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Revision 0

August 2004

IRIS Small Break LOCA Phenomena Identification and Ranking Table (PIRT)

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WCAP-16318-NP, Addendum 1

IRIS SMALL BREAK LOCA PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT)

ADDENDUM 1 – IRIS SMALL BREAK LOCA SENSITIVITY REPORT FOR PIRT DEVELOPMENT

August 19, 2004

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[ ] (a, c)

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1 INTRODUCTION

This document describes a set of sensitivity studies that have been performed to improve the understanding of the IRIS response to a small break LOCA (SBLOCA) event and define a sufficient knowledge basis for the development of an IRIS SBLOCA phenomena identification and ranking table (PIRT).

Based on the review of the preliminary SBLOCA results presented in WCAP-16082, Volume II, critical components and assumptions that were considered to have a potential influence on the IRIS SBLOCA response were examined, and a test matrix to evaluate these effects was developed. The rationale used to select the cases to be included in this matrix is discussed below in detail.

It should be noted that these analyses were also selected and used to support a final optimization of the design, and thus some of the considered cases explore beyond design basis conditions (i.e. multiple system failures) to evaluate the robustness of the IRIS safety concept. These cases were also included in the report since they can provide additional valuable information on the system response.

In these analyses the total reactor coolant system inventory and the peak containment pressure have been used as the relevant performance parameters, and the effect of each considered case on these two parameters is discussed in detail. The RCS inventory is used as performance parameter in place of more classical peak cladding temperature or oxidation limits, because these cases also show that the IRIS goal to always maintain the core covered with water thus maintaining core coolability during a small break LOCA is conservatively demonstrated. Also the collapsed liquid level is used, as a conservative index of core coverage, since as long as the collapsed liquid level is above the top of the core active fuel, the IRIS goal of no core uncovery is considered to be verified.

The RCS inventory relation to the collapsed liquid level is shown in Figures 1-1 and 1-2. Since the top of active fuel is at about 6.0m from the bottom of the vessel, a RCS inventory above about 120,000 kg guarantees core coverage at saturation conditions at atmospheric pressure. During a SBLOCA transient, the moment of minimum RCS inventory typically occurs when pressures and temperatures are higher than atmospheric conditions, and therefore a limit of 120,000 kg conservatively assures core coverage throughout the whole transient duration. In conclusion, if the predicted RCS inventory remains above 120,000 kg during the transient, then the criteria for no core uncovery can be considered conservatively verified. For cases where the RCS inventory drops below 120,000 kg, additional information (mixture level in the vessel, core void fractions, and collapsed liquid level) will be provided to verify that the core remains covered and thus cooled during the transient.

All analyses presented in this report were performed by the University of Zagreb, using a coupled RELAP/GOTHIC code as described in WCAP-16082. Plant Models are consistent with those adopted in the WCAP-16082 study, except for the modification and improvements discussed in Section 3.1.

The discussion presented in this reports assume that the reader is familiar with the IRIS safety concept and with the SBLOCA analysis presented in WCAP-16082.

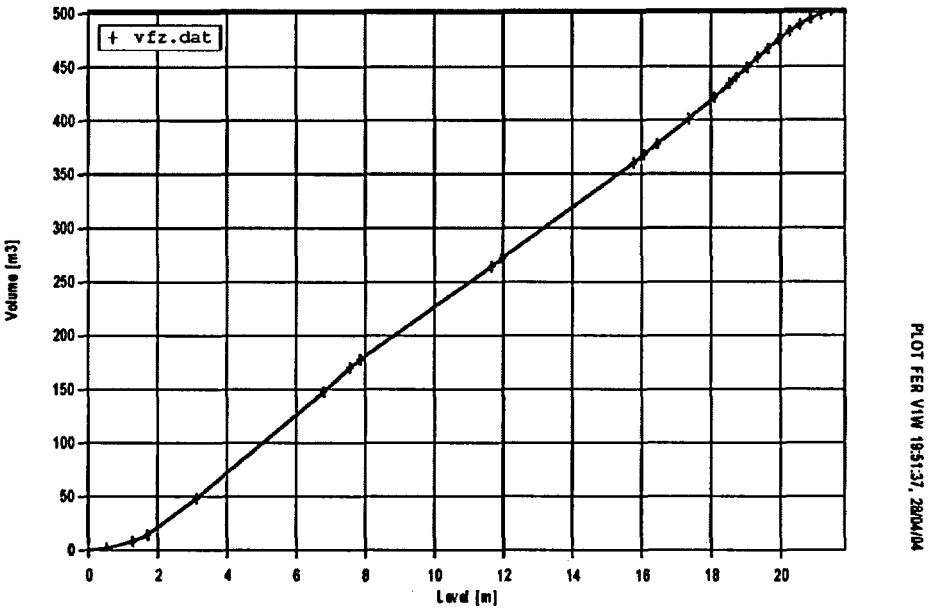

Figure 1 – IRIS Reactor Coolant System Volume as a function of elevation.

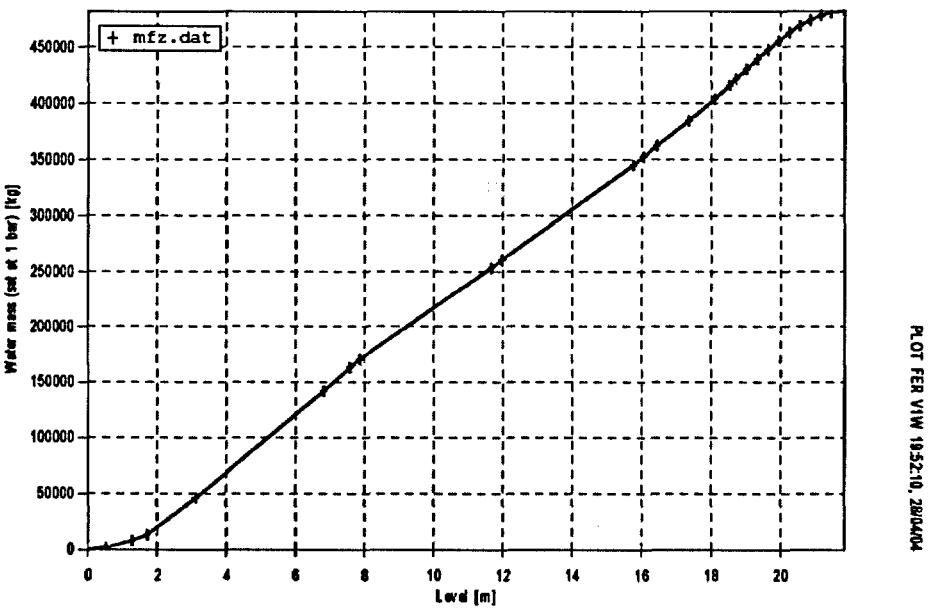

Figure 2 – IRIS Reactor Coolant System Inventory as a function of elevation (assuming saturation conditions at atmospheric pressure)

The following cases were included in the run matrix submitted to the University of Zagreb.

REVISED BASE CASES (SECTION 3.1)

As discussed in WCAP-16082 two base LOCA cases have been considered in the initial preliminary IRIS analyses: a 4" break in the CVCS line (upper break), and a 2" DVI guillotine break (lower break). Note that the two base cases discussed herein were analyzed with an updated plant model, as discussed in Section 3.1. The differences between the base case analyses presented in this study and those in WCAP-16082 are discussed in Section 3.1.

As part of the analysis some model limits that were identified during the development of this study were also investigated, and their impact on the overall transient response is discussed in Section 3.1.3.

Note that these two cases are referred throughout this report as "Base Cases".

Effect 1 - Sensitivity to EHRS performance (SECTION 3.2)

The EHRS is a critical component of the IRIS SBLOCA mitigation sequence. IRIS relies in fact on the EHRS to condense the steam generated in the core to depressurize the reactor coolant system and thus limit the loss of mass and maintain RCS inventory ("depressurization without loss of mass"). Thus, it was considered important to study the impact of the EHRS performance on the IRIS mitigation sequence.

Also, from a design point of view, the aggressive IRIS PRA goal (CDF goal of 10^{-8}) requires that SBLOCA mitigation be demonstrated, for realistic cases, with only three of four subsystems of EHRS available, and thus it was considered necessary to also include this case to provide the designers information to revise the sizing basis of the system (increasing, for example, the number of tubes) if necessary

The following sensitivity cases where therefore defined to address this effect:

[

]$^{(a,c)}$

[

]$^{(a,c)}$

Effect 2 - Sensitivity to Containment back pressure (SECTION 3.3)

Another critical component of IRIS SBLOCA evolution is the containment pressure since a higher pressure will in fact lead to a more rapid termination of the blowdown phase and thus to a higher reactor coolant system inventory during the transient. Also, separate studies were performed to explore the containment pressure calculations and these studies indicate that analytical and physical modeling can have a significant impact on the accurate determination of the containment pressure. These studies are included as Appendix A to this report, and discuss the importance of several modeling assumptions on the containment pressure prediction.

Given these considerations, it was considered important to verify the importance of the containment pressure on the IRIS SBLOCA response, as follows.

[

]$^{(a,c)}$

Effect 3 - Sensitivity to Suppression Pool (SECTION 3.4)

The containment pressure sensitivity to various assumptions is discussed in Appendix A. To complete that study, it was considered necessary to include some additional cases to explore the sensitivity to the suppression pool size.

Note that due to the additional information obtained from the Effect 2 sensitivity and from the analyses included in Appendix A, only a limited subset of these cases was performed as part of the final test matrix, as discussed below.

[

]$^{(a,c)}$

Effect 4 - Sensitivity to ADS (SECTION 3.5)

The ADS is designed to assist the EHRS in depressurizing the RCS during the blowdown phase of a SBLOCA in the low elevation DVI line. This limits the loss of mass from the RCS and allows coupling the RCS to the containment even though the break may be flooded, so that the water in the reactor cavity becomes available for core cooling by gravity.

The ADS was shown to be very important in the analyses performed in WCAP-16082 for the low break case, while it is expected to have a very limited impact on the upper break cases. Both to better understand the ADS impact on the IRIS SBLOCA sequence and to provide the designer with sufficient data for an optimization of the ADS size, the following sensitivity studies were performed.

[

]$^{(a,c)}$

Effect 5 - Sensitivity to Initial Conditions

All these cases were considered irrelevant for the purpose of this analysis and were therefore eliminated from the test matrix. Sensitivity to initial conditions will be a part of the evaluation model development, but compared to other effects considered in this analysis it does not add significant information on the IRIS SBLOCA event, and thus can be postponed to a later program phase.

[

]$^{(a,c)}$

Effect 6 - Miscellaneous Assumptions - Stored Energy, Decay Heat. (SECTION 3.6)

Decay heat and the initial stored energy in the fuel affect the energy release from the core to the RCS fluid and thus the rate of steam generation. The following analyses were included to verify the effect of different assumptions for these parameters on the overall SBLOCA response.

[

]$^{(a,c)}$

Effect 7 – Sensitivity to Break Size (SECTION 3.7)

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]$^{(a,c)}$

2 SUMMARY OF CONCLUSIONS

To provide sufficient information for the development of a small break LOCA (SBLOCA) PIRT of the IRIS reactor, a set of sensitity runs was performed to support the analysis documented in WCAP-16082, Vol II.

First, the two base cases presented in WCAP-16082 (an "upper break" in the 4" CVCS line, and a "lower break" in the 2" DVI line) were reviewed. Data for the IRIS reactor coolant system and reactor protection system were revised with the most updated plant data, and the results were compared with those provided in WCAP-16082. Since the results documented in WCAP-16082 were used to optimize the system design, results obtained in this revised calculations show an increased margin in the reactor coolant system inventory. The results were also reviewed to identify any limitation in the current model, and the plant model, especially for the lower break analysis, was revised to correct some unphysical behaviors in the treatment of the emergency boration tank (EBT) actuation.

The results for the base case analysis are discussed extensively in Section 3.1, and are summarized in Table 2-1.

The results of the sensivitity cases discussed in Section 1 are documented in Sections 3.2 to 3.7. The results for both upper and lower break cases are provided in Table 2-1. Results show that the upper break cases are relatively insensitive to a wide range of parameters, due to the fact that once the break uncovers early in the transient, only steam flows at the break and the loss of inventory is significantly milder. For the lower break cases however, the results were shown to be more sensitive to the variations in nominal conditions assumed in this sensitivity study.

In particular, the minimum RCS inventory for the lower break cases is shown to be especially sensitive to the containment pressure transient and to the break size.

In all considered cases, containment pressure has been shown to be sensitive to several plant parameters and modeling assumptions: additional informations on the containment pressure response are included in Appendix A to this report.

Finally, it should be noted that all the sensitivity cases explored confirm the large margin to core uncovery available for design basis SBLOCA events, and provide additional confirmation of the robustness of the proposed mitigation approach.

Table 2-1 Summary of Results for IRIS SBLOCA Sensitivities

Case	Upper Break		Lower Break	
	Containment Peak Pressure (MPa)	Minimum RCS Inventory (kg)	Containment Peak Pressure (MPa)	Minimum RCS Inventory (kg)

(a, c)

3 DETAILED SENSITIVITY STUDIES RESULTS

3.1 BASE CASE ANALYSES

As part of this study, the base case analyses for the 4" CVCS line break (upper break) and the 2" DVI line rupture (lower break), discussed in WCAP-16082, were repeated with an improved RELAP input deck. The differences from the previous input deck are summarized here, and an explaination of the differences between this revised analysis and the results presented in WCAP-16082 is provided in the following sections.

1. As discussed in WCAP-16082, the RSC (Riser Steam generator Check valves) were moved to a lower elevation to improve the system response and to maintain a single phase natural circulation path from the core to the SGs and back to the core for a longer period of time. [ ] (a, c). The same model was used for the steam generator shroud check valves.
2. As discussed in WCAP-16082, the ADS sparger has been moved from the suppression pool system to a dedicated quench tank. The reasons for this modification have been discussed in WCAP-16082 and are not explored further here. The ADS quench tank was modeled to initially contain 30 m^3 of water, and the tank is open to the containment atmosphere.
3. As more detailed design data on the IRIS integral reactor coolant system were made available, a new plant model was developed using these new design data to replace assumptions and simplifications used in the original model. Most of the modifications are negligible (e.g. small changes in the vessel dimensions, additional data on the downcomer geometry,...), but the increased normal operating water level in the pressurizer has a significant effect on the initial transient response. The large amount of high temperature water (i.e saturation conditions at 15.5 Mpa / 2250 psia) available in the pressurizer leads in fact to a slower RCS depressurization rate due to the larger inventory of water that is available for flashing. One of the main consequences of this modification is an increased operating margin, but also the fact that the actuation signals during the early part of the transient tend to be delayed compared to the analyses presented in WCAP-16082.
4. The EHRS model was updated with the latest design data. Based on the results of preliminary safety analyses, the EHRS design was in fact reviewed and optimized, and this revised model was used in the analyses. The effect of EHRS on the RCS response is discussed in more detailed in the following sections.
5. The Reactor Protection System model was improved for this revised analysis, correcting some actuation signals and including dynamic compensation terms on some protection functions. The effect on the RCS response is not significant for this analysis.

Figure 3.1-1 and 3.1-2 show the RCS and containment models used in the IRIS Analyses discussed in this report. Aside from the differences summarized above, the models are identical to those used for the study documented in WCAP-16082. Additional discussions on the containment and RCS model are provided in WCAP-16082.

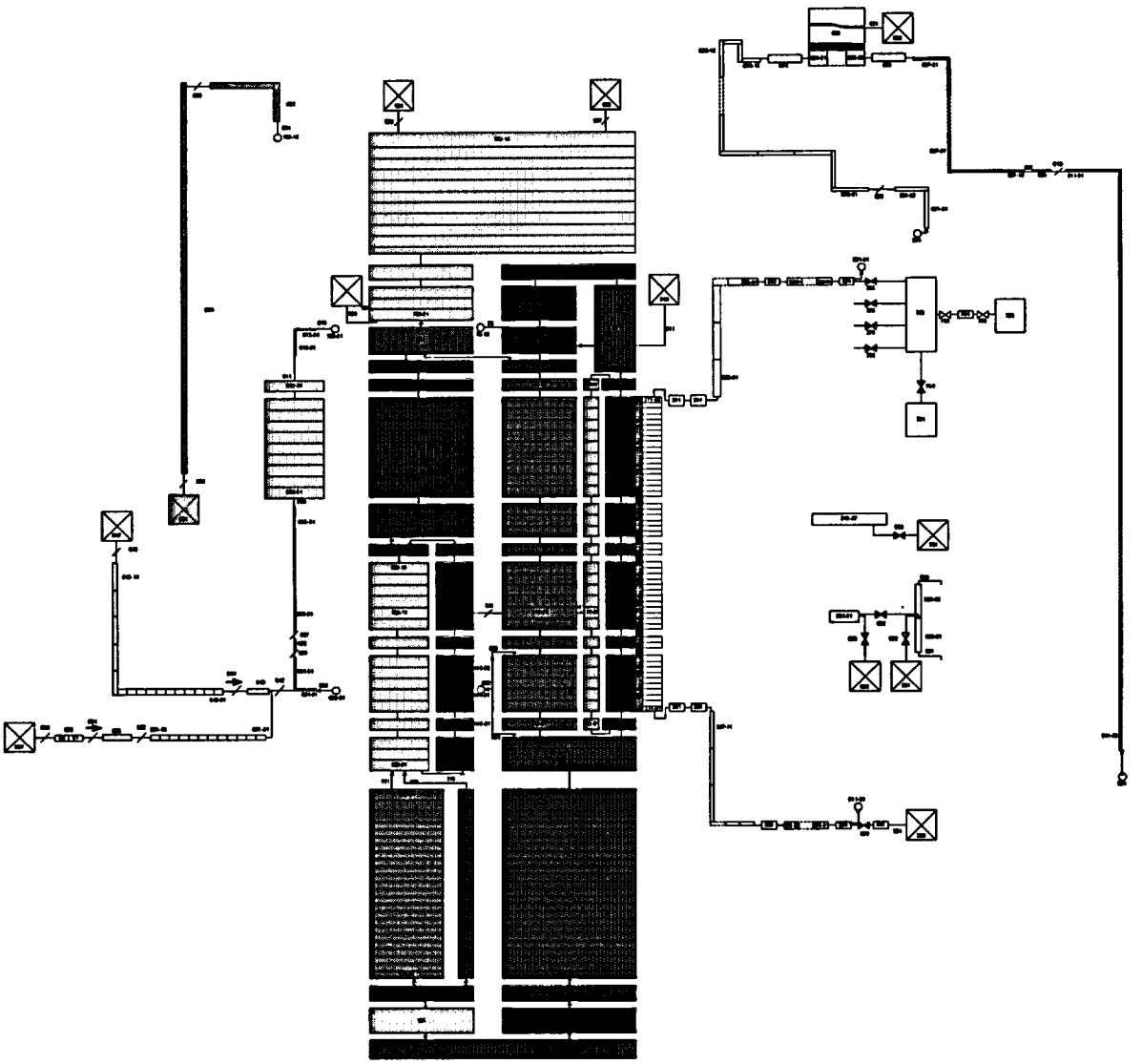

Figure 3.1-1
Reactor Coolant System Model for RELAP5 Used in IRIS Safety Analyses

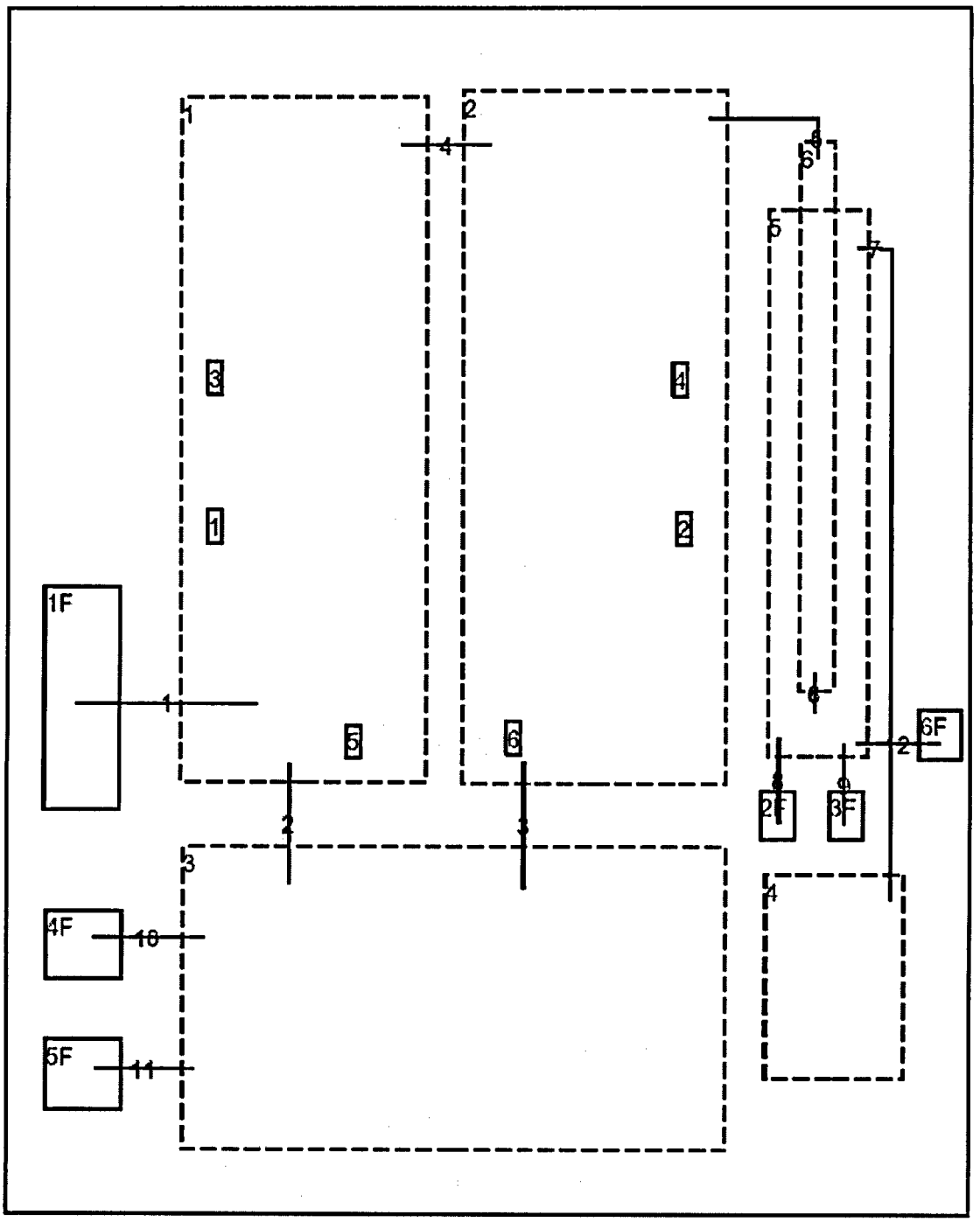

Figure 3.1-2
Simplified Containment Model for GOTHIC Used in IRIS Safety Analyses

Figures 3.1-3 to 3.1-38 show the transient evolution for the two Base Cases, while Tables 3-1 and 3-2 provide a sequence of events. Note that the qualitative transient response is in general very similar to the analyses documented in WCAP-16082, and only the significant differences between the analyses are discussed here.

3.1.1 Upper Break, Base Case

For the 4" CVCS line break (upper break), the overall response is in very good agreement with the calculations provided in WCAP-16082, with two important differences that are discussed here in detail.

The sequence of events for the revised calculation is significantly different in the early part of the transient from the WCAP-16082 analysis. Note that the differences only have a small effect on the overall transient response due to the long characteristic times of the IRIS SBLOCA. By reviewing Table 1, it is evident that the revised analyses present a slower initial depressurization rate, which delays the time to reach the low pressure reactor trip and LM setpoints. [

]$^{(a,c)}$

The results for the upper break Base Case are shown in Figures 3.1-4 to 3.1-20. [

]$^{(a,c)}$

[

]$^{(a,c)}$

[ ] (a, c)

Figure 3.1-3 – Primary and Secondary Liquid Levels during the depressurization phase (liquid=blue, steam=red)

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]$^{(a,c)}$

[

]$^{(a,c)}$

[

]$^{(a,c)}$

[

]$^{(a,c)}$

Note that in this analysis and in general for all the cases included in this report, little information is provided on containment parameters (such as liquid levels and nitrogen and steam masses). Detailed discussions on the containment response are provided in Appendix A and WCAP-16082, and was thus not considered necessary to provide additional discussions in the main body of this report.

3.1.2 Lower Break, Base Case

The results for the low break show some differences from those presented in WCAP-16082 due to the different layout of the ADS system. Figures 3.1-21 and 3.1-22 show the containment and pressurizer pressure transient. The peak containment pressure is reached at about 2,000 seconds at about 0.82 MPa, but it can be observed the the reactor coolant system pressure continues to be reduced until it reaches a minimum of 0.6 MPa at about 3,000 seconds. [

]$^{(a,c)}$

[

]$^{(a,c)}$

As expected, the loss of mass at the break continues throughout the transient, due to the injection of water from the suppression pool (Figure 3.1-36) that is sufficient to maintain the RCS level above the elevation of the break.

3.1.3 Review of Analysis Results and Model Limitations

During the review of the sensitivity analyses, some modeling assumptions were reviewed and found to be not appropriate for IRIS analyses, as follows:

1. [

]$^{(a,c)}$

2. [

]$^{(a,c)}$

3. [

]$^{(a,c)}$

3.1.3.1 ADS Actuation and Pressurizer – RCS Interface

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3.1.3.2 EBT draining for small breaks and low depressurization rates

[

]$^{(a,c)}$

Table 3-1
TIME SEQUENCE OF EVENTS FOR IRIS SBLOCA – Upper Break, Base Case

Accident	Event	New Results Time (seconds)	WCAP-16082 Results Time (seconds)
SBLOCA – 4" Break in CVCS Line	Break occurs	0.0	0.0
	Low Pressurizer Pressure Trip setpoint reached	(20.36)	10.6
	High Containment Pressure S-Signal setpoint reached, Reactor trip on S-Signal	12.0	
	Rods begin to fall into core	14.0	12.6
	LM (LOCA Mitigation) signal setpoint reached.	27.7	19.7
	EHRS actuation signal generated, ADS valves begin to open	29.7	21.7
	Reactor coolant pumps begin to coastdown	29.7	25.6
	EBT valve completely open	59.7	29.5
	EHRS valves completely open (all four subsystems)	41.7	33.7
	ADS valves completely open	29.7	51.7
	Containment and Pressure Vessel pressure equalizes (end of blowdown phase)	~1,600	~1,600
	Injection from the suppression pool to the reactor coolant system begins	~1,600	~1,600
	Suppression Pool vents fill and containment flooding begins	>>10,000	~3,000

Table 3-2
TIME SEQUENCE OF EVENTS FOR IRIS SBLOCA – Lower Break, Base Case

Accident	Event	New Results Time (seconds)	WCAP-16082 results Time (seconds)
SBLOCA – 2" DE Break in DVI Line	Break occurs	0.0	0.0
	Low Pressurizer Level Trip setpoint reached	(65.7)	36.1
	High Containment Pressure S-Signal setpoint reached, Reactor trip on S-Signal	46.5	
	Rods begin to fall into core	48.5	38.1
	LM (LOCA Mitigation) signal setpoint reached	95.2	50.2
	Reactor coolant pumps begin to coastdown	97.2	51.1
	EHRS actuation signal generated, ADS valves begin to open	97.2	52.2
	EBT valve completely open	127.2	63.2
	EHRS valves completely open (all four subsystems)	109.2	64.2
	ADS valves completely open	97.2	82.2
	Injection from the suppression pool to the reactor coolant system begins	~2100	~2600
	Containment and Pressure Vessel pressure equalizes (end of blowdown phase)	~2100	~2800
	Suppression Pool vents fill and containment flooding begins	N.A.	N.A.

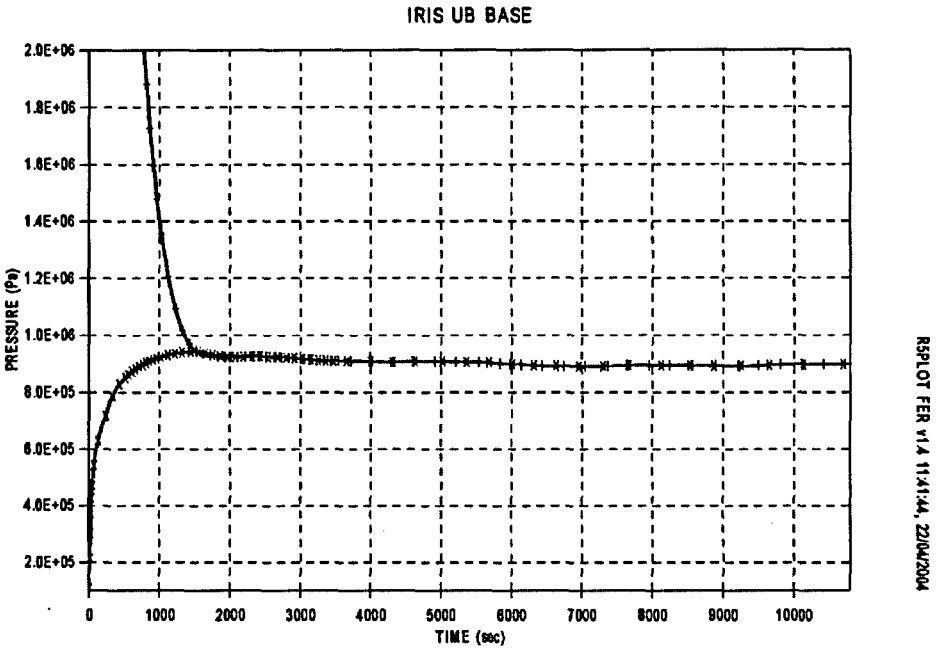

Figure 3.1-4 - Containment and Pressurizer Pressure Transient (Detail). Base Case, Upper Break

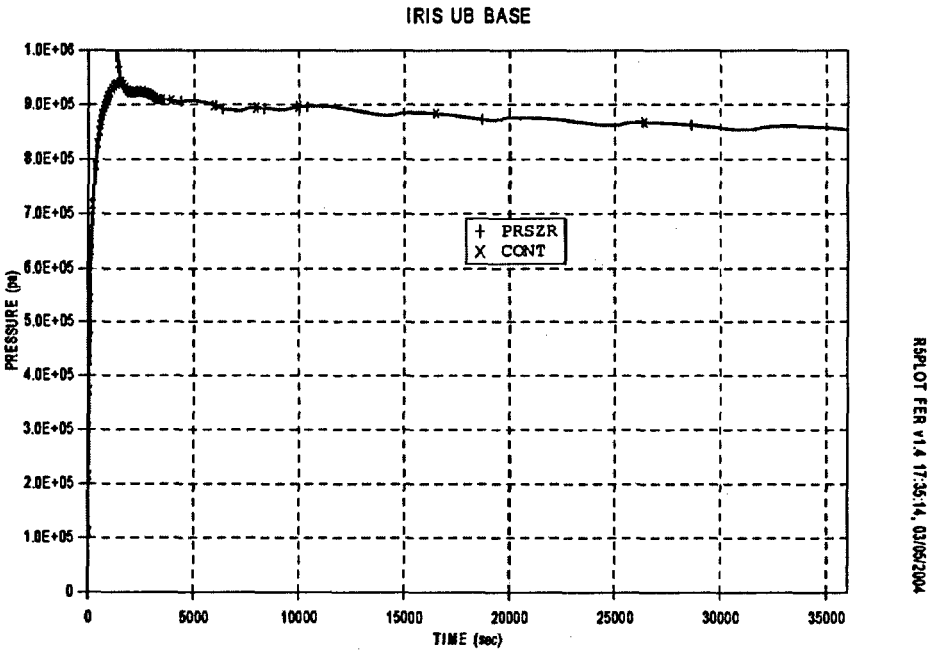

Figure 3.1-5 - Containment and Pressurizer Pressure Transient (Long Term Run). Base Case, Upper Break

