s the ICs absorb the decay heat, leading to a gradual depressurization, similar to other LOCA events.

As shown in Table 3.1, the scenarios of large and small break inside the containment lead to scram within a few seconds, isolation within 15 s or less, followed by depressurization, with a pressure of 10 bars reached about 200 to 600 s after the beginning of the transient.

Considering natural circulation flows during these blowdown transients, one is primarily concerned with potential low flow rates, which could lead to parallel channel flow instabilities

and/or local flow stagnation. The available SSAR transient results for large and small break LOCAs do not show any details of in-core flows. However, very recent simulations, made by other members of the BNL SBWR team, using the BNL-developed RELAP5 model for the SBWR (BNL-94), were made available. These transients consist of a

- Main Steam Line Break (MSLB), representative of a large break LOCA, and a
- Bottom Drain Line Break (BDLB), representative of a small break LOCA.

Some of the results from these transients are shown in Figures 3.1 to 3.6 for the MSLB, and in Figures 3.7 to 3.11 for the BDLB.

In the MSLB transient a reactor vessel pressure of 10 bar is reached at about 200 s, and the Automatic Depressurization System (ADS) is actuated at about 520 s. The MSLB simulations show initial decrease in core flow rates till about 70 s, followed by oscillating, but predominantly positive flow between 70 and 180 s, the oscillating period being about 18 s per cycle. Thereafter, more rapid oscillations of about 8 s per cycle are observed, with the minimum of the oscillations generally being below zero, indicating temporary flow reversal. The evaluation of these recent runs has not yet proceeded to the point, that the cause for the oscillations has been fully determined. However, similar runs at BNL, simulating a break transient for the PUMA test facility, show very similar oscillatory behavior. Oscillations of similar frequency were also observed in some of the SBWR simulations made at INEL (Ghan et al, 1992). For the 70 s to 200 s time period the average core inlet flow is about 900 kg/s, with about 1100 kg/s net upflow in the average core channels and 300 kg/s in the peripheral core channels, balanced by about 500 kg/s downflow in the core bypass channels. While there remains a significant buoyancy driven flow from the downcomer and through the core, there is additional internal circulation within the core, as anticipated.

During steady state operation the total core mass flow is 7600 kg/s, with 4800 kg/s through the average channels. Thus, during the MSLB transient, the average channel flow rate between 70 and 200 s is about 22% of steady state flow, at a time when the decay heat amounts to only 4 to 5% of steady state power. Even though flow reversal is observed in the average core channels, the mean flow remains relatively large, with good cooling conditions being maintained. The mean void fraction in the average channels remains at about 0.5, and no extremes are observed. Hot channel void fractions range between 0.5 and 0.6, again, without any extremes being observed. Under these conditions parallel channel instabilities or film boiling due to low flow are not expected to occur. Also, at these substantial flow rates, the available heat transfer correlation packages in RELAP5 and TRACG do not exceed their range of validity, which could occur at very low flow rates.

The BNL results for this transient, as well as for the BDLB differ significantly from the GE results of the SSAR. The main reason for the differing results is the modelling of the ICs in

the BNL runs, which were neglected in the GE evaluations. The primary effect of the ICs on the early blowdown transient is the return of cold liquid into the downcomer, with steam condensation, reduced steam dome pressure and, consequently, also reduced break flow. As mentioned above, a revision of the SSAR scenarios is anticipated to include the effects of ICs in the LOCA events. We would expect those results to correspond more closely with the ones presented here.

In the BDLB scenario, following an initial rapid decrease in pressure, isolation occurs at approximately 80 s on reaching Level L_2. Thereafter, with the inflow of cold liquid into the downcomer from the ICs, the pressure, now at about 20 bar, begins to decay relatively slowly. Correspondingly, the break flows from the bottom drain line and from the shutdown cooling system outlet remain relatively small, amounting only to about 25 kg/s after 100 s. A pressure of 10 bar in the steam dome is reached at about 870 s, after ADS actuation on reaching Level L_1 at 810 s. Actually, the water level in the reactor vessel came within 0.3 m of L_1 at 110 s. So a slightly different setting in some input parameter could have resulted in a much shorter transient. However, it is not anticipated, that the phenomena observed would have been significantly different.

In the core, after initial decreases and oscillations in core flow, a quasi-steady oscillatory flow is established around 270 s, with a period of about 4 s per cycle. The mean flow rates are 800 kg/s upflow in the average core channels, 200 kg/s upflow in the peripheral channels, balanced by 1000 kg/s downflow in the bypass channels. In this transient the core inflow from the downcomer of about 20 kg/s remains very small, approximately accounting for boil-off. Thus, in this case, there is hardly any natural circulation flow between core and downcomer, just a flow sufficient to absorb the remaining decay heat. However, there is a strong core-internal natural circulation flow, with the flow in the average channels being about 17% of full power steady state flow. The void fraction in the average core channels reaches an early peak of 0.55 at 80 s and decreases thereafter to about 0.27 for most of the transient. Again, with such strong internal circulation, no flow instabilities or local film boiling would be anticipated.

The results for this case differ even more from those of the SSAR, again, due to several differing assumptions. The SSAR figures for this case indicate that GE apparently assumed isolation and a rapid decrease of steam flow to begin at time zero. Also the feedwater flow down ramp is more rapid than the one assumed in the BNL run, which is based on different SSAR information. As a result of these differences, the SSAR simulations keep the pressure after isolation around 60 bar. In the BNL run, isolation on L_2 occurs only at 80 s, with more feedwater flow into the system, but also with significant steam outflow during this time period. In the BNL run a pressure of 20 bar is reached at about 50 s, with a correspondingly lower break flow. After isolation, the presence of the ICs in the BNL model accounts for the differences between the

results. While the durations of various time phases vary, based on sometimes only slightly different assumptions, the main effect shown here, is a strong tendency for the core to develop internal natural circulation, which should preclude fuel damage.