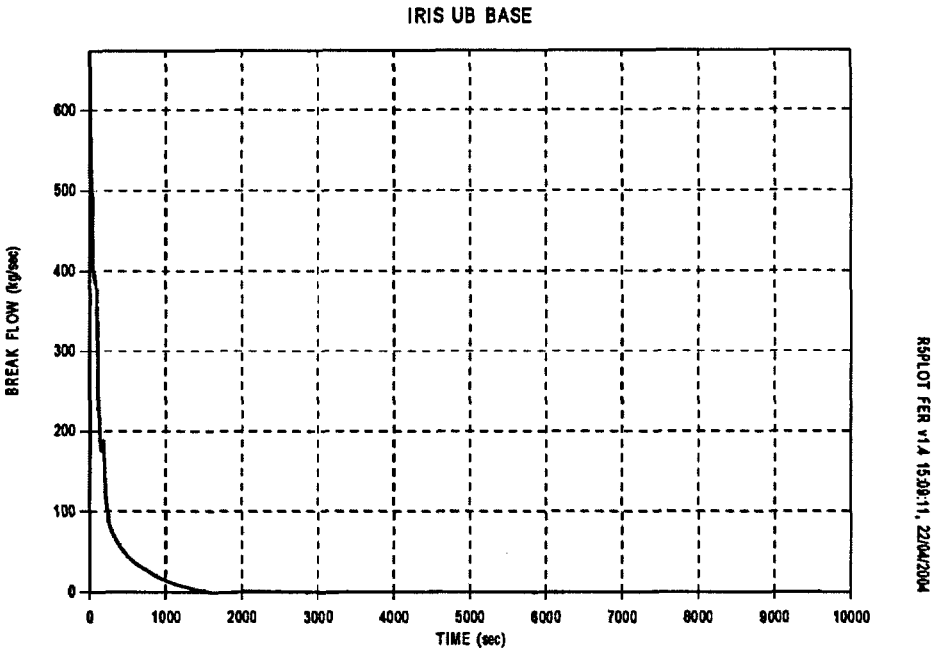

Figure 3.1-6 – Break Flow Rate Transient. Base Case, Upper Break

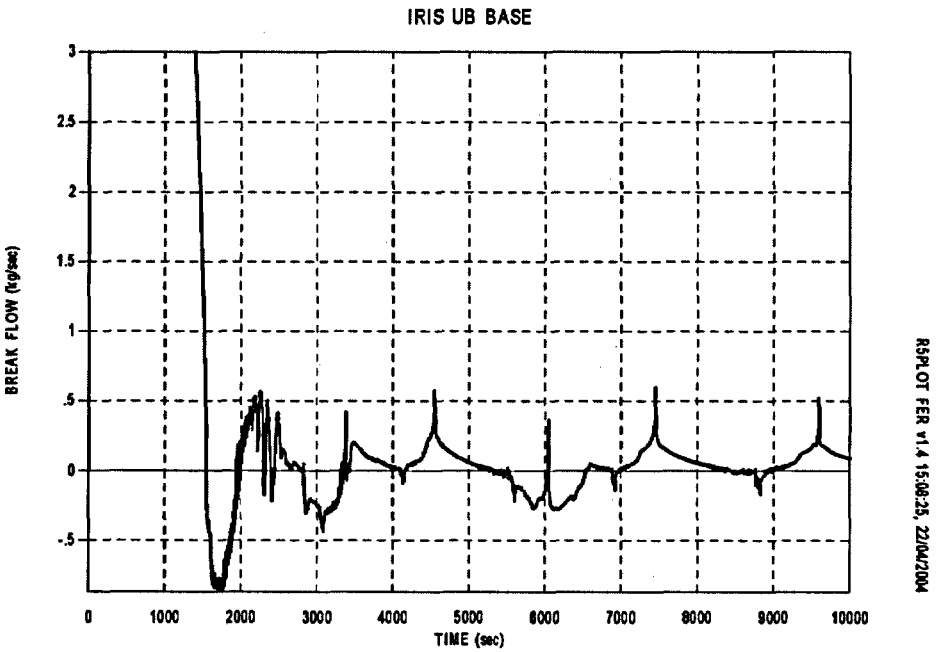

Figure 3.1-7 – Break Flow Rate Transient (Detail). Base Case, Upper Break

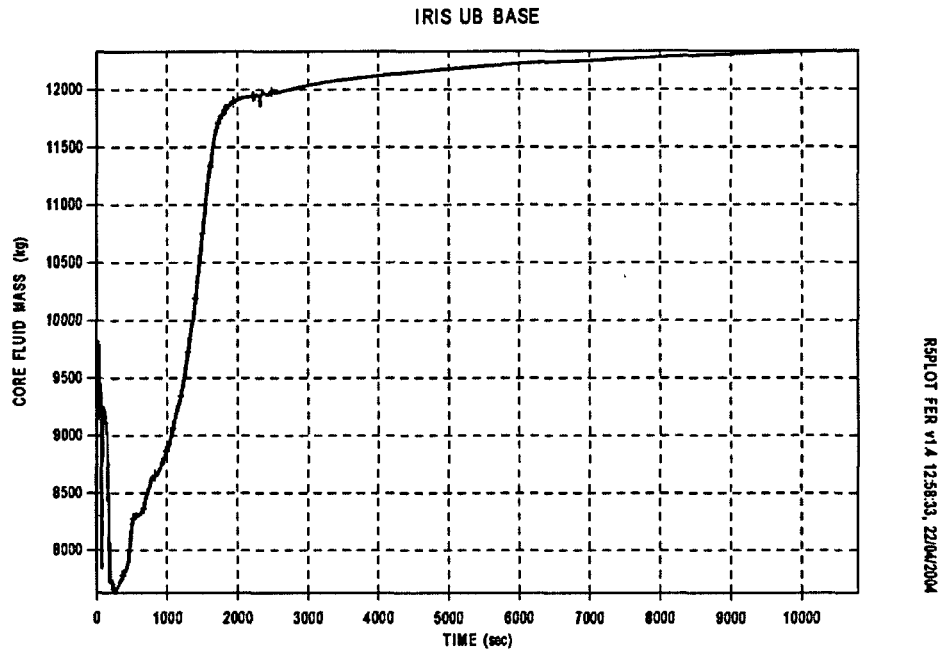

Figure 3.1-8 – Core Liquid Mass Inventory Transient. Base Case, Upper Break

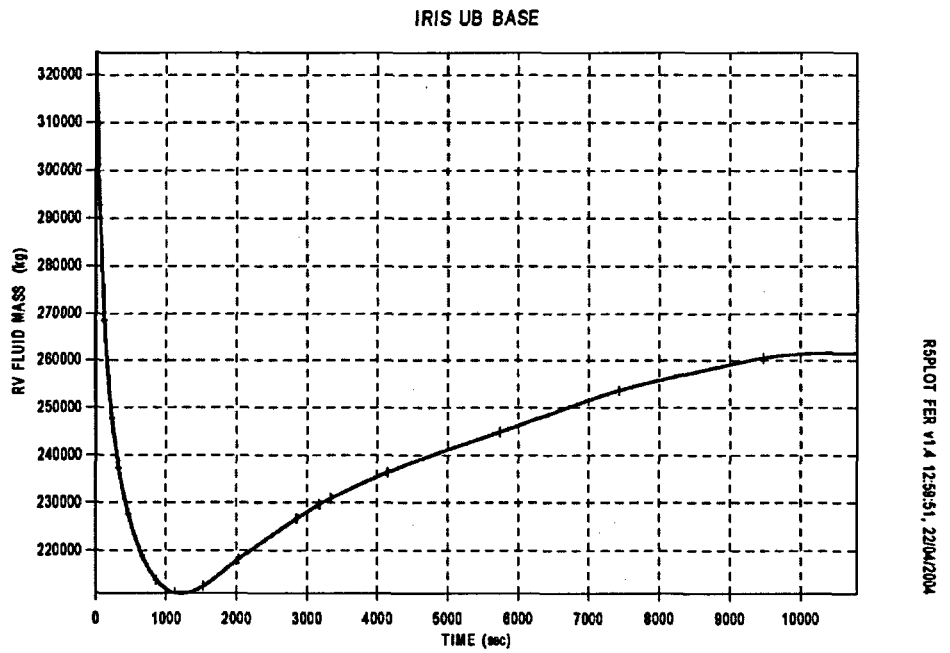

Figure 3.1-9 – Reactor Coolant System Inventory Transient. Base Case, Upper Break

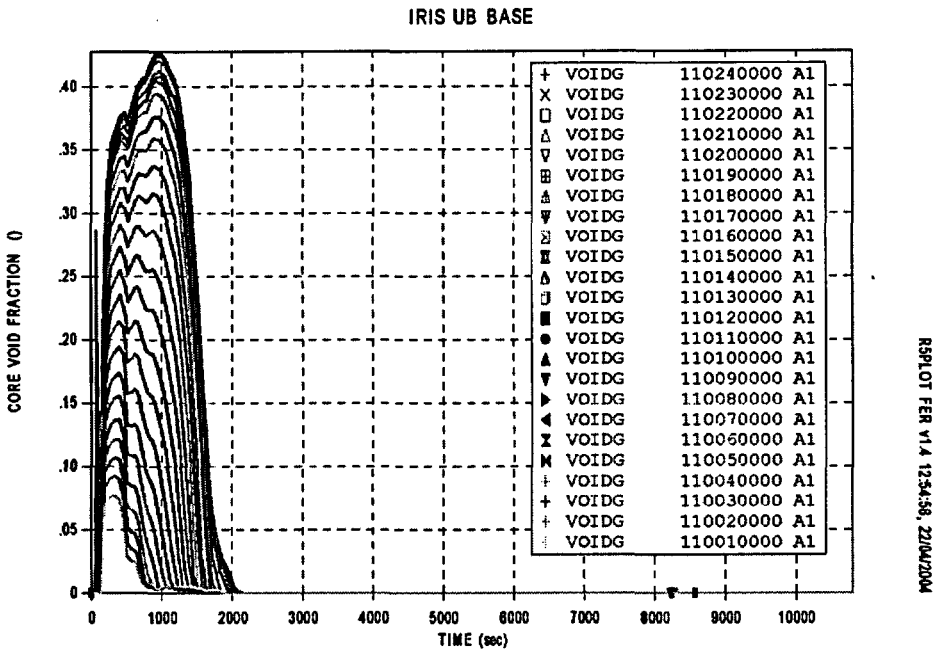

Figure 3.1-10 – Core Exit Void Fraction (volumes 1 to 24 in component 110) Transient. Base Case, Upper Break

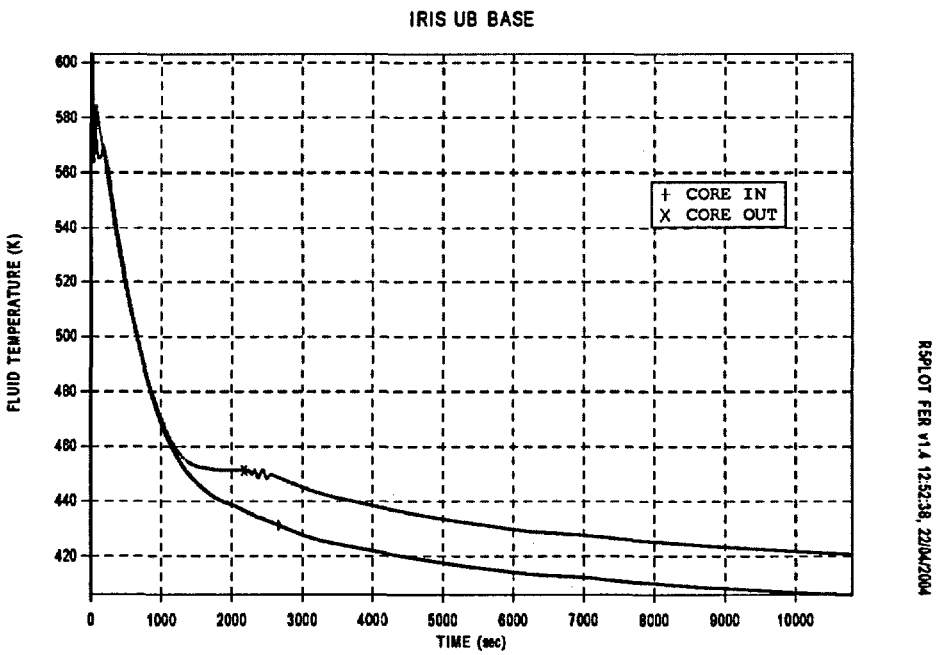

Figure 3.1-11 – Core Inlet and Outlet Temperature Transient. Base Case, Upper Break

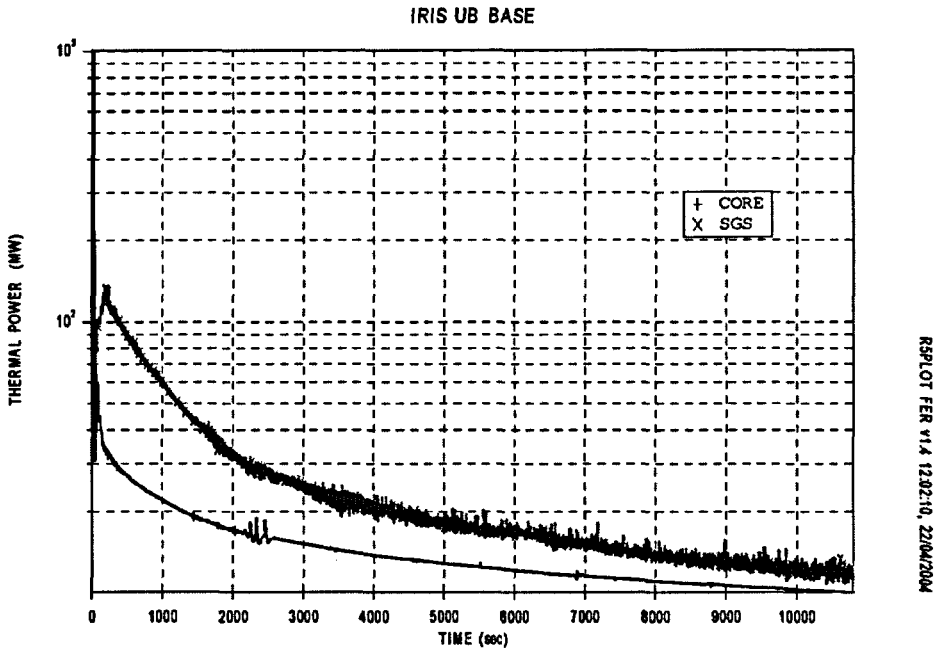

Figure 3.1-12 – Thermal Power in the core and total removed power in the SGs Transient. Base Case, Upper Break

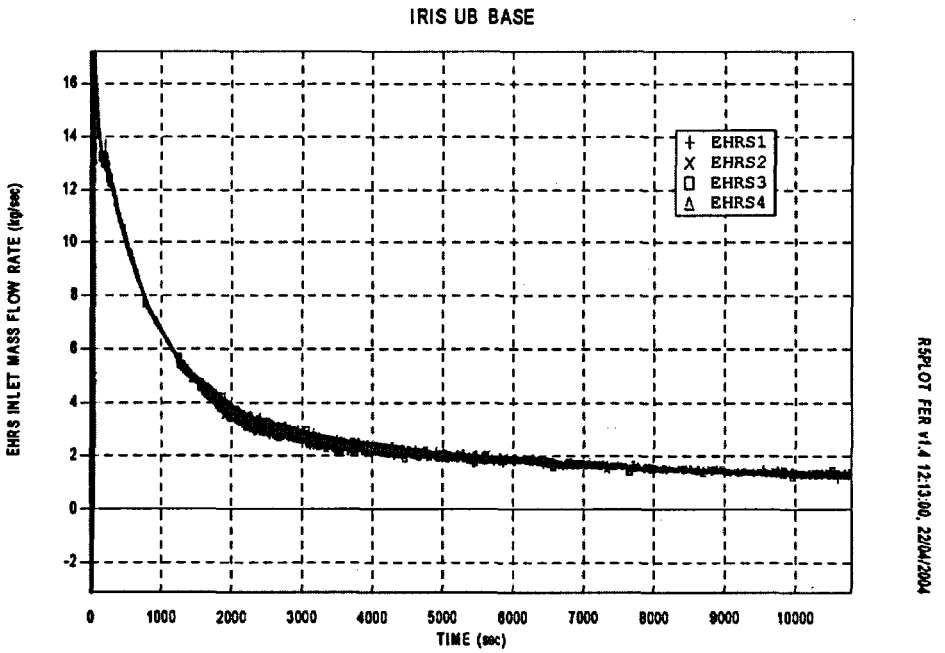

Figure 3.1-13 – EHRS System Flow Rate per loop Transient. Base Case, Upper Break

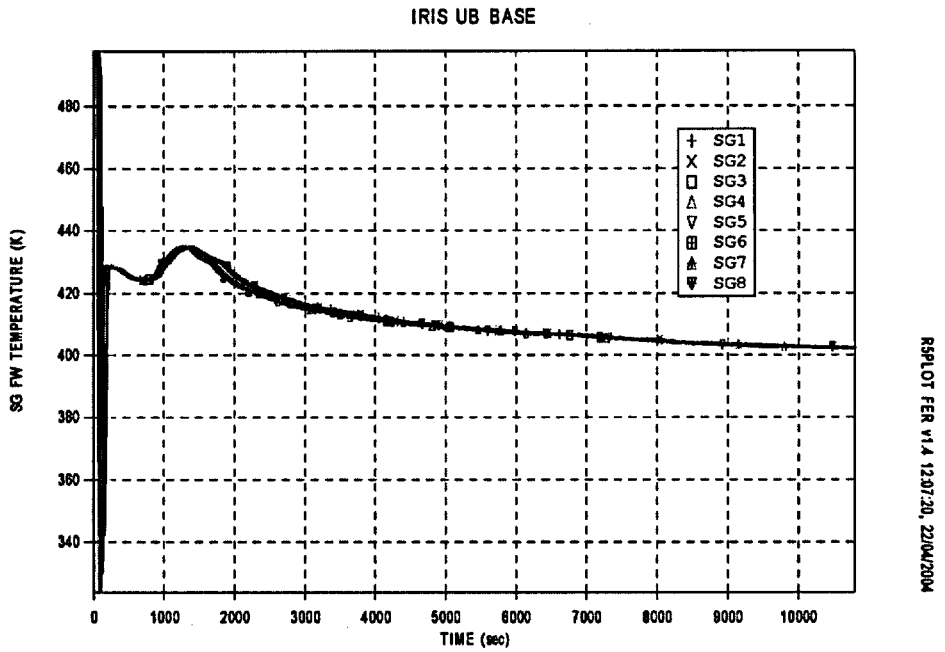

Figure 3.1-14 – EHRS temperature (SG inlet) Transient. Base Case, Upper Break

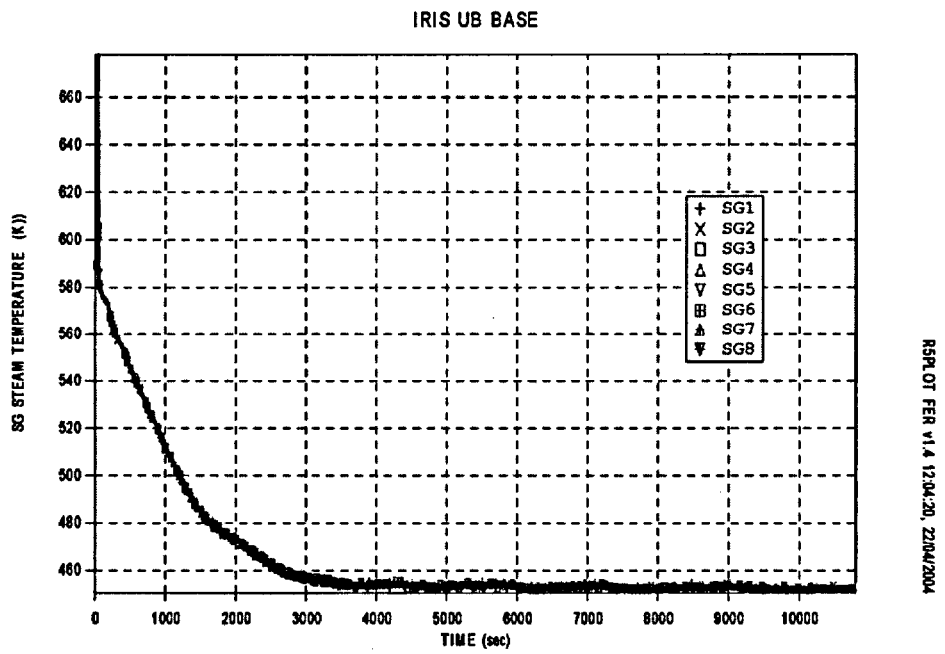

Figure 3.1-15 – EHRS Temperature (SG outlet) Transient. Base Case, Upper Break

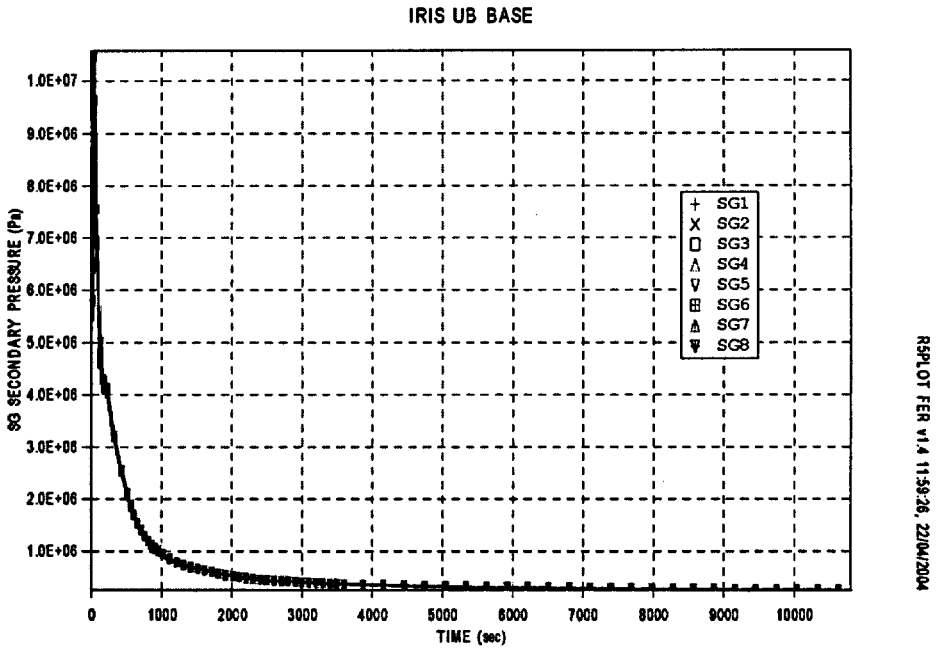

Figure 3.1-16 –EHRS Pressure Transient. Base Case, Upper Break

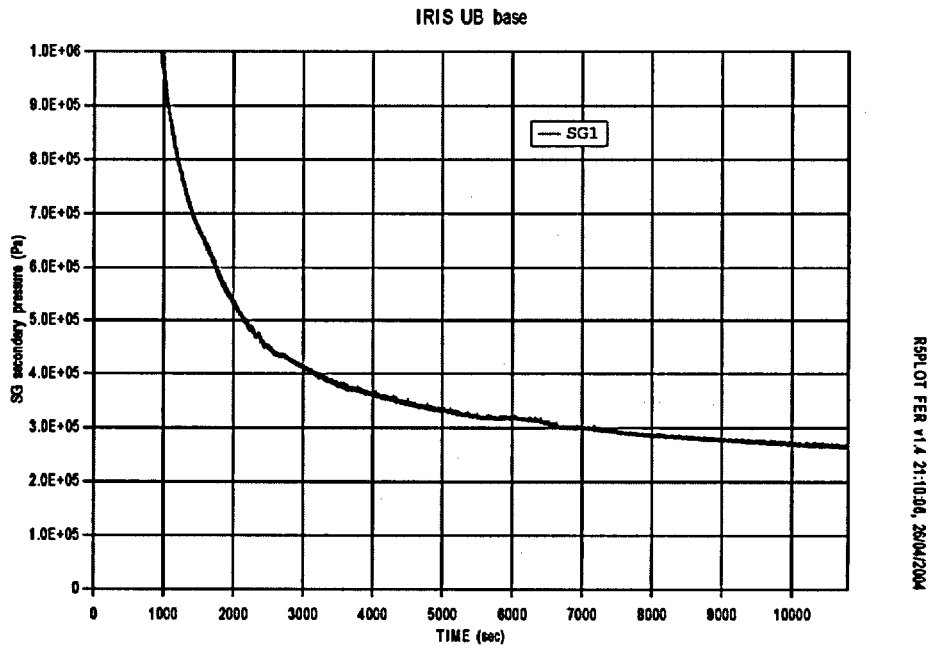

Figure 3.1-17 –EHRS Pressure Transient (Detail for SG#1). Base Case, Upper Break

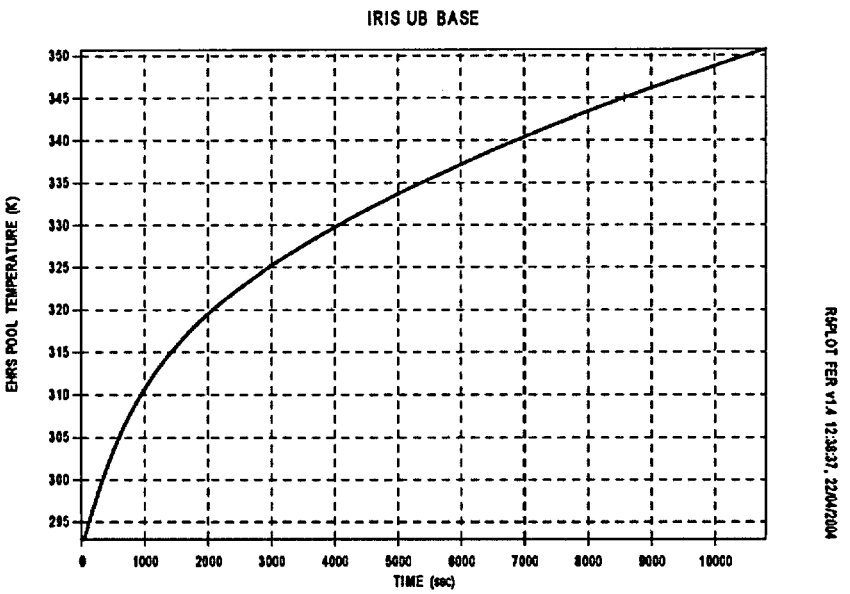

Figure 3.1-18 – RWST Pool Temperature Transient. Base Case, Upper Break

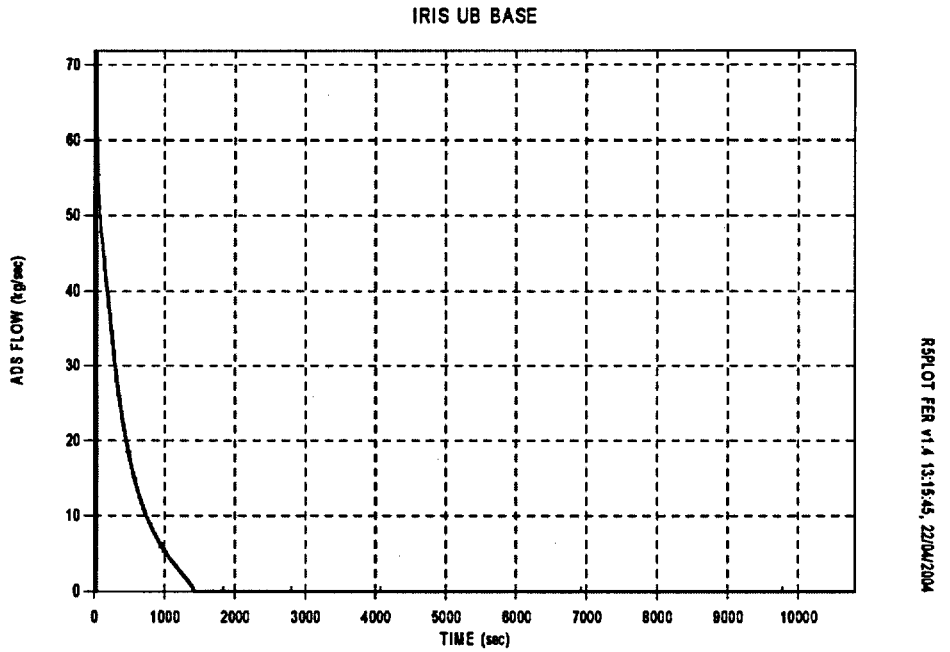

Figure 3.1-19 –ADS Flow Rate Transient. Base Case, Upper Break

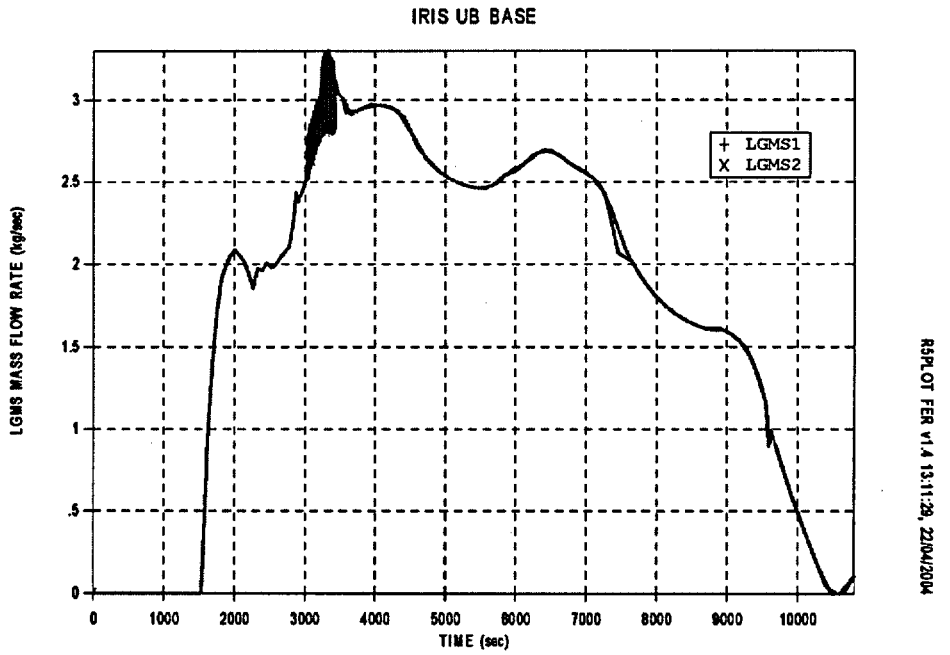

Figure 3.1-20 –LGMS Flow Rate (Suppression Pool to RCS) Transient. Base Case, Upper Break

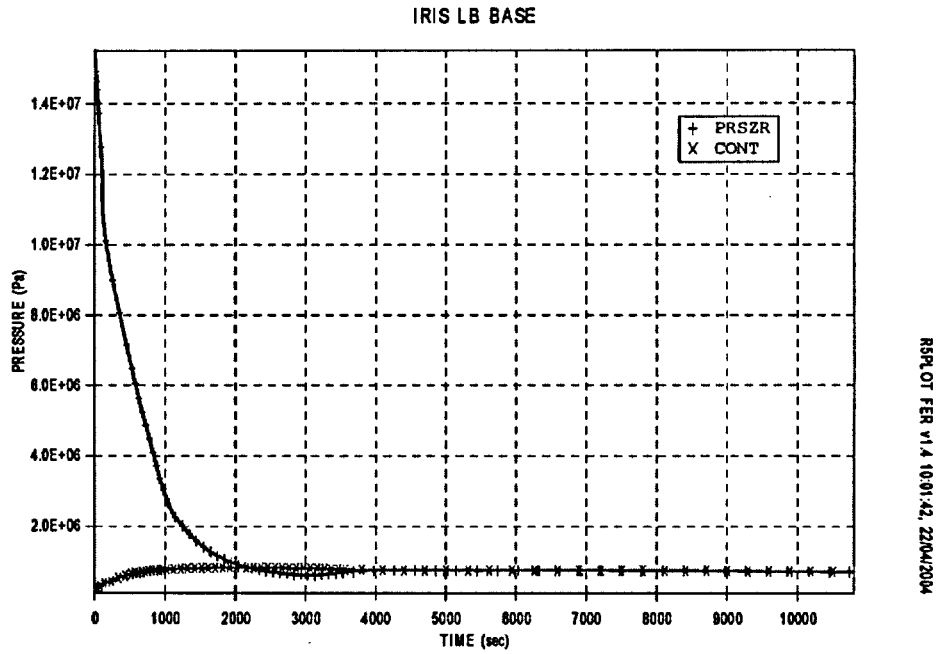

Figure 3.1-21 - Containment and Pressurizer Pressure Transient. Base Case, Lower Break

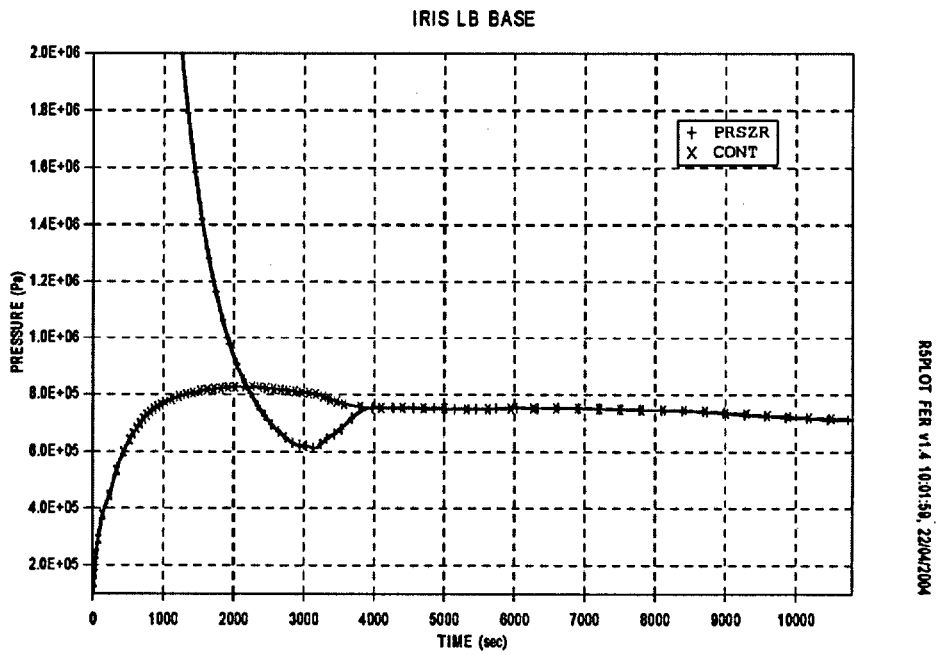

Figure 3.1-22 - Containment and Pressurizer Pressure Transient (Detail). Base Case, Lower Break