Full pressure natural circulation tests for BWR conditions were conducted in the FIST facility (Hwang, Alamgir & Sutherland, 1984, Sutherland et al, 1985), which includes a full size 8x8 fuel bundle as its core (with the fuel rods replaced by electrically heated rods). However, the given test data apply for a system very different from the SBWR, having jet pumps in the downcomer region and lacking a chimney above the core. The tests were essentially quasi steady runs at different power levels for each test, covering the power range from 0.5 to 3.0 MW (10 to 65% of normal operating power). By gradually and slowly reducing the water level in the downcomer, each test presents a set of data for flow rate versus buoyancy head, at a specific heat input rate.

While the FIST data are valuable for general code validation, they are not representative of SBWR blowdown transients, due to significant design differences and due to power levels above the range of decay heat. The observed natural circulation flow rates, with natural convection conditions less favorable than in the SBWR, were quite substantial. Similar to the BNL results above, internal core circulation was observed as the downcomer water level decreased, with return flow through the bypass channels. Thus, qualitatively the FIST experimental observations are in agreement with the BNL predictions for the SBWR.

3.2 Isolation Condenser Phenomena

While the concept of isolation condensers has been used before in natural circulation BWRs (Humboldt Bay, Unit 3; Dodewaard), their design significantly differs from the current SBWR design, which employs vertical tubes and includes drain lines for the removal of noncondensibles to the containment suppression pool. The major concerns with ICs , identified in the PIRTs of Chapter 2 were

- Isolation Condenser Tube Side Heat Transfer at High Pressure in the presence of Non-Condensibles
- Isolation Condenser and Downcomer Flow Interactions.

While the ICs will be less affected by noncondensibles than the PCCS, there are several sources for noncondensibles and their effect on the IC performance must be considered. Potential sources for noncondensibles are

- hydrogen and oxygen from radiolytic decomposition of water,
- noncondensible gases, dissolved in the feedwater, and
- air, entering the ICs during maintenance.

To investigate condensation heat transfer in the presence of noncondensibles for IC and PCCS conditions, experiments were conducted at University of California, Berkeley (UCB) (Vierow and Schrock, 1991), and at the Massachussets Institute of Technology (MIT) (Siddique, 1992, Siddique et al, 1993). The MIT experiments also investigated condensation heat transfer with noncondensibles on structural walls (Dehbi et al, 1991). Both sets of experiments were conducted at pressures of 3 to 5 bar, typical for PCCS and containment operating conditions, but not comparable to IC pressures. Correlations of heat transfer coefficients were developed, and the UCB correlations are currently in use in TRACG, apparently being used at all pressures. RELAP5 recently changed to a B&W modified Colburn-Hougen correlation (Kuo et al 1993, Colburn et al, 1934), which was also developed for low pressure applications. At the pressure level of the experiments, a strong increase in heat transfer coefficients with pressure was observed. Oscillatory behavior was observed at high concentrations of noncondensibles. However, the experimental apparatus did not include vent lines for noncondensibles, and flow stability conditions would be expected to differ significantly from IC operation. Tills (1994) shows potential inaccuracies in the predictions based on the above correlations, even at PCCS pressure levels.

The PIPER-ONE test facility at the university of Pisa, Italy, was used for an experimental investigation of IC condensation heat transfer, pool side heat transfer and IC/reactor interaction

(D'Auria et al, 1993). The IC/Reactor loop was operated at a pressure of 50 bar. Their IC consisted of 12 vertical tubes, of 20 mm ID, 400 mm long, immersed in a pool, apparently at atmospheric pressure. The authors attempted to correlate the experimental results with the RELAP5/Mod2 code, but their predictions of temperatures within the IC tubes remained unsatisfactory, for which they stated as a possible cause "code inadequacies in predicting condensation flow regimes (and heat transfer coefficients)".

It is clear from the above references, that data on condensation of water at high pressure in vertical tubes in the presence of noncondensibles would be highly desirable. Once available, the appropriate heat transfer correlations would have to be developed and implemented into the relevant codes. Kuo et al (1993) refer to the existence of recent, apparently unpublished B&W experimental data for high pressure condensation in the presence of noncondensibles. It would be desirable, to determine the range covered by these data, and their potential availability for this work.

An integral, full scale experiment of IC operation is currently being prepared at the PANTHERS facility of SIET in Italy. While this is not a separate effect heat transfer test facility, the results from these experiments are expected to be sufficient to ascertain satisfactory IC performance under LOCA conditions. The program is described by Masoni et al, (1993). The prototype IC to be used here is a single full sized module, two of such modules making up a complete SBWR isolation condenser. The objective of the full size module tests is to confirm the thermal and fluid flow performance of the IC and to demonstrate structural integrity. In particular, the heat removal capability, stable flow and heat transfer conditions, and proper operation of the noncondensible gas vents is to be confirmed.

The test program includes experiments, simulating normal operation, reactor heatup and cooldown without IC operation, i.e. the steam lines to the ICs are open, but the condensate return lines remain closed. Also planned are simulated ATWS events, with a rapid pressure rise and opening of the condensate return lines. The effects of non-condensible gases are included in some of the experiments. However, no experiments to simulate LOCA blowdown conditions are currently planned.

The first concern identified in the PIRTs, condensation heat transfer in the ICs during blowdown, may be sufficiently confirmed by these experiments, in establishing that the condensation performance is adequate to remove all decay heat. However, the data are not expected to be sufficiently detailed to yield correlations for heat transfer coefficients during condensation at high pressure in vertical tubes in the presence of noncondensibles. Such correlations would be of interest for model improvements. However, the demonstration of adequate heat removal rates, as currently planned in the PANTHERS program, may be considered sufficient to ascertain safe operation.