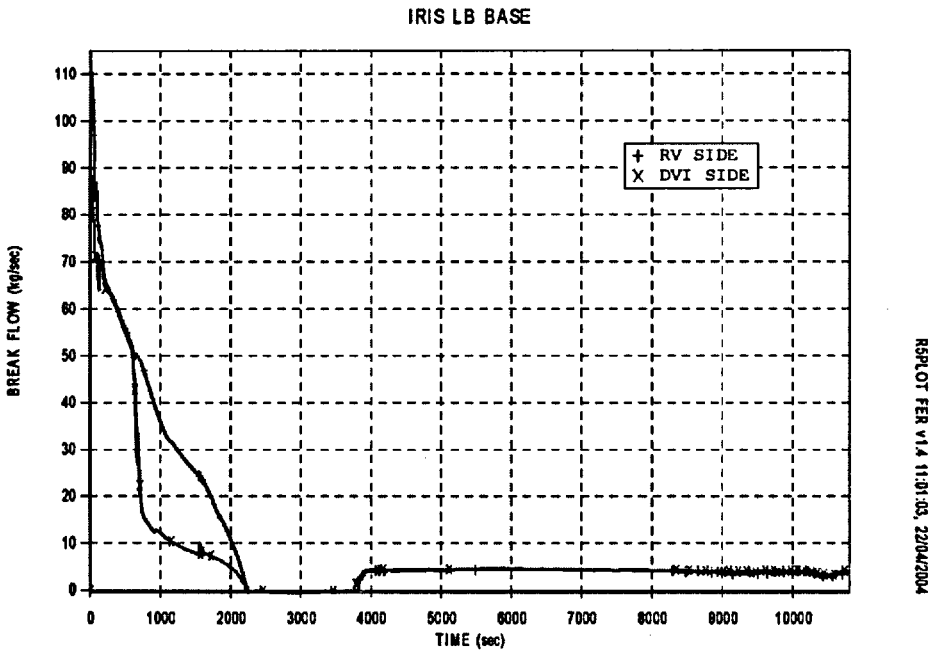

Figure 3.1-23 – Break Flow Rate Transient. Base Case, Lower Break

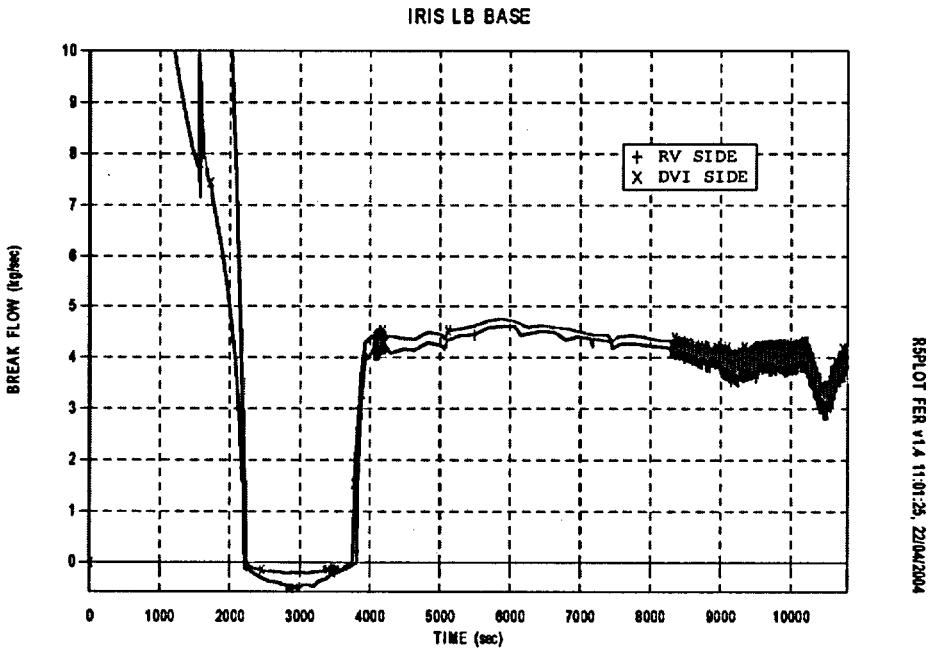

Figure 3.1-24 – Break Flow Rate Transient (Detail). Base Case, Lower Break

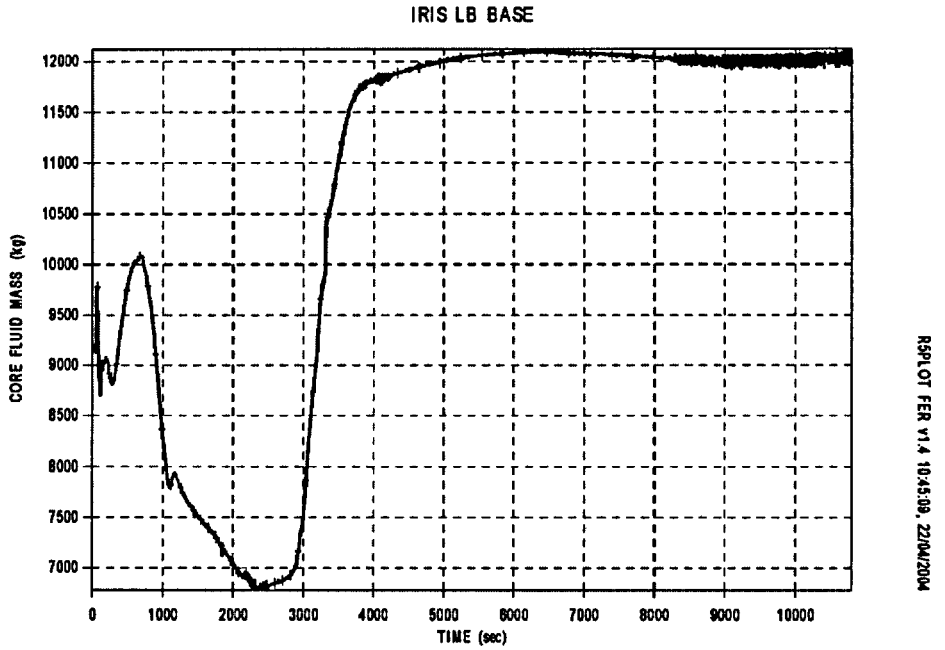

Figure 3.1-25 – Core Liquid Mass Inventory Transient. Base Case, Lower Break

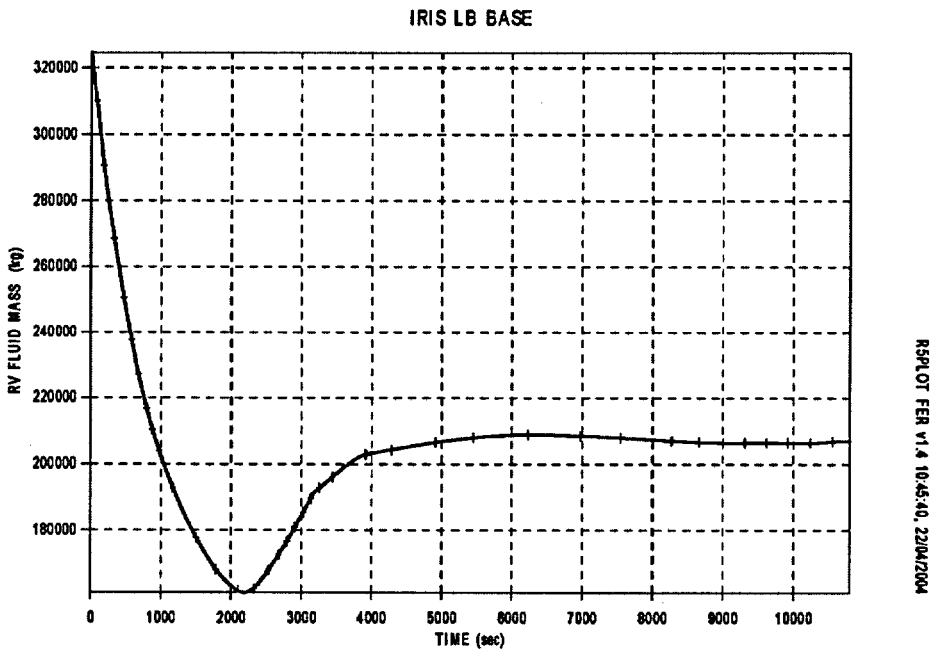

Figure 3.1-26 – Reactor Coolant System Inventory Transient. Base Case, Lower Break

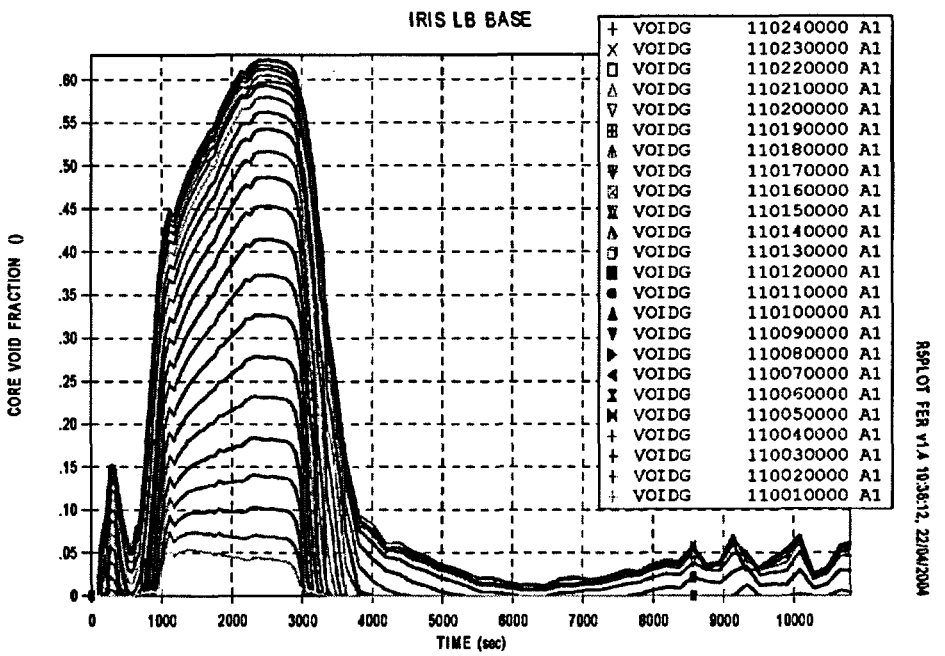

Figure 3.1-27 – Core Exit Void Fraction (volumes 1 to 24 in component 110) Transient. Base Case, Lower Break

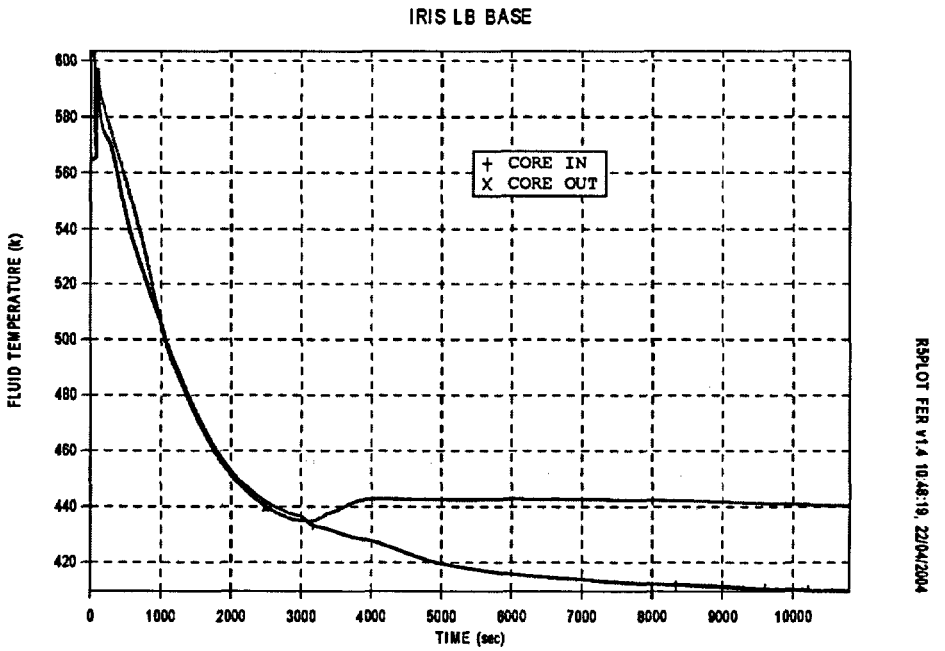

Figure 3.1-28 – Core Inlet and Outlet Temperature Transient. Base Case, Lower Break

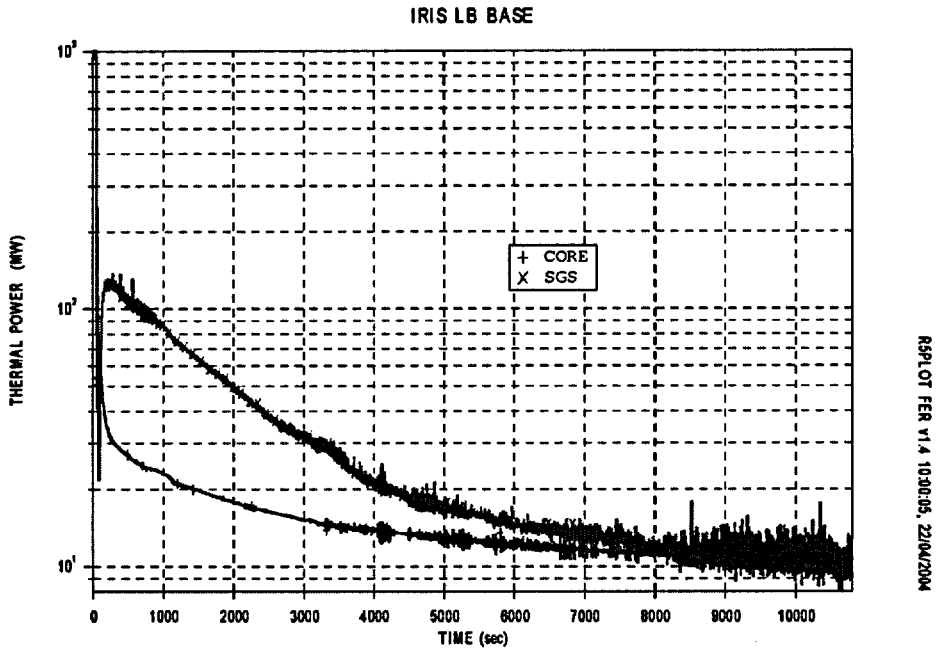

Figure 3.1-29 – Thermal Power in the core and total removed power in the SGs Transient. Base Case, Lower Break

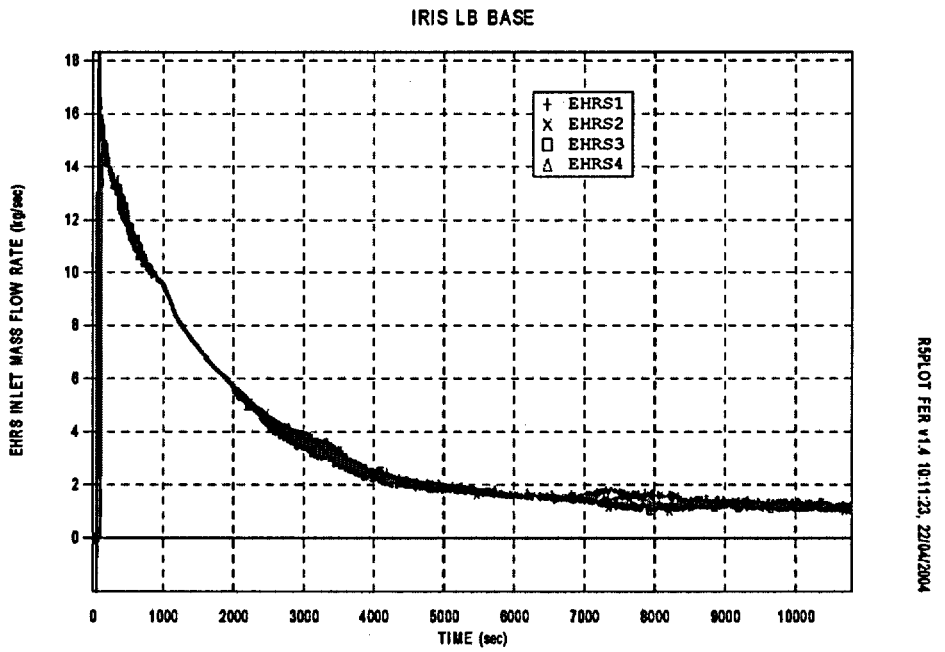

Figure 3.1-30 – EHRS System Flow Rate per loop Transient. Base Case, Lower Break

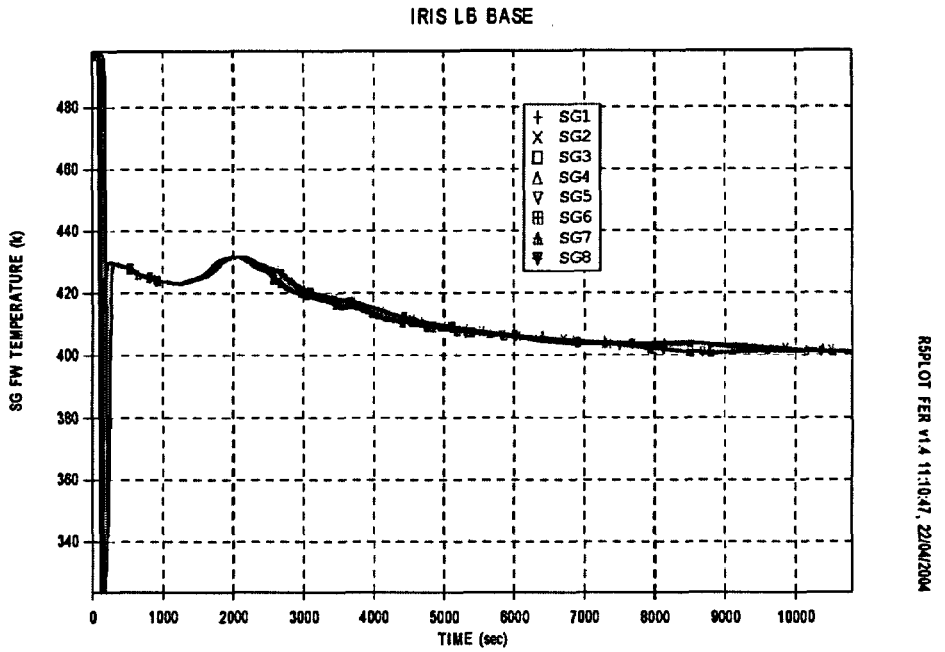

Figure 3.1-31 – EHRS temperature (SG inlet) Transient. Base Case, Lower Break

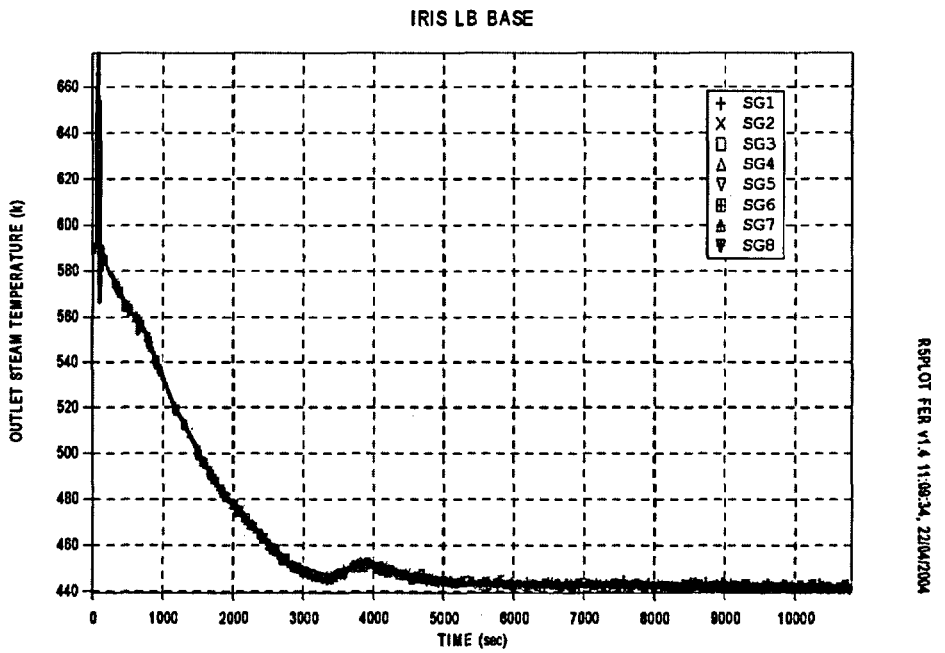

Figure 3.1-32 – EHRS Temperature (SG outlet) Transient. Base Case, Lower Break

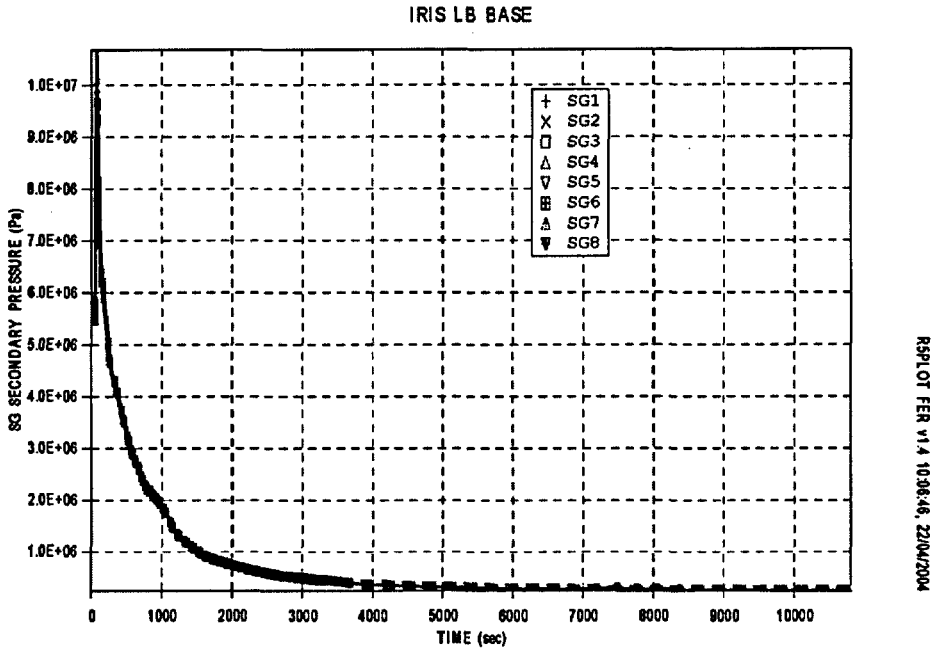

Figure 3.1-33 –EHRS Pressure Transient. Base Case, Lower Break

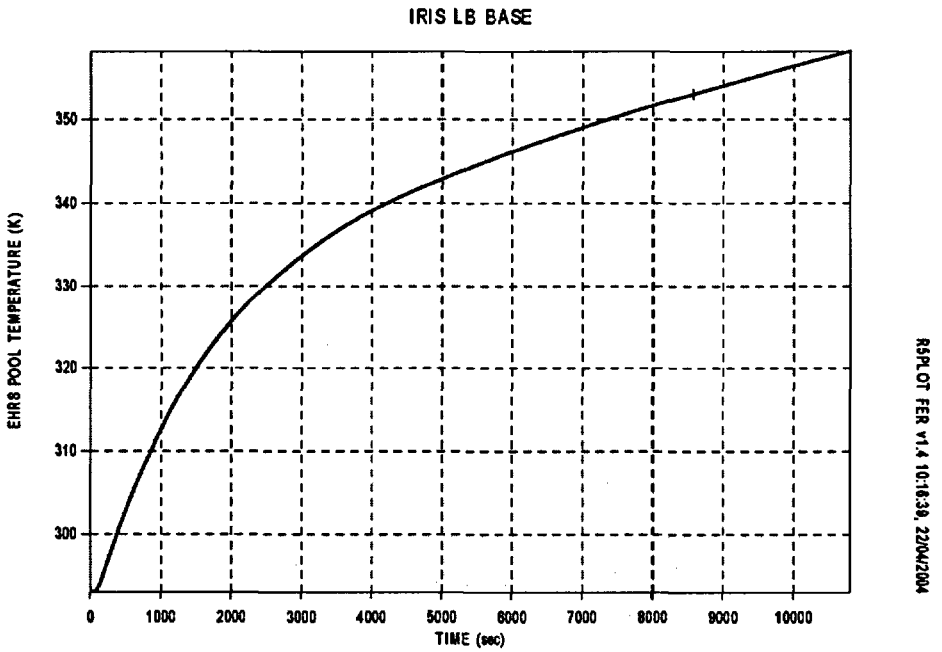

Figure 3.1-34 – RWST Pool Temperature Transient. Base Case, Lower Break

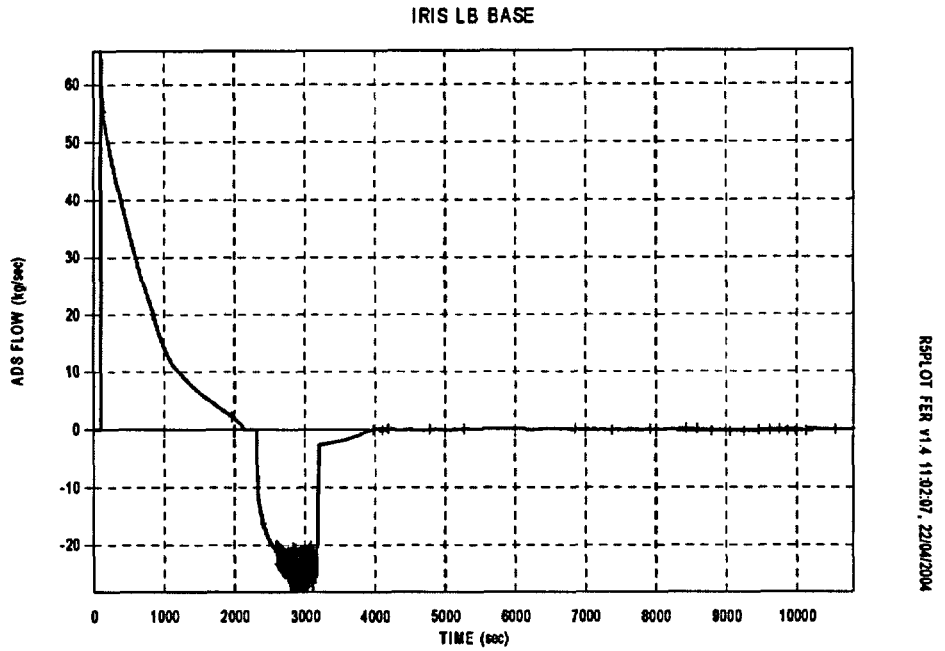

Figure 3.1-35 –ADS Flow Rate Transient. Base Case, Lower Break

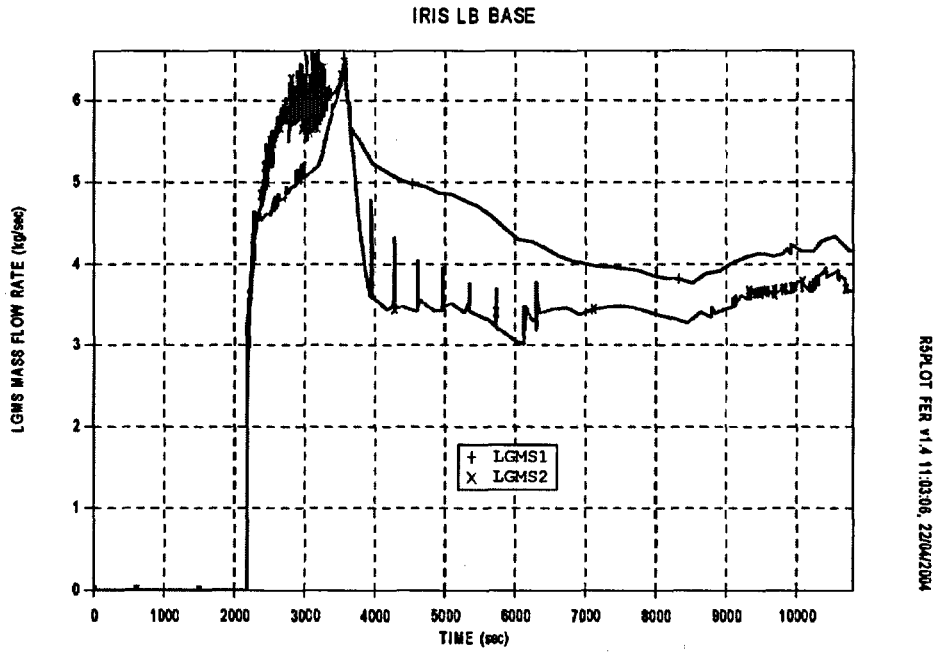

Figure 3.1-36 –LGMS Flow Rate (Suppression Pool to RCS) Transient. Base Case, Lower Break

[ ] (a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

[ ] (a, c)

[ ] (a, c)

(a, c)

3.2 EFFECT 1 – EHRS SENSITIVITY

As discussed in Section 1, the following cases where considered to evaluate the importance of the EHRS heat removal on the IRIS SBLOCA sequence:

[

]$^{(a,c)}$

The results for the evaluation of Effect 1 are shown in Figures 3.2-1 to 3.2-37.

3.2.1 Upper Break Analysis

[

]$^{(a,c)}$

3.2.2 Lower Break Analysis

[

]$^{(a,c)}$

(a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

(a, c)

3.3 EFFECT 2 – CONTAINMENT PRESSURE SENSITIVITY

In this section different cases were included to explore the sensitivity of the RCS inventory to the containment pressure. These cases are helpful in understanding the impact of containment pressure on the minimum RCS inventory, and were considered very important due to the complexity of predicting the containment pressure transients. As discussed in Appendix A, the containment pressure transient is in fact strongly influenced by the containment spatial geometry, by the composition of the steam/nitrogen mixture flowing to the PSS vents during the early blowdown, and by the amounf of mixing in the containment volume.

To evaluate the influence of the containment pressure on the RCS inventory, sensitivity cases were defined that can be divided in three different groups. [

]$^{(a,c)}$

The following cases are included in this assessment:

[

]$^{(a,c)}$

3.3.1 Upper Break Analysis

[

]$^{(a,c)}$

[

] (a, c)

[

]$^{(a,c)}$

[

] (a, c)

3.3.2 Lower Break Analysis

[

]$^{(a,c)}$

[

]$^{(a,c)}$

(a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

[ ] (a, c)

[ ] (a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

[ ] (a, c)

[ ] (a, c)

3.4 EFFECT 3 – SUPPRESSION POOL SENSITIVITY

The cases discussed in this section might be considered a subset of the overall sensitivity to containment pressure discussed in the previous Section. A separate section was chosen to focus on the pressure suppression system because of the importance that the suppression pool performance has on the containment response. In particular, the focus of this study was to identify the impact of different volumes and temperature of the pressure suppression system.

However, the results discussed in the previous section (that already provide a sufficient evaluation of the sensitivity of the RCS inventory to different containment pressurization scenarios) and of the analyses documented in Appendix A were considered to already provide a sufficient evaluation of the pressure suppression system effect on the containment pressure transient, and in particular identify phenomena other than volume and temperature of the pool as relevant to the pressure suppression system efficiency.

For this reason, a reduced test matrix was analyzed, as follows:

[

] (a, c)

3.4.1 Upper Break Analysis

[

] (a, c) These results confirm the conclusions provided in Appendix A, that identify the suppression pool and relative assumptions as having an important effect on the containment pressurization. Appendix A includes additional discussion of specific phenomena and model that affect the performance of the pressure suppression system.

As discussed in Appendix A, different assumptions relative to the relocation of nitrogen from the containment to the suppression pool can have a significant impact on the containment response. Specifically, if the nitrogen relocation is slow, the suppression pool will take longer to pressurize and more steam will be sparged into the PSS water, resulting in a decrease in the peak pressure. The current containment model is structured in such a way to force a very rapid relocation of nitrogen from the containment system to the pool. This is done by modeling the containment with two lumped volumes and connecting break and vents to different volumes. The effect of different assumptions relative to nitrogen relocation and the composition of the gas mixture entering the PSS during the early blowdown was assessed by moving the PSS inlet vents to the same volume where the break is located. As discussed in Appendix A and shown in Figure 3.4-5, the effect on the containment pressure is not negligible.

[

]$^{(a,c)}$

3.4.2 Lower Break Analysis

[

]$^{(a,c)}$

(a, c)

(a, c)

(a, c)

[ ] (a, c)

[ ] (a, c)

3.5 EFFECT 4 – ADS SENSITIVITY

The ADS performs the function of assisting the EHRS in depressurizing the RCS. The ADS is designed to provide adequate RCS depressurization capacity to mitigate the consequences of a break in the lower part of the reactor vessel.

[

]$^{(a, c)}$

3.5.1 Upper Break Analysis

[

]$^{(a, c)}$

3.5.2 Lower Break Analysis

[

]$^{(a, c)}$

(a, c)

(a, c)

[ ] (a, c)

(a, c)

(a, c)

[ ] (a, c)

3.6 EFFECT 6 – DECAY HEAT

3.6.1 Upper Break Analysis

The effect of an increased decay heat is similar to that of a reduced EHRS performance, as explored in Section 3.2. This is due to the fact that [

]$^{(a,c)}$

[

]$^{(a,c)}$

[

]$^{(a,c)}$

3.6.2 Lower Break Analysis

[

]$^{(a,c)}$

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

3.7 EFFECT 7 – BREAK SIZE SENSITIVITY

Break flow rate is naturally a critical influence on the IRIS SBLOCA scenario response where larger breaks tend to give a faster transient evolution, with higher containment peak pressures and lower RCS inventories. On the other hand, during a smaller break, the actuation of the mitigation features can be significantly delayed, and this can affect negatively the system response.

The following cases were included in the test matrix
[

] (a, c)

3.7.1 Upper Break Analysis

[

] (a, c)

3.7.2 Lower Break Analysis

[

]$^{(a,c)}$

(a, c)

(a, c)

(a, c)

[ ] (a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

[ ] (a, c)

(a, c)

[ ] (a, c)

[ ] (a, c)

[ ] (a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

(a, c)

APPENDIX A

The following Appendix contains a paper presented at the 2004 Dubrovnik Conference. This paper provides additional insights on the IRIS containment response during a SBLOCA and complements the sensitivity study documented in this report.

INTEGRATED ANALYSIS FOR A SMALL BREAK LOCA IN THE IRIS REACTOR USING MELCOR AND RELAP5 CODES

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ABSTRACT

The pressurized light water cooled, medium power (1000 MWt) IRIS (International Reactor Innovative and Secure) has been under development for four years by an international consortium of over 21 organizations from ten countries. The plant conceptual design was completed in 2001 and the preliminary design is nearing completion. The pre-application licensing process with NRC started in October, 2002 and IRIS is one of the designs considered by US utilities as part of the ESP (Early Site Permit) process.

This paper's focus is on the use of well known computer codes for integrated (reactor vessel and containment) calculations of the IRIS response to a small break loss of coolant accident (LOCA). In IRIS, large break LOCA events are eliminated by the use of a layout configuration in which the reactor vessel contains all the reactor coolant system components including the core, control rod drive mechanisms, pressurizer, steam generators, and coolant pumps. Thus the IRIS configuration has no large loop piping; also, no pipes with a diameter greater than 0.1 meters are part of the reactor coolant system boundary. For small break LOCAs, IRIS features an innovative mitigation approach that is based on maintaining coolant inventory rather than designing high and low pressure injection systems to provide makeup coolant to the reactor to maintain core cooling.

The novel IRIS approach requires development of evaluation models that are different from those used for the current generation of pressurized water reactors. An analysis of small break LOCAs for IRIS is documented in two companion papers, and has been developed using a preliminary evaluation model based on the explicit coupling of the RELAP5 and GOTHIC codes. The objective of this paper is to compare the results obtained via the coupled RELAP/GOTHIC code with different computational tools. A reference case from the preliminary IRIS safety assessment was selected, and the same small break LOCA sequence is analyzed using both RELAP5 and MELCOR codes in two integral separated calculations. The paper also describes the phenomenology of the containment spherical vessel and its strong interactions with the IRIS reactor vessel.

These two integrated analyses have been developed for IRIS in order to better identify the characteristics of the integrated plant response to a SBLOCA.

1 INTRODUCTION

The IRIS reactor [1], [2] is being developed by an international team, led by Westinghouse Science & Technology, that includes universities, research centres and industrial companies from ten countries. The plant design is characterized by a reactor coolant system that is fully integrated within a single pressure vessel, which makes the nuclear island more compact and eliminates the possibility of large releases of primary coolant. The emphasis is on utilization of existing nuclear technology, modularity, elimination (or, at least, reduction) of accident initiators, and passive systems to cope with the consequences of

accident events. The demonstration of the reactor's capability to successfully respond to a broad range of potential initiating events without the occurrence of appreciable core damage, requires a deep knowledge of all the thermal-hydraulic phenomena involved in each postulated sequence and the availability of appropriate computer tools, capable of performing such a kind of system analysis, characterised by an integrated approach to the whole reactor system. A preliminary study has been performed [1] that addresses the main design basis events for the IRIS reactor aimed to assist in the definition of requirements for IRIS evaluation models and the development of a complete set of phenomena identification and ranking tables. The results of these analyses for non-LOCA events are summarized in Reference [3], while the analysis of a small break LOCA event for IRIS is discussed in two companion papers [4, 5].

The present work is a contribution to the activities in progress to define an IRIS Evaluation Models for Small Break LOCA Safety Analyses, where particular emphasis is given to the development of an appropriate procedure and code selection to analyze the coupled IRIS reactor vessel and containment [6]. Due to the strong coupling that exists between the reactor coolant system (RCS) and containment system (CS) in the IRIS plant, the choice has been to develop two integral models of the plant as required by the RELAP5/Mod. 3.3 [7] and MELCOR 1.8.5 codes [8], [9]. One small break LOCA sequence simulated with RELAP/GOTHIC has been selected and is analyzed with both RELAP5 and MELCOR. These analyses allowed to better identify the characteristics of the integrated plant response to a SBLOCA and allowed to compare the thermal-hydraulics results of RELAP5, a well known system code which is not normally applied to containment studies, with the results from MELCOR, which was developed for source term studies. The results obtained using a third code, FUMO [10], a fast-running program aimed to provide a flexible support to the evaluation model development and assessment procedure of a nuclear plant, are also presented to further support the interpretation of the analysis results.

The objective of this study is to improve the understanding of the IRIS response following a small break LOCA event, by comparing code differences and identifying physical models that can have an impact on the system response. Also, by comparing the FUMO and MELCOR results with the RELAP/GOTHIC model, the possiblity of using these simpler models for fast running sensitivity studies was assessed. While the analysis presented in the companion paper [6] is focused on an overall review of the SBLOCA response of IRIS, the focus of these analyses is to expand on that initial study by focusing the attention on some key containment analyses issues that were identified during the benchmark of the different codes.

2. NODALIZATIONS OF THE IRIS PLANT

Models of the IRIS reactor coolant system and containment systems for the RELAP5, MELCOR and FUMO codes were developed for this study as discussed in the previous section, and are described in some detail as follows.

2.1 RELAP Integral Nodalization

A nodalization of the Westinghouse IRIS containment for the computer code RELAP5 has been realized in analysis for a postulated break of a 4" chemical volume and control system (CVCS) line, as discussed in Section 3. This effort represents a different approach to the SBLOCA analysis by using only the RELAP5 code, as compared to the IRIS RELAP5 nodalization created by the University of Zagreb which uses a coupled RELAP5/GOTHIC model as reported in reference [4].

While the overall structure is relatively simple and straightforward, the components discretization is sufficiently detailed in order to take into account all the important phenomena. A sliced approach was used in the discretization of the reactor vessel and it is maintained also for the containment system. It is specifically suitable for calculating scenarios that imply fluid stagnation in different parts of the system, or scenario characterized by low driving forces (i.e. natural circulation). The sliced approach is presented in Figure 1 where the complete nodalization is shown. The containment model has been developed using only "pipe" and "single junction" components and removing all the boundary conditions modeled with time dependent volumes used for the GOTHIC boundary conditions. The containment model can be divided in the following components: cavity, drywell (divided in two regions), vent, containment pressure

suppression system (CPSS) tanks, the air tank and the pipe connecting the system previously mentioned). This is the same approach used in the GOTHIC model described in Reference [4].

The cavity region has been modeled with a pipe component connected with the long gravity make-up system (LGMS) lines and with the drywell. The drywell region has been subdivided, consistent with the GOTHIC model, into two regions.

The vents are pipe components that connect the drywell with pressure suppression tanks. During the blow-down phase, in the LOCA event, the steam released to the drywell from the primary system is conveyed by the vents to the bottom of the suppression pools, where the steam is condensed. The ADS line has also been modeled.

All these systems are initialized considering the non-condensable gas (nitrogen) pressurization assuming 30% of humidity in the CPSS tank and vents. The total number of the volumes and the junctions are 2128 and 2192 respectively.

It is important to notice that, as discussed in Reference [5], a modified version of the RELAP code is being used in IRIS safety analyses. This version has only minor modifications from the RELAP5 Mod3.3 code [7], moslty to provide correction of some code deficiencies and to expand the capability of the control variables to allow simplifications in the input deck.

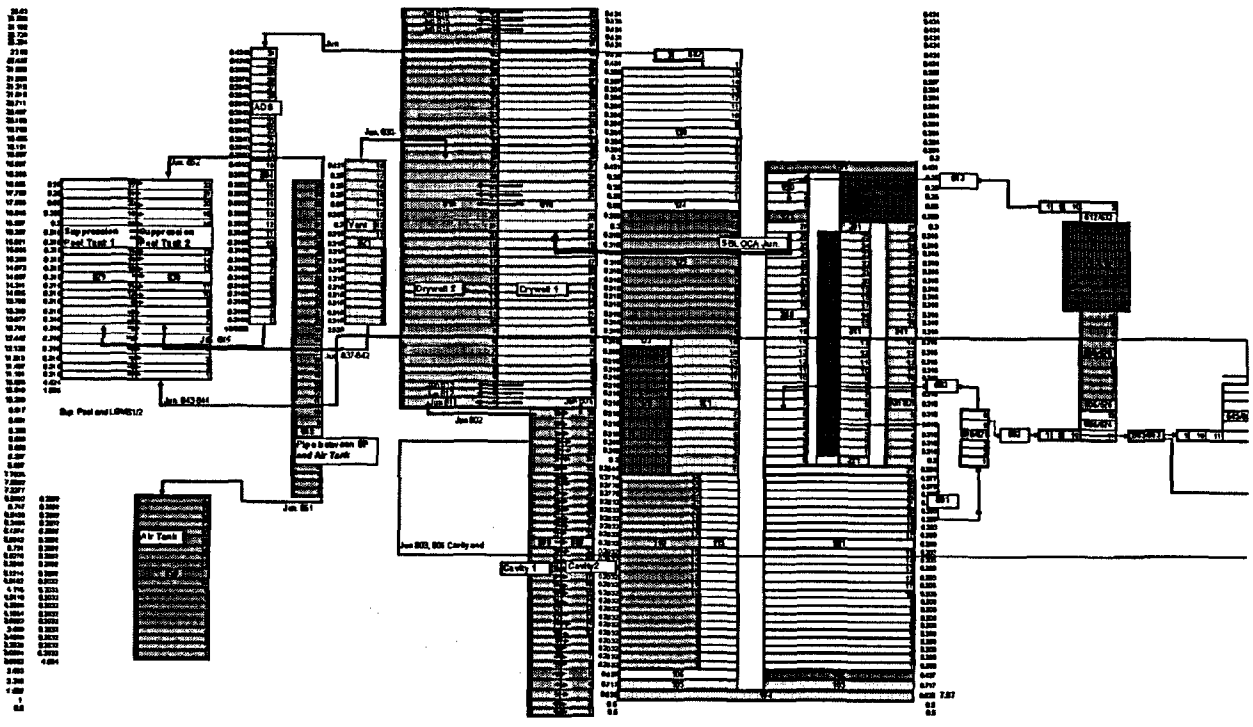

Figure 1. IRIS plant nodalization for RELAP integrated calculation.

2.2 MELCOR Nodalization

The IRIS nodalization for the MELCOR 1.8.5 code is presented in Figure 2. It is a rather detailed schematization of the plant which allows for an integral calculation of the RCS and CS. Four reactor coolant pressure/steam generator (RCP/SG) modules have been modeled, simulating respectively two single SGs, a module with two SGs and a double-module with four SGs, respectively. This choice was considered the best compromise between the calculation time requirements and the necessity to address possible asymmetries in coolant flow during an accident. The secondary system is modeled only up to the main feed and isolation valves, while the SGs were modeled with a high level of detail. Particular care has also been devoted to the core simulation. The core schematization is comprised of two radial rings and six axial levels ; four thermal hydraulic levels are used in each ring of the core region, with three core cells axially in each thermal-hydraulic control volume. Radial and axial flow paths in the core region allow for the prediction of 2-D flow patterns. Heat structures representative of the control rod guide tubes and upper tie plate in the upper plenum have been added to the model to permit condensation heat transfer and coolant recirculation inside the vessel. The input deck was developed using default MELCOR modeling parameters where possible, and allows for a complete description of a possible severe accident progression. Emergency safety features are also explicitly modeled, including the Emergency Heat Removal System (EHRS), the Emergency Boration Tanks (EBTs) the Refueling Water Storage Tank (RWST) and those IRIS safety features (automatic depressurization system (ADS), pressure suppression system, long term core make-up system) that cause a strong coupling between the integral RCS and the CS. The CS has been simulated with six control volumes (three volumes for the large spherical dome and cavity, two for the suppression tanks and air volumes and one for the venting pipes) using an approach similar to the one used for GOTHIC and FUMO. The results compared in this document were obtained considering an adiabatic CS, even though some calculations have also been performed considering a rough simulation of heat sinks.

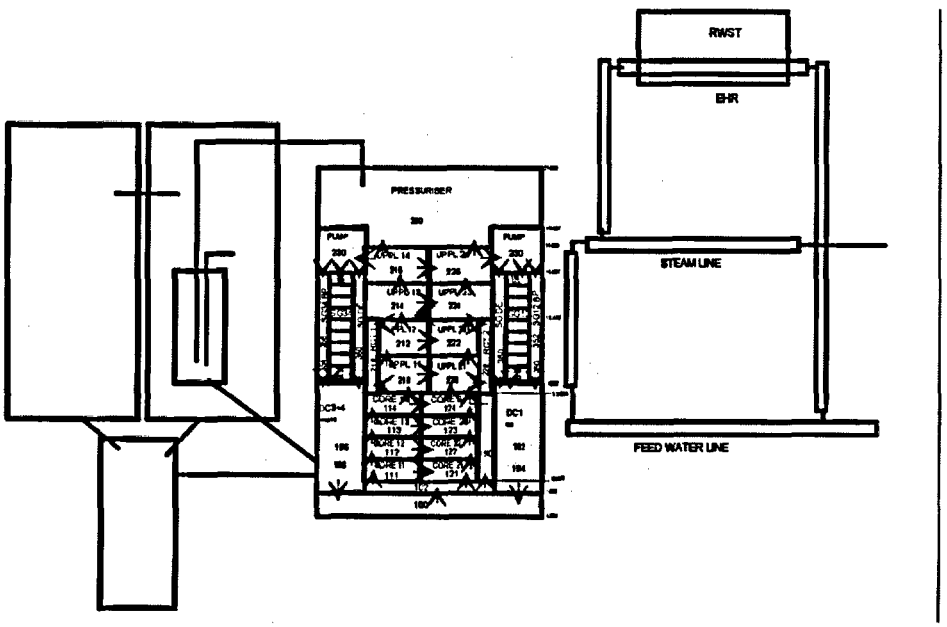

Figure 2. IRIS plant nodalization for the MELCOR code.

2.3 FUMO Nodalization

The FUMO code [10], a code initially intended to analyze dry containment systems, has been modified in order to allow the simulation of pressure suppression systems and has been integrated with other codes, specifically dedicated to the simulation of the primary system behavior. New models have been recently added in order to make it able to simulate the whole thermal-hydraulic transient in a Generation IV plant. In the framework of a research co-operation with the Westinghouse Technology Centre and the Polytechnic of Milan, FUMO has been employed in the simulation of some postulated accidental sequence in the IRIS plant. This activity highlighted the capabilities of the code as a valid support tool for both the design and the analysis of a new reactor concept.

The IRIS nodalization for the FUMO code is presented in the Figure 3. It is a simple schematization of the plant which allows for an integral calculation of the RCS and CS. The primary system with the eight RCP/SG modules has been modeled with two control volumes. Also the whole secondary side has been simulated with two control nodes, which are able to take into account the actuation of the EHR system and its operation. The EBT and ADS line are also modeled, along with a simplified actuation logic. The CS model is very similar to the MELCOR one and is able to simulate the long term gravity makeup system, with two junctions linking the suppression pool and the containment cavity to the RCS.

This simple nodalization allows a fast simulation of transients and provides a reasonably good qualitative level of detail of the plant response. Another characteristic of the simulation performed with FUMO is the reduced computational time (a ratio of about 1:100, with respect to MELCOR and RELAP codes has been generally verified), which allows for the execution of a large number of sensitivity analyses or different transients. Moreover, the detail of the FUMO nodalization can be easily increased, also including the simulation of the logic of actuation of various accident mitigation signals, or simple models of others containment emergency safety features.

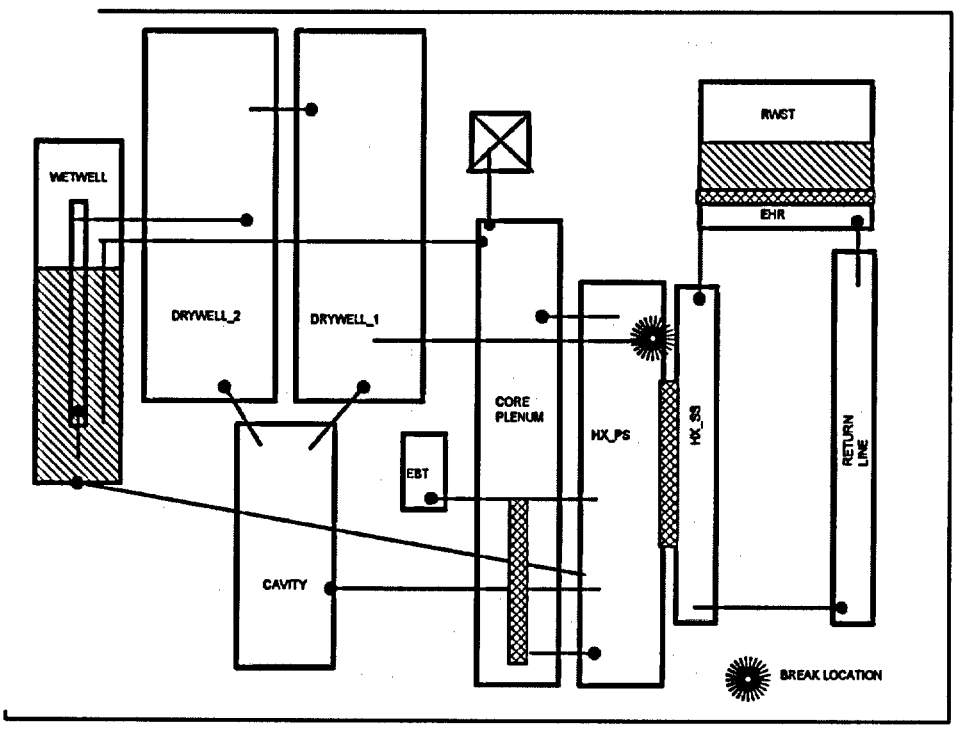

Figure 3. IRIS plant nodalization for the FUMO code.

3. RESULTS

The SBLOCA analyses for IRIS using the RELAP/GOTHIC code are discussed in two companion papers [4], [5] and are not discussed here in detail. A single case from those analyses was selected for this benchmark exercise: namely a 4" break of the chemical and volume control system, representing the rupture of the largest single pipe connected to the IRIS RCS. The break area is 4" (0.0081 m^2) and is located 15.6 m above the bottom of the vessel.

As discussed in Reference [5], the IRIS SBLOCA transient can be divided in three distinct phases. First, the blowdown phase is defined as the period during which the RCS pressure is reduced and the CS pressure increases until the RCS and CS pressures equalize. This phase can be further divided in two portions, before and after the transition from critical flow to subcritical flow through the break. The blow down phase is followed by a depressurization phase. During this phase the CS and RCS pressures are coupled and the coupled system is depressurized by the EHRS. From a phenomenological point of view, the depressurization phase should cover all the relevant phenomena that will occur during the depressurization of the coupled system (flooding of the cavity from the suppression pool, makeup of coolant from the suppression pool to the vessel). Finally, a long term cooling phase is established. This phase covers the long term cooling of the coupled CS and RCS and will be used to verify that sufficient water is provided by gravity driven flow from the reactor vessel cavity to the reactor vessel, and that the refueling water storage tank is correctly sized to provide seven days of heat sink for the EHRS, without requiring any operator action.

For all the calculations considered in this paper it was assumed no heat transfer from the containment atmosphere to the containment structures and to the external environment. This was considered an adequate simplification for this initial benchmark of the different codes, and additional cases with a complete CS model, including thermal structures, will be the object of future investigations. It is important to notice that simplification of the model allows a better comparison of the different relevant phenomena, by minimizing the potential for the various differences between the codes to compensate and offset each other.

In all calculations the large drywell is modeled by subdividing it in three sub-volumes, simulating the cavity and two similar halves of the upper region, in order to better address atmosphere natural circulation flows. The six suppression pools and the related vents are instead simulated with two single control nodes, respectively connected to the lower air tanks node and to the drywell. As explained below, this simplified nodalization of the IRIS CS can lead to quite different predicted responses to a SBLOCA sequence, depending on which of the two drywell control volumes the vents are connected to. The comparison of the base case presented in this paper is related to calculations performed with the same CS configuration, with the vents located in the drywell region that is the farther from the break.

3.1 Transient Overview and Sequences of Events

The sequence of events for a IRIS SBLOCA is discussed in detail in Reference [5], and only a brief summary is provided here. Following the initiating event, the LM-signal is rapidly actuated on a coincident low pressurizer pressure and high containment pressure. As a consequence the containment penetrations are isolated while the four EHRS subsystems are actuated by closing the main feed and steam isolation valves, and by opening the fail-open valves in the EHRS return lines from the EHRS heat exchangers connected to the SG feedlines. Finally the ADS is actuated and EBT discharge isolation valves are opened.

Table 1 shows the comparison between the four calculations relating to the sequence of events: the differences on the reactor trip signal are due to the differences on the break mass flow-rate predictions, affecting the level and pressure behavior in the primary system (both MELCOR and FUMO, as discussed later, predict a lower break mass flow rate and thus a slower sequence of events). Another difference is relative to the time to reach the LM-signal setpoint. The lower CS pressurization rate for the RELAP integrated model leads to a delay in the LM-signal on high CS pressure. The reasons for this lower CS pressurization rate predicted by RELAP are also discussed later in this paper. The different codes present some significant differences in the prediction of the end of the blowdown phase (RCS and CS pressure

equalization). The shorter duration of the MELCOR and FUMO blowdown phases would seem in contradiction with the lower break flow predicted by the codes as discussed above. However, as discussed in the following, both FUMO and MELCOR predict a higher heat removal through the EHRS than RELAP and this explains why for these two codes the lower mass flow rate at the break is offset by the increased heat removal and thus leads to a faster depressurization. A detailed analysis of the differences between the various codes is provided in the following sections.

Table 1. time sequence of events for IRIS SBLOCA in the CVCS line.

Event	RELAP/GOTHIC	RELAP	MELCOR	FUMO
Break occurs	0.0	0.0	0.0	0.0
Low pressurizer level setpoint	10.6	11.4	20.0	15.6
Low pressurizer pressure signal			16.6	22.7
High containment pressure signal			19.2	16.7
SCRAM beginning	12.6	13.4	18.6	17.6
LM signal setpoint reached	19.7	22.8	24.4	27.9
EHRS valves completely open	33.7	36.8	38.4	37.9
EBT valves completely open	29.5	39.	41.4	42.9
ADS valves completely open	51.7	54.8	56.4	57.9
CS and RCS pressure equalization	1600.	1820.	≈1150.	≈1600.
Injection from the SP to the RCS begins	≈1600.	≈1900.	≈1150.	≈1650.
Containment flooding from SP starts	≈3000.	≈2100.	≈1450.	≈2300.

It is important to notice that another set of differences between the different codes analysis is relative to minor input differences between the various codes. For example a difference in the main steam isolation valves model was identified: the RELAP/GOTHIC model discussed in Reference 4 includes a complete description of the reactor protection system, and thus predicts an early steam line isolation as soon as the high containment pressure signal is reached. While in the RELAP integral calculation the same trip logic has been used, in the other models developed for this exercise only a simplified protection system is modeled, and for example the main steam isolation valves signal is thus generated later only following the LM-signal as part of the EHRS actuation sequence. While the impact on the transient response is small, the effect on some local plant parameters can be significant: Figure 4 shows the steam generator header pressure, and it is evident that the RELAP/GOTHIC and RELAP models predict a much higher steam line pressure due to the early closure of the main steam isolation valves.

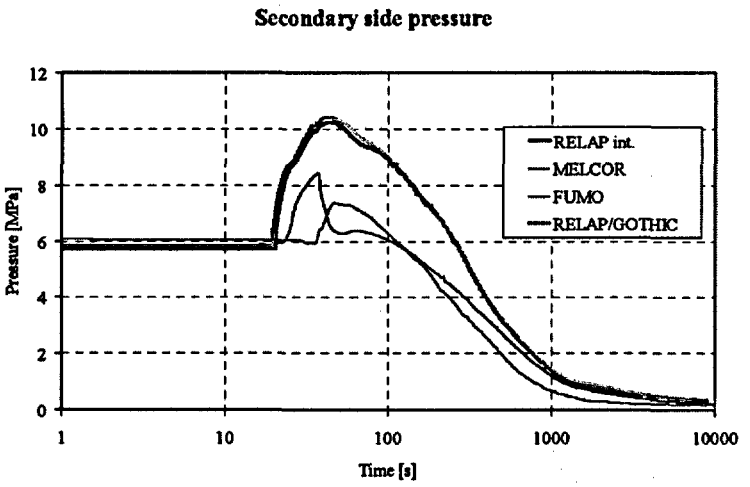

Figure 4. SBLOCA on the IRIS plant: secondary side pressure.

3.2 Detailed Comparisons of the Transient Evolution Predicted by the Different Codes

The blowdown phase is characterized by a steep decrease of the RCS pressure (Figure 5) due to the loss of mass at the break and at the ADS.

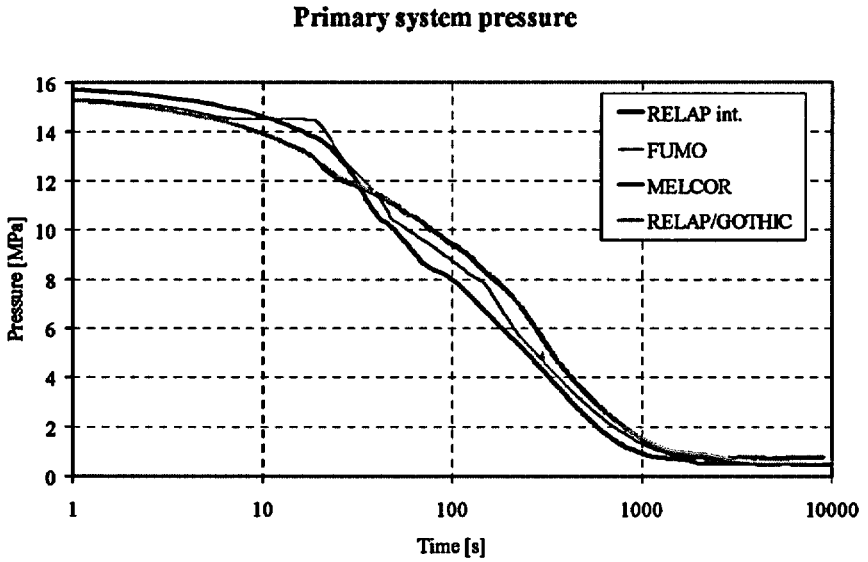

Figure 5. SBLOCA on the IRIS plant: primary system pressure.

The rate at which pressure decreases in the RCS is initially rapid, slows when the coolant in the RCS reaches saturation conditions, and then increases again when the mixture level drops below the break position. The imbalance between the heat transferred from the primary fluid through the SGs and the heat removed through the EHRS determines the depressurization of the secondary system (Figure 6).

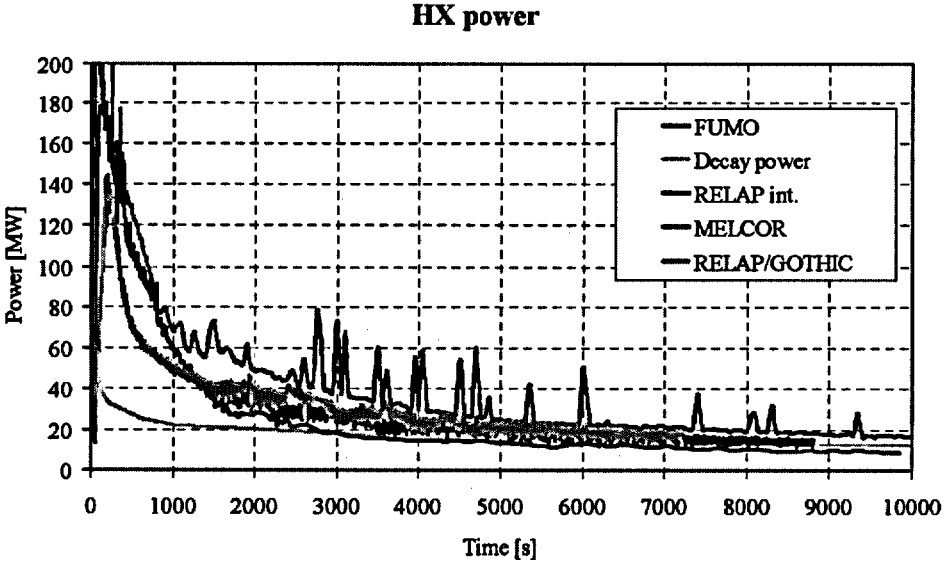

Figure 6. SBLOCA on the IRIS plant: Core Power (Decay Heat) and Removed Power at the SGs for different codes.

The comparison shows a qualitative agreement among the four calculations, but some significant differences exist. RELAP and RELAP/GOTHIC results are quite naturally in very good agreement given the fact that the RCS and EHRS models have an identical input and are analyzed by RELAP in both cases. Differences in the second part of the transient can be explained on the basis of the different transient evolution. FUMO and MELCOR both show significant differences from the RELAP calculations, and this is probably due to the fact that the model of the EHRS used in FUMO and MELCOR is simpler and the calculation is sensitive since it involves several different flow regime conditions and heat transfer modes. Qualification of the EHRS model and performance will be a focus of the IRIS testing program, and will lead to the development of an appropriate heat transfer package for the helical coil steam generators and for the EHRS heat exchangers. It can also be noted that both FUMO and MELCOR predict a slightly faster RCS depressurization, due to a more effective heat removal through SGs predicted (Figure 6). Once the pressure in the RCS and in the CS are equalized (end of the blowdown phase), the RCS continues to depressurize, albeit at a lower rate, due to the EHRS condensation of the steam produced in the core on the steam generator tube surface until a steady condition is reached where the EHRS power balances the core decay power. The CS pressure behavior presents the most significant differences among the four considered codes (Figure 7). These differences depend on a large number of variables, such as the mass and energy discharged through the break (Figures 8 and 9), the interaction between the water discharged through the break and the CS atmosphere and the thermal dynamics of the CS pools.