Concerning code capabilities, the data expected from this facility could provide a valuable data base for code validation, but they would not be expected to be sufficiently detailed, to serve for the development of revised correlations. Masoni et al (1993) provide detailed information on the planned PCCS instrumentation, which includes wall heat flux sensors. Very little is said with respect to the ICs. Apparently the main emphasis is on strain gauge measurements, but coolant and wall temperature measurements are included too.

Considering the IC/RV interactions, the main concern would be flow instabilities and oscillations due to simultaneous boiling and condensation within a loop, driven by buoyancy forces only. The PANTHERS tests are expected to provide significant insight in this regard. however, it would be of interest, if a simulated blowdown sequence could be included into the test matrix. Even then, the simulated reactor vessel lacks its own simulated core with a natural circulation loop between downcomer and core. In the absence of any specific flow instabilities that can be identified for this unique system, it would be desirable, to do more detailed code simulations of blowdown transients, with and without ICs, to establish whether there are further concerns, regarding IC/RV interactions.

As already noted in a review of the PANTHERS test program (Rohatgi, 1994), it would be desirable, to include simulation of the accumulation of noncondensibles in the IC tubes during normal standby operation, to ascertain proper operation as the condensate return lines are opened. In an investigation for the Oyster Creek BWR, which includes an IC with horizontal tubes, Moore (1988) found that such accumulations would be dissolved or carried under, as the ICs are started. However, it would be desirable, to have this established independently for the SBWR design. Apparently some of the IC tests will include noncondensible gasses, and the completed test program may answer all potential concerns.

3.3 Containment System Phenomena

The containment phenomena, identified as important and SBWR unique in the GE and BNL PIRTs of Chapter 2 can be summarized as

- Mixing and Non-Condensibles Distribution in the Drywell and Suppression Chamber
- Horizontal Vent Clearing
- Condensation with Non-Condensibles in Flow through Short L/D Horizontal Vent
- Flow Distribution with Slow Flows at Small Pressure Drops
- Condensation Heat Transfer with Non-Condensibles (IC, PCCS & Walls)
- Conduction Heat Transfer in Composite Walls.

Since the focus of this study is on high pressure blowdown, phenomena identified as important in the containment are only of interest if they do have a direct feed-back to the reactor transients during the high pressure phase. With reactor vessel pressures above 10 bar, the break flow as well as ADS flows through the DPVs or SRVs will be choked and there is essentially no feed-back from the containment to the reactor. The possible exception is the scram signal, which during an MSLB transient in the SBWR is generally expected to be on high dry well pressure, occurring within a fraction of a second (Arai & Nagasaka, 1991, Ghan et al 1992). Vent clearing is expected to occur around 1 to 4 s, i. e. after scram. Some of the small break scenarios could also include scram on high drywell pressure, within a few seconds from the beginning of the transient.

For modelling of the dry well pressure rise during the first few seconds of a transient, the only one of the above items, which is possibly of marginal importance, is condensation with noncondensibles on walls of the dry well. However, it is doubtful, whether this could have any measurable effect on the dry well pressure within such a short time. The effect of significant condensation would be a slower pressure rise, resulting in a later scram. Recent experiments at MIT (Dehbi et al, 1990 & 1991) have resulted in improved correlations for condensation heat transfer in the presence of noncondensibles on structural walls, including the range of drywell and suppression chamber pressure levels. While these correlations are available, for use in modelling of drywell and suppression chamber transients, their application to only the first few seconds or less of a break transient would be questionable, since the local distribution of the reactor coolant entering the drywell would be highly non-uniform and not well tractable for such a short time span.

Thus, any credible analysis of the first second or less of large break transients would require a completely different modelling approach, solving detailed multi-dimensional conservation equations in the drywell. This approach would not be cost effective, since

inaccuracies in the scram time prediction of a few seconds do not contribute significantly to the remainder of the transient, with its primary emphasis on peak clad temperature and reactor vessel coolant inventory. Furthermore, if the drywell scram signal were missed, scram on other signals (steam line differential pressure, low steam line pressure, turbine trip or low reactor water level (L_3)) would be observed promptly.

Table 3.1 Time Sequences for LOCA Scenarios (Based on SSAR)

Break	Break Size (cm^2)	Additional Failure	SSAR Figures 6.3-	Approximate Time of Event (s)					
				Scram (Level 3)[1] ($\approx$17.3 m)	Isolation (Level 2) ($\approx$14.4 m)	ADS Signal (Level 1) ($\approx$10.4 m)	ADS Observed	GDCS Flow Observed	RV Pressure at 10 bar
Steam Line, Outside CB	1955	DPV	65-72	few s (on steam flow or p)	few s (on steam flow or p)	3500	3500	3900	3700
DPV Stub Tube	1056	GDCS Valve	9-16	< 1 s (on p drywell)	$\approx$15 (on low p)	410 (?)[2]	250	430	290
Steam Line, Inside CB	977	DPV	17-24	< 1 s (on p drywell)	0.5 to 5.5 (on steam flow or p)	375 (?)[2]	370	510	280
Feedwater Line	390	DPV	25-32	0 (assumed)	< 10	120	125	300	230
RWCU/SDC Suction Line	295	GDCS Valve	33-40	0 (assumed)	< 10	230	230	510	370
IC Return Line	168	DPV	41-48	0 (assumed)	< 10	350	400	715	560
GDCS Injection Line	45.6	DPV	49-56	0 (assumed)	$\approx$10	100	150	500	320
Bottom Head Drain Line	20.3	DPV	57-64	0 (assumed)	$\approx$10	330	330	680	510

1 Normal Water Level NWL $\approx$18.3 m

2 Data extracted from SSAR figures. "?" indicates that times differ from expected values (ADS to follow ADS signal L_1, and GDCS to follow ADS signal + at least 150 s)