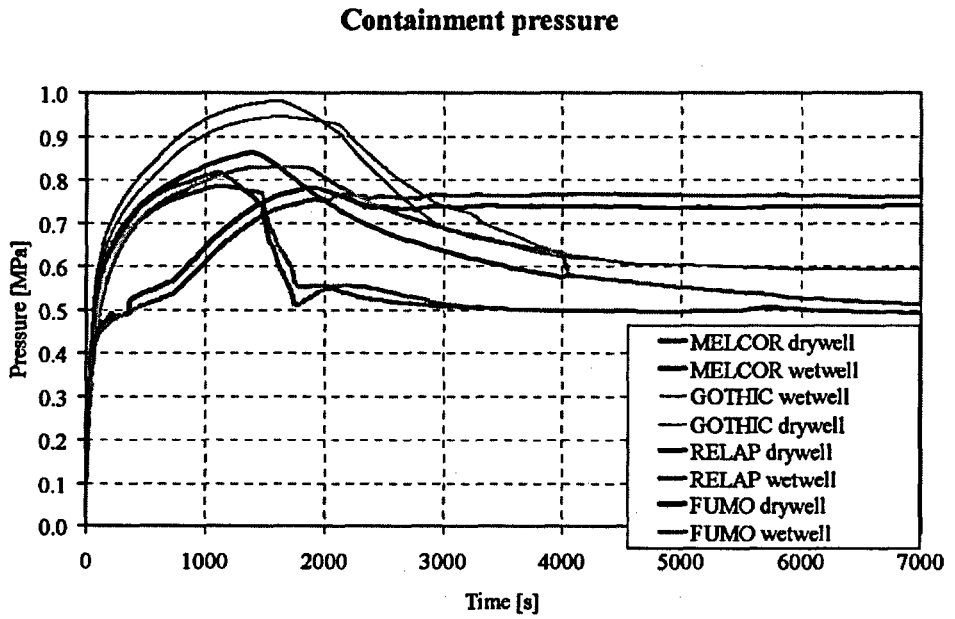

Figure 7. SBLOCA on the IRIS plant: containment atmosphere pressure.

The total break flow rate is provided in Figures 8 and 9. The default RELAP critical flow model (Henry-Fauske) with a discharge coefficient of 1.0 was employed to calculate critical flow and the break was simulated as an orifice with a length/diameter (L/D) ratio equal to zero. The FUMO critical flow model allows to take into account the effect of the L/D ratio and the calculation reported here has been performed with L/D = 0, in order to minimize differences with RELAP. When using a L/D ratio equal to 3 (consistent with a vessel thickness of about 30 cm) the FUMO break flow rate is predicted to be much more similar to MELCOR one. On the other side, the critical flow in MELCOR is evaluated by a polynomial interpolation and does not depend on the junction length (predicted data seem to be consistent with a L/D ratio larger than the asymptotic value of 3). Besides, RELAP and MELCOR, which have a more refined simulation of mixture level than FUMO, predict a longer time before the break uncovers. The lower break mass flow rate also explains the delay in MELCOR and FUMO sequence of events on the actuation of the reactor trip signal.

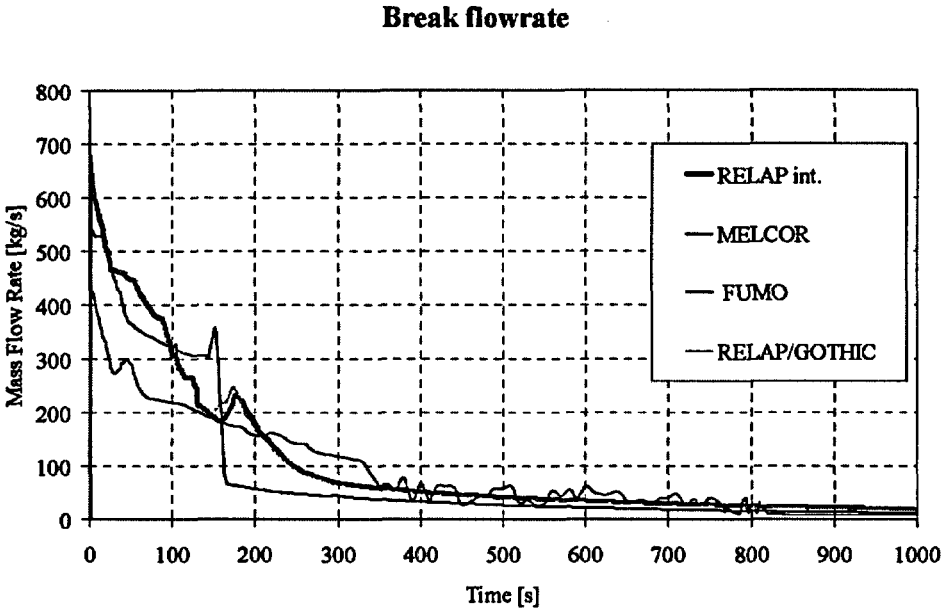

Figure 8. SBLOCA on the IRIS plant: break mass flowrate.

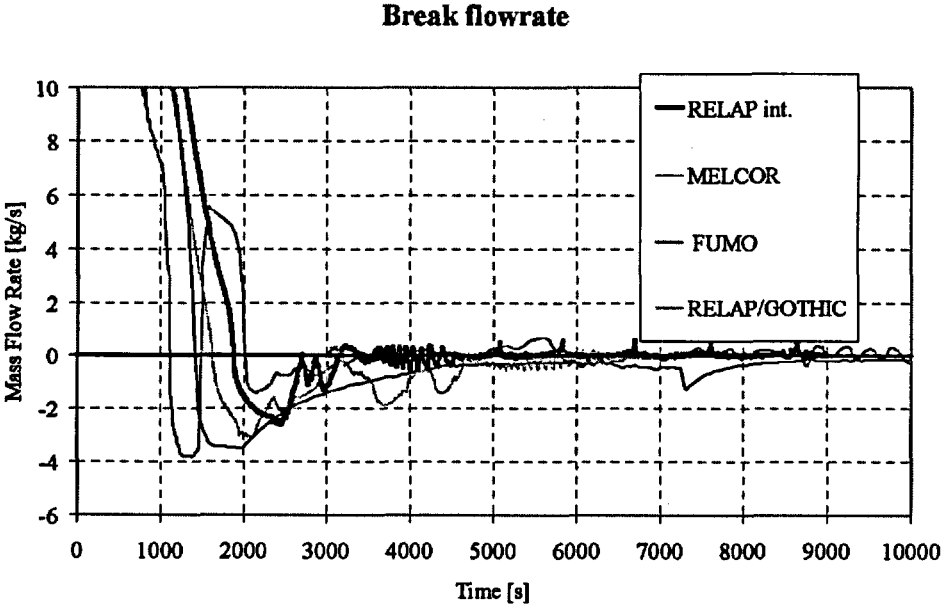

Figure 9. SBLOCA on the IRIS plant: break mass flow rate (detail)

Following the pressure equalization between the RCS and CS, the break flow rate reverses, due to the unique IRIS approach of providing heat removal inside the reactor vessel and thus RCS depressurization by the EHRS. All the calculations show this flow reversing through the break, even though the related mass flow rate is quite low and progressively decreases to almost zero.

The CS pressure transients requires some additional clarification, and the RELAP integral calculation shows the most interesting differences in that a slower CS pressurization and a lower pressure peak are obtained even though the mass and energy discharged through the break are almost identical to those predicted by the RELAP/GOTHIC calculation,. To explain this inconsistent behavior and also to allow a better comparison of the various containment models, an additional analysis was performed. In this analysis the CS models for each code (RELAP/GOTHIC, RELAP, MELCOR, FUMO) were uncoupled from the RCS models, and for each case the break and ADS mass and energy terms calculated by RELAP/GOTHIC were employed as a boundary condition. This study is discussed in detail in Section 3.3.

As shown in Figure 10, the liquid mass in the RCS stabilizes at the end of the blow-down phase and then increases during the depressurization phase. RELAP and RELAP/GOTHIC calculations predict similar trends for this variable that appears to be only weakly affected by CS behavior for this specific case; and the differences with respect to MELCOR and FUMO calculations can still be associated to the break mass flow rate as discussed above. For MELCOR, the lower break flow rate essentially corresponds to a smaller break and thus explains why the loss of RCS inventory is smaller than for RELAP and RELAP/GOTHIC. For FUMO, the reason for the reduced loss of mass is different and is due, as discussed above, to the fact that a lower mixture level is predicted (essentially the mixture level is coincident with the collapsed liquid level). Thus, the break uncovers more rapidly, and the loss of mass is significantly reduced when only steam is released at the break. The differences between the two codes are evident in the RCS mass transient, that shows how in MELCOR the loss of mass is under predicted from the beginning of the transient, while FUMO follows RELAP and RELAP/GOTHIC with good agreement at the beginning of the transient, until, at about 150 seconds, the break uncovers and only steam is discharged at the break.

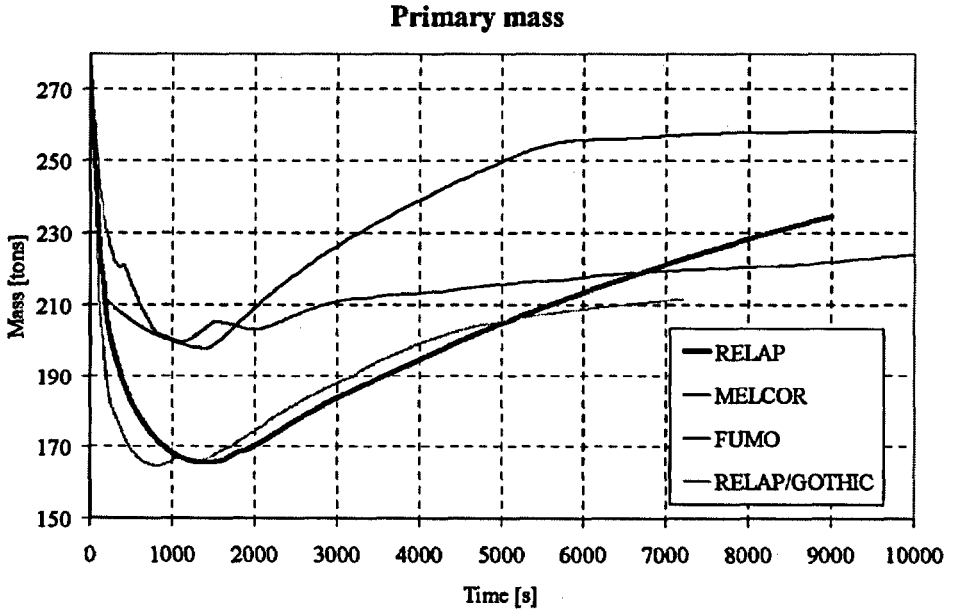

Figure 10. SBLOCA on the IRIS plant: primary coolant inventory.

The recovery of the liquid inventory is predicted by all the codes and is due to water injection from the suppression pools to the RCS through the makeup lines from the bottom of the suppression pool volume. While all codes qualitatively predict this phenomenon, some significant quantitative differences are evident in the inventory transient following the end of the blowdown phase. An explanation of these differences requires a review of the IRIS design and of the LGMS (long term core makeup system) and PSS (pressure suppression system) in particular. The suppression pool water level is in fact located at a

higher elevation than the core and this, combined with the slightly higher pressure in the suppression pool atmosphere during the initial part of the depressurization phase, results in injection flow to the RCS following the actuation of the valves on the makeup line that connects the suppression pool to the direct vessel injection line (Figure 11).

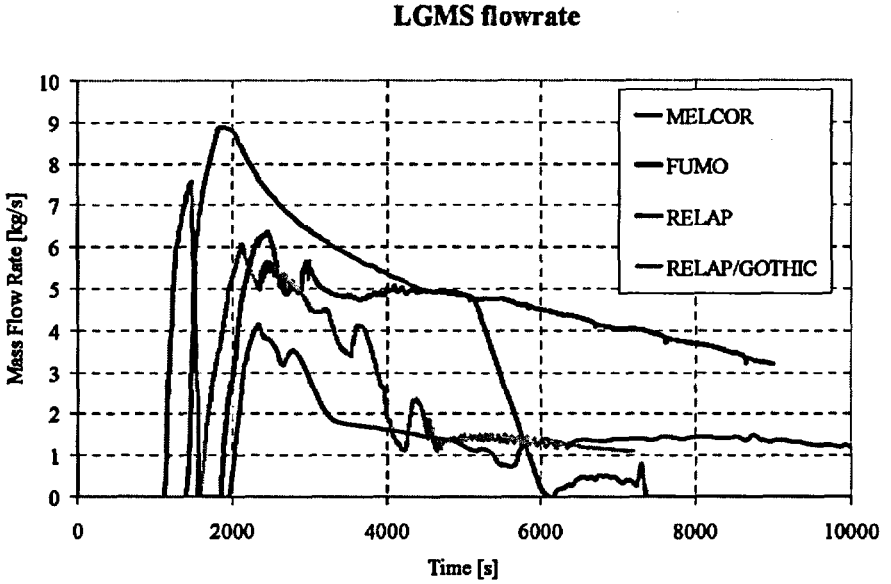

Figure 11. SBLOCA on the IRIS plant: LGMS mass flow rate (single line).

To explain the differences between the codes, it must be noted that once the RCS and CS pressures are equalized, the two systems are coupled and the CS pressure follows the same pressure transient as the RCS pressure decreases. The suppression pool atmosphere is not coupled to the rest of the CS due to the absence of vacuum-breakers as in conventional BWRs. As the CS pressure decreases and the suppression pool pressure remains essentially constant, the suppression pool water is discharged to the cavity through the vents. The flow of water from the suppression pool affects the depressurization rate of the CS in different ways for the different codes. The reason is that the depressurization rate is not only impacted by the amount of water discharged through the vents, but also by how the codes treat this injection. For example in MELCOR the interaction between the relatively cold water being discharged to the drywell and the CS atmosphere causes a very fast containment depressurization. This increases the water flow from the PSS to the CS and naturally limits the amount of water that is available from the pool for injection to the RCS, and this explains the slower mass recovery in the RCS predicted by MELCOR. Similar considerations could be made for the other analyses, and in particular the RELAP standalone results are of particular interest: given the different CS pressure prediction discussed above, little water is discharged from the suppression pool vents to the CS and more water is thus available for injection to the RCS.

The analysis of these results confirms the uncertainty already discussed in Reference [5] regarding the depressurization phase. The uncertainty related to the phenomena of the reflooding of the vents and the effect of the water discharge from the vents to the containment on the CS makes it very difficult at this stage of the investigation to suggest an adequate analytical approach. For conservative analyses, the following approach could be adopted. When the RCS is analyzed, the assumption should be that all the suppression pool water is delivered from the suppression pool to the CS, with the gravity makeup lines not credited. When the CS is analyzed, the opposite approach should be used, assuming that all the water is delivered to the RCS.

3.3 Containment Response Sensitivity Study

As previously discussed, significant differences were observed in the CS response, which can only be partially explained on the basis of the considerations provided in the previous sections. To allow a better comparison of the various CS models, an additional set of analyses was performed until 2000 seconds, about the time of the CS pressure peak, i.e. the initial phase where RCS and CS are uncoupled. In this analysis the CS models for each code (RELAP/GOTHIC, RELAP, MELCOR, FUMO) were uncoupled from the RCS models, and for each case the break and ADS mass and energy terms calculated by RELAP/GOTHIC were used as a boundary condition. To allow for a clearer comparison, results are presented here divided in two subsections: first the MELCOR and FUMO results are compared with GOTHIC, and then the RELAP results are separately discussed.

3.3.1 FUMO, MELCOR and GOTHIC Containment Response

The pressurization of the drywell and air tanks is presented in Figure 12. The agreement between the results confirms that the differences among integral calculations are basically due to different break discharge models, as expected. However, some difference between the codes still exists, and in particular a higher containment pressure is calculated by MELCOR.

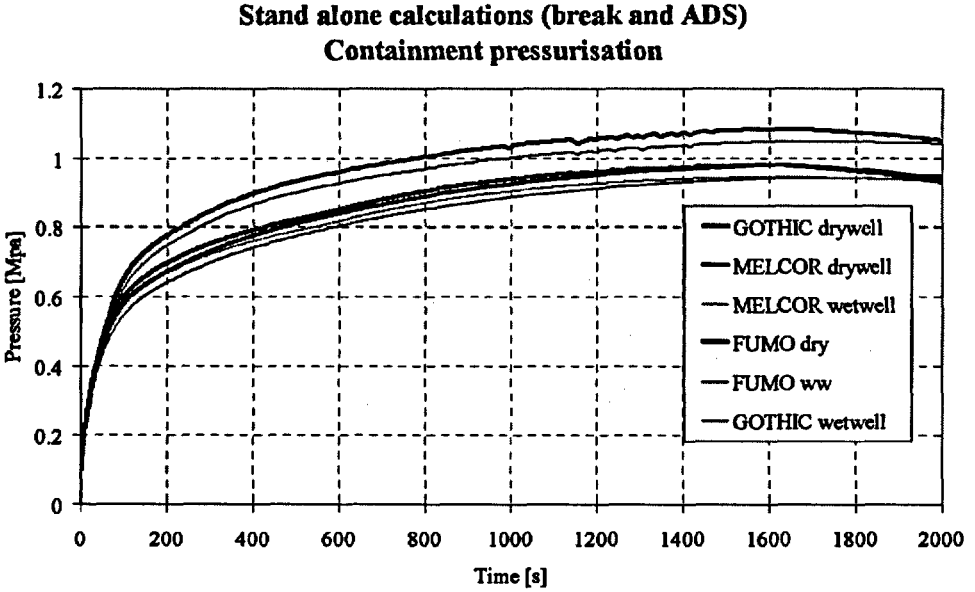

Figure 12. SBLOCA on the IRIS plant (stand alone): containment pressurization.

Review of the nitrogen and steam masses in the drywell and cavity showed that the larger pressurization of the CS predicted by MELCOR is due to a more efficient sweeping of nitrogen from the drywell to the air tanks during the initial blowdown phase. The amount of non-condensable gas as a function of time that is forced to flow through the vents during the early part of a LOCA play an important role in the IRIS containment pressurization, also affecting the medium and long-term system behavior.

This nitrogen relocation depends on several different variables, ranging from the blowdown characteristics (mass/energy and break location in the CS), the drywell spatial configuration (level of compartimentation, existence of dead-end zones, geometry of heat structures) to the number and location of the vents connecting the drywell to the suppression pools. Taking into account all these variables would require a computational three-dimensional analysis, but lumped parameter codes are able to show the importance of these effects on the performance of the suppression pool system. For example, the position of the vents with respect to the break has been investigated by performing two more sets of stand-alone calculations, in

which only the break contribution to the CS pressurization was considered in order to reduce to a minimum the possible differences in boundary conditions for the three containment codes. Figure 13 shows the pressurization of the IRIS CS, as predicted by the three codes when the vents are connected to the half drywell control volume farther from the break, while Figure 14 reports the results obtained with the vents located closer to the break. With the second configuration all the codes show a reduction in the CS pressurization rate and in the final peak, due to the different composition of mixture flowing through the vents, more rich in steam during the early part of the transient than in the former configuration. The nitrogen tends to be pushed on the other side of the drywell and towards the containment cavity and the amount of noncondensable gas forced thought he vents is smaller.

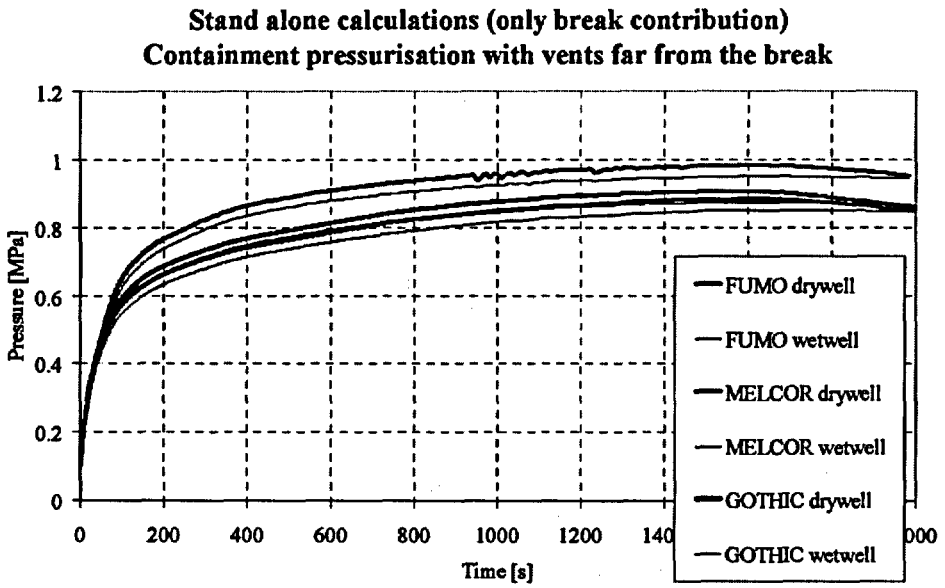

Figure 13. SBLOCA on the IRIS plant (stand alone): containment pressurization with vents far from the break.

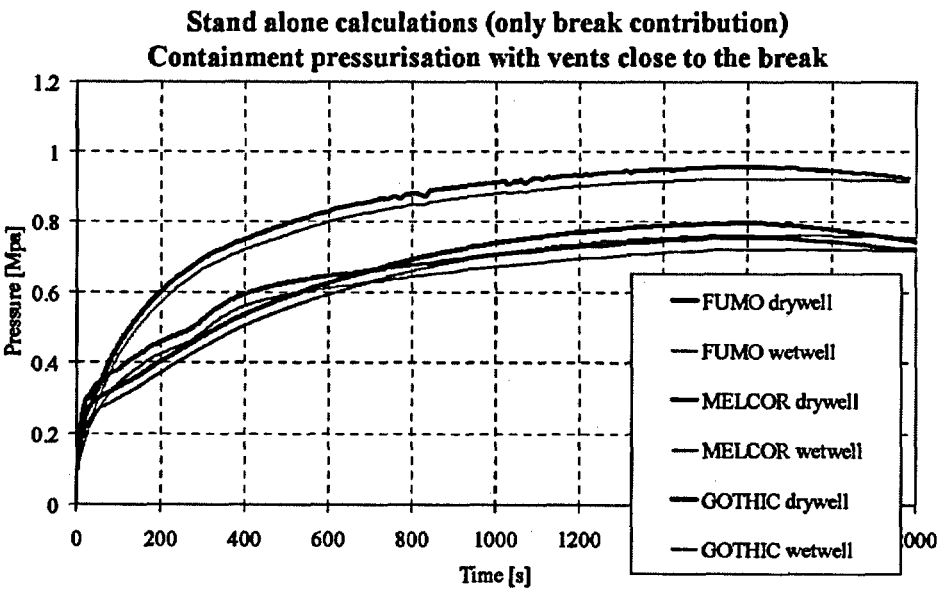

Figure 14. SBLOCA on the IRIS plant (stand alone): containment pressurization with vent close to the break.

The non-condensable gas dynamics inside the IRIS CS during a LOCA accident must be recognized to be a key issue for a realistic prediction of system pressurization, and is a known analytical problem in the analysis of pressure suppression CSs. A conservative approach for the evaluation of the containment peak pressure would assume that the non-condensable gas is completely relocated through the vents before steam can be discharged and thus condensed. However, it must be stressed that while such an approach is conservative for the calculation of the containment peak pressure, in the case of IRIS it would actually lead to a non-conservative analysis of the RCS inventory. Given the complexity of the phenomena involved, it is difficult to specify a best-estimate model for these analyses, and thus bounding calculations could be used for the purpose of determining the IRIS CS pressure transients, with two sets of analyses being performed, one to calculate conservatively the CS peak pressure, and the second to calculate conservatively the RCS inventory.

3.3.2 RELAP Containment Response

The importance of the compositions of the flow discharged through the vents to the suppression pool is further confirmed by RELAP analyses. As discussed in Section 3.2, the RELAP results are of difficult interpretation for the CS pressure: while the same break mass and energy terms of the RELAP/GOTHIC code is calculated for the break, the peak pressure of the RELAP containment model is significantly lower.

This difference can be explained in terms of the model used for the RELAP CS. As shown in Figure 1, the RELAP CS model was developed using 1-D components rather than a lumped parameter approach. The sliced modeling of the CS with RELAP, along with the particular configuration of vents leads in fact to the prediction of a "pocketing" of the noncondensable gas in the cavity and lower parts of the CS. This is similar to the behavior discussed above, but the use of a sliced nodalization approach enhances the localization of nitrogen in the lower drywell volumes, thus leading to a more effective suppression pool performance as an atmosphere richer in steam is delivered to the pool.

To confirm this explanation, a modified containment model was developed. In particular, a delayed opening of some of the junctions connecting the two drywell subvolumes allowed a more efficient sweeping of nitrogen through the vents towards the suppression pools and air tanks. In Figure 15 the CS pressurization obtained with this configuration is compared with the one obtained in the base case. Due to the larger amount of nitrogen mass pushed inside the air tanks (a similar amount with respect to the other calculations), the CS pressurization rate and the pressure peak at the end of the blowdown phase result are larger than the base case. Moreover, the depressurization phase results are also more similar to those predicted by the other containment codes.

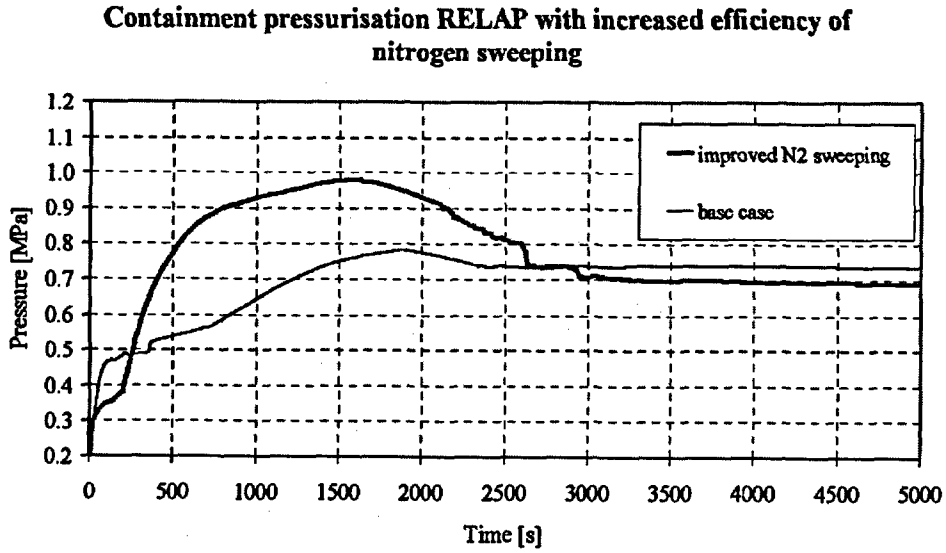

Figure 15. SBLOCA on the IRIS plant (RELAP sensitivity case): containment pressurization.

Another key difference between RELAP and the other codes that has not been extensively explored in this paper is the suppression pool thermal dynamics. While in MELCOR, FUMO and GOTHIC the water of the pool is perfectly mixed (as consistent with physical expectations during the blowdown phase), the sliced approach of a RELAP nodalization can lead to an unphysical thermal stratification, even in the presence of strong mixing forces. These effects have been highlighted in the simulation of the PANDA experiments [12], an important validation test about the modeling approach with RELAP for the containment suppression pools. As for the present IRIS simulations, to avoid an unphysical pool stratification, both the cavity and the suppression pool were subdivided into two parallel sub-volumes, allowing a homogenization of liquid and gas phases through natural circulation flows.

4 CONCLUSIONS

The behavior of the IRIS plant under a SBLOCA transient has been analyzed with three different codes and the RELAP/GOTHIC results presented in Reference [4] ad [5] were compared with results obtained using the MELCOR, RELAP and FUMO codes. The modeling of the IRIS RCS, safety systems and CS with these three codes has also been presented in this paper.

All the calculations confirmed the capability of IRIS plant, based on the "Safety-by-design" approach, to mitigate the consequences of a loss of coolant accident. The results demonstrate the possibility of maintaining the RCS inventory without the necessity of high and low pressure injection systems. The comparison between the four codes (RELAP/GOTHIC, RELAP, MELCOR and FUMO) predictions showed a good qualitative agreement, confirming the expected behavior of the plant during a SBLOCA event.

However, significant quantitative differences between the codes predictions were identified. The results were reviewed and used to identify important physical phenomena during the blowdown and depressurization phase of a IRIS small break LOCA event. In particular, most of the differences between the codes can be attributed to modeling of the following parameters:

1. Critical flow model at the break;
2. Composition of the flow mixture at the vents during the early blowdown phase (i.e. how fast the non-condensables are relocated from the drywell to the wetwell);
3. EHRS performance (heat transfer modeling);
4. Behavior and simulation of the interactions of the CS atmosphere and the liquid flow from the suppression pool to the containment during the depressurization phase.

For some of these effects a conservative, bounding approach could be used, relying on the intrinsic robustness of the IRIS design to obtain the required performance (i.e. the core remains covered throughout any postulated design basis LOCA). A conservative approach would naturally eliminate some of main concerns discussed in this paper regarding the analytical modeling and simplify the testing program required to validate the conservativeness of the proposed evaluation model. On the other hand, this approach may lead to increased performance requirements for the safety systems (EHRS, ADS) and thus to an increase in the cost of the plant. Both these aspects will have be considered and carefully balanced in the definition of a testing program and an evaluation models for IRIS SBLOCA.

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August 2004

IRIS Small Break LOCA Phenomena Identification and Ranking Table (PIRT)

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WCAP-16318-NP

IRIS SMALL BREAK LOCA PHENOMENA IDENTIFICATION AND RANKING TABLE (PIRT)

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Accordingly, the IRIS SBLOCA PIRT Panel members gratefully acknowledge the support and contributions given by these individuals to the panel's deliberations.

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LIST OF ACRONYMS

Acronym	Definition
ADS	Automatic Depressurization System
AOV	Air Operated Valve
BA	Boric Acid
CHF	Critical Heat Flux
CRDM	Control Rod Drive Mechanisms
CSAU	Code Scaling, Applicability and Uncertainty
CV	Containment Vessel
CVCS	Chemical and Volume Control System
DB	Design Basis
DVI	Direct Vessel Injection
EHRS	Emergency Heat Removal System
EBS	Emergency Boration System
EBT	Emergency Boration Tank
FOM	Figure of Merit
GT	Guide Tubes
HX	Heat Exchanger
IRIS	International Reactor Innovative and Secure
LBLOCA	Large Break LOCA
LGMS	Long Term Gravity Makeup System
LOCA	Loss of Coolant Accident
MOV	Motor Operated Valve
NRC	Nuclear Regulatory Commission
PIRT	Phenomena Identification and Ranking Table
PCCS	Passive Containment Cooling System
PSS	Pressure Suppression System
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Vessel
RVC	Reactor Vessel Cavity
RWST	Refueling Water Storage Tank
SBLOCA	Small Break LOCA
SG	Steam Generator
SoK	State of Knowledge
SS	Stainless Steel

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EXECUTIVE SUMMARY

The International Reactor Innovative and Secure (IRIS) is a modular pressurized water reactor with an integral configuration (all primary system components—reactor core, internals, pumps, steam generators, pressurizer, and control rod drive mechanisms—are inside the reactor vessel). The IRIS plant conceptual design was completed in 2001 and the preliminary design is currently underway. The pre-application licensing process with the United States Nuclear Regulatory Commission (US NRC) started in October 2002.

The first line of defense in IRIS is to eliminate event initiators that could potentially lead to core damage. If it is not possible to eliminate certain accidents altogether, then the design inherently reduces their consequences and/or decreases their probability of occurring. One of the most obvious advantages of the IRIS safety-by-design approach is the elimination of large break loss of coolant accidents (LBLOCAs), since no large primary penetrations of the reactor vessel or large loop piping exist. Handling of small-break loss of coolant accidents (SBLOCAs) is equally important, where the IRIS approach is to limit and eventually stop the loss of coolant from the vessel rather than to rely on active or passive systems to inject water into the reactor vessel (RV).

While the IRIS safety-by-design approach is a logical step in the effort to produce advanced reactors, the desired advances in safety must still be demonstrated in the licensing arena. With the elimination of LBLOCA, an important next consideration is to show the IRIS design fulfills the promise of increased safety for the SBLOCA. Accordingly, the IRIS Consortium, through Westinghouse, established the SBLOCA Phenomena Identification and Ranking Table (PIRT) Project that is documented herein. The IRIS program issue being addressed is the planning for continued development of the experimental data and analytical tools needed for safety analysis, particularly in the licensing arena, in a sufficient and cost effective manner. Thus the primary objective of the IRIS SBLOCA PIRT project was to identify the relative importance of phenomena in the IRIS response to SBLOCAs. This relative importance, coupled with the current relative state of knowledge for the phenomena, provides a framework for the planning of the continued experimental and analytical efforts.

To satisfy the SBLOCA PIRT Project objectives, Westinghouse organized an expert panel whose members were carefully selected to insure the PIRT results reflect internationally recognized experience in reactor safety analysis, and were not biased by program preconceptions internal to the IRIS Program. The panel employed the well proven PIRT Process to determine the range/conditions of SBLOCA scenarios that needed to be analyzed, and then applied that methodology. The panel concluded that analysis of a low elevation break in one of the Direct Vessel Injection lines (a 2",schedule 160 pipe), and a high elevation break in one of the 4" Chemical and Volume Control System lines (a 4", schedule 160 pipe) would satisfy the SBLOCA PIRT objectives. The selection process for the definition of the LOCAs scenario considered by the PIRT panel is summarized in Figure 1. These two scenarios were subsequently analyzed and the results are documented in this report. Main results of the phenomena ranking and identification process are summarized in Table 2, where phenomena are ranked (as High, Medium, Low and Insignificant phenomena) for each of the identified scenario phases (I: blowdown phase, II: depressurization phase, III: long term cooling phase) according to the figures of merit defined in Table 1.

The SBLOCA PIRT Panel concluded that continued experimental data and analytical tool development in the following areas, in decreasing level of significance, are perceived as important with respect to satisfying the safety analysis and licensing objectives of the IRIS Program:

1. Steam generator;
2. Pressure Suppression System, Containment Dry Well and their interactions;
3. Emergency Heat Removal System (this item and #2 immediately above are considered to have the same level of significance);

4. Core, Long Term Gravity Makeup System, Automatic Depressurization System, and Pressurizer;
5. Direct Vessel Injection System and Reactor Vessel Cavity.

FIGURE 1 - SCENARIO SELECTION PROCESS

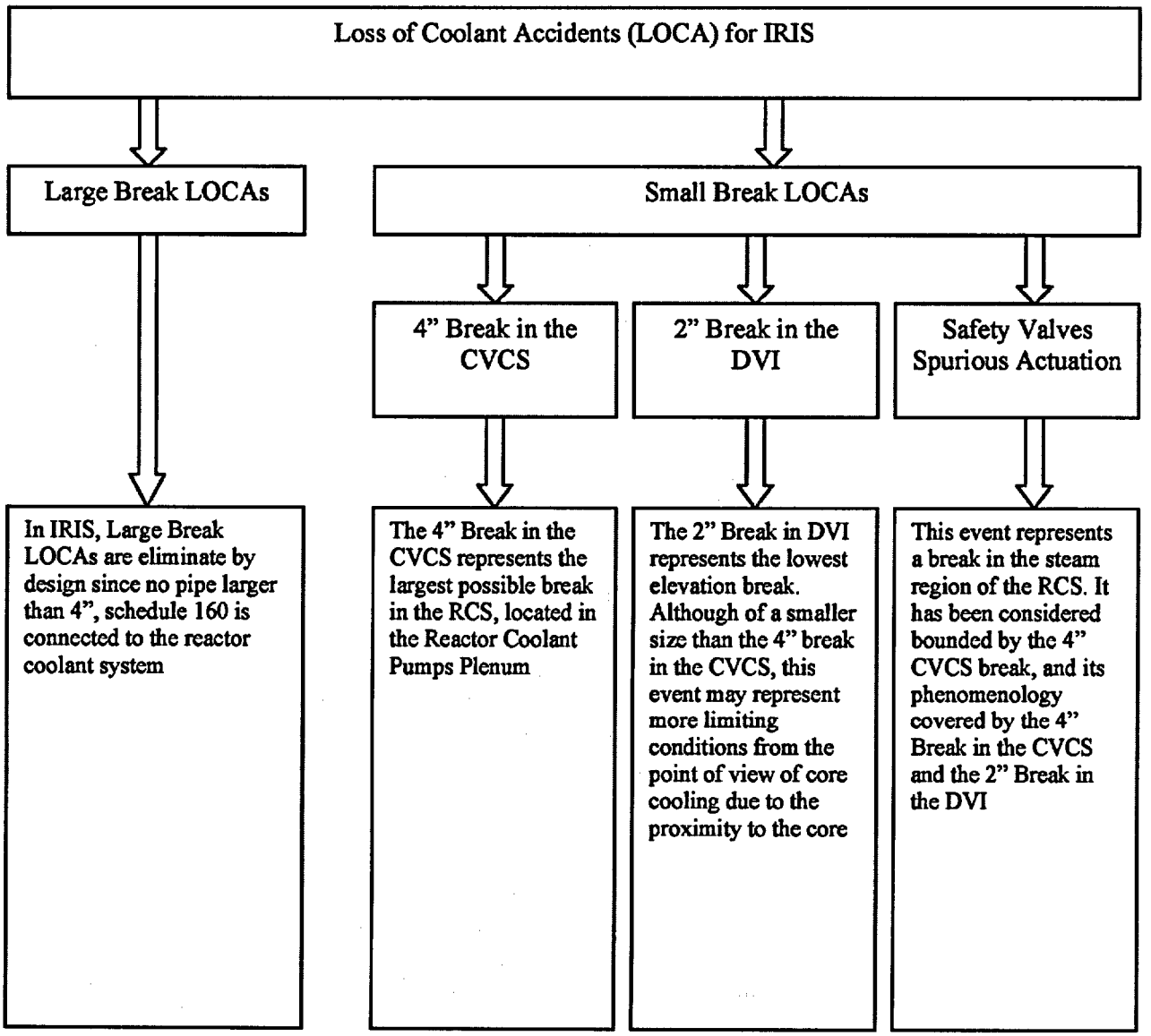

TABLE 1 - IRIS SBLOCA FIGURES OF MERIT

Figure of Merit	Rationale
RV coolant inventory	Maintain sufficient vessel coolant inventory to remove the initial stored energy and subsequent decay heat without significant fuel clad temperature excursions (Core Cooling)
Containment pressure and successful heat removal to the environment	Prevent an initial over-pressurization of the containment followed by containment depressurization.

TABLE 2 - IRIS SBLOCA PIRT RESULTS SIGNIFICANT TO CONTINUED EXPERIMENTAL DATA AND ANALYTICAL TOOL DEVELOPMENT (Entries with no parenthetical insertion in the phenomenon description denote the entry is equally applicable to the DVI and CVCS breaks. Entries with such insertions are applicable to the CVCS break only. SoK indicates the Panel state of knowledge assessment)

System	Component	Phenomenon	Phenomena Rank by Phase			SoK
			I	II	III	
Highly important phenomena with low state of knowledge						

(a, c)

Highly important phenomena with moderate state of knowledge

(a, c)

Highly important phenomena with high state of knowledge						

(a, c)

Moderately important phenomena with low state of knowledge						

(a, c)

							(a, c)
Moderately important phenomena with moderate state of knowledge							
							(a, c)

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1.0 INTRODUCTION

1.1 IRIS Approach to SBLOCA mitigation

IRIS is a pressurized water reactor that utilizes an integral reactor coolant system layout. The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components including pumps, steam generators, pressurizer and a neutron reflector. The IRIS integral vessel is larger than a traditional PWR pressure vessel, but the size of the IRIS containment is a fraction of the size of corresponding loop reactors, resulting in a significant reduction in the overall size of the reactor plant.

The plant conceptual design was completed in 2001 and the preliminary design is currently underway. The pre-application licensing process with NRC started in October 2002.

IRIS has been primarily focused on achieving a design with innovative safety characteristics. The first line of defense in IRIS is to eliminate event initiators that could lead to core damage. In IRIS, this concept is implemented through the "safety by design" approach, which can be simply described as "design the plant in such a way as to eliminate accidents, rather than coping with their consequences". If it is not possible to eliminate certain accidents altogether, then the design inherently reduces their consequences and/or decrease their probability of occurring (Refs 1.1, 1.2). The key difference in the IRIS "safety by design" approach from previous practice is that the integral reactor design is conducive to eliminating accidents, to a degree not possible in conventional loop-type reactors.

The elimination of the large break LOCAs is one of the most obvious advantages of the IRIS safety-by-design approach, since no large primary penetrations of the reactor vessel or large loop piping exist and a vessel break is not considered a credible event. An equally important example is given by the handling of small-break LOCAs, which historically have been most plaguing to PWRs. For small break LOCAs, IRIS features an innovative mitigation approach that is based on maintaining coolant inventory rather than providing high and low pressure injection systems to provide makeup coolant to the reactor to maintain core cooling. This is accomplished by taking advantage of the following three features of the design:

1. The large coolant inventory in the reactor vessel;

2. An emergency heat removal system (EHRS) employing the steam generators to remove heat directly from inside the RV, thus depressurizing the RV by condensing steam, rather than by discharging mass.

3. The compact, small-diameter, high design pressure containment, which during an SBLOCA accident becomes thermodynamically coupled with the vessel and assists in limiting the blowdown from the RV by rapidly equalizing the vessel and containment pressures.

An in depth review of the IRIS SBLOCA mitigation strategy is provided in Sections 3 and 4.

1.2 SBLOCA PIRT

While the IRIS safety-by-design approach is a logical step in the effort to produce advanced reactors, the desired advances in safety must still be demonstrated in the licensing arena. With the elimination of LBLOCA, an important next consideration is to show that the IRIS design fulfills the promise of increased safety for the SBLOCA. Accordingly, the IRIS Consortium, through Westinghouse, established the SBLOCA PIRT Project that is documented in this report. The objective of the project is three fold:

1. To provide expert technical evaluations of the relative importance of phenomena occurring during a SBLOCA,

2. To identify the "state of knowledge" relative to the important phenomena in the context of adequate, marginal and inadequate experimental databases and analytical tools with which to conduct safety analyses, and
3. Use the "state of knowledge" assessments to guide continued experimental database and analysis tool development.

To satisfy objectives 1 and 2, a PIRT panel was organized consisting of:

- Three members totally independent of W, and having extensive experience in reactor safety analysis,
- Three W members independent of the IRIS project, and having significant experience in other W plant designs (AP600, AP1000, etc.),
- One member from academia, with expertise associated with previous and ongoing tests in an AP600/AP1000 related experimental facility.

Additional information regarding the PIRT Panel charter and the qualifications of the panel members are provided in Appendix A.

Although not considered panel members, the project was supported by IRIS Experts. These individuals were W and IRIS consortium engineers responsible for various areas of the IRIS design. To insure no conflict of interest, the only role allowed to the IRIS Experts was to address any request for information from the PIRT Panel.

1.3 Report Structure

The PIRT methodology used for this SBLOCA application is described in Section 2. Section 2.1 focuses on the generalized PIRT process. Section 2.2 then expands the generalized process to those features common to the two SBLOCA scenarios addressed. Process features more specific to each scenario, particularly the scenario dependent PIRT application results, are provided in Sections 3 and 4. The significant conclusions drawn from the results are given in Section 5. Section 6 lists documents containing other information of general interest to this PIRT effort. More detailed information regarding how the project was conducted, and the results, is provided in Appendices A through E. In addition, Addendum 1 to this report documents SBLOCA sensitivity studies that helped the PIRT panel in its deliberations.

2.0 METHODOLOGY AND APPROACH

2.1 PIRT Methodology

The US NRC and its contractors developed the PIRT (Phenomena Identification and Ranking Tables) process in 1989 as part of the CSAU (Code Scaling, Applicability and Uncertainty) effort (Ref. 2.1). As originally conceived and applied to a PWR LBLOCA the process proved highly successful. However, the original process application was somewhat complex and labor intensive. As a first-of-a-kind product the process did not necessarily recognize future applications in which more limited objectives would be sufficient. In the intervening fourteen years the process has been applied and refined in more than fifteen projects, for example:

- Light water reactor accident scenarios:
  - Large break LOCA
  - Small break LOCA
  - Main steam-line break
- Steam generator tube rupture
- Debris transport in wet and dry containments
- Containment coatings
- High burnup fuel under accident conditions
- Burnup credit
- TRISO fuel (manufacturing, operation, and accident life-cycle phases)

As now evolved, the PIRT process provides for the identification and ranking of safety-significant phenomena and associated research needs through the sequential consideration of the elements shown in Fig. 2.1.

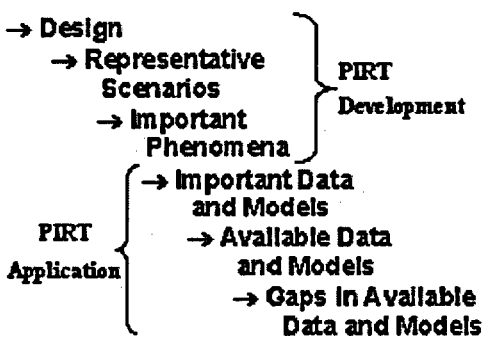

Figure 2.1. Elements considered in the PIRT process and application.

With respect to the elements illustrated in Figure 2.1, the PIRT process accommodates several key issues:

- Identification of ALL relevant phenomena,
- Recognition of phenomena RELATIVE importance including the associated rationales. That is, some "phenomena" ARE more important than others in complex and coupled physical systems. Conversely, some "phenomena" are NOT as important,
- Once important "phenomena" are identified, attention and resources can be focused more efficiently,
- Recognition of the RELATIVE state of knowledge for the phenomena, including the associated rationales,
- Must be well documented for the user.

The evolved nine step PIRT process is illustrated in Figure 2.2:

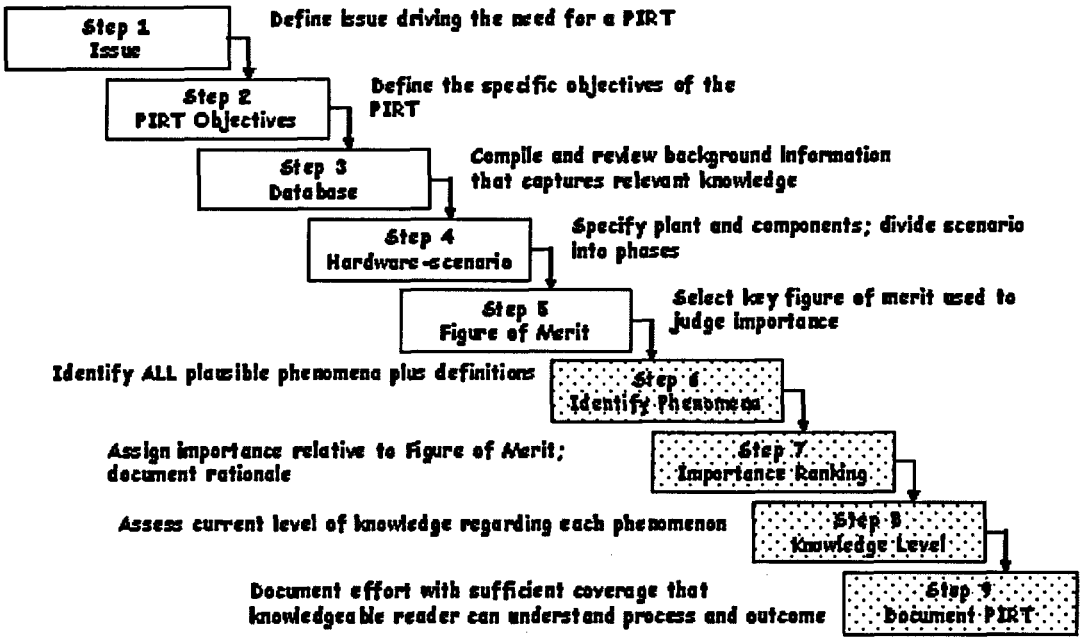

Figure 2.2 Nine-step PIRT process.

2.2 IRIS SBLOCA PIRT

The PIRT Panel followed the process illustrated in Figure 2.2. The general process features, common to the two SBLOCA scenarios addressed, are described in the remainder of this section.

2.2.1 Issue Definition

From the perspective of the IRIS Program, the primary issue is to insure that a sufficient experimental and analytical database exists to support the licensing process. And, this must be done in a cost effective manner. That is, the experimental and analytical database development should be planned to address design behavioral questions in a hierarchical sequence in which the plant responses that are postulated to be of the highest safety significance are explored first. In this context "safety significance" denotes the combination of how influential a behavior may be on a successful mitigation of an accident scenario and how well that behavior may be predicted via experimental data and/or analytical modeling. From this perspective it naturally follows that behaviors of "highest" safety significance are those that have the most influence on the plant response and are the least well understood and/or predicted with the current state of knowledge. The full range of decreasing safety significance determinations then progresses through the most influential behaviors that are moderately well predicted, moderately influential behaviors that are least well predicted, moderately influential behaviors that are moderately well predicted, etc. This is exactly the set of conditions for which the PIRT process was developed and has been applied as reported here.

2.2.2 SBLOCA PIRT Objectives

With the elimination of LBLOCA considerations via the safety-by-design approach, the next most logical set of considerations are those associated with SBLOCA. Based on the IRIS Program issue just discussed, the SBLOCA PIRT project had two primary and one subservient objectives:

1. To provide expert technical evaluations of the relative importance of phenomena occurring during a SBLOCA (primary objective),

2. To identify the "state of knowledge" relative to the phenomena in the context of adequate, marginal and inadequate experimental databases and analytical tools with which to conduct safety analyses (primary objective), and
3. To provide the framework whereby the phenomena relative importance and state of knowledge assessments can be used to guide continued experimental database and analysis tool development in a cost effective manner (secondary objective).

2.2.3 IRIS SBLOCA Data Base Review

The IRIS design development has continued for more than four years. Therefore, a large database exists related to the design, including several analytical studies to explore the potential design response to postulated accidents. Examples include References 2.2 – 2.8. Of particular interest is the recent sensitivity study included as Addendum 1 to this report. All of these documents, and more, were made available to the PIRT Panel members. It is important to stress that the information provided, and especially the SBLOCA analyses provided in Addendum 1, were reviewed critically by the panel, given the early stage of evaluation model development for the IRIS SBLOCA. That is, these sensitivity studies were only used by the panel to acquire a better familiarization with the design and the impact of the various safety systems on the SBLOCA evolutions.

2.2.4 System, Component and Scenario Specifications

The analysis and understanding of complex system behavior can be more successfully executed and achieved when the problem can be formulated into successively simpler sub-problems. This technique is recognized in the normal PIRT process in several ways, for example, a reactor design can be partitioned into systems and components within those systems. As noted in the previous section, a sufficiently mature design database existed to partition the plant into systems and components that provided a logical framework for the subsequent plausible phenomena identification (see Section 2.2.6). The systems and components are described in Appendix B.

For reasons already given, this PIRT effort was directed to SBLOCAs. In addition, only single failures were considered following accident initiation, to provide guidance to planning a cost effective experimental and analytical development program for design basis accidents. It was determined that a break in the lines between the pressurizer and safety relief valves, a break in the ADS piping, and an inadvertent safety relief valve opening are possible events that would lead to a loss of reactor coolant fluid. However, given the size of the valves and ADS lines (i.e. identical to the postulated break size in the CVCS) and the fact that these breaks would occur in the steam space of the integral RCS, these events were considered to be bounded by the postulated break of the 4" CVCS line. Accordingly, the panel agreed that its efforts should be focused on two SBLOCA scenarios: 1) A low elevation break in one of the 2" DVI (Direct Vessel Injection) lines, and 2) a high elevation break in one of the 4" CVCS (Chemical And Volume Control System) lines between the pressure vessel and an emergency boration tank. Of the two scenarios the panel considered the DVI break to be potentially more complex and more likely to encompass the full range of plausible phenomena. Therefore, it was decided to address the DVI break first and fully, with the expectation that the 4" CVCS break could be treated as an exception case, should it be determined this scenario contained significant behavioral differences from the DVI break.

Similar to the problem simplification by partitioning of the IRIS design into systems and components, the normal PIRT process partitions scenarios into logical time phases in which the phenomenological behaviors are reasonably consistent during a phase. This standard practice was used in this PIRT effort as is illustrated by the scenario descriptions provided in Table 2.1.

Table 2.1 - IRIS SBLOCA Scenario Description

Phase	Scenario Details
I - Blowdown Approx. duration: 4" CVCS pipe break: 1600s 2" DVI line break: 2100s	Following SBLOCA initiation, the mitigation sequence is initiated with the reactor trip and pump trip; the EBTs are actuated to provide boration; the EHRS is actuated to depressurize the primary system by condensing steam on the steam generators (depressurization without loss of mass); and finally, the ADS is actuated to assist the EHRS in depressurizing the RV. The containment pressure increase is limited by the Pressure Suppression System and the reduced break flow due to the EHRS heat removal from the RV. At the end of the blowdown phase the RV and CV pressure become equal (Pressure Equalization) with a CV pressure peak less than 8 barg. The break flow stops and the gravity makeup of borated water from the suppression pool becomes available.
II - RV/CV Depressurization Approx. duration = up to 12 hr	The coupled RV/CV system is depressurized by the EHRS (steam condensation inside the RV exceeds decay heat boiloff). In this phase the break flow reverses since heat is removed not from the containment, but directly from inside the vessel, and this increases the liquid level in the vessel. As steam from the containment is condensed inside the pressure vessel (RV and CV pressure reduced to less than 2 barg within 12 hours), and as the containment pressure is reduced a portion of suppression pool water is pushed out through the vents and assists in flooding the vessel cavity.
III - Long Term Cooling Duration from 12 hrs onward	The depressurization phase is followed by long term cooling where the RV and CV pressure is slowly reduced as the core decay heat decreases. During this phase of the accident recovery, gravity makeup of borated water from both suppression pool and RV cavity is available as required. Since decay heat is directly removed from within the vessel and the vessel and containment are thermodynamically coupled, the long term break flow does not correspond to the core decay heat, but it is in fact limited to only that induced by the containment heat loss.

2.2.5 Figures of Merit

Figures of Merit are those criteria against which the relative importance of each "phenomenon" is judged. Successful figures of merit have distinct characteristics, and in particular they are (1) directly related to the issue(s) being addressed by a PIRT, (2) directly related to the phenomena being assessed for relative importance, (3) easily comprehended, (4) explicit, and (5) measurable. In this context, the design goals of the IRIS design provide the basis for selection of suitable Figures of Merit. The design goals are to:

- Maintain sufficient vessel coolant inventory to remove the core initial stored energy and subsequent decay heat without significant fuel clad temperature excursions, and
- Prevent an initial over-pressurization of the containment followed by containment depressurization such that the desired vessel inventory is maintained.

Based on this design requirements, the Figures of Merit appropriate to the SBLOCA PIRT were defined as the RV coolant inventory as associated with successful removal of the initial stored energy and core decay heat, and the containment pressure and successful heat removal to the environment. These two FOMs are consistent throughout all three phases of the scenarios.

2.2.6 Phenomena Identification

In the PIRT process phenomena are broadly defined to include any physical behavior and/or process that may be displayed by a reactor design in its response to any postulated accident scenario. Further, "plausible" phenomena are those physical behaviors and/or processes that may have some influence in reactor plant's response. It is important to identify the plausible phenomena before a PIRT panel considers what importance (ranking) a phenomenon may have in influencing the plant response. That is, the panel first considers all possible physical behaviors and/or processes that may occur before evaluating whether each phenomenon has no, insignificant, or real consequences in the plant response. Twenty years of PIRT experience has shown that this is the most successful approach to insure no significantly influential phenomenon is overlooked.

Because of the similarity of the two SBLOCA scenarios considered in this effort, the PIRT Panel was successful in developing one set of plausible phenomena. Those results are provided in Appendix C. As might be expected, the plausible phenomena are cast in terms of the system and component partitioning, and scenario time phases already described. Descriptions of the phenomena are also provided in Appendix C.

2.2.7 Ranking of Phenomena Relative Importance

The ranking of phenomena relative importance is based on the safety criteria of interest (Figures of Merit) as described in Section 2.2.5. In addition, one of the first tasks of a PIRT Panel in this step is to agree upon an explicitly defined ranking scale. A number of such scales have been used over the course of the PIRT Process initial development and subsequent improvement. This considerable experience has demonstrated that the ranking scale described in Table 2.2 provides the best combination of sufficiency and efficiency. Sufficiency denotes that the scale contains enough detail (values) to capture the relative influence of a phenomenon. Efficiency denotes that the scale is not so large in values that the ranking becomes overly complex. Based on its proven success, the IRIS SBLOCA PIRT Panel adopted the scale described in Table 2.2.

Table 2.2 - IRIS SBLOCA Phenomena Ranking Scale

Rank	Definition	Application Outcomes
High (H)	Phenomena has controlling impact on Figure of Merit	Experimental simulation and analytical modeling with a high degree of accuracy is critical
Medium (M)	Phenomena has moderate impact on Figure of Merit	Experimental simulation and/or analytical modeling with a moderate degree of accuracy is required
Low (L)	Phenomena has low impact on Figure of Merit	Modeling must be present to preserve functional dependencies; however, almost any model will be adequate
Insignificant (I)	Phenomena has no, or insignificant impact on Figure of Merit	Modeling must be present only if functional dependencies are required
Functional dependence denotes that a phenomenon in itself may not be significantly important; however, its modeling must be present to adequately capture its effect on another phenomenon that may be of higher importance		

An operational practice that has proven its worth in the application part of a PIRT effort is for the panel to rank the relative importance of systems and components before ranking the phenomena. The same Figures of Merit are used as those applied to the phenomena ranking. Noting that a component cannot have a higher rank than the system in which it is located, and a phenomenon cannot have a higher rank than the component in which it occurs, the panel can more quickly determine the order in which

phenomena should be ranked. That is, the ranking sequence should progress through: phenomena located in highly ranked systems and components first, followed by phenomena located in moderately ranked components second, followed by phenomena located in low importance components third. It may be noted that phenomena occurring in a low importance component can only be of low importance, and therefore these are automatically ranked low once the component is ranked. Phenomena located in components ranked as insignificant can be eliminated from further consideration. The phenomena associated with low and insignificant ranked components are usually a reasonably large number of phenomena in the normally complex reactor designs being addressed by the PIRT process. Accordingly, ranking of systems and components first produces a significantly more efficient PIRT application. This practice was adopted by the IRIS SBLOCA PIRT Panel and, as expected, it again proved its worth. It may be noted that while a specific result is achieved in the system and component ranking, these results are eventually subsumed in the phenomena ranking tables. Therefore, it is not necessary to report the system and component ranking results separately from the phenomena ranking results. In the case of this PIRT effort, the phenomena ranking results are reported in Section 3.0 (2" DVI break scenario) and Section 4.0 (4" CVCS break).

2.2.8 Determination of the Phenomena Current State of Knowledge

The combination of a phenomenon's relative importance and its current state of knowledge provides the basis for determining how further efforts to conduct experimental programs and develop analytical tools can be accomplished in a sufficient and cost effective manner. That is, the experimental and analytical effort to achieve a certain level of knowledge is appropriate to the relative importance of a phenomenon.

Similar to phenomena ranking, a PIRT panel must adopt a state of knowledge ranking scale that is sufficient, efficient and explicit. Again, the IRIS SBLOCA PIRT Panel employed a scale that has proven its worth in other PIRT efforts, as described in Table 2.3.

Table 2.3 - IRIS SBLOCA State of Knowledge Ranking Scale

Rank	Meaning
High (H)	Fully known, small uncertainty in experimental data and analytical modeling
Moderate (M)	Known, moderate uncertainty in experimental data and analytical modeling
Low (L)	Marginally or inadequately known, large uncertainty in experimental data and analytical modeling

The state of knowledge ranking results are reported in Section 3.0 (2" DVI break scenario) and Section 4.0 (4" CVCS break) corresponding to the phenomena ranking results.

Note that the scope of the PIRT panel with regards to the determination of the state of knowledge was limited to a qualitative, experience-based assessment. It is the panel's expectation that Westinghouse will review and expand these recommendations in a more comprehensive state of knowledge review.

2.2.9 Document the PIRT

Detailed and complete documentation is a key to successfully completing a PIRT effort, making it as valuable as possible to the users. This document incorporates all of the elements that have proven worthwhile in other PIRT efforts. Therefore, this report is expected to serve the needs of the IRIS Program in planning continued experimental program and analytical tool development.

3.0 DVI 2" BREAK SCENARIO PIRT RESULTS

3.1 Introduction

The low elevation, 2" break in one DVI line is illustrated in the Figure 3.1 IRIS design schematic. The general scenario, including time phases, systems and components and plausible phenomena have already been described in Section 2.0 and are not repeated here. Additional information relative to the expected scenario progression is given in the following section.

3.2 Expected Scenario Progression

IRIS is designed to limit the loss of coolant from the vessel rather than relying on systems to inject water into the RV. This is accomplished by taking advantage of the following three safety-by-design features:

1. The initial large coolant inventory in the reactor vessel;
2. The capability of removing heat directly from inside the RV thus depressurizing the RV by condensing steam, rather than by discharging mass;
3. The compact, small diameter, high design pressure containment that assists in limiting the blowdown from the RV by providing a higher back-pressure in the initial stage of the accident and then rapidly equalizing containment and the vessel pressures.

After the LOCA initiation, the reactor vessel depressurizes and loses mass to the containment vessel causing the CV pressure to rise (Blowdown Phase). The mitigation sequence is initiated with the reactor trip and coolant pump trip, followed by actuation of the EBS to provide boration, actuation of the EHRS to depressurize the primary system by condensing steam on the steam generators (depressurization without loss of mass), and finally of the ADS to assist the EHRS in depressurizing the RV. The containment pressure is limited by the PSS and the reduced break flow due to the EHRS heat removal from the RV.

At the end of the blowdown phase the RV and CV pressures become equal (Pressure Equalization) with a CV pressure peak of approximately 0.8 MPag. The break flow stops and the injection of borated water from the suppression pool becomes available.

The coupled RV/CV system is then depressurized (RV/CV Depressurization Phase) by the EHRS (steam condensation inside the RV exceeds decay heat boiloff). In this phase the break flow reverses since heat is removed not from the containment, but directly from inside the vessel. Thus, the CV pressure is reduced following the RV depressurization as steam from the containment enters the reactor vessel and is condensed inside the reactor vessel (RV and CV pressure is reduced below 0.2 MPag in not more than 12 hours). As the containment pressure is reduced, a portion of suppression pool water is pushed out through the vents and assists in flooding the vessel cavity.

The depressurization phase is followed by the Long Term Cooling Phase where the RV and CV pressure is slowly reduced as the core decay heat decreases. During this phase of the accident recovery, injection of borated water from both the suppression pool and RV cavity is available as required. Since decay heat is directly removed from within the vessel, and the vessel and containment are thermodynamically coupled, the long term break flow does not correspond to the core decay heat, but in fact it is limited to only the mass equivalent of the steam condensed on the containment vessel surface due to heat losses from the containment walls.

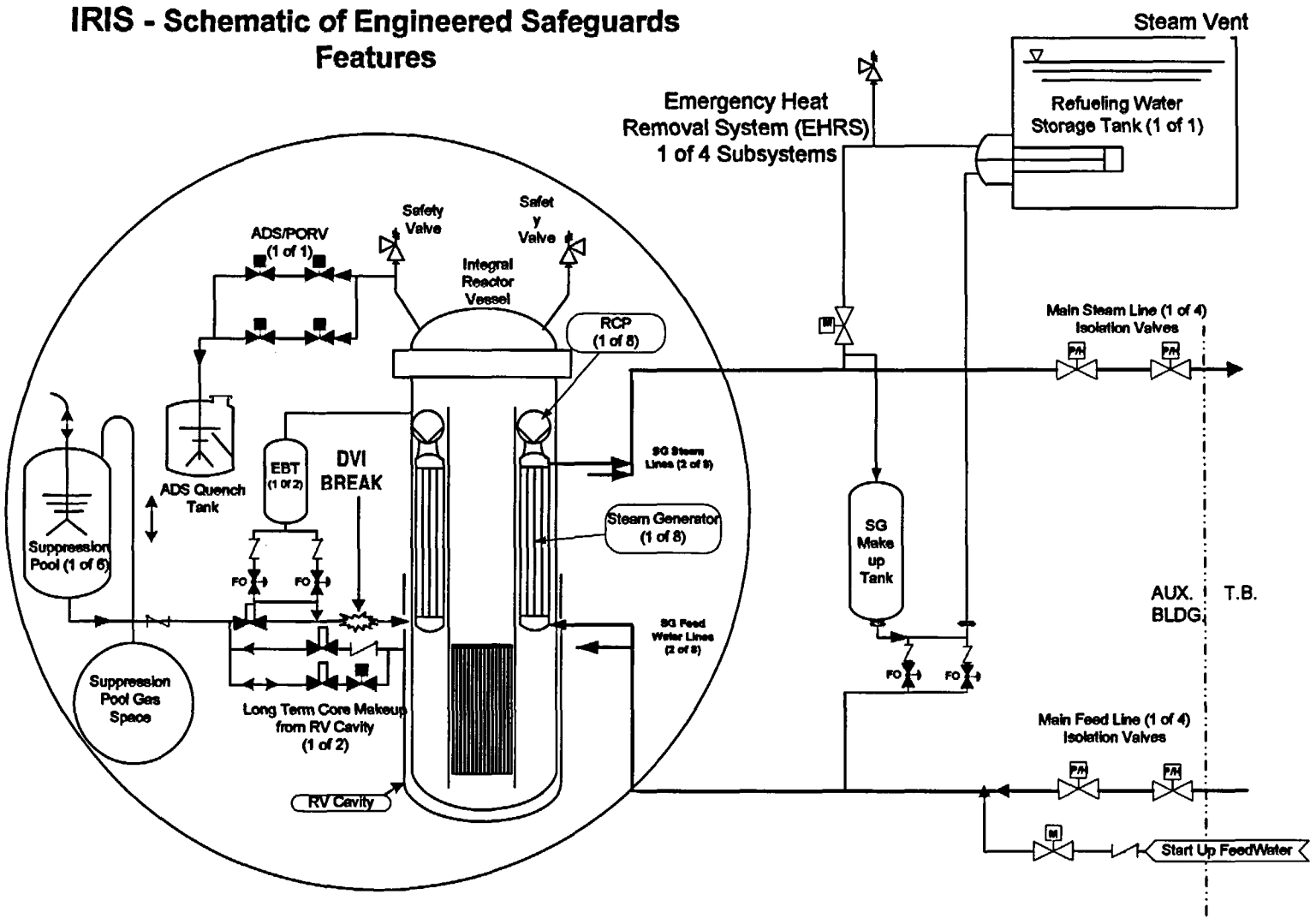

Figure 3.1 Illustration of direct vessel injection line break.