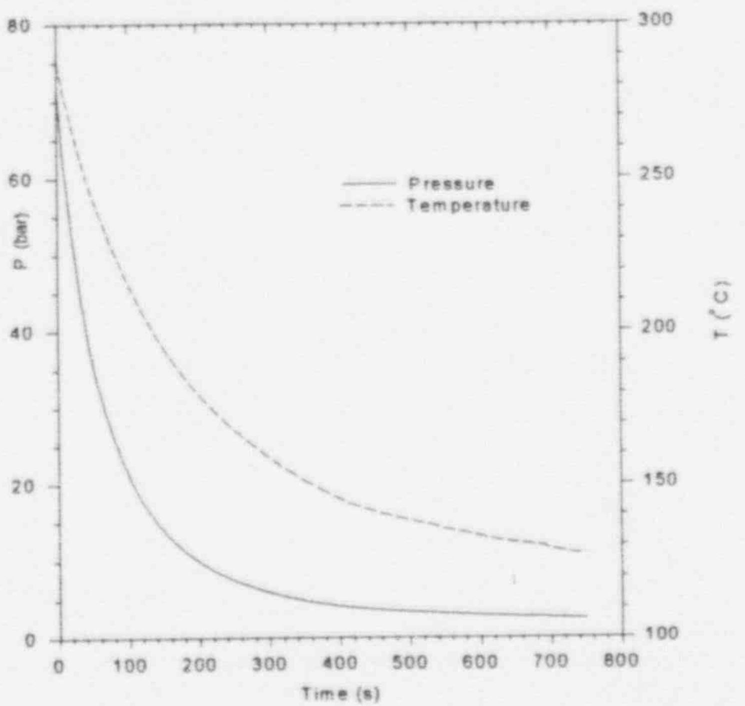

Figure 3.1 Steam Dome Pressure and Temperature During MSLB

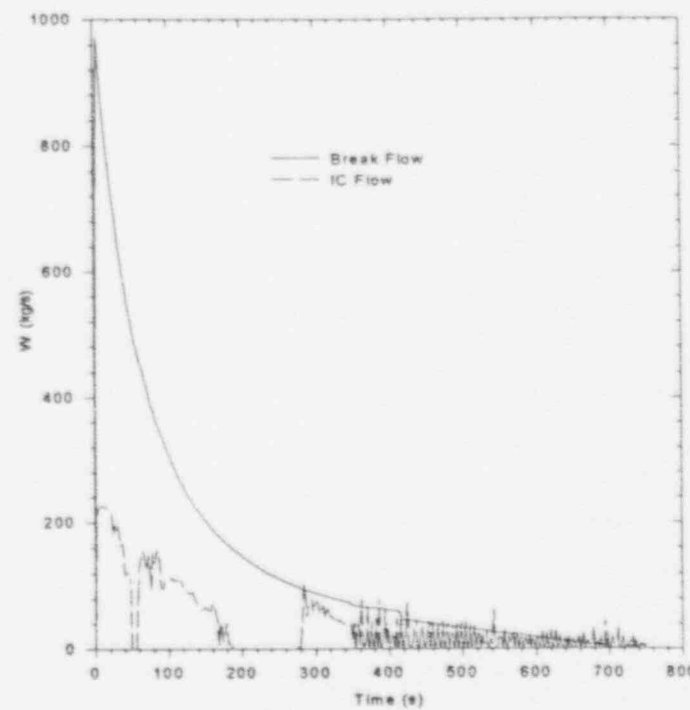

Figure 3.2 Break and IC Flow Rates During MSLB

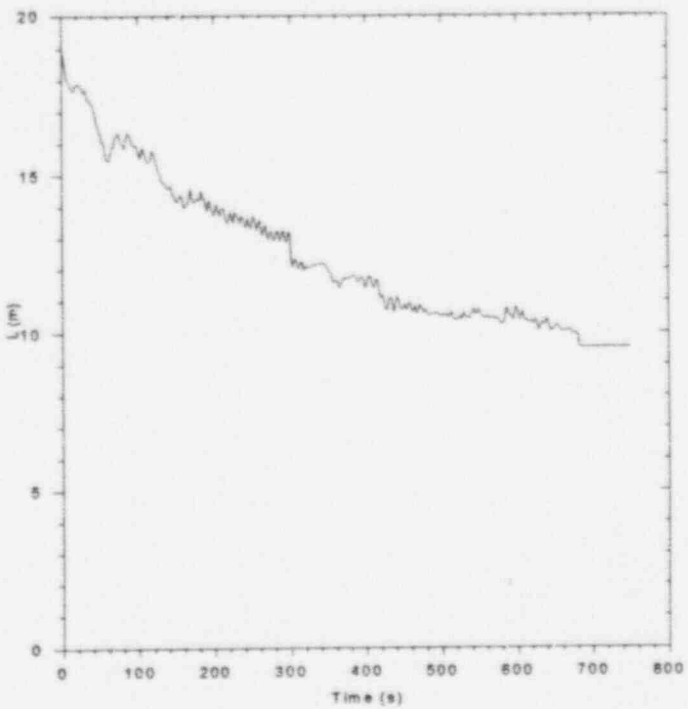

Figure 3.3 Downcomer Wide Range Level During MSLB

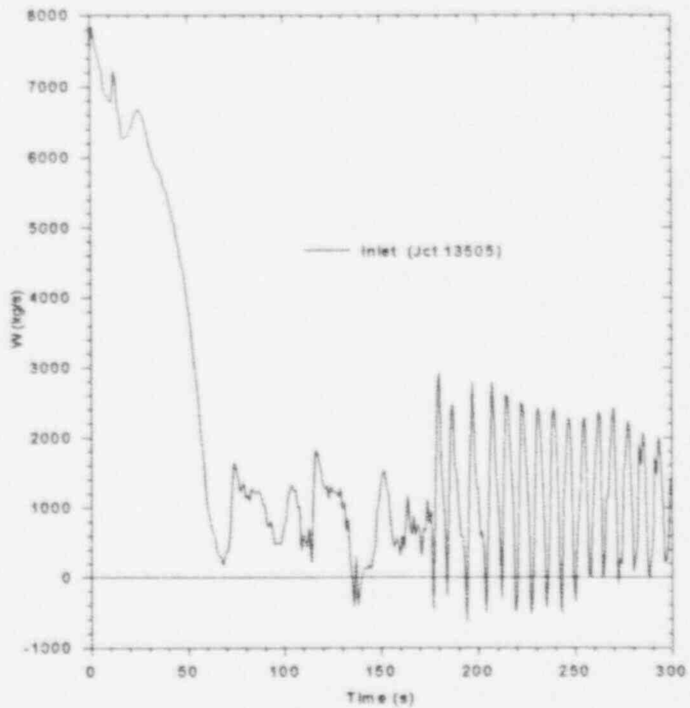

Figure 3.4 Core Inlet Flow During MSLB

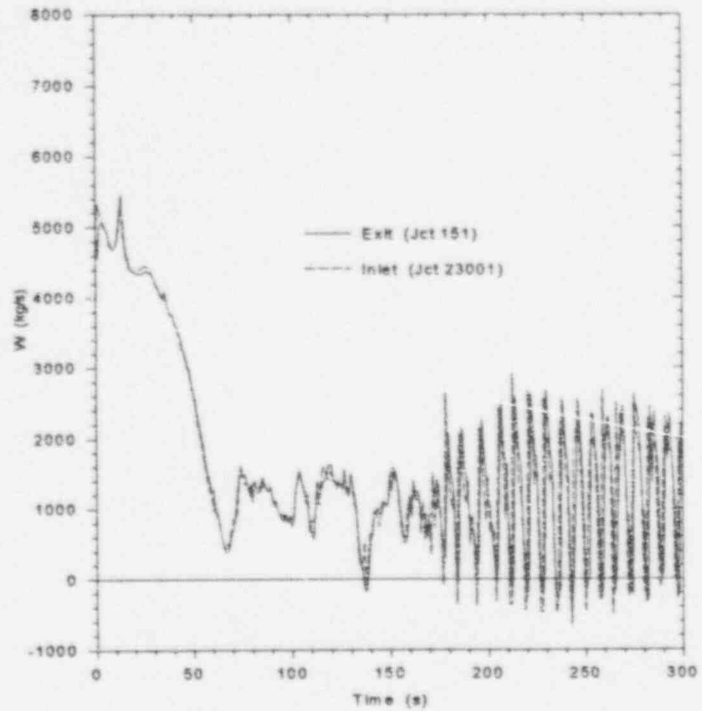

Figure 3.5 Core Average Channel Mass Flow During MSLB

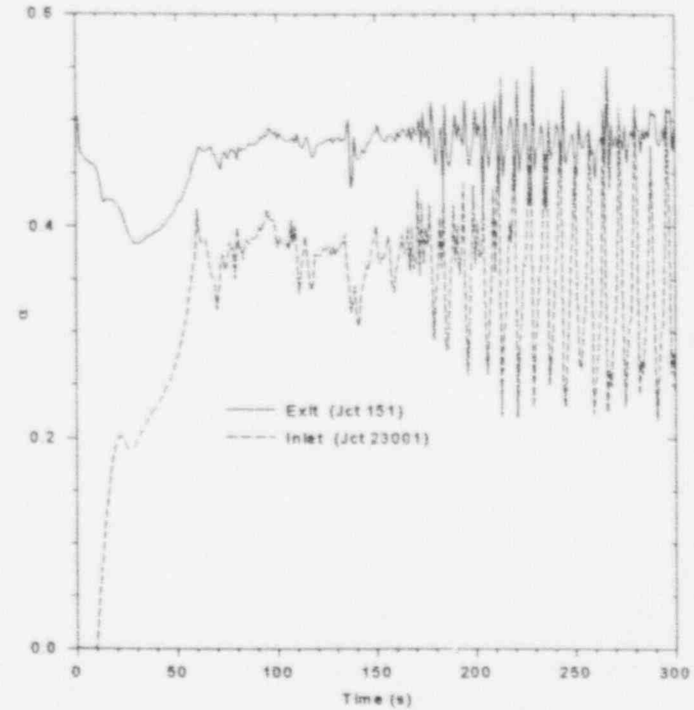

Figure 3.6 Core Average Channel Void Fractions During MSLB

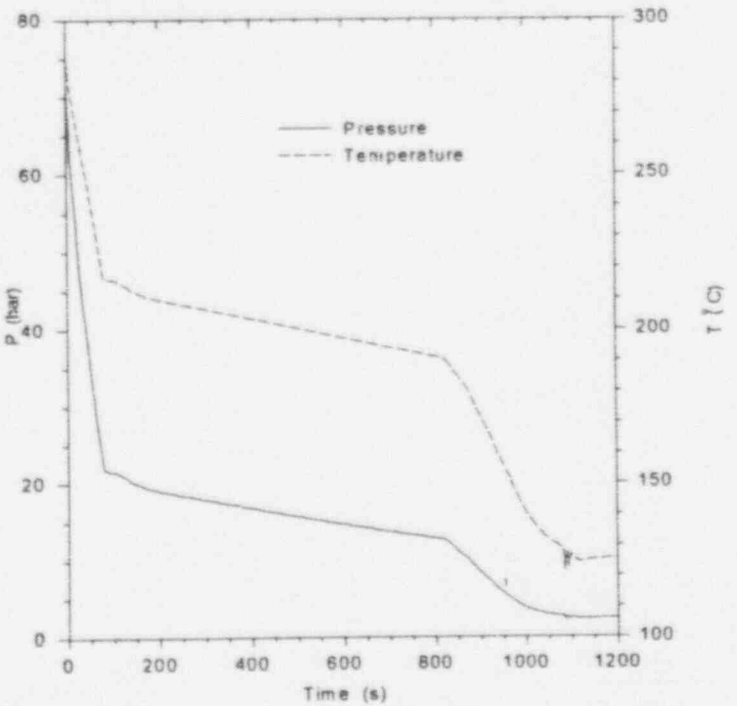

Figure 3.7 Steam Dome Pressure and Temperature During BDLB

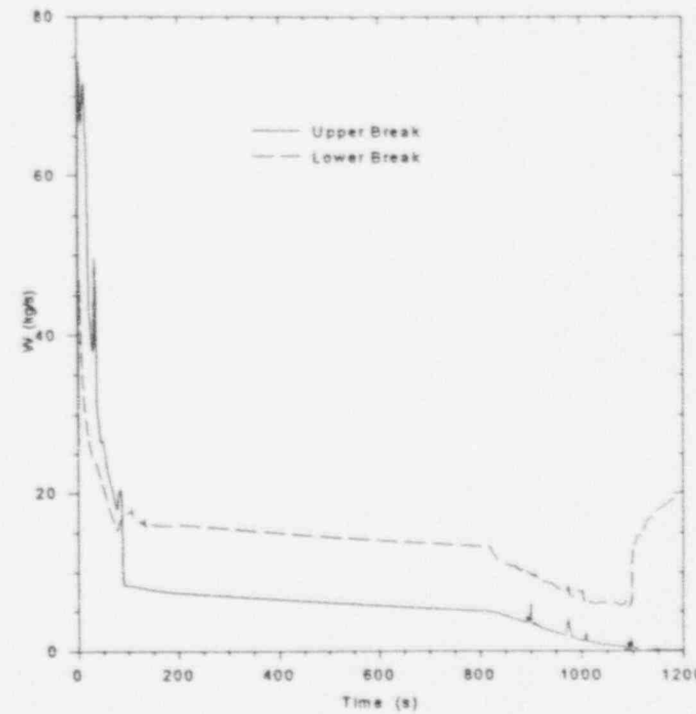

Figure 3.8 Break Flow Rates During BDLB

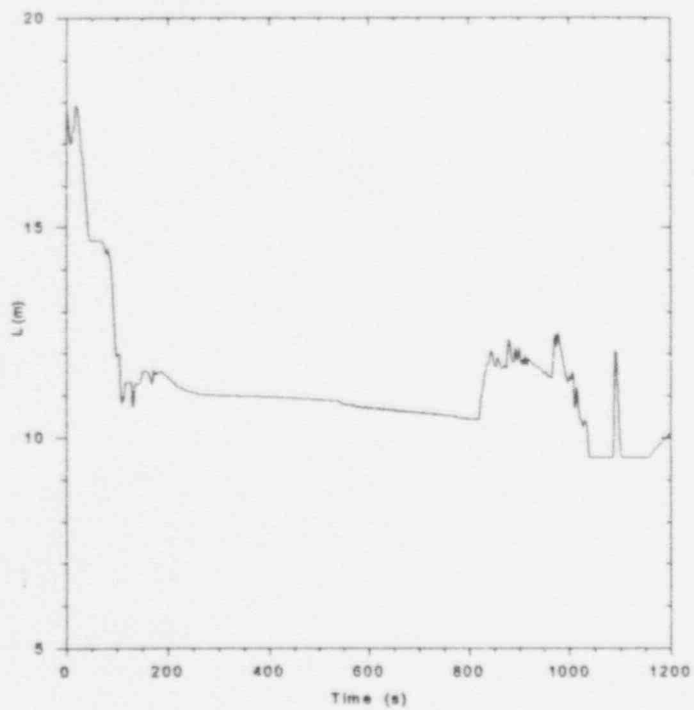

Figure 3.9 Downcomer Wide range Level During BDLB

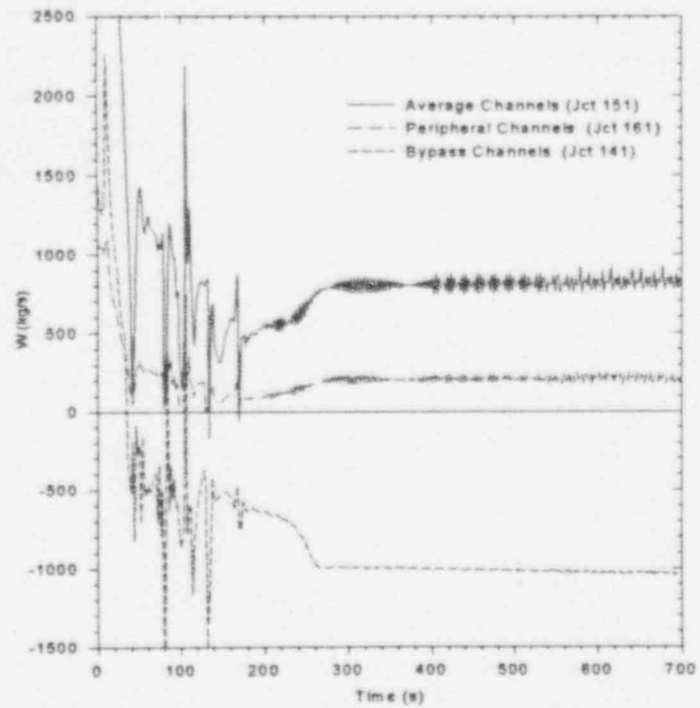

Figure 3.10 In-Core Mass Flows During BDLB

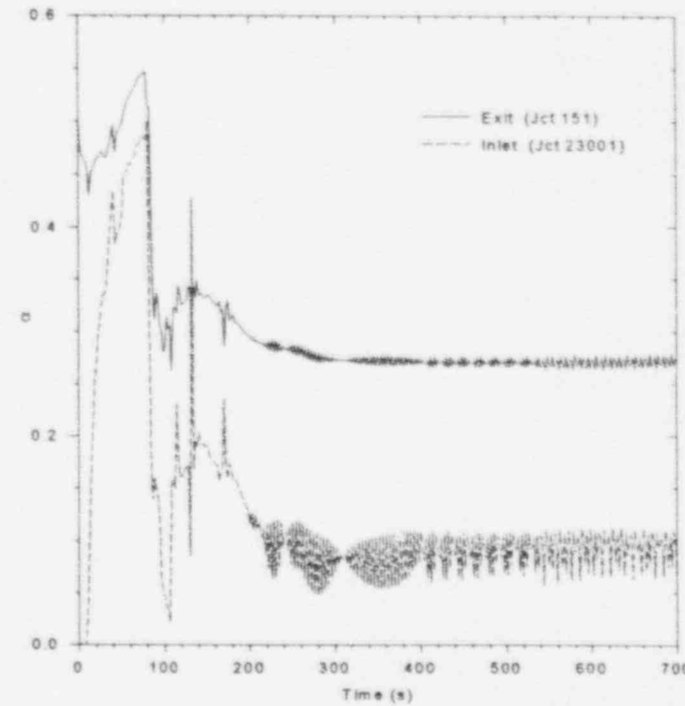

Figure 3.11 Core Average Channel Void Fraction During BDLB

4.0 Summary of Items for Desirable Data Base Extension or Further Attention

4.1 Reactor Vessel

It was shown in Section 3.1, that significant reactor vessel and/or core internal natural circulation flows provide good core cooling at all times, and no concerns for fuel damage were identified. The available heat transfer and pressure drop package remains valid at the relevant flow rates.