Table 3.1 lists the nominal (best estimate) sequence of events during blowdown from the most recent analytical simulation of this scenario (see Addendum 1 to this report).

Table 3.1 - IRIS DVI Break Blowdown Event Sequence

Event	Time (s)	Event	Time (s)
Break initiation	0	EHRS actuation signal generated, ADS valves begin to open	97
High containment pressure S-signal setpoint reached (reactor trip on S-signal	46	EHRS valves completely open in all four sub-systems	109
Rods begin to fall into core	48	EBT valve completely open	127
Low pressurizer level trip setpoint reached	66	Injection from the suppression pool to the reactor coolant system begins	2100
LOCA mitigation signal setpoint reached	95	Containment and pressure vessel pressure equalizes (end of blowdown)	2100
Reactor coolant pumps begin coastdown	97		

3.3 Ranking Results

The complete body of ranking results for the DVI break is provided in Appendix D. The results more significant to providing guidance to the continued IRIS experimental data and analytical modeling development are summarized in Table 3.2. These results are organized in four sets of data with each set of decreasing significance:

1. Highly important phenomena with low state of knowledge,
2. Highly important phenomena with moderate state of knowledge,
3. Highly important phenomena with high state of knowledge,
4. Moderately important phenomena with low state of knowledge,
5. Moderately important phenomena with moderate state of knowledge.

Additionally, the data within each set are further ordered by decreasing significance based on the duration in the scenario. For example, a highly ranked phenomenon with a low state of knowledge that occurs in all three phases, is considered more significant than a phenomenon with the same importance rank and state of knowledge, but occurring in two phases of the scenario.

Table 3.2 - IRIS DVI Break Summary of Significant Ranking Results

System	Component	Phenomenon	Phenomena Rank by Phase			SoK	
			I	II	III		
Highly important phenomena with low state of knowledge							
							(a, c)
Highly important phenomena with moderate state of knowledge							
							(a, c)

(a. c)

Highly important phenomena with high state of knowledge

(a, c)

							(a, c)
Moderately important phenomena with low state of knowledge							
							(a, c)
Moderately important phenomena with moderate state of knowledge							
							(a, c)

(a, c)

4.0 CVCS 4" BREAK SCENARIO

4.1 Introduction

The high elevation, 4" break in the CVCS line is illustrated in the Figure 4.1. The general scenario, including time phases, systems and components and plausible phenomena have already been described in Section 2.0 and are not repeated here.

While this scenario will present some significant difference in terms of the FOM from the 2" break discussed in the previous sections, from a phenomenological point of view the two scenarios were expected to be very similar. Therefore, the PIRT panel decided to address the upper break using the DVI break PIRT developed in Section 3 as a basis, and focusing only on the perceived differences between the two scenarios.

4.2 Scenario

The scenario description provided in Section 3.2 applies both to the DVI and CVCS breaks. The main differences between the two considered cases are summarized as follows:

1. For the upper break, the larger flow area (about 4 times that of the DVI break) and the higher position of the break in the RCS (that will lead to a break uncovery early in the transient and thus the release of steam from the RCS to the containment) lead to a faster RCS depressurization and containment pressurization. Thus, the blowdown phase tends to have a shorter duration, and the actuation of the safety features tend to occur earlier in this case.

   Table 4.1 lists the nominal (best estimate) sequence of events during blowdown from the most recent analytical simulation of this scenario (see Addendum 1 to this report).

Table 4.1 - IRIS CVCS Break Blowdown Event Sequence

Event	Time (s)	Event	Time (s)
Break initiation	0	EHRS actuation signal generated, ADS valves begin to open	30
High containment pressure S-signal setpoint reached (reactor trip on S-signal	12	EHRS valves completely open in all four sub-systems	42
Rods begin to fall into core	14	EBT valve completely open	60
Low pressurizer level trip setpoint reached	20	Injection from the suppression pool to the reactor coolant system begins	1600
LOCA mitigation signal setpoint reached	28	Containment and pressure vessel pressure equalizes (end of blowdown)	1600
Reactor coolant pumps begin coastdown	30		

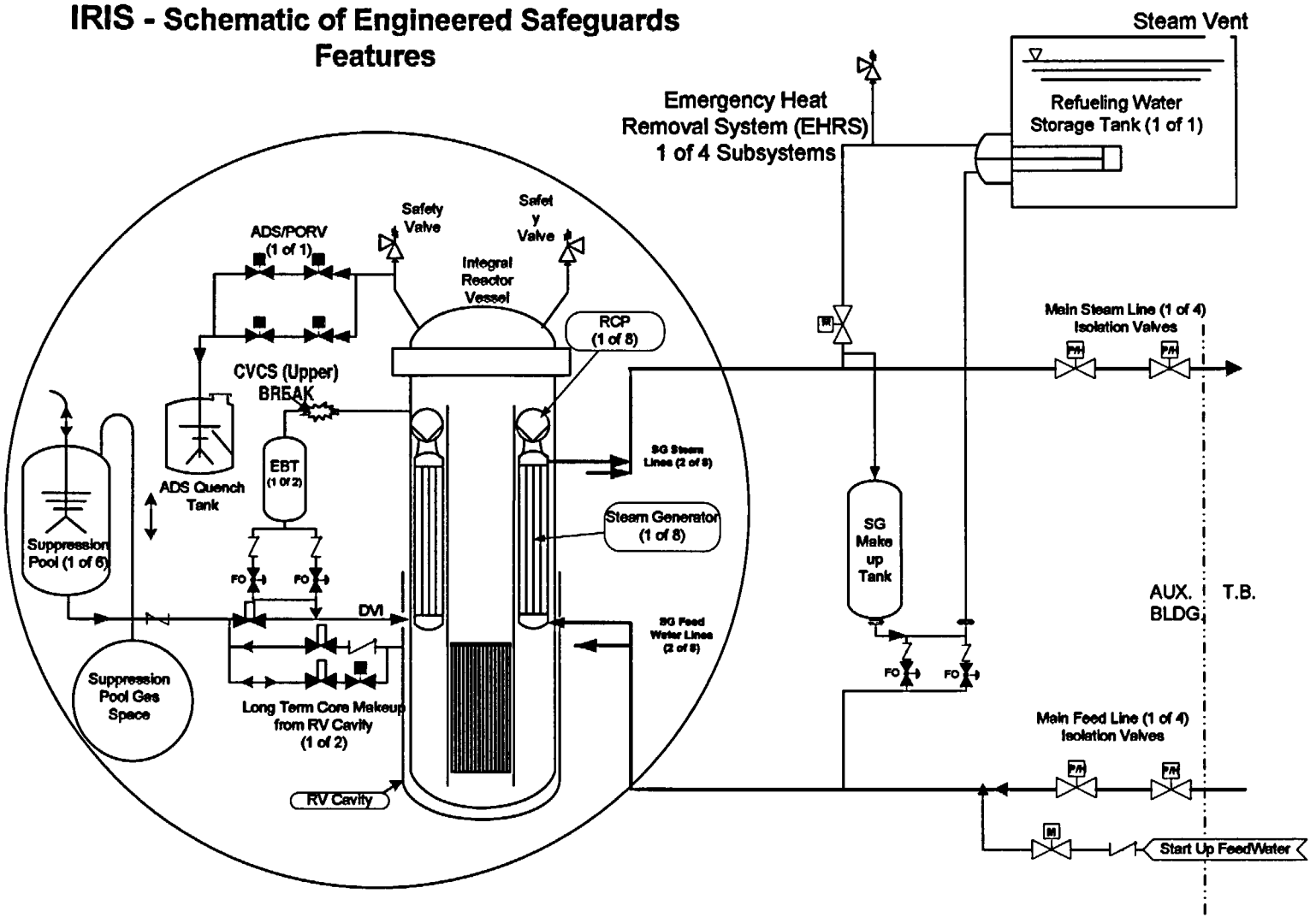

Figure 4.1 Illustration of CVCS line break.

2. For the upper break, the break itself uncovers relatively early in the transient and thus provides a venting path for the RCS to allow the equalization of the containment and RCS pressure. In the low break, this requires the actuation of the ADS. Therefore, the ADS operation in this scenario is typically less important than for the DVI break.

3. In general, the high break will lead to a more limiting scenario from the point of view of containment pressure (since the energy release at the break is higher) and to a less limiting scenario from the point of view of RCS inventory (since following the uncovery of the break, the loss of mass from the RCS is smaller than for the DVI break wherein the break does not uncover during the examined transients)

4.3 Ranking Results

The System and Component partitioning described in Appendix B applies to both the DVI and CVCS break, with the only difference related to the correct terminology for the break component. Based on this similarity at the system level, the following approach was used to develop the PIRT for the 4" Break:

1. The DVI break PIRT provided in Appendix D was used as a basis

2. The PIRT was reviewed at a system level, to identify if any difference was expected between the two SBLOCA scenarios.

3. For all cases where a difference was expected, the direction of this difference was examined: if the CVCS break suggested a lower ranking than that defined in the DVI break PIRT, the difference was ignored, since the DVI PIRT would become limiting. Thus, only for those systems/components where a potentially higher importance was expected for the CVCS break were reviewed at a phenomena level

The results of this assessment are documented in Section 4.5. Based on this assessment, it was concluded that the only significant (in the sense defined above) differences between the CVCS and DVI break are relative to the upper RCS region (Upper Pump Plenum, Reactor Coolant Pump, SG Annulus and Control Rod Drive (upper riser) Region): since the break is located near the bottom of the reactor coolant pumps region, and the mixture level during the transient can be expected to be relatively high, it is important to correctly predict the mixture level and the entrainment/de-entrainment phenomena in these region, since they will affect the composition of the fluid at the break and thus impact the overall transient and both the FOMs (containment pressure and RCS inventory).

Thus, the CVCS PIRT can be assumed identical to the DVI break PIRT except for the following changes:

Table 4.2 - IRIS CVCS Break Summary of Significant Ranking Results

System	Component	Phenomenon	Phenomena Rank by Phase			SoK
			I	II	III	

(a, c)

4.4 Ranking Rationale for the CVCS break

As discussed in the previous section, the CVCS PIRT was developed using the DVI break PIRT as a basis. The PIRT was analyzed at a system level, proceeding to the phenomena level only for those components for which the potential for higher importance to the FOMs compared to the DVI break was identified.

Following the same order used in Appendix D for the DVI break PIRT:

1. EBS. The panel did not identify any phenomenological difference between the two break scenarios for the intact EBT.

2. Reactor Coolant System. A critical area of difference between the DVI and the CVCS break is that in the CVCS break, the break will uncover earlier during the blowdown. Thus, the break mass and energy release to the containment (ranked of high importance) will be affected by the correct prediction of the mixture level in the volumes near to the break and by the composition of the water and steam mixture flowing to the break. Thus, level swell and entrainment/de-entrainment phenomena in the upper volumes in the RCS are expected to have a higher importance for this break during the blowdown phase. In particular, entrainment/de-entrainment in the reactor coolant pump, reactor coolant pump plenum and Upper Riser region were considered of high importance (versus a low or medium importance for the DVI break). Also, level swell in the riser region and the SG annulus region affects the mixture level just below the break position and are thus of high importance. Finally, flashing is the dominant effect that will lead to an increased mixture level in the in the annulus region.

   State of Knowledge for these phenomena was based on the equivalent phenomena already included in other regions in the DVI PIRT.

The remaining components of the RCS were evaluated as being essentially identical between the CVCS and DVI break. The only possible difference could be a lower importance of entrainment and level swell phenomena in the pressurizer, due to the fact that the ADS is expected to have a smaller impact on the transient in this case. Thus, the DVI break PIRT covers the CVCS break PIRT for this component and no additional considerations where necessary.

3. ADS. The ADS for the CVCS break is not going to be more important than for the DVI break, and thus the DVI break PIRT is judged sufficiently complete to cover also the CVCS break.

4. DVI. The break component of the DVI system should be removed from the DVI and separately defined as a CVCS break component. This is only a terminology issue and does not affect the ranking of the phenomena, which remain the same for CVCS and DVI breaks.

5. PSS, Containment and RVC. The Containment and related systems will be impacted by the different mass and energy release associated with the different breaks. While the transient will be different (especially from a timing point of view), this does not affect the ranking of the phenomena, that remains identical between the CVCS and DVI breaks.

6. EHRS. The EHRS phenomena are not impacted by the different breaks, and the DVI ranking applies.

7. Core. The Fuel Rods phenomena are not impacted by the different breaks, and the DVI ranking applies.

8. LGMS. One of the main differences between the CVCS and the DVI breaks is that for the CVCS break, both suppression pool gravity injection lines remain available for gravity makeup to the RCS. While it could be argued that the increased performance for the CVCS LGMS resulting from having two lines available reduces the importance of some phenomena, the DVI ranking appears adequate for the CVCS PIRT as is.

5.0 SUMMARY AND CONCLUSIONS

5.1 Summary

IRIS is a pressurized water reactor with an integral configuration (all primary system components—core, pumps, steam generators, pressurizer, and control rod drive mechanisms—are inside the reactor vessel). The IRIS plant conceptual design was completed in 2001 and the preliminary design is currently underway. The pre-application licensing process with NRC started in October 2002. The first line of defense in IRIS is to eliminate event initiators that could potentially lead to core damage. In IRIS, this concept is implemented through the "safety-by-design" approach, that is, eliminate accidents from occurring, rather than coping with their consequences. If it is not possible to eliminate certain accidents altogether, then the design inherently reduces their consequences and/or decreases their probability of occurring. One of the most obvious advantages of the IRIS safety-by-design approach is the elimination of large break LOCAs, since no large primary penetrations of the reactor vessel or large loop piping exist. An equally important example is given by the handling of small-break LOCAs, which historically have been most plaguing to PWRs. The IRIS approach is to limit and eventually stop the loss of coolant from the vessel rather than to rely on active or passive systems to inject water into the RV.

While the IRIS safety-by-design approach is a logical step in the effort to produce advanced reactors, the desired advances in safety must still be demonstrated in the licensing arena. With the elimination of LBLOCA, an important next consideration is to show that the IRIS design fulfills the promise of increased safety for the SBLOCA. Accordingly, the IRIS Consortium, through Westinghouse, established the SBLOCA PIRT Project that is documented herein. The IRIS Program issue being addressed is the planning for continued development of the experimental data and analytical tools needed for safety analysis, particularly in the licensing arena, in a sufficient and cost effective manner. Thus the primary objective of the IRIS SBLOCA PIRT project was to identify the relative importance of phenomena in the IRIS response to SBLOCAs. This relative importance, coupled with the current relative state of knowledge for the phenomena, then provides a framework for the planning of the experimental and analytical efforts.

To satisfy the primary objective of the SBLOCA PIRT Project, Westinghouse organized an expert panel. The members of this panel were carefully selected to insure that the PIRT results reflect internationally recognized experience in reactor safety analysis, and were not biased by program preconceptions internal to the IRIS Program. The seven member panel included three consultants totally independent of Westinghouse and the IRIS Program, three Westinghouse AP600/1000 engineers independent of the IRIS Program, and one member from academia with experience in AP600/1000 related experimental testing. This panel employed the well proven PIRT Process to determine the range/conditions of SBLOCA scenarios that needed to be analyzed, and then applied that methodology. The panel concluded that analysis of a low elevation break in one of the 2" Direct Vessel Injection lines, and a high elevation break in one of the 4" Chemical and Volume Control System lines would satisfy the SBLOCA PIRT objectives. These two scenarios were subsequently analyzed with the detailed results documented in the appendices in this report. The results more significant to the experimental and analytical planning are further summarized in the report main body. Lastly, the conclusions related to guidance for the experimental and analytical planning is given in the following Section 5.2.

5.2 Conclusions

The following conclusions are based in the PIRT results summarized in Sections 3.0 and 4.0, and are listed in their perceived decreasing order of significance to guiding the experimental and analytical efforts.

5.2.1 Steam Generator

[ ] (a, c)

5.2.2 Pressure Suppression System and Containment Dry Well

[ ] (a, c)

Table 5.1 - IRIS PIRT Results Pertinent to Further Containment Drywell and PSS Development

[

Component	Phenomenon	Phenomena Rank by Phase			SoK
		I	II	III	

] (a, c)

5.2.3 Emergency Heat Removal System

[ ] (a, c)

5.2.4 Core, Long Term Gravity Makeup System, Automatic Depressurization System, and Pressurizer

[ ] (a, c)

5.2.5 Direct Vessel Injection System and Reactor Vessel Cavity

[ ] (a, c)

Table 5.2 - IRIS PIRT Results Pertinent to Further EHRS Development

Component	Phenomenon	Phenomena Rank by Phase			SoK
		I	II	III	

(a, c)

Table 5.3 - IRIS PIRT Results Pertinent to Further Core, Long Term Gravity Makeup System, Automatic Depressurization System, and Pressurizer Development

Component	Phenomenon	Phenomena Rank by Phase			SoK
		I	II	III	

(a, c)

6.0 REFERENCES

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2.2 L. Cinotti, et al., "*Steam Generator of the International Reactor Innovative and Secure*", Proc. 10th Int. Conf. on Nuclear Engineering (ICONE-10), Arlington, USA, April 14-18, 2002, ASME.

2.3 J.M. Kujawski, et al., "*The IRIS Spool-type Reactor Coolant Pump*", Proc. 10th Int. Conf. on Nuclear Engineering (ICONE-10), Arlington, USA, April 14-18, 2002, ASME.

2.4 J. Collado, "*Design of the Reactor Pressure Vessel and Internals of the IRIS Integrated Nuclear System*", Proceeding of International Congress on Advanced Nuclear Power Plants (ICAPP03) Congress Palais, Córdoba (Spain), May 4-7, 2003

2.5 L. Oriani, et al., "Overview of Computational Challenges in the Development of Evaluation Models for IRIS", American Nuclear Society Topical Meeting in Mathematics & Computations (M&C), April 6-10, 2003, Gatlinburg, TN, USA

2.6 *IRIS Plant Description Document*, WCAP-16062-P and WCAP-16062-NP, March 21, 2003.

2.7 *IRIS Preliminary Safety Assessment*, WCAP-16082-P and WCAP-16082-NP, Vol. I , July, 2003.

2.8 *IRIS Preliminary Safety Assessment*, WCAP-16082-NP, Vol. II, October, 2003.

APPENDIX A

IRIS SBLOCA PIRT Panel Charter, Members and Organization

Charter

To help support the pre-application licensing effort, Westinghouse (Science and Technology Department) organized the IRIS SBLOCA PIRT project. The objective of this project is two fold:

1. To provide expert technical evaluations of the relative importance of phenomena occurring during a SBLOCA,

2. To identify the "state of knowledge" relative to the important phenomena in the context of adequate, marginal and inadequate experimental databases and analytical tools with which to conduct safety analyses. The state of knowledge assessments will be used by Westinghouse to guide continued experimental database and analysis tool development.

To insure no "conflict of interest", the Westinghouse IRIS management and staff organized a PIRT panel consisting of three members completely independent of Westinghouse, three Westinghouse members who were independent of the IRIS project, and one member from academia who, in the future, may perform IRIS related experiments as may be appropriate to the findings of the PIRT Panel. The Curriculum Vitae for the PIRT Panel members follow.

Curriculum Vitae

Independent Experts:

Thomas K. Larson - Idaho National Engineering and Environmental Laboratory (MS and BS Mechanical Engineering, Montana State University; 39 hours toward Ph.D, University of Idaho) is an Engineering Fellow at the INEEL. He has 25 plus years experience in nuclear reactor safety, thermal-hydraulic experiment design, PIRT development for reactor systems, two-phase flow scaling, analysis, measurement, instrumentation, and data acquisition, and application of expert systems, computer graphics/visualization, and systems analysis in the solution of engineering problems.

Recent work efforts include development and application of algorithms for near-field geophysics data interpretation and analysis, virtual environment applications for buried waste, development of earthquake engineering, seismic testing, and windstorm center needs related to natural hazard reduction, development and implementation of a data acquisition system and web site for the composite bridge testing effort at the INEEL, research associated with measurement and characterization of severe weather, particularly hurricane force winds, on structures as well as development/application of finite-element based analysis methods for residential structural analysis, and research in compute cluster applications and task scheduling.

He currently chairs the technical review committee for the INEEL Laboratory Directed Research and Development program. Mr. Larson has received recognition for achievements, including the Idaho State University "Outstanding Engineer Award" for the year 2000 and the Lockheed Martin Award for Excellence in recognition of exceptional quality performance and achievement for "Outstanding demonstration and defense before the Advisory Committee on Reactor Safeguards of the adequacy of RELAP5 for AP600 Safety Analysis," May 1997.

He is active in the American Nuclear Society and has held various elected governance positions including chair of two different Divisions in the Society. He has over 40 refereed publications in various technical areas.

Frederick J. Moody - Consultant (PhD and MS Mechanical Engineering, Stanford University; BS, Mechanical Engineering, University of Colorado) is an internationally recognized technical expert, educator and lecturer in nuclear reactor design, performance and safety analysis. He has 41 years experience with emphasis on fluid mechanics, thermodynamics, heat transfer, and coupled fluid-structure interaction, pertaining to reactor and containment technology. In addition to his consulting work, Dr. Moody is, and has been for 28 years, an Adjunct Professor in Thermosciences at San Jose State University. He also continues as an Instructor for General Electric's Advanced Engineering Programs and

for ASME's Continuing Education courses. He has been invited to present courses, seminars and lectures in numerous universities and national laboratories in Italy, Israel, Holland, Japan, India, Germany, Spain, Taiwan, Korea, and the U.S. on unsteady thermofluid behavior.

Dr. Moody's many awards include induction into the National Academy of Engineering (2001), Induction into Silicon Valley Engineers Hall of Fame (2000), ASME Pressure Vessels and Piping Award (1999), ASME Fellow (1981); George Westinghouse Gold Medal Award for work in two-phase flow and reactor accident analysis (1980), and Alfred Noble Award for technical paper on liquid-vapor blowdown discharge through pipes (1967).

He has served as committee chairman and co-chairman for the ASME Fluids Engineering Division, and the Pressure Vessels and Piping Division. He is an Associate Editor of ASME Journals. He has participated as an NRC-appointed peer reviewer for closing issues involving direct containment heating, severe accident progression, thermal-hydraulic studies, and application of Risk Oriented Accident Analysis Methodology. Dr. Moody is presently a consultant to the Thermal-Hydraulics Committee of NRC Advisory Committee on Reactor Safeguards (ACRS).

His recent and ongoing consulting assignments include NRC Severe Accident Scaling Methodology, High Burnup Fuel Panel, ARCO Alaska fluid transient loads in pipelines, NASA Microgravity two phase flow research peer review group, DOE Steam Explosion Study peer review, and NRC Waterhammer panel. Dr. Moody is participating in the National Academy of Engineering study on "The Safety and Security of Spent Fuel". For over 26 years he has been a Board Member, of the Medical Institutional Chaplains in Santa Clara County (a nonprofit organization devoted to placing and supporting chaplains in hospitals and other health/crisis institutions).

Dr. Moody is author of Introduction to Unsteady Thermofluid Mechanics, Wiley, 1990, and co-author of The Thermal-Hydraulics of a Boiling Water Nuclear Reactor, American Nuclear Society, 1995. He has more than 50 publications in technical journals, and symposium volumes.

Gary E. Wilson - KatJon Services, Inc. (MS Industrial Engineering, Colorado State University; BS, Aeronautical Engineering, University of Oklahoma) is the founder and president of KatJon Services, Inc. He has forty years experience in design, analysis and testing of thermal-hydraulic systems. The first 15 years were related to space launch vehicles, including ICBMs and space laboratories. The last 25 years were focused on safety analysis of nuclear reactors, including assessment of computer codes and their associated experimental databases, in the capacity of technical contributor and program manager.

Mr. Wilson was the INEEL program manager for the development and demonstration of the USNRC's Code Scaling, Applicability and Uncertainty (CSAU) evaluation methodology, including the embedded original version of the PIRT process used to support the CSAU initial application. CSAU provided the technical basis for revision of the US Code of Federal Regulation to optionally allow the use of best estimate plus uncertainty techniques in the licensing and regulation of nuclear reactors. Since 1987 Mr. Wilson has been involved in further development, application and improvement of the PIRT process. These activities include lead and support roles for PIRT process application to a B&W plant, Advanced Neutron Source production reactor, the NPR-MHTGR, Advanced Passive-600 reactor, debris transport in a BWR, Korean Next Generation Reactor, and severe accidents in the Advanced CANDUE-700.

Mr. Wilson is a member of ANS, including appointments to ANS Nuclear Installations Safety Division and Thermal Hydraulic Division National Program Committees (6 years each), election to THD Executive Committee (3 years), and election to THD Executive Committee Secretary (1996-1997). He was a member of the International Executive Committee for organization and conduct of the ANS NURETH-9 technical conference (Oct. 1999). He was Co-Chair for the organization and conduct of the ANS Best Estimate-2000 technical conference (Nov. 2000). Mr. Wilson is also Co-Chair for organization and conduct of the follow-on ANS Best Estimate-2004 technical conference (Nov. 2004).

In a private capacity, he organized and conducted a 1998 five-day workshop and lecture series on nuclear reactor safety analysis and licensing for graduate students at the University of Ljublijana, Slovenia. Mr.

Wilson has over thirty referred publications related to nuclear reactor safety analysis. He has been a Registered Professional Engineer since 1964.

Westinghouse AP1000 Experts:

William L. Brown - (MS, Mechanical Engineering, University of Pittsburgh; BAE, Architectural Engineering, Pennsylvania State University; 12 credits toward Ph.D, University of Pittsburgh) He is a Fellow Engineer at Westinghouse Electric Company and a Registered Professional Engineer licensed in the state of Pennsylvania. He has twenty-four years experience in the design, analysis and testing of thermal-hydraulic systems. The first 10 years were related to nuclear submarine design and testing for the U.S. Navy, including the SSN-21 Seawolf Class. The last 14 years were focused on the design, analysis, testing, and licensing of new commercial nuclear reactor plant designs, including the AP600 and the AP1000.

Mr. Brown's recent work includes APEX-1000 facility scaling and test analysis, PIRT development for the AP1000 and large break LOCAs in PWR containments, revisions to Appendix K small break LOCA safety analysis codes, and addressing testing, scaling, and licensing related issues in support of best estimate, small break LOCA codes and advanced plant designs such as APWR, SPWR, EPP, and PBMR.

Mr. Brown is a member of the ASME and he has co-chaired sessions at the ICONE-8 conference. He has been an invited lecturer to Tokyo Electric Power (TEPCO) and the Pennsylvania State University (PSU) on the subjects of passive plant design and scaling analysis and he has taught undergraduate engineering mechanics courses in the summer semester at PSU. Mr. Brown has received recognition for achievements, including the George Westinghouse Signature Award of Excellence in 2001 in recognition of exceptional individual quality performance and achievement for outstanding demonstration and defense before the Advisory Committee on Reactor Safeguards for the AP1000 scaling analysis. Mr. Brown has several refereed publications related to nuclear reactor scaling and safety analysis.

Cesare Frepoli - (PhD, Nuclear Engineering, Pennsylvania State University; MS, Nuclear Engineering, Polytechnic of Milano, Italy). Dr. Frepoli is currently Principal Engineer in the LOCA Integrated Services. His area of expertise includes reactor thermal-hydraulics, reactor safety, computational fluid dynamics, multi-phase flow and heat transfer.

Dr. Frepoli's recent work includes the development of a Best Estimate Large Break LOCA methodology which includes an automated statistical treatment of uncertainties method based on order statistics. He has performed several realistic thermal-hydraulic analyses of current PWRs and advanced nuclear systems such as APWR, APR1400 and AP600 and AP1000. Other activities in Westinghouse included development of advanced numerical schemes to be implemented in the WCOBRA/TRAC code to improve its stability and reduce calculation uncertainties.

Dr. Frepoli extensively supported the AP1000 team during the Design Certification process by providing resolution to thermal-hydraulic issues raised by the NRC, including various presentations to the NRC staff and ACRS committees. Previous to the AP1000 program, he was also involved with the Design Certification of the AP600. Those activities included the post-test analysis for the experiments performed in the SPES-2 and OSU (APEX) facilities. These experiments were conducted to study the AP600 behavior under Small Break LOCA and Long Term Cooling transient scenario.

As a research assistant (from 1997 to 2001) at the Mechanical and Nuclear Engineering Dept of Penn State University he was actively involved in the Rod Bundle Heat Transfer (RBHT) research program sponsored by the U.S. Nuclear Regulatory Commission. Dr. Frepoli was involved in the development of the supporting PIRT, support to the RBHT facility scaling and test design, preparation of reports, and several presentations to the NRC and the ACRS.

From 1990 to 1996 he worked at the Ansaldo Nuclear Division (Italy). His projects included CFD (Computational Fluid Dynamic) calculations and the utilization of thermal hydraulic codes (RELAP5, WCOBRA/TRAC, TRAC-GE) in the frame of nuclear safety analyses, fluid component and systems

design, plant operation services, and pre and post-test analysis. Dr. Frepoli participated in the EU-COMAS project in the context of the European Severe Accidents R&D program sponsored by the European Commission. These research activities focused on the ex-vessel core melt heat transfer, chemical interactions with surroundings and spreading behavior including oversight of prototypical corium melt spreading experiments performed in Germany (GRS, Siemens, Siempelkamp, Batelle, Achen-University).

Dr. Frepoli is an active member of the American Nuclear Society and author of various conference proceedings and referred publications related to nuclear reactor safety.

Josh Hartz - (BS Mechanical Engineering, University of Pittsburgh; 21 Credit hours toward MS in Nuclear Engineering, Pennsylvania State University with expected completion in 2005) is responsible for the Appendix K SBLOCA analysis evaluation model (NOTRUMP) and methodology. This includes standard and non-standard analyses for SBLOCA transients as well as other transient analysis to support a variety licensing applications for current operating Westinghouse and Combustion Engineering designed nuclear steam supply systems. His other responsibilities include thermal-hydraulic code maintenance and validation, 10 CFR 50.46 reporting, justification provisions for past and continued operation under un-analyzed conditions or plant equipment malfunctions, and training of employees and licensees in SBLOCA transient phenomena and behavior. Mr. Hartz has worked in this area for 5 years. Prior to that, he spent two years performing best estimate large break LOCA analysis for Westinghouse designed NSSS's. In his tenure at Westinghouse, Mr. Hartz has developed a strong working knowledge of the Westinghouse NSSS engineering design details and features and the effects of these on the SBLOCA transient. Previous to his Westinghouse employment, Mr. Hartz served as a senior design engineer at General Dynamics Electric Boat Division for a period of 6 years. This work involved concept, preliminary and detail design of a number of fluid systems in the S9G nuclear propulsion plant, including: condensate and feed, main and aux steam and a concept turbine control system. These are now complete and are under production in Virginia Class submarines. Mr. Hartz's credentials also include a two year period doing structural and electrical engineering work on mass transit cars for ABB Daimler-Benz Transportation.

Academia Expert:

Brian G. Woods - Oregon State University (PhD and MS, Nuclear Engineering, University of Maryland; BS, Mechanical Engineering, University of Virginia) is Assistant Professor, Department of Nuclear Engineering and Radiation Health Physics at OSU specializing in reactor thermal-hydraulics, reactor safety, computational fluid dynamics, multi-phase / multi-species flow and heat transfer. He is Co-Principal Investigator for the Advanced Plant Experiment Thermal Hydraulic Test Program that is performing certification testing for Westinghouse's AP-1000. Dr. Woods has sixteen years experience in thermal-hydraulic and neutronic analyses for fuel enrichment, control schemes for pressurizers, operation and maintenance of experimental reactors, investigation of rapid boron dilution transient phenomena, and development of data acquisition and control systems.

He has developed GOTHIC models for containment and emergency switchgear rooms and RETRAN models of various design basis accidents. Dr. Woods served as Chief Engineer of the USS GUARDIAN (MCM-5), and on the staff of the US Department of Energy's Office of Environmental Restoration. He is an active member of the ANS, American Physical Society and ASME. He has more than ten refereed publications related to nuclear reactor safety.