Heat conduction in the vessel walls was identified as more essential in the SBWR than in other BWR designs, due to its influence on downcomer fluid density and, thus, on the driving head for natural circulation flow. While this phenomenon is well understood and does not require further study, one might want to be aware of its added importance in SBWR modelling and ascertain that the model nodalization is sufficient for high fidelity predictions.

Therefore, we conclude, that the available data base is sufficient for high confidence modelling of in-vessel effects during the high pressure phase of blowdown scenarios.

4.2 Isolation Condensers

Concerns, identified in Sections 2 and 3, were heat transfer with condensation at high pressure and in the presence of noncondensibles, as well as flow interactions between the isolation condenser and the reactor vessel.

Considering the heat transfer with condensation, the PANTHERS test program is expected to demonstrate acceptable performance, and the effects of noncondensibles are scheduled to be included in the test program. While these tests will not be sufficiently detailed, to provide information for new or improved heat transfer correlations, they are expected to provide the assurance of sufficient heat removal capability.

Kuo et al (1993) mention recent B&W experiments on condensation with noncondensibles at high pressure. It would appear desirable to explore, whether these data could be made available and what their range is.

Considering potential flow interactions between the reactor vessel and the isolation condensers, specific concerns for flow instabilities have not been identified. However, it would have been of interest, if a simulated blowdown transient could have been included in the PANTHERS test program. Furthermore, additional code simulations, to search for potential oscillatory behavior and flow instabilities would be desirable.

Inclusion into the test program of a case with accumulation of noncondensibles in the ICs, prior to start during a blowdown sequence would be of interest and is apparently include in the test matrix. As indicated in Section 3.2, it is anticipated, that such accumulations would be purged by commencing coolant flow.

In summary, the PANTHERS program may well answer all concerns raised in Sections 2 and 3 about IC performance and IC/RV interactions during the high pressure phase of blowdown scenarios. A conclusive determination can, of course, only be made, once the test program is completed.

4.3 Containment

The break flow remains choked during the high pressure phase of blowdown transients, and feedback from the containment to the reactor vessel can only occur in form of the scram signal on high drywell pressure. This signal is anticipated within a fraction of a second for the main steamline break and within a few seconds for some small break scenarios. Condensation