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APPENDIX B

System and Component Descriptions

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

System		Component	
Identifier	Description	Identifier	Description

(a, c)

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APPENDIX C

Plausible Phenomena During the IRIS DVI and CVCS SBLOCAs

Table C1 – Plausible Phenomena

2" Direct Vessel Injection and 4" CVCS Line Breaks

(The phenomena descriptions [DX] are listed in Table C2)

System	Component	Phenomena	Code

(a, c)

Table C1 – Plausible Phenomena

2" Direct Vessel Injection and 4" CVCS Line Breaks

(The phenomena descriptions [DX] are listed in Table C2)

System	Component	Phenomena	Code

(a, c)

Table C1 – Plausible Phenomena

2" Direct Vessel Injection and 4" CVCS Line Breaks

(The phenomena descriptions [DX] are listed in Table C2)

System	Component	Phenomena	Code

(a, c)

Table C1 – Plausible Phenomena

2" Direct Vessel Injection and 4" CVCS Line Breaks

(The phenomena descriptions [DX] are listed in Table C2)

System	Component	Phenomena	Code

(a, c)

Table C1 – Plausible Phenomena

2" Direct Vessel Injection and 4" CVCS Line Breaks

(The phenomena descriptions [DX] are listed in Table C2)

System	Component	Phenomena	Code

(a, c)

Table C1 – Plausible Phenomena

2" Direct Vessel Injection and 4" CVCS Line Breaks

(The phenomena descriptions [DX] are listed in Table C2)

System	Component	Phenomena	Code

(a, c)

Table C2 - Plausible Phenomena Descriptions
2" Direct Vessel Injection and 4" CVCS Line Breaks

Code	Phenomenon	Description

(a, c)

Table C2 - Plausible Phenomena Descriptions
2" Direct Vessel Injection and 4" CVCS Line Breaks

Code	Phenomenon	Description

(a, c)

Table C2 - Plausible Phenomena Descriptions
2" Direct Vessel Injection and 4" CVCS Line Breaks

Code	Phenomenon	Description

(a, c)

APPENDIX D

2" Direct Vessel Line Break

Detailed Phenomena and State of Knowledge Ranking Results

Table D1 - Phenomena Importance

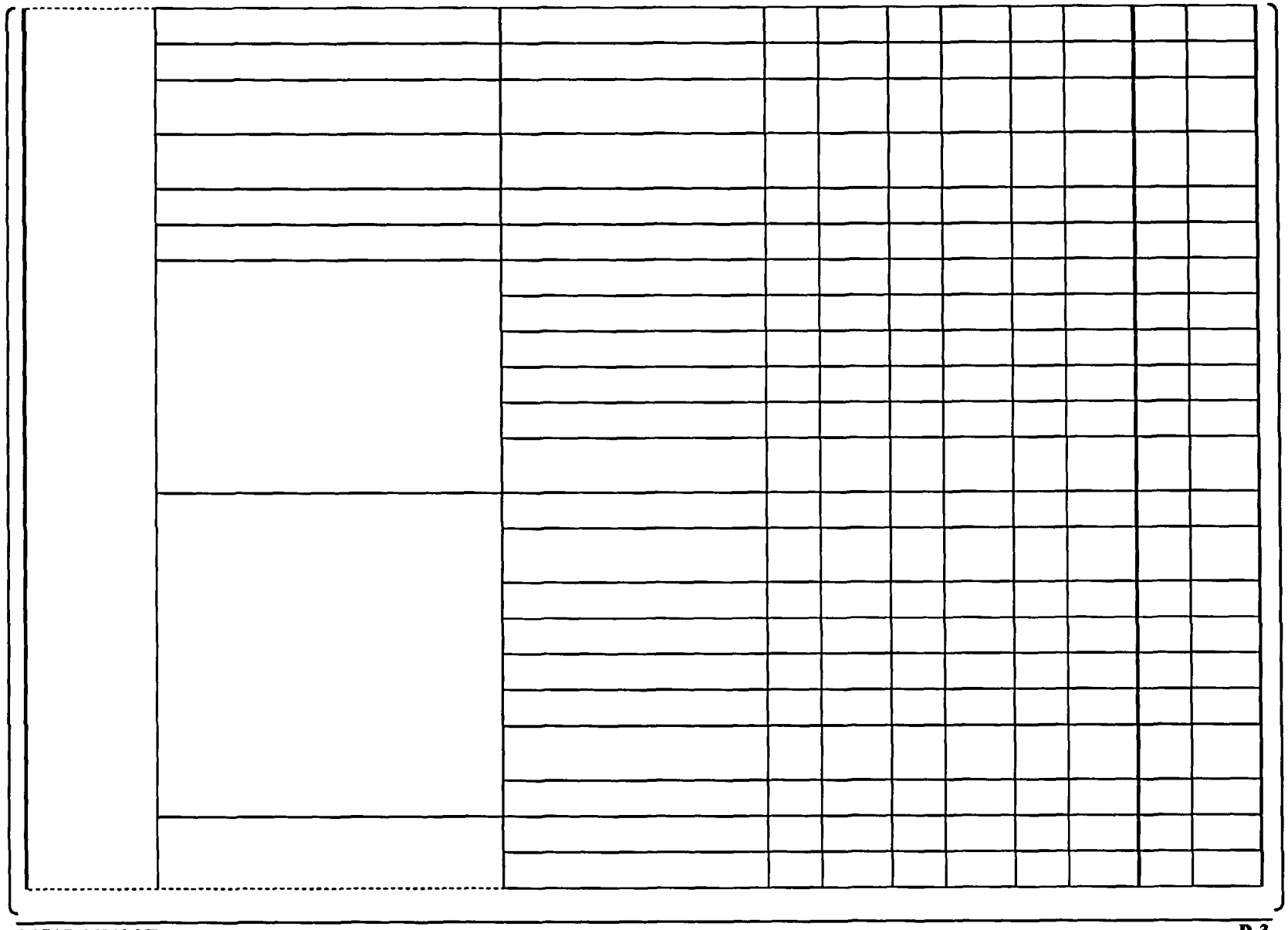
2" Direct Vessel Injection Line Break

(The phenomena ranking rationales [PX] are listed in Table D2)

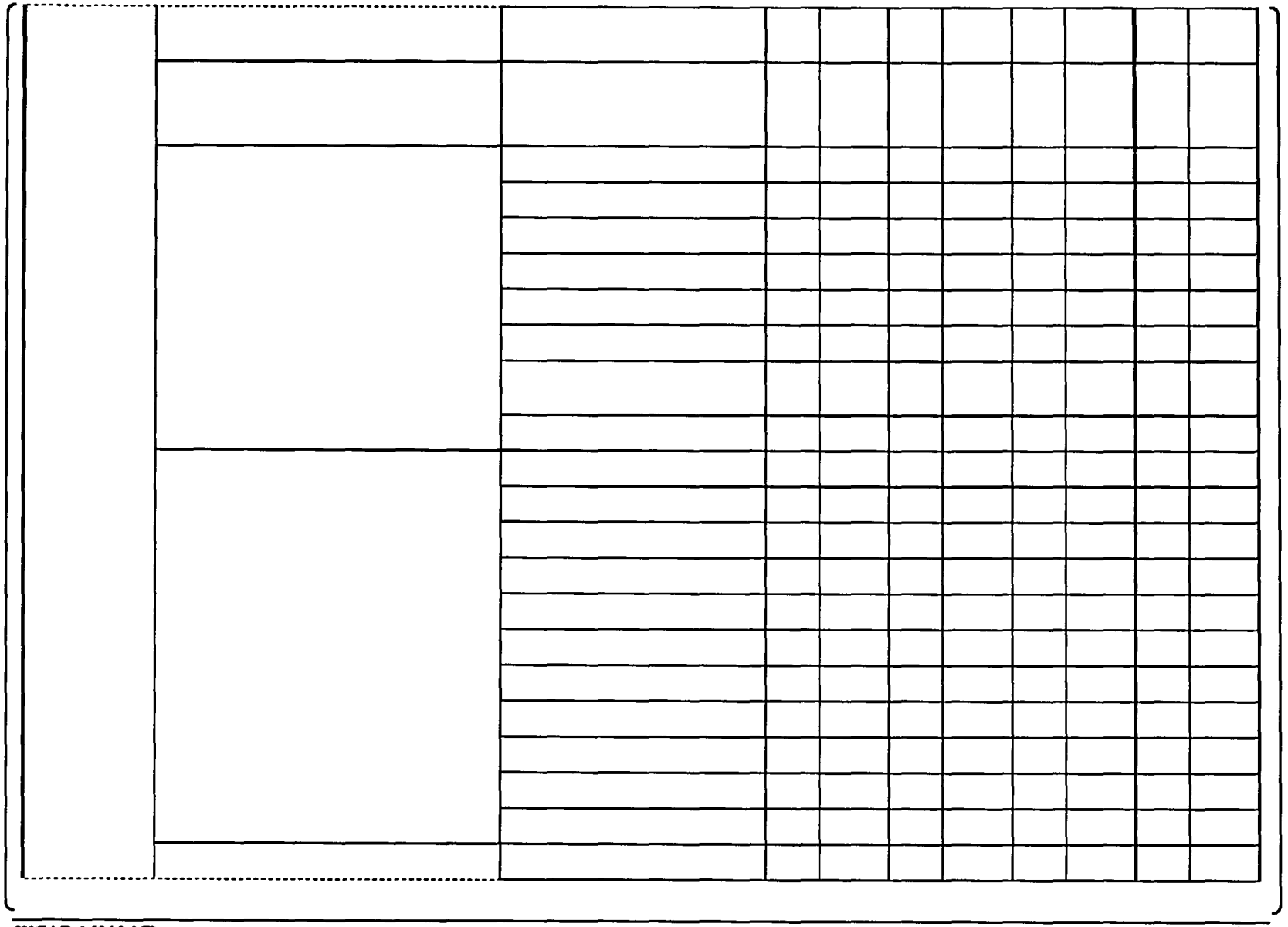
(The state of knowledge ranking rationales [SX] are listed in Table D3)

System	Component	Phenomenon	Phase I		Phase II		Phase III		State of Knowledge	
			Rank	Rationale	Rank	Rationale	Rank	Rationale	Rank	Rationale

(a, c)

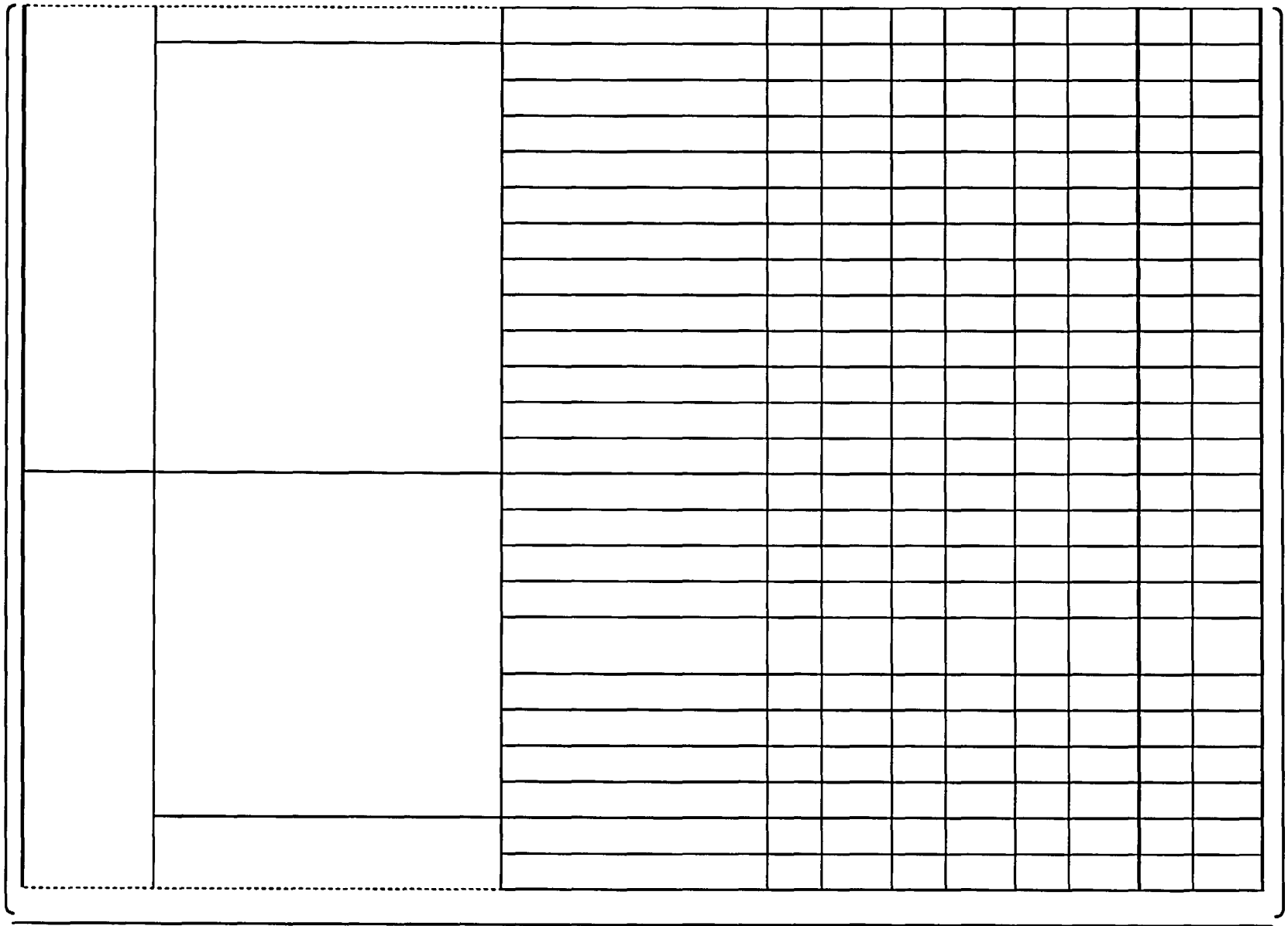


(a, c)



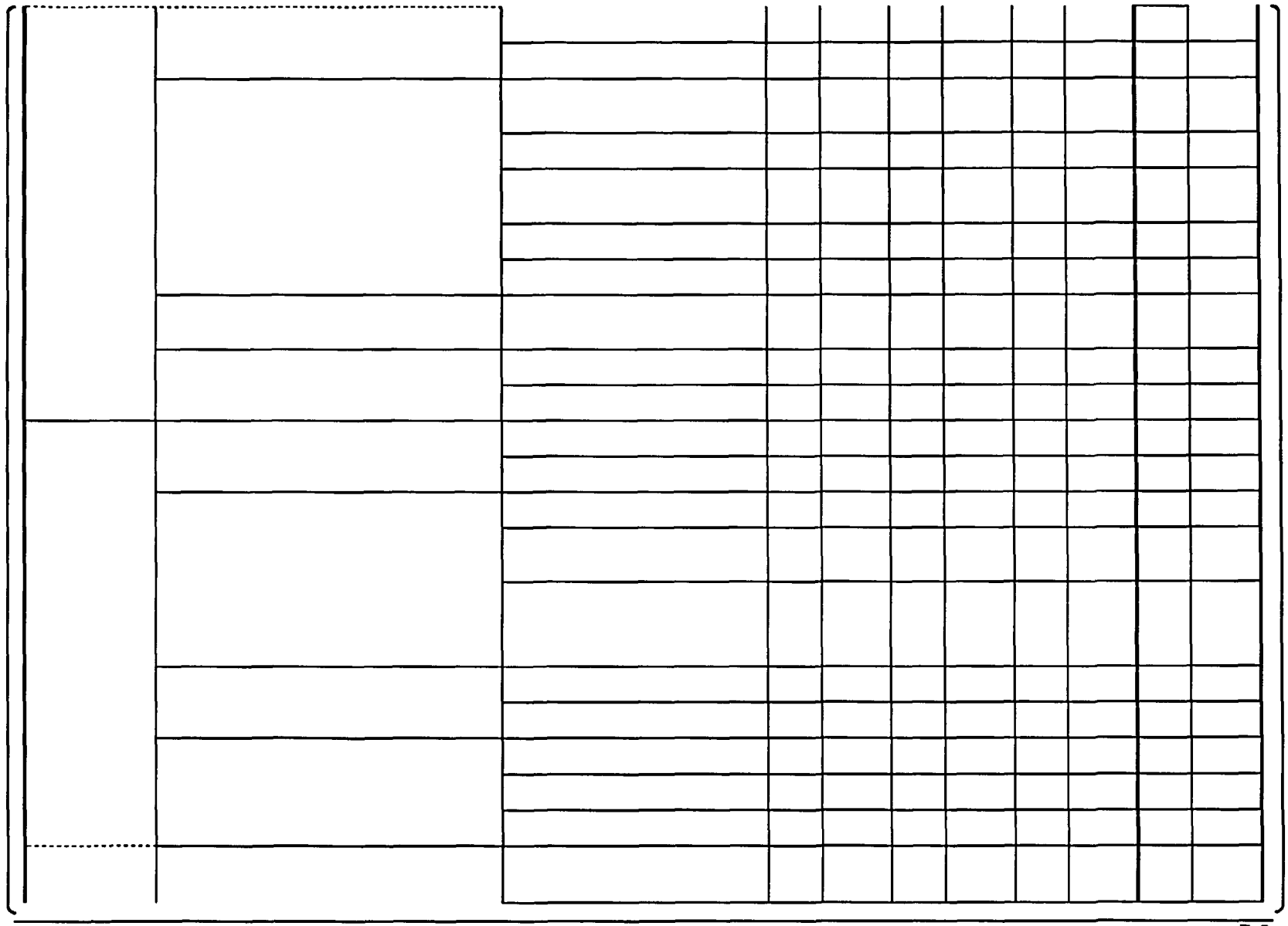
(a, c)

(a, c)



(a, c)

(a, c)



(a, c)

(a, c)

(a, c)

PHENOMENA RANKING SCALE:

H = High M = Moderate L = Low I = Insignificant or Inactive

STATE OF KNOWLEDGE RANKING SCALE:

H = Well known; adequate experimental data and computer modeling available

M = Reasonable well known; experimental data and/or computer modeling may have minor deficiencies and/or range limitations

L = Marginal or inadequate; experimental data and/or computer modeling has significant deficiencies and/or range limitations

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales

2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales

2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D2 - Phenomena Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

(a, c)

Table D3 – State of Knowledge Ranking Rationales
2" Direct Vessel Injection Line Break

Code	Rationale

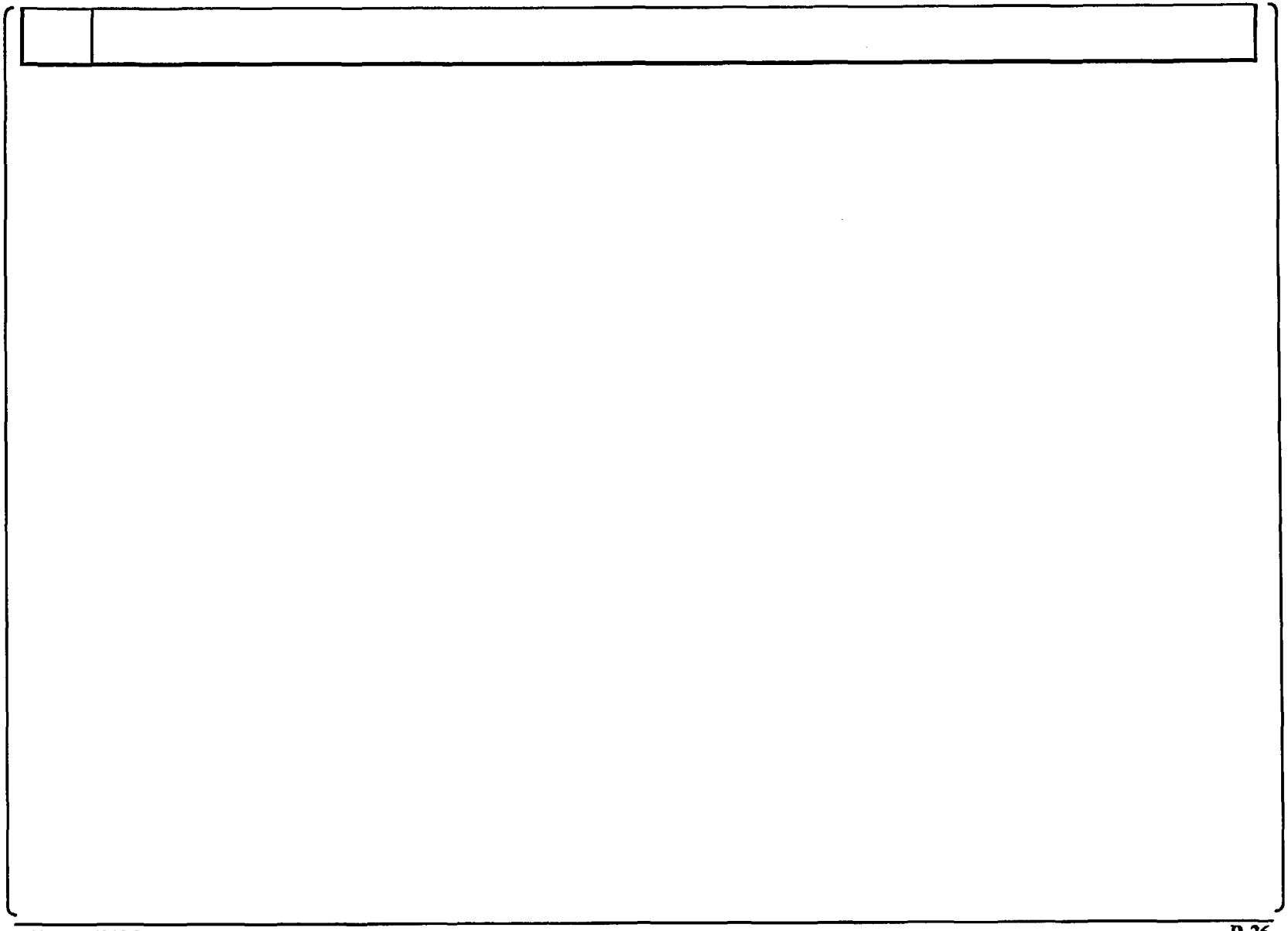
(a, c)

(a, c)

(a, c)

(a, c)

(a, c)



(a, c)

APPENDIX E

IRIS SBLOCA PIRT Panel Individual Perceptions Regarding Project Planning, Execution and Results

The intent of this appendix is to provide the PIRT panel member reactions to how the project was planned and conducted, including perceptions related to the completeness, accuracy and usefulness of the results. As a body, this information is expected to be of value to users in evaluating the PIRT results and how they may be applied.

INDEPENDENT EXPERTS:

Thomas K. Larson - I have been involved in many aspects of the PIRT process including development of and initial application of the process for the large break loss-of-coolant accident almost two decades ago as well as other applications of the process as it matured over the years. In the beginning, we struggled with how best to rank the phenomena, the components, the systems, etc. in a manner that arguably eliminates the majority of the biases that can unintentionally (or intentionally) creep into such an effort. Numerous approaches were tried including soliciting expert opinions from various sized groups (large and small) in various forums (semi-private, public, etc.) to use of an analytical hierarchical process that involved considerable effort and mathematical rigor in terms of setting up a pyramid scheme, conducting pair-wise rankings among and between the elements in the various levels in the pyramid, and then relying on software to produce the "right" answer in terms of who/what was more important in the grand scheme. These were all learning experiences and, indeed, in the latter approach, arguments regarding the software ancestry were every bit as ferocious as arguments over what was a process, a phenomena, basic physics, etc. were in the former. Note that, however, that such arguments paled in comparison to importance rankings "discussions" relative to what was a 1 or a 5 (the extremes of the scales)! What we realized in the beginning but chose to not completely accept was that simple fact that there is no single simple answer (for the most part and in contrast to most engineers' desire for THE ANSWER) and about the best one can do with this approach is exactly what one wishes it to do - that is provide a guidance framework.

Clearly, the process has matured to a point that reflects the previous lessons learned. For example, in the present application, use of the H, M, L, I scheme rather than numerical rankings seemed to work well. For whatever reason, this seems to smooth the process implementation and there were few arguments regarding whether this (or my) "pet interest" is more important relative to that (or your) "pet interest." Also helpful in the present application was the use of a strawman template at the beginning of the process so that every team member could start from the same point. Clearly the ample documentation (simulation code calculations, overall design information, detailed hardware component descriptions, etc.) helped streamline and strengthen the process even though the shear volume of information to a newcomer (nearly all involved were newcomers since IRIS is a conceptual design at this point) was overwhelming at times. The intense multi-day meeting to put the detail into the PIRT tables is, in my opinion, the best way to get the job done as it allows friendly technical interaction among a cross section of people with related but different skill sets and levels. This works in part because it is a stimulating learning experience for all involved (for me anyhow, enlightenment prevailed (the light bulb came on) several times and I found this to be rewarding) and, most importantly, it was fun. On reflection, one of the most frustrating parts of previous applications were arguments (and wasted time) over design or operation details that were contaminated with each participants less than complete (but totally correct none the less) understanding of said details. One of the pleasures of the present application was to have immediate access to IRIS design team personnel (since they were in the room) for explanations of why "it" was like that, what "its" design philosophy was, how "it" was expected to participate, etc. Without question such access streamlined the effort and helped produce a better product.

Fredrick J. Moody – The Westinghouse staff that prepared both preliminary and working background material for the PIRT panel to use provided information with optimum detail, slanted to an audience who had nuclear power plant background, but lacked in-depth familiarity with the IRIS system. Their

drawings and tables, and other descriptions made it easy to follow, and opened the way for sufficient understanding of the important processes, systems, and phenomena that play a major role in its performance.

The PIRT panel was led by G. Wilson, an excellent choice because of his extensive background in PIRT development. An atmosphere was established from the start that was particularly suited to encouraging discussion in a "brainstorming" mode so that no members felt restrained or embarrassed about expressing our thoughts, even if we thought it would reveal a lack of understanding or familiarity with the IRIS system, or some phenomenon which was not in our area of expertise.

One additional part of the PIRT development was that we could see the pieces coming together in a unified structure with a specific objective. The mix of people participating seemed to complement each other's backgrounds, and all personalities were accepting of the others – that is, no one dominated, or was overly opinionated. It soon was seen that each participant was contributing to the project to help meet the objectives. It was a rewarding team effort. Wherever strong opinions were expressed, reasons and background were offered so that every identified and ranked phenomenon can be explained and justified.

Gary E. Wilson – Based on experience in the original development of the PIRT, further refinement of the process and at least five significant applications of the methodology, the following summarizes the more significant perceptions formulated during this project.

- The high degree of openness and cooperation of the IRIS staff supporting the panel was most welcome and contributed strongly to a high level of accomplishment in a very short project schedule.
- The IRIS design/experimental/analytical database provided to the panel was impressive in its breadth and contributed largely to the panel obtaining sufficient knowledge to feel confident in its conclusions concerning potential plant behavior.
- The reactor safety knowledge embedded in the other panel members was impressive in its scope and content. It is estimated that some 200 man-years of reactor design, analysis and operational experience was available to the project.
- Although four members of the panel were either Westinghouse employees or an experimental subcontractor, their complete independence from the IRIS project and their objectiveness was noticeable. Thus, an original minor concern with a potential conflict of interest was quickly alleviated early in the project.
- As might be expected from a panel of highly informed and knowledgeable individuals, expert in a broad range of fields, there were occasional strong differences of opinions. Fortunately, the panel members shared the same high degree of professionalism and commitment to the project. Therefore, almost without exception, the panel was able to resolve these differences to come to an acceptable consensus.
- In summary, execution of this project is considered one of the best of the several such projects undertaken, in the context of efficiency, sufficiency, accuracy and usefulness of the results.

WESTINGHOUSE AP1000 EXPERTS:

William L. Brown - The expert panel brought a significant range of knowledge and experience to bear on IRIS SBLOCA phenomena. This resulted in a fairly comprehensive identification of potential phenomena expected for IRIS SBLOCA events that should be quite useful for purposes of IRIS test program and analysis code development. To provide additional insight into the relative importance of the phenomena identified by the PIRT panel, it is recommended that the expert opinions of the PIRT panel be supplemented with scaling analyses of the potential IRIS SBLOCA phenomena. Scaling is an effective tool to help confirm PIRT phenomena rankings and provides great insight in developing a test program.

Based upon past experience with PIRT, scaling, and testing efforts for new plant designs such as AP600 and AP1000, it should be recognized that the PIRT effort for any new plant design such as IRIS at this stage is still preliminary in nature. This is particularly true for phenomena associated with the interaction between the reactor coolant system and the containment. The IRIS approach to handling SBLOCAs is unique (relative to AP600 and AP1000) in using the containment to aid in limiting and eventually stopping the loss of coolant from the reactor vessel. Participating in the discussions of the IRIS PIRT panel has led me to recognize that this interaction between the reactor coolant system and the containment is important and very complex and is yet there is limited understanding of the interaction at this time, at least from an integral effects perspective. Therefore, it should be cautioned that the IRIS SBLOCA PIRT may be incomplete in this area and probably it is best to rely upon integral effects testing to improve the state of knowledge of the phenomena associated with this complex interaction.

Cesare Frepoli - The PIRT panel that was selected for the IRIS represented a good mix of different experiences and backgrounds. This contributed to establish, from the start, the right chemistry among the panel members for a very productive discussion. Despite the meeting was limited to few days, significant amount of work was accomplished where the main issues related to the safety of the plant thermal-hydraulic design were identified and phenomena ranked accordingly.

The framework for the PIRT development prepared in advance by the G. Wilson provided a very efficient and constructive walk through the systems, components and phenomena of the IRIS during a postulated Small Break LOCA.

Also, the approach from the panel members was to consider the existence of RELAP5 sensitivity studies only for guidance and as a tool to stimulate independent thinking, without necessarily putting too much faith in the results provided. As a result, the ranking of the phenomena was based on the expert opinion of panel members and not on the results of sensitivity studies. Results from of the sensitivity studies will be used in future analyses to confirm the assessment or identify code deficiencies and area of improvements. A similar approach was followed for the scaling analyses. For the same reason the PIRT can be used to guide the scaling analyses and not the opposite.

Obviously, the PIRT is still a qualitative assessment, even if based on expert opinions. The main purpose of the PIRT is to guide the development of new separate and integral tests, identify weak areas and deficiencies of current thermal-hydraulic tools which will be used to demonstrate the safety of the IRIS design. The recommendation is that, more quantitative analyses based on applicability of available tests, definition of new tests, scaling analyses and code sensitivity studies are necessary to provide a more robust conclusion about the safety of the design. Within this process, the PIRT needs to be seen as iterative in nature and will be further refined as more knowledge is brought into the process.

It was a general consensus that the coupling between the RCS and the containment is a key element of the IRIS safety concept. Compared to current operating PWRs and advanced plants such as the AP1000, this

is a unique feature. The interaction between the two systems is quite complex. In my opinion, this is an area where available T/H codes have limitations. Separate and integral effects on this type of interaction is limited. Time scales for the phenomena of interest span over a wide range and this represents a challenge in available T/H code.

Finally, it is recognized that the 'safety by design' concept built into the IRIS, represents a substantial mitigating factor where large volume of liquid are available for core cooling and it is difficult to postulate scenarios which lead to core uncovery. As a result, the implication of inaccuracies in modeling high important phenomena could be regarded as less severe than in traditional PWRs or advanced plant such as AP1000. However, despite these mitigating factors, for the safety analyses, it is recommended to maintain the focus on the impact on the selected figures of merit which are the containment pressure and vessel mass inventory for the IRIS. This will provide a robust defense of the safety case for the IRIS concept.

Josh Hartz - The PIRT panel and process for IRIS SBLOCA was a very noteworthy experience and this participant was honored to be in such good company. Everyone that participated brought a substantial amount of knowledge and experience to the table when the review was conducted. This experience included:

- An extreme, in-depth understanding of the physical phenomena that exist both in SBLOCA's and other operational transients as well. This understanding was essential to the panel's ability to effectively develop the IRIS PIRT.
- Knowledge on AP-1000/AP-600 plant designs and current operating plant designs (both pressurized water reactors and boiling water reactors) was utilized extensively. This was applied to the similarities and differences that exist within the IRIS design and how these would affect the design basis SBLOCA transients discussed for IRIS. This was especially true for the ADS system and the emergency heat removal system and how such concepts were implemented into AP-1000/AP-600. The knowledge base aspects of those configurations provided useful insight into the IRIS design and how IRIS was perceived to respond to the SBLOCA transients considered. In addition, any short comings that evolved in the design, analysis and test programs of these features in AP-1000/AP-600 were thoroughly considered when they were applied to the IRIS PIRT process.
- The group used an un-biased approach to conceptualize the behavior of the transients considered. That is, preliminary RELAP results were available from runs that simulated the design basis SBLOCA transients for IRIS. These could have easily been used to sway the judgment of the panel, but they were not used in this capacity. Instead, the panel used all the combined knowledge and experience available to perform the PIRT process with little input from the model results other than to see if it was in agreement with the conclusions of the panel, and only in some cases.
- Experience with many thermal-hydraulic test programs, both more recent as well as in the past was applied to the PIRT process. This experience was used as a reference for some of the expected responses of the SBLOCA transients considered for the IRIS PIRT. Shortfalls in these test programs were also used to evaluate the knowledge base for a particular component phenomena.
- Several panel members have been involved in previous PIRTS for other programs. This experience was helpful in the development of the IRIS SBLOCA PIRT.

In summary, this panel member is satisfied with all the contributions brought by the various panel members to develop the IRIS PIRT and considers this PIRT a strong foundation for the licensing and acceptance of IRIS w/respect to its ability to effectively mitigate an SBLOCA transient.

ACADEMIA EXPERT:

Brian G. Woods - The whole PIRT process was conducted in a highly professional manner. I believe that the PIRT documented in this report is complete and accurate. It is expected that this information will be extremely useful for further IRIS certification and testing activities. One aspect of this PIRT process that was of particular value is the work that was completed prior to the actual meeting of the PIRT panel which included a conference call and preliminary ranking of systems and phenomena. This preliminary work allowed for more useful discussion during the actual meeting of the panel.