of reactor coolant on the containment walls of the drywell was identified in Sections 2 and 3 as a phenomenon of importance, but its prediction during this very short time period would be very difficult and does not appear essential or warranted, in particular in the presence of several other fast scram signals. If it were to be modelled, there is a sufficient data base, but alternate modelling approaches, with finer nodalization, would be required.

Therefore, we conclude, that the available data base is sufficient for high confidence modelling of the relevant containment phenomena during the high pressure phase of blowdown scenarios.

5.0 References

BNL-93

Presentation by Brookhaven National Laboratory to NRC Staff, "PIRT for SBWR Transients", December 13 & 14, 1993.

BNL-94

Presentation by Brookhaven National Laboratory to NRC Staff, "Preliminary Assessment of PUMA Facility", March 9, 1994.

GE-92

"Implementation of TRAC-G for Licensing Analysis", J. L. Rash (GE) to R. C. Jones (US NRC), March 9, 1992, JSC 92-010, MFN 042-92. (View Graphs of Presentation by General Electric to NRC Staff, TRAC-G, January 29, 1992)

GE-93-1

Presentation by General Electric to NRC Staff, "TRAC-G Application to BWR Analysis, Introductory Meeting", June 15, 1993

GE-93-2

Presentation by GE Nuclear Energy, P.F. Billig et al, Overview of GE's SBWR Testing Programs, October 1, 1993, at Purdue University, West Lafayette, IN.

K. Arai, H. Nagasaka,

"Analytical Study on Drywell Cooler Heat Removal Performance as a Passive Containment Cooling System", Proceedings of the 1st JSME/ASME Joint International Conference on Nuclear Engineering, Tokyo, Japan, pp. 281-287, 1991.

B. Boyack et. al.,

"Quantifying Reactor Safety Margins - Application of Code Scaling and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident ", NUREG/CR-5249; EGG-2552, October 1989.

A. P. Colburn, O. A. Hougen,

"Design of Cooler Condensers for Mixtures of Vapors with Noncondensing Gases," Industrial and Engineering Chemistry, Vol 26, pp 1178-1183, 1934.

F. D'Auria, P. Vigni, P. Marsili,

"Application of RELAP5/MOD3 to the Evaluation of Isolation Condenser Performance," ASME/JSME Nuclear Engineering Conference, Volume 1, pp 173-183, ASME 1993.

A. A. Dehbi, M. W. Golay and M. S. Kazimi,

"A Theoretical Modelling of the Effects of Noncondensable Gases on Steam Condensation Under Turbulent Natural Convection," Seventh Proceedings of Nuclear Thermal

Hydraulics, ANS Winter Meeting, pp 229-236, 1991.

A. A. Dehbi, M. W. Golay and M. S. Kazimi,
"The Effect of Air in Steam Condensation Under Turbulent Natural Convection," Trans ANS, 62, pp 704-705, Washington DC, 1990.

L. Scott Ghan, Roy A. Shaw, Jeffrey A. Borkowski, John C. Determan,
"Initial Application of RELAP5/MOD3 to the Simplified Boiling Water Reactor", INEL, EGG-NE-10475, September 1992.

C. H. Heath, C. M. Kullberg, J.A. Borkowski, K. R. Jones,
"Operating Experience of Passive, Natural Circulation Core Cooling in Boiling Water Reactors", INEL Letter Report, December 1992.

W. S. Hwang, Md. Alamgir, W. A. Sutherland,
"BWR Full Integral Simulation Test (FIST) Phase I Test Results", General Electric Co., NUREG/CR-3711, GEAP-30496, September 1984.

M. Ishii, S. T. Revankar, R. Dowlati, I. Babelli, W. Wang, H. Prokharna, M. L. Bertodano, V. H. Ransom,
"Preliminary Scientific Design of Purdue University Multi-Dimensional Integral Test Assembly (PUMA) for GE SBWR", School of Nuclear Engineering, Purdue University, West Lafayette, IN 47907, February, 1994.

H. T. Kim, W. Marquino, F. M. Paradiso, J. R. Fitch,
"Application of TRACG to SBWR Licensing Safety Analysis", General Electric, NEDE-32178P, Rev 0, February 1993.

Craig M. Kullberg, C. Don Fletcher,
"Survey of Experimental Data Base for the Simplified Boiling Water Reactor", INEL, EGG-NE-10474, September 1992.

H. H. Kuo, R. W. Shumway, G. W. Johnsen,
"Software Design and Implementation Document, Shah and Colburn-Hougen Correlations", INEL NRCL2537-14, Rev. 0, October 1993.

W. Marquino, J. G. Munthe-Andersen,
"Results of TRACG Analysis of SBWR Transients, Accounting for Model Uncertainty", Ninth Proceedings of Nuclear Thermal Hydraulics, ANS Winter Meeting, 1993, Pg. 202, 1993.

P. Masoni, S. Botti, G. W. Fitzsimmons,
"Confirmatory Tests of Full-Scale Condensers for the SBWR", Proceedings of the Second ASME/JSME Nuclear Engineering Joint Conference, San Francisco, CA, pp. 735-744.

J. P. Moore,
"Evaluation of Isolation Condenser Performance with Noncondensable Gases in Steam," GPU Nuclear, Topical Report 056, Rev. 0, December 1988.

F. M. Paradiso, A. I. Yang, C. D. Sawyer,
"SBWR Loss-of-Coolant Accident Analysis,", ASME/JSME Nuclear Engineering Conference, Volume 1, pp.313-315, 1993

U. S. Rohatgi,
"Preliminary Comments on PANTHERS Tests", Brookhaven National Laboratory, Note to T. M. Lee, USNRC, April 4, 1994.

B. S. Shiralkar, Md. Alamgir, J.G.M. Andersen,
"Thermal Hydraulic Aspects of the SBWR Design," International Conference on Design and Safety of Advanced Nuclear Power Plants (ANP 92), Paper 31.1, Tokyo, Japan, 1992.

Mansoor Siddique,
"The Effects of Noncondensible Gases on Steam Condensation under Forced Convection Conditions", Ph. D. Thesis, MIT, January 1992.

M. Siddique, M. W. Golay, M. S. Kazimi,
"Local Heat Transfer Coefficients for Forced Convection Condensation of Steam in a Vertical Tube in the Presence of a Noncondensable Gas," Nuclear Technology, Vol 102, pp. 386-402, June 1993.

W. A. Sutherland, Md. Alamgir, J.A. Findlay, W. S. Hwang,
"BWR Full Integral Simulation Test (FIST) Phase II Test Results and TRAC-BWR Model Qualification", General Electric Co., NUREG/CR-4128, GEAP-30876, October, 1985.

Jack Tills,
"Letter Report on PCCS Modeling for SBWR", Sandia National Laboratories, March, 1994.

K. M. Vierow, V. E. Schrock,
"Condensation in a Natural Circulation Loop with Non-Condensable Gases, Part I - Heat Transfer," Proc. Intl. Conf. on Multi-Phase Flow, Tsuruba, Japan, pp.183-186, September 1991. (Also Japan-US Seminar on Two-Phase Flow Dynamics, Berkeley, CA, July 5-11, 1992).